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Rockville, Maryland 20852 USA

Nuclear Pressure Equipment Expertise and Regulation 2005 Symposium
June 22 – 24, 2005

REGULATORY PERSPECTIVE ON STEAM GENERATOR TUBE OPERATING EXPERIENCE

Introduction

The United States Nuclear Regulatory Commission (NRC), as part of its mission to protect public health and safety and the environment. In fulfilling this mission the NRC
(1) monitors and assesses steam generator tube operating experience and (2) reviews industry proposals to revise steam generator tube inspection and repair requirements in the technical specifications. This paper reviews significant occurrences, trends, and issues in the recent past relating to steam generator tube integrity from a regulatory perspective. This paper focuses on (1) the status of replacing steam generators, (2) recent experience at plants with thermally-treated Alloy 600 tubes, (3) recently issued or planned generic communications, and (4) the status of revising the steam generator portion of the technical specifications. Each of these focus topics are discussed in the following sections.

Steam Generator Replacement Status

There are 69 operating pressurized water reactors in the United States. Each of these pressurized water reactors have two to four steam generators, and each steam generator can contain anywhere from 3,000 to 16,000 tubes. These tubes have an important safety role because they constitute one of the primary barriers between the radioactive and
non-radioactive sides of the plant. For this reason, the integrity of the tubing is essential in limiting the leakage of radioactive water to the environment.

The susceptibility of steam generator tubes to degradation is affected by various factors, including the steam generator design, the operating environment (temperature and water chemistry), and operating and residual stresses. Two of the most important factors affecting the susceptibility of a tube to degradation are the tube material and the tube's heat treatment.

During the early-to-mid 1970s, when all plants in the United States, except one, had mill-annealed Alloy 600 steam generator tubes, tube thinning was the dominant cause of tube degradation. In the mid-to-late 1970s, tube denting became a primary concern. Denting resulted from the corrosion of the carbon steel tube support plates and the buildup of corrosion products in the crevices between the tubes and the tube support plates. The extensive tube degradation at pressurized water reactors with mill-annealed Alloy 600 steam generator tubes has resulted in tube leaks, tube ruptures, and in some cases has caused the NRC to require mid-cycle steam generator tube inspections. It has led licensees to replace steam generators at 39 plants in the United States, as indicated in Table 1. In addition, extensive tube degradation contributed to the shutdown of Haddam Neck, Maine Yankee, Trojan, Zion 1, Zion 2, and San Onofre 1.

As mill-annealed Alloy 600 steam generator tubes began exhibiting degradation in the early 1970s, the industry pursued improvements in the design of future steam generators to reduce the likelihood of corrosion and other service-induced degradation. One of these improvements involved subjecting the Alloy 600 tubes to a high temperature thermal treatment (approximately 705°C) for 10 to 15 hours to promote carbide precipitation at the grain boundaries and diffusion of chromium to the regions adjacent to the grain boundaries. This thermal treatment process was intended to reduce the susceptibility of the material to stress corrosion cracking. In addition to the thermal treatment process, other design improvements to increase the tubes' resistance to degradation included
(1) expanding the tubes into the tubesheet by hydraulic means rather than by mechanical rolling or explosive methods, (2) use of stainless steel rather than carbon steel for the tube support plates, and (3) use of quatrefoil-shaped holes instead of round shaped holes.
This latter improvement limits the contact between the tube and the support plate to four narrow lands, minimizing local dryout and chemical concentration.

The first steam generator replacement in the U.S. took place at Surry Unit 2 in 1980. These replacement steam generators have thermally-treated Alloy 600 tubes and all of the design improvements mentioned above. In the early 1980s, approximately one plant per year replaced their steam generators. All of these replacement steam generators had thermally-treated Alloy 600 tubes. In 1989, the first steam generators with
thermally-treated Alloy 690 tubes were placed into service in the United States. Alloy 690 is a nickel-chromium-iron alloy that is similar to Alloy 600. The principal differences between Alloy 690 and Alloy 600 are the increase in chromium content from about 16 percent to 30 percent, a decrease in the nickel content from 72 percent minimum to 58 percent minimum, and a decrease in carbon content from 0.15 percent maximum to 0.05 percent maximum.

