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Proceedings of the CNRA/CSNI Workshop on Steam Generator Tube Integrity in Nuclear Power Plants

Held in
Oak Brook, Illinois
October 30–November 2, 1995

U.S. Nuclear Regulatory Commission

Prepared by
D. R. Diercks

Argonne National Laboratory

Sponsored by
The Committee on Nuclear Regulatory Activities
The Committee on the Safety of Nuclear Installations

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Abstract

An International Workshop on Steam Generator Tube Integrity in Nuclear Power Plants, sponsored by the Committee on Nuclear Regulatory Activities (CNRA) and the Committee on the Safety of Nuclear Installations (CSNI) of the OECD-NEA, was held at Oak Brook (suburban Chicago), Illinois, on October 30–November 2, 1995. The USNRC Office of Nuclear Regulatory Research served as host. The objective of the workshop was to provide a working forum for the exchange of information by contributing experts on current issues related to PWR steam generator tube integrity. One hundred persons from 15 countries attended the workshop, including 36 from regulatory and nuclear policy agencies, 28 from research and development laboratories, 18 from nuclear vendors and consulting firms, and 18 from electrical utilities.

The workshop opened with a plenary session; the first part of the session covered international steam generator regulatory practices and issues, featuring speakers from regulatory bodies in Belgium, France, Japan, Spain, and the United States. In Part 2 of the plenary session, comprehensive technical overviews on steam generator tubing degradation, inspection, and integrity were presented by authorities in these fields from the United States, France, and Belgium. Parallel working sessions on the second and third days of the workshop then developed findings and recommendations in the areas of (1) tubing degradation, (2) tubing inspection, (3) tubing integrity, (4) preventative and corrective measures, and (5) operational aspects and risk analysis. On the final day of the workshop, the working-session facilitators presented summaries of their sessions to the workshop attendees.

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Executive Summary

An International Workshop on Steam Generator Tube Integrity in Nuclear Power Plants, sponsored by the Committee on Nuclear Regulatory Activities (CNRA) and the Committee on the Safety of Nuclear Installations (CSNI) of the OECD-NEA, was held at Oak Brook (suburban Chicago), Illinois, on October 30-November 2, 1995. Serving as host was the Office of Nuclear Regulatory Research of the USNRC. The objective of the workshop was to provide a working forum for the exchange of information by contributing experts on current issues related to PWR steam generator tube integrity. One hundred persons from 15 countries attended the workshop, including 36 from regulatory and nuclear policy agencies, 28 from research and development laboratories, 18 from nuclear vendors and consulting firms, and 18 from electrical utilities.

The workshop opened with a plenary session; the first part of the session covered international steam generator regulatory practices and issues, featuring speakers from regulatory bodies in Belgium, France, Japan, Spain, and the United States. In Part 2 of the plenary session, comprehensive technical overviews on steam generator tubing degradation, inspection, and integrity were presented by authorities in these fields from the United States, France, and Belgium. Parallel working sessions on the second and third days of the workshop then developed findings and recommendations in the areas of (1) tubing degradation, (2) tubing inspection, (3) tubing integrity, (4) preventative and corrective measures, and (5) operational aspects and risk analysis. On the final day of the workshop, the working-session facilitators presented summaries of their sessions to the workshop attendees. Extended versions of these summaries, which were prepared by the facilitators subsequent to the workshop, are included in these proceedings.

Several of the workshop sessions highlighted the need for more examinations of pulled tubes and tube/tube support plate assemblies from operating and retired steam generators. Such examinations were considered important from the standpoint of better defining crevice chemistry conditions, providing improved reference standards for NDE, providing prototypic flawed specimens for tube integrity testing, and evaluating tubing repair techniques such as sleeving. Other significant conclusions and recommendations include the following: (1) an expanded data base on ODSCC crack initiation and growth rates is needed, (2) techniques for determining and possibly monitoring crevice chemistries should be developed, (3) a better understanding of the influence of local stress and strain distributions on crack initiation and growth is needed, (4) international cooperation on realistic "universal" samples should be pursued to develop improved NDE techniques for flaw detection and sizing, particularly for circumferential cracks, (5) improvements in data analysis are needed for current voltage-based tube plugging criteria, (6) an international data base on tube pressure and leakage tests and destructive examinations should be developed, (7) new steam generator designs are generally well-designed to avoid known degradation processes, (8) good sludge control is essential to preventing steam generator

degradation, (9) qualification guidelines should be established for repair and maintenance techniques, and (10) leakage monitoring is very important to evaluating steam generator degradation and improving operator response to a tube rupture event.

Acknowledgments

The International Workshop on Steam Generator Tube Integrity in Nuclear Power Plants was sponsored by the Committee on Nuclear Regulatory Activities (CNRA) and the Committee on the Safety of Nuclear Installations (CSNI) of the OECD-NEA and was hosted by the Office of Nuclear Regulatory Research of the U.S. Nuclear Regulatory Commission. The following persons served on the Workshop Program Committee:

J.-P. Clausner, OECD-NEA
J. Figueras, CSN, Spain
G. Hedner, SKI, Sweden
T. Iwase, NUPEC, Japan
I. D. Moussebois, AIB-Vinçotte, Belgium
J. Muscara, USNRC, United States
G. Turluer, IPSN, France

Workshop arrangements were handled by D. R. Diercks, J. S. Brunsvold, and D. K. Moores of Argonne National Laboratory.

1 Introduction

Steam generator tubing has exhibited a wide variety of degradation mechanisms. As a result, considerable effort has been expended in addressing the safety and economic implications of these degradation processes. These efforts have resulted in improved inspection techniques, the development of defect-specific tube repair criteria, enhanced primary-to-secondary leakage monitoring programs, and implementation of various preventative and corrective measures. Nevertheless, steam generator tube integrity continues to be a major issue for operators, vendors, and regulators of nuclear power plants, and efforts continue to be directed at addressing issues related to steam generator tube integrity.

The present International Workshop on Steam Generator Tube Integrity in Nuclear Power Plants, sponsored by the Committee on Nuclear Regulatory Activities (CNRA) and the Committee on the Safety of Nuclear Installations (CSNI) of the OECD Nuclear Energy Agency (NEA), was held at Oak Brook (suburban Chicago), Illinois, on October 30-November 2, 1995. The USNRC Office of Nuclear Regulatory Research served as host. The purpose of this workshop was to provide a unique forum for the exchange of information on topics related to steam generator tube integrity, with emphasis on regulatory and safety issues. One hundred contributing experts from 15 countries were given the opportunity to meet their counterparts from around the world to discuss regulatory and research issues on this topic. The participants included 36 from regulatory and nuclear policy agencies, 28 from research and development laboratories, 18 from nuclear vendors and consulting firms, and 18 from electrical utilities.

The workshop opened with a plenary session on international steam generator regulatory practices; Part 1 of this session covered regulatory practices and issues and featured speakers from regulatory bodies in Belgium, France, Japan, Spain, and the United States. In Part 2, comprehensive technical overviews on steam generator tubing degradation, integrity, and inspection were presented by authorities in these fields from the United States, France, and Belgium. Parallel working sessions on the second and third days of the workshop then developed findings and recommendations in the areas of (1) tubing degradation, (2) tubing inspection, (3) tubing integrity, (4) preventative and corrective measures, and (5) operational aspects and risk analysis. On the final day of the workshop, the working-session facilitators presented summaries of their sessions to the workshop attendees. Extended versions of these summaries, which were prepared by the facilitators subsequent to the workshop, are included in these proceedings.

These proceedings contain the eight plenary session papers presented at the workshop, as well as the 22 technical papers presented in the working sessions. Also included are summaries of the discussions in the working sessions and the conclusions resulting from these sessions. A brief review and summary of some of the major conclusions and recommendations resulting from the workshop is included at the beginning of these proceedings.

2 Workshop Overview and Summary

D. R. Diercks, Argonne National Laboratory

Working sessions were conducted in the areas of (1) tubing degradation, (2) tubing inspection, (3) tubing integrity, (4) preventative and corrective measures, and (5) operational aspects and risk analysis. Detailed summaries of the conclusions from these working sessions, as prepared by the working session facilitators, are presented in Sections 5 through 9 of these proceedings. A review of these summaries makes it apparent that a number of significant issues and concerns were raised that cut across the specific interests of the individual working sessions. These include concerns about circumferential cracking, the need for a better understanding of crevice chemistry conditions, the desirability of additional pulled tubes, the need for improved NDE techniques for better flaw detection and sizing, the need for more and better crack-growth-rate data, and the importance of leakage monitoring. These common issues and concerns are discussed in more detail here.

2.1 Circumferential Cracking

Session 1 on tube degradation noted that circumferential cracking of steam generator tubes is potentially a major safety issue and that the occurrence of circumferential outer diameter stress corrosion cracking (ODSCC), particularly near the top of the tube sheet, is becoming increasingly common. In both Sessions 1 and 2, it was observed that circumferential cracks are often difficult to detect, and that an unambiguous distinction between OD- and ID-initiated circumferential cracks cannot always be made. The probability of detection appears to be higher for ID-initiated circumferential cracks than for OD-initiated cracks, and the French experience cited in Session 1 indicates that OD cracks can propagate through $\approx 40\%$ of the wall thickness before becoming detectable by rotating pancake coil (RPC) eddy current technology (ECT). A median detection limit of $\approx 50\%$ through-wall was suggested for RPC ECT technology in Session 2. There was general agreement that conventional bobbin coil probes cannot detect circumferential cracks, and that development of NDE techniques with inspection speeds similar to bobbin coils but with improved circumferential crack detection capabilities is an important need. It was suggested that ultrasonic technology (UT) in conjunction with ECT appears to offer the best hope for the improved detection of circumferential cracks.

In Session 3 on structural integrity, it was noted that there is no worldwide uniform position on leaving tubes with circumferential cracks in service. Belgium has implemented criteria for leaving tubes with circumferential cracks in the roll transition zone (RTZ) above the tubesheet in service, and Canada permits some circumferential cracks in the U-bends. As noted in Session 2, these criteria are based on examinations of large numbers of pulled tubes in Belgium and the testing of laboratory-generated cracked tubes in Canada. The regulatory bodies in other countries generally require plugging of circumferentially cracked tubes upon detection. In both Sessions 2 and 3, the need for the improved probability of

detection and sizing of circumferential cracks was underscored. With this capability, an improved basis for determining the serviceability of tubes with such cracks could be developed.

2.2 Crevice Chemistry

In both Sessions 1 and 4, the need for developing improved characterization of crevice chemistry conditions and an improved understanding of the influence of crevice chemistry on stress corrosion cracking was considered in some detail. It was noted in Session 1 that a better characterization of the local water chemistry in crevices would permit more meaningful autoclave experiments on tube SCC to be conducted with bulk water chemistries corresponding to those present in the crevices. Additional studies on crevice chemistry were urged as a part of a larger activity to better understand the mechanisms of SCC in steam generator tubes.

Participants in both Sessions 1 and 4 agreed that pulled tubes with intact tube support plate crevices provide a very valuable resource for characterizing crevice chemistry, but participants in both sessions noted that such tube pulls are expensive and the number of samples available is limited. Instrumented model boiler tests, ideally using a side-stream boiler connected directly to an operating PWR, were proposed as another means for determining local crevice chemistry conditions. The efforts of Jesse Lumsden at Rockwell International in using an instrumented autoclave system to determine crevice chemistries were mentioned in both Sessions 1 and 4. There was agreement that additional work is clearly needed on the characterization of crevice water chemistries, and Session 4 participants noted that the present lack of knowledge of specific crevice conditions makes it difficult to evaluate proposed water chemistry remedies.

2.3 Pulled Tubes

Discussions in Sessions 1 and 4 on the value of pulled tubes and tube/tube support plate assemblies in assessing crevice water chemistry conditions were mentioned above. Participants in Session 1 noted that in addition to the high cost of such tube pulls, the opportunities for obtaining pulled tubes are generally limited. A number of retired steam generators have already been lost with respect to tube pulls because the units have been buried or otherwise disposed of. Accessibility to retired units that have been placed in long-term storage is also generally quite limited, and these units are not always laid up under conditions that leave them in a useful state. Nonetheless, participants in both Sessions 1 and 4 recommended vigorous pursuit of tube pulls from operating and retired steam generators and the "mining" of tube/tube support plate samples from retired units.

The desirability of obtaining pulled tubes and tube/tube support plate assemblies from operating and retired steam generators was underscored in the other working sessions as well. Participants in both Sessions 2 and 3 noted that such pulled tubes are needed to provide a source of realistic defects for evaluating and calibrating NDE techniques, and international collaboration to obtain more

pulled tubes was urged. Testing of tubes pulled from service to verify structural integrity models was also strongly recommended in Session 3. The need for correlations of NDE parameters to leakage rates in cracked tubes was also cited, since some plugging/repair criteria permit leaking tubes to remain in service under some conditions.

2.4 Improvements in NDE

The shortcomings associated with present NDE probes and the need for the continued development of advanced technology were cited in several of the working sessions. The limitations of both bobbin coil and RPC eddy current probes cited in Session 1 and 2 have already been noted in the preceding discussion of circumferential cracking. It was observed in Session 2 that improved inspection techniques that have the inspection speed of the conventional bobbin coil but with improved detection and sizing capabilities for circumferential cracks are needed. This same statement clearly applies to other types of defects as well. Session 1 participants noted that periodic NDE is generally not useful in the detection of fatigue cracks, which usually grow relatively quickly. Instead, plants must rely on the concept of leak-before-break to cope with unexpected fatigue degradation.

Participants in Session 2 noted that if flawed tubes are to be left in service under some circumstances, improvements in probability of detection and flaw sizing are needed. They cited the recent implementation of multicoil probes as promising, but stated that more experience with these probes and further improvements in analysis tools and data presentation are needed before these probes become widely accepted. Replacement of the current widely used voltage-based criteria for assessing axial flaws will depend in part on advances in probe technology. While UT has the potential for greater spatial resolution and sensitivity, widespread application of this technique is presently limited by problems with false positive calls and relatively slow inspection rates. In Session 3, it was similarly noted that some repair criteria considered under design specific management approaches to tube degradation require higher confidence in NDE and defect growth rate, and that more data are necessary to verify NDE capability and reliability and to correlate burst pressures and leak rates to NDE parameters.

Participants in Session 4 on preventative and corrective measures also considered the development of improved NDE techniques, and in particular UT, for a different reason. They noted that Ni plating is being used in Belgium and France to repair ID cracking from PWSCC. It has been most extensively used in Belgium in the RTZ, and several years of favorable operating experience have been obtained. However, the ferromagnetic Ni-plated layer complicates eddy current inspection, and the Belgians routinely use UT as a part of their inspection procedure. As a result, they are actively pursuing an extensive development program on improved UT techniques.

2.5 Crack-Growth-Rate Data

Related to the need for improved NDE techniques is the need for better crack-growth-rate data, particularly in the degradation-specific management approach to managing tube degradation. In order to justify operation with tubes containing defects, one must be able to confidently detect and size the defects and to accurately predict their growth rates. Thus, as noted in Session 3, the successful implementation of a degradation-specific management approach requires both improved NDE and better data on crack growth rates. The participants in Session 1 similarly concluded that SCC crack propagation data were needed for both circumferential and axial cracks, particularly under prototypical heat flux and local chemistry conditions. i.e., these data should be determined in model boilers rather than in isothermal autoclave systems. It was noted that for the case of ODSCC in particular, a strong dependence of crack growth rate on local chemistry might be expected. Because crevice chemistry conditions are not well understood and may vary with time and location even in a given steam generator, large variations in crack-growth-rate behavior are possible even under nominally similar bulk conditions.

2.6 Leakage Monitoring

Primary-to-secondary leakage monitoring was cited in two of the working sessions as an important tool for managing tube degradation and avoiding tube rupture. Participants in Session 5 on operational aspects and risk analysis noted that all plants use some form of leakage monitoring, but not necessarily to its full advantage. They recommended more extended use of and further improvements in N-16 detectors on main steam lines to enhance the ability of operators to promptly respond to tube rupture events. They also recommended improved leakage-to-degradation correlations for various modes of tube degradation and urged the establishment of a more robust technical basis for operational leakage limits. Participants in Session 3 also noted the importance of leakage monitoring while simultaneously recognizing that not all tube degradation processes necessarily lead to leak before break. It was recommended that leakage monitoring programs at plants include limits on the absolute leakage value, as well as on the trend.

2.7 Severe Accidents

Participants in both Sessions 3 and 5 discussed tube integrity during accident situations such as a main steam line break or a core melt event. It was noted in Session 5 that tube degradation could adversely impact the ability of the tubes to fulfill their fission-product containment function during a core damage event, but it was not clear how much such a tube rupture event would contribute to overall plant risk in comparison with other potential contributing events. Implicit in these discussions was the need for more data on the burst behavior of flawed and unflawed tubes under a core melt accident event.

Participants in Session 3 dealt with the question of using the differential pressure associated with a postulated main steam line break event as a design basis loading. It was agreed that such a break may be considered a bounding case for secondary-side depressurization events in general. Related questions discussed in Session 3 include leakage rates through axial cracks in the roll transition zone at postulated accident differential pressures, the potential for tube support plate displacement during accidents, and, as discussed above, the issue of leaving circumferential cracks in service.

3 Opening of Meeting

- 3.1 *Welcome*, J.-P. Clausner, OECD/NEA
- 3.2 *Welcome*, J. Muscara, USNRC/RES
- 3.3 *Welcome*, A. C. Thadani, USNRC/NRR
- 3.4 *Welcome*, D. Morrison, USNRC/RES

CNRA/CSNI International Workshop on Steam Generator Tube Integrity in NPPs
Oak Brook, IL. (USA) October 30-November 2, 1995
Welcoming Address by Jean-Pierre Clausner, NEA Administrator

Good morning ladies and gentlemen. My name is Jean-Pierre Clausner and, on behalf of the OECD Nuclear Energy (NEA), I am very pleased to address a warm welcome to all of you to this CNRA/CSNI Workshop on Steam Generator Tube Integrity. I would like also to thank the USNRC for hosting the meeting and for its support together with the Argonne National Laboratory in organizing the meeting.

As you know, the meeting has been organized in the framework of the NEA's activities, but may be for those of you who are not familiar with that organization, I should briefly explain both its role and its activities.

The NEA is one of the fifteen bodies that make up the Paris based Organization for Economic Cooperation and Development (OECD). Since the majority of the OECD's activities are oriented towards economics, the role of the Agency is less known and may be sometimes confused with that of the International Atomic Energy Agency (IAEA) in Vienna.

In connection with safety, the charter of the NEA calls for '... the promotion of the safety of nuclear installations...', '...the establishment of joint services for the prevention of accidents..', and for '.. the dissemination of information.. on the safety and regulation of nuclear activities..'.¹

With some 80% of the 430 reactors operated around the world, Members have set up and, over the years, developed a close operational feedback process in order to meet the primary objective of the Agency which aims to promote cooperation between regulatory authorities of its Member countries, contribute to nuclear safety and regulations of nuclear activities and, ultimately, to help in the development of nuclear energy as a safe environmentally acceptable energy source.

Activities of the NEA in the vast domain of nuclear safety and regulation include: operating experience and human factor, primary coolant system behavior, reactor component integrity, probabilistic safety assessment, severe accident management, nuclear fuel cycle, regulatory activities and some specific project as PISC, the OECD Halden Project and the RASPLAV Project conducted in collaboration with Russia. Most of these activities are determined by the two standing committees which deal with safety matters: the CSNI (Committee on the Safety of Nuclear Installations), and the CNRA (Committee on Nuclear Regulatory Activities) which both consist of delegates representing the NEA Member countries¹.

While the CSNI has been responsible for more than 20 years for the exchange of information on safety research programs enabling participating countries to develop common approach on identified safety issues, the CNRA, created in 1989, aims at exchanging information and developments which could affect national regulatory practices, and at achieving a common understanding on possible new regulatory requirements.

In brief, the role of these committees is neither to impose nor judge but, rather, through exchanging and understanding, to reach a common technical opinion and develop international consensus based on a better knowledge of technical issues and regulatory practices.

¹The OECD Nuclear Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. Its received its present designation on 20th April 1972 when Japan became its first non-European full member. NEA membership today consists of all European Member countries of the OECD as well as Australia, Canada, Japan, Mexico, Republic of Korea and the United States.

With respect to steam generators, the attention of the nuclear community has been attracted to the safety issues posed by the operation of this equipment for more than ten years. In 1984, a specialist meeting organized the CSNI Principal Working Group NY1 on Operating Experience and Human Factor (PWG1), concluded that problems related to steam generators (SGs) had evolved from availability concerns to safety concerns and therefore, that techniques of inspections should be developed as well as international cooperation and exchange of technical information. In September 1991, a second Specialist Meeting held in Brussels, concluded with an increase in the frequency of steam generator tube ruptures and identified some divergence in licensing issues and in understanding of defects and problems induced by corrective actions. Based on these findings, the CNRA at its third meeting in November 1991, expressed its concern regarding the types and frequency of degradation observed during SG inspections and agreed to hold a special issue meeting to deal with questions related to plugging criteria, reliability of inspection techniques, water chemistry, aging and corrective measures.

The meeting which took place in Paris in June 1992 concluded that neither common practices nor policies could be drawn, in particular because both techniques and policies were still evolving. It was recommended to continue to closely monitor the evolution of degradation in steam generators and define an appropriate forum whenever the need would be identified, to discuss regulatory practices and research needs, and, ultimately to develop conclusions and proposals for areas which would require more attention. This is the purpose of this workshop and I am sure that with the group of experts gathered here today and through your exchanges and discussions, the meeting will meet its objective and hopefully your expectations.

In closing, I would like to reiterate my thanks to our hosts as well as to you all for your contribution in the success of the workshop. I wish you a very interesting and productive meeting.

Welcoming Speech -- J. Muscara, USNRC

Good morning. My name is Joe Muscara, and I am from USNRC Office of Nuclear Regulatory Research. As Chairman of the Program Committee for this workshop, I wish to extend my welcome to all of you.

This workshop, which is being sponsored by the OECD Nuclear Energy Agency and hosted by the Office of Nuclear Regulatory Research of the U.S. Nuclear Regulatory Commission, was conceived as providing a unique international forum for the exchange of operating experience and ongoing activities related to steam generator tube integrity. In keeping with the interests of the supporting agencies, the emphasis in this workshop will be on regulatory and reliability issues rather than economic concerns. This emphasis will be underscored later this morning in a series of presentations that will provide an international overview of steam generator regulatory practices and issues. Following that, three international experts will provide comprehensive summaries of our current understanding of steam generator tube degradation, integrity, and inspection. The purpose of these plenary presentations is to provide a common background of information for the subsequent development of overall integrity and safety assessments in the working sessions.

After this opening plenary session, the participants will divide into five parallel working sessions that will deal with the topics of tubing degradation mechanisms, inspection, integrity, preventative/corrective measures, and operational aspects/risk analysis. The objective of these working sessions is to identify technical issues and to discuss, debate, and provide recommendations for the resolution of these issues. The working sessions will be informal in nature, and the technical discussions should be free and open. The concluding session will consist of summary presentations by the facilitators of each workshop session, followed by an integration session in which the facilitators and the CNRA representatives will develop overall summary conclusions and recommendations on regulatory and research issues relevant to regulators.

The Workshop Program Committee worked diligently to develop the structure and program for this workshop, which I believe will enable us to meet our objectives. I wish to thank the Program Committee members for their contributions:

Jean-Pierre Clausner of the NEA
José Figueras of the CSN in Spain
Gert Hedner of the SKI in Sweden
Toshihiko Iwase of NUPEC in Japan
Kenneth Karwoski of the USNRC
Dominique Moussebois of AIB-Vinçotte in Belgium
William Shack of Argonne National Laboratory, and
Guy Turluer of the IPSN in France

I also want to thank the staff of Argonne, and in particular, Dwight Diercks, for participating in the planning meetings and making the conference arrangements. I urge all of you as participants to help make this workshop successful by actively contributing to your sessions. Best wishes for a productive four days, and I look forward to hearing the summary conclusions from our deliberations that will be presented in the integration session.

**WORKSHOP ON STEAM GENERATOR TUBE
INTEGRITY IN NUCLEAR POWER PLANTS**



ASHOK C. THADANI

**ASSOCIATE DIRECTOR FOR TECHNICAL REVIEW
OFFICE OF NUCLEAR REACTOR REGULATION**

OCTOBER 30, 1995

HISTORICAL PERSPECTIVE

1984 CSNI SPECIALIST MEETING IN STOCKHOLM

Exchange of technical information was beneficial

1991 CSNI SPECIALIST MEETING IN BRUSSELS

Differences in regulatory criteria and operating experience noted

1992 CNRA SPECIAL ISSUE MEETING

Tube integrity continues to be a major issue

MANY CHANGES HAVE OCCURRED SINCE 1992 CNRA MEETING

Regulatory developments

Alternate tube repair criteria

Research developments

Eddy current probes

Operating experience

Tube rupture at Palo Verde

TUBE INTEGRITY IN THE U.S.

MANAGED THROUGH A VARIETY OF ELEMENTS SUCH AS

Design

Inspection

Repair

Leakage monitoring

**PLANT IS DESIGNED FOR MITIGATING A SINGLE STEAM
GENERATOR TUBE RUPTURE**

Emergency operating procedures

Trained operators

Offsite radiological dose consequences

CURRENT REGULATORY FRAMEWORK IN THE U.S.

DEVELOPED WHEN PREVALENT FORMS OF DEGRADATION INVOLVED GENERAL WALL THINNING

Forms of degradation have changed

Inspection technology has improved

CURRENT INSPECTION AND REPAIR CRITERIA ARE OUTDATED AND PRESENT FRAMEWORK LEADS TO FREQUENT PLANT-SPECIFIC DECISION MAKING

OPERATIONAL IMPLICATIONS OF CURRENT FRAMEWORK

Increased plugging/sleeving and associated costs

Potential for power reduction as steam generator plugging increases

Shortened operating life of steam generators/continued facility viability

Large expenditure of resources

RULEMAKING EFFORT

USNRC IS DRAFTING A NEW STEAM GENERATOR RULE TO ADDRESS CURRENT SITUATION

Performance based

Adaptable

Provide Incentives

Integrated - systems, materials, and radiological considerations

Framework for degradation specific management

PERFORMANCE BASED RULE WOULD ESTABLISH HIGH LEVEL CRITERIA

Accompanying regulatory guide would provide guidance on industry steam generator programs

SG RULE MUST CONFORM WITH COMMISSION SAFETY GOAL GUIDANCE

Containment bypass for severe accident scenarios

Frequency of high pressure core melt sequences

Severe accident conditions (pressure, temperature)

Tube performance under the postulated conditions

Containment bypass probability

RULEMAKING EFFORT (cont.)

**ENFORCEMENT ACTION IS TAKEN WHEN USNRC REGULATIONS
ARE NOT MET**

CURRENT SCHEDULE

Draft rule and regulatory guide for public comment: June 1996

Final rule and regulatory guide: January 1997

**USNRC OFFICE OF RESEARCH HAS INITIATED A NEW STEAM
GENERATOR TUBE INTEGRITY PLAN TO SUPPORT/SUPPLEMENT
RULEMAKING EFFORT**

WELCOMING ADDRESS BY DAVID MORRISON
Director, NRC Office of Nuclear Regulatory Research

Good Morning. My name is David Morrison, and I am Director of the NRC Office of Nuclear Regulatory Research. I would like to welcome all of you to this Workshop on Steam Generator Tube Integrity in Nuclear Power Plants, sponsored by the Committee on Nuclear Regulatory Activities and the Committee on the Safety of Nuclear Installations of the OECD Nuclear Energy Agency.

It's no secret to anyone in this room that steam generator reliability and performance are serious problems in the operation of pressurized water reactors. Steam generators have been variously described as the "weak link"¹ and "one of the most troublesome components"² in the PWR design. In particular, steam generator tubing is subject to a variety of degradation processes that can lead to tube cracking, wall thinning, and potential leakage or rupture. This workshop will deal with these degradation processes and the related issues of tubing inspection, tubing integrity, preventative and corrective measures, and operational aspects and risk analysis.

Corrosion problems in PWR steam generator tubes can be traced back at least as far as 1957 to the Shippingport reactor, which was the first commercial PWR operated in the U.S. One of the four Shippingport steam generators developed leaking cracks in two tubes after only 150 hours of full-power operation.^{3,4,5} The tubes in this early design were made of Type 304 stainless steel, and the failures were attributed to stress corrosion cracking produced by free caustic and steam blanketing on the secondary side. The tubes were plugged, and the use of a modified phosphate water chemistry was instituted to control secondary water pH. As you know, these phosphate additions were subsequently found to produce their own set of corrosion problems, and corrosion-related degradation of steam generator tubes has been a serious problem ever since, in spite of the use of a number of water chemistries, tube materials, and steam generator designs.

The impact of these tube degradation problems has been enormous. Since the plugging of two tubes in Shippingport in 1957, more than 100,000 steam generator tubes have been plugged to date around the world.⁶ In 1993, more than half the operating PWRs included in the EPRI worldwide survey were required to plug steam generator tubes. Fifty steam generators in 18 PWRs around the world had been replaced by the end of 1993 because of serious tubing degradation, and the replacement of 85 additional steam generators at 26 plants was planned.⁶ The cost associated with such steam generator replacements now typically ranges between 100 and 200 million dollars per plant. This figure does not include power replacement costs.⁷ In early 1993, the Trojan nuclear plant in the United States was decommissioned at a projected cost of more than \$400 million in large part because of severe steam generator tube degradation problems.⁷

It is clear from these numbers that steam generator tubing degradation imposes a serious economic burden on PWR plant owners. In addition, the NRC and corresponding regulatory bodies around the world remain concerned about the potential safety implications of this problem. In terms of surface area, steam generator tubes make up the majority of the primary pressure boundary in a PWR. Propagation of tube flaws can result in tube rupture, particularly in the event of a main steam line break, which would result in a large differential pressure.

Steam generator tube rupture is one of the design basis accidents considered in the NRC safety review of nuclear power plants.⁸ It is a significant safety issue for two

reasons. First, it can result in containment by-pass leading to release of radionuclides to the environment. Second, tube rupture results in the loss of primary reactor coolant which can significantly exceed the make-up capacity of the charging pumps.

Starting with the Point Beach incident in 1975,² at least 11 incidents of steam generator tube rupture have been reported worldwide. Reported maximum primary-to-secondary leak rates ranged from 74 gallons per minute (280 liters per minute)^{8,9} to as high as 760 gallons per minute (2,900 liters per minute).¹⁰ The principal causes of the ruptures were ODSCC, PWSCC, loose parts, and high-cycle fatigue, with other incidents produced by wastage and a failed tube plug. In all cases, only a single tube ruptured and reactor shutdown was accomplished without serious loss of coolant or significant external release of radioactivity.

In response to the safety concerns raised by steam generator tube degradation, the NRC Office of Nuclear Regulatory Research has sponsored a series of research programs on tube integrity and inspection. These programs have produced a number of significant results. For example, in the area of steam generator tube integrity, quantitative relationships have been developed for predicting the failure pressure of degraded steam generator tubing as a function of degradation type and severity. Relationships have also been developed for predicting the leak rates of through-wall cracked tubes as a function of flaw length and pressure. These relationships are useful for evaluating the integrity of degraded tubes under normal and accident conditions, and for evaluating various tube plugging criteria. In the area of inservice inspection, the probabilities of flaw detection and flaw sizing accuracy have been evaluated and quantified for inspection techniques used to detect wastage, pitting, and cracking. The effectiveness of various sampling plans for detecting significantly degraded tubes has also been evaluated. Performance demonstration criteria have been recommended for qualifying NDE systems used for detection and sizing of flaws. Finally, advanced multi-frequency eddy current procedures were developed, evaluated, and field tested.

The results obtained under these research programs have helped to ensure the safe operation of nuclear steam generators. However, in recent years, tube degradation modes have changed with intergranular stress corrosion cracking degradation becoming more widespread and the emergence of circumferential cracking. While the degradation modes have changed and the incidence of degradation has increased, inspection techniques have improved. These changes have resulted in the need for new methods to evaluate the significance of the various degradation modes to assure safe continued operation.

In view of the changing degradation experience and improved inspection techniques, the NRC plans to make significant changes in its regulatory criteria concerning steam generator tubing integrity by developing a performance-based rule and regulatory guide. This new rule will require that the performance or condition of steam generator tubes be monitored against safety related performance criteria in a manner sufficient to provide reasonable assurance that the steam generator tubes will maintain their structural and leakage integrity.

The new rule will give the licensee latitude in developing and implementing defect specific management based on NRC approved performance criteria. The NRC has recently initiated a new research program at Argonne National Laboratory to provide data and methodologies needed to independently evaluate and assess industry's proposals for implementation of the rule.

Integrity evaluations and assessments often revolve around calculations of potential leak rates under postulated accident conditions over the next operating cycle for degraded tubes remaining in service. To accomplish this, information is needed on inspection reliability, degradation processes, and on integrity of degraded tubes. Our research program is addressing each of these technical areas.

The NRC's research program includes efforts to assess inspection reliability by characterizing the probability of detection and sizing accuracy. Round robin testing will be conducted on a tube bundle mock-up containing laboratory degraded tubes using commercial teams and techniques and advanced or emerging techniques. These studies will be followed by round robin tests on service degraded tubes followed by destructive examination to validate the POD and sizing accuracy of practiced and emerging ISI techniques.

The research to predict degradation generation and progression in steam generator tubes, includes tests to bound environmental, loading, and metallurgical parameters that affect crack initiation, growth, and arrest. In addition, studies of crevice conditions as a function of bulk environment and thermal hydraulics will be conducted. Once the environmental and crevice chemistry parameters that have a significant effect on crack initiation and growth are identified, comprehensive studies will be conducted under prototypic conditions to quantify the effects and variabilities, and to gain an understanding of degradation and crack evolution. This information will provide a basis for assessing predicted inservice degradation of original and repaired tubes and for selecting appropriate models for predicting integrity at different stages in the progression of degradation.

To develop correlations for evaluating integrity and leakage of steam generator tubes, pressure and leak rate tests will be conducted under normal operating, accident, and severe accident conditions on tube specimens that contain laboratory produced axial and circumferential flaws that simulate cracks of current interest. Additional tests will be conducted on tubes removed from service to validate and update the integrity correlations.

Data, results, correlations, and models from the above studies will be used to assemble an integrated, mechanistically-based model for assessing predictions of failure, rupture, and total leak rate from inservice inspection results for a given steam generator under normal operating, accident, and severe accident conditions.

The research program I have just described represents the most recent in a series of NRC-sponsored programs that address the issue of steam generator safety. Ultimately, of course, we expect that the older trouble-prone steam generators will be replaced with more modern (and hopefully more trouble-free) designs and materials. However, approximately 90 PWRs around the world presently operate with mill-annealed Alloy 600 steam generator tubing.⁶ In many cases, strong economic incentives exist to extend the life of these units as long as possible, and at least some units are likely to be in service for many years to come. It is the responsibility of the NRC in the United States, and the corresponding regulatory bodies in other countries, to ensure that the continued operation of these older units, as well as the operation of newer steam generators, is carried out consistent with public health and safety. We at the NRC are eager to bring about international cooperation in research that will help all of us to accomplish this common goal. It is my hope that this Workshop will further enhance this cooperation.

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4 Plenary Session, Part 1: International Regulatory Practices and Issues

- 4.1 *Status of the Steam Generator Tube Circumferential ODSCC Degradation Experienced at the Doel 4 Plant*, G. Roussel, AVN
- 4.2 *French Regulatory Practice and Experience Feedback on Steam Generator Tube Integrity*, G. Sandon, J. Deschamps, and P. Chartier, DSIN/BCCN, and G. Turluer, IPSN/DES
- 4.3 *Recent Operating Experiences with Steam Generators in Japanese NPPs*, S. Yashima, JAPEIC
- 4.4 *Steam Generator Regulatory Practices and Issues in Spain*, C. Mendoza, C. Castelao, J. Ruiz-Colino, and J. M. Figueras, CSN
- 4.5 *Steam Generator Issues in the United States*, J. R. Strosnider, USNRC

STATUS OF THE STEAM GENERATOR TUBE CIRCUMFERENTIAL ODSCC DEGRADATION EXPERIENCED AT THE DOEL 4 PLANT

G. ROUSSEL
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ABSTRACT

Since the 1991 outage, the Doel Unit 4 nuclear power plant is known to be affected by circumferential outside diameter intergranular stress corrosion cracking at the hot leg tube expansion transition. Extensive non destructive examination inspections have shown the number of tubes affected by this problem as well as the size of the cracks to have been increasing for the three cycles up to 1993. As a result of the high percentage of tubes found non acceptable for continued service after the 1993 in-service inspection, about 1,700 mechanical sleeves were installed in the steam generators. During the 1994 outage, all the tubes sleeved during the 1993 outage were considered as potentially cracked to some extent at the upper hydraulic transition and were therefore not acceptable for continued service. They were subsequently repaired by laser welding. Furthermore all the tubes not sleeved during the 1993 outage were considered as not acceptable for continued service and were repaired by installing laser welded sleeves. During the 1995 outage, some unexpected degradation phenomena were evidenced in the sleeved tubes. This paper summarizes the status of the circumferential ODSCC experienced in the SG tubes of the Doel 4 plant as well as the other connected degradation phenomena.

1.0 BACKGROUND

1.1 Key information on the Doel 4 steam generators and related plant features

The Doel Unit 4 nuclear power plant started commercial operation in 1985. It is a 3-loop pressurized water reactor of 1,000 MWe output and has steam generators of Westinghouse Model E1 design. These steam generators are of the counterflow preheater design type with 100% of the feedwater flow entering the preheater (i.e., only very low flow going to the auxiliary feedwater nozzle for conditioning purposes). All of the 4,864 tubes of each steam generator are made of low temperature mill annealed Alloy 600 and have a 3/4 inch (19.05 mm) nominal outside diameter and 0.043 inch (1.09 mm) nominal wall thickness. The tube support plates are of the drilled plate style made of type 405 ferritic stainless steel. The flow distribution plate is of the slotted hole type made of type 405 ferritic stainless steel. The tubes are full tubesheet depth hard rolled, with kiss roll at the top of the tubesheet. The hot leg expanded sections of the tubes were treated by rotopeening prior to plant operation. Shotpeening of the cold leg was performed in April 1988 after three fuel cycles, but limited to the upper 120 mm (4.72 inch) of the expanded section.

Primary and secondary faces of the tubesheet are clad with Inconel 600 type weld metal. Hot leg temperature was originally 330°C (626°F) but it was changed to 326°C (617°F) in 1990. Secondary system has no copper alloy heat exchangers tubes but the condenser has a copper alloy tubesheet. Chemical cleaning was undertaken in 1992 on the three steam generators using a high temperature EPRI/SGOG process (iron step only). Again in 1993 chemical cleaning was performed, but limited to steam generator B, using KWU process with both an iron and a copper process. The secondary system originally operated with all-volatile treatment (AVT) chemistry but in 1992 the Doel 4 unit shifted to phosphate water chemistry, with a low molar ratio (Na/PO₄ : 2.0 to 2.2).

1.2 Summary of the SG tube degradations at the Doel 4 plant

Besides the circumferential ODSCC phenomenon, the steam generator tubes of the Doel 4 plant are mainly affected by outside diameter IGA/IGSCC at the tube support plate intersections and by outside diameter stress corrosion cracking in the free spans.

The Doel 4 steam generators have experienced at the tube support plate (TSP) intersections tube degradation which has been attributed to outside diameter intergranular attack and outside diameter intergranular stress corrosion cracking (IGA/IGSCC). This degradation was evidenced in the hot leg of steam generator B for the first time during the 1990 refuelling outage (i.e., after 5 cycles) by a leak observed on five tubes during the secondary side fluoresceine pressure test. The characterization of the degradation as IGA/IGSCC is supported by the laboratory metallographic examinations on tube sections pulled out during the 1991 and 1992 outages. This degradation most strongly affects steam generator B. This steam generator contains traces of lead, which are believed to come from lead shielding inadvertently left behind after maintenance in 1986. The steam generators R and G are less affected and degradations have only been detected in the hot leg. Bobbin coil inspections up to 1992 have shown the number of tubes with TSP indications in steam generator B to have increased and the degradation to have extended to higher TSPs in the hot leg. Since the 1993 outage, inspection has shown the degradation growth rate as measured by the bobbin coil voltage amplitude growth between two inspections to have decreased. Tube plugging was initially required in 1990 and at the end of the 1995 outage (i.e., after 10 cycles), the number of plugged tubes was 278. No tube has been required to be plugged in steam generators R and G.

The tubes in steam generator B were found to be affected since 1991 (i.e., after 6 cycles) by axial outside diameter stress corrosion cracking (ODSCC) in the free spans. Three of the tubes with indications at the TSPs which were pulled out during the 1991 outage for metallographic examination showed ODSCC in the free spans. These degradations had not been previously detected by the NDE inspection. During the 1992 outage, the free span defects were confirmed in the hot leg of steam generator B by the bobbin coil inspection. Laboratory examinations on tube sections pulled out during the 1991, 1992 and 1993 outages allowed to characterize the degradation as initiating from the secondary side and being intergranular. On most of the affected tubes, the degradation morphology is characterized by short (less than 1 mm) axial cracks. The number of cracks along a circumference is variable but it can range over several tens. Deepest observed cracks are maximum 40% through wall and are usually very fine and without branching. The ODSCC initially affected the hot leg of steam generator B but during the 1993 outage the degradation was found to have been extended in the cold leg. No tube plugging has been required up to the 1992 outage when 119 tubes in steam generator B were required to be plugged. Since the 1993 outage, inspection has shown the degradation growth rate as measured by the bobbin coil

voltage amplitude growth to have decreased. No tube has been required to be plugged since the 1992 outage.

1.3 Circumferential ODSCC degradation history up to the 1993 outage

Since the 1991 outage (i.e., after 6 cycles), it was found that tubes were affected by circumferential outside diameter intergranular stress corrosion cracking (ODSCC) at the hot leg tube expansion transition. The utility reported that all of the four tubes which were selected for pull out during this 1991 outage for purpose of laboratory examinations on tube sections with indications of IGA/IGSCC at the TSP were found to show these cracks. The circumferential cracks that had not been previously identified by the NDE inspection ranged in size from about 30% through wall to 100% through wall. During the 1991 outage, only steam generator B was found to have experienced circumferential ODSCC. Extensive NDE inspections since 1991 using Eddy Current (EC) Rotating Pancake Coil (RPC) probe and Ultrasonic (UT) rotating transducer have shown the number of tubes affected by this problem as well as the size of the cracks to have been increasing for the three cycles up to 1993. During the 1992 outage (i.e., after 7 cycles), the hot leg of steam generators R and B was also found to be affected and during the 1993 inspection, all the three steam generators were found to be affected to about the same amount. At the end of the 8th cycle (1993 outage), the circumferential ODSCC at roll transition affected about 50% of the tubes in each of the three steam generators.

During the 1991 and 1992 outages, 6 tubes in steam generator R, 11 tubes in steam generator G, and 33 tubes in steam generator B have been required to be plugged. During the 1993 outage (i.e., after 8 cycles), 962 tubes (19.8%) in steam generator R, 488 tubes (10.0%) in steam generator G, and 838 tubes (17.2%) in steam generator B were found to require plugging or repairing.

Laboratory examinations on tubes pulled out during the 1991, 1992 and 1993 outages allowed to characterize the circumferential crack mechanism as initiating from the outside surface and being intergranular. Circumferential ODSCC at the hard roll/kiss roll transition is located in a narrow band (typically 2 to 3 mm width) exhibiting a network of multiple cracks of very short length (microcracks) and variable orientation (from circumferential to axial). The whole circumference may be affected by cracking. The microcracks eventually aggregate to form longer (up to 360 degrees) and deeper (up to 100% through wall) cracks. However cracks may be fairly long in some cases without being through-wall. In steam generator B circumferential cracks are also found at the kiss roll/free span transition. The circumferential ODSCC at the kiss roll transition is characterized by multiple pure circumferential cracks extending all around the circumference with nearly constant depth. The crack depths range from a few percent through wall to about 30% through wall.

1.4 HEJ sleeving campaign in 1993

As a result of the high percentage of tubes found non acceptable for continued service after the 1993 in-service inspection, significant sleeving effort has been decided by the utility. During the 1993 outage, mechanical sleeves were installed in 815 tubes in steam generator R, in 309 tubes in steam generator G, and 614 tubes in steam generator B.

These sleeves are 36-inch or 30-inch Westinghouse tubesheet full length HEJ (Hybrid Expansion Joint) sleeves made of Thermally Treated Alloy 690. These sleeves extend the full length of the tube within the tubesheet. The lower sleeve-to-tube joint is located at the tubesheet bottom. It

consists of a section which is hydraulically expanded into the tube and a shorter section at the bottom of the hydraulically expanded zone which is roll expanded (hard rolled). This type of joint is known as an hybrid expansion joint (HEJ). The upper joint also consists of a section which is hydraulically expanded into the tube and a shorter section centered within the hydraulically expanded zone which is subsequently roll expanded. This joint is also referred to as an HEJ. Both lower and upper joints are designed to ensure the leak tightness and load carrying capability of the sleeve-to-tube attachment. The upper joint of the 36 inch sleeves is provided with a overlength (hop-off length or safe end) of 6.5 inch, i.e. a length of unexpanded sleeve outboard of the upper edge of the HEJ which should limit lateral displacement of the tube in the event of a circumferential break of the parent tube in the upper joint region. The hop-off length is limited to 0.5 inch in the 30 inch sleeve.

Approximately 92% of the HEJ sleeves installed in the steam generators were 36 inch sleeves and 8% were 30 inch sleeves. Following the installation of sleeves, the equivalent plugging level in steam generators R, G and B was 4.4%, 4.4% and 13.2% respectively at the end of the 1993 outage.

2.0 SG TUBE DEGRADATION PROBLEMS DURING THE 1993-1994 CYCLE

2.1 Circumferential ID cracking at the upper hydraulic transition of the HEJ sleeves

On March 17, 1994, six days prior to the scheduled refueling outage, the primary-to-secondary leak rate in steam generator G raised to about 5 kg/hr (31.9 gpd) as measured by the Nitrogen-16 monitors, while the unit was at 97% power. The primary-to-secondary leakage prior to the event was approximately 0.4 kg/hr (2.5 gpd). The leakage occurred 6 days prior to the scheduled refueling outage. The days after, the leak rate in steam generator G has been decreasing to about 1 kg/hr (6.4 gpd) but it raised again suddenly to about 21 kg/hr (133.8 gpd) on March 20, 1994 before it decreased to about 11 kg/hr (70.1 gpd). The reactor was shut down on March 23, 1994 at the planned date.

The leaking tube was identified by visual examination of the tubesheet during secondary side fluoresceine pressure test. A wet tube end was identified in the hot leg of steam generator G indicating apparent leakage from the secondary side. A HEJ sleeve had been installed in this tube during the 1993 outage. Re-performing the pressure test while adapting the level of water filling the secondary side allowed to locate the leak at the upper junction of the sleeve. The tube was pulled out and laboratory examinations showed that leakage occurred at the upper hydraulic transition of the upper joint where the tubes expanded during the installation of the sleeves return to normal diameter. The leak resulted from a circumferential through-wall crack in the parent tube extending 180 degrees around and initiating from the ID. Following the identification of the leakage cause, removal of another sleeved tube section from steam generator G (selection at random amongst the peripheral tubes) was performed to test the generic nature of the degradation. Upon examination it was found to have at the upper hydraulic expansion a circumferential crack 90% to 100% through wall and extending over 30 degrees around the parent tube.

Taking benefit of the NDE inspection of the sleeved tubes in the upper joint area to identify all the tubes affected by circumferential cracking at the hydraulic transition was out of the question. The NDE inspection methods previously used for the steam generator tube in-service inspection are

indeed known to have a limited effectiveness in the sleeve expansion joints and even worse in the parent tube wall behind the hop-off length.

The utility concluded that all the tubes sleeved during the 1993 outage with Westinghouse HEJ sleeves should be considered as potentially cracked to some extent at the upper hydraulic transition and were therefore not acceptable for continued service.

2.2 Evolution of the circumferential ODSCC at roll transition (hot leg)

The steam generator tube in-service inspection planned for the 1994 outage (i.e., after the 9th cycle) included the inspection of the roll transition of all the non-sleeved tubes in the hot leg of the three steam generators. This examination involved the use of the rotating UT transducer. The utility modified the original inspection scope during the outage as additional information gathering from the on going tube inspections became available. The final program was limited to 1,208 tubes in steam generator R (31.2% of the planned number), 4,356 tubes in steam generator G (100% of the planned number), and 80 tubes in steam generator B (2.1% of the planned number). As the UT inspection progressed in the steam generators, it was established that the defect growth for some tubes was faster than anticipated. The utility decided then to limit the UT inspection scope, to put an end to the inspection in fact, as the current results indicated that the observed ODSCC growth rates led to consider *all* the tubes as unacceptable for continued service.

The plugging limit for circumferential defects at roll transition is defined as the maximum acceptable cross sectional area of degraded material. It is based on the minimum cross sectional area of sound material to meet the required margin against bursting under accident conditions. This is calculated by the "collapse load" or "net section stress" theory. Such a plugging criterion assumes the availability of an inspection method allowing defect sizing. The morphology of the circumferential ODSCC at the roll transition makes difficult its detection by NDE techniques. The very small opening of the microcracks and their variable orientation are the main factors limiting the detectability of the circumferential ODSCC. Rotating UT transducer was found to ensure better detection than the RPC probe. Laboratory destructive examinations of pulled out tube sections showed that rotating UT transducer could detect defects deeper than 30% as long as the microcracks are nearly circumferentially oriented. Defect depth sizing with rotating UT transducer is based on a calibration curve determined from EDM notches. Comparing defect depths obtained by on-site UT measurements to the values given by laboratory destructive examinations of the same tube sections after removal from the steam generators led to conclude that defect depth sizing with rotating UT transducer can be underestimated. Moreover some tubes with confirmed leakage at the roll transition also showed UT measured defect depths lower than 100% through wall. The evidence of the underestimation of defect depth sizing with rotating UT transducer led the utility to define a correction factor to be applied to the UT measured defect depth to determine a conservative estimate of the defect depth, known as the corrected depth. During the 1993 outage, the available data at this date supported to allocate a value of 3.0 to the correction factor for any detected signal. The *corrected ODSCC degraded (cross sectional) area* takes into consideration an uniformly 30% through wall degraded area outside of the zones where defects have been detected by UT inspection and the corrected area of the UT detected defects. The corrected area of the UT detected defect is obtained by multiplying the UT measured length of the defect by its corrected depth (i.e., the UT measured depth multiplied by the correction factor). The corrected ODSCC degraded area is said to be obtained from the UT measured area by applying the 3.0/30% correction.

The ODSCC growth rate is obtained from the evolution of the corrected ODSCC degraded areas between two successive inspections. During the 1993 outage, the set of the available data from the last two inspections (1992 and 1993 outages) allowed the utility to conclude that the upper bound of the ODSCC growth is 50%, independently of the initial estimated ODSCC degraded area. At the end of the 1993 outage, the corrected end-of-cycle ODSCC degraded area was therefore obtained by multiplying the corrected begin-of-cycle ODSCC degraded area by 1.5.

The assurance of the structural integrity of an ODSCC degraded tube for the forthcoming cycle is demonstrated when the corrected end-of-cycle ODSCC degraded area is found smaller than the maximum acceptable cross sectional area of degraded material.

The results of the limited UT inspections performed during the 1993 outage on the three steam generators shows for a significant number of tubes ODSCC growth faster than anticipated. This unpredicted growth rate has been evidenced by (i) defects identified as acceptable for one cycle service at the end of the 1993 outage but not evidencing at the end of the 1993-1994 cycle the required margin to ensure the structural integrity (i.e., the corrected end-of-cycle ODSCC degraded area is larger than the maximum acceptable degraded area to ensure the tube structural integrity) and (ii) by defects non-identified by the 1993 outage UT inspection and identified as degraded by the 1994 outage inspection but not evidencing the required margin to ensure the structural integrity.

The utility concluded that *all* the tubes not sleeved during the 1993 outage could not be demonstrated as having the required margin to ensure the tube structural integrity during the forthcoming cycle and were therefore not acceptable for continued service.

3.0 HEJ REPAIRING AND LASER WELDED SLEEVING CAMPAIGN IN 1994

3.1. Elevated laser welded sleeves

Following the results of the in-service inspection, the utility decided to install Laser Welded Sleeves (LWS) in every remaining unsleeved tubes in order to bridge the roll transition area. These sleeves are Westinghouse 12-inch elevated laser welded sleeves made of Thermally Treated Alloy 690. This type of sleeve extends over one-quarter of the tube length within the tubesheet. The lower sleeve-to-tube joint is located approximately 17 inch above the tubesheet primary face and is of the HEJ type (identical to the lower joint of the 30- or 36-inch HEJ sleeves). The upper joint consists of a section which is hydraulically expanded into the tube and a autogenous weld between the sleeve and the tube which is made approximately in the middle of the hydraulically expanded zone using a laser welding process. This type of joint is referred to as a laser welded joint (LWJ). The utility reported that the tube diametrical expansion was limited to 0.075 mm in the upper joint instead of 0.25 mm as exhibited in the hydraulically expanded section of the HEJ sleeve upper joint. Should the weld be found unacceptable a reweld would be made about one-third of an inch under the initial weld. The upper joint is provided with a overlength of approximately 2 inch. In steam generator B, some 30-inch tubesheet laser welded sleeves (demonstration sleeves) were installed instead of elevated laser welded sleeves. The utility decided to stress relief (heat treat) the weld joint (weld and reweld, if present) and the upper transition of the tube upper hydraulic expansion. Holding temperature between 1,400°F and 1,600° (760°C and 871°C) for 5 minutes is required. The oscillating heater stress relief procedure was applied to both the weld and the upper transition. The oscillation stroke was 1.38 inch (35.05 mm) for both areas.

Of the installed 11,577 laser welded sleeves, 11,346 were available for active service. The number of tubes with 12-inch LWS accepted for service after the sleeving campaign was 3,649 in steam generator R, 4,183 in steam generator G and 3,365 in steam generator B. The number of tubes with 30-inch LWS accepted for service was 149 in steam generator B.

3.2 Repair of the 36 inch HEJ sleeves

The utility decided to repair the existing HEJ joints of all the 36 inch HEJ sleeves by performing in the available hop-off length a laser welded joint of the same configuration as the upper joint of the 12 inch laser welded sleeves. The utility also decided to heat treat the weld joint and the upper transition of the tube upper hydraulic expansion, using the same procedure as for the 12 inch elevated laser sleeves.

Of the repaired 1,628 HEJ sleeves, 1,088 were accepted for service. The number of tubes with repaired 36-inch HEJ sleeves accepted for service after the repair campaign was 609 in steam generator R, 127 in steam generator G and 352 in steam generator B.

3.3 Plugging of the 30 inch HEJ sleeves

The utility decided to plug all the tubes sleeved during the 1993 outage with 30-inch HEJ sleeves. He reported that due to the actual tolerances in tool positioning, the fruitfull performance of the weld joint in the short hop-off length of the 30 inch HEJ sleeves at an acceptable distance outboard of the potentially degraded area (upper hydraulic transition) was not sufficiently substantiated.

3.4 Summary of the plugged/repaired tubes after the 1994 outage

After the repair/sleeving campaign performed during the 1994 outage, the equivalent plugging level in steam generators R, G and B was 14.7%, 13.5% and 22.5% respectively.

4.0 SG TUBES 1995 IN-SERVICE INSPECTION

4.1 In-service inspection program

The final in-service inspection program required the 100% Cecco-3 inspection of all sleeved tube end sections remaining in service in all the three steam generators. The inspection zones (area of concern) in the sleeved tube end section are the parent tube above the weld, the weld (sleeve and parent tube) and the sleeve under the weld. It should be noted that a 0.3 inch length parent tubing section (7 to 8 mm) at the sleeve top end is not inspectable due to geometrical discontinuity problem. All indications evidenced by the Cecco-3 inspection were additionally inspected by Zetec "+ Point" probe. In addition, a random sample of 214 tubes (with no Cecco-3 indications) was also inspected by the Zetec "+ Point" probe in steam generator B.

The original in-service inspection program also included the removal of two tube sections with repaired HEJ sleeves and two tube sections with laser welded sleeves. Six sleeved tube sections were additionally removed from steam generator R.

4.2 1995 in-service inspection results

An eddy current inspection was performed in 1994 subsequent to the LWS installation and HEJ repair campaign using the Cecco-3 probe. The bobbin segment data were used for process verification, including detection of hydraulic and hard roll expansions, presence of the weld and performance of the two heat treatments. The Cecco segment (transmit-receive array) data were collected as a baseline archive.

Evaluation of the indications identified by the Cecco-3 inspection during the 1995 in-service inspection was based on the reconciliation with the baseline data obtained during the summer of 1994. The Cecco-3 baseline data represent the condition of the sleeve joint after the HEJ repair and LWS installation campaign. This condition was accepted on the basis of the process verification work performed in 1994.

The Cecco-3 indications found during the 1995 outage were therefore considered as acceptable at the condition that they showed no significant change relative to the 1994 baseline data. The evaluation of an indication against the baseline examination should include the evaluation of the circumferential length/pattern of the indication and the amplitude of the coil responses.

In steam generator R, one significant change was observed in the reweld of a tube and three significant changes were observed at the original HEJ joint (out of the area of concern). In steam generator G, no significant change was observed in the area of concern. In steam generator B, no significant change was observed in the area of concern but three significant changes were observed at the original HEJ joint (out of the area of concern).

A total of 20 tubes (8 in SG-R, 3 in SG-G and 9 in SG-B) had restrictions with the standard .530 inch probe (nominal ID of sleeve is .562 inch). With the .500 inch probe, 0 of the 8 tubes in SG-R, 0 of the 3 tubes in SG-G and 2 of the 9 tubes in SG-B had restriction. Restriction of the 2 tubes in steam generator B was confirmed by endoscopy.

The Zetec "+ Point" probe was primarily used by as a diagnosis tool intended to provide additional information on the LWS joints for which the Cecco-3 probe had identified indications. No acceptance criteria was defined for the "+ Point" indications.

In the random sample of 214 tubes (SG-B), no clear indication of circumferential or axial defect was evidenced by the Zetec "+ Point" probe in LWS joint. One doubtful indication was detected in the weld of a repaired 36-inch HEJ sleeve. No indication (neither clear nor doubtful) of circumferential or axial defect was evidenced in the HEJ joint of the eight repaired 36-inch HEJ sleeves included in the sample.

4.3 Tube removal

To meet the requirements set forth in 1994 in the authorisation for startup and operation of the Döel 4 plant, a minimum of two tube sections with 36-inch repaired HEJ sleeves and two tube sections with 12-inch LWS had to be removed during the 1995 outage.

Four tubes of steam generator B were selected for removal : two repaired 36-inch HEJ sleeves, one 30-inch LWS and one 12-inch LWS. The selection was based on the Cecco-3 indications confirmed by the results of the + Point inspection.

On none of the removed tubes circumferential corrosion cracking either in the weld or in the upper hydraulic transition was evidenced.

However the examination of the four tubes led to identify three unexpected phenomena :

(i) Bulging of the parent tube in the hydraulic expansion above the weld and also above the expansion zone. Three of the four tubes clearly suffered this bulging phenomenon. The maximum increase of the parent tube OD was found at the upper hydraulic expansion and reached 0.03 inch (0.8 mm). The bulging phenomenon was not evidenced by the NDE inspection.

(ii) Corrosion of the sleeve-tube assembly in a zone (360° extension and 5mm height) above the upper hydraulic expansion (HEJ joint) in one of the repaired HEJ sleeve. The corrosion was found on both the ID of the parent tube (penetration less than 10% thru-wall) and the OD of the sleeve (25% thru-wall). The corrosion at the sleeve OD is general corrosion combined with IGA. The corrosion at the parent tube ID is general corrosion. The general corrosion of the sleeve-tube assembly above the upper hydraulic expansion (HEJ joint) of the repaired HEJ sleeve was identified by the Cecco-3 probe as a circumferential indication.

(iii) Circumferential cracks in the weld of the 36-inch HEJ sleeved tube showing a circumferential indication at the weld. The longest crack was found to extend over about 120° and to be thru-wall. The cracked area is about 50% of the cross sectional area of the weld. The cracks initiated from the sleeve OD. Metallographic examination of the crack allowed the utility to conclude that the crack was a fabrication defect. No propagation either by corrosion or by fatigue was evidenced.

Fabrication cracks were known to potentially affect the welds, specially these of the HEJ repaired sleeves. However based on tests performed in 1994, the cracks were believed to be small cracks not questioning the structural integrity of the sleeve/tube assembly. The Cecco-3 inspection guidelines for the weld were defined to identify the small fabrication cracks. The circumferential indication in the weld of the 36-inch HEJ sleeved tube was detected by the Cecco-3 probe both in 1994 and 1995 but the signal was not interpreted as a crack.

The utility accepted to remove six additional sleeved tubes from steam generator R. The results of the laboratory examination performed on the pulled tubes of the second series can be summarized as follows :

(i) None of the removed tubes exhibited circumferential corrosion cracking either in the weld or in the upper hydraulic transition.

(ii) The bulging phenomenon of the parent tube above the weld was found on all the pulled tubes and was so confirmed to be a generic problem. None of the tubes exhibited however stress corrosion in the bulged region.

(iii) The tube of steam generator R showing a significant Cecco-3 change in the reweld of a tube only exhibited fabrication cracks in the weld region without any in-service growth.

(iv) General corrosion of the sleeve-tube assembly at the upper hydraulic transition of the HEJ joint was confirmed to be a potential problem for the 36 inch repaired HEJ sleeves.

4.4 Summary of the revised inspection results

Following the laboratory examinations performed on the pulled tubes and the subsequent revision of the inspection guidelines, the Cecco-3 indications were reanalyzed to identify those indications corresponding to fabrication cracks in the weld region and general corrosion at the upper hydraulic expansion (HEJ joint) of the repaired HEJ sleeves.

The following table summarizes the results of this reanalysis.

Number of tubes with indications of crack and general corrosion

	SG-R	SG-G	SG-B
<u>Clear and doubtful crack indications in weld region</u>			
12 inch LWS	6	5	22
30 inch LWS			1
36 inch repaired HEJ sleeves	79	6	32
<u>General corrosion at upper HE (HEJ joint)</u>			
36 inch repaired HEJ sleeves	8	0	8
<u>Crack indication in parent tube at upper HE (HEJ joint)</u>			
36 inch repaired HEJ sleeves	11	4	6

The utility decided to plug *all* the repaired HEJ sleeves as well as the LWS sleeves with crack indication in weld region.

4.5 Summary of the plugged/repaired tubes after the 1995 outage

At the end of the 1995 outage, the equivalent plugging level in steam generators R, G and B was about 26.7%, 16.1% and 30.1% respectively.

5.0 CONCLUSION

The Doel 4 nuclear power plant will have its steam generators replaced in an outage planned for summer 1996. The decision for replacement has been taken by the utility in 1992 at time when the SG tubes were mainly affected by outside diameter IGA/IGSCC at the tube support plate intersections and axial ODSCC in the free spans. The present report only aims at giving the status of the circumferential ODSCC at the hot leg tube expansion transition. It shows the unexpected flaw growth and the unexpected degradation phenomena connected to the repair processes.

Other important aspects such as the qualification of the repair processes, the qualification of the NDE inspection techniques, and the conditions for authorizing start-up and operation were deliberately not covered in this report.



CNRA WORKSHOP - ARGONNE - ILLINOIS



30th October to 2nd November 1995

1/23

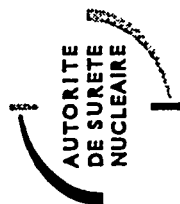
FRENCH REGULATORY PRACTICE AND EXPERIENCE FEEDBACK ON STEAM GENERATOR TUBE INTEGRITY

BY

G. SANDON French Safety Authority

PLAN☐ INTRODUCTION☐ TYPES OF DEGRADATIONS☐ METHODOLOGY☐ TEST METHODS USED☐ SURVEILLANCE PROGRAM☐ PREVENTIVE AND CORRECTIVE MEASURES☐ CONCLUSION

INTRODUCTION



3/23

IN FRANCE :

➤ 54 PWR OPERATED BY ONE UTILITY (E.D.F)

34 PWR 900 MWe

20 PWR 1300 MWe

➤ PRODUCING 80 % OF THE FRENCH ELECTRICITY CONSUMPTION

➤ THIS REPRESENTS OVER 180 STEAM GENERATORS AND ABOUT 800,000 TUBES

CNRA - Oct / Nov 1995

DSIN/95-9668/PhE

TYPES OF TUBE DEGRADATIONS

THE MAIN CONCERN

→ PRIMARY WATER STRESS CORROSION CRACKING, (PWSCC)

➤ AXIAL AND CIRCUMFERENTIAL CRACKS

- AT ROLL TRANSITION ZONE
- IN SMALL RADIUS U-BENDS (ROWS 1 AND 2)

THE SECOND ONE

→ SECONDARY SIDE CORROSION

➤ IGA AND CRACKS OF IGSCC (NETWORK OR ISOLATED)

- INITIATED IN CONCENTRATION ZONE

SLUDGE ZONE

UNDER CREVICES OF TUBE SHEET OR TUBE SUPPORT PLATES

OTHERS

→ VIBRATION FATIGUE CRACKS (MITIGATION OF 800 TUBES)

→ WEAR CAUSED BY

- LOOSE PARTS ABOVE TUBE SHEET
- ANTIVIBRATION BARS

→ DEFORMATION

- IN SLUDGE ZONE
- AT UPPER TUBE SUPPORT PLATE INTERSECTIONS

OTHERS SG DEGRADATIONS



5/23

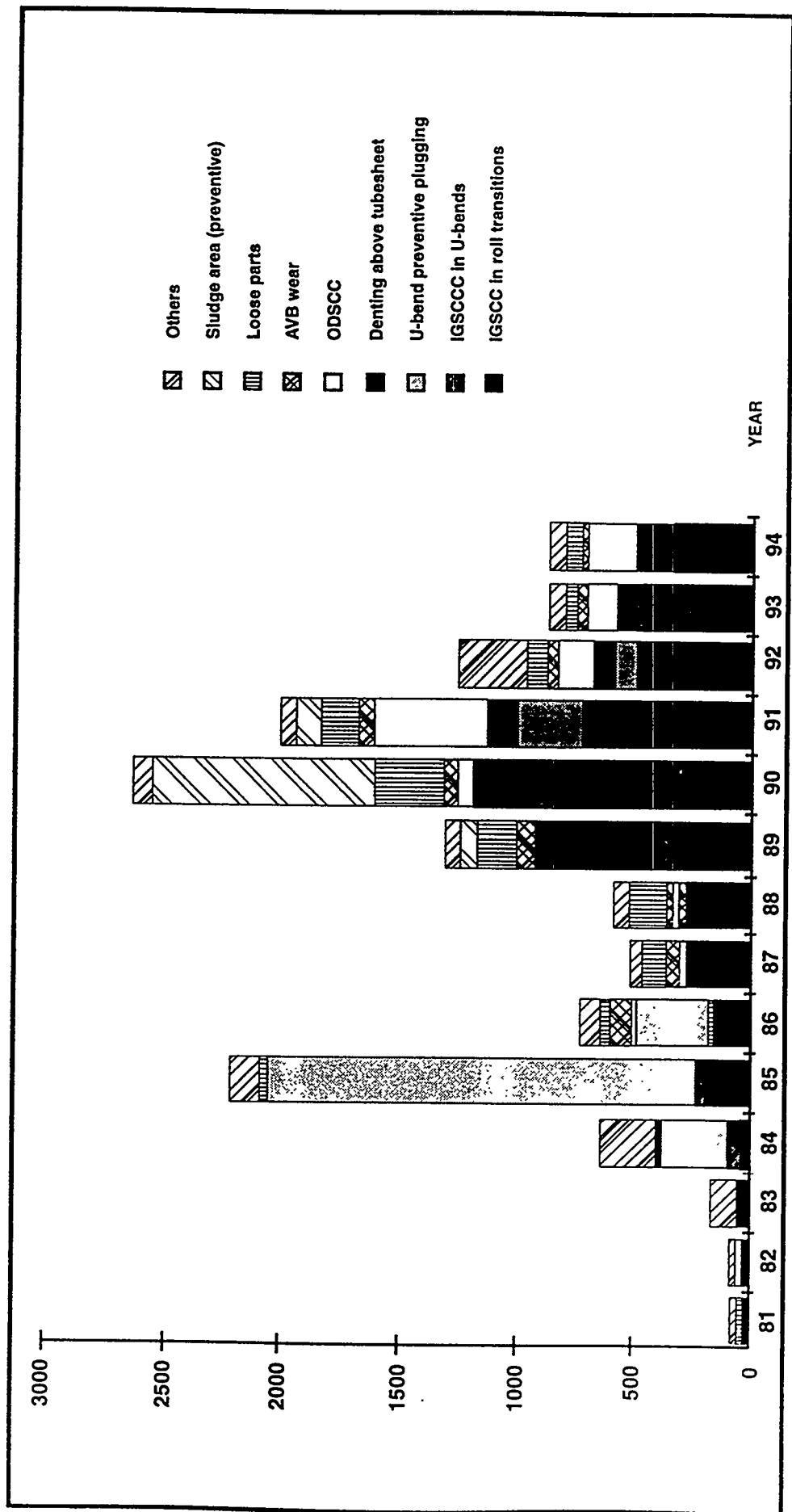
→ DISPLACEMENT OF THE BUNDLE WRAPPER

- OBSERVED IN JUNE 1994 ON 2 STEAM GENERATORS
OF BLAYAIS 3

→ DAMAGE TO TUBE SUPPORT PLATES

- OBSERVED RECENTLY ON SOME STEAM GENERATORS
MODEL 51A AND 51M

NUMBER OF PLUGGED TUBES PER YEAR



METHODOLOGY



▷ THE GOAL = AVOID STEAM GENERATOR TUBE RUPTURE

➤ USING A DETERMINISTIC APPROACH

BASED ON

UNDERSTANDING DEGRADATION PHENOMENON

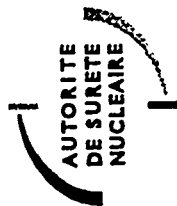
ANTICIPATION USING PREVENTIVE AND CORRECTIVE MEASURES

METHODOLOGY

▷ THE MEANS

- ☐ PERMANENT IN SERVICE MONITORING
- ☐ IN SERVICE INSPECTION DURING OUTAGES
- ☐ PREVENTIVE AND CORRECTIVE MEASURES

METHODOLOGY



BCCN

9/23

▷ FUNDAMENTAL PRINCIPLE

PLUGGING THE TUBES FOR WHICH CRITICAL SIZE OR MORPHOLOGY
OF DEFECT WOULD BE OBTAINED AFTER GROWTH DURING
THE NEXT CYCLE OF OPERATION

METHODOLOGY

 THIS REQUIRES A SOUND KNOWLEDGE OF

▷ THE MORPHOLOGY OF THE DEFECTS

- NDE CAPABILITIES ?
- LEAK BEFORE BREAK CONCEPT ?

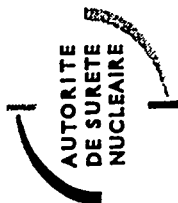
▷ THE GROWTH RATE

- ISI POLICY AND FREQUENCY

▷ DEGRADATION MECHANISMS

- PREVENTIVE AND CORRECTIVE MEASURES

METHODOLOGY



**NO PERMANENT SET RULE BUT ONLY THE
PROCESS OF DEVELOPMENT OF THE RULES**

**FOR THAT REASON A YEARLY REVISION OF THE POLICY
AND IN SERVICE INSPECTION PROGRAM IS MADE AFTER
ANALYSIS OF OPERATING EXPERIENCE FEEDBACK**

TEST METHODS USED



12/23

NUREG/CP-0154

METHODS	PERFORMANCES
▷ BOBBIN COIL PROBE BASIC INSPECTION IGA - WEAR WHOLE LENGTH	• WEAR DEPTH $\geq 20\%$ WALL THICKNESS • AXIAL - CRACKS LENGTH ≥ 5 mm • CIRCUMFERENTIAL CRACKS NO DETECTION
▷ ROTATING PANCAKE COIL PROBE IGSCC ROLL TRANSITION ZONE IGSCC SMALL RADIUS U-BENDS IGA AND ODSCC	• DEFORMATION $\Delta r \geq 70$ microns • AXIAL CRACKS LENGTH ≥ 5mm • CIRCUMFERENTIAL CRACKS ANGULAR EXTENSION $> 50^\circ$ DEPTH ≥ 600 microns
▷ OTHERS ULTRASONIC TESTING HELIUM TESTS VIDEO INSPECTION	• DEFORMATION $\Delta r \geq 70$ microns • U. BEND CRACKS LENGTH ≥ 6mm DEPTH ≥ 600 microns USED FOR SPECIFIC EXAMINATIONS SMALL THROUGH WALL DEFECTS USED FOR SPECIFIC COMPLEMENTARY EXAMINATIONS

CNRA - Oct / Nov 1995

DSIN/95-9677/PhE

TEST METHODS USED

▷ THE FRENCH SAFETY AUTHORITY INSISTS ON

→ QUALIFICATION

- EXAMINATION OF PULLED TUBES (MORE THAN 300 TUBES)
- QUALITY AND RELIABILITY IN THE INTERPRETATION OF SIGNALS (AUTOMATION)

→ ANTICIPATION

- DEVELOPMENT OF NEW METHODS

SURVEILLANCE PROGRAM

▷ IN SERVICE LEAK MONITORING - **METHODS**

- MEASUREMENT OF ACTIVITY OF GASES EXTRACTED FROM
THE CONDENSOR
- MEASUREMENT OF WATER ACTIVITY IN THE BLOWDOWN
SYSTEM OF EACH STEAM GENERATOR
- MEASUREMENT OF ACTIVITY DUE TO NITROGEN 16 IN
THE STEAM LINE OF EACH STEAM GENERATOR

SURVEILLANCE PROGRAM



15/23

▷ IN SERVICE LEAK MONITORING - METHODOLOGY

→ FLAWS OBSERVED ON TUBES DO NOT ALLOW TO FULLY TRUST LEAK BEFORE

BREAK CONCEPT

- REMOVED TUBES SHOWED SMALLER LEAKAGE THAN THEORETICAL RESULTS
- BORIC ACID, LITHIUM AND SLUDGE MIGHT PARTIALLY BLOCK UP THE CRACKS

→ PREVENTION OF STEAM GENERATOR TUBE RUPTURE CANNOT RELY ONLY ON

LEAK MONITORING

→ HOWEVER, THE EARLY DETECTION OF INCREASING LEAKS IS PROBABLE ENOUGH
TO MAKE LEAK MONITORING AN ESSENTIAL COMPONENT OF DEFENSE IN DEPTH

SURVEILLANCE PROGRAM

▷ IN SERVICE LEAK MONITORING - SHUTDOWN CRITERIA

- NEW RULES SINCE JULY 1995 BASED ON NITROGEN 16 DETECTION
- LEAK AS LOW AS POSSIBLE DURING NORMAL OPERATION
- IMMEDIATE SHUTDOWN IF THE N16 DETECTOR INDICATES
 - A LEAK ≥ 20 l/h
 - A LEAK INCREASING > 3 l/h in 24 hours
 - A DIFFERENCE OF LEAKS BETWEEN 2 SG > 3 l/h

SURVEILLANCE PROGRAM

▷ IN SERVICE LEAK MONITORING - SHUTDOWN CRITERIA

→ SHUTDOWN WITHIN A WEEK IF MANUAL MEASUREMENTS CONFIRM A LEAK

DETECTED BY N16 GREATER THAN A FEW LITRES PER HOUR

- 10 l/h FOR THE MOST DEGRADED SG (14 plants)
- 3 l/h FOR SG WITH THE FEWEST DEGRADED TUBES (40 plants)

SURVEILLANCE PROGRAM

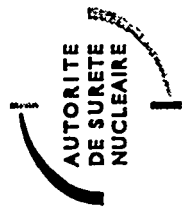
▷ IN SERVICE INSPECTION - DURING OUTAGE

FREQUENCY	SG 900 MWe		SG 1300 MWe
	51 A - 51 M	51 B	
BOBBIN COIL PROBE BASIC PROGRAM SLUDGE ZONE TSP INTERSECTION AVB WEARS TUBES WITH DEFECT	100 % / 2 cycles 100 % / cycle 100 % / 2 cycles 100 % / 2 cycles EVERY YEAR	25 % / cycle 100 % / cycle after 60,000 h — 50 % / cycle EVERY YEAR	25 % / cycle — — 25 % / cycle EVERY YEAR
ROTATING PROBE BASIC PROGRAM (RTZ) SLUDGE PILE (RTZ) SMALL U-BEND R1 TUBES WITH DEFECT	100 % / cycle 100 % / cycle 100 % / cycle EVERY YEAR	25 % / cycle 100 % / cycle 100 % / cycle EVERY YEAR	25 % / cycle 100 % / cycle RANDOM EVERY YEAR
HELIUM TEST	EVERY TEN YEARS OR WHEN LEAK P/s ≥ 3 L / Hour		

SURVEILLANCE PROGRAM

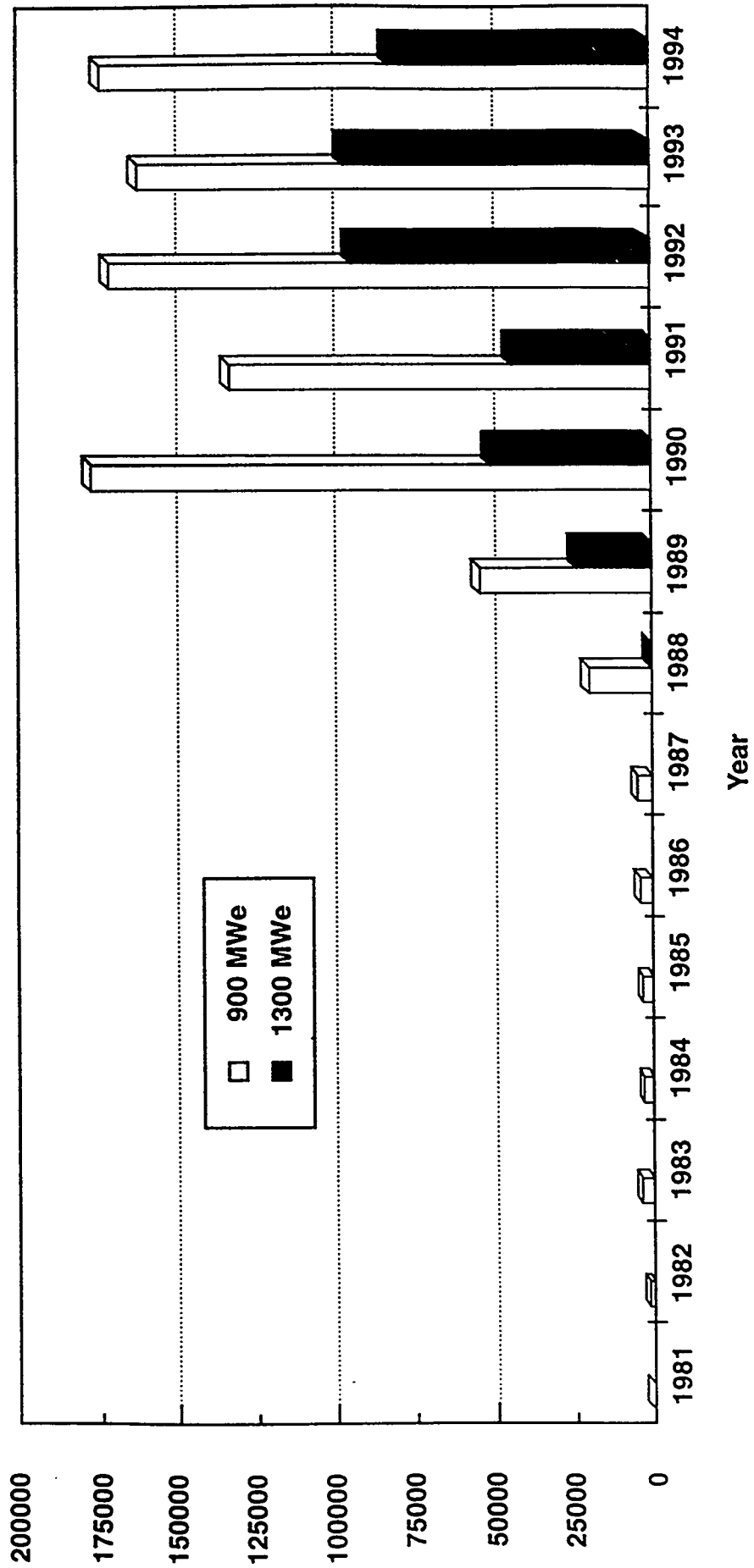


19/23



▷ IN SERVICE INSPECTION - DURING OUTAGE

E.C. RPC INSPECTIONS / NUMBER OF INSPECTED TUBES (P.S.I. NOT INCLUDED)



IN SERVICE INSPECTION - DURING OUTAGE
PLUGGING CRITERIA

if degradation	Non destructive testing (*)	Frequencies	Critical size	Kinetics	Plugging criteria	Comments
nsition zone	MRPC	100 % every outage	(1) 16,5 mm (2) 360° - 80 % thickness 280° - through wall	2,5 mm / cycle ?	13 mm All tubes affected	(1) axial cracks (2) circumferential cracks
sealed tubes kiss-rolling	MRPC	100 % every outage	(1) 16,5 mm (2) 360° - 80 % thickness 280° - through wall	2,5 mm / cycle ?	13 mm All tubes affected	
with kiss rolling inealed	BC + MRPC	25 % every outage 100 % sludge pile every outage (3)	(1) 16,5 mm (900 MWe) (1) 14,5 mm (1300 MWe) (2) 360° - 80 % t 280° - through wall	2,5 mm / cycle Lower than 2,5 mm / cycle ?	13 mm Length < 9 mm in 1995 All tubes affected	(3) after 60,000 operating hours
nsition zone	BC + MRPC	100 % every other outage	860 mV (70 % of thickness)	250 mV	500 mV 200 mV (4)	(4) If longitudinal crack close to the ODSCC zone
inealed tubes	BC	25 % every outage				
sealed tubes	BC	100 % every outage	6 V (50 % of thickness)	2 V	2 V 1 V	For TSP number 7 and 8
tube support intersections + IGA	FRPC	100 % every outage Row 1 or 2	?	?	All tubes affected	
h drilled holes	BC	100 % every outage on peripheral tubes	75 % of thickness	15 % max / cycle	40 % All indications	When loose part can't be removed

11/ Nov 1995

(*) BC = Bobbin Coil

MRPC = Motor Rotating Pancake Coil

FRPC = Flexible Rotating Pancake Coil

DSIN/95-9685/PHE

PREVENTIVE MEASURES



21/23

ON OPERATING STEAM GENERATORS

- IN ROLL TRANSITION ZONE
 - SHOT PEENING
 - SLUDGE CLEANING DURING EVERY OUTAGE
 - LOWER PRIMARY TEMPERATURE (1300 MWe)
 - INVENTORY AND PLUGGING OF ALL TUBES WITH A ROLL TRANSITION DOUBLE ANOMALY
- IN SMALL RADIUS U-BENDS
 - HEAT TREATMENT FOR THE BEND SECTION
 - PREVENTIVE PLUGGING OF ALL SMALL RADIUS U-BENDS FROM WESTINGHOUSE
- IGA ON SECONDARY SIDE
 - CHEMICAL CLEANING
 - INJECTION OF BORIC ACID

ON NEW STEAM GENERATORS

- TUBES MADE OF HEAT TREATED ALLOY 690 (SINCE 1990)
- TUBE SUPPORT PLATES MADE OF STAINLESS STEEL AND WITH BROACHED HOLES
- HYDRAULIC EXPANSION IN THE TUBE SHEET

CORRECTIVE MEASURES

PLUGGING

LESS THAN 1000 TUBES WERE PLUGGED IN 1994

SLEEVING IS NO LONGER USED

ONLY 12 SLEEVES ARE STILL IN OPERATION AT TRICASTIN 2

SG REPLACEMENT

- DAMPIERRE 1 IN 1990
- BUGEY 5 IN 1993
- GRAVELINES 1 IN 1994
- ST LAURENT B1 } IN 1995
- DAMPIERRE 3 }
- ONE PER YEAR UNTIL 2000 FOR THE MOST AFFECTED SG

→ THE CHOICE IS MADE BY THE OPERATOR ON A TECHNICO - ECONOMICAL BASIS

CONCLUSION

▷ THE FRENCH SAFETY AUTHORITY ALWAYS PAYS GREAT ATTENTION
TO ALL THE EFFORTS EDF IS MAKING IN ORDER TO :

- UNDERSTAND EACH MECHANISM OF DEGRADATION
- EVALUATE THEIR GROWTH RATE
- ELABORATE A NEW PROGRAM OF INSPECTION EVERY YEAR
- DEVELOP AND QUALIFY NDE METHODS
- USE PREVENTIVE AND CORRECTIVE MEASURES
(THE BEST ANTICIPATION = SG REPLACEMENTS)

OECD/NEA - CNRA/CNSI
International Workshop on
Steam Generator Tube Integrity in NPPs

Recent Operating Experiences with
Steam Generators in Japanese NPPs

Seiji Yashima
Japan Power Engineering and Inspection Corp.
Tokyo, Japan

30 Oct. - 2 Nov., 1995
Argonne National Laboratory
Chicago, Ill., U.S.A.

1. Introduction

In 1994, the Genkai-3 of Kyushu Electric Power Co., Inc. and the Ikata-3 of Shikoku Electric Power Co., Inc. started commercial operation, and now 22 PWR plants are being operated in Japan. Since the first PWR plant was started to operate, Japanese PWR plants have had an operating experience of approx. 280 reactor-years.

During that period, many tube degradations have been experienced in steam generators (SGs). And, in 1991, the steam generator tube rupture (SGTR) occurred in the Mihama-2 of Kansai Electric Power Co., Inc. However, the occurrence of tube degradation of SGs has been decreased by the instructions of the MITI as regulatory authorities, efforts of Electric Utilities, and technical support from the SG manufacturers.

Here we describe the recent SGs in Japan about the following points.

- * Recent Operating Experiences
- * Lessons learned from Mihama-2 SGTR
- * SG replacement
- * Safety Regulations on SG
- * Research and development on SG

2. Recent Operating Experiences

2.1 Tube Degradation History

As of Oct. 1995, 22 PWR plants are being operated and the total number of SGs is 64. Since the Mihama-1 as the first PWR plant was started to operate in 1970, it has accumulated operating experience of approx. 280 reactor years. During that period, as shown in Fig. 1, mainly the tube degradations as indicated below have been experienced.

First half of 1970's	Wastage (OD)
Latter half of 1970's	TS Crevice SCC (OD)
	U-Bend SCC (ID)
1980 to date	TS/TSPL IGA (OD)
	TS SCC (ID)
	AVB Fretting (OD)
	Pitting (OD)

Many tube degradations were found in SGs when the plants were inspected during outage due to incidents and failures that must be reported from Electric Utilites to the MITI by law. As shown in Table 1, they occurred 5 to 9 times every year since 1982. However, the replacement of SGs with many tube degradations was started in 1993 and thus the number of failure reports on SGs in 1994 was reduced to only three. It is expected that it will be reduced further in the future.

2.2 Recent Topics

2.2.1 SG Tube Leakage at Mihama-1

- (1) The Mihama-1 is the first PWR plant in Japan that was started to operate in Nov. 1970. It has two SGs of Combustion Engineering Inc. (CE)-type, their tube material is Alloy 600 MA, and upper tube bundle support has a specific structure different

from that of WH-type.

- (2) In Feb. 18, 1994 when the plant was operated at full power, leakage from the primary system to secondary system was detected slightly, and the plant was shutdown. As the result of Helium Gas Leak Test, it was confirmed that the leakage occurred from one tube near the edge of the scalloped bar at the horizontal portion of the upper U-bend. The schematic of the leaked portion is shown in Fig. 2.
- (3) The cause of the leakage was investigated and the following was found.
 - a. As the result of ECT using Bobbin Coil, any significant indication was not detected at the defected area.
 - b. As the result of 8x1 ECT and Rotating Pancake Coil ECT, the circumferential indication on the O.D. surface was detected at the outer edge of the scalloped bar, and its depth of indication was almost 100% through wall.
 - c. As the result of visual inspection of the upper tube support structure, no trace of sliding was noticed at the sliding portions of the upper tube support structure assembly and the wrapper. Therefore it was considered that, during the operation, the tube bundle and wrapper were positioned in such a state that their thermal expansion difference could not be absorbed.
 - d. As the result of measurement of the tube bend, the leaked tube was noticed to be lifted up and bent at the scalloped bar portion due to accumulation of sludge.
 - e. As the result of investigation of the secondary water chemistry history, the chemical environment was acid in the past and, at present, is neutral to weak acid.
 - f. As the result of vibration evaluation, it could not be

considered that it was damaged by fluid elastic vibration.

- (4) As the result of stress evaluation, it was noticed that an axial stress of approx. 22 to 28 kg/mm² occurred on the leaked tube and also corrosive environment was formed on the secondary side and, thus it was estimated that SCC occurred on the outer surface of the tube.
- (5) The SGs of the Mihama-1 unit were planned to be replaced with those of advanced type and the replacement work was planned to be started from the end of 1995. However the schedule was advanced and the work is now under way, and expected to be completed at the end of this year.

2.2.2 SG Tube Leakage at Ohi-2

- (1) The Ohi-2 plant is the PWR plant with 4-loops. The SGs of it were manufactured by MHI, and their tubes of Alloy 600 MA were manufactured by Sumitomo Metal.
- (2) In Feb. 25, 1995 when the plant was being operated at full power, alarms for N-16 leak monitor and others set in the B-main steam line were dispatched, and the operation of the plant was stopped. To locate the leaked tube by the secondary side hydraulic pressure test, water was filled to the secondary side of the SG and leakage was found from the cold leg side of one of the Row-1 tubes.
- (3) As the results of ECT on the leaked tube, the following was obtained.
 - a. As the result of ECT using the bobbin coil type U-bend probe, an indication was detected at the cold leg bend portion.
 - b. As the result of ECT using the rotating pancake coil probe, an axial indication of approx. 30 mm was detected at the cold leg bend portion.
- (4) As the results of examination by cutting off of the leaked tube,

the following was obtained.

- a. A line of short axial intergranular cracks were found near the flank of the cold leg by approx. 17 mm in the the outer surface and approx. 40 mm in the inner surface.
- b. The ovality was 6% maximum, but it was noticed that it was varied at the cracked portion in longitudinal direction.

(5) Results of stress evaluation

It was evaluated that the residual stress by bending was high, and thus the PWSCC was considered to be produced by that stress together with the working stress during the operation.

- (6) To prevent this, all the Row-1 tubes were plugged. Also the SGs in which tubing of Alloy 600 MA is used and of which U-bend tubes are not stress-relieved are to be inspected for all Row-1 and -2 U-bends with a plus point type rotating probe. In the Ohi-2 plant, it is planned to replace the SG in 1997.

2.2.3 Others

Mechanical plug degradation and degradation at sleeved tubes that were experienced overseas have never been experienced in Japan.

For the broach hole blockage of a broached type tube support plate, as the result of ECT, an indication showing the adhesion of scale was noticed on some SGs.

3. Lessons Learned from Mihama-2 SGTR

In the Mihama-2 plant in which SGTR occurred in Feb. 9, 1991, the SGs were replaced and the preventive measure against re-occurrence of SGTR was taken and, in Oct. 13, 1994, the operation was re-started. This SGTR gave us many lessons useful to maintain the safety of NPPs.

3.1 Preventive measures against re-occurrence by MITI

To prevent the re-occurrence of SGTR, the MITI will take the measures described below based on the lessons so as to further improve the safety regulations.

- (1) AVB will be added to examination items of construction plan and inspection.
 - a. Items on AVB will be added for the examination of construction plan and pre-use inspection.
 - b. Structural standards on AVB will be added to the technical standards.
- (2) Improvement in periodical inspection
 - a. Inspection to check AVB insertion and instalation conditions will be added.
 - b. Inspection to check deformation of SG TSPL portion will be added.
 - c. System integrity confirmation inspection before starting reactor and hot function inspection for main steam isolation valves (MSIVs) will be added.
 - d. Ideal periodical inspection for aging plants will be considered (under consideration).
- (3) Improvement in safety evaluation

The safety evaluation on SGTR will be changed to the evaluation in which the time taken to complete the closing of the MSIVs is

extended.

- (4) Reinforcement of guidance and supervision on quality assurance activities

At each stage ranging from design to operation, guidance and supervision on the quality assurance activities of Electric Utilities will be further reinforced.

- (5) The classification of the abnormal conditions in safety review will be studied.

The classification of the abnormal conditions including the SGTR in safety review will be studied.

- (6) Contents of the installation permission application will be reviewed.

The standard application manual will be prepared for the instruction to Electric Utilities.

- (7) Technological advancement, accumulated experiences, and others will be reflected properly.

Periodic safety review (PSR) will evaluate reflection of operational experience and advanced technical knowledge.

- (8) Plant operation and maintenance standard (POMS) will be reviewed.

a. The aging of plant will be evaluated properly and POMS to maintain the integrity will be arranged (under study).

b. Based on the aging of plant, how to perform periodical inspection will be studied (under study).

- (9) Safety research will be further promoted.

a. Validation test on fluid induced vibration by two phase flow will be performed (under test).

b. Reliability evaluation method and aging evaluation method of materials and equipment will be developed (under development).

c. Advanced inspection techniques will be developed (under

development).

3.2 Preventive measures against re-occurrence by Electric Utilities

Electric Utilities took the measures described below based on the instruction from the MITI.

- (1) Reinforcement of autonomous safety assurance
 - a. Reinforcement of quality assurance activities
 - b. Improvement in maintenance control method
 - c. Improvement in operation manual
 - d. Improvement in monitoring system and instrument and control system.
- (2) Promotion of technological development
 - a. Development of advanced inspection techniques
 - b. Development of human error preventing techniques
 - c. Improvement in fluid induced vibration evaluation method
- (3) Improvement in emergency response to occurrence of abnormalities
 - a. Improvement in information communication system
 - b. Prompt and appropriate assistance to visitors to NPP
 - c. Prompt estimation of radiation release
- (4) Measures to enhance the reliability of nuclear power generation
 - a. Scheduled replacement of AVB
 - b. Reinforcement of monitoring of SG leakage

3.3 Preventive Measures against re-occurrence in the Mihama-2

In the Mihama-2, the following measures were taken before re-starting of the operation.

- (1) In light of economy and the general public, the old SGs were replaced with new ones of advance type.
- (2) N-16 radiation monitors (Scintillation detectors) were installed in each main steam piping so that SG leakage could

be detected rapidly and securely and also be located.

- (3) Monitors (High range: Ionization chamber detectors, Low range: Scintillation detectors) were installed in each main steam piping so that radiation release could be estimated in the occurrence of SG leakage.
- (4) Thermometer type reactor water level gauge was installed so as to diversify the monitoring methods of plant parameters during accident occurrence.
- (5) To improve the closing function of MSIVs, actuators were improved.
- (6) To increase the reliability of the working air system of the pressurizer relief valve, the instrument air system was doubled and set independently.
- (7) The processing capability of the plant computers was increased.
- (8) Even when safety injection signal is generated, a line capable of supplying power directly to the start up transformer was newly connected to the common bus bar.

4. SG Replacement

4.1 Background of SG Replacement

The integrity of SGs is maintained through strict inspections and appropriate repairs. Also the safety of the plant is secured by continuous leak monitoring during the operation. In Japan, however, not only extremely minute leakage of primary coolant to the secondary side due to degradation of tubes but also the detection of degradation of tubes by inspection make the general public anxious about safety and reliability of nuclear power generation.

At the same time, through the inspections and repairs of SGs, the outage period of the plant was prolonged and direct and indirect costs were increased resulting in economical inefficiency.

On the other hand, as the result of R&D based on the past experiences, advanced type SGs were developed, and also the replacement technique was improved through the experiences of replacement performed overseas. Therefore the replacement of SGs became reasonable.

4.2 Operation of SG Replacement

The SG replacement is performed by Electric Utilities to maintain the safety, maintainability, and social reliability of NPPs when they judge that it is desirable to replace the SGs causing tube degradation frequently and those designed and manufactured in the same period as the degraded SGs with those of advance type.

So far the SG replacement has been completed in the four plants: the Takahama-2, Mihama-2, Genkai-1, and Ohi-1, and under way in the Mihama-1 plant.

In addition, the SG replacement is planned in the three plants: the Takahama-1, Mihama-3, and Ohi-2. The list of the completed and

planned SG replacements is shown in Table 2.

4.3 Outline of Regulations on SG Replacement

In Japan, the regulations on the construction, maintenance, and operation of commercial NPPs are performed by the MITI based on "The Law for Regulations of Nuclear Source Materials, Nuclear Fuel Materials and Reactors" (LRNR) and "The Electric Utility Industry Law" (EUIL). Though SG replacement is planned and performed by the autonomous judgment of Electric Utilities, the MITI examines the validity from the view point of safety assurance and conformance with technical standards based on the above-mentioned laws and requirements of the Modification Permission for Installation and Approval of Construction Plans, and confirms that it meets the requirements for permission and approval by Pre-Use Inspection and Welding Inspection.

5. Regulations on SG

5.1 Inspection Criteria

In the EUIL, it is specified that NPPs shall be subjected to periodical inspections by the MITI. Also, in the Enforcement Regulations of the EUIL, the time and facilities for the periodical inspection are specified.

SGs are specified as one of the facilities that shall be subjected to periodical inspection performed annually. In the Standard Manual of Periodical Inspection prepared by the Safety Administration Division, it is also specified that all SG tubes shall be inspected through the full length using ECT.

Types of probes used in the ECT are specified properly according to the types and locations of degradations. Table 3 shows the types and area of application of probes and the form of degradations to be detected. Therefore the probes used depend upon plants. Table 4 shows types of probes which each plant is using.

5.2 Flaw Acceptance Criteria

The technical standard on nuclear power plants is not only the standard on the design and manufacturing of facilities but also the standard on the level at which the facilities must be maintained and operated. Therefore, if any defect is found through inspection, the defective tube cannot be used, and the MITI requires to repair it by plugging or sleeving.

The specific acceptance criteria are as follows.

- * For defects such as wastage of which depth can be specified, their depth shall be 20% or less.
- * For defects such as IGA and SCC of which depth is difficult to be evaluated quantitatively, their indication signal shall be

smaller definitely than noise level.

The criteria are shown in Table 5.

5.3 Primary to Secondary Leakage Monitoring and Limit

On all PWR plants, the following three types of leak monitors are mounted to detect the primary to secondary leakage caused by tube degradation of SGs at the early stage of minute leakage.

R-15: Condenser Air Exhaust Gas Monitor (Scintillation Detector or GM Counter Tube)

R-19: SG Blowdown Water Monitor (Scintillation Detector)

N-16: High Sensitivity Main Steam Monitor (Scintillation Detector)

The arrangement of these monitors is shown in Fig. 3.

In Japan, SG leak operation is not permitted. Therefore, if any one of the indication values on the above monitors arrives at 120% of the background level value, the plant operation must be stopped.

5.4 Technical Specification

The safety of nuclear power plants in Japan is secured not only by the regulation of the MITI but also the autonomous safety assurance control of Electric Utilities. Based on the LRNR and EUIL, the Electric Utilities must specify the items necessary for the autonomous safety assurance control in "Technical Specification", and the specification must be approved by the MITI.

6. Research Projects on SG

6.1 National Projects

6.1.1 Verification Test on SG Tubing Reliability

The following tests and researches were performed over five years from 1989 to 1993 so as to confirm the effectiveness of various measures taken for IGA prevention and verify the reliability of defective tubes during the operation.

- * Identification of SG Tubing Material's Susceptibility to IGA
- * Verification of IGA Countermeasures such as Environmental, Material and Design Improvement
- * Confirmation of Reliability of Tubing with Undetected IGA

6.1.2 Development of SG Tube Inspection Technology

The following tests were performed over seven years from 1992 to 1998 so as to verify that SG tubing degradation could be detected satisfactorily by ECT and the safety of NPPs could be secured.

- * Verification Test of Inspection Technology for Circumferential Defects
- * Verification Test of Precision Inspection Technology
- * Verification Test of Inspection Technology for Mechanical Plug
- * Development of Micro ECT

6.1.3 Development of NDE Technology to Evaluate Fatigue Damage of SG Tubing

The following tests were performed over six years from 1992 to 1997 so as to develop the technologies to detect and diagnose, at the initial stage, the metal fatigue damage occurred on SG tubing.

- * Development of NDE Technology
- * Construction of Data Base relating to Inconel Fatigue

6.1.4 Verification Test on Flow Induced Vibration in U-Bend Tube Bundle of SG

To advance the evaluation method for flow induced vibration of SG at the U-bend portions, the test on thermohydraulic and flow induced vibration was performed over five years from 1993 to 1997 using a large Model SG which simulated the actual SG in construction and thermohydraulic condition.

6.2 Electric Utility's Projects

Electric Utilities are performing the following researches in cooperation with NPP manufacturers.

- (1) Research and development of inspection technology on SG tubing
- (2) Research on advancement of ECT signal evaluation
- (3) Research on analysis of flow induced vibration of SG tubing
- (4) Research on fatigue damage of SG outermost peripheral tubing at U-bend portions
- (5) Research on SCC of SG tubing
- (6) Research on effect of lead on SG tube degradation
- (7) Research on technology for removal of scale from SG TSPL
- (8) Research on adhesion of corrosion products to broached hole of SG TSPL

7. Conclusion

Most SGs of PWR plants in Japan are of advanced type by beginning operation of NPPs of advanced design or SG replacement.

Therefore it is considered that tube degradation experienced in the past will be hardly occurred under the present water chemistry environment.

In order to maintain the integrity of SGs, however, it is mandatory that secondary water chemistry be controlled properly and inspection be performed appropriately. As for inspections in the future, the following is now considered:

- * From the results of researches performed so far, the possibility of degradation of SGs using Alloy 600 MA as tube material experienced in the past cannot be denied except for some SGs. Therefore careful inspection must be performed in the future also.
- * For SGs using Alloy 600 TT, based on the results of researches, the integrity is considered to be maintained except for some degradation modes. Therefore careful inspection must be performed continuously in the future also.
- * For SGs using Alloy 690 TT, the integrity is considered to be maintained against any degradation mode. Therefore reasonable inspection is desired in the future. Also, because its life is still short, it is required to know the aging change and its tendency.
- * Sleeved tubes must be inspected continuously in the future also.

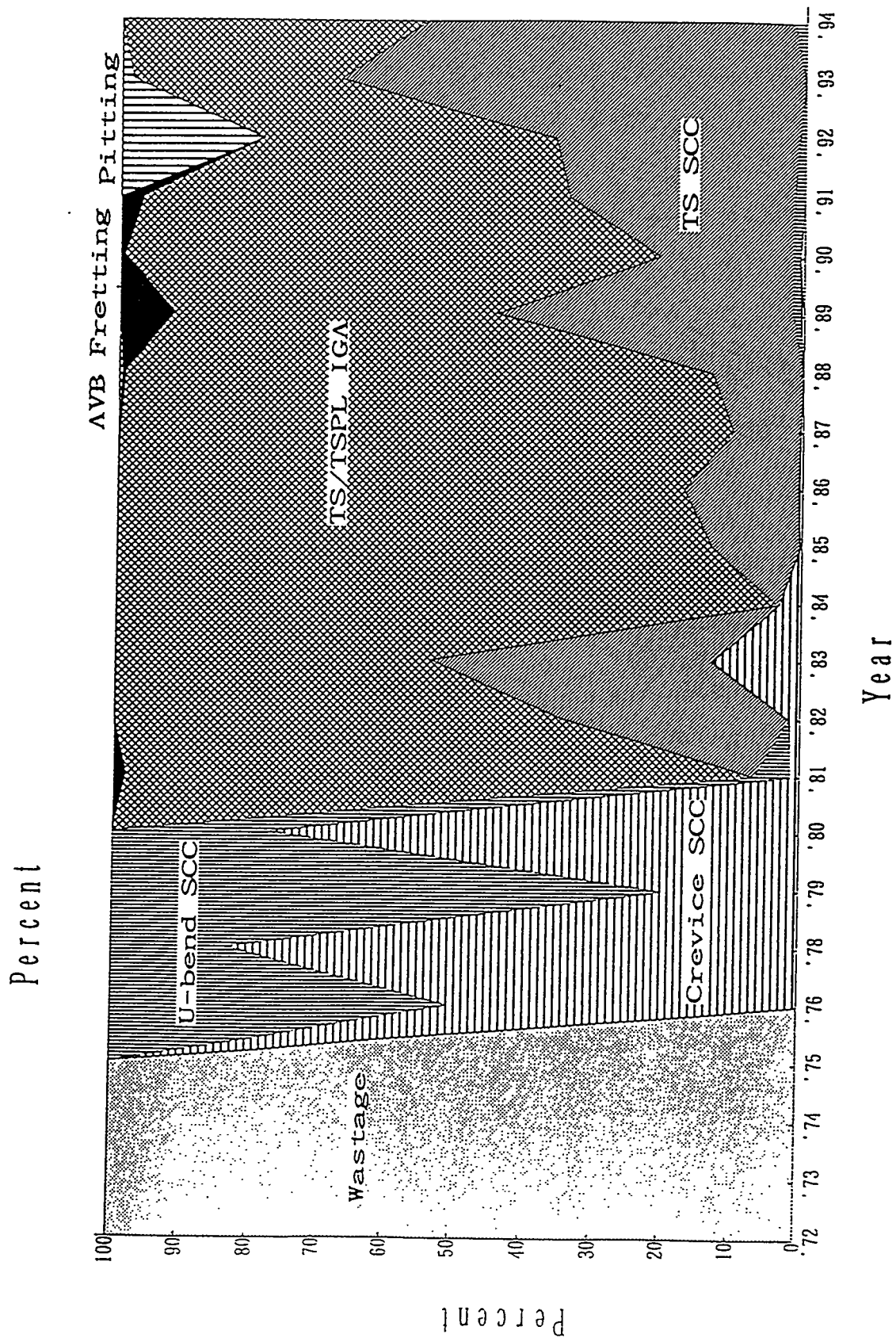


Fig.1 SG TUBE DEGRADATION EXPERIENCE IN JAPAN

Fig. 2 Schematic of defected area

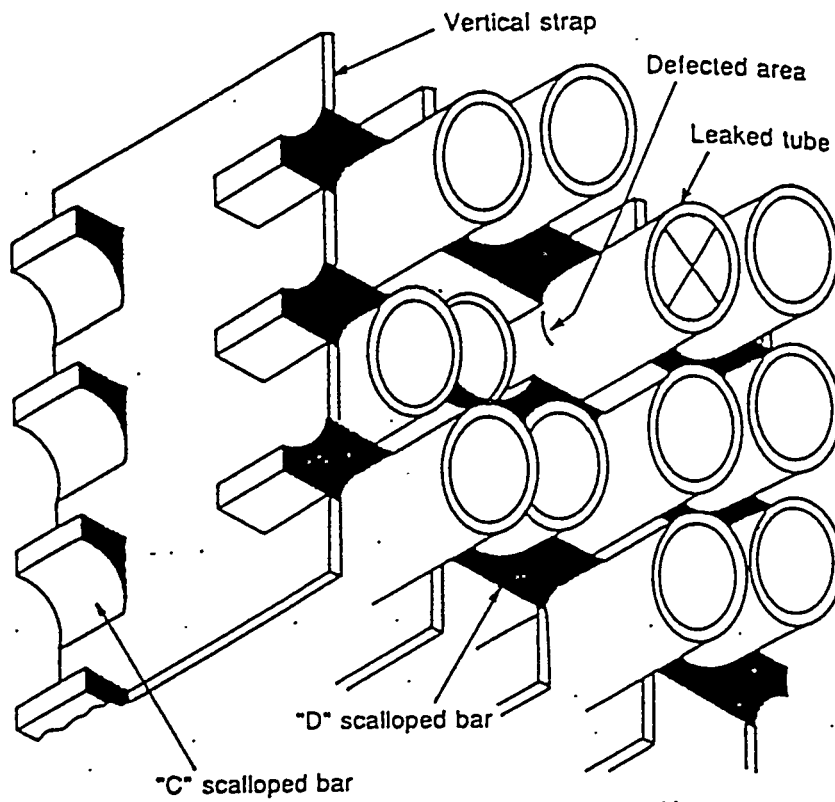
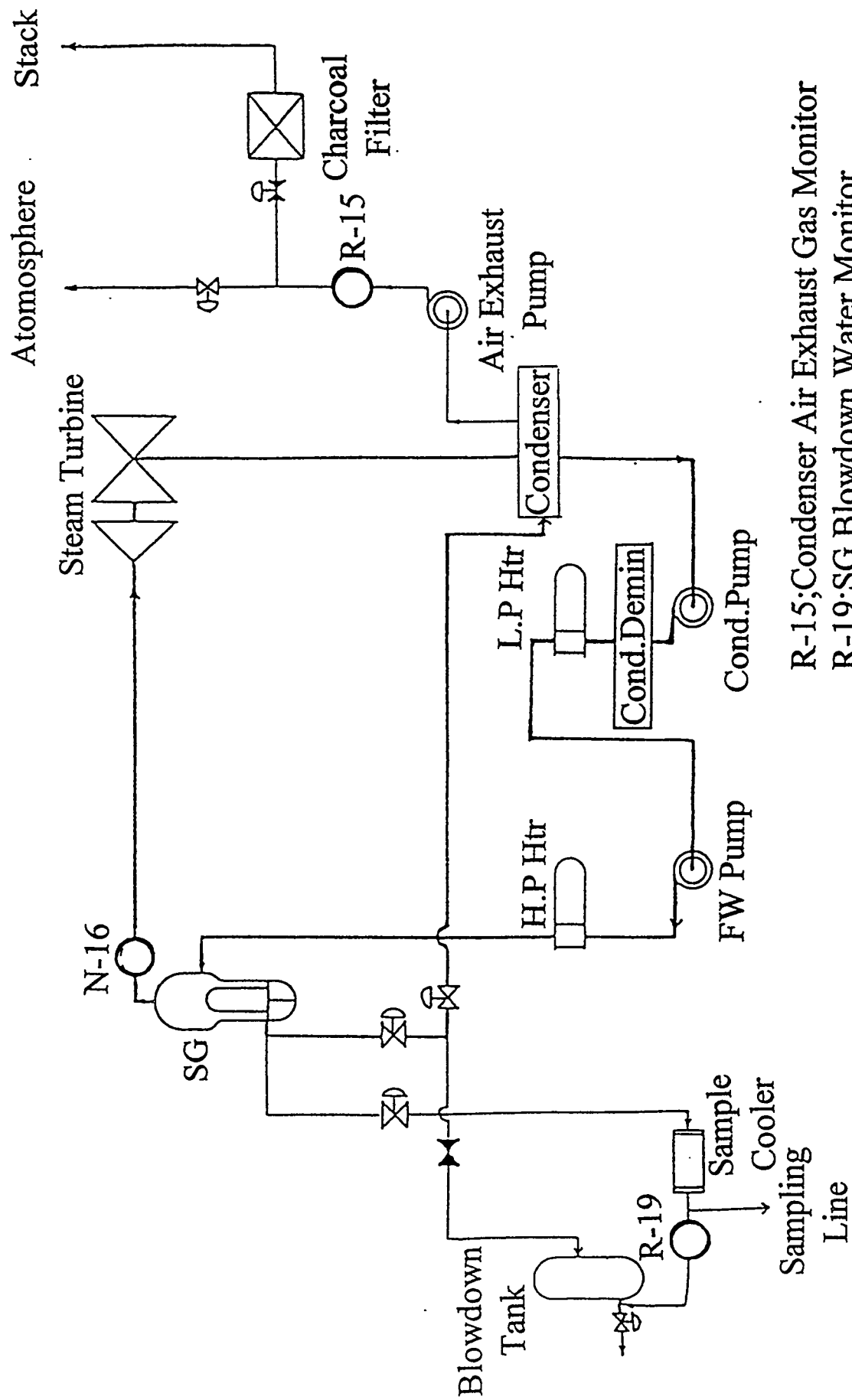


Fig.3 SG LEAK MONITOR SYSTEM



R-15; Condenser Air Exhaust Gas Monitor
 R-19; SG Blowdown Water Monitor
 N-16; High Sensitivity Main Steam Monitor

TABLE-1 NUMBER OF REPORTED INCIDENTS AND FAILURES

(Under the Provision of Laws)

Fiscal Year	81	82	83	84	85	86	87	88	89	90	91	92	93	94
Occured During Operation	Automatic Shutdown	13	7	11	4	4	5	4	1	4	4	4	0	3
	Manual Shutdown	12	10	5	3	8	6	9	10	9	6	10	10	7
	Power Decrease	0	0	0	0	0	0	0	0	0	0	0	0	0
	Tube Degradation of SGs	2	5	6	5	6	7	5	9	9	7	5	6	3
Discovered During Shutdown	Except SGs Tubing	7	3	5	4	1	1	5	1	2	3	0	1	0
	Others	2	1	0	2	0	0	2	1	0	0	1	0	1
Total		36	26	27	18	19	19	23	22	24	20	20	17	14
No. of Plants		23	24	25	28	32	33	36	37	39	41	42	46	48
No. of Reports/Plant		1.6	1.1	1.1	0.6	0.6	0.6	0.5	0.6	0.6	0.5	0.5	0.4	0.3

TABLE-2 STEAM GENERATOR REPLACEMENT

COMPLETED

PLANT	NO. SG	COMM. OPER.	OPER. (YEARS)	REPL. METHOD	REPL. SG	TUBE MATERIAL	COMP. DATE
TAKAHAMA-2	3	NOV.1975	19	PIPE CUT (Entire SG)	MHI	I-690TT	Aug.1994
MIHAMA-2	2	JUL.1972	19	PIPE CUT (Entire SG)	MHI	I-690TT	Oct.1994
GENKAI-1	2	OCT.1975	19	PIPE CUT (Entire SG)	MHI	I-690TT	Nov.1994
OHI-1	4	MAR.1979	15	PIPE CUT (Entire SG)	MHI	I-690TT	May.1995

UNDER REPLACEMENT

MIHAMA-1	2	NOV.1970	24	PIPE CUT (Entire SG)	WH	I-690TT	(Jan.1996)
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PLANNED

TAKAHAMA-1	3	NOV.1974	-	PIPE CUT (Entire SG)	MHI	I-690TT	(Aug.1996)
MIHAMA-3	3	DEC.1976	-	PIPE CUT (Entire SG)	MHI	I-690TT	(Feb.1997)
OHI-2	4	DEC.1979	-	PIPE CUT (Entire SG)	MHI	I-690TT	(Aug.1997)

TABLE-3 PROBES USED IN ECT

PROBE	TESTING AREA	DEGRADATION MODE
Dual Function Probe	All Tubes, Full Length	SCC, IGA, Fretting Wear
Single Coil Probe	TS Crevice	Long Axial SCC
Pancake Coil (8x1)Probe	Roll Transition	SCC
Plus Point Rotating Probe	Tight U-Bend	SCC
U-Bend Probe	Tight U-Bend	SCC
Sleeve Probe	Brazed or Laser Welding Sleeve	SCC

TABLE-4

PROBES USED FOR EDDY CURRENT TESTING

Plant	Dual Function Probe	U-bend Probe	Single Coil Probe	Brazed Sleeve Probe	Pancake Coil Probe (for TS)	Pancake Coil Probe (for TSPL)	Plus Point Coil Probe (for small U-bend)
Mihama -1	○				○		
-2	○	○					
-3	○	○			○	○	○
Takahama-1	○	○	○	○		○	
-2	○	○					
-3	○	○				○	
-4	○	○				○	
Ohn	○	○					
-2	○	○			○	○	○
-3	○	○					
-4	○	○					
Ikata	○	○				○	○
-2	○	○				○	○
-3	○	○					
Genkai	○	○					
-2	○	○				○	○
-3	○	○					
Sendai	○	○				○	
-2	○	○				○	
Tomari	○	○				○	
-2	○	○				○	
Tsuruga	○	○				○	

TABLE-5 Acceptance Criteria of ECT

Location, Mode	Probe, Frequency	Acceptance Criteria
TSPL etc. IGA	DF Probe (Large Pitch) 100-400kHz	<Noise Level ($<1.5V$)
Expansion SCC	DF Probe (Small/Large Pitch) 400kHz	<Noise Level ($<0.5V$) (Small) No Signal Change (Large)
Expansion Circ. SCC	8 X 1 Probe 400kHz	<Noise Level ($S/N < 2$)
U-Bend SCC	U-bend Probe 100kHz	<TSPL Signal $\times 1/2$ No Signal Change
	Plus Point Coil Probe 300kHz	$S/N < 2$
TSPL etc. Wastage	DF Probe (Large Pitch) 100-400kHz	<20% Wall Thickness
AVB Fretting Wear	DF Probe (Large Pitch) 100-400kHz	<20% Wall Thickness
Top of TS Pitting	Rotating Pancake Coil 100-400kHz	$S/N < 2$
Deposit Signal All Modes	DF Probe (Large Pitch) 100-400-600kHz	<Noise Level ($<1.2V$)

OECD/NEA - CNRA/CSNI International Workshop on Steam Generator Tube Integrity in NPPs

**Recent Operating Experiences with
Steam Generators in Japanese NPPs**

**Seiji Yashima
Japan Power Engineering and Inspection Corp.
Tokyo, Japan**

**30 Oct.- 2 Nov., 1995
Argonne National Laboratory
Chicago, Ill., U.S.A.**

INTRODUCTION

Recent Operating Experiences with SG
Lessons Learned From M-2 SGTR
SG Replacement
Safety Regulations on SG
Research and Development on SG

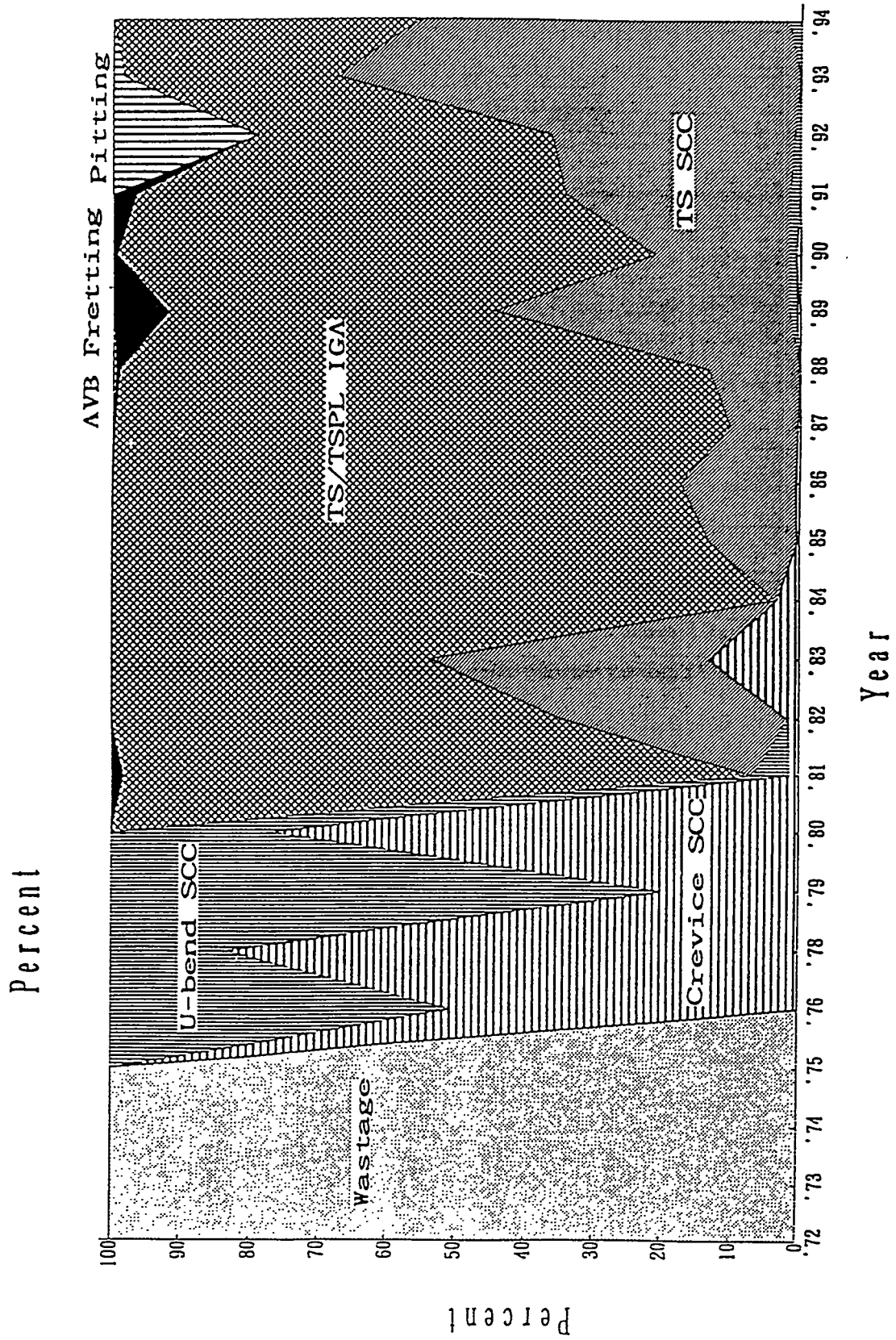
RECENT OPERATING EXPERIENCES WITH SG

Japanese PWR plants

- 22 PWR Plants are being Operated (Oct.1995)
- They have Total 64 SGs
- They have Operating Experiences of Approx. 280 reactor years

Tube Degradation History

- | | |
|-------------------------|--|
| • First Half of 1970's | Wastage (OD) |
| • Latter Half of 1970's | TS Crevice SCC (OD)
U-Bent SCC (ID) |
| • 1980~ | TS/TSPL IGA (OD)
TS SCC (ID)
AVB Fretting (OD)
Pitting (OD) |



SG TUBE DEGRADATION EXPERIENCE IN JAPAN

NUMBER OF REPORTED INCIDENTS AND FAILURES

(Under The Provision of Laws)

Fiscal Year		81	82	83	84	85	86	87	88	89	90	91	92	93	94
Occurred During Operation	Automatic Shutdown	13	7	11	4	4	5	4	4	1	4	4	4	0	3
	Manual Shutdown	12	10	5	3	8	6	7	9	10	9	6	10	10	7
	Power Decrease	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	Tube Degradation of SGs	2	5	6	5	6	7	5	5	9	9	7	5	6	3
Discovered During Shutdown	Except SGs Tubing	7	3	5	4	1	1	1	5	1	2	3	0	1	0
	Others	2	1	0	2	0	0	2	0	1	0	0	1	0	1
Total		36	26	27	18	19	19	19	23	22	24	20	20	17	14
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RECENT OPERATING EXPERIENCES WITH SG

Recent Topics

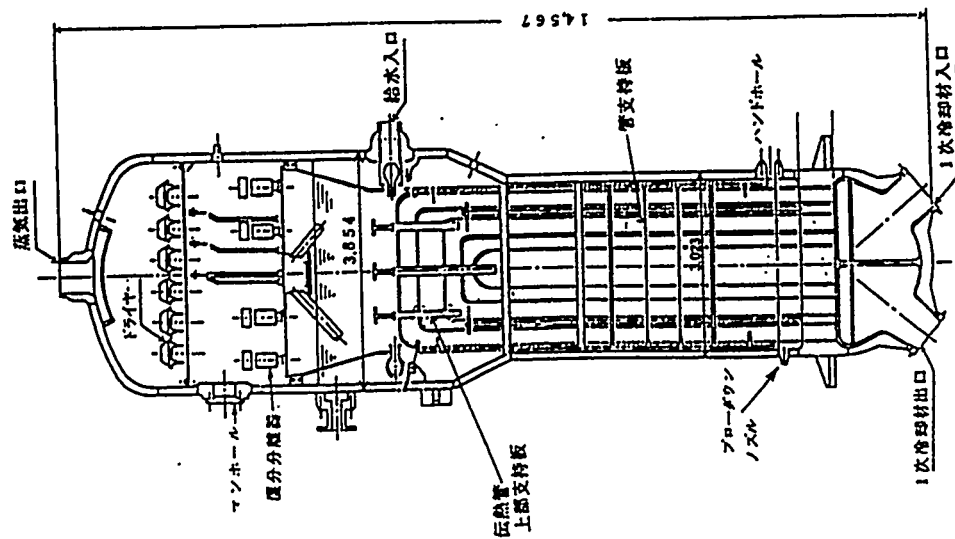
SG Tube Leakage at Mihama-1

Phenomenon : SCC at Upper Tube Support Structure

**Cause : High Stress by Bending and
Corrosive Environment**

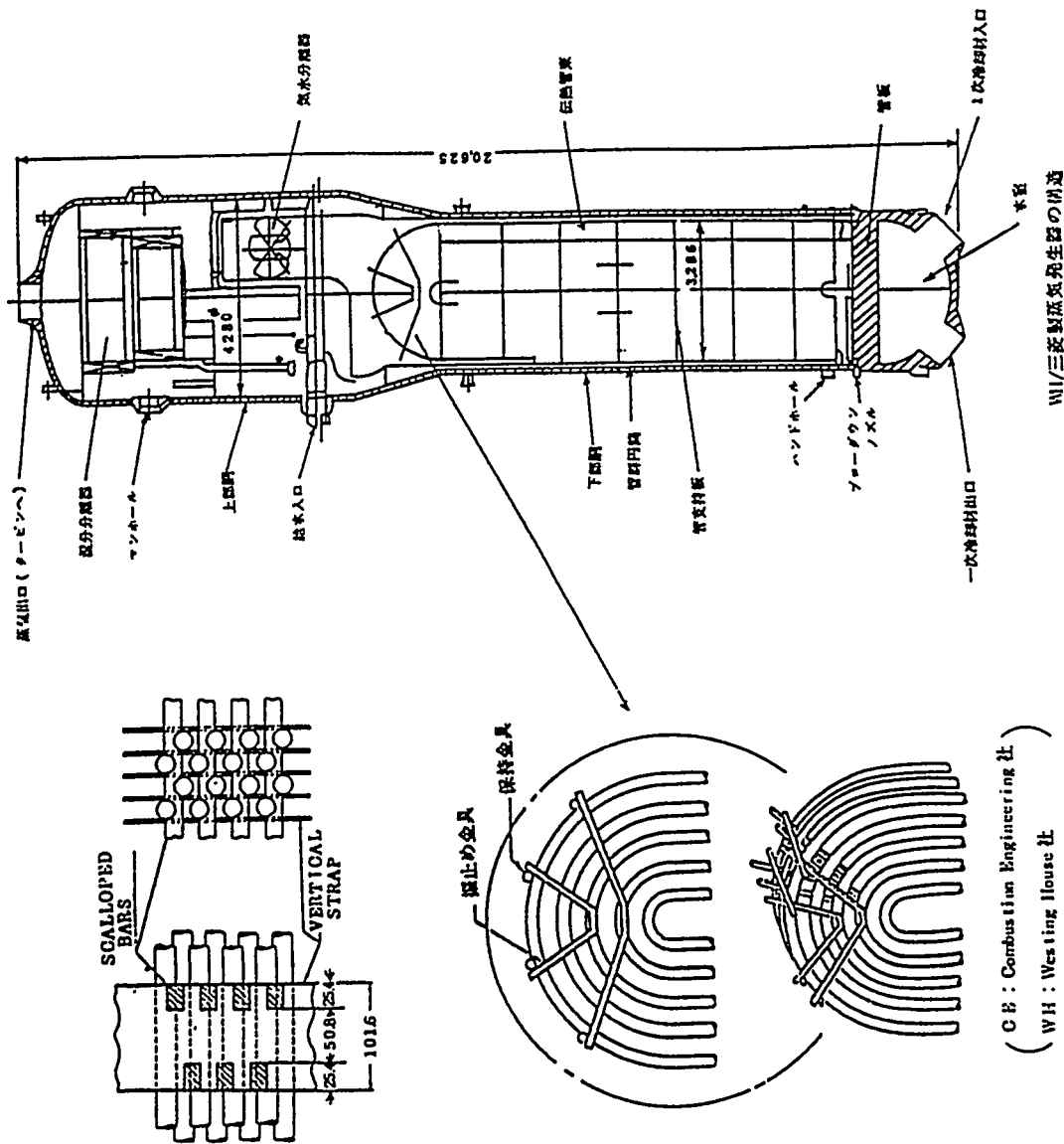
Countermeasure : SG Replacement

1-2) 蒸気発生器の構造



GE製蒸気発生器の構造
(英氏1号機)

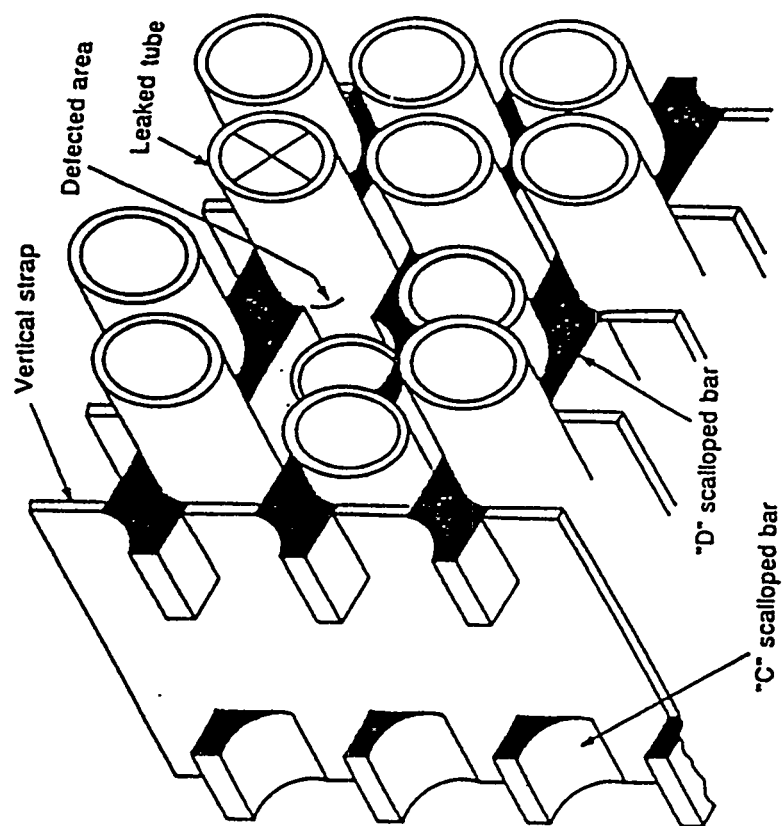
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GE: Combustion Engineering 社
(WH: Westing House 社)

— 5 —

SCHEMATIC OF DEFECTED AREA



RECENT OPERATING EXPERIENCES WITH SG

Recent Topics

SG Tube Leakage at Ohi-2

Phenomenon : SCC at Row-1 Tube

Cause : High Residual Stress

**Countermeasures : Plugging for All Row-1 Tubes
SG Replacement (1997)**

LESSONS LEARNED FROM MIHAMA-2 SGTR Preventive Measures by MITI

1. Addition AVB To Examination of Construction Plan and Pre-Use Inspection
2. Improvement in Periodical Inspection
3. Improvement in Safety Evaluation
4. Reinforcement of Guidance and Supervision on QA Activity
5. Study on Classification of Abnormal Condition in Safety Review
6. Review on Contents of Installation Permission Application
7. Reflection of Technical Advancement and Accumulated Experiences
8. Review on Plant Operation and Maintenance Standard (POMS)
9. Promotion of Safety Research

LESSONS LEARNED FROM MIHAMMA-2 SGTR

Preventive Measures by Electric Utilities

1. Reinforcement of Autonomous Safety Assurance
2. Promotion of Technological Development
3. Improvement in Emergency Response in Occurrence of Abnormalities
4. Measures to Enhance Reliability of Nuclear Power Generation

PREVENTIVE MEASURES IN MIHAMA-2

1. SG Replacement to Advanced Type
2. Installation of N-16 Radiation Monitors on MSL
3. Installation of Monitors on MSL for Radication Release Estimation
4. Installation of Thermometer Type Water Level Gauge on Reactor Vessel
5. Improvement of MSIVs Actuators
6. Improvement of Working Air System to PRVs
7. Increasing of Plant Computers Processing Capability
8. Modification of Power Supply System

STEAM GENERATOR REPLACEMENT

COMPLETED

PLANT	No. SG	COMM. OPER.	OPER. (YEARS)	REPL. METHOD	REPL. SG	TUBE MATERIAL	COMP. DATE
TAKAHAMA-2	3	Nov.1975	19	Pipe Cut (Entire SG)	MHI	I-690TT	Aug.1994
MIHAMA-2	2	Jul.1972	19	Pipe Cut (Entire SG)	MHI	I-690TT	Oct.1994
GENKAI-1	2	Oct.1975	19	Pipe Cut (Entire SG)	MHI	I-690TT	Nov.1994
OHI-1	4	Mar.1979	15	Pipe Cut (Entire SG)	MHI	I-690TT	May.1995

UNDER REPLACEMENT

MIHAMA-1	2	Nov.1970	24	Pipe Cut (Entire SG)	WH	I-690TT	(Jan.1996)
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PLANNED

TAKAHAMA-1	3	Nov.1974	-	Pipe Cut (Entire SG)	MHI	I-690TT	(Aug.1996)
MIHAMA-3	3	Dec.1976	-	Pipe Cut (Entire SG)	MHI	I-690TT	(Feb.1997)
OHI-2	4	Dec.1979	-	Pipe Cut (Entire SG)	MHI	I-690TT	(Aug.1997)

REGULATIONS ON SG REPLACEMENT

- LRNR : Modification Permission for Installation
(Basic Design Examination)
- EUIL : Approval of Construction Plan
(Detailed Design Examination)
Pre-Use Inspection
Welding Inspection

REGULATIONS ON SG

- Inspection Criteria
 - Frequency : Every Periodical Inspection
(Within 12 months $\pm$ 1 month)
 - Objects : All Tubes and Full Length
- Flaw Acceptance Criteria
 - Defective Tubes can not be Used
- Leakage Monitoring and Limit
 - Monitoring : R-15, R-19, N-16
 - Leak Operation is not Permitted
- Technical Specification

PROBES USED IN ECT

PROBE	TESTING AREA	DEGRADATION MODE
Dual Function Probe	All Tubes, Full Length	SCC IGA Fretting Wear
Single Coil Probe	TS Crevice	Long Axial SCC
Pancake Coil (8 × 1) Probe	Roll Transition	SCC
Plus Point Rotating Probe	Tight U-Bend	SCC
U-Bend Probe	Tight U-Bend	SCC
Sleeve Probe	Brazed or Laser Welding Sleeve	SCC

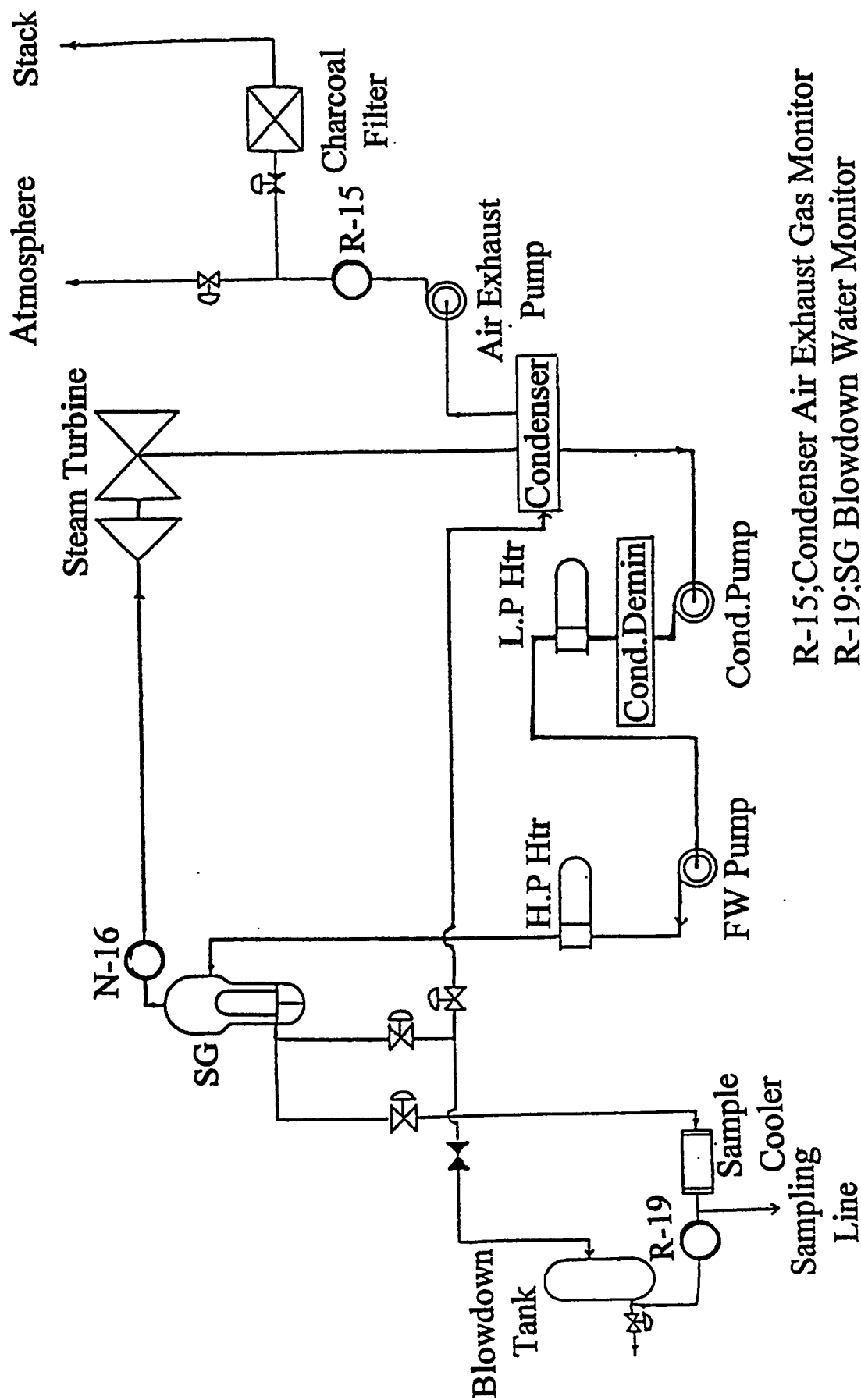
PROBES USED FOR EDDY CURRENT TESTING

Plant	Dual Function Probe	U-bend Probe	Single Coil Probe	Brazed Sleeve Probe	Pancake Coil Probe (for TS)	Pancake Coil Probe (for TSPL)	Plus Point Coil Probe (for small U-bend)
Mihama -1	O				O		
-2	O	O					
-3	O	O			O	O	O
Takahama-1	O	O	O	O		O	
-2	O	O					
-3	O	O				O	
-4	O	O				O	
Ohn	O	O					
-1	O	O					
-2	O	O			O	O	O
-3	O	O					
-4	O	O					
Ikata -1	O	O				O	O
-2	O	O				O	O
-3	O	O					
Genkai -1	O	O					
-2	O	O				O	O
-3	O	O					
Sendai -1	O	O				O	
-2	O	O				O	
Tomari -1	O	O				O	
-2	O	O				O	
Tsuruga -2	O	O				O	

Acceptance Criteria of ECT

Location, Mode	Probe, Frequency	Acceptance Criteria
TSPL etc. IGA	DF Probe (Large Pitch) 100-400kHz	<Noise Level ($<1.5V$)
Expansion SCC	DF Probe (Small/Large Pitch) 400kHz	<Noise Level ($<0.5V$) (Small) No Signal Change (Large)
Expansion Circ. SCC	8 x 1 Probe 400kHz	<Noise Level ($S/N < 2$)
U-Bend SCC	U-bend Probe 100kHz	<TSPL Signal $\times 1/2$ No Signal Change
	Plus Point Coil Probe 300kHz	$S/N < 2$
TSPL etc. Wastage	DF Probe (Large Pitch) 100-400kHz	<20% Wall Thickness
AVB Freighting Wear	DF Probe (Large Pitch) 100-400kHz	<20% Wall Thickness
Top of TS Pitting	Rotating Pancake Coil 100-400kHz	$S/N < 2$
Deposit Signal All Modes	DF Probe (Large Pitch) 100-400-600kHz	<Noise Level ($<1.2V$)

SG LEAK MONITOR SYSTEM



R-15; Condenser Air Exhaust Gas Monitor
 R-19; SG Blowdown Water Monitor
 N-16; High Sensitivity Main Steam Monitor

RESEARCH PROJECTS ON SG

National Projects

- Verification Test on SG Tubing Reliability
(1989~1993)
- Development of SG Tube Inspection Technology
(1992~1998)
- Development of NDE Technology to Evaluate Fatigue
Damage of SG Tubing
(1992~1997)
- Verification Test on Flow Induced Vibration in
U-Bend Tube Bundle of SG
(1993~1997)

RESEARCH PROJECTS ON SG

Electric Utility's Projects

- Research and development of inspection technology on SG tubing
- Research on advancement of ECT signal evaluation
- Research on analysis of flow induced vibration of SG tubing
- Research on fatigue damage of SG outermost peripheral tubing at U-bend portions
- Research on SCC of SG tubing
- Research on effect of lead on SG tube degradation
- Research on technology for removal of scale from SG TSPL
- Research on adhesion of corrosion products to broached hole of SG TSPL

CONCLUSION

Inspection for SG in Future

- 600MA Tubing
- 600TT Tubing

} : Careful Inspection Continue

- 690TT Tubing

: Reasonable Inspection and Aging Change
Inspection

OECD / NUCLEAR ENERGY AGENCY

CNRA/CSNI WORKSHOP

ON

**STEAM GENERATOR TUBE
INTEGRITY**

IN

NUCLEAR POWER PLANTS

**PAPER: "STEAM GENERATORS REGULATORY PRACTICES
AND ISSUES IN SPAIN"**

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**Argonne N.L., Chicago, IL, USA
30 Oct./ 2 Nov. 1995**

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ABSTRACT.-

This paper presents the actual status of spanish Steam Generator's tubes, actions developed by PWR plants owners (licensees) and submitted to C.S.N., and regulatory activities related to tube degradation mechanisms analysis; NDT tube inspection techniques; tube, tubesheet and TSPs integrity studies; tube plugging/repair criteria; preventive and corrective measures including whole S.G.s replacement; tube leak measurement methods and other operational aspects.

