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Research Perspectives on the Evaluation of Steam Generator Tube Integrity

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Abstract

Industry efforts have been largely successful in managing degradation of steam generator tubes due to wastage, pitting, and denting, but fretting, SCC and intergranular attack have proved more difficult to manage. Although steam generator replacements are proceeding there is substantial industry interest in operating with degraded steam generators, and significant numbers of plants will continue to do so. In most cases degradation of steam generator tubing by stress corrosion cracking is still managed by "plug or repair on detection," because current NDE techniques for characterization of flaws are not accurate enough to permit continued operation. This paper reviews some of the historical background that underlies current steam generator degradation management strategies and outlines some of the additional research that must be done to provide more effective management of degradation in current generators and provide greater assurance of satisfactory performance in replacement steam generators.

1. Introduction

Steam generators have historically been among the most troublesome of the major components in commercial pressurized water reactor (PWR) nuclear power plants around the world. They have been described as the "weak link"[1] in the PWR design, and their premature deterioration has been characterized as "one of the most persistent challenges facing utilities with pressurized water reactors."[2] For the past decade or more, steam generator problems in the United States have ranked only behind refueling outages as the most significant contributor to lost power generation.[3] Beyond the reliability and economic issues facing utilities, steam generator problems raise potentially significant regulatory issues within the U.S. Nuclear Regulatory Commission.

The magnitude of the steam generator tube degradation problem is illustrated by the fact that, to date, various forms of degradation have resulted in the plugging of more than 100,000 tubes worldwide. In 1998, the last year for which complete data are available, 45% of 230 operating PWRs in the EPRI survey of steam generator performance were required to plug tubes. In addition, 150 steam generators in 51 PWRs around the world had been replaced because of severe tube degradation, including 68 steam generators in 22 U.S. plants.[3] Replacements are continuing in both the U.S. and abroad.

The present paper discusses research perspectives related to steam generator tube degradation, with emphasis on problems associated with stress corrosion cracking (SCC). Such SCC can be either axial and circumferential in orientation and can occur at various locations in steam generators, initiating at both the inner and outer surfaces of the tubes. The history of steam generator tube degradation is briefly reviewed, and the evolution of technologies for the nondestructive examination of steam generator tubes is summarized. Also considered are the effects of flaws and flaw morphologies on the structural and leak integrity of steam generator tubing, and the difficulties in detecting, characterizing, and analyzing the structural effects associated with the more complex crack geometries observed in recent years is discussed. The potential for tube failure under severe-accident conditions is also considered.

2. Corrosion Degradation of Steam Generator Tubes

Corrosion problems have afflicted steam generators from the very introduction of the PWR technology. Shippingport, the first commercial PWR operated in the United States, developed leaking cracks in two Type 304 stainless steel (SS) steam generator tubes as early as 1957, after only 150 h of full-power operation.[4-6] The cracks were attributed to stress corrosion cracking (SCC) produced by free caustic in the secondary water and steam blanketing of the tubes at the top inlet portion of the steam generator, leading to concentration of the caustic. The leaking tubes were plugged, and the use of a modified sodium phosphate water chemistry was instituted to control secondary water pH.

Because austenitic SS steam generator tubes were found to be susceptible to SCC from both chlorides and free caustic, the decision was made in the late 1960s to instead use Alloy 600 tubes in the United States and most of Europe and Alloy 800 tubes in Germany.[1] The decision to use Alloy 600 was made on the basis of its known high resistance to chloride attack, based largely on petrochemical plant experience. However, it ignored the work of Coriou et al.[7,8] which showed as long ago as 1959 that this and similar nickel-base alloys were subject to stress corrosion cracking in deionized water at 300-350°C. Alloy 600 steam generator tubes have undergone a series of successive failure modes since that time.

Because Alloy 600 is subject to cracking at high caustic concentrations, early steam generators with Alloy 600 tubes used phosphate additions to the secondary water to provide a buffering capability. This was based on prior experience with fossil-fired boilers. However, rapid caustic cracking was observed in several early steam generators. This problem was successfully controlled by reducing the sodium-to-phosphate molar ratio, but severe tube wastage problems were almost immediately experienced. By the early-to-mid 1970's, wastage was by far the leading cause of tube plugging in the U.S.

In response to the tube wastage problem, most U.S. plants switched to all volatile water treatment (AVT) around 1974. In this approach, ammonia, morpholine or similar additions were added to control pH and hydrazine or similar additions were added for oxygen scavenging. Effective use of AVT water chemistry requires very high purity levels in the feedwater, since no buffering agents are present to prevent excessive acidic or caustic conditions in regions of impurity concentration. It should be noted that once-through steam generators have always used AVT water chemistry to avoid the deposition of chemical solids on tube surfaces in their boil-dry design.

