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STEAM GENERATOR OPERATION AND MAINTENANCE

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INTRODUCTION

Corrosion of steam generator tube has resulted in the need for extensive repair and replacement of steam generators. Over the past two decades, steam generator problems in the United States were viewed to be one of the most significant contributor to lost generation in operating PWR plants. When the SGOG-I(Steam Generator Owners Groups) was formed in early 1977, denting was responsible for almost 90% of the tube plugging. By the end of 1982, this figure was reduced to less than 2%. During the existence of SGOG-II(from 1982 to 1986), IGA/SCC(Intergranular Attack/Stress Corrosion Cracking) in the tube sheet, primary side SCC, pitting, and fretting surfaced as the primary causes of tube degradation. Although significant process has been made with wastage and denting, the utilities experience shows that the percentage of reactors plugging tubes and the percenage of tubes being plugged each year has remained relatively constant. The diversity of the damage mechanisms means that no one solution is likely to resolve all problems. The task of maintaining steam generator integrity continues to be formidable and challenging.

As the older problems were brought under control, many new problems emerged. SGOG-II(Steam Generator Owners Group program from 1982 to 1986) has focused on these problem areas such as tube stress corrosion cracking(SCC) and intergranular attack(IGA) in the open tube sheet crevice , primary side tube cracking, pitting in the lower span, and tube fretting in preheated section and anti-vibration bar(AVB) locations.

Primary Water Stress Corrosion Cracking(PWSCC) in the tube to tubesheet roll transition has been a wide spread problem in the Recirculation Steam Generators(RSG) during this period. Although significant progress has been made in resolving this problem, considerable work still remains. One typical problem in the Once Through Steam Generator(OTSG) was the tube support plate broached hole fouling which affects the OTSG steam generating capacity due to excessive pressure drop across the tube support plates. OTSG owners group has developed both mechanical and chemical cleaning process and an upgraded secondary water chemistry in resolving these problems. The OTSG performance has been greatly improved since OTSG plants implemented chemical cleaning and morpholine water chemistry.

The SGOG project officially ended December 31, 1986. A six year Steam Generator Reliability Program(SGRP) under the EPRI base program began January 1, 1987. SGRP continued to address the generic steam generator problems facing nuclear utilities. In order to develop appropriate strategies to cope with the tube degradation problems, SGRP has performed the statistical evaluations to model the progression of damage mechanism aimed at accurate prediction of the defect growth rate of various mechanisms such that long term trends can be developed. Analysis of the behavior of group of plants indicate that insights on the potential behavior of a specific plant may be developed from the observed behavior at other plants.

SGRP has provided utilities with tube inservice inspection guidelines(ISI Guideline) including ISI Performance Demonstration program to help utilities to improve tube inspection accuracy and sensitivity. SGRP has also updated the Secondary Chemistry Guidelines(Rev.3) and worked on the advanced amine application guidelines to better protect the steam generator tube from corrosion.

SGRP has promoted Steam Generator Degradation Specific Mnangement(SGDSM) program in the past two years. The USNRC indicates support of the owners group generic approach. The objective is by leaving defective tubes in service with acceptable small amplitube eddy current indication(s) within the tube support plate or the tube sheet. Some European utilities have implemented these alternate tube plugging critera in a much aggressive manner. The industry has identified the "lead plants" and will submit the generic alternate repair criteria(ARC) for USNRC approval. The success of the ARC depends on utilities ability to demonstrate the eddy current inspection performance quality and the reliability and repeatability of the on line leakrate monitoring technique.

The steam generator owners group program has been restructured and become Steam Generator Management Project(SGMP) since March, 1993. Approximately half of the estimated expense will be contributed from EPRI's base program and the remaining from member co-funding and Tailored Collaboration(TC) contributions. SGMP-I will focus on pro-active projects such as corrosion inbihitors, advanced water chemistry, sludge and fouling control, tube ISI improvement, thermal hydraulic code enhancement, advanced leak detection methods, and specific damage-form detection and prevention. The pro-active program will maintain the tube integrity to assure reliable operation of the steam generators for the design life of the plant, while the alternate tube repair criteria are only buying times for the defective steam generators. The pro-active approach is the optimum way to manage the Steam Generators from both an economical and regulatory standpoint.

The history of SGOG/SGRP work from 1977 through 1992 is an indicator of the steam generator problems and performance during this period.