Paragraph 1 describe briefly all **tube degradation mechanisms** observed in spanish S.G.s, putting specific emphasis on ID/OD SCC circumferential (RTZ) and OD SCC axial (TSP) cracking.

Paragraph 2 is devoted to **detection, discrimination and sizing** capabilities of various E.C.T. coils used for inspection of defects, specially ID/OD SCC circumferential and OD SCC axial cracking.

Paragraph 3 discusses **tube plugging/repair criteria** applied and based on the degree of confidence on crack growth mechanisms and rates, on tube collapse/burst resistances (explosive failure) and on the developed leak rate/crack opening correlations.

Paragraph 4 is related to **leak detection** capabilities and implementations due to the new developed LBB criteria, emphasizing the general high confidence on N-16 monitors accuracy.

Paragraph 5 describe briefly all **preventive and corrective measures** implemented up to now in Spain, with more or less success, which in four plants led to a final decision of **S.G.s replacement** in order to recover the primary-to-secondary heat transfer initially designed capacity.

1.- DEGRADATION TYPES OBSERVED IN SPANISH S.G.s

Actually, there are seven (7) NPPs PWR-type operating in Spain, with very different ages and supply origin, as can be seen in Table 1. This Table also shows all S.G. models in operation, from the 27 years old and unique model W-24 in Zorita, the twelve (12) S.G.s model W-D3 installed initially in Almaraz 1&2 and Ascó 1&2, till the newer three (3) S.G.s model W-F in Vandellós 2 and, finally, the three (3) S.G.s model KWU-GT 54 in Trillo.

In an extensive replacement program, lasting three years (1995 - 1997), all 12 S.G.s in Almaraz 1&2 and Ascó 1&2 will be substituted by newer S.G.s, designed from SIEMENS-FRAMATOME Consortium model S-F/D3, incorporating many changes in materials (i.e: Incoloy 800 Mod., Stainless steel, ...), design features (i.e: upper F.W. J-pipe injection, tube supporting, tube expansion method, ...) and in operation (i.e: T "window", heat transfer area, ...), fully described in other paper during this CNRA-CSNI Workshop.

Operating experience of those nineteen (19) S.G.s has been quite different, dependent on the models and its features: tube and other internals materials, tube-to-tubesheet joint, tube-to-support plates geometries, water chemistries in primary and secondary sides, stress levels, temperatures, etc.

Main causes of degradation, some of them leading to plug, were in the past, loss of tube wall thickness by wear (fretting) in some tubes in contact with AVBs (model W-F mainly) and at plate locations of tubes in preheater sections (only model W-D3), denting due to corrosion of carbon steel TSPs (some S.G.s model W-D3), random thinning on top of TS (model W-24) and some other minor problems, involving a reduced number of tubes, due to loose parts/ foreign objects (model KWU-GT 54 mainly), dings, bulging or plugging errors, as shown in Tables 2 and A.

Nowadays, tubing degradations which lead to plug or to apply plugging criteria are attributed mainly to SCC from primary (PW) and secondary (SW) sides with preferent locations, as roll transition zones (RTZ) and tube-to-support plates (TSP) intersections, observing both ID/OD axial and circumferential cracking, in all models W-24 and W-D3, as is also shown in Tables 2 and A.

In relation with ID/OD SCC axial and circumferential (RTZ) and OD SCC axial (TSP) cracking, those phenomena has revealed mainly on all twelve W-D3 model S.G.s tubing, leading to a bulky plugging of more than 4600 tubes in Almaraz 1&2 and Ascó 1&2.

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At these units, 3254 tubes account for OD SCC axial cracking in TSP, while 1402 tubes show ID/OD SCC circumferential and axial cracks in the RTZ. As will be discussed later, in paragraph 3, no circumferential cracks at RTZ are allowed in Spain, so tubes are consequently plugged or repaired, while some axial cracking have taken profit of LBB or similar criteria, and depending on their location, morphology and size can remain in service.

2.- DEFECTS DETECTION, DISCRIMINATION AND NDT SIZING TECHNIQUES.

The inservice inspection plans of the steam generator's tubes follow the requirements of R.G. 1.83. However, these plans were modified due to the number of tubes affected for different degradations forms, and actually only two NPPs inspect the 3% required for the R.G.. The rest inspect 100% full length with bobbin coil and 100% on the tubesheet area with MRPC. All indications detected with bobbin coil on the TSPs (axial indications due to stress corrosion cracking) are reinspected with MRPC, as shown in Table B.

Bobbin coil can detect indications with a great level of confidence in the free tube zone, so it is possible to detect defects as pitting, denting, IGA, ODSCC. However this coil has not been able to detect indications, specially circumferentially oriented, in the RTZ. From 1.990 on, in this zone the MRPC probe has been extensively used. Even though the MRPC level of detection is not very high, being its threshold estimated in a 60% loss of wall thickness and 30 ° length, it is assumed as the most effective coil for circumferential indications detection and also for discrimination in the RTZ. With this probe circumferential cracks on the secondary side (ODSCC) of the tubes can be detected and sized, while those initiated from the primary side (IDSCC) can only be detected.

Even though this probe is able to detect and size ODSCC cracks, the method is more effective if the evaluation is carried out with the 3D display (terrain plots), because the voltage of the ECT signal of these cracks is very low, being very difficult for evaluators their detection, as is explained in more detail in other paper during this Workshop. This is the reason why some circumferential cracks were missed to be detected in some inspections. The CSN performed some audits on this way and, as a result, some licensees decided to take corrective measures such as to review records of previous inspections, the use of two different teams evaluating in a 3D monitor display (terrain plots) and to improve the specific training of the personnel involved in the evaluation.

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The MRPC probe is also used to discriminate between axial and circumferential indications at the TSPs intersections previously detected with bobbin coil, as well as those detected at the RTZ, being able to determine the length of the axial cracks with an accuracy of ± 1 mm. (tubes pulled out). In any case, axial cracks length is only important for those cracks at the RTZ where LBB criterion is applied.

Bobbin coil has always been used to define the loss of wall thickness on an amplitude & phase bases; up to now, the voltage repair criteria has not been applied yet, in Spain, by any utility nor is intended to apply in the short term.

Ultrasonic inspection is not regularly applied in Spain, being used only for confirmation of E.C. test signals, in scarce cases.

3.- TUBE PLUGGING AND REPAIR CRITERIA

The general plugging or repair limit for all steam generator models, except model 24, is 40% loss of wall thickness, as established in R. G. 1.121. For model 24 this criterion is 50%, because of the greatest thickness of the tubes.

Model F has also an specific plugging limit of 55% loss of wall thickness for indications coincident with the position of the former AVBs.

However, due to the unexpected defectologies found during operation on model D3, other plugging or repair limits were licensed in Spain: Leak Before Break (LBB) and P* (see Table C).

Leak Before Break (LBB) criterion

This criterion is based in the fact that any risk of breakage, even in the most severe conditions, of a steam generator tube due to PWSCC in the RTZ, is always preceded by an admissible leak in normal operation.

The LBB criterion was licensed only for PWSCC type defects, with recognized morphology, location and growth rate, for which some experience exists in the nuclear field. This criterion applies only for axially oriented cracks; circumferential or complex arrays of flaws are not allowed. Also, the maximum number of axial cracks in a tube must be less than twenty.

"STEAM GENERATOR REGULATORY PRACTICES AND ISSUES IN SPAIN"

This criterion allows tubes to remain in service if they present:

- * Axial cracks located within the RTZ and up to 18 mm. above the top of the tubesheet, if their length is less than 8 mm., or less than 7.6 mm. if the crack reaches the top of the tubesheet.

This criterion was licensed taking into consideration the following:

- a) The critical crack length is about 13 mm. This has been obtained by means of Folias "bulging factor" and was validated by burst testing carried out in Spain.
- b) Tubes with these kind of defects have shown, statistically speaking, a growth rate of 4 mm. between inspections (ie. 12 months) as average.
- c) Eddy Current tests performed in tubes pulled out of spanish steam generators have shown an uncertainty of about 1 mm. on crack length determination, for this kind of defects.

As a result of the application of the LBB criterion, many tubes which present axial oriented cracks remain in service, so the plant's licensees that apply this LBB criterion were required to perform a primary to secondary leak control, using a N-16 system. This aspect is developed in the next paragraph of this paper.

P' criterion

This criterion allows circumferential cracks located below 38 mm. from the top of the tubesheet, except for the outermost rows. It is based on the fact that, even if a total rupture of the tube happens (guillotine rupture), it will be supported by the tubesheet and no whip effect will be produced.

Even though a leak from an existing circumferential crack at the tubesheet (below P' zone) is not expected, should it happen it will promptly be detected in the secondary system in normal operation and corrective measures could be taken. The partial or total rupture of a tube would be detected by the same means, and the safety margins will be respected in any case.

The application of this criterion requires a 100% inspection with RPC on each refueling outage.

4.- LEAK DETECTION SYSTEMS.

Since beginning of operation, in above mentioned spanish plants, all typical leak detection systems have been applied (i.e. off-gas, blowdown and grab sample analysis) with medium to high effectiveness, as shown in Table D.

But when, in the late 80's and after, ID/OD SCC axial and circumferential (RTZ) and OD SCC axial (TSP) cracking arrived to be the major contributors to S.G.s tube plugging in some W models, and this fact drove to propose alternative plugging criteria as P* or LBB, the evaluation performed in CSN resulted in the need of a more effective surveillance system, for leak detection during operation, than "classical" systems, as the plants were allowed to continue full operation cycles with some cracking in the pressure barrier.

That led to installation of N-16 monitors in all steam lines of Almaraz 1&2 and Ascó 1&2, as plants more affected.

With the LBB and P* plugging/repair criteria, primary to secondary leakages can be due either to axial oriented cracks (LBB zone) or to circumferential cracks (below P* zone), and, even though leaks from circumferential cracks at the tubesheet are not very likely to happen, the leakages due to these kind of cracks will be considered for the limits established for the LBB criterion, being in a modified scenario which makes more conservative the LBB criterion adopted for axial cracks.

The Standard Technical Specifications establish a primary to secondary leakage limit of 78 litres/hour per each steam generator. Because of the LBB and P* plugging/repair criteria, other leakage limits were imposed to plants with models 24 and D3. These leakage limits were established initially in 5 litres/hour above the maximum stabilized leakage (measured after 15 days of operation, which was established in about 5 litres/hour), being later augmented up to 15 litres/hour. Also if the N-16 system detects an increase greater than 3 litres/hour, in any 72 hours period, the plant must go to the shutdown mode; confirmation of values by blowdown analysis is always required in those situations.

Operating experience of such leak surveillance methodology has been successful up to now in spanish plants, as is fully described in other paper during this Workshop. Only minor installation problems (i.e: shielding and calibration) have been observed at the beginning in one of the plants that have N-16 monitors, but actually solved. Figures 1 and 2 show two time graphics, for two different plants, displaying leak rate values observed during the operating cycle.

5.- PREVENTIVE AND CORRECTIVE MEASURES. S.G.s REPLACEMENT.

Operating experience of steam generators in the spanish NPPs has not been similar due, mainly, to the different features of each model. In particular, D3 model has been the most affected due to the different types of degradations (loss of wall thickness by wear in different zones, stress corrosion cracking from the primary and secondary side in zones as TSP intersections and RTZ and others), and have obliged the licensees of these NPPs to take a number of preventive and corrective actions.

5.1.- PREVENTIVE ACTIONS

Preventive measures implemented up to now in spanish S.G.s are well associated to the degradation types observed. (see Table E).

The preventive measures adopted by the licensees, up to date, are the following:

- a) The first preventive measure done in Spain (1.983) in D3 model was to perform a modification in the inlet feedwater nozzle, due to problems that had been detected in other abroad NPPs with the same steam generator, consisting in a manifold installation to reduce the flow-induced vibration and, thus, the wear rate of the tubes in this area (preheater).
- b) In 1986, during the refuelling outages of Almaraz and Asco NPPs (both D3 model) it was applied the shot-peening process in all the length of the tube sheet in the hot leg in order to mitigate the degradation by SCC in this area.

According to the laboratory tests this process is effective if it is performed during the two first years of operation. The results obtained in Almaraz after the shot-peening process did not confirm this prevision, being Almaraz 2 more affected than Almaraz 1, when the shot-peening process was applied during the second outage (before two years of operation) and third outage (after two years of operation) respectively.

- c) In 1897, the licensees of NPPs with D3 models performed a heat treatment in the U-bend small radius tubes, for stress relief, reducing the probability of cracking in this area.

Up to date no cracks have been detected in this area inducing to give reliance in the high effectiveness of this preventive process.

- d) In order to prevent corrosion problems due the iron and/or copper species (corrosion products) which accumulate in sludge form on the tubesheet or scale by deposition on the tubes surface, some licensees with AVT chemistry adopted measures to remove such products. The methods adopted were chemical cleaning and boron injection in the secondary. The effectiveness of these measures was very high, as indicated in Table E, being the amounts of corrosion products very low comparing with previous values in earlier outages.

On the other hand, and in order to reduce the amount of copper in the secondary, these licensees took the decision of changing the condenser tubes, with a high percentage of copper, as well as other parts of the secondary, replacing them with materials with a high content of titanium.

5.2.- CORRECTIVE ACTIONS

Corrective measures, as shown in Table E, are related with plugging and sleeving and also with replacement or modification of internal components of S.G.s.

The corrective measures adopted up to date in Spain are the following:

- a) The heavy degradation of tubes by wear caused by fretting with the AVBs in the U-bend zone detected in the F model caused the plugging of a quite large number of tubes up to 1.992 (4th refueling), as can be seen in Table 2. In this outage the old AVBs were replaced by others of a new design; since then no tubes have been plugged by this cause and a new plugging criterion was licensed for this model (see paragraph 3 above).
- b) The most common corrective measure adopted in Spain to deal with all kind of defects was the plugging of tubes, as it is shown in Table 3.

However, due to the big amount of tubes with defects, some licensees, specially those with D3 model, decided to use the sleeving process to keep those tubes in service. This process could be applied on those tubes with defects in the tubesheet area and up to the first tube support plate. Table 4 shows the behaviour of the sleeved tubes, which can not be considered as good as it was expected because a high percentage of these tubes had to be plugged afterwards for different causes (defects in other areas of the parent tube and, mainly, defects that appeared in the upper joint between the parent tubes and the sleeves. This measure was not widely used and, for that reason, actually sleeving is not a concern in Spain.

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- c) Finally, and due to various reasons, among them the increase in the plugging rate caused by ID/OD SCC axial and circumferential (RTZ) and, specially, OD SCC axial (TSP) cracking in Almaraz 1&2 and Ascó 1&2, which reaches in a case 16% of plugged tubes, being 18% the limit licensed by the CSN, and because the sleeving process was not an effective measure, the licensees decided the whole replacement of all 12 type W-D3 S.G.s by new S-F/D3 Consortium designed S.G.s. Replacement of the first 3 S.G.s in Ascó 1 has finished just some days ago (october 95), successfully. The other 9 S.G.s will be replaced in 1996 (Almaraz 1 and Ascó 2) and 1997 (Almaraz 2).

6.- CONCLUSIONS.

The paper present a general description of actual S.G.s problems and regulatory concerns in Spain. Although defectologies as ID/OD SCC axial and circumferential (RTZ) and OD SCC axial (TSP) cracking exist in 13 spanish S.G.s, there is a comfortable degree of confidence in their operation.

Replacement of 12 S.G.s by less susceptible materials and a better design, in very short term; a low degradation rate (6% plugged tubes in 27 years) in Zorita W-24 model and a general good behaviour in the Vandellós 2 and Trillo S.G.s up to now, gives, actually, a reasonable status from the regulatory view.

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DEGRADATION MECHANISMS	OBSERVED	CURRENT	PRIORITY	UNDERSTANDING	FUTURE ACTIONS
ID SCC, CIRCUMFERENTIAL, RTZ	YES	YES	HIGH	HIGH	RESEARCH
OD SCC, CIRCUMFERENTIAL, RTZ	YES	YES	HIGH	HIGH	RESEARCH
ID AXIAL PWSCC	YES	YES	MEDIUM	HIGH	-
ODSCC AT TUBE SUPPORT PLATES	YES	YES	HIGH	HIGH	RESEARCH
SLEEVED TUBE DEGRADATION	YES	NO	LOW	HIGH	-
FREESPAN CRACKING	YES	NO	LOW	HIGH	-
OTHERS (SPECIFY) *	YES	YES (LOW)	LOW	HIGH	-

* FRETTING AVBs, FR. PREHEATER, IGA TSPs, DENTING TSPs

TABLE A

"STEAM GENERATORS REGULATORY PRACTICES AND ISSUES IN SPAIN"

INSPECTION	TECHNIQUE	DETECTION	SIZING	CHARACTERISATION OF INTEGRITY	FUTURE ACTIONS
ID SCC, CIRCUMFERENTIAL, RTZ	BOBBIN + RPC	MEDIUM	LOW	MEDIUM	RESEARCH
OD SCC, CIRCUMFERENTIAL, RTZ	BOBBIN COIL + MRPC + 3- COIL MRPC	MEDIUM	LOW	MEDIUM	RESEARCH
ID AXIAL PWSCC	"	HIGH	HIGH	HIGH	-
ODSCC AT TUBE SUPPORT PLATES	BOBBIN COIL/ RPC	HIGH	HIGH	HIGH	RESEARCH
SLEEVED TUBE DEGRADATION	CROSS-WOUND	HIGH	HIGH	HIGH	VENDOR
FREESPAN CRACKING	BOBBIN COIL	HIGH	HIGH	HIGH	-
ALL KIND OF DEFECTS (FRETTING AVBs, PITTING,...)	BOBBIN COIL	HIGH	HIGH	HIGH	-
	HELIUM LEAK TEST (UNUSUAL)	HIGH	NA	NA	-

TABLE B

"STEAM GENERATORS REGULATORY PRACTICES AND ISSUES IN SPAIN"

INTEGRITY	REPAIR LIMIT	BASIS	CONFID. ON GROWTH RATE	CGR BASIS	CONFID. ON BURST PRESSURE PREDICTION	CONFID. ON LEAK RATE	PROBABILISTIC DETERMINISTIC	FUTURE ACTIONS
ID SCC, CIRCUMFERENTIAL, RTZ	ALL (*)	NOT ALLOWED	LOW	NOT ALLOWED	LOW	LOW	DETERMINISTIC	RESEARCH
OD SCC, CIRCUMFERENTIAL, RTZ	ALL (*)	NOT ALLOWED	LOW	NOT ALLOWED	LOW	LOW	DETERMINISTIC	RESEARCH
ID AXIAL PWSCC	CRACK LENGTH	PHYSICALLY	MEDIUM	LAB * FIELD	HIGH	MEDIUM	DETERM. * PROBABILISTIC	RESEARCH (LEAK)
ODSCC AT TUBE SUPPORT PLATES	CRACK DEPTH	"	HIGH	LAB * FIELD	HIGH	MEDIUM	DETERMINISTIC	RESEARCH (LEAK)
SLEEVED TUBE DEGRADATION	"	"	HIGH	LAB * FIELD	HIGH	HIGH	DETERMINISTIC	-
FREESPAN CRACKING	"	"	HIGH	FIELD	HIGH	HIGH	DETERMINISTIC	-
FRETTING AVBS	CRACK DEPTH	PHYSICALLY	HIGH	FIELD	HIGH	HIGH	DETERMINISTIC	-

TABLE C

"STEAM GENERATORS REGULATORY PRACTICES AND ISSUES IN SPAIN"

OPERATION ASPECTS/ RISK ANALYSIS	APPLIED/ USED	ACCURACY EFFECTIVENESS	LIMITS	FUTURE ACTIONS
N-16	YES	HIGH	LBB/TUBE INTEGRITY	REGULATORY
OFF-GAS ANALYSIS	YES	MEDIUM-LOW	DOSE	REGULATORY
BLOWDOWN ANALYSIS	YES	HIGH	TUBE INTEGRITY/DOSE	REGULATORY
GRAB SAMPLES	YES	MEDIUM	DOSE	-
OTHERS (SPECIFY)	NO	-	-	-

TABLE D

"STEAM GENERATORS REGULATORY PRACTICES AND ISSUES IN SPAIN"

PREVENTIVE/ CORRECTIVE MEASURES	APPLIED/ USED	EFFECTIVENESS	QUALIFIED	FUTURE ACTIONS
SLUDGE LANCING	YES	HIGH	YES	VENDOR
CHEMICAL CLEANING	YES	HIGH	YES	VENDOR
SLEEVING	YES	LOW	YES	VENDOR
SHOT PEENING	YES	LOW	YES	VENDOR
WATER CHEMISTRY	YES (1)	HIGH	YES	VENDOR
AVBs REPLACEMENT	YES (2)	HIGH	YES	VENDOR
PREHEATER MODIFICATION	YES	HIGH	YES	-
SG REPLACEMENT	YES (3)	-	YES	VENDOR

- (1) INCLUDING BORON INJECTION IN THE SECONDARY
(2) PERFORMED IN VANDELLOS 2 NPP (MODEL W-F)
(3) THREE SG REPLACED IN ASCO 1 NPP (MODEL W-D3)

TABLE E

SPANISH SGs' CHARACTERISTICS

	J. CABRERA	ASCO I/II	ALMARAZ I/II	VANDELLOS II	TRILLO I
S.G. MODEL	24 (W)	D3 (W)	D3 (W)	F (W)	KWU
S.G.s	1	3	3	3	3
PLANT STARTUP	1968	1983/85	1981/83	1988	1988
TOTAL TUBES/S.G.	2604	4674	4674	5626	4086
MATERIAL	INCONEL 600	INCONEL 600 MA	INCONEL 600 MA	INCONEL 600 TT	INCOLOY 800
ROLLING	PARTIAL	FULL DEPTH MECHANICAL	FULL DEPTH MECHANICAL	FULL DEPTH HIDRAULIC	PARTIAL MECHANICAL
PREHEATER	NO	YES	YES	NO	YES
FEEDWATER INLET	TOP	BOTTOM	BOTTOM	TOP	TOP &
2RY. CHEMISTRY	PHOSPHATES	AVT	AVT	AVT	AVT BOTTOM

(*) modified per S.G. replacement

TABLE 1

DEGRADATIONS IN SPANISH S.G.s NUMBER OF PLUGGED TUBES									
PLANT	S.G.	THINNING	FRETTING AYBs	FRETTING PREHEAT	TUBESHEET ID-OD SCC	S. PLATES OD SCC	IGA	OTHERS	TOTAL
J. Cabrera W-24	1	19	21	0	77	12	0	26	155 (5.95%)
Almaraz 1 W-D3	1	0	17	46	125	138	0	26	352 (7.50%)
	2	0	15	1	200	218	0	68	502 (10.7%)
	3	0	10	20	154	143	0	26	353 (7.50%)
Almaraz 2 W-D3	1	0	2	24	201	335	0	11	574 (12.3%)
	2	0	6	1	180	75	0	2	264 (5.60%)
	3	0	0	30	46	60	1	3	140 (3.00%)
Ascó 1 W-D3	1	0	1	18	158	547	0	18	742 (15.9%)
	2	0	20	12	153	278	0	5	468 (10.0%)
	3	0	14	0	65	565	0	10	656 (14.0%)
Ascó 2 W-D3	1	0	6	0	22	463	0	14	505 (10.8%)
	2	0	2	0	55	264	0	16	337 (7.21%)
	3	0	3	0	105	384	0	4	496 (10.6%)
Vandellós 2 W-F	1	0	97	0	0	0	0	3	100 (1.78%)
	2	0	63	0	0	0	0	7	70 (1.24%)
	3	0	42	0	0	0	0	5	47 (0.84%)
Trillo KWU	1	0	0	0	0	0	0	0	0
	2	0	0	0	0	0	0	12	12
	3	0	0	0	0	0	0	2	2

OCTOBER 95

NO TUBES PLUGGED BY DENTING OR PITTING

TABLE 2

DEGRADATIONS IN SPANISH SGs

NUMBER OF PLUGGED TUBES

PLANT	PLUGGED TUBES
JOSE CABRERA (W-24)	155 (5.95%)
ALMARAZ 1 (W-D3)	SG1 352 (7.50%) SG2 502 (10.7%) SG3 353 (7.50%)
ALMARAZ 2 (W-D3)	SG1 574 (12.3%) SG2 264 (5.60%) SG3 140 (3.00%)
ASCO 1 (W-D3)	SG1 742 (15.9%) SG2 468 (10.0%) SG3 656 (14.0%)
ASCO 2 (W-D3)	SG1 505 (10.8%) SG2 337 (7.21%) SG3 496 (10.6%)
VANDELLOS 2 (W-F)	SG1 100 (1.78%) SG2 70 (1.24%) SG3 47 (0.84%)
TRILLO (KWU)	SG1 0 SG2 12 SG3 2

MAXIMUM LICENSED PERCENTAGE: 18%

OCTOBER 95

TABLE 3

BEHAVIOUR OF SLEEVED TUBES

ALL SLEEVED TUBES ARE INSPECTED BY EDDY CURRENT (CROSSWOUND) ON EACH REFUELING

TOTAL INSTALLED		1992	1993	1994	REMAIN
ALMARAZ 1	SG1	20	1 (1) 19 (2)		0
	SG2	57	8 (1) 47 (2)		2
	SG3	20	20 (2)		0
ASCO 1	SG1	115	1 (1) 29 (5)	10 (3) 4 (4)	69
	SG2	0			0
	SG3	145	1 (4) 29 (5)	5 (1) 4 (3) 2 (4)	101

- (1) PLUGGED FOR DEFECT IN THE TUBE
 (2) PLUGGED FOR RECOMMENDATION OF B&W
 (3) PLUGGED FOR LOSS OF THICKNESS IN THE SLEEVE
 (4) PREVENTIVE
 (5) PLUGGED IN THE SAME REFUELING OUTAGE AS THEY WERE INSTALLED

TABLE 4

NUCLEAR POWER PLANT A

Primary-Secondary Leakage

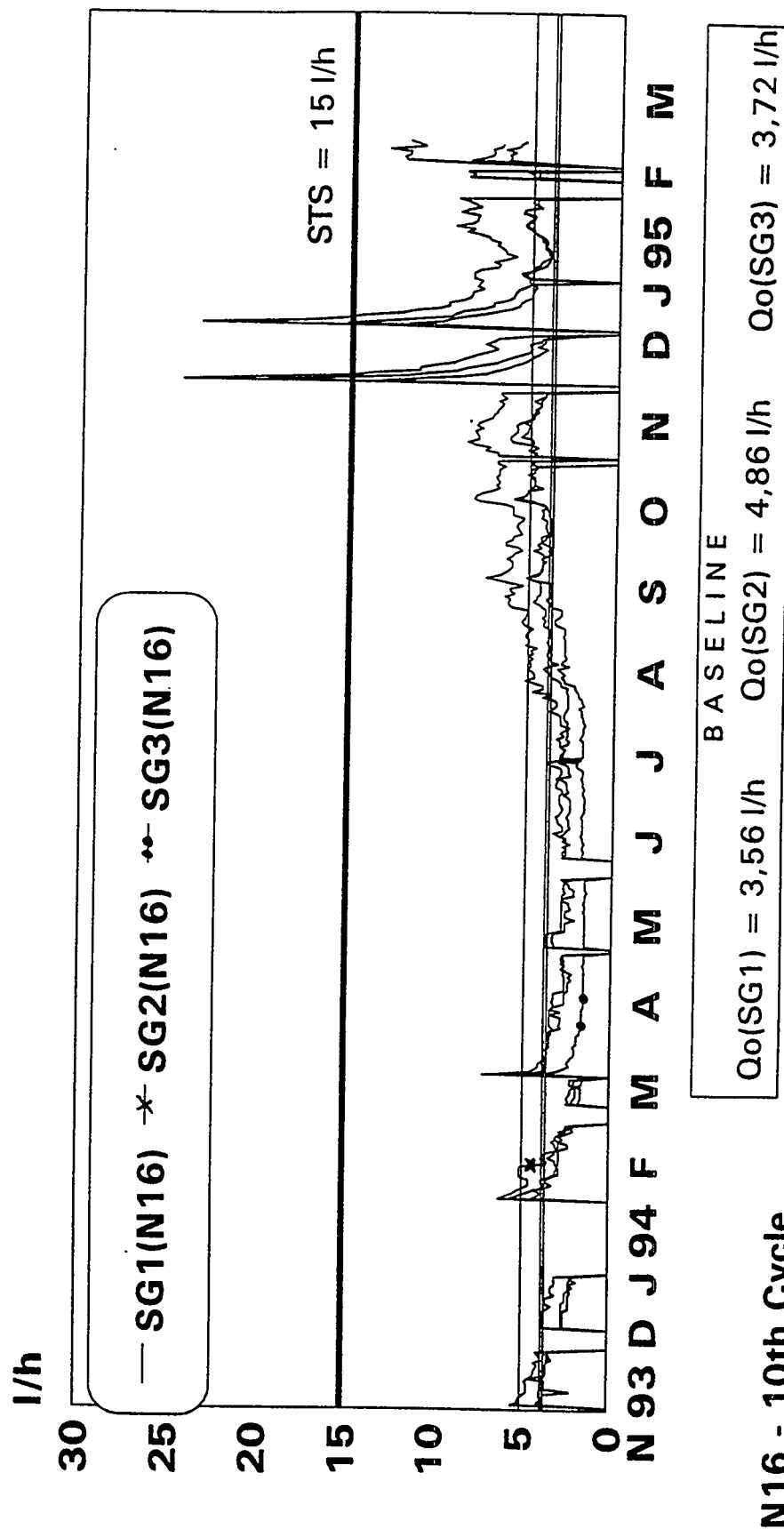
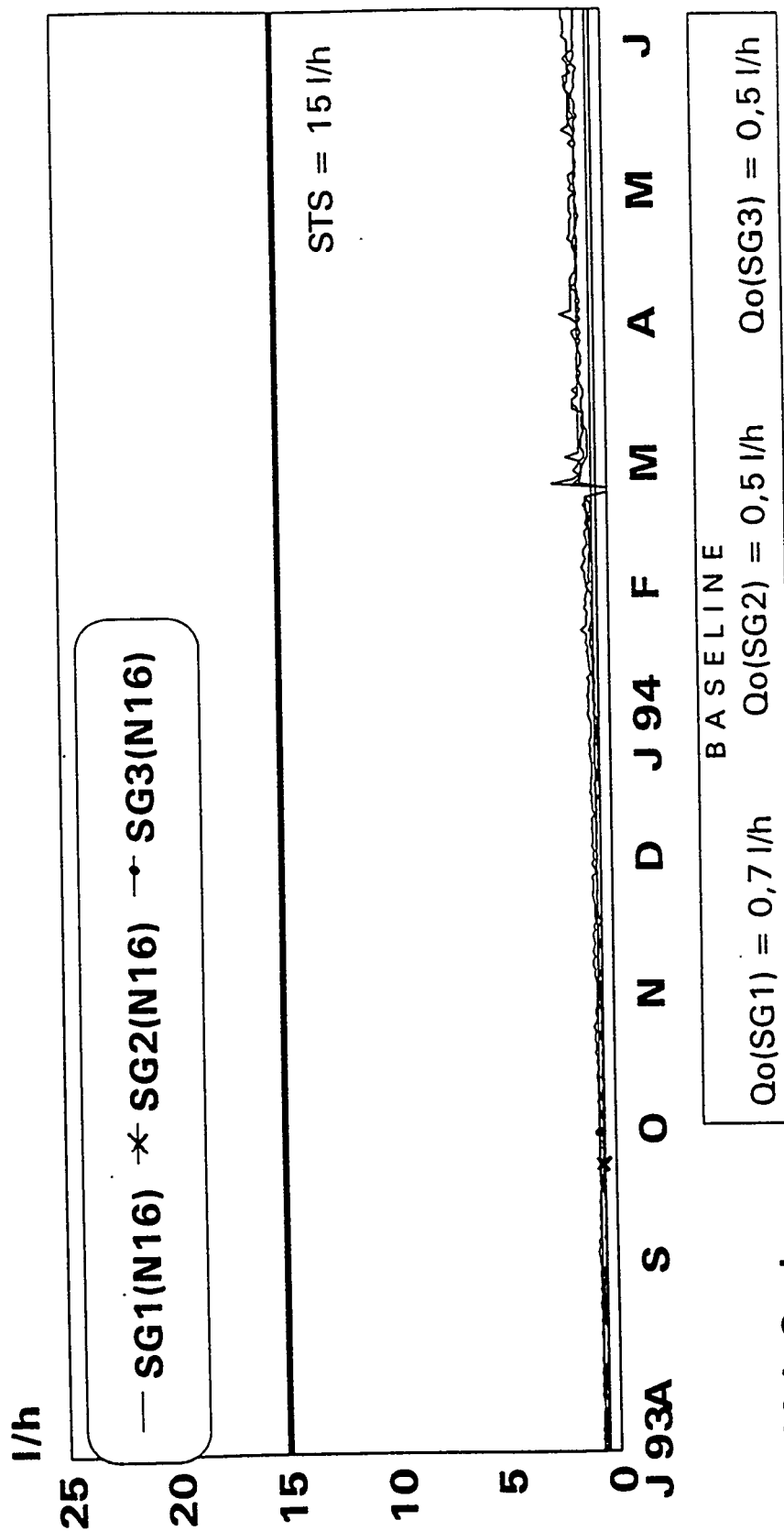


Figure 1

NUCLEAR POWER PLANT B

Primary-Secondary Leakage



N16 - 10th Cycle

Figure 2

WORKSHOP ON STEAM GENERATOR TUBE INTEGRITY IN NUCLEAR POWER PLANTS

Steam Generator Issues in the United States



Jack R. Strosnider

Chief, Materials and Chemical Engineering Branch

October 30, 1995

DEGRADATION MECHANISMS

ALLOY 600 STEAM GENERATOR TUBES IN THE U.S. HAVE EXHIBITED DEGRADATION MECHANISMS SIMILAR TO THOSE OBSERVED IN OTHER COUNTRIES

EFFECTIVE PROGRAMS HAVE BEEN IMPLEMENTED TO ADDRESS SEVERAL DEGRADATION MECHANISMS INCLUDING

Wastage

Mechanical wear

Pitting

Fatigue

THESE DEGRADATION MECHANISMS ARE FAIRLY WELL UNDERSTOOD AS INDICATED BY THE ABILITY TO EFFECTIVELY MITIGATE/MANAGE THEM

DEGRADATION MECHANISMS (cont.)

**STRESS CORROSION CRACKING (SCC) IS THE DOMINANT
DEGRADATION MECHANISM IN THE U.S.**

Axial SCC

Tube support plates (TSPs) - OD initiated

NRC Generic Letter 95-05

Dented TSP intersections - ID initiated

Free span - OD initiated

**Circumferential SCC (NRC Generic Letter 95-03, Information
Notices 94-88 and 95-40)**

Expansion Transition (PWSCC and ODSCC)

Dented TSP intersections

Parent tube at sleeve joints

U-bend of small radius tubes

**SCC POSES SIGNIFICANT INSPECTION AND MANAGEMENT
CHALLENGES TO THE INDUSTRY AND THE REGULATORS**

UNDERSTANDING OF DEGRADATION MECHANISMS

A BETTER UNDERSTANDING OF SOME OF THE CRITICAL PARAMETERS THAT AFFECT SCC INITIATION AND PROPAGATION MAY BE USEFUL TO BOTH THE INDUSTRY AND THE REGULATORS

Incubation times for various forms of SCC

Commencing more intensive examinations at location(s) of interest

Determining the type of SCC a location is most susceptible to [ID or OD; axial or circumferential]

Permits the optimization of the examination method at a given plant

IDENTIFICATION AND DISCUSSION OF THESE CRITICAL PARAMETERS MAY PROVIDE INSIGHTS, FOR EXAMPLE, ON THE EFFECTS OF

Deposits and sludge on susceptibility to ID/OD cracking

Expansion technique on the susceptibility to ID/OD initiated cracking at the expansion-transition

Tube heat treatment and the microstructure of the tubing and the susceptibility to SCC

Temperature and general water chemistry on SCC initiation and propagation

Mechanical or corrosion induced tube deformation (i.e., denting) on the initiation and propagation of SCC

INSERVICE INSPECTION

**EXISTING REGULATIONS IN THE U.S. WERE DEVELOPED IN 1970s
WHEN**

General wall thinning was the dominant degradation mechanism

Readily detected and sized

Extensive amounts of degradation were not anticipated

Initial sample sizes were limited

Inspection criteria were very prescriptive

**A PLANT'S TECHNICAL SPECIFICATIONS SPECIFY THE INITIAL
SAMPLING SCOPE AND EXPANSION CRITERIA**

Differences exist between the plants

Probe type is not specified

EVEN WITH THE CURRENT REGULATORY FRAMEWORK

**Extensive sampling plans have been developed by both the
regulators and the industry**

**Probes have also been developed and optimized to detect the
degradation mechanisms currently being observed in the field**

**Such an effort was necessary in order to ensure tube integrity
per the U.S. regulations**

**Utilities whose steam generators have exhibited
widespread degradation are essentially required to adopt
more robust inspection plans and techniques**

CAPABILITIES OF INSPECTION TECHNIQUES

RELIABLE FLAW DETECTION AND SIZING ARE NECESSARY FOR ENSURING STEAM GENERATOR TUBE INTEGRITY

Periodic tube removal for destructive examination

In-situ pressure testing

IMPORTANT CONSIDERATIONS IN DEVELOPING AND EVALUATING AN INSPECTION PROGRAM, WHICH MAY BE USEFUL TO DISCUSS, INCLUDE

Probe selection and qualification

Performance demonstration requirements and criteria

Effects of tube geometry (dents, expansion transition) and interfering signals (noise, deposits, etc.) on the capabilities of a probe

Methods for minimizing the effects of these interferences and their limitations

Probability of detection for a specific flaw type and a specific set of analyst guidelines

**Ability to distinguish between interfering signals and true flaws
- use of multiple techniques**

Sampling strategies (overall tube examinations, expansion transition, dents, etc.) including the frequency of examinations

Sizing accuracy and identification of the appropriate parameter to assess tube integrity (length, depth, volume, etc.)

TUBE INTEGRITY

ALL PLANTS IN THE U.S. HAVE A DEPTH-BASED TUBE REPAIR CRITERIA

Applicable to all forms of degradation unless an alternative is approved by the USNRC

DUE TO THE EXTENT OF DEGRADATION, ALTERNATE TUBE REPAIR CRITERIA HAVE BEEN DEVELOPED AND IMPLEMENTED

F* criteria is applicable to defects located below the expansion transition. It establishes minimum distance below which the degradation has minimal effect on tube integrity.

Voltage-based limits for predominantly axially oriented ODSCC at the TSPs. Structural and leakage integrity analyses do not rely on the TSPs.

REPAIR CRITERIA INCLUDE ALLOWANCES FOR FLAW GROWTH AND NON-DESTRUCTIVE EXAMINATION (NDE) UNCERTAINTY

Developed from conservative estimates of the growth rates and NDE uncertainty

Conservative margins on structural loading applied (1.4 times postulated accident condition loadings and/or 3 times the normal operating condition loadings)

TUBE INTEGRITY (cont.)

U.S. IS BEGINNING TO USE PROBABILISTIC ANALYSES TO SUPPLEMENT THE DETERMINISTIC TUBE REPAIR CRITERIA TO

**Address the probability of tube rupture and estimate the
leakage under postulated accident conditions**

**Alternate tube repair criteria proposals that may leave
through-wall indications in service have implications for
leakage under postulated accident conditions**

**Account for the use of non-bounding values in the deterministic
analyses**

TUBE INTEGRITY (cont.)

DEVELOPMENT OF A TUBE REPAIR CRITERIA MAY INVOLVE

Developing a correlation between an NDE parameter and tube structural integrity

Physically based (e.g., depth based) or empirically based (e.g., voltage-based) correlations

Estimating the growth rate and NDE uncertainty

Laboratory estimates and/or field observations

Worst case or average values

Assessing whether leak before break is applicable for a specific flaw type

Using probabilistic methods to supplement the deterministic assessments

Evaluating leakage integrity

TUBE REPAIR CRITERIA DIFFER FROM COUNTRY-TO-COUNTRY

A better understanding of the regulatory goals and the methodology and technical basis for the various tube repair criteria could provide mutual benefits

USE OF IN SITU PRESSURE TESTING FOR DEMONSTRATION OF LEAKAGE AND STRUCTURAL MARGIN

PREVENTIVE/CORRECTIVE MEASURES

PREVENTIVE/CORRECTIVE ACTIONS ARE GENERALLY NOT ADDRESSED IN U.S. REGULATIONS WITH THE EXCEPTION OF

Tube repair - plugging

Water chemistry program

UTILITIES, HOWEVER, TAKE THESE ACTIONS AS A RESULT OF ECONOMIC CONCERNS

PREVENTIVE/CORRECTIVE ACTIONS IMPLEMENTED INCLUDE

Sludge lancing

Chemical cleaning

Sleeving - process must be approved by the USNRC

Roto/shot peening

Water chemistry control

U-bend heat treatment

Replacement of copper bearing components in the secondary system

Steam generator replacement

Replacement of Alloy 600 plugs

Rerolling steam generator tubes (full depth expansion)

MITIGATING ACTIONS

MITIGATING ACTIONS ARE IMPORTANT IN ENSURING TUBE INTEGRITY

SLEEVING CONTINUES TO BE A MAJOR FOCUS OF ATTENTION IN THE U.S.

Can provide interim fix until plant shutdown or steam generator replacement

Service induced degradation of the parent tube at B&W kinetically expanded sleeve joints and Westinghouse hybrid expansion sleeve joints has been observed

Sleeve integrity has had a significant drain on USNRC and industry resources with respect to approving and installing the sleeves, respectively

Maine Yankee sleeve installation: ~17,000 tubes for expansion transition defects

Welding and the UT inspection of these welds is complete: >99% acceptance rate

Heat treatment of welds is commencing

PLUG INTEGRITY

Westinghouse Alloy 600 plug replacement schedule has been accelerated as a result of cracking

~5 units have experienced in-service leakage with replacement Alloy 690 plugs as a result of tooling used to remove previous plug

MITIGATING ACTIONS (cont.)

STEAM GENERATOR REPLACEMENTS

13 replacement projects have been completed

8 more replacement projects anticipated in the next 5 years

One of the replacement Catawba steam generators derailed during shipment

USNRC is monitoring utilities recovery plan

Initial visual assessments to be performed in November

EFFECTIVENESS OF VARIOUS MITIGATING ACTIONS CONTINUES TO BE AN ISSUE

Operating experience with Alloy 690

Longevity of sleeve repairs

Improvements in water chemistry

Effectiveness of roto/shot peening and U-bend heat treatment

OPERATIONAL ASPECTS

VARIETY OF PRIMARY-TO-SECONDARY LEAKAGE MONITORING TECHNIQUES ARE USED IN THE U.S.

Blowdown, air ejector, and N-16 main steam line radiation monitors

Grab samples - variety of isotopes are used

IN THE U.S., A SPECIFIC METHOD OF MONITORING FOR LEAKAGE IS NOT SPECIFIED; HOWEVER, LIMITS ON ALLOWABLE PRIMARY-TO-SECONDARY LEAKAGE ARE IMPOSED

TYPICAL LIMIT IN PLANT TECHNICAL SPECIFICATION IS 500 GPD THROUGH ANY ONE STEAM GENERATOR

Most plants administratively adopt much more restrictive limits

Plants which implement certain alternate repair criteria implement 150 gpd leakage limits in their technical specifications

In general, leak before break is not relied upon for any of the degradation mechanisms; however, leakage monitoring is considered an important element for defense-in-depth

LEAKAGE DURING POSTULATED ACCIDENT CONDITIONS IS LIMITED IN THE U.S. TO ENSURE THAT OFFSITE DOSE CONSEQUENCES ARE WITHIN REGULATORY LIMITS

Evaluated with alternate repair criteria proposals that could potentially leave through-wall defects in service

OPERATIONAL ASPECTS (cont.)

IN ADDITION TO LEAKAGE MONITORING, OTHER ELEMENTS WHICH ENSURE SAFE PLANT OPERATION WITH RESPECT TO TUBE INTEGRITY INCLUDE

Assessing the potential for single and multiple tube ruptures and evaluating the associated risk

Training operators to properly diagnose and react to increasing steam generator tube leakage (up to and including single and multiple tube ruptures)

Evaluating expected plant response during single and multiple tube ruptures

Ensuring the effectiveness of emergency operating procedures to address a variety of tube leakage scenarios

EMERGING ISSUES IN THE U.S.

PARENT TUBE CRACKING AT SLEEVE JOINTS

Degradation observed at

Westinghouse hybrid expansion joint sleeves

B&W kinetically expanded sleeves

Long term integrity continues to be an issue

**Alternate tube repair criteria has been developed for
Westinghouse HEJ sleeves**

DETECTION AND REPAIR OF CIRCUMFERENTIAL CRACKS

**Performance demonstration of existing techniques continues to
be an issue**

Development of alternate repair criteria is being pursued

Limitations of inspection techniques must be understood

Growth rates must be quantified

EMERGING ISSUES IN THE U.S.

FREE SPAN CRACKING

Observed at Palo Verde and McGuire units

Several indications recently detected at ANO-2 - licensee is still evaluating these indications

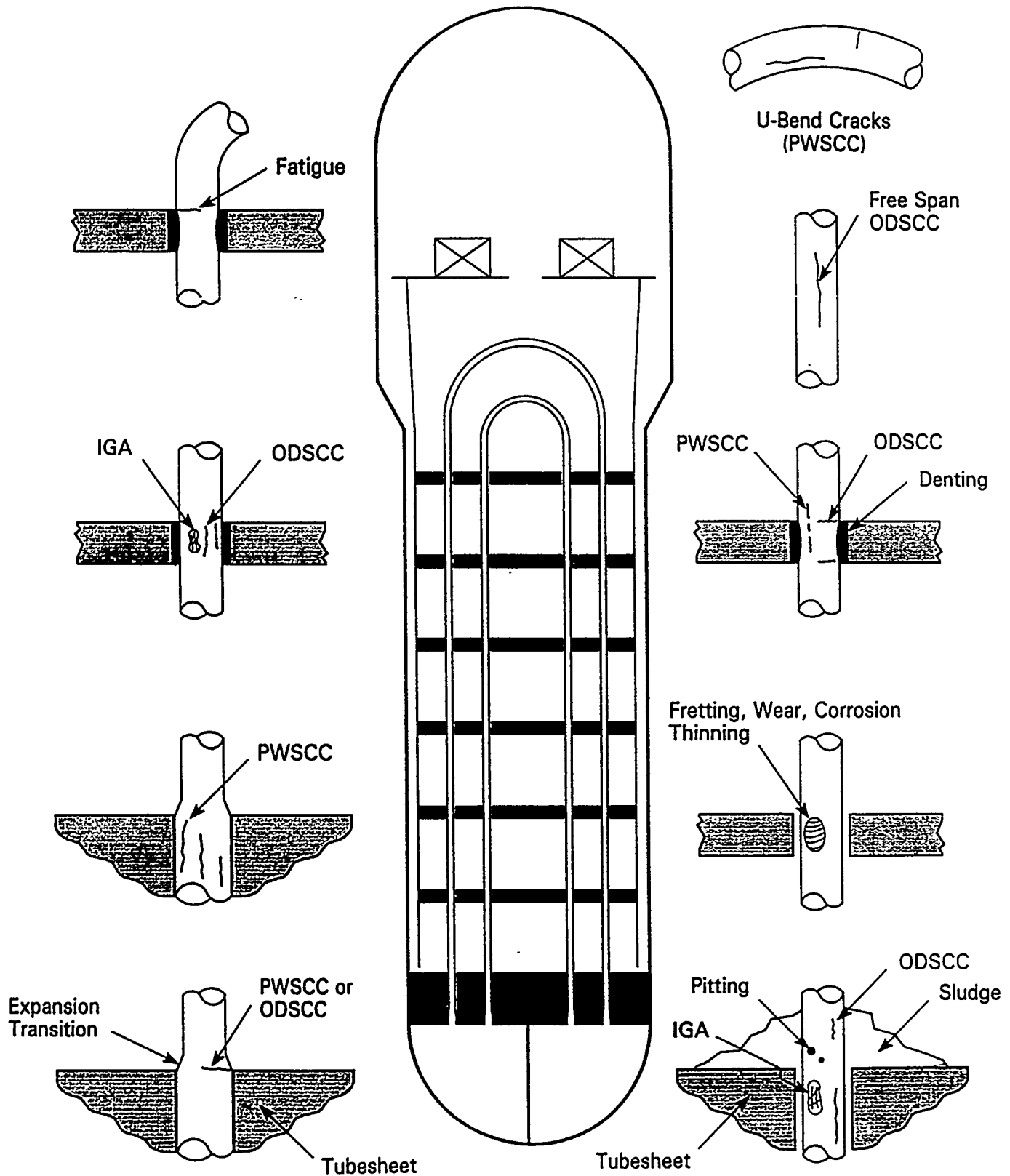
INSPECTION AND CRACKING OF DENTED REGIONS

Axial PWSCC and/or circumferential ODSCC has recently been found at dented TSP intersections

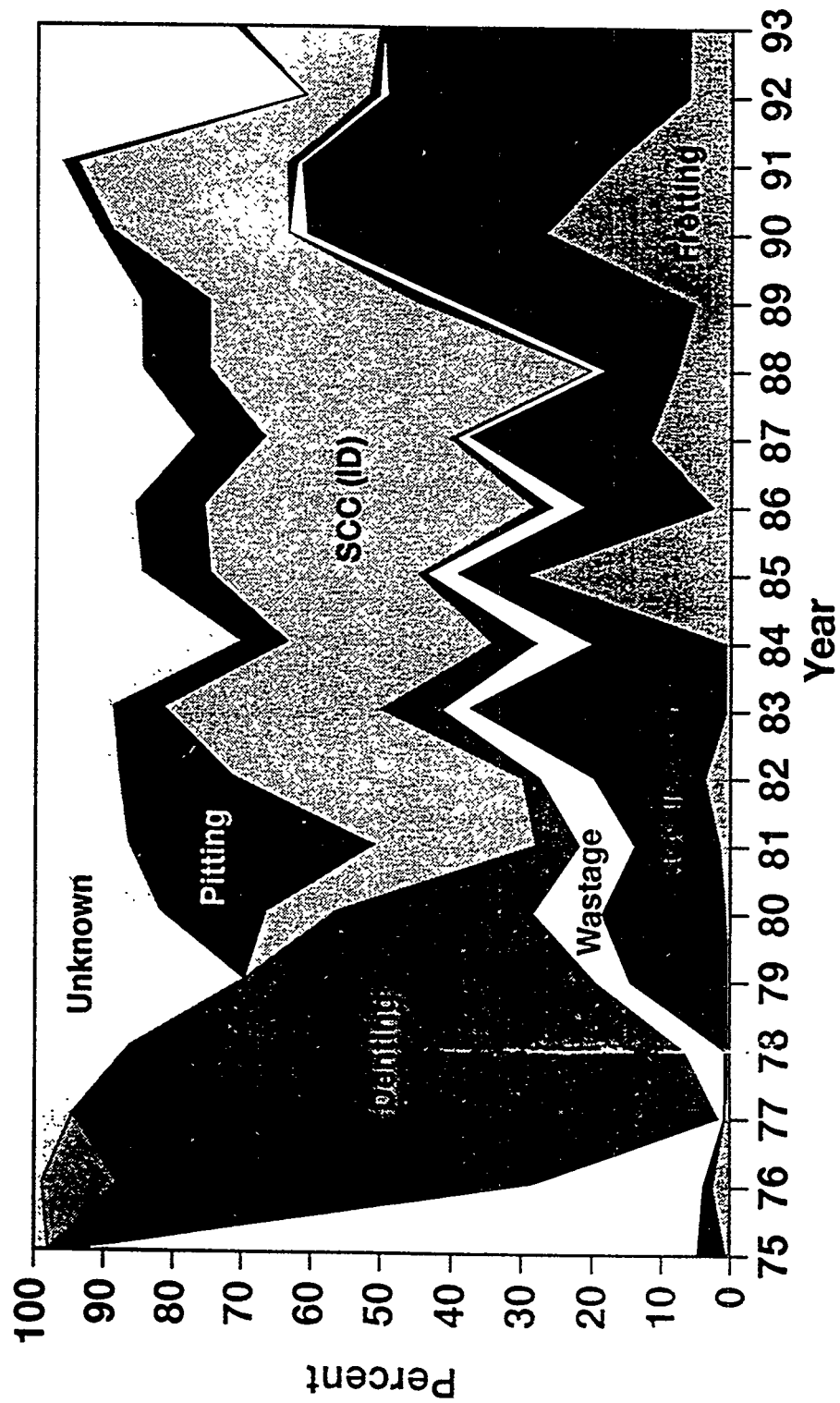
Axial PWSCC has been found at small magnitude dent signals (less than 5.0 volts per Generic Letter 95-05 calibration)

SEVERE ACCIDENT ISSUES

EXAMPLES OF SG TUBE DEGRADATION MECHANISMS



U.S. CAUSES OF STEAM GENERATOR PLUGGING



STEAM GENERATOR REPLACEMENTS

Plant Name	No. of Loops	SG Manufacturer/Model		Completion Date
		Original	New	
Surry 2	3	W/51	W/51F	9/80
Surry 1	3	W/51	W/51F	7/81
Turkey Point 3	3	W/44	W/44F	4/82
Turkey Point 4	3	W/44	W/44F	5/83
Point Beach 1	2	W/44	W/44F	3/84
H.B. Robinson 2	3	W/44	W/44F	10/84
D.C. Cook 2	4	W/51	W/54F	3/89
Indian Point 3	4	W/44	W/44F	6/89
Palisades	2	CE	CE	3/91
Millstone 2	2	CE-67	BWC	1/93
North Anna 1	3	W/51	W/54F	4/93
V.C. Summer	3	W/D3	W/D75	12/94
North Anna 2	3	W/51	W/54F	5/95

NEAR-TERM PLANNED SG REPLACEMENTS

<u>PLANT</u>	<u>YEAR</u>
Catawba 1	1996
Point Beach 2	1996
Ginna	1996
St. Lucie 1	1997
McGuire 1	1997
McGuire 2	1997
Braidwood 1	1998
Byron 1	1999

5 Plenary Session, Part 2: Degradation, Inspection, and Integrity Overviews

- 5.1 *Steam Generator Degradation: Current Mitigation Strategies for Controlling Corrosion*, P. Millett, EPRI
- 5.2 *Steam Generator Tubes Integrity: In-Service Inspection*, R. J. Comby, EDF/GDL/NDE
- 5.3 *Structural and Leakage Integrity of Tubes Affected by Circumferential Cracking*, P. Hernalsteen, Tractabel

Steam Generator Degradation: Current Mitigation Strategies for Controlling Corrosion

Peter Millett
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Presented at CNRA/CSNI Workshop on Steam Generator Tube Integrity in Nuclear Power Plants

October 30-November 2, 1995

Argonne National Laboratory

INTRODUCTION

Steam Generator degradation has caused substantial losses of power generation, resulted in large repair and maintenance costs, and contributed to significant personnel radiation exposures in Pressurized Water Reactors (PWRs) operating throughout the world. EPRI has just published the revised Steam Generator Reference Book¹, which reviews all of the major forms of SG degradation. This paper discusses the types of SG degradation that have been experienced with emphasis on the mitigation strategies that have been developed and implemented in the field. SG degradation is presented from a world wide perspective as all countries operating PWRs have been effected to one degree or another. The paper is written from a US. perspective where the utility industry is currently undergoing tremendous change as a result of deregulation of the electricity marketplace. Competitive pressures are causing utilities to strive to reduce Operations and Maintenance (O&M) and capital costs. SG corrosion is a major contributor to the O&M costs of PWR plants, and therefore US utilities are evaluating and implementing the most cost effective solutions to their corrosion problems. Mitigation strategies developed over the past few years reflect a trend towards plant specific solutions to SG corrosion problems. Since SG degradation is in most cases an economic problem and not a safety problem, utilities can focus their mitigation strategies on their unique financial situation. Accordingly, the focus of R&D has shifted from the development of more expensive, prescriptive solutions (e.g. reduced impurity limits) to corrosion problems to providing the utilities with a number of cost effective mitigation options (e.g. molar ratio control, boric acid treatment).

Many of the mitigation strategies involve changes to the secondary and primary water chemistry, since these are often the most cost effective. In developing new water chemistries, a systems engineering approach is being pursued. In the past, a water chemistry would be designed to solve a specific problem in a single component. Today, it is recognized that the chemistry needs to be optimized for the entire power plant, not just the SG's. Since the primary and secondary systems contain a number of different materials of construction which are exposed to varied environmental conditions, no one chemistry can be best for the entire system. The concept of optimization applied to mitigation strategies means that the overall economic impact on the entire plant is

considered when pursuing a given strategy. Much emphasis is being focused on developing tools which the PWR operators can use to perform these evaluations.

Figure 1 shows the loss of capacity factor attributed to corrosion problems in US PWRs for the years 1980 through 1994. Pressurized water reactors (PWRs) experienced an average capacity loss of more than 5% and a peak of nearly 8% in 1982. It is estimated that SG corrosion problems have cost the US nuclear utility industry billions of dollars. The data in Figure 1 shows that despite the fact that the PWR fleet is aging, capacity factor losses have continually improved, although the levels have somewhat stabilized over the past few years. It is anticipated that further improvements will be realized as a result of the implementation of improved water chemistry treatments and improved inspection and repair techniques. Radiation exposure trends in US LWRs are shown in Figure 2. The radiation exposure trend which has steadily decreased reflects in part reduced maintenance activities associated with SG corrosion and improvement in SG replacement programs. The challenge for the future is to improve upon the successes of the past illustrated in Figures 1 and 2, while simultaneously reducing costs.

Corrosion of steam generator tubes in PWRs has been the most important in terms of their impact on power production. Figure 1 shows that SG corrosion has been the major cause of loss of power production in PWRs. SG corrosion has resulted in the permanent shutdown of one PWR in the US due to IGSCC of the SG tubing. The changing economic climate of the utility industry may force other older PWR utilities to shutdown rather than replace their aging steam generators. For most utilities, the preferred strategy is to delay major repair or replacement decisions until improved financial conditions exist. Many of the mitigation strategies discussed in this paper are based on this philosophy.

In the remainder of this paper, the types of SG degradation which have been experienced and mitigation strategies developed are discussed. Special emphasis is placed on advances in water chemistry technology to remediate corrosion damage, and new techniques for analyzing the corrosive environment in SG crevices. This paper focuses on recent advancements. The reader is referred to reference 1 for a more historical review.

STEAM GENERATOR TUBE DEGRADATION

Steam generators are large and expensive components that were designed for an expected service life of up to 40 years. Two types of steam generators are in use in PWRs: recirculating steam generators (RSGs) and once-through steam generators (OTSGs). This paper focuses primarily on RSGs since this design is more universally employed. Most, if not all, of the remedial actions and mitigation strategies, however, are applicable to both designs. Some service-induced damage to steam generator tubes was anticipated, however, by the mid-1970s, steam generator tubes in many RSGs were being plugged at a rate which indicated that the design margin would be exceeded well before their 40 year design life. Since 1977, EPRI under the direction of the US and foreign utilities has conducted a major research program to help utilities improve the performance of their steam generators. As a result, most of the initial types of damage have been eliminated. Improved steam generators incorporating design enhancements derived from both laboratory and field R&D have operated for well over a decade with virtually no

corrosion related problems. Current new and replacement steam generators are anticipated to operate for the remaining life of the plants and will support life extension in many cases. However, many older design units remain in service. While some operate successfully with minimum degradation, others have deteriorated to such an extent that some utilities now face a particularly difficult and expensive decision: whether to replace their current steam generators at costs of around \$100-200 million per plant, to de-rate the plant due to a reduction in heat removal capability as steam generator tubes are plugged, or to shut down certain plants entirely.

Numerous steam generators have been replaced world wide. The timing of these replacements is based on operational and economic factors, which are influenced by both regulatory and political considerations. A marked trend towards more rapid, and less costly replacements coupled with lower radiation doses has occurred as more units are replaced.

DEGRADATION FORMS AND MITIGATION STRATEGIES

PWSCC

Degradation of the SG tubing occurs on both the primary and secondary side. Primary side corrosion has effectively been limited to stress corrosion cracking (SCC) of the tubing in regions of high tensile stresses, including the tube to tubesheet expansion zone and inner row u-bend regions. Operationally induced stresses associated with denting and sleeving have also resulted in PWSCC. From a mechanistic view, PWSCC or pure water cracking is not completely understood. Primary water chemistry is believed to have at most a second order effect on PWSCC. The motivation for optimizing primary water chemistry is to control radiation fields in the primary system.

R&D programs conducted throughout the world have led to the identification of materials and heat treatments that are much more resistant to PWSCC than the mill-annealed Alloy 600 used in most original equipment generators. These improved materials including Alloy 690 and Alloy 800 are being used in new and replacement units. To date these new materials have not yet experienced PWSCC. These materials also have proved to be more resistant to secondary side attack in plants that have replaced SG's. To prolong the life of the older generators, stress improvement processes (annealing and shot peening) have been developed and widely used. PWSCC is very temperature sensitive, and at some plants the primary system temperature has been reduced to further extend generator life. Sleeving also has been employed as a repair method to avoid the need to plug damaged tubes. Another repair method which looks promising involves insitu welding of the tube at the damage location and also nickel plating.

More recently, zinc additions have been shown to be effective in reducing primary water SCC rates in laboratory testing. Zinc additions also have the side benefit that radiation fields are reduced. The first demonstration of zinc additions to the primary system of a PWR was temporarily discontinued due to unusual fuel deposits. The investigation of the fuel deposits is ongoing, but preliminary results suggest that the deposits are not

deleterious. The pH in the primary system is controlled by the lithium and boron levels. It has been found that the activation and subsequent transport of corrosion products to out of core surfaces can be minimized by operating at elevated pH's. Recent evaluations have shown that the elevated lithium concentrations have little impact on PWSCC. EPRI has recently published revised Primary Water Chemistry Guidelines 2 addressing this and other primary side chemistry issues.

Secondary Side Degradation

The primary focus of this paper is secondary system degradation since many forms of corrosion have effected SGs on the secondary side and several mitigation strategies have been developed and implemented. Most forms of secondary side SG degradation have been associated with the concentration of impurities locally in flow starved regions, such as the crevice formed between the tube and support plates, or underneath corrosion product deposits which form on tube supports, tube sheet surfaces, and free span surfaces. The concentration of impurities in these regions can be orders of magnitude greater than the bulk water. These aggressive solutions have caused a number of forms of corrosion. Since changing the material properties in situ is difficult, most of the remedial actions have focused on modifying the crevice chemistry to control the corrosion of the SG tubing. An understanding of the crevice solution is necessary to mitigate corrosion, so much effort has focused on developing insights into how crevices concentrate solutions, what chemistries exist in crevices and how potential inhibitors might work in the crevice. In the remainder of this paper, a brief historical review of SG degradation forms effecting PWR operators over the past two decades is followed by a more detailed discussion of the current degradation problems. Because control of SG secondary side degradation ultimately involves modifying the crevice chemistry, the paper includes a review of crevice chemistry and a description of some new diagnostic techniques developed to interpret the crevice chemistry in the field. These techniques lead to several options for controlling SG degradation.

Historical Overview of Steam Generator Degradation Forms

Figure 3 shows the evolution of major SG degradation mechanisms in world wide PWRs. Initially, PWRs adopted sodium phosphate based water chemistry on the secondary side based on fossil experience. Beginning in 1974, many utilities ceased using sodium phosphate due to general wastage of the Inconel SG tubes. Wastage was believed to result from either a concentrated phosphoric acid solution or an insoluble sodium phosphate solution phase which forms when sodium phosphate solutions concentrate in the crevice. Wastage was essentially eliminated with the introduction of All-Volatile Secondary Water Treatment (AVT). AVT originally consisted of adding hydrazine to the water to scavenge dissolved oxygen and ammonia to control pH. AVT water treatment solved the wastage problem but another corrosion issue, denting, soon followed.

Denting of steam generator tubes at tube support plate intersections occurs as a result of corrosion of the carbon steel support plate in the crevice between it and the tube. Research showed that denting was associated with acid chloride attack and strongly oxidizing conditions caused by reducible copper oxides. It was found that denting could be prevented or arrested by improved control of secondary water chemistry. As a result,

PWR operators improved their water quality, specifically by retrofitting condensate polishers, replacing copper alloys, and by paying strict attention to condenser leaks. Early versions of EPRI's Secondary Water Chemistry Guidelines incorporated these and other improvements such as boric acid treatment, and as illustrated in Figure 3, proved to be very effective at controlling denting.

During the 1980s, several slower-developing forms of tube degradation emerged including pitting, vibration-induced fretting and fatigue, and secondary side stress corrosion cracking and intergranular attack (ODSCC/IGA). Intergranular corrosion that originates on the outside of tubes, particularly at or near tube support plates as well as axial and circumferential SCC at the top of the tubesheet has become a leading concern. If not controlled soon, it is likely that this secondary side IGA/IGSCC could limit the useful life of the steam generators involved to 20-25 years. At that age, the economic decision of whether to install new steam generators becomes particularly difficult, since nuclear plants are licensed for 40 years and there may not be enough remaining life for some utilities to recover the additional investment.

Mitigation Strategies for Secondary Side IGSCC

In terms of prevention, progress has been reported in reducing secondary side IGA/IGSCC through even more aggressive water treatment. Figure 4 shows the pace of implementation of several recent initiatives to control IGA/IGSCC. These initiatives were established in 1992, when EPRI revised the secondary water chemistry guidelines³ to provide assistance to those utilities experiencing rapid IGA/SCC of Alloy 600 tubing. Table 3 taken from the EPRI Secondary Water Chemistry Guidelines shows current SG blowdown water chemistry specifications to control IGA/IGSCC and other localized forms of corrosion.

Table 1
Recirculating Steam Generator Power Operation (> 5% Reactor Power) Blowdown
Sample

CONTROL PARAMETERS

Parameter	Frequency	Median Value	Action Level		
			1	2	3
pH @ 25°C	continuous	(c)	(c)		
Cation Conductivity, $\mu\text{S}/\text{cm}$ non-ammonia AVT)	continuous	0.15	>0.8	>2	>7
Cation Conductivity, $\mu\text{S}/\text{cm}$ (non-ammonia AVT)	continuous	0.5	>0.8	>2	>7
Sodium, ppb	continuous	2	>20	>100	>500
Chloride, ppb	daily	2	>20	>100	
Sulfate, ppb	daily	3	>20	>100	
Boron, ppm ^a	daily	6	(6)		
Phosphate ^a	(b)	(b)	(b)		

DIAGNOSTIC PARAMETERS

Parameter	Consideration
Specific Conductivity	Consistency between Amines pH, amine, boron and specific conductivity is important.
Hideout Return Analysis	Perform each planned shutdown. Cation-to-Anion Ratio
Silica	Control to prevent formation of highly acidic or alkaline crevices. Crevice chemistry, steam quality and impurity source considerations

Footnotes: (a) If used. (b) As specified in site-specific chemistry program. (c) Per station pH program.

A detailed discussion of some of the major initiatives in the EPRI Water Chemistry Guidelines follows. In addition to the guidelines, EPRI has developed a number of computer tools to help utilities select and implement the appropriate water chemistry program. These tools are used in conjunction with the water chemistry guidelines to perform specific analyses to support the development of hideout return, MRC and pH control programs.

Molar Ratio Control

The revised water chemistry guidelines formalized a methodology known as Molar Ratio Control (MRC) for reducing the growth rates of IGA/SCG. The practice of MRC is based on the assumption that the crevice pH influences crack initiation and growth and can be modified by controlling the ratio of strong acid anions and strong basic cations in the steam generator bulk water. By correlating the bulk water chemistry to the crevice chemistry, appropriate changes can be made to the bulk water to proactively control the crevice chemistry. Since the pH in the crevice cannot be directly measured, the first step in this methodology is to establish the appropriate target crevice chemistry. The target must consider

both the parameters which affect corrosion of the SG tubing, and be based on parameters which can be reliably measured or estimated in practice.

Reference 4 provides a review of the technical basis for MRC. Specifically, a correlation between crevice pH and IGSCC crack growth rate has been developed based on crack growth rates measured in a number of laboratory studies over the past two decades. Figure 5 taken from reference 4 shows the relationship between the at temperature pH (nominally 300°C) and the IGSCC rate of mill annealed alloy 600 steam generator tubing. Although the data is scattered, it can be concluded that the lowest crack growth rate is achieved in the pH range from 5-9. A target crevice chemistry based on its measured pH would be desirable. Since the crevice pH cannot yet be directly measured, the pH must be predicted from the composition of the solution accumulated there. The composition of the solution must be estimated from either bulk water chemistry measurements or hideout return measurements as will be described later.

Molar Ratio Control (MRC) has now been adopted at nearly 50 percent of US PWRs with older SG designs. Reference 3 suggests operating at a target SG Na/Cl ratio of nominally 0.5 as a starting point. This ratio should be modified based on evaluation of hideout return data, and can include other species such as potassium or sulfate. Many plants historically operated with a Na/Cl ratio greater than one during the time period when IGSCC was first detected. Although, the preferred method for reducing the molar ratio is by the reduction of sodium, in many cases this requires major modifications to the condensate demineralization system. Alternatively, MRC is achieved by adding ammonium chloride at a low level to the SG. Reference 4 provides an upper limit for the SG chloride concentration of 5 ppb during normal operation. It is believed that this limit in conjunction with elevated hydrazine (see next section) and boric acid treatment will insure that acid chloride forms of corrosion are not reinitiated in PWRs practicing MRC. MRC has also been used in Japan for some time. To date, no firm conclusions can be drawn as to the effectiveness of MRC in reducing IGSCC in US plants. The preliminary conclusion reported in reference 4 is that the use of MRC in conjunction with other chemistry control measures such as, elevated hydrazine, boric acid treatment and as low as reasonably achievable (ALARA) chemistry give the highest probability of controlling IGSCC. These other chemistry control measures are discussed in the following sections.

Elevated Hydrazine

In addition to MRC, the use of an elevated hydrazine concentration in the feedwater is also recommended to insure reducing feedwater conditions. In the US, elevated hydrazine control defined as >100 ppb in the feedwater is used in nearly half of the PWRs. The use of condensate polishers and residual copper alloyed components in the remainder of the PWRs prevent adoption of higher hydrazine levels. The initiation and growth of IGA and IGSCC in Inconel 600 Steam Generator (SG) tubing has been shown to be a strong function of the local electrochemical potential (ECP). Corrosion is accelerated by oxidizing conditions in caustic, acid and neutral environments. The ECP in the crevice cannot be easily measured and is difficult to calculate. It is reasonable to anticipate that it will be strongly influenced by redox species, such as oxygen, hydrazine

and hydrogen, and reducible metal oxides (e.g. copper oxides and hematite) if they are present in the crevice.

Hino, et. al. ⁵ measured the effect of various oxidants on the ECP of Inconel 600 at 280°C in a 40% caustic solution. They found that the reducible oxides (CuO, Cu₂O, Fe₂O₃, and PbO) can raise the potential by 100-550 mv into a region where rapid IGSCC is observed in laboratory tests. Magnetite did not raise the potential above the baseline in these tests. This data suggests that the most conservative approach is to minimize the amount of hematite and copper oxides being transported into the SG's .

The amount of excess hydrazine required to maintain reducing conditions may be plant specific. Data from direct electrochemical measurements at the Biblis A plant in Germany indicate that the ECP of Alloy 800 continues to decrease up to a feedwater concentration of 100 ppb hydrazine ⁶. The feedwater ECP was also measured at the St. Lucie plant⁷. The measurements at St. Lucie indicated that the minimum hydrazine concentration should be 7-8X of the condensate oxygen concentration. More recent measurements at several US PWRs have suggested that if the condensate oxygen concentration is low (e.g. < few ppb), the feedwater ECP is independent of the ratio of hydrazine to oxygen. ECP measurements from several plants throughout the world are summarized in Figure 6. In all cases, the minimum ECP appears to correlate with the expected potential of the hydrogen electrode under feedwater conditions.

Although the ECP may reach a minimum at 100 ppb hydrazine or less, the oxidation states of the metal oxides is also important. Many Japanese PWRs operate with up to 600 ppb hydrazine in the feedwater based on the observation that hematite is not found in the feedwater or blowdown in plants that operated at these elevated hydrazine concentrations. EPRI has been compiling data on the composition of iron corrosion products transported throughout the secondary system. Figure 7 shows the fraction of magnetite in SG blowdown corrosion product samples plotted against the ratio of feedwater hydrazine to condensate oxygen. Although the data is scattered, there is a clear trend which indicates that elevated hydrazine concentrations promote the formation of magnetite in the SG blowdown. The data seems to indicate that a ratio of feedwater hydrazine to condensate oxygen of at least 30-40 is required to convert all of the reducible iron oxides (hematite, etc.) to magnetite.

Inhibitors

Boric Acid Treatment (BAT) was originally used to control SG tube denting. Subsequent laboratory work⁸⁻¹⁰ indicated that the use of boric acid may also be beneficial in controlling secondary side IGA and/or SCC. The exact mechanism for inhibition is not known. It may be neutralization of a caustic environment and/or modification of the oxide film on the tubing making it more protective. Figure 4 shows that approximately 30 percent of US PWRs are currently using BAT. Most of these plants indicate that they are using BAT for IGSCC control. Although data from the laboratory testing of BAT is quite positive, the field data is less clear. Weibull statistics have been used to evaluate the time to failure of SG tubes before and after BAT application¹¹. The interpretation of the field data in reference 11 suggests that on average a small benefit has been realized

due to the use of BAT. Conventionally, BAT is applied by maintaining 5-10 ppm boron in the SG during operation. Many plants conduct a low power soak during startup. This procedure evolved from the use of BAT for denting control. Although, the benefits are not absolute, many utilities feel that the risks and costs of using BAT are quite small in comparison to the potential benefits..

More recently, Lumsden¹² has shown that the use of titanium oxides strongly inhibit caustic IGSCC in static autoclave testing. In C-Ring testing, Lumsden found that the number of cracked specimens increased at a given test duration as the applied ECP was increased to 200 mv above the free corrosion potential. The use of various titanium oxide compounds was able to completely inhibit cracking in these tests where boric acid was only partially effective. Follow on tests performed in another laboratory¹³ using the constant extension rate test (CERT) confirmed Lumsden's results. The static autoclave tests were followed by a series of model boiler tests¹⁴. Since, titanium compounds have extremely low solubilities, it was postulated that the inhibitor would not be able to penetrate fouled crevices. In fact, the titanium additives were found to be less effective in the model boiler testing when the crevices were prepacked to simulate fouling. The titanium additives were more effective at inhibiting cracking in initially open crevices. These results suggest that implementation of a titanium based inhibitor may be more effective immediately following a chemical cleaning or before the crevices are significantly fouled. Several plants have gone forward with full scale additions of titanium oxides. To address the question of access of the crevice to titanium, a destructive examination of a tube support/tube intersection removed from a recently retired SG is planned. Titanium was injected prior to shutdown of the plant. As with BAT, the potential benefits of adding the inhibitor greatly outweigh the small risk of deleterious side effects. One potential side effect is the fouling of the SG tubes due to deposition of the inhibitor on the tubing. This potential adverse effect is currently being evaluated. It should be noted that testing in neutral or acid conditions has not yet been performed. Laboratory work continues to identify other colloidal titanium compounds that may be able to penetrate SG crevices. Other inhibitors based on cerium chemistries have also been investigated.

Corrosion Product Control

Corrosion products which deposit in the crevices and on tube support and tube sheet surfaces aggravate corrosion by promoting the concentration of impurities. Organic amines are being used in the secondary water to reduce the transport of iron, which contributes to sludge buildup. Amines such as, ethanolamine, provide better pH control and thereby reduce carbon steel corrosion in the balance of plant¹⁵. Figure 8 shows the required feedwater concentrations of several amines qualified for use in PWRs. The objective is to raise the pH in two phase regions of the steam cycle, such as the moisture separator reheaters, where most of the carbon steel corrosion occurs. Figure 8 shows the advantage of several alternate amines qualified for use in PWRs. Since much lower concentrations of these amines need to be added to the feedwater compared to ammonia, and less is carryover to the condenser, loading of the pH control agent on the demineralizers is reduced. This reduces the amount of impurity leakage of aggressive species such as sodium and chloride from the demineralizers. pH control using

advanced amines is a good example of how a systems approach to chemistry control is being pursued to address SG corrosion control issues.

Since the generation and transport of corrosion products cannot be completely eliminated, another approach to damage prevention is removal of corrosion products once they are deposited in the SG's. Corrosion product fouling in both types of steam generators (recirculating and once-through) has been identified as a major cause of heat transfer degradation in PWR plants, with the output at some plants being decreased to as low as 80% of full power. EPRI research has led to development of a process and general guidelines -- now widely used -- for chemical cleaning of sensitive areas in steam generators, particularly the surfaces of tube sheets and support plates. In addition, joint work with Consolidated Edison Company and the Empire State Electric Energy Research Company (ESEERCO) led to development of a remotely operated robot, known as CECIL, that can reach between tubes and perform a variety of maintenance tasks. Equipped with a miniature video camera and fiber optics for illumination, CECIL can inspect tube exteriors directly, drill into sludge piles for samples, and direct a high-pressure water lance at hard-to-reach crevices.

Improved Diagnostics for Interpreting and Implementing Mitigation Strategies

As previously indicated, most forms of SG degradation have occurred in regions where impurities concentrate in SGs such as, tube to support crevices, sludge piles and under thick corrosion product tube deposits. The effectiveness of the mitigation strategies is dependent on the crevice chemistry. As such, much effort has focused on developing an understanding of the underlying processes in crevices. This work has led to some new diagnostic techniques which can be used by plant personnel to evaluate the crevice chemistry and identify appropriate corrosion mitigation strategies.

Local Concentration Processes

Figure 9 shows the locations where impurities accumulate within PWR SG's. In this paper, these locations are collectively referred to as crevices, and the local chemistry within these local regions as the crevice chemistry. In crevices and on vertical and horizontal tube surfaces, corrosion products and particulate matter accumulate in the form of porous deposits. The SG water contains impurities at extremely low levels (ppb). Low levels of non-volatile impurities, however, can be efficiently concentrated in fouled crevices and sludge piles by a thermal hydraulic mechanism. The temperature gradient across the SG tube coupled with local flow starvation, produces boiling of the pore solution. Since mass transfer processes are inhibited by the porous corrosion product, the residual pore liquid becomes enriched in the non-volatile species which are present in the SG water. This concentration process can also occur in the absence of the corrosion product in open crevices if the tube is contacting the support or tube sheet, such as in a broached tube support plate. The presence of corrosion product aggravates the situation.

The formation of concentrated solutions is the precursor to corrosion and therefore, the detailed mechanism must be understood so SG designs and operating procedures can be improved to minimize corrosion. Since this concentration process (hideout) is the

precursor to corrosion, the mechanism for this phenomenon has been studied by many investigators. A recent review article of concentrating mechanisms was written by Millett and Fenton¹⁶. Experimental studies of crevice concentration processes have stimulated the development of a number of mathematical models which describe how impurities accumulate in local regions of the SG¹⁷⁻²¹. The models and experimental data suggest that the mass of impurity in the crevice can be correlated to the bulk water concentration multiplied by the exposure time (ppm x hrs). After the crevice is nearly full of concentrated solution, the accumulation rate goes to zero. This simple concept has been shown to be adequate for describing the accumulation rate in both packed and dented support plate crevices and sludge piles. On the other hand, the models and experimental data suggest that although the accumulated mass is proportional to the bulk water concentration, the concentration in the crevice reaches extremely high levels in equilibrium with the available superheat very quickly, independent of the bulk water concentration. The models imply that corrosive conditions can exist in the crevice, albeit very locally, almost immediately after the heat flux is initiated. This explains why ALARA chemistry has not been completely effective in stopping SG corrosion. The concept of MRC is to insure that this concentrated solution is always benign. In adding impurities to the system to balance the molar ratio, this is done at the expense of accumulating more impurity mass in the crevice. In practice, at normal impurity levels, it is unlikely that the crevices ever completely fill with concentrated solutions, since the plant is shutdown prior to the time required for the crevices to fill completely.

Crevice Chemistry

All of the concentration models discussed so far describe the thermal hydraulic and mass transfer processes for a single specie concentrating in the crevice. These models can in theory be extended to more complex crevice solutions, however, this would make the models nearly intractable. As a first approximation, EPRI has addressed this issue in the MULTEQ²² code by providing several simplified models of the concentration process. These models assume that the system is in thermodynamic equilibrium at all times. Rate processes such as the reaction kinetics of the solid metal surface interacting with the local environment are not included in the present models. The thermodynamic model does account for complex ion formation, solute precipitation, and distribution of the volatile neutral species into the steam phase. Redox reactions are considered in one version of the code, so that the redox potential can be calculated. These thermodynamic processes are important and are believed to dictate the solution composition. Unfortunately, the concentration model in MULTEQ is not rigorous enough to be used directly with bulk water chemistry measurements to predict the crevice chemistry. Additionally, the impurities within the SG are often at or are near the limit of detection. Uncertainties associated with measurements at these low levels make the modeling calculations extremely uncertain.

Two alternative approaches can be taken to predict the crevice chemistry. The first approach is to measure the bulk water impurities on a real time basis and through empirical measurements, assess the rate at which the bulk water impurities are concentrating in the crevices. This rate is then integrated over time to yield the crevice inventory. The crevice pH is calculated directly from the inventory. In this method, the average "hideout" or concentration rate is measured for the entire steam generator²³. Measurements in operating

plants have shown that the average "hideout" rate is proportional to the bulk water concentration for most impurities, in agreement with the models of crevice concentration as previously mentioned. This approach has been incorporated into an algorithm known as CREV-SIM²⁴ designed to track the crevice chemistry on a real time basis. So far, this approach has not been entirely successful. As the models suggest, it is likely that the "hideout rate" is both a function of the crevice chemistry (e.g. pH, potential, etc.) and the physical condition of the steam generator crevices (e.g. porosity, permeability, etc.). Many of these variables change quickly enough that frequent measurement of the "hideout" efficiency may be required to improve this technique as a method of predicting crevice chemistry.

Another approach is based on an evaluation of the "hideout" return chemistry in the steam generator when the plant is shut down. When the heat flux is reduced, the concentrated solution can diffuse out of the crevice and into the bulk water. In practice, this is observed and is detected as an increase in concentration of impurities in the bulk water. This approach to assessing the crevice chemistry has two limitations. First, only an average crevice chemistry across the thousands of crevices can be inferred, and secondly, the hideout return can only be measured perhaps once per year or less frequently when the plant shuts down. Nonetheless, evaluation of hideout return data is believed to yield the most useful insight into the crevice chemistry at this time, although there is not complete agreement as to the source of the hideout return in the plant.

A Molar Ratio Index (MRI) is defined based on hideout return data as follows:

$$\text{MRI} = (\text{Na} + \text{K}) / (\text{Cl} + \text{SO}_4) \quad (1)$$

The MRI is based on the relative ratio of impurities that are believed to be returning from the crevice solution during hideout return. The excess sulfate term takes into account the fact that sulfate in excess of that required to produce a precipitate of calcium sulfate may exist in the crevice. If sulfate is not found to be in excess of calcium, sulfate is excluded from the MRI. Use of the MRI does not discount the effect of the other insoluble impurities on the crevice pH(t). A detailed discussion into the theory of the MRI is beyond the scope of this paper. Interested readers are referred to reference 4 and 28.

It is believed that attempts to modify the crevice solution to produce a MRI as close as possible to 1 gives a higher probability of achieving a crevice pH within the target range for minimizing localized corrosion. The calculated inventories to be used in the MRI can be assessed from a diffusion based hideout return model. Although this approach is believed to be more conservative than the conventional technique using tools like MULTEQ, it is still useful to employ MULTEQ to develop insight into the composition of the crevice chemistry. If careful analysis of the hideout return data and MULTEQ warrants targeting an MRI other than 1, this course of action should be pursued cautiously with an appropriate study to evaluate the sensitivity of all species used in the calculations.

Status and Future Prospects for Steam Generator Tube Degradation

R&D to develop remedies for the various forms of steam generator tube degradation has been very beneficial to US utilities. Average capacity factor losses (see Figure 1) have been decreased from 5.5% (1980-1984) to 3% (1987-1994), average steam generator life has been substantially extended and the forced outage rate due to tube leaks has been reduced by a factor of ~7 from its peak in 1976. US utility estimates put the present value of these improvements at several billions of dollars.

However, for the past several years, worldwide tube plugging rates which were trending down in the early 1980s, have been increasing slowly and currently are at values which suggest that the average steam generator life will be 20 to 25 years. Thus, most original equipment steam generators may have to be replaced within the 40-year plant license life. Because the average age of installed steam generators in the US currently is ~15 years, many replacements may be necessary in US plants within the next 10 to 15 years unless remedial measures and/or improved repair methods can be developed and deployed for ODS/IGA, which currently appears to be the damage form that will limit the life of most generators.

The challenge for the near future is to make further improvements in capacity factors as the nuclear fleet continues to age. As competitive pressures continue to influence decisions that effect corrosion damage in PWRs, the challenge is to make further improvements in a cost-effective manner. This means that utility remedial actions will become more plant specific as individual utilities weigh the risks and benefits of the remedial actions available to them. This is true especially for water chemistry treatment programs which must be optimized on a plant specific basis. Materials of construction, the existence and capacity of cleanup and ion exchange systems dictate which mitigation strategies can be pursued. The R&D programs so far have led to a good but incomplete knowledge of the mechanisms for concentrating impurities in SGs and the underlying corrosion processes themselves. Mitigation strategies and techniques for evaluating what conditions exist in the SG crevices have been developed and are now being implemented based on the current understanding of these phenomenon.

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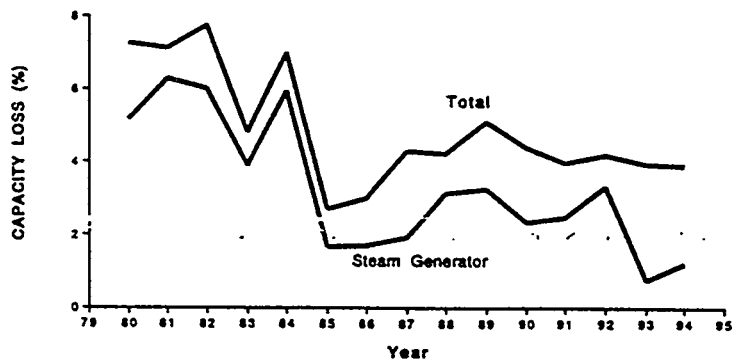


Figure 1. PWR capacity factor losses due to corrosion.

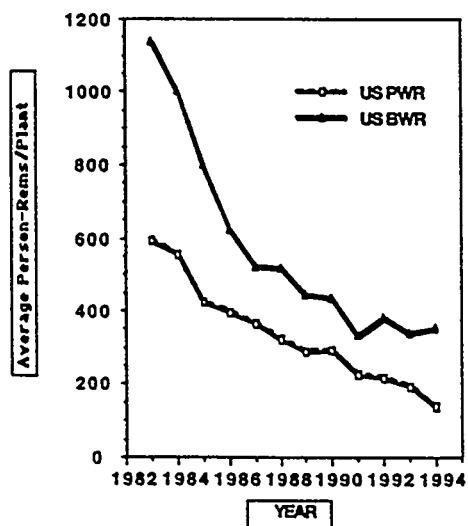


Figure 2. Radiation exposure in US nuclear plants.

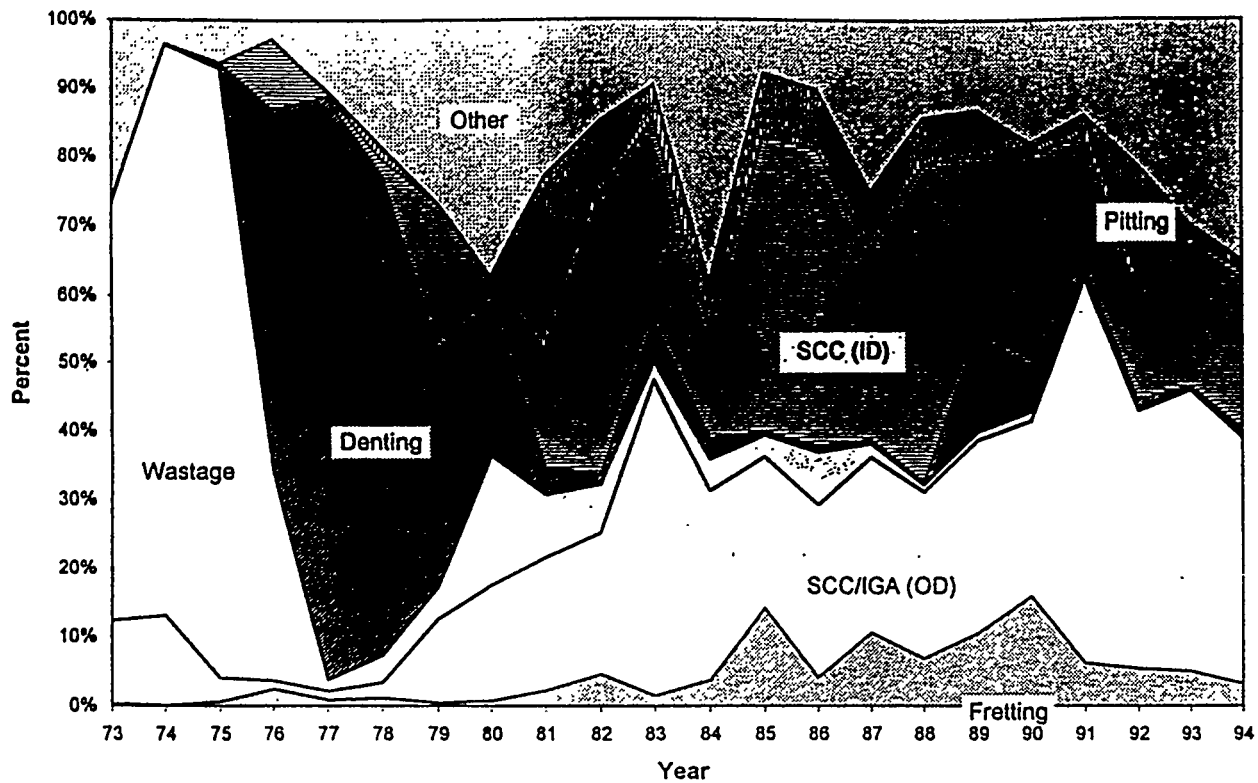


Figure 3. World wide causes of SG tube plugging.

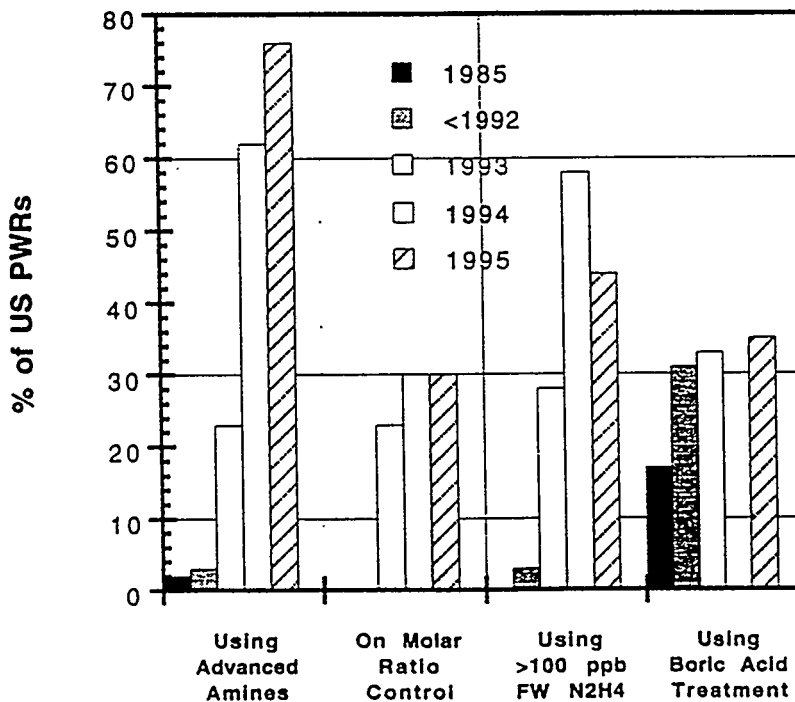


Figure 4. Implementation of water chemistry initiatives in US PWRs.

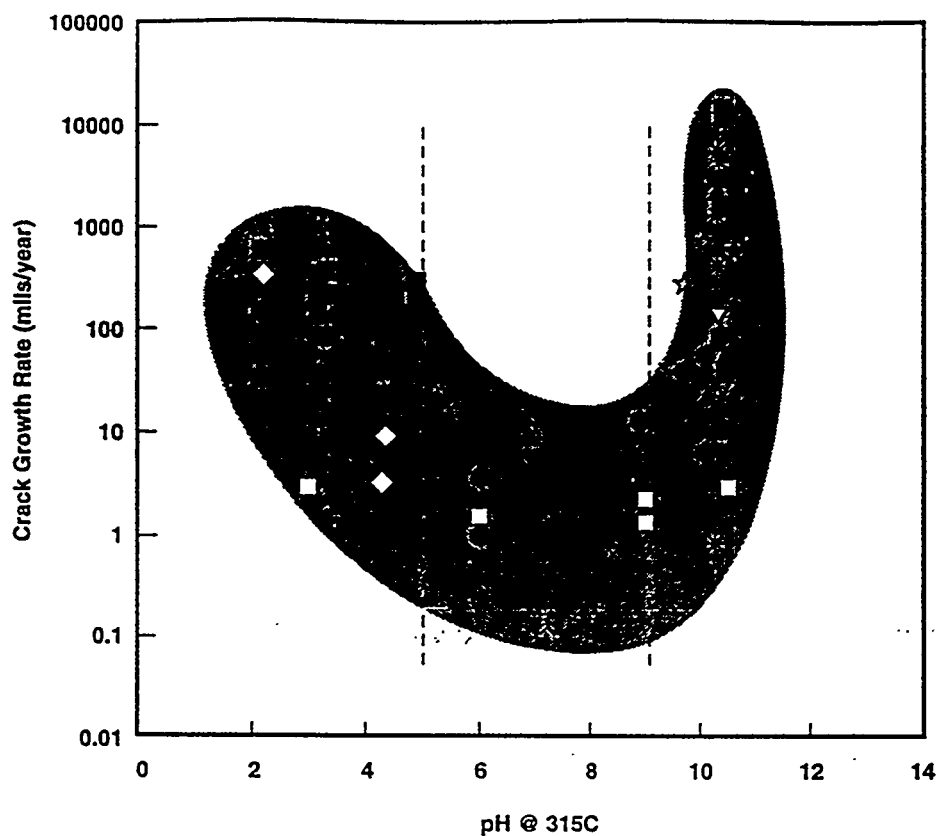


Figure 5. Crack growth rates of mill annealed Inconel 600 (taken from reference 4).

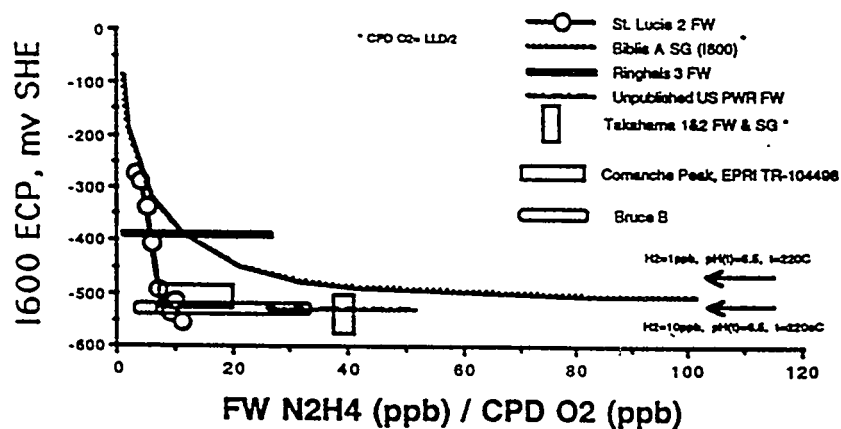


Figure 6. Electrochemical corrosion potential measurements in the feedwater of PWRs.

Relationship between % Magnetite
in FW Iron to Ratio of FW N₂H₄ to CPD O₂

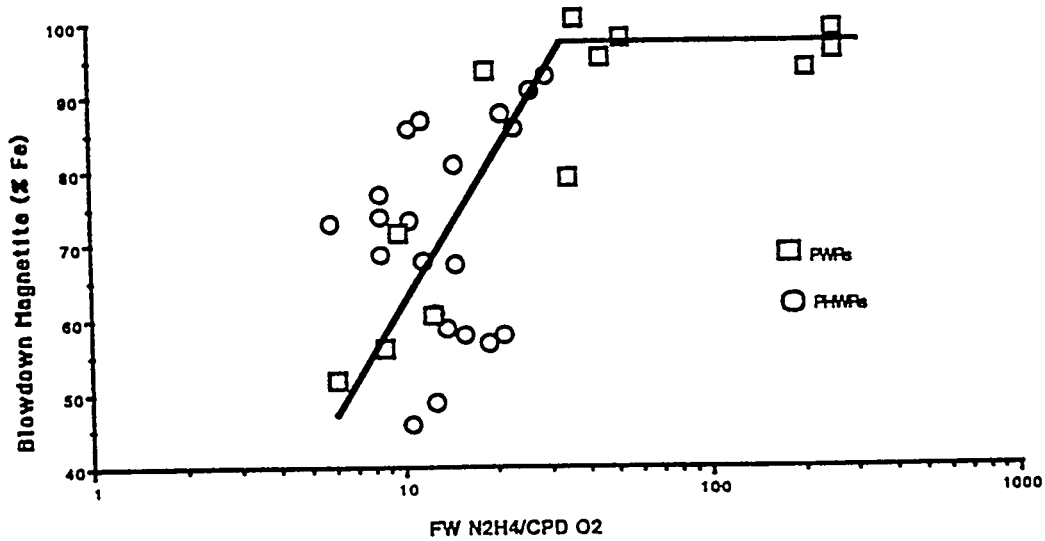


Figure 7. Relationship between percent magnetite in blowdown corrosion product samples with hydrazine and oxygen levels in PWRs.

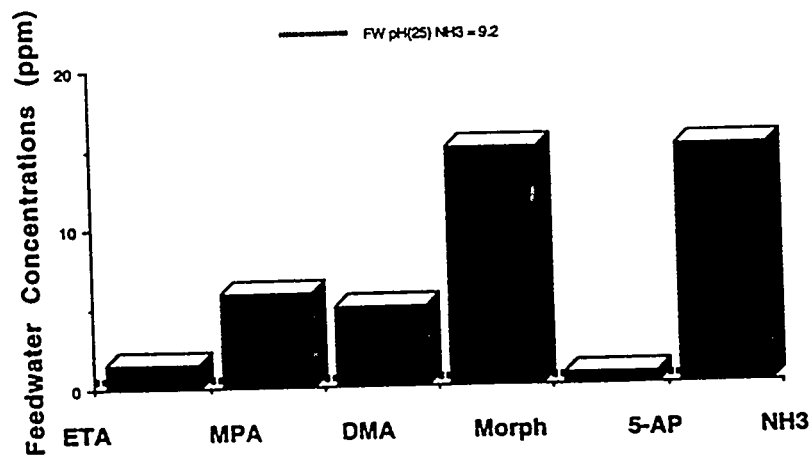


Figure 8. Required concentration of alternate amines to achieve a pH(t) of 6.6 at the moisture separator/reheater in a PWR.

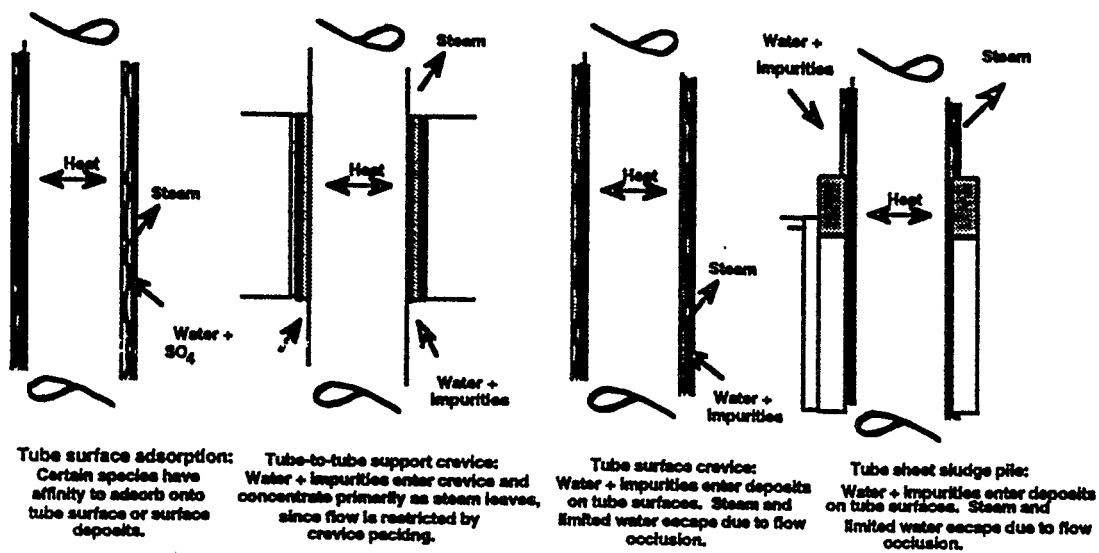


Figure 9. Locations where impurities concentrate in PWR SG's.

Steam Generator Degradation and Mitigation Strategies

**Peter J. Millett
EPRI**

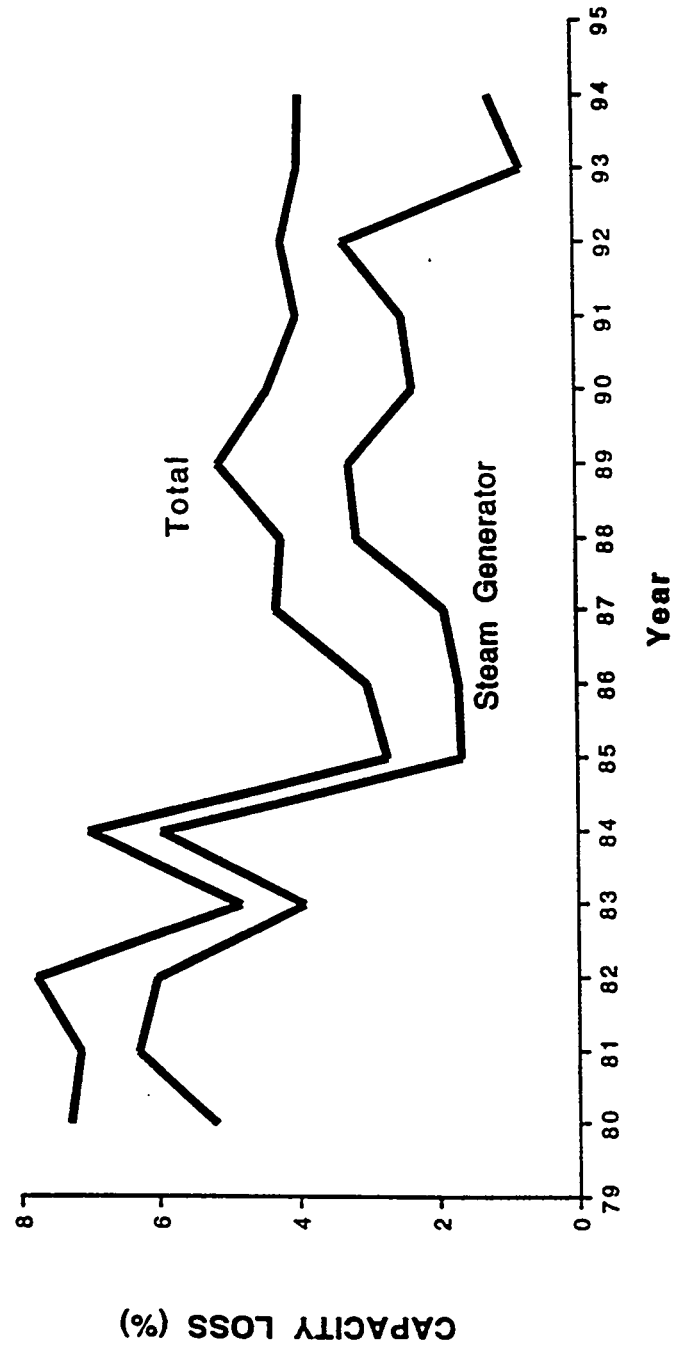
**presented at
CNRA/CSNI Workshop on Steam Generator
Tube Integrity in NPPs**

**Argonne
October 30-November 2, 1995**

Overview of Presentation

- Status of SG degradation worldwide
- Historical review of SG degradation forms
- Mechanisms and environments responsible for corrosion (primary and secondary)
- Current mitigation strategies being used in the field (secondary side corrosion)
- Control and Optimization of secondary water chemistry
- What research needs to be done to develop improved understanding of corrosion and chemistry
- Summary

PWR CAPACITY FACTOR LOSSES DUE TO CORROSION



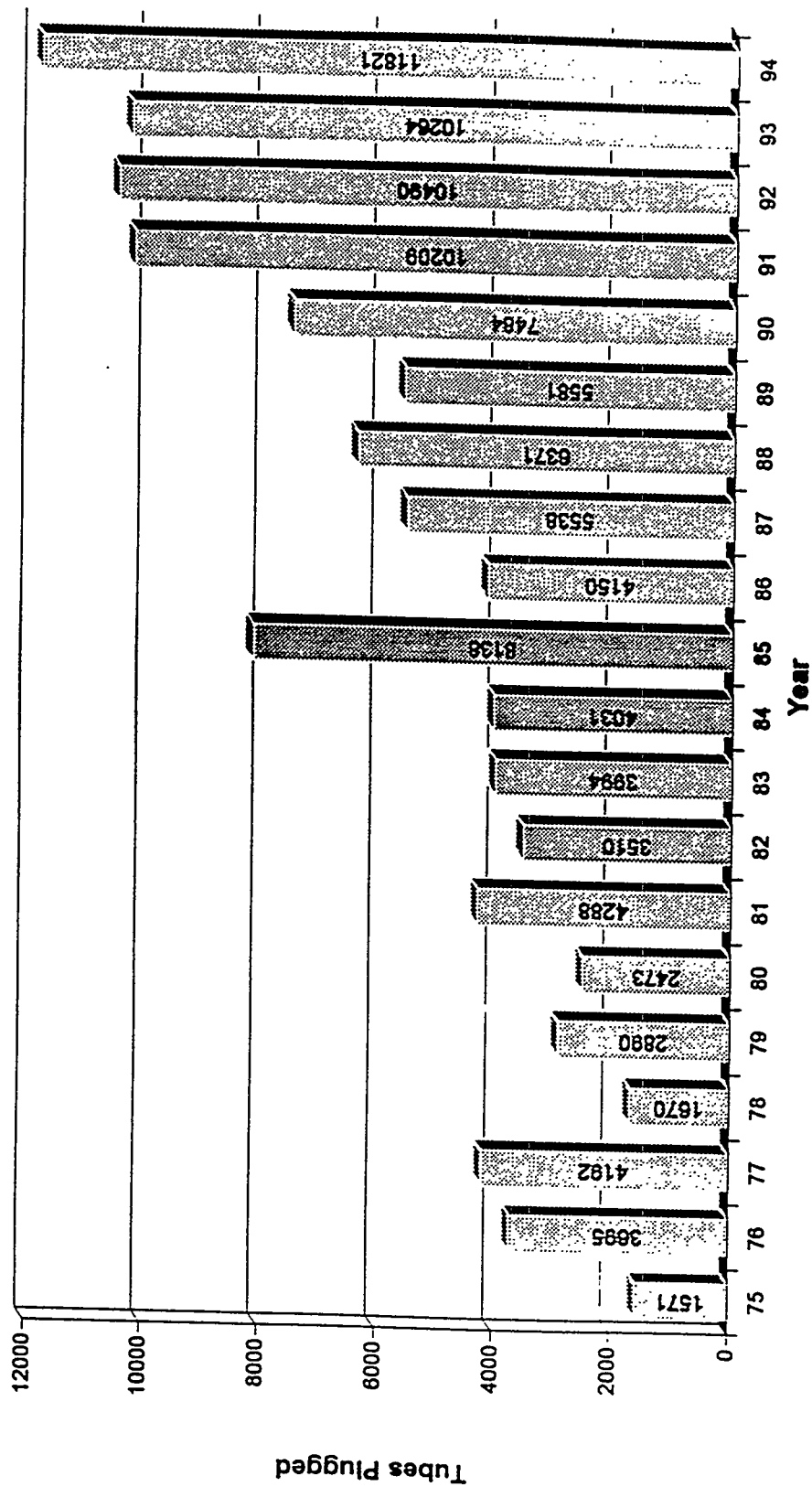


Figure 2-3
World Wide Number of Steam Generator Tubes Plugged

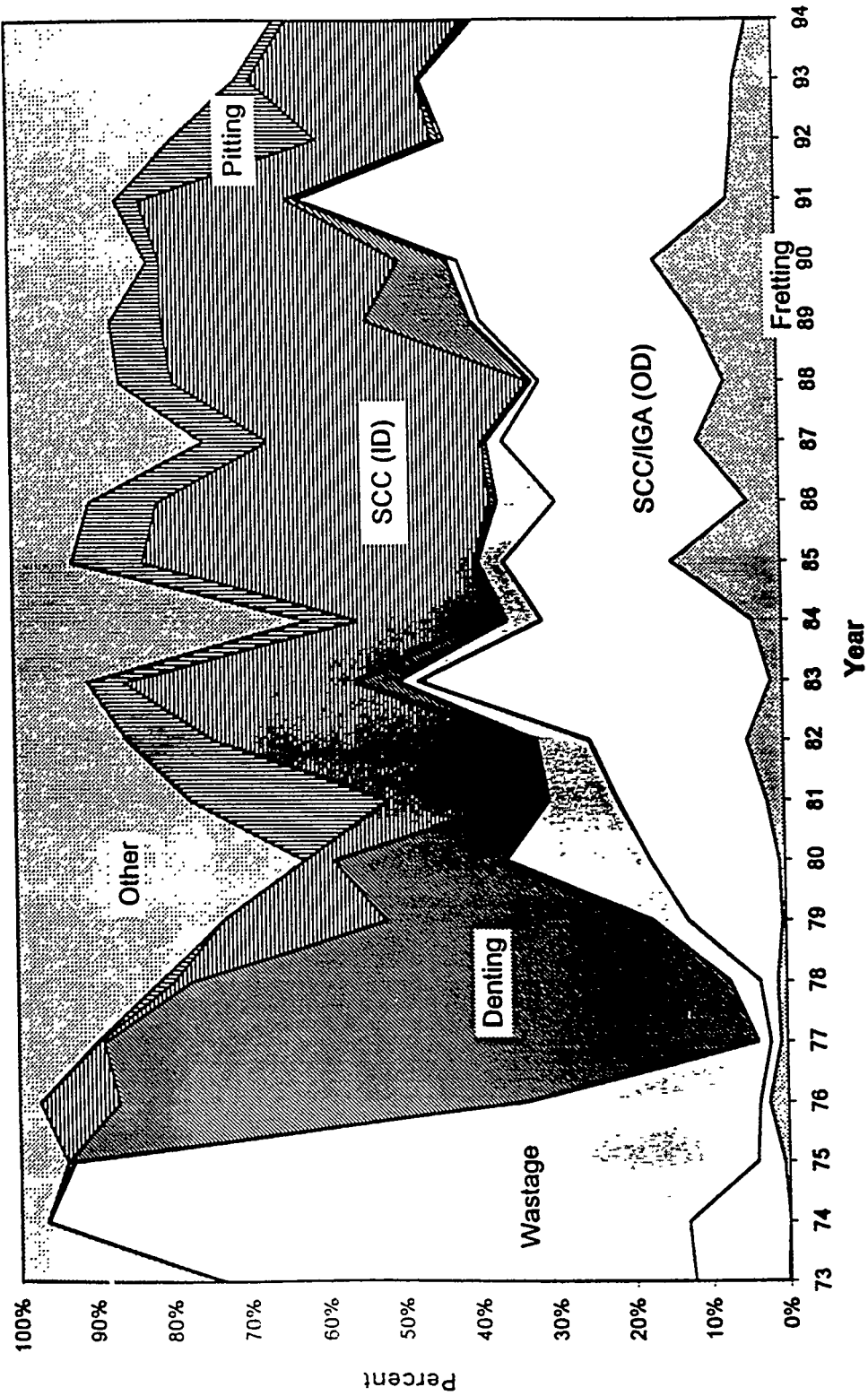
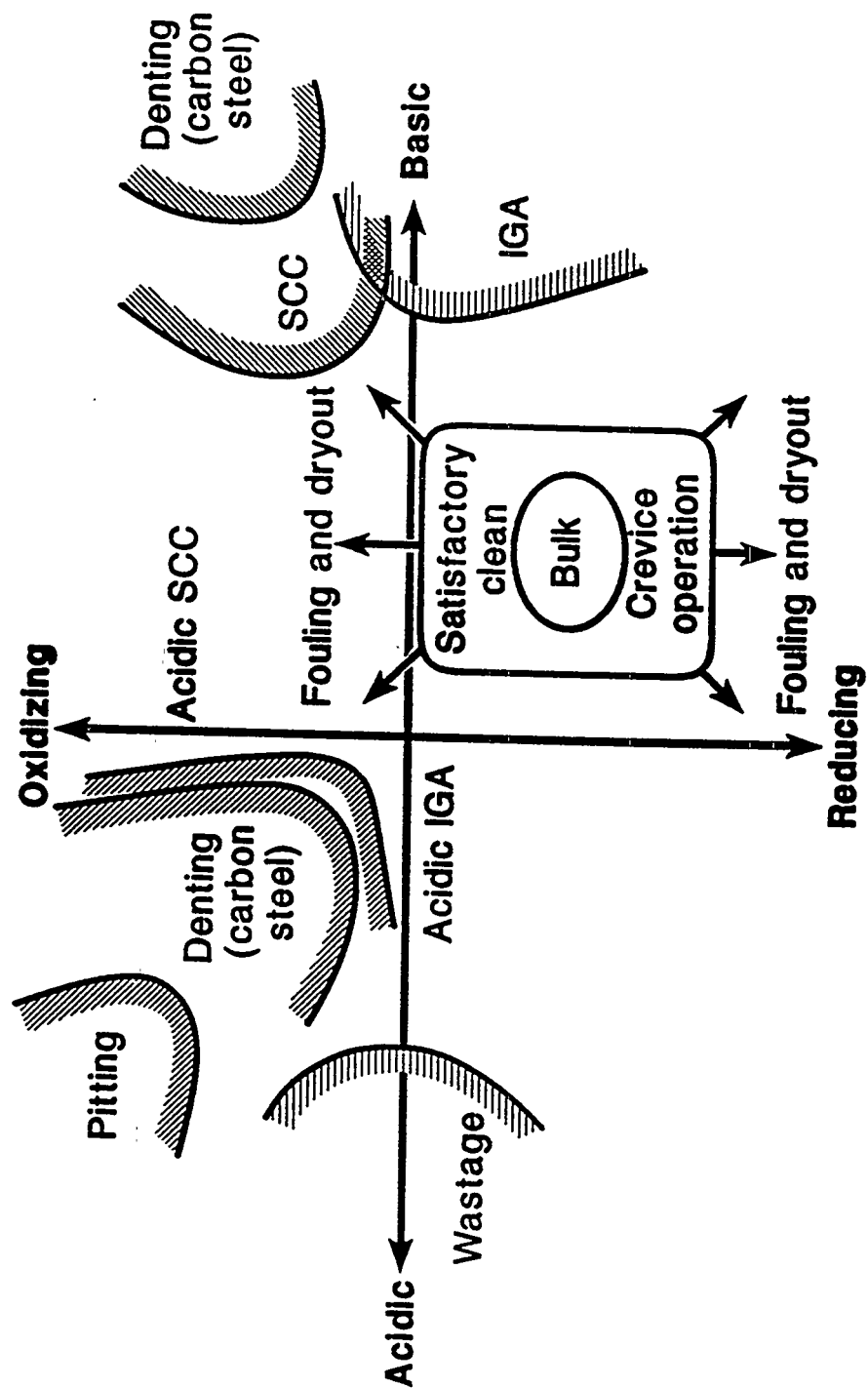


Figure 2-4
World Wide Causes of Steam Generator Tube Plugging

SECONDARY-SIDE CORROSION OF TUBING



R.W. Staehle, "Occurrence of Modes and Submodes of IGC and SCC," presented at workshop on Improving the Understanding and Control of Corrosion on the Secondary Side of Steam Generators, Airlie, October 9-13, 1995.

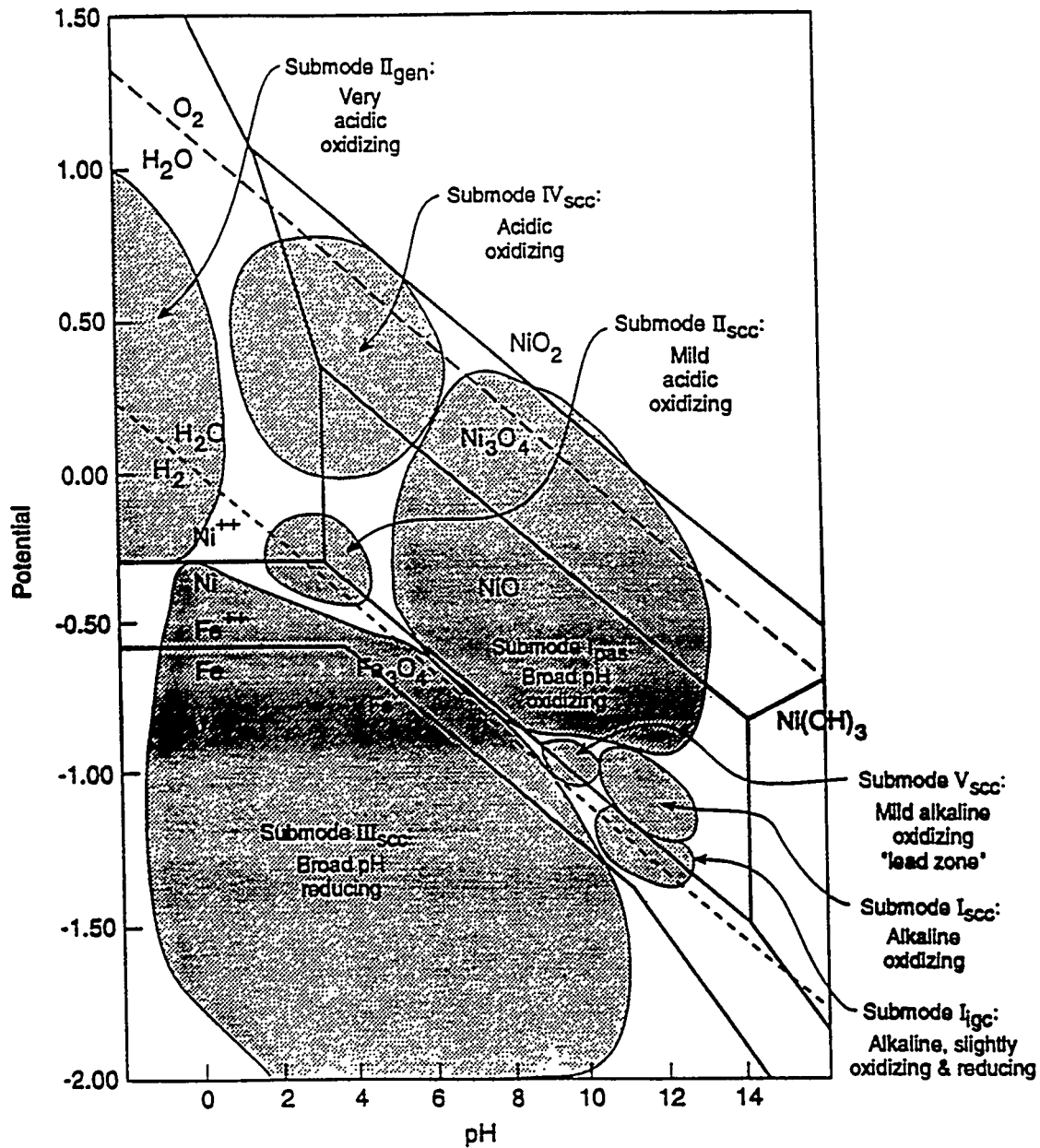
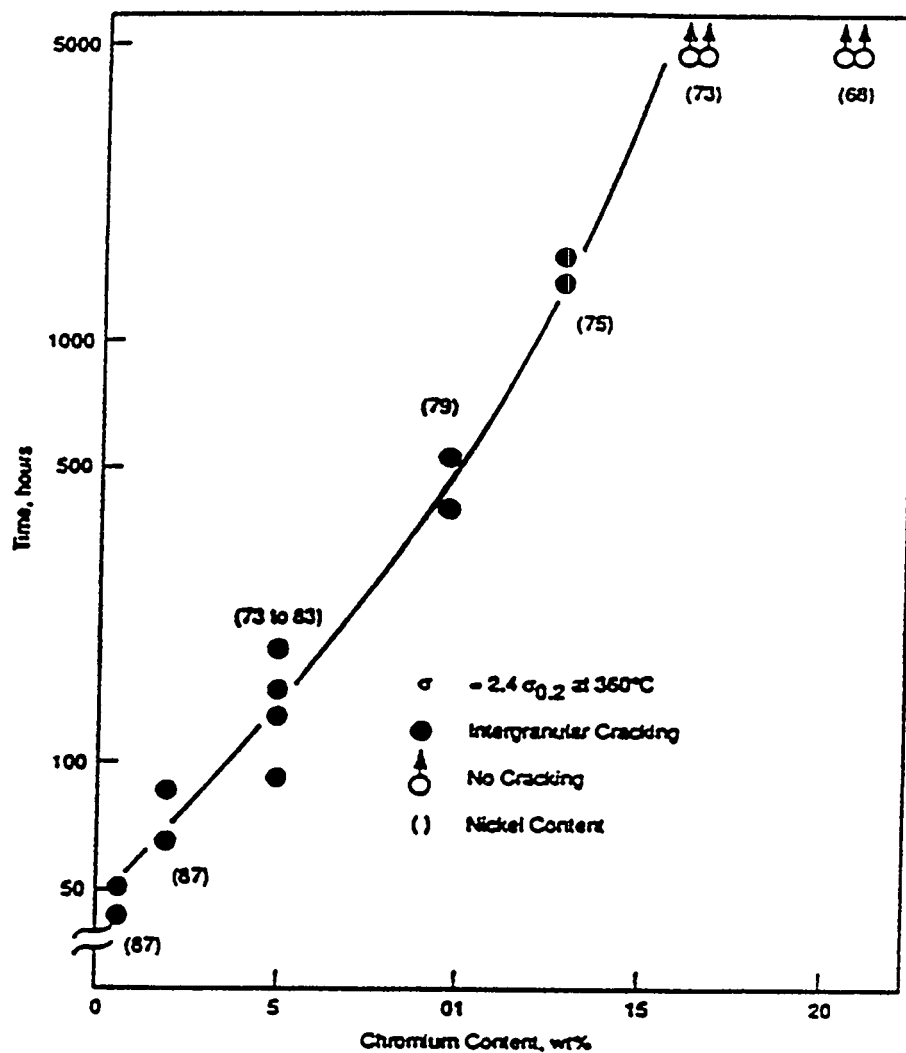


Figure 1. Integrated view of SCC data for Inconel 600MA in high temperature water generally in the range of 300 to 350°C. Ranges of submodes are based on experimental work and judgment about integration. Submodes of corrosion are identified and are numbered starting from right. From Staehle.⁴



G.S. Was, V. Thaveepringsriporn, and J. Hertzberg, "Effects of Grain Boundary Properties on SCC," presented at workshop on Improving the Understanding and Control of Corrosion on the Secondary Side of Steam Generators, Airlie, October 9-13, 1995.

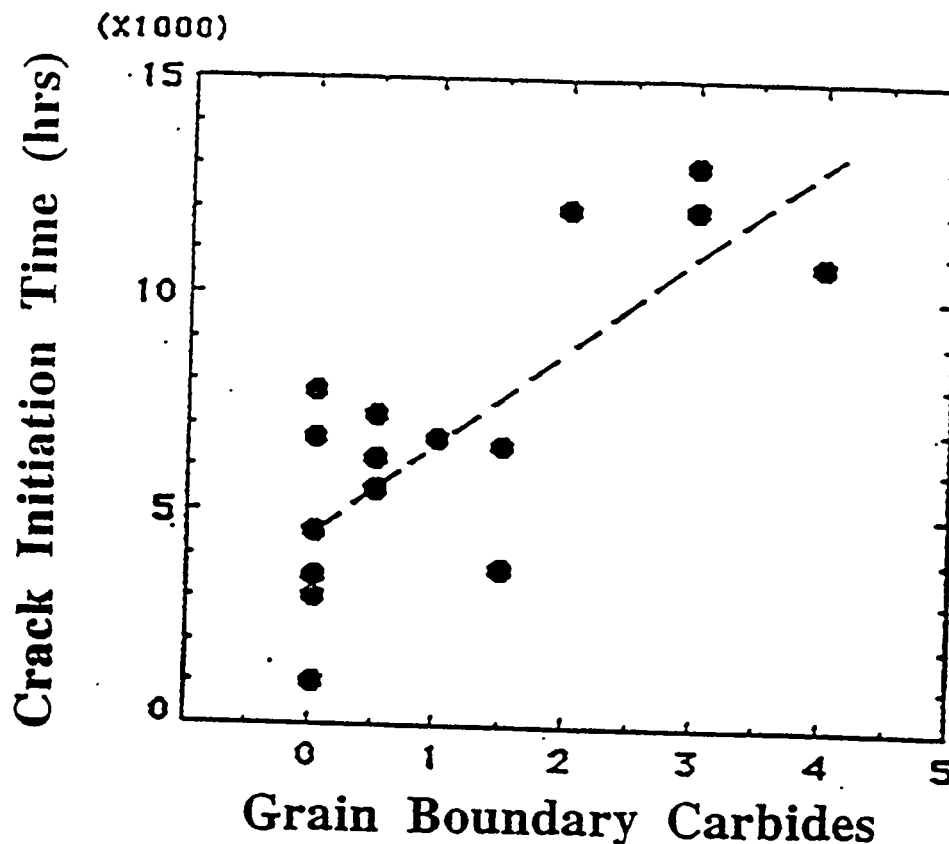


Figure 13. Influence of grain boundary carbides (0 = no carbides; 5 = complete coverage with carbides) on the crack initiation time of alloy 600 in RUB specimens immersed on high purity water with 50 ml H₂/kg H₂O at 365°C (from reference 59).

G.S. Was, V. Thaveepriingsriporn, and J. Hertzberg, "Effects of Grain Boundary Properties on SCC," presented at workshop on Improving the Understanding and Control of Corrosion on the Secondary Side of Steam Generators, Airlie, October 9-13, 1995.

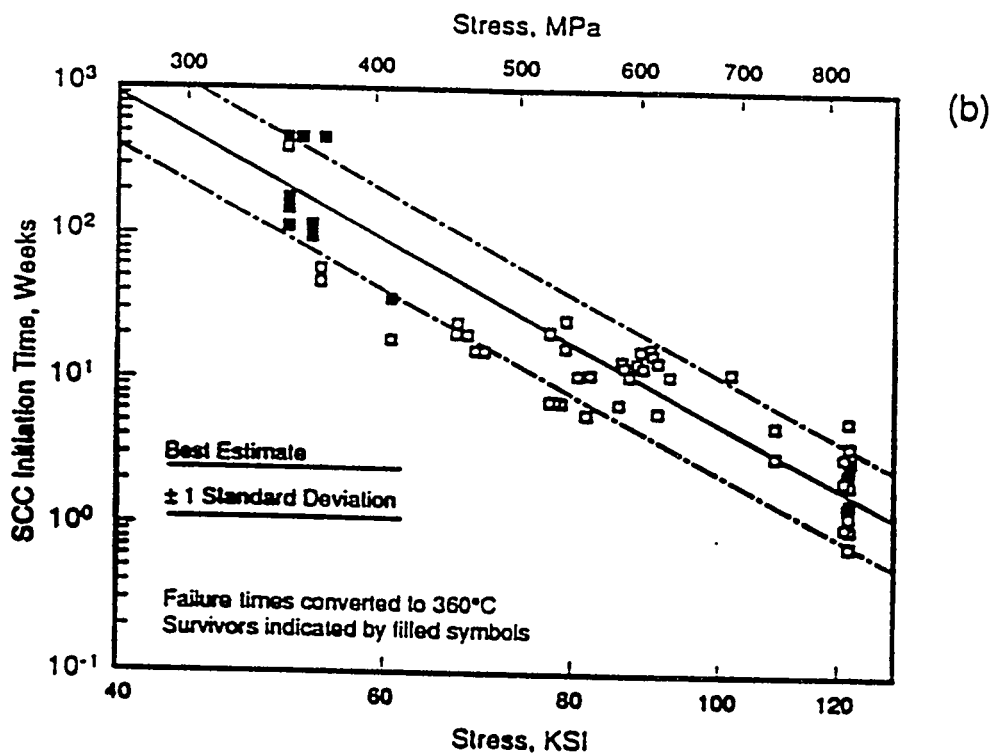
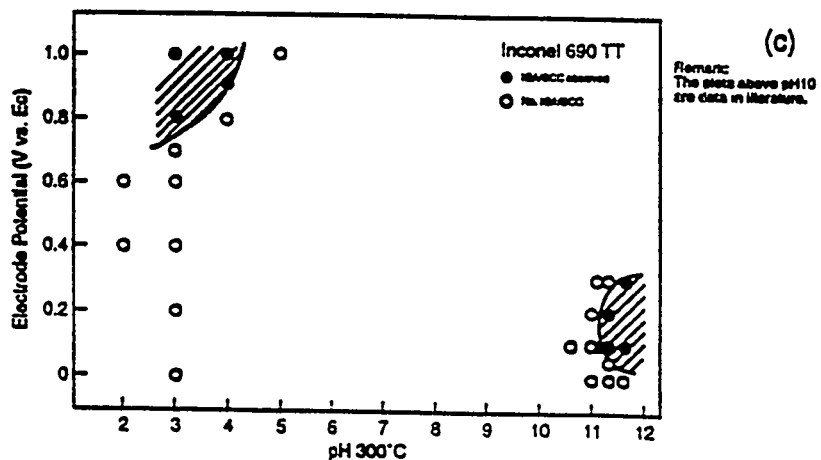
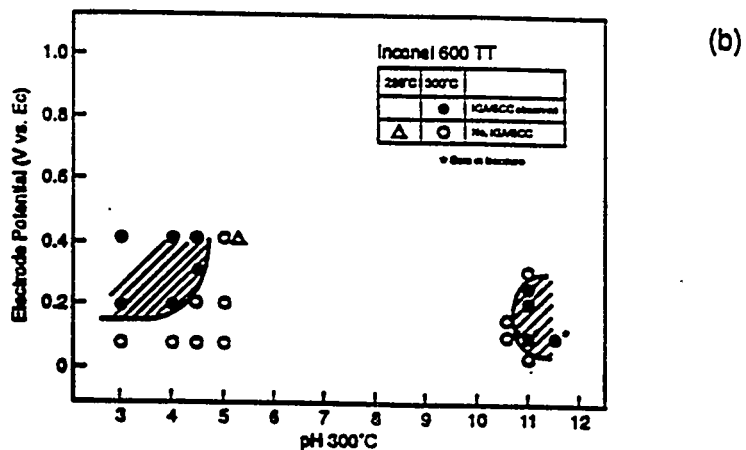
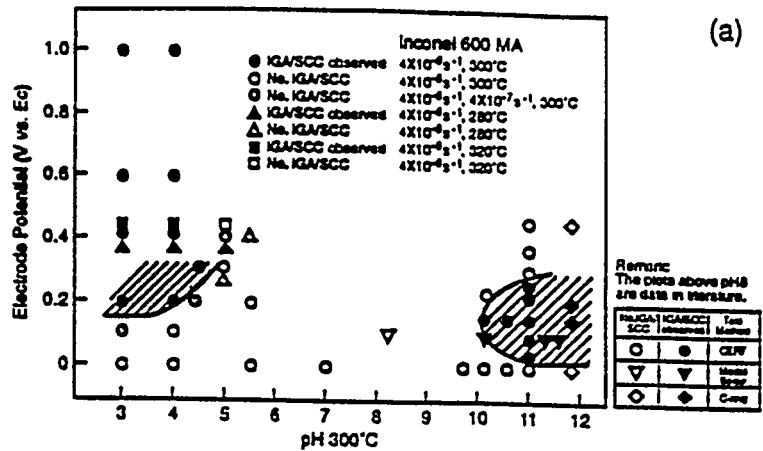


Figure 48. (a) Applied stress as a fraction of the yield stress vs. time to failure for Inconel 600MA at 365°C in pure water. Stress exponent was about -4. From Bandy and van Rooyen.⁹⁵ (b) SCC initiation time vs. applied stress for Inconel 600MA tested at 360°C. The stress exponent was about -5.7. From Webb.⁹⁴

R.W. Staehle, "Occurrence of Modes and Submodes of IGC and SCC," presented at workshop on Improving the Understanding and Control of Corrosion on the Secondary Side of Steam Generators, Airlie, October 9-13, 1995.