The wide-spread change to AVT water chemistry resulted in a dramatic decrease in tube plugging due to wastage, but this problem was soon replaced by severe tube denting problems in many plants. Tube denting was eventually brought under control by improved controls on feedwater chemistry, improved condenser integrity to eliminate the inleakage of oxygen and other impurities, and, in some cases, the use of condensate polishers or boric acid additions.

Since about 1980, steam generator tube degradation in the U.S, and elsewhere has been dominated by stress corrosion cracking. Unlike wastage and denting, which occur exclusively on the secondary side (outer diameter) of the tubes, stress corrosion cracking can occur on either the primary or secondary side. Primary water stress corrosion cracking (PWSCC) is most likely to occur at regions of high residual stress, as at the tube expansion transition at and immediately above the tubesheet, at U-bends (particularly the small-radius U-bends on the inner-row tubes, and in tube regions deformed by secondary-side denting at the tube support plates.

A number of design changes have been implemented over the years to address the PWSCC issue. These include shot peening or rotopeening of the ID surfaces of the tubes in the roll transition zone to produce compressive residual stresses, improved processes for expanding the tubes into the tubesheet to reduce residual stresses in this region, and in-situ thermal treatment of U-bends on older plants and thermally treated U-bends in newer plants to reduce residual stresses in this region. All of these processes have proven at least somewhat beneficial, but PWSCC continues to be a problem in PWR steam generators.

Outer-diameter stress corrosion cracking (ODSCC) and intergranular attack (IGA) commonly occur in crevices or under corrosion product scales, where conditions are such that incomplete wetting by secondary water occurs, and the consequent alternate wetting and drying result in substantial local buildup of corrosive species. Such locations include the tube support plate crevice, the region near the top of the tube sheet, free span areas under corrosion products or deposits, and regions under sludge build-up. Calculations of local crevice chemistry predict concentration factors approaching 10^8 and crevice pH values ranging from < 2 to > 10 at operating temperatures, depending upon the impurity species in the secondary water. Again, remedial actions have been taken over the years to address this problem. However, ODSCC, like PWSCC, continues to be a leading cause of steam generator tube plugging and repair.

In the 1980s, PWSCC and ODSCC problems were almost entirely confined to low-temperature mill annealed (LTMA) tubing found in Westinghouse steam generators. However, starting about 1990, SCC problems also began to significantly affect the high-temperature mill annealed (HTMA) tubes used in the Combustion Engineering steam generators.[3,9,10] More recently, SCC is increasingly observed in Babcock & Wilcox steam generators, which use tubes that have been stress relieved (SR) tubes after a similar high-temperature 1065-1090°C mill anneal.[3]

Around 1980 for replacement units and in the mid-1980s for the new Model D-5 and Model F steam generators, Westinghouse began using thermally treated (TT) Alloy 600 tubing, which has a microstructure more resistant to SCC, and the oldest of these units (e.g., Surry 1 and 2) have operated for 20 years with virtually no SCC.[3] However, numerous SCC failures have been observed in Alloy 600 TT mechanical plugs, an effect attributed to certain susceptible heats of material,[11-17] and PWSCC of the Alloy 600 TT tubes in the Ulchin 1 and 2 steam generators in Korea has led to extensive tube sleeving, and replacement of these steam generators is now under consideration.[18-20]

Since about 1989, thermally treated Alloy 690 has been the tubing material of choice for replacement steam generators. After up to 11 years of service, no incidents of SCC have been reported for any of these tubes in the field. While laboratory studies have also been unable to produce SCC in Alloy 690 in primary water chemistries, numerous studies have demonstrated the ability to crack this alloy under conditions that approximate the water chemistries and impurities expected in steam generator crevices.[21-24]

3. Steam Generator Integrity

3.1 Nondestructive Evaluation

To be able to ensure the integrity of steam generator tubing, it is important to be able to detect and characterize the degradation. Up to the early 70s the inservice inspection of PWR steam generators was carried using single-frequency eddy current (EC) bobbin coils. This inspection technology was adequate for detection of volumetric degradations but not for cracks. Part of the problem was a low signal to noise ratio for cracks, and in the late 70's, two-frequency EC equipment was introduced to help reduce noise signals from probe wobble and the tube support plate.

