History of Westinghouse Model 44 Steam Generators

[Indian Point 2 Event | Nuclear Reactors | NRC Home Page]

Plants	Commercial Operation Dates	Replacement Dates
Ginna	07/1970	06/1996
Point Beach 1	12/1970	03/1983
H. B. Robinson	03/1971	10/1984
Point Beach 2	10/1972	12/1996
Turkey Point 3	12/1972	04/1982
Turkey Point 4	09/1973	05/1983
Indian Point 2	08/1974	N/A
Indian Point 3	08/1976	06/1989

Degradation Mechanisms in Westinghouse Model 44 Steam Generators

Model 44 Steam Generators (SGs) use alloy 600 material for tubing. The tubes are supported vertically by the thick tube sheet at the bottom of the steam generator and laterally by drilled-hole tube support plates which are made of carbon steel. Corrosive products (typically magnetite) tend to accumulate in the crevice between tube and tube support plates because of the drilled-hole configuration and the susceptibility of the carbon steel to corrosion. The corrosive products have caused tube denting which resulted in axially and circumferentially oriented stress corrosion cracking in tubes. This degradation mechanism was the predominant reason for tube plugging in model 44 SGs.

The tubes in Model 44 SGs were mechanically rolled for only a few inches into the tubesheet starting at the primary (lower) face of the tubesheet. As a result, a crevice exists between the tube and the tubesheet for the portion of the tube within the tubesheet that is not expanded. Since the tube was mechanically rolled into the tubesheet, the expansion transition is commonly referred to as a roll transition. Plants with SG tubes that are only roll expanded for a small distance within the tubesheet are referred to as partial-depth hardroll plants. The crevice region has been a site for corrosion. Degradation in the crevice and/or at the roll transition has led some plants to re-roll the tubes above the original rolled region to provide a new pressure boundary free of any detectable defects. By so doing, the licensees have been able to apply alternate repair criteria, such as the F-star criterion, in which tubes with degradation in a certain specified distance below the top of the tubesheet or the bottom of the roll transition (whichever is lower), may remain in service. Tubes accepted for continued service on the basis of the F-star criterion have structural and leakage integrity consistent with regulatory criteria.

In general, plants with SGs that have only been partially expanded into the tubesheet have exhibited very little circumferentially oriented degradation either at the expansion transition or at the top of the tubesheet.

DB 24

Most of these plants have exhibited axially oriented primary water stress corrosion cracking at the roll transition, axially oriented outside diameter stress corrosion cracking in the tubesheet crevice region, and/or intergranular attack in the tubesheet crevice region.

The small u-bend regions of the tubes have exhibited axial and circumferential cracking. The degradation is caused by excess ovality during tube fabrication process, residual stresses, and tube denting at the top tube support plate. However, some plants had preventively plugged the row of tubes with the smallest u-bends (innermost row).

There were wastage problems in early operation of model 44 SGs (see tube rupture events below). With improved secondary side chemistry, the tube wastage problem was essentially eliminated.

[**Indian Point 2 Event** | **Nuclear Reactors** | **NRC Home Page**]

Steam Generator Tube Operational Experience

[Indian Point 2 Event | Nuclear Reactors | NRC Home Page]

Early, commercial nuclear power plant steam generator tubes were formed from Alloy 600, a corrosion-resistant high-nickel alloy. Early supplies of the Alloy 600 tubes were provided in a mill-anneal condition. Most of the steam generator tubes which have required plugging over the years have been mill-annealed Alloy 600 tubes. Information on the operational experience with Alloy 600 tubes can be found in the following NRC Information Notices:

- IN 98-27, "Steam Generator Tube End Cracking"
- IN 97-88, "Experiences During Recent Steam Generator Inspections"
- IN 97-49, "B&W Once-Through Steam Generator Tube Inspection Findings"
- IN 97-26, "Degradation in Small-Radius U-Bend Regions of Steam Generator Tubes"
- IN 96-38, "Results of Steam Generator Tube Examinations"
- IN 94-62, "Operational Experience on Steam Generator Tube Leaks and Tube Ruptures"
- IN 90-49, "Stress Corrosion Cracking in PWR Steam Generator Tubes"

Subsequently, steam generator tube manufacturers determined that by thermally treating the Alloy 600 material for later tubes, the corrosion resistance of the tubes was improved. Industry experience with thermally treated Alloy 600 steam generator tubes suggests that these tubes are more resistant to in-service cracking than mill-annealed Alloy 600 tubes. A few indications of degradation in thermally treated Alloy 600 tubes have been reported, however, and one is discussed in IN 97-26, listed above.