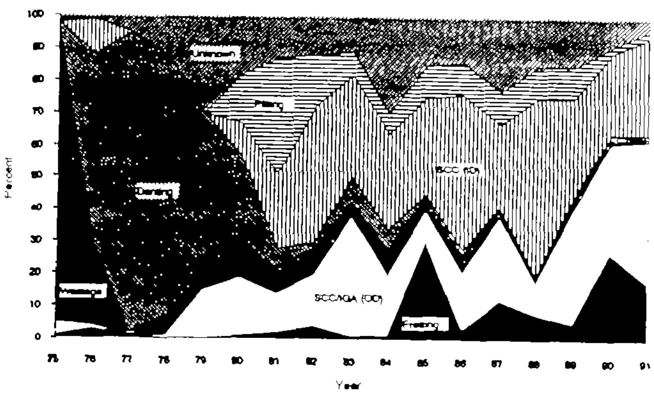
To maintain steam generator integrity is a challenging and expensive task; however, it is manageable. Each nuclear utility should have their steam generator strategic management program in place at the begining of plant life in order to avoid the pre-matured permanent plant shutdown or the costly steam generators replacement.

CURRENT STATUS OF STEAM GENERATOR OPERATION AND PERFORMANCE

Figure 1 shows the causes of steam generator plugging from 1973 to 1991. In the early years denting was responsible for almost 90% of the tube plugging. A significant progress has been made to combat denting by improving the secondary water chemistry in the RSG having drilled tube support plates. In recent years, denting has been decreased to less than 2% as shown in Figure 1. From 1982 to 1986, primary side IGA/SCC, pitting, PWSCC(Primary Water SCC) in the tube sheet, pitting, and fretting surfaced as the primary causes of steam generator tube degradation. This trend continued from 1987 to 1991.

Figure 1

United States Causes of Steam Generator Plugging



Last year, the small amplitude "signal/ noise" indications became the center stage of steam generator tube problem. The Torjan problem and Palo Verde incident brought new concerns to the nuclear industry. Tube Intergranular Ataack/Stress Corrosion Cracking(IGA/SCC) at the tube support plate, in the tubesheet crevice, and in the sludge piles or under the deposit layer will be the dominant issues facing the nuclear utilities in the coming years.

A number of PWRs in the United States, as well as in Japan and Germany, have experienced reductions in steam generator pressure, which has been severe enough to result in decreased electrical output. At a number of plants the trend of steam generator pressure over time has been characterized by step change decreases in pressure, typically associated with a reactor trip or shutdown followed by a gradual partial recovery of pressure. This also happened to the Once Through Steam Generator in that the downcomer level increases due to tube bundle fouling and results in decreased unit power output.

Approximately one-third of the PWRs are experiencing a degradation of steam generator performance. The preheater-type steam generators tend to expeience degradation more quickly than the nonpreheater types. The fouling factor calculations that were performed by one of EPRI contractor suggest that a fouling factor of 1.5 x 10E-04 h sq.ft. F/Btu could result in a decrease in steam generator pressure of approximately 75 psi. This is comparable to the magnitude of the pressure losses being experienced by operating units. Data supplied by Mishubishi Heavy Industry(MHI) provides a correlation between fouling factor and scale thickness. This correlation suggests that a deposit with a thickness of 8 mils would results in a fouling factor of 1.5 x 10E-04 h sq.ft. F/Btu.

The performance and reliability of steam generators can be assessed by the statistical data from the experience of tube plugging, lost of generation, unit forced outages, and the thermal hycraulic performance.

(1) TUBE PLUGGING

During the last 10 years the average percentage of reactors plugging steam generator tubes has been 45%. The trend has not changed significantly. In 1991, 50% of the operating units were required to plug steam generator tubes. The number of tubes in service has risen from less than 200,000 to more than 2.9 million in 1991. The percentage of tubes plugged per year over the last 10 years has averaged 0.23 percent. This trend has decreased from an average of 0.39% from 1971 through 1979, to 0.23% from 1982 through 1991. However, steam generator tube degradation is still widely occurring, and utilities are required to take expensive remedial action to maintain the integrity of the operating steam generators.

(2) LOST GENERATION

During the last ten years, the capacity factor loss for U.S. PWR's due to steam generator problems has averaged 3.4%. In 1991 the capacity factor loss was 2.4%. The average capacity factor of all U.S. PWR units in 1991 was a record high of 71.8%. The capacity factors of OTSG units were greater than 80% in 1992, with two plants approaching 100%.

A number of U.S. and non-U.S. nuclear power plants have experienced a gradual degradation in steam generator performance and reported a loss in reduction of power output ranging from 3 to 8 megawatts. The step changes in steam pressure which have been observed in these units may be the results of partial exfoliation of tube OD deposits. These step changes in pressure are typically associated with reactor trips or shutdowns. Each exfoliation event(cooldown or thermal transient) produces a step change reduction in steam pressure, and then it is followed by a gradually , but incomplete recovery of lost pressure during subsequent operation.