Control of PWSCC in Operating PWRs

- Material properties much more important than operating chemistry
 - Stress relief
 - operating temperature
 - Improved materials and heat treatment
- Chemistry effects
 - small effect of lithium over the range of 0.7 ppm to 3.5 ppm (10-35% decrease in lifetime)
 - pH effect small: 10-25% in life from 6.9-7.4
 - No effect of hydrogen between 25-50 cc/kg, small benefit below 25 cc/kg
- Population of susceptible tubing decreasing in older units. Optimize chemistry for radiation field control



R.W. Staehle, "Occurrence of Modes and Submodes of IGC and SCC," presented at workshop on Improving the Understanding and Control of Corrosion on the Secondary Side of Steam Generators, Airlie, October 9-13, 1995.

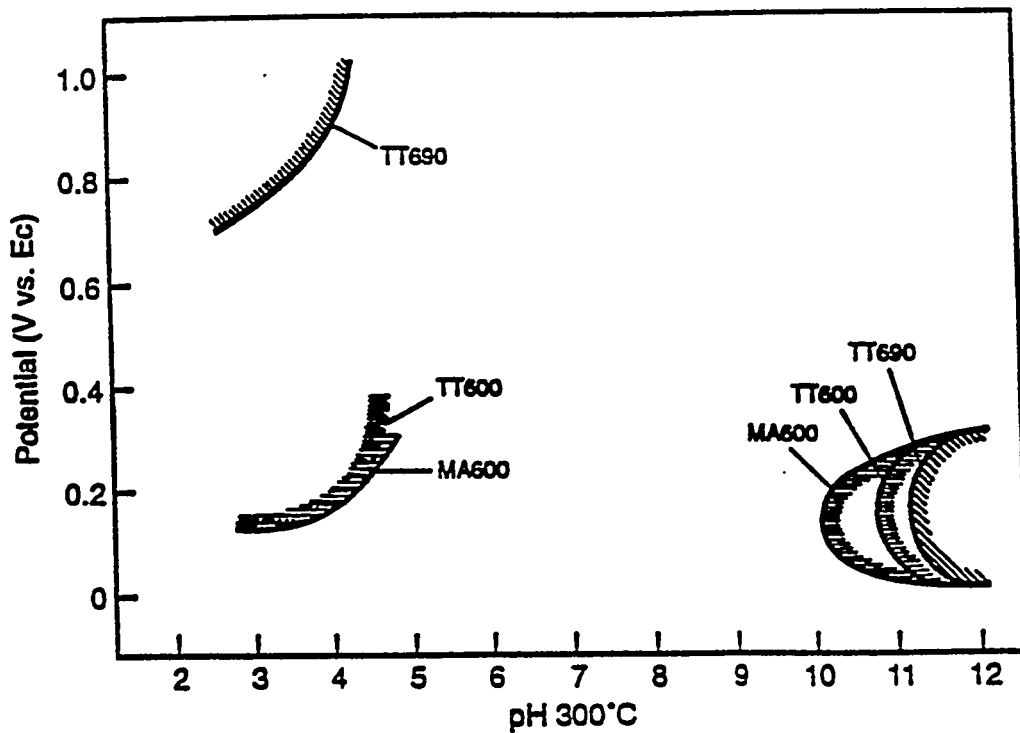
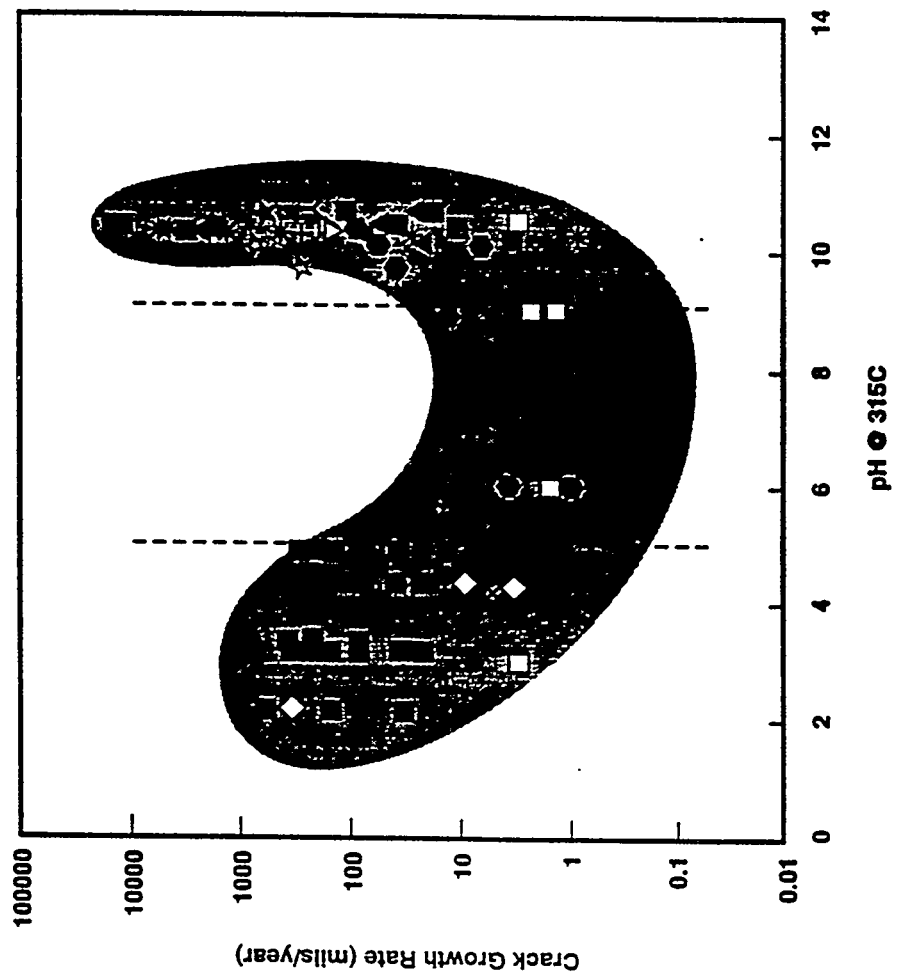


Figure 15. Comparison of IGA susceptibility among Inconel 600MA, Inconel TT, and Inconel 690TT in the range of 280 to 320°C as a function of electrode potential and pH taken at 300°C. From Tsujikawa and Yashima.⁴⁴

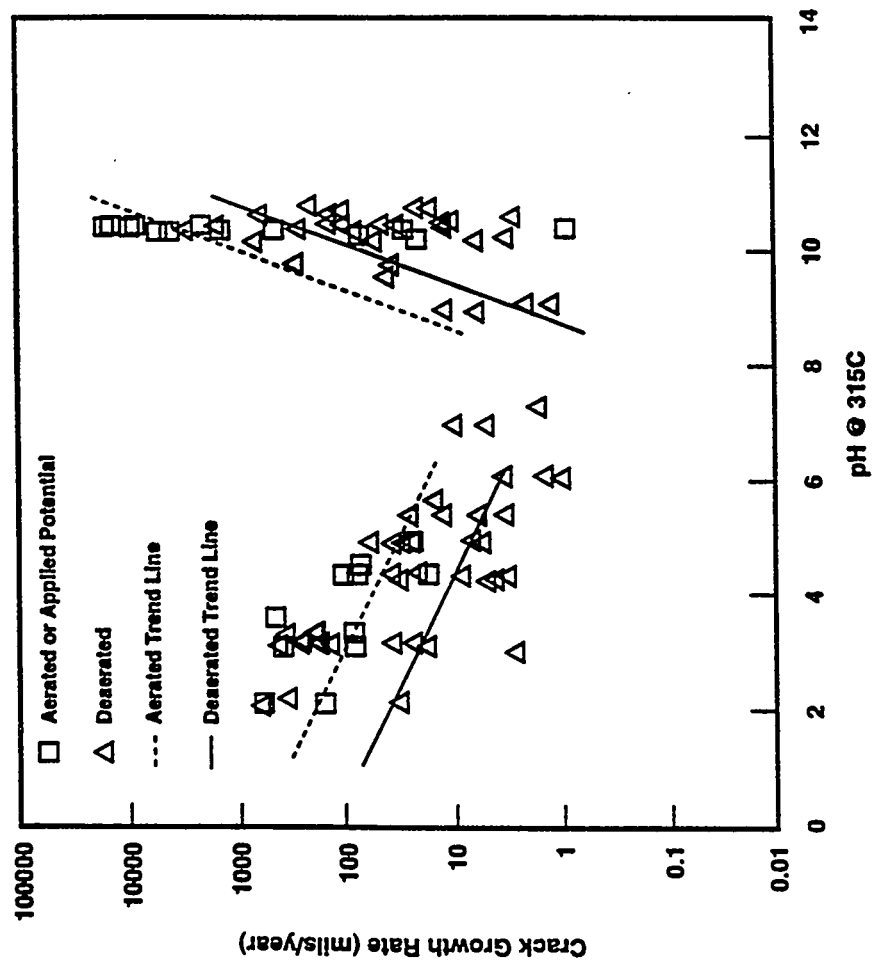
R.W. Staehle, "Occurrence of Modes and Submodes of IGC and SCC," presented at workshop on Improving the Understanding and Control of Corrosion on the Secondary Side of Steam Generators, Airlie, October 9-13, 1995.

MA I600 Crack Growth Rates

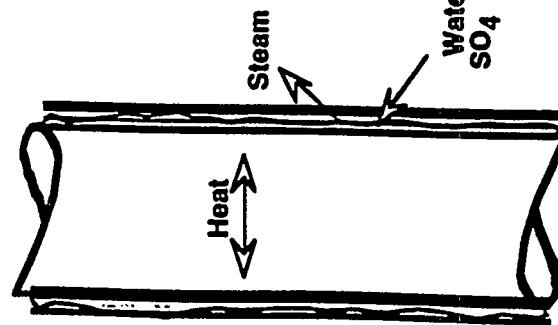
(from EPRI TR-104811 Vol 1, Jan. 1995)



Influence of Oxygen of IGSCC of I600 (from EPRI TR-104811 Vol 1, Jan. 1995)

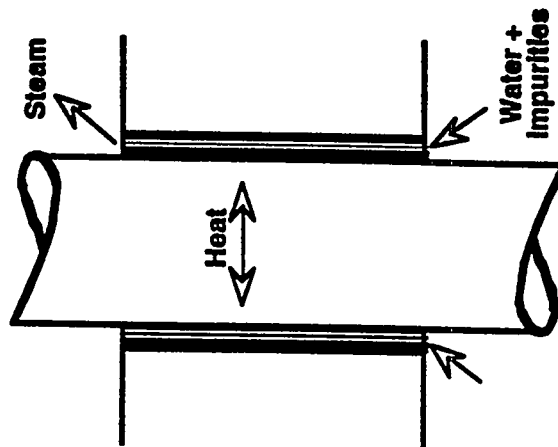


Concentration Sites in SGs



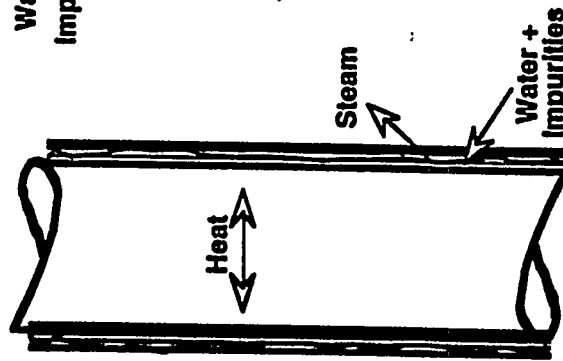
Tube surface adsorption:
Certain species have affinity to adsorb onto tube surface or surface deposits.

Figure 1-a



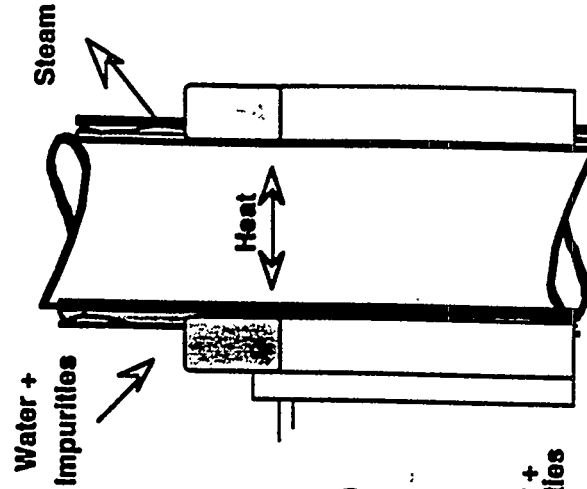
Tube-to-tube support crevice:
Water + impurities enter crevice and concentrate primarily as steam leaves, since flow is restricted by crevice packing.

Figure 1-b



Tube surface crevice:
Water + impurities enter deposits on tube surfaces. Steam and limited water escape due to flow occlusion.

Figure 1-c

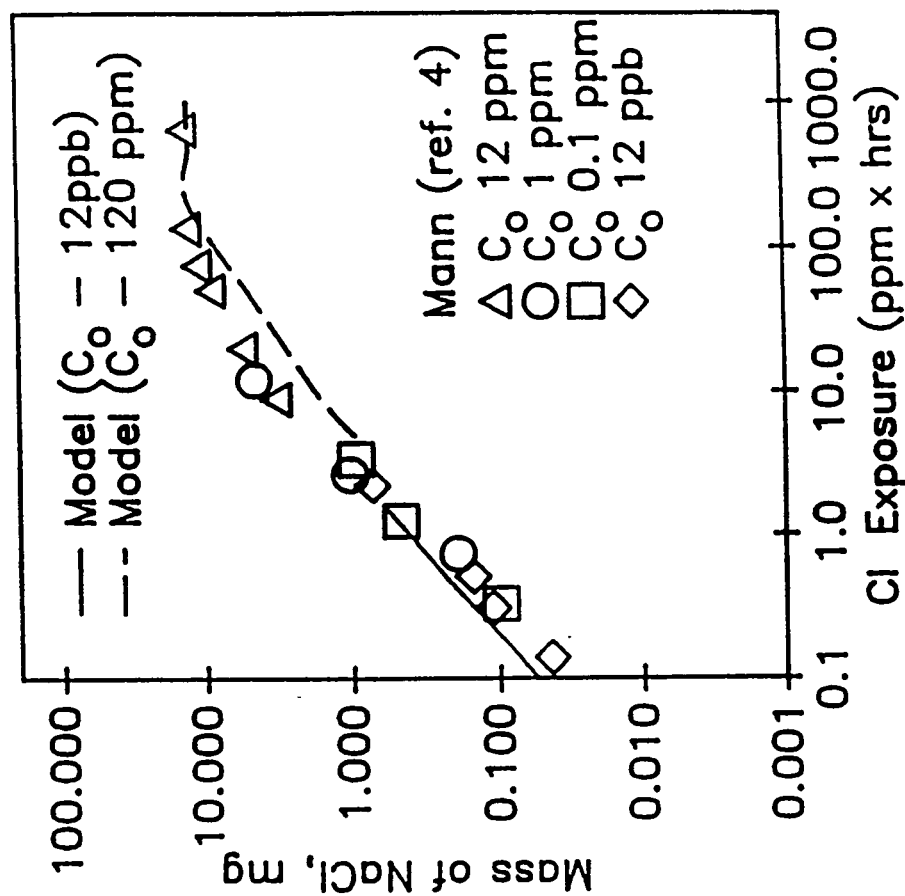


Tube sheet sludge pile:
Water + impurities enter deposits on tube surfaces. Steam and limited water escape due to flow occlusion.

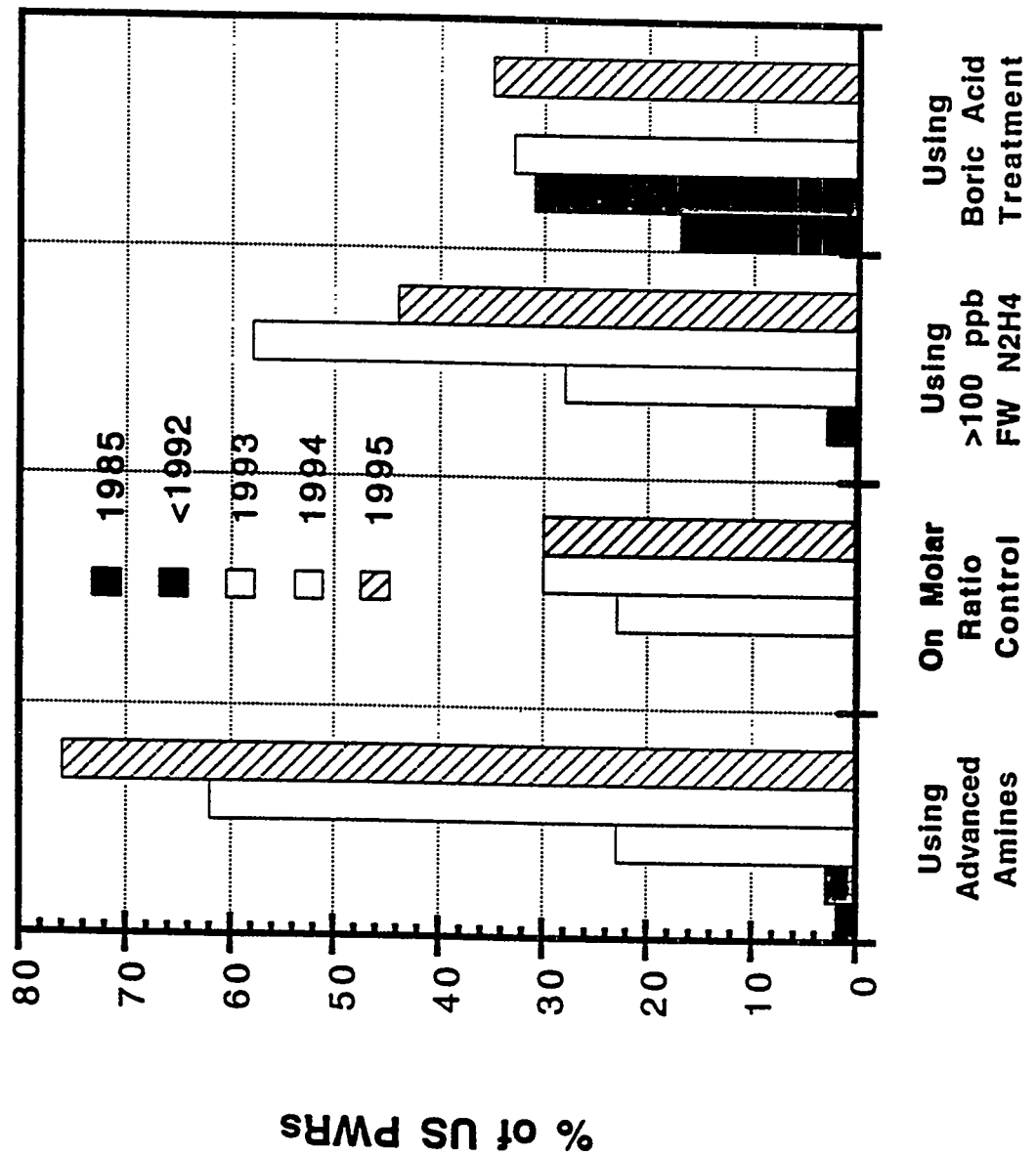
Figure 1-d

Accumulation Rate of NaCl in Packed Crevice

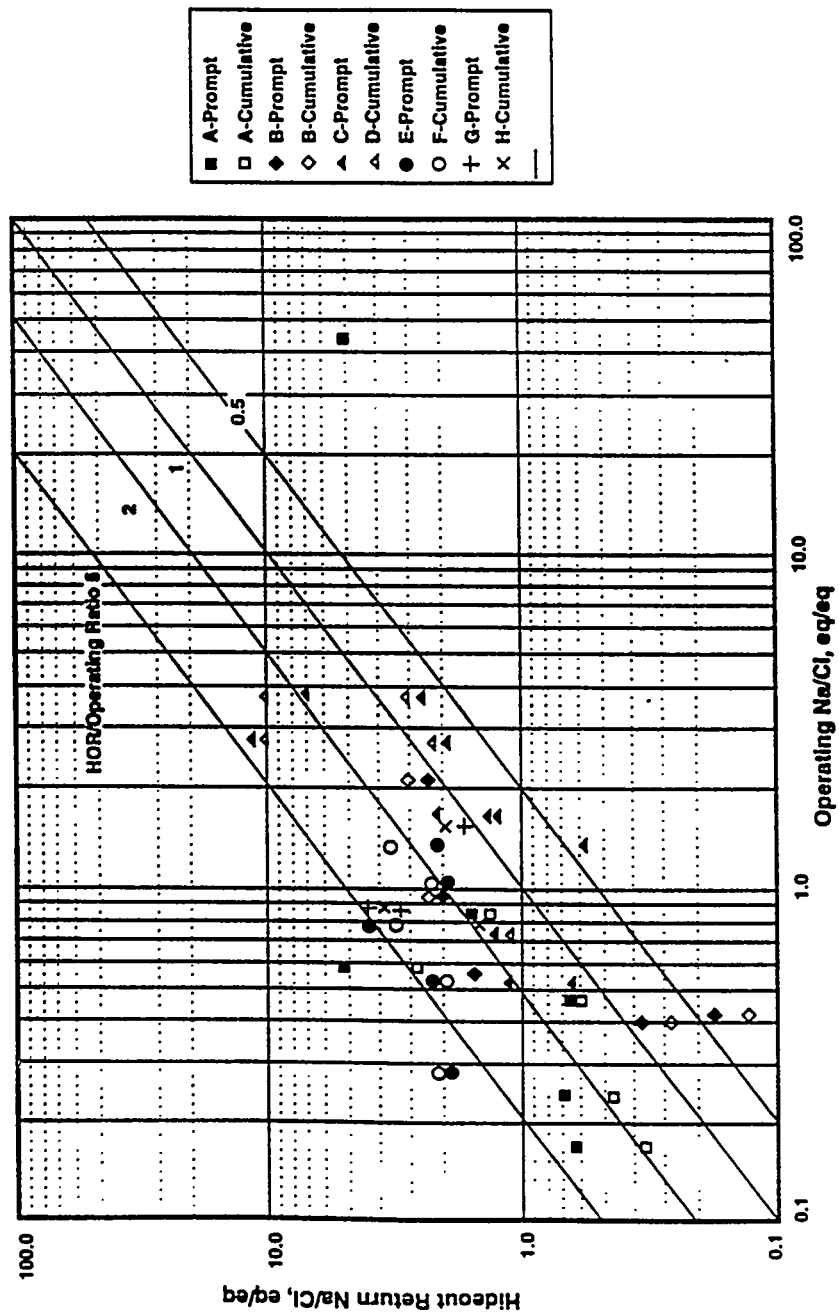
Comparison of Experimental Data (Mann) with Crevice Concentration Model (Millett and Fenton)



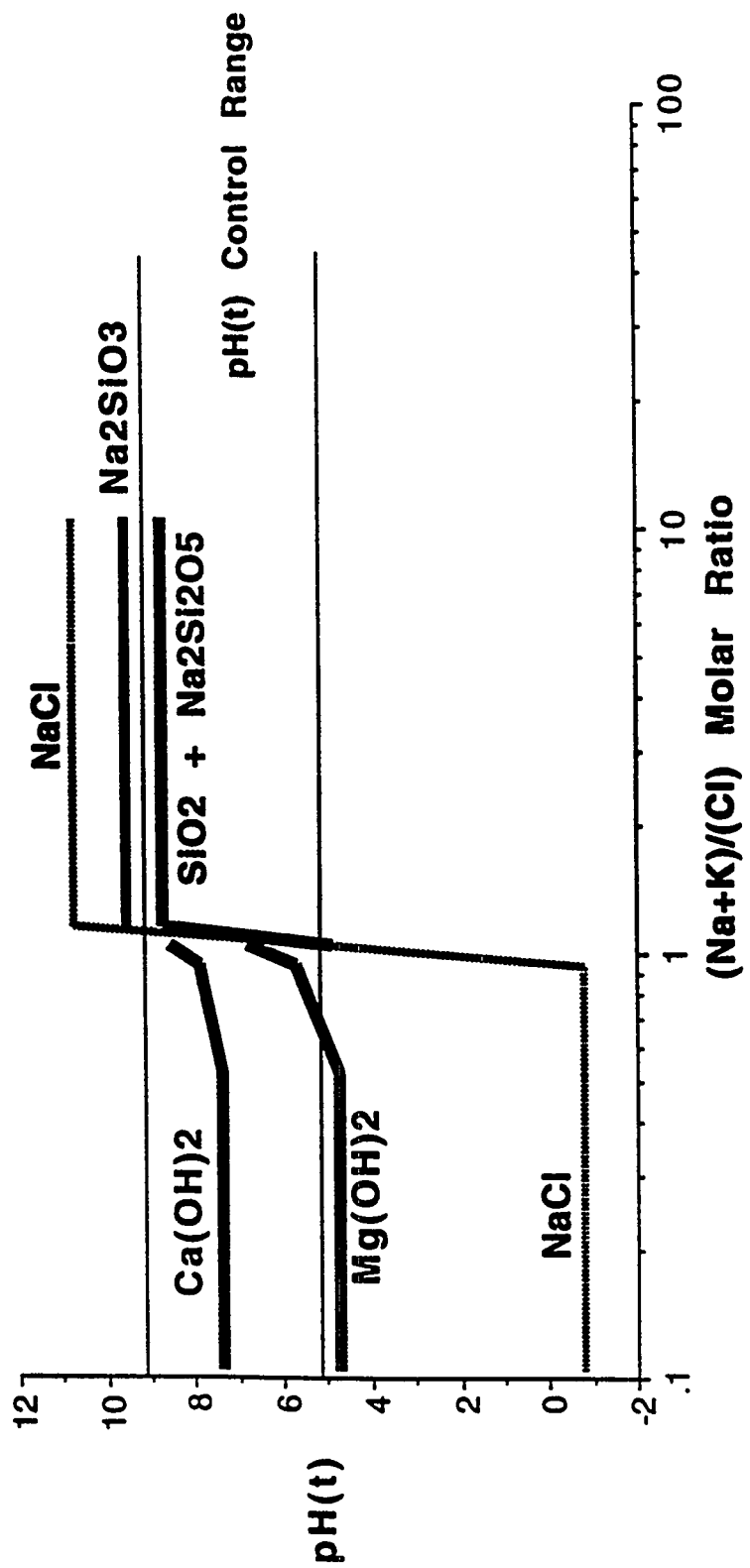
Implementation of Rev. 3 Secondary Water Chemistry Guidelines in US PWRs



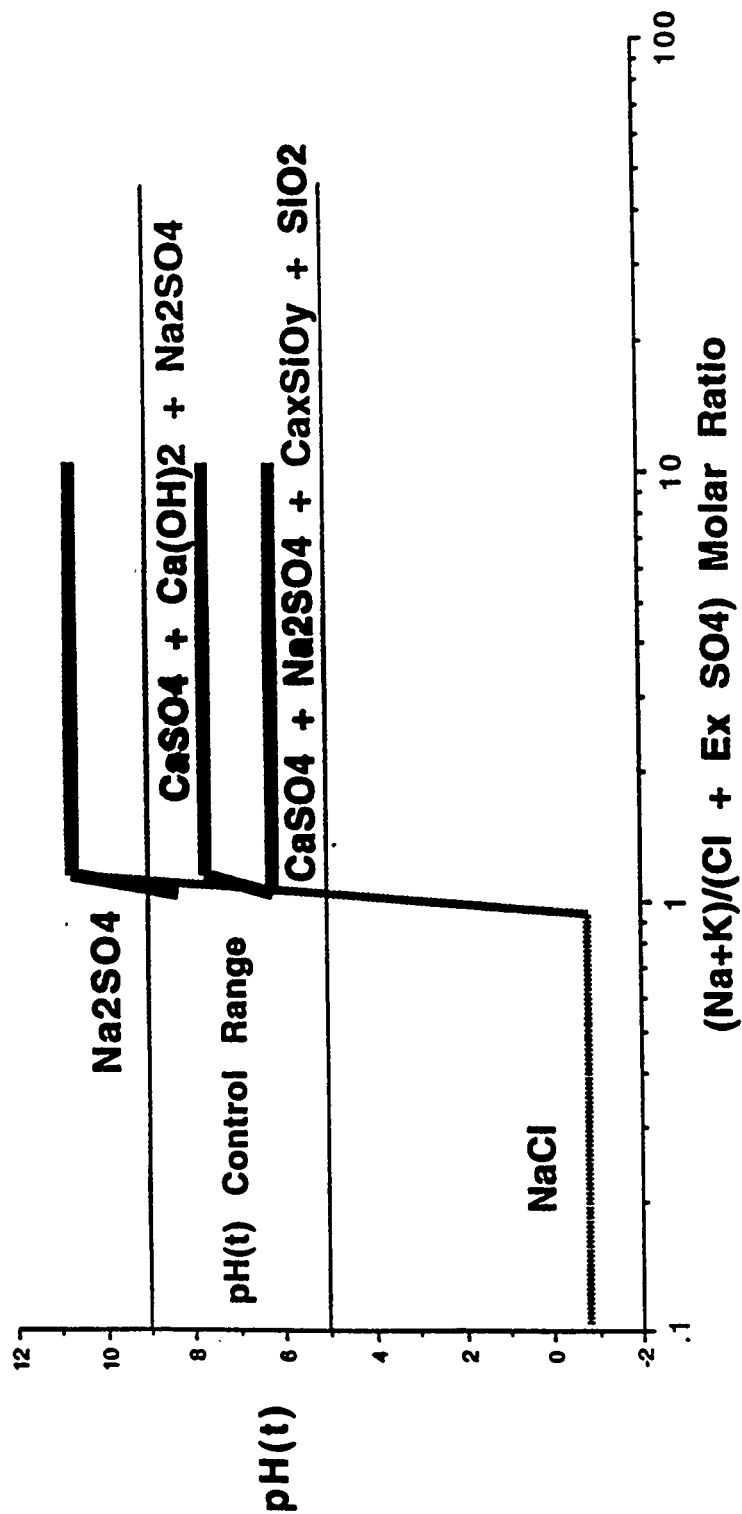
Relationship of Na/Cl in SG Blowdown to Na/Cl in Hideout Return (from EPRI TR-104811 Vol 1, Jan. 1995)



Influence of Precipitates on pH(t) for NaCl Solutions



Impact of Precipitates on pH(t) for SO4 containing solutions

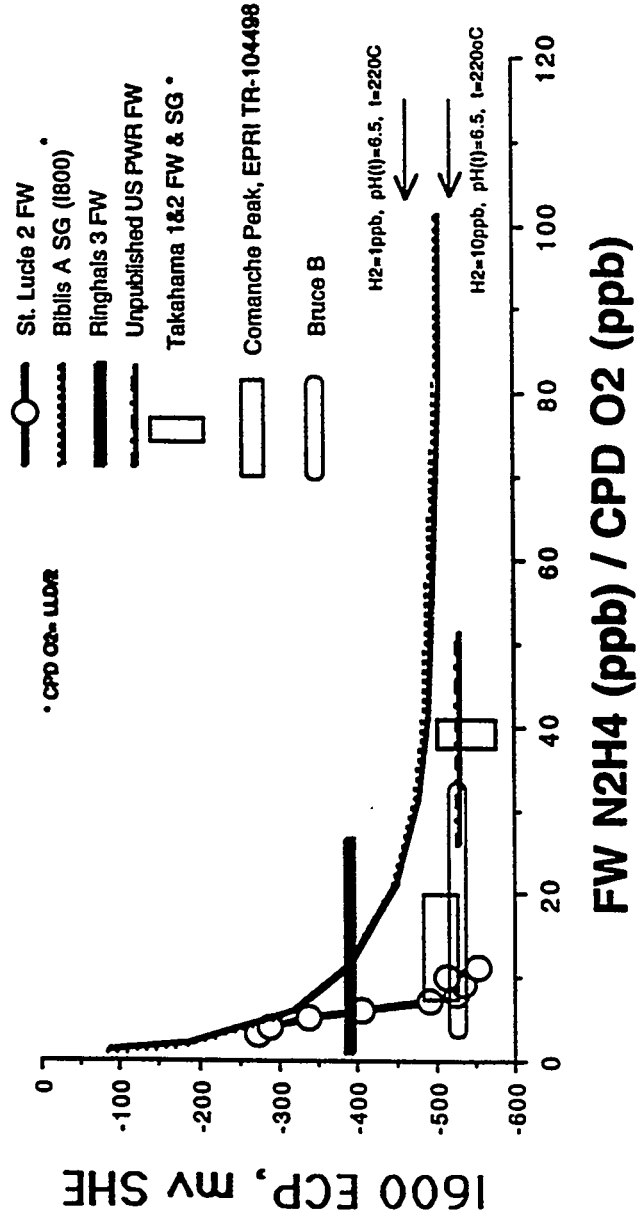


Crevice Chemistry

- **Control of crevice chemistry based on deterministic models is difficult**
- **Corrosion literature suggests a range of pH's where crack growth rates of 1600 can be minimized**
- **A Molar Ratio Indicator close to one appears to give the highest probability that the crevice pH is in the target range**
- **New methodologies for evaluating hideout return data may yield best data for determining the Molar Ratio Indicator**

Feedwater and Steam Generator

ECP Measurements

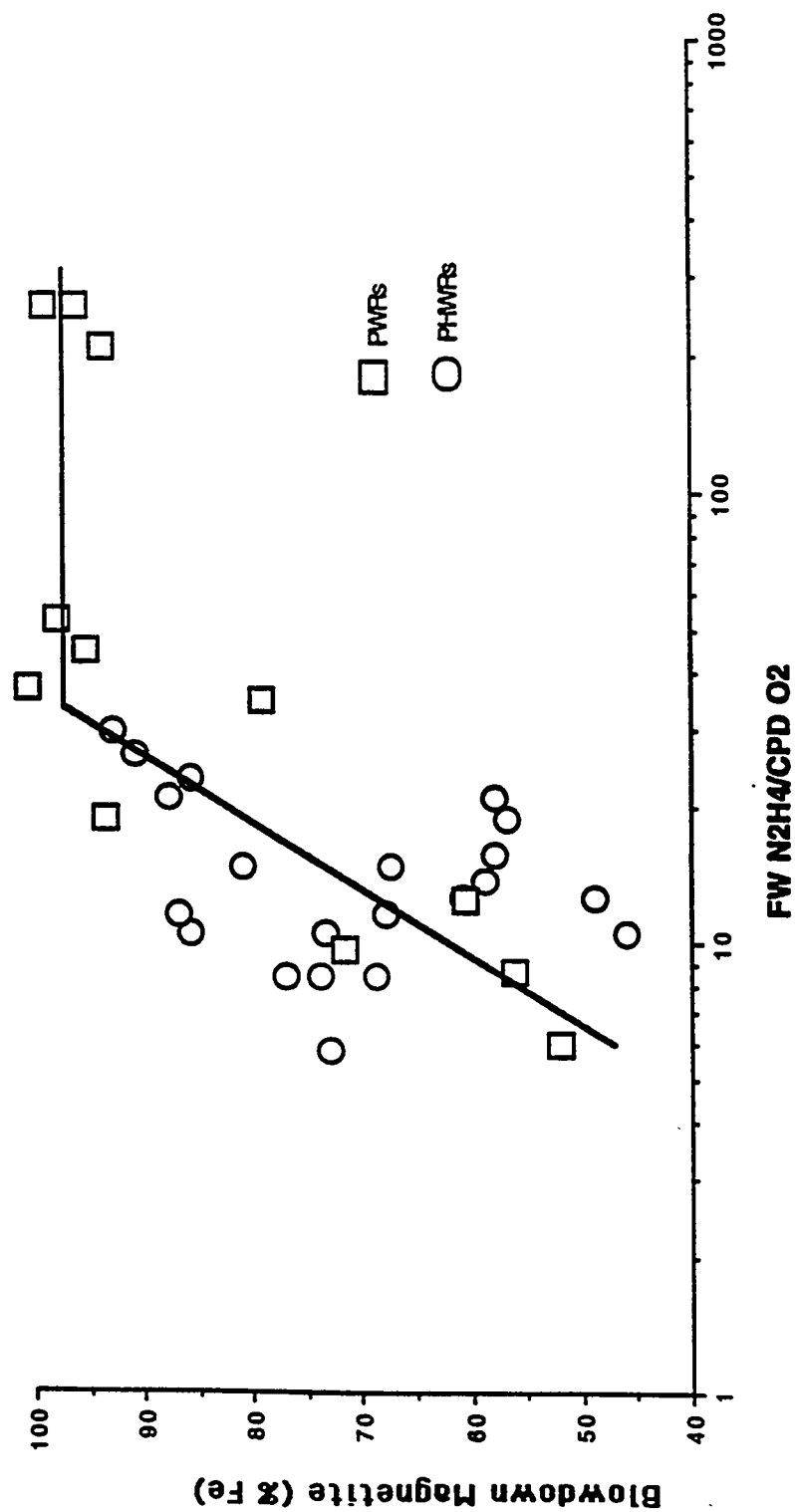


Effect of Oxidants on Alloy 600 ECP in 280 C, 40% Caustic

Oxidant	~ ECP (above baseline)
40% Caustic	0
+ Cu	0
+ Cu2O	+350
+ CuO	+550
+ Fe3O4	0 + small amount
+ Fe2O3	+100
+Cr2O3	0 + small amount
+ NiO	0
+ PbO	+150

Source: Y. Hino et. al, Water Chemistry Control in PWR Nuclear Plant, Thermal and Nuclear Power, Vol. 43 No. 433 p186-202, 1993.

Relationship between % Magnetite in FW Iron to Ratio of FW N2H4 to CPD O2



9/1995 Jt. Mtg. 8

Table 3. Stress Corrosion Cracking Results in 50% Caustic Solutions. The Reference Environment is 50% NaOH + 1% Na₂CO₃. Exposure Time were 2 Days in the Reference Solution and with the Boric Acid Additive; All Others were 1 Week.

Environmental/Potential (mV)	0	100	150	200
Reference (50% NaOH)- 2 days	No SCC	No SCC	SCC	SCC
Reference + 5% H ₃ BO ₃ - 2 days	No SCC	No SCC	SCC	SCC
Reference + 15% H ₃ BO ₃ - 2days	No SCC	No SCC	SCC	SCC
Reference + 0.1% Ca(OH) ₂ -1wk	No SCC	SCC	SCC	SCC
Reference + 2% Na ₂ ZrO ₃ -1wk	No SCC	SCC	SCC	SCC
Reference + 2% ZnO-1wk	No SCC	No SCC	No SCC	SCC
Reference + 2% Zn(PO ₄) ₂ -1wk	No SCC	No SCC	No SCC	SCC
Reference + SAT H ₂ TiO ₃ - 1 wk	No SCC	No SCC	No SCC	SCC
Reference + 0.5% SiO ₂ -TiO ₂ -1wk	No SCC	No SCC	No SCC	SCC
Reference + 1% TiO ₂ (TiLAC)-1wk	No SCC	No SCC	No SCC	No SCC
Reference + 0.8% TiO ₂ (TiLAC)-1wk	No SCC	No SCC	No SCC	No SCC
Reference + 0.5% TiO ₂ (TiLAC)-1wk	No SCC	No SCC	No SCC	No SCC
Reference + 0.1% TiO ₂ (TiLAC)-1wk	No SCC	No SCC	No SCC	SCC
Reference + 1.0% TiO ₂ (TiOE)-wk	No SCC	No SCC	No SCC	No SCC
Reference + 1% Anatase-1wk	No SCC	No SCC	No SCC	SCC
Reference + 0.5% TiB ₂ -1wk	No SCC	No SCC	No SCC	No SCC
Reference + Amorph TiO ₂ -1wk	No SCC	No SCC	No SCC	No SCC
Reference + Rutile-1wk	No SCC	No SCC	SCC	SCC
Reference + ZnTiO ₃ -1wk	No SCC	No SCC	SCC	SCC
Reference + 2% Ce(AC) ₃ -1wk	No SCC	No SCC	No SCC	SCC
Reference + 2% Ce(Cl) ₃ -1wk	No SCC	No SCC	No SCC	SCC
Reference + 2% Ce(NO ₃) ₃ -1wk	No SCC	No SCC	No SCC	No SCC

Development of Inhibitors for Secondary Side IGSCC

- TiO₂ shown to be effective for inhibiting IGSCC in caustic environments (Lumsden-C rings & Miglin-CERT)
- Model boiler tests showed difficulty in transporting TiO₂ to packed crevices (Daret)
- Application issues being addressed (chemical cleaning and tube fouling)
- Two US plants currently adding TiO₂

Control of ODSCC in Operating PWRs

- **Material properties may be much more important than operating chemistry**
 - **operating temperature**
 - **Improved materials and heat treatment**
- **Chemistry effects**
 - **existing inhibitors (BAT), Molar Ratio Control, elevated hydrazine, ALARA chemistry should be pursued. May be more effective at controlling initiation than growth**
 - **need to better understand crevice chemistry**
 - improved diagnostics, tube pulls, in-situ monitors**
 - crack growth data in real crevice environments**

Airlie Specialist Meeting

Meeting Objective: Have a select group of recognized experts review plant experience and laboratory test results to determine if models can be developed for secondary side degradation(s).

Purpose: Establish priorities with respect to the secondary side for a) any actions which should be taken in the operation of steam generators to minimize or prevent corrosion related damage, and b) technical work which should be undertaken to improve the understanding of corrosion processes and the factors which control them.

M & C

ARM 10/95 TAG 8

EPRI/NPG

Airlie Specialist Meeting

Modeling Session Reports:

- SCC in Acidic, Neutral and Alkaline Environments by Peter Andresen/GE (21 pages)

Issues

- Nature & predictability of Crevice
 - » pH
 - » isolated by steam(no ionic path)
 - » isolated permanently
- Synergy/relationship to other corrosion phen.
 - » crack blunting if pitting, hi general corrosion
- Fundamental importance of strain rate in passive metals
- Are we really sure that low or high pH signifies “disaster”
 - » local environment fairly constant vs. time (within 1 cycle)?

M & C

Airlie Specialist Meeting

Modeling Session Reports:

- Denting at TTS and TSP Locations by Peter Scott/Framatome (14 pages)

Slow Denting is the Issue

- TSP denting observed after approx. 10 years(Ohl 1, McGuire 1 and other similar well managed plants)
- Continued low denting rates seen at some plants affected by early fast denting, following water chemistry improvements (MY, IP2)
- slow structural deformation
 - » Bruce A scallop bars
 - » Tie rods in CE plants
 - » TSP cracking/deformation (Surry, IP2)
- In well managed plants we assume that oxidizing agents are not present

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EPRI/NPG

Airlie Specialist Meeting

Modeling Session Reports:

- **Denting at TTS and TSP Locations by Peter Scott/Framatome (continued)**
 - The gap to be filled prior to denting is about 200 microns (either TSP or TTS)
 - What is relevant rate of corrosion of carbon steel in the crevice?
 - What other deposits accumulate in the crevice and at what rate?
 - Under ideal conditions at 300 C, pH~7, no contaminants, the carbon steel corrosion rate is about 2 microns/year.
 - Oxide volume/metal consumed = 2.3 to 3.
 - Therefore, crevice is filled in 50 years in the absence of other deposits(shorter if crevice fills with deposits)
 - Dampierre 1 crevices show 200 microns, not 20 microns, of metal consumed in 10 years

M & C

Airlie Specialist Meeting

Modeling Session Reports:

- Action of Lead in Locally Concentrating and Producing SCC by Jacqueline Stubbe/Laborelec (12 pages)
 - There is no phenomenological or mechanistic models available.
 - Possible Mechanisms:
 - destruction of passive layers
 - galvanic effect -- enhanced anodic dissolution
 - liquid/solid metal embrittlement
 - Hydrogen embrittlement
 - vapor phase attack-- adapt existing mechanistic models?

M & C

EPRI/NPG

Airlie Specialist Meeting

Modeling Session Reports:

- Action of Lead in Locally Concentrating and Producing SCC by Jacqueline Stubbe/Laborelec (continued)

Needed Information

- How is Pb transported?
 - liquid vs. gas; ionic vs. neutral, organic Pb
- Pb synergistic effect
- (Is Pb different than As, Sb?)
- Electrochemical bounds to assess PbSCC
- More physical chemistry on Pb compounds
- Embrittlement mechanisms(H₂, LME, SME, why TGSCC?)
- Vapor phase attack- vapor pressure vs. temperature

M & C

Airlie Specialist Meeting

• Local Chemistry in Tube Support Crevice

Priority should be to evaluate fouled crevices with main features:

- porous packed corrosion product
 - concentrated solutions in thermodynamic equilibrium with superheat
 - slowly filling, memory effect
 - chemical equilibrium in solution, not between solids and solution
- Important issues which need to be understood
 - Is crevice environment electrically isolated from bulk water?
 - How much hydrogen is in crevice?

M & C

EPRI/NPG

Airlie Specialist Meeting

- **Local Chemistry in Tube Support Crevices (Continued)**

Priorities for further work include:

- model development (ECP and T/H); build on existing models
- heated crevice work to calibrate models
- compile results of tube pulls (new and old)
- section TSP's from retired SG's (e.g. Dampierre exam)
- expand MULTEQ database to include plant deposit chemistry
- develop capabilities for insitu crevice measurements

****conductivity, ECP, raman, direct sampling****

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Airlie Specialist Meeting

Local Chemistry in Top of the Tubesheet Crevices and Sludge

Key features of process include:

- Crevices are small compared to TSP
- Only hot leg effected so far (high superheat)
- can have sharp thermal gradients
- local environment more variable than TSP to due to better communication with bulk
- variable stresses and environment make attack more difficult to predict
- Modeling work needed to address chemistry in these regions:
 - without sludge, TTS crevice special case of TSP
 - with sludge more complex models needed
 - need thermal model to define tube temperature ^{M & C}

EPRI/NPG

Airlie Specialist Meeting

• Local Chemistry and Deposit Formation at Upper Bundle Free Span Region

Key questions include:

- Is free span cracking under surface deposits due to similar mechanism as cracking in TS and TSP regions (e.g. impurity concentration processes)?
- Is cracking at scratches and/or high stress regions in absence of surface deposits, pure water cracking?
- Modeling of chemistry must first include good model of particulate deposition
- For apparent pure water cracking need K (SCC) and crack growth data as a function of potential, pH and temperature

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Airlie Specialist Meeting

Summary of Research Priorities:

Roger Staehle/ University of Minnesota

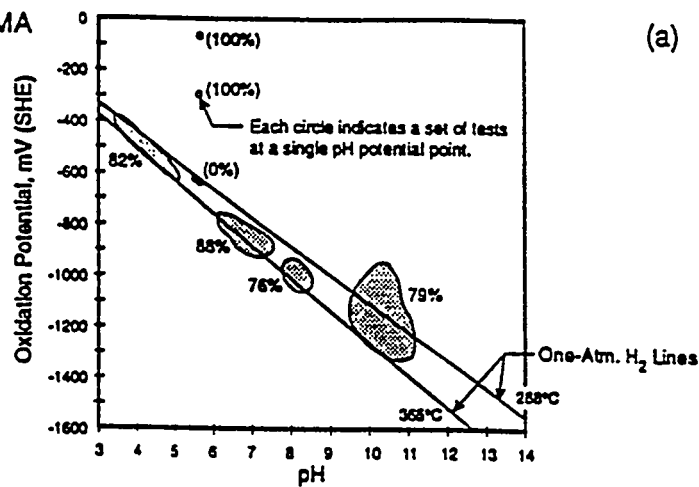
- Define environment
 - Tube and SG examinations
 - In situ SG monitoring
 - Laboratory monitoring
- Define corrosion properties
 - Emphasize initiation and propagation studies in pH 5 -9 range
- Determine the importance of slow TTS and TSP Denting

M & C

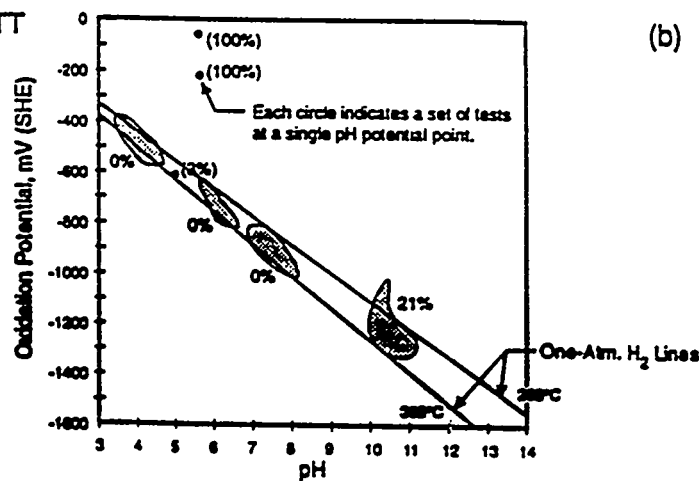
Summary

- **Most forms of SG degradation have been brought under control**
- **Many of the key features of corrosion and water chemistry are understood, but some questions still need to be answered**
- **The development of predictive mechanistic models is a goal, but semi-analytical (semi-empirical) probabilistic models are more reasonable short term goal**
- **Collaborative R&D efforts are needed to focus on key unanswered questions**

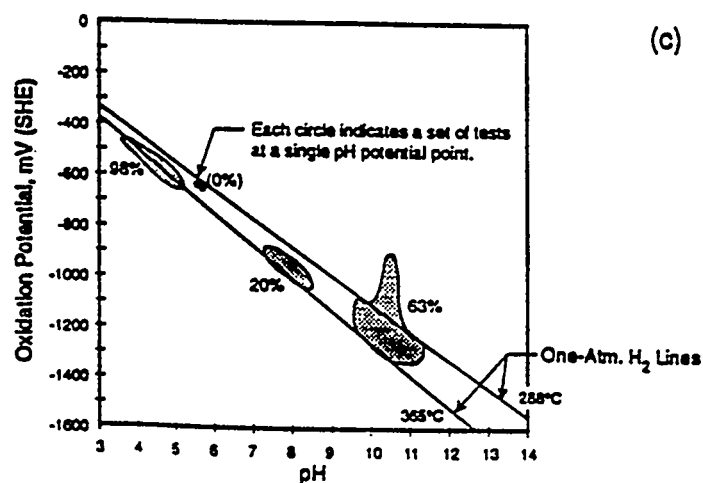
Inconel 600MA



Inconel 690TT



Incoloy 800



R.W. Staehle, "Occurrence of Modes and Submodes of IGC and SCC," presented at workshop on Improving the Understanding and Control of Corrosion on the Secondary Side of Steam Generators, Airlie, October 9-13, 1995.



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NDE Engineering Department

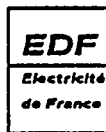
CNRA Workshop - Oak Brook (IL) - Oct. 30th to Nov. 2nd, 1995

Steam Generator Tubes Integrity

In-Service-Inspection

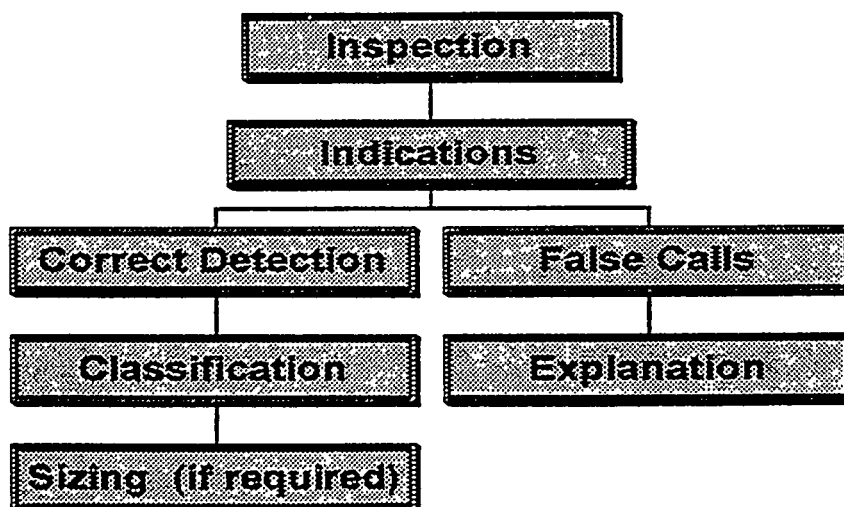
Robert J. Comby

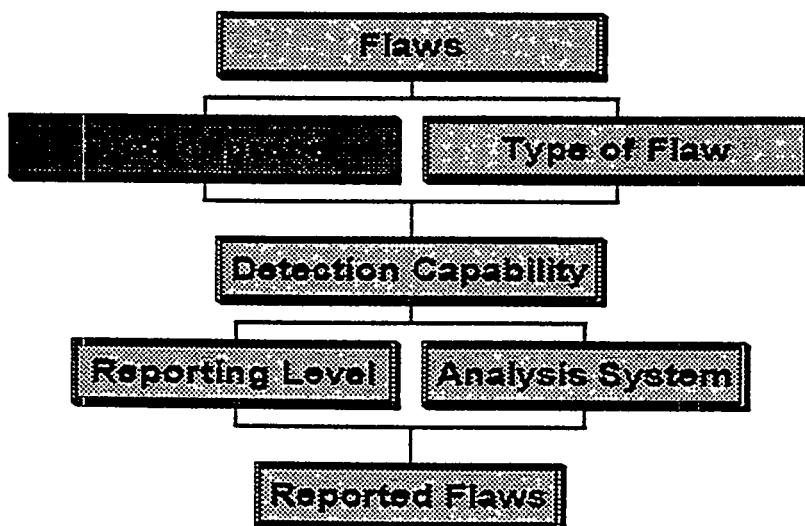
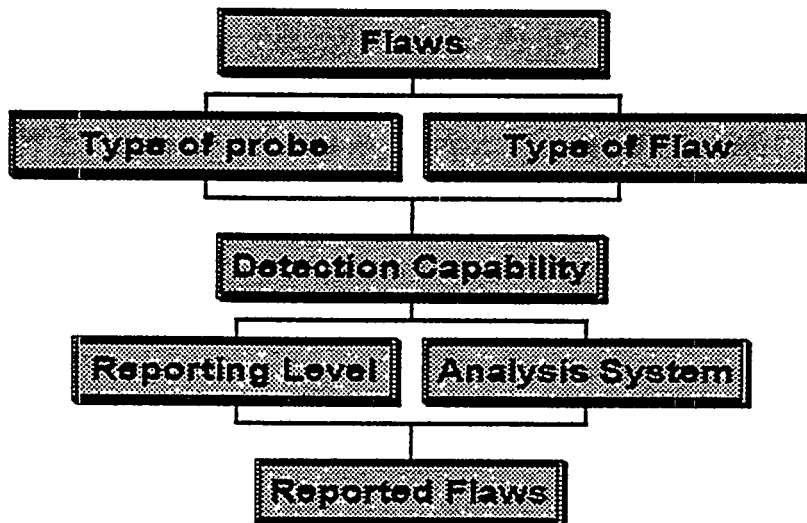
EDF / GDL / NDE Engineering Dpt



Central Laboratories

NDE Engineering Department





TYPE OF PROBE

1/4

- **ECT Bobbin Coils:** widely used as a multi purpose probe for detection of major types of flaws.
- **ECT Rotating Probes:** Often used for specific flaws :
 - Detection and / or sizing of Axial IGSCC
 - Detection and sizing of Circumferential IGSCC

TYPE OF PROBE

2/4

- **ECT Array Coils:** can be used on an economic basis to replace ECT rotating probes.
- but, detection capabilities and sizing accuracy to be compared with those obtained by rotating probes.



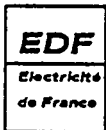
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TYPE OF PROBE

3 / 4

- **Emerging Technologies :**
 - **Transmit - Receive probes**
(Array probes or Rotating probes)
 - **" Multi Coil Rotating Probes "**
 - **Rotating Field Probes**



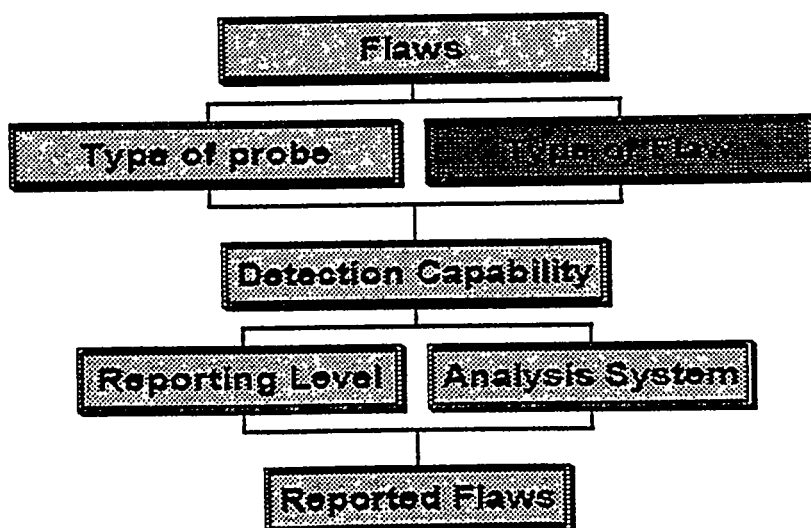
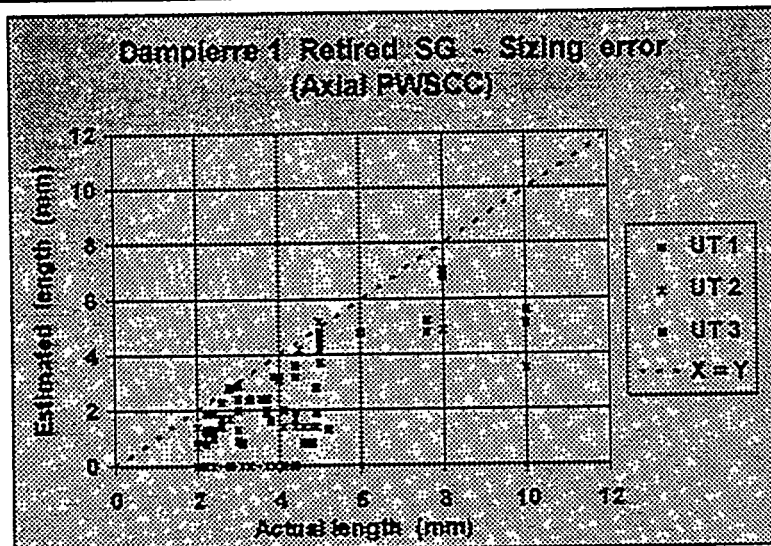
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TYPE OF PROBE

4 / 4

- **UT Probes :**
 - **Should improve detection of small depth cracks .**
 - **Best sizing methodology to be defined .**
 - **But sometimes , problems have been reported**
(false calls , sizing errors)

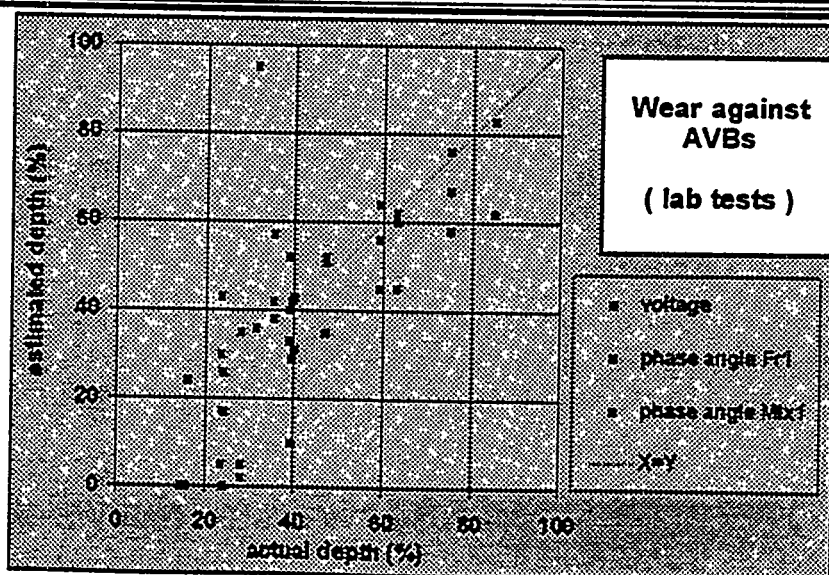


Flaw Specific Analysis

Example 1: Wear Against AVBs

- Use of single frequency phase angle
- Use of Mix phase angle
- Use of specific voltage calibration curve

(EDF tests)





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Dampierre (French PWR) - Unit 1 - Retired SG

Inspection of large radius U-bends with wear at AVBs

4 procedures using Bobbin coil probes

3 procedures using RPC

Tube R31C35 - To be pulled out in 1996



Central Laboratories

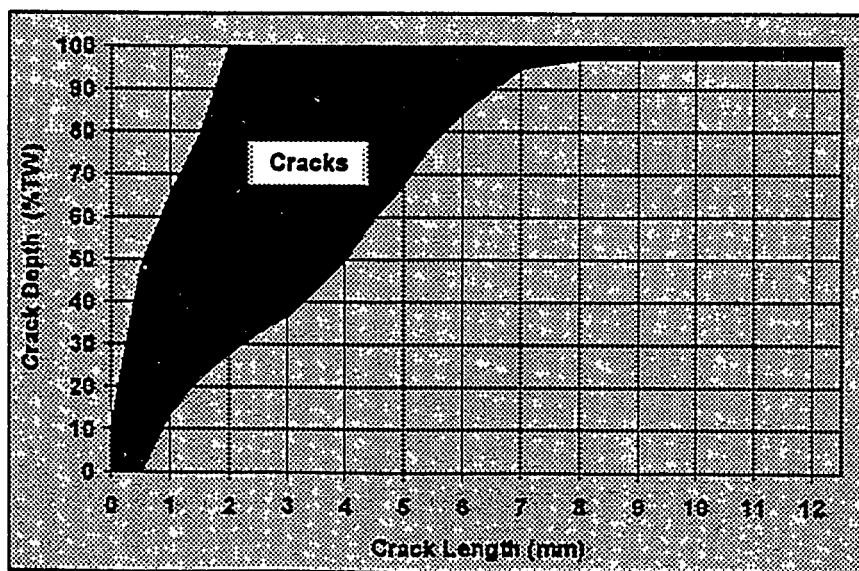
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Technique	Estimated Depth	Analysis based on	Used in the field
Bobbin 1	37 %	voltage	Yes
Bobbin 2	36 %	voltage	Yes
Bobbin 3	98 %	phase	Yes
Bobbin 4	76 %	phase	No
RPC 1	89 %	phase	No
RPC 2	12 %	voltage	No
RPC 3	40 %	phase	No

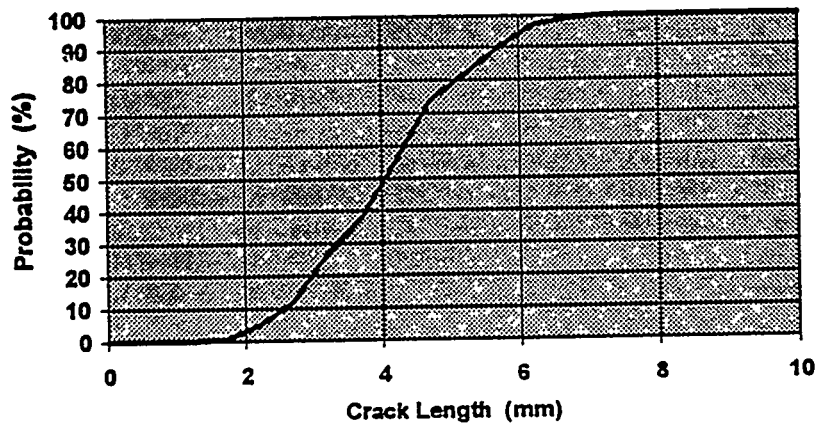
Flaw Specific Analysis

Example 2 : Axial PWSCC in Roll Transition

- Morphology of cracks ?
- Detection capabilities of probes ?
- Sizing performances ?



**Probability for a crack for reaching 100% TW
(based on 250 real cracks in roll transition)**



Flaw Specific Analysis

Example : Axial PWSCG in Roll Transition

- Cracks reach 100% TW very quickly .
-
- Detection capabilities of probes ?
- Sizing performances ?



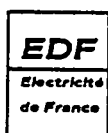
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Flaw Specific Analysis

Example : Axial PWSCC in Roll Transition

- Cracks reach 100% TW very quickly.
- Bobbin coil detects only deep cracks.
- ECT/RPC or UT inspection doesn't guarantee detection of cracks with a depth $< 40\%$.
- Sizing performances?



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Flaw Specific Analysis

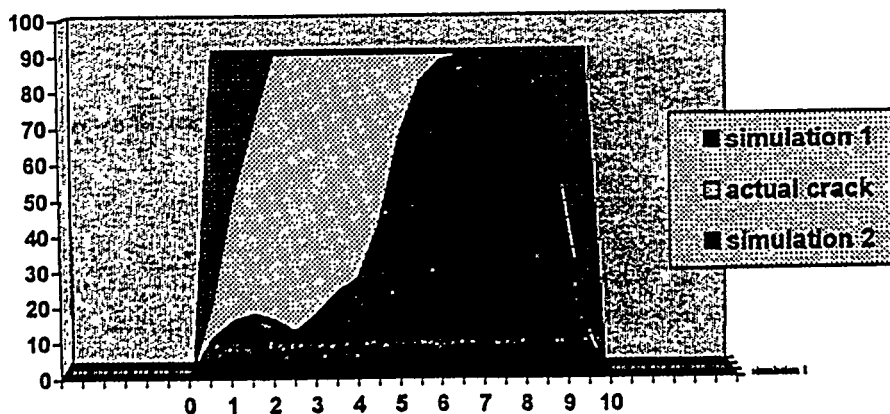
Example : Axial PWSCC in Roll Transition

- Cracks reach 100% TW very quickly.
- Bobbin coil doesn't detect partial TW cracks.
- Cracks detected by ECT/RPC or UT are often nearly 100% TW, due to their shape.
- ECT/RPC or UT probes can measure the length but sizing accuracy must be demonstrated.

Sizing performance for axial cracks

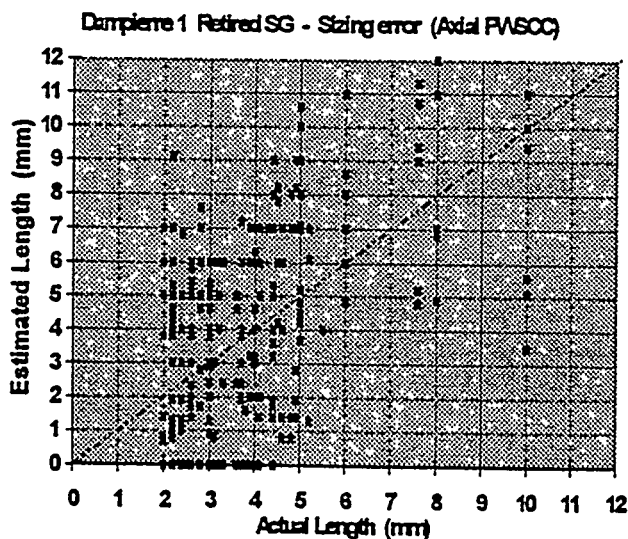
Lab samples must be representative of real defects in their morphology.

Use of pulled tubes destructive examination results is necessary.



In the previous example :

- **Simulation 1 will probably underestimate the actual length .**
- **Simulation 2 should lead to good results , but needs to be validated with real flaws .**

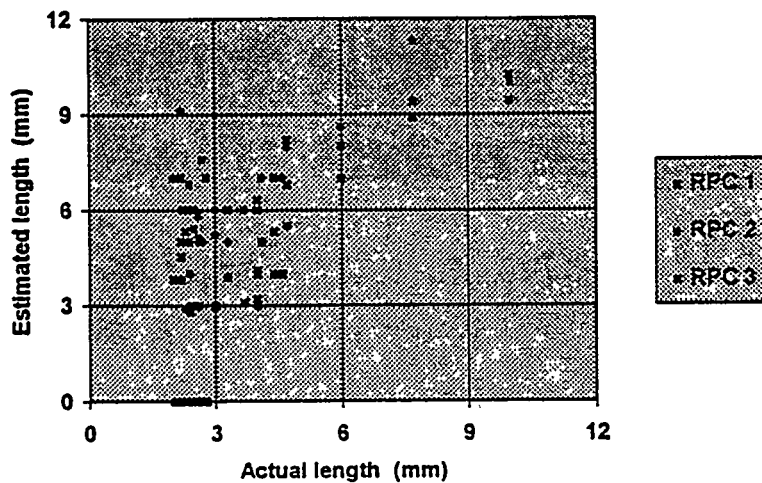


Overall results

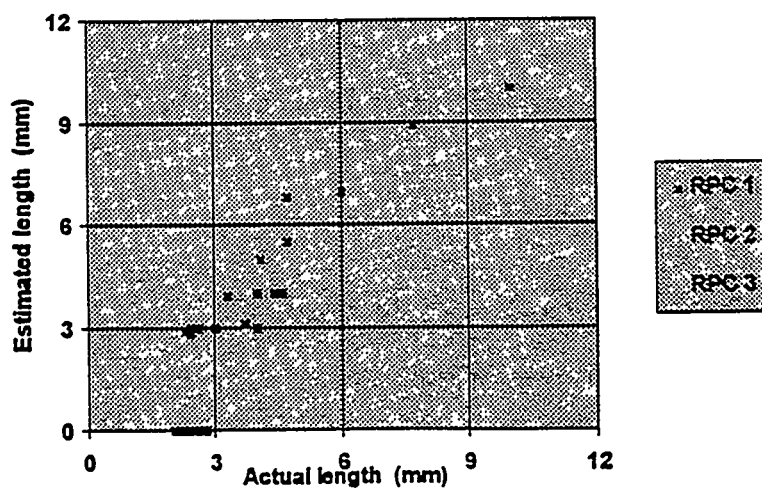
7 procedures

Including ECT & UT

Dampierre 1 Retired SG : Sizing error (Axial PWSCC)



Dampierre 1 Retired SG : Sizing error (Axial PWSCC)

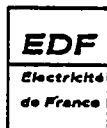




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- **Sizing methodology must be established or, at least validated on representative flaws or real ones (when possible).**
- **Analysis of statistical results must be done carefully because these results can include non qualified techniques or procedures and lead to false conclusions.**



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Flaw Specific Analysis

Example : Circumferential cracking

- **Difficult problem due to :**
 - **different origins of cracking .**
 - **different morphologies .**
 - **Few knowndge about growth rate .**



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Flaw Specific Analysis

Example : Circumferential cracking

- **First "philosophy" :**
 - **Repair all degraded tubes .**
 - > **Detection level and inspection program must be in accordance with critical size .**



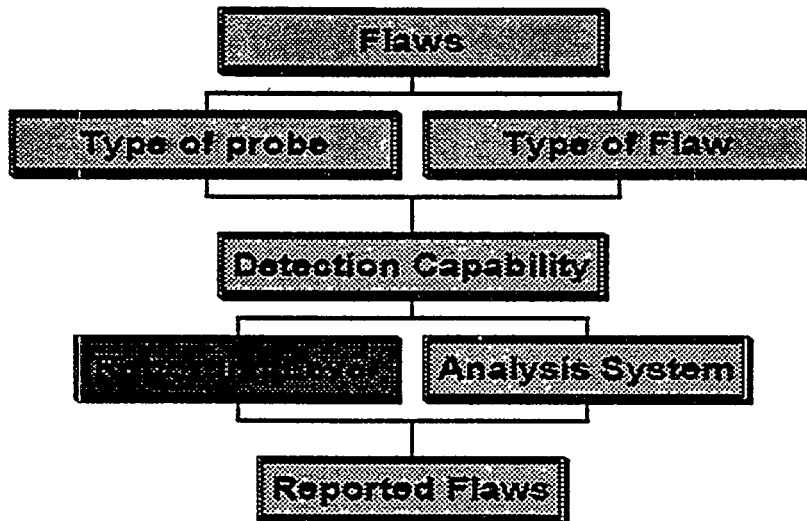
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Flaw Specific Analysis

Example : Circumferential cracking

- **Second "philosophy" :**
 - **Use of an alternate repair criteria .**
 - > **Detection level sufficiently low .**
 - > **Good sizing capabilities in depth .**



REPORTING LEVEL

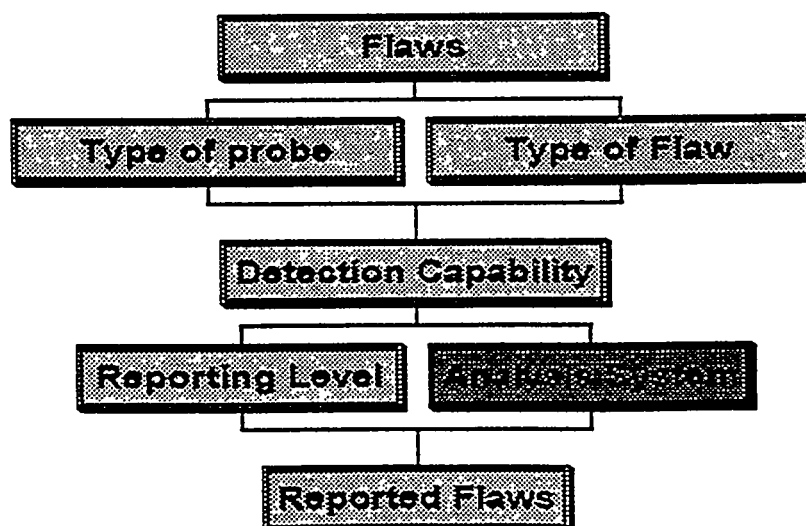
Advantages:

- prevent from large number of false calls
- limit the risk of classification error
- reduce the time required for analysis
- is required for automatic data screening

REPORTING LEVEL

But :

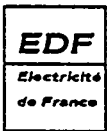
- **Must be established for each type of flaw**
- **Must be compatible with repair criteria**
- **Needs a correct classification by NDE**





1. "Manual" Analysis

- Strongly dependant on data analyst qualification
- Effect on human reliability : environment ,
physiological conditions of analysts .
- Independant two-party analysis increase detection probability .
- Is two-party analysis sufficient ?



2. Computer Data Screening

- Can be "Simple-Threshold Data Screening"

But, can lead to missed indications if the threshold is not context related , or when the threshold value is not supported by a strong feedback experience .



2. Computer Data Screening

- Can be "Rule-based Data Screening"

But, is experienced based. Such system has not been trained to recognize all conditions (new type of flaw, poor knowledge of some flaws, etc...)



2. Computer Data Screening

- "Rule-based Data Screening" can be used with further manual analysis on optimized visualization of signals corresponding to all tube zones, on screen or printings. Indications missed by the computer must be easily and quickly detected by using this visualization for final analysis.



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CONCLUSION

**No International simple rules can be fixed to adopt
an universal inspection methodology because of
various concepts related to experience , leak acceptance ,
Leak before Break approach , etc..**

**Flaw specific management is probably the most reliable approach
as a compromise between safety , availability and economic
issues . In that case , NDE capabilities have to be in accordance
with informations required by Structural Integrity demonstration .**

**STRUCTURAL and LEAKAGE
INTEGRITY
of TUBES AFFECTED by
CIRCUMFERENTIAL CRACKING**

— — — — —

presented by
P. Hernalsteen (TRACTEBEL/Belgium)

at the NEA/CNRA/CSNI International Workshop
on Steam Generator Integrity in Nuclear Power Plants
Oak Brook, Illinois, October 30 - November 2, 1995

OUTLINE

- ♦ CONTEXT AND OBJECTIVE
- ♦ STRUCTURAL ANALYSIS
- ♦ INTEGRATED BURST PRESSURE DATABASE
- ♦ LEAKAGE ANALYSIS
- ♦ SGDSM IMPLEMENTATION
- ♦ SUMMARY
- ♦ CONCLUSION

CONTEXT and OBJECTIVE

- ◆ CIRCUMFERENTIAL CRACKS GENERALLY CONSIDERED UNACCEPTABLE
(more dangerous than axial cracks ?)
- ◆ NEED TO DIFFERENTIATE TWO FACETS :
 - ① Sizing } → Knowledge of EOC size
Growth }
 - ② Structural Strength } from known
Leakage potential } crack size
- ◆ PURPOSE OF THIS PAPER :
Show that concern not justified for facet ②
- ◆ STRUCTURAL STRENGTH (burst pressure)
 - Analytical model (NSCL based)
 - Validation on large burst data base
 - Extrapolation from testing to service conditions
 - SGDSM implementation
- ◆ LEAK RATE MODEL
- ◆ COMPARISON TO AXIAL CRACKS
 - Burst Strength
 - Leak Rate

STRUCTURAL ANALYSIS PRINCIPLES

♦ FLAW MORPHOLOGY

- Main segment
 - 100 % TWD, or
 - partial TW (ligament t_0)
- Partial penetration outside main flaw (ligament t')

♦ STRUCTURAL MODEL (3 components)

- "Net Section Collapse Load" for degraded cross section
- "Beam modelization" for tube lateral restraint (TSP, FDB, EC, ...)
 - Stiffness index K (0.6 to 0.9)
- "Equivalent pressure" for load on crack lips
 - local reinforcement in testing conditions
 - fluid pressure in service conditions
 - Load index η (-1 to +1)

♦ FAILURE MODE

- Bending failure mode
- 2 limit cases

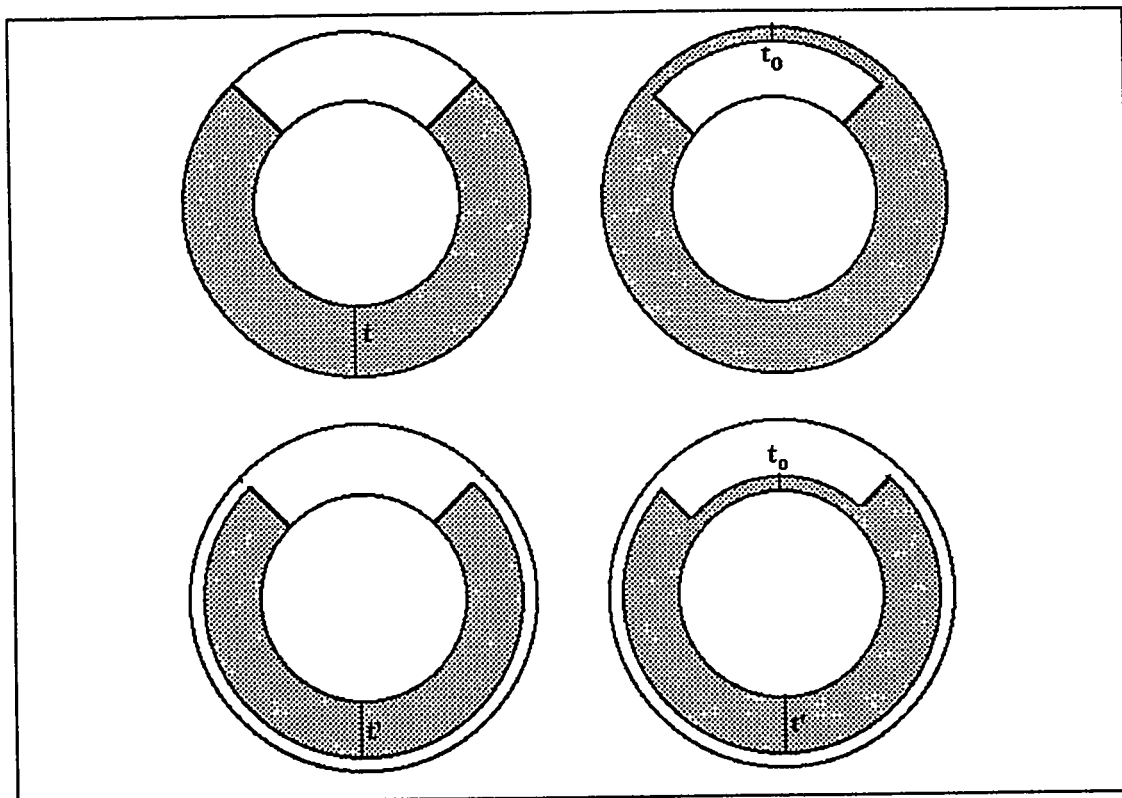
{	axial burst (short cracks)
	tensile failure (long cracks)

♦ NORMALIZATION

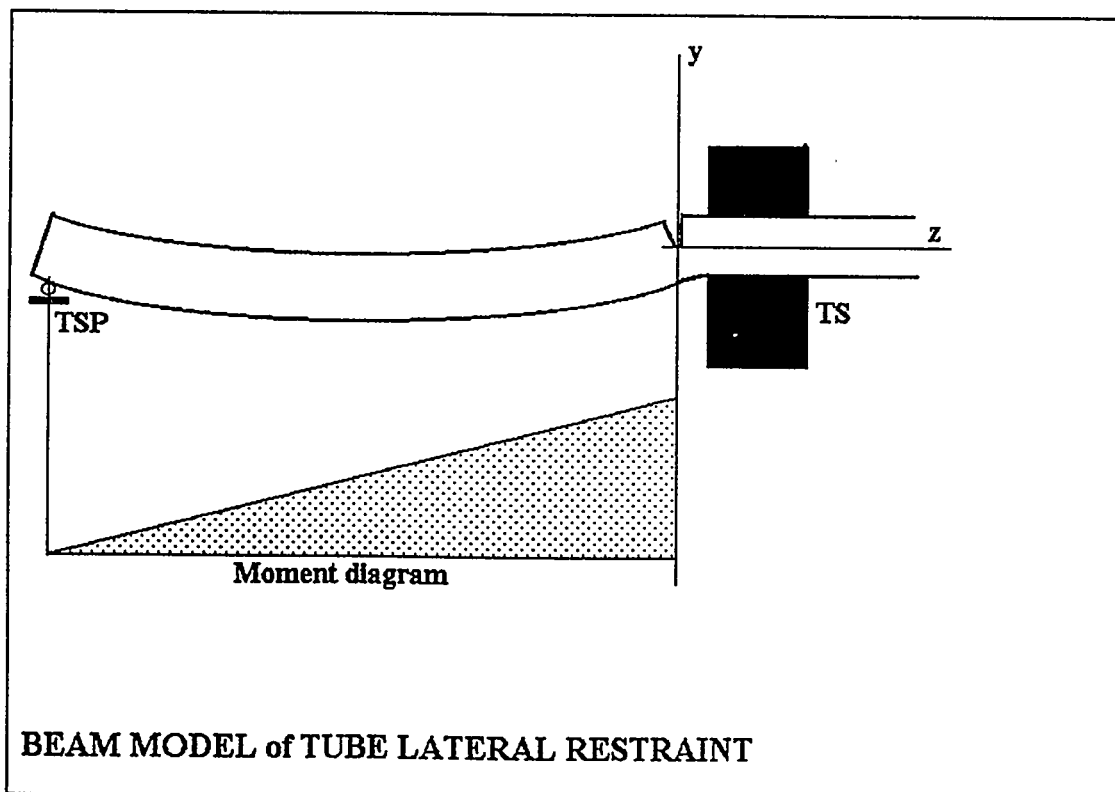
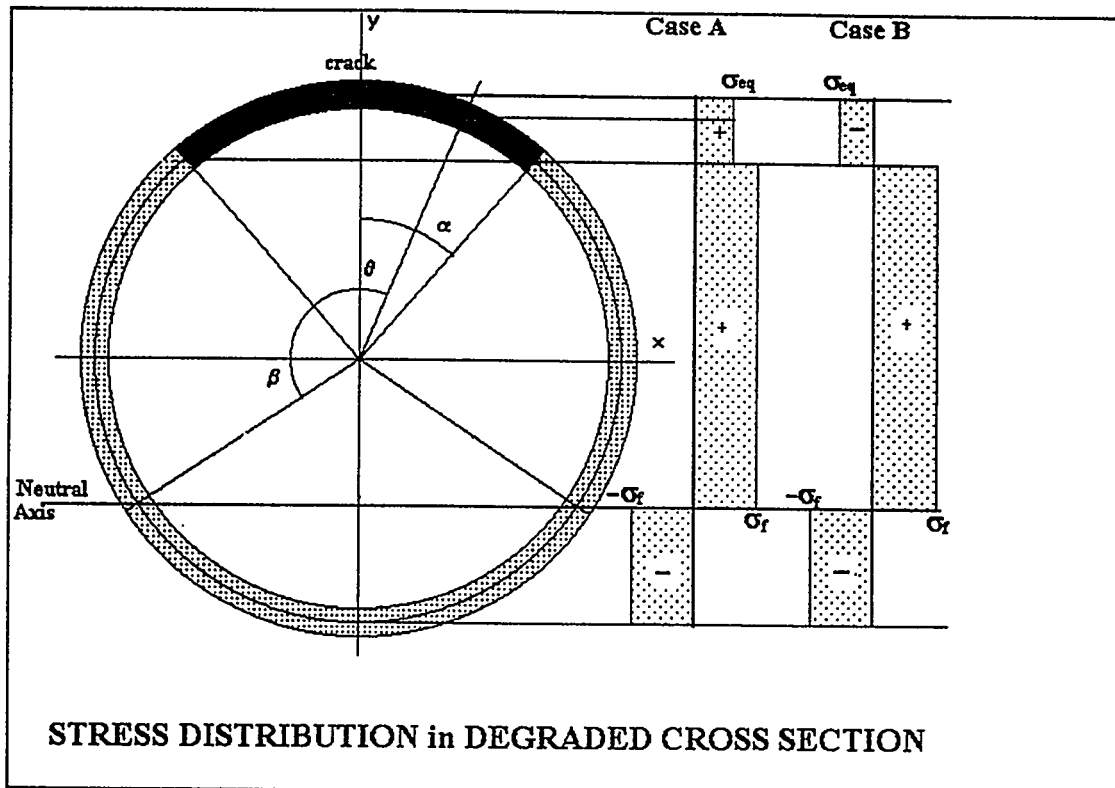
- Burst Pressure normalized to material flow stress

$$P_n = P_{\text{burst}} / \sigma_f \text{ with } \sigma_f = 0,53 (\sigma_u + \sigma_y)$$

FLAW MORPHOLOGY



STRUCTURAL MODEL



STIFFNESS INDEX (K)

- ◆ DEFINITION : $K = \sigma_{burst} / \sigma_f$

with σ_{burst} = maximum axial stress
(membrane + bending) in tube,
near the degraded section

- ◆ BEAM MODEL ASSUMPTION

BENDING FAILURE REQUIRES LARGE
ANGULAR DEFLECTION ASSOCIATED WITH
INCIPIENT PLASTICITY IN RESTRAINED TUBE

→ K IS CONSTANT FOR A GIVEN RESTRAINT
CONFIGURATION, CHARACTERIZED BY A
ROTATIONAL STIFFNESS

$S = M / (y' E I)$ with M = applied moment to
tube
 $y' =$ resulting angular
deflection
E = modulus of elasticity
I = second moment of
area

- ◆ MEASUREMENT of the STIFFNESS INDEX

$$K = \frac{\left[\frac{M_{burst}}{\pi R^2 t} + \frac{P_{burst} \left(\frac{R}{t} - 1 \right)}{2} \right]}{\sigma_f}$$

- ◆ CORRELATION between K and S
(Calculation of S → K)

LOAD INDEX (η)

◆ EQUIVALENT STRESS

ASSUMED UNIFORM STRESS ON CRACK LIPS,
UNDER

- LABORATORY TESTING CONDITIONS
(friction load from metal foil used as a local reinforcement of 100 % TWD flaws)
- SG SERVICE CONDITIONS
(fluid pressure acting on lips of an ID or TWD crack)

◆ LOAD INDEX

$$\eta = \sigma_{eq} / P$$

= $f w / t$ under testing conditions (foil of width $2w$, with friction factor f)

Typical value = 1

($\eta > 0 \rightarrow$ increased burst pressure)

= $\sim c P$ under service conditions

Typical value = - 0.5

($\eta < 0 \rightarrow$ decreased burst pressure)

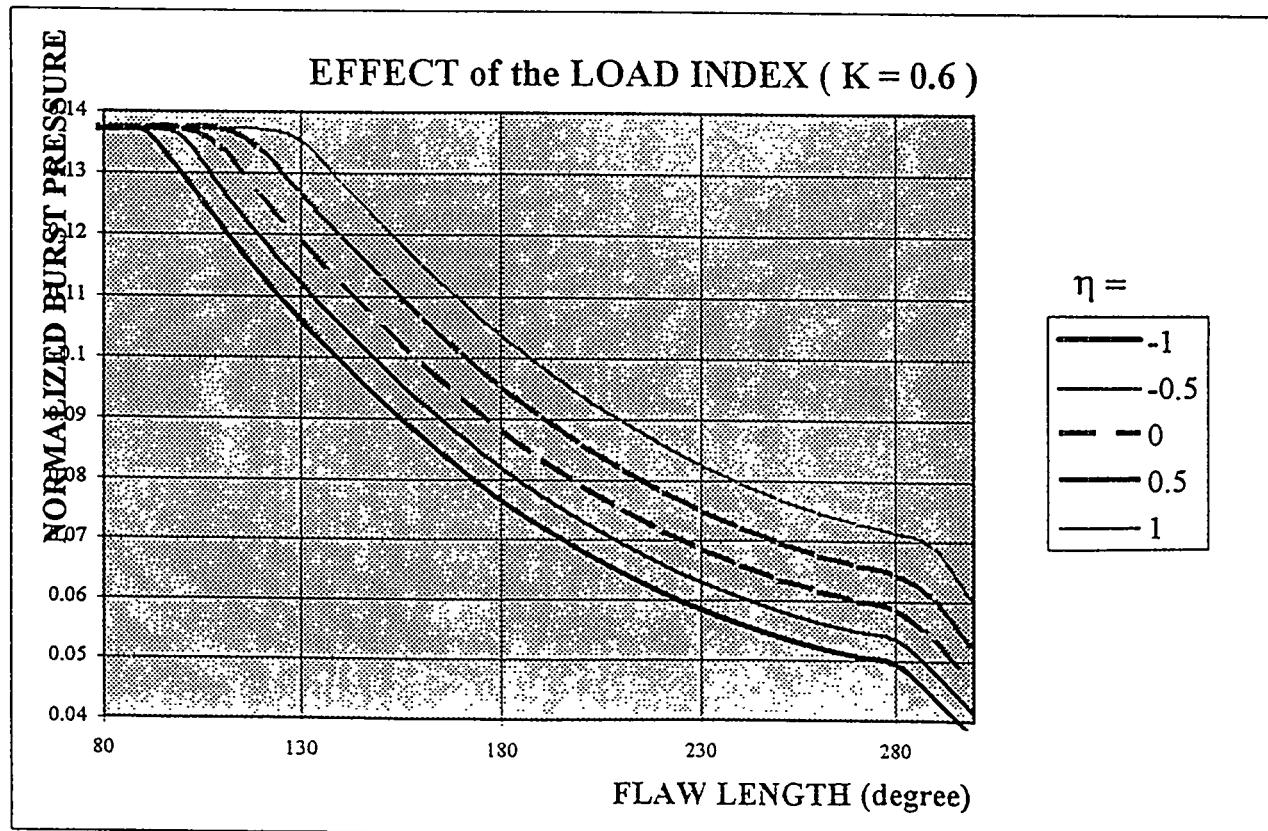
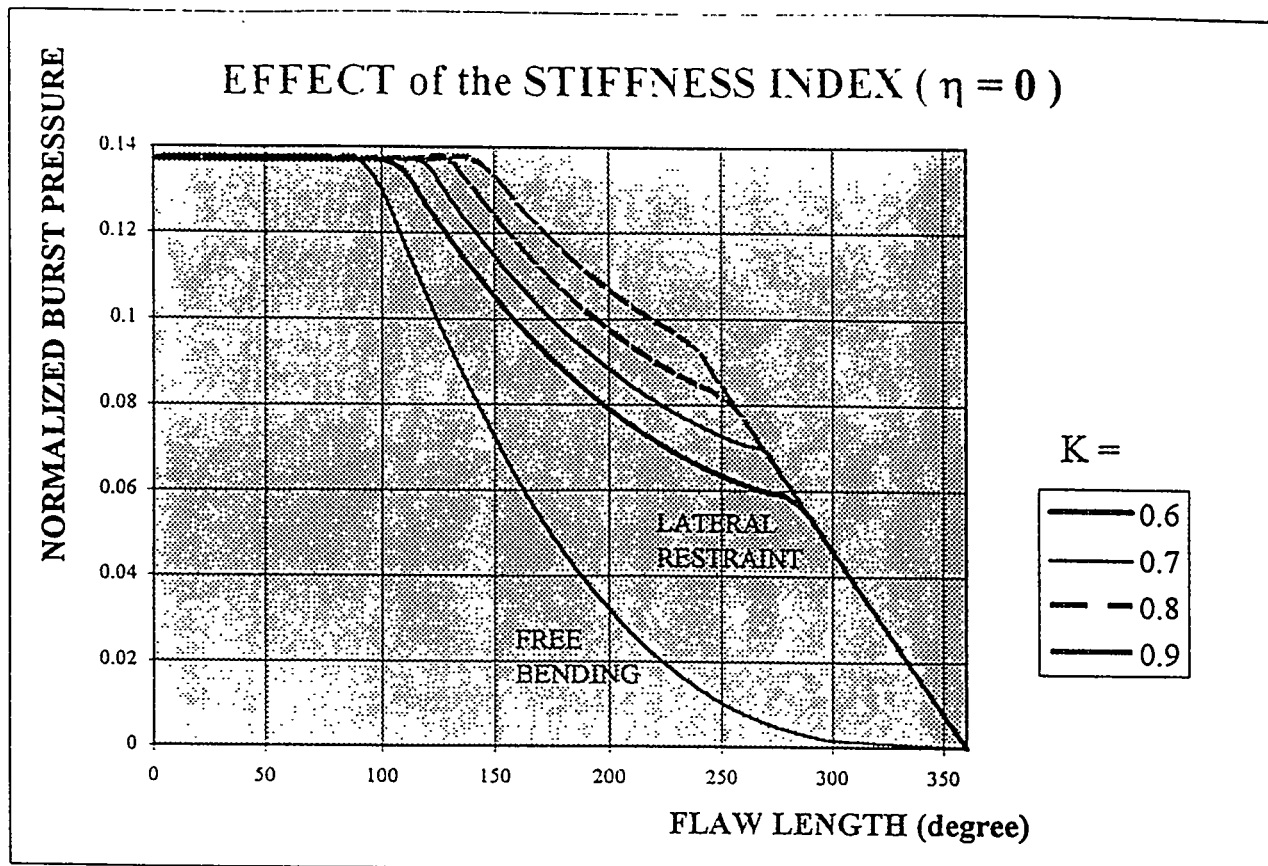
STRUCTURAL ANALYSIS RESULT

- ◆ SIMPLE PROCEDURE for SOLVING the NORMALIZED BURST PRESSURE P_n as a function of :

- (main) FLAW LENGTH : 2α
- STIFFNESS INDEX : K
- LOAD INDEX : η
- LIGAMENT in main flaw : t_o/t
- PENETRATION outside main flaw : t'/t

- ◆ TABLES and DIAGRAMS for predefined parameters

DEPENDANCE of NORMALIZED BURST PRESSURE ($P_n = P_{burst} / \sigma_f$) on FLAW LENGTH (100 % TWD)



BURST PRESSURE DATA INTEGRATED DATABASE

◆ PRESENT STATUS

- about 100 data points from :
 - Belgian Laboratory Test Program / Laborelec (1982 - 89)
 - Millstone 2 Laboratory Test Program / CE and W (1989 - 91)
- To be further extended (EPRI)

◆ WIDE VARIATION IN TESTING PARAMETERS

- 2 tube sizes (3/4" - 7/8")
- 2 R/t ratios (W - CE)
- 4 flaw morphologies (+ real cracks)
- Free or restrained bending (+ axial and tensile failure limit cases)
- 8 lateral support configurations
- range of main flaw length : 100 to 300 °
- range of lateral support offset : 0 to 0.8 "
- 3 independent testing laboratories

INTEGRATED BURST DATA BASE

TEST MATRIX (number of data points)		
	BELGIUM (LABORELEC)	MILLSTONE 2 (CE + Westinghouse)
Free Bending	17 (0)	18 (0)
Restrained Bending	29 (0)	14 (2)
Axisymmetric flaw (or tensile test)	11 (11)	16 (0)
TOTAL	58 (11)	48 (2)
GRAND TOTAL	106 data points	
Note : # of real cracks (pulled tubes) are mentionned in brackets		

BURST PRESSURE DATA STRUCTURAL MODEL VALIDATION

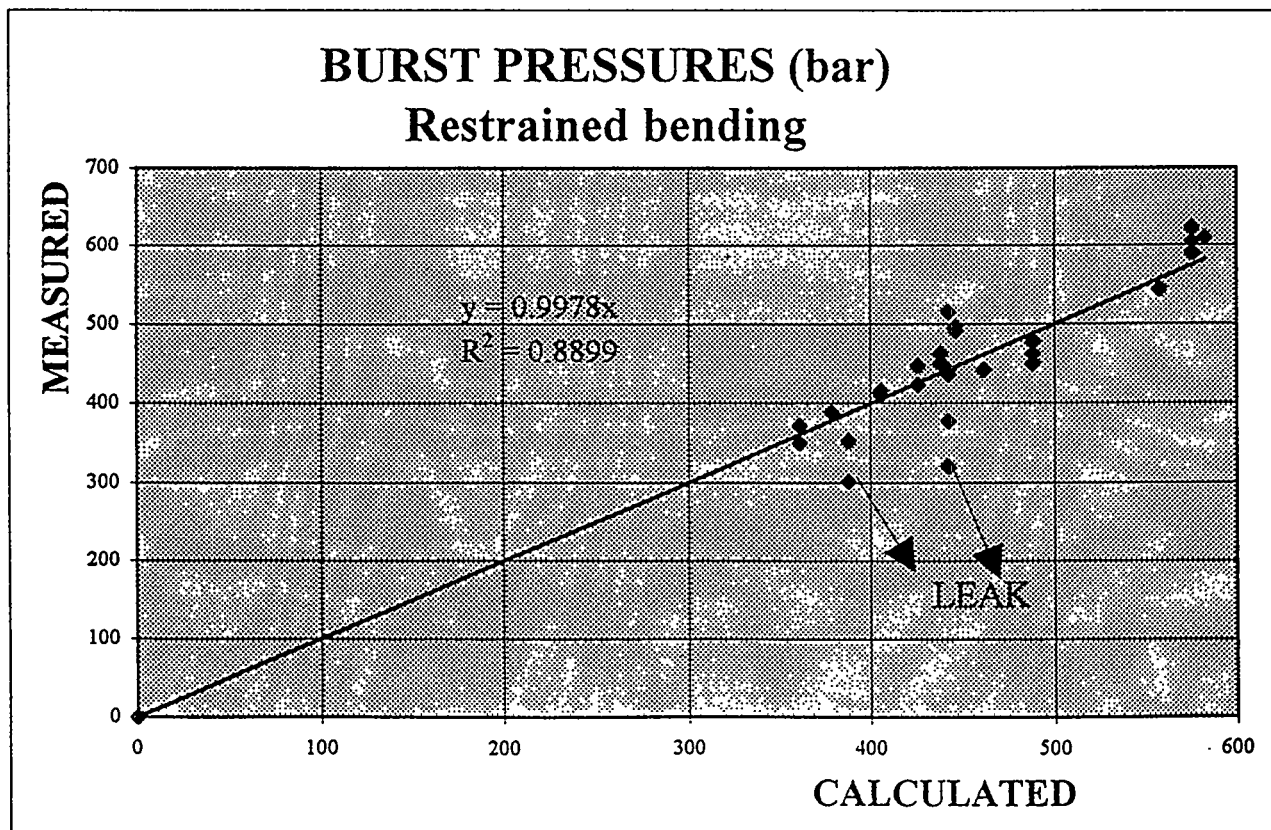
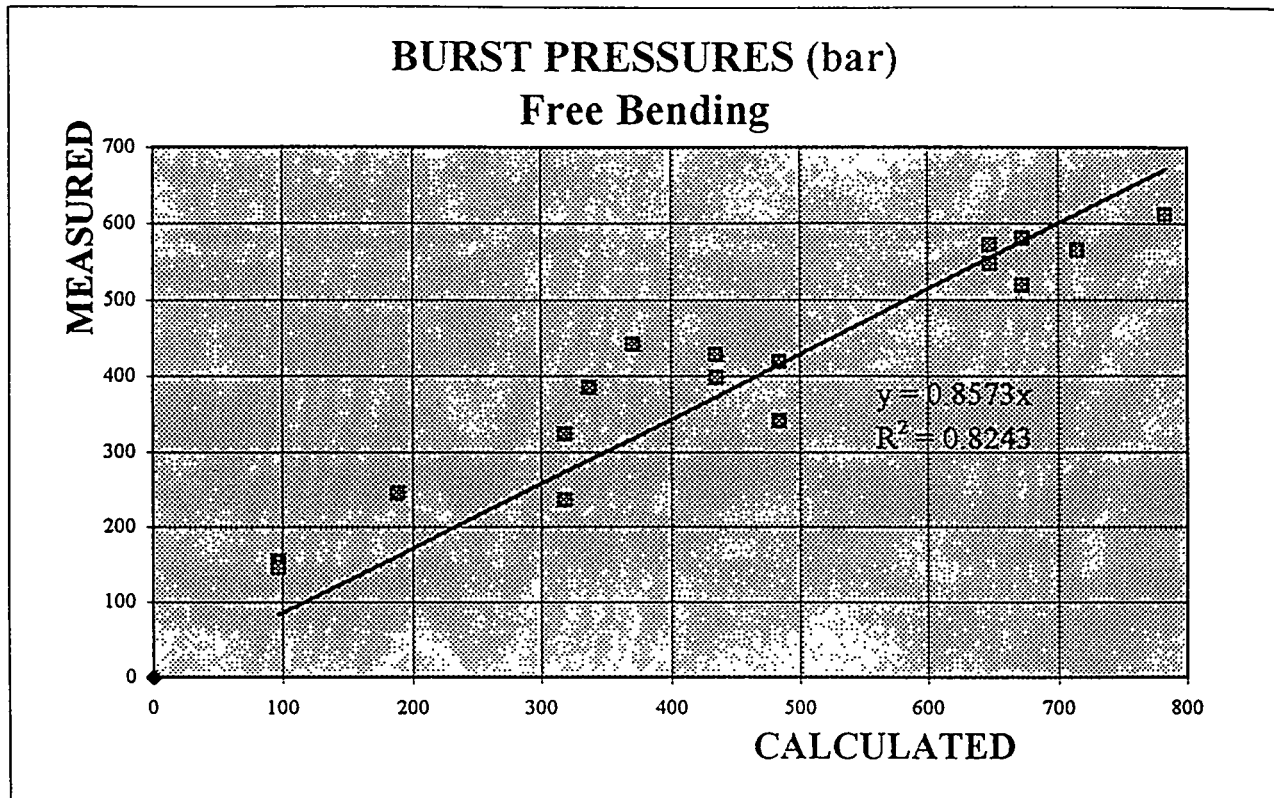
- ◆ **COMPARISON of MEASURED/CALCULATED DATA**
 - Excellent agreement in average (mean ratio $\cong 1$)
 - Moderate scatter in free bending
 - Low scatter in restrained bending (ratio standard deviation $\cong 0.08$)

- ◆ **ADEQUATE and SUFFICIENT VALIDATION SUPPORT for STRUCTURAL MODEL**

VALIDATION OF STRUCTURAL MODEL

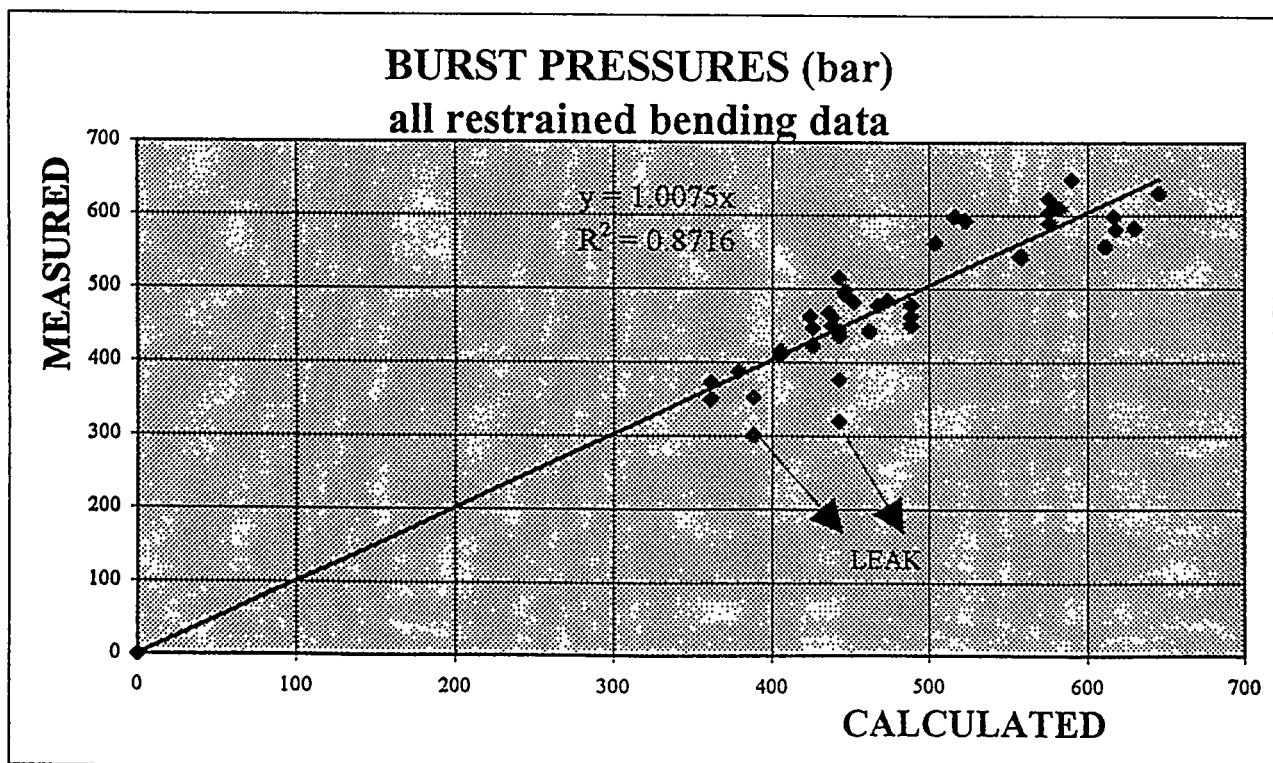
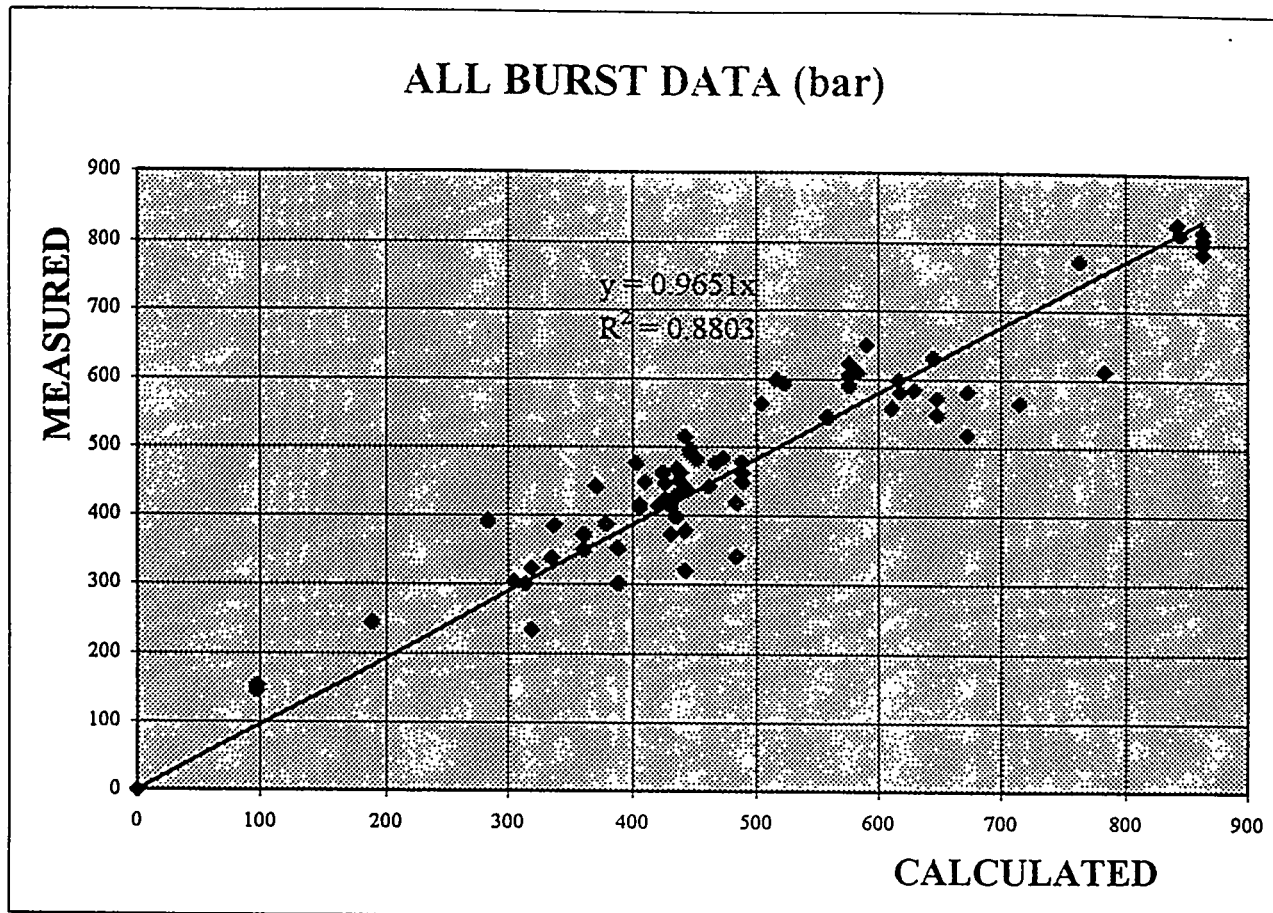
Comparison of burst pressure measurements against predicted values

Belgian data



VALIDATION OF STRUCTURAL MODEL

Comparison of burst pressure measurements against predicted values



LEAKAGE ANALYSIS PRINCIPLES

- ♦ METHODOLOGY : same as already used for axial cracks
 - In Belgium (TRACTEBEL / LABORELEC)
 - in USA (EPRI report NP 6864-L)

- ♦ BASE MODEL

$$Q = K A \sqrt{2 \rho (\Delta p)_{\text{eff}}}$$

- ♦ FLOW DISCHARGE COEFFICIENT K
= Function of mean crack width
(same as for axial cracks)

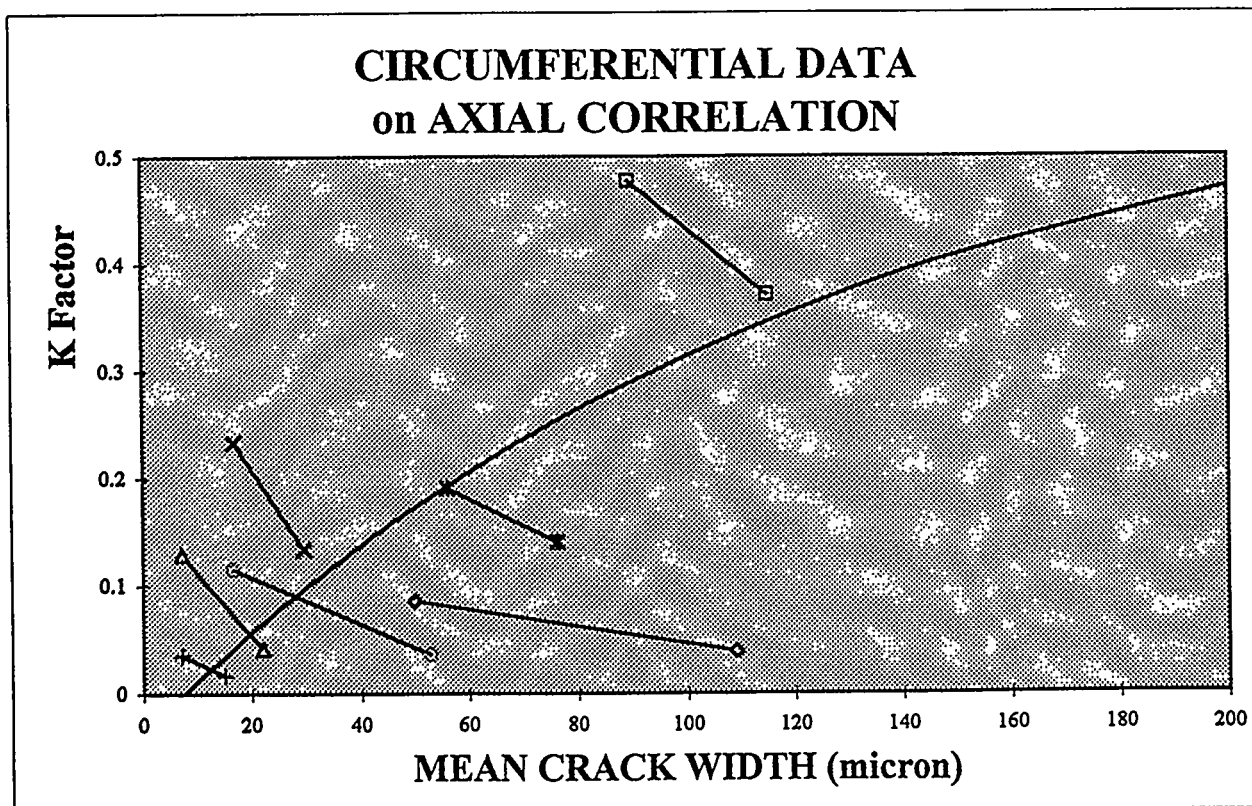
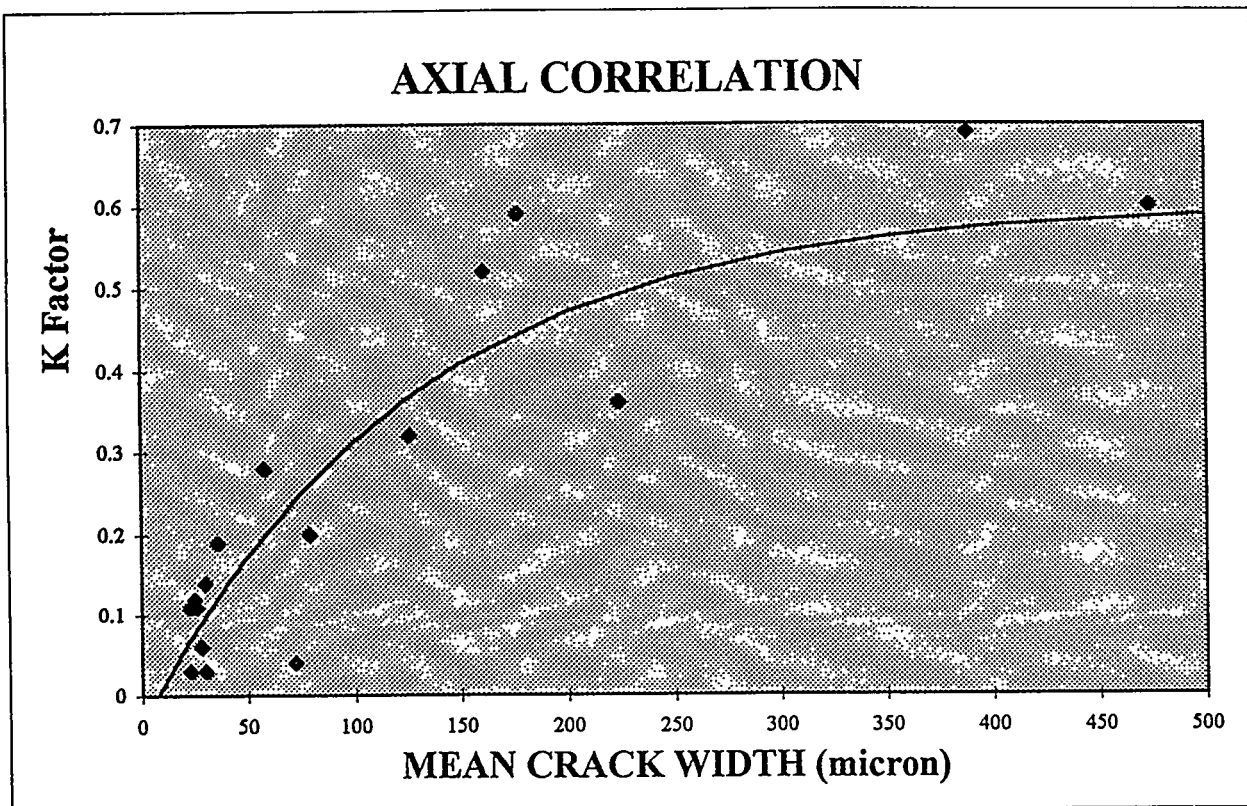
- ♦ LEAKAGE AREA A

$$A = A_{\text{el}} + A_{\text{cor}} + A_{\text{pl}}$$

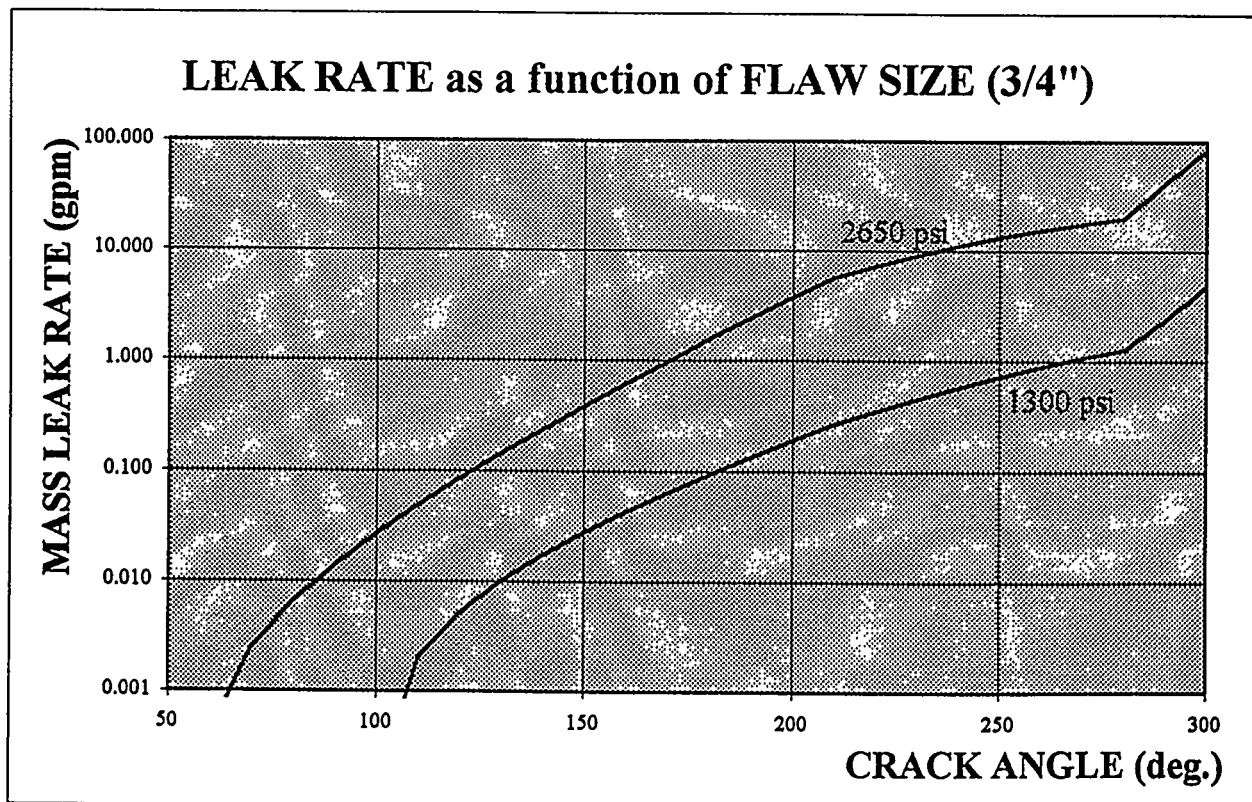
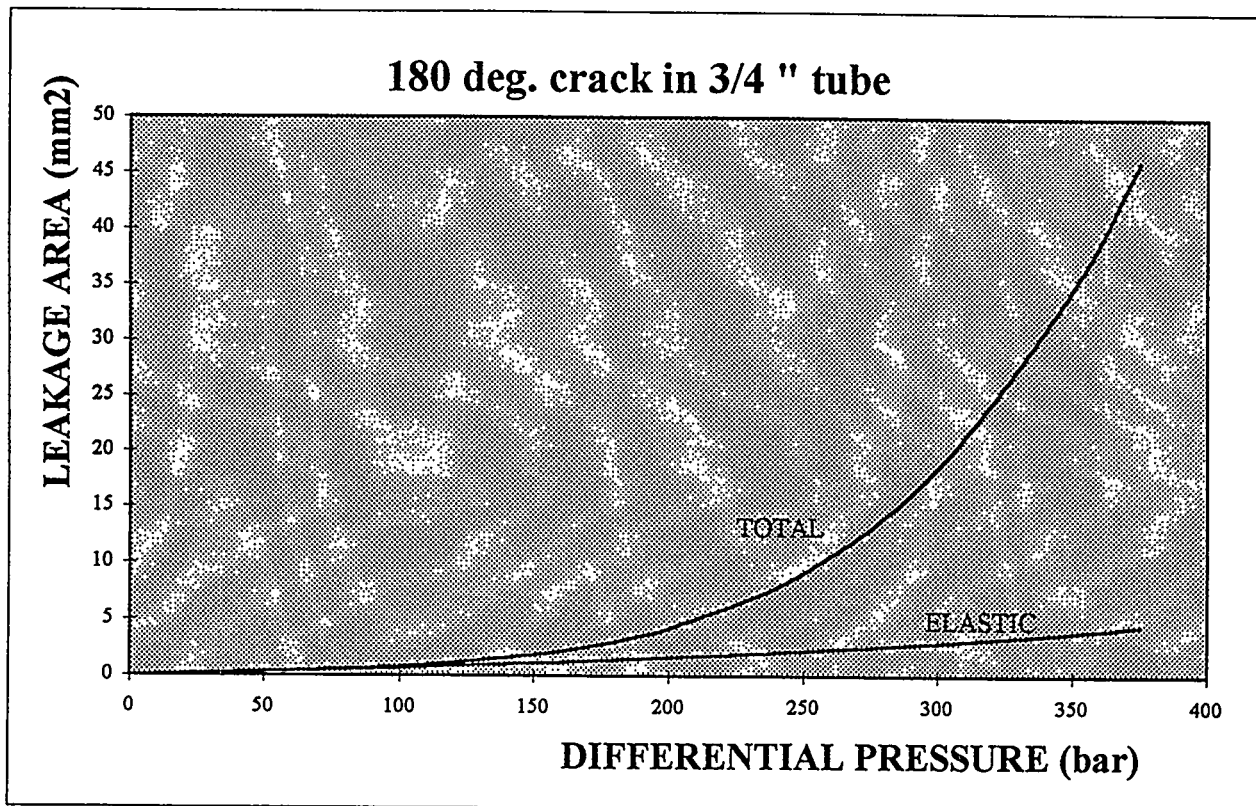
$$\left\{ \begin{array}{l} A_{\text{el}} = \text{elastic component} \\ A_{\text{cor}} = \text{small scale yielding correction} \end{array} \right\} \begin{array}{l} \text{from theory} \\ \text{(literature)} \end{array}$$
$$\left\{ \begin{array}{l} A_{\text{pl}} = \text{large scale plasticity from} \end{array} \right\} \begin{array}{l} \text{structural model} \\ \text{empirical support} \end{array}$$

NB : Main source of difference from axial cracks

FLOW DISCHARGE COEFFICIENT as a function of the MEAN CRACK WIDTH



LEAKAGE AREAS and LEAK RATES as a function of CRACK LENGTH DIFFERENTIAL PRESSURE



LEAKAGE ANALYSIS VALIDATION

- ◆ **MODEL VALIDATION**
 - **Components level**
 - K : same as for axial cracks
 - Δp_{eff} , A_{el} and A_{cor} : theory based
 - A_{pl} : limited laboratory support (pressure dependence)
 - **Global level** : limited number of reliable, documented field data

- ◆ **PRESENT STATUS IS SUFFICIENT TO CONCLUDE IN LOW LEAK RATES (as compared to axial cracks)**

STEAM GENERATOR DEFECT SPECIFIC MANAGEMENT SGDSM IMPLEMENTATION

◆ DEFINITION OF A REFERENCE CURVE

**= Single lower bound correlation between
burst pressure and flaw size**

◆ BASED on STRUCTURAL MODEL and SIMPLE (not over) CONSERVATIVE APPROXIMATION

- Total flaw area analysed as single TWD
segment (→ single size parameter =
average flaw depth)**
- LTL 650 °F mechanical properties**
- Lowest lateral restraint stiffness ($K = 0.6$)**
- Half differential pressure acting on crack
lips ($\eta = 0.5$)**

SGDSM IMPLEMENTATION (cont'd)

- ◆ **USE of REFERENCE CURVE for DEFINITION of STRUCTURAL LIMIT (allowable EOC flaw size)**
 - Intersection with MAX (3 NOP ; $\sqrt{2}$ MSLB)

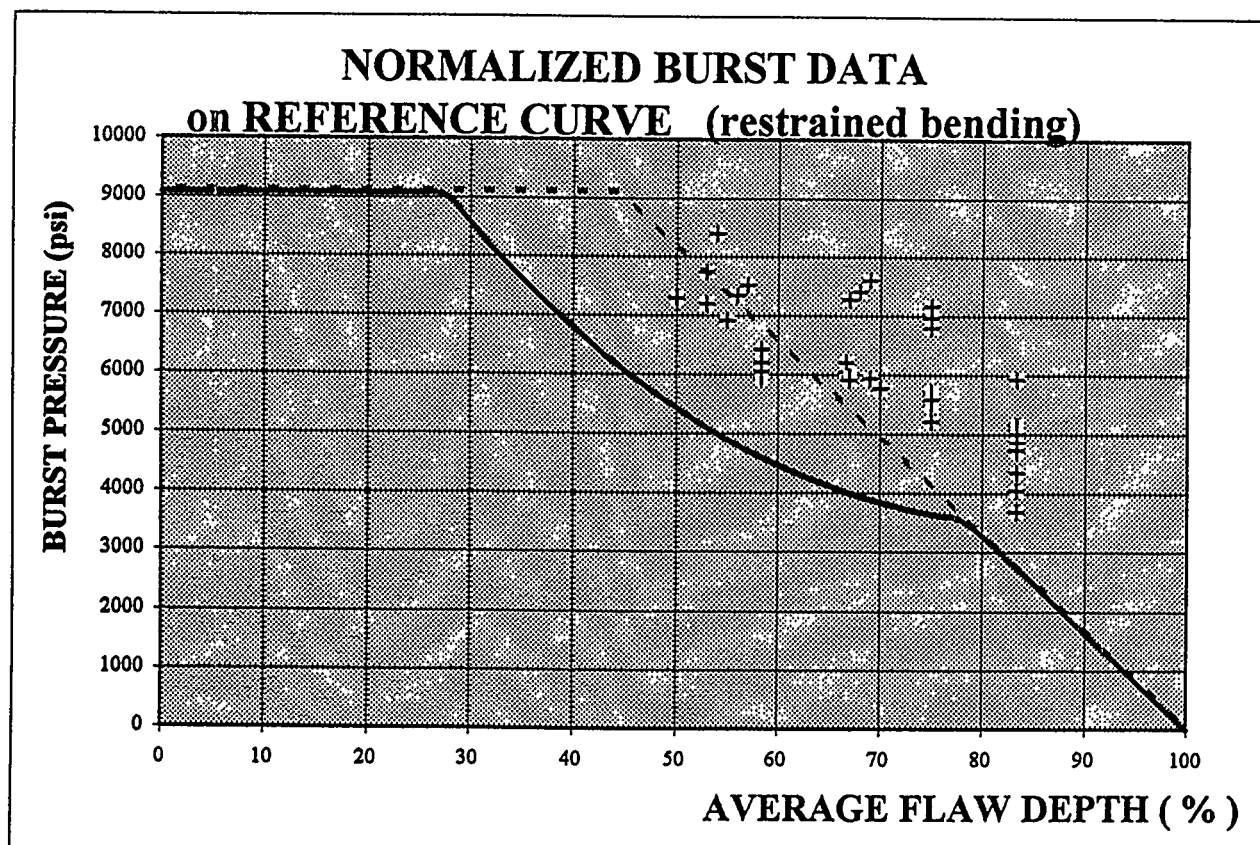
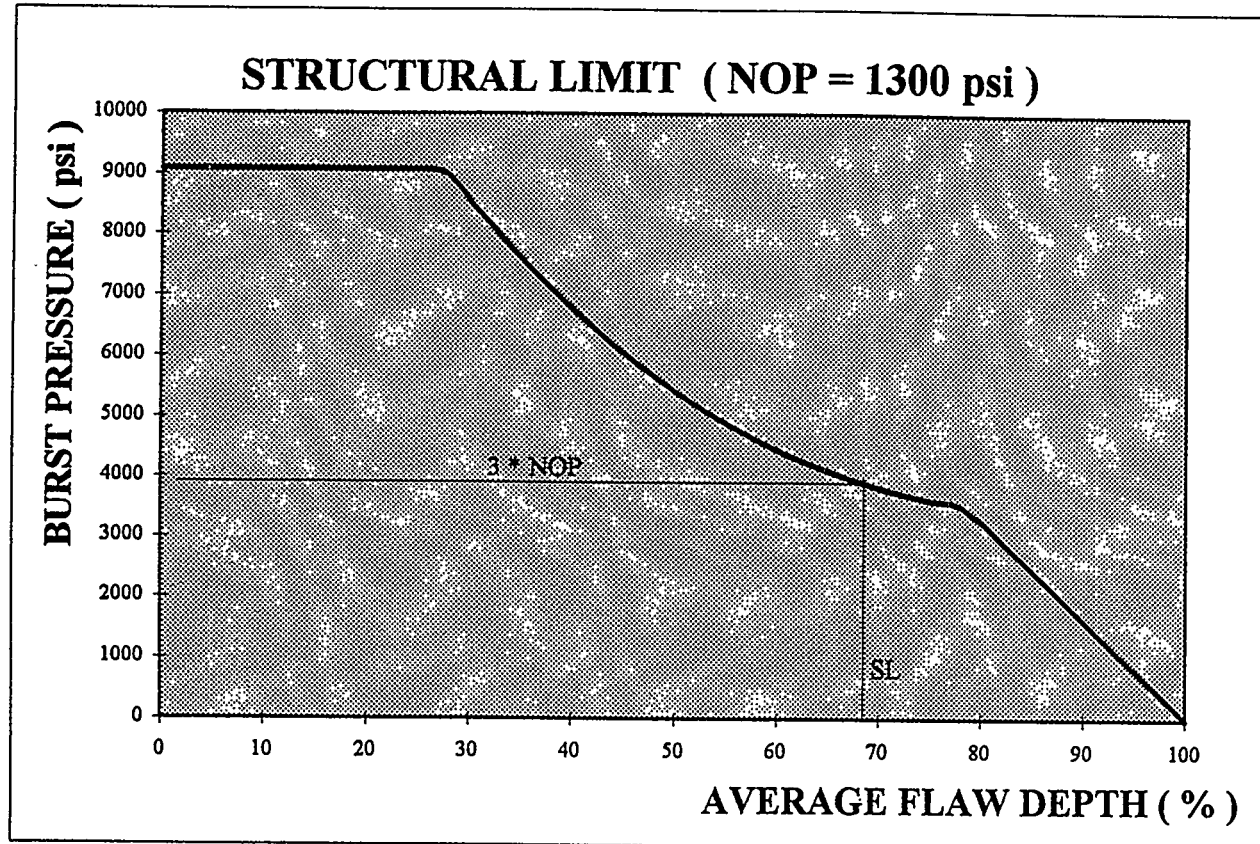
- ◆ **USE of STRUCTURAL LIMIT for CALCULATION of PSGTR**
 - Population of BOC average flaw depths
 - Statistical distributions of uncertainties for NDE sizing and growth rate
 - Probability of exceeding SL

- ◆ **USE of LEAK RATE MODEL for CALCULATION of ACCIDENT LEAKAGE (probability density function)**

- ◆ **DEFINITION of REPAIR LIMIT MAY REQUIRE IMPROVED CONTROL on SIZING and GROWTH**

PROPOSED S G D S M IMPLEMENTATION

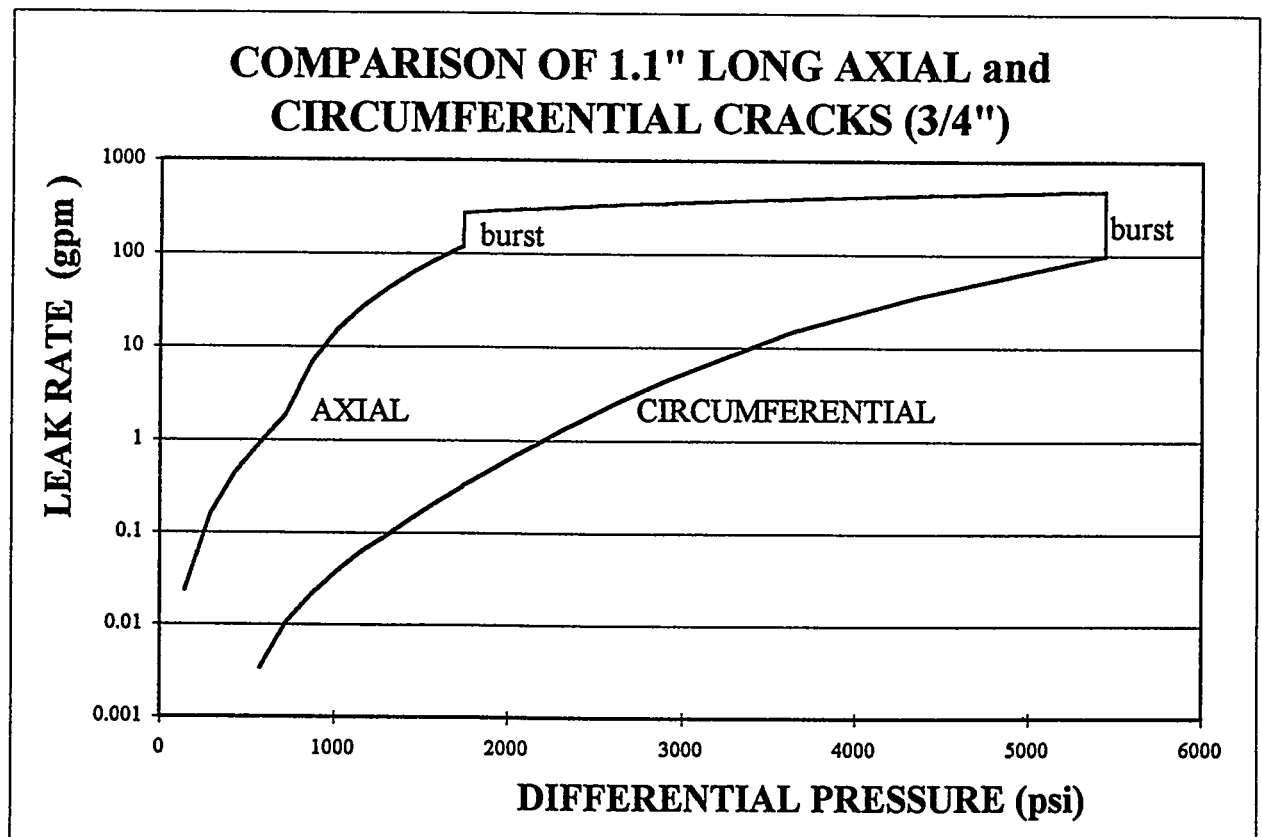
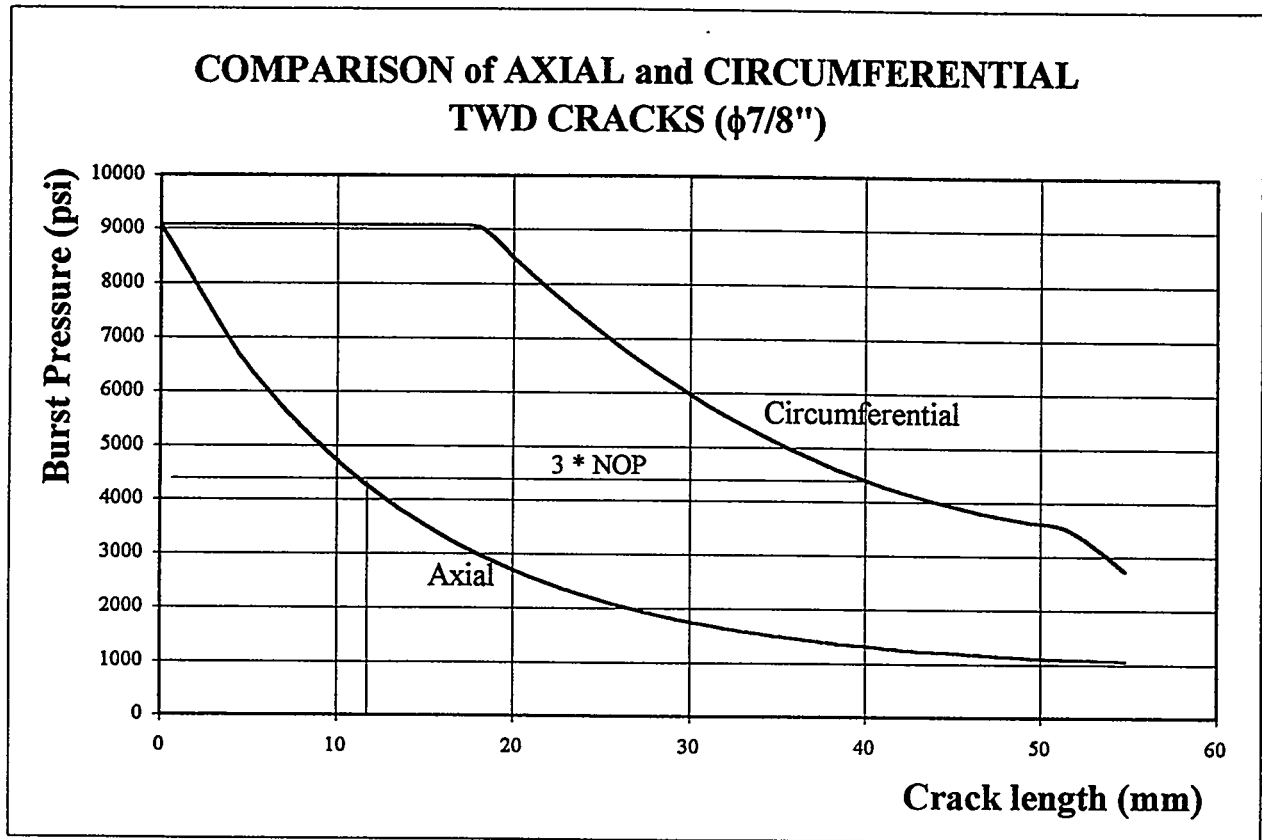
The REFERENCE CURVE concept



SUMMARY

- ◆ STRUCTURAL MODEL ALLOWS SIMPLE YET ACCURATE PREDICTION of BURST PRESSURE under :
 - laboratory testing } conditions
 - actual SG service }
- ◆ VALIDATION PROVIDED by LARGE INTEGRATED BURST DATA BASE with wide coverage of all relevant parameters
- ◆ SGDSM IMPLEMENTATION PROPOSED on BASIS of SIMPLE (not over) CONSERVATIVE APPROXIMATION
- ◆ REFERENCE CURVE → STRUCTURAL LIMIT
→ REPAIR LIMIT and PSGTR
- ◆ LEAK RATE MODEL AVAILABLE
(should be improved by further testing)
- ◆ COMPARISON WITH AXIAL FLAWS
 - much higher burst pressure
 - much lower leakage

CIRCUMFERENTIAL CRACKS have **HIGHER BURST PRESSURE**
and **LOWER LEAK RATE** than **AXIAL CRACKS**



CONCLUSION

**SUBJECT TO IMPROVEMENT
IN NDE CAPABILITIES
(DETECTION, SIZING, GROWTH),
SGDSM CAN BE USED TO ALLOW
CIRCUMFERENTIALLY DEGRADED
TUBES TO REMAIN IN SERVICE**

6 Working Session 1: Tubing Degradation

G. Kharshafdjian, AECL, and G. Turluer, IPSN

6.1 Participants

The Working Group (WG) was led by cofacilitators Gary Kharshafdjian (AECL, Canada) and Guy Turluer (IPSN, France). Participants in the session were:

F. Cattant (EDF, France)	M. Mirzai (Ont. Hydro, Canada)
P. Chow (AECL, Canada)	M. Mayfield (NRC, U.S.A.)
K. Colgan (Northeast Util., U.S.A.)	J. Muscara (NRC, U.S.A.)
J. Daret (CEA, France)	R. D. Nicholson (Nucl. Install. Inspect., UK)
D. Diercks (ANL, U.S.A.)	C. Rodríguez-Monroy (UNESA, Spain)
J. Esposito (Westinghouse, U.S.A.)	P. Rush (NRC, U.S.A.)
D. Gómez-Briceño (Ciemat, Spain)	P. Verón (ENSA, Spain)
K. Gott (SKI, Sweden)	C. Welty (EPRI, U.S.A.)
D. Hughes (TVA, U.S.A.)	