The higher chromium content in Alloy 690 when compared to Alloy 600 reduces the degree of sensitization (i.e., the amount of chromium depleted in areas adjacent to the metal grain boundaries), thus increasing resistance to corrosion attack at the metal grain boundaries. The heat treatment, which is intended to improve the stress corrosion cracking resistance of the material, involves mill-annealing at temperatures sufficient to put all the carbon into solution, followed by a thermal treatment to precipitate carbides on the metal grain boundaries into the tube metal microstructure. [Resistance to stress corrosion cracking in this material is greatest when the metal grain boundaries are fully populated with carbides.] Alloy 690 is more resistant to both primary and secondary side stress corrosion cracking, pitting, and general corrosion. The superior resistance of

Alloy 690 to intergranular attack, pitting corrosion, primary water stress corrosion cracking, and outside diameter stress corrosion cracking has been attributed mainly to the higher chromium content.

Since 1989, most of the replacement steam generators installed in the United States have primarily used tubes fabricated from thermally-treated Alloy 690. The few exceptions are Palisades, Salem 1, and Indian Point 2 as shown in Table 1. Of the 69 currently operating pressurized water reactors in the United States, 22 (32 percent of plants) have steam generators with mill-annealed Alloy 600 tubes, 17 (25 percent of plants) have steam generators with thermally-treated Alloy 600 tubes, and 30 (43 percent of plants) have steam generators with thermally-treated Alloy 690 tubes as of April 2005. Figure 1 shows the percentage of plants with a given tube material and heat treatment have evolved with time. As shown in the figure, there are currently more plants that have thermally-treated steam generator tubes than there are plants with mill-annealed tubes.

The decrease in the number of plants with mill-annealed steam generator tubes is expected to continue. Currently, approximately 2 to 4 plants with mill-annealed Alloy 600 tubes are replacing their steam generators per year. For example, in the fall of 2005, Callaway, Arkansas Nuclear One 1, and Palo Verde 1 are planning to replace their steam generators. In 2006, four additional plants with mill-annealed Alloy 600 tubes plan on replacing their steam generators.

It is important to evaluate the operating experience with replacement steam generators in order to identify the need for additional (or more frequent) steam generator tube inspections or repair. Although thermally-treated Alloy 600 is no longer the material of choice for new or replacement steam generators, its operating experience can provide insights into the future behavior of newer steam generators with Alloy 690 tubes.
The operating experience associated with thermally-treated Alloy 600 steam generator tubes is discussed in the following section.

Thermally-Treated Alloy 600 Steam Generator Tube Experience

There are currently 17 plants with thermally-treated Alloy 600 steam generator tubes in the United States. An additional plant has a small fraction of its tubes fabricated from thermally-treated Alloy 600. As discussed above, thermally-treated Alloy 600 tubes were first placed into service in the United States in 1980. There are approximately 281,000 thermally-treated Alloy 600 tubes in the 17 plants with this tube material. All the steam generators in the United States with thermally-treated Alloy 600 tubes were designed and fabricated by Westinghouse.