(3) FORCED OUTAGES

Forced outages not only cause a loss of production, but also are expensive. In 1991 there were four forced outages in the U.S. due to steam generator tube leaks. This compares with five forced outages in the U.S. in 1990 and an average of eleven forced outages per year from 1976 to 1991. The statistic shows an improving trend, the long range goal of utilities is to have no forced outage due to steam generator tube leak. Most of tube failures in the RSGs occur at the upper tube bundle near AVB, U-Bend, or top support palte. Tube failure in the OTSG units usually occurs in the top span near the upper tubesheet or the 15th tube support plate, due to high cycle fatigue. A small tube imperfection can propagate to a tube rupture in a relatively short duration without any precursory indications. Figures 2 and 3 illustrate the characteristic of primary to secondary leakrate during tube rupture or tube failure events of different kinds happened in recent years. A tube rupture event is a tube failure when the leakrate exceeding the makeup or charging pump capacity. Early detection of a tube leak and timely shutdown of the unit will prevent the undesirable tube rupture event.

Figurre 2

MIHAMA-2 AND PALO VERDE SGTR EVENTS

P/S LEAKRATE PRIOR AND DURING SGTR

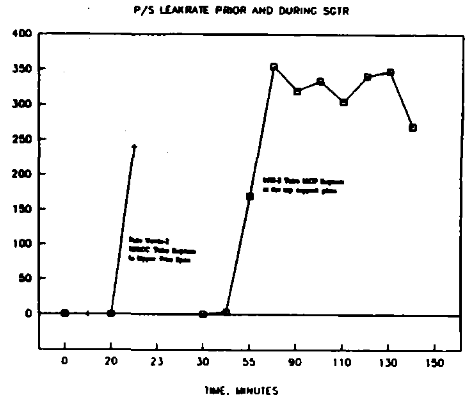
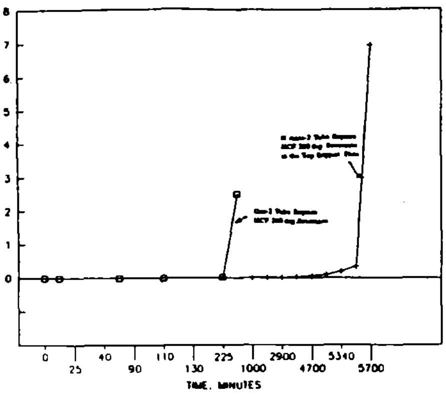


Figure 3
TUBE RUPTURE P/S LEAKRATE
North Anna-2 and Oconee-2 SGTR Events



(4) THERMAL HYDRAULIC PERFORMANCE

Steam generator operation and performance problems can be catagorized into two groups, namely tube integrity and thermal hydraulic performance. Degraded thermal hydraulic performance has been mainly due to secondary side fouling of the steam generators with deposits built up on the tube outside surface or in the tube support plates. The effect on the RSGs is decreasing steam generator outlet pressure and lost of generated power. The secondary side fouling in the OTSGs will result in an increase of the OTSG downcomer level and reduction of unit output.

(a) Once Through Steam Generator(OTSG)

Thermal Hydraulic performance of the OTSGs would be significantly affected by tube bundle fouling due to corrosion product transport to the steam generator from the balance of the plant. When the level encroaches the feedwater nozzles located at the top of the downcomer, the heating steam is blocked and the power has to be reduced in order to bring downcomer level to below the maximum allowable limit.

All of the OTSG units which encountered severe tube support plate fouling have executed partially tube bundle chemical cleaning and changed to morpholine water chemistry. Chemical cleaning has been very effective by removing all of the deposits from the broached holes of the support plates and restored the tube bundle cleanliness to a like new condition. Morpholine all volatile water treatment minimizes the iron transport to the steam generator, thus reduced the fouling rate of the tube bundle. In a moderately fouled OTSG, morpholine chemistry has been able to retard the fouling in the tube bundle, thus postponed the need for chemical cleaning. Those OTSGs which have been chemically cleaned and changed to morpholine chemistry were operating at a like new condition with a controlled fouling rate of less than 4% per year.

Most of the OTSGs have experienced thermal hydraulic instability at partial load operation. The instability range were from 60% to 70% of full power. When the unit is operated in this range, the OTSG pressure, downcomer level and reactor coolant temperatures fluctuate, which resulted in an unstable operating situation. The downcomer orifice opening and system static and dynamic unbalance are believed to be the causes of the problem.