These individuals represented utilities, vendors, researchers, consultants and regulatory authorities. The major subjects discussed by the group included stress corrosion cracking, local deformations, and NDE past experience and degradation assessment.

Below is a brief summary of the discussions and conclusions.

6.2 Introduction to Session 1

- **Objective:** Provide R & D needs and regulatory proposals on steam generator tube (SGT) degradations.
- **Listing of potential SG tube degradation** offered by the facilitators; review of types, locations, and characteristics of tubing degradations in steam generators; synergistic effects of material variables; environmental factors; and stress and strain.
- **Overview by Peter Millet, EPRI** (Plenary) - chemical aspects and degradation from secondary side (corrosion, IGA, ODSCC) emphasized.
- **Report on the conclusions of Oct. 95 EPRI Expert Workshop on secondary side corrosion.**

A general introductory overview of the purpose of the group and the general subject area of SG tubing degradation was given by the facilitator. The purpose of the session was described as to "develop conclusions and proposals on regulatory and technical needs required to deal with the issues of SG tubing degradation."

Types, locations and characteristics of tubing degradation in steam generators were briefly reviewed. The well-known synergistic effects of materials,

environment, and stress and strain/strain rate, subsequently referred to by the acronym "MESS" by some of the group members, were noted. The element of time (i.e., evolution of these variables with time) was emphasized.

It was also suggested that the group might want to consider the related topics of inspection capabilities, operational variables, degradation remedies, and validity of test data, and some background information in these areas was provided.

The presentation given by Peter Millet during the Plenary Session was reviewed; Specifically, the chemical aspects and the degradation from the secondary side of the steam generator were noted.

The main issues discussed during the October 1995 EPRI meeting on secondary side corrosion were reported, and a listing of the potential SG tube degradations was provided and discussed.

The group agreed to focus on one or two of the most important degradation issues, since to discuss the entire "universe" of tube degradation processes would require much more time than was available.

6.3 Main Degradation Modes Considered by the WG (Common Concerns)

It was stated the SCC was the principal concern, at least in the U.S. and probably for most of the remaining international SG community. A discussion of SCC should include the effects of mild denting and other local deformation on initiating SCC, because this is becoming more commonly observed in the past few years. It was observed that the major factors influencing the SCC process were stress and strain rate.

Fatigue was described as a lesser, though real, concern that was for the most part manageable by the industry. The consensus of the group was that SCC was indeed the degradation mode of primary interest.

6.3.1 Stress Corrosion Cracking (SCC) of Mill Annealed (MA) Alloy 600

The focus has been placed on low temperature mill annealed Alloy 600 tubing, which is considered more sensitive to PWSCC.

SCC, the main concern of utilities today, was judged by the WG as the major cause for SG tubing degradation. Parameters affecting SCC initiation and growth, such as initiation time, tube locations susceptible to SCC, water chemistry, and temperature, were discussed.

With regard to impact of crack orientation on safety, it was agreed that circumferential cracks and axial cracks should be treated separately, although the degradation mechanism for both axial and circumferential cracks may be similar.

For PWSCC, it was considered that material properties stress and temperature were more important than the primary water operating chemistry variations. The effects of stress relief and heat treatment were noted.

6.3.2 Circumferential Cracking: Both ID- and OD- Initiated

On the issue of OD & ID SCC circumferential cracking, the following were discussed.

- **Safety issues:** Circumferential cracking is a major safety issue considered both by the utilities and the regulatory bodies.
- **Intrinsic difficulty of detection by NDE:**
 - distinction between OD and ID initiated circ-cracks cannot be unambiguously made in many instances.
 - probability of detection of ID circ-cracks is reported to be higher than that relative to OD circ-cracks of similar cross-section fraction.
 - from the examination of a significant number of pulled tubes from units affected by OD circ SCC, it has been concluded in France that SCC degradation up to 40% in cross section initiated at the OD can develop before being detected by RPC probes.
- **Urgent need for crack propagation rates**, which should take into account the various morphologies and distribution of individual cracks or attack i.e., single cracks, vs. multiple short cracks.
 - single-crack growth.
 - crack coalescence and subsequent growth transient.
 - IGA, and influence of IGA on discrete crack growth.
- **The increasing occurrence of tube ODS** and particularly with a **circumferential orientation** in the vicinity of the top of the tube sheet (TTS) is now a greater source of concern.
- Therefore the combinations of factors that determine the stress magnitude, orientation, distribution, particularly those factors enhancing circumferential cracking, and crack growth, should be more thoroughly investigated; in particular, the **relative contribution of the following factors should be assessed:**
 - features of the expansion process,
 - other additional deformations or stress generated i.e. by bending, slow denting, drilled hole profile.

- cold work, scratches etc.
- operational and thermal stresses.
- **Possible SCC enhancement by dynamic effects** such as denting even with slow or moderate denting growth rate.

6.3.3 Axial Stress Corrosion Cracking (SCC) and Intergranular Attack (IGA)

Axial PWSCC at dent locations and axial ODSCC at free spans have been reported. The following were indicated as needs:

- Flaw sizing in length.
- Detection and repair at TSP intersections.

6.3.4 Mechanisms, Environments, and Ways to Assess ODSCC

The importance of crevice chemistry was discussed. It was noted that the tube-support plate (TSP) crevices from pulled tubes obtained from Dampierre, Surry, Ringhals, and Point Beach, among others, had been extremely helpful sources of information. However, tube pulls are expensive, and this resource is limited by the availability of funding to pay for it. The desirability of obtaining pulled tubes and intact tube/TSP assemblies from retired steam generators was then discussed. It was noted that a number of retired units had already been "lost" insofar as obtaining tubes was concerned because the units were buried or otherwise disposed of. Even when the retired units are not buried, their accessibility is highly limited. Several persons noted the difficulties associated with obtaining samples from retired units, since the utilities replacing the steam generators typically entomb the old units in inaccessible concrete buildings and are generally not interested in subsequently accessing them. Also, the units may not be "laid up" in a condition that leaves them in a useful state.

It was suggested that the crucial information needed was the crevice chemistry, and that knowing this could enable meaningful autoclave experiments to be conducted with a bulk water chemistry corresponding to this crevice chemistry. It was also noted that obtaining crack initiation data in real time was difficult, but that meaningful crack growth rate measurements could be made because of the high sensitivity of the experimental techniques used to measure crack growth rates in the laboratory.

The WG agreed that the secondary water local chemistry in the crevice was of great interest but was very difficult or impossible to measure directly. The group agreed that the pH and chemistries of the crevice regions could be deduced by several means, including analyses of the deposits obtained from tube pulls, calculated values from codes such as MULTEQ (though some questioned the reliability of these calculations), hide-out returns, etc. In addition to the use of hideout returns and the analysis of crevice deposits from pulled tubes as ways of

deducing crevice chemistry, the attempts by Lumdsen and others to determine crevice water chemistries directly were also noted.

Discussion and suggestions about secondary side corrosion testing

The use of the following was suggested:

- Corrosion tests in autoclaves, usually for screening and comparative purposes in assumed concentrated corrosive media.
- Model boiler tests for corrosion and chemistry assessment in sludge and crevice areas under heat flux.
- Alternatively, sidestream experiments to replicate SCC in crevice configuration and/or for crevice chemistry condition monitoring.

However, autoclave tests usually devised to provide cracking in a reasonable time frame rely principally on doped corrosive water chemistries (strongly acidic or caustic) and thus might become irrelevant to many SG operating conditions.

Some model boiler tests might be more appropriate for closer replication of concentration and diffusion processes occurring in SGs, particularly if information from crevice side stream monitors in actual plants becomes available.

As already noted at the Airlie EPRI conference, all approaches, if combined, might contribute to better assess the tricky phenomena accounting for ODS CC and OD local crevice chemistries.

However, the group emphasized the need to concentrate more on model boiler tests and/or on sidestream corrosion/chemistry monitoring in plants.

On the **effect of lead and sulfate** it was reported that; the susceptibility presented by 1600 MA and 1600 TT in AVT + lead oxide is higher than in caustic + lead oxide environments. Boric acid seems to mitigate the effect of lead (but the evidence for this is weak), but doesn't prevent the initiation of cracks, and in acidic sulfates, Incoloy 800 is more susceptible than Inconel 690 TT.

The **stress redistribution and relaxation** during normal steam generator operation was discussed. It was noted that residual stresses would basically relax to the yield stress of the tube material at the operating temperature, and this yield stress was not substantially lower at the operating temperature; other relaxation processes (i.e., inelasticity) were negligible.

While stress and water chemistry were highly significant variables in influencing SCC behavior, **material variables were also significant**. The commonly observed phenomenon was noted of two plants having nominally identical steam generator and operated under similar water chemistry conditions exhibiting wide differences in cracking and denting behaviors. This has also been observed in Saint-Laurent B1 in France, where only one steam generator of three

had tube batches more sensitive to ODSCC at the TSP intersections. The significance was noted of sufficiently characterizing heat-to-heat variations and microstructural features to enable one to accurately predict the susceptibility of a given heat of material to ODSCC, features which might well differ from those known to enhance PWSCC.

It was agreed that while some of the more significant microstructural variables were understood (e.g., carbide distribution), one could still not reliably predict the susceptibility of a given heat of material from "first principles".

6.3.5 Conclusions on Understanding of Mechanisms and State of Knowledge on Tube Degradation by SCC

SCC mechanisms are not yet fully understood or identified (both PWSCC and ODSCC).

Primary water chemistry variations, usually limited, known, and controlled, are a second-order parameter of PWSCC, which is mainly governed by high stress, material, and temperature dependences

Secondary side corrosive environments, which are one of the main controlling factors of ODSCC, can vary widely in service along with time, locations, SG design, chemical control programs, sludge, and impurity ingress, etc.

In addition, complete inventories of all species (including noxious) and the relative concentration factors in occluded heated areas such as crevices or under sludge or tube deposits are still poorly identified, despite current progress.

Therefore, only fragile deductions can so far be derived from hide-out returns, crevice or tube deposit analyses on pulled tubes, and tentative thermohydraulic calculations.

Consequently, reliable crevice chemistry determination was judged by the WG to constitute an ongoing challenge that adds to the intrinsic burden of ODSCC initiation and propagation prediction.

Because actual reliable ODSCC crack growth data are few or missing, the suggestion was made to use past field experience through NDE instead, a view that was not fully shared by all. Actually, unless substantial progress in sensitivity and reliability is made, those NDE methods might remain blind for ODSCC flaws shallower than mid-thickness.

6.3.6 NDE Past Experience and Degradation Kinetics Assessment

As a follow-up of the above discussion, it was also acknowledged in the group that replication of actual tube degradation by SCC observed on operating SGs can be achieved only with difficulty and with some limitations on laboratory setups or even on loops incorporating model boilers, the main reasons being:

- Time constraints for SCC systems with long incubation periods.
- Need to deliberately resort to noxious impurity injections, sometimes at high concentration levels, which might lead to poorly representative conditions.
- Difficulty in replicating the realistic combination of heat flux + thermohydraulic + fouling conditions on the same setup.

Therefore during this discussion, the WG judged that past field degradation experience assessed by NDE techniques should be taken into account but cannot be solely relied upon to determine and predict SCC crack growth rate (CGR) as illustrated by the following considerations and examples:

- Industrial NDE performed so far on a standard outage schedule and sampling ratio is not usually best fit and sensitive enough to size all types of cracks or flaws for GCR determination,
- Circumferential through wall PWSCC at TTS, which occurred for example during the first fuel cycle at Nogent 1 in France, could not be anticipated from past experience.
- Occurrence of a through-wall, 360° ODSCC circumferential crack, which affected 74% of a tube cross section in Bugey 4, went undetected at the previous outage by RPC inspection even with the benefit of hindsight.
- Sudden evolution of a large number of ODSCC cracks at the TSP intersection at Saint Laurent B1 with a significantly large bobbin coil voltage increase could not be anticipated from past experience in France.
- In the case of ODSCC, some strong crack growth rate (GCR) dependence on local chemistry should be anticipated, and therefore large variations in CGR cannot be excluded because chemical factors can vary in nature and concentration with time or operational hazards.

6.3.7 Identified Needs

SCC crack propagation data on tube with heat flux conditions, such as model boilers rather than under isothermal conditions.

Some suggestions to leave cracks in the SG to monitor their evolution were challenged, in particular for circumferential cracks.

Bulk and crevice chemistry, corrosivity monitoring, monitoring tools in field and in model boilers, developing instrumentation to monitor bulk and crevice chemical conditions, ODSCC crack initiation and propagation (crack coalescence of small cracks, which is believed to contribute most to macroscopic flaw growth, should be taken into account).

More pulled tubes and use of retired SGs (SG sample extraction, also to remove tubes and their environment, i.e., sludge, deposits, full crevice such as TSP section + tube). Past instances of TSP crevices from pulled tubes obtained from Dampierre, Surry, Ringhals, and Point Beach had been extremely helpful sources of information.

Assess with more scrutiny the material variability effect, especially batch-to-batch variations.

Better understanding of local stress/strain distribution is required.

Substantial improvement of NDE detection and sizing capabilities in particular for OD degradations and combination of flaws.

It was suggested that better NDE techniques were needed to detect crack initiation, both in the field and in laboratory tests. NDE is potentially a very useful tool in monitoring the progress of laboratory tests. Acoustic emission was suggested as a possible tool for detecting crack initiation. It was noted that NRC had shown success in using this tool for detecting crack initiation in coolant piping. Some group members expressed skepticism about its use in a SG environment.

6.4 Effect of Local Deformation on Initiating SCC

Local deformations such as denting are stress raisers that enhance PWSCC and ODSCC. Even moderate and slowly growing dents might entail preferential circumferential SCC; therefore dents should be surveyed with more scrutiny.

Immediate plugging of tubes severely dented in the past (up to 160 mils at Connecticut Yankee) probably has resulted in rather modest PWSCC occurrence, while less-conspicuous slow denting might have contributed to significant PWSCC and/or OD circumferential cracks with dents as small as a few mils, as observed at Sequoyah 1 and suspected at other units.

It was suggested that the amount of strain at the dent was not as important as the strain rate in determining subsequent PWSCC. It was also observed that tube inspections could not generally obtain a good correlation between strain and voltage in eddy current profilometry.

Several persons made the observation that denting was typically arrested after the first (or, at most, after the first few) fuel cycles, and that PWSCC did not generally appear until significantly later. It was argued that the critical issue was therefore the time required for the stress field associated with denting to result in crack initiation. Others countered that tubes are not normally plugged upon detection of an initiated crack, and therefore crack growth rates were also important.

Stress, strain, and strain rate need better assessment; there is a need to develop models to correlate the local deformations (e.g., profilometry measurements) to the stress/strain fields.

Effect of tube-support interaction such as tube locking at TSP should be assessed more thoroughly, in view of:

- Tube bulging or bending after local heat treatment on sleeved tube: avoid circumferential SCC.
- Potential stress raisers at support structures such as TSP, and scallop bars, causing circumferential ODSCC.
- Effect of ODSCC circumferential flaws on vibration-induced fatigue crack propagation

6.5 Potential Tube Degradation by Vibration-Induced Fatigue (Concerns that May Not be Primary to All)

The consensus of the WG was that fatigue is a safety issue, but it can be addressed by various actions or dispositions including the following:

- **Comprehensive assessment of adequate fitness of the tube support structures to prevent potential tube vibration;** this is illustrated by:
 - verification of adequate AVB positioning by NDE as published by Japan and France.
 - current investigations in France to assess the TSP soundness in some SG models, as a consequence of various defects detected.
- a) missing TSP support, locally at Fessenheim 2 following a chemical cleaning applied locally outside qualification bounds.
- b) TSP ligament degradation in France and in the USA (Sequoyah and Indian Point 2), some of which evolved in service (as observed in 1995, at Gravelines 2).
- **Preventive or corrective measures;** as an alternative to repairing the inadequate supports, effects of potential vibration-induced fatigue can be prevented by tube plugging and installation of a dampening cable as currently practiced in France.
- **Refined vibration analysis.**
- **Strengthening primary-to-secondary leak rates requirements.**
- **Implementation of more sensitive and responsive improved leak monitoring (N-16)**

North Anna: the fatigue cracking at North Anna 1 that lead to a tube rupture in 1989 was discussed extensively. It was noted that in this incident, like almost all fatigue incidents, fatigue crack growth following initiation was rapid, and periodic inspections at refueling outages would have little hope of seeing such a crack before it went through-wall. It was then argued that had present operating procedures been used in the North Anna incident, the plant would have been shut down during the leakage stage before the crack had reached a critical size, and rupture would never have occurred. **Nevertheless, the root-cause determination is still being investigated.**

Preventing potential tube-vibration-induced fatigue may be a source of potential concern, as currently illustrated by recent implementation of repair techniques and plugging on tubes for which the quality of the supports could be questioned (e.g., displacement of AVB from nominal design position and TSP ligament cracking).

6.6 Potential Tube Degradation as a Result of Remedial Actions

The following examples were reported on the issue of adverse effects of remedial actions.

- **Circumferential PWSCC on sleeved tubes**, without an adequate stress relief heat treatment.
 - hazards linked to tube bending or bulging as a result of stress relief heat treatment for those tubes locked in first TSP.
- **Alloy 600 plugs experiencing cracking**; this might result in:
 - plugged tube bulging and bursting.
 - potential missile effect on degraded tube U-bend.
- **TSP degradation** triggered by chemical cleaning applied outside qualification bounds, which by inadequate corrosion inhibition would entail tube support deficiency, as discovered at Fessenheim.

6.7 Miscellaneous Discussions

AT SG operating temperatures as well as at 400° C, there is a substantial **diffusion of hydrogen from the primary side** through the 600 MA tube wall to the secondary side crevice, particularly at hot spots. This hydrogen may influence corrosion potential and induce IGSCC at the OD of the tube wall when high tensile stresses and steam are present: perhaps half the hydrogen on the secondary side diffuses through the tube from the primary side.

Pb contamination is clearly detrimental; the mechanisms by which Pb enhances SCC are not well understood, and more data are needed on the effects of specific variables. Tests were conducted on SG tube materials in a 1 ppm sulfate

sludge containing 0.5% Pb and 1% Pb. Alloy 600 TT was less susceptible than Alloy 600 MA, and Alloy 690 TT showed no cracking.

6.8 Topics not Covered in Session 1

Due to time limitations the following topics, although proposed, were not fully discussed in Session 1:

- Possible limitations and return of experience for Alloy 690 and Alloy 800.
- Other causes of plugging of tubes (mainly in Alloy 600 amounting to 30% plugging ratio).
- Other types of degradation (pitting, wear, impact of loose parts, etc.).

The first of these is the fact that the system is not a simple one. It is a complex system, and the behavior of the system is not predictable. The second is that the system is not a simple one. It is a complex system, and the behavior of the system is not predictable. The third is that the system is not a simple one. It is a complex system, and the behavior of the system is not predictable. The fourth is that the system is not a simple one. It is a complex system, and the behavior of the system is not predictable. The fifth is that the system is not a simple one. It is a complex system, and the behavior of the system is not predictable. The sixth is that the system is not a simple one. It is a complex system, and the behavior of the system is not predictable. The seventh is that the system is not a simple one. It is a complex system, and the behavior of the system is not predictable. The eighth is that the system is not a simple one. It is a complex system, and the behavior of the system is not predictable. The ninth is that the system is not a simple one. It is a complex system, and the behavior of the system is not predictable. The tenth is that the system is not a simple one. It is a complex system, and the behavior of the system is not predictable.

**SUSCEPTIBILITY OF STEAM GENERATOR
TUBES IN SECONDARY CONDITIONS**

EFFECTS OF LEAD AND SULPHATE

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**CNRA/CSNI Workshop on
Steam Generator Tube Integrity
in Nuclear Power Plants**

**30 Oct / 2 Nov, 1995
Argonne National Laboratory**

IGA/SCC on the secondary side of steam generators is increasing every year, being the cause of some of the steam generator replacements.

Until recently, caustic and acidic environments have been accepted as causes of IGA/SCC:

- . In Sludge Pile on the Tube Sheet
- . At Support Crevices
- . In Free Span

Lead and sulphur have been identified as significant impurities.

Now it is considered that some IGA/SCC at support crevices may have occurred in nearly neutral or mildly alkaline environments.

In order to understand the influence of lead and sulphur in the behaviour of steam generator tube alloys, an experimental program has been carried out with:

- . C-ring samples in static autoclaves
- . Model Boilers

TEST CONDITIONS. C-RINGS TESTS**PbO Concentration**

	no PbO	%			ppm			
		2	0,2	0,04	100	80	50	10
10% NaOH	(1)	(1)	(1)					
4% NaOH	(2)		(2)	(1) (2)		(2)		
AVT			(2)	(2)	(2)		(2)	(2)

(1) 350°C
(2) 320°C

Up to 2000 hours have been accumulated in most tests, with intervals every 500 hours to inspect the specimens and renew the solutions.

Visual inspection and metallographic examination have been performed.

Deposit analyses have been carried out by EDX, AUGER and ESCA

In AVT + 10 ppm PbO, Inconel 600 TT shows no cracks after 2000 hours while Inconel 600 MA cracks after 500 hours.

In caustic + PbO, the observed morphology for Inconel 600 MA and Inconel 600 TT has been IGA.

In AVT + PbO, the observed morphology has been IGSCC for Inconel 600 MA and TG + IGSCC for Inconel 600 TT.

INCONEL 690 TT

After 2000 hours of testing, cracks have not been detected neither in caustic nor in AVT with lead oxide.

INCOLOY 800

For the Incoloy 800 heats (CW, SP), the caustic solution with the minor concentration of PbO has been the most aggressive (80 ppm PbO). No cracks have been found in the more concentrated solutions.

No cracks have been detected in AVT with lead oxide.

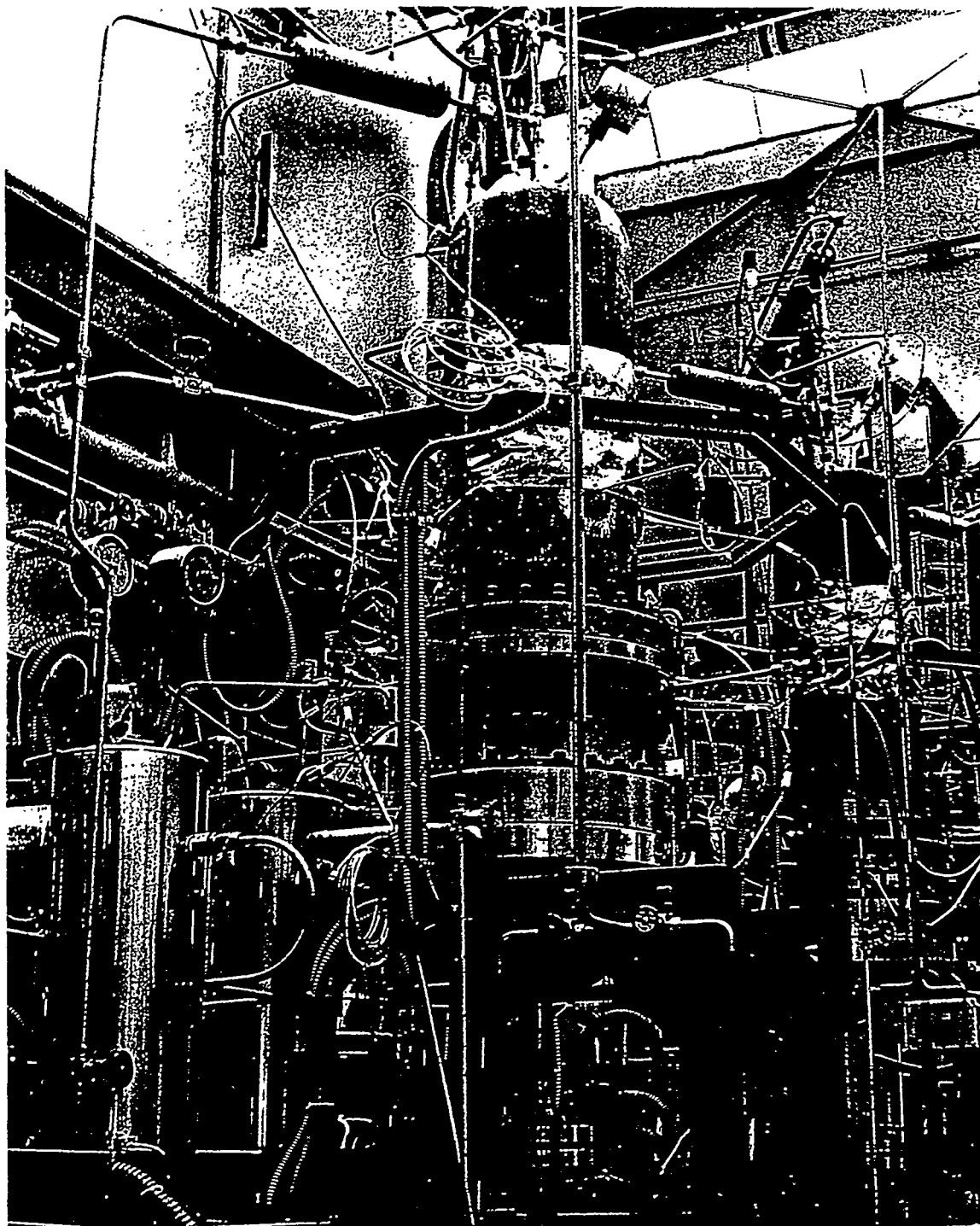
MODEL BOILER TEST**SCOPE**

- **Effect of lead oxide on Inconel 600 MA SCC susceptibility.**
- **Mitigation effect of boric acid.**
- **Influence of sulphate in the behaviour of steam generator tube materials.**

CIEMAT



MODEL BOILER FACILITIES



MODEL BOILER

EFFECT OF LEAD OXIDE MODEL BOILER TESTS

- Two model boilers (called MB2 and MB3) with seven tubes of Inconel 600 MA each have been used.
 - The tubes had accumulated 6000-8000 hours in caustic environments before the test with lead oxide.
 - Operating conditions are similar for both model boilers.
 - Chemical conditions:
 - MB2 - AVT+6 ppm CO_3^{2-} +2 ppm NH_3 .
 - MB3 - AVT+6 ppm CO_3^{2-} +2 ppm NH_3 +boric acid.
 - PbO was incorporated to the simulated sludge placed in some of the support plate crevices (Pb^{2+} in solution: 1-2 ppm).
-

E.C. INSPECTION RESULTS

After 2.300 hours of tests, E.C. inspection was performed with the following results:

- No cracks were found in the support crevices in which lead oxide was placed.
- Most cracks were found in the tubes that had not lead oxide in the crevices.

DESTRUCTIVE EXAMINATION

- Stress corrosion cracking and intergranular attack have been observed.
- The morphology of the cracks has been always intergranular.
- The measured depth of the cracks has been larger than the one detected by E.C.

DEPOSIT ANALYSES

EDX and ESCA techniques have been used

- **At the fracture surface**

Lead has been found in some of the cracks (~ 3% at).

The rate Ni/Cr is similar to the one in the base material

- **On the OD surface**

A small amount of lead has been detected in the internal layer of deposits (~ 1% at).

Boron concentration is higher in the free span (11% at) than at the support plate intersection (5% at).

**EFFECT OF "NEUTRAL" SULPHATE
MODEL BOILER TESTS**

- Inconel 600 TT and Inconel 690 TT have been tested.
 - Quatrefoil support plates were used in these tests.
 - Model boiler feed water:

AVT + 2 ppm SO_4Ca + 2 ppm $\text{Ca}(\text{OH})_2$
 - A concentration factor x2.5 for sulphate and x9 for calcium have been observed in the model boiler water samples.
-

"NEUTRAL" SULPHATE TESTS

RESULTS

- After 9000 hours of operation, E.C. inspection showed no cracks or any other type of defects in any of the material tested.
- Destructive examination confirmed the E.C. results.

RESULTS

- After 9000 hours of operation, E.C. inspection showed one indication in the roll transition zone in one of the tubes of Incoloy 800 SP.
 - The destructive examination:
 - * Confirmed the E.C. indication, that was identified as wastage (max depth 40°C).
 - * Found other types of degradation:
 - Multiples circumferential cracks in the roll transition zone of two other tubes of Incoloy 800 SP (max. dept 23%).
-

- Wastage (max depth 8%) and intergranular attack (max depth 5%) at the support crevice in the Incoloy 800 SP tubes. Minor defects in free span.
- Wastage in the roll transition zone (max depth 12%) and at the support crevices (max depth 4%) in the Inconel 690 TT tubes.

DEPOSIT ANALYSES RESULTS

- The ratio Ni/Cr is lower in OD surface deposits than in the base material.
 - Ni depletion in the cracked tube is higher than in the tube with wastage.
 - S is incorporated in the entire oxide layer at O.D. surface.
 - At the fracture surface has been detected:
 - . A high S concentration that decreases towards the crack tip.
 - . Ni enrichment that disappears towards the crack tip
 - . Cr enrichment that increases from the mouth to the tip of the cracks.
 - . Sulphur has been identified as sulphate, sulphide and intermediate species.
-

EFFECT OF SULPHATE

CONCLUSIONS

- Significant amount of sulphates can be tolerated in the secondary water whenever the pH is maintained "neutral".
- In acidic sulphate, Incoloy 800 SP is more susceptible than Inconel 690 TT.
- The presence of cracks in the roll transition zone of the Incoloy 800 SP tubes places doubt on the validity of shot peening to increase the SCC resistance in the roll transition zone where the material is plastically deformed.
- Reduced sulphur species can be formed after sulphate inleaks into the steam generator if acidic pH is established.

**CNRA/CSNI Workshop on Steam Generator Tube
Integrity**

“Experiences on IGSCC crack manufacturing”

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ABSTRACT

We present our experience in manufacturing IGSCC realistic defects, mainly in INCONEL 600 MA Steam Generator Tubes.

From that experience we extract some knowledge about this cracking (influence of chemistry in the environment, stress state, crack growth rate, and occurrence in laboratory condition of break before leak).

1.- INTRODUCTION

The reliability of damaged components requires frequent and extensive NDE examinations, structural assessments and remedy evaluations.

For that purpose samples with realistic defects (reproducing actual cracks) serve as:

- Mock-ups for E.N.D.
- Test samples to determine the existing margin to failure.
- Test samples to evaluate the capability of proposed remedies to cope with the problem.

Within the Spanish PISGV programme, EQUIPOS NUCLEARES, S.A. began to develop with full success his own procedures to manufacture these realistic defects that reproduce the cracks found in Spanish S.G., not only in its location and macrofeatures, but also in the microscopic scale and its corresponding NDE signal.

Afterwards, when the RPV head penetration cracking phenomenon was acknowledged we took advantage on the experience gained with the thin S.G. tubes to generate IGSCC defects in the CRDM penetrations.

Nowadays we are working in the field of very shallow cracks that help to improve early detection of IGSCC occurrence.

2- MANUFACTURED DEFECTS IN STEAM GENERATOR 600 MA TUBES

a) **Primary Side:**

- Longitudinal cracks at transition expansion
- Circumferential cracks at transition expansion
- U shaped crack at transition expansion
- At dented areas ($\Delta\phi=0,4$ mm)
- Longitudinal crack at free span
- Circumferential crack at free span
- At U bend

b) **Secondary Side:**

- Circumferential cracks at transition expansion
- Longitudinal crack at tube support plate
- At dented areas ($\Delta\phi=0,4$ mm)
- Longitudinal crack at free span
- Circumferential crack at free span

The environment was in some cases 10% caustic soda deaerated water solution, and, in others, doped steam plus hydrogen at 400°C.

To accelerate the cracking process, the maximum permissible mechanical loads compatible with a minimum plastic strain were used. The cracking seems to be perpendicular to the maximum stress applied. Crack growth rate is 9×10^{-9} m/s at 425°C (determined on thick pieces of inconel, namely RPV Head Penetrations).

3.-OUTSTANDING OBSERVATIONS FROM OUR IGSCC WORK

From our work in generating cracks, we highlight the following:

- IGSCC of S.G. alloy 600 MA tube and RPB head penetration seem to have the same nature. Both metals are susceptible to the same environment and the crack growth rates found are within the values given by bibliography.
- The hydrogen concentration in steam is not a critical parameter within molar fractions between 0,005 and 0,02.
- According to our experience with S.G. tubes, it seems that at 400°C hydrogen is able to diffuse through the allow 600 MA tube and induce IGSCC on the other side of wall, if high tensile stresses and steam are present there.

In that case, in the actual S.G., we might expect a hydrogen diffusion from primary to secondary side specially at hot spots, that would enrich in hydrogen the rather isolated environment of the crevices and would give cause to secondary IGSCC. This hypothesis may partly explain the good service record of alloy 800 (immune to hydrogen) compared with alloy 600.

- Under laboratory condition a few inconel 600 MA tubes under primary IGSSC bursted instead of leaking. The inner compressible pressurising medium (steam at 400°C) and the rather high pressure used justify these facts.

US PWR Steam Generator Management, an Overview

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Presented at CNRA/CSNI Workshop on
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in Nuclear Power Plants

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ABSTRACT

This paper provides an overview on the status of steam generator management activities in US PWRs, and includes: 1) an overview of the impact of steam generator problems; 2) a brief discussion of historical damage trends and the current damage mechanism of most concern; 3) a discussion of the elements of "steam generator management"; and 4) a description of the approach being followed to implement a degradation-specific protocol for tubing inspection and repair.

This paper was prepared in conjunction with another paper presented during the Plenary Session of this Conference, "Steam Generator Degradation: Current Mitigation Strategies for Controlling Corrosion" (Ref. 1), and is provided as a supplement to that material.

IMPACT

Historically, steam generators have been the component with the single highest impact on lost capacity in US PWRs. This result from several causes: 1) forced and extended outages due to tube leak and/or rupture events; 2) identification of new and wide-spread damage where there is ambiguity in the inspection results and the approaches to damage repair (i.e., tube plugging and/or sleeving, revised plugging criteria, etc.); and 3) extended outages to effect major repairs and/or steam generator replacement. For the years 1980-1984 the average annual capacity factor loss in US

PWRs due to steam generator problems was 5.7% (Ref. 2).

In addition to the impact on plant performance reflected by the capacity loss figures, economic considerations have resulted in major sleeving repair campaigns at three PWRs and steam generator replacement at thirteen plants with currently scheduled replacement at eight more plants prior to the year 2000 (Ref. 2).

Three major goals of the industry-wide steam generator effort are to reduce the capacity factor loss figure to less than 2.5% annually, to achieve replacement outage duration's of less than 100 days, and to provide replacement designs and operational guidance to assure that no more than one replacement is required during the life of the plant. The first two goals are currently being met, and, based on experience to date with replacement steam generators, there is reason to believe that the latter goal will be met.

DAMAGE MECHANISMS

As described in detail in Reference 3, the general chronology of steam generator tubing corrosion damage for plants with alloy 600 tubing has been general wastage and thinning, followed by denting due to carbon steel tube support plate and tube sheet corrosion, followed by pitting-corrosion in tubesheet sludge pile regions, followed by intergranular attack in tubesheet crevices and deep sludge pile regions, followed by both ID and OD initiated stress corrosion cracking (SCC). The time sequence is generally explained by the changes in water chemistry that corrected one form of damage, allowing for the emergence of the next form. The latter of the damage forms, stress corrosion cracking, is currently the dominant degradation mechanism, and the one that is driving programs addressing steam generator management. References 1 and 3 describe the known causes and remedial measures for each of the damage mechanisms.

Circumferential Cracking

At this writing circumferential cracking in plants with recirculating steam generators is receiving a great deal of attention. Though not a new phenomenon, initial occurrences were reported in the mid to late 1980s, as generators have aged and inspection techniques improved, both the number of plants reporting such damage, and the extent of damage within certain plants has increased. In general the cause of this damage is known. Specifically, experience and testing show that mill annealed alloy 600 used for steam generator tubing is susceptible to SCC in steam generator environments (ID or OD initiated) when the combination of material susceptibility, tensile stress and environment (including temperature) exceed a threshold exposure time (Ref. 3). ODSCC can occur at much lower stresses than IDSCC, probably due to the more aggressive chemical and electro-chemical environments that can exist on the secondary side of steam generators. For circumferential cracking to occur (rather than axial), a situation where axial stresses dominate over hoop stresses is required. Field experience indicates that this can be the case for certain roll transition expansions, for some cases where denting at tube-tube-support-plate intersections exists, where tube-sleeve joints create a particularly susceptible condition, and on certain short-radius U-bend tubes. Based on recent tube-pull data, the cracking is generally a matrix of short, non-co-planar cracks of varying depths, that exhibits substantial structural capability when analyzed or tested to conditions postulated to exist under accident loads - this is probably due to the significant portion of the cracked region that has in-tact ligaments. Current work is underway to more accurately define the specific conditions that can lead to this phenomenon, as well as to understand how the morphology grows over time and the impact of that growth on structural capability.

The phenomenon is of particular concern for at least two reasons. First, it presents a

unique set of challenges to NDE processes and systems. These challenges are being addressed, and while detection is not considered to be a major issue, standardized capabilities for sizing, are not currently in field use. This limits the available data on growth rates as well the ability to use a size-based plugging/repair limit, which results in a need to "plug-on-detection" without regard to the potential structural consequences, or lack thereof, when circumferential cracks are detected. A second concern is the observed location of the cracks in unconstrained regions of the tube such as in roll transitions at the top-of-the-tubesheet (TTS) and at the top and bottom edges of dented tube support plate intersections. This contrasts with axial SCC which has generally been found in regions where there is at least partial restraint provided by the tubesheet or tube support plate.

Remedial measures for ODSCC (axial or circumferential) include improved water chemistry control to reduce sludge transport and accumulation, bulk water molar-ratio-control (MRC) to control crevice chemistry pH, the addition of bulk water boric acid buffer-treatment, and the application of advanced sludge removal techniques (e.g., chemical cleaning, improved water lancing, etc.). In-situ stress relief and/or peening and Thot reduction have been used to control certain occurrences of IDSCC.

STEAM GENERATOR MANAGEMENT

Steam generator management, the goal of which is to optimize steam generator operation relative to safety, reliability and cost-effective maintenance, is accomplished through three elements, "Mechanism Management", "Defect Management" and "Life Extension/Replacement Management" defined as follows:

Mechanism Management - determines the causes of and remedies for steam generator tubing degradation; provides input to the "defect management" element on damage morphology, growth rates and

susceptible regions of the steam generator; provides input to the "life extension/replacement management" element on improved designs and materials.

Defect Management - removes from service damaged tubes that could leak or rupture during the next operating cycle. Optimum "defect management" integrates periodic ISI with tube plugging criteria to assure that all tubes that could rupture or leak unacceptably are removed from service, while all tubes that do not pose a rupture or leakage concern remain in service.

Life Extension/Replacement Management - focuses on long range planning and the decision processes involved in steam generator replacement; provides improved designs, materials and operating recommendations.

The tie among the three elements is: (1) to use the product(s) of mechanism management to prevent tube damage; (2) to apply defect management to prevent tube failure under conditions when the mechanism management products are not available or effective; and (3) to resort to life-extension/replacement management when the other two elements are no longer cost-effective.

Steam Generator Degradation Specific Management (SGDSM)

(The following material is extracted from and essentially quoted, with minor updates, from Reference 4. It is provided here as a complete description of the SGDSM protocol.)

The tube plugging limits in most plant Technical Specifications were developed early in the commercial nuclear power program. They were developed around the ASME Code requirements for new tubing, and generally consider the reduction in structural capability from large (or long) uniform wall-reduction-type flaw. They

include conservative safety factors on tube loading, have an allowance for NDE measurement uncertainty, and, based on experience, they are generally conservative.

The existence of multiple damage forms which vary in both length and volume of material impacted, leads to the fact that different defect types will have differing effects on structural integrity. Using a single plugging limit based on uniform wall loss may not be appropriate for all defect types. This, coupled with improvements in ISI, provide the opportunity to reduce some of the over-conservatism that may be inherent in existing plugging limits.

Steam generator degradation specific management (SGDSM) is an approach to steam generator management that is receiving increasing attention in the PWR community. Essentially, SGDSM optimizes the integration of ISI and plugging criteria on a mechanism or defect specific basis (e.g., ID stress corrosion cracking in the tubesheet expansion zone, OD pitting in the tubesheet sludge pile, OD stress corrosion cracking in the tube support plates, etc.) to limit unnecessary tube plugging and sleeving while still assuring steaming generator safety and reliability. In addition to the potential for premature plant derating or steam generator replacement, unnecessary plugging and sleeving increases personnel radiation exposure, direct maintenance costs, and the plant downtime required for steam generator maintenance.

The rationale for SGDSM is that different defects/damage forms impact tubing structural integrity in different ways at different rates, and therefore, have different limits at which they could lead to unacceptable failure. SGDSM differs from current U.S. practice, in which tube defect management is generally nonspecific and can be characterized as follows:

1. assumes all forms of damage are of equal risk;

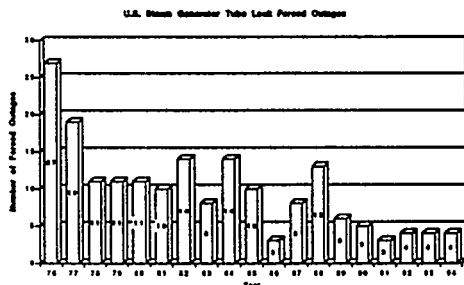
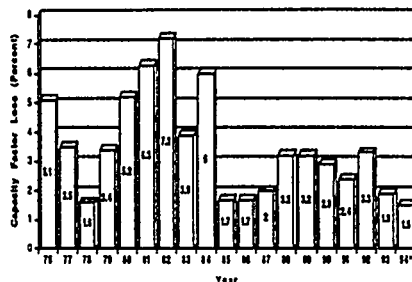
US PWR Steam Generator Management, an Overview

C. S. Welty, Jr.
Manager, Steam Generator Program
Electric Power Research Institute

October 31, 1995

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U.S. Capacity Factor Loss Due to Steam Generator Problems



STATUS

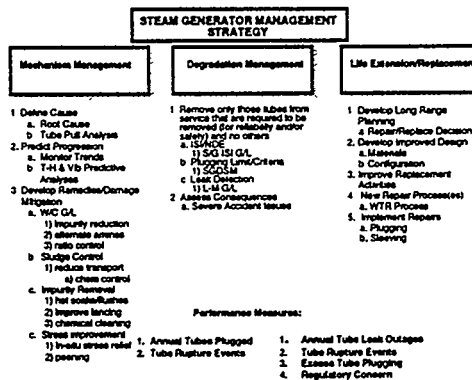
- Some Damage Eliminated
 - Denting, pitting, wastage, sludge-pile IGA
 - chemistry control measures
 - U-bend primary cracking, AVB wear
 - in-situ repairs
- Some Damage Managed
 - Low and steady rate of tube plugging, causes known
 - expansion zone primary cracking, tube sheet IGA
- Secondary Side Stress Corrosion Cracking
 - Not currently under control
 - Molar ratio chemistry control and inhibitor development work show promise

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STATUS (cont'd)

- Replacement Units
 - Minimal damage in units replaced to date (>12years)
 - Much less susceptible
 - improved tubing materials, support plate materials and design, thermal-hydraulic analysis and design
 - Improved SG management practiced by PWR operators
 - Improved chemistry control

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2. considers any tube with an ECT-measured defect depth greater than a certain through-wall limit (usually 40%) as a tube that could rupture regardless of defect length, volume of material loss or cause;
3. accepts that some defects may penetrate the tube wall and leak during an operating cycle, and allows for this condition by establishing a shutdown-leak-rate-limit based on a leak-before-break analysis of a single axial or circumferential through-wall crack;
4. assumes that the combined application of remedial measures, the inherent leak-before-break nature of Inconel for most defects and the mandated inspection provide assurance that the consequences of tube damage are acceptable.

Removing tubes from service with a depth-based criteria is generally conservative, but while this approach has proved acceptable, it may lead to unnecessary tube plugging (particularly in the case of small volume defects such as isolated pits and short stress corrosion cracks).

SGDSM is predicated on being able to reliably discriminate among defect types, which is accomplished by a combination of understanding the cause of damage, knowing its likely location and the existence of suitable NDE capabilities. Once discrimination is assured, justification for the appropriate plugging criteria must satisfy the following:

1. avoid conditions that lead to excessive leakage resulting in forced shutdown during normal operation;
2. avoid rupture during normal operation;
3. avoid rupture during postulated accidents (e.g., steam line break);

4. avoid excessive leakage during postulated accidents.

In applying SGDSM, the defense-in-depth concept must be addressed/maintained, and there is an implicit commitment to the following:

1. applying known remedies to limit the number of expected defects;
2. evaluating the primary-to-secondary leak rate limit with respect to leak-before-break (LBB) capability;
3. performing appropriate NDE (scope and technique/procedure) of the region of expected damage;
4. plugging tubes with defects greater than a conservatively established limit which includes:

- use of a rupture* correlation which bounds tube burst-test-to-defect-geometry data
- application of appropriate safety factors and material properties
- allowance for growth between inspections
- allowance for NDE measurement uncertainty

5. calculation of the potential leak rate during postulated accidents from defective tubes that remain in service, and insuring that the

* Tube rupture is the condition of a tube segment with a flaw at the maximum pressure it can support under actual steam generator conditions. Generally, rupture has occurred by significant tube deformation and/or flaw extension.

TTS Circ Cracking

- Management of Phenomenon is "Highest" Priority Activity
 - Occurred at several plants - identified in mid-1980s
 - Europe and US
 - Need to establish technical viability to *NOT* "plug-on-detection"
 - experience shows not a safety issue
 - structural limits not exceeded
 - RC analysis and/or in-situ pressure tests
- Understand Fundamental Cause
 - Alloy 600 is susceptible to SCC in SG environments
 - orientation depends on direction of principal stresses
 - need to complete analysis of recent events
 - extent at MY
 - differences between EdF and Belgian experience

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TTS Circ Cracking

- Westinghouse SG's (US)
 - 51 operating plants
 - 19 have reported cracking
 - 5 have pulled tubes
 - 6 categories
 - MA A600
 - PD HR - 3 OF 10, PD HR - 7 of 10, Western - 8 of 2, PD Hyd - 1 of 2
 - TT A600 & A690
 - PD Hyd - 6 of 20
- CE SGs (US)
 - 15 operating plants (plus 2 replacements)
 - 12 have reported cracking (include the 2 replaced)
 - 4 plants have pulled tubes (19 tubes)
 - 9 reported leakier outgas in 4 plants (7 in the original Palisades and Millstone, 2 SCC)

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TTS Circ Cracking

- Plants w/Extensive Involvement
 - Doel 4 (Model E), MY (CE)
 - Tihange 3 (Model E) to a lesser extent
 - Doel 4 and Tihange 3 have A600 clad on secondary face of TS
- EdF Experience
 - Extensive MRPC NDE since late 80s
 - sludge pile region of 900 and 1300 MW units
 - 571 "indications", all plugged, 84 tube pulls
- Experiencing an "Inspection Transient"
 - To be expected
 - improvements in NDE
 - circumferential flaws are more of a challenge to NDE
 - operating with a material susceptible to SCC
 - Results in large "Weibull" slopes

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TTS Circ Cracking

- Observations
 - Explosive expansions may create susceptible stress condition (more than other expansions)
 - Structural limits have not been exceeded
 - Units w/preheaters may be more susceptible
 - mechanical and/or T-H conditions (?)
 - SCC is usually a matrix (or band) of short "micro" cracks of varying depth
 - may be axial, circumferential or both
 - substantial strength against rupture
 - Sludge pile is implicated
 - ID and OD

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Structural Integrity Issues

- Three Items/Questions Being Addressed
 - Do "unseen" cracks compromise integrity?
 - NDE instrument threshold issue
 - experience indicates not an issue w/existing MRPC
 - Do "undetected" cracks pose significant risk?
 - POD issue
 - relates to ISI process/system
 - potential for leaving "large" cracks in service
 - experience indicates not an issue
 - Can a "rational" repair/plugging limit be developed?
 - alternate to "plug/repair-on-detection"
 - morphology, ID v OD, etc.
 - use "ARC methodology"
 - existing 40% TW TS limit

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Technical Detail

- Specific Objectives
 - Provide capability for TI assessment based on NDE
 - return to power and cycle length
 - Develop probe sizing capabilities (and associated analysis guidelines) to support both ISI and TI assessment
 - pre-App H
 - Assess feasibility of alternative to "plug-on-detection"
 - w/ existing TS NDE size limit (<40% TW)
 - ARC per SGDSM methodology

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leak rate is compatible with design basis safety analyses.

ISI process reliability, and the capability of the NDE system used in the process, is a crucial element in the SGDSM approach. This requires the development and implementation of standardized and consistent approaches to improved inservice inspection, including inspection sampling and expansion criteria; procedure, equipment, technique qualification and performance demonstration; and data analyst qualification and performance demonstration. Such a protocol has recently been approved in a limited sense for axial ODSCC at tube support plate intersections in plants with drilled-hole, carbon steel support plates (Ref. 5). The USNRC is proposing rule making which, when completed sometime in the 1997 time frame, would provide for routine application of SGDSM in US PWRs.

Relative to circumferential cracking, the near-term work in the SGDSM area is to develop consistent recommendations for NDE data acquisition and analysis. This will in turn allow preliminary growth rate data to be obtained, and sizing and structural assessment capabilities to be developed, evaluated and refined. The goal, if technically supportable and justified, is to develop a repair criteria that is an alternative to "plug/repair-on-detection".

REFERENCES

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2. "Steam Generator Progress Report", Revision 11, November 1995, Electric Power Research Institute - Steam Generator Strategic Management Project, Palo Alto, California.
3. "Steam Generator Reference Book", Revision 1, EPRI Report TR-103824, Volume 1, December 1994.
4. C. S. Welty, Jr.. "The Maintenance and Inspection Aspects of PWR Steam Generator Management", Canadian Nuclear Society Steam Generator and Heat Exchanger Conference, Conference Proceedings, Volume 1, Toronto Canada, 1990.
5. USNRC Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking", August 1995.

ISI/NDE

□ Circumferential Cracking

- Detection
 - "probe" versus "coil"
 - "configuration" may be a better term
 - existing RPC technology historically "adequate"
 - when analyzed properly
 - newer configurations
 - improved technical and economic features
 - ISI and Zetec had qualified for specific applications
 - App H "Peer Review" 5/95
 - reviewed 45 configurations (systematic/in/frequency)
 - systems - Mix 18 & 30, TCE700
 - coil/lymbes - 0.50617, 0.113 P/C, + Point, Cocco 3 & 5, Chant 3
 - damage - PWSCC, ODSCC

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ISI/NDE (cont'd)

□ Circumferential Cracking

- Detection (cont'd)
 - large # qualified
 - all coils, probes and systems
 - lower threshold (30%-40%)
 - "With qualified techniques and analysts (both available), detection of circumferential cracks is/should not be an issue"
- Length sizing
 - prior performance w/available techniques - ± 35 RMSE
 - newer coils (+ Point) - < 25 RMSE
 - flaw matrix
 - ID & OD, varying depths up to TW
 - EDM (113), laser cut (22), lab (4), pulled tubes (7) - total 144

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ISI/NDE (cont'd)

□ Circumferential Cracking

- Depth sizing
 - currently no "qualified" techniques
 - not required w/"plug-on-detection" scenario, difficult issue
 - current work
 - flaw profiling
 - some software available
 - need adequate cal scale (RPC phase angle)
 - automation of analysis
 - specimen fabrication
 - pulled tube verification

□ Note: Absence of "formal" App H qualification does not signify lack of capability and should not be regarded as an impediment

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STEAM GENERATOR DEGRADATION SPECIFIC MANAGEMENT (SGDSM)

□ PROCESS for ISI and Plugging/Repair of SG Tubes Based on Specific Damage Form

- Defines reliable NDE process and measurable parameter, and relates to appropriate structural need
 - results in more realistic tube plugging criteria
- Documents (submitted for NRC review/approval 8/93)

□ Technical Approach

- Modeled after European approach
 - Prepared by Ad-hoc Committee

□ USNRC Proposing "Rule-Making"

- Schedule - resolve technical issues in 1995 and complete rule-making in 1997
- NEI S/G Issues Working Group
 - Supported by Rule Making Task Force

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SG "Rule"

□ Performance Based

- Develop and implement program
 - monitor condition of SG tubes
 - assure capability to fulfill intended safety functions
 - function as part of RCFB
 - prevent/mitigate consequences of accidents - <10CFR 100
 - establish performance criteria
 - factor in industrywide experience
 - describe performance criteria
 - provide balance of measures
 - prevention, inspection, repair and monitoring
 - include measures
 - assurance of effective detection, monitoring and evaluation
 - maintain measures
 - mitigating consequences of gross rupture

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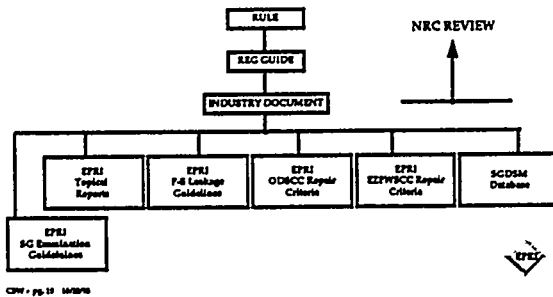
SG "Rule"

□ Principal Performance Criteria

- Reg Guide OR Industry Document
 - structural
 - adequate integrity, w/margin, for entire cycle
 - accident leakage
 - accident leakage/doses meet 10CFR 100 G/La
 - operational leakage
 - monitoring and SD prior to SGTR

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Framework for SG Rule



Repair Criteria Development

□ Approach

- Determine impact on structural tube integrity (TI)
 - leakage during operation *and/or*
 - rupture during operation *and/or*
 - rupture under accident loads *and/or*
 - leakage under accident loads
- Determine physical condition and parameter(s) that impact TI
 - volume of material loss for volumetric flaws
 - e.g., wastage, thinning, wear, IGA
 - matrix of small (TW or near TW) cracks
 - e.g., TSP ODS/CC, EZ/FWSCC, non-coplanar circ cracks
 - length and depth (possibly orientation) of single free span cracks

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Repair Criteria

□ Approach (cont'd)

- Relate physical parameter(s)/condition to an NDE parameter or parameters that can be *reliably* determined
 - detected at a threshold below which structural limit is not exceeded, and meaningfully "measured" (related to degree of TI-loss)
 - e.g., bobbin coil phase angle can be related to *depth* of structural wall loss for volumetric damage, array and/or pancake coil length and orientation for SCC, bobbin coil voltage for TSP ODS/CC RPC length of longest axial crack for EZ/FWSCC, etc.
- Relate *change* in NDE parameter(s) to *growth* in loss of TI
- Address uncertainties
 - POD, NDE characterization/sizing, growth rate, leak and burst correlations

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Repair Criteria

□ When Approach Can't Be, or Has Not Been, Met

- Plug/repair on detection
 - implicit status for TTS circ cracking
- Rely on defense-in-depth
 - operational LR monitoring, EOPs, precursor conditions, etc.
 - e.g., fatigue, loose parts

□ Complex Issue for Crack-like Damage!

- Usually dealing w/a matrix of *small* cracks
 - varying lengths and depths
 - TW depth is *NOT* the relevant physical parameter
 - ECT phase-angle is *NOT* the (angle) relevant NDE parameter
 - *potential "misdiagnosis" problem*
- Defects that are TW or that can go TW under accident loads introduce leakage as a design basis issue (beyond SGTR issues)

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7 Working Session 2: Tubing Inspection

J. Guerra, Tecnatom, and R. L. Tapping, AECL

This session was attended by delegates from 10 countries, and four papers were presented. A wide range of issues was tabled for discussion. Realizing that there was limited time available for more detailed discussion, three topics were chosen for the more detailed discussion: circumferential cracking, performance demonstration (to focus on POD and sizing), and limits of methods. Two other subsessions were organized: one dealt with some challenges related to the robustness of current inspection methods, especially with respect to leaving cracked tubes in service, and the other with developing a chart of current NDE technology with recommendations for future development. These three areas are summarized in turn, along with conclusions and/or recommendations.

During the preliminary discussion, several people brought forward items that probably deserve attention, but were not as high-priority as the major topics. Several of these lower-priority topics are probably worth further discussion, for instance standardized inspection calibration, methodology and performance demonstration, how to deal with changing forms of degradation (this could probably be covered by the development of probes to detect such a variety), and how to deal with NDE uncertainty.

During the discussions there were four presentations. There were two (Canada, Japan) on eddy current probe developments, both of which addressed multiarray probes that would detect a range of flaws, one (Spain) on circumferential crack detection, and one (JRC, Petten) on the recent PISC III results.

7.1 Circumferential Cracking

This was recognized as a complicated issue, and is in reality a set of problems. NDE signals and inspection techniques are known to vary depending on the location and geometry of the cracks; for instance PWSCC at the RTZ, ODS CC at U-bends, TSPs, and the TTS have all been identified at various stations and each case requires specific attention. It was recognized that the crack growth rate of such defects is important to know, even for those stations plugging on detection, which is the practice in most countries (excepting Canada and Belgium). However, the uncertainty on crack growth rates measured by NDE is not well known. The spectrum of root causes for circumferential cracking tends to lead to a lack of confidence in relying on NDE results alone, so most countries plug on detection.

It was noted that UT technology is probably best used for circumferential crack detection in conjunction with ECT. The average crack detection limit for UT, in favorable cases, is about 30% through wall. RPC is the most commonly used ET technique, and a median detection limit of about 50% is quoted. Bobbin coil probes cannot detect circumferential cracks, so the objective is to develop probes as fast as the bobbin coil but with the ability detect circumferential cracks, and

preferably cracks with depths much shallower than currently possible. This will allow more confidence for leaving cracked tubes in service and will provide crack growth data if the probes can also improve their reproducibility in measuring crack depth.

On sizing, both Canada and Belgium have "qualified" practices acceptable to their regulatory authorities. This "qualification" is based on a large number of removed tubes, and on laboratory-generated crack samples. Qualification is in progress for some ET techniques in the U.S. It was recognized that length sizing is possible only for that portion of the crack above the detection limit, so crack lengths may be under-estimated. Improved crack depth detection will improve crack length estimation.

The use of EDM notches to simulate in-service cracks was discussed, and, at this time, it was concluded that they can be used reliably only for probe development and procedure calibration. Once again it was stressed that more realistic defects are needed, along with more pulled tubes.

Those steam generator owners with Incoloy 800 or Inconel 690 tubing wondered if there is any need for improved NDE capability, given that currently cracking is not a concern with these materials. Given the lack of understanding of the cracking mechanisms of Inconel 600, and their applicability to other tubing alloys, it is probably unwise to assume that the lack of cracking in Incoloy 800 or Inconel 690 can be assumed to continue for the life of the station. Hence, continued NDE probe improvements and procedures will be needed there also.

New probe designs appear to promise faster detection of a wide variety of degradation mechanisms and are currently dealing with the need to simplify the presentation of data from these multiarray detectors. It was noted that C-scan displays make it much easier for the analyst to disposition any defects, and this and other improvements in software are needed so that NDE data analysis does not become the critical path.

The PISC III analysis, which was designed to evaluate inspection capability for simulated defects for a number of probes, was compromised somewhat by the lack of circumferential cracking generated in the test samples. Overall, there were some interesting results but it seemed clear that the PODs developed for notches and the few (≈ 10) SCC cracks were not good enough to determine that notches could be used for qualification.

7.2 Challenges

As noted in Section 1, if cracked tubes are to be left in service, an improvement in POD and sizing capabilities is required. This will require more sensitive probes, more accurate sizing, and an increased number of pulled tubes (or retired steam generator sections) against which to qualify procedures and techniques, along with laboratory-generated cracks. It was suggested that tubes

from retired steam generators be used. However, the cost of obtaining these tubes is likely to be significant and no actions were taken to pursue this further. French experience with the retired Dampierre steam generators was mixed, with good practice for evaluation of inspection capabilities but more expensive than anticipated.

The current practice of bobbin coil and RPC ET probes applied in series for detecting/sizing flaws in susceptible steam generators was deemed to be an acceptable approach for now, especially given the increasing inspection efficiency due to technology improvements.

However, multi-coil probes are becoming available for detection of all currently-known defects. Some of these probes use transmit/receive or multiarray coils and currently operate at nearly the speed of bobbin coils. However, considerably more data are generated from the multicoil probes and improvements in analysis tools and data presentation will be necessary before these probes are widely accepted. These improvements need to simplify flow diagnoses and ensure that the analysis does not become the critical path.

Discussion on the use of the voltage criterion to assess axial cracks in TSPs was extensive and controversial, and concluded that the approach, which had different origins depending on the jurisdiction, and hence different calibrations, has weaknesses but is an acceptable approach for now. It was recognized that a more robust approach may be needed in the future, and that this would be somewhat dependent on advances in probe technology.

Overall, it was recognized that inspection technology is rapidly improving and that regulatory requirements need to be flexible in order to provide steam generator operators with flexibility as well as safe operation.

7.3 Chart of Capabilities

In order to summarize and capture the various practices used by the participants, and to determine future needs, a chart of "capabilities" was developed.

The attached chart gives the NDE capabilities for circumferential cracking for each country participating in the Workshop. It is evident that improvements in POD sizing are needed for those who wish to leave cracked tubes in service, and an improvement in detection of OD cracks is needed. There appears to be little work focusing on development of UT as a stand-alone robust inspection technique; for the foreseeable future, it will likely continue to be used as a complementary technique to ET.

There were two clear divisions. Those with Alloy 600 tubing who require the technology that will enable them to live with the various degradation mechanisms that plague that alloy, and those with Alloy 800 or 690 tubing who seem less

concerned about the need for significant improvements in inspection technology. This is based on the belief that these two alloys will continue to demonstrate good resistance to cracking, in particular. However, it will be unwise to assume that this situation will continue for the design life of the steam generator, and hence there is a need for continued routine inspections of samples of this tubing sufficient to determine the health of the steam generator, and to have technology capable of detecting cracks before significant damage has occurred.

Table 1. International Comparison of Circumferential Crack Inspection

Country	Technique	Detection Limit (min. depth)	Sizing	Inspection Scope	Future Actions
USA	EPRI guidelines are employed. Use RPC (+Point, Pancake) and array probes (CECCO)	Limit is plant-specific. Detection threshold is 40-50%, ID & OD	- No qualified sizing techniques expected until 1996 - Current method for length is based on RPC. Phase and amplitude used for depth. +Point appears best for length.	Scopes are plant dependent. In general, 20% of tubes are inspected.	Qualified sizing by EPRI Guidelines is expected. More tubes to be pulled for performance-based qualification. Alternate repair criteria and better probes and software needed.
France	RPC (absolute and differential) using EdF Procedures. Qualification based on pulled tubes.	Based on 72 recently pulled tubes, the POD is 95% for OD cracks with a 35% cross section and 95% for ID cracks with a 29% cross section. Threshold is 50% through-wall for OD cracks.	No depth sizing. - OD circ. length not accurate. Only ID crack lengths are sized. (RPC accuracy is $\pm 30^\circ$).	Inspect 100% of MA-1600 tubes (hot leg) in roll trans. with RPC each year (OD). For TT-1600 tubes, inspect 100% sludge pile, and 25% at random. Nothing done for 1690 tubes.	- Qualifying UT equipment but no plans for industrial use. - Improve OD circumferential crack detection (sizing is not a primary objective). - Automated analysis.
Japan	Detection with 8x1 probes	50% for ID cracks	No data	100% at mechanical expansion using 8x1 (hot leg)	New probe for faster inspection (rotating field probe). Automated analysis and double manual analysis for future.
Canada	CECCO-3 is used for detection and sizing (amplitude-based). Technique qualified on pulled tubes. RTZ qualification based on lab samples.	40-50% for OD cracks	Use CECCO data from BRUCE pulled tubes, estimate $\pm 15\%$ of through-wall depth, 5 mm length for OD cracks.	Inspection 100% with active degradation 20% for plausible degradation Only small samples for 1-800 SGs.	- Auto analysis of CECCO probe data. - Improved analysis using C-scan plots. - Work on UT probe development and qualification. - Validation program with lab/pulled tubes.
Germany	BC and RPC but no cracks	NA	NA	10% every 4 years with BC, covering 100% of length	No need

Table 1. International Comparison of Circumferential Crack Inspection (Cont'd.)

Country	Technique	Detection Limit (min. depth)	Sizing	Inspection Scope	Future Actions
Belgium	RPC+UT and +Point used for inspections. Techniques qualified on EDM notches and pulled tubes.	30-40% for OD cracks using ET+UT.	Estimation of corrected degraded cross-sectional area from raw UT and TOFD data.	100% in plants with cracks (all hot leg tubes, some cold leg).	- Continue to improve ET+UT technique. - Continue to develop sleeve inspection technique. - Assess new transmit/receive probes.
Spain	RPC with qualification on lab samples and pulled tubes.	50% for OD cracks	No sizing.	100% of plants with cracks (hot leg) and 3% of others, except 0 for 1-800 tubes.	- Replace SGs that have circ cracks. - Use +Point and pancake RPC for detection.
Switz.	Use BC and RPC on crevice, tube sheet, etc. If nothing seen with BC, then no RPC is used. No formal technique qualification required.	50%	Use reference notches, but change of length is most important for OD and ID cracks. Length criteria is 13 mm.	3% of 1-800 and 1-690 tubes and 100% for 1-600 tubes (hot leg).	- Assess new multielement probes to be tried for fast screening. - Need for better understanding of growth rate and undefined signals.
Finland	BC for general inspection with RPC on tube sheet. No cracking problems.	N/A	No sizing.		- Assess single-pass probe for detection.

TRANSMIT-RECEIVE EDDY CURRENT PROBES FOR DEFECT DETECTION AND SIZING IN STEAM GENERATOR TUBES

L.S. Obrutsky, V.S. Cecco and S.P. Sullivan

ABSTRACT

Inspection of steam generator tubes in aging Nuclear Generating Stations is increasingly important. Defect detection and sizing, especially in defect prone areas such as the tubesheet, support plates and U-bend regions, are required to assess the fitness-for-service of the steam generators. Information about defect morphology is required to address operational integrity issues, i.e., risk of tube rupture, number of tubes at risk, consequential leakage.

A major challenge continues to be the detection and sizing of circumferential cracks. Utilities around the world have experienced this type of tube failure. Conventional in-service inspection, performed with eddy current bobbin probes, is ineffectual in detecting circumferential cracks in tubing. Other alternative NDT techniques such as rotating pancake probes or ultrasonic testing are sensitive but very slow.

It has been demonstrated in CANDU steam generators, with deformation, magnetite and copper deposits that multi-channel probes with transmit-receive eddy current coils are superior to those using surface impedance coils. Transmit-receive probes have strong directional properties, permitting probe optimization according to crack orientation. They are less sensitive to lift-off noise and magnetite deposits and possess good discrimination to internal defects. A single pass C3 array transmit-receive probe developed by AECL can detect and size circumferential stress corrosion cracks as shallow as 40% trough-wall. Since its first trial in 1992, it has been used routinely for steam generator in-service inspection of four CANDU plants, preventing unscheduled shutdowns due to leaking steam generator tubes. The C3 probe has provided the information about crack depth and crack length required to address licensing issues.

More recently, a need has surfaced for simultaneous detection of both circumferential and axial cracks. The C5 probe was designed to address this concern. It combines transmit-receive array probe technology for equal sensitivity to axial and circumferential cracks with a bobbin probe for historical reference. This paper will discuss the operating principles of transmit-receive probes, along with inspection results.

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TRANSMIT-RECEIVE EDDY CURRENT PROBES FOR DEFECT DETECTION AND SIZING IN STEAM GENERATOR TUBES

L.S. Obrutsky, V.S. Cecco and S.P. Sullivan

1. INTRODUCTION

Detection and sizing of defects in steam generator (SG) tubes are important licensing issues. Information about defect length and depth is required for structural analyses of degraded tubes, to assess the fitness-for-service of the steam generators.

Cracks in SG tubes can occur due to a variety of mechanisms such as stress corrosion cracking (SCC), fatigue and inter-granular attack (IGA). They can initiate from the tube ID or OD and can be circumferential, axial or branching. They occur most frequently at tubesheet transition, support and U-bend regions.

Detection of circumferential cracks in SG tubes continues to present a major challenge to the nuclear industry. Conventional eddy current inspections are performed with bobbin probes. Since they are insensitive to circumferential cracks in tubing, specialized probes, such as the motorized rotating pancake probes (MRPC), are required. However, they are very slow and detection is often hampered by the presence of tube deformation and conducting or ferromagnetic deposits [1,2,3]. Although ultrasonic testing (UT) can be used to detect such cracks, its inspection speed is also very slow.

In 1991 Unit 2 at Bruce Nuclear Generating Station A (BNGS-A) was shut down because of steam generator tube failures due to circumferential stress-corrosion cracking at the 40° hot leg U-bend support plate. Eddy current testing (ET) was performed using the best commercially available technology at that time which used pancake impedance coil probes. It was demonstrated that impedance probes were only able to detect some cracks that had propagated 100% through the tube wall. No partial through-wall cracks were ever detected by impedance probe technology in this application. Even through-wall cracks in tubes identified as leakers were only barely detected [4,5].

To address this problem AECL developed, under the sponsorship of CANDU Owner's Group (COG), a new probe specifically designed for detecting circumferential SCC [4].

This probe, denoted as C3, is a transmit-receive multi-coil array probe with four or eight transmit-receive units (T/R units). Since the first field trial on 1000 tubes, it has been used routinely to inspect SG tubes at four CANDU plants for detection and sizing of SCC [4]. The probe was able to detect OD cracks as shallow as 40% deep [6].

Recently, a new transmit-receive probe has been designed to address the need of detecting circumferential and axial cracks simultaneously. The C5 probe design incorporates transmit-receive array technology with equal sensitivity to both type of cracks as well as a bobbin probe for historical comparison.

Transmit-receive (T/R) array probes with up to 24 T/R units have been used to inspect 12.9 mm to 22.2 mm diameter tubing for external (OD) as well as internal (ID) SCC in CANDU and PWR steam generators.

This paper will discuss the operating principles of transmit-receive eddy current probes. It will describe field experience with detection and sizing of SCC with the C3 probe. Additionally, it will explain the features of a C5 probe and its applications.

2. THEORY

2.1 Background Information

Detection of circumferential cracks is one of the most challenging aspects of eddy current tube inspection. Bobbin type probes have very low sensitivity to circumferential cracks, because the eddy currents flow parallel to the circumferential coil windings and do not interact with circumferential cracks [1].

The usual solution is to use MRPC probes or probes consisting of an array of small pancake coils. These are usually impedance probes that have good sensitivity to internal circumferential cracks. However, they tend to have low sensitivity to external defects, and they generate large lift-off noise that complicate signal analysis and reduce sensitivity [2,3]. These probes are normally spring loaded to reduce lift-off effects but that in turn makes them prone to mechanical failure.

Research at Chalk River Laboratories, established that transmit-receive probes have much lower sensitivity to lift-off noise than impedance probes. Computer simulations [6,7] of small defect signals and lift-off for impedance and a transmit-receive probes are compared in Figure 1. For the same defect, signal-to-noise with a transmit-receive probe can be 5 to 10 times that of an impedance probe.

2.2 Normalized Voltage Diagram

Normalized voltage diagrams can be used to display voltage changes of transmit-receive probes [7]. Figures 2b and 2c show response of an absolute transmit-receive probe to various test parameters at various frequencies: lift-off (LO), internal and external wall thinning (ID and OD), internal and external magnetite deposits (ID Mag, OD Mag), external copper (Cu) deposits, and carbon steel (CS) tube supports. Figure 2d shows computer derived signals from internal and external cracks with a differential probe.

Probe response to conducting and ferromagnetic deposits and support plates increases with decreasing frequency, which is typical of all eddy current probes. Fortunately, unlike surface impedance probes, C3 probe signals from internal magnetite deposits are nearly in phase with lift-off signals at frequencies higher than 160 kHz. This is an important advantage of the C3 over surface impedance probes since CANDU SG tubes often have internal magnetite deposits with high magnetic permeability.

Computer modelling permits paper studies of probe/tube response such as in Figure 2; it is very useful in optimizing probe design and is indispensable for signal analysis studies.

3. PROBE CHARACTERISTICS

3.1 Probe Features

The C3 probe is a differential multi-coil transmit-receive probe. Figure 3a illustrates a probe with eight sets of transmit-receive (T/R) coil units capable of operating at four multiplexed frequencies. Unlike a 6x1 or 8x1 spring loaded surface probe, it has no moving parts permitting single pass inspections as fast as bobbin type probes. The eight T/R units are contained in two separate bodies. Each probe body is encased in a metallic sleeve, with centering guides. The probe bodies are physically separated by a section of flexible cable, because a single solid body would be too long to go around tight U-bends. This is a robust design but still flexible enough for it to pass around U-bends as small as 150 mm radius without difficulty.

To obtain eight T/R units, the transmit coils adjacent to either side of the receive coil pair are active at different multiplexed times. This allows each receive coil pair to detect a signal from one transmit coil at a time [4] and results in eight separated 45° windows of sensitivity.

Computer modelling and laboratory tests have demonstrated that probe sensitivity is maximum between transmit and receive coils and drops to zero at the centre of the coils [6,7]. Circumferential coverage of each probe body (coil array) can be represented by four windows of approximately 45° of high sensitivity areas located between transmit and receive coils, with four windows of low sensitivity areas located at the centres of the coils [4]. It follows that each probe body is capable of only 50% coverage of the tube circumference. To achieve 100% coverage in a single scan the second probe body is rotated 45° with respect to the first one. Figure 3b illustrates the eight windows of sensitivity.

The C3 probe requires an instrument that can operate in transmit-receive mode. The Zetec MIZ-18 requires a modification board designed for this application. Newer instruments such as the Zetec MIZ-30 or the RD Tech TC-6700 have this capability incorporated in the basic instrument.

3.2 Directional Properties

A transmit-receive probe's maximum response corresponds to variations in the induced magnetic field in the region between the transmit and receive coils. They have directional properties, being sensitive primarily to defects in line with the transmit-receive coil pairs.

The differential C3 probe design was originally optimized for detecting circumferential cracks, however, it retained some sensitivity to cracks which are not perfectly circumferential.

4. SIGNAL ANALYSIS

4.1 Display of Data

A usual concern, when new eddy current technology is introduced, is whether signal analysis is more complicated than that of standard technology. Since the C3 is a differential probe, each channel yields the familiar figure-8 signal, making the probe "user friendly". Additionally, signal analysis done with a system such as Zetec-Eddynet or Westinghouse-ANSER, which produces multi-channel presentation of data, is relatively easy, since they allow analysts to study the signals from multiple T/R units or frequencies at the same time. Calibration tubes containing concentric grooves are used rather than localized defects as in the ASME standard, to allow all separate T/R

units to be set-up simultaneously. The calibration tubes need to contain a concentric indentation which is used to establish signal phase. That signal is set horizontally and moving to the left as the probe is retracted through the tube past the indentation, so the contribution of tube deformation to the vertical defect signal component is minimized.

The data shown in Figure 4 illustrates signals from a C3 probe with eight T/R units, at a test frequency of 250 kHz. Figure 4a shows the Lissajous and strip-chart signals for one T/R unit from a 3 mm long EDM notch located in an eccentrically deformed section of the tube. Signal-to-noise is excellent. Figure 4b illustrates the strip-chart signals from the eight units of the probe, only one unit out of eight detected the EDM notches, implying that circumferential resolution is about 5 mm.

4.2 Defect Sizing

4.2.1 Crack Depth

Typically, eddy current testing uses signal phase (orientation angle on the Lissajous display) to predict defect depth. However, in the BNGS-A SG, the tube deformation at the 40° hot leg scallop bar caused large signal phase distortion. Therefore signal phase was not considered a reliable parameter for predicting crack depth in these tubes. Vertical signal amplitude (V_{mx}) had to be used to estimate crack depth.

Since the eddy current signal amplitude from SCC is affected by electrical contact between the walls of the crack and factors such as crack morphology, artificial defects such as EDM notches and machined grooves cannot be used to generate accurate calibration curves. The calibration curves for C3 probes were derived by correlating eddy current signals with results of destructive examination of pulled tubes from BNGS-A, Unit 2 as well as laboratory induced SCC. Figure 5a shows a calibration curve for the C3 eight T/R unit probe for the 250-70 kHz mix. It contains separate data points for ten pulled tubes and the average normalized signal amplitude for repeated scans of tubes with laboratory induced SCC. The solid line represents 'the best-fit' curve through these data points. The inserted table shows the voltage values and the corresponding depth used to build the curve[8].

Figure 5b presents the sizing accuracy using the calibration curve for the 250-70 kHz mix. The solid lines represent ideal sizing of eddy current predicted depth versus the metallography measurement. In most cases, the ET estimates fall inside the $\pm 15\%$ error band.

4.2.2 Crack Circumferential Extent

Defect length is often an important licensing issue. The combined information about circumferential crack extent and crack depth are required to address concerns about the risk of tube rupture during fitness-for-service assessment of steam generators.

The C3 probe with eight T/R units generates eight outputs, each covering a detection window of approximately 45° of the tube circumference, thus crack length can be estimated by the number of units detecting the crack. Furthermore, the output from each unit can be plotted in sequence around the circumference at equal intervals to produce a two dimensional depth profile of a crack.

C3 probes were used to assess circumferential extent of cracking in about 50 tubes[9]. To validate the method, eddy current predictions were compared to destructive examination results of ten pulled tubes from BNGS-A and 20 tubes with laboratory induced SCC. Figure 6 contains results from a typical pulled tube. Figure 6a shows the C3 signals from the 250-70 kHz mix, the

Lissajous and strip-chart display of one T/R unit shows the eddy current estimates of defect depth. Figure 6b shows a two dimensional profile generated with the eddy current data of the crack from a pulled tube, superimposed on the actual crack profile obtained by metallography. The solid line, which was generated using a polynomial equation, represents a 'best-fit' curve through the discrete values of the eddy current estimates. Figure 6c is a cross-section of the tube. Note the good agreement between the two profiles.

The information provided to the station was crack extent at a given defect depth level, typically 40%. Figure 7 illustrates the accuracy of the crack extent predictions. The solid line represents the ideal sizing versus the metallography measurements with a ± 5 mm error band. The tubes with laboratory induced SCC were scanned several times at different relative probe positions to determine repeatability. The plot shows that, in almost all cases the predictions fall within the ± 5 mm error band.

5. PROBABILITY OF DETECTION

The C3 probe has been extensively used to inspect 12.9 mm diameter, Inconel 600 steam generator tubes at BNGS-A. The probes were able to detect and size cracks at the area of interest, where conventional eddy current probes had previously failed to detect any cracks that had not propagated completely through the tube wall.

Detectability with the C3 probe was not significantly affected by deformation, yielding a clear vertical signal from defects. However, the presence of copper and magnetite deposits, in some cases, reduced signal-to-noise and complicated signal analysis.

Results from the destructive examination of 100 pulled tubes was used to generate probability of detection (POD) histograms of circumferential cracks with the C3 four T/R unit probe. Figure 8 shows that cracks as shallow as 40% are likely to be detected, while the POD decreases rapidly for shallower cracks [6].

6. OTHER APPLICATIONS OF TRANSMIT-RECEIVE ARRAY PROBES

C3 probes have been used to inspect 22.2 mm diameter Inconel 600 tubes in PWR SGs for internal SCCs (PWSCC) at the U-bend transition. The probe, first evaluated in short U-bend samples with laboratory induced SCCs, was able to detect cracks as shallow as 20% deep; field evaluation showed that the single pass C3 probe had defect detectability as good or better than MRPC probes.

A growing area of interest has been the simultaneous detection of both circumferential and axial cracks. Computer modeling studies were conducted to design a probe with equal sensitivity to circumferential and axial cracks [6]. This new probe, denoted as C5, comprises three bodies: the front and middle bodies contain 8 to 24 T/R units with equal sensitivity to circumferential and axial cracks, as well as volumetric defect detection capabilities, and the back body contains conventional bobbin coils to obtain historic signals for comparison. Figure 9a illustrates the areas of sensitivity for a single C5 T/R unit to short circumferential and axial cracks. Figure 9b shows C5 probe response to a defect free tube in the tubesheet area. Figure 9c shows C5 probe response to an axial crack at the tubesheet expansion region. These probes have been used to inspect tubes from 15.9 mm (5/8") to 22.2 mm (7/8") diameter. The C5 probe has been accepted by the US Nuclear Regulatory Commission for providing SG fitness-for-service.

Recently, new software has been developed to generate C-scan displays from array probe data (tube mapping). This type of display provides the analyst with a very useful tool to rapidly visualize tube artifacts.

7. SUMMARY/CONCLUSIONS

A single-pass C3 probe improves the detection of circumferential cracks in SG tubes. Its characteristics/capability includes:

- It detected circumferential SCC, previously undetected by conventional impedance probes.
- Comparison between predicted depths and destructive examination indicated good detection and sizing for circumferential SCC as shallow as 40% through-wall.
- The speed of inspection is similar to that for standard bobbin probe inspection.
- Signal analysis is relatively easy, with multi-channel presentation of data.
- The probe yields the familiar figure-8 signals with similar phase separation as that of standard bobbin probes.
- The probe has no moving parts, making it more durable than surface riding probes.
- SCC can be sized with the C3 probe at highly deformed areas to an accuracy of $\pm 15\%$ in depth and ± 5 mm in length.

Computer modelling can be used to predict probe/sample responses, significantly increasing confidence in this new technology.

The C5 transmit-receive probe provides equal sensitivity for both axial and circumferential cracks in SG tubes including cracks at tubesheet expansions.

8. ACKNOWLEDGMENTS

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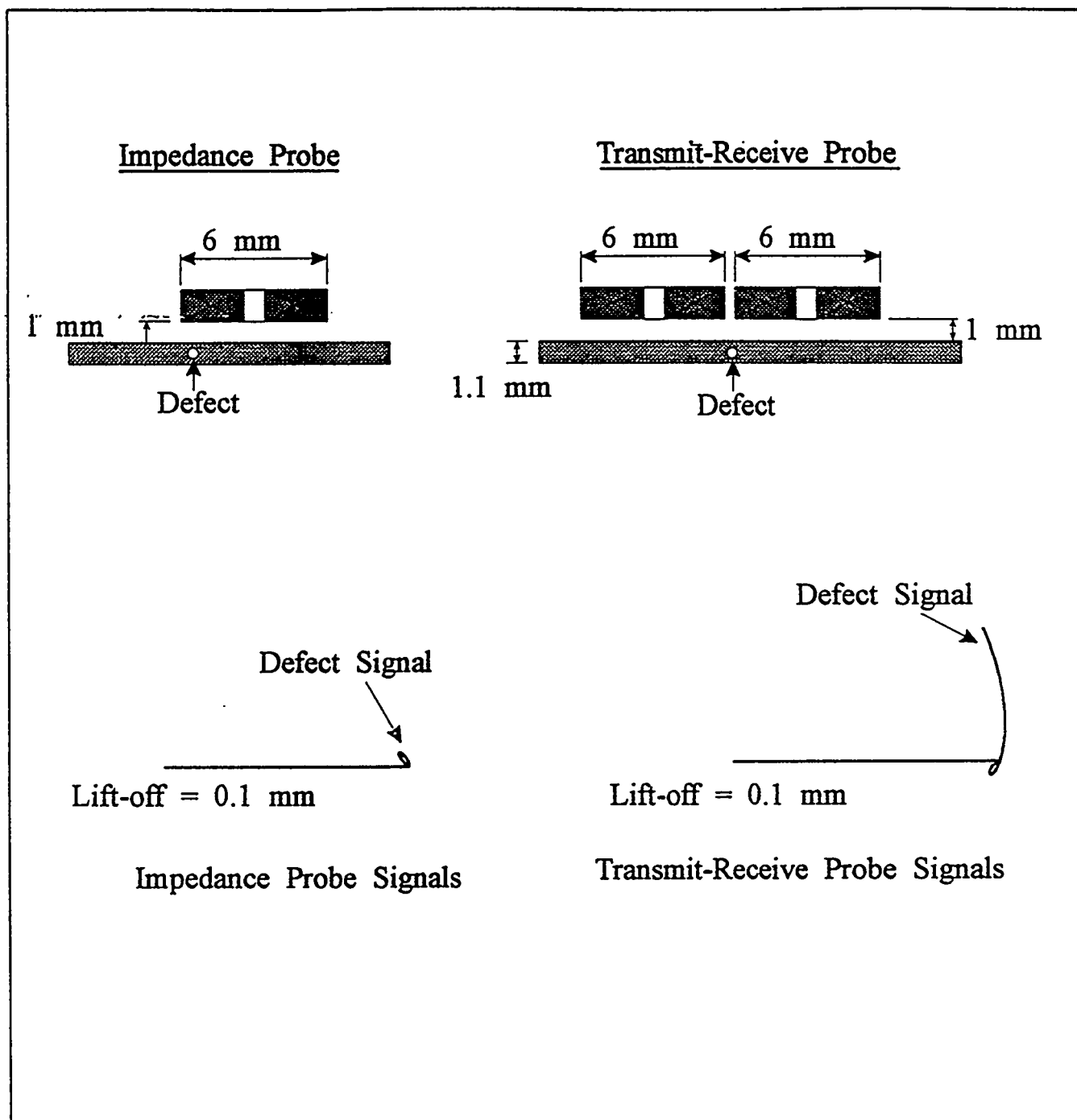


Figure. 1: Signal-to-Noise Comparison Between an Impedance Probe and a Transmit-Receive Probe. The "Signal" is Due to a Small Crack. The "Noise" is Due to a 0.1 mm Variation in Lift-off.

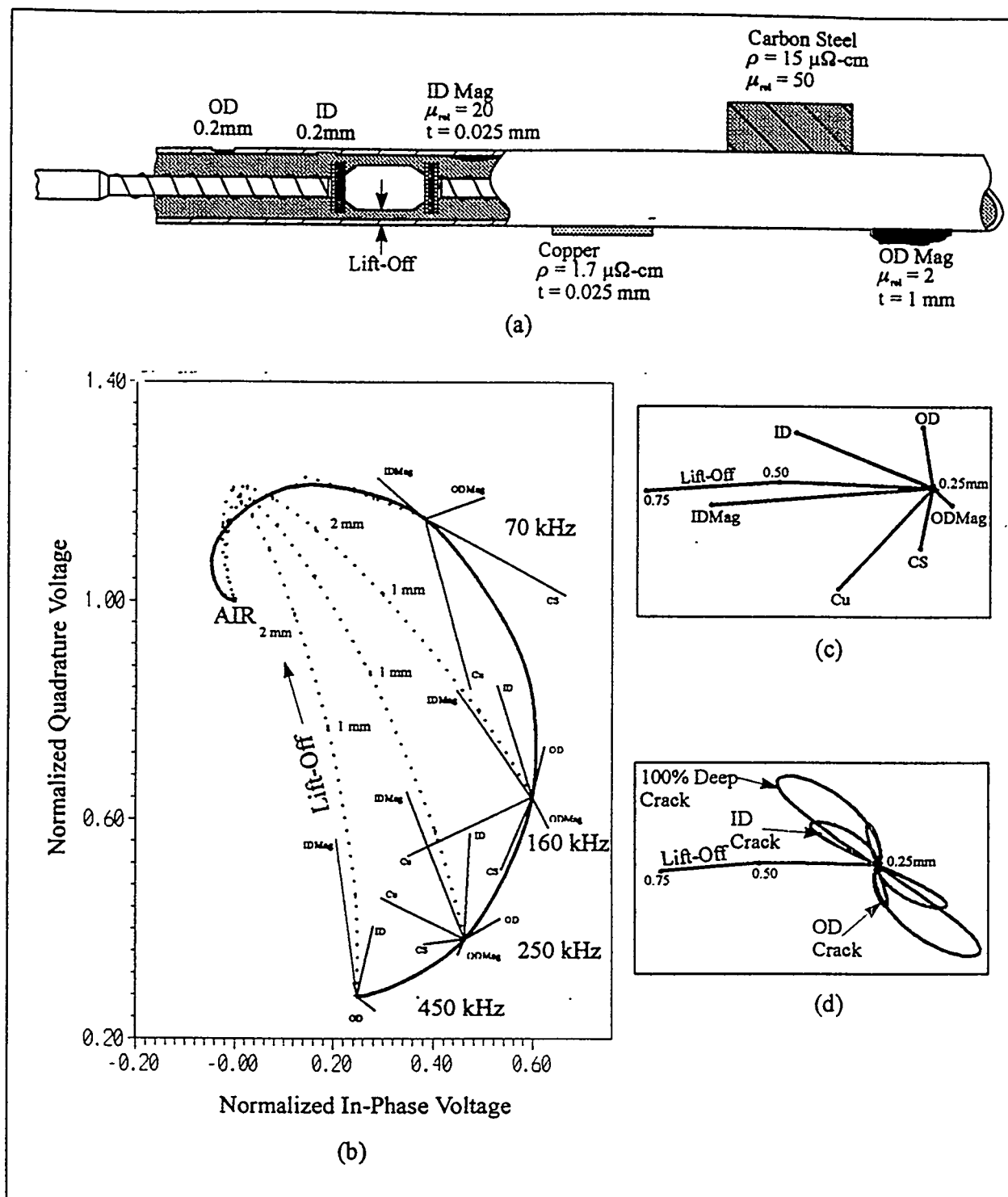


Figure 2 : (a) Tube with Simulated Defects and Deposits.
 (b) Computer-Derived Voltage Display Illustrating the Effect of Various Test Variables at 70 to 450 kHz.
 (c) Absolute Probe Signals at 250 kHz.
 (d) Differential probe signals of defects at 250 kHz.

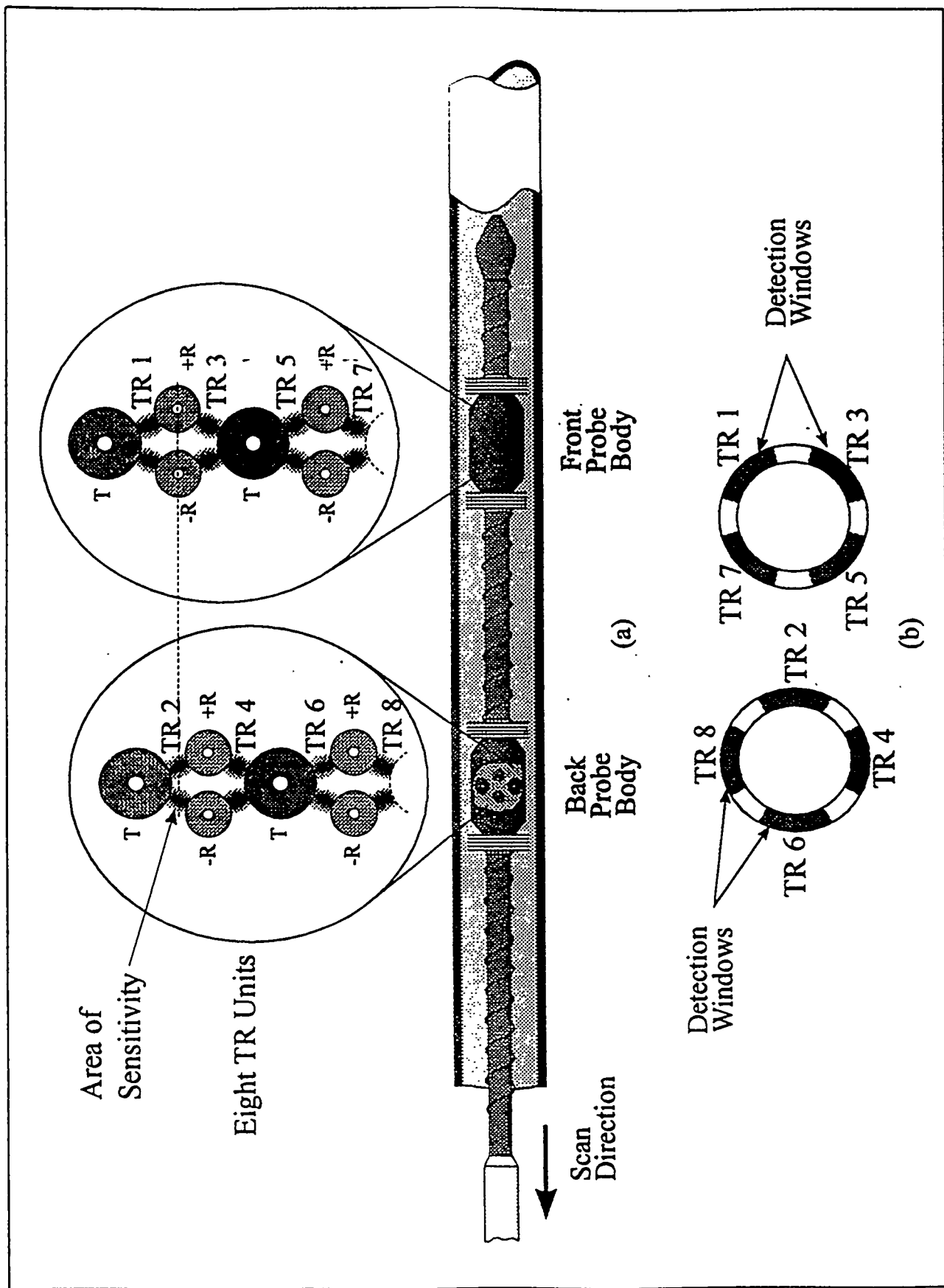
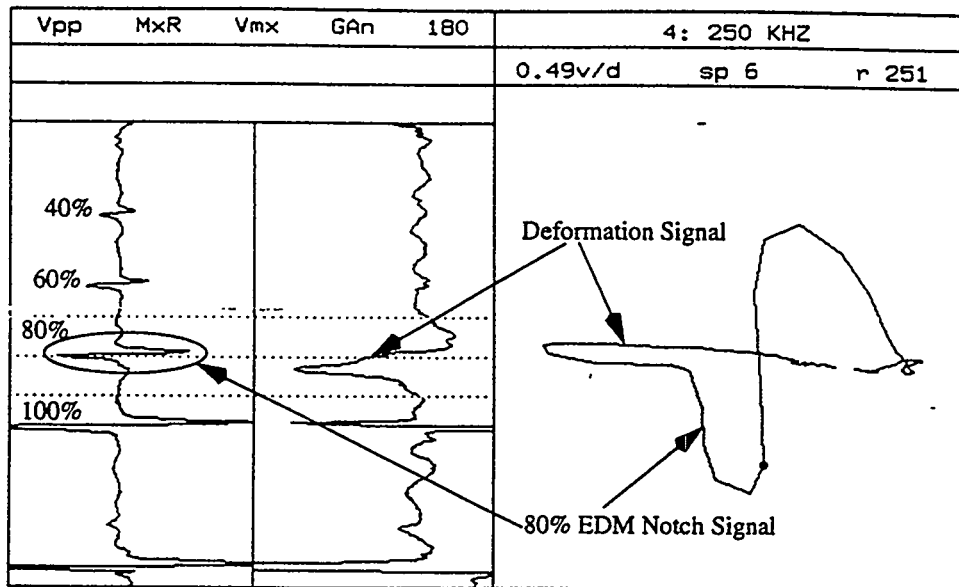
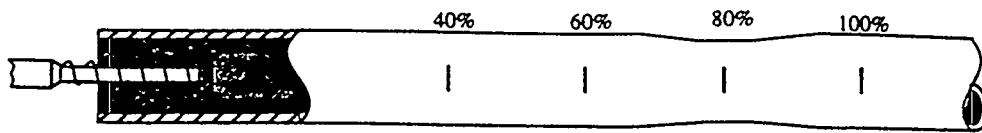
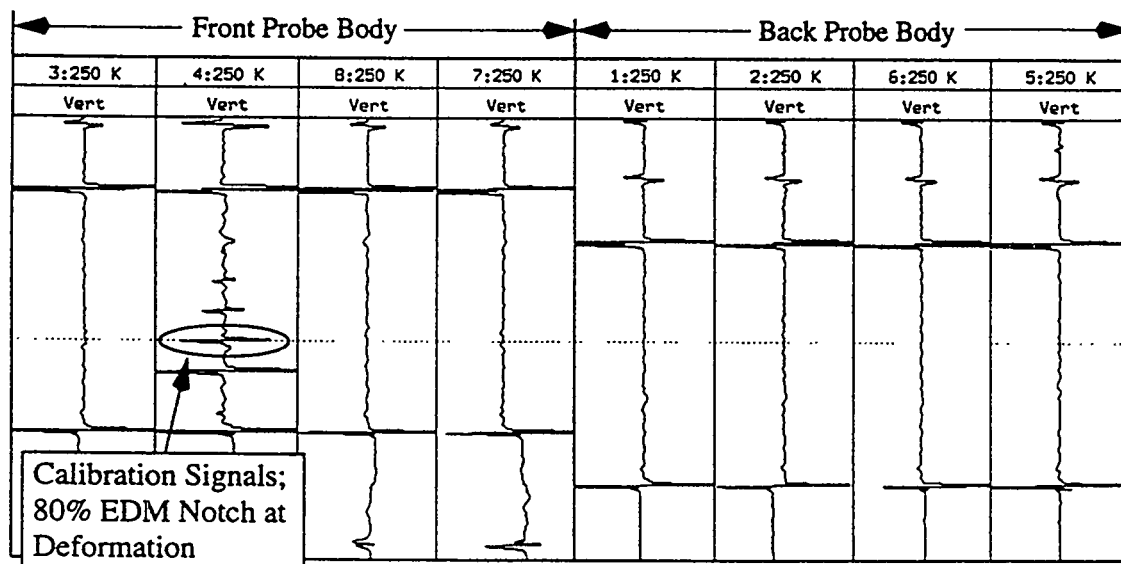


Figure 3: C3 Probe Showing T/R Coil Configuration and Detection Windows.



(a)



(b)

Figure 4: Signals with an Eight T/R Unit C3 probe from Circumferential EDM Notches and Deformation.
 (a) X-Y and Strip-Chart Signals from the 80% EDM Notch plus Deformation.
 (b) Eight-Channel Strip-Chart Display of Calibration EDM Notches, 40 to 100% Deep, 3mm Long

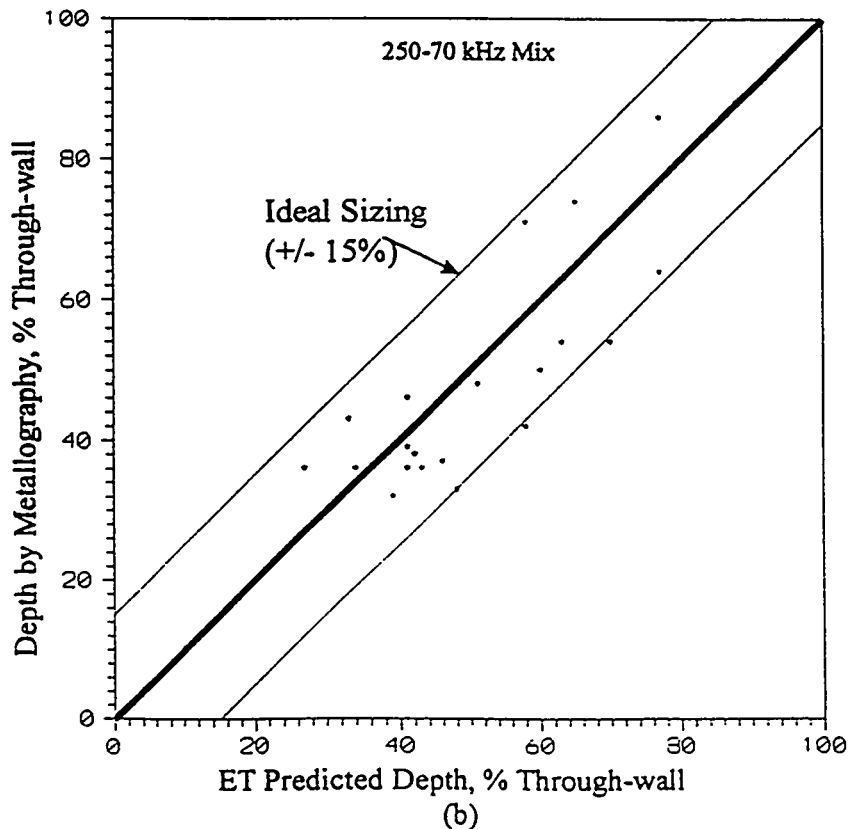
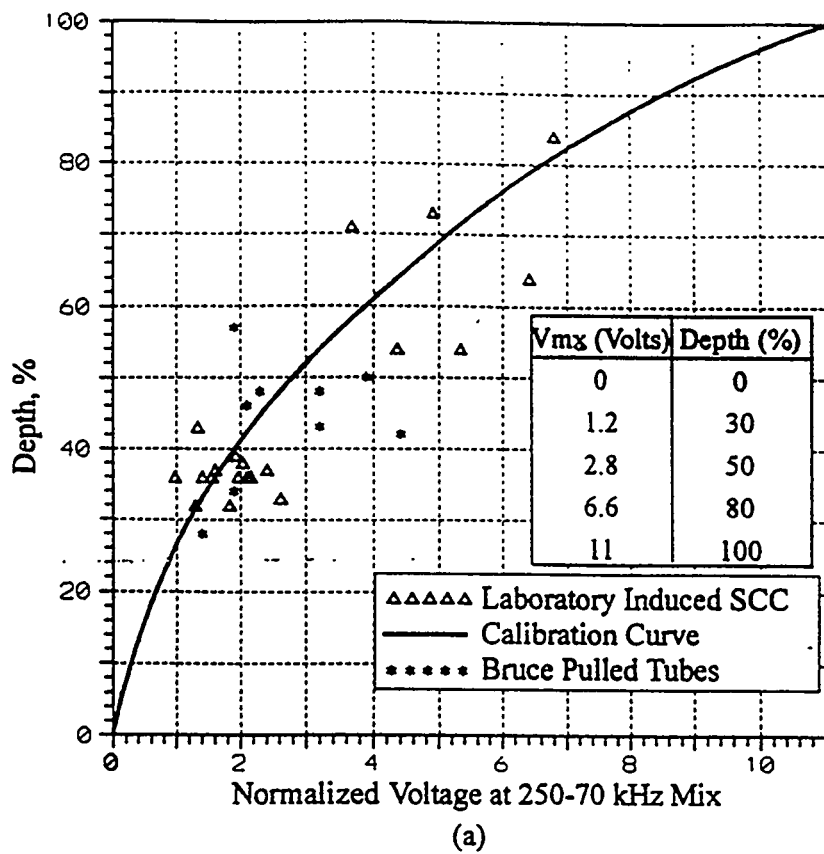


Figure 5: (a) Empirical Calibration Curve for a C3 Eight T/R Unit Probe Based on Average Probe Responses to Laboratory Induced SCC and Pulled SG Tubes from BNGS-2.
 (b) Sizing Accuracy Plot.

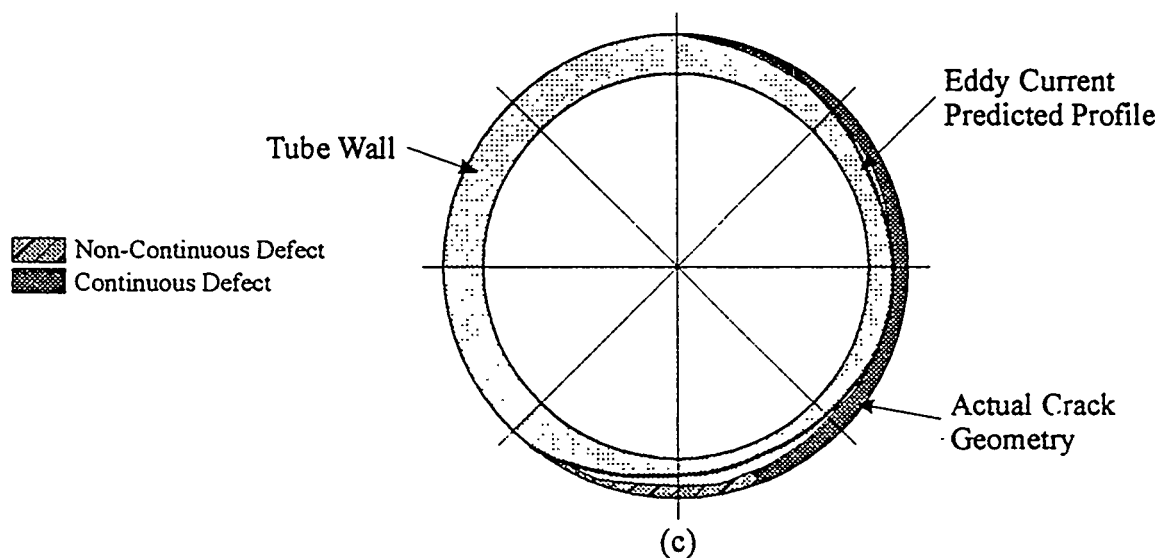
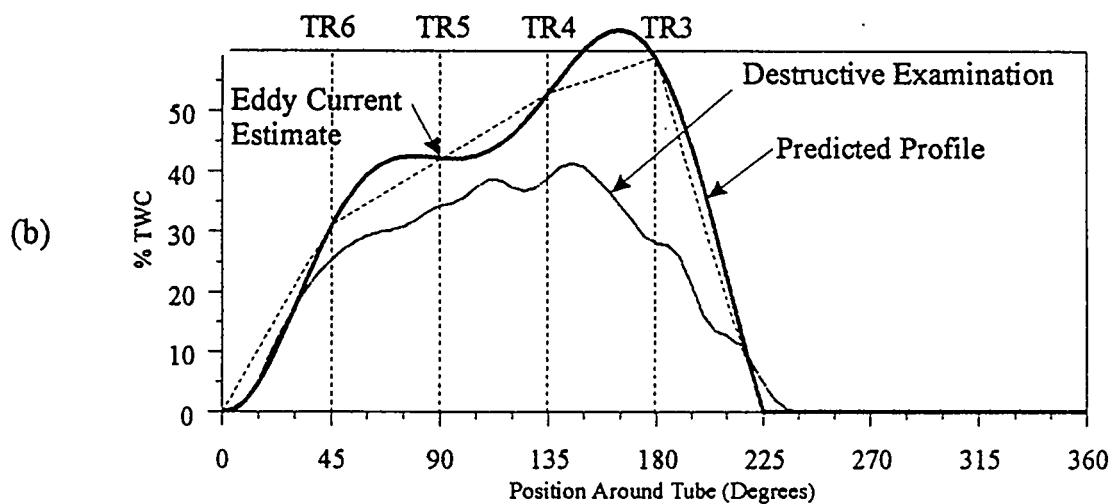
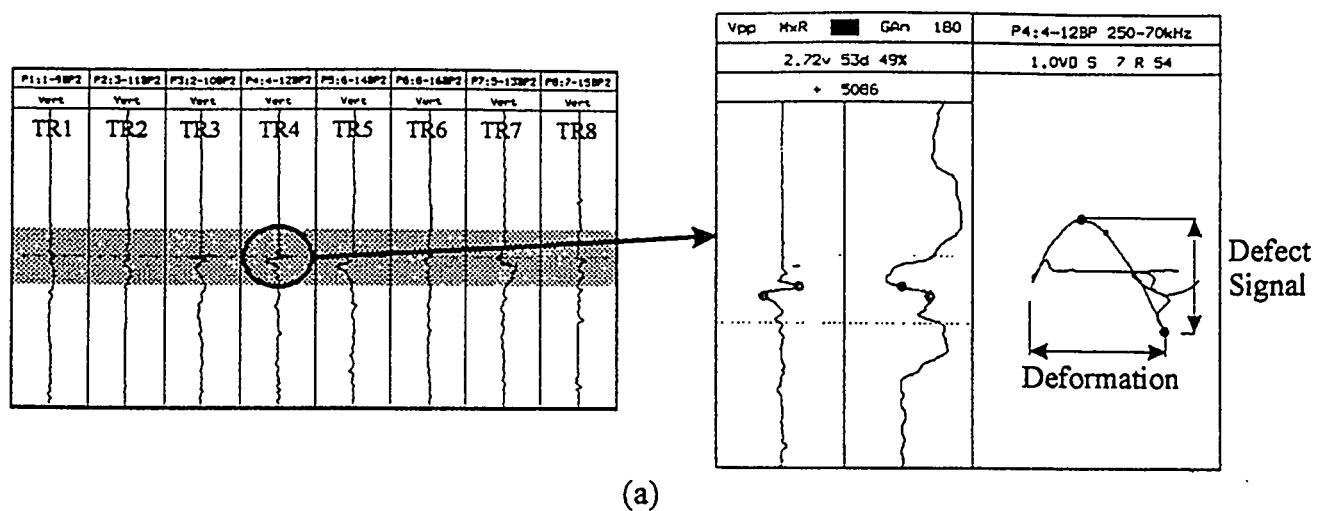


Figure 6: (a) C3 Probe Response to a Circumferential OD Stress Corrosion Crack at the 40° U-bend Scallop Bar.
(b), (c) Comparison Between the Actual Crack Geometry and the Eddy Current Predicted Profile.

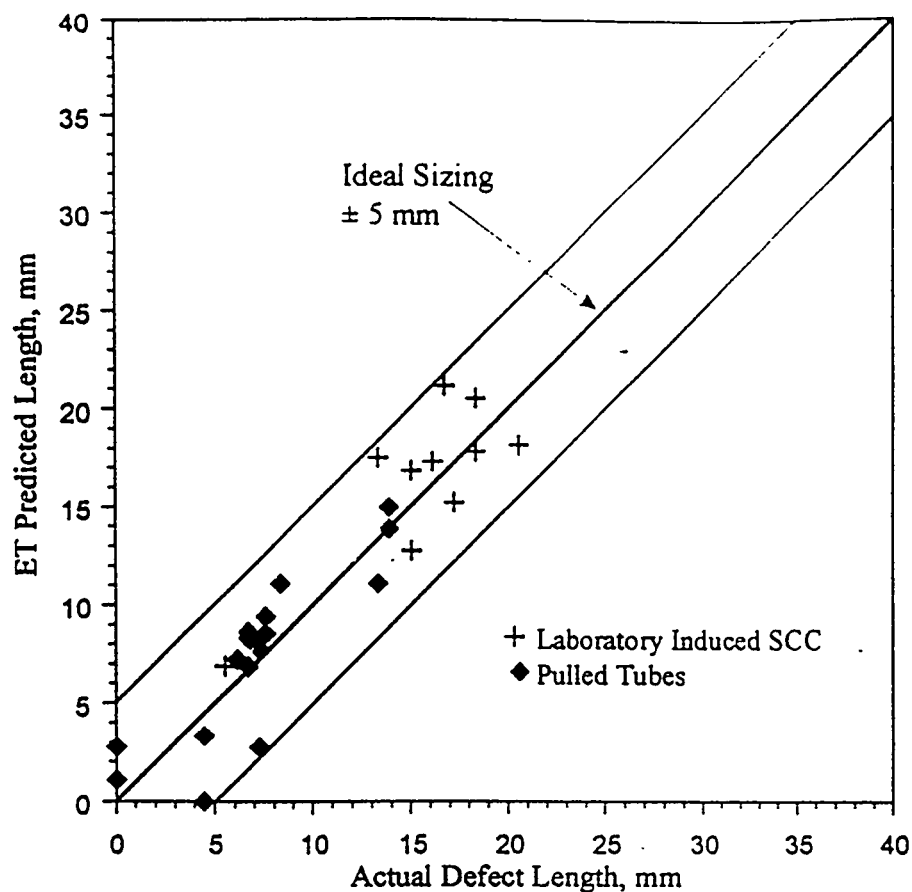


Figure 7: Comparison of ET Predicted and Actual Defect Length of Laboratory Induced SCC (Average Value for Repeated Scans) and Pulled Tubes from BNGS-A.

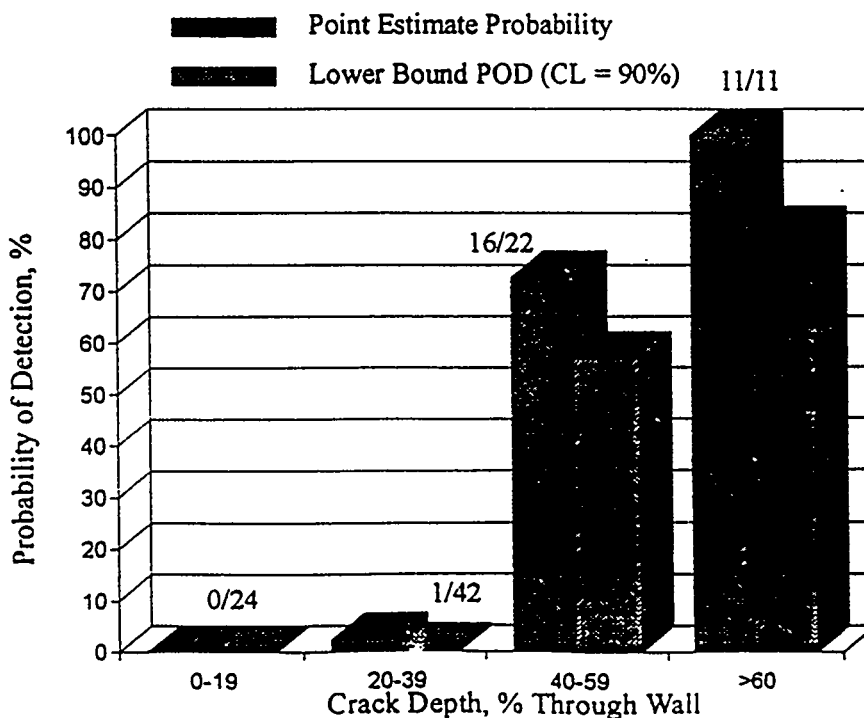


Figure 8: POD Histogram for C3 Probe Detecting SCC in Inconel 600 SG Tubes. This Data is Based on Pulled Tube Destructive Examinations Performed in 1992-94.

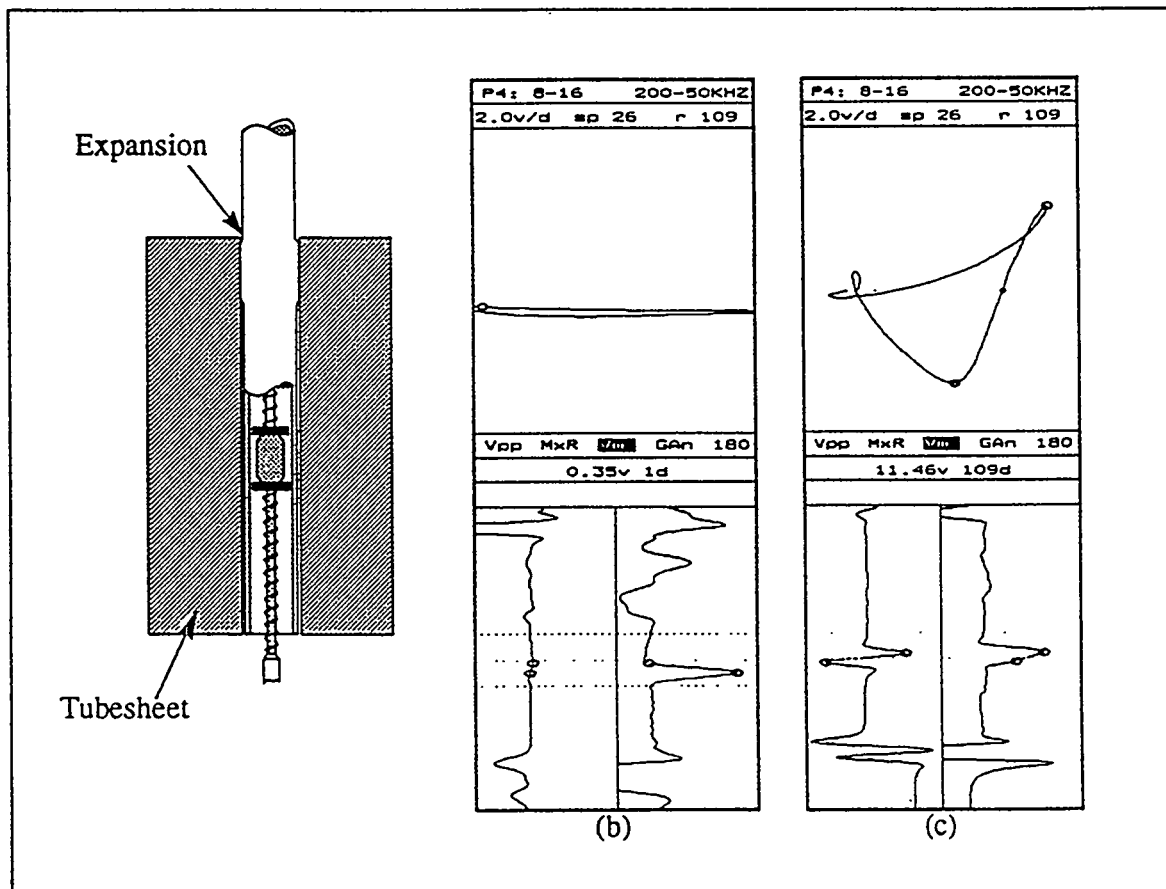
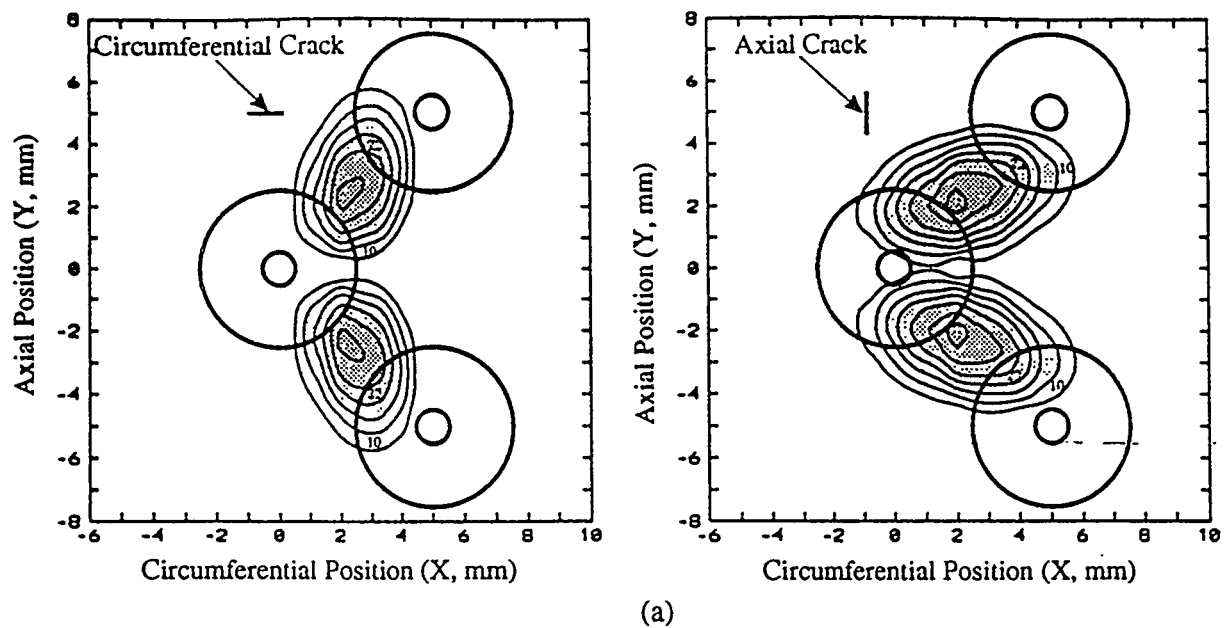


Figure 9: (a) Computer-Predicted Areas of Sensitivity for C5 Probe to Short Circumferential and Axial Cracks.
 (b) C5 Response to a Defect Free Tube in the Area of the Tubesheet.
 (c) Tube with an Axial Crack.

OECD/NEA

**CNRA/CSNI Workshop on
Steam Generator Tube Integrity in Nuclear Power Plants**

Steam Generator Tube Inspection in Japan

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**October 30 - November 2, 1995
Chicago, Illinois USA**

Abstract

Steam generator tube inspection was first carried out in 1971 at Mihama Unit-1 that is first PWR plant in Japan, when the plant was brought into the first annual inspection. At that time, inspection was made on sampling basis, and only bobbin coil probe was used. After experiencing various kinds of tube degradations, inspection method was changed from sampling to all number of tubes, and various kinds of probes were used to get higher detectability of flaw.

At present, it is required that all the tubes shall be inspected in their full length at each annual inspection using standard bobbin coil probe, and some special probes for certain plants that have susceptibility of occurrence of flaw. Sleeve repaired portion is included in this inspection.

As a result of analyses of eddy current testing data, all indications that have been evaluated to be 20% wall thickness or deeper shall be repaired by either plugging or sleeving, where flaw morphology is to be a wastage or wear. Other types of flaw such as IGA/SCC are not allowed to be left inservice when those indications are detected.

These inspections are performed according to inspection procedures that are approved by regulatory authority. Actual inspections are witnessed by the Japan Power engineering and inspection corporation(JAPEIC)'s inspectors during data acquisition and analysis, and they issue inspection report to authority for review and approval.

It is achieved high safety performance of steam generator through this method of inspections, however, some tube leakage problems were experienced in the past. To prevent recurrence of such events, government is conducting development and verification test program for new eddy current testing technology.

1. History of Steam Generator Tube Inspection in Japan

The first steam generator tube eddy current inspection had been performed in 1971 when Mihama Unit-1 that is first PWR plant in Japan, was in first annual inspection. This inspection was carried out in sampling basis for about 500 tubes, and this inspection revealed no significant indications.

After about a half year of operation, tube leakage incident occurred in this plant, all the tubes in service had been inspected using same inspection technique as first inspection.

The cause of tube leakage was due to phosphate thinning, and was easily detectable by conventional eddy current technique. However, as increasing the operating plants, various kinds of tube degradations had been experienced. Eddy current inspection techniques had been improved for better flaw detection capability and easy interpretation of eddy current data.

In early days of eddy current inspection, only bobbin coil probe was used with low frequency. When the dominant mode of degradation was wastage, to evaluate the depth of wastage was prime concern, so high frequency technique using 400kHz had been applied for better depth sizing. However, the wastage was observed at tube support plate annulus or top of tubesheet, it was very difficult to distinguish flaw signal from support structure signal. In this connection, two frequency

method was developed. Low frequency of 100kHz was used for flaw detection, and high frequency of 400kHz was used for flaw depth sizing. During the period between mid '70s and early '80s, it was not available of multifrequency eddy current equipment, equipment was switched to low frequency when probe was pushed and to high frequency when probe was pulled. When multifrequency eddy current equipment became available, multifrequency method was commonly adopted.

As for inspection probe, bobbin coil probe was mainly used in early days. After experiencing various kinds of degradations, many kinds of probe were introduced for specific types of flaw. So-called U-Bend probe which has smaller diameter of coil and top centering guide for better insertability for small radius u-bend portion was introduced after revealing the occurrence of stress corrosion cracks at tight radius u-bends. Next probe was four segment surface riding type probe for better sensitivity for primary water stress corrosion cracks at tubesheet expansion region. This probe was combined with standard bobbin coil probe to form a Dual-Function probe (DF probe) which has two sets of bobbin coils, and has functions of standard bobbin coil probe and four segment probe. This probe is commonly applied for general purpose inspection for full length of tube and tubesheet expansion region inspection at same time without changing probe.

Table-1 and 2 show the history of eddy current inspection technology in Japan.

2. Eddy Current Probes used in Inspection

Eddy current probes used in steam generator tube inspection are as follows.

Name of probe	Applied for
DF probe	full length, general purpose inspection
U-Bend probe	tight radius u-bends
8x1 probe	top of tubesheet / hot leg side (for specific plants)
Rotating pancake coil probe	top of tubesheet (for specific plants)
Rotating plus point coil probe	tight radius u-bends (for specific plants)
Sleeve probe	brazed sleeves and laser welded sleeves

These probes are divided into two categories. First is used for general purpose inspection probes which are used regardless of plant condition, and second is specific purpose probes which are used for plants having susceptibility to certain types of flaws.

The outline of each probes is shown in Table-3, and selection of probe for each type of steam generator is shown in Table-4.

3. Acceptance Criteria

The technical standard for operation and maintenance of nuclear power plant is the same as that of governing design and fabrication of plant, and in principle, no defective materials can be used for pressure boundary. So, this concept is applicable to steam generator tubes after being placed plant

operation. Generally, the following criteria are applied.

For flaws such as wastage and wear that can be evaluated quantitatively by eddy current method, the depth of flaw which exceed 20% of tube thickness should be removed from service or repaired by sleeving.

For flaws such as SCC or IGA that cannot be evaluated quantitatively precisely, all signals regardless of depth which exceeds noise level or shows meaningful change from previous inspection should be removed from service or repaired by sleeving.

Table-5 shows acceptance criteria for specific mode and location of degradation in conjunction with probe used.

4. Execution of Inspection

The standard periodical inspection manual is established by Nuclear Power Safety Administration Division of Agency of Natural Resources and Energy, Ministry of International Trade and Industry (MITI). This manual describes the purpose of inspection and standard procedures of inspection. Steam generator tube inspection procedure is also described in this manual. On the basis of this standard inspection manual, utility submits his specific inspection manual to MITI for approval before execution of inspection.

Examination will be carried out by utility and Japan Power Engineering and Inspection Corporation (JAPEIC) will supervise and witness the examination as the third party inspection agency. Witness point of examination includes performance check of examination such as calibration confirmation, and reviewing of all the analyzed data. After finishing of examination, JAPEIC issues the certificate of examination and MITI will confirm the record of examination results to complete the inspection. In this certificate of examination, only tubes that are having no defects, i.e. requiring no repair, are described. For those tubes that are not described in the certificate of examination, will be repaired with formal procedure.

5. New Inspection Technology

It is achieved high safety performance of steam generator in Japan through above mentioned procedure, however, some tube leakage incidents had occurred in the past. To prevent recurrence of such events, it is essential to improve the inspection technology to reliably detect the flaws before which become critical. In this view point, government is conducting development and verification test programs for improving steam generator reliability.

"Verification test program for eddy current test technology for steam generator tubes" is one of these programs. Objectives of this program are to further enhance the detectability of circumferential flaws in steam generator tubes and to verify the detectability using full scale mockup. This program is seven years program to develop and verify a fast inspection technology for circumferential flaws. As a first phase, it was developed a new type of probe having fourteen sets of pancake coils for circumferential flaw detectability. Fig.-1 shows this probe. In second phase, much sophisticated eddy

current probe is to be developed for both axial and circumferential flaws. This probe will have more than thirty coils around the probe body and switched electronically for faster inspection speed than rotating probe, while keeping the flaw detectability of rotating probes. Fig.-2 shows the concept of this probe.

6. Conclusions

The practices of steam generator tube inspection in Japan including historical change in inspection procedures, inspection probes used in inspection, acceptance criteria, and regal procedure of executing inspections are introduced.

Table-1
Technical Change in Eddy Current Tests

Year	Eddy Current Techniques	Reasons
1971	Sampling Inspection	
1973	100% Inspection	Tube Leakage occurred at Mihama-1
1977	U-Bend-probe	SCC occurred at tight U-bend
1982	Four Segment Probe	SCC occurred at Expansion Region
1983	Single Coil Probe	Long SCC occurred at TS Crevice
1984	Brazed Sleeve Probe	Brazed Sleeve Adopted
1986	Dual Function (DF) Probe	Shorten Inspection Time
1988	3 Frequency Method	Separate Deposit Signal
1988	8x1 Probe	Circ. SCC at Exp. Transition
1992	Rotating Pancake Coil	Better Pitting Detection in Sludge
1995	Plus Point Probe	Better U-Bend SCC detection

Table-2
History of
Tube Degradations and Detection Techniques

	71	73	75	77	79	81	83	85	87	89	91	93	95
Wastage													
TS Crevice SCC													
U-Bend SCC													
TS SCC													
TS/TSPL IGA													
AVB Fretting													
Pitting													
Std Bobbin Coil Probe													
U-Bend Probe													
Four Segment Probe													
Dual Function Probe													
8x1 Probe													
Rotating Pancake coil Probe													
Rotating Plus Point Probe													

Table-3

Eddy Current Probes

JAPFIC

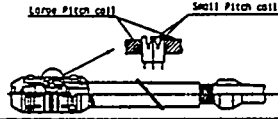
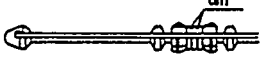
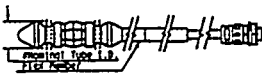
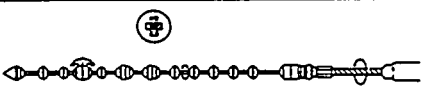
Type	Applied for	Shape
DF Probe (100-400kHz)	Full Length	
DF Probe (100-400-600kHz)	Deposite Signal Area	
U-Bend Probe	Tight Radius U-bend (Row1-3)	
8x1 Probe	Top of TS	
Rotating Pancake Coil	Top of TS	
Plus Point Coil Probe	Tight Radius U-bend (Row1-2)	

Table-4

Probe Selection for SG Types

JAPFIC

SG Type	DF	U-Bend	8x1	RPC	+ Point
CE	○		○	○	
51(Partial Roll)	○	○			
51(Full Depth Roll)	○	○	○		○
51M	○	○			○(MA600)
51F	○	○			
Replace Model	○	○			

Table-5

Acceptance Criteria of ECT

JAPSAIC

Location, Mode	Probe, Frequency	Acceptance Criteria
TSPL etc. IGA	DF Probe (Large Pitch) 100-400kHz	<Noise Level ($<1.5V$)
Expansion SCC	DF Probe (Small/Large Pitch) 400kHz	<Noise Level($<0.5V$)(Small) No Signal Change(Large)
Expansion Circ. SCC	8×1 Probe 400kHz	<Noise Level ($S/N<2$)
U-Bend SCC	U-bend Probe 100kHz	<TSPL Signal × 1/2 No Signal Change
	Plus Point Probe 300kHz	$S/N<2$
TSPL etc. Wastage	DF Probe (Large Pitch) 100-400kHz	<20% Wall Thickness
AVB Fretting Wear	DF Probe (Large Pitch) 100-400kHz	<20% Wall Thickness
Top of TS Pitting	Rotating Pancake Coil 100-400kHz	$S/N<2$
Deposit Signal All Modes	DF Probe (Large Pitch) 100-400-600kHz	<Noise Level ($<1.2V$)

8 Working Session 3: Tubing Integrity

C. Cueto-Felgueroso, Tecnatom, and J. Strosnider, USNRC

Twenty-three individuals representing nine countries (Belgium, Canada, the Czech Republic, France, Japan, the Slovak Republic, Spain, the U.K., and the U.S.) participated in the session on tube integrity. These individuals represented utilities, vendors, consultants and regulatory authorities. The major subjects discussed by the group included overall objectives of managing steam generator tube degradation, necessary elements of a steam generator degradation management program, the concept of degradation specific management, structural integrity evaluations, leakage evaluations, and specific degradation mechanisms. The group's discussions on these subjects, including conclusions and recommendations, are summarized below.

8.1 Objectives

The group agreed that the overall objective of a steam generator degradation management program is to assure public health and safety while providing economic plant operation. It was further agreed that appropriate goals to accomplish this overall objective are:

- No steam generator tube ruptures and
- Minimize steam generator leakage and provide a high confidence of not exceeding established off-site radiological dose limits.

It was noted that achieving a record of no steam generator tube ruptures was an appropriate goal, but not a requirement. This was an acknowledgment of actual operating experience in which steam generator tube ruptures have occurred and have been mitigated such that the health and safety of the public was not adversely affected. Nonetheless, avoiding steam generator tube ruptures is important from both a safety and economic point of view.

8.2 Program Elements

It was agreed by the group that an effective program for managing steam generator tube integrity should include the following elements:

- Prevention.
- Inspection and repair.
- Leakage monitoring.
- Mitigation.