Table 1: Plants with Replacement Steam Generators

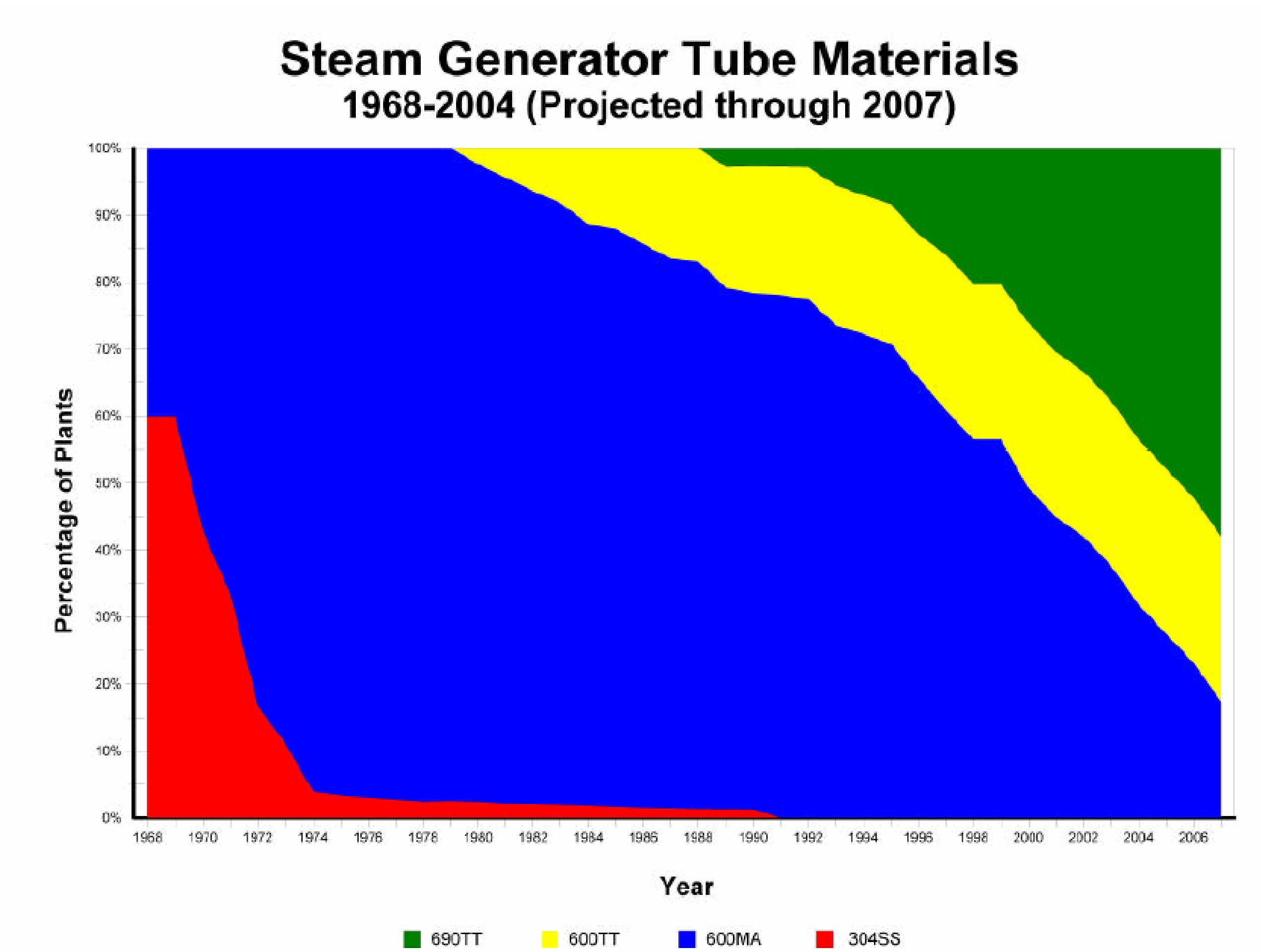
Plant Name	Year Installed	Tube Material[1]
Surry 2	9/80	600 TT
Surry 1	7/81	600 TT
Turkey Point 3	4/82	600 TT
Turkey Point 4	5/83	600 TT
Point Beach 1	3/84	600 TT
Robinson 2	10/84	600 TT
Cook 2	3/89	690 TT
Indian Point 3	6/89	690 TT
Palisades	3/91	600 MA
Millstone 2	1/93	690 TT
North Anna 1	4/93	690 TT
Summer	12/94	690 TT
North Anna 2	5/95	690 TT
Ginna	6/96	690 TT
Catawba 1	9/96	690 TT
Point Beach 2	12/96	690 TT
McGuire 1	5/97	690 TT
Salem 1	7/97	600 TT
McGuire 2	12/97	690 TT
St. Lucie 1	1/98	690 TT

Table 1: Plants with Replacement Steam Generators Continued

Plant Name	**Year Installed**	**Tube Material**[1]
Byron 1	1/98	690 TT
Braidwood 1	11/98	690 TT
South Texas Project 1	5/00	690 TT
Farley 1	5/00	690 TT
Cook 1	12/00	690 TT
Arkansas Nuclear One 2	12/00	690 TT
Indian Point 2	12/00	690 TT
Farley 2	5/01	690 TT
Kewaunee	12/01	690 TT
Harris	12/01	690 TT
Calvert Cliffs 1	6/02	690 TT
South Texas 2	12/02	690 TT
Calvert Cliffs 2	5/03	690 TT
Sequoyah 1	6/03	690 TT
Palo Verde 2	12/03	690 TT
Oconee 1	1/04	690 TT
Oconee 2	6/04	690 TT
Prairie Island 1	11/04	690 TT
Oconee 3	12/04	690 TT

[1]TT = thermally-treated, MA = mill-annealed

Figure 1: Percentage of Plants with a Given Tube Material and Heat Treatment as a Function of Time



As discussed in NUREG-1771, “U.S. Operating Experience with Thermally-Treated Alloy 600 Steam Generator Tubes”, only 0.5 percent of the thermally-treated Alloy 600 tubes were plugged as of December 2001. The dominant degradation mode (and cause for steam generator tube plugging) in thermally-treated Alloy 600 tubes is wear. Of the approximate 1400 tubes plugged, 53 percent of the tubes were plugged as a result of wear. None of the tubes plugged prior to 2002 were plugged as a result of a confirmed crack in a tube.