By the mid-80s additional modes of degradation such as pitting and intergranular attack (IGA) had to be addressed. Pancake coils were introduced in the 80's to improve detection of IGA in the tube sheet crevice. In addition, three-frequency mixing of bobbin coil signals was introduced to improve flaw detection. Dodd and Deeds[25] showed how to eliminate the tube support plate (TSP) signal by using magnitude and phase in a least-square analysis of data at different frequencies. Steam generator inservice inspection guidelines (ISI) guidelines were introduced by EPRI in the 80's that included qualification requirements for techniques and analysts that focussed on performance with a requirement that the inspector demonstrate an 80% probability of detection (POD) for flaws > 60% throughwall rather than mere adherence to procedures.

By 1990, motorized rotating pancake coils (MRPC) with single or multiple probe heads and isometric displays of the eddy current response were being used to supplement bobbin coil inspections. The 90's saw extensive use of MRPC for better characterization of cracks in U-bends, TSP, and the roll transition zone (RTZ). In addition to the extensive use for supplementary inspections in locations susceptible to cracking, MRPC were used for primary examinations in some cases such as the detection of circumferential cracking. Differential MRPC designs like the +Point probe were introduce to provide improved signal to noise ratios in many cases. Despite improvements in detection capability, sizing, however, is still a problem in many cases.

3.2 Failure Models for Steam Generator Tubes

Extensive work by NRC[26] and industry[27-28] during 70's and 80's has developed and verified models for failures of flawed steam generator tubes under normal operating temperature (300°C) and pressures up to the failure of unflawed tubes (10,000–11,000 psi). Failure of steam generator tubes under such conditions is controlled by the flow stress of the material. A significant body of failure data on flawed steam generator tubes currently exists and has been the basis for the development of various flow stress models.

Most of this work focused on the potential for tube failure during design basis accidents like a main steam line break (MSLB). Risk assessment studies,[29] however, show that a significant portion of the risk due to steam generator tube failures is associated with tube failures due to severe accidents, during which tube temperatures can increase to 650–750°C. Under such conditions the strength of Alloy 600 decreases significantly as shown in Fig. 1, and creep becomes a potential failure mechanism for the tubes and the potential for increased leakage through flaws due to opening of existing throughwall flaws by creep deformation must be considered.

The research program at ANL has developed a data base, correlations and methodologies for predicting the failure of flawed and unflawed steam generator tubes under severe accident conditions.[30] It is well known[31] that high-temperature failure is controlled by thermal creep at low strain rates and by flow stress at high strain rates. In the most structurally challenging severe accidents, the coolant pressure remains high (e.g., close to the safety relief set point) whereas the temperature of the tubes rises at rates varying from 5–10°C/min. Tests have shown that under these conditions, tube failure is best described by a creep model.[30]

Such models can also be extended to consider the potential for the failure of repaired tubes under severe accident conditions. Analyses and tests have been performed to study the behavior of tubes repaired by the Electrosleeve™ process under severe accident conditions.[32]

Stress corrosion cracks on the primary side and due to high caustic concentrations on the secondary are mainly planar in nature. The cracks of primary interest currently, particularly on the secondary side, are segmented and have many ligaments between small segments of the crack. These cracks behave differently structurally from planar cracks. Tubes with these kinds of cracks exhibit higher burst pressures than one would predict using the correlations based on a planar bounding crack.

Mechanistic approaches to prediction of failure require the characterization of flaw geometries. In some cases it is possible to develop empirically based Alternate Repair Criteria (ARC) that do not explicitly consider flaw geometry. For example, in the case of ODSCC in Westinghouse steam generators with drilled-hole tube support plates, Generic Letter 95-05 provides repair criteria in terms of the peak bobbin coil voltage that do not explicitly address crack length or depth. Such empirical approaches presume that the crack geometries in the tubes used to develop the data base for the empirical correlation are representative of those actually encountered in reactor.

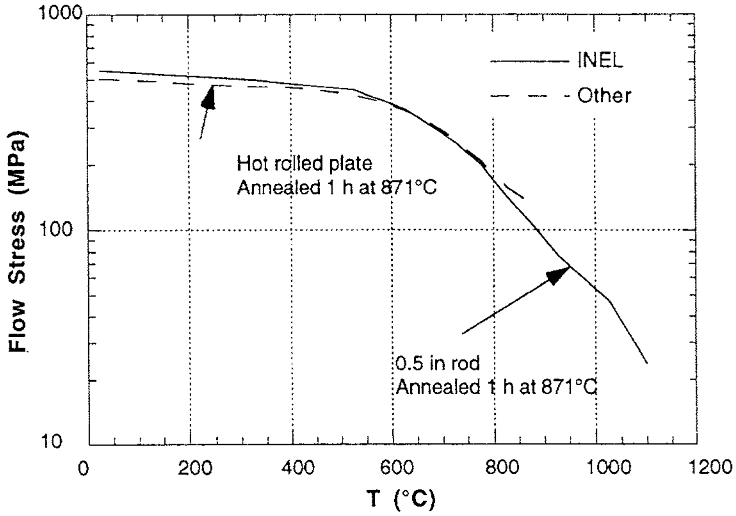


Figure 1.
At high temperatures the flow stress of Alloy 600 decreases markedly and creep effects become significant