It was noted that these elements can be used in various combinations or with different emphases and that different methods of implementation can be used to provide effective steam generator degradation management programs. In this regard, no single program is ideal for all countries, but the programs and the methods of implementation are tailored based on economic, regulatory, and political considerations. A comparison of practices in different countries supported this observation. For example, countries in which the economics support steam generator replacement may place less emphasis on tube repair than countries in which steam generator replacement is difficult to justify economically.

8.3 Degradation Specific Management

Degradation Specific Management (DSM) represents the concept of using the inspection and repair criteria most appropriate for a specific form of degradation. Under this concept, different inspection methods, e.g., probes or analysis techniques, and different repair criteria, e.g., flaw length, flaw depth, or voltage amplitude, are used for different types of degradation. Based on the groups discussions, the DSM concept is in fact being practiced to some degree in most countries. There was a consensus that the DSM concept has merit and can help to optimize the effectiveness of a program by allowing flexibility to adopt to changing technology e.g., NDE techniques, and operating experience. This has advantages from both a safety and economic perspective. In this regard, **the group recommended that regulatory requirements should be flexible to allow adapting to changing technology and operating experience, e.g., allow implementation of DSM (recommendation 3-1).**

The group noted, however, several issues that need to be considered in the implementation of DSM. Some repair criteria considered under DSM require higher confidence in NDE and defect growth rate and may require addressing potential leakage and off-site radiological doses under postulated accident conditions. Also, tighter primary-to-secondary leakage monitoring may be appropriate. It was noted that laboratory data may not be representative of degradation rates that occur in operating steam generators and that plant-specific data are the most reliable. Therefore, the group concluded that robust data bases are necessary to support development and implementation of the DSM approach. Specifically, the group concluded that data are necessary to verify NDE capability and reliability and to correlate burst pressures and leakage rates to NDE parameters. The group concluded that examination of tubes removed from service is essential for verification of NDE capability and reliability. It was agreed that although the uncertainties associated with structural integrity models are less those associated with NDE, testing of tubes removed from service is also useful for verifying structural integrity models. The need for correlations of NDE parameters to leakage rates was tied to the acknowledgment that some repair criteria could allow leaving through-wall cracks in service, and that potential primary-to-secondary leakage may need to be evaluated.

Based on the above, **the group recommended that an international data base of steam generator tube pressure tests, leakage tests, and destructive examinations be developed (recommendation 3-2).** The purpose of the data base would be to maximize the amount of data available to participants to assist in their development and qualification of inspection methods and repair criteria.

In conjunction with this recommendation, **the group recommended pursuing acquisition of tubes from retired steam generators for testing and utilization of in-situ pressure testing to add to the existing data base (recommendation 3-3).**

8.4 Structural Integrity Criteria

The general approach used to establish steam generator tube repair criteria involves (1) defining a set of design basis loading conditions, (2) establishing a critical value of a parameter related to the failure load of the tube and (3) applying appropriate margins to the critical parameter to account for continued degradation during the next period of operation and NDE and other uncertainties. The design basis loads typically include the pressure differential associated with a postulated rupture of the main steam line. The parameter related to failure load of the tube can be a physical characteristic of the defect, e.g., length or depth, which generally comes from a structural integrity model such as limit load analysis. Alternatively, an NDE parameter that can be correlated to the failure load, e.g., eddy current voltage amplitude, may be used. In some cases, allowances for continued degradation, NDE, and other uncertainties are explicitly considered, while in other cases an overall factor of safety intended to account for all the uncertainties is applied. The group discussed structural integrity models, design basis loads, and factors of safety.

Regarding structural integrity models, the group concluded that existing models, in particular limit load analyses, are generally reliable. Limit load analysis is applicable because of the high ductility and toughness of the materials used in steam generator tubing, and has been demonstrated through extensive laboratory testing. However, it was noted that care may need to be taken in the application of limit load analyses in some cases such as that of extremely deep flaws. It was also noted that the parameter used for establishing repair limits should be carefully chosen based on the capabilities of NDE. For example, although structural integrity models exist that consider both crack depth and length, limitations in the ability to accurately measure depth may support use of a criterion based solely on crack length.

As indicated above, design basis loadings generally include the differential pressure associated with a postulated main steam line break. There was some discussion of whether or not it is reasonable to postulate a main steam line break. In this regard, it was noted that the postulated main steam line break may be considered a surrogate event to bound all other potential secondary side depressurization events, and it is not clear what it should be replaced with if it were

eliminated. However, it was also suggested that a great deal of experimental information exists on the actual failure modes, e.g. break time, of piping and that this information may provide a basis for reducing the dynamic loadings associated with an assumed instantaneous double-ended pipe rupture.

With regard to factors of safety, it was noted that if the ASME code factors of safety are used to assess tube integrity, as is done in many countries, the factor of safety of three on normal operating pressure may be more limiting than that associated with the postulated main steam line break. Some countries do not consider the factor of safety of three on normal operating pressure, and it was suggested that a reassessment of the ASME code factors of safety as applied to steam generator tubes may be appropriate.

Finally, there was some discussion regarding the use of probabilistic criteria for assuring tube integrity. Probabilistic criteria are still used primarily to complement deterministic criteria. Although it was noted that the USNRC did establish criteria for conditional probability of tube rupture under a postulated main steam line break in the recently issued Generic Letter 95-05 on voltage-based criteria for outside stress corrosion cracking at tube support plates.

8.5 Leakage Evaluations

The group discussed leakage evaluations in the context of monitoring primary-to-secondary leakage under normal operating conditions and assessing potential leakage under postulated accident conditions.

There was a consensus that normal primary-to-secondary leakage monitoring should be an important part of any program for managing steam generator tube degradation. It was agreed that not all tube degradation will lead to leak-before-break and therefore primary-to-secondary leakage monitoring cannot be relied upon exclusively to assure steam generator tube integrity. However, it was noted that review of operating experience in the U.S. indicates that prompt action in response to primary-to-secondary leakage can help in avoiding tube ruptures. Based on their discussion, the group recommended that primary-to-secondary leakage monitoring programs should include a limit on the absolute leakage value and on the trend in leakage.

As indicated above, one of the objectives of a steam generator degradation management program is to provide a high level of confidence that any potential leakage under postulated accident conditions is low enough to satisfy established radiological dose limits. The importance of this issue is emphasized by the acceptance of repair criteria that may leave through-wall defects in service. Various methods are used to accomplish this objective. They include establishing repair criteria sufficiently conservative to "preclude" leakage, demonstrating leak tightness through hydrostatic testing, and analyses of potential leakage. Analyses of potential leakage and off-site doses are generally statistically based, and it was suggested that care should be taken not to propagate undue conservatism through

the analysis, e.g., it may not be necessary to use 95/95 values for all the random variables.

8.6 Specific Degradation Mechanisms

8.6.1 Wear at Antivibration Bars

The group concluded that this degradation mechanism can be effectively controlled using current approaches.

8.6.2 Axial ODSCC at Top of Tubesheet

Length-based criteria for leaving axial cracks in service at the roll transition at the top of the tubesheet are being implemented in some countries. The French, in particular, have extensive experience implementing this criteria and have demonstrated through hydrostatic testing of numerous steam generators that such defects leak only very small amounts at differential pressures in excess of postulated accident differential pressures. It was noted that in-situ leakage and pressure tests may be useful in plant specific applications of this criteria.

8.6.3 Axial ODSCC at Tube Support Plate Intersections

Both eddy current voltage amplitude and depth-based criteria are used to manage this degradation mechanism. It was noted that voltage-based criteria should consider circumferential tube failure due to the potential formation of bands of circumferentially oriented intergranular attack. Differences in voltage limits between countries are the result of different eddy current calibration techniques and credit for restraint provided by tube support plates. **It was recommended that a data base on tube "lock-up forces" at support plate intersections be developed (recommendation 3-4).** These data would be useful in assessing the potential for support plate displacement under postulated accident conditions.

8.6.4 Circumferential Cracks

There was no uniform position on leaving circumferential cracks in service. The Belgians have implemented a criterion for leaving circumferential cracks in service in the roll transition at the top of the tubesheet, and the Canadians have implemented criteria for leaving circumferential cracks in the U-bends in service. None of the other countries represented in the group had implemented criteria to leave circumferential cracks in service. Nonetheless, there was a consensus that improving the capability and reliability to size circumferential cracks is important. In this regard, it was noted a large penalty is paid in order to account for sizing uncertainty and that even if circumferential cracks are not allowed to remain in service, there is value in accurately sizing them in order to assess the as-found condition of the steam generator.



CNRA/CSNI International Workshop - OAK BROOK - ILLINOIS

October 30 - November 2 , 1995

STEAM GENERATOR TUBE INTEGRITY FLAW ACCEPTANCE CRITERIA

335

NUREG/CP-0154

B. COCHET

**STEAM GENERATOR ENGINEERING MANAGER
ENGINEERING SERVICES DIVISION**



FLAW ACCEPTANCE CRITERIA

THERE ARE SEVERAL APPROACHES WHICH VARY SOMEWHAT BETWEEN DIFFERENT COUNTRIES.

■ OBJECTIVES

- AVOID THE UNSCHEDULED SHUTDOWN OF THE REACTOR IN NORMAL OPERATION
- AVOID SGTR
- AVOID EXCESSIVE LEAK RATE IN THE EVENT OF ACCIDENTAL OVERPRESSURE (SLB - FLB)

■ FOR EACH DEGRADATION, THE FOLLOWING QUESTIONS MUST BE ANSWERED :

- DOES LBB CONCEPT APPLY ?
- WHAT ARE THE PERFORMANCES OF NDT ?
- WHAT ARE THE KINETICS OF DEVELOPMENT ?
- WHAT IS OUR UNDERSTANDING OF THE PHENOMENON ?

■ FROM THESE ANSWERS, THE POLICY IS DEDUCED :

- FREQUENCY OF IN-SERVICE INSPECTION
- PLUGGING CRITERIA BASED ON :
 - CRACK DEPTH MEASUREMENT
 - CRACK LENGTH MEASUREMENT
 - BOP_N VOLTAGE



FLAW ACCEPTANCE CRITERIA

■ DEFINITION

$$L_T = L_C - L_P - E$$

L_T : ALLOWABLE CRACK LENGTH

L_C : CRITICAL CRACK LENGTH

L_P : CRACK PROPAGATION

E : MEASUREMENT ERROR OF CRACK LENGTH DETERMINATION

■ METHODOLOGY

⇒ UNDERSTANDING OF DAMAGE MECHANISMS BASED ON EXPERIMENTAL WORKS AND PULLED TUBE EXAMINATIONS

- Type and location of flaws
- Flaw propagation kinetics



EVALUATION OF PRIMARY WATER STRESS CORROSION CRACKING OF THE STEAM GENERATOR TUBES

❑ INTRODUCTION

MRPC RESULTS SUMMARY

❑ ANALYSIS OF THE EVOLUTION OF THE NUMBER OF CRACKED TUBES

**ANALYTICAL METHOD (WEIBULL DISTRIBUTION)
FRENCH OPERATING EXPERIENCE
APPLICATION TO A SPECIFIC PLANT**

❑ ANALYSIS OF HEAT TO HEAT SENSITIVITY TO PWSCC

**NUMBER OF CRACKED TUBES PER HEAT/LOT
ASSIGNMENT OF PWSCC SENSITIVITY INDICES
MONTE CARLO SIMULATIONS**

❑ ANALYSIS OF CRACK GROWTH RATES

**INSPECTION DATA
COMPARISON WITH PULLED TUBE EXAMINATION**

❑ CONCLUSIONS

//-

FLAW ACCEPTANCE CRITERIA

⇒ DETERMINATION OF CRITICAL FLAW DIMENSIONS

⇒ DATA/PARAMETERS

- PRESSURE AND TEMPERATURE
- TUBE DIMENSIONS
- MECHANICAL PROPERTIES OF MATERIAL
- LOCATION OF THE FLAWS

⇒ VALIDATION OF MECHANICAL STRENGTH CRITERIA

- AXIAL AND CIRCUMFERENTIAL CRACKS
- MULTIPLE AXIAL AND MIXED DEFECTS
- DEFECTS WITH COMPLEX MORPHOLOGY (no sizing by NDT)

⇒ JUSTIFICATION OF THE TUBE BEHAVIOR IN ACCIDENT CONDITIONS

⇒ THEORETICAL ANALYSIS OF THE SEVERITY OF FLAWS

Based on the most unfavourable combination of material data and geometry of the tubes under postulated accident conditions(SLB - FLB)



FLAW ACCEPTANCE CRITERIA

■ BURST TESTS

PARAMETERS LIKELY TO INFLUENCE THE BURST PRESSURE OF TUBES :

- INFLUENCE OF THE SEALING SYSTEM
- LOAD TIME HISTORY
- DYNAMIC EFFECT OF CRACK PROPAGATION FOLLOWING THE RUPTURE OF THE REMAINING LIGAMENT OF A PART-THROUGH FLAW



**INFLUENCE OF PRESSURE APPLIED TO THE SIDES OF THE DEFECT
(COMPARISON OF RESULTS OBTAINED FOR ONE HEAT)**

Ø mm	DEFECT LENGTH mm	WITHOUT PRESSURE EFFECT ON DEFECT SIDES	WITH PRESSURE EFFECT ON DEFECT SIDES	DECREASE IN INTERNAL PRESSURE %
		P_a bar	P_a bar	
19.05	7	-	458 ± 10	-
22.22		472 ± 4	424	10
19.05	15	285 ± 16	247 ± 8	13
22.22		298 ± 8	268 ± 9	10
19.05	25	184 ± 20	144	22
19.05	30	170 ± 4	127 ± 6	25
22.22		169 ± 3	-	-



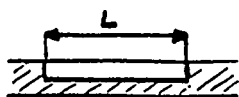
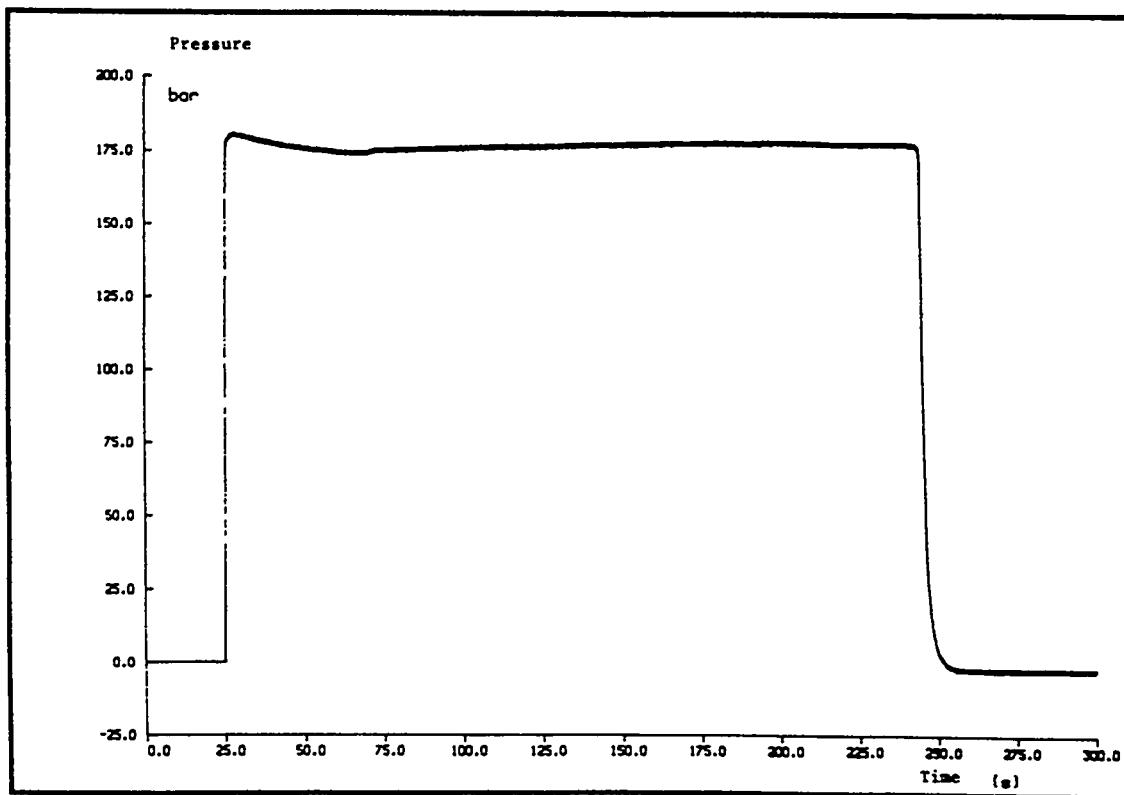
INFLUENCE OF PRESSURE BUILD UP RATE

DEFECT LENGTH (mm)	MEAN RATE (bar/mn)	INSTABILITY LIMIT PRESSURE Pa (bar)
7	75 1800	458 ± 2 % 452
15	75 ± 25 % 970 ± 5 % 215 ± 12 % 234 930	298 ± 4 % 299 ± 11 % 270 ± 1 % 269 266
30	69 315 > 1000	126 ± 5 % 128 ± 5 % 121



■ TEST CONDITIONS

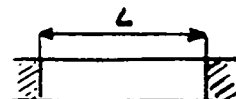
- TESTS AT ROOM AND ELEVATED TEMPERATURE :
 $328^{\circ}\text{C} \leq \theta \leq 350^{\circ}\text{C}$
- MAXIMUM PRESSURE > 182.5 BAR



UNSTABLE
PART-THRU CRACK



TUBE BREAKTHROUGH
WHEN P



STABLE
THROUGH-WALL CRACK
after Breakthrough

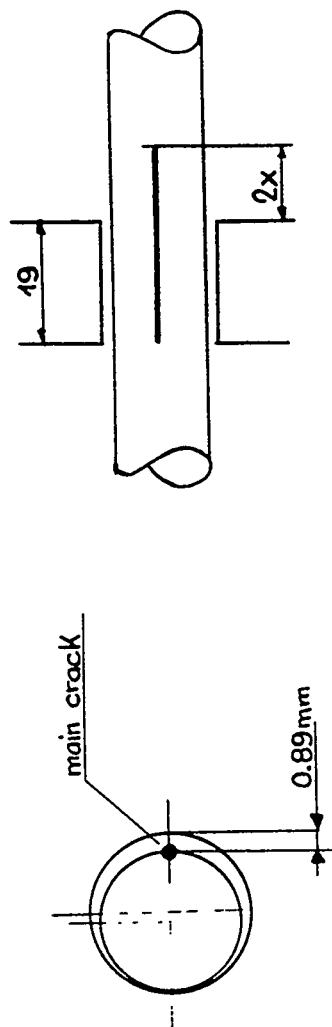
$L < L_c$



BURST TESTS AT ELEVATED TEMPERATURE

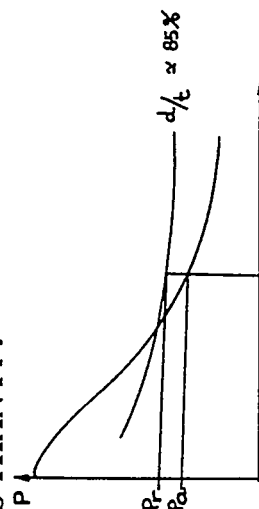
TESTS WERE PERFORMED WITH DEFECT OR CRACKS LOCATED SO AS TO MINIMIZE THE INFLUENCE OF THE SUPPORT PLATE.

- FLAW LOCATION



- ELEVATED TEMPERATURE TESTS TAKE INTO ACCOUNT :

1. THE DYNAMIC EFFECT WHEN P_a OF THE THROUGH-WALL CRACK RESULTING FROM RUPTURE OF THE REMAINING LIGAMENT IS LESS THAN P_r .



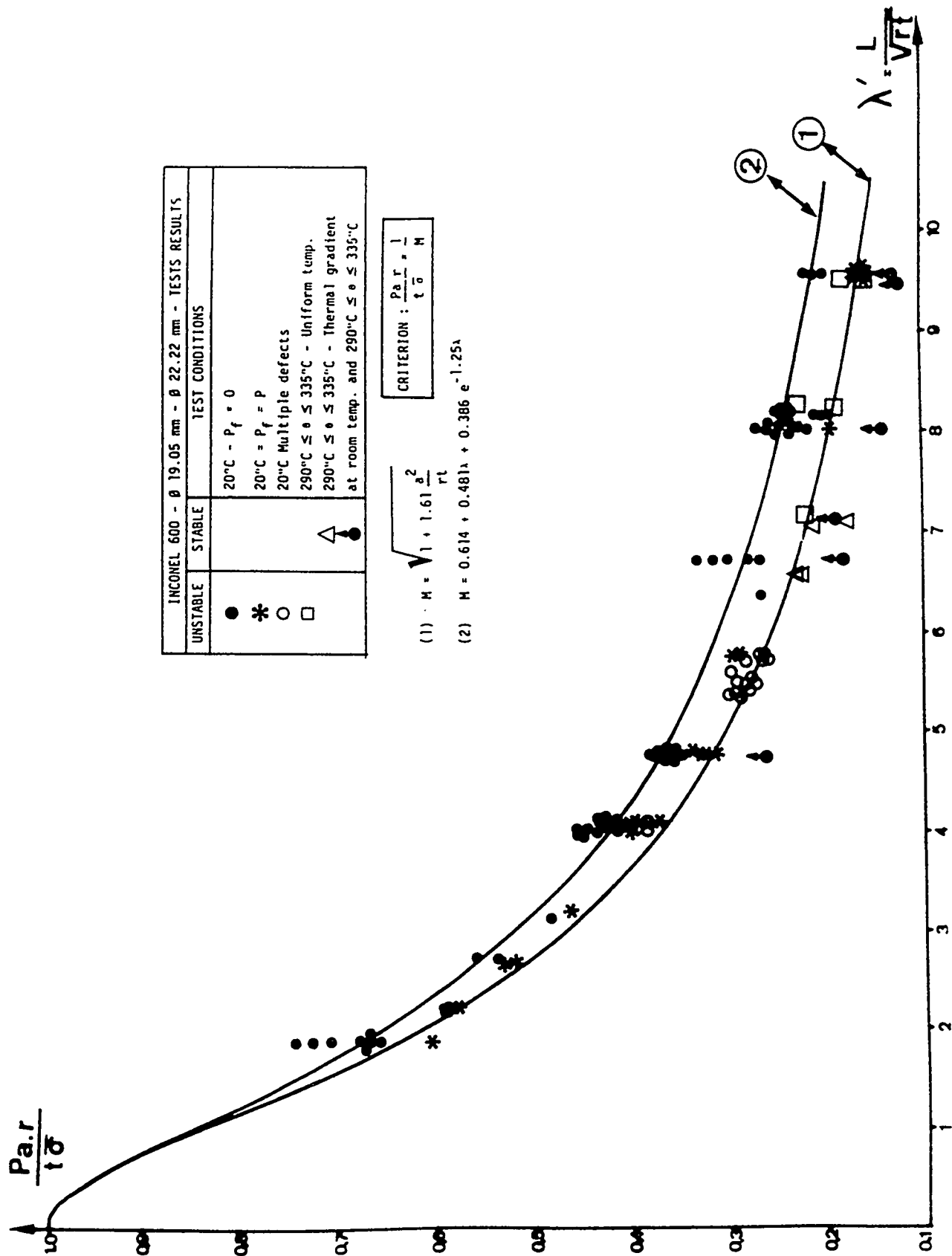
2. THE JET EFFECT AFTER BREAKTHROUGH OF THE TUBE.

IF INSTABILITY OCCURS, THE RUPTURE MAY BE DIVERTED CIRCUMFERENTIALLY.

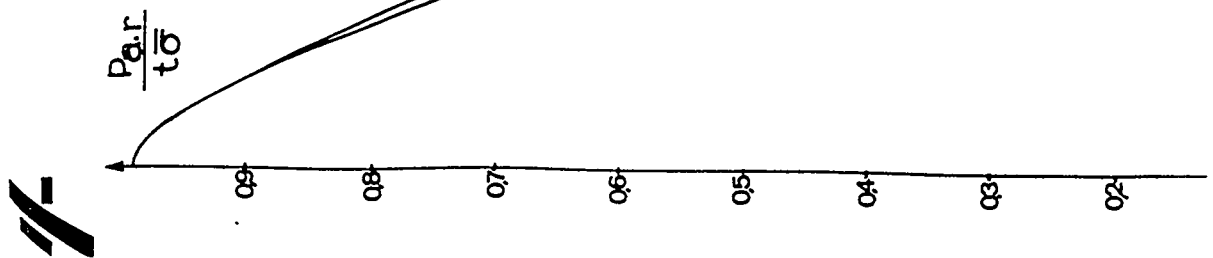
3. MAINTAINING A LARGE LEAK RATE THROUGH THE POTENTIALLY UNSTABLE CRACK.

THE TEST CONDITIONS (WITHOUT SEALING DEVICE) ALLOW THE ACTUAL PRESSURE TO BE EXERTED ON THE SIDES OF THE CRACK, THUS FAVORING INSTABILITY.

AXIAL THROUGH-WALL DEFECTS BURST PRESSURE OF SG TUBES AT ROOM TEMPERATURE AND AT ELEVATED TEMPERATURE



MULTIPLE PARALLEL AXIAL DEFECTS IN THE ROLL TRANSITION ZONE BURST TEST RESULTS ON ALLOY 600 TUBES



LEGEND (19.05 mm DIA.)							LEGEND (22.22 mm DIA.)						
SYMBOL	2a (mm)	HB (mm)	V (mm)	d/t HB %	NUMBER OF DEFECTS	X (mm)	SYMBOL	2a (mm)	HB (mm)	V (mm)	d/t HB %		
■	10	10	3	100	20	0	◇	12	8	5	100		
	15	15	3	100	20	0	◆	12	10	5	50		
✱	18	15	3	100	1 + 19	0	◇	12	10	5	70		
★	25	17	3	100	1 + 19	-7	○	12	10	4	100		
							⊕	12.2	12	5	100		
							⊗	14	10	5	100		
							▲	16	8	5	50		
							△	16	8	5	70		
							▽	16	8	5	100		
							●	16	8	4	100		
							●	16	10	5	50		
							●	16	10	5	70		
							□	16	10	5	100		
							□	16	10	4	100		
							□	16	12	5	100		

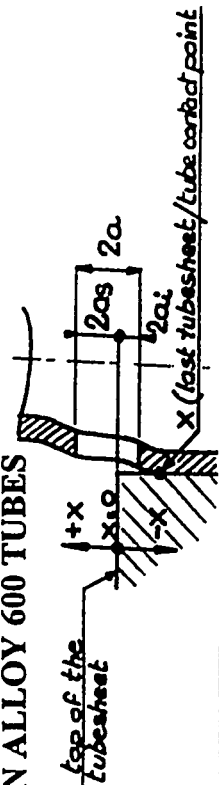
SKETCH		SYMBOL	2a (mm)	HB (mm)	V (mm)	d/t HB %
		◇	12	8	5	100
		◆	12	10	5	50
		◇	12	10	5	70
		○	12	10	4	100
		⊕	12.2	12	5	100
		⊗	14	10	5	100
		▲	16	8	5	50
		△	16	8	5	70
		▽	16	8	5	100
		●	16	8	4	100
		●	16	10	5	50
		●	16	10	5	70
		□	16	10	5	100
		□	16	10	4	100
		□	16	12	5	100

L = 2a : Through-wall defect
 HB : Through-wall or part-through defect
 d/t HB : Depth of defect (%)

(*) Last tubesheet/tube contact point

$$\lambda = \frac{2as}{\sqrt{r^2 - t^2}}$$

AXIAL THROUGH-WALL DEFECTS IN THE ROLL TRANSITION ZONE BURST TEST RESULTS ON ALLOY 600 TUBES



LEGEND (19.05 mm and 22.22 mm DIA.)

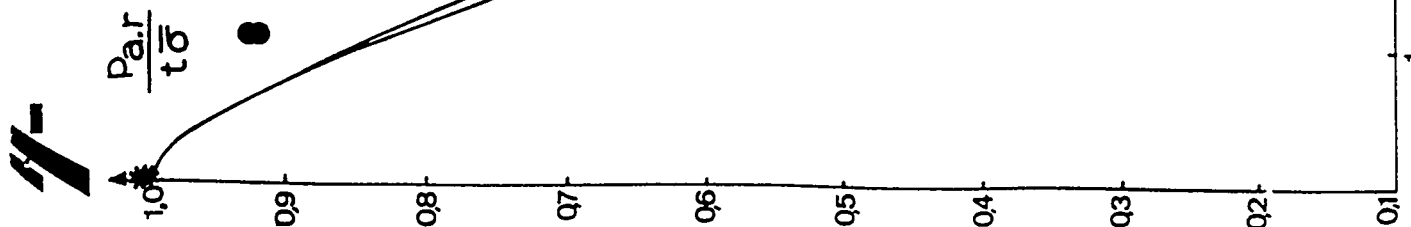
NUMBER OF TESTS	SYMBOLS	X (mm)	2a _i (mm)	2a _s (mm)	2a (mm)	NUMBER OF TESTS	SYMBOLS	X (mm)	2a _i (mm)	2a _s (mm)	2a (mm)
2	●	0	5	10	5	2	▲			13	13
5	○		10	15	10	2	△		0	14	14
3			13	16	13	1				15	15
2			14	17	14	5		-7	7	16	16
2			15	18	15	2				17	17
2			16	19	16	2				18	18
2			17	20	17	1				19	19
2			18	21	18	5				20	20
2			19	22	19	2				21	21
2			20	23	20	2				22	22
2			21	24	21	2				23	23
2			22	25	22	2				24	24
2			23	26	23	2				25	25
3	*	+2	10	15	10	3				13	13
3	★	-2	13	15	13	2				14	14
3	▲	-4	16	16	16	2				15	15
3	☆	-5	18	16	18	4		-10	10	16	16
3			20	20	20	2			13	13	13
2			25	25	25	3		-13	13	16	16
2			41	41	41	3			13	16	16

2a : Crack length inside the tubesheet
2a_i : Crack length above the top of the tubesheet
X : Last tubesheet/tube contact point

RUPTURE CRITERIA : Longitudinal through-wall crack in the roll transition zone.

$$\frac{Pa \cdot r}{t \bar{\sigma}} = \frac{1}{H (\lambda^* t_{ip})} \begin{cases} (1) H = [1 + 1.61 a_s^2 / (r t_{ip})]^{1/2} \\ (2) H = 0.614 + 0.481 \lambda + 0.386 e^{-1.23 \lambda} \end{cases}$$

NUREG/CP-0154





FLAW ACCEPTANCE CRITERIA

□ DEGRADATION INSIDE THE TUBESHEET

- Type of flaw : Axial part-through and through-wall flaw
- Location : Upper tip of the flaw located at the top of the tubesheet
- Flaw length : Up to 200mm
- Max. tube/tubesheet gap : 0.64mm

TEST RESULTS

- ☞ Test results shows that the burst pressure for tubes with axial part-through flaws can be obtained using a plastic flow instability criterion
- ☞ After breakthrough , the potentially instable flaw does not propagate and remains confined within the tubesheet



FLAW ACCEPTANCE CRITERIA

□ DEGRADATION IN THE ROLL TRANSITION ZONE

• TYPES OF DEGRADATION MECHANISMS

- | | |
|------------------------------------|---------------------------|
| — PRIMARY SIDE CORROSION (PWSCC) | ☞ THE MAIN CONCERN |
| — SECONDARY SIDE CORROSION (IGSCC) | ☞ IN CONTINUING EVOLUTION |

• PLUGGING CRITERION BASED ON CRACK DEPTH MEASUREMENT

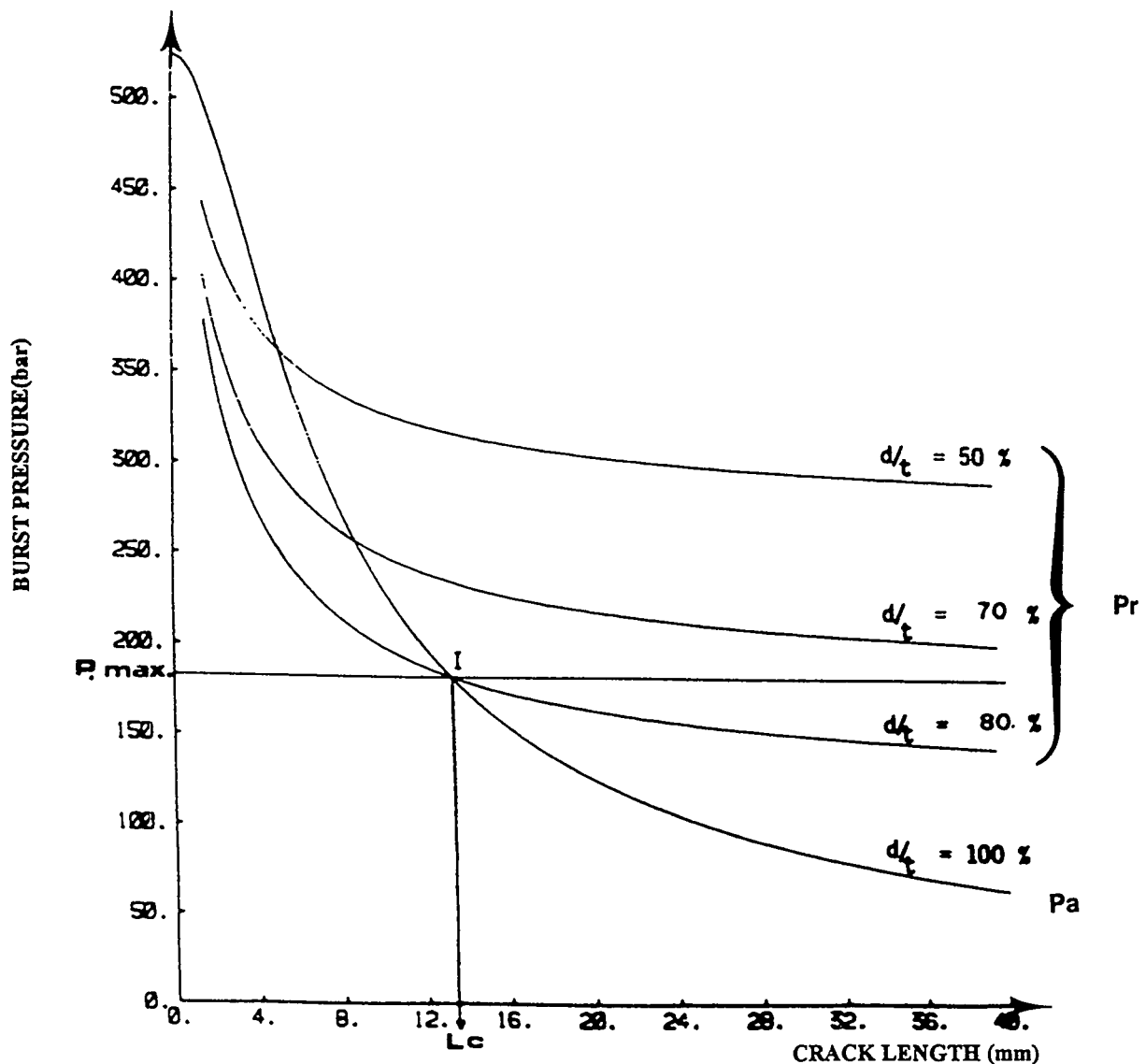
THE 40% PLUGGING CRITERION IS NOT APPROPRIATE FOR THE FOLLOWING REASONS :

- THE AVAILABLE IN-SERVICE INSPECTION METHODS ARE INSUFFICIENT TO DETERMINE WITH A GOOD ACCURACY THE CRACK DEPTH IN THE TUBE THICKNESS DIRECTION,
- THE SAFETY MARGINS IN TERM OF DEPTH WITH RESPECT TO TUBE BREAKTHROUGH OR CRITICAL CRACK DEPTH ARE SMALL : A CRACK CAN BECOME THROUGH-WALL IN A CYCLE,
- IF FOR ANY REASON , THE LENGTH OF THE ACCEPTABLE 40% PART-THROUGH CRACK IS LONGER THAN THE CRITICAL CRACK LENGTH, THE RISK TO GET AN UNSTABLE CRACK IN THE EVENT OF ACCIDENTAL OVERPRESSURE CANNOT BE RULED-OUT.



FLAW ACCEPTANCE CRITERIA

- **PLUGGING CRITERION BASED ON CRACK LENGTH MEASUREMENT**
 - **AXIAL CRACKS : DETERMINATION OF THE MINIMUM CRITICAL CRACK DIMENSIONS (L_C) CORRESPONDING TO POSTULATED ACCIDENT CONDITIONS : SLB - FLB**



- **PLUGGING CRITERION**

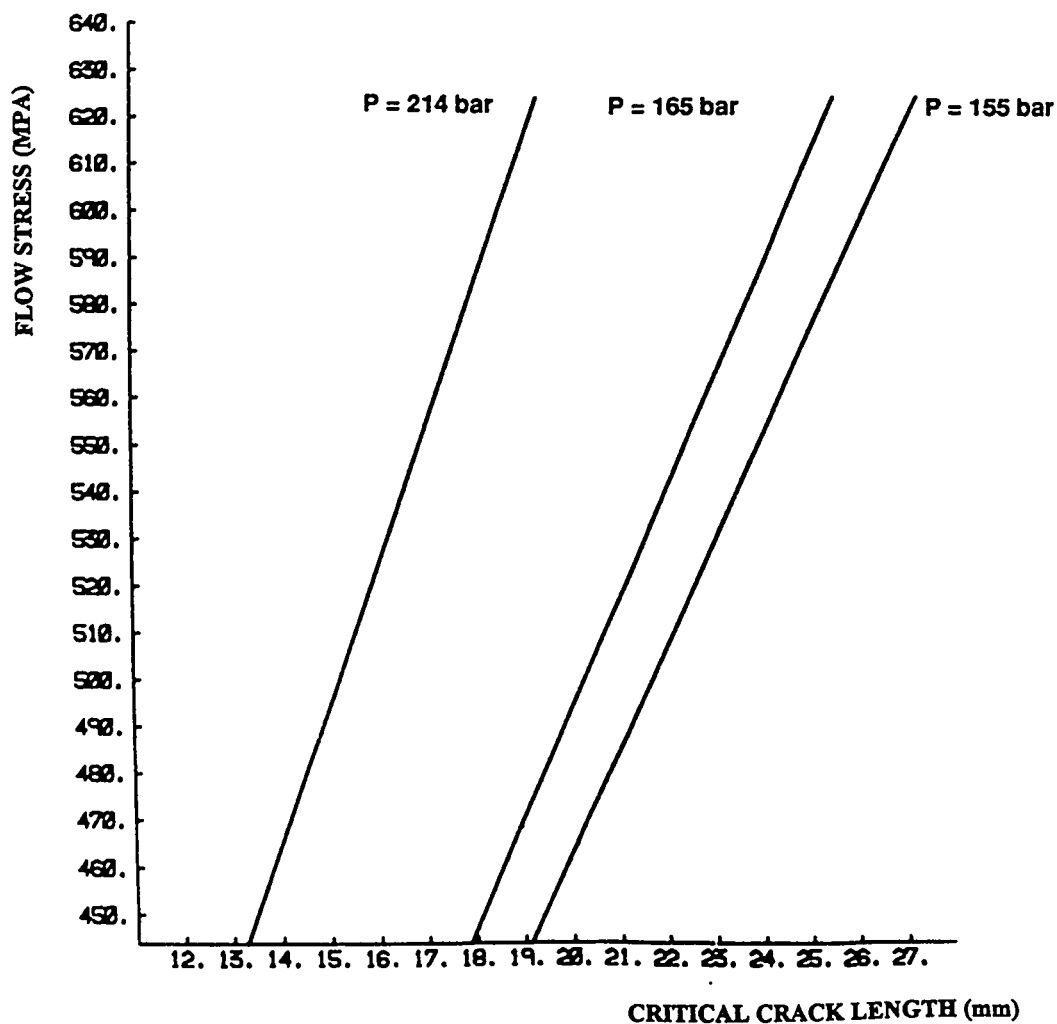
$$L(\text{PLUGGING}) = L_C - L_P - E$$



FLAW ACCEPTANCE CRITERIA

- INFLUENCE OF MATERIAL PROPERTIES ON THROUGH-WALL CRITICAL CRACK LENGTH

- TUBE DIMENSIONS $\left\{ \begin{array}{l} D = 22.22 \text{ mm} \\ t = 1.27 \text{ mm} \end{array} \right.$
- LOCATION: STRAIGHT SECTION





FLAW ACCEPTANCE CRITERIA

• CIRCUMFERENTIAL CRACKING AT THE TOP OF THE TUBESHEET

1. Type : OD or ID circumferential cracks

2. Root Causes :

- Rolling anomalies ,
- Interaction between the tube and the lower tube support plate or flow distribution baffle plate ,
- Denting at the top of the tubesheet : Crud build up in the crevice or Sludge with metallic particles which oxidate and form a hard and resistant crust which imprisons the tubes in the roll transition zone ,
- Sludge deposit areas : The sludge pile affects heat transfer through tube walls which modifies the distribution and magnitude of the thermal stresses in the roll transition zone ,
 - Increase in wall temperature
 - Deterioration of the local chemical conditions on the secondary side
- Tube support plate blockage : tubes are locked at the TSP location .

1-

FLAW ACCEPTANCE CRITERIA

3. Circumferential crack - Burst test results :

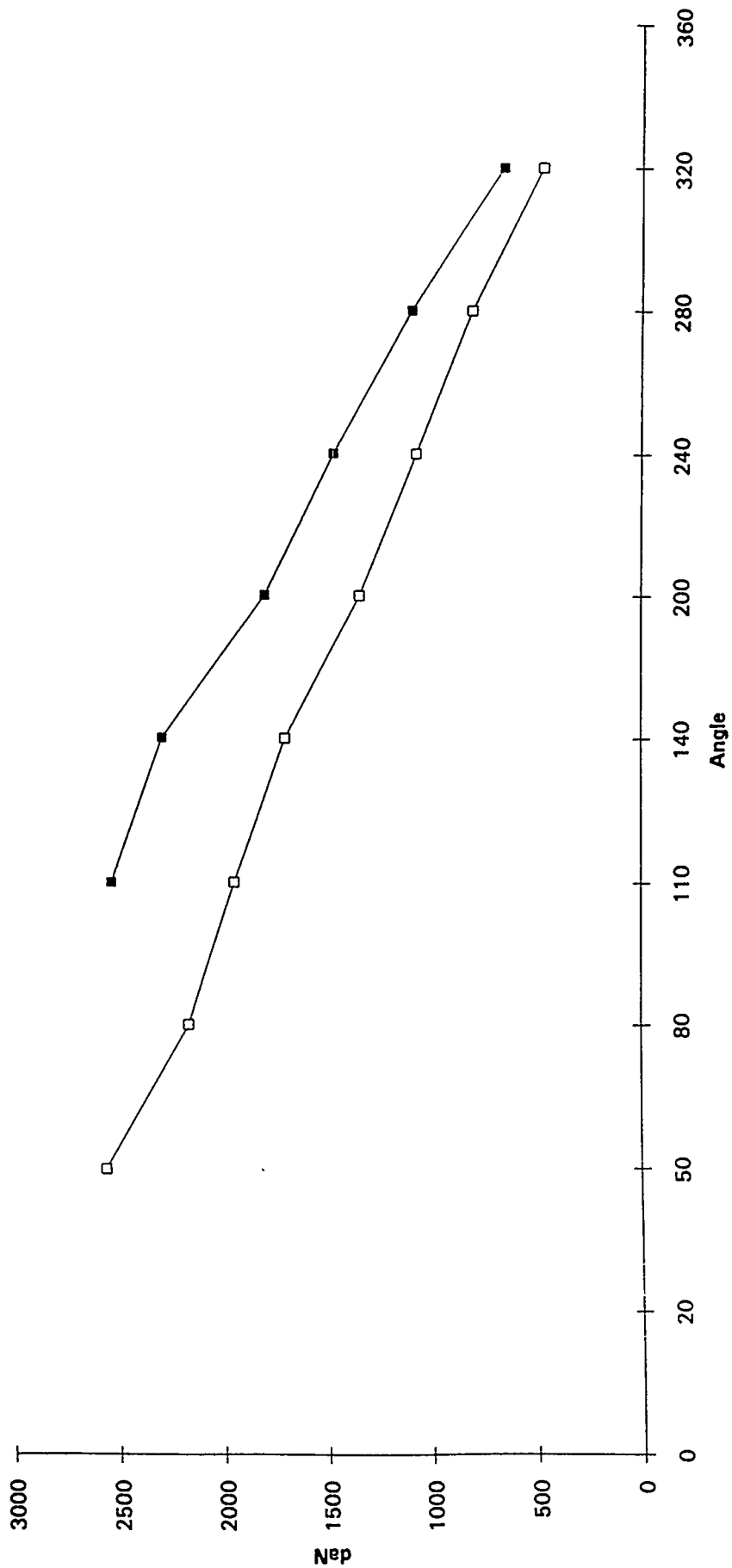
- THE END EFFECT EXERTED BY INTERNAL PRESSURE AND AXIAL LOADS INDUCE TUBE BENDING WHICH INCREASES WITH THE LENGTH OF THE DEFECT.
- FOR AN ANGLE GREATER THAN 250° , THE BENDING MOMENT IS NO LONGER DOMINANT AND RUPTURE OCCURS DUE TO TENSION OF THE NET SECTION

BURST PRESSURE OF A CIRCUMFERENTIAL FLAW				
TUBE DIMENSIONS	TEST CONDITIONS	DEFECT ANGLE Degree	TEST RESULTS Pr(bar)	PREDICTION Pa(bar)
22.22 x 1.27mm	room	250	490	500
19.05 x 1.09mm	room	250	506	518
22.22 x 1.27mm	265°C	320	148	162

- A CIRCUMFERENTIAL 90° DEFECT , THROUGH-WALL OR PART-THROUGH , REMAINS STABLE.
- A CIRCUMFERENTIAL PART-THROUGH DEFECT ,WHOSE DEPTH IS LESS THAN OR EQUAL TO 50% OF THE TUBE WALL THICKNESS OVER 360° REMAINS STABLE.



CIRCUMFERENTIAL THROUGH-WALL FLAWS - TEST RESULTS : TENSILE STRENGTH





FLAW ACCEPTANCE CRITERIA

□ DEGRADATION AT TUBE/TSP INTERSECTIONS : AXIAL CRACKS

- THE TSP HAS NO INFLUENCE ON THE RUPTURE OF THE REMAINING LIGAMENT OF THE PART-THROUGH FLAW.
RUPTURE CONFORMS TO PREDICTIONS MADE USING CRITERIA VALIDATED FOR CRACKS AWAY FROM SUPPORTS.
- AFTER BREAKTHROUGH OF THE TUBE, THE SUPPORT PLATE LIMITS BULGING OF THE CRACKED TUBE ONCE THE GAP BETWEEN THE TUBE AND THE SUPPORT PLATE HAS BEEN FILLED.
- RUPTURE OF A TUBE CRACKED AT SUPPORT PLATE LEVEL CANNOT OCCUR UNDER NORMAL OR ACCIDENT CONDITIONS SO LONG AS THE DEGRADED REGION IS LIMITED TO THE THICKNESS OF THE SUPPORT PLATE.
- SO LONG AS THE FREE LENGTH OF THE CRACK OUTSIDE THE SUPPORT PLATE REMAINS LIMITED TO 11MM, INSTABILITY OF A CRACK WITH A MAXIMUM LENGTH OF 25MM CANNOT BE PRODUCED BY THE MAXIMUM OVERPRESSURE OF 182.5 bar.
- THE PRESENCE OF MULTIPLE STRESS CORROSION CRACKS ALL ROUND THE TUBE CIRCUMFERENCE HAS NO EFFECT ON THE BEHAVIOR OF THE POTENTIALLY UNSTABLE CRACK (LONGEST CRACK).

TUBE / TUBE SUPPORT PLATE INTERACTION

BURST TEST RESULTS AT ELEVATED TEMPERATURE
ALLOY 600 - OD : 19.05mm

TEST No	TYPE OF DEFECT	MAIN DEFECT OR CRACK GEOMETRY		TEST RESULTS			PREDICTIONS - RIFTURE CRITERION: THROUGH-WALL CRACK AWAY FROM SUPPORTS $P_a = \frac{t}{r} \frac{\sigma}{M}$ Where $M = \sqrt{1 + 1.61 \frac{a^2}{rt}}$ (bar)
		INITIAL LENGTH (mm)	DEPTH d/t (%)	MAX. PRESSURE (bar)	TEMPERATURE (°C)	STABILITY (WITH T.S.P.)	
6M	Multiple corrosion cracks + 1 part-through defect	25.1	87.7 then through-wall	182	346	Stable	139
7M	Multiple corrosion cracks	20.8	Partially through-wall	186	328	Stable	164
7H	Multiple corrosion cracks	18.1	through-wall	185	346	Stable	186
6H	Multiple corrosion cracks	24.3	through-wall	183	347	Stable	141
7B	Multiple corrosion cracks + 1 part-through defect	25.06	86 then through-wall	185	350	Stable	139
369	1 part-through defect	25.01	89.8 then through-wall	184	338	Stable Final crack length : 25.09 mm	128
373	1 part-through defect	19.01	84 then through-wall	191	348	Stable	167
6B	Multiple corrosion cracks + 1 part-through defect	25.02	86.4 then through-wall	146	340	Unstable (without T.S.P.)	140





FLAW ACCEPTANCE CRITERIA

■ CONCLUSION

- FROM BURST TESTS ON ALLOY 600 TUBING, IT IS DEMONSTRATED THAT PLASTIC INSTABILITY RUPTURE CRITERIA ARE RELIABLE TO SAFELY PREDICT THE BURST PRESSURE OF DAMAGED TUBES OR TO DETERMINE THE CRITICAL CRACK DIMENSIONS.

RESULTS OF BURST TESTS ON PULLED OUT SG TUBES DO NOT CALL INTO QUESTION THE MECHANICAL STENGTH CRITERIA.

- THE KNOWLEDGE OF THE LENGTH OF FLAW IS ONE OF THE MOST IMPORTANT FACTOR TO BE CONSIDERED FOR ESTABLISHING PLUGGING CRITERIA.

- BURST TESTING ON DEGRADED TUBES PULLED FROM SGs IS A KEYSTONE FOR THE QUALIFICATION OF REPAIR LIMITS APPLICABLE TO COMPLEX DEFECT MORPHOLOGIES.
CORRELATIONS CAN BE ESTABLISHED BETWEEN BURST/LEAK TEST DATA AND CORRESPONDING EDDY CURRENT BOBBIN COIL SIGNAL AMPLITUDE.

ALTERNATE TUBE PLUGGING CRITERIA FOR STEAM GENERATOR TUBES

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1 BACKGROUND

The tubing of the Steam Generators constitutes more than half of the reactor coolant pressure boundary. Specific requirements governing the maintenance of steam generator tubes integrity are set in Plant Technical Specifications and in Section XI of the ASME Boiler and Pressure Vessel Code.

The operating experience of Steam Generator tubes of PWR plants has shown the existence of some types of degradatory processes. Every one of these has an specific cause and affects one or more zones of the tubes. In the case of Spanish Power Plants, and depending on the particular Plant considered, they should be mentioned the Primary Water Stress Corrosion Cracking (PWSCC) at the roll transition zone (RTZ), the Outside Diameter Stress Corrosion Cracking (ODSCC) at the Tube Support Plate (TSP) intersections and the fretting with the Anti-Vibration Bars (AVBs) or with the Support Plates in the preheater zone.

The In-Service Inspections by Eddy Currents constitute the standard method for assuring the SG tubes integrity and they permit the monitoring of the defects during the service life of the Plant. When the degradation reaches a determined limit, called the plugging limit, the SG tube must be either repaired or retired from service by plugging.

Customarily, the plugging limit is related to the depth of the defect. Such depth is typically 40% of the wall thickness of the tube and is applicable to any type of defect in the tube, when in its origin, that limit was established for tubes thinned by wastage, which was the predominant degradation in the seventies. The application of this criterion for axial crack-like

defects, as, for instance, those due to PWSCC in the roll transition zone, has lead to an excessive and unnecessary number of tubes plugged.

For this reason, it became evident the necessity for developing defect specific plugging criteria, based on both experimental tests and analytical calculations, which were able to assure the integrity of the pressure boundary.

As an evidence of this trend towards the application of defect specific plugging criteria, there were the experiences in European Countries, and more recently some initiatives undertaken in the U.S. Moreover, the latest editions of the Section XI of the ASME code include the article IWB-3630, according to which tubes with defects whose depth is greater than the acceptance standards may be left in service if the evaluation by analyses is acceptable to the Regulatory Authority having jurisdiction at the plant site.

2 PWSCC IN THE ROLL TRANSITION ZONE

A comprehensive experimental program was performed in France in order to validate a plugging criterion for this type of degradation based on the leak before break (LBB) approach, or more precisely the leak before risk of break (LBRB), which has been applied for several years. With the support of the french manufacturer and utility similar criteria were developed for the Spanish Plants. With this aim, specific experimental program and analytical studies were performed in Spain. The methodology is shown in Fig. 1.

The criteria are based also in the LBB concept which takes advantage of the ductile inherent-safe behaviour of the axial cracks in Alloy 600 tubes. Unlike the french standard practice, however, the approach followed in Spain relies more in NDT inspections. In fact, 100% of the roll transition zone susceptible to PWSCC is inspected every outage and the plugging limits account for crack evolution during the cycle as well as the crack size uncertainty by EC.

Primary to secondary leak rates are monitored during operation (including N-16 detectors) and must remain below a conservative limit (see below) established by the Consejo de Seguridad Nuclear (CSN), the Regulatory Authority in Spain.

Generally speaking the practice followed in Spain conforms the latter EPRI inspection-based approach, in which consideration of leak before break is made in order to obtain additional assurance of safety.

2.1 Determination of the critical crack size

The steps covered are as follows:

- Analysis of operational transients

The more severe conditions for the tube are the Feedwater Line Break (FLB) and the Main Steam Line Break (MSLB). The secondary side of the SG was assumed to depressurize instantaneously in order to maximize the differential pressure on the tube.

- Tube Rupture Criterion

The tube burst tests performed for the more representative configurations of cracks validated a plastic instability criterion, whose expression is:

$$\sigma_o = \sigma M^*$$

where:

σ_o is the flow stress

σ is the hoop stress

M^* is the Folias bulging factor, corrected to account for the reinforcement provided by the tubesheet which extends up to 18 mm above it.

- Mechanical properties

Yield and ultimate strength for the tube bundles were evaluated at the maximum temperature during FLB transient. The values used in the critical crack size evaluations were based in the minimum values specified in the ASME Code at room temperature and conservatively corrected to 343 °C (650 °F), what led to yield and ultimate strength lower than those specified at that temperature in the ASME Code. In statistical terms, the values selected for the calculations were lower than the mean minus three times the standard deviation of the distributions of mechanical properties in the tube bundles.

- Tube geometry

The geometry of tube selected for the analysis (maximum outside diameter and minimum thickness based on manufacturing tolerances) maximized the hoop stress in the tube.

- Results

All the hypothesis above were assumed to occur simultaneously what yielded conservative critical crack sizes.

The axial critical crack size in the roll transition zone when the rolling is in accordance with manufacturing tolerances, is 13 mm. The critical crack length above the tubesheet when the expanded zone of the tube ends up to 7 mm below the top of the tubesheet is 12.6 mm.

2.2 Leak rate evaluation

According to the leak rate tests performed through corrosion cracks in Alloy 600 tubes, it was shown that the flow rate coefficient K can be determined from the following expression:

$$Q = KS \sqrt{\frac{2\Delta p}{\rho}}$$

where:

- Q is the leak rate
- S is the crack opening area, numerically calculated on basis to finite element results.
Correction for plastic zone was applied
- ρ is the fluid density
- Δp is the differential pressure

The scatter of the test results was relatively large. A lower bound of the leak flow rate could be obtained with $K = 0.10$.

The primary to secondary leak rate evaluation through the critical crack by means of the above criterion was calculated to be 30 liters/hour. However, in practice, the allowable leakage rates for the Spanish NPPs are lower than the calculated limit (See 2.3.2).

2.3 Surveillance requirements

2.3.1 Surveillance requirements during shutdown

In-Service Inspection by Eddy Currents of 100% of tubes in the roll transition zone are performed at each outage.

- Inside the tubesheet, the plugging criterion is the P^* . Axial cracks are allowed in this zone. Circumferential cracks below the P^* distance from the top of the tubesheet are permitted. Typically, the P^* distance is 38 mm or greater for most of the non-peripheral tubes.
- In the roll transition zone, extending up to 18 mm above the tubesheet, the maximum acceptable lengths for axial PWSCC are as follows:

$L_a = 8$ mm for correct rolling

$L_a = 7.6$ mm for rolling elevation up to 7 mm below the top of tubesheet.

The plugging criterion above was obtained by subtracting from the critical crack lengths the crack growth during cycle (4 mm) and the uncertainty in crack sizing (1 mm).

- Other zones of the tube. All indications other than the defect type and location specific defects discussed above are judged against the 40% wall thickness criterion.

2.3.2 Surveillance requirements during plant operation

Primary to secondary leak rate is monitored during operation, including N-16 detectors. The limit established by the Consejo de Seguridad Nuclear (CSN) the Regulatory Authority in Spain, is 5 liters/hour above a steady leak at the beginning of cycle not higher than 5 liters/ hour. Therefore, the allowable total leakage, is lower than the calculated value of 30 liters/hour.

3 ODSCC AT THE TUBE SUPPORT PLATE INTERSECTIONS

Another type of degradation which is the cause of the plugging of an important number of tubes, is the outside diameter stress corrosion cracking (ODSCC) at the Tube Support Plates (TSPs) intersections.

The justification of a plugging criterion for this type of degradation, has required the fulfilment of another experimental programme including burst and leak tests, and metallographic examination of pulled tubes and model boiler specimens, as well as analytical calculations.

Before, the EPRI undertook its work towards an APC, a plugging criterion which significantly increased the allowable depth of the crack inside the Support Plate thickness was developed. Later, the EPRI approach was followed, which essentially consists in defining the

plugging limit in terms of the amplitude of the indication by Eddy Currents, regardless of its depth. The collaboration with the EPRI has made possible an exchange of data, thus increasing the experimental data base of the criterion. For this reason part of the data base points on which the EPRI criterion is based, belongs to tubes pulled from Spanish Plants and model boiler specimens representative of the degradation found in these plants.

The methodology and Surveillance Requirements are shown in Figures 2 and 3 respectively.

Recently the U.S. Nuclear Regulatory Commission (NRC) has issued the Generic Letter 95-05 to give guidance to licensees who may wish to request a licence amendment to plant technical specifications to implement voltage-based alternative steam generator tube repair criteria for this type of degradation.

4 DEFECT DUE TO FRETTING WITH ANTIVIBRATION BARS (AVBs)

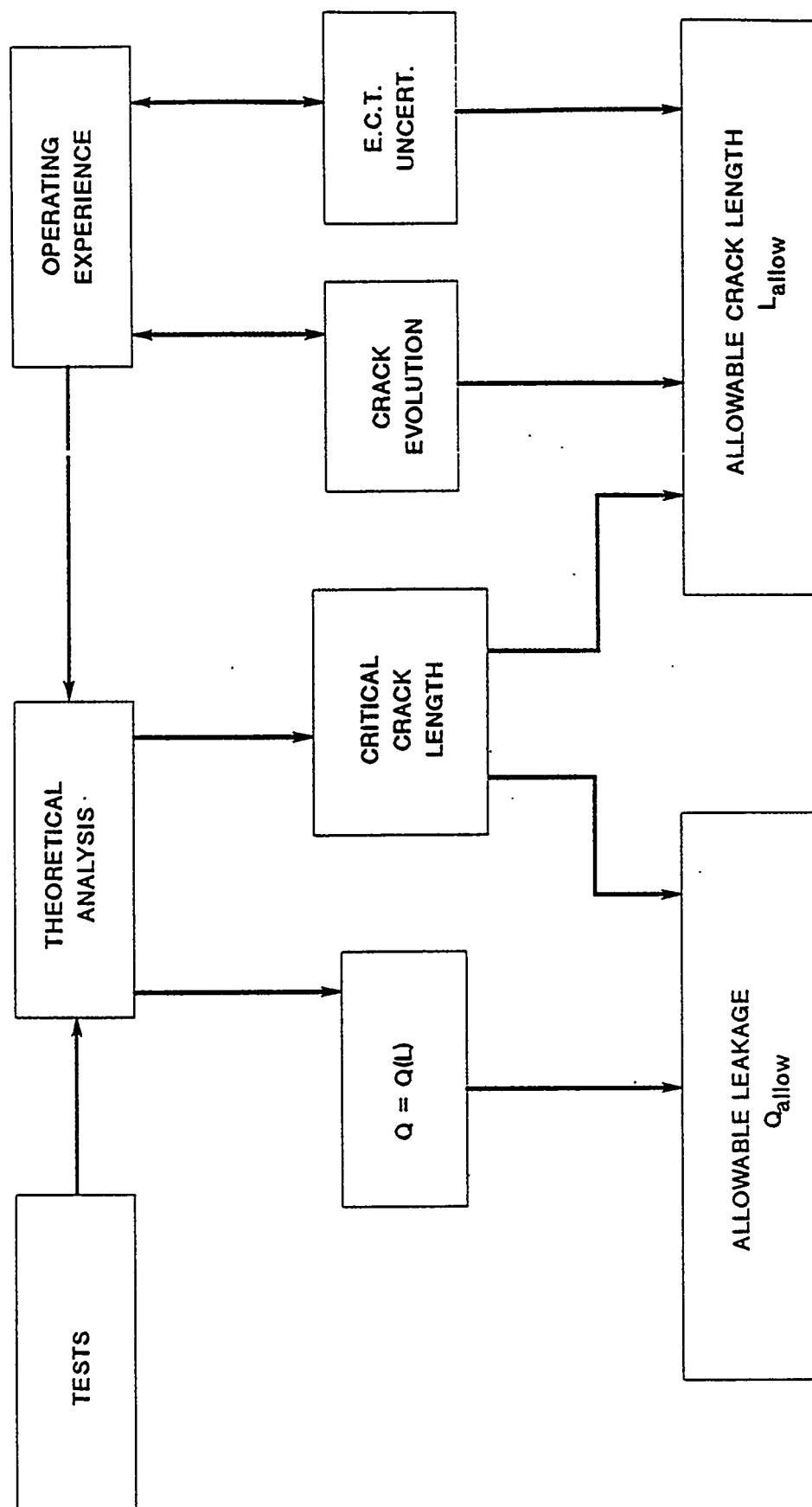
Finally, in order to deal with the locally thinned tubes by fretting with AVBs and with Support Plates in the preheater zone, another plugging criterion was developed. As in other types of degradation, the criterion is supported by tests, which confirm the theoretical models used in the calculations, whose aim is to assure the integrity of the pressure boundary. The methodology used is shown in Fig. 4.

5 REFERENCES

1. USNRC Regulatory Guide 1.121, 1976, "Bases for Plugging Degraded PWR Steam Generator Tubes". Draft for Comment, August.
2. Cochet, B. 1988, Application fo the Leak Before Break Concept to Steam Generator Tubes, NuSAFE 88, Avignon, France, October.

3. Cueto-Felgueroso C. and García Alonso J.M., 1989, Plugging Criteria for SG Tubes in Model D3 Steam Generators Based on the Leak-Before-Break Criterion Applicable to Stress Corrosion Cracking, Tecnatom report: EPRI-CT-D3, July.
4. Committee for Alternate Repair Limits for EZ PWSCC 1993, Technical Support Document for Expansion Zones PWSCC in Roll Transition, Draft Report, EPRI NP-6864-L Rev. 2, August.
5. Committee for Alternate Repair Limits for ODSCC at TSPs, 1993, PWR Steam Generator Tube Repair Limits- Technical Support Document for Outside Diameter Stress Corrossion Cracking at Tube Support Plates, EPRI TR-100407 Rev. 1 Draft Report, August.
6. USNRC Generic Letter 95-05, 1995 "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking", August.

PLUGGING CRITERION FOR PWSCC IN R.T.Z.- METHODOLOGY



TECNATOM, S.A.

FIGURE 1

CALCULATION OF PLUGGING LIMIT FOR ODSCC AT TSPs - METHODOLOGY

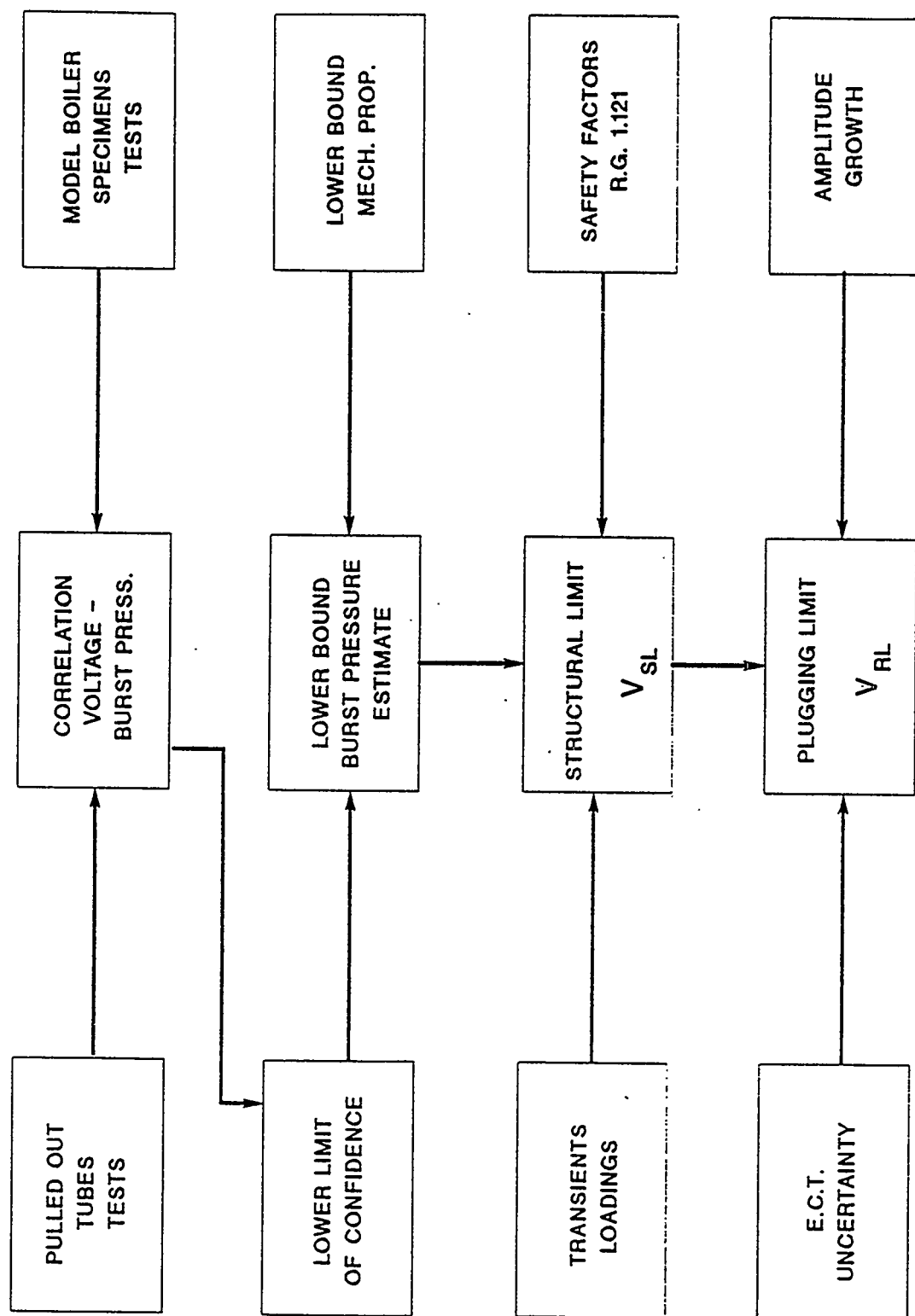
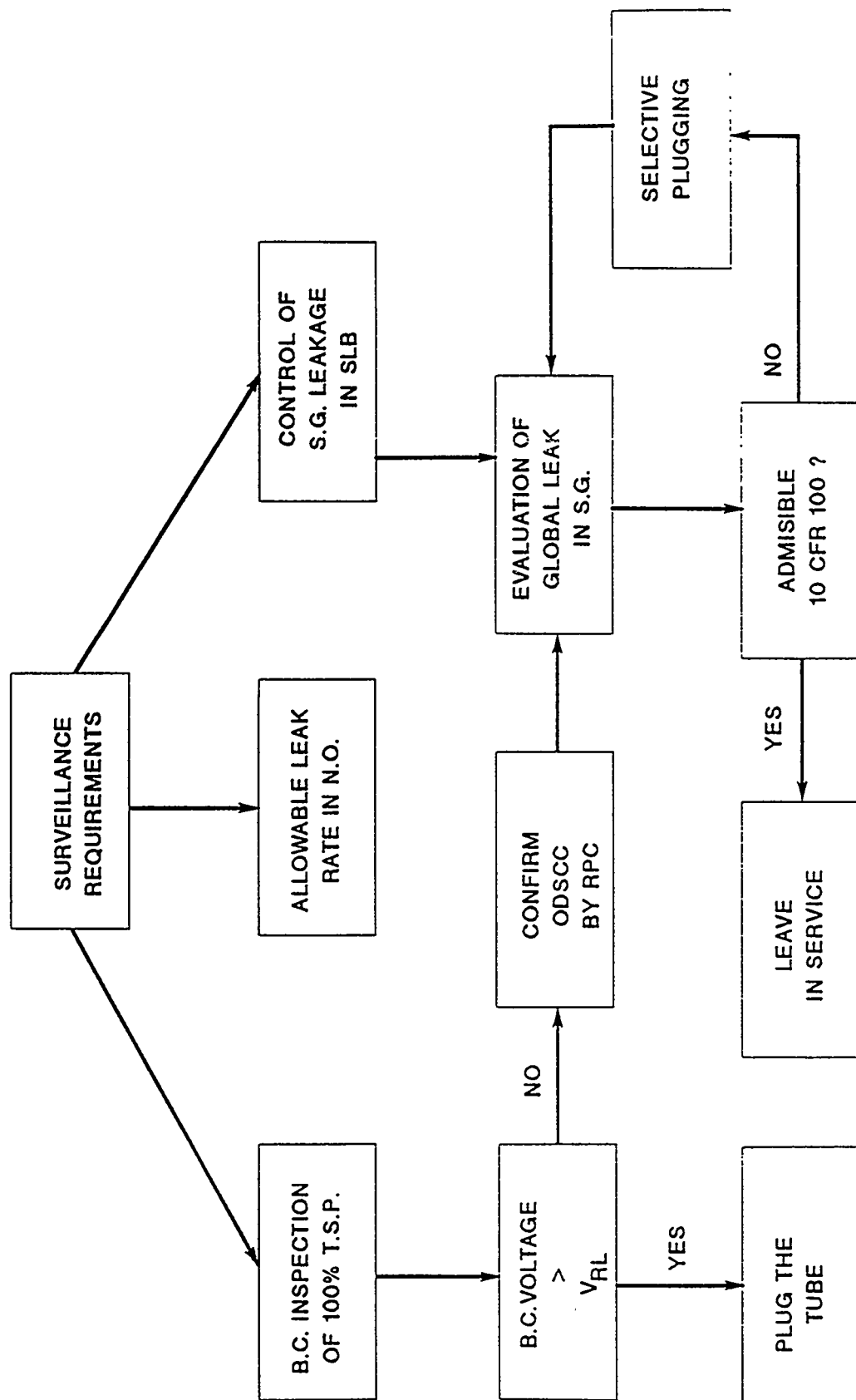


FIGURE 2

TECNATOM, S.A.

SURVEILLANCE REQUIREMENTS – ODSCC AT TSPs



TECNATOM, S.A.

FIGURE 3

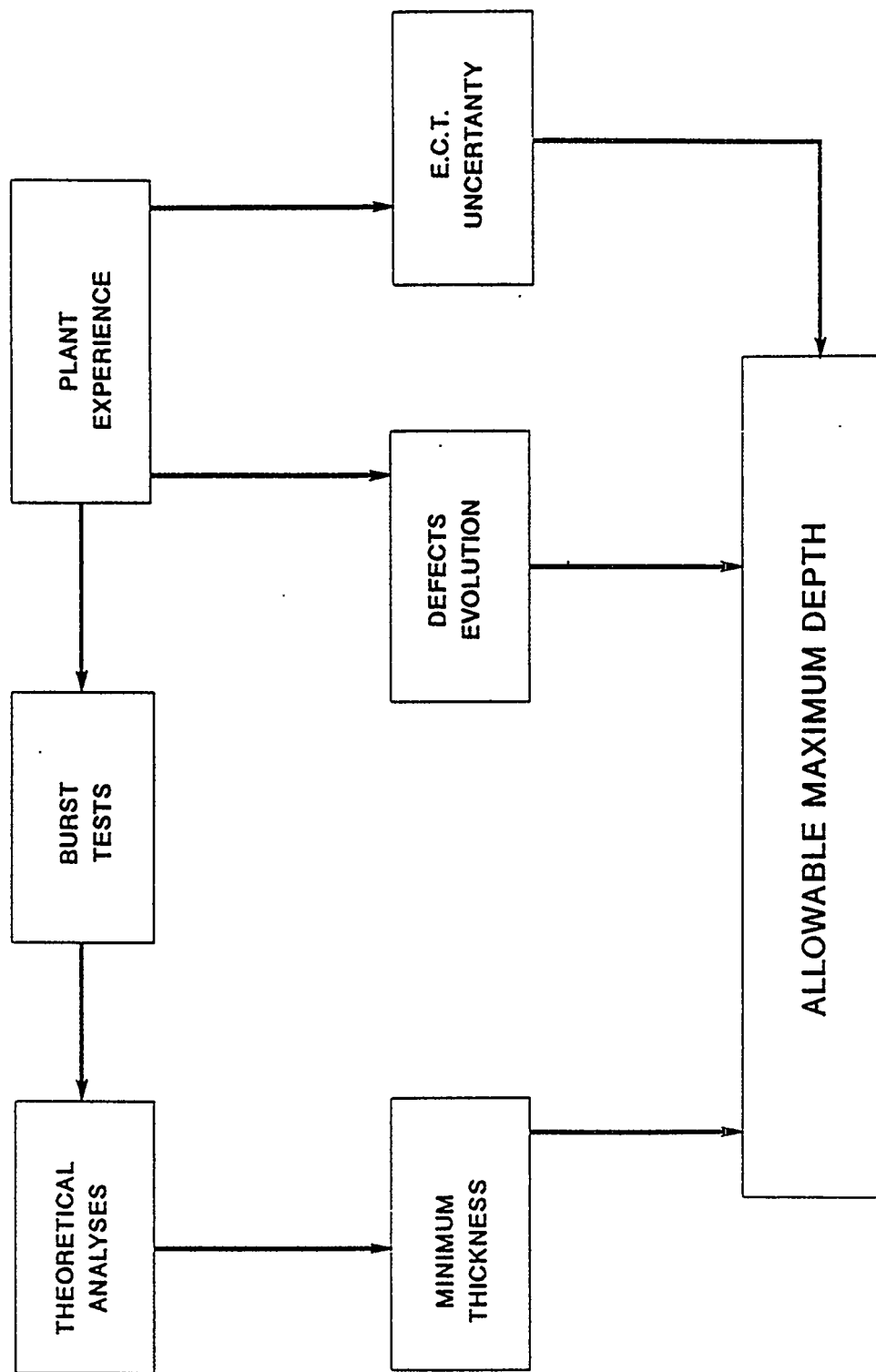


FIGURE 4

TECNATOM, S.A.

**CNRA / CSNI International Workshop on
Steam Generator Tube Integrity Issue**

Reliability of Steam Generator Tubing

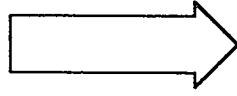
Ei Kadokami

Mitsubishi Heavy Industries, Ltd.

Oct. 30~Nov. 2, 1995

Objective

- All SG tubes, in which any defect was detected, should be repaired either by plugging or sleeving in Japanese practice
- However, there exists some possibility that undetected defect, whose size is small as compared with the limit of detectability of current NDE technology, may propagate under plant operating condition



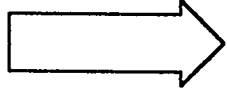
- To confirm the reliability and safety of degraded SG tubes

Test / Investigation Program

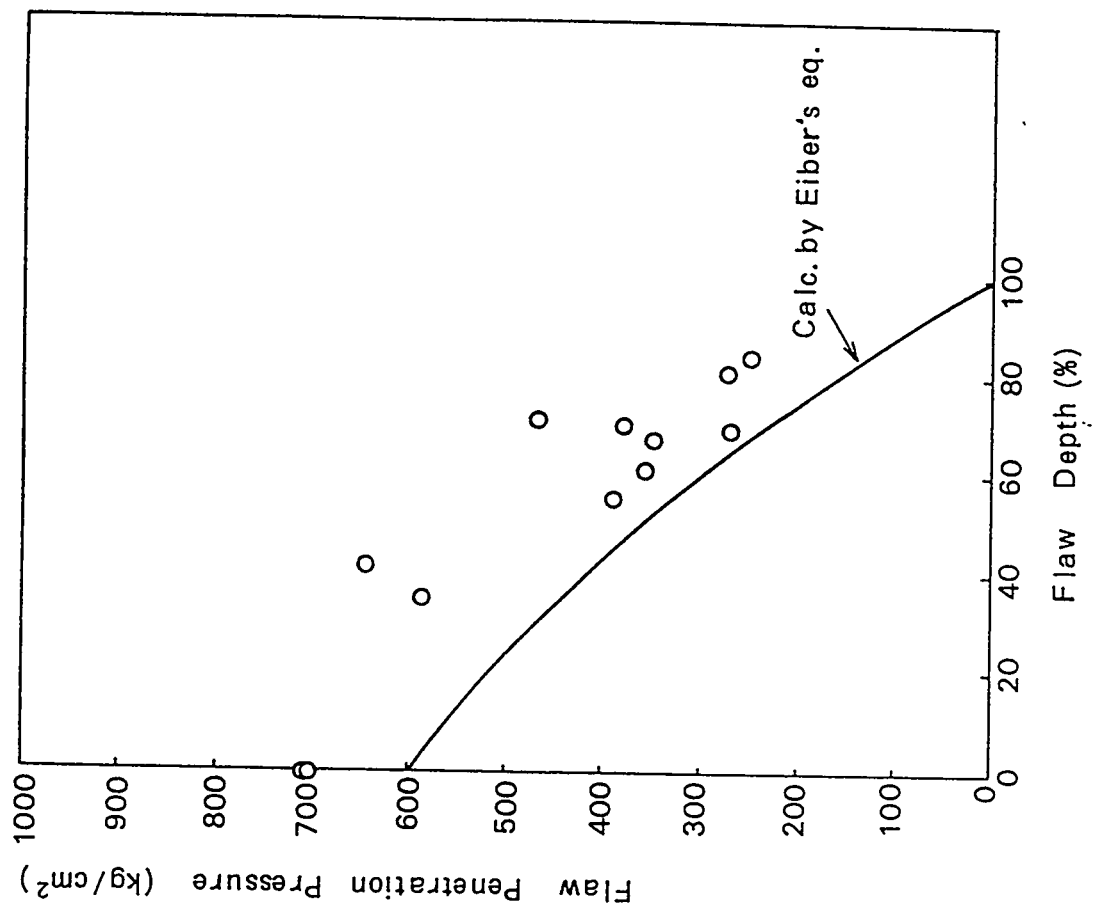
- **Assessment of penetration and opening behavior of tube flaws by internal pressure**
 - Penetration behavior of the tube flaws by internal pressure
 - Primary water leakage behavior from the through - wall flaw
 - Opening behavior of the through - wall flaw
- **Assessment of reliability of the tube with flaws during plant operation**
- **Assessment of behavior for the case of tube rupture accident**

o Penetration Behavior of the Tube Flaws by Internal Pressure

- Test piece tube with part - through IGA was internally pressurized



- Correlation between penetration pressure and initial flaw depth was obtained
- It was confirmed that Eiber's equation was applicable to assess the flaw penetration behavior by internal pressure



Flaw Penetration Pressure and Flaw Depth

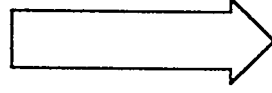
o Primary Water Leakage Behavior from the Through - Wall Flaw

- Test piece tube with through - wall IGA was internally pressurized

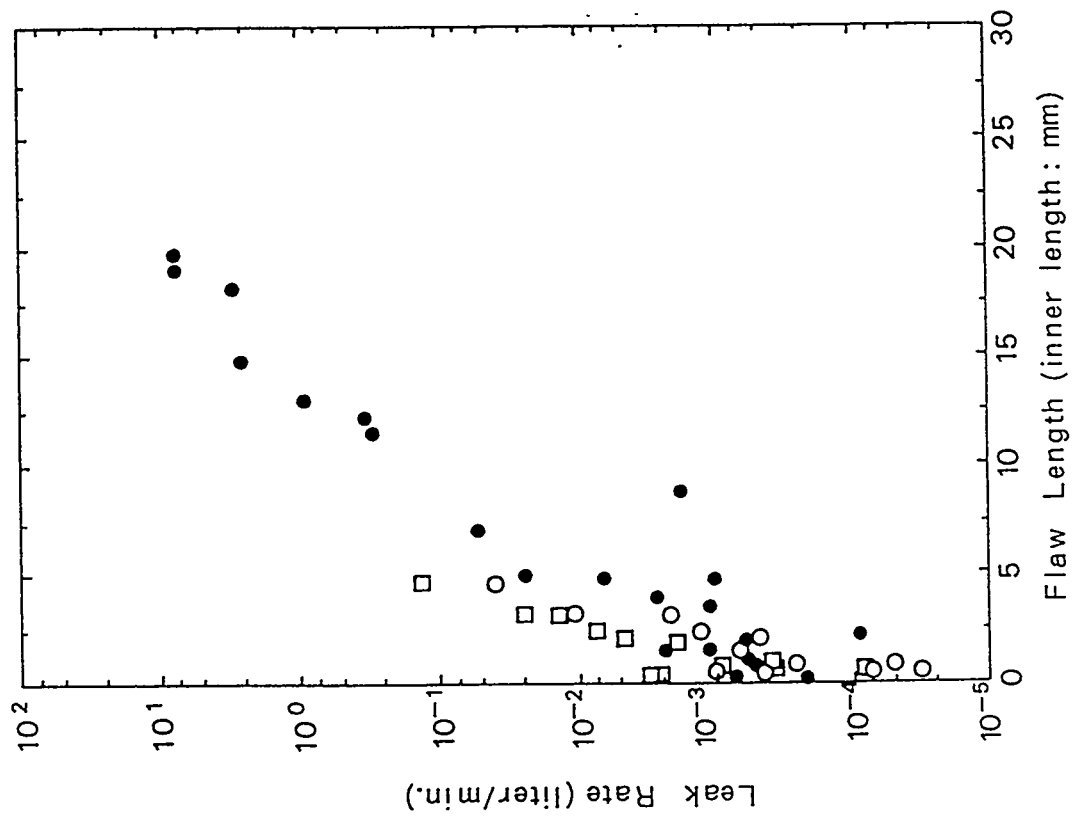
- Leak rate was measured at pressure difference for

normal operation ($\Delta P=100\text{kg/cm}^2$)

MSLB condition ($\Delta P=180\text{kg/cm}^2$)



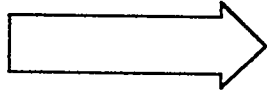
- Leak rate increases for longer crack length, but does not increase with time



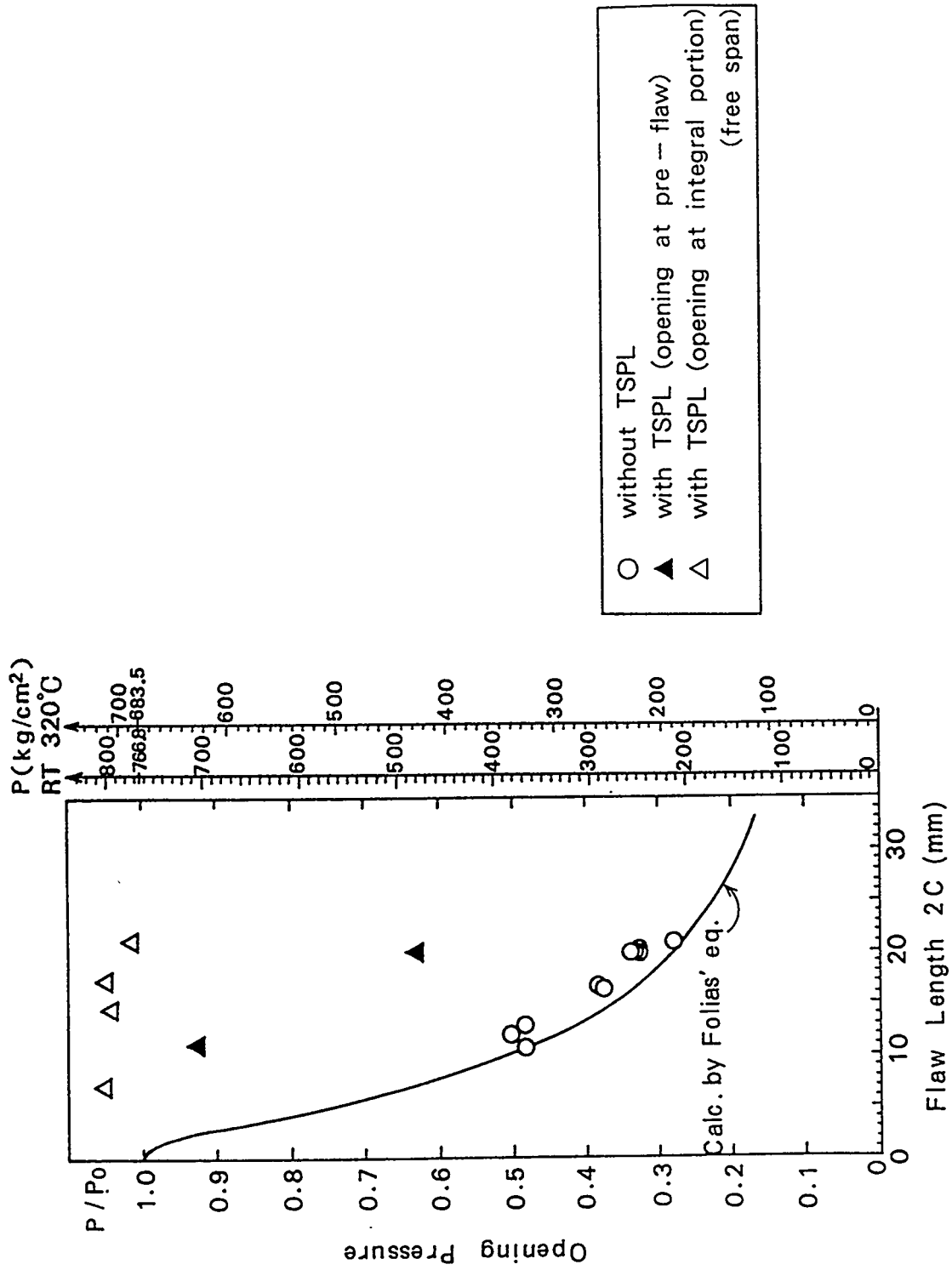
Leak Rate and Flaw Length

o Opening Behavior of the Through - Wall Flaw

- Test piece tube with through - wall IGA was internally pressurized to confirm the pressure level for opening the through - wall flaw



- With tube support plate, opening pressure becomes higher by its reinforcing effect
- Without tube support plate, Folias' equation is applicable to assess the opening behavior of through - wall flaw

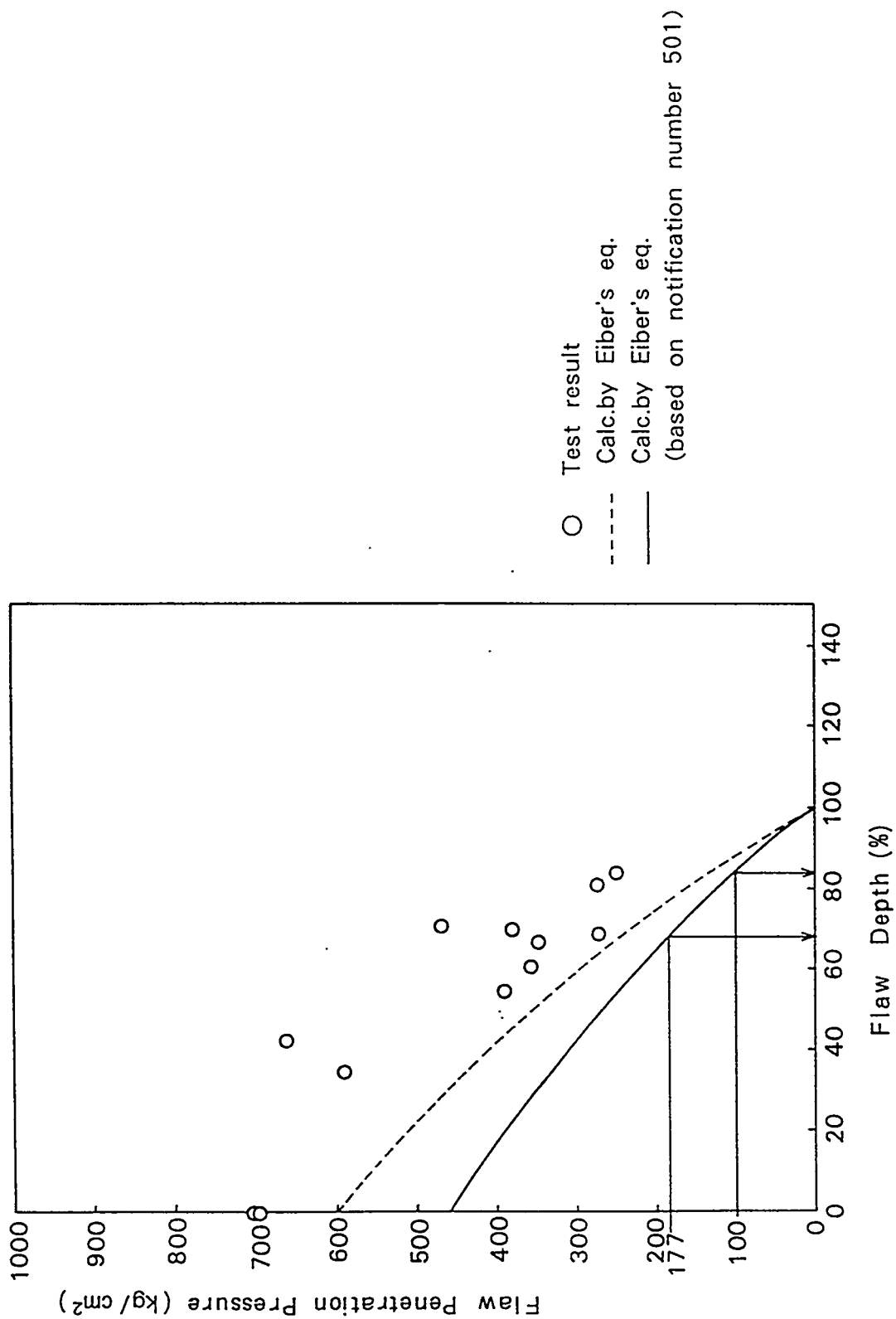


Opening Pressure of Through – wall Flaw
 (P_o : rupture pressure of tube without flaw)

Assessment of Reliability of the Tube with Flaws during Plant Operation

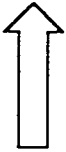
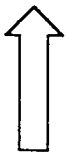
maximum undetected IGA depth (40%WT)
+
propagation during one operating cycle (10000Hr)
||
expected IGA depth at EOC (~59%WT)

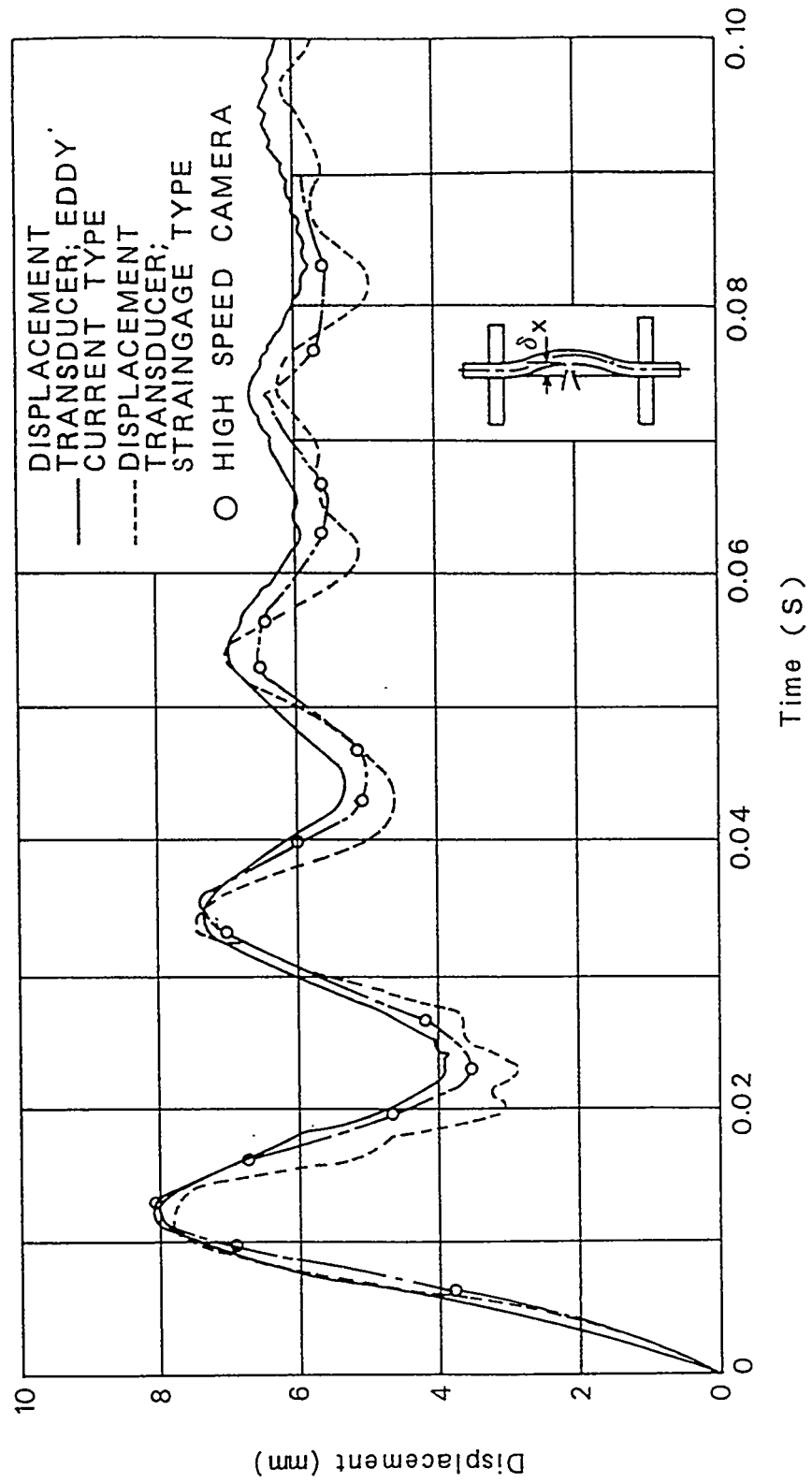
- Expected IGA depth at EOL (~59%TH) is smaller than the threshold depth to induce flaw penetration at normal operation (~83%WT) and at MSLB (~68%WT)



Flaw Penetration Pressure and Flaw Depth

Assessment of Behavior for the Case of Tube Rupture Accident

- Erosion behavior of ruptured tube and adjacent tubes by jet flow
 no significant effect on both parts is induced
- Dynamic behavior of ruptured tube and adjacent tubes
 contacts to the adjacent tubes occur occasionally,
but no significant effect is induced



Displacement of Ruptured Tube

Cnclusions

- Tube with undetected IGA can maintain its integrity during normal operation and even under accident (MSLB) condition
- Even if the flaw penetration happens, crack opening does not occure by tube support plate reinforcing effect and leak rate does not increase

ASSESSMENT OF THE LEAK TIGHTNESS INTEGRITY OF THE STEAM GENERATOR TUBES AFFECTED BY ODS CC AT THE TUBE SUPPORT PLATES

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ABSTRACT

An EPRI report gives a method for predicting a conservative value of the total primary-to-secondary leak rate which may occur during a postulated steam generator depressurization accident such as a Main Steam Line Break (MSLB) in a steam generator with axial through-wall ODS CC at the TSP intersections. The Belgian utility defined an alternative method deviating somewhat from the EPRI method. When reviewing this proposed method, the Belgian safety authorities performed some calculations to investigate its conservatism. This led them to recommend some modifications to the EPRI method which should reduce its undue conservatism while maintaining the objective of conservatism in the offsite dose calculations.

1. INTRODUCTION

Evaluation of the Outside Stress Corrosion Cracking (ODSCC) affecting Steam Generator (SG) tubes at the Tube Support Plates (TSP) requires to demonstrate the adequate leakage integrity of degraded SG tubes under postulated accident conditions. This is achieved by showing that for the most limiting accident, the resulting primary-to-secondary leakage will not lead to unacceptable offsite doses.

According to the applicable Standard Review Plan [1], the offsite dose calculations are intended to be conservative. The calculations are deterministic and use conservative estimates of the influencing parameters.

Usually a maximum allowable leak rate under accident condition is defined which allows to meet the criterion for maximum acceptable offsite dose. Adequate leakage integrity of the SG tubes is then demonstrated if the predicted primary-to-secondary leak rate does not exceed the maximum allowable accident leak rate.

An EPRI report [2] gives a method for predicting the total primary-to-secondary leak rate which may occur during a postulated steam generator depressurization accident such as a Main Steam Line Break (MSLB) in a steam generator with axial through-wall ODS CC at the TSP intersections. The Belgian utility defined an alternative method [3] deviating somewhat from the EPRI method. When reviewing this proposed method, the Belgian safety authorities performed

some calculations to investigate its inherent conservatism. This led them to recommend the utility to implement some modifications which should make acceptable the proposed method for predicting the total leak rate under accident conditions. The efforts undertaken by the Belgian safety authorities are summarized hereafter.

2. STATEMENT OF THE PROBLEM

2.1 Basis

Steam generator B of the Doel 4 plant and all steam generators of the Tihange 3 plant to a lesser extent are affected by ODSCC at the tube support plates. During each refuelling outage, all the tubes in the affected steam generators are inspected with the bobbin coil to monitor the extent and the growth of this degradation. Each recorded voltage is then compared to the specific voltage-based plugging limit to determine its acceptability for the upcoming cycle. The voltage-based plugging criterion is defined in order to ensure both the structural integrity of the tubes with the required margins and the leakage integrity of the tube bundle, taking into account the expected defect growth up to the end of the upcoming cycle.

It should be noted that the Belgian utility is using for the bobbin coil inspection of the SG tubes a calibration standard different from the ASME calibration standard. However a limited sample of ODSCC indications was inspected on site using both calibration methods and the linear regression analysis of the recorded voltage amplitudes allowed the Belgian utility to define a relationship between the two types of voltage amplitude. For the sake of simplicity, all the bobbin coil voltage figures given in the following paragraphs are expressed as ASME calibrated voltages.

2.2 Plugging limit for ensuring structural integrity

The basis for determining the plugging limit as applied to the Belgian plants to ensure the required structural integrity, referred to as the structural limit, is developed hereafter. The structural limit should ensure a safety factor of 3.0 under normal operating pressure differentials and a safety factor of 1.4 for faulted conditions.

Laboratory examination of tubes removed from service allowed to determine the ODSCC degradation mechanism affecting the Alloy 600 tubes at the TSPs. Axially oriented microcracks initiate somewhat randomly on the OD surface of the tube. They grow and link together to form larger macrocracks having numerous ligaments. When wall penetration is nearly achieved, a single dominant crack is usually formed.

Burst testing of axially cracked Inconel tubes shows that no crack growth precedes the bursting but large ductile deformation (bulging) is observed before the axial crack becomes unstable. If the axially cracked area is located within a tubesheet or a tube support plate, the axial failure mode is precluded by the constraint against tube wall deformation provided the tubesheet or TSP. This led the Belgian utility to conclude that the axial failure mode of the ODSCC affected tubes was no safety concern in normal operating conditions and so this failure mode should only be considered for the postulated accident conditions. This conclusion was endorsed by the Belgian safety authorities. For the Belgian plants, the limiting accident condition, i.e. the accident condition leading to the highest pressure differential, is the FeedWater Line Break (FWLB).

Finite element structural analysis of the tube bundle supporting system of the Doel 4 and Tihange 3 steam generators allowed the utility to conclude that the axial deflection of all the tube support

Plates but TSP nr 9 under the dynamic load resulting from a secondary line break was limited (less than the critical length of axial cracks in free span region). The axial failure mode should therefore be considered only when assessing the structural integrity of the tubes affected at TSP nr 9. For all the other TSPs for which the degraded area remains confined within the TSP during the postulated accident, the circumferential failure mode is the only structural concern. An empirical correlation of residual cross sectional area of sound material vs. bobbin coil voltage was developed by the Belgian utility from laboratory examination and/or tensile testing of removed tubes. This correlation is used to define the voltage-based plugging limit when the circumferential failure mode is the structural concern. When the axial failure mode is to be considered, the voltage-based plugging limit is based on the empirical correlation of bursting pressure vs. bobbin coil voltage developed by the EPRI [2]. For all the TSPs, but TSP nr 9, the plugging limit is about 11.0 V. For TSP nr 9, the plugging limit is 0.0 V. This zero volt limit results from the applicable value of the upper bound of the recorded voltage growth which is higher than the acceptable Begin-Of-Cycle (BOC) voltage.

2.3 Plugging limit for ensuring leakage integrity

The plugging limit for ensuring the leakage integrity, referred to as the leak tightness limit, is determined such that the predicted End-Of-Cycle (EOC) primary-to-secondary leakage through the whole tube bundle of the most degraded steam generator under accident conditions does not lead to an unacceptable offsite dose.

For plants with moderately degraded steam generators and/or for plants using a structural limit with a low value (1.0 to 2.0 V range), it is usually verified that the use of the structural limit allows to ensure also the leakage integrity. For the Belgian steam generators affected by ODSCC at the TSP, combination of severe degradation and relatively high value of the structural limit leads the leak tightness limit to be potentially the leading plugging limit. Moreover the rather low value of the maximum acceptable offsite dose at the Belgian plants is also a penalizing factor which reinforces this conclusion. For the 1995 outages, the leak tightness limit was about 11V (equal to the structural limit) for the Doel-4 SG and 5.5 V for the Tihange-3 SG.

2.4 Requirements for assessing the leakage integrity

Paragraph 2.3 hereabove shows the necessity for the Belgian plants affected by ODSCC at the TSP to define an adequate method for predicting the primary-to-secondary total leak rate under the most limiting accident condition. The method should be conservative enough to meet the intent of the applicable Standard Review Plan for offsite dose calculations [1] but it should also not be overly conservative to avoid the unnecessary plugging or repairing of tubes meeting the requirements for structural integrity.

3.0 COMPARISON BETWEEN EPRI AND LEAK-2 INDIVIDUAL LEAK RATE PREDICTIONS

3.1 Basic principles

The Belgian utility developed a method, referred to as LEAK 2 method, generally similar to the EPRI method [2].

Both EPRI method and LEAK 2 method aim at predicting the primary-to-secondary leakage which may occur in a steam generator during a line break postulated in the secondary system (FWLB or MSLB) when the steam generator tubes are affected by ODSCC at the tube support

plates. Both methods rely on a empirical correlation of individual tube leak rates vs. eddy current bobbin coil voltage . The data were obtained from pulled out tubes and specimens tested in a model boiler facility. Both types of specimens exhibited ODSCC and were leak tested to different pressure differentials across the tube without the presence of simulated TSP.

The relationship between the individual conditional leak rate ($\dot{Q}_i$) and the measured bobbin voltage (V_i) is assumed to be

$$\log_{10}(\dot{Q}_i) = \beta_0 + \beta_1 \log_{10}(V_i) + \varepsilon \quad (1)$$

An essential difference between the EPRI method and LEAK 2 method is found in the calculation of the conditional leak rate ($\dot{Q}_i$). The way the uncertainties of the linear regression analysis are handled is also different.

The EPRI method aims at giving the upper limit on leak rate for an individual tube exhibiting an ODSCC defect of a given begin-of-cycle voltage. The 95/95 probability/confidence level obtained from a Monte Carlo analysis is taken as the upper limit. In order to take into account adequately the uncertainties present in the leak rate vs. voltage correlation, the joint sampling distribution of the two parameters of the linear regression analysis (intercept β_0 and slope β_1) is considered as well as the sampling distribution of the prediction error.

The LEAK 2 method aims at giving the mean value of leak rate for an individual tube. To achieve this goal, the relationship between the mean conditional leak rate ($\dot{Q}_{i, \text{mean}}$) and the measured bobbin voltage (V_i) as inferred from eq (1) is used

$$\dot{Q}_{i, \text{mean}} = \exp \left[\frac{(\ln 10 \times \hat{\sigma}_\varepsilon)^2}{2} \right] 10^{\hat{\beta}_0} V_i^{\hat{\beta}_1} \quad (2)$$

where $\hat{\beta}_0$ and $\hat{\beta}_1$ are the estimates of β_0 and β_1 given by the least-squares analysis.

The relationship (2) is obtained from (1) by assuming that $\log_{10}(\dot{Q}_i)$ is normally distributed with mean equal to $\hat{\beta}_0 + \hat{\beta}_1 \log_{10}(V_i)$ and variance $\hat{\sigma}_\varepsilon^2$ equal to the estimate of the variance of the disturbance term ε .

3.2 Assessment of the best estimate value of the conditional leak rate

The EPRI method is believed to provide a correct statistical analysis and is therefore considered as the reference method to which the LEAK 2 method will be compared.

There are two main reasons for suspecting the mean conditional leak rate ($\dot{Q}_{i, \text{mean}}$) obtained by a deterministic equation in the LEAK 2 method to differ from the mean value obtained by a statistical analysis as in the EPRI method :

- (i) the LEAK 2 method does not consider the influence of the sampling distribution of the parameters β_0 and β_1
- (ii) the variance of the prediction error is known to increase non linearly as $\log_{10}(V_i)$ departs from the mean value of the $\log_{10}(V)$ data found in the data base.

As the linear regression analysis uses a Log-Log model, it is believed that the mean conditional leak rate calculated by the LEAK 2 method ($\dot{Q}_{i, \text{mean}}^{\text{LEAK2}}$) underestimates the mean value calculated by the EPRI method ($\dot{Q}_{i, \text{mean}}^{\text{EPRI}}$).

In order to assess the suspected underestimation of the mean leakage by the LEAK 2 method, the probability distribution of the conditional individual leak rate $\dot{Q}_i$ has been calculated by performing a Monte Carlo analysis accounting for uncertainties as in the EPRI method. The mean value of the conditional leak rate ($\dot{Q}_{i, \text{mean}}^{\text{EPRI}}$) is then compared to $\dot{Q}_{i, \text{mean}}^{\text{LEAK2}}$. Calculations were performed for 3/4-inch and 7/8-inch tubes under a pressure differential of 2650 psi. The leak rate data base for 3/4-inch and 7/8-inch tubing provided in the EPRI report is used. Each $\dot{Q}_{i, \text{mean}}^{\text{EPRI}}$ value is obtained by means of 1000 Monte Carlo trials. It should be noted that these calculations do not consider any flaw growth.

Figure 1 and 2 show the ratio $\dot{Q}_{i, \text{mean}}^{\text{EPRI}} / \dot{Q}_{i, \text{mean}}^{\text{LEAK2}}$ for the 3/4-inch and 7/8-inch tubes respectively. These results show that in the voltage range centered at the mean value of the $\log_{10}(V)$ data found in the data base (8.7 V for 3/4-inch tubes and 14.4 V for 7/8-inch tubes), the ratio $\dot{Q}_{i, \text{mean}}^{\text{EPRI}} / \dot{Q}_{i, \text{mean}}^{\text{LEAK2}}$ is nearly constant and is equal to about 1.2 and 2.0 for 3/4-inch and 7/8-inch tubes respectively. For lower values of bobbin voltage ($< 2V$ for 3/4-inch tubes and $< 4V$ for 7/8-inch tubes), the ratio $\dot{Q}_{i, \text{mean}}^{\text{EPRI}} / \dot{Q}_{i, \text{mean}}^{\text{LEAK2}}$ is increasing (not shown on Figures 1 and 2).

Therefore the underestimation of the conditional leak rate by using the LEAK 2 method depends heavily on the applicable plugging limit. If the voltage-based repair criteria for Westinghouse steam generator tubes proposed by the US NRC are used (i.e. 2V for 7/8-inch tubing and 1V for 3/4-inch tubing), the use of the LEAK 2 method leads to a high (and unacceptable) underestimation of the mean leak rate. For the specific case of the steam generators of DOEL-4 and Tihange 3 (3/4-inch tubing), where an higher plugging limit is used (i.e., about 7V) the ratio $\dot{Q}_{i, \text{mean}}^{\text{EPRI}} / \dot{Q}_{i, \text{mean}}^{\text{LEAK2}}$ is close to 1.0 and the use of the LEAK-2 method to calculate the conditional mean leak rate does not lead to an unacceptable underestimation.

3.3 Upper bound vs. mean leak rate

In order to assess the difference between the 95/95 value and the mean value, the probability distribution of the conditional individual leak rate $\dot{Q}_i$ as obtained in paragraph 3.2 hereabove by performing a Monte Carlo analysis is used. The upper limit on conditional leak rate as defined in the EPRI report [2], i.e. $\dot{Q}_i^{\text{EPRI}} [95 | N_{\text{trials}}=1000]$, is compared to the mean value $\dot{Q}_{i, \text{mean}}^{\text{EPRI}}$. Figures 2 and 3 give $\dot{Q}_i^{\text{EPRI}} [95 | N_{\text{trials}}=1000]$ and $\dot{Q}_{i, \text{mean}}^{\text{EPRI}}$ for the 3/4-inch and 7/8-inch tubes respectively. These figures also give the mean value of the leak rate obtained by the LEAK 2 method, i.e. $\dot{Q}_{i, \text{mean}}^{\text{LEAK2}}$. Figures 4 and 5 give the ratio $\dot{Q}_i^{\text{EPRI}} [95 | N_{\text{trials}}=1000] / \dot{Q}_{i, \text{mean}}^{\text{EPRI}}$ for the 3/4-inch and 7/8-inch tubes respectively. The results show that the upper limit of the conditional individual leak rate exceeds the mean value by a factor of about 4.0 within the 2.0 to 14.0 V voltage range.

On the basis of these results, the Belgian safety authorities considered the use of the predicted mean leak rate as unacceptable.

4.0 PREDICTION OF THE CONDITIONAL TOTAL LEAK RATE

4.1 EPRI method

In the EPRI method [2], the predicted conditional total leak rate $\dot{Q}_p$ for the postulated accident condition is obtained summing the contribution of all indications. Each contribution to the total leak rate is equal to the upper limit of the corresponding predicted individual leak rate.

Let the number of indications within a voltage interval i specified by $[V_i - \delta V/2, V_i + \delta V/2]$ be designated by m_i . The predicted total leak rate is then expressed as

$$\dot{Q}_p^{EPRI} = \sum_{i=1}^n m_i \dot{Q}_i^{EPRI} [961 | N_{trials} = 1000] \quad (3)$$

This procedure is obviously conservative as it overestimates the 95/95 probability/confidence level of the predicted total leak rate. This results from the fact that the standard deviation of the probability distribution of the sum of N independent random variables is less than the summation of the individual standard deviations of the probability distribution of each individual random variable.

In order to assess the overestimation of the 95/95 probability/confidence level of the predicted total leak rate by the EPRI method, two alternative procedures have been investigated. The numerical implementation of these procedures assumes the histogram of ODSCC indications shown in Figure 7. This histogram is typical of a Belgian steam generator seriously affected by ODSCC. It should be noted that this histogram depicts the SG condition before the required plugging is performed.

For this typical histogram, the predicted total leak rate calculated in accordance with the EPRI method is equal to 24,128.0 l/hr. Once again the calculation does not consider any flaw growth.

4.2 First alternative procedure for predicting the total leak rate

This first alternative procedure aims at determining the probability distribution of the predicted total leak rate, the upper limit of which is the predicted total leak rate given by the EPRI method.

The procedure considers that the probability distribution of the total leak rate can be obtained by a Monte Carlo summation of the n probability distributions of the leak rate for each voltage interval. The probability distribution of the predicted leak rate for a given voltage interval, i.e. the probability distribution of the predicted leak rate resulting from the m_i indications of the voltage interval i is obtained by scaling by a factor equal to m_i the probability distribution of the predicted individual leak rate as given by the EPRI method (1000 trials).

This procedure was applied to predict the 95/95 probability/confidence level of the predicted total leak rate for the typical histogram shown in Figure 7. The probability distribution of the predicted total leak rate was obtained by means of a 1000 summation trials and is shown on Figure 8. The upper limit of the predicted total leak rate is calculated as

$$\dot{Q}_p = \dot{Q} [961 | N_{trials} = 1000]$$

and is equal to 25,015.0 l/hr. No flaw growth was considered in the calculation.

4.3 Second alternative procedure for predicting the total leak rate

In the second alternative procedure, the probability distribution of the predicted total leak rate is obtained by a Monte Carlo summation of the probability distributions of the leak rate for each individual indication. The probability distribution of the predicted total leak rate was obtained by means of a 1000 trials and is shown on Figure 9. The probability distribution of the individual leak rate is calculated in accordance with the EPRI method. The upper limit of the predicted total leak rate is calculated as

$$\dot{Q}_p = \dot{Q} [961 | N_{trials}=1000]$$

and is equal to 9,664.0 l/hr. No flaw growth was considered in the calculation.

4.4 Comparison of the alternative procedures with the EPRI procedure

The upper limits of the predicted total leak rate determined by the EPRI method and the first alternative method should theoretically be identical. This is confirmed by the calculations performed for the typical histogram of ODSCC indications shown in Figure 7. The calculations show that both upper limits are close to each other. The difference between the results (less than 4 %) is due to the limited number of Monte Carlo trials.

Comparing the probability distribution the predicted total leak rate as determined by the second alternative method (see Figure 9) to the distribution given by first alternative method (see Figure 8) shows the conservatism of the EPRI method for predicting the conditional total leak rate. The conservatism is also illustrated by comparing the upper limit of the total leak rate determined by the second alternative method to the value calculated in accordance with the EPRI method. The calculations performed for the typical histogram of ODSCC indications show that the EPRI method overestimates the upper limit of the total leak rate by a factor of about 2.5.

5.0 PREDICTION OF THE OFFSITE DOSE

Offsite dose calculations are required to be conservative. The Belgian safety authorities however believe that the objective of the calculations is to predict the 95/95 probability/confidence level of the offsite dose without requiring that the 95/95 probability/confidence level value be assigned to each influencing parameter.

The Time Integrated Concentration (TIC) can be expressed as :

$$TIC = (1 - \alpha) \left(\frac{\lambda}{Q} \right)_1 \int_0^T q(t) t^{-\alpha} dt$$

where $q(t)$ is the activity release rate. If the accident induced spiking phenomenon is neglected, the activity release rate $q(t)$ is the product of the coolant gross activity A_c by the primary-to-secondary total leak rate $\dot{Q}_p(t)$

$$q(t) = A_c \dot{Q}_p(t)$$

Furthermore the primary-to-secondary total leak rate $\dot{Q}_p(t)$ can be expressed as the primary-to-secondary total leak rate at the initiating time of the accident $\dot{Q}_p(0)$, i.e. under the maximum pressure differential, multiplied by a deterministic time-dependent function $f(t)$.

$$\dot{Q}_p(t) = \dot{Q}_p(0) f(t)$$

The Time Integrated Concentration (TIC) can be written as

$$TIC = (1 - \alpha) \left(\frac{\chi}{Q} \right)_1 \dot{Q}_p(0) A_c \int_0^T f(t) t^{-\alpha} dt \quad (4)$$

Considered as a random variable, TIC is the product of the two independent random variables $\left(\frac{\chi}{Q} \right)_1$ and $\dot{Q}_p(0)$. The density function of $\dot{Q}_p(0)$ is known from the calculation of the predicted primary-to-secondary total leak rate. The density function of $\left(\frac{\chi}{Q} \right)_1$ should be known from the onsite meteorological measurement program. The density function of the product $\left(\frac{\chi}{Q} \right)_1 \dot{Q}_p(0)$ can be determined by a Monte Carlo simulation. The 95/95 prediction value of the product $\left(\frac{\chi}{Q} \right)_1 \dot{Q}_p(0)$ can then be assessed. By using eq(4), the 95/95-percent prediction value of the time integrated concentration is then calculated.

This procedure would allow to use for the predicted total primary-to-secondary leak rate $\dot{Q}_p(0)$ a value lower than the 95/95 prediction value if the 95/95 percent prediction value of $\left(\frac{\chi}{Q} \right)_1$ is adopted. Preliminary calculations performed on a typical case showed that value of the upper limit of the predicted total primary-to-secondary leak rate $\dot{Q}_p(0)$ could be lowered by about 30%.

6.0 CONCLUSION

An EPRI report [2] gives a method for predicting the total primary-to-secondary leak rate which may occur during a postulated steam generator depressurization accident such as a Main Steam Line Break (MSLB) in a steam generator with axial through-wall ODSCC at the TSP intersections. The method is however believed to be overly conservative and some proposals are made to modify it while maintaining the objective of conservatism in calculating offsite dose under accident condition.

7.0 REFERENCES

- [1] Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, Section 15.1.5, NUREG-0800, July, 1981.
- [2] EPRI Report TR 100407 Rev.1, *PWR Steam Generator Tube Repair Limits - Technical Support Document for Outside Diameter Stress Corrosion Cracking at Tube Support Plates.*

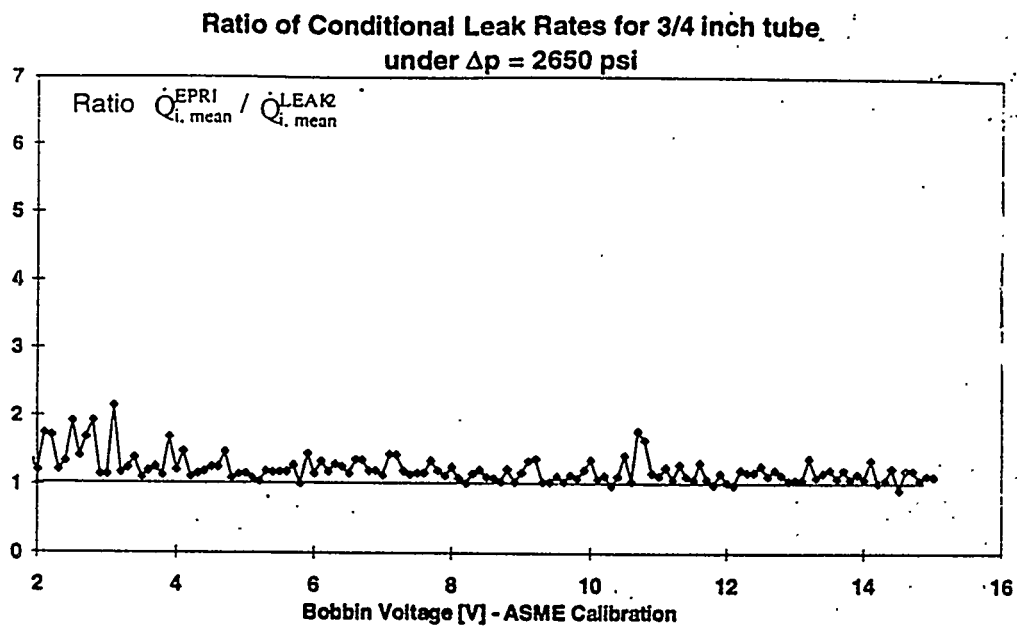


Figure 1

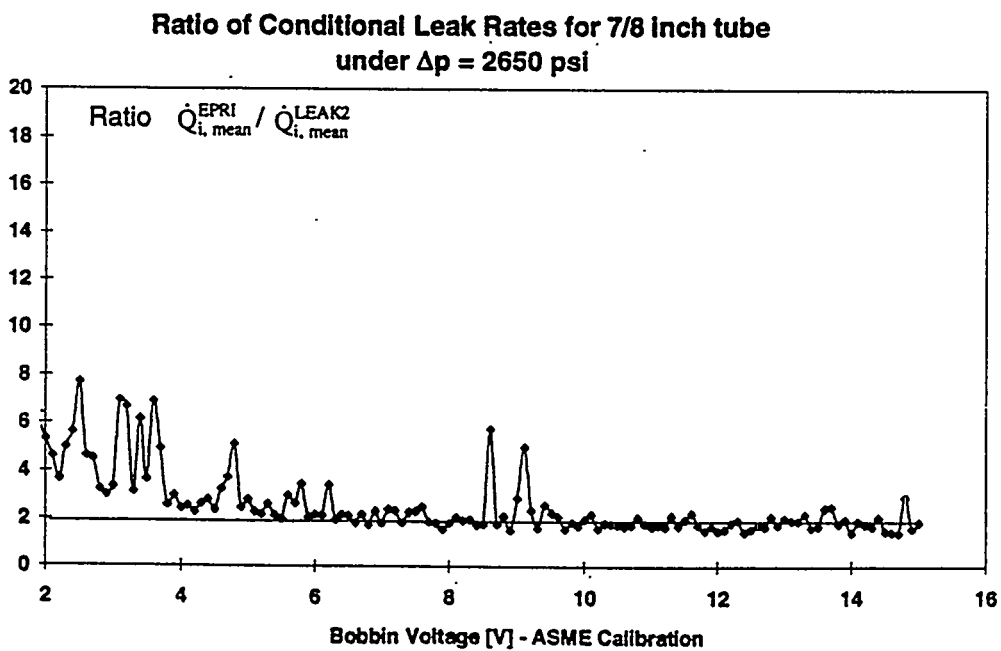


Figure 2

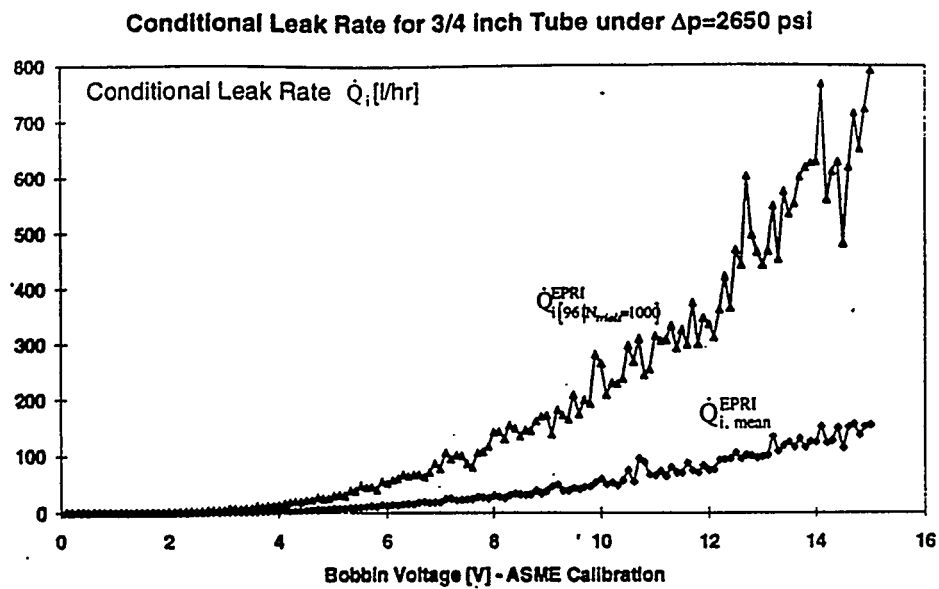


Figure 3

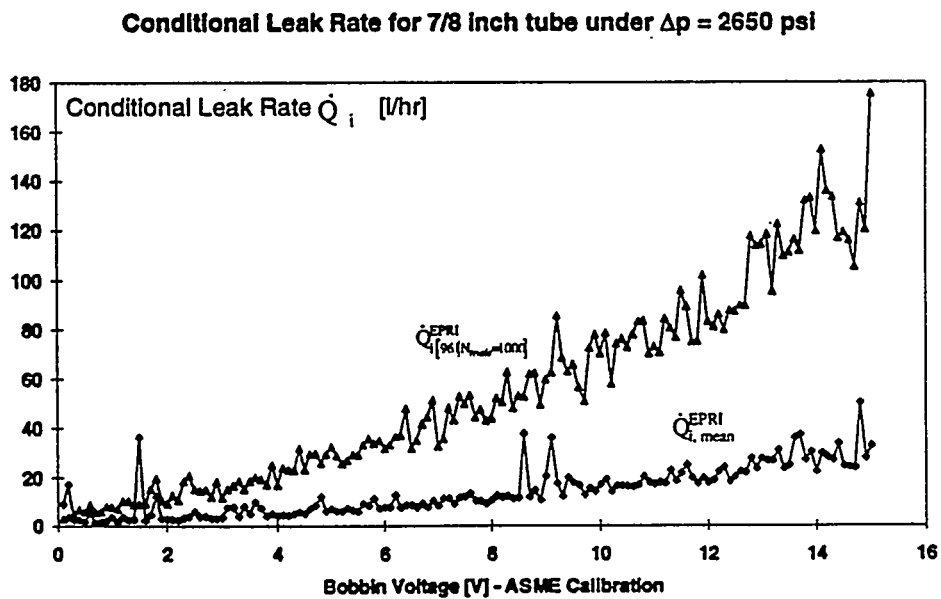


Figure 4

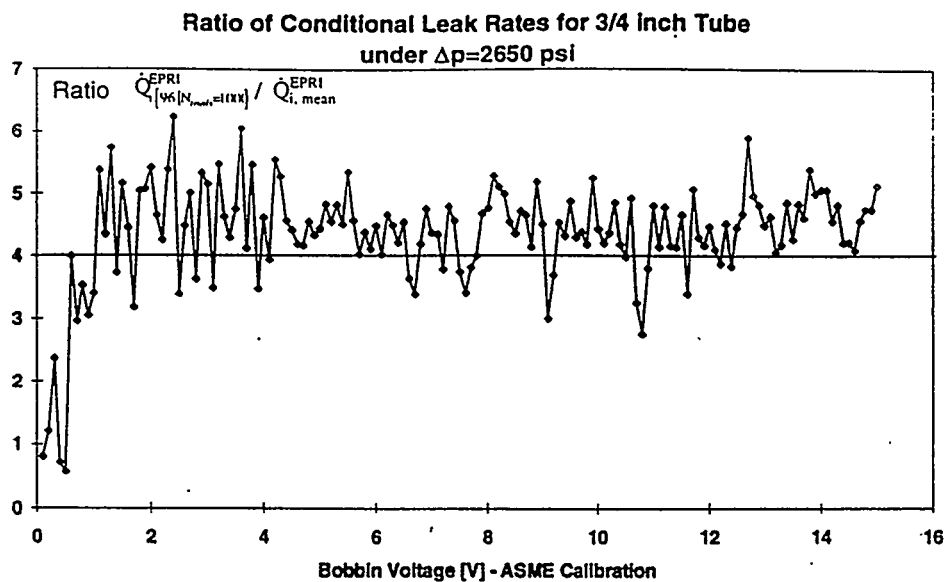


Figure 5

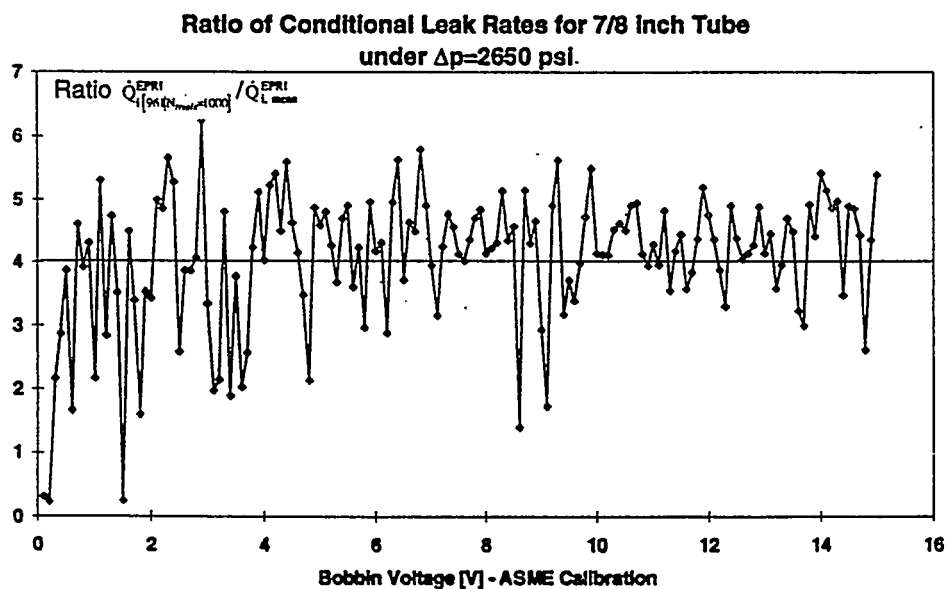


Figure 6

Typical ODSCC Voltage Indication Distribution in Belgian Steam Generator
for 3/4 inch tubing

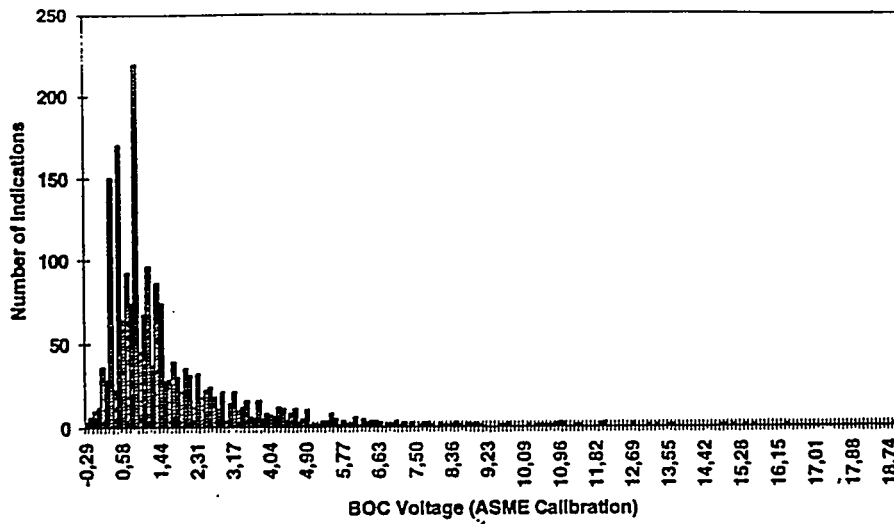


Figure 7

Predicted Total Leak Rate for Assumed Typical Voltage Indication
Distribution (First Alternative Procedure)

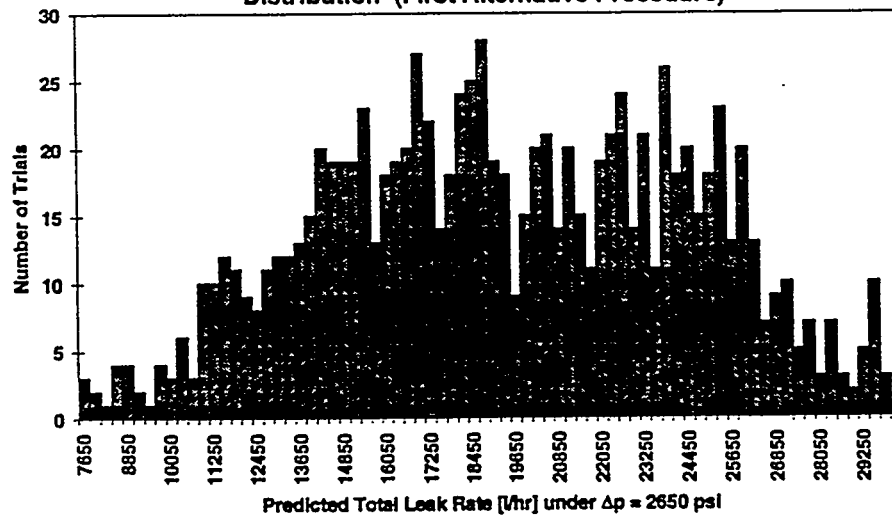


Figure 8

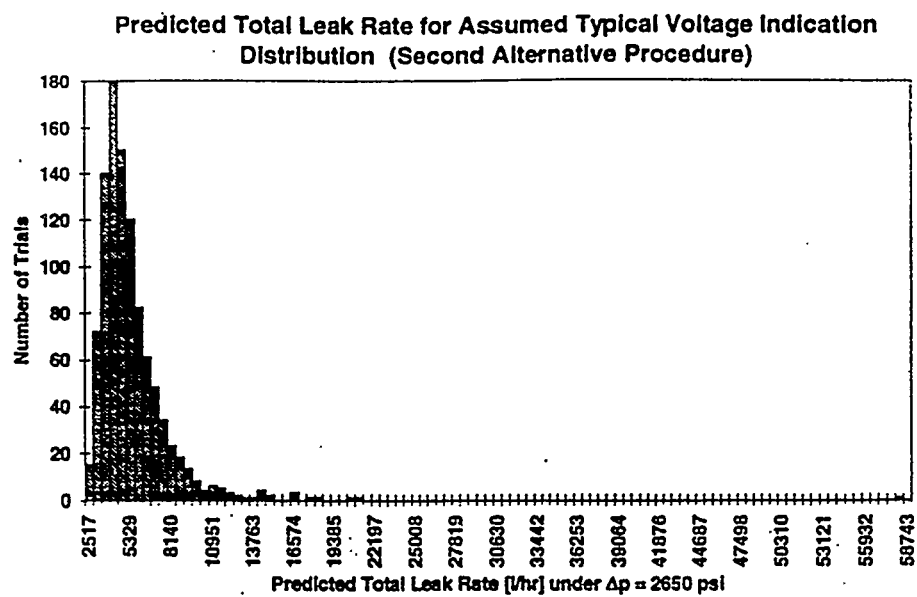


Figure 9

TITLE: SPANISH APPROACH TO RESEARCH AND DEVELOPMENT APPLIED TO STEAM GENERATOR TUBES STRUCTURAL INTEGRITY AND LIFE MANAGEMENT

AUTHORS JOAQUIN LOZANO (C.N. ASCÓ)
GUSTAVO J. BOLLINI (TECNATOM, S.A.)

ABSTRACT

1 INTRODUCTION

The operating experience acquired from certain Spanish Nuclear Power Plant steam generators shows that the tubes, which constitute the second barrier to release of fission products, are susceptible to mechanical damage and corrosion as a result of a variety of mechanisms, among them wastage, pitting, intergranular attack (IGA), stress-corrosion cracking (SCC), fatigue-induced cracking, fretting, erosion/corrosion, support plate denting, etc.

These problems, which are common in many plants throughout the world, have required numerous investments by the plants (water treatment plants, replacement of secondary side materials such as condensers and heaters, etc.), have meant costs (operation, inspection and maintenance) and have led to the unavailability of the affected units.

In identifying and implementing all these preventive and corrective measures, the Spanish utilities have moved through three successive stages:

- In the initial stage, the main source of information and of proposals for solutions was the Plant Vendor, whose participation in this respect was based on his own Research and Development programs.
- Subsequently, the Spanish utilities participated jointly in the EPRI Steam Generator Owners Group, collaborating in financing.
- Finally, the Spanish utilities set up their own Steam Generator Research and Development program, while maintaining relations with EPRI programs and those of other countries through information interchange.

2 THE STEAM GENERATOR RESEARCH AND DEVELOPMENT PROJECT

Although the Spanish utilities had previously collaborated in specific research projects, in 1987 a multidisciplinary project was initiated with a view to ensuring integrated consideration of the most relevant aspects, the main objectives being to improve plant availability, reduce operating and maintenance costs and increase the

technological capacity of the participating companies.

Achievement of these objectives also implied extending the operating lifetime of these components while at the same time maintaining the necessary levels of safety.

The organizations participating in the project include both the nuclear power plants and utilities and CIEMAT, ENSA and Tecnatom. The Sarriá Institute of Chemistry (IQS) has also participated in development of certain project activities. The total budget for the first phase of the project (1987-1991) amounted to 14 million dollars.

The activities performed within the framework of this project, the first phase of which was completed in June 1991, were as follows:

- . Development of a database containing complete information on each steam generator and installed in all the Spanish nuclear power plants, and of a documentary database which has made it possible to centralize all the information generated by this project, other similar projects and overseas organizations.
- . Optimization and Validation of the eddy current in-service inspection techniques applied to the steam generator tubes.
- . Examination of tubes removed from steam generators in laboratories especially prepared for this purpose.
- . Study of the corrosion behaviour of tube materials in response to primary and secondary side chemical environments.
- . Specific Plugging Criterion applicable to each type of defect, which is currently being applied in the Plants as an alternative to the requirements of the operating technical specifications.
- . Evaluation of mitigation and repair techniques, such as chemical cleaning, plugging and sleeving.

3 THE NEED FOR A SECOND PHASE OF THE STEAM GENERATOR RESEARCH AND DEVELOPMENT PROJECT

As the Steam Generator Research Project developed, it became increasingly clear that a second phase would be advisable for the following reasons:

- To finalize activities that it had not been possible to complete during the first phase.
- The appearance of new problems not foreseen in the first phase.
- The possibility to initiate more innovatory developments on the basis of the capabilities acquired during the first phase.