As alluded to above, the relatively good operating experience of plants with thermally-treated Alloy 600 steam generator tubes can be attributed to several factors besides the heat treatment of the tubes. For example, the hydraulic expansion of the tubes into the tubesheet, the quatrefoil design of the tube support plates, and the stainless steel material used to fabricate the tube support plates contribute to the good operating experience.

In 2002, the first confirmed instance of stress corrosion cracking affecting thermally-treated Alloy 600 tubing was reported in the United States. This cracking occurred at Seabrook. Since this initial finding, three other plants (Braidwood Unit 2, Catawba Unit 2, and Vogtle Unit 1) with thermally-treated Alloy 600 tubing have found cracking indications. Each of these instances of cracking is described in greater detail below.

Seabrook Station is a Westinghouse four-loop pressurized water reactor with Model F steam generators. The unit had operated for approximately 9.7 effective full-power years as of May 2002. During an outage in May 2002, axially oriented linear indications were detected on the outer diameter tube surface of the tube at a number of tube-to-tube support plate intersections. The maximum depth of the indications was estimated to be 62 percent through-wall and the lengths ranged from 0.3 to 0.75 inch. All tubes with cracks were plugged. These findings are noteworthy because Seabrook was the first plant with thermally treated Alloy 600 to observe cracking in the U.S. despite having operated for less time than most other plants with thermally-treated Alloy 600.

An evaluation was performed to determine the root cause of the cracking. The principal cause was determined to be elevated residual stresses in the degraded tubes that made them more susceptible to corrosion in the operating environment. The elevated residual stress levels were attributed to non-optimal tube processing. The precise processing steps responsible for the adverse stress state could not be conclusively determined from a review of the tube processing records. During the root cause investigation, a unique offset or shift on the low frequency absolute channel between the straight leg portion of the tube and the U-bend region was observed in the cracked tubes. This offset was attributed to changes in the residual stresses in the tube. No offset in the eddy current data had been expected since the U-bend region of the cracked tubes had been stress relieved after bending.

Following the 2002 inspections, six additional tubes with the unique offset were determined to exist at Seabrook. None of these tubes had cracks during the 2002 outage. During Seabrook's next outage in October 2003, these six tubes were inspected. Of these six tubes, there were three tubes with nine indications of outside diameter stress corrosion cracking at the tube support plate elevations. All six of these tubes were plugged (i.e., even those with no crack indications). Additional details regarding the findings at Seabrook are contained in NRC Information Notice 2002-21, Supplement 1.

Subsequent to the findings at Seabrook, plants with thermally-treated Alloy 600 tubing began to review their eddy current data for offsets similar to that observed at Seabrook. Although several of the plants found tubes with the offset signal, only one plant (Braidwood 2), to-date, has found cracks in these tubes (it appears that). Braidwood Station Unit 2 is a Westinghouse four-loop pressurized water reactor with Model D5 steam generators. The unit has operated for approximately 10.9 effective full-power years as of their 2003 outage. During their 2003 outage, a total of four hot-leg tube support plate intersections in three tubes with the offset signal were identified as containing outside diameter stress corrosion cracking. All tubes were plugged.

Prior to 2004, most (if not all) of the crack indications in thermally-treated Alloy 600 tubes in the United States occurred at two plants and were attributed to non-optimal tube processing. These crack-like indications occurred in the region of the tube where it passes through the tube support plate. In 2004 and early 2005, several plants with thermally-treated Alloy 600 steam generator tubes found cracks in the portion of the tube contained within the tubesheet. The circumstances surrounding these findings are discussed below.