4. Regulatory Guidance

Regulatory guidance for steam generator tube plugging and repair was developed in the 1970s (Reg. Guide 1.121, *Bases for Plugging Degraded PWR Steam Generator Tubes*). This guidance was based on deterministic depth based plugging criteria that did, however, attempt to account for degradation growth and NDE uncertainty. The specific estimates for degradation growth and NDE uncertainty in Reg. Guide 1.121 were based on engineering judgment. Because the uncertainties associated with the integrity assessments are strongly dependent on the specific mode of degradation, regulatory guidance has changed with time from deterministic depth-based plugging criteria that apply to all flaw types toward performance-based risk-informed degradation-specific plugging criteria. One example of such guidance is Generic Letter 95-05, which as noted previously, provides plugging and repair criteria for a specific degradation, ODSCC at tube support plates in Westinghouse steam generators with drilled-hole tube support plates. The NRC staff has considered more broadly applicable performance-based risk-informed guidance in DG-1074, *Steam Generator Tube Integrity*, and such an approach is reflected in current industry guidance for tube integrity assessments (NEI 97-06).

One major outcome of regulatory activity over the past 10 years to develop guidance for tube integrity assessments is the development and implementation of two key concepts, condition monitoring and operational assessment. Condition monitoring is an assessment of the current state of the steam generator relative to the performance criteria of structural integrity. An operational assessment is an attempt to assess what will be the state of generator relative to the structural integrity performance criteria at the end of the next inspection cycle. The predictions of the operational assessment from the previous cycle can be compared with the results of the condition monitoring assessment to verify the adequacy of the methods and data used to perform the operational assessment.

The reliability of such assessments and projections depends critically on the reliability of the NDE techniques used to establish the flaw distribution both in terms of detection of flaws and characterization of flaws, the capability to assess the impact of these flaws on the structural integrity of the tubes, and the ability to predict crack initiation, evolution, and growth. In most cases degradation of steam generator tubing by stress corrosion cracking is still managed by "plug or repair on detection," because current NDE techniques for characterization of flaws are not accurate enough to permit continued operation. This is very conservative in many cases, since flaws less than 40% throughwall or even deeper, short flaws have very little impact on tube integrity. On the other hand, current inspection technologies and procedures can miss flaws that will lead to steam generator tube ruptures.

5. Ongoing and Future Research

5.1 NDE Round Robin

An independent assessment of steam generator inspection reliability is being developed through an NDE round–robin on a steam generator mockup at Argonne National Laboratory. The purpose of the round robin is to assess the current state of SG tubing ISI reliability, determine the probability of detection (POD) as function of flaw size or severity and assess flaw sizing capability. Eleven teams have participated in analyzing bobbin coil and rotating coil mock-up data collected by qualified industry personnel. The mockup contains hundreds of cracks and simulations of the artifacts such as corrosion deposits, tube support plates, etc. that make detection and characterization of cracks more difficult in operating steam generators than in most laboratory situations. An expert group from ISI vendors, utilities, EPRI, ANL, and the NRC have reviewed the signals from the laboratory grown cracks used in the mockup to ensure that they provide reasonable simulations of those obtained from real cracks. The number of tubes inspected and number of teams in the round robin are intended to provide better statistical data on the probability of detection (POD) and characterization accuracy than is currently available from industry performance demonstration programs.

5.2 Advanced NDE

Other current research in EC NDE involves the development of advanced modern imaging and analysis algorithms[34]. Codes have been developed that permit more efficient and flexible analyses of rotating coil data. Simplification of interpretation of data is provided though improved enhanced visualization and automated analysis methodologies. It is now possible to produce NDE profiles of large sections of tubing in a fraction of the time that is needed for manual analysis.

Manual analysis of multiple frequency eddy current data is a tedious and challenging process. Signal distortion by interference from internal/external artifacts in the vicinity of flaw further complicates discriminating of flaw signals from noise. In comparison to high-speed bobbin coil inspections, high-resolution multi-coil rotating and array probes generate enormous amounts of data over comparable scan lengths. Rotating probe ISI of SG tubing is thus generally restricted to areas that are historically predisposed to known damage mechanisms and sections of particular interest that are flagged by the initial bobbin coil examinations. More extensive application of such probes for improving NDE reliability rests in part on automating various stages of the data screening process. Computer-aided data analysis is the only viable way to overcome many of the challenges associated with reliable processing of data acquired with high-resolution probes.