(b) CE Steam Generator System 80

EPRI study has identified an area in the CE System 80 steam generators having very high void fraction and high velocity field, which has led to serious under deposit corrosion and resulted in a fish-mouth tube rupture. EPRI's thermal hydraulic analysis has determined the conditions in the upper hot log region conductive to fouling and sludge deposition. The transient analysis was also be able to evaluate the effect of downcomer oscillaions of Parlo Verde-2 steam generators which had been observed for a few months prior to March 1993 tube rupture. Initial analysis indicated that high velocity regions in the upper hot leg side as well as high void fraction of approximately 90%, which may create a re-circulation zone in the upper tube bundle region and result in heavy deposit buildup with the detrimental sulfate enviroment. The undesirable thermal hydraulic performance may indirectly result in under deposit IGSCC and tube fatigue or wear failure. Parlo Verde will pull tubes to determine the root cause of tube rupture and may perform 100% eddy current inspections of tubes in two other C-E plants on the same site.

ATHOS3 thermal hydraulic code can also be used to perform flow field calculation for tube fatigue and wear analysis. The results have been verified with the actual steam generator performance. This Code has been used to evaluate effects of different blowdown rates on void fraction and velocity field in the tube sheet region in one type of Westinghouse recirculation steam generator as well as for C-E System 80.

(c) Westinghouse Design Steam Generators

There were ocasionally tube failure or rupture at the top support plate when the anti-vibration bar(AVB) in the region was not propoerly installed. Japanese study concluded that the causes of tube rupture and damage to the adjacent tubes at Mihama-2 were that these tubes experienced fatigue break or wear thinning as fluid elastic vibration occurred at the U-bend section of the tubes and the vibration amplitude had not been restricted, because of the significant lack of insertion of AVBs. Inspection after the tube rupture event, revealed that high cycle fretting fatigue caused crack at the No.6 tube support plate, while the crevice of the tube to No.6 support plate had been filled with iron oxide(a fixed support condition). The flow elastic vibration analysis showed that the maximum stability ratio of the broken tube with the dispersion range considered (with impropered AVB insertion) was more than 1. Therefore, there was a potential for fluid elastic vibration to occur under this condition. Based on utilities operating experience, most of the tube failures at the top support plate and AVB region were due to high cycle fatigue caused by fluid elastic vibration.

(d) OTSG Lane and Wedge Tube Failure

Corrosion assisted high cycle fatigue(HCF) has been the damage mechanism responsible for leaking tubes in OTSG's, and most of the leaker outages were caused by lane region leakers. The lane and wedge tubes are located in a relatively high cross steam flow area and potentially high moisture and contaminant carry-over zone. B&W has recommended that tubes located in these potential fatigue failure area be preventively sleeved to the top tube support plate. Sleeve tubes will have better fluid elastic instability margin and improved HCF strength. Since the OTSGs utitilites implemented the suggested preventive measures and improved the secondary water chemistry starting 1986, tube fatigue failure in the Once Through Steam Generator has deminished.

MAJOR SPECIFIC DEFECT MECHANISM

The following are the major specific tube degradation mechanism , which may imply alternative tube repair criteria:

(1) Primary Water Stress Corrosion Cracking(PWSCC- P*/F*)
(2) Secondary Side IGA/SCC at TSPs or TS Crevice
(3) Sludge Pile IGA/SCC

One effective method of holding down IGA/ SCC due to electrochemical corrosion potential is to add an oxygen scanvenger to the feedwater. Since IGA/SCC accelerates in the oxidizing enviroment, the reducing condition produced by hydrazine or other advanced chemicals decelerates these form of corrosion.

Boric acid on the other hand, can reduce plugging or sleeving rates in plants with IGA/ SCC problems. Boric acid appears to be especially helpful in plants with copper alloys in the balance of plant. For AVT chemistry plants, the volatile ammonia goes in the steam phase in two-phase flow regions, reducing its effectiveness in wet-steam areas; therefore, increase corrosion product transport into the steam generators. Now, some plants have switched over to morpholine, which is less volatile than hydrazine. EPRI research has identified other amines that have more favorable properties than morpholine, which are currently under tests in three U.S. PWR plants. Prelinimary results indicated a higher pH from a smaller amount of an advanced amine, reduced iron input, and a small increase in cation conductivity. The result is promissing.

Chemical additives to the caustic enviroment can change IGSCC behavior markedly. The rate of intergranular crack growth in alloy 600 in a caustic enviroment depends on the electrochemical potential. The presence of sulfate in a caustic enviroment increase the SCC crack growth rate, but does not alter the SCC potential range. Titanium oxide, zinc phosphate, cerium chloride,and zinc oxide increase the resistance to SCC by increasing the potential, at which there is onset of SCC, from approximately 150 mV to 200 mV. EPRI's pilot test program is currently on going. Titania-Silica sol gel and Defussa Anatase formulated by Rockwell is being tested in two U.S. PWR plants and one Belgium plant. Model boiler tests will continue for at least 2 years with shutdowns and tube NDE examinations, every 4-6 months. Cerium acetate plant test is planned for October 1993 under EPRI steam generator management program.