Catawba Nuclear Station Unit 2 is a Westinghouse four-loop pressurized water reactor with Model D5 steam generators. The unit has operated for approximately 14.7 effective full-power years as of their 2004 outage. During fabrication of the steam generators, a portion of the U-shaped tubes are inserted into a thick plate called a tubesheet. The tubesheet is approximately 21-inches thick and has two holes for each tube (one hole on the hot-leg side of the steam generator and one hole on the cold-leg side). The lower ends of the tubes were tack-expanded into the tubesheet for approximately 0.70-inch. This tack expansion is performed to facilitate welding of the tube to the primary side of the tubesheet. In the case of Catawba Unit 2, this region is frequently referred to as the tack roll region since the tack expansion was accomplished by mechanically rolling the tube into the tubesheet. Following welding, the tubes were hydraulically expanded for the full depth of the tubesheet.

During an outage in 2004, three discrete circumferential indications were found in an overexpanded region within the tubesheet region of one tube at Catawba 2. Overexpanded regions such as this are sometimes referred to as bulges or tubesheet anomalies. The indications were located approximately 7-inches below the top of the hot-leg tubesheet.

In addition to the three indications found in the bulged region of one of the tubes, nine tubes were found to have circumferentially oriented indications in the tack roll region and several hundred tubes were found to have indications in the tube-to-tubesheet weld. In six of the tubes with tube-to-tubesheet weld indications, the indications extended into the parent tube. The indications in these tubes consisted of either single or multiple cracks. The findings are very noteworthy because potential crack-like indications were first found in manufacturing anomalies (e.g., bulges or tubesheet anomalies) rather than at other tube locations such as the expansion transition or U-bend region. Expansion transitions and U-bends have high residual stresses and are routinely considered to be the leading indicator that cracking is occurring in the steam generator tubes. Additional information pertaining to the findings at Catawba Unit 2 can be found in NRC Information Notice 2005-09.

Subsequent to the findings at Catawba Unit 2, a few cracks were also found at Vogtle Unit 1. Vogtle Unit 1 is a Westinghouse four-loop pressurized water reactor with Model F steam generators. During an outage in 2005, three circumferential indications were found on the inside diameter of two of the tubes. The indications were associated with bulges or overexpansions within the tubesheet region.

In general, the operating experience with thermally-treated Alloy 600 tubing has been favorable in the U.S. To date, only a limited number of tubes at a few plants have exhibited cracking. In addition, the operating experience at the 30 plants that have thermally-treated Alloy 690 tubes has been favorable with no reported incidence of cracking.

As noted in the examples given in this section, the NRC uses generic communications to inform the industry of recent operating experience. The NRC anticipates that recipients will review the information for applicability to their facilities and consider taking actions to avoid similar issues. The most recent generic communications related to steam generator tube integrity are discussed in the following section.

Steam Generator Generic Communications

Generic communications have recently been issued on a number of topics related to steam generator tube integrity. Information notices were recently issued on experience with loose parts in steam generators, tube leakage due to a fabrication flaw in a replacement steam generator, and problems with computerized eddy current data analysis. A generic letter was recently issued on the interpretation of technical specification requirements in conjunction with Title 10 of the *Code of Federal Regulations*, Appendix B, pertaining to the inspection for cracks in the portion of the tube within the steam generator tubesheet. A draft generic letter was recently released for public comment on steam generator technical specifications and bending loads. Details regarding the aforementioned generic communications are provided below.

Generic Letter 2004-01, Requirements for Steam Generator Tube Inspections:
Generic Letter 2004-01 was issued August 30, 2004, to advise the industry of the NRC's interpretation of the technical specification requirements in conjunction with Title 10 of the *Code of Federal Regulations*, Part 50, Appendix B. The NRC's interpretation of these requirements as they pertain to steam generator tube inspections is that plants must use probes capable of detecting the forms of degradation that may exist along the length of tube required to be inspected by the technical specifications. In the event that a plant does not want to use such probes to inspect certain portions of the tube, relief from the requirements can be granted by the NRC.

Information Notice 2004-10, Loose Parts in Steam Generators:
Information Notice 2004-10 was issued May 4, 2004, to inform the industry about loose parts found in steam generators. Loose parts are often introduced into steam generators from maintenance activities or degradation in primary- or secondary-system components. These loose parts may result in steam generator tube degradation (i.e., through mechanical interaction between the loose part and the tube or through introduction of chemical impurities into the steam generator) and in some cases lead to tube leakage. This information notice included several examples where loose parts had been identified in steam generators. For example, a piece of weld slag located on the top of the cold-leg tubesheet at Braidwood Unit 2, manufacturing fit-up bars on top of a preheater baffle plate at Braidwood Unit 2, and a guide tube support pin nut and a locking device found in the primary side of a steam generator at Wolf Creek. Loose parts and their locations are not limited to the previously discussed items and locations. The information notice discussed the importance of performing engineering evaluations in cases where the loose part cannot be retrieved to determine whether the part will impair tube integrity if it is left in service. The information notice also discussed procedures for precluding the introduction of loose parts into the primary- and secondary-system. These included maintenance operation tools and equipment accountability, cleanliness requirements, accountability procedures for components and parts removed from major components, and post-maintenance inspections.