In order to characterize flaws in a SG, a variety of characterization methods are being examined. An automated imaging and analysis algorithm for the analysis of RPC data has been developed. The basic elements of the algorithm include automated calibration of the data, filtering and deconvolution to improve the signal to noise ratio, use of a rule-based expert system to classify indications, and the use of multifrequency, multiparameter correlations for flaw size.

The method also provides a graphical display which helps visualize cracking especially in cases like the roll transition where geometry greatly complicates analysis. The results can be presented directly in terms of depth profiles as a percentage of the tube wall thickness. Reconstruction of helically scanned data into C-scan format allows for the observation of sizing results from any azimuth and elevation view angle and for any axial or circumferential cross section of the tube. Typical examples of the graphical display are shown in Figs. 2a and 2b.

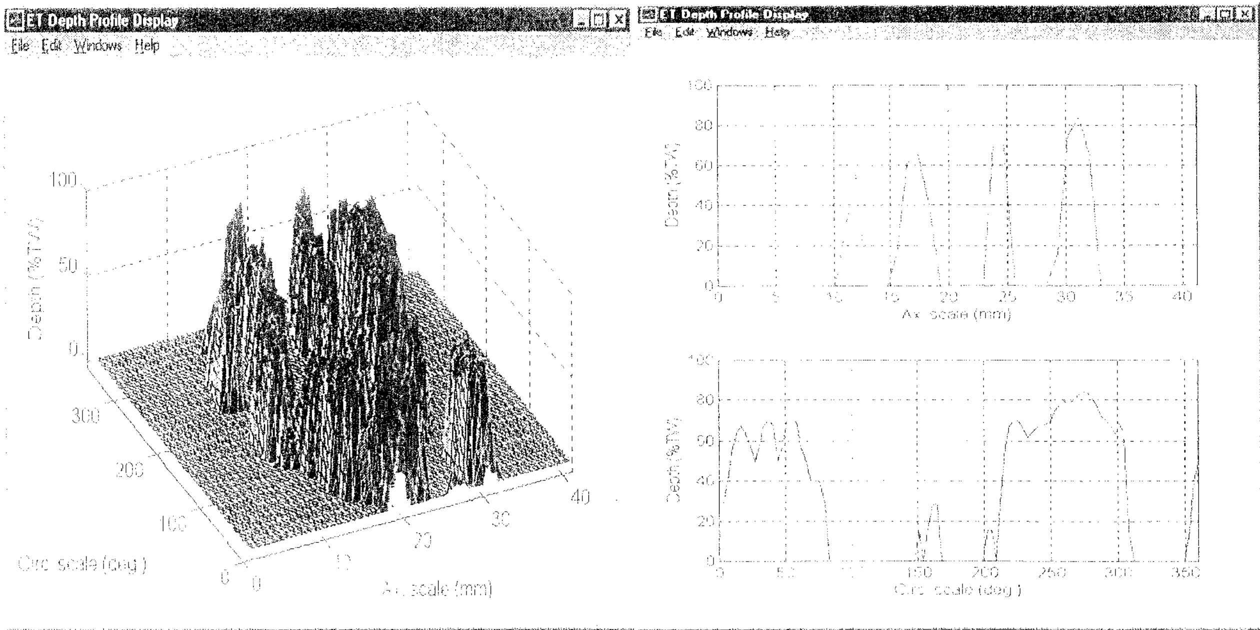
In the development of the multiparameter algorithm the results from the algorithm have been compared to fractographic results on a wide variety of SCC cracks and EDM and laser notches. To provide an objective benchmark, however, additional SCC cracks were produced and used for a blind test of the predictions of the algorithm against fractographic measurements of the crack geometry. Examples of the comparison of the NDE results with fractographic measurements are shown in Fig. 3.

5.3 Structural Integrity

As noted previously, well-established criteria for predicting ligament rupture and unstable burst pressures of tubes with relatively long rectangular flaws exist. Some modifications of these criteria have been made for short and deep flaws based on recent tests at ANL [13]. Although we can currently predict with some confidence failure pressures of tubes with flaws that are rectangular in shape, such a morphology is not characteristic of much of the cracking that is currently being observed in steam generators. Stress corrosion cracks in steam generator tubes are generally non-planar, ligamented, and can have highly complex geometry. Procedures for predicting ligament rupture for such complex cracks, using an "equivalent rectangular crack" approach (Fig. 4), has recently been developed[32,33]. Limited tests at ANL on steam generator tubes with laboratory generated stress corrosion cracks have shown the usefulness of such an approach. The use of the "equivalent rectangular crack" to predict leak rates through laboratory generated stress corrosion cracks has also proven to be promising. Additional leak rate and failure tests on tubes with stress corrosion cracks that are generated in the laboratory as well as on pulled tubes from a retired SG are currently being planned to further validate the approach.