Now, most steam generator tubes for new steam generators are being fabricated from thermally treatec Alloy 690, an alloy with a higher amount of chromium and reduced amount of nickel. Alloy 690 is considered to be very resistant to the types of degradation experienced by the Alloy 600 tubes. At this time, the NRC staff is not aware of any primary or secondary side cracking in Alloy 690 tubing.

Some tubes have also been plugged due to fretting and denting, degradation mechanisms due to the design of the support plates and antivibration bars and the presence of loose parts, rather than the tubing material.

In the 20 years preceding the failure at Indian Point 2 on February 15, 2000, there had been seven steam generator tube ruptures in commercial nuclear power plants, all in tubes fabricated from mill-annealed Alloy 600. The last tube rupture occurred in 1993 at Palo Verde, Unit 2. It is considered a tube rupture rather than merely tube leakage when tube leakage of reactor coolant exceeds the makeup capacity. Sec Table 1 for information about the seven tube ruptures.

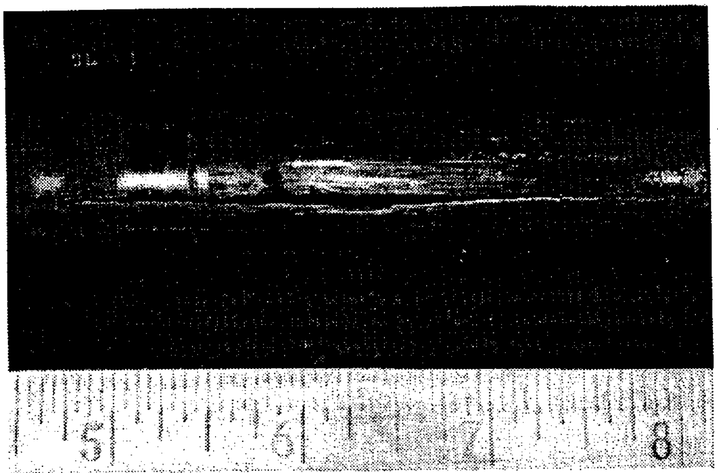

Figure 1. Example steam generator tube rupture at McGuire, Unit 1

There are other cases where defects have grown through the thickness of the tube wall, and caused the tube to leak. In these cases, either the plant was shut down due to the tube leakage, or the through-wall defect was detected as a result of the tube inspection during a plant outage.

Table 1. Steam Generator Tube Ruptures (from NUREG/CR-6365)

Date	Plant, SG Model	Maximum Leak Rate GPM	Degradation Mechanism	Rupture Size	Rupture Location	Stressors and Contributing Factors
2/26/75	Point Beach-1, W-44	125	Wastage	2 adjacent ruptured bulges each about 20 mm long and wide	Slightly above the tubesheet, outer row on the hot leg side	Large sludge pile, ineffective cleaning
9/15/76	Surry-2, W-51	330	Primary water stress corrosion cracking	114.3 mm long axial crack	Top of U-bend in Row 1, Column 7	High stresses and ovalization caused by inward movement of the legs due to support plate deformation
10/2/79	Prairie Island-1, W-51	336	Loose parts wear	38 mm long axial fishmouth opening	Tube bundle outer surface, 76 mm above the tubesheet on the hot leg side, Row 4, Column 1	sludge lancing equipment left in the steam generator
1/25/82	Ginna, W-44	760	Loose parts wear, fretting	100 mm long axial fishmouth opening	127 mm above the tubesheet on the hot leg side, Row 42, Column 55 (third row in from the bundle periphery)	Loose parts (baffle plate debris) left in the steam generator, wear of peripheral tubes, fretting of inner tubes
7/15/87	North Anna-1, W-51	637	High-cycle fatigue	360 circumferential break	Top of the 7^{th} upper tube support plate on the cold leg side, Row 9, Column 51	High-cycle vibration, denting, lack of anti-vibration bar support
3/7/89	McGuire-1, W-D2	500	Outer diameter stress corrosion cracking	95 mm long axial crack in a 645 mm long groove, 9.5 mm wide at the maximum point	711 mm above the tubesheet at the lower tube support plate on the cold leg side, Row 18, Column 25	Long shallow groove, possibly a contaminant
3/14/93	Palo Verde-2, CE-80	240	Outer diameter stress corrosion cracking	65 mm long axial fishmouth opening in a 250 mm long axial crack	Freespan region between the 08H and 09H tube support structures on the hot	Tube-to-tube crevice formation, bridging deposits, caustic secondary water

					leg sid, Row 117, Column 144	chemistry, susceptible material

[Indian Point 2 Event | Nuclear Reactors | NRC Home Page]