- The need to replace the steam generators in certain Spanish plants.

This second phase has the same objectives, participants and internal organization as the first, the foreseen period for its development being January 1991 to June 1996.

The second phase of the Steam Generator Research Project includes nine activities:

- . Steam generator tube plugging criteria for secondary side stress corrosion cracking in the support plate zone, circumferential cracks in the roll transition zone and fretting defects in the anti-vibration bar zone.
- . Generation by accelerating methods of defects similar to those found in steam generator tubes, in order to have available manufacturing procedures and test pieces for burst tests, leaks, etc. and for the validation and optimization of inspection techniques.
- . Ultrasonic technique steam generator tube inspection system, as a complement to the eddy current technique, for complex crack configurations and for the inspection of the sleeves installed on the tubes.
- . Optimization and validation of the eddy current, steam generator in-service inspection techniques applied to defects of complex configuration (axial-circumferential).
- . Research into new, alternative materials for steam generator tubes (Inconel-690 TT and Incoloy 800 N.G.).
- . Creation of an advanced steam generator tube degradation prediction model.
- . System based on laser techniques for inspection of steam generator tubes from the inner surface.
- . Development of an optimized procedure for chemical cleaning of the secondary side of Spanish steam generators.
- . Corrosion testing of the materials typically used for steam generator tubes (Inconel 600 MA and 600 TT), which will continue the work on tube material behaviour initiated during the first phase.

The budget for this second phase will amount to approximately 12 million dollars.

This paper presents the main conclusions reached in the different activities of Phase I and II and their applications to the Spanish Nuclear Power Plants.

EVALUATION OF STEAM GENERATOR WWER 440 TUBE INTEGRITY CRITERIA

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Introduction

The main corrosion damage in WWER steam generators under operating conditions has been observed on the outer surface of these tubes. An essential operational requirement is to assure a low probability of radioactive primary water leakage, unstable defect development and rupture of tubes.

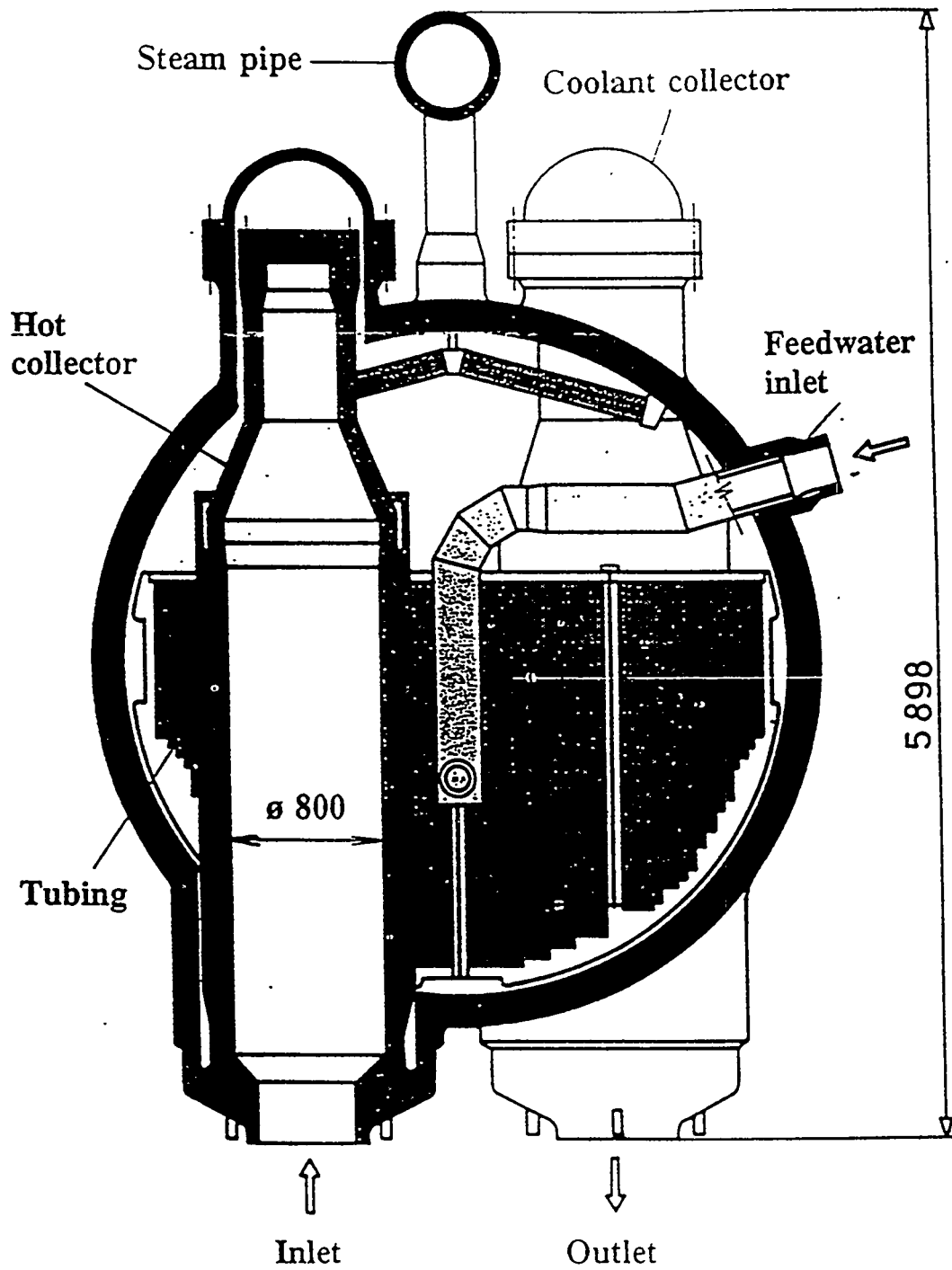
In the case of WWER 440 steam generators the above requirements led to the development of permissible limits for data evaluation of the primary-to-secondary leak measurements and determination of acceptable values for plugging of heat exchange tubes based on eddy current test (ECT) inspections.

Tube integrity

Evaluation of the heat exchange tube integrity in operating steam generators implies complex procedures and data acquisition, involving, in-service measurement and non-destructive inspection of tubes, determination of the extent of damage, evaluation of the measurement results based on limit values for tube operation and plugging and existing and proposed measures to reduce the degradation effects. For the evaluation of in-service inspections and operational measurements, structural and leakage limits have been established involving three factors;

- permissible tube wall thickness that limits continued operation of the tubes with defects and assures their integrity upon loading under normal and accident conditions. This includes an additional margin for tube wall thinning due to crack growth during the period between two non-destructive inspections,
- permissible size of a through-wall crack that is sufficiently stable under operating conditions in relation to the critical crack length and does not aggravate secondary coolant activity above limit values,
- permissible value of the secondary water radioactivity.

The WWER steam generators have in comparison with the vertical types a lower frequency of corrosion damage, but possess a limited possibility to pull out heat exchange tubes. Therefore, the limit values for the primary coolant leakage and plugging of damaged tubes have been determined on tubes with artificial cracks prepared by SCC method in a MgCl_2 melt. Such cracks are very similar to the defects observed on the tubes in operation.



STEAM GENERATOR WWER 440

TUBE CORROSION DAMAGE

CORROSION TYPE :

- SECONDARY SIDE CORROSION
- PITTING, TG SCC,IG SCC

LOCATION :

- TUBE SUPPORT PLATE (90 %)
- FREE SPAN

DEFECT ORIENTATION :

- AXIAL AND CIRCUMFERENTIAL CRACKS

WALL PENETRATION :

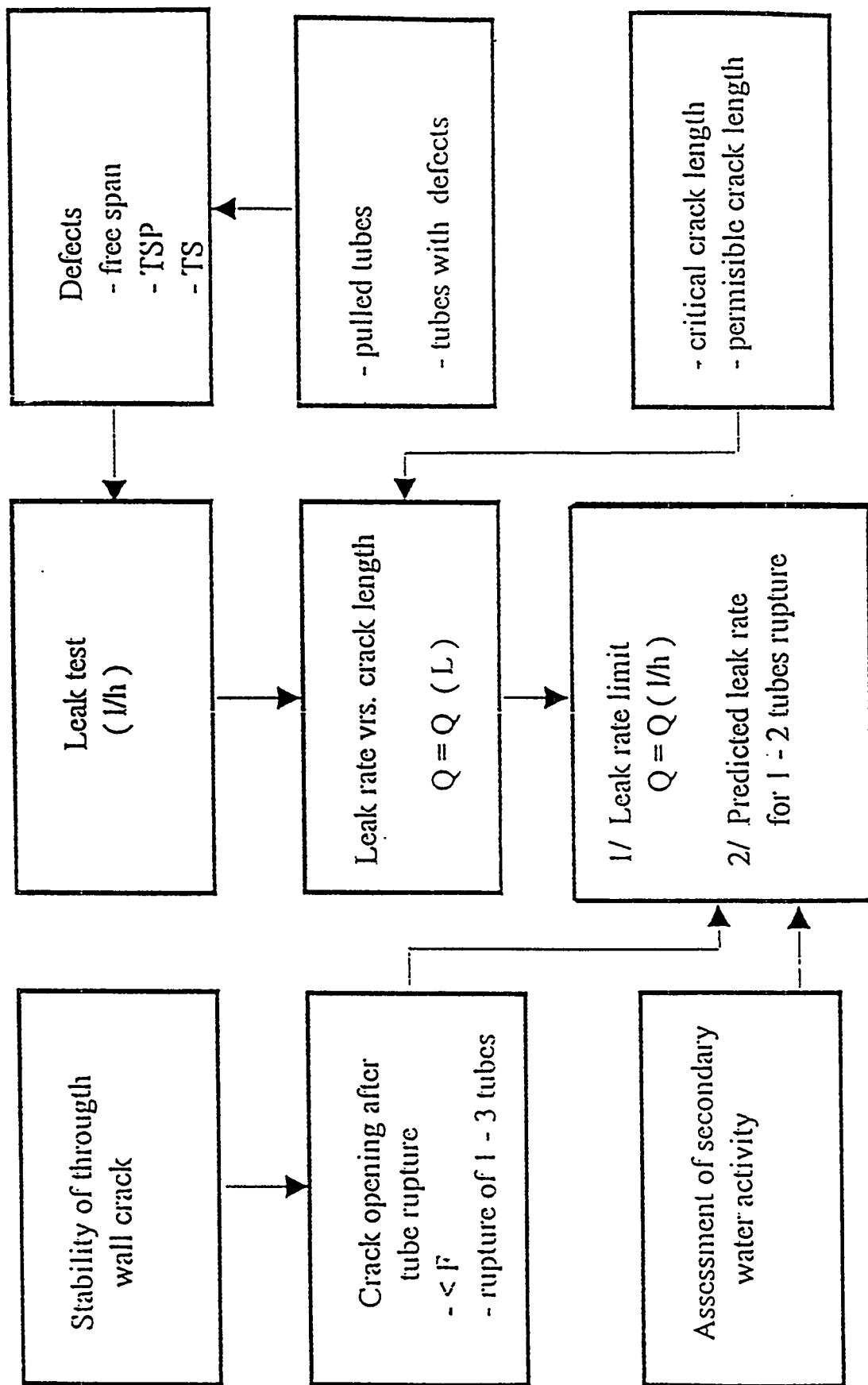
- PITTING < 0.4 mm
- NOT THROUGH - WALL CRACKS
- THROUGH - WALL CRACKS

CRACK GROWTH :

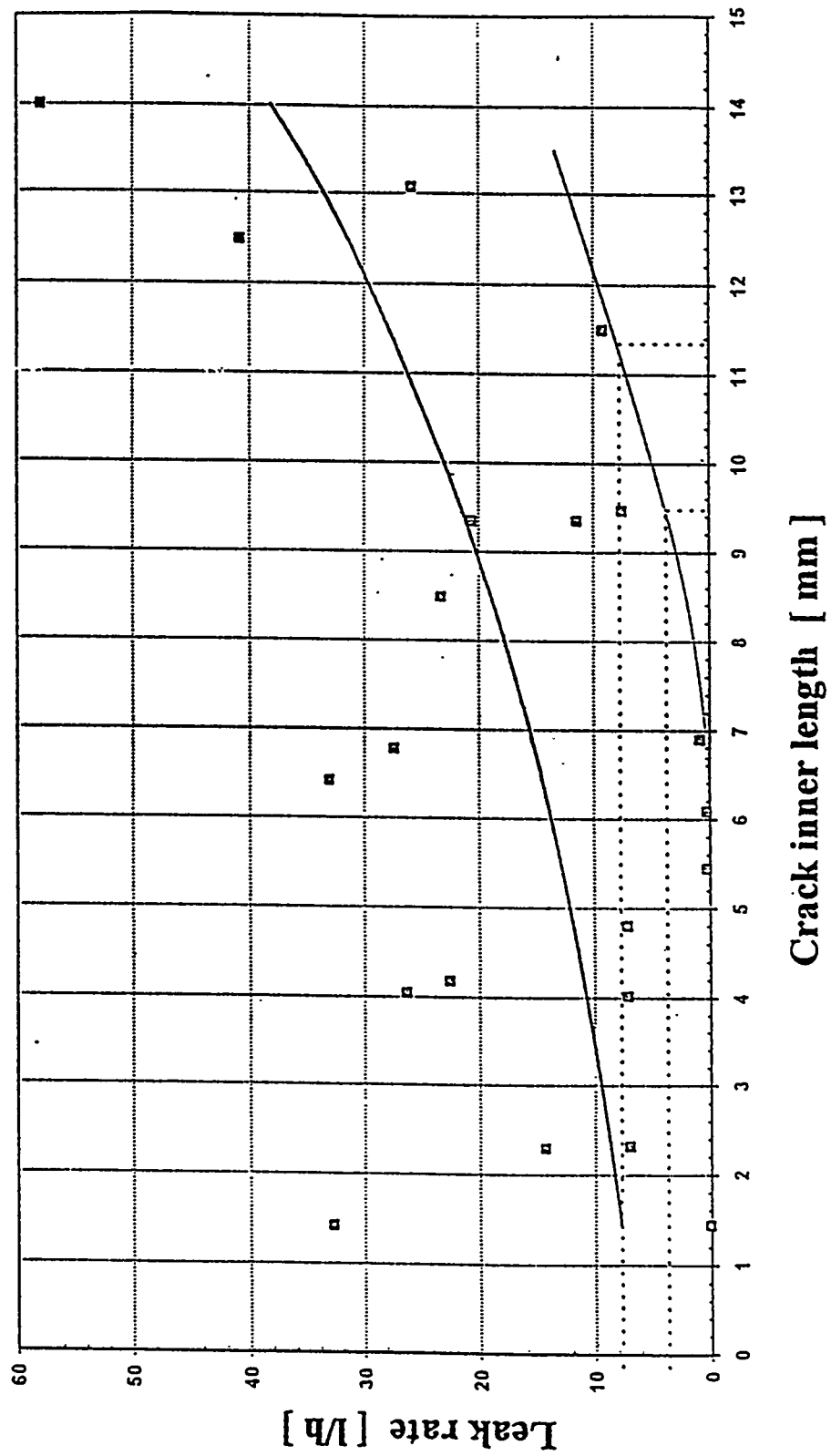
- TG SCC (IG SCC)
- BRANCHING OF CRACKS
- CRACK CLOGGING
- EXPOSURE TIME ENLARGE CRACK WIDTH

MAIN INFLUENCING FACTORS : Cl⁻, Cu, SO₄⁻, pH, Δt

SG TUBE LEAK ASSESSMENT



LEAK RATE TEST - SG WWER 440 MW



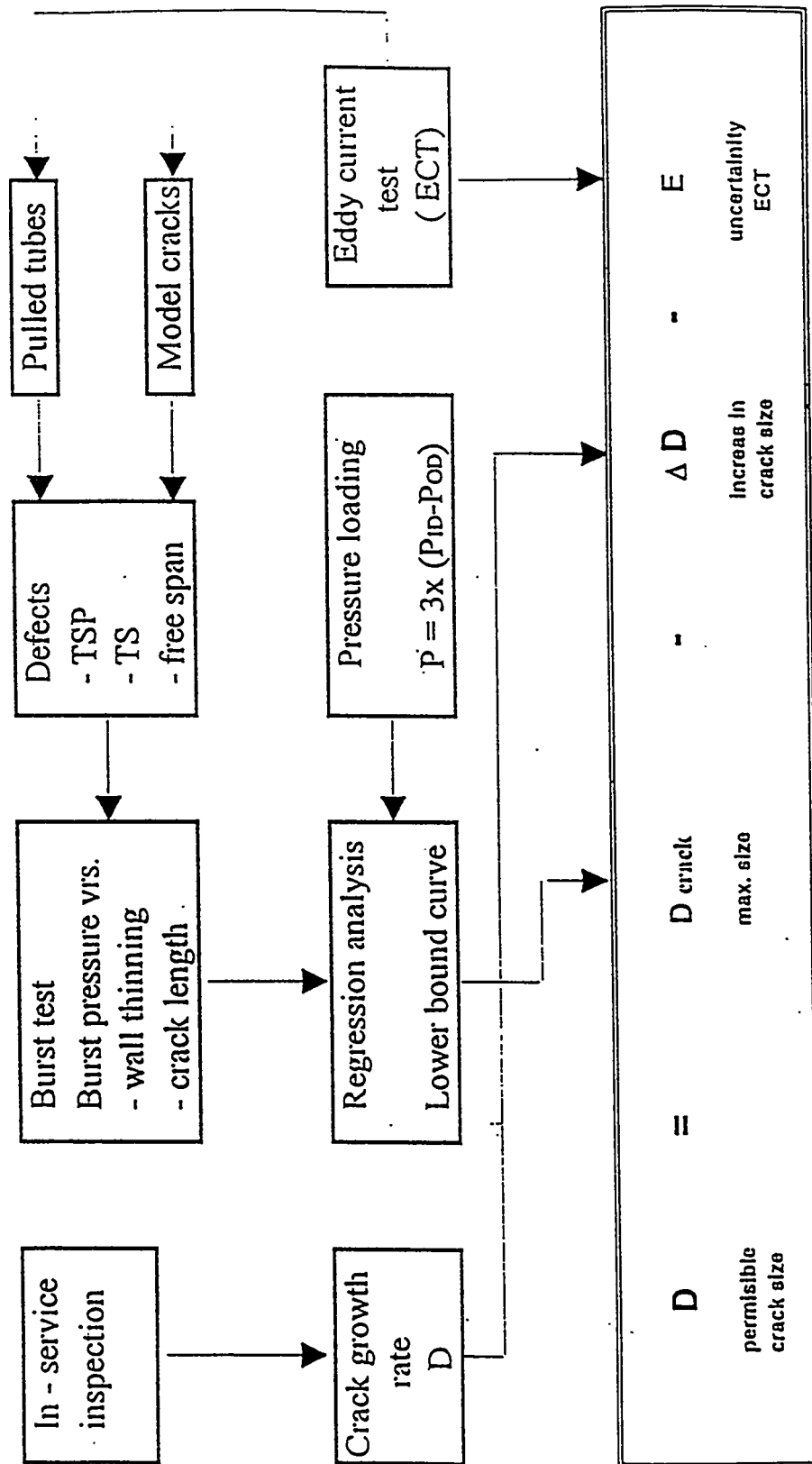
RELATION BETWEEN CRACKS SIZE AND LEAKAGE

Number of tube specimens	Cracks		Leakage
	Inner width [μm]	Inner length [mm]	
8	1.3 - 5	1.4 - 6.0	without leakage
20	5 - 20	2.3 - 13.0	blocking of leakage
5	20.0 - 86.0	1.4 - 14.0	leakage

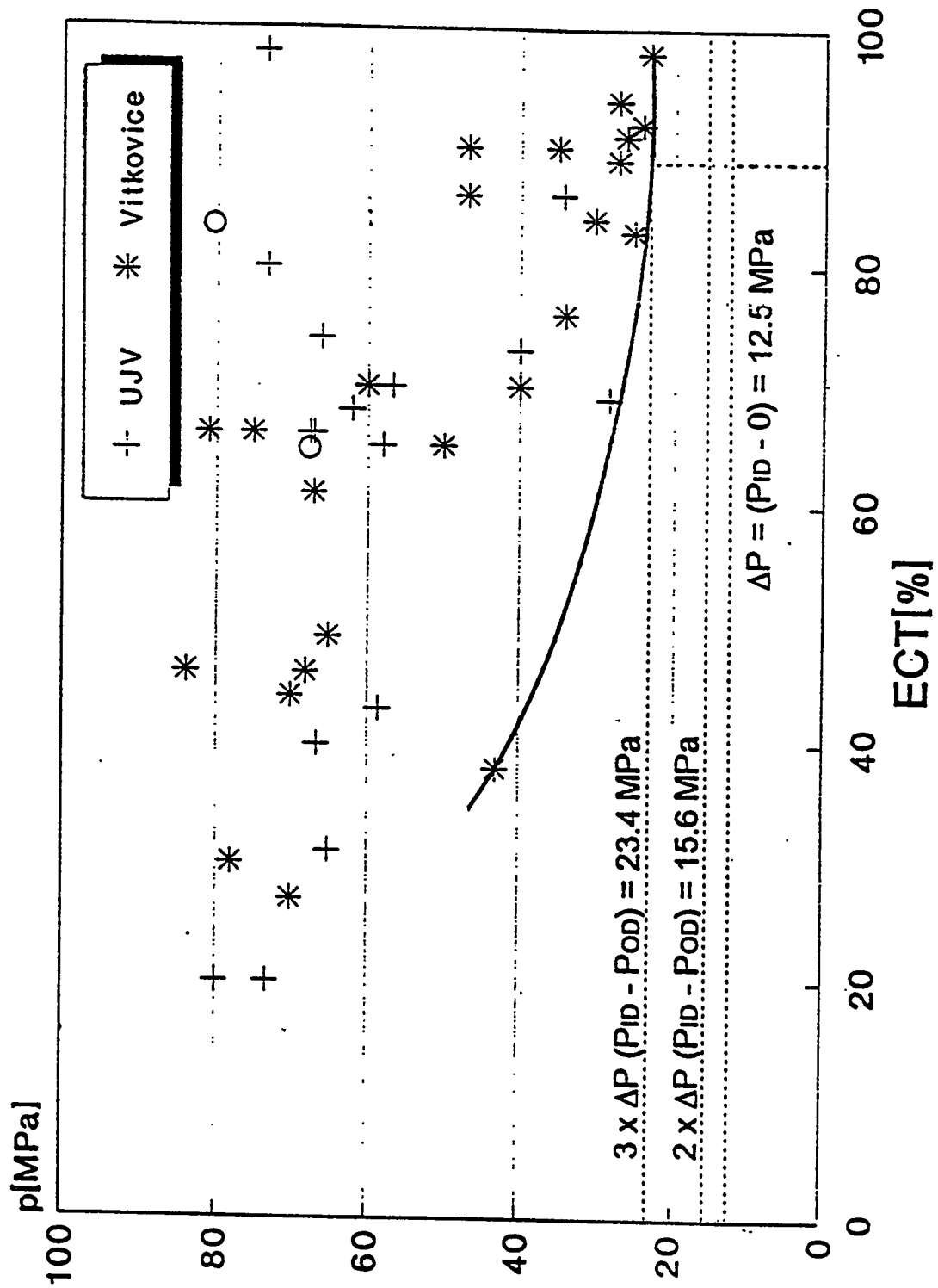
WWER 440 PRIMARY WATER PARTICLE SIZES

Power plat	Operating time [hours]	Range of partical size [μm]	Average partical size [μm]
EDU	cca 14 000	0.6 - 4.4	1
EBO	7 000 - 22 000	1 - 6	4.5 - 5.0
KKW	60 000	1 - 30	(cca 5.0)
Reactor water loop	11 000	0.5 - 30	7.3

STEAM GENERATOR TUBE PLUGGING



BURST TEST - SG WWER 440 MW



LEAK RATE LIMITS

Loading [MPa]	Critical crack length [mm]	Safety margin	Crack length [mm]	Leak rate [l/h]
$\Delta P = 7.8$	-	3 $\Delta P = 23.4$	11.3	8
12.5	28.5	$\sqrt{2}$	20.1	

PLUGGING LIMITS

PERCENTAGE OF TUBE THINNING

Loading [MPa]	Safety factor	Evaluated value [%]	Accuracy ECT	Suggested value	Used value TSP
$\Delta P = 3 \times 7.8$ $= 23.4$	3	88	15 %	75 %	70 - 75 %
$\Delta P = 2 \times 7.8$ $= 15.6$	2	< 100	15 %	85 %	
$P_D = 12.5$	-	< 100	15 %		

CONCLUSIONS

- the permissible value for the primary medium leak rate of 8 l.h^{-1} derived for one tube in the whole bundle;
- narrow through-wall cracks $3\text{-}5 \text{ }\mu\text{m}$ of width on the inner tube surface are impenetrable, narrow through-wall cracks about $5\text{-}20 \text{ }\mu\text{m}$ of inner width reduce the leak rate under conditions, where the inner crack width corresponds to the size of corrosion product particles or other compounds;
- 75 % wall thinning is a permissible value that sufficiently assures tube integrity. The use of this value is supported by the following facts ;

the critical pressure for tubes with wall thinning above 80 % is higher than the threefold value of primary-to-secondary pressure difference

through-wall cracks initiated under operation condition are narrow and do not cause an excessive leakage of primary water, they do not initiate unstable rupture of tubes under burst test pressure up to 29 MPa

primary water leakage is monitored by N^{16} measurement or other similar sensitive methods

- the corrosion cracking and pitting of the 08ChN10T steel is affected mainly by copper and chlorides, it is recommended to enhance the reliability of operational measurement of condenser tubes, and possibly substitute copper based materials in the secondary circuit.

STATUS OF STEAM GENERATOR TUBING INTEGRITY AT JASLOVSKÉ BOHUNICE NPP

Mr. Štefan Čepček

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Abstract:

Steamgenerator represents one of the most important component of nuclear power plants. Especially, loss of tubing integrity of steam generators can lead to the primary coolant leak to secondary circuit and in worse cases to the unit shut down or to the PTS events occurrence.

Therefore, to ensure the steamgenerator tubing integrity and the current knowledge about tube degradation propagation and development is of the highest importance.

In the paper the present status of steam generator tubing integrity in operated NPP in Slovak Republic is presented.

1. INTRODUCTION

Nuclear power plants operated and constructed are represent by WWER-440/230 and WWER-440/213 reactor type units.

In Bohunice NPP there are under operation 4 units,

2 units of WWER - 440/230 type

2 units of WWER - 440/213 type

Terms of commissioning:

Unit 1: 1978

Unit 2: 1980

Unit 3: 1984

Unit 4: 1985

Each unit consists of 6 loops with horizontal steamgenerators, each loop is equipped by two main isolation valves, one on hot and one on cold leg. Valves enable the isolation and draining of steamgenerator for inservice inspection and repairs.

The operation of unit from the point of view of steamgenerators is up to now very reliable.

No one leakage has been occurred at unit 1, unit 2 and unit 4 steamgenerator tube bundles. only at 4 steamgenerators of unit 3 were leakages, as it is showed in next paragraphs.

2. WWER 440 STEAM GENERATOR DESIGN

The WWER 440 steam generator is horizontal pressure vessel with two integrated vertical headers (collectors) and with two horizontal tube bundles.

The tube sheet is created in fact by the cylindrical wall of the headers.

The number of the heat exchange tubes per one steam generator is 5 536 pieces. From this amount, 18 % of heat exchange tubes represent the reserve for plugging and/or for heat transfer efficiency decreasing.

The "rolling" of SG tubes in header was performed by using explosion technology and after then tubes were welded to the headers.

The tube bundles are supported by 26 supports.

The material of the heat exchange tube is an austenitic chromium-nickel steel stabilized by titanium, 08 KH 18 N 10T, the headers are made of the same steel quality.

3. IN-SERVICE INSPECTION

In accordance with the original (former Soviet Union) documentation, the in-service inspection was restricted to the visual inspection of the tube bundle in accessible areas from the secondary side and to the visual inspection of the tube-to-header welds in the headers. Only, in the case of activity occurrence in secondary circuit, leak test was prescribed.

For in-service inspection four years inspection interval is prescribed.

In 1988, the Eddy-Current method was introduced for SG tubes inspection and the inspection is regularly performed from 1989.

The scope of Eddy-Current inspection according to the inspection procedure is set out to 10 % of all tubes, but nowadays it was supplemented to 100 % of all tubes.

The scope of SG tubes inspection performed on Bohunice NPP units is shown in table 1.

It must be pointed out that the first inspection by using Eddy-Current method was performed after 10 years of operation. Therefore, in order to have an overview about SG quality to the disposition and in order to have the basis for next ISI planning and the reject criteria for plugging development, 100 % scope of SG tubes is applied.

The results of inspection are quite appropriate, the number of defected tubes is relatively low, especially concerning defects above 50 % of wall thickness reduction.

At unit 3 only, 29 tubes have been plugged based on results of Eddy-Current-inspection.

During last inspections, it was observed that the Eddy-Current indication are mostly on outer surface in support area.

Due to a small inner diameter and quite small bending radius of the tubes ($R = 50$ mm) it is problem to introduce inspection by using rotation pancake coil. Nowadays, in Nuclear Power Plants Research Institute (VÚJE) Trnava, Slovak Republic, the module for this examination is under development. Expected length of the tube to be inspected by using this module is 1 m approximately.

According to the defected tube distribution in tube sheet, it is not possible to confirm the dependency of the defect on region.

4. OPERATIONAL EXPERIENCE

Operational experiences reflect the condition of the tube bundle of steam generators.

From this point of view, no one shut down of unit 1,2 and 4 have been caused by problems with the tube integrity.

Only, at unit 3 some repairs during outages and during operation were made due to a leakages. It is supposed that the presented defects can have their origin in manufacturing - so called technological defects, which was partially confirmed during analysis of removed tubes.

A quite serious problem is the identification of leak tube. In such cases, the pressure test from secondary side of SG is performed and the leakages are observed by underwater TV camera (SG headers and tubes are filled by water).

The activity limit in secondary side of SG is 370 kBq .m³ in blowdown water and this limit represents the leakage corresponding about 1 litre/hour. Of course this threshold limit can be revised according to on actual situation.

The possibility to limit the primary - to secondary SGs side leakage to 5 litres per hour is considered.

Due to a rather complicated procedure for detection of exact SG and the delay in leak evaluation, it is planned to install N¹⁶ activity measurement in steam.

5. PLUGGING CRITERIA

At present, no reject criteria for steam generator tubes plugging exist.

The plugging is performed in following cases:

- if in blowdown water is an activity identified - the tube with leakage has to be find, e.g. by using leak test
- if the results of in-service inspection are assessed as not appropriate for further operation, the decision for plugging is accepted after the assessment of:
 - the actual damage of the tubes (wall thickness reduction, Eddy-Current parameters)
 - number of detected indications
 - following steam generator inspection.

This year a quite wide research programme has been started with the aim to develop criteria for tube plugging. The research is performed by Nuclear Power Plants Research Institute (VÚJE) Trnava-Slovak Republic, Nuclear Research Institute (ÚJV) Řež, Czech Republic and

Vitkovice Steel Works Research Institute, Vitkovice, Czech Republic.

In the first step, the goal of programme is to develop a database of:

- results of in-service inspection
- results of material research of steam generator tubes
- operational loadings
- water chemistry during the operation and outages, etc.

In the second step, the reject criteria will be established and they will consist of.

- maximum acceptable wall thickness reduction of the tube
- procedure for following steam generator in-service inspection planning

It is expected to finish this programme in February, 1996.

6. INTEGRITY ASSESSMENT

In previous several years, some work relating to steam generator tubes integrity took place, which can be very briefly described as follows.

- Experimentally was verified the leak rate of through - wall cracked tubes.
- The comparison of real (measured) wall thickness reduction and the wall thickness reduction obtained from Eddy-Current testing.
- Pressure test of the pre-cracked tubes (artificial defects) up to the tube break
- Crack propagation rate (0.5 CT specimens were used).
- Assessment of collapse of the tubes at the loss of coolant condition.

The main results:

It was stated that the leak rate of through-wall crack depends on the geometrical parameters and on corrosion products in the coolant and in the cracks.

- The comparison of measured wall thickness reduction versus Eddy-Current testing results showed relatively good agreement, up to 70 % of wall reduction, below this value, EC testing gives smaller values.
- Pressure tests of the tubes up to the tube break showed very high value of critical pressure, e. g.: for defectless tube at 290 °C, $p = 93,4$ MPa for defect of 22,5 mm for length and 70% of wall thickness at 290 °C, $p = 37$ MPa.
- The rates of crack propagation in laboratory condition were obtained at the different values of O_2
- It was stated that the collapse of steam generator tubes at the loss of coolant condition are not of importance.

At present, as it was mentioned, a very comprehensive programme for reject criteria development for the SG tube plugging is under way.

Another task under way, which was ordered by ACT No 1/94 of Nuclear Regulatory

Authority of the Slovak Republic is the assessment of steam generator tube bundle integrity at the steam line break condition.

8. CONCLUSION

1. Steam generators of Bohunice NPP showed very good condition of heat exchange tubes. Up to now leakages appeared at unit 3 only and total amount of plugged tubes represents 0.1%.
2. The reject criteria for the steam generator tubes plugging are under development.
3. The assessment of steam generator tube bundle integrity at the steam line break condition is under way.

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Řež, 1993.
2. Šplichal K. and others: WWER-440 Steam Generator Tubes Integrity Assessment.
Řež, 1993.
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UNIT NO	SG number and percentage of inspection						
	1989	1990	1991	1992	1993	1994	1995
1	13-10 %	11-7 %	12-15 %	13-100% 16-15 %	11-100% 15-100%	12-100% 16-100%	14-100%
2	24-10 %	25-10 %	22-15 % 26-15 %	-	21-100% 24-100%	25-100% 26-100%	22-100% 23-100%
3	31-10 %	31-40 % hot leg	31-100% 36-40 % hot leg	31-10 % 32-40 % 36-100%	33-100%	34-100% 35-100%	31-100% 32-100% 33-* 34-* 35-* 36-100%
4	41-10 %	44-12 %	43-15 % 45-15 %	42-100% 46-100%	41-100% 43-100%	44-100% 45-100%	42-100%

* selected scope

Tab. 1. Inspection of Bohunice steam generator exchange tubes

The reasons fro plugging	Number of tubes					
	SG31	SG32	SG33	SG34	SG35	SG36
sampling for material analysis	6	1	3	1	2	-
leakage of the tubes	22	6	-	-	11	41
Eddy-current results	13	2	2	2	3	7
other reasons	9	-	-	3	-	1
total No per SG	50	9	5	6	16	49
total - unit 3	135					

Table 2: Unit 3 - Plugging of SG tubes

9 Working Session 4: Preventative and Corrective Measures

R. Clark, Golder Associates, and G. Slama, Framatome

9.1 Introduction

The Preventive and Corrective Measures working session included 13 members from France, Germany, Japan, Spain, Slovenia, and the United States. Attendee experience included regulators, utilities, three steam generator vendors, consultants and researchers. Discussions centered on four principal topics: (1) alternate materials, (2) mechanical mitigation, (3) maintenance, and (4) water chemistry. New or replacement steam generators and original equipment steam generators were separately addressed. Four papers were presented to the session, to provide information and stimulate various discussion topics. Topics discussed and issues raised during the several meeting sessions are provided below, followed by summary conclusions and recommendations on which the group was able to reach a majority consensus. The working session was composed of individuals with diverse experience and varied areas of specialized expertise. The somewhat broad range of topics addressed by the group at times saw discussion participation by only a few individuals. As in any technical meeting where all are allowed the opportunity to speak their mind, straying from an individual topic was not unusual. Where useful, these stray topics are also presented below within the context in which they occurred.

9.2 Alternate Materials

The session was initiated with presentation by Mr. Lozano of the paper *Spanish Approach to Research and Development Applied to Steam Generator Tubes Structural Integrity and Life Management*, authored by Mr. Lozano and Mr. Bollini. The paper deals with a number of topics, including degradation history in Spanish steam generators and application of various degradation specific nondestructive testing (NDT) techniques. Programs for development of plugging criteria for primary water stress corrosion cracking (PWSCC) at the roll transition zone (RTZ) and outer diameter stress corrosion cracking (ODSCC) at tube support plates are mentioned. Mitigation and repair techniques under research include a cooperative program with Belgium and Sweden on qualification of sleeving and a program evaluating removable tube plugs. The rest of the paper and much of the session discussion dealt with the processes leading to choice of materials and design for replacement steam generators.

The Spanish have decided to replace steam generators at Almaraz 1 and 2 and Asco 1 and 2 with Siemens-Framatome steam generators tubed in Incoloy 800 NG. The original generators were Westinghouse Model D-3 preheater type units tubed in mill-annealed Alloy 600. The decision to replace steam generators was based on ODSCC degradation; top of tubesheet (TTS) cracking was not considered to be an issue leading to excessive tube plugging due to adaptation of a leak before break (LBB) plugging criteria in Spain.

A test program, on candidate alternate steam generator tubing material was conducted. Mr. Lozano noted that the testing at Ciemat showed that Alloy 690 was better overall than Alloy 800, but the test results did not eliminate either material as a viable candidate. The ultimate choice of Incoloy 800 was based on the extended favorable service experience with this alloy plus the willingness of the vendor to warranty this material for 15 years. Warranty provisions include vendor payment for plugging and sleeving if necessary to return to rated thermal capacity.

The design of the nonpreheater type replacement units includes a triangular tube pitch and higher tube bundle, providing sufficient heat transfer surface within the original steam generator envelope to eliminate the preheater and provide for uprating. The replacement units operate at 610°F, and provide 10°F margin for uprating or plugging. Because there is less water in the secondary system a lower margin for transients was mentioned. Also, there is some remaining concern with the possible effects of lead contamination and of tube fouling.

Discussions on the experience factor brought up issues of how similar the Spanish replacement units are to the KWU standard steam generator product. It was mentioned that the replacement units will have different tube spacing, bundle height, and blowdown rates. The Spanish specified the thermal-hydraulics operating window and ran ATHOS code evaluations to assess plant compatibility.

It was generally agreed that Alloy 690TT and Alloy 800 are both probably adequate for the lifetime of steam generators, and that further work to examine and qualify another material is not called for. Most participants agreed that Alloy 690TT is preferable, but that Alloy 800 is adequate.

Design of replacement steam generator tube-to-tubesheet expansion was discussed. The replacement designs all utilize hydraulic expansions. The participants in this discussion appeared to agree that the hydraulic tube expansion in new steam generators leaves relatively low residual stresses, that when combined with increased tubesheet hole tolerance control and improved material resistance, cannot be improved upon.

Mr. Lozano's presentation mentioned that there was no problem with the Vandellós II plant, which employs Westinghouse Model F steam generators with Alloy 600TT tubing, or with the Trillo plant utilizing KWU steam generators and Incoloy 800 tubing. Discussion then diverged into differences between water chemistry philosophy between KWU design plants and the U.S. approach. The U.S. position was characterized as counteracting secondary side erosion/corrosion with pH control. The German (KWU) approach is to allow sufficient oxygen in the secondary system to control erosion. Steam generator feedwater is then deaerated ahead of the final feedwater preheater. KWU systems do not generally use condensate polishing, emphasizing condenser leak tightness, but have condensate polishers available on standby at brackish sites. German water chemistry control has only two control parameters, cation conductivity and sodium concentration (Na+), at much higher limits (5-8 ppb in the steam generator), than the current

EPRI water chemistry recommendation. Also discussed was the issue of lead contamination and levels of importance. No conclusion was reached relative to the lead issue.

The paper *Actual Operation and Regulatory Activities on Steam Generator Replacement in Japan*, by Mr. Hitoshi Saeki, was presented with the assistance of Mr. T. Iwase. The paper starts with a summary of nuclear power plant status, operational and under construction, by type of plant. It continues with a review of current capacity factors, with a mean of $\approx 70\%$. A chart of annual inspection durations drew some comment, especially concerning the very long annual outages: 8 of 19 plants shown over 100 days. These were explained in terms of repair but no specifics provided. The types of degradation at each plant were next listed, along with percentages of tubes plugged and sleeved. The paper then provided an extensive discussion on planned Japanese steam generator replacement, listing plants, dates for replacement, and providing details of planning and procedures. The concluding portions of the paper dealt with the regulatory process and procedures associated with the steam generator replacements and prestart inspections.

Following this presentation, there was discussion on Japanese inspection and tube plugging criteria. A 100% inspection is required at each annual outage. No leakage is allowed and all damaged tubes are either plugged or sleeved. The definition of "damaged tube" was sought and a response given in terms of eddy current bobbin coil voltage. However, it was unclear to this reviewer as to the voltage calibration and how the damaged tube voltage criteria might correlate to plugging criteria in other countries. Questioned about sleeving, Mr. Iwase responded based on his experience at Genkai, where laser-welded sleeves with no subsequent heat treating were installed in 1989, with no subsequent problems.

Replacement steam generator design used in Japan was explained based on the Genkai unit 1 replacement. The unit, which is a feedring type recirculating steam generator, employs Alloy 690TT tubing supported at broached concave-land stainless steel tube support plates. The number of antivibration bars (AVBs) had been increased from two in the original units to three in the replacement steam generators. Blowdown capability was described as between 1.3 and 2.3%

Questioned concerning Japanese plans to pursue research utilizing retired steam generators, Mr. Iwase responded that there is no plan to conduct research on the generators or to take samples from them. The replaced units are being stored sealed in on-site mausoleums.

The workshop continued with general discussion on issues of possible continuing concern, specifically as related to replacement steam generator design features.

There was some repetition of discussion on Alloy 690TT vs. Incoloy 800, specifically related to lead contamination. It was recognized that losses directly

attributable to lead were associated with large ingress, and that there is insufficient understanding concerning lead limits related to tubing degradation. Tube support plate design and remaining crevices were considered. Comparing the flat-land broached and grid TSP designs, no significant difference in performance was expected, as long as appropriate materials were used. Some questioned why the Japanese replacements used concave broached holes, if this is indeed the case.

The experience with broached hole deposit plugging at Surry and Callaway and with their successful cleaning was reviewed. This led into considerable deliberation concerning the need for efforts to minimize sludge inputs into steam generators, as the key to avoiding most other problems. A broad consensus was achieved on this issue. Input from a KWU representative indicated that some of their units had run 12 years without sludge lancing, because none was present under their operating chemistry. It was still recommended by KWU that sludge lancing be conducted every 4 years to avoid formation of a hard scale at the TTS.

9.3 Maintenance

The next topic taken up by the working group was maintenance. The session opened with a paper entitled *Steam Generators Secondary Side Chemical Cleaning at Point Lepreau Using the Siemens High Temperature Process*, by K. Verma, C. MacNeil, S. Odar, and K. Kunke. The paper was presented by Dr. Odar. The Point Lepreau plant is a pressurized heavy water reactor (PWR) with four B&W Ltd. fabricated steam generators employing Incoloy 800 tubing. Tube support plates are broached-hole stainless steel. The secondary side was essentially copper-free. Initial operation in 1982 utilized a secondary side congruent phosphate treatment. In 1988, secondary chemistry was converted to an all volatile treatment (AVT), ammonia plus morpholine. Starting in 1986, full-flow condensate polishing was added to protect from seawater leakage at the condenser. After 12 years service, deposits had built up on the primary side due to carbon steel in the primary circuits. Secondary side sludge buildup had resulted in partial blockage of the first two tube support plate (TSP) broached holes, along with initial blockage at upper TSPs. Tube corrosion in the sludge regions and a loss of heat transfer capacity led to the chemical cleaning decision for the secondary side. The primary side was mechanically cleaned.

The paper provides information on sludge pile chemistry. A description of the selection procedure involving evaluation of all commercially available chemical cleaning processes followed. Dr. Odar then presented the process application steps. Principal features of the process were the hot application temperature, 160-200°C, allowing chemical mixing by steam off. Alkaline and reducing conditions were maintained for corrosion control. Chemicals were injected stepwise as the liquid level was raised. Excess solvent EDTA was added, with no additional inhibitors. The cleaning resulted in removal of 124 kg of salts, including 70 kg of phosphates. In addition, over 250 kg of magnetite were removed from each of the generators. Corrosion test results were reported.

Among other results no selective corrosion was observed, such as pitting or stress corrosion cracking. The tube surfaces were described as cleaned to shiny metal. A few TSP trefoil broaches showed some remaining deposits, and limited areas of tubesheet deposits were also present.

Group discussion followed on the question of assuring adequate circulation during chemical cleaning. The concern was pitting in the stagnant area, specifically the shell weld heat affected zone, between the shroud and shell, since there is no local boiling in the downcomer region. Application under reducing environments seeks to avoid issues of galvanic corrosion. It was expressed among the group that pitting has been an issue in stagnant zones. The overall issue in corrosion prevention was to assure that all exposed regions of the generator remained within the window of process qualification.

The Callaway experience with Westinghouse pressure pulse cleaning was presented by one of the workshop participants. This cleaning was necessitated by pressure drop attributed to blockage of broached quatrefoil TSP holes, significant at or above TSP 5. It was reported that 17,000 lb. of sludge were removed in the iron cleaning step and 120 pounds during the copper cleaning step. There was a concern, but no apparent problem, with pitting in 533 to 508 welds. The cleaning resulted in total cleaning of the broached holes. Discussion continued with mention that application of Westinghouse pressure pulse cleaning to Sequoyah 1, with drilled hole carbon steel TSPs, resulted in cleaning of crevices which were not initially fully restricted or dented, but had no effect on fully packed or dented crevices.

An experiment was described in which a crevice tube support plate plus tube section removed from Dampierre 1 was included as a sample in the chemical cleaning of Ringhals 3 using the EPRI chemical cleaning process. The sample was subsequently sectioned and 50 to 70% of the crevice was described as cleaned.

The remaining sludge at Pt. Lepreau, estimated at 5%, was reported as very hard and containing difficult-to-dissolve silicates and phosphates. This led to group discussion on the use of chemical cleaning as a preventative measure, applied before serious accumulation and TSP plugging. This concept was taken further, to the thought that replacement steam generators might usefully be designed with a chemical cleaning margin comprised of a suitable corrosion margin and possibly a margin to fatigue from pulse cleaning, to allow cleaning every X years. The potential usefulness of developing mechanical cleaning technique(s) for upper TSP regions was also put forward.

9.4 Mechanical Mitigation

The next topic of discussion was sleeving. This topic was initiated by a presentation of the paper *Repair Boundary for Parent Tube Indications Within the Upper Joint Zone of Hybrid Expansion Joint (HEJ) Sleeved Tubes*, authored and presented by Mr. William Cullen. This paper provides a description of HEJ sleeve

and its manner of installation. The hybrid expansion joint consists of a hydraulically expanded sleeve end with a roll expansion within the hydraulically expanded region. The paper continues with discussion of service degradation experienced with a subpopulation of the $\approx 10,000$ such sleeves remaining in service. It points out that 98% of such service degradation is associated with cracking in the parent tube located at the upper joint in either the lower end of the hydraulic expansion or the lower end of the hard roll transition. Mr. Cullen attributed the failures to water entering the gap between the tube and sleeve. He indicated that cracking was typically $\approx 90^\circ$ of circumference.

The presentation and paper continued with description of the HEJ sleeved tube and leakage integrity testing program. An F* type analysis was described. (F* refers to an evaluation based on the length of tubing expansion necessary to resist the pull-out forces under normal and accident conditions, capable of preventing a circumferentially severed tube from being pulled free; this method has been applied to tube cracking in the tubesheet expansion region). The reported analysis showed that 360°, 100% TW (through wall) cracks at least 1.1 in. (2.8 cm) below the bottom of the upper joint hardroll upper transition would meet the structural integrity criteria of USNRC Regulatory Guide 1.121 "Bases for Plugging Degraded PWR Steam Generator Tubes." The paper reported leak test bounding values under steam line break conditions. Significant leakage was described as being possible only with postulated axial displacement separation of the parent tube. Such displacement is resisted by the sleeve and the sleeve-to-parent tube joints. Based on the analysis provided in the paper, a modified tube repair criterion for parent tubing of HEJ sleeves was proposed to NRC.

The presentation continued with description of individual plant experiences. Kewaunee sought a licensing amendment based on limiting angular extent of cracking. This was reported as rejected by NRC due to limited time constraint for arriving at a decision consistent with the planned outage schedule. A similar approach was applied for at Point Beach Unit 2. Evaluation of the approach led to rejection based on uncertainties in eddy current ability to measure crack angle, uncertainties in crack growth rates, and hypothesized potential for rapid crack growth rates. D.C. Cook applied for a licensing amendment based on a modified pressure boundary classification, described in the paper. This used a P* criteria, based on the limited amount of possible tube lift resulting in limited leakage potential. The licensing amendment application was reported as being rejected due to time constraints. The author indicated that Westinghouse supports the HEJ sleeve design, which provides the advantages of rapid installation and lower cost than competing sleeve designs.

The workshop continued with a discussion on sleeving topics. Individual plant experiences were expressed by those present with direct knowledge, for a limited number of plants. Experience at Krsko and Asco indicated some unsuccessful installations, plus early failures. These experiences were with qualified sleeving situation, per the Belgian, Swedish, Spanish research program. There was a discussion of the recent experiences at Doel 4, with no consensus on what the

significant factors were leading to failures in sleeved tubes. Mentioned was the tube bending associated with tubes locked into the first TSP and the possible thermal expansion effects due to use of an oscillating (traveling) heater for weld stress relief.

Concern was expressed regarding nondestructive inspection capability of the parent tube in sleeved tubes. Vendor representatives in the meeting claimed that the parent tube can be seen behind the sleeve. Discussion continued into acceptance criteria for a sleeving service vendor qualification program. The following points regarding sleeving were generally arrived at:

- Qualification needs to take specifics of each plant situation into account, for example locking of tubes into adjacent TSPs, and other issues of susceptibility specific to each unit (without further elaboration at this time).
- Qualify NDE of sleeved parent tubing, to the extent sufficient to ensure adequate safety decisions.
- A checklist of qualification specificity should be developed.

Further discussion on sleeving involved the issue of sleeve life expectancy. It was generally felt that users wished to be assured that sleeves will survive the remainder of steam generator operating life.

The final point of discussion on sleeving considered verification of sleeving performance. In particular, a polling of knowledge concerning how the performance of sleeves and information on sleeve life expectancy might be achieved. Aside from in-service inspection, nondestructive methods, tube pulls, and secondary side video were suggested.

The workshop session continued with discussion of other repair or tube modification processes. Direct tube repair considered electroplating techniques, weldment additions, and an autogenous remelt process. The Belgians and French have developed a nickel plating technique. This technique involves electroplating nickel over the inner diameter surface of tubing in regions subject to PWSCC. The Belgians have produced papers which indicate that the repair prevents crack initiation and also stops further crack growth, even for through-wall defects. Nickel electroplating is widely applied only in Belgium, where large numbers of tubes in the tubesheet expansion transition zone have been treated and several operating years of favorable experience gained. Limited application in other countries was related to the need for ultrasonic (UT) inspection, since eddy current will not work through Ni. The Belgians have conducted extensive UT development and routinely use UT in their inspections. Time constraints and complications of adding UT inspection, where it is not routinely a part of in-service inspection, were presented as reasons for limited application outside of Belgium.

The French utility has decided, based on economic analysis for their situation, that plugging and steam generator replacement is their preferred maintenance path. French plants also have the advantage of common design, and have a set of replacement steam generators available on short notice, which could address rapid degradation appearing at a specific plant. French participants at the meeting also voiced some concern over NDE inspectability of sleeve joints.

The Canadians have recently developed a 30-mil-thick structural sleeve, which is placed in-situ and is claimed to leave no residual stress. This repair was represented as being applicable at any location within the tube. No one in the working session had detailed knowledge concerning the process parameters.

Mention was made of the EPRI-J.A. Jones direct tube repair process which has apparently reached a commercial licensing state. The method per the working groups understanding involves in-situ placement of weld material on the tubing inner diameter to arrive at a near-original nondegraded wall thickness, from weldment plus tubing remelt. The weld material is chosen to result in an Alloy 690 composition net weld composition.

Autogenous tube repair through in-situ remelting of the tubing from the inner diameter was briefly discussed. This is a technique Westinghouse sought to develop. The current Westinghouse approach is to melt a thin sleeve into place, resulting in an equivalent Alloy 690 material. The process is apparently relatively time-intensive and presents inspectability issues arising from surface roughness (affecting UT) and magnetic effects with eddy current testing due to heating of the parent tube. While much less shrinkage was ascribed to the thin tube approach than with direct tube melting, it produces more shrinkage than laser welding of sleeves. This in turn led to discussion of the issue of far field stress effects.

Direct tube repair methods indicated advantages of less flow restriction and the ability to repair above existing sleeves. However, methods requiring stress relief would not work in the tubesheet region, since stress relief is not possible in this location. Remaining issues include far field stress effects, inspectability, and definition of what constitutes the primary system boundary in the repaired region.

Peening (shot-peening or roto-peening) as a mechanical remediation was next discussed. The discussion was initiated by raising the question of peening life expectancy. There was no disagreement that experience has so far shown that plants peened prior to service have not suffered from significant primary side degradation in the peened region. The position expressed by French participants was that plants peened after service exhibit at least a few years of benefit, less PWSCC in the expansion transition region, but then may return to a degradation pattern as if no peening had been applied. Peening, because it affects only a very shallow surface region, would not prevent eventual propagation of cracks which have initiated at the time of peening application. The issue on after-service application is concerned with service time versus crack initiation time. The experience at some plants appears to indicate for their conditions (tubing

sensitivity and operating temperature) significant initiation had occurred within the first operating cycle. Doel 4, for example, intentionally left a sample of tubes in nonpeened condition; these tubes cracked during the first two operating cycles.

Peening for replacement steam generators was discussed. Apparently replacement steam generators are not peened in the expansion transition region. The thoughts presented in the working session were that reduced stresses from maintaining close tolerances on tubesheet holes and the use of hydraulic expansion, coupled with the apparent immunity of Alloy 690TT to PWSCC, eliminated the need to peen these steam generators.

9.5 Water Chemistry

Workshop session discussions related to water chemistry were somewhat restrained by the limited number of experienced participants, although a couple of session attendees had considerable depth in this area.

As previously mentioned in discussion following presentation of the Spanish paper, there is a considerable difference between the secondary side water chemistry recommended by KWU-Siemens for their units and the EPRI-recommended water chemistry guidelines. These differences in approach however were seen to have the common goal of seeking minimization of sludge forming contaminants entering the steam generator, the German approach utilizing higher secondary side oxygen concentration prior to entering the steam generator through a deaerator and the other approach based on chemical control of erosion in a deoxidized secondary side, e.g., hydrazine plus morpholine.

Most of the group discussion centered on the issue of crevice chemistry, discussed largely in the context of whether molar ratio control (MRC) is a reasonable approach. MRC refers to a method of secondary side water treatment that seeks to control impurity content, particularly strong acid-forming anions and strong base forming cations, at relative levels such that when these impurities concentrate in crevice regions a near-neutral pH crevice chemistry is formed.

The comment from a KWU-Siemens water chemistry expert was that a variety of crevice conditions exist within every steam generator. Altering crevice-chemistry-forming conditions may cause a group of crevices to enter a corrosive regime. Therefore KWU-Siemens does not recommend an MRC chemistry approach.

Comments from French participants in the working session indicated that the French do not use MRC per se. It was stated that there is no way of knowing the actual crevice conditions achieved through the concentration of bulk impurities. Not knowing these conditions affects the ability to postulate an appropriate MRC regime.

Discussion then continued into how a better characterization of crevice conditions could be achieved. The group strongly recommended mining of samples, including tube-to-support-plate intersections, from retired steam generators. Discussion favored mining sufficient samples to assess how crevices differ within a single generator, as well as a comparison of samples from different generators, e.g., those removed from different plants, although it was felt there may also be significant differences between generators in a single plant.

The other path toward establishing crevice chemistry is in the laboratory, such as model boiler testing. Discussion on different approaches appeared to indicate the preferred methods would utilize a side stream boiler, that is a small pressure vessel with appropriate samples which accepts feed directly from the reactor, and/or utilizing the autoclave type system that Jesse Lumsden at Rockwell appears to have developed.

Further discussion on water chemistry next addressed inhibitors. The injection of TiO_2 was considered. It was felt there was lack of knowledge as to whether this or other inhibitors could actually get into a crevice. There is some plant application now; TiO_2 was injected after chemical cleaning at the Callaway plant, which has Model F Westinghouse generators, on the basis that the inhibitors cannot hurt. Boric acid addition raises the issue of pH changes, potentially increasing the amount of sludge. Uncertainty was expressed about the lack of knowledge as to the precise method whereby boric acid works.

9.6 Summary

The final portion of the workshop session sought to establish summary conclusions that were more or less the consensus group opinion, for presentation to the whole audience of conference attendees. Summary conclusions of the group were:

- **Minimize sludge.** Secondary side degradation in steam generators results from occluded areas which allow chemical concentration: (1) Efforts must concentrate on minimizing the ingress of sludge into steam generators, (2) changes in chemistry/operating parameters must consider the effects (e.g., leading to sludge generation) on the entire secondary circuit, and (3) consideration should be given to the use of preventative chemical cleaning to avoid deposit occlusion in addition to its use as a curative fix for already occluded regions.
- **New Steam Generators.** Design features that minimize crevices, eliminate or reduce residual tubing stresses, and employ enhanced materials in new/replacement steam generators should preclude current forms of tubing degradation during a nominal 40-year lifetime for units operated within accepted chemistry bounds and kept free of deposit buildup. There is a remaining crevice in the stressed region at the top of tubesheet expansion zone that can lead to chemical concentrations if sludge is allowed to build up.

Relatively low stresses from hydraulic tube expansion, coupled with increased material resistance, should provide a corrosion degradation margin. However chemically concentrating conditions should be avoided by limiting top of tubesheet sludge accumulation. Incoloy 800NG and Inconel 690TT are sufficient, and there is no need to research and qualify additional alloy materials for steam generator service. Designs should consider a sufficient corrosion margin for periodic chemical cleaning.

- **Maintenance.** Chemical cleaning is currently limited in its ability to effectively clean occluded drilled tube support plate (TSP) holes. Methods do not address silica removal, which ongoing current research suggests may be a significant factor in TSP region tubing corrosion. Excessive degradation can be experienced without careful maintenance of conditions within the process qualification bounds, such as unexpected or unaccounted for zones of solution stagnation or too high a flow rate. Numerous chemical cleaning operations have been successfully and safely applied.
- **Mechanical Mitigation.** (1) Sleeves: Early failures are often associated with application outside the window of process qualification. It is essential that steam generator conditions of each application have been sufficiently identified and addressed in qualification. Nondestructive examination techniques can provide adequate detection of degradation at safety limits. Validation of NDE, of sleeve/parent tube lifetime, and of sleeving effects on adjacent points of stress reaction needs further effort. There is insufficient information to reach a conclusion concerning expected sleeving service lifetimes. (2) Tube repair: Inner diameter nickel plating has shown positive results to date. The process has advantages of leaving no mechanical or thermal residual stresses and being applicable throughout the tube length. A disadvantage is the requirement for ultrasonic examination. Insufficient information exists to conclude service lifetime. (3) Tube repair: Direct tube repair through addition of weld material on the tube inner diameter is developmental and under current qualification. The method can be applied above an existing sleeve, requires thermal stress relief, and necessitates UT inspection. A potentially significant concern that requires addressing is the far field stress effects that could be generated as a result of thermal stress relief over the relatively large weld area. (4) Peening: Peening has proven effective when applied to plants prior to service for PWSCC suppression in the tube expansion zone. There have been no indications that the induced inner diameter compressive stress loses effectiveness with time. Peening of plants that have seen previous service cycles provides a degradation delay depending on tube condition at the time of application. However, damage initiated previous to application will proceed. Serious consideration should be made to cold leg peening in plants where PWSCC is the primary degradation concern.
- **Water Chemistry.** Characterization of steam generator crevices, chemistry, and morphology is inadequate. In addition, crevices through the steam generator, as well as individual crevices, are likely not a homogeneous

population. It is therefore difficult to evaluate remedial measures, molar ratio control, inhibitors, and alternate amines in light of the lack of a fundamental understanding of what conditions are being modified, as well as issues of accessibility. The session was unable to reach consensus on any particular chemistry regime, other than ensuring minimal corrosion product input into the steam generators.

Summary recommendations of the group for additional study were:

- Retired steam generators should be used for tube/tube support plate intersection removals to characterize (1) the range of crevice conditions (morphology of deposit and chemistry) in a steam generator and (2) differences between steam generators experiencing a range of degradation history. In addition, removal of samples for validation of repairs (sleeving) and far field effects of sleeving should be pursued.
- Qualification guidelines should be established for repair and maintenance techniques, e.g., chemical cleaning and sleeving.

**REPAIR BOUNDARY FOR PARENT TUBE INDICATIONS
WITHIN THE UPPER JOINT ZONE OF HYBRID EXPANSION JOINT
(HEJ) SLEEVED TUBES**

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Repair Boundary for Parent Tube Indications Within the Upper Joint Zone of Hybrid Expansion Joint (HEJ) Sleeved Tubes

Introduction

In the Spring and Fall of 1994, and the Spring of 1995, crack-like indications were found in the upper hybrid expansion joint (HEJ) region of Steam Generator (S/G) tubes which had been sleeved using Westinghouse HEJ sleeves. As a result of these findings, analytic and test evaluations were performed to assess the effect of the degradation on the structural, and leakage, integrity of the sleeve/tube joint relative to the requirements of the United States Nuclear Regulatory Commission's (NRC) draft Regulatory Guide (RG) 1.121. The results of these evaluations demonstrated that tubes with implied or known crack-like circumferential parent tube indications (PTIs) located 1.1" or farther below the bottom of the hardroll upper transition, have sufficient, and significant, integrity relative to the requirements of RG 1.121. Thus, the purpose of this report is to provide background information related to the justification of the modified tube repair boundary.

In accordance with Plant Technical Specification requirements, steam generator tubes are periodically inspected for degradation using nondestructive examination (NDE) techniques. If established degradation acceptance criteria are exceeded, the indication must be removed from service by plugging or repairing the tube. Tube sleeving is one repair technique used to return a tube to an operable condition.

In the sleeving technique, a smaller diameter tube, or sleeve, is positioned within the parent tube so as to span the degraded region. The ends of the sleeve are then secured to the parent tube forming a new pressure boundary and structural element between the attachment points. Sleeves may be positioned at any location along the straight length of a tube, but are typically placed to repair tube degradation at the top of, or within the tubesheet, or at tube support plate (TSP) intersections. Sleeves may be of various lengths and may be attached to the parent tube in a variety of ways. In the case of Westinghouse sleeve designs, the method of attachment is generally restricted to either a leak limiting mechanical HEJ or a hermetic Laser Welded Sleeve (LWS) joint. The type of the particular joint configuration is a function of the date of installation and/or the customer's needs and the current plant operating conditions. Figure 1 shows a schematic of a typical HEJ tubesheet sleeve installation. Figure 2 illustrates the details of the upper joint along with terminology used in this report. An illustration of modified repair boundary is provided on Figure 3.

A chronology of HEJ sleeve installations is provided in Table 1. Since its inception in 1980, the HEJ sleeve has been successfully used to restore over 28,000 steam generator tubes to operational status. Due to decommissioning of plants, replacement of steam generators, and plugging of sleeved tubes for parent tube indications, only about 11,000 domestic HEJ sleeves remain in service. The HEJ sleeves listed in Table 1 were installed between April 1983 and May 1993, and have operated in the United States without incidence of significant leakage through the upper joint, however, Doel 4 (Belgium) experienced leakage through an upper joint crack in a parent tube in April of 1994.

Until March of 1994 there had been no reports of degradation of the parent tubes or the sleeves. In 1994, degradation of the parent tube of HEJ sleeved tubes was detected at the Kewaunee, Point Beach 2, and Doel 4 power plants. At Kewaunee the indications were predominantly located in the hardroll lower transition (HRLT), while at Point Beach 2 the indications were predominately in the hydraulic expansion lower transition (HELT), and at Doel 4 the two confirmed indications were located at the hydraulic expansion upper transition (HEUT). Additional degradation was reported at Kewaunee in the Spring of 1995, and three sleeve/tube joints were removed from the "B" SG for laboratory examination. A summary of the elevation location of the detected indications at Kewaunee, Point Beach Unit 2 and Doel 4 is provided on Figure 4.

HEJ SLEEVED TUBE STRUCTURAL INTEGRITY REQUIREMENTS AND CHARACTERISTICS

The original qualification report for HEJ sleeves specified the entire tube hydraulically expanded length (including the roll expanded region) as part of the primary pressure boundary and thus necessitated its inspection using NDE techniques. Operability of parent tubes with detected degradation in the HEJ area would be assessed against the tube plugging limit of 40% (by NDE). At the time of the generation of the original qualification it was believed that the double-walled condition created by the sleeve would help to preclude any further degradation of the parent tube.

Evaluation of the 40% Depth Based Plugging Criteria with Regard to HEJ Sleeved Tubes

Considering the HEJ sleeved tube geometry, axial tube loads are developed primarily by the end cap load from the primary-to-secondary pressure difference. Tube loads can also be influenced by thermal expansion, and to some extent, the installation process. If it is assumed that the NDE technique was truly accurate, then uniform circumferential indications above the hardroll area

in the hydraulic expansion region of up to 80% throughwall would not be expected to fail during normal operating conditions based on the available non-degraded ligament cross-sectional area. These indications would be expected to be protected from burst by the presence of the sleeve. Uniform circumferential indications above the hardroll area of up to 50% throughwall would not be expected to fail at end cap loads equal to the RG 1.121 most limiting condition of 3 times the normal operating pressure differential, based on available non-degraded ligament area. Axial cracks in this region would be reinforced from burst by the presence of the sleeve, and would not be subject to failure by end cap loads.

For circumferential cracking below the hardroll area, the residual preload of the interference fit between the tube and sleeve due to the roll expansion provides inherent protection from axial displacement of the tube, even for a circumferentially separated tube, and forms the basis for the modified parent tube repair boundary. If the interference forces created by the roll expansion and HEJ geometry are greater than the maximum applied end cap loads, then tube burst is precluded for all plant conditions. For the HEJ geometry, reactor coolant system release rates approaching assumed rates used in the Final Safety Analysis Report can only be achieved if the tube is axially displaced by approximately 3 inches.

Impact of HEJ Sleeved Tube Geometry Upon Development of the Modified Tube Repair Boundary

The geometric interaction between the tube and sleeve can provide for increased margins of safety when the inherent structural and leakage characteristics of the joint are examined.

For the HEJ sleeved tube, structural integrity testing was conducted by applying a tensile load to the parent tube above the sleeve and to the sleeve/tube joint simulating the roll expansion within the tubesheet region. These tests showed that the interference forces developed by the combination of the roll expansion and the converging conical shape of the tube/sleeve at the top of the hardroll lower transition provide structural characteristics such that the structural integrity recommendations of RG 1.121 are met. For this configuration, a tube completely circumferentially separated at a distance of 1.1" below the bottom of the hardroll upper transition will provide for axial load bearing capability of nearly twice the most limiting RG 1.121 pressure end cap loading. The converging tube section below the hardroll flat length must be outwardly deformed by the sleeve as the tube is pulled over the sleeve, similar to a drawn-over-mandrel tube manufacturing process, thereby requiring significant force application to deform the converging tube end.

For tubes with throughwall EDM notches extending over 240° arc around the circumference of the tube at an elevation of ~ 1" below the bottom of the hardroll upper transition, axial load bearing capability is influenced both by the interference forces in the roll expanded region and by the plastic overload capability of the non-degraded tube ligament. For this condition the failure load of the ligament is enhanced by the roll expansion, and the loads required for ligament separation are again approximately twice the most limiting RG 1.121 pressure end cap loading.

The structural characteristics of PTIs of any extent located greater than 1.1" below the bottom of the hardroll upper transition provide for a resistive load capability such that tube displacement is precluded during all plant conditions. Therefore, structural integrity will be provided for all plant conditions, and leakage outside of the bounds of the following leak test results would not be expected.

Leakage Aspects

Primary to secondary leakage testing was also performed for completely circumferentially separated tubes and EDM slitted tubes. For the cases where the tube is circumferentially separated at 1.1" below the bottom of the hardroll lower transition or where the tube is slitted throughwall for an arc length of 240°, leakage is bounded by approximately 0.016 gpm (0.061 liters per minute).

In license amendment requests to the U.S. NRC for the above described criteria, leakage was conservatively adjusted to 0.033 gpm per indication, regardless of the size or elevation location of the indication.

DESTRUCTIVE EXAMINATION RESULTS

In 1995, three HEJ sleeved tubes were removed from the Kewaunee Nuclear Plant. The tubes were cut from the secondary side, just above the top of the tubesheet and several inches above the top of the sleeve. The tube/sleeve sections were then removed from the secondary side of the steam generator. Two of the three Kewaunee specimens were destructively examined while one was archived.

Kewaunee Removed HEJ Sleeved Tube NDE Results

The field NDE of the Kewaunee tubes was performed using the Plus Point probe and indicated 360° degradation with expected throughwall distances. Laboratory NDE was performed using the Plus Point, Cecco-5 transmit-receive probe, and 3-coil RPC. The laboratory results for the Plus Point and Cecco-5 were consistent, while the 3-coil indicated circumferential arc lengths of ~270°. No cracking of the sleeves, and no cracking in the upper hydraulic or hardroll upper transitions of the tubes was found.

Tube Degradation Observed by Destructive Examination

The crack depths for both destructively examined specimens were strikingly consistent. In both tubes the maximum depth of crack penetration was 92 % throughwall, while the average crack depths were 60 and 62 % throughwall. The crack morphology indicated multiple crack initiation sites at several closely spaced elevations with numerous ductile ligaments separating the crack initiation sites.

Structural and Leakage Testing Performed on Kewaunee Removed HEJ Sleeved Tubes

One of the removed HEJ sleeved tubes was leak tested at normal operating and SLB conditions. No leakage was observed. Since it was believed, based partly on the NDE results, that throughwall degradation existed in the parent tube in the hardroll lower transition, the leak test configuration was set up to look for leakage from this region, not between the tube and sleeve.

Both destructively examined HEJ Sleeved tubes were tensile loaded to open the crack faces. Ligament failure loads of 10,350 lb and 10,700 lb were observed. In both cases the peak frictional loads recorded after ligament separation exceeded the RG 1.121 most limiting loading condition.

DEVELOPMENT OF MODIFIED TUBE REPAIR BOUNDARY

Based on the laboratory testing work described in addition to the results of the destructive examination results of the Kewaunee removed HEJ sleeved tubes, a HEJ sleeved tube crack indication operability assessment program was developed. This program is summarized by the following points:

1. Any circumferential crack indication existing 1.1" or greater below the bottom of the HEJ hardroll upper transition is acceptable for continued service. Circumferential crack indications existing less than 1.1" below the bottom of the hardroll upper transition are to be removed from service.
2. HEJ sleeved tubes determined to be acceptable for continued service are assigned a SLB leakage allowance of 0.033 gpm, regardless of the size of the indication.
3. HEJ sleeved tubes determined to be acceptable for continued service whose circumferential crack elevation is located greater than 1.3" below the bottom of the hardroll upper transition are assumed to leak at such small amounts during a postulated SLB that leakage from these indications can be neglected.
4. HEJ sleeved tube SLB leakage is to be compared against a site specific offsite dose limiting leakage allowance which results in offsite doses at the site boundary consistent with the licensing basis, or 10% of the 10 CFR Part 100 guidelines.

Eddy Current Elevation Uncertainty

The ability of the eddy current inspection process to accurately locate the indication with respect to the bottom of the hardroll upper transition must also be factored into the ultimate decision on tube operability. Westinghouse has performed mockup testing using the Plus Point, Cecco-5, and 3-coil RPC to determine the elevation location uncertainty for HEJ configurations. The results of these tests indicate that for all cases, the uncertainty is bounded by about 0.06 inch. Therefore, this uncertainty allowance must be added to the 1.1" repair boundary to compensate for any uncertainties applied through the location inspection process.

Accident Condition Allowable Leak Rate

The accidents that are affected by primary-to-secondary leakage are those that include, in the activity release and offsite dose calculation, modeling of leakage and secondary steam release to the environment. The postulated steam line break (SLB) accident represents the most limiting case due to the potential for increasing leakage due to the steadily increasing primary-to-secondary pressure differential during recovery from the accident and the direct release path to the environment provided by the break in the steam pipe.

Establishment of the repair boundary includes calculation of the maximum permissible steam generator primary-to-secondary leak rate during a steam line break outside of the containment building. Standard Review Plan (NUREG-0800) methodology is used to establish the maximum permissible leak rate. This methodology has been used to justify primary-to-secondary leak rates greater than the value of 1.0 gpm normally assumed in the plants' FSAR. This methodology has been utilized previously by the NRC for the licensing of the steam generator tube support plate voltage based plugging criteria, described in Generic Letter 95-05. NUREG-0800 limits the thyroid dose to 10% of the 10 CFR 100 limit of 300 Rem for the accident initiated iodine spike case. The repair boundary established in this report considers a conservative SLB per tube leakage allowance based on test data which accounts for potential leakage from PTIs left in service due to application of the criteria. The total S/G SLB leak rate from all sources (including calculated leakage from TSP intersections which are addressed by Generic Letter 95-05) is summed when comparing the estimated leak rate against the value established using the methodology of NUREG 0800.

An earlier license amendment request proposed by Wisconsin Electric Power Co. utilized a differing approach. This approach limited the circumferential indication arc length based on a non-degraded ligament plastic overload philosophy. In this program the apparent interference loads resultant from the hardroll interaction were neglected for conservatism. A growth rate allowance and conservative eddy current evaluation criteria were suggested. This program allowed indications a minimum of 1" from the bottom of the hardroll upper transition.

Other Contributing Factors to the Overall Safety Aspect of Steam Generator Tube Integrity Programs

When developing and evaluating proposed tube integrity assessments, other salient points must be considered. These include;

- a. inspection history and inspection trends,
- b. observed plant operating parameters including primary to secondary leak rate during operation, and
- c. other inherent tube restraints provided by the design of the steam generator.

Examining the inspection results for U. S. plants, it is observed that the overwhelming majority of HEJ sleeved tube indications have occurred in either the hardroll lower transition or in the lower hydraulic expansion transition. This biasing of detected crack indications suggests that

the installed condition of an HEJ sleeve affects the tube in such a manner that if indications are to develop, that the likelihood is that they will occur in the hardroll lower or lower hydraulic transition. This suggests that rapidly propagating cracks at the hardroll upper transition or upper hydraulic transition would not be expected, and that evaluation of the proposed criteria should not be dictated by "anticipated" cracks, but by previously observed crack indications. Observed primary to secondary leakage can also be used as a qualitative measure to assess the potential for rapidly propagating cracks in the upper transitions. There have been no reported incidents of primary to secondary leakage from the upper transitions in U. S. plants, in over 14 years of operation with HEJ sleeves. The condition of the secondary side of the steam generator must also be, as a minimum, qualitatively considered when evaluating the sensitivity of particular aspects of the program upon overall safety. The Kewaunee and Point Beach Unit 2 plants have reported large numbers of dented TSP intersections, and it is believed that those intersections that are not dented are firmly packed with TSP corrosion products. The axial restraint provided by dented or packed intersections would act to counter tube end cap loads.

Status of Proposed License Amendment Requests

In April 1994, Wisconsin Public Service proposed a license amendment request for a limited circumferential indication arc length approach. In September 1994, Wisconsin Electric proposed a similar program, however, in this case the results of the laboratory test program were incorporated. In August 1995, American Electric Power proposed the modification to the pressure boundary approach. To date, the U. S. NRC has not approved either approach.

Cost Benefit Impact of the Tube Repair Boundary Limits

Without successful implementation of the criteria, large numbers of tubes with PTIs in HEJ sleeved tubes have been unnecessarily removed from service. The distribution of indication elevations provided on Figure 5 indicates that roughly 90% of the indications are below the 1.1" limit value (including an eddy current uncertainty allowance). The cost-benefit impact of the criteria can be shown to represent a substantial financial burden. The cost of removing these tubes from service, loss of revenue due to reduced electrical output and other costs associated with outage extension due to expanded scope tube plugging operations should also be weighed. At an assumed cost of tube plug installation of \$2000 per tube, approximately \$1,700,000 for additional tube plugging has been incurred between the Kewaunee and Point Beach Unit 2 plants. When the loss of revenue due to reduced electrical output and outage extension are considered, the implementation of the criteria can be shown to represent a substantial financial savings.

Figure 1
Typical HEJ Sleeve Installation

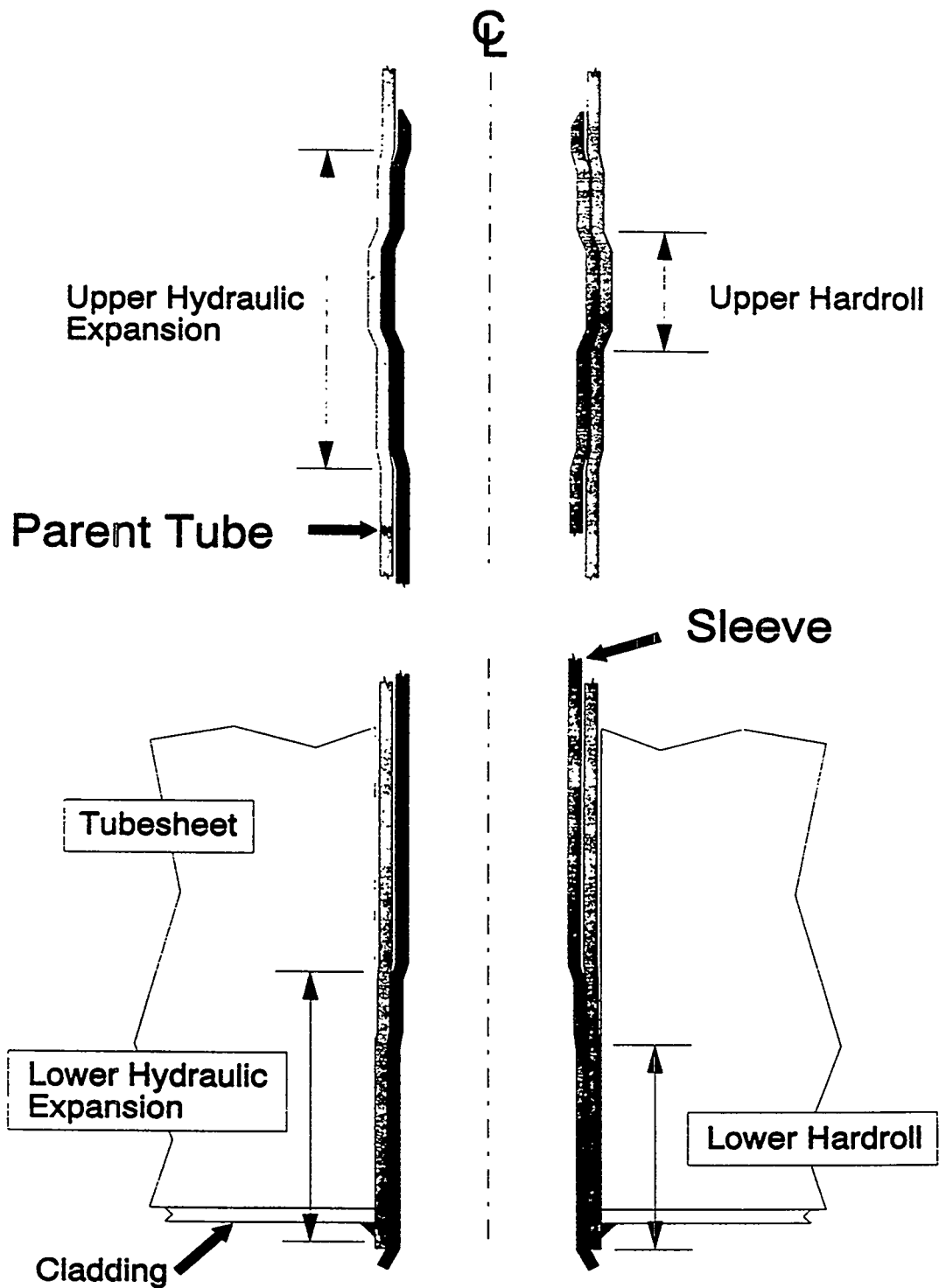


Figure 2
HEJ Sleeved Tube Terminology

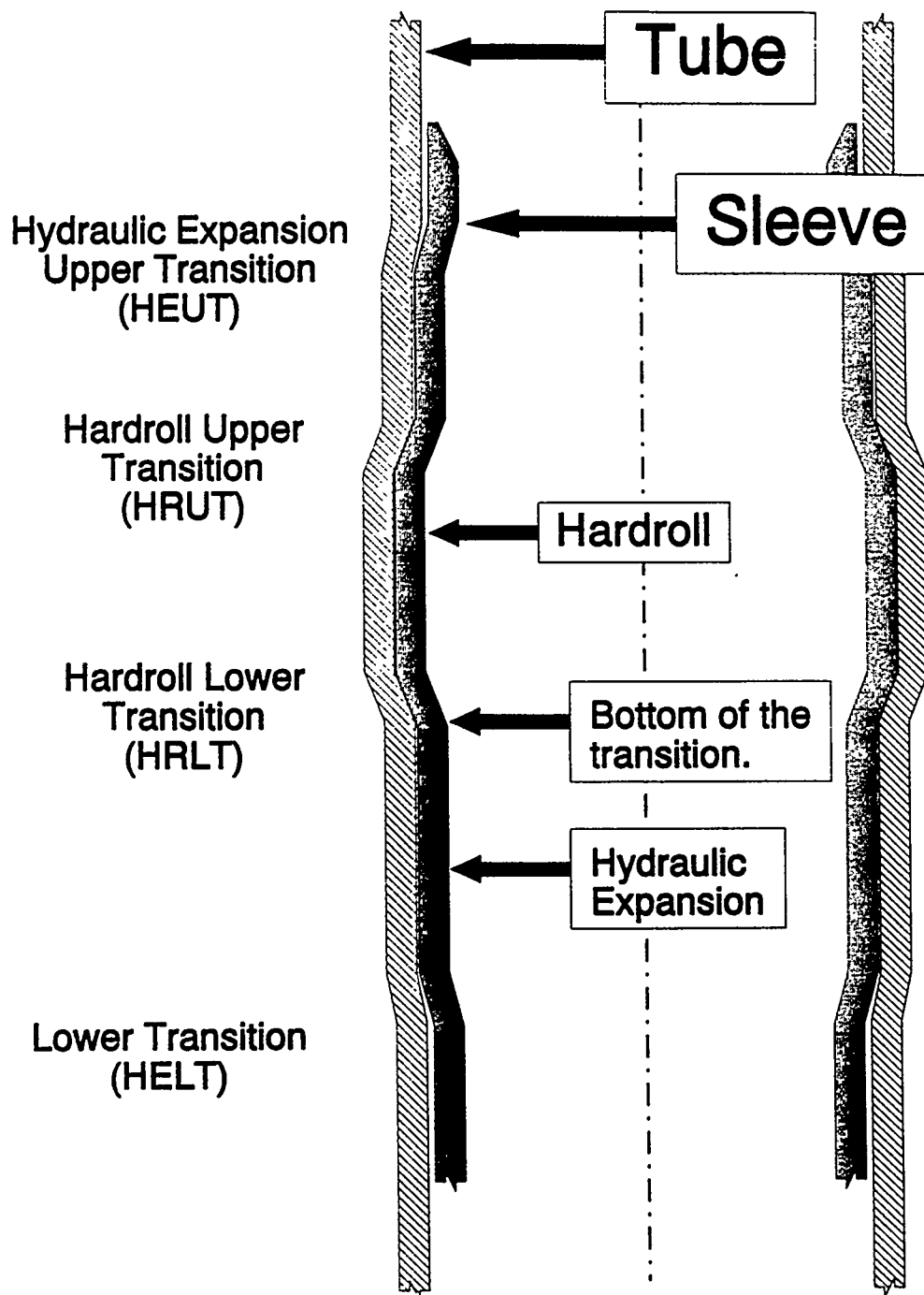


Figure 3
Illustration of Modified Tube Repair Boundary Limits

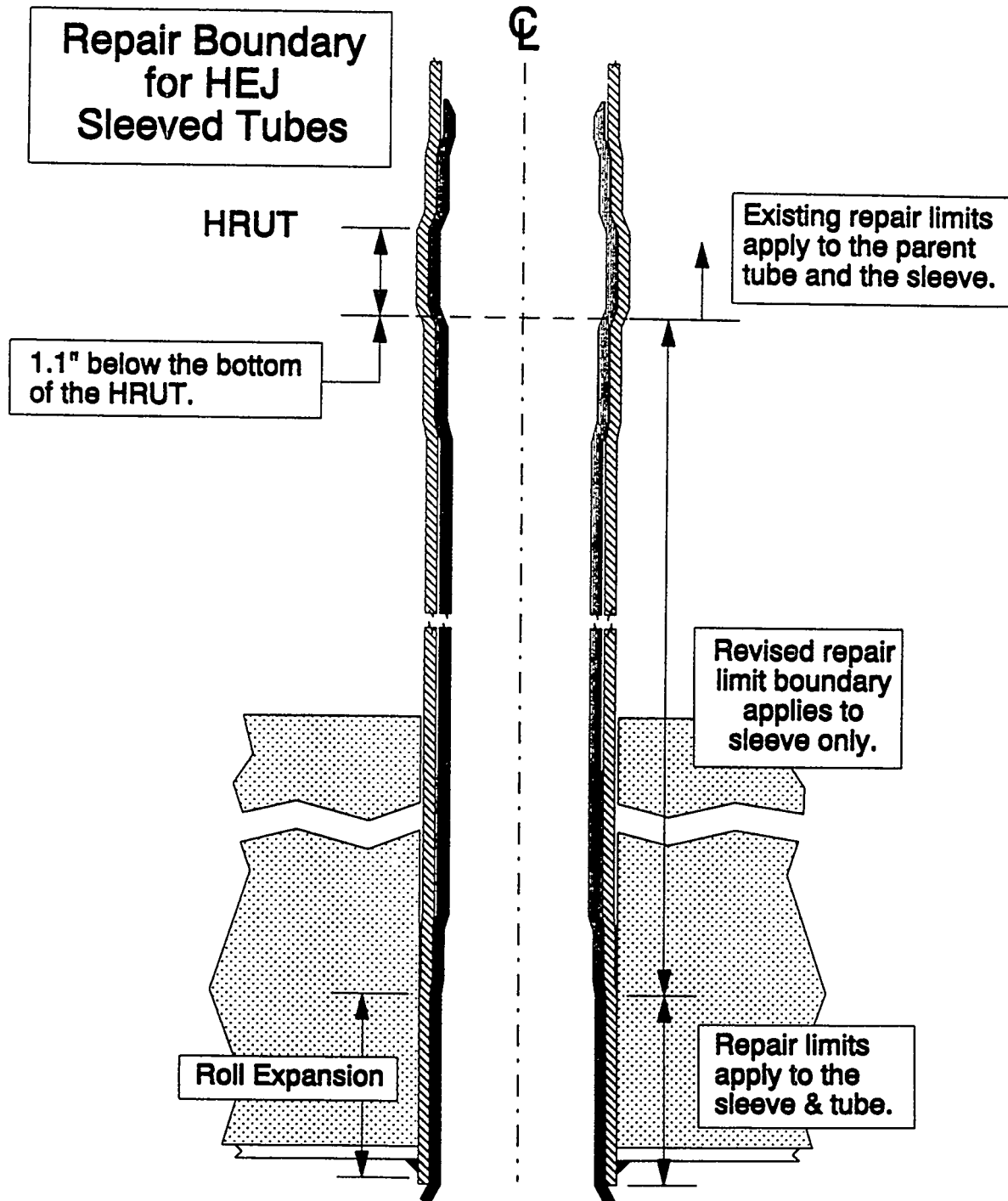


Figure 4
Summary of HEJ Sleeved Tube Parent Tube Indications

Distribution of All Circumferential PTIs by Transition

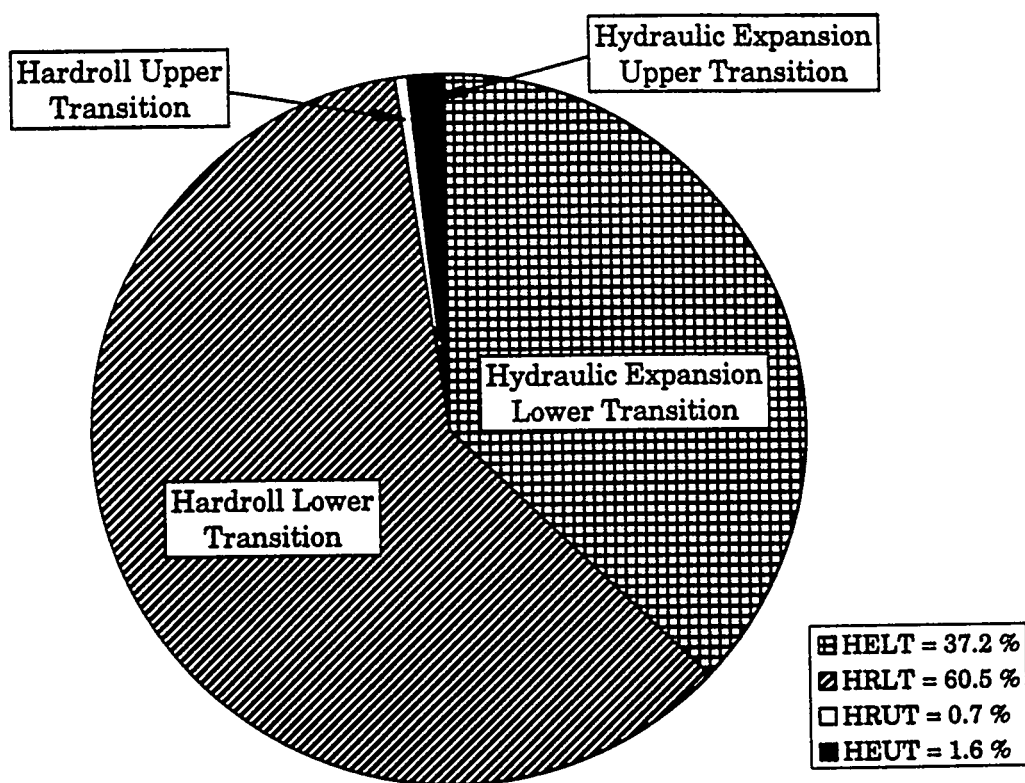


Figure 5
 Distribution of Indication Elevations with Respect to Bottom of HRUT

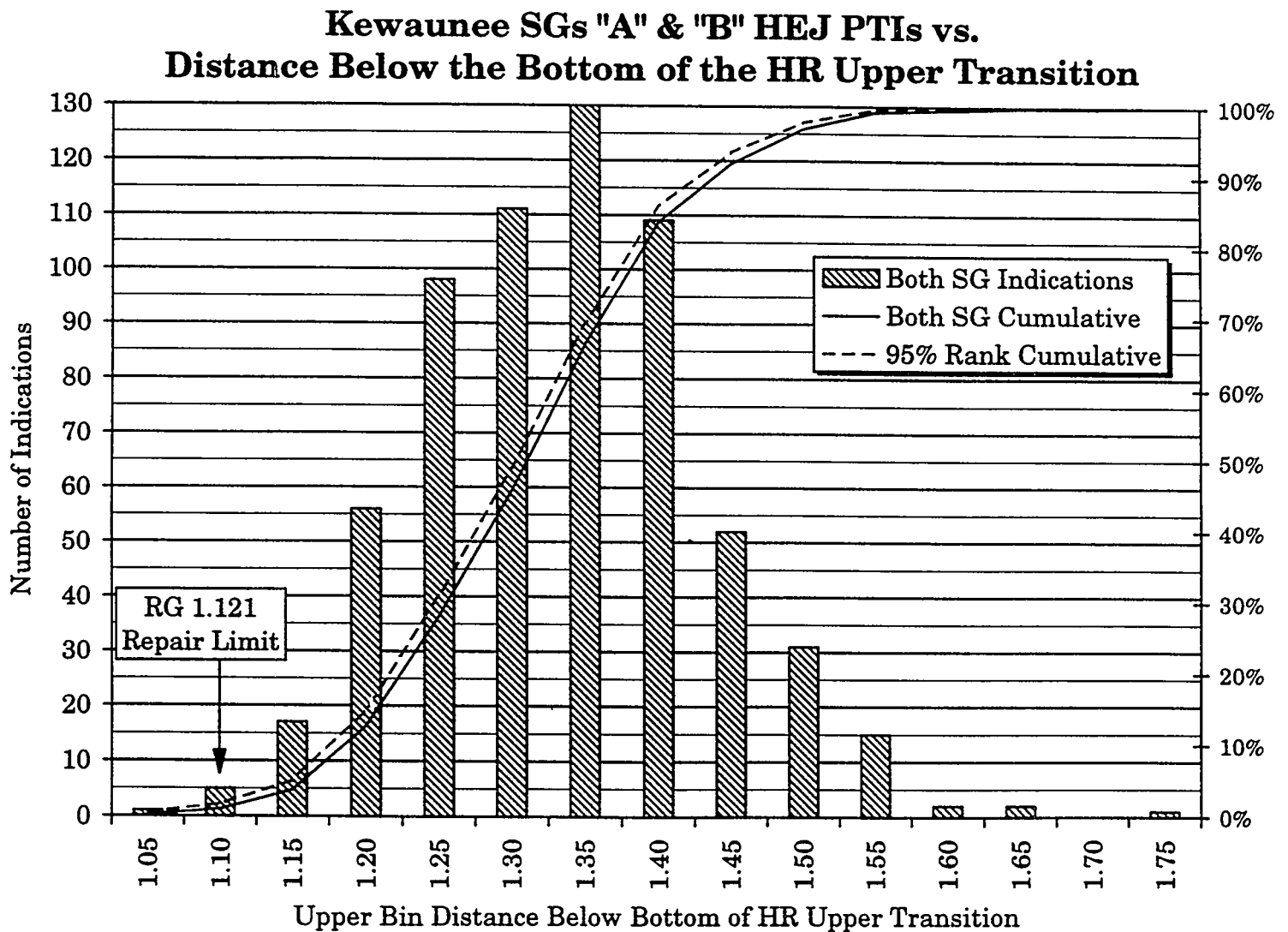


Table 1
Summary of HEJ Sleeve Installations

Westinghouse Sleeving Experience Chronology							
Plant	Date Installed	S/G Type	Sleeve Type ⁽¹⁾	Number of Sleeves	Sleeve Characteristics		
					Material	Length (in)	Type
San Onofre 2	10/80-6/81	W-27	TS	6929	600TT	27,30,36	HEJ
Point Beach 1	11/81	W-44	TS	13	600TT	36	HEJ ⁽²⁾
Indian Point 3	10/82-1/83	W-44	TS	2971	600TT	36,40,44	HEJ
Point Beach 2	4-6/83	W-44	TS	3000	600TT	36	HEJ
Millstone 2	7-9/83	CE-67	TS	2036	625/690TT	40	HEJ
Ringhals 2	05/84	W-51	TS	17	690TT	30	Braze
Millstone 2	03/85	CE-67	TS	2926	625/690TT	40	HEJ
Indian Point 3	07/85	W-44	TS	635	600TT	36,40,44	HEJ
Millstone 2	11/86	CE-67	TS	225	625/690TT	40	HEJ
Point Beach 2	10/87	W-44	TS	87	690TT	36	HEJ
Kewaunee	03/88	W-51	TS	1940	690TT	30,36	HEJ
Zion 1	03/88	W-51	TS	58	690TT	30	HEJ
Doel 3	06/88	W-51	TS	54	690TT	30	Laser ⁽³⁾
Point Beach 2	10/88	W-44	TS	509	690TT	30	HEJ
Kewaunee	03/89	W -51	TS	1698	690TT	30,36	HEJ
Point Beach 2	10/89	W-44	TS	298	690TT	36	HEJ
Kewaunee	03/91	W-51	TS	691	690TT	27,30,36	HEJ
Farley 2	3/92	W-51	TSP	69	690TT	30	Laser ⁽⁴⁾
Farley 2	3/92	W-51	TS	30	690TT	12	Laser ⁽⁴⁾
D. C. Cook 1	7/92	W-51	TS	1840	690TT	30,27	HEJ
Farley 1	9/92	W-51	TSP	148	690TT	30	Laser ⁽⁴⁾
Farley 1	9/92	W-51	TS	446	690TT	12	Laser ⁽⁴⁾
Doel 4	5/93	W-E1	TS	1752	690TT	30,36	HEJ
Doel 4	1994	W-E1	TS	> 11000	690TT	12	Laser ⁽⁴⁾
			Total	> 39000			
Notes: (1) TS = tubesheet sleeve & TSP = tube support plate sleeve. (2) Brazed sleeves also installed. (3) CO ₂ laser used for the welding process. (4) YAG laser used for the welding process.							

INTERNATIONAL WORKSHOP ON STEAM GENERATOR TUBE INTEGRITY

30 OCT.- 2 NOV, 1995
ARGONNE, ILLINOIS, USA

TITLE: SPANISH APPROACH TO RESEARCH AND DEVELOPMENT APPLIED
TO STEAM GENERATOR TUBES STRUCTURAL INTEGRITY AND
LIFE MANAGEMENT

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ABSTRACT

1 INTRODUCTION

The operating experience acquired from certain Spanish Nuclear Power Plant steam generators shows that the tubes, which constitute the second barrier to release of fission products, are susceptible to mechanical damage and corrosion as a result of a variety of mechanisms, among them wastage, pitting, intergranular attack (IGA), stress-corrosion cracking (SCC), fatigue-induced cracking, fretting, erosion/corrosion, support plate denting, etc.

These problems, which are common in many plants throughout the world, have required numerous investments by the plants (water treatment plants, replacement of secondary side materials such as condensers and heaters, etc.), have meant costs (operation, inspection and maintenance) and have led to the unavailability of the affected units.

In identifying and implementing all these preventive and corrective measures, the Spanish utilities have moved through three successive stages:

- In the initial stage, the main source of information and of proposals for solutions was the Plant Vendor, whose participation in this respect was based on his own Research and Development programs.
- Subsequently, the Spanish utilities participated jointly in the EPRI Steam Generator Owners Group, collaborating in financing.
- Finally, the Spanish utilities set up their own Steam Generator Research and Development program, while maintaining relations with EPRI programs and those of other countries through information interchange.

2 THE STEAM GENERATOR RESEARCH AND DEVELOPMENT PROJECT

Although the Spanish utilities had previously collaborated in specific research projects, in 1987 a multidisciplinary project was initiated with a view to ensuring integrated consideration of the most relevant aspects, the main objectives being to improve plant availability, reduce operating and maintenance costs and increase the

technological capacity of the participating companies.

Achievement of these objectives also implied extending the operating lifetime of these components while at the same time maintaining the necessary levels of safety.

The organizations participating in the project include both the nuclear power plants and utilities and CIEMAT, ENSA and Tecnatom. The Sarriá Institute of Chemistry (IQS) has also participated in development of certain project activities. The total budget for the first phase of the project (1987-1991) amounted to 14 million dollars.

The activities performed within the framework of this project, the first phase of which was completed in June 1991, were as follows:

- . Development of a database containing complete information on each steam generator and installed in all the Spanish nuclear power plants, and of a documentary database which has made it possible to centralize all the information generated by this project, other similar projects and overseas organizations.
- . Optimization and Validation of the eddy current in-service inspection techniques applied to the steam generator tubes.
- . Examination of tubes removed from steam generators in laboratories especially prepared for this purpose.
- . Study of the corrosion behaviour of tube materials in response to primary and secondary side chemical environments.
- . Specific Plugging Criterion applicable to each type of defect, which is currently being applied in the Plants as an alternative to the requirements of the operating technical specifications.
- . Evaluation of mitigation and repair techniques, such as chemical cleaning, plugging and sleeving.

3 THE NEED FOR A SECOND PHASE OF THE STEAM GENERATOR RESEARCH AND DEVELOPMENT PROJECT

As the Steam Generator Research Project developed, it became increasingly clear that a second phase would be advisable for the following reasons:

- To finalize activities that it had not been possible to complete during the first phase.
- The appearance of new problems not foreseen in the first phase.
- The possibility to initiate more innovatory developments on the basis of the capabilities acquired during the first phase.

- The need to replace the steam generators in certain Spanish plants.

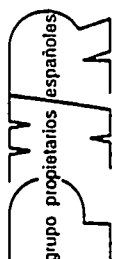
This second phase has the same objectives, participants and internal organization as the first, the foreseen period for its development being January 1991 to June 1996.

The second phase of the Steam Generator Research Project includes nine activities:

- . Steam generator tube plugging criteria for secondary side stress corrosion cracking in the support plate zone, circumferential cracks in the roll transition zone and fretting defects in the anti-vibration bar zone.
- . Generation by accelerating methods of defects similar to those found in steam generator tubes, in order to have available manufacturing procedures and test pieces for burst tests, leaks, etc. and for the validation and optimization of inspection techniques.
- . Ultrasonic technique steam generator tube inspection system, as a complement to the eddy current technique, for complex crack configurations and for the inspection of the sleeves installed on the tubes.
- . Optimization and validation of the eddy current, steam generator in-service inspection techniques applied to defects of complex configuration (axial-circumferential).
- . Research into new, alternative materials for steam generator tubes (Inconel-690 TT and Incoloy 800 N.G.).
- . Creation of an advanced steam generator tube degradation prediction model.
- . System based on laser techniques for inspection of steam generator tubes from the inner surface.
- . Development of an optimized procedure for chemical cleaning of the secondary side of Spanish steam generators.
- . Corrosion testing of the materials typically used for steam generator tubes (Inconel 600 MA and 600 TT), which will continue the work on tube material behaviour initiated during the first phase.

The budget for this second phase will amount to approximately 12 million dollars.

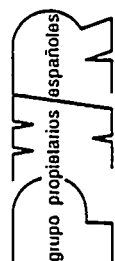
This paper presents the main conclusions reached in the different activities of Phase I and II and their applications to the Spanish Nuclear Power Plants.

	INTERNATIONAL WORKSHOP ON STEAM GENERATOR TUBE INTEGRITY – ARGONNE, ILLINOIS, USA	Fecha: 30 OCT – 2 NOV, 1995
SPANISH APPROACH TO RESEARCH AND DEVELOPMENT APPLIED TO STEAM GENERATOR TUBES STRUCTURAL INTEGRITY AND LIFE MANAGEMENT AUTHORS: JOAQUÍN LOZANO – C.N. ASCÓ GUSTAVO J. BOLLINI – TECNATOM, S.A.		

	INTERNATIONAL WORKSHOP ON STEAM GENERATOR TUBE INTEGRITY – ARGONNE, ILLINOIS, USA	Fecha: 30 OCT – 2 NOV, 1995
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SPANISH PWR NUCLEAR PLANTS

	PLANT	MW	NSSS	S.G. MODEL CONSTR.	S.G. TUBE MATERIAL	COMMERCIAL OPERATING
1 st . GENERATION	ZORITA	160	W	24	I-600 MA	1968
	ALMARAZ I	930	W	D-3	I-600 MA	1981
2 nd . GENERATION	ALMARAZ II	930	W	D-3	I-600 MA	1983
	ASCO I	930	W	D-3	I-600 MA	1984
	ASCO II	930	W	D-3	I-600 MA	1985
3 rd . GENERATION	TRILLO	1.030	KWU	54-GT	I-800	1988
	VANDELLÓS II	980	W	F	I-600 TT	1988



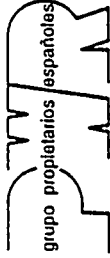
**INTERNATIONAL WORKSHOP ON STEAM GENERATOR
TUBE INTEGRITY – ARGONNE, ILLINOIS, USA**

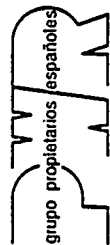
Fecha:

30 OCT – 2 NOV, 1995

**IMPLEMENTING STAGES OF PREVENTIVE AND CORRECTIVE
MEASURES**

- ◆ IN THE INITIAL STAGE, SOLUTIONS WERE BASICALLY PROPOSED BY THE PLANT
VENDOR.
- ◆ SPANISH UTILITIES PARTICIPATED IN EPRI S.G. PROGRAMS (SGOG I AND II),
COLLABORATING IN FINANCING.
- ◆ PHASE-I SET UP OF THE SPANISH S.G. RESEARCH PROGRAM (1987–1991), WHILE
PARTICIPATING IN THE EPRI S.G. RELIABILITY PROJECT (SGRP).
- ◆ PHASE II SET UP (1992–1997), WHILE PARTICIPATING IN THE EPRI S.G. STRATEGIC
MANAGEMENT PROGRAM (SGMP–1).

	INTERNATIONAL WORKSHOP ON STEAM GENERATOR TUBE INTEGRITY – ARGONNE, ILLINOIS, USA	Fecha 30 OCT – 2 NOV, 1995
<u>MAIN TASKS OF PHASE I AND II</u> ☐ <u>S.G. DATA BASE AND DEGRADATION EVOLUTION MODEL</u> ▪ PWSCC AT ROLL TRANSITION ZONE ▪ IGA/ODSCC AT SUPPORT PLATES ☐ <u>N.D.T. TECHNIQUES</u> ▪ EDDY CURRENTS ▪ ULTRASONICS ▪ LASER (PWSCC) ☐ <u>FABRICATION OF REALISTIC FLAWED TUBE SAMPLES</u> ☐ <u>PULLET-OUT TUBE EXAMINATION</u> ☐ <u>CORROSION TESTS</u> ▪ PRIMARY ENVIRONMENTS ▪ SECONDARY ENVIRONMENTS ▪ ALTERNATE S.G. TUBE MATERIALS ☐ <u>PLUGGING CRITERIA</u> ▪ PWSCC AT ROLL TRANSITION ZONE ▪ IGA/ODSCC AT SUPPORT PLATES ☐ <u>EVALUATION OF MITIGATION AND REPAIR TECHNIQUES</u> ▪ SECONDARY SIDE CHEMICAL CLEANING ▪ REMOVABLE PLUGS ▪ SLEEVES		



**INTERNATIONAL WORKSHOP ON STEAM GENERATOR
TUBE INTEGRITY – ARGONNE, ILLINOIS, USA**

Fecha:

30 OCT – 2 NOV, 1995

S.G. DATA BASE AND TUBE DEGRADATION EVOLUTION MODEL

ANALYSIS PERFORMED AS FUNCTION OF EFPY:

- TUBES WITH PWSCC AT R.T.
- TUBES WITH ODSCC INDICATION AT T.S.P.
- PWSCC AT R.T. PLUGGED TUBES
- ODSCC AT T.S.P. PLUGGED TUBES
- MODEL PER STEAM GENERATOR FOR EACH UNIT
- MODEL FOR ALL THE STEAM GENERATOR FOR EACH UNIT
- ALMARAZ, UNITS 1 AND 2
- ASCÓ, UNITS 1 AND 2

	INTERNATIONAL WORKSHOP ON STEAM GENERATOR TUBE INTEGRITY – ARGONNE, ILLINOIS, USA	Fecha: 30 OCT – 2 NOV, 1995			
<u>N.D.T. TECHNIQUES</u> ☐ <u>E.C. TECHNIQUE</u> OPTIMIZATION AND VALIDATION OF: <table border="0"> <tr> <td> ■ BOBBIN COIL ■ RPC AND 3XRPC </td> <td align="center"> TYPE OF DEFECTS } } </td> <td> – IGA – ODSCC (FREE SPAN AND SUPPORT PLATES) – PWSCC – DENTING – MIXED MODE CRACKING </td> </tr> </table> ☐ <u>U.T. TECHNIQUE</u> ■ MIXED MODE CRACKING (AXIAL AND CIRCUMFERENTIAL) ■ SLEEVE WELD INSPECTION ☐ <u>LASER TECHNIQUE: (UNDER DEVELOPMENT)</u> ■ PWSCC MIXED MODE CRACKING ■ VERY HIGH RESOLUTION			■ BOBBIN COIL ■ RPC AND 3XRPC	TYPE OF DEFECTS } }	– IGA – ODSCC (FREE SPAN AND SUPPORT PLATES) – PWSCC – DENTING – MIXED MODE CRACKING
■ BOBBIN COIL ■ RPC AND 3XRPC	TYPE OF DEFECTS } }	– IGA – ODSCC (FREE SPAN AND SUPPORT PLATES) – PWSCC – DENTING – MIXED MODE CRACKING			



INTERNATIONAL WORKSHOP ON STEAM GENERATOR
TUBE INTEGRITY – ARGONNE, ILLINOIS, USA

Fecha:

30 OCT – 2 NOV, 1995

PRIMARY WATER STRESS CORROSION CRACKING TEST

☐ **OBJECTIVE**

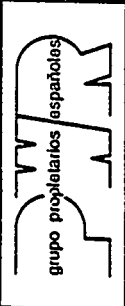
TO ESTABLISH THE INFLUENCE OF LI CONCENTRATION AND pH IN THE
GENERATION AND PROPAGATION OF PWSCC

☐ **TEST CONDITIONS**

- OPERATING TEMPERATURE: 330°C
- CONCENTRATIONS (p.p.m.):

B (ppm) Li (ppm)	1400	740	385	0	H ₂ (cc kg/H ₂ O)
	*	*	*		23
3,5		*			45
2,0	*		*	*	23
	*		*		45
0,7	*		*	*	23
					45

*: COMPLETED

	INTERNATIONAL WORKSHOP ON STEAM GENERATOR TUBE INTEGRITY – ARGONNE, ILLINOIS, USA	Fecha: 30 OCT – 2 NOV, 1995
<u>INTERGRANULAR ATTACK IN SECONDARY SIDE</u> ● <u>OBJECTIVE</u> TO STUDY THE EFFECT OF CONTAMINANTS PRESENT IN THE S.G. SECONDARY SIDE IN THE GENERATION OF IGA. ● <u>TEST CONDITIONS</u> – PRIMARY SIDE: TEMPERATURE : 325°C PRESSURE : 140 Kg/cm² MAXIMUM FLOW : 3.2 m³/h – SECONDARY SIDE: TEMPERATURE : 285°C PRESSURE : 75 Kg/cm² MAXIMUM FLOW : 10 l/h		

☐ D-3 MODEL MOCK-UPS (2)

- STUDY OF IGA/ODSCC MITIGATION BY BORIC ACID ADDITION

☐ 24 MODEL MOCK-UP

- WATER WITH FOSFATES; RATIOS OF Na/PO₄ BETWEEN 2 AND 3

☐ F MODEL MOCK-UP

- AVT + ALKALINE ENVIRONMENT (Ca)

☐ KWU MODEL MOCK-UP

- AVT + ALKALINE ENVIRONMENT (SULPHUR)

☐ DENTED MOCK-UP

- GENERATION OF ODSCC BY DENTING (AVT + Cu + Cl + O₂)

	INTERNATIONAL WORKSHOP ON STEAM GENERATOR TUBE INTEGRITY -- ARGONNE, ILLINOIS, USA	Fecha: 30 OCT - 2 NOV, 1995
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ALTERNATE S.G. TUBE MATERIAL STUDY

☐ **OBJECTIVE**

TO STUDY COMPARATIVELY THE SUSCEPTIBILITY OF INCONEL 690 TT AND INCOLOY 800 Mod.
IN CAUSTIC AND ACIDIC ENVIRONMENTS WITH SULPHUR, COPPER AND LEAD.

☐ **TEST CONDITIONS**

TEST SAMPLES : C-ring; 2% STRAIN
TEMPERATURE : 350°C
ENVIRONMENTS : CAUSTIC AND ACIDIC

☐ **TUBE MATERIALS**

- INCONEL 690 TT
- INCOLOY 800 AND 800 SP
- INCONEL 600 MA (AS REFERENCE)

11/11

	INTERNATIONAL WORKSHOP ON STEAM GENERATOR TUBE INTEGRITY – ARGONNE, ILLINOIS, USA	Fecha: 30 OCT – 2 NOV, 1995
<u>PLUGGING CRITERIA:</u> ● <u>PWSCC IN R.T.Z. (L.B.B.)</u> <u>OBJECTIVE:</u> ANY RISK IN THE MOST SEVERE CONDITIONS OF S.G. TUBE DUE TO PWSCC IN THE R.T.Z., IS ALWAYS PRECEDED BY AN ADMISSIBLE LEAK IN NORMAL OPERATION. <u>HYPOTHESIS:</u> ALL THE PRIMARY TO SECONDARY LEAK IN THE S.G. IS COMING FROM ONE CRACK. <u>SPECIFIC EXPERIENCE:</u> TYPE OF DEFECTS AND EVOLUTION. <u>SPECIFIC STUDIES:</u> THEORETICAL ANALYSIS AND EXPERIMENTAL TESTS. <u>SURVEILLANCE RULES:</u> DURING PLANT OPERATION (LEAK RATE) AND REFUELLING OUTAGE (CRACK LENGTH).		

	INTERNATIONAL WORKSHOP ON STEAM GENERATOR TUBE INTEGRITY – ARGONNE, ILLINOIS, USA	Fecha: 30 OCT – 2 NOV, 1995
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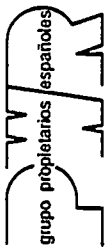
ODSCC AT TUBE SUPPORT PLATES

☐ **OBJECTIVE:**

TO LICENSE A PLUGGING CRITERIA FOR SINGLE OR MULTIPLE AXIAL ODSCC AT TUBE
SUPPORT PLATES FOR S.G. MODEL D-3 (W).

☐ **MAIN TASKS:**

- ANALYSIS OF ISI RESULTS
- ANALYSIS OF TUBE INTEGRITY:
 - BURST TESTS AND VOLTAGE AMPLITUDE CORRELATION
- LEAK TESTS AND VOLTAGE AMPLITUDE CORRELATION
- SAFETY ANALYSIS
- PLANT AVAILABILITY
- PLUGGING CRITERIA

	INTERNATIONAL WORKSHOP ON STEAM GENERATOR TUBE INTEGRITY – ARGONNE, ILLINOIS, USA	Fecha: 30 OCT – 2 NOV, 1995
<u>EVALUATION OF THE INFLUENCE OF THE PRESENCE OF CIRCUMFERENTIAL CRACKS AT ROLL TRANSITION ZONE</u> <u>OBJECTIVES:</u> a) TO REVIEW ALL THE AVAILABLE DATA REGARDING CIRCUMFERENTIAL CRACKING IN R.T.; BOTH IN SPAIN AND ELSEWHERE. b) TO STUDY THE INFLUENCE OF THE PRESENCE OF CIRCUMFERENTIAL CRACKS IN THE APPLICATION OF THE PLUGGING LIMIT OF S.G. TUBES WITH AXIAL PWSCC IN ROLL TRANSITION.		

	INTERNATIONAL WORKSHOP ON STEAM GENERATOR TUBE INTEGRITY – ARGONNE, ILLINOIS, USA	Fecha: 30 OCT – 2 NOV, 1995
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EVALUATION OF MITIGATION AND REPAIR TECHNIQUES

☐ **REMOVABLE PLUGS:**

▪ **OBJECTIVE:**

TO EVALUATE THE PERFORMANCE OF S.G. TUBE PLUGS REGARDING
REMOVABILITY, TIGHTNESS IN OPERATION AND DISPLACEMENT IN WORST
PRESSURE CONDITION.

▪ **VENDORS PARTICIPATION:**

FRAMATOME:	INCONEL 690	
WESTINGHOUSE:	INCONEL 600 (SHORT AND LONG MODELS)	
COMBUSTION ENG.:	INCONEL 690	
A.B.B.:	INCONEL 690/INCOLOY 800	
SIEMENS-KWU:	X22 CrNi 17 (Din 1.4057)	

	INTERNATIONAL WORKSHOP ON STEAM GENERATOR TUBE INTEGRITY – ARGONNE, ILLINOIS, USA	Fecha: 30 OCT – 2 NOV, 1995
☐ <u>BELGIUM, SWEDISH AND SPANISH SLEEVING QUALIFICATION PROGRAM</u> ■ OBJECTIVES: THE EVALUATION WILL MAINLY ADDRESS THE LIFE EXPECTANCY OF THE UPPER SLEEVE JOINT AND FEATURES SUCH INSTALLATION SPEED AND EXPECTED DOSE RATES. THE PROGRAM WILL ALSO INCLUDE EXAMINATION OF SLEEVES. REMOVED FROM RINGHALS UNIT 2 AFTER SEVERAL YEARS OF SERVICE. ■ VENDORS PARTICIPATION: – BABCOCK AND WILCOX – COMBUSTION ENGINEERING – FRAMATOME – SIEMENS-KWU – A.B.B. – MITSUBISHI (LASER) – WESTINGHOUSE (LASER)		

	INTERNATIONAL WORKSHOP ON STEAM GENERATOR TUBE INTEGRITY – ARGONNE, ILLINOIS, USA	Fecha: 30 OCT – 2 NOV, 1995
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☐ **SCOPE:**

- **SWEDISH STATE POWER BOARD:**
 - MANUFACTURING OF SINGLE AND MULTIPLE TUBE MOCK-UPS
 - VENDOR'S COSTS FOR INSTALLATION OF SLEEVES IN THE MOCK-UPS
 - EXTRACTION OF SLEEVES FROM RINGHALS UNIT 2
- **SPANISH UTILITIES (UNESA-PISGV):**
 - PROVIDE TUBE MATERIAL FOR USE IN MOCK-UPS
 - PERFORM THE STRESSES AND CORROSION INVESTIGATIONS, EXCEPT FOR X-RAY DIFFRACTION MEASUREMENTS.
- **LABORELEC:**
 - RESULTS FROM PREVIOUS PERFORMED SLEEVING QUALIFICATION WORK.
 - PERFORM THE X-RAY DIFFRACTION MEASUREMENTS.
 - PERFORM METALLOGRAPHIC ON SLEEVES (≈ 10) EXTRACTED FROM RINGHALS UNIT 2.

	INTERNATIONAL WORKSHOP ON STEAM GENERATOR TUBE INTEGRITY – ARGONNE, ILLINOIS, USA	Fecha: 30 OCT – 2 NOV, 1995
<u>SECONDARY SIDE CHEMICAL CLEANING</u> ☐ <u>OBJECTIVE</u> TO QUALIFY THE KWU PROCEDURE FOR SECONDARY SIDE CHEMICAL CLEANING. ☐ <u>TEST PERFORMED</u> – EFFICIENCY TESTS FOR IRON AND COPPER REMOVAL – CORROSION TEST: pH INFLUENCE AND OXYGEN INFLUENCE – OPTIMIZATION TESTS FOR MINIMUM CORROSION LEVEL – CREVICE DEPOSITS REMOVAL – REMOVAL OF COPPER DEPOSITS ON TUBES		

	INTERNATIONAL WORKSHOP ON STEAM GENERATOR TUBE INTEGRITY -- ARGONNE, ILLINOIS, USA	Fecha: 30 OCT - 2 NOV, 1995
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STEAM GENERATOR REPLACEMENT

☐ **MODEL D-3 FROM C.N. ASCO, UNITS I AND II AND C.N. ALMARAZ, UNITS I AND II:**
TOTAL OF 12 STEAM GENERATORS.

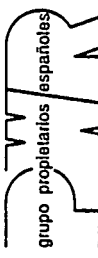
☐ **BIDS EVALUATION**

WESTINGHOUSE
 BABCOCK-WILCOX
 SIEMENS-FRAMATOME

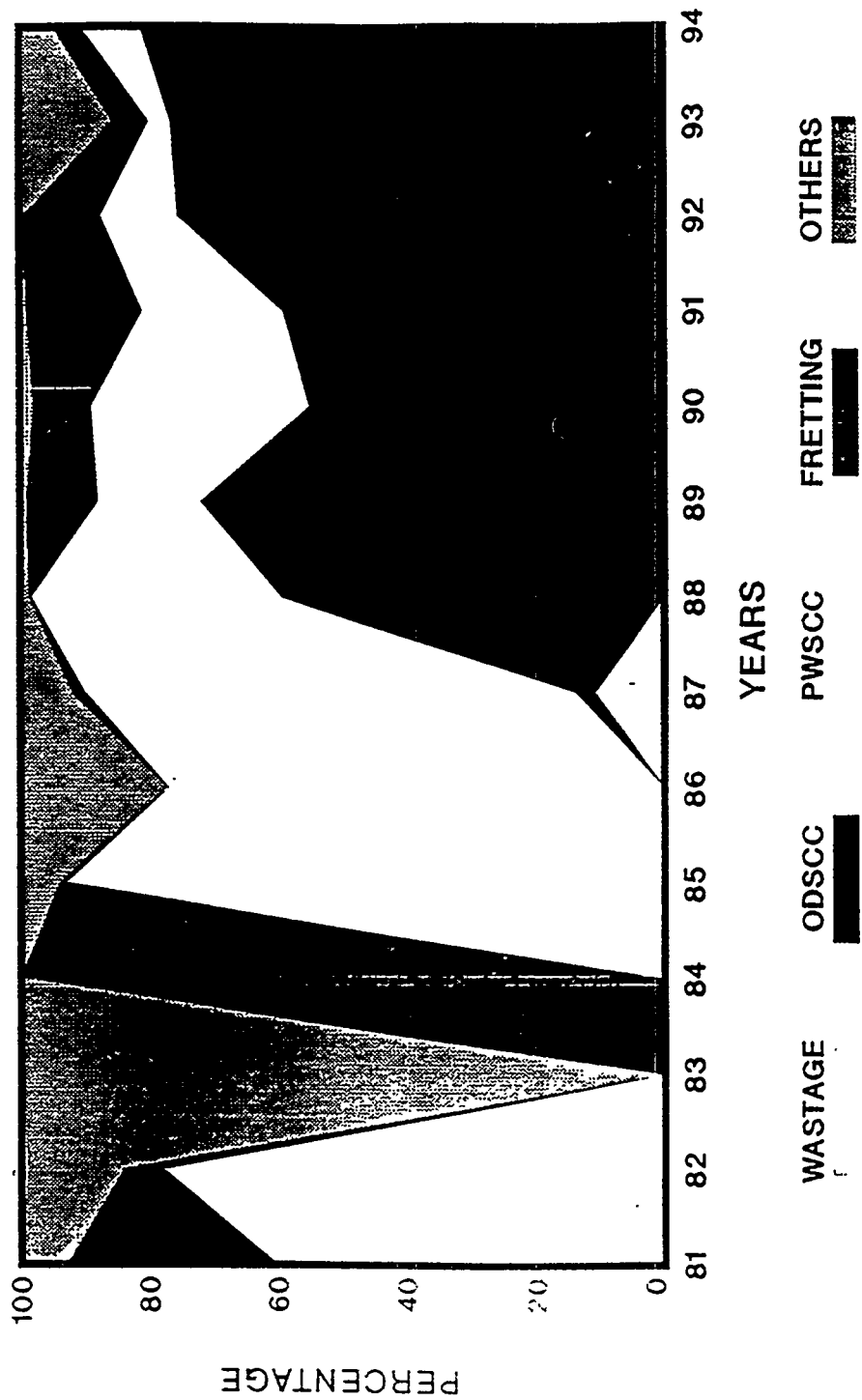
☐ **S.G. MODEL SELECTED: SIEMENS WITH TUBE MATERIAL: INCOLOY-800 N.G.**

☐ **SCHEDULE:**

ASCO I: 1995
 ALMARAZ I AND ASCO II: 1996
 ALMARAZ II: 1997

	INTERNATIONAL WORKSHOP ON STEAM GENERATOR TUBE INTEGRITY – ARGONNE, ILLINOIS, USA	Fecha: 30 OCT – 2 NOV, 1995
<u>STEAM GENERATOR REPLACEMENT</u> ☐ <u>MODEL D-3 FROM C.N. ASCO, UNITS I AND II AND C.N. ALMARAZ, UNITS I AND II:</u> TOTAL OF 12 STEAM GENERATORS. ☐ <u>BIDS EVALUATION</u> WESTINGHOUSE BABCOCK-WILCOX SIEMENS-FRAMATOME ☐ <u>S.G. MODEL SELECTED: SIEMENS WITH TUBE MATERIAL: INCOLOY-800 N.G.</u> ☐ <u>SCHEDULE:</u> ASCO I: 1995 ALMARAZ I AND ASCO II: 1996 ALMARAZ II: 1997		

SPANISH NUCLEAR POWER PLANTS (PWR's) **EVOLUTION OF PLUGGED TUBES BY CAUSES**



Steam Generators Secondary Side Chemical Cleaning at Point Lepreau Using the Siemens High Temperature Process

By

**Kamal Verma
Cyril MacNeil
Suat Odar
Klaus Kuhnke**

**NB Power Point Lepreau N.G.S.
NB Power Point Lepreau N.G.S.
Siemens/KWU
Siemens/KWU**

**Presented at CNRA/CSNI International Workshop on
Steam Generator Tube Integrity**

Hosted by USNRC/NRR, in Argonne, Illinois, 30 Oct.-2. Nov. 1995

Point Lepreau SG Chemical Cleaning

PtLChC1a



PLANT INFORMATION

- * Point Lepreau is a seawater cooled 680 MWe PHWR located on the Bay of Fundy.
- * The unit went into service in 1982 and has approximately 4214 EFPD.
- * The station employs 4 SGs designed and built by B&W Ltd with I-800 tubes, stainless steel broached hole TSPs and integral pre-heaters.
- * The secondary side has an all ferrous copper free feedtrain except Aluminum Bronze condenser tubesheet.

Point Lepreau SG Chemical Cleaning

PLCNC02



SIEMENS

NB POWER

PLANT CHEMISTRY

- * The SG chemistry control since start-up has employed the congruent phosphate technique.
- * The feedwater pH control using ammonia was switched to morpholine in 1988, to counteract erosion/corrosion in piping systems.
- * To protect SG's from sea water ingress due to condenser leaks, a full flow Condensate Polisher consisting of mixed bed ion exchange columns was added in late 1986.

Point Lepreau SG Chemical Cleaning

PLCNC03



STEAM GENERATOR SECONDARY SIDE SLUDGE COMPOSITION

Sludge samples have been obtained from sludge pile during tube removals and during tubesheet lancing. The chemical composition is:

- * Main elements were Fe, P and Na.
- * Minor elements (<2wt%) were Ca, Mg, Mn, Cr, Zn, Cl, Cu, Ni, Ti, Si and Pb.
- * Compounds of Magnetite (90 wt%) and phosphates were predominant. Silicates were in trace amounts.

Point Lepreau SG Chemical Cleaning

PILCHG04



SIEMENS

NB POWER

REASONS FOR THE CHEMICAL CLEANING

After 12 years of operation the SG inspections have revealed:

- * Hard caked sludge accumulation at the TS hot leg side.
- * Partial blockage of the 1st & 2nd TSP broached holes.
- * Onset of blockage of broached holes of some top TSPs.
- * OD tube corrosion (Pitting, Wastage) in the sludge pile region on the hot leg side TS and at the 1st & 2nd TSP intersections (appx.70 tubes).
- * An increase of SG outlet temp. was observed.

Point Lepreau SG Chemical Cleaning

PILCHG05



OBJECTIVES of the CHEMICAL CLEANING

The objective of chemical cleaning was to effectively:

- * Remove all the sludge from the tube sheet,
- * Clean all the TSP broached holes including crevices at the TSP lands,
- * Clean OD side heat transfer surface area of all tubes.

Point Lepreau SG Chemical Cleaning

PLCNC04



SIEMENS

NB POWER

SELECTION of the CHEMICAL CLEANING PROCESS

All the current commercially available chemical cleaning processes were evaluated in order to select the most suitable process for Point Lepreau. The selection criteria were:

- * Corrosiveness and effectiveness of different processes,
- * A process which has been applied successfully in conditions similar to those at Point Lepreau,
- * A non-complex process application which can be applied using plant heat at Point Lepreau,

Point Lepreau SG Chemical Cleaning

PLCNC07



SELECTION of the CHEMICAL CLEANING PROCESS (cont.)

- * Compatibility with the critical path of the annual plant outage,

The work was done in conjunction with:

- * An independent and comparative testing of different processes (by AECL),
- * A review and assessment of the data base of different vendors and their processes (by AECL),
- * A witnessing of chemical cleaning applications at different sites.
- * Discussions with utilities in N. America, Europe.

Point Lepreau SG Chemical Cleaning

PLOG08



SELECTION of the CHEMICAL CLEANING PROCESS (cont.)

Evaluation results:

- * Both the Siemens hot temperature and the EPRI-SGOG crevice cleaning processes fulfill the requirements in terms of effectiveness and minimizing corrosion,
- * Subsequent detail evaluation by NB Power with respect to the overall cost for the chemical cleaning, application time, equipment/material required and feedback from other utilities on their experiences with different vendors and cleaning processes resulted in the selection of Siemens hot temperature process.

Point Lepreau SG Chemical Cleaning

PLOG08



SIEMENS HIGH TEMPERATURE CHEMICAL CLEANING PROCESS

General Application Procedure:

- * The Iron Removal Process is applied at temp. 160-200°C lasting up to approximately 5 hours.
- * The high temperatures are achieved by using primary side heat during plant shut-down or start-up.
- * The chemicals in the SGs are mixed by steam-off, which replenishes the solvent in the crevice regions and as a consequence enhances the cleaning effectiveness.

Point Lepreau SG Chemical Cleaning

PLOC18



SIEMENS HIGH TEMPERATURE CHEMICAL CLEANING PROCESS

Process Chemistry:

- * The process uses EDTA solvent with concentrations equivalent to sludge amount.
- * The cleaning solvent is injected stepwise, which limits the free EDTA concentrations.
- * Due to limiting effects of low EDTA conc. and the short exposure time of free EDTA with SG internals no corrosion inhibitors are needed.
- * The corrosion control is enhanced in addition by alkaline and reducing conditions.

Point Lepreau SG Chemical Cleaning

PLOC18



PRE CHEMICAL CLEANING ACTIVITIES AT POINT LEPREAU

- * A comprehensive materials and weldment survey of the Point Lepreau SGs.
- * A detailed corrosion allowance determination.
- * Installation of corrosion coupons in a SG two years before the chemical cleaning application.
- * A dry-run using plant equipment during the 1993 annual outage.
- * Installation of chemical injection, steam and water sampling during the 1993 outage prior to the outage scheduled for chem. cleaning (1995).

Point Lepreau SG Chemical Cleaning

PLCNC12



SIEMENS

NB POWER

QUALIFICATION PROGRAM

Tests were conducted to cover the following plant specific aspects:

- * Application time, temperature, cleaning sequence, solvent injections, steam-off operations, SG heat-up and cool-down rates.
- * Sludge dissolution efficiency and EDTA conc.
- * Material compatibility of SG internals (Base metals, weldments and galvanic coupons).
- * Calibration of the H₂/N₂ corrosion monitoring system.

Point Lepreau SG Chemical Cleaning

PLCNC13



PROCESS AND CORROSION CONTROL

The process and corrosion control is performed by H_2/N_2 measurements:

- * The corrosion monitoring system measures the amount of corrosion hydrogen as a consequence of base metal corrosion.
- * This technique delivers on-line indication for corrosion, independent of the location within SG. It covers the corrosion aspects of entire SG.
- * After calibration the technique provides also an indirect measurement of the average corrosion of the exposed materials in μm .

Point Lepreau SG Chemical Cleaning

PLC/C14



SIEMENS

NB POWER

APPLICATION OF THE CHEMICAL CLEANING PROCESS

- * Two SGs were cleaned at a time. The SG pair to be cleaned was drained to lowest possible level. The water level of the other two SGs were maintained at preset levels for heat sink reasons.
- * Application Temperature: $160^{\circ}C \Rightarrow 175^{\circ}C$
- * Cleaning time per SG pair: 2,5 to 3,0 hours
at $160^{\circ}C$: 1 hr
 $\Rightarrow 175^{\circ}C$: 1 hr
 $\Rightarrow 160^{\circ}C$: 0,5 hr
- * The chemicals were injected in several steps, while monitoring corrosion in-between steps.

Point Lepreau SG Chemical Cleaning

PLC/C15



APPLICATION OF THE CHEMICAL CLEANING PROCESS

- * Final solvent concentration: 4,2-5,9 % EDTA
- * After the cleaning of the first SG pair, the temp. was lowered to 160°C and the entire chemical cleaning was repeated for the other SG pair.
- * The cleaned SG pair was kept full with the waste solvent for heat sink reasons.
- * After all 4 SGs were cleaned, the plant was cooled down, the SGs were drained and rinsed into temporary tanks for processing of the spent solvent and rinse solution.

Point Lepreau SG Chemical Cleaning

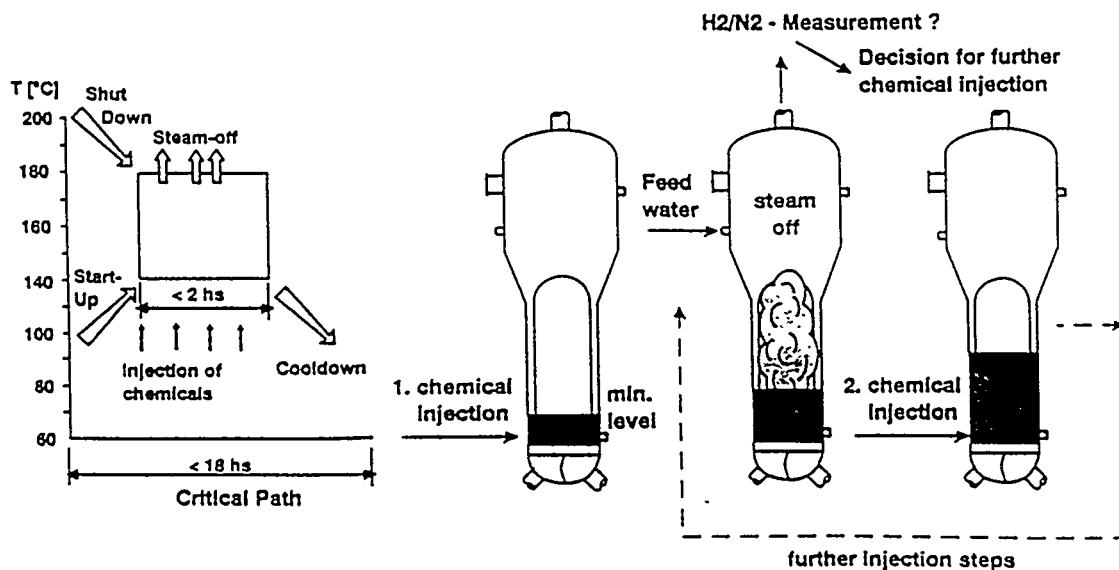
PLCNC18



SIEMENS

NB POWER

PROCESS APPLICATION AND CONTROL

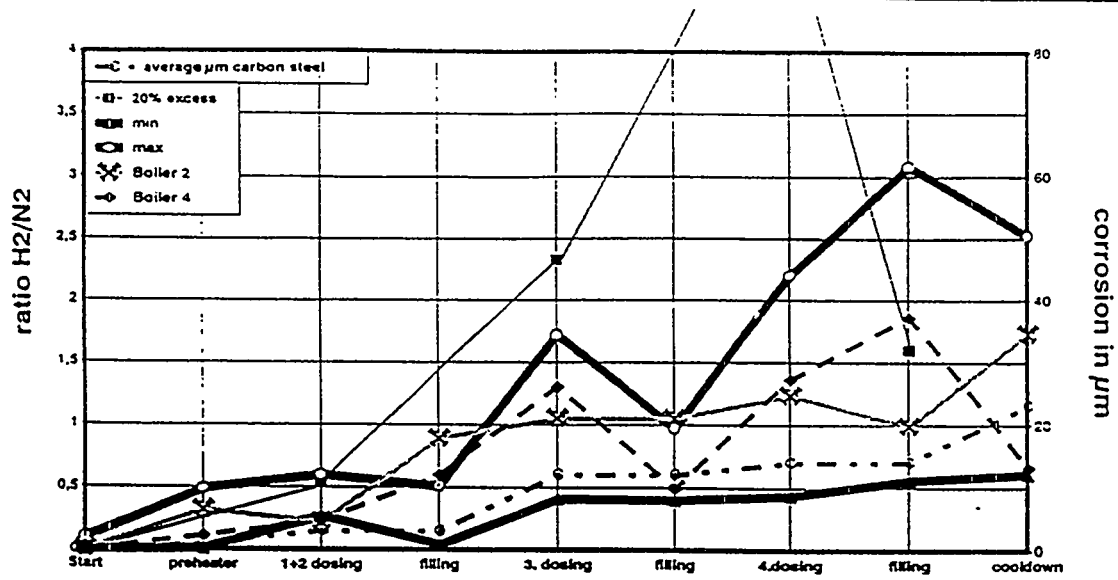


Point Lepreau SG Chemical Cleaning

PLCNC18



CORROSION MONITORING SYSTEM USING H₂/N₂ MEASUREMENTS

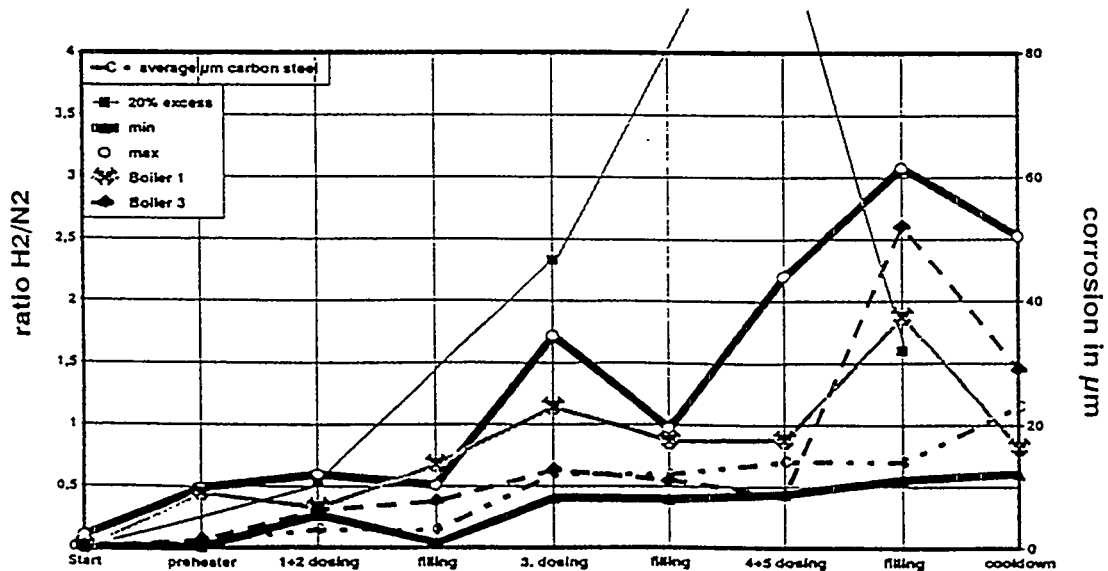


Point Lepreau SG Chemical Cleaning

Pcc02.gr1



CORROSION MONITORING SYSTEM USING H₂/N₂ MEASUREMENTS



Point Lepreau SG Chemical Cleaning

Pcc01.gr1



CHEMICAL CLEANING RESULTS

Amount of deposits removed [kg]:

<u>Deposits</u>	<u>SG1</u>	<u>SG2</u>	<u>SG3</u>	<u>SG4</u>
Magnetite:	259	313	266	274

The sludge removed included:

Salts:	32	33	36	23
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In total 124 kg of salt was removed from four SGs.
Out of this 70 kg were phosphates.