5.4 Materials Degradation

Although the performance of Alloy 690TT material has been excellent to date, further work is needed to characterize its potential for SCC. As stated previously, laboratory studies have identified a variety of environments in which SCC of Alloy 690 TT occurs. It is commonly



(a) (b)

Figure 2. (a) Lab-grown circumferential ODSCC ≈360° staggered cracking with maximum depth > 80% TW. (b) Axial or circumferential cross-sections can be taken to get profiles of the crack

observed in mildly acidic and slightly oxidizing solutions and may be aggravated by the presence of chlorides or sulfates.[35] These mildly acidic conditions are particularly relevant because molar ratio control as well as sea water contamination can produce mildly acidic crevices in steam generators.

It has been observed that concentrations of lead in the ppm range in mildly acidic, neutral, and alkaline environments produce or substantially accelerate the SCC of Alloy 690.[21,36] Despite efforts to reduce lead contamination, it still persists in deposits inside steam generators in ranges substantially greater than the levels required to produce SCC in the laboratory. However, this Pb contamination has not yet produced SCC in the field, and the reasons are unclear.

Sulfur in the form of sulfides is another contaminant that is known to accelerate SCC in Alloy 690 TT, based on experiments in alkaline solutions. Sulfur is sometimes introduced as sulfates as feedwater contamination or in resin fines. These sulfates, in turn, can be reduced to sulfides either by hydrazine or by direct reaction with Alloy 690TT, and this reduction process and its consequences have not been adequately characterized.

Additional research is needed to address these and other issues. The crevice chemistries at various locations in steam generators must be determined, and the chemistry of Pb-containing deposits and their possible relationship to the occurrence of SCC should be evaluated. In addition, the conditions under which lower-valence sulfur compounds can form in crevice environments due to reactions with hydrazine or with Alloy 690 requires further