Information Notice 2004-16, Tube Leakage due to a Fabrication Flaw in a Replacement Steam Generator: Information Notice 2004-16 was issued August 3, 2004, to inform the industry about the potential for steam generator tubes to be damaged during fabrication and packaging. The information notice discussed the small primary-to-secondary leak that was observed at Palo Verde Unit 2 during the first cycle of operation with their replacement steam generators. The plant was shut down when the leakage increased to 11 gallons per day in order to identify the source of the leak. The leaking tube was identified during a secondary-side pressure test. During the root cause analysis, the licensee fabricated a series of mock-up dents, reviewed manufacturing records for the steam generators, reviewed steam generator packing records, and reviewed the preservice examination results for the new steam generators. It was determined that one tube was discarded during the fabrication of the replacement steam generators since it was damaged by a packing screw. This finding contributed to the conclusion that the damage to the leaking tube occurred during the packing of the tubes into a shipping crate.

The information notice stressed the importance of monitoring the fabrication process including packing procedures for the tubes and the receipt inspections performed at the fabrication facility. In addition, the information notice stressed the importance of communicating non-conforming conditions observed during fabrication to the individuals responsible for the preservice examination so that these individuals can further ensure that such conditions do not exist in the steam generator.

Information Notice 2004-17, Loose Part Detection and Computerized Eddy Current Data Analysis in Steam Generators: Information Notice 2004-17 was issued August 25, 2004, to inform the industry about challenges associated with the detection of loose parts and recent experience related to applying computerized data screening algorithms in the evaluation of steam generator tube eddy current data. At Shearon Harris, three tubes were identified as being damaged by a loose part. One of these tubes had a through-wall flaw and was leaking during operation. The leaking tube was identified through a secondary side pressure test. Even with the information from the secondary-side pressure test indicating that a through-wall flaw existed in the tube, the standard bobbin coil analysis techniques could not readily detect the flaw. The masking of the flaw signal was attributed to the proximity of the flaw to the expansion transition which is located at the top of the tubesheet. While evaluating the sequence of events that led to the damage to these three tubes, it was determined that the computerized data screening algorithm used during the prior inspection used improper settings. These improper settings resulted in the computerized data screening skipping the evaluation of a small portion of tubing. As a result of these findings, the information notice stressed the possibility that tube damage from loose parts may not always be identified with standard bobbin coil analysis as a result of the presence of interfering signal. In addition, the information notice stressed the importance of properly setting computerized data screening parameters for eddy current data analysis.

Information Notice 2005-09, Indications in Thermally-treated Alloy 600 Steam Generator Tubes and Tube-To-Tubesheet Welds: Information Notice 2005-09 was issued April 7, 2005, to inform industry about recent operating experience with degradation in steam generator tubes and tube-to-tubesheet welds. Details regarding this information notice were provided in the previous section on thermally-treated Alloy 600 operating experience.

As a result of operating experience and research, the nuclear industry's understanding of steam generator tube degradation has improved. This improved understanding has led to changes in the design and operation of the steam generators. Given these improvements, the NRC embarked on an effort to update the requirements governing steam generator tube inspections. The status of this effort is discussed in the following section.

Status of New Steam Generator Technical Specifications

Given that improvements could be made to the existing requirements pertaining to steam generator tube integrity, the NRC staff embarked on an effort to improve its regulatory requirements. Currently, this effort is focused on improving the technical specifications. The technical specifications at many plants are modeled after a generic standard technical specification; however, each plant in the U.S. has its own unique technical specifications. Nonetheless, the steam generator portion of most plants' technical specifications was developed in the 1970s. As a result, these technical specifications do not reflect the current understanding of steam generator tube degradation and the improvements in steam generator design. In addition, these technical specifications have some unnecessary prescriptive attributes.