Point Lepreau SG Chemical Cleaning

PLCNC17



SIEMENS

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CHEMICAL CLEANING RESULTS (cont.)

Corrosion assessment:

- * The H₂/N₂ Corrosion monitoring system indicated an average carbon steel corrosion of 20 to 25 µm, which is in good agreement with the qualification results.
- * The insitu corrosion coupons after 2 years of operation and chemical cleaning experienced:
 - ° Negligible corrosion of stainless steel and SG tubing material,
 - ° Carbon steel corrosion of < 34 µm. after considering the contribution of 2 years of operation to be 30 µm,
 - ° No selective type of corrosion.

Point Lepreau SG Chemical Cleaning

PLCNC18



CHEMICAL CLEANING RESULTS (cont.)

Post Cleaning Inspections:

The visual inspections were performed using a fiberscope through the SG handholes:

- * All tube surfaces were cleaned to shiny metal.
- * Few TSP trefoil broaches showed some deposits, but less than before.
- * In some limited small areas of the TS some deposits remained but were significantly reduced. This was also confirmed by ECT (MRPC).

Point Lepreau SG Chemical Cleaning

PLCNG18



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WASTE HANDLING

- * After cleaning the spent solvent and the rinse water were drained to temporary storage tanks for processing at a later date.
- * All waste was characterized and classified to be a non-radioactive waste and was handled in the following manner:
 - ° Conventional disposal of the spent solvent via incineration was completed by a commercial contractor.
 - ° The rinse water was neutralized and disposed of via normal plant disposal pathways.

Point Lepreau SG Chemical Cleaning

PLCNG18



CONCLUSIONS

- * The Siemens high temperature chemical cleaning process could be applied without any problems and without any major impact on the outage critical path.
- * The majority of the objectives of the chemical cleaning were successfully met as evidenced by appx. 95% sludge removal and very low overall corrosion.
- * The process corrosion control using H_2/N_2 was effective and useful. A good agreement was obtained between this technique and the actual insitu corrosion coupon evaluation.

Point Lepreau SG Chemical Cleaning

PILCNC23



SIEMENS

NB POWER

WASTE HANDLING

- * After cleaning, the spent solvent and the rinse water were drained to temporary storage tanks for processing at a later date.
- * All waste was characterized and classified to be a non-radioactive waste and was handled in the following manner:
 - ° Conventional disposal of the spent solvent via incineration was completed by a commercial contractor.
 - ° The rinse water was neutralized and disposed of via normal plant disposal pathways.

Point Lepreau SG Chemical Cleaning

OECD/NEA

CNRA/CSNI Workshop on
Steam Generator Tube Integrity in Nuclear Power Plants

Actual Operation and Regulatory
Activities on Steam Generator
Replacement in Japan

Hitoshi Saeki
Kyushu Electric Power Co., Inc.
Fukuoka, Japan

October 30 - November 2, 1995
Chicago, Illinois USA

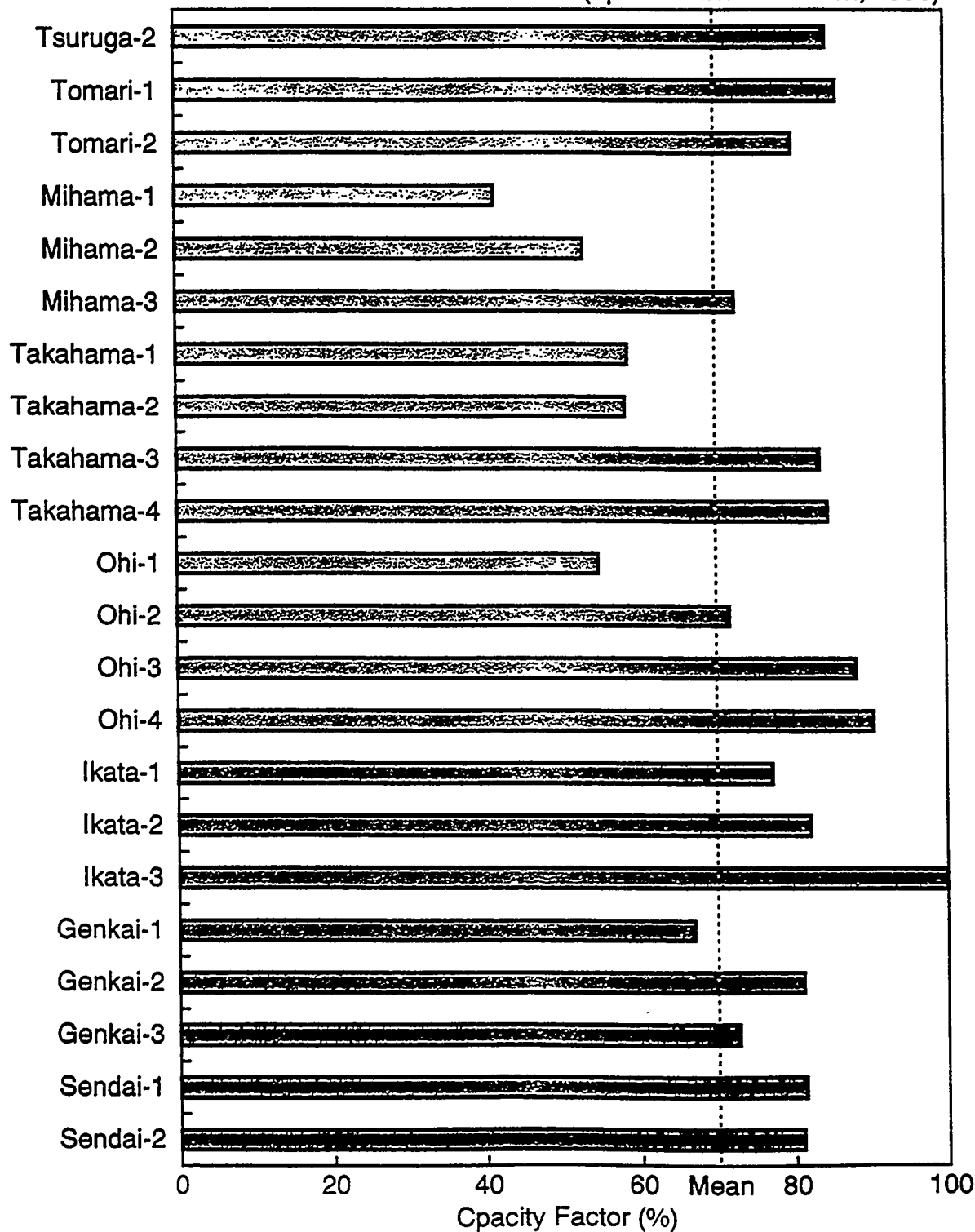
Status of Nuclear Power Plants in Japan

(As of the end of December 1994)

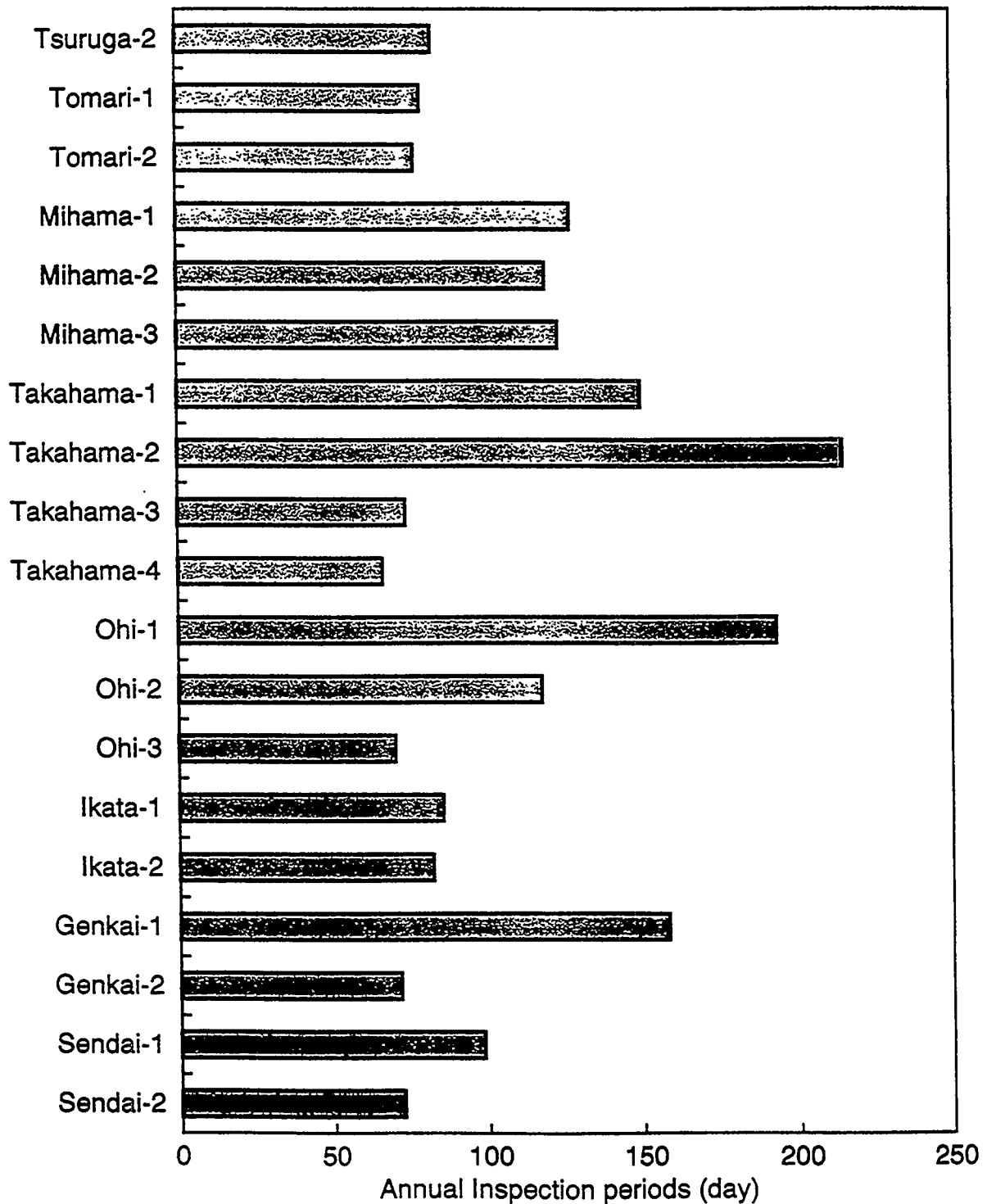
		PWR	BWR	GCR	Total
In operation	Units	22	25	1	48
	MW	18,186	22,014	166	40,366
Under construction	Units	1	3	—	4
	MW	1,180	3,537	—	4,717
Total	Units	23	28	1	52
	MW	19,366	25,551	166	45,083

Operating Status of PWR Nuclear Power Plants

(up to the end of March, 1995)



Annual Inspection Periods of PWR Nuclear Power Plants



Degradation Status in Japan

(No degradation in other plants, as of end of December 1994)

Power Plants	SG Type	Number of tubes and SGs	Number of plugged tubes	Number of sleeved tubes	Degradation Mode
Mihama-1 Note)	CE	4426X2	1891	0	Wastage, Pitting
Mihama-2 Note)	44	3260X2	411	41	Wastage, ODSCC, PWSCC, Fretting
Mihama-3	51	3388X3	797	0	PWSCC, IGA, Fretting
Takahama-1	51	3388X3	1095	77	Wastage, ODSCC, PWSCC, IGA
Takahama-2 Note)	51	3388X3	1610	4425	IGA
Takahama-3	51F	3382X3	26	0	Fretting
Takahama-4	51F	3382X3	21	0	Fretting
Ohi-1 Note)	51A	3388X4	2253	5511	PWSCC, IGA
Ohi-2	51A	3388X4	704	0	Fretting, PWSCC
Ikata-1	51	3388X2	405	13	PWSCC, Fretting
Ikata-2	51M	3382X2	4	0	Fretting
Genkai-1 Note)	51	3388X2	683	3164	IGA
Sendai-1	51M	3382X3	17	0	Fretting
Sendai-2	51F	3382X3	19	0	Fretting
Tsuruga-2	51FA	3382X4	2	0	Fretting

Note) Number of plugged and sleeved tubes show the final numbers before SG replacement.

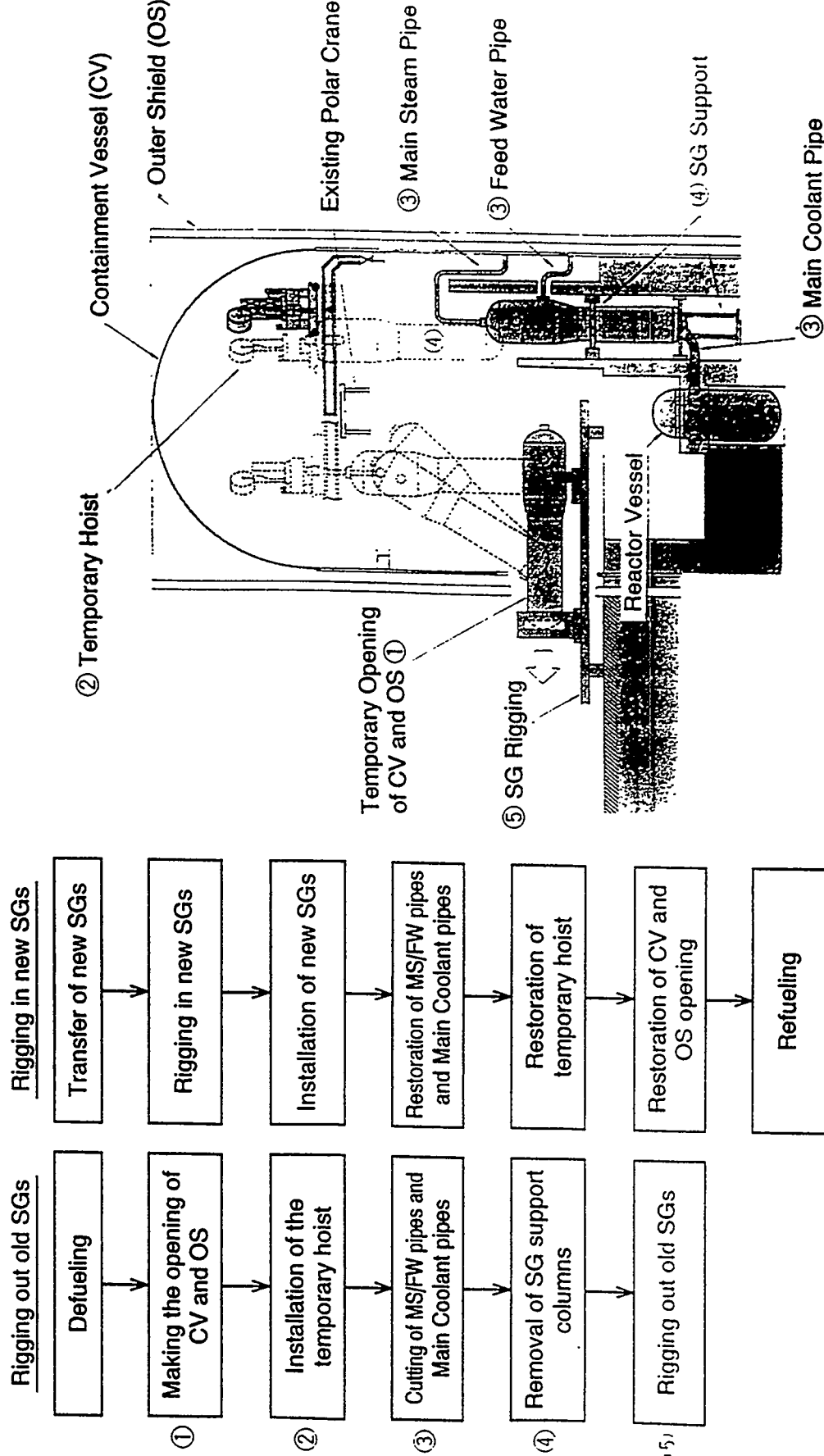
SG Replacement Schedule in Japan

	1993	1994	1995	1996	1997
Takahama-2 (826MWe)					
Mihama-2 (500MWe)					
Genkai-1 (559MWe)					
Ohi-1 (1175MWe)					
Mihama-1 (340MWe)					
Takahama-1 (826MWe)					
Mihama-3 (826MWe)					
Ohi-2 (1175MWe)					

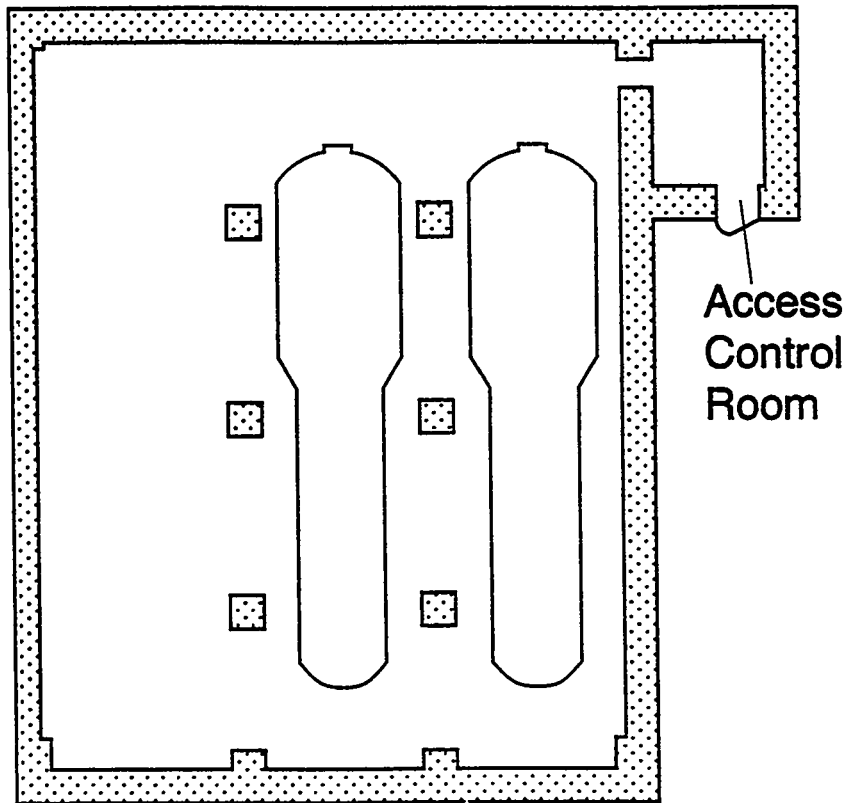
SG Replacement Characteristics in Japan

Plant	① The Condition of SG	② OS, CV, CW Opening	③ Removal of SG room	④ Rigging	⑤ MCP Cut	⑥ Decontam- ination	Remarks	
Takahama-2	One Piece	OS, CV	_____	(Inside CV) Wire-Jack U/D ender (Outside OS) Gantry Hoist Installation	3Cut	Mechanical Decontam- ination (Alumina)	_____	
Mihama-2		OS, CV, CW		Wire-Jack Existing P/C Co-lifting	A-Loop ; 3Cut *1 B-Loop ; 4Cut		*1 To obtain samples for welding	
Genkai-1		OS, CV		Semi-Frame +Wire-Jack +U/D ender	3Cut		_____	
Ohi-1		Equipment Hatch	Partial Removal of Concrete Ceiling and Wall	Chain Jack +U/D ender	3Cut		_____	
Mihama-1		OS, CV, CW	_____	Semi Frame +Wire Jack +U/D ender	*2 4Cut		*2 To remove the loop isolation valves	
Takahama-1		OS, CV	Partial Removal of Concrete Wall	Wire Jack +U/D ender	3Cut		_____	
Mihama-3		Equipment Hatch	_____	P/C Improvement +U/D ender			_____	
Ohi-2			Partial Removal of Concrete Ceiling and Wall	Chain Jack +U/D ender			_____	

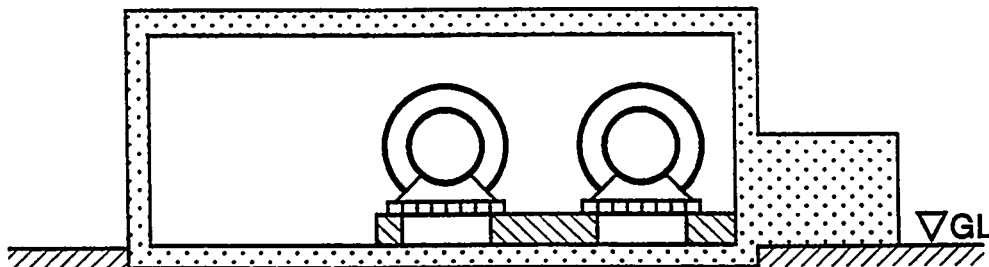
SG replacement work procedure



SG storage facility



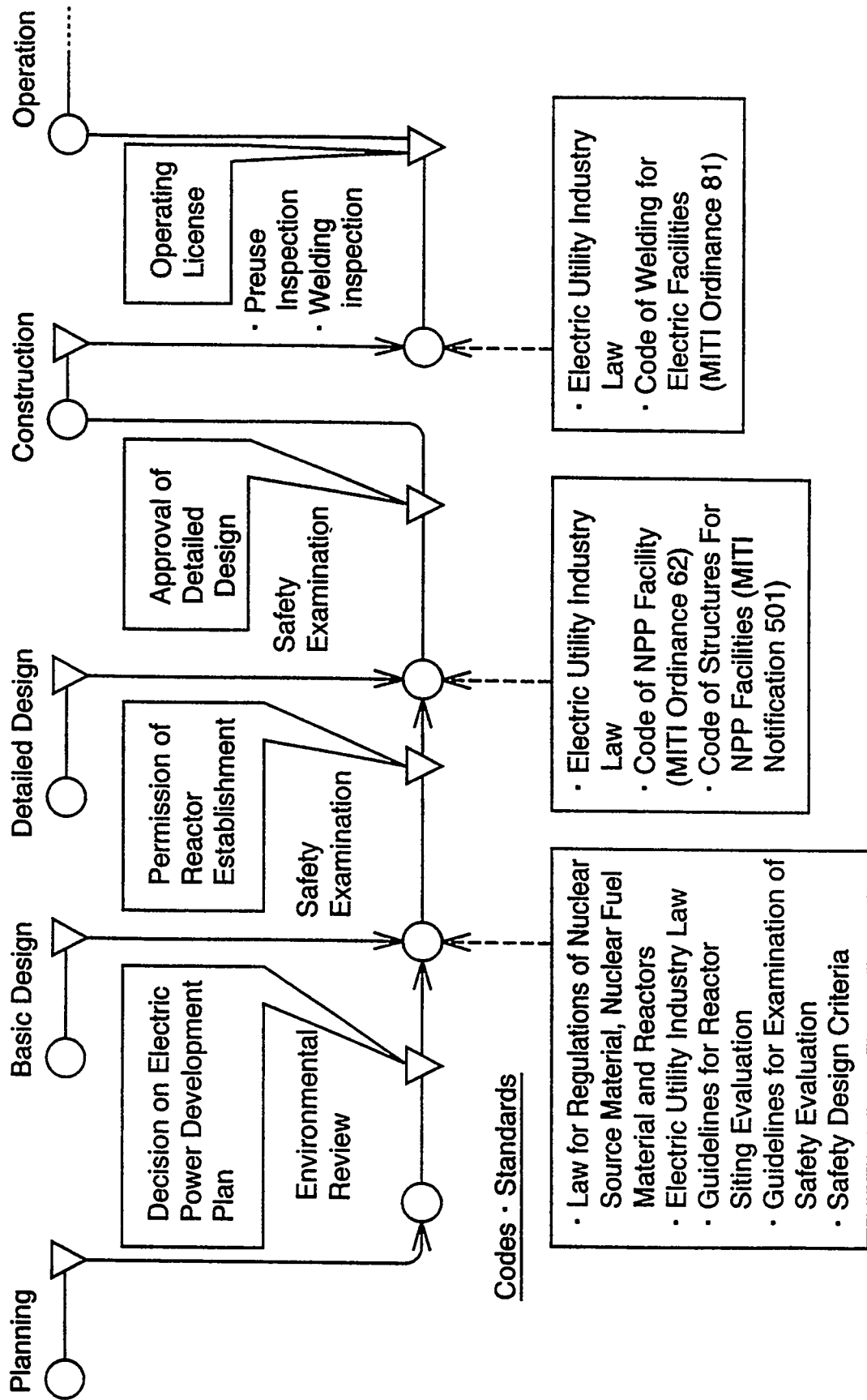
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(Section)

Licensing Procedures and Applicable Codes and Standards

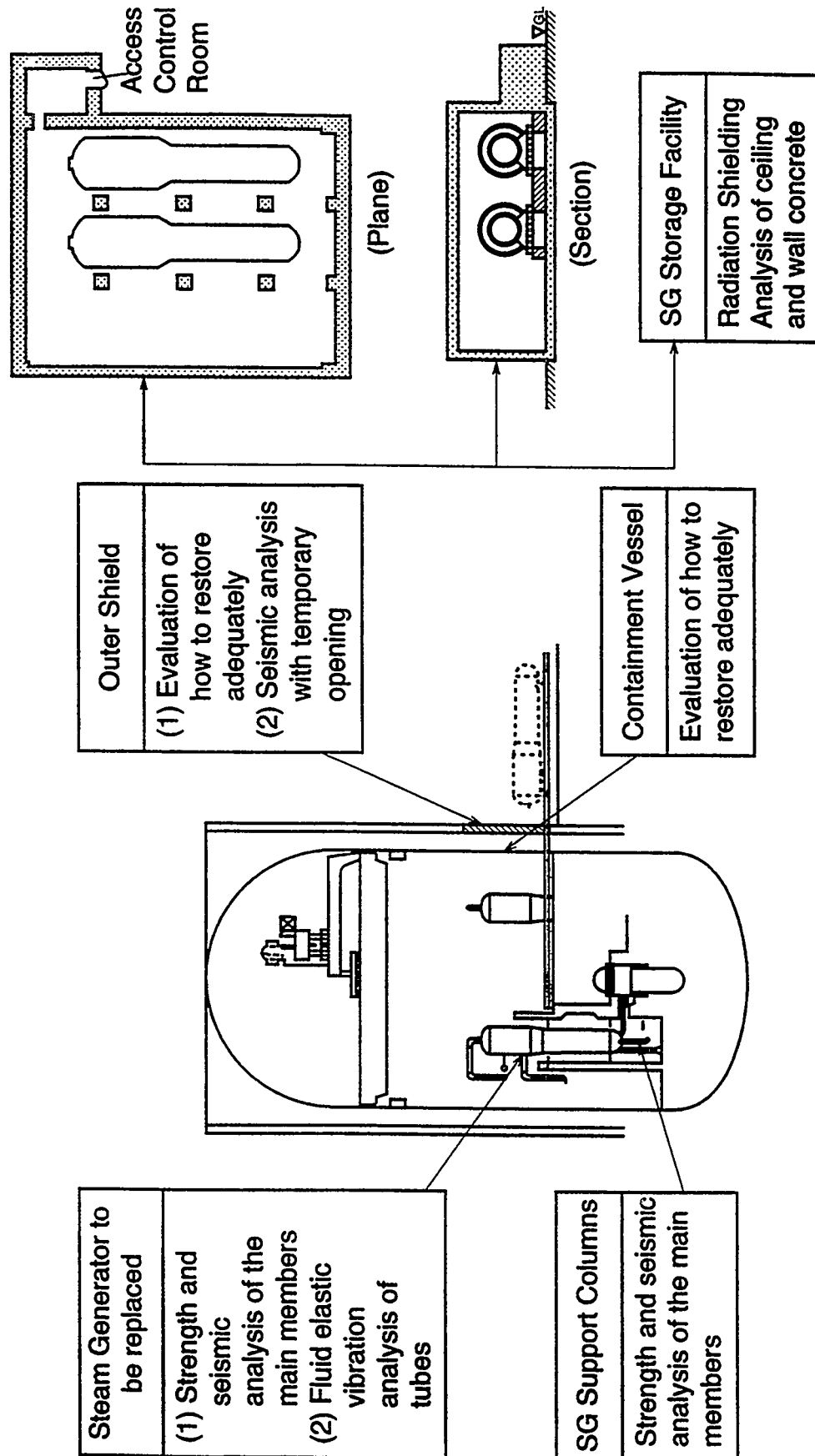
Nuclear Power Plant Licensing Procedures



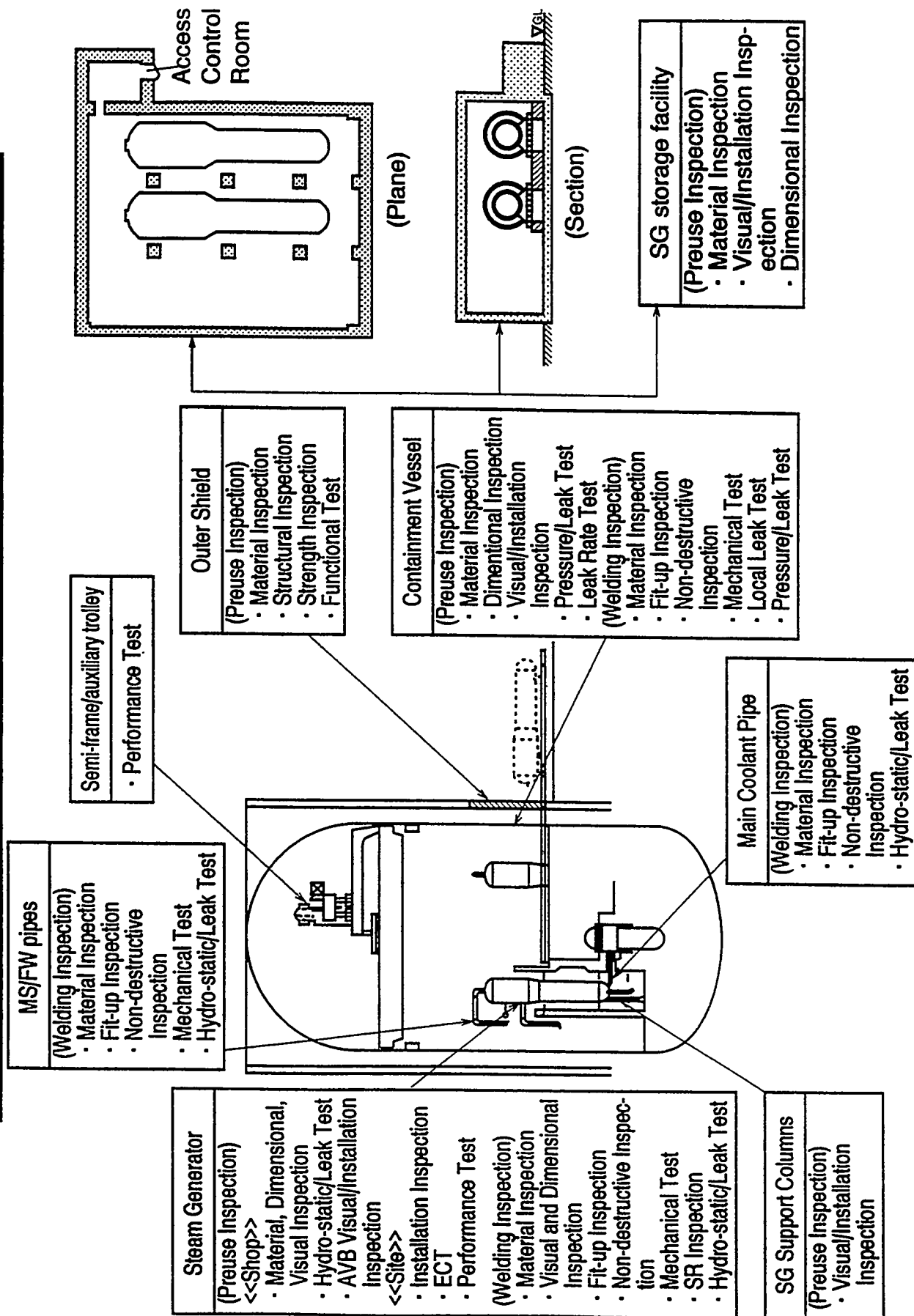
Safety Examination

Subjects	Safety Review Items
1. Steam Generator to be replaced	① Material and structural design
	② Countermeasures for past degradation case
	③ Design standard for pipe rupture and whipping
	④ Influence for safety evaluation
2. SG replacement technology	① SG replacement procedure
	② SG transportation on the site
	③ Cutting and welding MCP
	④ SG support replacement and improvement
	⑤ Cutting and restoration of Containment Vessel
	⑥ Cutting and restoration of Outer Shield Concrete
	⑦ Main Coolant Pipe decontamination
3. Waste Disposal	① Amount of waste and how to store
	② Waste disposal ability
	③ Situation and structure
	④ Radiation shield design
4. Countermeasures for radiation exposure	① Cumulative radiation exposure during SGR
	② Influence for radiation level around site boundary

Review of construction plan



Preuse Inspection etc.



10 Working Session 5: Operational Aspects and Risk Analysis

L. Cizelj, Jozef Stefan Institute, and J. Donoghue, USNRC

A general observation is that both operational aspects and risk analysis cannot be adequately discussed without information presented in other sessions. Some overlap of conclusions and recommendations is therefore to be expected.

Further, it was assumed that recommendations concerning improvements in some related topics were generated by other sessions and are not repeated here. These include:

- Knowledge on degradation mechanisms (initiation, progression, and failure).
- Modeling of degradation (initiation, progression, and failure).
- Capabilities of NDE methods.
- Preventive maintenance and repair.

We should note here, however, that all of these directly affect both operational and risk aspects of affected plants.

A list of conclusions and recommendations is based on available presentations and discussions addressing risk and operational experience. We aimed at reaching as broad a consensus as possible. We should note here that there is no strict delineation between operational and safety aspects of degradation of steam generator tubes. This is caused by different risk perceptions in different countries/societies.

The conclusions and recommendations were divided into four broad groups, as listed below. The basic reasoning behind the recommendations and conclusions is briefly outlined.

10.1 Human Reliability

PRAs/PSAs of SGTR events have shown that human reliability (HR) can be a significant contributor to the outcome of such events. In particular, fast and reliable diagnosis of the SGTR event is considered the most crucial action of the operators. Two recommendations follow:

- **Determine the need to improve human reliability information and methods used in PRA(PSA) of SGTR events.** Much of the HR information available as input to the PRA/PSA of SGTR events is based on engineering judgment rather than assessment of operator response from actual or simulated events. More effort is needed to determine which SGTR scenarios require accurate understanding of operator response.

- **Determine how to improve methods/information.** Training of operators on simulators is considered an appropriate method to study the response of operators.

10.2 Leakage Monitoring

All operating plants use leak rate monitoring. However, its role is not emphasized uniformly. Some plants depend on monitoring as a means to detect degrading tubes, while others use it only to help operators respond to tube failure. The following topics were considered the most important during the session:

- **On-line leakage monitoring can improve operator diagnosis and response to SGTR events.** Leak monitoring systems, especially N-16 detectors on main steam lines, can enhance the operator's ability to diagnose and promptly respond to the SGTR events. It is recommended to use and further improve such systems.
- **Address increased leak rates observed after startups following transients at some plants.** Some questions regarding observed leak rates under plant transient conditions could be more fully explained. In particular, spurious increases of measured leak rates (N-16) following plant transients have been reported. The initial peak is followed by an exponential decrease until a stable leak rate value is reached. This decrease becomes faster toward the end of the fuel cycle. Although theories have been presented to explain the effect, better understanding of such factors affecting leakage indication could help to improve its acceptability as a diagnostic and response tool.
- **Improve leakage-to-degradation correlations for various modes of tube degradation.** The scatter in the experimental leakage-to-degradation correlations (e.g., an alternate plugging criterion based on bobbin coil signal amplitude) is large. Further work in reducing this scatter could significantly improve the estimated risk due to excessive leakage. On the other hand, some plants are using primary-to-secondary leakage as an indication of tube conditions. However, the connection between leak rate and flaw progression is not accurately defined for essentially all known modes of degradation. To more fully rely on leakage monitoring, more experimental data are required to strengthen the leakage-to-degradation correlations.
- **Attempt to refine the technical basis for operational leakage limits.** Various limits for primary-to-secondary leakage have been adopted by plants. For example, most of the European plants use leakage limits on both absolute value and trend of leakage. Such limits are mainly based on operational experience and seem to offer benefits in detecting and mitigating tube failures. However, to understand the safety implications of a particular leakage limit, a more robust technical basis should be established. This may improve the determination of the safety benefit of the leakage limits.

10.3 Risk Impact

The effect on risk of operating plants with degraded SG tubes must be better understood in the context of reactor coolant pressure boundary integrity requirements. Many recently reported modes of tube degradation were not known when the original safety analyses were done and were therefore not accounted for.

- **Improve appreciation of the safety role of SG tube degradation in establishing plant risk.**

In terms of core damage frequency:

Degraded tubes could impact the expected core damage frequency if the degradation leads to a significantly higher potential for tube failure during normal operation and transients. Further, the degradation could impact the ability of tubes to fulfill their fission-product containment function during a core melt damage event. This may allow tube failure that would present a containment bypass path and could result in a large release of radionuclides. However, clarification is needed as to how much a postulated SGTR event contributes to the overall plant risk as compared with other potential contributing events.

Influence of degradation management:

Degradation management is essentially the only tool we have to control the safety and reliability of degraded steam generator tubes. The implementation of any degradation management could therefore have a significant effect on plant risk. This is largely an issue of understanding the level of degradation that may be permitted under a certain management plan, and then addressing the risk questions. Some probabilistic techniques to evaluate the contributions of the maintenance to the plant risk already exist. Further development in this field is recommended.

10.4 Consequence Assessment

- **Explore alternate dose assessment approaches to understand the margin in current methods.** Radiological dose consequences are often determined using conservative or bounding assumptions to ensure that an acceptable margin of safety is maintained. A comprehensive understanding of the safety implications of tube degradation should include alternate ("best estimate") dose assessment so that the margins of safety are apparent. Some ideas have included using statistically based methods to account for the expected distributions of important parameters rather than using conservative (i.e., worst case) point estimates.

SUMMARY

N-16 MONITORS:

ALMARAZ NPP EXPERIENCE

Almaraz Nuclear Power Plant has installed N-16 monitors -one per steam generator- to control the leakage rate through the steam generators. Tubes after the application of leak before break (LBB) criteria for the top tube sheet (TTS).

After several years of operation with the N-16 monitors, Almaraz NPP experience may be summarized as follows: N-16 monitors are very useful to follow the S.G. leak rate trend and to detect an incipient tube rupture; but they do not provide an exact absolute leak rate value, mainly when there are small leaks.

The evolution of the measured N-16 leak rates varies along the fuel cycle, with the same trend for the 3 steam generators. This behaviour is associated with the primary water chemistry evolution along the cycle.

N-16 MONITORS:

ALMARAZ NPP EXPERIENCE

In order to carry out the continuous surveillance of steam generators leaks and to control their evolution allowing to advance the tubes failures, Almaraz NPP has diverse methods, as well as direct surveillance, as grab samples and analysis.

a) Steam Generators Continuous Blowdown

Continuous surveillance: Is performed by means of a NaI monitor, placed for the measurements of the total γ activity in the grab samples collector of this system. For their situation, cannot be identify the steam generator affected by direct method, been necessary their sequential isolation, to identify the leaking steam generator.

Grab samples and analysis: Independently and with a daily frequency, it is carried out a high sensibility isotopic analysis in each one of the steam generators blowdown, making the leak calculation with the ^{133}I values of reactor coolant and blowdown.

The document "PWR Primary-to-Secondary Leak Guidelines Final Report," EPRI TR-104788, published in May 1995, proposes in order to effect this measurement, several isotopes presents in the reactor coolant, taking notice of the use of the iodine isotopes. for their possible reaction with the sludge accumulated in the steam generators. It is the Almaraz case, then ^{133}I is used, since to being the most abundant, it is the unique isotope that are detected more frequently in blowdown samples.

b) Condenser Off Gas

Direct surveillance: Is carried out by means of a NaI monitor provided in the vacuum pumps discharge of this system. As the previous case, it cannot be determine which of the steam

generators produces the increment.

Grab sample and analysis: On the other hand and with a daily frequency, a high sensitivity isotopic analysis is performed, that allows to quantitative evaluation of the three steam generators leak combined.

Both determinations suffer a lack of precision, as a consequence of the difficulty in the flow measurement of the air condenser removal.

c) Main Steam N-16 Monitors

Direct surveillance: Almaraz has N-16 monitors in the main steam pipes of each one steam generators. The measurement is continuous, giving a quantitative evaluation of the individualized leak of each steam generators.

The installation was made up after July 1988, as a consequence of a reactor coolant leak through the steam generator # 1 of the Unit 1. That leak was detected quickly by the increment activity in the continuous blowdown monitor, quantifying the leak by means of grab samples, as off gas condenser, as blowdown and, by means of these last results, identifying the steam generator for their isolation.

STEAM GENERATORS LEAK RATE MONITORING

Generalities:

Almaraz fuel cycles have a duration of 18 months, due to this, the operation of both units is started with elevated concentrations of boric acid (1800 ppm like B). During the minimal required time it is worked with a pH of 6.9 (calculated at operating temperature), decreasing coordinately B and Li concentrations. Once 2.2 ppm of Li are reached, this concentration stays constant, allowing the pH increment until 7.4, and then it is restarted the coordination between B and Li, in order to maintain constant this value until the end of the cycle. See figure 1.

Due to the application of LBB (leak before break) criterion in the top tube sheet transition zone, the operation with little defects is allowed. The Unit 1 begins the operation of each cycle with small leaks, and

it allows the surveillance of the leak rate evolution during these cycles. The experience of the unit 2 introduces an essential difference with the other, because the historical leaks in the beginning of the cycle are not detectable, starting their detection very advanced in the operation.

For both units and along the cycles during which the N-16 monitors have been in operation, their operative experience has been excellent, although these monitors have a limitation, because of being very sensitive to the load variations and to the unit trips, in spite of being calibrated for power corrections.

The trip incidents influences is very accused with two different effects. After each trip and when the normal operation is reached, it is produced a big increment of the leak rate. The previous tendency does not recover until 20 or 30 days after the nominal power rate is achieved on stable basis. These effects are observed in both determinations, as N-16 monitors as the continuous blowdown grab samples analysis.

Leak indications evolution discussion:

At the beginning of the cycle the leak values are elevated, decreasing in a gradual form but rapidly. These values remain constant, until the B concentration in the reactor coolant reaches the 900 - 800 ppm, equivalent to a pH of 7.0 - 7.1. From this moment, the leak rates begins a light and continued increment, until the reactor coolant Li concentration reduction is restarted, in order to maintain 7.4 pH constant, by means of the coordination with the B reduction. See figure 2.

With the exception of the steam generator # 1 Unit 1 evolution in the 8th cycle, the rest of the steam generators surveillance curves maintains constant the differences between them, except for small and discreet but punctual increments, that establishes a new difference, that after it maintains permanent. See figure 3.

The steam generators continuous blowdown grab samples analysis evolution maintains a total parallelism with the N-16 monitors measurements, but with a value between 2 and 6 times lower. This possible difference could be due to iodine isotopes reaction with the sludges accumulated in the top tubes sheet, near to the transition zone .

According to figure 2, at the beginning of 9th cycle, the discrepancy between both determinations is lower, may be due to the smaller accumulation of sludge, since at the end of previous cycle, a sludge lancing was performed.

CONCLUSIONS

- 1.- The N-16 monitors give an excellent answer for steam generators leak rates surveillance, by means of the tendency checking, and also is useful to promptly detect a tube failure, due to the different response of the affected steam generator in relation to the two others.
- 2.- The operation with small cracks in the steam generators tubes, due to the application of the LBB criterion at the top tube sheet, give place to very reduced leaks, that vary along the cycle, in accordance with the observed measurements of the N-16 monitors, or by the manual analysis in the steam generators blowdown.
- 3.- The evolution of the N-16 monitors indication, between the different steam generators and/or between the diverse cycles, maintains a parallelism. The leak attributed to the transition zone of the tube sheet, decrease initially until they achieve a stable value, then this value is maintained until the moment that in the reactor coolant is reached a pH value of 7.0 - 7.1, at that time the leak rate begins to increase lightly and continuously. This situation remains until the 7.4 pH is reached, later in leak rate decreases again.
- 4.- Since the steam generators blowdown grab samples have less sensitivity than the N-16 monitors, as a consequence of the sludge accumulation in the transition zone, the leak rate evolution has the same behavior, although in minor grade.
- 5.- The repetition of this behavior in both measuring methods along several operation cycles, denote an influence of the reactor coolant chemistry in the leak rates of the steam generators tubes, when these leaks are due to small cracks located in the tubes sheet transition zone.

Unit 1: 9th Cycle

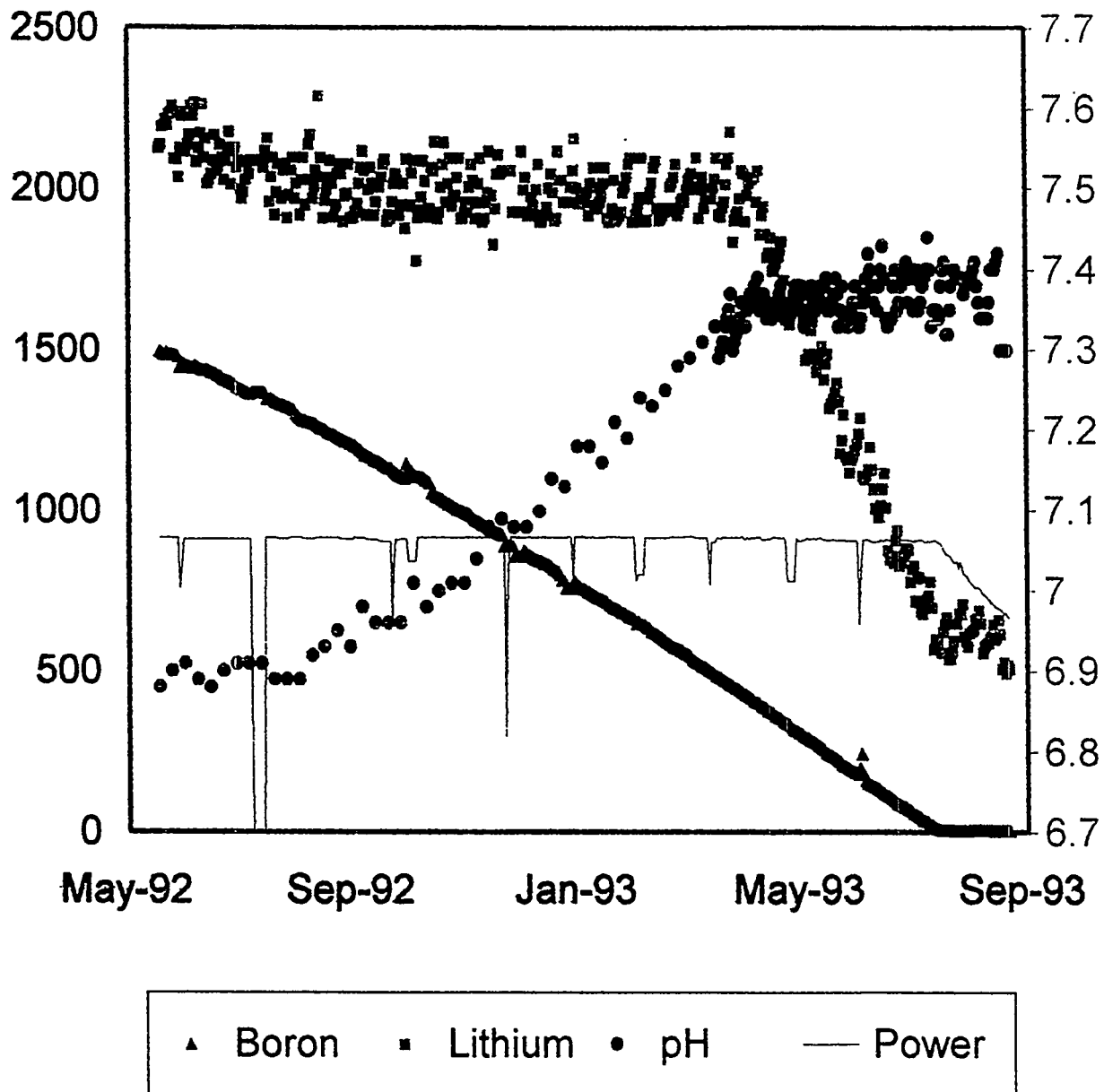


Fig. 1.- Reactor Coolant Chemistry

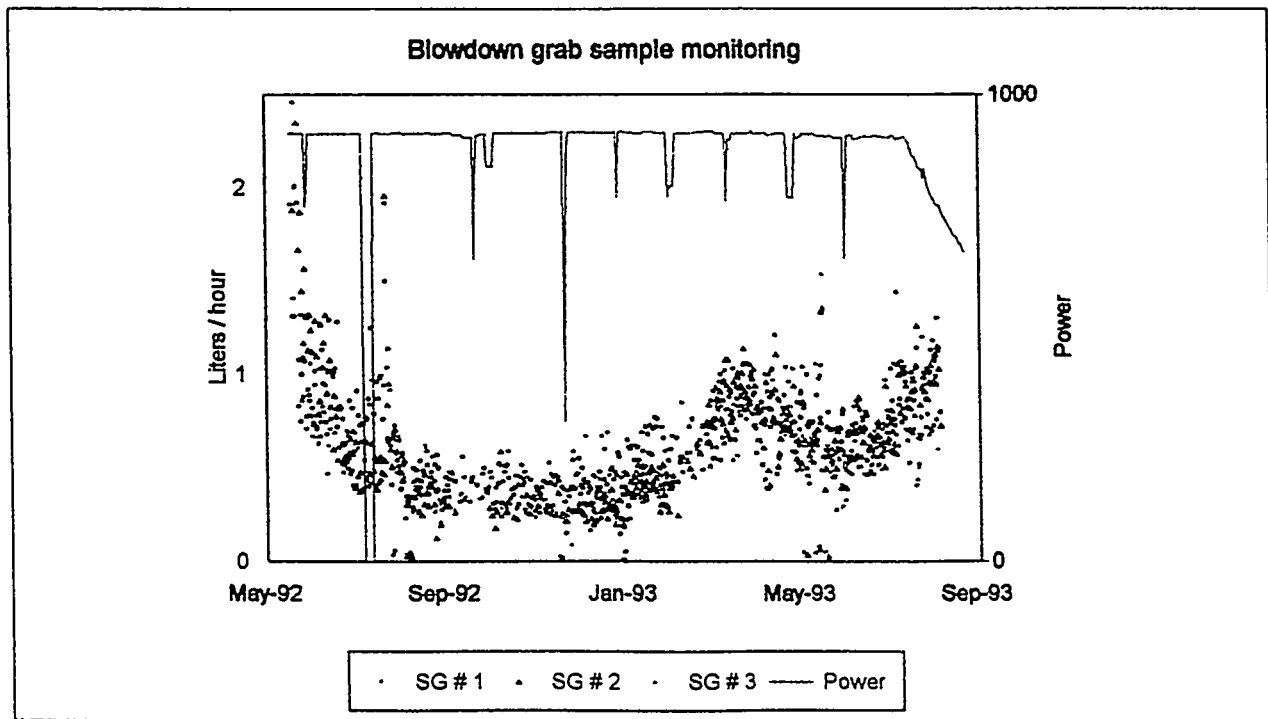
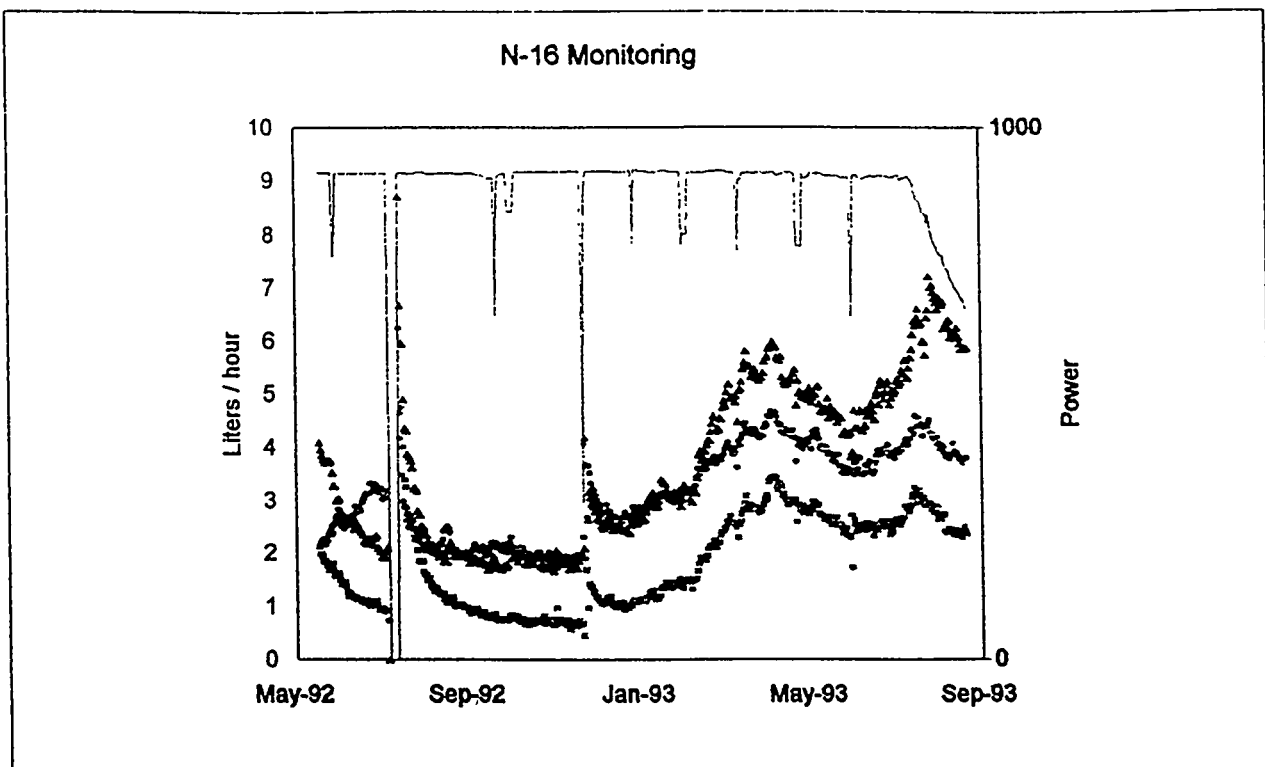


Fig. 2.- UNIT 1: 9th CYCLE. LEAK RATE EVOLUTION

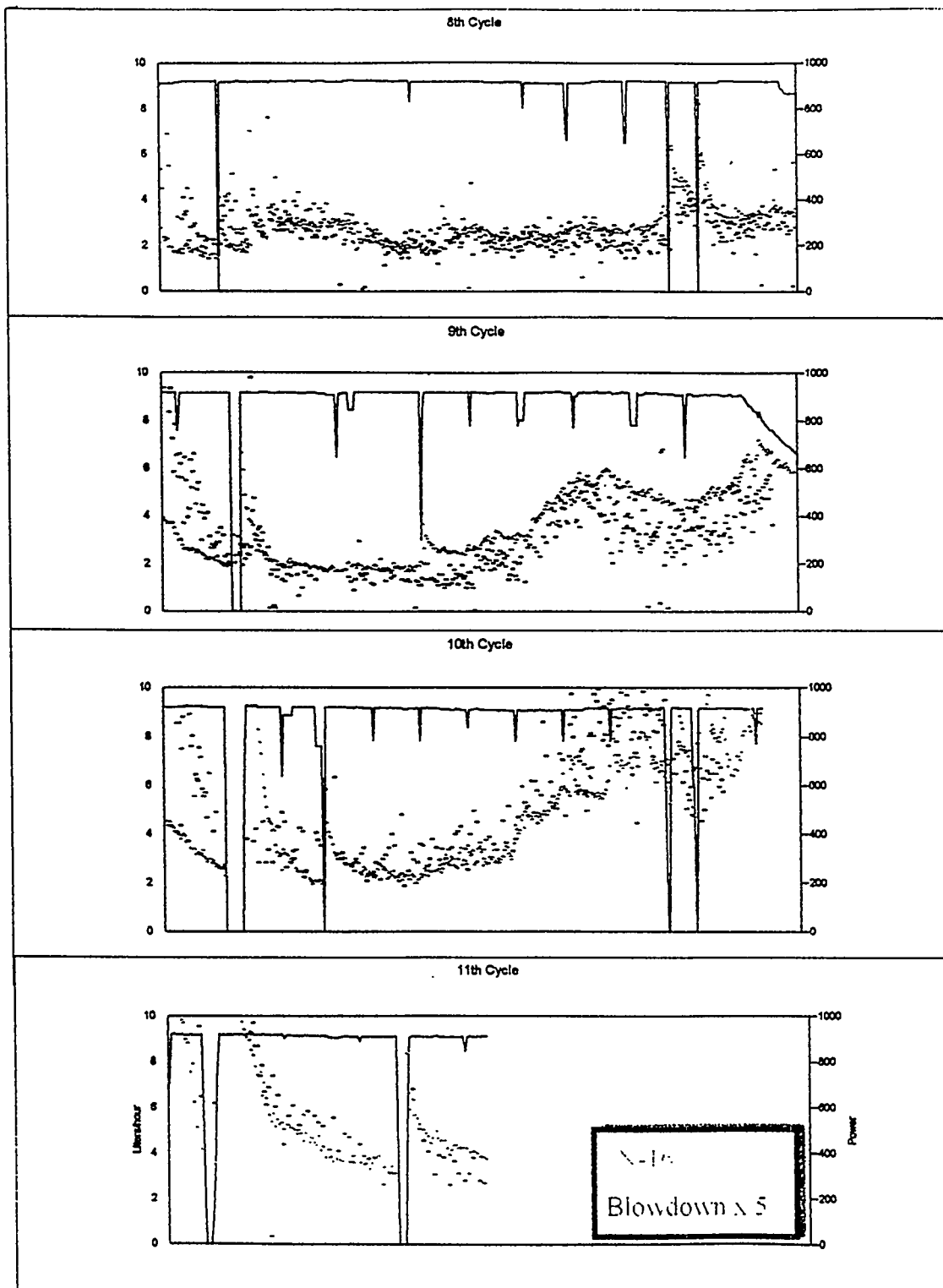


Fig. 3.- Steam Generator #2, leak rate evolution



N - 16 MONITORS
ALMARAZ NPP EXPERIENCE

CNRA/CSNI International Workshop on
Steam Generators Tube Integrity in Nuclear Power Plants

Argonne, Illinois, USA

October 30 - November 2



ALMARAZ NNP CHARACTERISTICS

2 UNITS 930 MW

3 STEAM GENERATORS PER UNIT.

WESTINGHOUSE D-3 MODEL.

LEAK RATES MONITORING

BLOW DOWN.

OFF GAS CONDENSER

N-16 AT MAIN STEAM (After 8th Cycle unit 1 and 7th Cycle Unit 2)



AUTOMATIC MONITORING

CONTINUOUS BLOW DOWN

- 1 Monitor at grab samples collector. (Leak rate surveillance).**
- 2 Monitor at blow down collector .**

OFF GAS CONDENSER

- 1 Monitor at air removal vacuum pumps exhaust. (Leak rate surveillance).**

MAIN STEAM

- 3 N-16 monitors. (Leak rate surveillance).**

ALM-1 SEPT 93



GRAB SAMPLES MONITORING

OFF GAS CONDENSER

DAILY. 3 Steam Generators Global Value.

BLOW DOWN

DAILY. Individual Steam Generators Value.

ALM-4, SEPT 93



FUEL CYCLE CHEMISTRY

COORDINATED B - LI:

Starting with [B] ~ 1800 ppm and [Li] for 6.9 pH minimum.

Stage 1: [Li] decreasing coordinately until reach 2.2 ppm

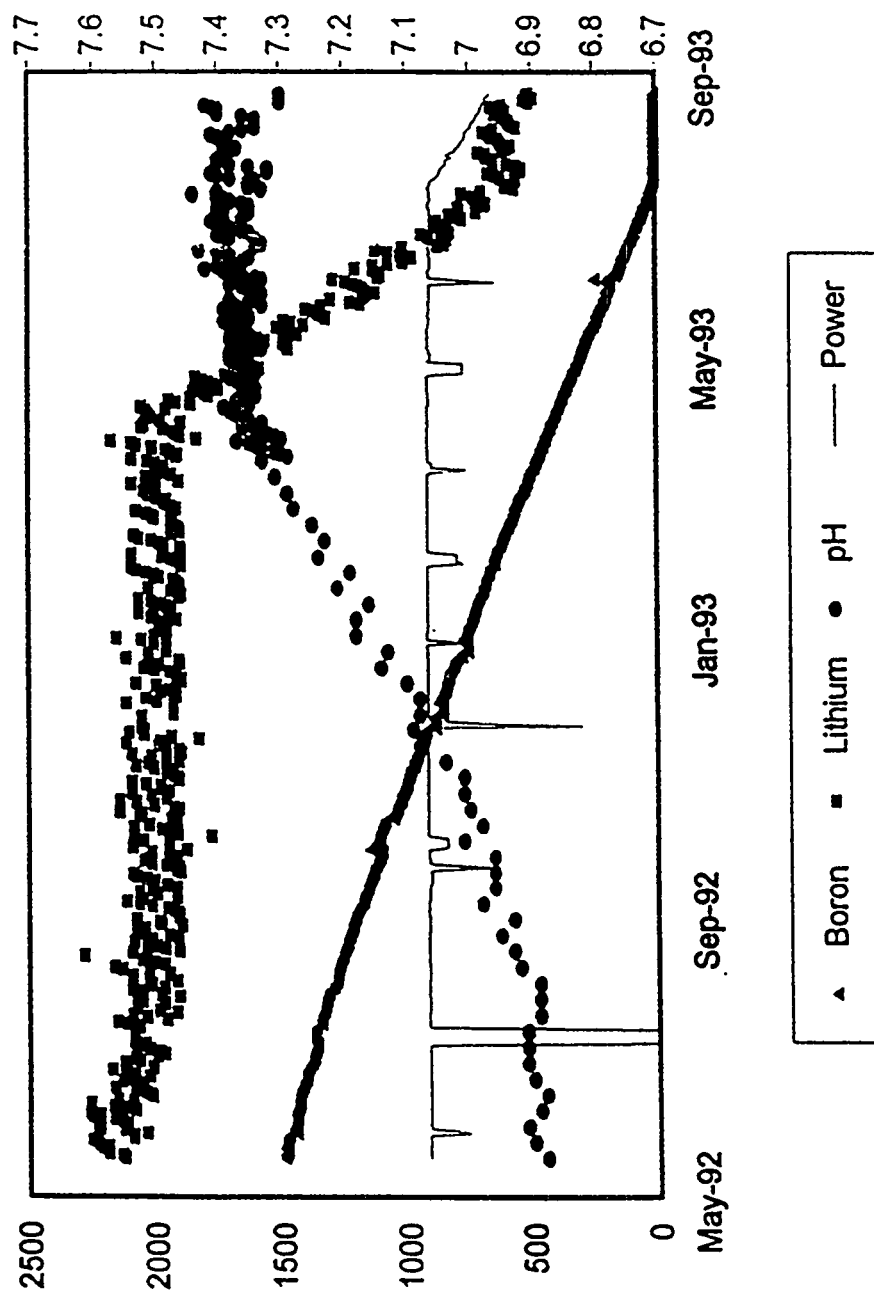
Stage 2: [Li] = 2.2 ppm and pH increasing between 6.9 and 7.4

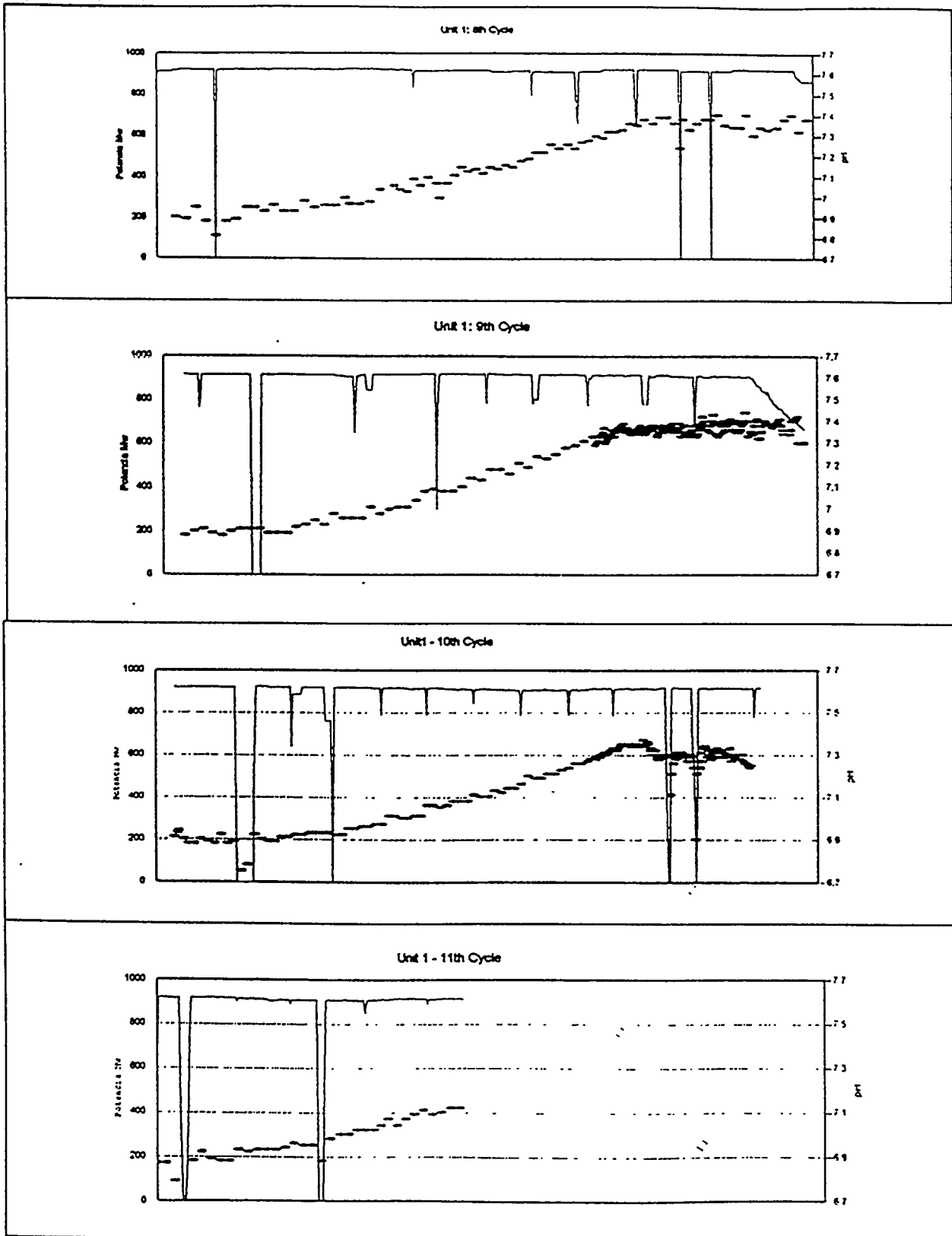
Stage 3: [Li] decreasing coordinately with [B] for pH = 7.4

ALM-3, SEPT 83



Unit 1: 9th Cycle







LEAK RATES AT ALMARAZ

**BOTH UNITS APPLIED THE LBB CRITERIUM AT THE TOP TUBE SHEET
& OPERATE WITH VERY SMALL TUBE CRACKS**

**ONLY UNIT 1 STARTS AND OPERATES WITH SMALL LEAKS OVER LLD
DETECTED BY N-16 & BLOWDOWN MONITORS**

GRAB SAMPLES DETECTION DEPENDS ON REACTOR COOLANT ACTIVITY

BOTH METHODS ARE VERY SENSIBLES TO POWER VARIATIONS

After each event needs 20 or 30 days to reach previous values



UNIT 1: LEAK RATES EVOLUTION

ELEVATED AT THE BEGINING OF THE CYCLE

LITTLE AND CONTINUOUS DECREASE DURING pH 6.9 STAGE

STABLE "PLATEAU" BETWEEN 6.9 AND 7.1 pH

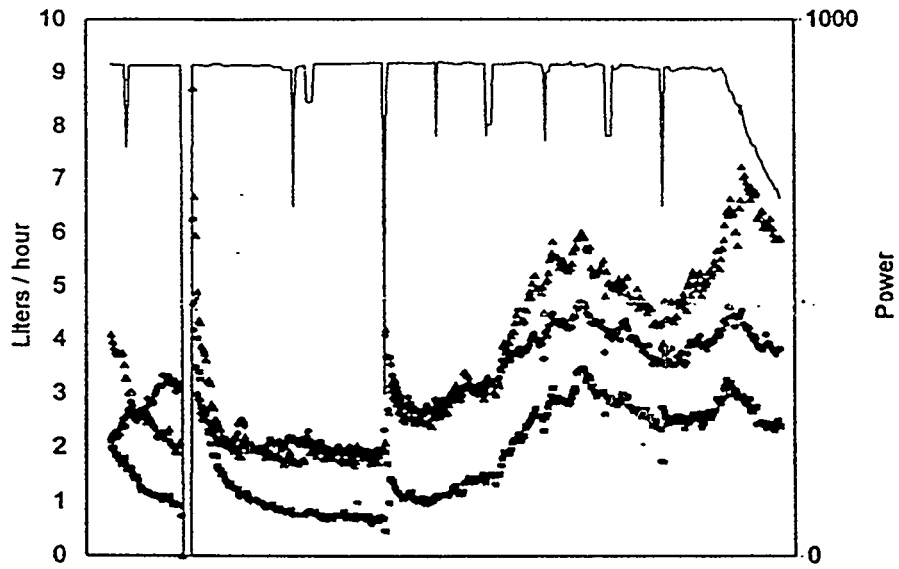
CONTINUOUS AND SMALL INCREASING BETWEEN 7.1 AND 7.4 pH

STABLE OR DECREASING VALUES UNTIL REFUELING

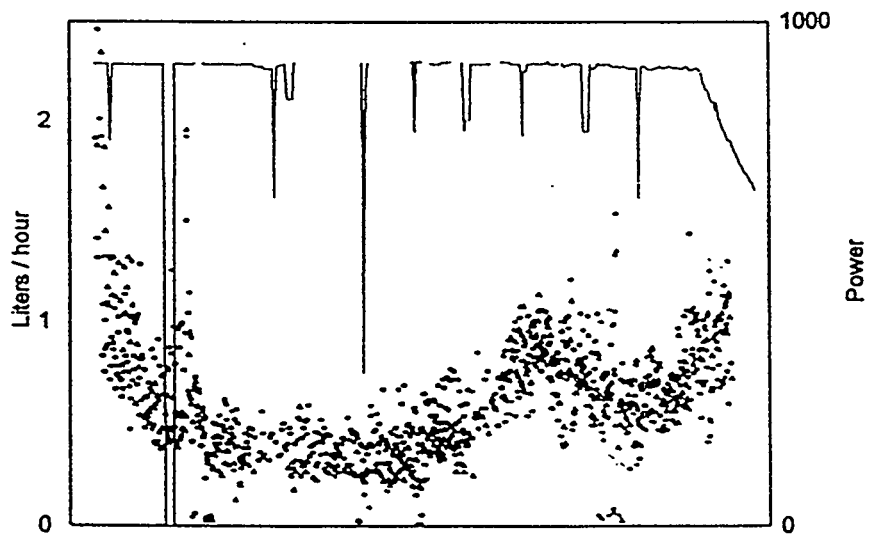
ALM-9, SEPT 93

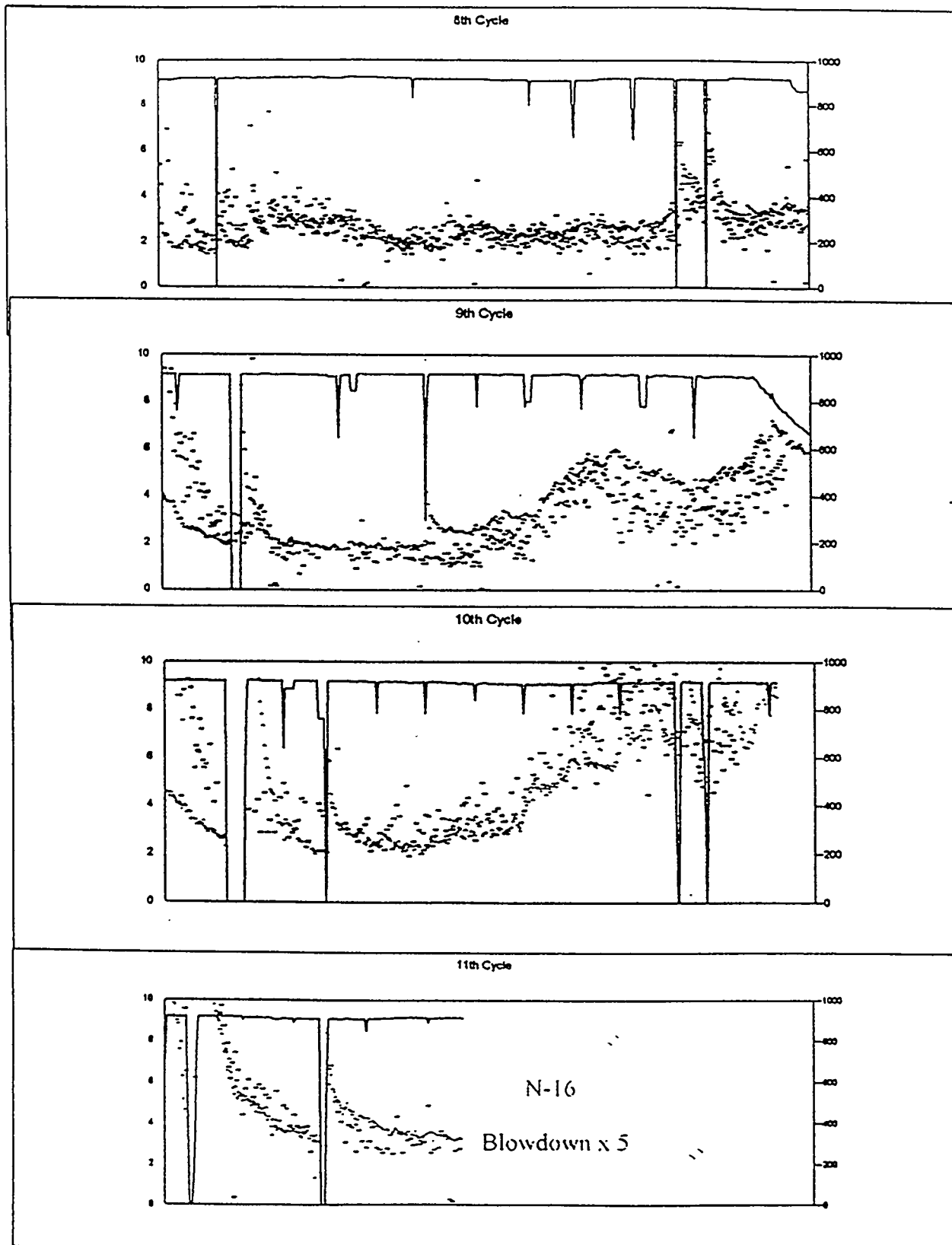


N-16 Monitoring



Blowdown grab sample monitoring







CONCLUSIONS

N-16 MONITORS:

- a) Excellent Answer to Tube Failure
- b) Very Fast Response to Load Variations.
- c) Steam Generator Identification by Differences with the others.
- d) High Leak Rates Values in Comparison with Blow Down Measurements.

BLOW DOWN & OFF GAS CONDENSER MONITORS

- a) Very Fast Response to Leak Variations.
- b) Does not Allow Neither Leak Quantification Nor Steam Generator Identification

BLOW DOWN GRAB SAMPLES SURVEILLANCE

- a) Steam Generator Leak Identification & Quantification.
- b) Values Affected by Sludge Accumulation.

ALM-12, SEPT 95



CONCLUSIONS II

N-16 MONITORS & BLOW DOWN GRAB SAMPLES

ANALOGIES

Shows High Values at BOC
Leak Rate Maintain Plateau until Reach pH 7.1
Continuous Increase Between 7.1 & 7.4 pH
Stable or Slightly Decrease Values to EOC

DIFFERENCES

The N-16 Monitors Leak Indications are 2 to 6 Times Higher

ALM-11, SEPT 93



FINAL CONCLUSIONS

THIS ITERATIVE BEHAVIOUR ALONG SEVERAL CYCLES SHOWS:

**N-16 MONITORS ARE AN EXCELLENT METHOD TO ADVANCE PROMT TUBE FAILURE,
FAST DETECTION & QUANTIFICATION**

**EVOLUTION INDICATE THAT REACTOR COOLANT CHEMISTRY HAVE AN
IMPORTANT INFLUENCE WHEN THE LEAK IS DUE TO SMALL CRAKS
AT TOP OF TUBE SHEET TRANSITION ZONE**

DEFECT SPECIFIC MAINTENANCE OF SG TUBES- HOW SAFE IS IT?

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ABSTRACT

The efficiency of the defect specific plugging criterion for outside diameter stress corrosion cracking at tube support plates is assessed. The efficiency is defined by three parameters: (1) number of plugged tubes, (2) probability of steam generator tube rupture and (3) predicted accidental leak rate through the defects. A probabilistic model is proposed to quantify the probability of tube rupture, while procedures available in literature were used to define the accidental leak rates. The defect specific plugging criterion was then compared to the performance of traditional (45%) plugging criterion using realistic data from Krško nuclear power plant. Advantages of the defect specific approach over the traditional one are clearly shown. Some hints on the optimization of safe life of steam generator are also given.

1 INTRODUCTION

Tubes in steam generators (SG) are exposed to thermal and mechanical loads combined by aggressive environmental conditions. Rather severe corrosion damage results in tubes made of Inconel 600, which may affect the integrity of degraded tubes. The SG tubes represent more than one half of the reactor coolant pressure boundary. Consequently, excessive degradation of tubes might imply reduced plant availability and safety [1].

The appropriate level of plant safety and reliability is maintained by periodical inspection of tubes. Tubes with defects which exceed certain allowable size are then repaired (e.g., sleeved) or removed from service (e.g., plugged). Traditionally, the allowable defect size is defined in terms of tube wall thickness, usually about 40%, and is termed generic plugging criterion [2], [3].

The extent and morphology of the recent types of corrosion damage (Primary Water Stress Corrosion Cracking-PWSCC and Outside Diameter Stress Corrosion Cracking-ODSCC) required more specific approaches. Basically, the conservativities inherent in the development of the generic plugging criterion were replaced by dedicated inspection and implementation of defect specific failure models. This led to the development and implementation of defect specific SG maintenance approaches. In particular, the plugging criterion for PWSCC in expansion transitions

was defined in terms of the length of axial crack, while the ODSCC (at tube support plates only) criterion relies directly on the signal amplitude obtained from a bobbin coil eddy current inspection. A review on the development work and underlying assumptions can be found in [1] and [4].

The analyses addressing the change in plant reliability and safety level due to the implementation of defect specific maintenance are rather scarce and at present limited to the axial PWSCC in expansion transitions (e.g., [5], [6] and [7]). Such analyses are usually based on probabilistic fracture mechanics techniques and clearly showed the advantage of defect specific over traditional plugging criterion, both in terms of probability of tube rupture and in the number of tubes plugged.

In this paper we propose a method which evaluates the impact of defect specific maintenance on steam generator operational safety in the case of ODSCC. A probabilistic approach already implemented in the case of axial PWSCC in expansion transitions [6] is extended to cover the specific points of the ODSCC approach. Essentially, we follow the methodology developed by EPRI [8] for ODSCC at tube support plate intersections. The main result is the efficiency of defect specific maintenance as compared to the generic maintenance in terms of: (1) tube rupture probability, (2) expected accidental primary to secondary leak rate and (3) number of plugged tubes. These three parameters also represent a basis for the optimization of safe steam generator life time. The results of the numerical example clearly show the advantages of the defect specific criterion.

An additional topic covered in this paper is the relatively fast numerical method implemented in the calculations of the probability of tube rupture, as opposed to the computationally very intensive Direct Monte Carlo simulation, which is acceptable to the NRC [9]. Approximate methods called First- and Second Order Reliability Methods (FORM and SORM, respectively, for the implementation see Refs. [7] and [10]) were used in the calculations of tube rupture probabilities.

2 COMMENTS ON THE MODELING

A brief description of the calculation procedure is given below. The main body of the calculations follows the methodology defined for the alternate voltage criteria by EPRI (for details see [8], [9] and references therein), which is not repeated herein.

2.1 Probability of tube rupture

The probability of failure P_f in probabilistic fracture mechanics is usually defined by:

$$P_f = \int_{g(x_1, x_2, \dots, x_n) \leq 0} f_1(x_1) f_2(x_2) \dots f_n(x_n) dx_1 dx_2 \dots dx_n \quad (1)$$

The statistically independent basic variables and their probability densities, n in number, are denoted by $x_1, x_2, \dots, x_n$ and $f_1(x_1), f_2(x_2), \dots, f_n(x_n)$. The failure is defined in terms of failure function $g(\cdot)$, which is by definition negative for all failure states.

The failure function in the case of ODSCC specific maintenance can be defined simply as:

$$g(\Delta p_{SLB}, \Delta p_{EOC}) = \Delta p_{SLB} - \Delta p_{EOC} \quad (2)$$

Δp_{SLB} denotes the pressure difference obtained from the limiting accidental state such as Steam Line Break (SLB). Δp_{EOC} represents the burst pressure of the tube containing a defect at the end of inspection cycle (EOC). A tube is therefore considered to be failed (ruptured) when its burst pressure becomes less than the highest expected accidental pressured difference (e.g., at SLB). Δp_{EOC} is obtained from the correlation between burst pressure and bobbin coil signal amplitude [8] as a function of the End of Cycle defect size. This in turn is obtained by a stochastic combination of basic variables describing:

- distribution of defects at the beginning of inspection cycle (BOC). The plugging criterion may be included here by imposing appropriate upper limit to the distribution function;
- defect progression until the end of inspection cycle (EOC);
- uncertainties inherent to the Eddy Current Technique (ECT) measurement variability.

The calculation of Δp_{EOC} accounts also for uncertainties inherent to the correlation between burst pressure and bobbin coil signal amplitude.

The value of P_f (eq. (1)) was obtained using the First and Second Order reliability methods as implemented in the ZERBERUS code [7],[10]. It represents the fraction of failed tubes in the population of all defective tubes. The observed steam generator is then represented as a random sample of size N . The probability of having i tubes failed $p(i)$ then follows from the Poisson distribution:

$$p(i) = \frac{(N \cdot P_f)^i}{i!} \exp(-N \cdot P_f) \quad (3)$$

This enables the calculation of single and multiple tube rupture probabilities.

2.2 Maximum expected leak rate

The estimation of the maximum expected leak rate at the end of operating cycle (EOC) under accidental conditions follows the procedure outlined in [8] and [12]. Some possible improvements of the procedure are discussed in [11]. They are however not implemented in the present analysis.

2.3 Number of plugged tubes

Suppose that inspection of the tubes revealed N defects. Distribution of defect sizes (in terms of bobbin coil signal amplitudes) is denoted by $f(V_{BOC})$. Imposing a plugging criterion with value of V_{PLG} then the number of plugged tubes N_{PLG} can be obtained from the relation:

$$\frac{N_{PLG}}{N} = \frac{\int_{V_{PLG}}^{\infty} f(V_{BOC}) dV_{BOC}}{\int_0^{\infty} f(V_{BOC}) dV_{BOC}} \quad (4)$$

3 NUMERICAL EXAMPLE

The data used in the following numerical example is specific for the Slovenian NPP at Krško. Detailed description of data and methodology used is given in [12].

The comparison of the efficiency of both different maintenance strategies (defect specific and generic) is presented as a function of voltage plugging criterion. The efficiency was evaluated through

- predicted total leak rate through one steam generator during the postulated limiting accident (SLB) at the end of cycle;
- estimated probability of tube rupture (single or multiple) at the EOC, given postulated limiting accidental conditions (FLB);
- number of tubes which will be plugged using one or another maintenance strategy.

Differences in maintenance strategies were described by elimination of plugged tubes from the inspection results. Thus, setting the voltage criterion to 1.5 V simply caused setting 1.5 as an upper limit of the distribution describing the BOC defect sizes. On the other hand, the list of tubes plugged because of the 45% plugging criterion was available. Therefore, all plugged tubes were eliminated from the inspection results, yielding an estimated distribution of defect sizes without an upper limit.

3.1 Maximum expected leak rate

An evaluation of leak rate for different maintenance strategies was performed. Leak rate prediction was performed for four sets of data obtained from two consecutive inspections of both steam generators. A postulated SLB accident with a conservative pressure difference of 182.7 bar (2650 psi) was assumed in the analysis.

The defect size (BOC) and defect growth data were fitted using lognormal distributions. Different plugging criteria are reflected through different distributions of BOC defect sizes. Only defects on tubes which will remain in operation after implementing particular plugging criterion were used to define BOC defect size distributions.

ECT uncertainties were assumed to be normally distributed with zero mean. Standard deviation of uncertainties was assumed to be proportional to the defect size. No upper limit was imposed on the ECT uncertainties in the calculation. This is a conservative assumption.

The efficiency of a defect specific plugging criterion was estimated in the following way: First, the total leak rates were predicted for all four sets of data corresponding to the 45% plugging criterion. Then, the total leak rates for the defect specific plugging criterion were predicted for all four data sets. A ratio between corresponding leak rates was then plotted as a function of defect specific plugging criterion (Fig. 1).

The predicted total leak rates show similar behavior in the qualitative manner. A considerable scatter can however be observed. This is essentially caused by different *number* and *distribution* of defects included in data sets available. A practical conclusion is that such analyses should be preformed after *each* inspection of a particular steam generator.

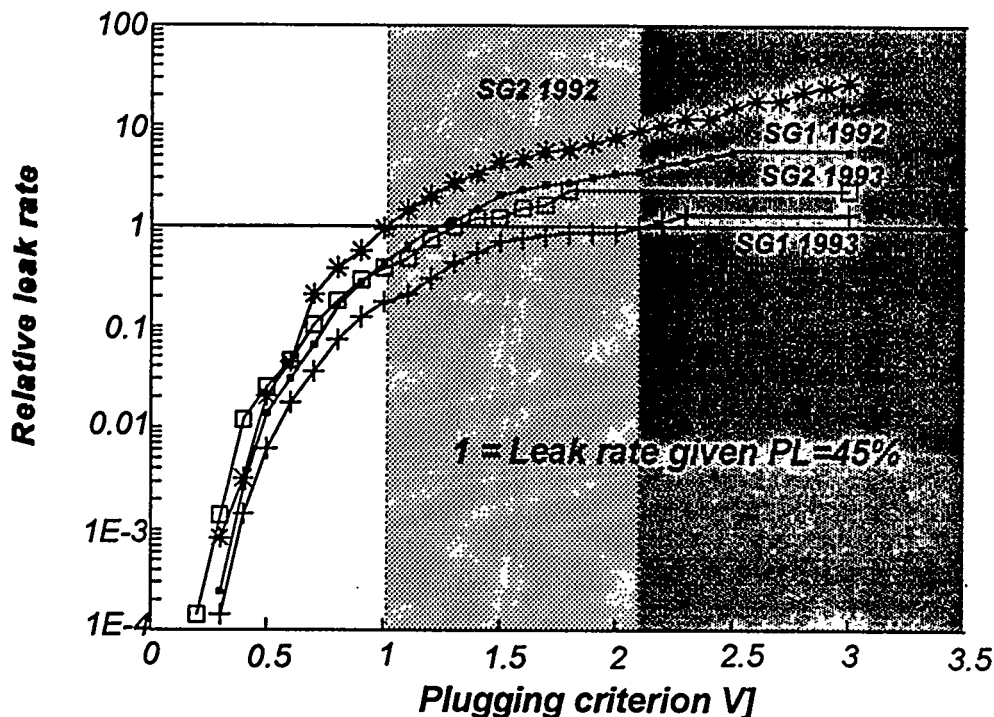


Figure 1 Comparison of relative total leak rates

Three distinct regions are shown in Fig. 1. First, all data sets show better performance of the defect specific approach for plugging criteria below about 1 V. This is therefore the region of better performance of defect specific plugging criterion.

The second region (between about 1 and 2 V, Fig. 1) is an intermediate region. Better efficiency is a strong function of data in this region.

In the third region (above about 2V), the defect specific plugging criterion is approaching the state of *no plugging at all*. It is therefore appropriate to say that in this region, the defect specific approach is not efficient at all.

We should stress here again that the above results strongly depend on the particular steam generator analyzed. This results can therefore not be generalized either to another steam generators or other inspection results of the same steam generators.

3.2 Tube rupture probability

To estimate tube rupture probability a postulated Feed Line Break (FLB) accident was assumed with differential pressure of 195.6 bar (2850 psi).

In this analysis we only present the single tube rupture probabilities. As to the absolute values, they were consistently less than 1%. These values are, as stated above, conditional given a postulated FLB accident. The probability of multiple tube ruptures was at least two orders of magnitude lower than for single tube rupture for all four cases analyzed. Thus, the multiple tube rupture event was not considered to be of particular importance here.

Again, the results obtained from all four data sets show similar behavior in the qualitative manner. Therefore, only the results based on the most recent data will be given here. Results which were obtained by implementation of 45% plugging criterion are again set to unity.

Fig. 2 (curve denoted by Probability of SGTR) depicts the effect of the defect specific plugging criterion on the probability of single tube rupture. Setting the defect specific plugging criterion below about 1.3 V is shown to outperform the traditional approach (45%). However, above 1.3 V we are again approaching the inefficient state of *not plugging at all*. It should be mentioned here that this using traditional 45% criterion requires some tubes to be plugged, but does not significantly contribute to the plant safety. This observation can of course only be made for this particular set of data.

3.3 Toward the optimization of SG life time

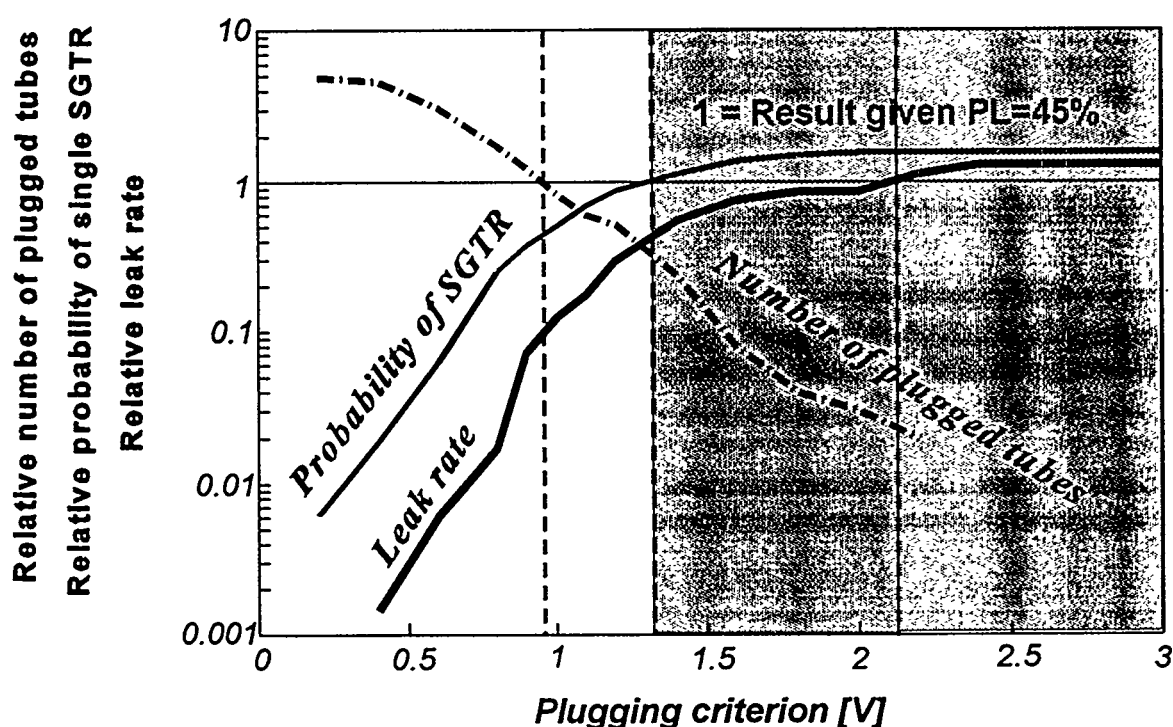


Figure 2 Towards the optimization of safe life time of steam generators

The number of tubes required for plugging is the third important parameter in the present analysis. In practice, we can decide to plug some additional tubes to comply with some preset acceptable probability of tube rupture and total leak rates.

Generally, more tubes plugged should reduce both the probability of SGTR and the maximum expected total leak rate. Unfortunately, excessive plugging reduces the lifetime of the steam generators. We are therefore seeking for some method enabling optimization between safe and long operation of steam generators. A step toward such optimization procedure is depicted in Fig. 2, which again has a relative scale with unity representing the performance of the traditional 45% plugging criterion.

The general behavior of all three curves in Fig. 2 was already explained in above discussion. We shall now examine some quantitative properties. It is obvious that defect specific plugging criteria below approx. 1V can provide lower SGTR probabilities and expected accidental leak rates. This increase in performance is essentially obtained by excessive plugging of tubes.

The region between 1 and 1.3 V represents the region with best performance of the defect specific plugging criterion. In this region, lower likelihood of tube bursting and excessive leakage is obtained with fewer tubes plugged as compared to the traditional approach. This is also the region where the present values of defect specific criteria reside (1V for 3/4" tubes [9]).

Between 1.3 and 2.1V, the SGTR probability is raised to its asymptotic level (given the defect population), while there can still be significant reduction in expected leak rates as compared to the traditional approach. This region may be very useful if we are more interested in preventing excessive leak rates (without tube rupture) than the tube rupture event. It should be however noted here that this effect may be caused only by the uncertainties in the correlations defining tube bursting and individual leak rates. Some more detailed physical modeling may be therefore required to clarify this effect.

We shall state here again, that the quantitative basis of above considerations strongly depends on a particular steam generator and operating cycle considered.

4 CONCLUSIONS

The efficiency of defect specific plugging criteria for steam generator tubes was studied based on the comparison with the traditional plugging criterion allowing for 45% of the tube wall loss. In particular, the alternate plugging criterion for the ODSCT under tube support plates which is based on the bobbin signal amplitude was taken as an example.

Three parameters were defined as a measure of the efficiency: (1) number of plugged tubes, (2) probability of having a tube rupture and (3) predicted accidental leak rate through damaged steam generator tubing.

The methods used in the comparison essentially follow the probabilistic fracture mechanics approach already implemented in the case of primary water stress corrosion cracking. They were appropriately modified to accommodate the present analysis.

The efficiency of the defect specific plugging criterion was found to depend strongly on the number and distribution of the defects in the steam generator tubing and value of the criterion implemented. However, in all cases analyzed we found that the current implementations of defect specific plugging criteria outperformed the traditional with respect to all of the parameters compared.

The topics which should get closer attention in the future are more accurate or more physical-based modeling of the individual leak rates.

5 ACKNOWLEDGMENTS

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ON THE PROBABILITY OF EXCEEDING ALLOWABLE LEAK RATES THROUGH DEGRADED STEAM GENERATOR TUBES

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ABSTRACT

This paper discusses some possible ways of predicting the behavior of the total leak rate through the damaged steam generator tubes. This failure mode is of special concern in cases where most through-wall defects may remain in operation. A particular example is the application of alternate (bobbin coil voltage) plugging criterion to Outside Diameter Stress Corrosion Cracking at the tube support plate intersections. It is our aim to discuss some possible modeling options that could be applied to solve the problem formulated as: Estimate the probability that the sum of all individual leak rates through degraded tubes exceeds the predefined acceptable value.

The probabilistic approach is of course aiming at reliable and computationally bearable estimate of the failure probability. A closed form solution is given for a special case of exponentially distributed individual leak rates. Also, some possibilities for the use of computationally efficient First and Second Order Reliability Methods (FORM and SORM) are discussed. The first numerical example compares the results of approximate methods with closed form results. SORM in particular shows acceptable agreement. The second numerical example considers a realistic case of NPP in Krško, Slovenia.

1 INTRODUCTION

The steam generator tubes in nuclear power plants with pressurized water reactors comprise most of the reactor coolant pressure boundary. The tubes made of Inconel 600 are susceptible to corrosion damage. This may have detrimental consequences on the plant safety and reliability. Two potential failure modes of the tubes are of special concern [1]:

- single or multiple tube rupture and
- excessive leak of the reactor coolant to the secondary side (without tube rupture).

A sufficient safety margin against tube rupture used to be the basis for a variety of maintenance strategies (inspection and repair of tubes [1, 2]) developed to maintain suitable level of plant safety and reliability. Originally, the defects not deeper than 40% of the tube wall thickness were

allowed [3]. Therefore, all tubes with through-wall defects were assumed to be removed from operation (e.g., plugged) or repaired (e.g., sleeved). This essentially precluded the second failure mode (excessive leaks).

Recently, the defect specific maintenance and fitness-for-service criteria for degraded steam generator tubes are being implemented on a worldwide basis [1], [4]. An important consequence of such approach is that through-wall defects in tubes may remain in operation. This requires the quantification of the expected leak rates and a demonstration of the available safety margin.

A new probabilistic approach aimed at predicting the probability of excessive leaks is proposed in this paper. The application of probabilistic methods that explicitly account for rare events is inspired by the need to account for uncertain parameters governing the calculation of leak rates. An additional consideration is that the inspection of tubes actually reveals the state of individual defects on tubes, which in turn enables prediction of individual leak rates through a single defect. On the other hand, the monitoring during operation and hypothetical accidents depends on the total leak rate through one steam generator. Therefore, we are going to estimate the *probability that the sum of individual leak rates exceeds the allowable leak rate* (given pressure difference and the distribution of defects in the steam generator). The probabilistic approach proposed here is based on the analogy with probabilistic models estimating the tube burst probability of axial through wall cracks [5], [6].

Approximate numerical methods called First- and Second Order Reliability Methods (FORM and SORM, respectively) are briefly introduced. Their applicability in safety assessment of cracked steam generator tubes has already been shown [7]. Additional tests against the available closed form results were conducted for excessive leaks. SORM in particular showed reasonable agreement and should be further considered in the analysis of practically relevant cases in the future.

2 FORMULATION OF THE PROBLEM

In the following discussion we assume an existing deterministic model describing the individual leak rate as a function of defect size. The uncertainties inherent to this model and its parameters are assumed to be known and quantified at least to the extent of leak rate models proposed to support the alternate bobbin coil voltage plugging criterion (see [8], [4] and references therein).

Then, the individual leak rate through a single defect can be described as a random variable, denoted by Q_i . Its probability density function is denoted by $f_i(Q_i)$. Some additional useful assumptions are:

- all individual leak rates are members of the same population. This leads to a unique probability density function:

$$f(Q_i) = f(Q_1) = f(Q_2) = \dots f(Q_n) \quad (1)$$

- individual leak rates are statistically independent This simplifies the joint probability density function:

$$f(Q_1, Q_2, \dots, Q_n) = \prod_{i=1}^n f(Q_i) \quad (2)$$

The total leak rate Q_T is simply a sum of all individual leak rates Q_i . Let us therefore define the failure of a steam generator by the event "total leak rate Q_T exceeds allowable leak rate Q_{MAX} ." The failure probability P_f is then given by:

$$P_f = P(Q_T \geq Q_{MAX}) = P\left(\sum_{i=1}^n Q_i \geq Q_{MAX}\right) \quad (3)$$

Using the well-known procedures from statistical reliability theory, the failure probability P_f can be calculated by the integration of the probability content of the failure space ($Q_T > Q_{MAX}$):

$$P_f = \int_{g(Q_1, Q_2, \dots, Q_n) < 0} \prod_{i=1}^n (f(Q_i) dQ_i) \quad (4)$$

$g(\bullet)$ represents the failure function, which is by definition positive in the safe space and negative in the failure space ($Q_T > Q_{MAX}$). Therefore, it is given here as:

$$g(Q_1, Q_2, \dots, Q_n) = Q_{MAX} - \sum_{i=1}^n Q_i \quad (5)$$

The hyper surface obtained by setting $g(\bullet) = 0$ represents a boundary between safe and failure space and is usually called failure surface (also limit state surface).

3 ON THE POSSIBLE SOLUTIONS

Closed form solutions of eq. (4) are possible for some special types of individual leak rate distributions. Especially exponential distribution yields a very convenient solution. It is given below because it may serve very well as a reference case for the verification of numerical solutions, which are required to solve the cases with arbitrary distributions.

Among available numerical solutions, the direct Monte Carlo simulation is a very convenient and straightforward method to implement. It is however computationally very intensive. For example, a realistic case with 10^3 individual leak rates and failure probability in the order of 10^{-4} would require about 10^8 evaluations of the leak rate model. This may be computationally prohibitive. It is therefore not convenient for routine use, but may serve very well for the verification purposes. It should be mentioned here that direct Monte Carlo is considered an "exact" method in the literature and is also accepted as a tool in the assessment of reliability of steam generator tubes by the U.S. NRC [4].

Alternative approximate methods such as First and Second Order Reliability Methods (FORM and SORM, respectively [9], [10]) were recently successfully implemented in safety assessment of cracked steam generator tubes [7]. In particular, the failure probabilities of axially cracked tubes were analyzed using FORM and SORM [6]. Thus, implementing them also in assessment of excessive leak rates may be worthwhile, as described below.

3.1 Closed form solution

A rather straightforward closed form solution is possible for exponential $f(Q_i)$:

$$f(Q_i) = \frac{1}{\beta} \exp\left(-\frac{Q_i}{\beta}\right) \quad (6)$$

where β represents the mean of all individual leak rates. Here, the failure probability (eq. (4)) is simply given as [11]:

$$P_f = \exp\left(-\frac{Q_{\max}}{\beta}\right) \sum_{i=0}^{n-1} \frac{1}{i!} \left(\frac{Q_{\max}}{\beta}\right)^i \quad (7)$$

This may be interpreted as a cumulative Poisson distribution with parameter $Q_{\max}/\beta$. The parameter $Q_{\max}/\beta$ represents a number of average individual leak rates required to exceed the allowable leak rate.

3.2 First Order Reliability Method

The basic idea behind FORM is to transform all basic random variables to the space of uncorrelated standard normal variables. A point on the transformed failure surface with the minimum distance toward the origin of the space of uncorrelated standard normal variables is to be found. This point is traditionally called design point. Finally, the failure surface is approximated by a hyperplane passing through the design point and being normal to the position vector of the design point. Then, a simple solution of the failure integral (eq. (4)) exists:

$$P_{f, FORM} = \Phi(-\beta_f) \quad (8)$$

$\Phi(\bullet)$ is standard normal distribution and β_f the reliability index, which represents the minimum distance between the design point on the failure surface ($g(\bullet)=0$) and the origin of the space of standard normal variables. Further details on the implementation of the method are described in [7], [12] and references therein.

The assumptions on the individual leak rates (eqs. (1) and (2)) allow for some simplifications of FORM. Recalling eq. (5) and identical probability densities of individual leak rates, the design point in the physical space is given as:

$$Q_1 = Q_2 = \dots = Q_n = \frac{Q_{\max}}{n} \quad (9)$$

Essentially, this means that exceeding the limit by a sum of smaller leaks is more likely than by extreme individual leaks. The coordinates of the design point are then transformed to the standard normal space by [9]:

$$u_{1,n} = u_{2,n} = \dots = u_{n,n} = \Phi^{-1} \left[F \left(\frac{Q_{\max}}{n} \right) \right] \quad (10)$$

This suggests a strong dependence on n and defines reliability index β_f as:

$$\beta_f = u_{1,n} \sqrt{n} \quad (11)$$

The FORM approximation of the failure probability is therefore given by:

$$P_{f,FORM} = \Phi(-\beta_f) = \Phi \left\{ -\sqrt{n} \Phi^{-1} \left[F \left(\frac{Q_{MAX}}{n} \right) \right] \right\} \quad (12)$$

For $n=1$, an exact solution is obtained. The case with $n=2$ shows an underestimate. This n -dependence is a serious drawback of FORM. It can be illustrated by considering the case $u_{1,n} \approx u_{1,n+1}$. It is rather straightforward to show that this requires the condition of $F(Q_{\max}/n) \approx F(Q_{\max}/(n+1))$. Recalling eqs. (11) and (12), FORM could obviously assign a lower failure probability to a case with more leaks. Therefore, the application of FORM would be of limited practical value.

3.3 Second Order Reliability Method

Second order reliability method approximates the failure surface in the design point by a quadratic hyper surface (instead of hyperplane in FORM). The design point and reliability index β_f obtained by FORM is generally used for this purpose. The main curvatures of the quadric are identical with main curvatures of the failure surface in the design point. For a rotationally symmetric quadric, the approximate value of failure probability is obtained by [10]:

$$P_{f,SORM} = \int_0^{\infty} \Phi \left(- \left(\frac{t}{2R} + \beta_f \right) \right) \chi^2_{n-1}(t) dt \quad (13)$$

$\chi^2_n(\bullet)$ is chi-square probability density with n degrees of freedom. R is the radius of the curvature in the design point. The hyper quadric is rotationally symmetric with respect to the position vector of the design point and has its apex in the design point.

4 NUMERICAL EXAMPLE

4.1 Numerical vs. closed form solutions

Consider the exponential distribution of individual leak rates with mean value of .03 l/h. The allowable leak rate through the steam generator is set to 25l/h. The failure probability is plotted as a function of number of individual leaks in **Figure 1**.

Failure probability, as expected, strongly depends on the number of individual leaks. A relatively small increase in the number of leaks can cause a few orders of magnitude higher failure

probabilities. This is especially important in the connection with nondetection probability of the inspection hardware used.

FORM clearly underestimates the exact results and is not recommended to be used as a standalone method in further analysis. It is however essential as a base for the SORM solution that shows reasonable agreement with exact results.

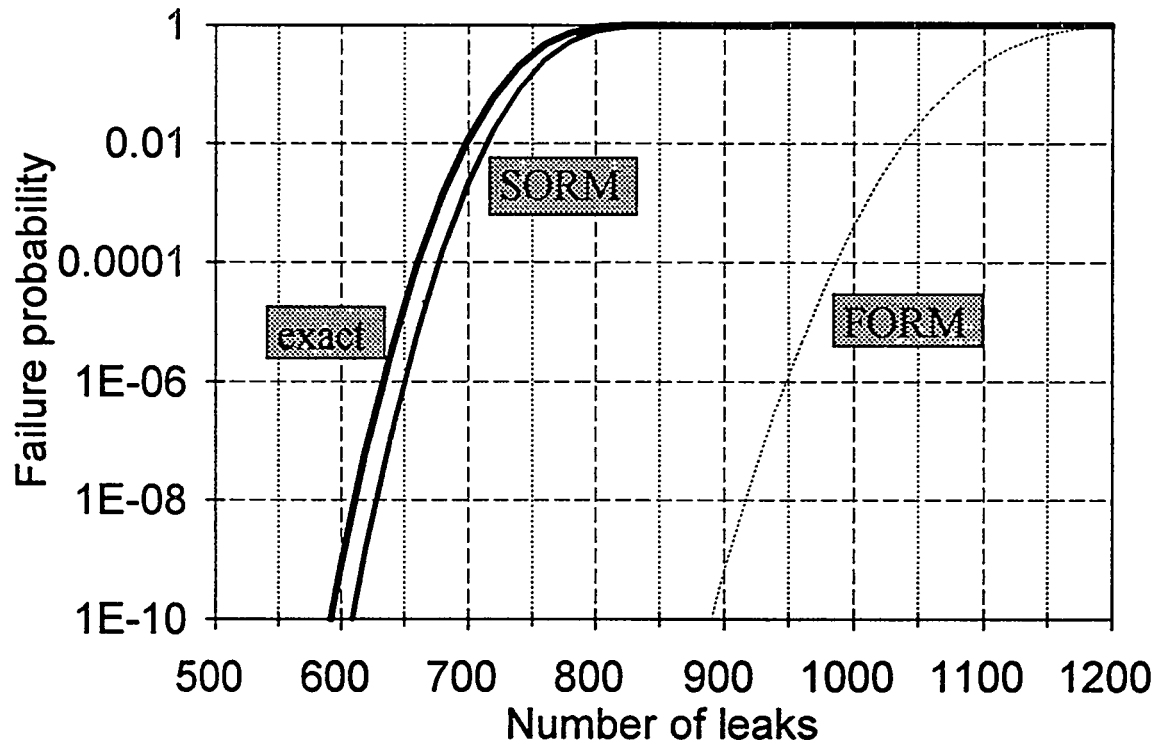


Fig. 1 Comparison of FORM and SORM results with exact solution

4.2 A realistic example

Recently, a specific fitness-for-service criterion has been proposed [13] and accepted [4] in the USA. It addresses the outside diameter stress corrosion cracking at tube support plates. The demonstration of safety against both failure modes is based on a rather simple regression analysis of defect sizes vs. burst pressures and individual leak rates. The data base used in regression analysis was obtained from a sample of tubes pulled out from operating steam generators and model boiler facility [13]. In the following, only the excessive leak rates will be addressed. The tube failure probabilities are addressed elsewhere [14].

A linear regression analysis resulted in the following leak rate model:

$$\log(Q) = a + b \log(V) + \epsilon \quad (14)$$

Where Q represents the individual leak rate through a defect. Defect size V is expressed simply as the signal amplitude (in Volts) obtained from a bobbin coil. An extensive analysis was also done

to characterize the confidence intervals of the regression model parameters (intercept a , slope b and error term ϵ) [13].

The total leak rate means a sum of individual leak rates through all detected defects in a given steam generator. To obtain a conservative estimate of the upper limit, the measured defect sizes are grouped together in small intervals (e.g., 0.1 V). Each class contains m_i defects with maximum size V_i . The upper 95% limit (with 95% confidence) of the total leak rate is defined as [13]:

$$Q_{\Sigma} = \sum_{i=1}^n m_i Q(V_i)_{961|1000} \quad (15)$$

The subscript 961|1000 means that 1000 values of $Q(V_i)$ are generated using Monte Carlo simulation and then sorted in ascending order. The 961th value of this ordered set is then chosen as a representative value for the entire class of defects. Allowances for defect growth and measurement errors are also provided and affect the actual value of V_i . Using the Krško power plant specific values (for beginning of cycle defect size distribution see Figure 2), a value of 25 l/h was obtained for the most unfavorable accidental conditions.

The statistical stability of the 95% total leak rate estimate was analyzed using Monte Carlo simulation (Table 1) and was considered reasonable. A big scatter is however observed in sums obtained by averaging over 1000 samples of each data class or simply taking their maxima. This scatter is mainly caused by the large influence of the residual term in the leak rate model (eqn. (14)). We should stress here that the maximum expected leak rate as predicted by the EPRI model always exceeded the 95% upper limit by at least one order of magnitude.

Total leak rate based on eqn. (15):	Results from 100 numerical experiments[l/h]				
	Mean	Median	Std.Dev.	Minimum	Maximum
min (i=1)	<10 ⁻⁴	<10 ⁻⁴	<10 ⁻⁴	<10 ⁻⁴	<10 ⁻⁴
median (i=501)	0.36	0.36	0.01	0.33	0.38
95% (i=961)	26.27	26.3	1.32	23.3	29.5
max (i=1000)	4296	2478	5347	872	33432
averaged 1000	10.	8.32	5.6	6.0	38.9

Table 1 Statistics of selected parameters obtained from EPRI procedure for the total leak rate

The distribution of defect sizes is given in Figure 2. In the calculations, both empirical and lognormal distributions were used. Both distributions gave essentially the same results. This is caused by rather large uncertainty of the regression model. The number of defects was set to 495 for the EPRI point estimate calculations outlined above. The value of 25 l/h was obtained as a 95%

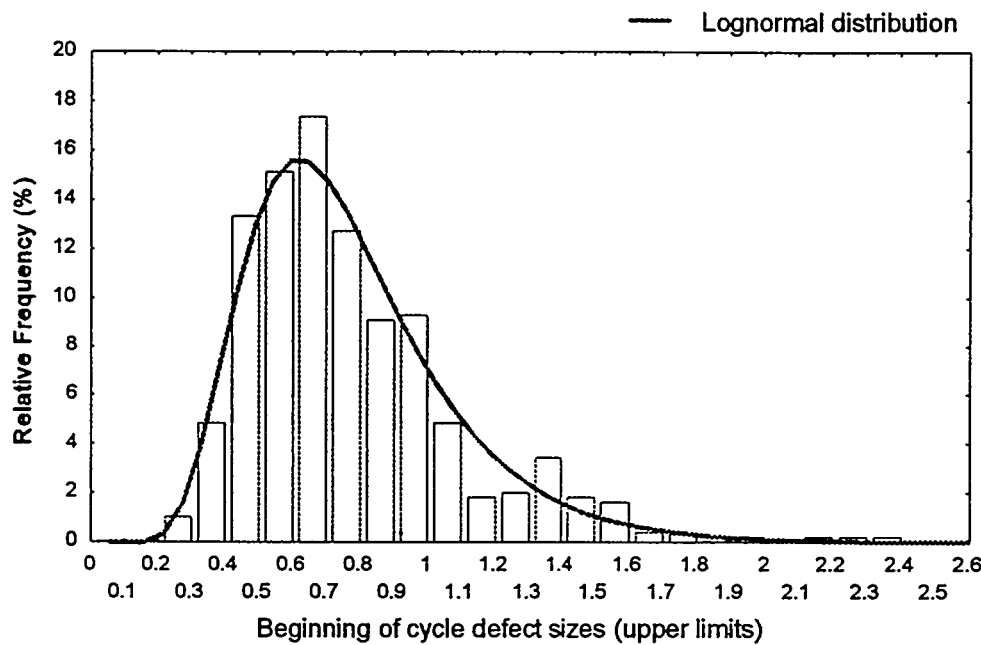


Figure 2 Distribution of the beginning of cycle defect sizes (in Volts)

upper limit. By setting the allowable leak rate value to 25 l/h, this essentially suggests failure probability of 5%.

The probabilities of exceeding the maximum leak rate (set to 25 l/h) are given in **Figure 3**. EPRI point estimate clearly coincides very well with the direct Monte Carlo (DMC) simulation results. SORM slightly underestimates the DMC results, while FORM tends to give a rather serious

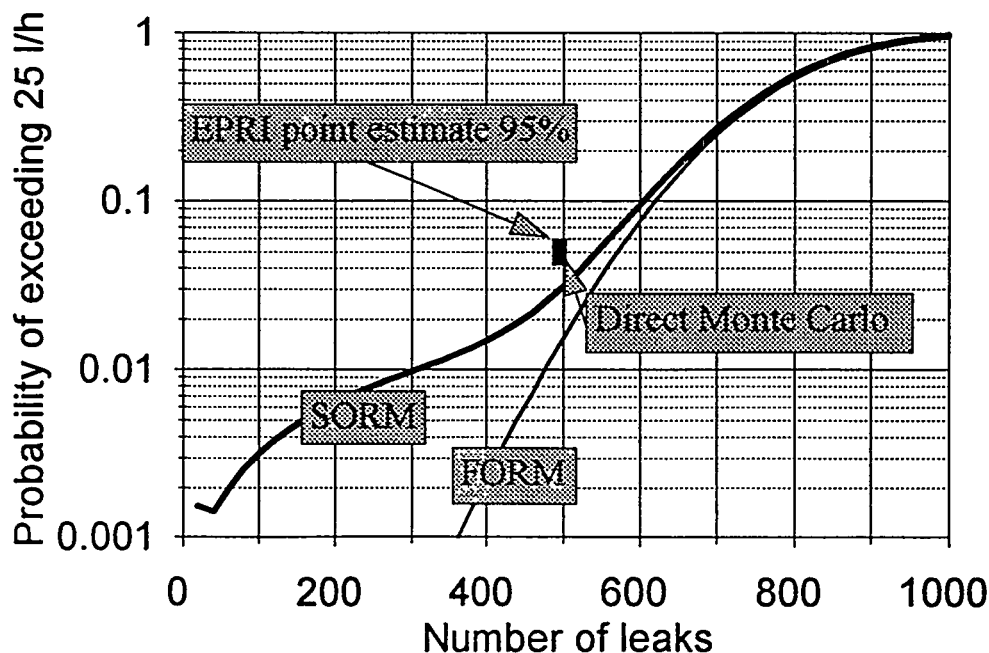


Figure 3 Comparison of different results

underestimation. At the 95% reliability level, targeted by the EPRI procedure, both methods give the exact order of magnitude of the failure probability.

Another interesting result is that although the EPRI point estimate is intended to be conservative, direct Monte Carlo gives virtually the same value of the failure probability. The main difference between the procedures is of course in the fact that Monte Carlo simulation does not accumulate the conservative estimates from different steps of computation.

A probable explanation of this paradox is given in Table 1. Simply, it can be expected that the uncertain regression model scatters the results (very large maxima as compared to the average or median values). Thus, single large leak values dominate the sum of all leak rates, which was verified by preliminary analyses of (simulated) data.

A good correlation that exists between the maximal individual leak and the sum of all leaks within a tube bundle suggests future use of extreme value statistics. In addition, some improvements in the leak rate modeling may be crucial for the success of probabilistic studies.

5 CONCLUSIONS

The problem of predicting the probability that a random total leak rate that is a sum of random individual leak rates exceeds allowable value is addressed in this paper. A closed form solution for the case of exponentially distributed individual leak rates is given. Some numerical solutions are developed, based on First and Second Order Reliability Methods. The results obtained were compared with the exact ones, showing acceptable performance of SORM.

A numerical example with data from Krško NPP in Slovenia was analyzed. It is shown that the intentioned built in conservativity of the deterministic approach is not exactly reached. This is caused by rather large uncertainty of the leak rate model, which is based on a regression analysis. Basically, use of extreme value statistics is suggested besides the improvements of the leak rate model.

The main topic for the future work is to extend the numerical solutions to an arbitrary distribution of individual leak rates. This may involve use of extreme value statistics and some additional work in numerical methods like FORM and SORM.

6 ACKNOWLEDGMENTS

This work was partially supported by the Ministry of science and technology of the Republic of Slovenia, Nuclear Power Plant Krško, Slovenia and International Office of FA Jülich, Germany.

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Asociación Nuclear Ascó, A.I.E.

INTERNATIONAL WORKSHOP ON STEAM GENERATOR TUBE INTEGRITY

30 Oct - 2 Nov, 1995
ARGONNE, ILLINOIS, USA

TITLE: ASCO STEAM GENERATORS OPERATING EXPERIENCE. SAFETY CRITERIA FOR DEFECT MANAGEMENT AND EFFECTIVENESS OF PREVENTIVE MEASURES.

PRESENTED BY: EMILIO LÓPEZ TORIBIO (ASOCIACION NUCLEAR ASCÓ A.I.E.)

ABSTRACT

ASCO NPP is a two W-PWR 930 Mwe Units. Each Unit is provided with three Westinghouse Model D3 steam generators (Fig.1) which are of preheater type and Inconel 600 MA as tube material. The Secondary side was designed and erected with copper alloys. Unit I: 81.072 EFPH (11 th R.O. July 1995), and Unit II: 69.720 EFPH (9 th R.O. April 95).

The results of the Eddy Currents Inspections performed during the first refueling outage showed Denting at tube support plates and PWSCC at roll transition zone in Unit I and Denting in Unit II. Later inspections showed other types of damages, such as (Fig. 2):

- ODSCC at tube support plates intersections.
- Circunferential cracks OD and ID at roll transition zone.
- Wear at antivibration bars and preheater baffles level.

Consequently, in order to limit the plugging rate, A.N. ASCO decided to license new plugging criteria in addition to the 40% depth criterion included in Technical Specification (Tec. Spec). The licensing criteria and surveillance requirements according to tube zone, are the following (see Fig. 3):

- Full length of tube (Technical Specifications)



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Tubes with indications 18 mm above the top of tubesheet must be repaired or plugged when the indications are greater than 40% of the wall thickness.

Surveillance requirements: All tubes in their full length must be tested with eddy current bobbin coil (BC) in each refueling outage.

- Inside of tubesheet.

The P^* criterion does not allow any tube with circumferential indications to remain in-service if the indications are located at a distance less than P^* -value. This criterion is based on the fact that tube motion in the vertical direction is restricted by an surrounding nondefective tube. The P^* value varies from 38 mm for the majority of tubes to the whole length of tubesheet for peripheral tubes.

Surveillance requirements: All tubes of the hot leg must be tested in the P^* value with motor rotating pancake coil (MRPC) eddy current probe in each refueling outage.

- Up to 18 mm above the top of tubesheet.

This criterion is based on the Leak Before Risk of Break approach, which assumes that accurate leak rate measurement during normal operation will be able to detect crack growth before the crack reaches the critical length. The largest allowable crack length is 8 mm, which is based on a critical crack length of 13 mm minus the average crack growth of 4 mm per fuel cycle and a measurement uncertainty of 1 mm.

- . All tubes with indications equal or longer than 8 mm must be repaired or plugged.
- . The maximum allowed number of parallel axial cracks in the same tube is 20.
- . Circumferential cracks are not allowed.

Surveillance requirements: All tubes of the hot leg must be tested up to 18 mm above of top of tubesheet with MRPC eddy current probe.

Limit Flow Rate.



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Licensing of this criterion involves a Limiting Condition for Operation of a maximum primary to secondary leak rate per SG limited to 5 l/h above the steady leak rate value at the beginning of cycle also limited to less than 5 l/h (78 l/h was the former leak rate Technical Specifications limit).

Surveillance requirements: The primary to secondary leak rate must be surveilled by Nitrogen 16 detectors.

The Inspection Technics were also improved.

Simultaneously, the following remedial actions to avoid the corrosion problems were implemented in both Units:

- Tubesheet Sludge Lancing in all outages.
- Boric Acid addition for denting prevention.
- Shotpeening in roll transition zone (Hot Leg).
- U-Bends Heat Treatment in two first rows.
- SG Copper Chemical Cleaning.
- Secondary Side Copper removal.
- New Boric Acid Addition procedure for ODSCC.

As a consequence of the copper removal, the secondary side chemistry was improved to high pH (10) with high Hydrazine (> 250 ppb).

In spite of all efforts performed, the ODSCC growing process in support plates has not been stopped.

Due to the progressive degradation of SG's tubes experienced, a Mathematical Prediction Model developed by PWR Spanish Owners Group, predicted that the limit of 18% of plugged tubes could be exceeded in 1.995. So A.N. Ascó decided the SG's replacement in 1.990.

The ASCO I SG's have been replaced in July 1995 and the ASCO II SG's will be replaced in July 1996.



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CONCLUSIONS

- Results of Eddy Current Inspections performed during refueling outages showed:
 - . Denting at tube support plates.
 - . Axial (PWSCC) and circumferential cracks in roll transition zone.
 - . ODSCC at tube support plates intersections.
 - . Wear at antivibration bars and preheater baffles.
- To limit plugging rate, ASCO licensed new criteria in addition to the 40% depth criterion (Tec. Spec.), such as: P* and the criterion based on LBRB, both to be applied in roll transition zone.
- In spite of the applied remedies, all defects have kept on its growing process, mainly the ODSCC in support plates area.
- The evolution of plugged tubes showed the need to replace the SG's in 1.990.
- The ASCO I SG's have been replaced in July 1995 and ASCO II SG's will be replaced in July 1996.

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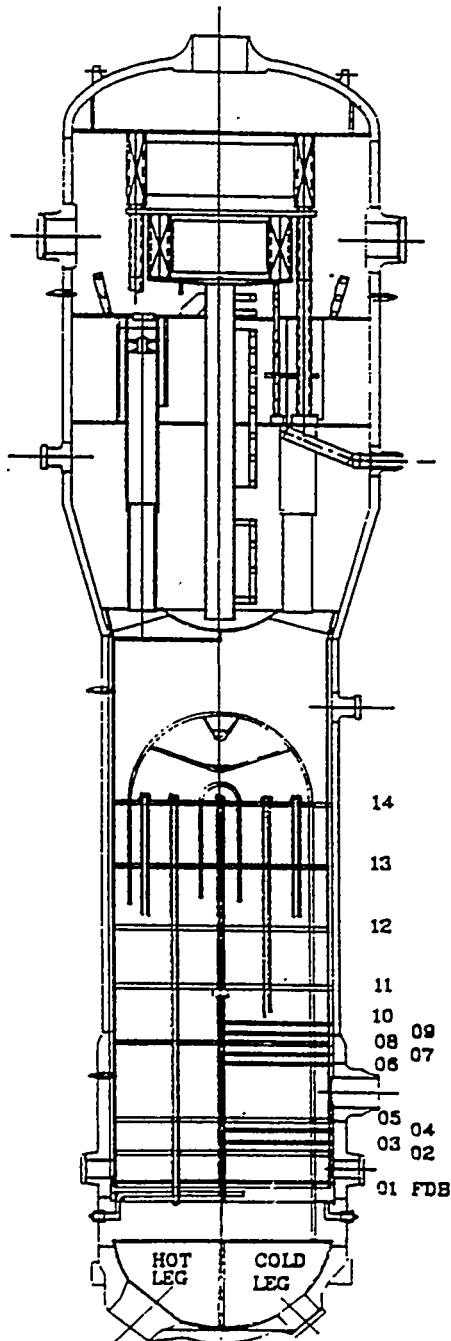
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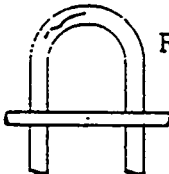
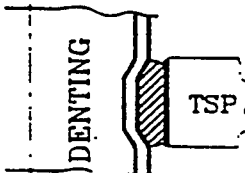
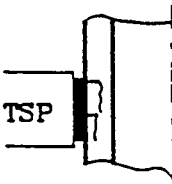
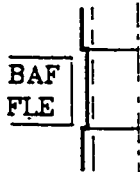
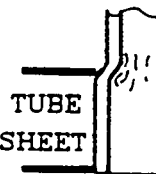


- FIG.1 -

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STEAM GENERATOR WESTINGHOUSE TYPE MODEL : D3 PLUGGING CAUSES AND REMEDIES APPLIED IN ASCO N.P.P.



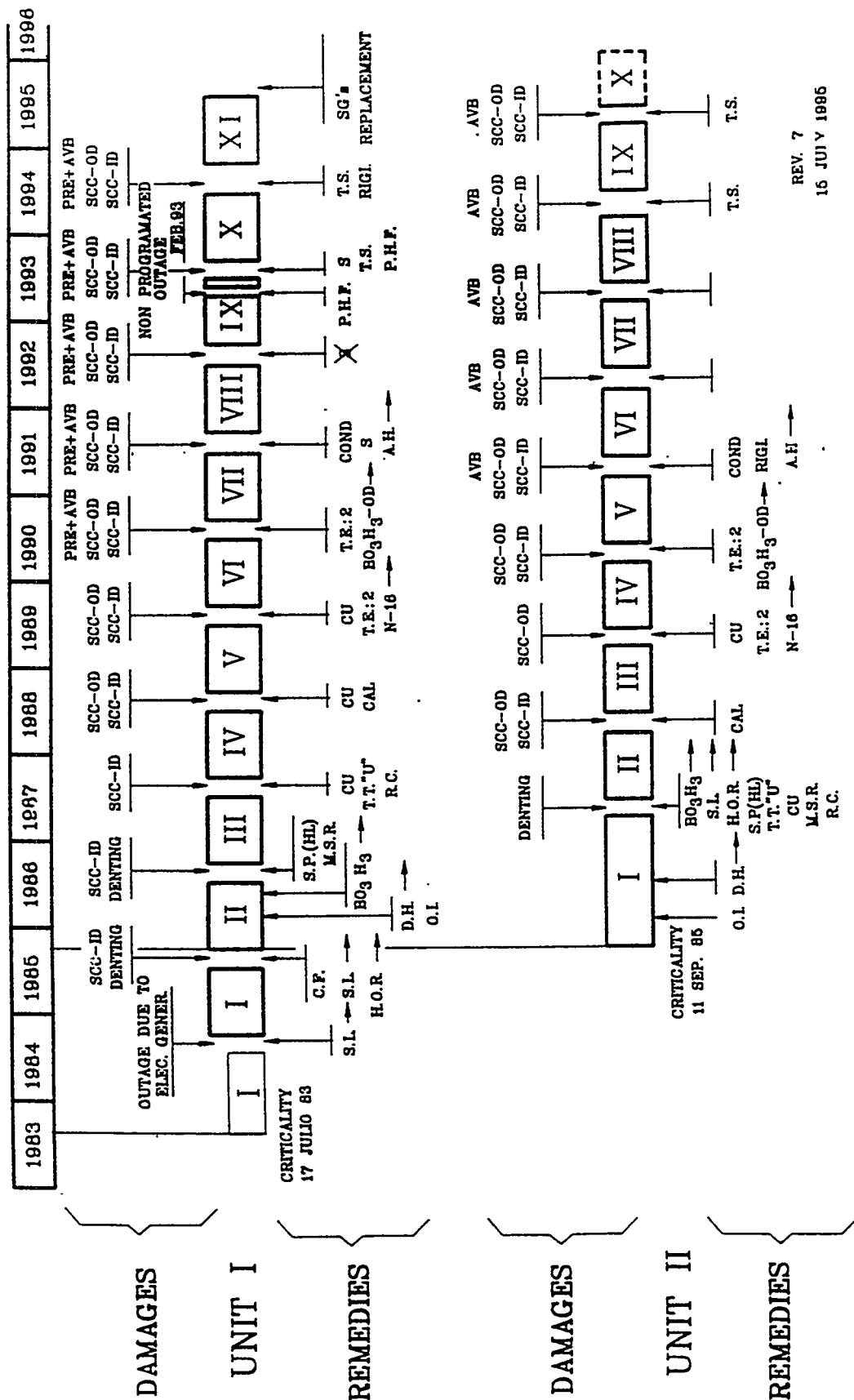
TYPE OF DAMAGE	REMEDIES APPLIED
 TUBE/AVB WEAR	PLUGGING
 ROWS 1&2 CRACKING	THERMAL TREATMENT
 DENTING TSP	BORIC ACID ADDITION WATER CHEMISTRY IMPROVEMENT COPPER REMOVAL
 TSP ODSCC	BORIC ACID ADDITION NEW PROCEDURE SLEEVEING PLUGGING
 BAF FLE TUBE WEAR AT PREHEATER BAFFLES LEVEL	PLUGGING
 TUBE SHEET PWS AT ROLLING T Z	LBBR CRITERION TO LIMIT THE PLUGGING RATE SLEEVEING PLUGGING SHOTPEENING



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- FIG. 2 -

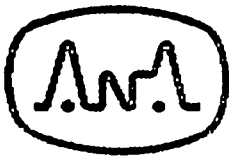
MAIN DAMAGES AND MITIGATION REMEDIES IMPLEMENTED IN SG'S ASCO NPP



REV. 7

15 JULY 1996

CC1/JC-9509



ABBREVIATIONS

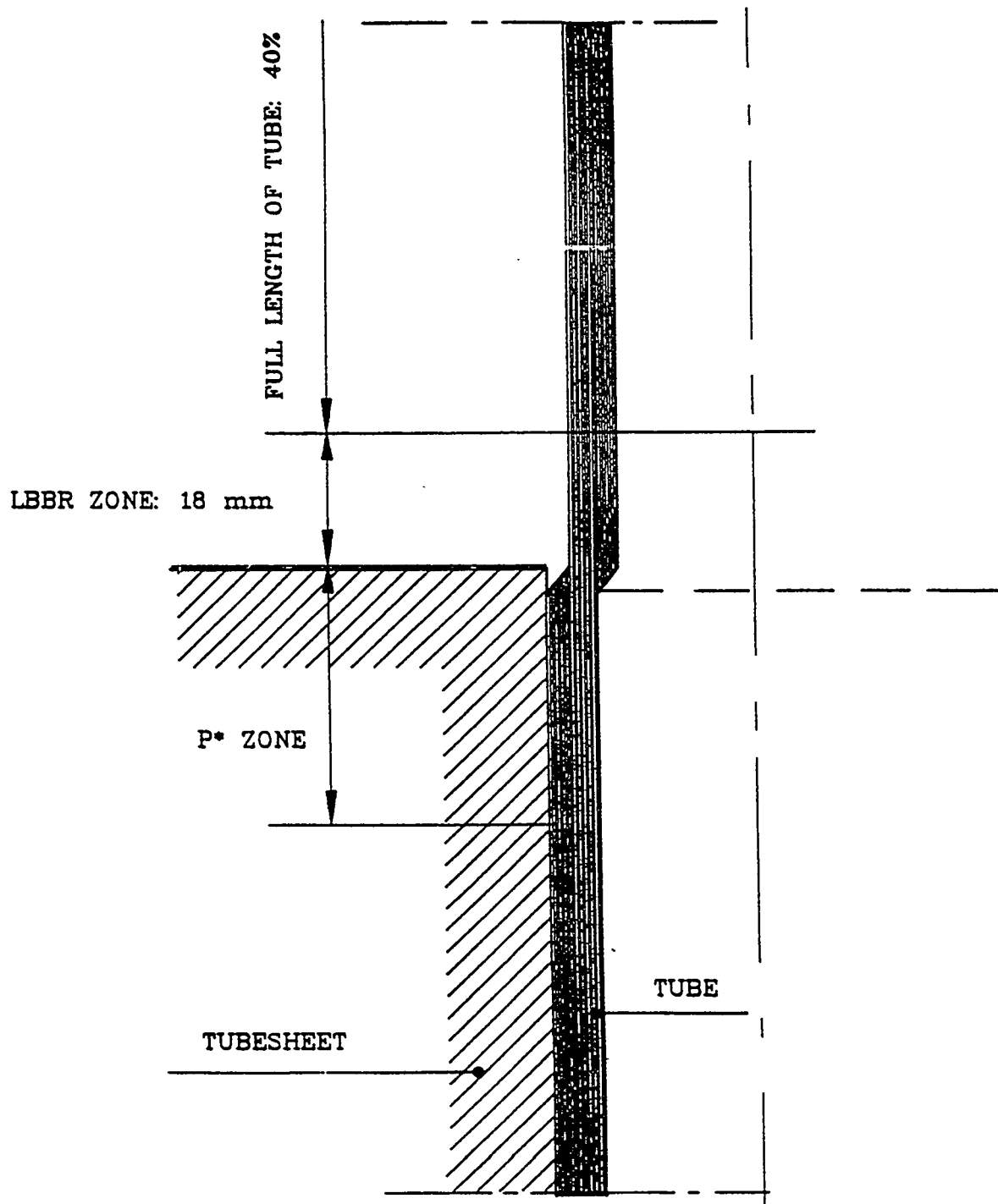
BORIC ACID ADDITION - DENTING	BO_3H_3
BORIC ACID ADDITION - ODSCC	$\text{BO}_3\text{H}_3\text{-OD}$
SHOTPEENING (HOT LEG)	S.P. (H.L.)
U-BENDS HEAT TREATMENT. ROWS 1 & 2	T.T.
COPPER CHEMICAL CLEANING	CU
SLUDGE LANCING	S.L.
COPPER REMOVAL MSR (INOX)	M.S.R.
HEATERS REPLACEMENT (INOX)	CAL
CONDENSER REPLACEMENT(TITANIUM)	COND
REVERSE OSMOSIS	O.I.
SLEEVING	S.
HIDE-OUT RETURN	H.O.R.
CREVICE FLUSHING	C.F.
NITROGEN 16	N-16
HIDROGEN MONITORING	D.H.
CONDENSATE RECIRCULATION	R.C.
PULL TUBES	T.E.
HIGH HIDRACINE (>250 PPB)	A.H.
STABILIZERS	RIGI
WESTINGHOUSE SUSCEPTIBLE PLUGS REPLACEMENT	T.S.
HYDRAULIC TEST WITH FLUORESCINE	P.H.F.