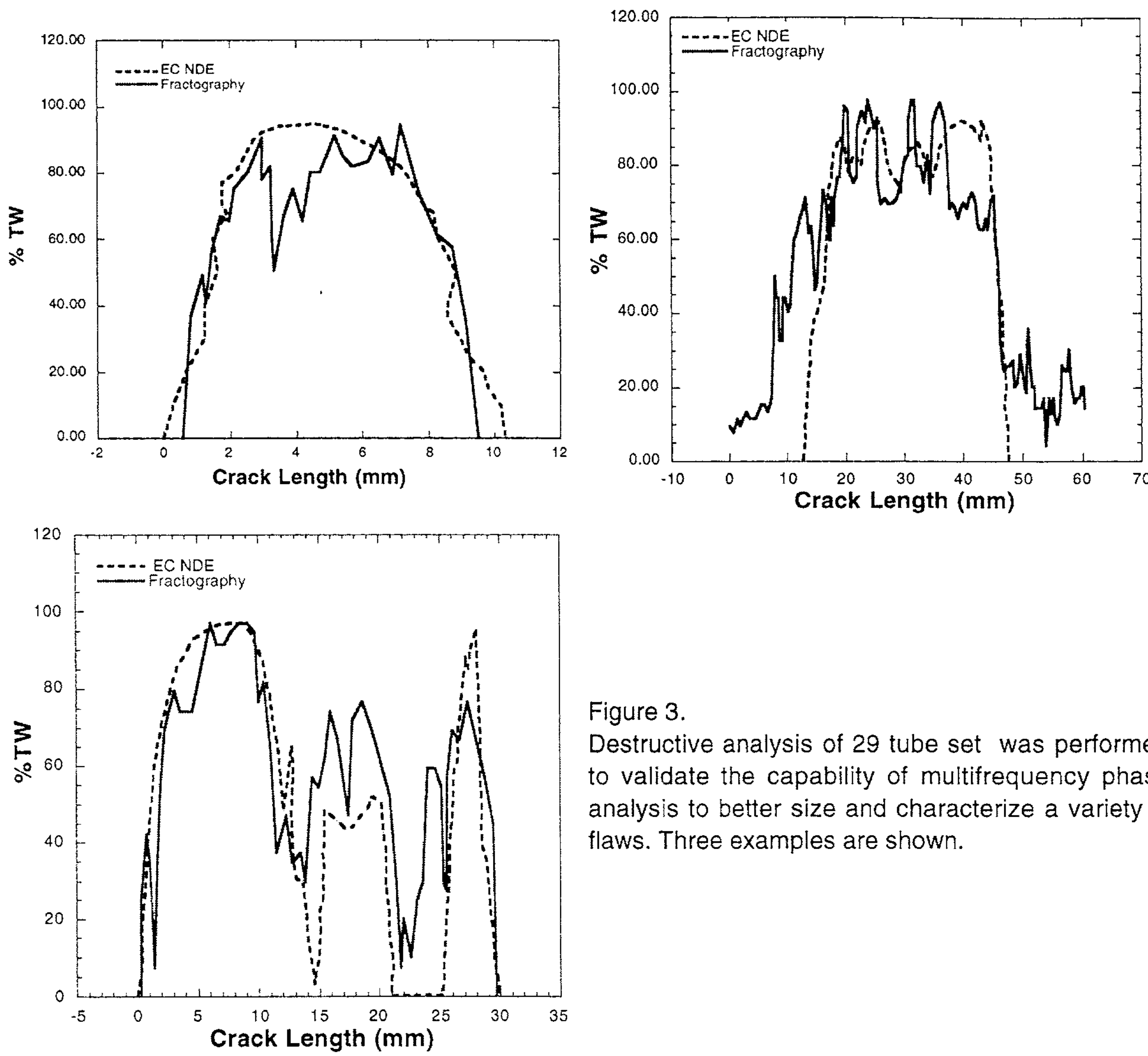


Figure 3.
Destructive analysis of 29 tube set was performed to validate the capability of multifrequency phase analysis to better size and characterize a variety of flaws. Three examples are shown.

study. In all of these studies, it is desirable to conduct the tests in the appropriate crevice chemistry environments.

Finally, studies on the behavior of Alloy 600 are still important, even though replacement steam generators are using Alloy 690 tubes. We have extensive and very valuable field experience with Alloy 600 that can be coupled with laboratory data. The knowledge gained in this coupling process can provide a bridge between laboratory data and expected field behavior for Alloy 690. The two alloys should be studied under similar conditions, and a better understanding of crack initiation, evolution, and growth under realistic crevice chemistry conditions is needed for both materials.

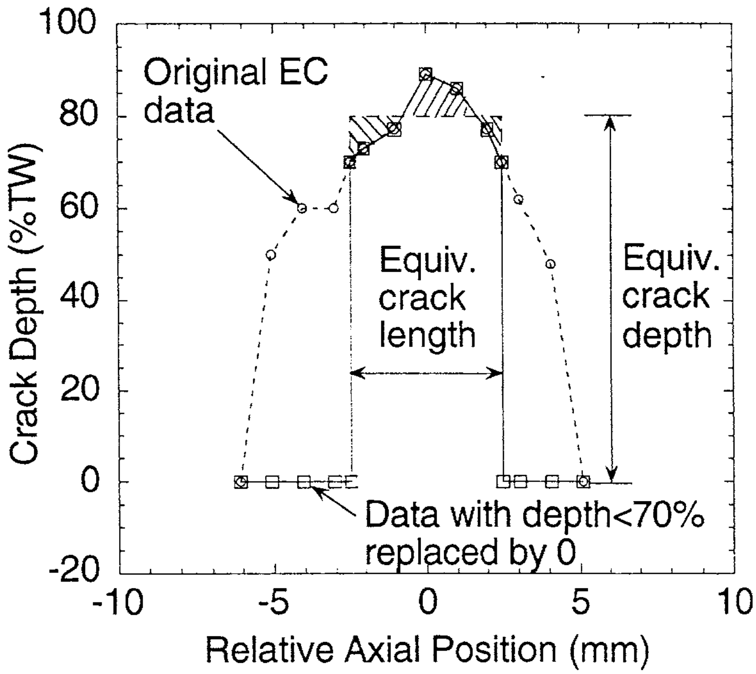


Figure 4.
"Equivalent rectangular crack" methods are promising, but more validation work on a wider variety of crack geometries is needed.

6. Observations and Recommendations

6.1 NDE

There is a need for a more robust screening of SG tubing. The development of array probes, which have better resolution than bobbin coils and enable rapid detection of both axial and circumferential cracks, may be the route to improved screening.

The industry has developed inspection technologies, performance demonstration and qualification programs that have improved the effectiveness and reliability of steam generator inspection programs, but improvements are needed in inspection guidelines and performance demonstration required to qualify techniques and analysts. The current passing criteria is 80% detection rate. Is it acceptable to have 20% of deep flaws remaining in service?