Recently, the NRC approved modifications to the steam generator portion of the technical specifications at six plants. These technical specifications are consistent with those developed under the Nuclear Energy Institute's 97-06 initiative. The six plants that have adopted the new steam generator portion of the technical specifications are Catawba Units 1 and 2, Farley Units 1 and 2, and South Texas Project Units 1 and 2. In addition, several other plants have requested NRC approval to use these new steam generator technical specifications (Arkansas Nuclear One Unit 1, Callaway Unit 1, and Salem Unit 1).

The new technical specifications are risk informed and performance based. In addition, they reflect the current understanding of tube degradation and the improvements incorporated into newer steam generators. In the specification, the goals of the tube integrity program are defined in terms of performance criteria. There are criteria associated with structural integrity, leakage during normal operation, and leakage during postulated accident conditions.

To facilitate the adoption of new technical specification requirements related to steam generator tube integrity, the industry's Technical Specification Task Force requested NRC approval of a generic revision to the steam generator portion of the technical specifications. In response to this request, the NRC staff published a draft generic safety evaluation on the industry's proposal for public comment in the Federal Register on March 2, 2005. The public comment period expired on April 1, 2005. After reviewing the public comments, the NRC plans to issue a "Notice of Availability" of this Technical Specification Task Force proposal to allow plants to adopt the new steam generator portion of the technical specifications under the consolidated line item improvement process. The consolidated line item improvement process streamlines the process for modifying plant-specific technical specifications.

In the meantime, on October 7, 2004, the staff also issued in the Federal Register a generic letter (Generic Letter 2004-xx, Steam Generator Tube Integrity and Associated Technical Specifications) for a 60 day comment period. If finalized, this generic letter will request licensees (1) to discuss the adequacy of their steam generator tube integrity program and their plans for modifying their technical specifications to ensure they are representative of their program and (2) to discuss how bending loads are assessed in their evaluations of tube integrity. The licensees that have adopted the new steam generator portion of the technical specifications will not be required to respond to the generic letter.

The public comment period on the draft generic letter expired on December 6, 2004, and the staff is currently reviewing and incorporating these comments into the draft generic letter. One of the comments requested that the NRC withhold issuing the generic letter for some time period after the "Notice of Availability" is issued for the Technical Specification Task Force proposal. This would permit plants to voluntarily modify their technical specifications; thereby, avoiding having to respond to the generic letter.
The staff anticipates issuing the "Notice of Availability" in May or June 2005.

Conclusion

As a result of tube degradation associated with original steam generator designs, many plants have replaced their steam generators. The operating experience associated with the newer steam generator designs has been favorable with only a few instances of cracking being reported. The NRC and the U.S. nuclear industry continue to improve their programs for managing steam generator tube integrity. In addition, the NRC and the industry are making advances in improving the regulatory requirements pertaining to tube inspections.

References

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2. Letter from M.E. Warner, FPL Energy Seabrook Station, to the NRC dated October 12, 2004, "Seabrook Station Steam Generator Inservice Inspection." (ML042940501)

3. Leter from NRC to J.L. Skolds, Exelon Nuclear dated January 15, 2004, "Summary of Conference Call with Exelon Nuclear Regarding the 2003 Steam Generator Inspections at Braidwood Unit 2 (TAC NO. MC1367)." (ML033580377)

4. Letter from T.P. Joyce, Exelon Nuclear, to the NRC dated February 12, 2004, "Braidwood Station, Unit 2 Tenth Refueling Outage Steam Generator Inservice Inspection Summary Report." (ML040540452)

5. U.S. Nuclear Regulatory Commission Information Notice 2005-09, *Indications in Thermally Treated Alloy 600 Steam Generator Tubes and Tube-to-Tubesheet Welds*, April 7, 2005. (ML050530400)

6. “Summary of Conference Call with Vogtle Unit 1 regarding their 2005 Steam Generator Tube Inspections.” (ML051020152)

7. U.S. Nuclear Regulatory Commission Information Notice 2004-10, *Loose Parts in Steam Generators*, May 4, 2004. (ML041170480).

8. U.S. Nuclear Regulatory Commission Information Notice 2004-16, *Tube Leakage due to a Fabrication Flaw in a Replacement Steam Generator*, August 3, 2004. (ML041460357)

9. U.S. Nuclear Regulatory Commission Information Notice 2004-17, *Loose Part Detection and Computerized Eddy Current Data Analysis in Steam Generators*, August 25, 2004. (ML042180094)

10. U.S. Nuclear Regulatory Commission Generic Letter 2004-01, *Requirements for Steam Generator Tube Inspections*, August 30, 2004. (ML042370766)