STEAM GENERATOR MAINTENANCE

(1) Steam Generator Tube Plug Integrity

There are many types of plugs being used in the steam generators to remove defective tubes from service, namely: Mechanical Expanded Plug(Ribbed Plug), Rolled Plug, Explosive Welded Plug, Manually Welded Plug, Remote Welded Plug, and Tapered Plug. In the 70s and 80s, thermally treated Alloy 600 material were used for all types of plugs installed in the steam generators. During routine maintenance operations, several utilities reported observing water dripping from steam generator tube ends plugged with Westinghouse mechanical plugs in the fall of 1988. Plants using B&W rolled or ribbed plugs also reported detection of stress corrosion cracking in the same time frame.

The time characteristic for the initiation of PWSCC is a function of residual stress, plug operating temperature, and material microstructure. Typically, for mechanically expanded plugs, when cracking occurs it is mostly axially oriented or; if circumferentially oriented, the cracking has been limited in axial extent in the plug shell when it progresses through wall. For rolled plugs, most of the indications were found in the heel transition region. The indications appeared to be single and multiple cracks in both the circuferential and axial direction.

Beginning in 1983, with increased cracking in mill annealed(MA) Alloy 600, more detailed studies were undertaken to arrive at a clearer understanding of the factors which contribute to PWSCC susceptibility. It is also found that microstructures characterized predominantly by intergranular carbides display greater PWSCC resistance than those with less intergranular and more intragranular carbide distribution. The relative grain carbide distribution depends directly on the carbon content,the grain size, the final mill annealed temperature and the subsequent thermal treatment of the final product.

Westinghouse has performed a series of corrosion tests in late 1988, and a micro-structural classification of the grain boundary carbide distribution for a number of plugs undergone tests were categorized. These studies consistently indicated that plugs with continuous grain boundary carbide had a greater resistance to PWSCC than those with semicontinuous grain boundary carbides which were in turn generally more resistance to PWSCC than plugs with discontinuous grain boundary carbides.

Since the detection of Alloy 600 plug problem in late 80s, both B&W and Westinghouse have expeditly developed Alloy 690TT plugs to replace the defective Alloy 600 plugs. Based on a review of corrosion test results performed by EPRI, B&W and others, it can be concluded that Alloy 690 displays superior corrosion resistance than Alloy 600. No indication of intergranular stress corrosion cracking has been observed in Alloy 690 samples that have been subjected to various corrosion tests as of to date.

(2) STEAM GENERATOR TUBE SLEEVING

Tube sleeving is an alternate tube repair method to keep the defective tubes in service and preserve the capacity of the steam generator when massive repair is required. The "killer" tube degradation mechanism in RSG is IGA/SCC in tubesheet or support plate crevices or at roll transitions. The most frequent tube failure in the Once Through Steam Generator(OTSG) has been high cycle fatigue failure in the top span of the tube bundle located in the lane and wedge area. The sleeve keeps tubes in service and eliminates the thermal degradation that would occur if the tubes were plugged. Preventive sleeving of the OTSG lane and wedge tubes in the potential fatigue failure area will eliminate costly and unscheduled tube leaker outages and prevent tube rupture due to corrosion assisted high cycle fatigue. There are various sleeve designs offered by different vendors. The following are four bacic types of sleeve being used in the nuclear steam generators:

(a) Hybrid Expansion Joint Sleeve
(b) Mechanical Rolled Joint Sleeve
(c) TIG Welded Sleeve
(d) Laser Welded Sleeve

The sleeve material shall be made from Alloy 690 in accordance with ASME SB-163, Code Case N-474-1, and with EPRI specification for Alloy 690 tubing material. After the final tube reduction, all tubing shall be thermally treated in accordance with EPRI specification. Metallographic evaluation shall be performed in each lot of tubing after final mill annealing and thermal treatment. It shall be evaluated to verify that the desired carbide microstructure has been achieved. The grain size shall be ASTM 5-9 per ASTM E-112. All tubing shall be UT and eddy current examined. A moderate S/N ratio of less than 7 is preferable in order to enhance the inspection sensitibity, which is especially important when a small amplitude imperfection existed in the eddy current low sensitivity region. Meeting these material requirements will ensure that the installed sleeve will last for the design life.

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