Currently, there are no passing criteria for sizing. Qualification for sizing is needed using a sample sets with realistic cracks and other flaws. Until better sizing can be achieved, degradation of steam generator tubing by stress corrosion cracking may have to be managed by "plug or repair on detection."

Replacement SGs with 690 tubing may require a different approach to ISI. In general larger samples are needed for early detection of developing degradation. Thus it may be preferable to have 100% sampling rather than the 3% or 20% often proposed currently. The larger inspection sample could be balanced by a lower frequency of inspection, particularly in early years of operation. In addition, a higher POD for smaller flaws is needed for early detection.

6.2 Integrity

Currently, reliable correlations for predicting structural integrity and leakage of tubes with single well-defined rectangular cracks or notches are available. These models can be used to conduct conservative calculations by replacing actual cracks by bounding rectangular cracks. However, these evaluations can sometimes be overly conservative. To obtain more realistic assessments, these models have to be extended and/or modified.

The "equivalent rectangular crack" appears to be a reasonable approach towards a more realistic description of planar cracks with irregular shapes.[32,33] The current approach projects the crack depth profile as measured from NDE on to a single plane and treats the crack as a planar crack. But tests show that such planar cracks tend to have lower ligament rupture and burst pressures and higher leak rates compared to cracks that are non-planar and segmented separated by ligaments. A key to developing more realistic rupture and leak rate criteria is to determine the behavior of such ligaments as a function of their size and width.

An interesting recent observation made on deep stress corrosion cracks is the significant time-dependent increase of leakage under constant pressure hold, indicating an increase of throughwall crack length due to time-dependent ligament rupture. Such behavior has been observed in tests at room temperature as well as at 282°C - a temperature regime where time-dependent creep deformation is generally accepted to be negligible. Such a time-dependent behavior under constant pressure suggests that the ligament rupture pressure of deep cracks under a constantly rising pressure test may be dependent on the pressurization rate. An analogous effect of pressurization rate on ligament rupture pressure has also been observed for deep planar machined notches with variable ligament width. Currently, we have no model to account for such time-dependent ligament rupture phenomenon.

To select appropriate integrity model used in the operational assessment, we need to better predict how flaws develop, evolve, and grow from more complex infant cracks to planar cracks. Again, the behavior of ligaments under pressure and corrosive environment is the key to developing such predictive models.

Nucleation and early growth of stress corrosion cracks are controlled by various factors – some of them are mechanical (e.g., stress), some are environmental (e.g., temperature), some are chemical (e.g., pH) and some are metallurgical (e.g., carbide morphology). Currently, the complex interactions between these various factors are not clearly understood. Also, crack morphology during this period can be very complex (e.g., cellular cracks rather than a single dominant crack). As a result, mechanistic models for predicting crack initiation are currently lacking and empirical models based on stress are often used for predicting crack initiation. With continued service exposure, often a dominant single crack emerges for which the mechanical component of the crack driving force becomes controlling. Fracture mechanics-based models can then be used to calculate the growth of such cracks and crack growth rate data for alloy 600 under primary and secondary water environments have been generated for this purpose.

6.3 Materials Degradation

As noted previously service experience to date with thermally treated Alloy 690 has been good. After up to 11 years of service, no incidents of SCC have been reported in operating steam generators. However, although laboratory studies have also been unable to produce SCC in Alloy 690 in primary water chemistries, numerous studies have demonstrated the ability to crack this alloy under conditions that approximate chemistries that could occur under crevice conditions on the secondary side of steam generators. In addition, it should be noted that widespread instances of SCC with Alloy 600 tubes did not occur until after ≈10 years of service. The situation has some similarity to that in the late '60s when Alloy 600 was thought to be the solution to steam generator corrosion problems. Although designs and water chemistry controls have improved and Alloy 690 is clearly a more resistant material, it should not be assumed that the SCC problem in PWR steam generators has been permanently solved through this choice of materials.

Studies of Alloy 600 behavior are still important, even though replacement steam generators use Alloy 690 tubes. There is extensive field experience with Alloy 600 that can be coupled with laboratory data to help understand and validate the relation between laboratory data and behavior in actual steam generators. This information can be used to provide a bridge between laboratory data and field behavior for Alloy 690.

Substantial progress is being made in improving NDE capability and developing a better understanding of the structural behavior of flawed tubes. However, we still need to gain a better understanding of crack initiation, evolution, and growth under realistic crevice chemistry conditions for both Alloy 600 and 690 to carry out more realistic operational assessments of steam generator integrity.

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