Asociación Nuclear Asda, ALE

- FIG. 3 -

APPLICATION ZONES OF PLUGGING CRITERIA IN SG's TUBES



QCI/JC-8806

THE P* VALUE VARIES FROM 38 mm FOR THE MAJORITY OF TUBES
TO THE WHOLE LENGTH OF TUBESHEET FOR PERIPHERIC TUBES

CNRA/CSNI Meeting Handout

Health and Safety Impact of Steam Generator Tube Degradation

by

Theodore Marston, Ph.D.
Vice President,
PLG, Inc.
Newport Beach, CA

**CNRA/CSNI International Workshop
on Steam Generator Tube Integrity - Session 5
30 Oct - 2 Nov 1995 Argonne, IL USA**



CSNI/CNRA
Meeting

Presentation Overview

-
- Role of the Steam Generator (SG) in the PSA
 - PSA chronology
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 - SGTR as an initiating event
 - SGTR as a conditional event
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 - What is needed for the PSA to handle SGTR?
 - SG safety conclusions



Slide 2

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Steam Generators in the PSA

- Normal operation
- Upset conditions
- Accident conditions
- SG related issues on risk
 - small contribution to CDF (~5%)
 - large contribution to LER (>50%)
 - severe accident contribution
- Degradation mechanisms

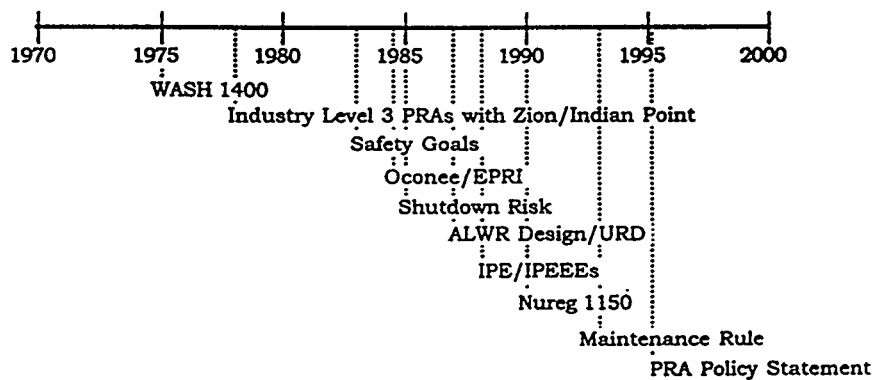
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PRA History in US



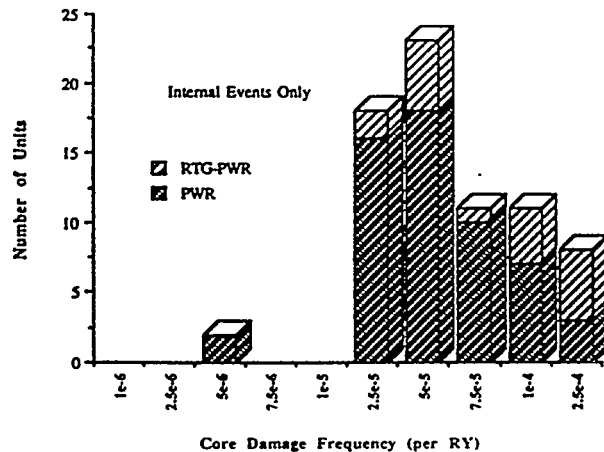
Slide 4



CNRA/CSNI Meeting Handout

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Meeting **Not All Plants and Not All PSAs are Created Equal**



Slide 5



CSNI/CNRA

Meeting **Goals of SG Management**

- Minimize the likelihood of a tube leak during the operating cycle from reaching the tech spec leakage limit.
- Minimize the likelihood of a tube rupture during the operating cycle
- Assure that the contribution of a SGTR to the overall safety of the plant and health of the public is not compromised
- Maximize the resource-effectiveness of the program

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Steam Generator Tube Rupture

- Historical perspective of a SGTR
- Single vs multiple tube rupture events
 - multiple tube ruptures not observed (to date)
 - assume that the $P(MT/ST) \sim 0.1$, i.e. one in every ten SGTR will involve >1 tube
 - while the leakage is much greater and the operator response time is shorter, the chance of a wrong diagnosis or selection of the wrong SG is very much reduced, therefore the impact on CD is less for multiple tubes
- Inspection and maintenance programs can help with most damage forms, save foreign object damage and fatigue

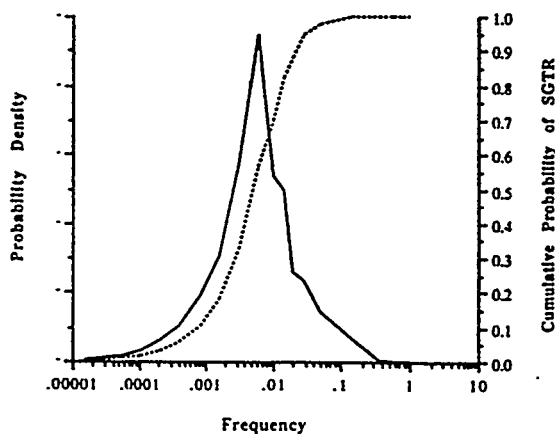


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SGTR as an Initiating Event



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SGTR as a Conditional Event

- Case where the tube rupture is a collateral event
- The probability of SGTR in this case is based upon the condition of another event, i.e. an event that would increase the pressure differential across the tube
- Two primary cases
 - Main steamline break (MSLB), where the secondary pressure drops
 - Anticipated transient without scram (ATWS), where the primary pressure increases, maximum pressure function of time into cycle
- LOCAs were determined not to be a realistic contributor to SGTRs

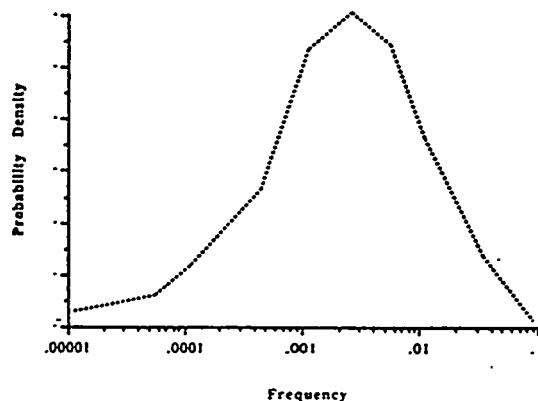
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Main Steamline Break Outside Containment



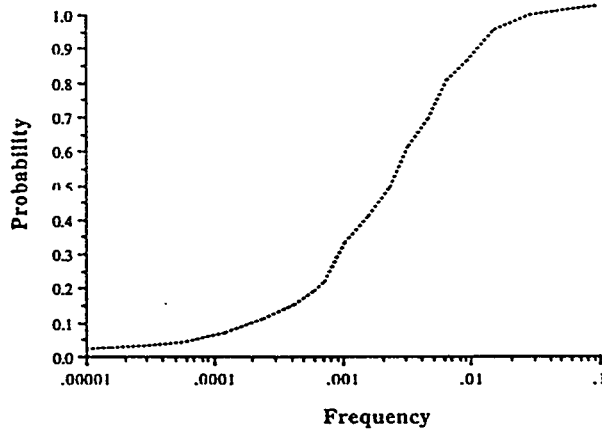
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Main Steamline Break Outside Containment



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Anticipated Transient Without Scram

- Identify those initiating events (i.e., those that challenge the reactor protection system) with the highest frequency
- Develop a plant specific model to determine the likelihood of core damage due to an ATWS
- This retains any dependencies between initiating events and the probability of RPS and/or mitigating systems failures, i.e. DC bus failure
- Others sum the probabilities of the initiating events and multiple that by the likelihood of RPS failure, ignoring the dependencies
- $10^{-6}/RY \gg CD[ATWS] > 10^{-7}/RY$

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Meeting **Severe Accident Induced SGTR**

-
- This is an accident that is significantly beyond design basis
 - In the very unlikely event of a core melt event, some scenarios have the primary circuit intact with the temperature and pressure increasing
 - If not mitigated, the primary circuit finally fails, due to either:
 - primary piping creep rupture,
 - reactor vessel melt through, or
 - SGTR creep failure
 - SGTR is the most undesirable due to containment by-pass

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SG Safety Conclusions

-
- Real safety is most likely better than apparent safety
 - Use best estimate with uncertainties vs. bounding (licensing) calculations
 - Look beyond the SG degradation to assess safety, i.e. what independent and dependent events exacerbate the health and safety effects of SGTR
 - it may be more cost effective to deal with other actions, systems, structures and components
 - make sure we don't violate the time-proven 80/20 rule
 - Always keep in mind that it is the Safety Goals, not the CDF or the LERF, that determine when 'safe is safe enough'

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NRC Safety Goals

- Quantitative Health Objectives:
 - Prompt fatality goal = 0.5×10^{-6} /year for a person living within one mile of plant (0.1% of normal accident risk)
 - Latent fatality goal = 2.0×10^{-6} /year for a person living within 10 miles of plant (0.1% of normal cancer risks)
- Guidelines (not an objective)
 - $LRG \leq 10^{-6}$ /reactor year
 - $CDF \leq 10^{-4}$ /reactor year

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What is Needed for the PSA

- In order to integrate properly SG degradation into the PSA model, the risk analyst needs the following:
 - Probability of SGTR as a function of plant operating time to deal with the initiating events
 - Probability of SGTR as a function of plant operating time and pressure and temperature to deal with the collateral events
 - Creep failure conditions for degraded tubes to deal with the severe accident induced SGTRs
- Some PSA models must be enhanced to accommodate SG degradation
- Level 3 calculations are necessary to assess dose and health consequences

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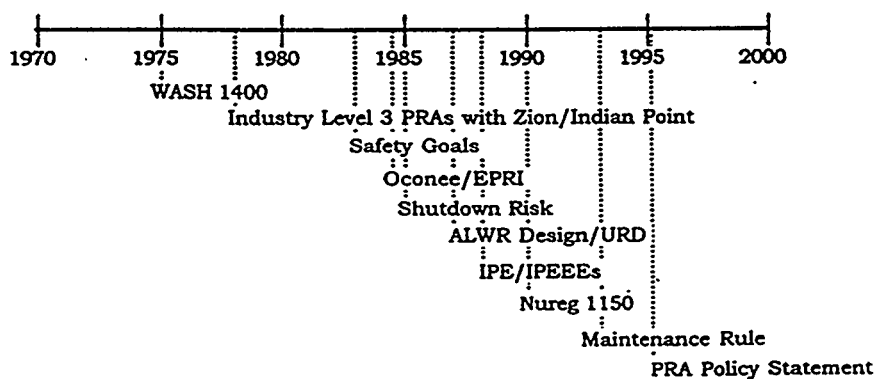


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PRA History in US

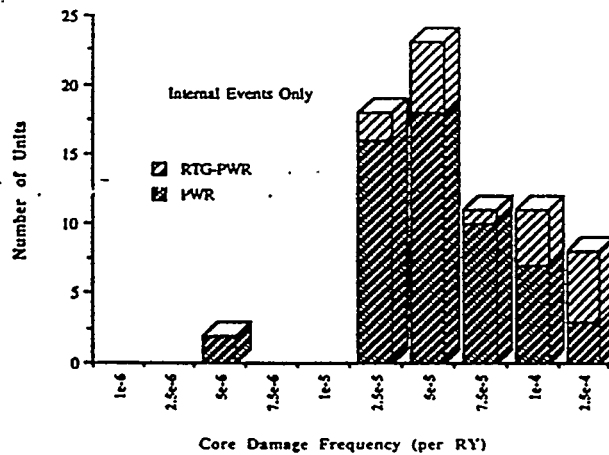


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- Minimize the likelihood of a tube leak during the operating cycle from reaching the tech spec leakage limit.
- Minimize the likelihood of a tube rupture during the operating cycle
- Assure that the contribution of a SGTR to the overall safety of the plant and health of the public is not compromised
- Maximize the resource-effectiveness of the program

Slide 6



CNRA/CSNI Meeting Handout

CSNI/CNRA

Meeting

Steam Generator Tube Rupture

- Historical perspective of a SGTR
- Single vs multiple tube rupture events
 - multiple tube ruptures not observed (to date)
 - assume that the $P(MT/ST) \sim 0.1$, i.e. one in every ten SGTR will involve >1 tube
 - while the leakage is much greater and the operator response time is shorter, the chance of a wrong diagnosis or selection of the wrong SG is very much reduced, therefore the impact on CD is less for multiple tubes
- Inspection and maintenance programs can help with most damage forms, save foreign object damage and fatigue

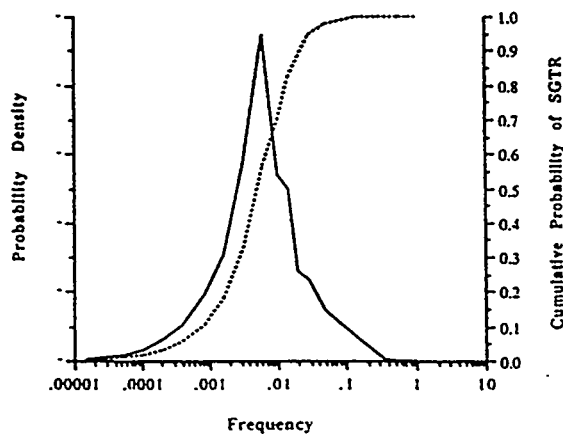


Slide 7

CSNI/CNRA

Meeting

SGTR as an Initiating Event



Slide 8

CNRA/CSNI Meeting Handout

CSNI/CNRA

Meeting **SGTR as a Conditional Event**

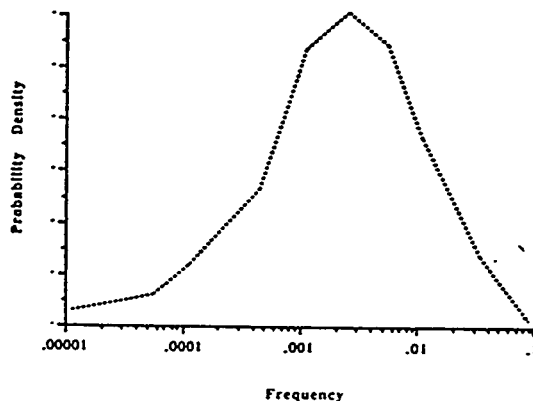
- Case where the tube rupture is a collateral event
- The probability of SGTR in this case is based upon the condition of another event, i.e. an event that would increase the pressure differential across the tube
- Two primary cases
 - Main steamline break (MSLB), where the secondary pressure drops
 - Anticipated transient without scram (ATWS), where the primary pressure increases, maximum pressure function of time into cycle
- LOCAs were determined not to be a realistic contributor to SGTRs

Slide 9

PLG

CSNI/CNRA
Meeting

Main Steamline Break Outside Containment



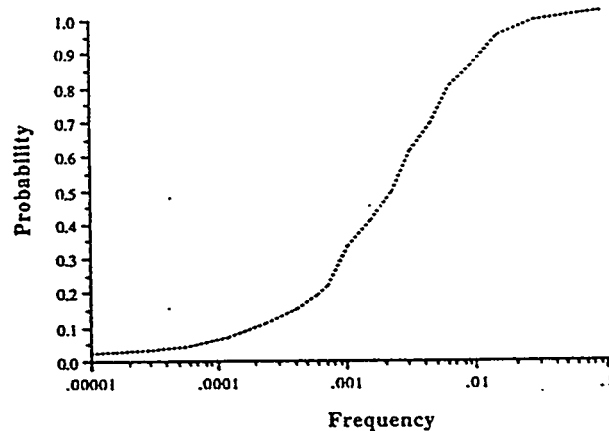
Slide 10

PLG

CNRA/CSNI Meeting Handout

CSNI/CNRA
Meeting

Main Steamline Break Outside Containment



Slide 11



CSNI/CNRA
Meeting

Anticipated Transient Without Scram

- Identify those initiating events (i.e., those that challenge the reactor protection system) with the highest frequency
- Develop a plant specific model to determine the likelihood of core damage due to an ATWS
- This retains any dependencies between initiating events and the probability of RPS and/or mitigating systems failures, i.e. DC bus failure
- Others sum the probabilities of the initiating events and multiple that by the likelihood of RPS failure, ignoring the dependencies
- $10^{-6}/RY \gg CD[ATWS] > 10^{-7}/RY$

Slide 12



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Meeting **Severe Accident Induced SGTR**

-
- This is an accident that is significantly beyond design basis
 - In the very unlikely event of a core melt event, some scenarios have the primary circuit intact with the temperature and pressure increasing
 - If not mitigated, the primary circuit finally fails, due to either:
 - primary piping creep rupture,
 - reactor vessel melt through, or
 - SGTR creep failure
 - SGTR is the most undesirable due to containment by-pass

Slide 13



CSNI/CNRA

Meeting **SG Safety Conclusions**

-
- Real safety is most likely better than apparent safety
 - Use best estimate with uncertainties vs. bounding (licensing) calculations
 - Look beyond the SG degradation to assess safety, i.e. what independent and dependent events exacerbate the health and safety effects of SGTR
 - it may be more cost effective to deal with other actions, systems, structures and components
 - make sure we don't violate the time-proven 80/20 rule
 - Always keep in mind that it is the Safety Goals, not the CDF or the LERF, that determine when 'safe is safe enough'

Slide 14



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NRC Safety Goals

- Quantitative Health Objectives:
 - Prompt fatality goal = 0.5×10^{-6} /year for a person living within one mile of plant (0.1% of normal accident risk)
 - Latent fatality goal = 2.0×10^{-6} /year for a person living within 10 miles of plant (0.1% of normal cancer risks)
- Guidelines (not an objective)
 - $LRG \leq 10^{-6}$ /reactor year
 - $CDF \leq 10^{-4}$ /reactor year

Slide 15



CSNI/CNRA
Meeting

What is Needed for the PSA

- In order to integrate properly SG degradation into the PSA model, the risk analyst needs the following:
 - Probability of SGTR as a function of plant operating time to deal with the initiating events
 - Probability of SGTR as a function of plant operating time and pressure and temperature to deal with the collateral events
 - Creep failure conditions for degraded tubes to deal with the severe accident induced SGTRs
- Some PSA models must be enhanced to accommodate SG degradation
- Level 3 calculations are necessary to assess dose and health consequences

Slide 16



STAC - A new Swedish code for statistical analysis of cracks in SG-tubes

by

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presented at

NEA/CNRA/CSNI International Workshop
on
Steam Generator Tube Integrity in Nuclear Power Plants

ARGONNE, IL., USA
30 Oct - 2 Nov 1995

STAC - A new Swedish code for statistical analysis of cracks in SG-tubes

1 Introduction

Steam generator (SG) tubes in pressurized water reactor plants are exposed to various types of degradation processes, among which stress corrosion cracking in particular has been observed. To be able to evaluate the safety importance of such cracking of SG-tubes one has to have a good and empirically founded knowledge about the scope and the size of the cracks as well as the rate of their continuous growth. The basis of experience is to a large extent constituted of the annually performed SG-inspections and crack sizing procedures. On the basis of this experience one can estimate the distribution of existing crack lengths, and modify this distribution with regard to maintenance (plugging) and the predicted rate of crack propagation. Finally, one can calculate the rupture probability of SG-tubes as a function of a given critical crack length.

On account of the Swedish Nuclear Power Inspectorate an introductory study has been performed (Pörn, 1994) in order to get a survey of what has been done elsewhere in this field. The study resulted in a proposal of a computerizable model to be able to estimate the distribution of true cracks, to modify this distribution due to the crack growth and to compute the probability of tube rupture. The model has now been implemented in a computer code, called STAC (Statistical Analysis of Cracks), (Pörn, 1995). This paper is aimed to give a brief outline of the model to facilitate the understanding of the possibilities and limitations associated with the model.

In the model proposed one considers the fact that the detectability, the sizing accuracy and the crack growth are dependent on the actual crack size. Maintenance actions like plugging of tubes will have an influence on the crack length distribution. For a given critical crack length, which is assumed constant, the model can be used to estimate the probability of SG-tube rupture during an operation period. Further, one can use the model to evaluate existing or proposed maintenance rules and their impact on the tube rupture probability. Thereby, the primary objective of the model is safety oriented. The model will also enhance the possibility to systematically utilize the continuously growing amount of experiences.

The work on the model construction has been influenced by the French computer code COMPROMIS (Barnier et al, 1992 and Granger et al, 1992), developed by Electricité de France. A decisive difference between the two models is that Bayes' method is proposed here to solve the central problem of estimating the distribution of true crack lengths. This feature will allow a more complete treatment of various uncertainties, and consequently lead to a more robust estimation of the tube rupture probability. In the context of reliability data, the Bayesian approach is clearly advocated in (Pörn, 1990) and systematically applied in (T-Book, 1994).

2 Assumptions and model

The technique used to detect and size any single crack in the tubes is not perfect with regard to reliability and accuracy. This source of uncertainty must be considered in a model by which one tries to estimate the distribution of true cracks, the change of this distribution due to the crack propagation and to compute the probability of tube rupture. In contrast to deterministic models all uncertainty of this kind is treated probabilistically. Thus probability distributions are used to describe, for example, the detectability and the accuracy performance of the inspection technique as well as the crack growth during a certain time interval.

The model developed can be structured into three main modules:

- to estimate the distribution of crack lengths immediately after an inspection, partial or complete, and to estimate the influence of maintenance measures on this distribution
- to estimate the evolution of this distribution in time due to the crack growth
- to compute the probability of tube rupture on demand, based on a specified critical crack length

Before going to the basic assumptions let us introduce some notations:

A, a	true cracklength, stochastic variable & given value respectively
M, m	measured cracklength, stochastic variable & given value respectively
D	the event that an existing crack is detected in inspection
$p(a)$	distribution (density function) of true cracklengths
$p(m)$	distribution of measured cracklengths
$p_d(a)$	distribution of true cracklengths that are detected
$P(D a)$	probability that a crack of length a is detected
$v(a)$	crack growth as function of crack length

Among the more fundamental assumptions or judgements made in the construction of the model we want to mention the following:

- 1 **The true crack lengths are, presumptively, lognormally distributed .**
- 2 **The probability that a crack is detected in an inspection (detectability) is zero for cracklengths $< a_0$ and one for cracklengths $> a_1$, otherwise following a smooth (cubic) curve in between.**
- 3 **The accuracy of the crack length measurement is described by a normal distribution, the mean value and standard deviation of which determine the systematic and random errors respectively.**
- 4 **The growth or the propagation rate of a crack during a certain time follows a lognormal distribution the parameters of which are based on previous measurement records taking into account the accuracy of the measurement.**

The initiation of cracks or time to cracking is not treated at all in the model. Thus it cannot be used for a long term prediction of the SG-life time. Instead, it is intended to be used after each inspection to update the distribution of real crack lengths and to predict the short term behaviour only.

3 **The basic problem**

As mentioned above, presumptive distributions of crack lengths are lognormal, $LN(a|\alpha, \beta)$, where the parameters α and β denote the mean value and the standard deviation respectively of the logarithms of crack lengths. This choice of distribution type can, to some extent, be justified with theoretical arguments (Mann et al, 1974). If we briefly put $\theta = (\alpha, \beta)$ and reminding of the notations above we have, according to Bayes' theorem,

$$p(\theta|m) \sim p(\theta) p(m|\theta), \quad (1)$$

where $p(\theta)$ stands for the apriori distribution of θ and $p(m|\theta)$ is the likelihood function, based on the real measurements m . This likelihood function is obtained as the product of the likelihoods of individual observations

$$p(m|\theta) = \prod_j p(m_j|\theta) \quad (2)$$

where, disregarding the index j

$$p(m|\theta) = \int p(m|a) P(D|a) p(a|\theta) da / \int P(D|a) p(a|\theta) da \quad (3)$$

Here $p(a|\theta)$ denotes the distribution of real crack lengths when the parameters θ are known, i.e. the lognormal distribution $LN(a|\alpha, \beta)$ mentioned above. Similarly, $p(m|a)$ denotes the distribution of the measured crack length when the real length is a .

As we see above the solution of the inferential problem requires the specification of a prior distribution $p(\theta)$, $\theta=(\alpha, \beta)$, before we have obtained any observations at all. For the apriori distribution $p(\theta)$ we use the non-informative distribution $p(\alpha, \beta) \sim 1/\beta$, derived according to the principle of "data translated likelihood" (Box & Tiao, 1973)

Thus we are able to calculate the posterior distribution $p(\theta|m)$ according to (1)-(3). Then, by the Law of total probability, we convert from the space of parameter θ to the space of true crack lengths through the relation

$$p(a|m) = \int p(a|\theta) p(\theta|m) d\theta \quad (4)$$

The equations (1)-(4) describe the procedure through which we are able to estimate the distribution of true crack lengths on the basis of measured values and taking into account the uncertainty in detection and measurement.

For the detectability $P(D|a)$ we apply Assumption 2. Similarly, Assumption 3 of normal distribution is used for the measurement accuracy, which means that $p(m|a)$ is written as

$$p(m|a) = N(m; E(M|a), \sigma(a)) \quad (5)$$

where the difference between the sample mean value $E(M|a)$ and the true crack length a , $E(M|a) - a$, is the systematic error, while the standard deviation $\sigma(a)$ is a measure of the random error. Obviously, both of these types of error may be dependent on the real crack length a . However, in the STAC code we make the simplifying assumption that this dependence is linear in case of $E(M|a)$, i.e.

$$E(M|a) = c_1 \cdot a + c_0 \quad (6)$$

and that the standard deviation is constant $\sigma(a) = \sigma$.

4 Distribution after plugging

It is easily seen that the distribution of true crack lengths, $p(a|m)$, is influenced by maintenance actions like plugging, whereby strongly degraded tubes are taken out of operation. Let a_p be the plugging criterion, i.e. tubes having measured crack lengths $m > a_p$ are plugged. The probability that a tube with a given crack length will be plugged is a function of several other probabilities: probability of tube being inspected, probability of crack being detected, probability that measured length exceeds a_p and probability that such a tube is really plugged. Thus the searched probability is a product of the following conditional probabilities:

$$P(\text{plugging a tube } |a) = P(\text{inspected \& detected \& } m > a_p \text{ \& plugging conducted} |a). \quad (7)$$

where

$$P(\text{inspection}|a) = 1/K. \quad (\text{Sampling } 1/K)$$

$$P(\text{detection}|a) = P(D|a)$$

$$P(m > a_p | a) = \int_{a_p}^{\infty} p(m|a) dm$$

$$P(\text{plugging is conducted} | a, m > a_p) = 1 - z$$

The distribution, $p_P(a)$, of crack lengths remaining after inspection (I) and plugging (P) is

$$p_P(a) = p(a|m) \cdot [1 - P(\text{tube plugged}|a)] / \int p(a|m) \cdot [1 - P(\text{tube plugged}|a)] da \quad (8)$$

5 Distribution after crack propagation

The distribution above may also be denoted with $p_P(a;t)$, indicating that the distribution describes true crack lengths at time t immediately after inspection and plugging. During the next operation period Δt a crack of length a will grow with an amount $v(a)$. This crack propagation will vary from crack to crack according to a distribution $p_v(v|a)$, which is assumed to depend on the original crack length. Because each individual crack at time $t + \Delta t$ has the length $a + v(a)$, the distribution $p(a;t + \Delta t)$ is obtained by convolution:

$$p(a;t + \Delta t) = \int_0^a p_P(u;t) \cdot p_v(a-u|u) du \quad (9)$$

According to Assumption 4 is the crack growth lognormally distributed

$$p_v(a-u|u) = \text{LN}(\cdot | \mu, v) \quad (10)$$

with parameters μ and v depending on the original crack length u . The estimate of these parameters is based on previous measurement records. Again, the problem of this approach is associated with the measurement uncertainty. The severity of this problem is easily understood if we consider the fact that the crack growth during a cycle of operation might be of the same magnitude as the measurement error.

Let us assume a crack that has the length a_1 och a_2 at the beginning and at the end of the operation period. The corresponding measured lengths are m_1 and m_2 respectively. The records of crack propagation rate is usually presented in the form of mean value $E(m_2 - m_1 | m_1)$ and variance $V(m_2 - m_1 | m_1)$ for each millimeter of original, measured crack length m_1 . Based on the distribution of measurement error, (5)-(6), it is possible to derive a relation between these characteristics and the corresponding characteristics of the true crack propagation, $E(a_2 - a_1 | m_1)$ and $V(a_2 - a_1 | m_1)$. The derivation (Pörn, 1995) results in

$$E(a_2 - a_1 | m_1) = 1/c_1 \cdot E(m_2 - m_1 | m_1)$$

och (11)

$$V(a_2 - a_1 | m_1) = [(1/c_1)^2 + (1 - 1/c_1)^2] \cdot V(m_2 - m_1 | m_1) + 2 \cdot (\sigma/c_1)^2$$

Thus we have obtained point estimates of $E(a_2 - a_1 | m_1)$ and $V(a_2 - a_1 | m_1)$ based on the corresponding observed quantities $E(m_2 - m_1 | m_1)$ and $V(m_2 - m_1 | m_1)$ respectively. These point estimates are then transformed into the parameters μ and ν of the crack propagation distribution in (10).

6 Probability of tube rupture

Calculating the probability of tube rupture and of a complete SG failure we make following assumptions, similar to those of (Mavko and Cizelj, 1991).

The behaviour of the observed crack does not depend on other cracks in the same tube or in other tubes. The failure of a tube is defined as the event that the maximum crack length in the tube exceeds the critical crack length. A single-tube failure causes a complete SG failure.

Using these assumptions the failure probability of a SG tube bundle containing N cracks at time $t + \Delta t$ is defined by

$$Q = 1 - \exp(-N \cdot Q_1) \quad (12)$$

where Q_1 denotes the probability that a single crack leads to tube rupture. Assuming a constant critical crack length a_c we can write

$$Q_1 = \int_{a_c}^{\infty} p(a; t + \Delta t) da \quad (13)$$

7 Application

To illustrate the model presented here for the statistical analysis of SG tube cracks an application is presented, based on crack lengths observed at a Swedish nuclear power plant during the annual revision period in 1994. The observed data are given in Table 1. The application is focussed on SG1 that seems to have the longest cracks among the three SGs. The crack propagation rates are estimated on the basis of average measurement records, displayed in Figure 1, from all three SGs.

The estimated probability density function (pdf) of true crack lengths is shown in

Figure 2, where this function can be compared to the relative frequencies of observed lengths. The substantial difference between pdf and frequencies is mainly caused by the systematic measurement error described by eqn.(5) and (6). The hypothetical assumption of no systematic errors at all results in a pdf shown in Figure 5. Further, the presumptive lognormal distributions do not seem to fit so well for short cracks. The set of relatively many cracks of length 1 mm might be composed of two subsets, of which one represents stable cracks that don't propagate at all or very little. Leaving all cracks of 1 mm out of consideration results in a pdf shown in Figure 6.

The influence of plugging with a fix plugging limit of 9 mm is shown in Figure 3. The area of influence on the pdf is rather wide, which is a consequence of the great measurement uncertainty, in particular for long cracks. Figure 4 displays the change of crack length pdf due to crack propagation during one year. Here it is to be noted that initiating cracks are not treated in the model.

8 Some conclusions

In the Bayesian methodology applied in STAC the choice of a specific lognormal distribution for crack lengths is considered uncertain, and this uncertainty is described by a probability distribution. This approach means, in the particular case, that all crack length distributions in the selected family are treated as candidates that are assigned probability weights according to their goodness of fit to existing observations. Therefore, we think we have achieved greater robustness of the final distribution compared to the results of traditional methods like maximum likelihood.

However, the application presented in this report seems to indicate that the frequency function of observed crack lengths is bimodal where very short crack lengths constitute one mode. Thus the family of lognormal distributions is not appropriate for the whole range of observed crack lengths, while it seems to be quite adequate for cracks longer than one millimeter. The question of which distribution family should be selected has to be investigated further.

The STAC code is expected to become an important tool for the statistical follow-up and analysis of observed crack lengths in SG-tubes. It enhances the possibility to study the influence of various parameters like detectability, measurement uncertainty and different maintenance criteria, and to provide an improved support for decision-making. Coming applications of the code will certainly disclose needs for further improvement.

Acknowledgement

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LIJFALL AB Kungälv

Angeneratorstatistik

RINGHALS 3 SG 1-3 RA-94 TOTAL NUMBER OF CRACKS AS FUNCTION OF CRACKLENGTH AND SG.

Tot LN	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	>=20
SG1	468	198	933	2105	1585	676	258	95	44	19	17	4	4	4	4	4	4	4	1	2
SG2	355	155	671	1246	805	365	141	39	14	3	4	4	4	4	4	4	4	4	1	2
SG3	265	162	699	1443	1207	624	206	68	29	14	7	2	2	2	2	2	2	2	1	2
Tot No	1088	515	2303	4794	3597	1665	605	202	87	36	28	6	4	4	4	4	4	4	1	2

Total No. of cracks:	14944
Total No. of Tubes excl. those with only Circ cracks:	4327
Average Crack Length:	4.27
Average Number of Cracks per Tube:	3.45

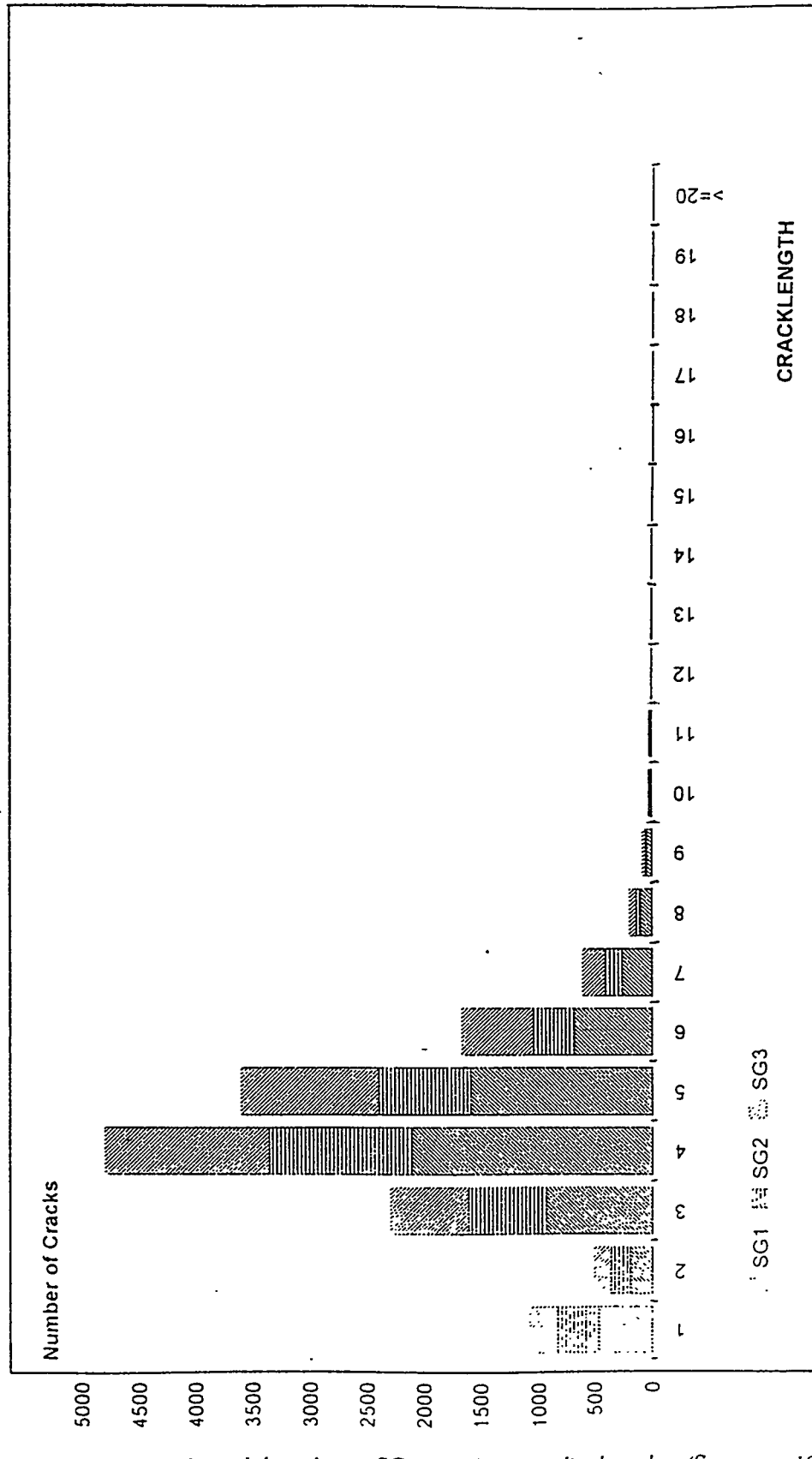


Fig. 1.1. Observed crack length in SG 1 as they are displayed in (Svensson 1991)

PCRep95-8

(Allman data, R3, RA94)
BILAGA 4

1994-01-11, Rev 0, 1994-11-03/ROWA
C:\K01\JFC\MGM1\EC_DATA\STATRES\RA94

RINGHALS 3 SG 1-3 RA-94

AVG. LENGTH PROPAGATION 1993 TO 1994 AS FUNCTION OF LENGTH 1993.

Length RA-93		2	3	4	5	6	7	P	No of tubes		Length		Std dev	
No of cracks		402	417	2369	4260	1369	484	173	No of ind		Average		0.03	
Average Prop.		1.58	0.72	0.46	0.22	-0.02	-0.08	-0.03	Average		0.28		0.66	
													0.66	
													0.66	

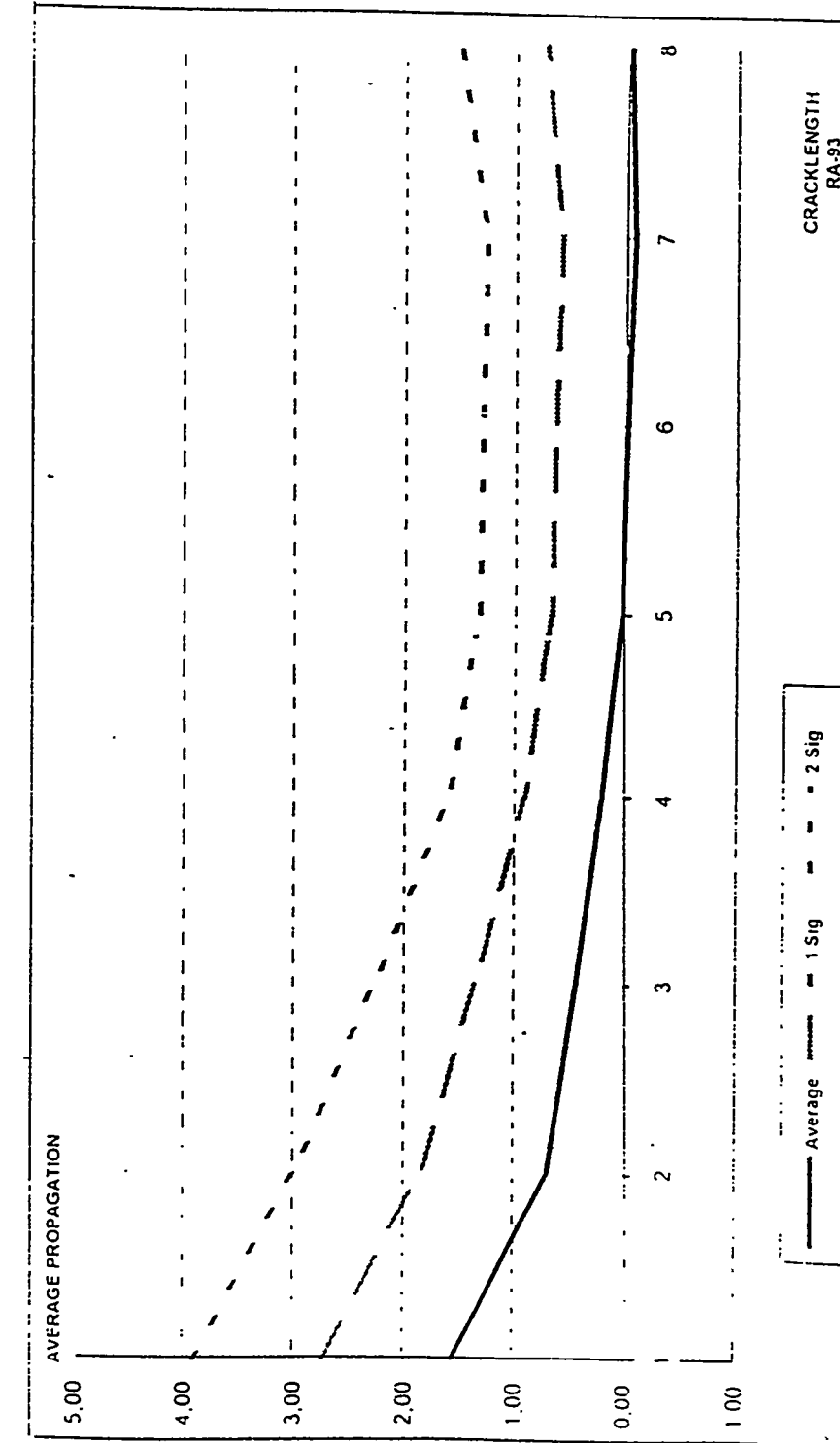


Figure 1. Crack length propagation 1993 to 1994 as function of length 1993, taken as average from all three SGs (Svensson, 1994).

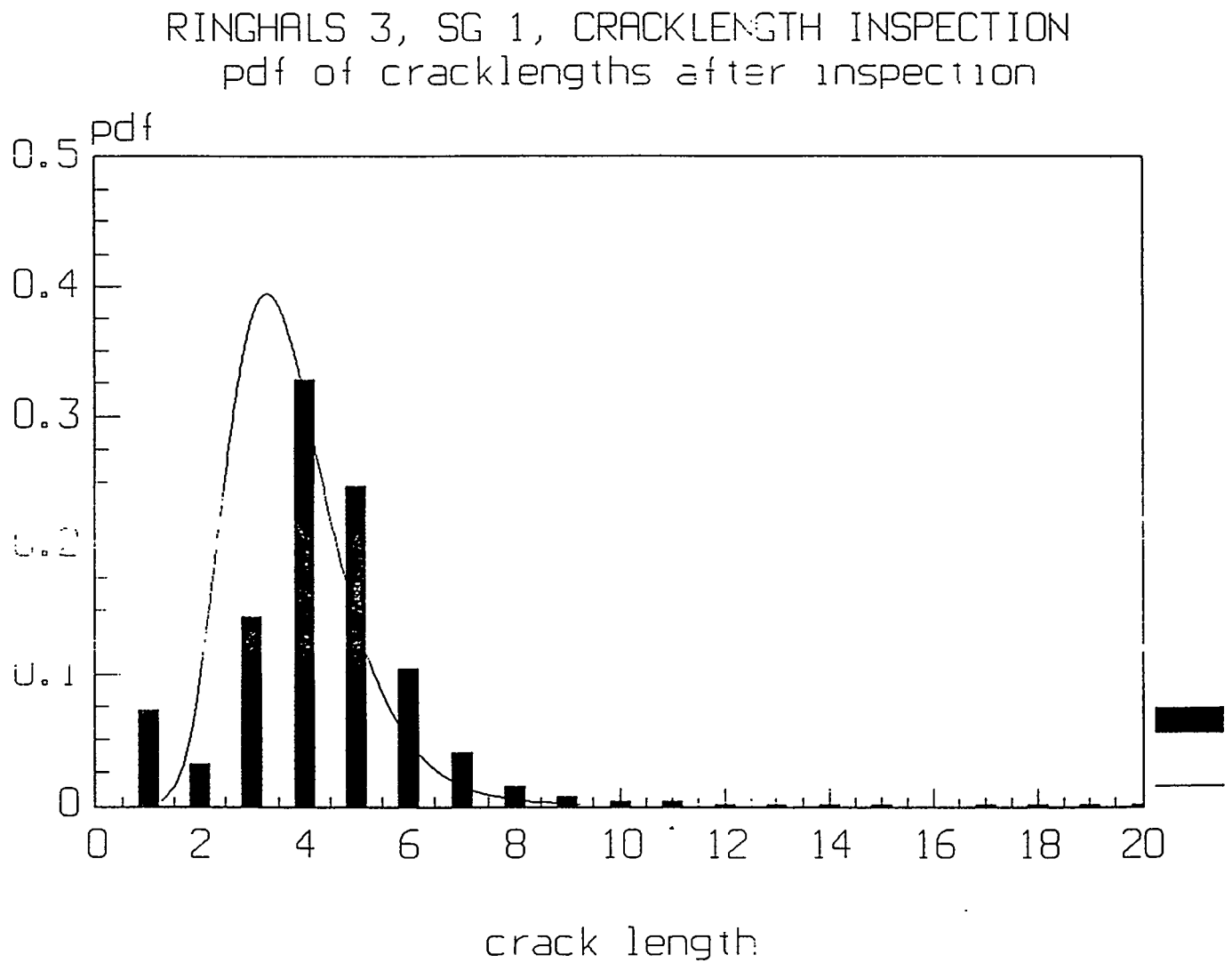


Figure 2. Pdf of crack lengths vs relative frequencies of observed lengths for SG1 in Table 1.

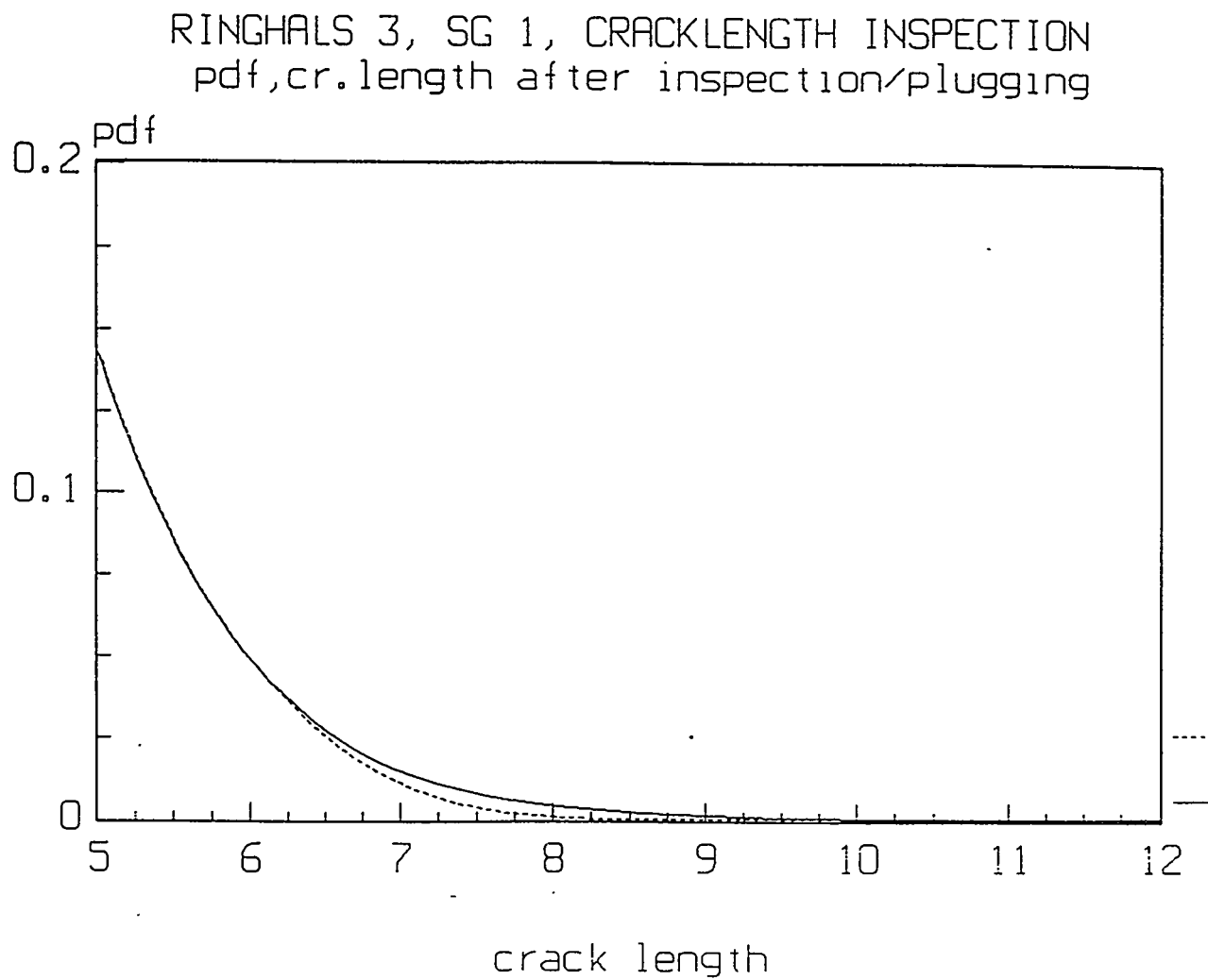


Figure3. The influence of plugging on pdf of crack lengths, with a plugging limit of 9mm.

RINGHALS 3, SG 1, CRACKLENGTH INSPECTION
pdf, cracklength after inspection/growth

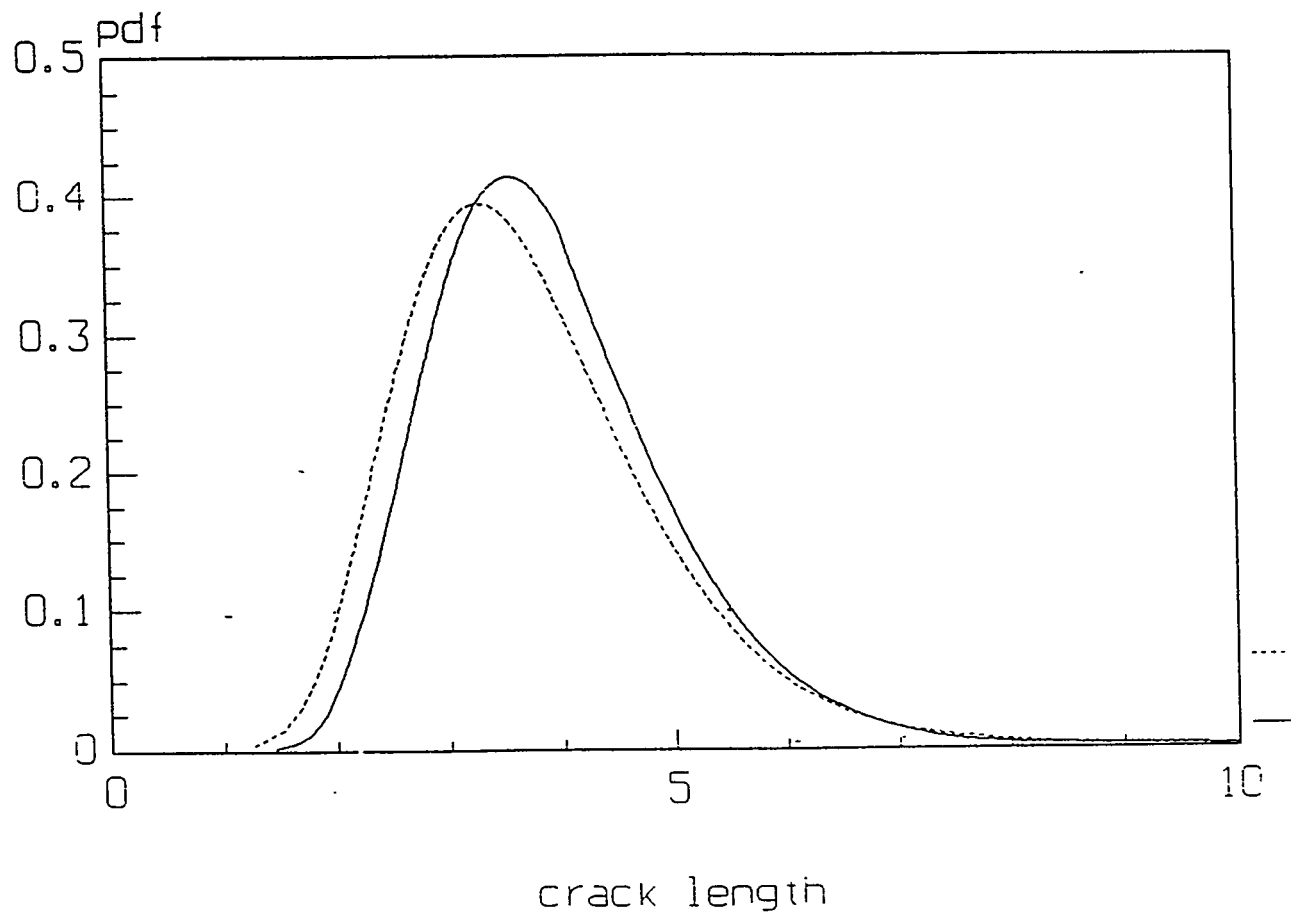


Figure 4. *Change of pdf of crack lengths due to propagation during one year*

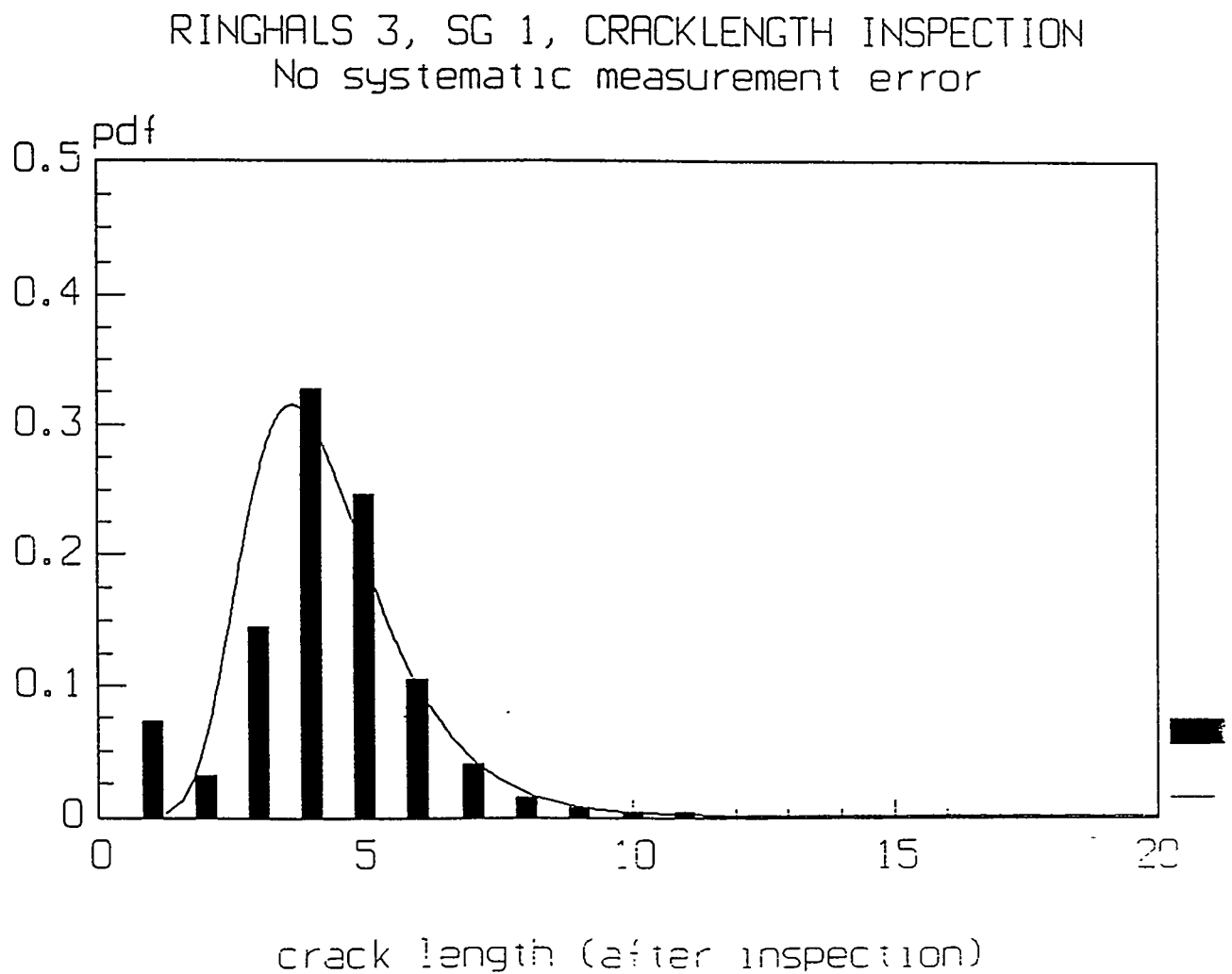


Figure 5. Pdf of crack length vs observed lengths if systematic measurement errors could be deleted

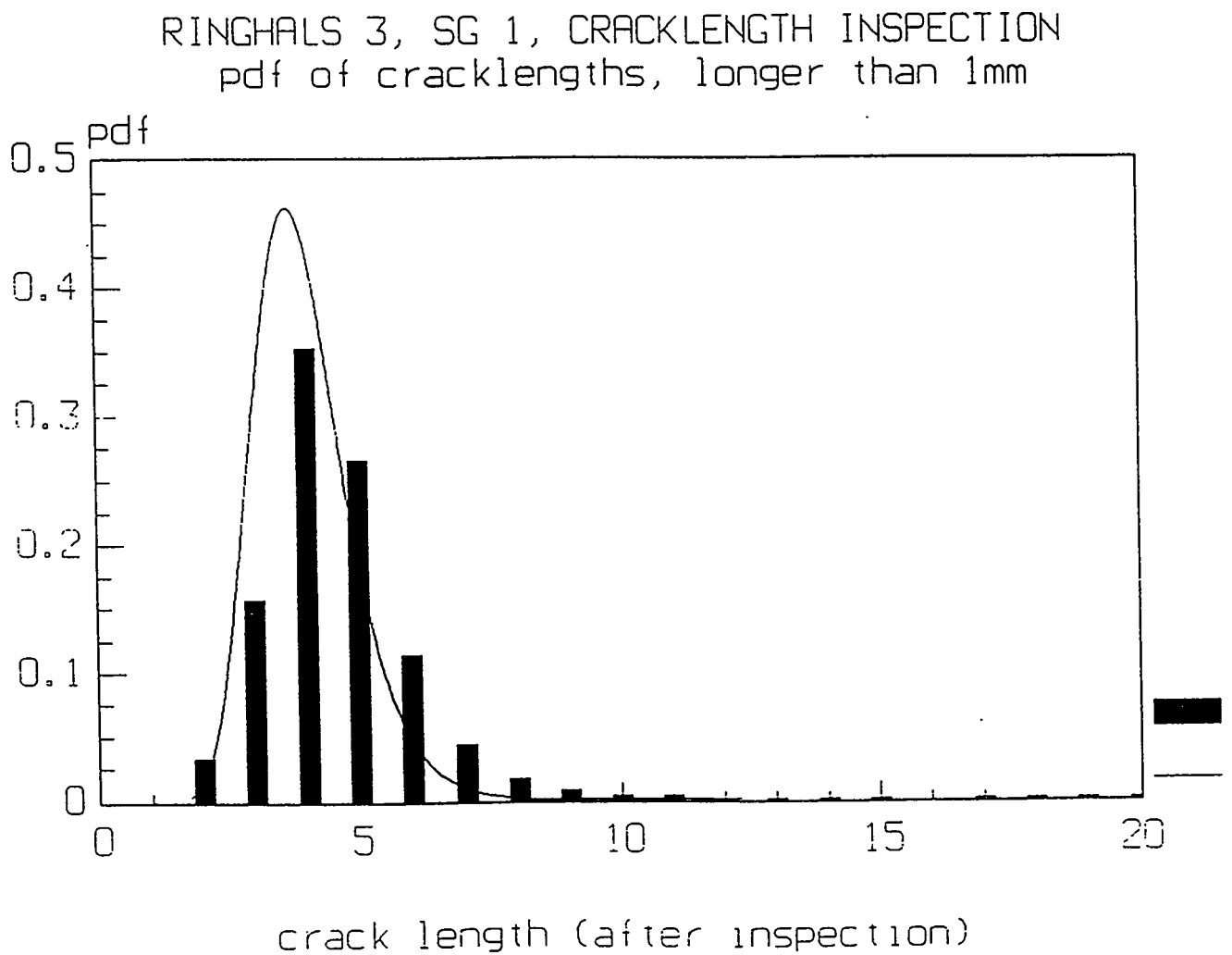


Figure 6. Pdf of crack lengths based on observed cracks longer than 1mm

STEAM GENERATOR TUBE DEGRADATION AT THE DOEL 4 PLANT INFLUENCE ON PLANT OPERATION AND SAFETY.

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ABSTRACT

The steam generator tubes of Doel 4 are affected by a multitude of corrosion phenomena. Some of them have been very difficult to manage because of their extremely fast evolution, non linear evolution behaviour or difficult detectability and/or measurability.

The exceptional corrosion behaviour of the steam generator tubes has had its drawbacks on plant operation and safety. Extensive inspection and repair campaigns have been necessary and have largely increased outage times and radiation exposure to personnel. Although considerable effort was invested by the utility to control corrosion problems, non anticipated phenomena and/or evolution have jeopardised plant safety.

The extensive plugging and repairs performed on the steam generators have necessitated continual review of the design basis safety studies and the adaptation of the protection system setpoints. The large asymmetric plugging has further complicated these reviews.

During the years many preventive and recently also defence measures have been implemented by the utility to manage corrosion and to decrease the probability and consequences of single or multiple tube rupture.

The present state of the Doel 4 steam generators remains troublesome and further examinations are performed to evaluate if continued operation until June '96, when the steam generators will be replaced, is justified.

BACKGROUND INFORMATION ON THE DOEL 4 STEAM GENERATORS.

The Doel Unit 4 is a 3 loop PWR nuclear power plant with a Westinghouse design NPSS and a 1000 MWe output. It started operation in 1985. The steam generators are of the Westinghouse E1 type, provided with special features such as: full feedwater flow into a preheater at the bottom of the cold leg side, stainless steel support plates with drilled holes, inconel cladding on the secondary side of the tubesheet.

Tube material is mill annealed Inconel 600 with an outside diameter of 3/4 inch. Tubes are hard rolled in the full length of the tubesheet, with a kiss roll at the top of tubesheet (TTS).

Similar steam generators of the E1 type in operation world-wide are at the South Texas 1 Unit and the Tihange 3 Unit. The Tihange 3 steam generators are an exact replica, while the South Texas steam generators do not possess features such as the Inconel cladding on the TTS and the stainless steel support plates.

Prior to operation rotopeening was performed on almost all tubes at the hot leg inner surface of the roll transition.

1. CORROSION PHENOMENA AND THEIR HISTORY.

1.1. AX/PWSCC/TTS: Axial primary water stress corrosion cracking at the top of tubesheet.

Axial primary water stress corrosion cracking at the top of tubesheet on the hot leg side (AX/PWSCC/TTS-HL) was already observed after the first cycle (1986). The degradation has shown a steady but slow growth and only a limited number of tubes had to be plugged for it. Today this degradation is no longer of any concern since all tubes have been sleeved on the hot leg side of the top of tubesheet.

Before sleeving was carried out, about 4% of the tubes treated with rotopeening and 85% of the tubes not rotopeened were measured to be affected. The beneficial effect of rotopeening is clear.

On the cold leg side AX/PWSCC is also developing but much slower and until now no tubes had to be plugged for it.

The safety concern for this type of degradation is axial tube burst. The plugging criterion is based on the length of the indication.

The degradation is easily measurable with an EC rotating pancake probe or with a UT pulse echo probe. Since 1991 and until sleeving was performed on all tubes, a 100% inspection at the hot leg roll transition of the three steam generators was performed each outage.

1.2. IGA-ODSCC/TSP: Outer diameter intergranular attack and stress corrosion cracking at the tube support plates.

Axial corrosion cracking at the tube support plates was first evidenced in 1990 after cycle 5. During the last two months of cycle 5 primary to secondary leak rate gradually increased on steam generator B to a value of 25 kg/h at the stop for refuelling. Five leaking tubes were identified by a pressure test with fluoresceine on the secondary side. Two of the leaks could be identified as originating from large axial cracks at the first tube support plate.

During the 1991 outage four tubes exhibiting this phenomenon were pulled. The degradation was characterised to be a combined intergranular attack/axial stress corrosion cracking on the outer surface at the tube support plates.

Steam generator B is the most strongly affected, on the other steam generators this degradation has until now not become a safety concern. There is a clear correlation with temperature, since the hot leg side is most affected and the number of defects decreases with the elevation of the support plate. Since 1992, after preventive measures were taken, the growth rate has gradually decreased and has become very small. In 1994 about 40% of the tubes of steam generator B were affected and 6% of the tubes were plugged for this degradation.

Since all axial cracks are situated within the TSP borders, axial tube burst is not a safety concern. Plugging limits are based on requiring sufficient margin with respect to circumferential tube rupture and limiting off site doses (by primary to secondary leakage) in the event of a secondary line break. Plugging limits were determined by leakage and rupture tests on pulled tubes. Because of the large extent of the problem (small but many cracks) the plugging limit based on the off site dose limit has been the most restricting since 1993.

The degradation is easily measurable by the eddy current bobbin coil technique and since 1991 a 100% inspection is performed each year on steam generator B.

1.3. AX/ODSCC/FS: Axial outer diameter stress corrosion cracking in the free span.

Axial stress corrosion cracking on the outer surface in the free length of the tubes was first discovered on tubes pulled for IGA-ODSCC/TSP in 1991. In 1992 about 120 tubes had to be plugged for this degradation. After preventive measures in 1992 the evolution has steadily decreased and the plugging limit has been increased on the basis of tests on pulled tubes, such that only few more tubes had to be plugged.

The degradation affects only steam generator B. Laboratory examinations on pulled tubes showed that the corrosion consists of a dense network of small (1 mm long) axial cracks with a depth of maximum 40% of the tube wall. Only the density of the cracks seems to be a variable parameter. The density of cracks seems to be correlated to the primary temperature since degradation is strongest on the hot leg side and decreases with increasing elevation.

The safety concern is axial tube burst. Based on tests on pulled tubes, the burst pressure is correlated to the indication amplitude.

The degradation is easily measurable with the EC bobbin coil. Since 1992 a 100% inspection was performed each outage on steam generator B.

1.4. CI/ODSCC/TTS-HL: Circumferential outer diameter stress corrosion cracking at the top of tubesheet hot leg.

Circumferential stress corrosion cracking on the outer surface at top of tubesheet on the hot leg (CI/ODSCC/TTS-HL) was first evidenced by chance on pulled tubes during the 1991 outage. The degradation consists of a small band (2 mm width) of circumferential cracks situated in the roll transition area. The small cracks join to form larger circumferential cracks. Due to their very closed appearance and complex structure they are difficult to detect by EC techniques and even more difficult to quantify by UT techniques. The three steam generators are equally affected.

According to non destructive measurements this degradation phenomenon, since its discovery in 1991, has shown a rapid increase in both number and growth rate, such that in 1993 the utility had to resort to sleeving because of the large number of tubes (2250) that had to be treated. One year after, in 1994, new indications were discovered for which tube integrity could not be guaranteed and all tubes had to be sleeved at the top of tubesheet on the hot leg. Since, this degradation no longer constitutes a safety concern.

The safety concern for this degradation is circumferential tube rupture. Plugging criteria are expressed as the cross section area required to guarantee tube integrity, growth rate within an inspection interval taken into account. On two occasions plant safety was clearly jeopardised by this phenomenon: in 1993 tubes were pulled for which the required unaffected cross section area was insufficient to guarantee tube integrity in the event of a secondary line break, and in 1994 non destructive inspection results indicated the same safety problem for a large number of tubes. The uncontrolled situation resulted the first time from measurement uncertainties, the second time increased growth rate was not anticipated.

1.5. CI/ODSCC/TTS-CL: Circumferential outer diameter stress corrosion cracking at the top of tubesheet cold leg.

In 1994 indications of CI/ODSCC were found on a sample of tubes inspected by UT at the top of tubesheet on the cold leg side. The inspections were extended to all tubes and a large number showed CI/ODSCC indications. Fourteen tubes were pulled, only one small circumferential crack was confirmed. It was concluded that the high sensitivity of the UT technique to phenomena such as scratches or external deposits was responsible for the indications.

In 1995 55% of the tubes were inspected again on the cold side at the top of tubesheet with a combined UT/EC rotating probe, no evolution of the UT indications was found and EC did not confirm the indications.

These findings demonstrate the difficulty to detect and quantify CI/ODSCC and to discriminate them from spurious indications.

1.6. CI/PWSCC/HEJ: Circumferential primary water stress corrosion cracking at the hybrid expansion joint.

One week before the planned outage of 1994 suddenly the primary to secondary leak rate of steam generator G step wise increased from zero to 21 kg/h on N16 measurement. The leaking tube was identified by a pressure test with fluoresceine on the secondary side to be a tube repaired with a HEJ (hybrid expansion joint) sleeve in 1993. The tube was pulled and a circumferential primary water stress corrosion crack, through wall over 180°, was found in the parent tube at the upper hydraulic expansion transition of the HEJ. A second tube pulled at random in steam generator G displayed the same type of defect, only smaller in size.

All HEJ were potentially degraded by this phenomenon and no technique was available to measure it by non destructive inspection. The utility decided to have all 36" HEJ sleeves (≈ 1700) repaired by the Westinghouse LWS (Laser Welded Sleeve) technique.

1.7. Potential CI/PWSCC/LWJ: Circumferential primary water stress corrosion cracking at the laser weld joint.

During the 1994 outage all remaining tubes (without a plug or HEJ sleeve) were treated with 12" laser welded sleeves (11197 LWS) for CI/ODSCC/TTS-HL and all HEJ sleeves still in service were repaired for CI/PWSCC/HEJ with a laser welded joint above the HEJ (1088 HEJ). Both the laser weld and the upper transition of the hydraulic expansion have been heat treated to relief residual stresses and thereby preventing PWSCC.

Corrosion tests on LWS and repaired HEJ samples have demonstrated the parent tube at the laser weld to be the most susceptible to CI/PWSCC, although failure times were comfortably large. The 1994-1995 cycle has been limited to 8 months, after which continued operation had to be justified by an extensive ISI program.

A 100% inspection of the laser welded joints by Cecco 3 and destructive examination of 10 pulled joint (4 LWS and 6 repaired HEJ) have not revealed any stress corrosion cracking in the suspected area's (at laser weld or expansion transition). On the other hand two new phenomena were discovered on the pulled tube sections: a wastage of the sleeve at the upper end of the HEJ and a bulging of the parent tube starting at the weld, with a maximum amplitude at the expansion transition

and extending one to two inches above the expansion. The first condition was explained to result from the boric acid that was concentrated at this location during the vacuum drying for the repair of the HEJ sleeves, that can form a very aggressive environment at operating temperature. The second condition was explained by the hypothesis that tubes were locked at the tube support plates during heat treatment. This situation must have resulted in large far field axial stresses in the tubes after heat treatment. Simulations of the locked tube condition were performed and after heat treatment axial far field stresses of about 300 MPa were measured.

The utility has decided to plug all tubes still in service with repaired HEJ sleeves, because they presented by far the largest bulging. By this action, the wastage problem also no longer constituted a safety concern.

A test program was started to examine the corrosion resistance of bulged LWS samples. Meanwhile the unit was returned to service for a limited time (two months). On the basis of the results of this corrosion test program continued operation until January 15th has been granted.

2. PLANT OPERATION.

2.1. Primary to secondary leakage.

In contrast to the troublesome degradation behaviour of Doel 4 steam generators, the primary to secondary leakage has posed only limited operational burden. No forced outage (in the sense of technical specification limitation) has occurred and only on one occasion the utility decided an unplanned outage to inspect and repair steam generators (November 1992), because of a sudden increase of primary to secondary leakage (to about 37 kg/h) through a CI/ODSCC/TTS-HL crack. Only on two other occasions an increase of primary to secondary leak rate has occurred, each time relatively short before a planned outage. In 1990 the leak rate in steam generator B started to increase two months before the outage to attain 25 kg/h when the plant was shutdown. The leak was identified during the outage to originate from AX/PWSCC/TTS and IGA-ODSCC/TSP. In 1994 one week before the planned outage leak rate step wise increased in steam generator G to 21 kg/h. The utility decided to continue plant operation until the planned shutdown, at that moment the leak rate had decreased to 11 kg/h. The leak was identified during the outage to originate from CI/PWSCC in the parent tube at the upper transition of the hydraulic expansion of a HEJ sleeve installed during the previous outage.

Although from an operational point of view this behaviour -low probability for leakage- of the Doel 4 steam generators is advantageous, is certainly is not from a safety point of view (see point 3.1).

2.2. Outages.

The troublesome degradation behaviour of the Doel 4 steam generators has resulted in an important increase of the duration of annual outages. A normal refuelling outage at the Doel 4 plant with a partial quick refuelling takes about 23 days. Figure 1 shows the duration of annual outages. It clearly shows an increase with time as degradation problems become more extensive and complex.

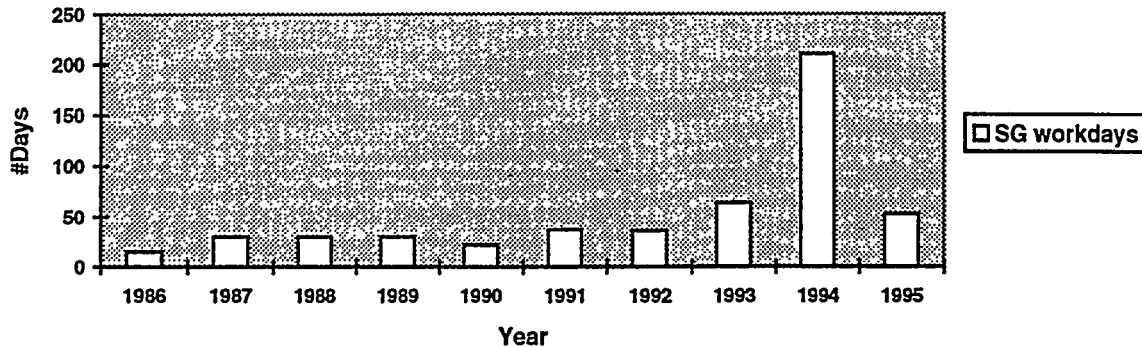
Extensive in service inspection campaigns have been necessary to identify degraded tubes and to justify continued operation. Although these inspection programs have been managed with great efficiency by the utility, for instance a hot lab was installed at the Doel site to perform examinations on pulled tubes, they have appreciably increased outage times.

The extensive sleeving repair campaigns that were found to be necessary during the 1993 and 1994 outages resulted in an important increase of outage duration. In 1994 when all tubes had to be repaired with either a new 12" laser welded sleeve or a laser welded joint above the hybrid expansion joint of an already installed sleeve, outage duration peaked at 210 days. The sleeving itself took the majority of the outage time, about 145 days.

Table 1: Duration of steam generator work during annual outages.

	1986	1987	1988	1989	1990	1991	1992	1993	1994	1995
SG workdays	15	30	30	30	22	37	36	64	210	53

Figure 1: outage duration in days



2.3. Radiation doses.

The total radiation dose of a normal annual refuelling outage at the Doel 4 plant is typically between 400 and 600 man-mSv, work on the steam generators excluded. A minimum program on the steam generators consists of non destructive inspections according to regulations (ASME XI) and usually sludge lancing, this program takes about 200 manmSv.

Figure 2 shows the annual radiation doses due to steam generator work over the years. The higher doses in 1988 and 1989 are the result of preventive actions -shotpeening and heat treatment (see par. 2.5)- that were performed. From 1991 on annual doses steeply increase as degradation becomes more widespread and acute. Factors adding to this dose increase are the extensions of the in service inspection programs (destructive and non destructive) and in 1993 and 1994 the sleeving repairs campaigns. In 1994 the dose peaked at about 1300 man-mSv.

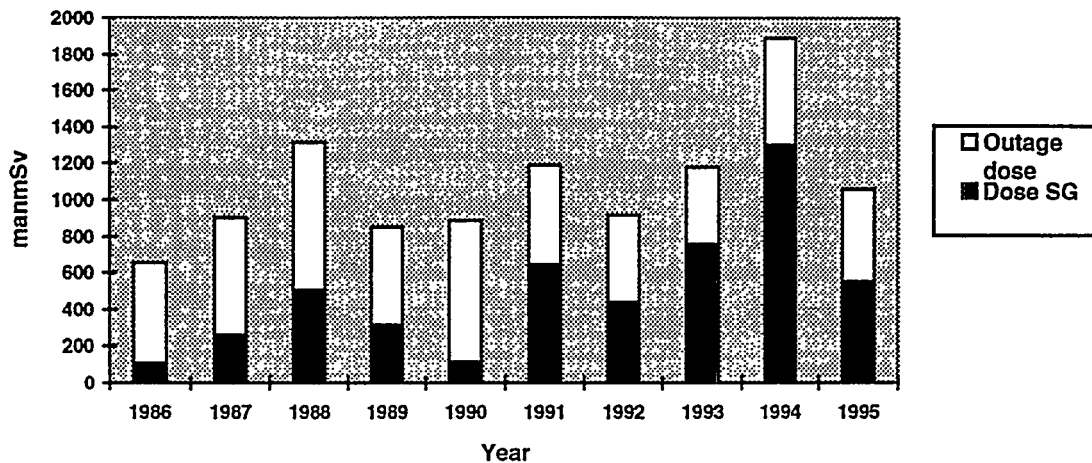
It should be stressed that a lot of effort has been spent on limiting radiation dose due to steam generator work. Intensified training and increased automation have continually and appreciably decreased the dose per manhour over the years. Actions have been undertaken in order to decrease radiation in the steam generators (typically 30 to 40 mSv/h) but they have not been very successful.

The radiation dose due to the extensive sleeving repair campaign in 1994 (1300 manmSv) compares to the radiation dose of a steam generator replacement (1700 manmSv at Doel 3 in 1993). Early steam generator replacement is thus also advantageous from the ALARA point of view.

Table 2: radiation dosis during annual outages (manmSv)

	1986	1987	1988	1989	1990	1991	1992	1993	1994	1995
Dose SG	108	258	506	317	114	643	441	759	1303	555
Total dose	654	902	1314	852	890	1190	922	1179	1892	1063

Figure 2: outage doses



2.4. Plugging and power output.

Extensive plugging over the years (see figure 3) has had an influence on the plant power output. After the 1994 outage thermal power output had decreased to about 97% of nominal power. This was the maximum thermal power reached with all turbine admission valves fully opened. After the 1995 outage the thermal power output further decreased to about 93.5%.

2.5. Preventive actions.

2.5.1. Actions to improve chemical environment.

Sludge lancing has been performed almost every outage. The quantity of sludge removed has always been relatively small, between 1 and 20 kg per steam generator.

In 1989 the pH of the secondary water was slightly increased from 9.4 to 9.6.

In 1991 for the first time tubes were pulled from steam generator B. Chemical examination evidenced the presence of lead, sulphate and organic molecules. The hide out return confirmed the high sulphate concentration. The lead was believed to originate from a radiation protection shield that was accidentally left in steam generator B after the 1986 outage. The sulphate could only originate from regeneration resins in the blow down purification. The organic compounds were present in the make up water and originate from an open air intermediate storage of town water

supply. The environment on the secondary side of the steam generator was calculated to be very acid (2 pH).

As a consequence of these establishments a number of actions was taken immediately to improve secondary chemistry. The regeneration resins were changed and their rinsing after regeneration was increased. The quantity of make up water was minimised by obtaining auxiliary steam from the other Doel units. In 1992 supplementary actions were taken. A chemical cleaning was performed on the three steam generators (iron step only, EPRI process), and the secondary water treatment was changed from AVT to sodium phosphate treatment ($\text{Na}/\text{PO}_4=2$ and later 2.2), at the same time the hydrazine injection was increased.

Examination on 11 tubes pulled from steam generator B in 1992 evidenced that lead was still present and Cu was also widespread. The Cu is believed to originate from the tubesheet of the condenser, the only component in the secondary system made of a Cu containing alloy. It was decided to perform a second chemical cleaning of steam generator B in 1993 with a KWU process including a Cu step.

In 1993 osmotic membranes were installed to improve the purity of make up water. This purification has largely reduced the TOC of make up water and consequently in the steam generators.

The effect of the preventive actions cannot be clearly established from the behaviour of the corrosion processes. Both AX/ODSCC/FS and IGA-ODSCC/TSP in steam generator B have shown a significant decrease in evolution rate since 1992. This was however not the case for CI/ODSCC/TTS that on the contrary has shown a dramatic increase in the three steam generators since 1992.

In 1996 the condenser will be replaced by one containing no copper.

2.5.2. Actions to reduce material stresses.

Before the plant was put into operation rotopeening was performed on the inner surface of the tubes at the hot side of the top of tubesheet. This measure has proved to be very effective in preventing PWSCC in this region and it was decided to perform shotpeening on the cold leg side in 1988.

In 1989 the U-bends of tubes in the first two rows have been thermally treated. Until today no corrosion problems have been detected in this region.

2.5.3. Temperature reduction.

In 1990 the mean primary temperature was decreased from 312 to 308 °C. The hot leg temperature became 326 °C which is still relatively high.

This action was not only taken to prevent corrosion, the main reason was to reposition turbine admission valves in a less erosive configuration.

3. PLANT SAFETY.

The degradation behaviour of the Doel 4 steam generators at several occasions has raised serious questions about plant safety.

Until 1994, when all tubes were sleeved at the top of tubesheet on the hot leg side, the CI/ODSCC/TTS-HL had been the most troublesome degradation. It had demonstrated to be extremely difficult to measure and to predict its evolution. In 1993 tubes have been pulled that no longer presented the structural strength required to sustain a secondary line break. In operation these tubes did not present a sufficiently large leakage to be alarmed. As a consequence of these findings the UT detection technique was recalibrated on the basis of a large number of pulled tubes (13 in total) and large correction factors had to be applied on the measured crack dimensions. Until 1993 the measured surface growth rate of the indications within one cycle was bounded by 50% of their initial surface. In 1994 however a lot of indications were measured that had apparently grown much faster than anticipated and even new indications were found had grown beyond the required structural margin within one cycle. It should be noted that these conclusions have been derived from UT measurements and that no direct evidence exists that proof them. Anyhow the measurability or the growth rate had become beyond control and the utility decided to sleeve all possible tubes with elevated tubesheet Laser Welded Sleeves, the remaining tubes were plugged.

In 1994 an in service leak was identified to originate from a CI/PWSCC crack at the upper hydraulic transition of a parent tube that was repaired the year before with a HEJ sleeve. The crack was through wall over 180°. A second HEJ pulled at random displayed similar cracks of smaller dimension. It was concluded that all sleeved tubes were potentially degraded by this extremely fast PWSCC. No effort was made to measure the degradation, but it is possible that tubes were present at the end of cycle that no longer possessed the required structural strength.

In 1995 all tubes were inspected at the laser weld with a Cecco 3 EC probe and no degradation phenomena were detected at the laser weld joint. Ten sleeved were pulled and showed no degradation at the laser weld joint either. However an unexpected phenomenon was discovered that raised questions about the qualification of the repair process. All pulled tubes displayed a bulging of the parent tube above the weld. This bulging was explained by considering the tubes locked at the tube support plates during the heat treatment of the laser weld joint. This condition was not taken into account, because TSP are made of stainless steel and no evidence of blocked tubes existed. Laboratory tests showed that this phenomenon can have produced large axial far field stresses in the tubes, that were not accounted for in the qualification of the repair technique and could well produce CI/PWSCC, most probably at the weld.

The troublesome degradation behaviour of the Doel 4 steam generators has led to a number of actions to improve plant safety. This chapter gives a summary of the different actions that were taken together with their motivations.

The extensive plugging and sleeving of steam generator tubes has continually changed the physical properties of the steam generators. The plant design safety studies had to be reviewed and the protection system adapted at several occasions. Paragraph 3.5 gives an overview of this process.

3.1. Accident preventive actions.

Apart from the operational preventive actions listed under par. 2.5 other preventive actions have mainly relied on the leak before break principle.

In 1990 nitrogen 16 monitors have been installed on the three main steam lines. They have proven to be a very sensitive and fast method to monitor primary to secondary leakage. In November 1992 and February 1994 they detected the step like leakage increase, that seems to be a property of the leakage behaviour of circumferential cracks. The same year integrating gamma ray detection chains have been installed on the three generator sampling lines. They are slower in response but remain functioning after a reactor scram.

Since October 1994 the operability of nitrogen 16 monitors is required by plant Technical Specifications. At the same time the operational limits for primary to secondary leakage have been reduced: stationary leak rate has been limited to 20 kg/h per steam generator and a separate limit for the maximum leak rate increase (10 kg/h in one hour) was included in Technical Specifications. This last limit is based on the leakage history of Doel 4 steam generators: all leakage's in the past originating from circumferential cracks would have exceeded it.

In 1994 it was agreed with the utility not to perform any voluntary power transients (e.g. house load operation), in order to minimise solicitation of the steam generator tubes.

On two occasions the cycle length has been limited in order to perform in service inspections with an increased periodicity. Cycle 10 (1994-1995) was limited to 8 months, after which an extensive in service inspection program was carried out to justify continued operation. Presently cycle 11 is limited to 5 months, until January 1996.

3.2. Accident mitigating actions.

Already at the beginning of plant operation (1985) accident procedures for beyond design basis accidents involving steam generator tube rupture were adopted. They are based on the Westinghouse ERP for steam generator tube rupture together with a secondary line break or loss of primary coolant.

Since 1992 special attention has been paid to accidents involving steam generator tube rupture during the training courses of operators on the plant full scope simulator and operators have been made aware (but not too aware) of the troublesome condition of the Doel 4 steam generators.

In 1994 accident procedures were tested on the plant full scope simulator for beyond design accidents. A multiple tube rupture of 5 tubes was combined with a non isolated steam leak or a steam line rupture. On a nuclear plant analyser the accident procedures were tested for a multiple rupture of 10 tubes. The accident procedures were found adequate to cope with these accidents, only minor additions were necessary.

In 1994 the probabilistic safety assessment study of Doel 3 (completed in 1993) was examined to identify the main factor contributing to core melt in the event of steam generator tube rupture. The failure to close of the atmospheric discharge line was identified to be the main contributor to core melt for this scenario. To improve the availability of this system it was included in the plant technical specifications with an allowed outage time of 72 hours. The periodic functional testing was increased to monthly for both the discharge valve and its isolation valve.

3.3. Accident consequences mitigating actions.

In 1990 integrating gamma ray detection chains were installed at the three main steam lines. They serve to evaluate the radiological consequences of a tube rupture, such that appropriate actions can be taken with respect to public safety.

In 1994 the operational limit in technical specifications for iodine primary activity has been reduced by a factor of ten (stationary max. 420 MBq/ton of dose equivalent I-concentration).

In 1995 the transient Iodine concentration was further reduced by a factor of 4 (9.25 GBq/t dose equivalent I-concentration).

3.4. Corrective actions.

In June 1996 the steam generators will be replaced. The replacement was already decided at the end of 1992, mainly because of IGA-ODSCC/TSP and AX/ODSCC/FS.

3.5. Safety studies.

The initial design safety studies were carried out without any margin for plugging of steam generator tubes. It could be demonstrated however that they covered a plugging fraction of 2% in each steam generator.

In 1991 the actual plugging fractions were still very limited (0%, 0%, 1.9%) but it became clear that the justified plugging limits would not much longer cover the actual situation. It was then decided to review the safety studies to justify a plugging fraction of 15% in each steam generator. The affected accidents were mainly DNB accidents such as 'Uncontrolled rod withdrawal', primary overpressure accidents and to a lesser extend the LOCA.

As a consequence of this review the overpower and overtemperature protections had to be adapted.

In 1993 the actual plugging fractions became 4.4%, 4.3% and 13.2%. It was clear that the limit of 15% in each steam generator wouldn't be satisfied much longer. The asymmetric degradation behaviour resulted in a much higher plugging fraction of steam generator B, the already present unbalance in primary coolant flow at zero plugging added to this asymmetry. Therefore a second review of the safety studies was performed for a plugging configuration of 10%, 10% and 30%.

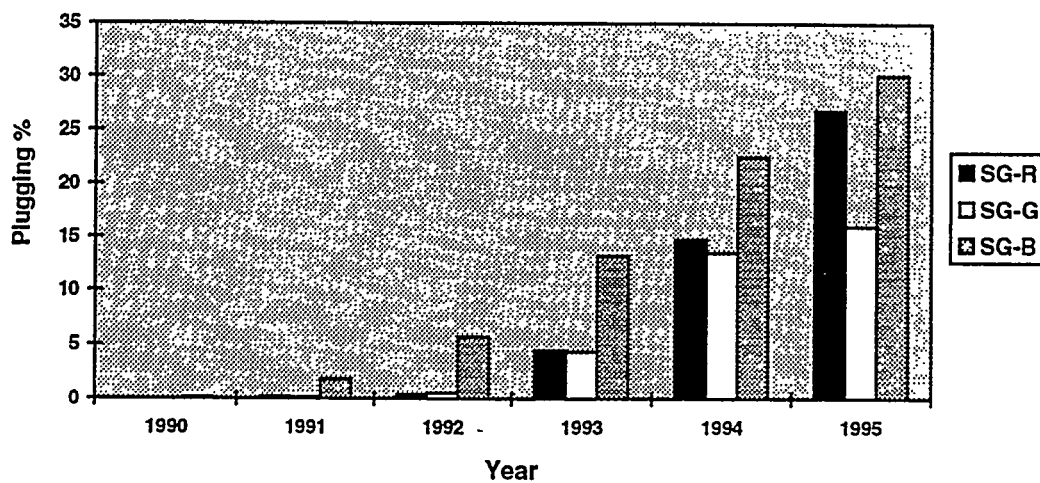
In 1994 plugging reached 14.7%, 13.5% and 22.5%. This configuration was not fully covered by the reviewed safety studies but could be justified by the available margins. The asymmetric flow in the primary loops necessitated a conservative adaptation of the calibration of overpower and overtemperature protection with respect to delta temperature and mean temperature.

In 1995 plugging fractions further increased to 26.8%, 16.0% and 30.1% and continued operation was justified by limiting the maximum allowed power to 96% of nominal thermal power. All protection parameters were set relative to the newly determined maximum power. In addition the pressuriser high level scram setpoint had to be reduced for overpressure protection of the primary system.

Table 3: equivalent plugging fractions in %.

	1990	1991	1992	1993	1994	1995
SG-R	0.0	0.2	0.4	4.4	14.7	26.8
SG-G	0.0	0.1	0.5	4.3	13.5	16.0
SG-B	0.1	1.8	5.7	13.2	22.5	30.1

Figure 3: equivalent plugging fractions.



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Instructions: Place pages 34a through 34kk in front of the view graphs for
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In Chapter 4, Section 4.2, hardcopy of the view graphs for the
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on pages 35 to 57. The text of the paper, which was inadvertently
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**CNRA WORKSHOP, ARGONNE, ILLINOIS
30th OCTOBER TO 2nd NOVEMBER 1995**

**FRENCH REGULATORY PRACTICE
AND EXPERIENCE FEEDBACK
ON STEAM GENERATOR
TUBE INTEGRITY**

BY

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Figure 1: Primary/secondary leak

Figure 2: Primary/secondary leak

1 - INTRODUCTION

Almost 80% of the electricity consumed in France is generated by the 54 pressurised water nuclear reactors operated by Electricité de France (EDF). Of these 54 reactors, all from a standardised serie, 24 have an installed capacity of 900 MWe, and the remaining 20 have an installed capacity of 1300 MWe. Two new 1450 MWe reactors are currently being started up and two others are under construction.

This represents over 180 Westinghouse design steam generators, most of them containing Alloy 600 tubes, either mill-annealed or heat treated. The new steam generators, which were completed in 1990, are fitted with heat treated Alloy 690 tubes.

The first signs of damage to the Alloy 600 tubes appeared in the early eighties. Since then, the number of tubes affected, and the different types of damage, have increased continually. Each type of damage was carefully analysed to determine a suitable in-service inspection campaign and preventive or curative treatments which will be discussed in detail in the remainder of this document.

Appendix 1 sets out the progression in the number of tubes plugged each year, together with the reasons for the plugging.

2 - GENERAL METHODOLOGY

The aim is to keep the risk of steam generator tube rupture as a result of various degradation mechanisms, at an acceptable level, i.e. compatible with the accident probabilities adopted in the design process.

Prevention of tube rupture therefore constitutes a large proportion of the analysis work performed by the operator and the safety authority.

However, one should not under-estimate the effort made to reduce the consequences of such an accident, whether this is through studies of accident sequences, surveillance and reduction of radioactivity in the reactor coolant, operating resources and procedures implemented to detect which steam generator has suffered from a tube rupture, or finally the training given to operators to cope with such a situation.

For the purposes of prevention, the fundamental principle applied in France consists in withdrawing from service any tube containing a defect for which there is a high probability of its reaching critical size under accident conditions, and hence of becoming unstable during the next operating cycle.

The way in which this principle is applied to the various types of degradation encountered will be mentioned in detail below. It should, however, be stipulated now that application of such a principle requires a sound knowledge of the phenomena observed, particularly of the respective degradation kinetics, of the areas which are preferentially affected and the morphology of the defects encountered.

Such knowledge of the damage has a considerable importance on the non-destructive examination (NDE) method requirements, on the degree of accuracy and reliability these tests must have, and on the frequency of the tests to be conducted.

Any uncertainty regarding one of these factors should lead the safety authority to adopt safety margins, and also lead the operator to increase the precision of these analyses and supplement its knowledge of the damage to avoid plugging too many tubes.

This principle has been applied in France for over ten years. It requires frequent questioning of the validity of the knowledge acquired, through detailed analysis of annual experience feedback and detection of any specific event which could herald further damage or a new phase of development for a known type of damage. It makes it mandatory to take account of all new data, especially that obtained from non-destructive tests and from examinations performed on tubes removed from service and of all additional analyses carried out in parallel.

There is therefore no definite set rule, but rather an ongoing process of development of the rules drawn up for each type of damage encountered.

Surveillance during reactor outage is one of the cornerstones of this approach. It is supplemented by an operating monitoring system based mainly on detection of loose parts using acoustic sensors, and on the measurement of changes in the rate of leakage from the reactor coolant system to the secondary system. The rules are being strengthened in this field too, so that the early signs of tube rupture can be detected as soon as possible, despite the increase in the number of tubes affected. The section 5 of this document will be devoted to this issue.

3 - DEGRADATIONS OBSERVED

3.1 - Stress corrosion cracking in the reactor coolant system environment (PWSCC)

This phenomenon, which first appeared in France in the reactor at Unit 1 of Fessenheim Power Plant in 1981, is still today the predominant cause for tube plugging. It affects tubes in areas where stresses are high, i.e. mainly the roll transition zones and the bent parts of the first two rows of tubes.

It occurs mainly in two forms: axial or circumferential cracking.

Roll transition zone

It should be borne in mind with regard to the transition zone, that of the five first reactors built in France, only one integral rolling operation was carried out over the full thickness of the tube sheet. On the subsequent steam generators, additional rolling, the so-called kiss-rolling was carried out to try to reduce longitudinal stresses in the outer tube wall.

Axial cracks are formed at the level of the roll transition zone, and mainly develop axially.

Apparently, the first steam generators are only slightly affected by this phenomenon. However, it is seen that kiss-rolling had a particularly adverse effect on subsequent steam generators with respect to this cause of cracking.

Steam generators on which kiss-rolling had been performed and with mill-annealed Alloy 600 tubes have so far a mean ratio of tubes cracked by PWSCC close to 35 %. A considerable amount of data is now available on this form of cracking : crack initiation time, morphology of the cracks and speed of propagation, taken from the regular inspections performed in the last fifteen years, and from the results of tests and examinations carried out on tubes removed specifically for that purpose. (Since the first PWRs entered operation, EDF has removed some 300 tubes).

These cracks are likely to become through-wall cracks as soon as their length exceeds 4 to 5 mm. From this moment, they can therefore contribute to primary to secondary leaks. Usually, crack propagation rates tend to decrease as the cracks become longer.

Certain specific configurations which can generate considerable residual stress cause the formation of circumferential cracks.

Particular mention can be made of certain rolling anomalies in the transition zone, of local strain produced, for instance, by denting cause by swelling sludges on top of the tube sheet, or simply the thermal effect of the sludge which modifies the distribution and magnitude of the stresses in the internal tube wall skin.

This type of flaw does not satisfy the leak before break criterion and data on the kinetics are rare, since circumferential cracks depth can not be properly estimated with the NDE methods currently used.

Compared with the axial cracking, circumferential cracking although considered more serious, affects only a smaller number of tubes. About 600 tubes have been recorded as having circumferential cracks, out of the 800,000 steam generator tubes currently in service. Appendix 2 shows the distribution of these cracks on 31/12/94.

In certain cases, owing to the likelihood of rapid formation of circumferential cracks, the tubes affected were plugged as a preventive measure.

This category includes tubes with a double rolling anomalies (protruding rolling and excessive expansion during kiss-rolling) and those subject to denting process above the tube sheet in the sludge pile.

U-Bends with a low bend radius

Cracking in the U-bends in Row 1 caused eleven forced outages, following major increases in the rate of leakage between the reactor coolant system and the secondary system between 1982 and 1986.

It was very quickly decided to plug the Westinghouse tubes in Row 1 as a preventive measure, and then to perform heat treatment on site for the unplugged tubes in Rows 1 and 2 which had not been heat treated in the factory after bending.

At present, there is insufficient exact information regarding the morphology and the propagation kinetics for these cracks, since it has not proved possible to remove and examine the areas concerned, and also because of the uncertainty of the NDE methods used. Cracks which form in this area may be either axial or circumferential.

An expert appraisal is currently underway on a series of bends from Row 1 taken from one of the steam generators removed at Dampierre 1. The findings should be available at the end of 1995.

3.2 - Tube degradations by corrosion on the secondary side (outer diameter)

Various types and morphologies of tube degradation by outer diameter (OD) corrosion processes can occur preferentially in confined areas such as crevices, or under deposits where the build up of chemicals and pollutants in the secondary coolant can produce concentrated corrosive local environments.

Thus various OD corrosion forms can be mainly found at the drilled TSP/tube intersections, in areas where sludge collects on the tube sheet or on some TSPs, and under deposits on tube free span OD surfaces.

OD corrosion degradations can show up under various morphologies and distributions which can be briefly categorised by some of the following types observed alone or in combination with other forms.

- * Discrete Intergranular Stress Corrosion Cracks (OD IGSCC)

- . either axial
- . or circumferential

- * Network of axial OD IGSCC, or of short circumferentials cracks

- * Networks of cracks of combined orientation both axial and circumferential, also called multidirectional cracking

- * Patches of "generalised" Intergranular Attack called IGA.

Corrosion at the tube/tube support plate intersection (TSP intersection)

The crevices and deposits build up at the intersection between the tube and the TSP constitute a preferential site for corrosive species to concentrate and degrade Alloy 600 OD tubing.

It is difficult, using current test equipment, to assess the real effect on the tube, owing to the complex nature of the damage (mixture of IGA and ODSCC).

Many NDE and destructive examinations were carried out on removed tubes, particularly after the rapid spread of this phenomenon on one of the steam generators at Saint-Laurent B1 between 1990 and 1991 (the cracking ratio - % of tube cracked to the whole tube population associated with this phenomenon - rose from 1% to 13% in the space of one fuel cycle).

These examinations provided better characterisation of the flaws observed and were used to establish a plugging criterion based on the amplitude of the signal measured by an axial probe. Many correlations were needed to determine this criterion.

Appendix 3 shows the increase in the number of tubes plugged for this reason against time.

It should be borne in mind that during the examination of tubes removed from the steam generator, the cracks observed were mainly axial, and deeper in the mid plane of the tube/TSP intersection. However in some tubes tiny circumferential shallow (typically less than 25 % wall thickness) net works of cracks were observed at the top or bottom boundaries of the TSP intersection, possibly opened up as a result of the high load required to achieve the extraction process. These circumferential cracks have been so far observed only in presence of more conspicuous IGA and deeper axial ODSCC occurrences which causes the main reason for bobbin coil calls. The cause of this phenomenon is not yet fully established, however it was taken into consideration in determining the plugging criteria.

Corrosion under deposits and in the sludge pile

This phenomenon was first observed at Bugey 3 in 1988.

Once again, it took the form of IGA patches and/or of networks of multiple fine axial and circumferential cracks in which large axial cracks could develop.

As this corrosion developed on the outer diameter wall, mainly next to the roll transition zone, the combined effect of cracking in the primary side and of corrosion in the secondary side had to be incorporated in the plugging criterion.

A few cases of corrosion were observed where deposits had built up above the tube support plates.

As with corrosion at the intersection between the tube and the tube support plate, and because of the similarity of the morphology of the damage, the plugging criterion was based on the amplitude of the eddy current signal measured by the bobbin coil probe, after correlation with the results of tube examinations with a large number of removed tubes. The procedures used made it possible to differentiate between the signal indicating deposits from the signal indicating corrosion.

The rotating pancake probe (MRPC) can be used to characterise axial cracks which have formed in the networks.

Appendix 4 shows the change in the number of tubes plugged for this reason. It can be seen that a large number of the preventive plugging operations in 1991 were carried out in response to an acceleration of the phenomenon and because of the difficulty of determining a reliable plugging criterion. Safety margins therefore had to be adopted.

External circumferential cracking below the roll transition zone

This phenomenon was expected in the sludge area on tubes which had not undergone kiss-rolling. It was first detected in 1988 using a MRPC.

However, a phenomenon of the same type was discovered outside the sludge area on steam generator tubes which had undergone kiss-rolling in 1990 at the Dampierre 3 reactor.

In this case, circumferential cracking formed just above the last point of contact between the rolled tube and the tube sheet.

For the moment, this phenomenon only affects mill-annealed Alloy 600 tubes, and the cause is not yet certain.

In view of the lack of kinetic data, mainly because of difficulties in measuring the crack depth initiated from the OD, and because the leak before break principle is not applicable here (these cracks have complex morphologies with kinetics that vary across the circumference of the tube), this phenomenon alone considerably increases the likelihood of steam generator tube rupture.

For this reason, all the steam generator tubes concerned have been inspected at each outage using the rotating probe since the start of 1995. All the tubes where this type of crack is detected are immediately plugged.

Appendix 5 sums up, in grey, the change in the number of tubes plugged for this reason since 1990.

In light of this alarming phenomenon, the safety authority has asked the operating organisation to rapidly develop a more sensitive inspection method, i.e. one able to detect this type of cracking at the earliest possible stage.

This request is currently being analysed by the operator.

3.3 - Wear

Wear caused by loose parts remains one of the main causes of forced outages affecting EDF reactors triggered by a primary to secondary leak rate increase.

The morphology of this wear can vary, mainly depending on the nature, shape and mass of the loose part as well as on the way it moves.

Since it is impossible to predict the type of loose part that could be transported to the tube sheet by the feed water, vigilance is essential to cope with this type of problem, both at the edge of the tube bundle and at the centre.

Wear caused by contact with anti-vibration bars is very common on first generation steam generators. Improvements in the tubing conditions for steam generators in the factory have ensured that the spacing between the tubes and the anti-vibration bars were not more closely controlled. Furthermore, the best possible material was chosen for the bars to reduce the effects of friction.

This phenomenon is regularly monitored. It has a slow but steady development kinetic (on average 2% of the thickness of the tube in one cycle, with a maximum of 17%).

A few cases of wear at the top of a bend were reported on 1300 MWe steam generators, when some of the anti-vibration bars are placed far too high with respect to the tube to be held in place.

3.4 - Vibration fatigue

Following rupture of a tube in the Mihama 2 reactor, caused by the incorrect positioning of an anti-vibration bar (vibration fatigue rupture of a poorly supported tube fixed in the upper tube support plate), inspections were carried out in France to ensure that the anti-vibration bars were correctly positioned.

The first inspections showed that in the case of certain bars, there was considerable displacement from their theoretical position.

Extending these inspections to cover all the steam generators in service showed that this was a generic problem, and one that should be rectified.

Vibration calculations, carried out as part of these inspections, identified those tubes which might represent a risk of rupture by vibration fatigue, given their position in the steam generator and any aggravating factors (fixed in the upper tube support plate, corrosion on the tube secondary side at the upper TSP intersection, specific configurations of adjacent anti-vibration bars).

Since plugging these tubes would not reduce the risk of fatigue rupture, an alternative solution had to be found instead of replacing the anti-vibration bars, which was judged too costly. A cable was therefore installed inside these tubes, before they were plugged, to damp vibrations and hence reduce the vibrational instability of the tube.

This solution is currently being implemented in 1995, and work will continue into 1996. This problem affects some 800 tubes.

3.5 - Other types of damage

Tube deformation at some TSP intersection

This type of deformation was observed long ago on certain steam generators. It affects some of the outer tubes at the upper tube support plates, generally opposite the aseismic blocks.

In 1991, a considerable change in the signal picked up by the axial probe on one of the tubes in Gravelines 1 triggered a massive inspection campaign to detect other cases of deformation, characterise them and ensure that cracks had not formed.

These cases of deformation, which cause a local increase in longitudinal stresses, could lead to the formation of circumferential cracks.

In order to detect them, an inspection was performed with a rotating probe on those tubes which had indicated the greatest amplitudes with the axial probe.

Since the root cause of the problem had not been determined, and since the phenomenon had not been proven to be harmless, all tubes showing a reduction in diameter of over 450 microns were plugged.

Tube bulbing above the tube sheet in certain outer tubes

This phenomenon was discovered in 1991 at Bugey 5, after a forced outage caused by a large-scale leak between the reactor coolant system and the secondary system. In some tubes, the swelling only affected a limited area of the tube wall. These tubes were all located close to a loose part.

No explanation has yet been found, although an attempt was made to relate the phenomenon to the presence of the loose part.

Destructive examinations are to be carried out on the tubes affected at Bugey 5 in an attempt to explain this phenomenon.

In the meanwhile, tubes have been found with this same type of deformation on other steam generators. They were all plugged as a safety measure.

Displacement of the tube bundle wrapper

In 1993, during an operation to clean the tube sheet of a steam generator at Blayais 3, problems encountered in introducing tooling indicated that the bundle wrapper had dropped about 20 mm.

Additional examinations carried out showed that the supporting parts of this wrapper had broken (weld failure). After this incident, a specific inspection programme was initiated in 1995 on all steam generators.

Today, only two of the three steam generators at Blayais 3 are affected. The worst damage was repaired by constructing a new support frame for the bundle wrapper in the top of the steam generator.

The studies carried out to explain this phenomenon should be completed before the end of 1995.

Tube support plate degradations

This problem was discovered on two steam generators at Fessenheim 2 at the start of 1995. A part of the upper tube support plate had vanished and some twelve tubes were no longer supported in this area. The chemical cleaning carried out in 1992 on these steam generators is suspected to be the cause.

The results of the tests carried out on the other steam generators containing drilled carbon steel support plates, highlighted two other types of phenomena, mainly affecting the upper tube support plates.

Some of the plates seemed to be broken locally and ruptures appear to have been present since the steam generators in question first entered service.

Other plates, most recently at Gravelines 2, have been found to contain many damaged ligaments. Drilling anomalies made during manufacture of the plates could be one of the root cause of this problem.

Since the risk of vibration of tubes poorly supported by the damaged upper tube support plate cannot be discounted, the tubes concerned are plugged for the moment, pending the results of fuller analyses requested of the operator.

4 - TEST METHODS USED

The integrity of the steam generator tubes is periodically assessed using the eddy current method.

Wherever possible, all degradation, whether expected or not, are detected using a bobbin coil probe (BC) able to check the entire tube very rapidly in a single pass. If necessary, any damage indicated is characterised using a rotating pancake probe (MRPC), initially developed and used to measure the length of axial cracks at the roll transition zone.

Actually, these two types of probe have very different capabilities ; this leads in particular to the exclusive use of the rotating probe in order to find circumferential cracks.

Furthermore, mention should be made of other non-destructive testing methods which are used for different purposes:

- ultrasound testing which is still considered to be in the process of development on account of its unreliability ;
- video inspections, mainly used in the secondary side to detect loose parts and to examine abnormal phenomena detected using other methods ;
- helium leaktest checks carried out as part of the ten-yearly hydrostatic tests, to detect and plug tubes likely to display a detectable leak in the next cycle and which could lead to a forced outage.

Finally, many tubes have been removed and some will have to be in order to :

- contribute to the understanding of the new phenomena,
- determine the nature and characteristics of new degradation process,
- measure important parameters such as leak rate, burst strength, metallurgical and chemical properties of the tubes and of the deposits adhering either to the tubes or even to the tube support plate and the tube sheet,
- correlate non-destructive testing results with the real characteristics of the damage.

4.1 - Bobbin coil probe

Description of the probe

The probe used is of the axial symmetry differential type. It consists of:

- two identical coils placed very close to each other, and with an axis as near as possible identical to the tube axis during the inspection and with a diameter slightly less than the internal diameter of the tube to be checked,
- a device (permanent magnet or special coil) used to saturate the magnetic areas.

The two coils in the probe are powered by an AC generator which provides permanently a complex voltage consisting of three fundamental and different frequencies.

The two coils are inserted into a measuring system which is used to measure:

- differences between the apparent impedance of the two coils for the three fundamental frequencies (differential mode),
- variations in the apparent impedance of one of the coils for one of the fundamental frequencies (absolute mode).

One of the advantages of simultaneously using different frequencies is that this allows the elimination of certain unwanted signals (tube support plate, geometric background noise) by combining the signals obtained at the different frequencies.

Signal processing principles

The main parameter used to determine the origin (internal or external) and the nature (loss of material, deformation, sludge, conductive deposits etc.) is the signal phase. The phase can also be used to evaluate the depth of the flaws by using phase/depth correlation curves plotted by experimental means for each frequency and when possible for each specific type of damage.

The amplitude of the signals is of only secondary importance; it is used to determine the extent of the damage once the nature of the damage has been established. The amplitude values for the notation and plugging thresholds can differ greatly according to the nature of the flaws.

Finally, in the most complex or most recent cases, the signals are processed in detail in the impedance view for each frequency and for each mode.

Reference tubes

It is a piece of tube, known as the reference tube, of the same metallurgical grade as the tubes to be inspected, through which the probe passed to check each tube.

This reference tube contains several artificial flaws and geometrical particularities; this makes it possible in one operation and each time a tube is inspected to:

- adjust amplitudes and phases for both differential (including mixes) and absolute mode,
- verify the quality of the signals.

4.2 - Rotating pancake coil probe (MRPC)

Description of the probes

This type of probe is available in several different models, to make allowance not only for the internal diameter of the tubes, but also and above all of that part of the tube to be examined (free span or U-bend).

The tube is examined along a helical path with a pitch of 1 mm per revolution. The speed of rotation is about equal to 5 revolutions per second.

Free-Span and TSP elevation

The probes are not in contact with the tube and only their system of propulsion differs according to the area of the tube to be examined:

- worm-screw method for the base of the tube,
- translational and rotational motion imparted by a hose for the areas located above the tube support plate.

The sensitive part of the probe consists of two parallel coils arranged on ferrite rods 1 mm in diameter, placed at an angle of 45° to the generatrix of the tube. There is no device today to saturate the magnetic areas.

As with axial probes, the method used is multi-frequency and multi-mode.

U-bend

The probe is kept in contact with the inner wall of the tube. It contains only one measurement coil. The technique is still multi-frequency, but, obviously, only the absolute mode is available. An identification system is used to locate the signals along the azimuth.

Reference tubes

As with the axial probe, a reference tube is passed by the probe prior to each tube inspection. However, this tube contains only two man-made flaws: an axial through-wall notch and an external axial notch 80% through wall.

Signal processing principles

The purpose of the inspections, depending on the part of the tube being examined, is to :

- detect, size and locate crack-type flaws (both axial and circumferential),
- detect wear,
- detect external corrosion-type flaws,
- detect deformation,
- assess the presence and elevation of the sludge or deposits on the tubes.

Processing is greatly simplified by the use of a program which proposes a diagnosis of the flaw; this is then systematically checked by an analyst.

As a general rule, information is not provided on the depth of the cracks and whether they stem from the inside or the outside because the assessment of these indicators is not considered reliable enough.

Detection threshold

Appendix 6 summarises the type of flaw sought in each area of the tubes, the detection and characterisation facilities used and the value of the corresponding detection threshold.

5 - MONITORING AND SURVEILLANCE PROGRAMME

Risk prevention depends primarily on the design choices made during construction of the steam generators and thereafter on monitoring of changes in the state of the appliances over time.

In France, this surveillance is carried out at two complementary levels:

→ First of all, ongoing in-service monitoring which relies primarily on measurement of changes in the rate of leakage between the primary and the secondary system.

Despite the many through-wall axial cracks present in the roll transition zone, it was found that the leak rates during operation remained well below those estimated using theoretical models. Considerable caution must therefore be exercised when using the leak-before-break principle.

As regards corrosion of tubes at the intersection with the tube support plates, the effect of the plate and the deposits leads also in this case to a considerable reduction in leaks.

Furthermore, an analysis of experience feedback on tube ruptures which have arisen throughout the world has showed that the leak rate could change very rapidly and that steps had to be taken at a very early stage to prevent rupture.

In this context, it would therefore seem necessary to be able to detect the slightest change in a potentially serious leak very rapidly.

In France, although the technical specifications for operation permitted a leak rate of up to 70 l/h from a steam generator, the safety authority obliged the utility to control leaks and imposed much stricter limits, in the region of some litres an hour.

This, of course, required modification of the current two measurement methods:

- * one continuous measuring system with three different and independent measuring channels,
 - one nitrogen-16 monitor on each steam discharge line,
 - one radioactivity measurement device on the blowdown line of each steam generator,
 - one all-purpose measure per reactor placed on the air ejector of the condenser.
- * one discontinuous with three different types of periodic sampling system to cross-check the values obtained by the previous devices.

The rules applicable in France are currently being amended. Their underlying principle is summed up in Figures 1 and 2 for the two main types of steam generator considered.

The apparently complex nature of these rules stems from the usual change in leak rates observed on steam generators depending on the different operating phases of the reactor.

The following phenomena are generally observed :

- a considerable increase in the flow rate during a restart after refuelling. The gradient of the increase in the flow rate seems to be proportional to the rate of power escalation,
- a steady reduction in the leak rate until a relatively low plateau is reached,
- rapid changes during certain load variation transients,
- a steady increase at the end of the cycle, most probably the result of the drop in the boron content of the coolant.

An extrapolation of this approach leads us quite naturally to consider that steam generators with Alloy 690 tubes where no damage is expected in the medium term should be operated with a non detectable (close to zero) leak rate. In this case, any measured leak initiation must be considered suspect.

→Then on surveillance during outages, based on non-destructive eddy current testing :

- the axial probe along the entire length of the tubes inspected (this is the basic inspection for detecting further damage);
- the rotating probe in specific areas (transition zone, intersection with the tube support plates, small-radius bends, deformed areas etc.).

The inspection programme is revised each year to provide for:

- analysis of experience feedback on the previous year,
- improved understanding of the corrosion observed,
- uncertainty regarding the propagation rate of these phenomena.

Appendix 7 sets out the inspection method, the periodicity and the plugging criteria used to date for each different type of damage.

This programme is based on two fundamental principles:

- a sampling of at least 12,5 %, deduced from simple probabilistic studies, to indicate, with a high probability of success, any damage which would affect only about ten tubes on the steam generators inspected,
- a periodic inspection of all the tubes affected, and a sampling of the other tubes to monitor the development of the different types of damage and to plug any tubes with flaws which exceed the criteria adopted.

This surveillance during outages is supplemented once every ten years by:

- a helium leak test to detect any leaky tubes and to correlate these leaks with the defects observed during eddy current tests. Any through-wall defects not detected during eddy current tests could, however, be detected during this leak test,
- a hydrostatic test up to a pressure differential between the reactor coolant system and the secondary system of 207 bar. In this case, a measurement of the rate of leakage from the tube bundle could provide indications on what might happen during accident. Caution is, nonetheless, recommended because none of the correlations attempted so far between leak rates during operation and those measured during this proof-test have been conclusive. The theoretical studies are continuing to provide us with a better picture of the phenomena, but many phenomena which are difficult to quantify can affect these leak rates :

- . temperature,
- . the presence of deposits inside the cracks,
- . the sludge deposits can hide cracks in the transition zone or in the holes in the tube support plates.

The plugging criteria were drawn up from knowledge acquired on the problems encountered, especially:

- the morphology of the defects and the levels of performance of the non-destructive testing methods,

- the damage mechanisms,
- the development kinetics which determine the in-service inspection policy and hence the frequency of the inspections.

The principle consists in withdrawing a tube from service when the defect it contains might reach the critical size during the next cycle of operation.

To illustrate this principle, three examples can be described in detail:

a. For axial cracks in the roll transition zone, a rotating probe can be used to measure the crack length to within an accuracy range of about 1 mm.

The considerable amount of data acquired on this type of defect was used to determine, by statistical means, the increase in the length of these cracks with each cycle.

Under these conditions, it is possible to estimate the development of this defect over time and to predict the moment at which the defect could reach the theoretical critical size, calculated for accident conditions.

The plugging criterion can be deduced from this critical size by taking away the kinetic value observed in one cycle along with the accuracy of the measurement of the length of the crack using the rotating probe.

The plugging criterion for this type of flaw is therefore established as 13 mm for 900 MWe steam generator tubes.

More complex configurations, making allowance for combined defects, were studied and gave rise to certain variations on this plugging criterion.

It is obvious that any increase observed in the kinetics would lead to a downward revision of this plugging criterion.

Ongoing analysis of any advances in the knowledge acquired therefore remains indispensable.

b. For circumferential cracks initiated at the inner or outer wall, it is difficult to establish such a criterion for two main reasons:

- we still do not know how to measure the depth of the cracks. Tests conducted with ultrasonic sensors are not very conclusive,

- it is hence impossible to assess the propagation kinetics for these cracks. Unlike axial cracks which only develop in one direction, above 4 to 5 mm in length, since they become through-wall cracks, circumferential cracks can spread in both directions (in length and in depth); it has so far proved impossible to establish a relationship between these two directions of spreading.

Given these conditions, any tube containing such a crack is plugged as soon as the crack is detected.

Here again, poor knowledge of the damage mechanism and kinetics should incite us to attempt to detect these faults at the earliest possible stage and hence to have an inspection method which is well suited to this type of detection.

c. For corrosion developing in the secondary side, particularly at the tube support plates elevation, the task of establishing a plugging criterion has been complicated for two reasons:

- the complexity of the damage which can result in the combination of several axial or circumferential OD IGSCC and possibly IGSCC networks,
- the impossibility of obtaining a direct measurement of the extent of the damaged area, and hence the impossibility of directly determining the weakening of the tube for the damaged area.

A correlation therefore had to be made between the amplitude of the signal obtained with the axial probe, giving a good idea of the overall damage (loss of material), and the observations and tube destructive examinations carried out on a large number of removed tubes or on mock-ups representative of this damage.

In this case, the crack propagation kinetics can be measured using the variation in the amplitude of the eddy current signal.

The critical size was determined by considering the risk of circumferential cracks appearing in the damaged area beyond a tube thinning threshold equal to 50% of the thickness of the tube all over the entire circumference (360°).

The tests carried out on mock-ups showed that a tube affected by IGA to this depth over the entire circumference gives an axial probe amplitude of 6 V with the calibration used in France. This is the critical threshold which must not be attained.

A tube affected to a depth of only 25% over 360° gives a signal of 2 V.

Since axial IG ODSCC isolated cracks usually observed at TSP intersections display a higher in depth propagation rate compared to IGA, they tend to contribute substantially and increasing with time to the axial probe signal magnitude. This adds a certain margin relative to the real extent of the IGA. Indeed, burst tests carried out in tube inserted at the defect level in TSP intersection mock ups showed that cracks which did not extend beyond this plate remained stable under accident conditions.

However, with the upper tube support plates of steam generators, which can be deflected considerably in the event of a main steam line break, the risk of cracks being discovered is high. A new criterion should therefore be established, should corrosion be found in these areas. For the moment, this phenomenon generally occurs on the lower tube support plates.

6 - PREVENTIVE AND CORRECTIVE MEASURES

It should first be borne in mind that the ongoing analysis of experience feedback since the first damage was found in 1981, has led the operator, in conjunction with the manufacturer, to improve the design of new steam generators.

Appendix 8 sums up this change which successively included heat treatment of Alloy 600 (TT 600), the selection of stainless ferritic steel as a TSP material, replacement of drilled holes by quattrifoil broached hole, replacement of Alloy 600 with heat-treated Alloy 690 and the use of hydraulic expansion instead of kiss rolling.

Corrective or preventive measures were gradually implemented for steam generators in service in an attempt to resolve the problems found.

This section gives a brief description of the different measures developed and the conditions in which they were implemented.

6.1 - Preventive measures

Shot peening and rotopeening in the roll transition zone

These are mechanical processes applied to reduce stress on the inner surface of steam generator tubes at the roll transition zone, to limit the initiation of stress corrosion cracking on the primary side.

The methods consist in using surface hardening to compress the primary side ID superficial layer of the tubes by peening with miniature shot.

They were used from 1985 to 1988, in the hot leg of 900 MWe steam generators with Alloy 600 mill-annealed tubes, and thereafter on the other 900 and 1300 MWe steam generators in the population of reactors between 1989 and 1993.

The effectiveness of these processes has been the subject of many technical publications. Experience feedback has shown that it has a positive effect on crack initiation time, but that this effect diminishes rapidly as soon as the crack has been initiated.

Stress relief heat treatment for small radius bends

This process applies heat to reduce the residual manufacturing stress remaining in small-radius bends, which causes stress cracking in the primary coolant environment.

It involves raising the bent area of the tube to a temperature of over 700°C for a few minutes using a flexible heating tube containing an electric heating element.

In the case of steam generators containing bends which were not treated after bending operations, this heat treatment was carried out on site on the tubes in Rows 1 and 2.

The treatment was started in 1987 and completed in 1994.

Here again, the effectiveness of the process is clearest in the crack initiation phase. Since new flexible rotating probes have been in use, new cracks have been discovered several cycles after this heat treatment on site. In this case, it is believed that the crack had already been initiated before the treatment, but that the bobbin coil probe had been unable to pick it up.

Reducing the temperature of the reactor coolant system

Stress corrosion cracking in Alloy 600 is activated by heat, hence the temperature of the reactor coolant is an important factor. It was therefore decided to reduce the temperature of the reactor coolant by 4.2°C on 1300 MWe reactors, without any noticeable effect on the thermal output.

This modification was implemented from 1989 onwards. It is adopted on all reactors which have started up since 1990.

Sludge lancing on the tube sheet

Given the major role of these deposits in the development of corrosion on tubes in the confined environment of the secondary side, and also in view of their thermally induced possible effect on the stress field in the rolling transition zone, it was decided to clean the tubesheet. This was carried out with a pressurised water lance starting from the tube lane during each refuelling outage for the 900 and 1300 MWe units.

After discovery of the "denting" phenomenon, which appeared above the tube sheet at the bases of the tubes in certain 1300 MWe steam generators, a new tool, called CECIL, was used to locally eliminate the hard and very compact deposits which caused the deformation and circumferential cracking observed. It has been in use since 1991, and is applied every other cycle on steam generators of the 51 A type (900 MWe) and, since 1993, every third cycle on 1300 MWe steam generators.

For reasons associated with differences in design, this tool cannot be used on other steam generators. Research is being conducted into the possibility of miniaturising the technology

Chemical cleaning

Secondary corrosion is very often related to the aggressive nature of the secondary side environment in confined areas. The sludge built up on the tubesheet or occasionally on the tube support plates and deposit on tube free span contribute considerably to the rapid spread of this phenomenon.

Chemical cleaning is carried out to eliminate these deposits in areas it is difficult to reach using the aforementioned cleaning tools.

The process consists in two successive chemical formulations, one capable to dissolve iron oxides, and the other to oxidise and dissolve metallic copper.

Three chemical cleaning operations have been performed on steam generators in France, at Nogent 1 in 1989, at Saint-Alban 2 in 1990 and at Fessenheim 2 in 1992.

The use of chemical cleaning has been so far totally suspended in view of its high cost, the relative inefficiency of the process to clean the narrow crevices between the tubes and tube support plates which have become filled with deposits, and also because of corrosion control problems during this treatment on the steam generators at Fessenheim 2 which might have contributed to the observed degradation of the upper TSP.

Injection of boric acid into the secondary side

The chemical treatment is aiming at inhibiting intergranular corrosion which develops on the outer wall of steam generator tubes.

This is carried out in two stages, firstly soaking in view of allowing the high concentration of boric acid to migrate into the confined areas, followed by continuous injection which maintains the boron content between 5 and 10 ppm at the blow down of the steam generator.

The first injection took place in 1994 on reactor 4 at Bugey.

The reactors at Fessenheim 2, Bugey 2 and Fessenheim 1 were treated in 1995. Bugey 3 is to be treated in 1996.

A first review of the effectiveness of this process will be carried out on Bugey 4 in 1996. If the findings of this review are positive, the process could be implemented on all steam generators containing drilled tube support plates made of carbon steel.

Installation of damper cables

Following the tube rupture at Mihama 2, a survey was carried out to identify tubes which were poorly supported by anti-vibration bars.

The studies conducted after this discovery showed that the risk of vibrational instability on certain tubes could not be totally discounted, particularly on steam generators of types 51 A and M which contained drilled tube support plates (an aggravating factor).

These tubes will be fitted with an internal cable to damp vibrations. They are also withdrawn from service by plugging.

EDF has committed itself to treating all the tubes concerned (about 800 in all) within two years. Cable introduction operations began in early 1995.

At present, the unlikely fixing of the tubes in the upper TSP support plate, there are over the 51 BI and subsequent steam generators is anticipated and will be further checked in a coming investigation programme.

6.2 - Corrective measures

Plugging

This process consists in plugging the two ends of an affected tube, in the hot side and in the cold side.

Initially, the plugs used were of the Westinghouse type made of Alloy 600. After the rupture of a plug led to a tube rupture at the North Anna plant in the USA, new plugs were made from Alloy 690.

Alloy 600 plugs installed on steam generators were temporarily fitted with "plug in plug" devices plates to prevent the production of a "projectile" if the plug were to crack.

Under pressure from the safety authority, EDF has committed itself, since 1994, to a programme to replace all the Alloy 600 plugs in the hot side of steam generators. This involves some 3000 plugs.

Currently there are three types of plug that can be used:

- the Westinghouse type of mechanical plug,
- a modified version of the Combustion Engineering type plug,
- the Framatome removable plug.

EDF was asked to verify the state of mechanical plugs made from Alloy 600 and installed in the cold side of steam generators. These are not fitted with "plug in plug" devices.

Sleeving

This is an alternative to the plugging process and consists in placing a sleeve between the tube and the reactor coolant system at the roll transition zone containing the crack in order to make the tube leaktight once again.

The sleeve is made from Alloy 690 and is secured to the tube by mechanical or hydraulic expansion and then welded using the TIG or laser process; the stress in the upper joint is relieved by heating.

Experience in France has been deliberately limited in this area owing to difficulties encountered in performing in-service inspections of the weld.

Currently, only Tricastin 2 is fitted with about twelve sleeves. However, these steam generators are to be replaced in 1996.

In view of the difficulties of inspecting the weld area and also because of difficulties encountered recently in other countries, there are no plans to employ this process for the time being.

Steam generator replacement

Steam generator replacement is a major maintenance operation, in terms of its cost and the time it takes; it is a means of installing new steam generators of a new design.

In France, this operation has already been carried out on three 900 MWe reactors: Dampierre 1 in 1990, Bugey 5 in 1993 and Gravelines 1 in 1994. It is currently being performed this year at the Saint-Laurent B1 and Dampierre 3 reactors.

Given the large number of inspections to perform on damaged steam generators, EDF has committed itself to replace steam generators at a rate of at least one reactor per year. As a consequence, it has placed the necessary orders to procure these new steam generators in due time.

It is EDF that decided whether or not to replace steam generators on the basis of mainly financial and technical criteria. The safety authority limits its involvement to requiring that tubes which exceed the plugging criteria be withdrawn from service. The replacement threshold is reached at a mean plugging ratio close to 12.5%.

7 - CONCLUSION

Since 1980, the instances of degradation observed on French pressurised water reactor steam generator tubes have increased steadily.

Although improvements have been made in the design of new steam generators, the lengthy process of detecting and surveying all of the instances of damage found has led EDF to implement a comprehensive programme of inspections and major, costly preventive and corrective measures.

Additional efforts must still be made, however:

- to understand all the damage mechanisms detected in order to assess their progression,
- to anticipate the progression of these phenomena by developing more powerful inspection resources and suitable maintenance processes.

The French safety authority pays particular attention to advances in knowledge available in these areas and does its utmost to ensure that the monitoring, surveillance and maintenance strategies proposed by the operator are adequate.

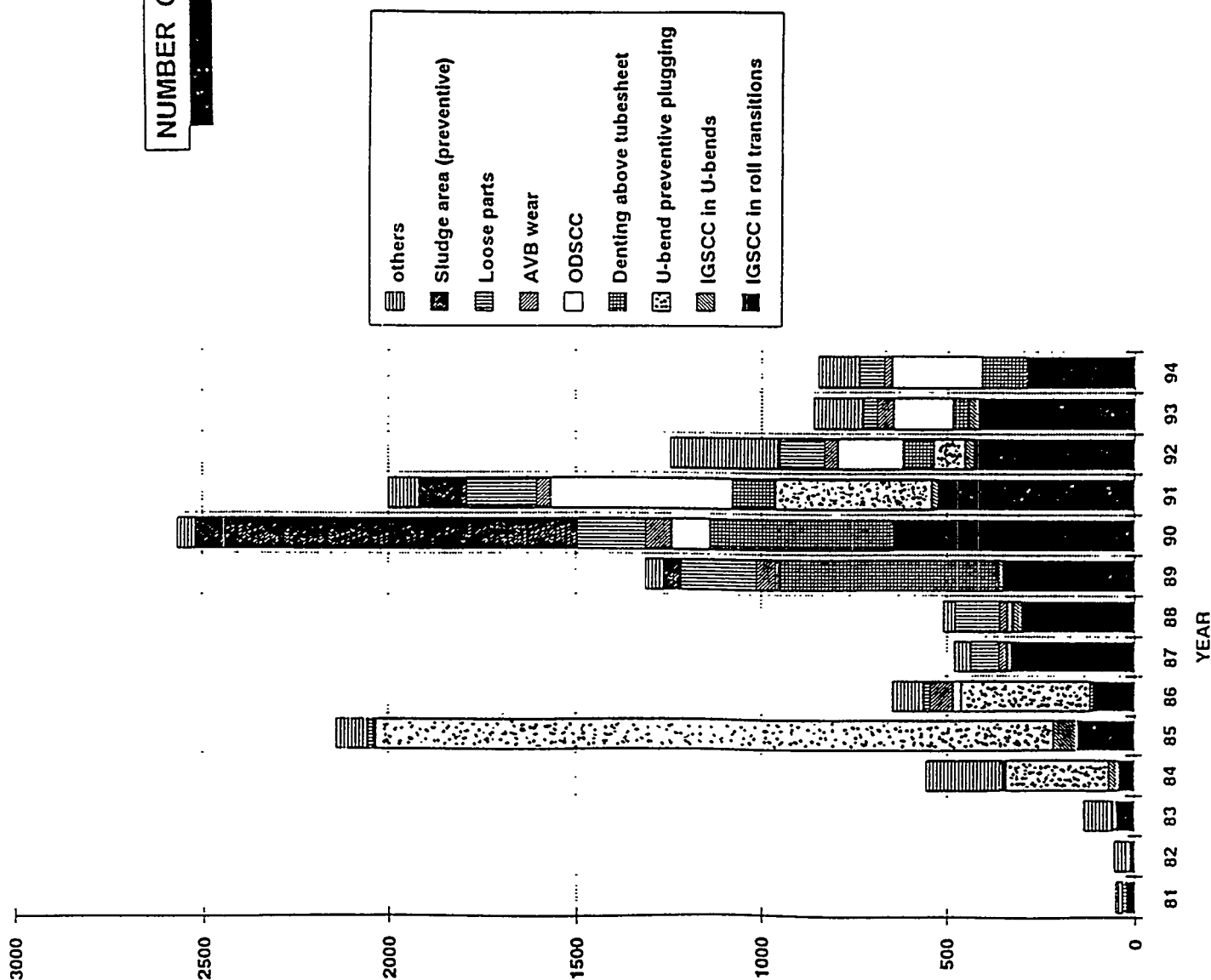
The programme to replace all the older generation steam generators gradually and as a preventive measure marks a step towards achieving greater safety.

In the meanwhile, the ageing of the installations is reason for continued vigilance.

This vigilance and the responsiveness of the operator have certainly contributed to the fact that France has so far not had any steam generator tube rupture accidents.

APPENDIX 1

NUMBER OF PLUGGED TUBES PER YEAR



APPENDIX 2

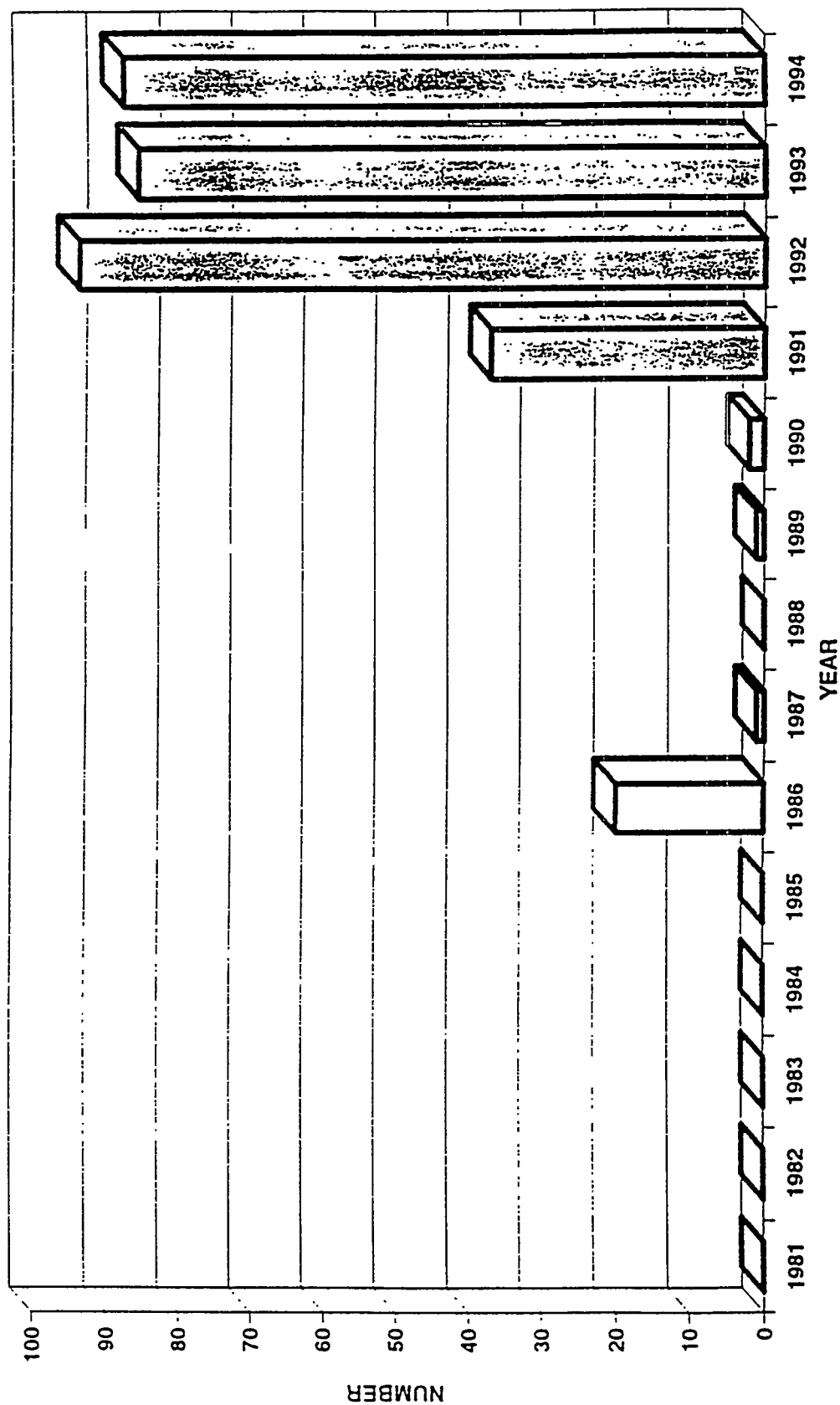
CIRCUMFERENTIAL CRACKS IN EDF STEAM GENERATOR TUBES

NUMBER OF TUBES PLUG WITH CIRCUMFERENTIAL CRACKS

In sludge pile			Out of sludge pile	With roll transition anomalies	Total
With denting	Unknown denting	Without denting			
162	77	80	239	12	570

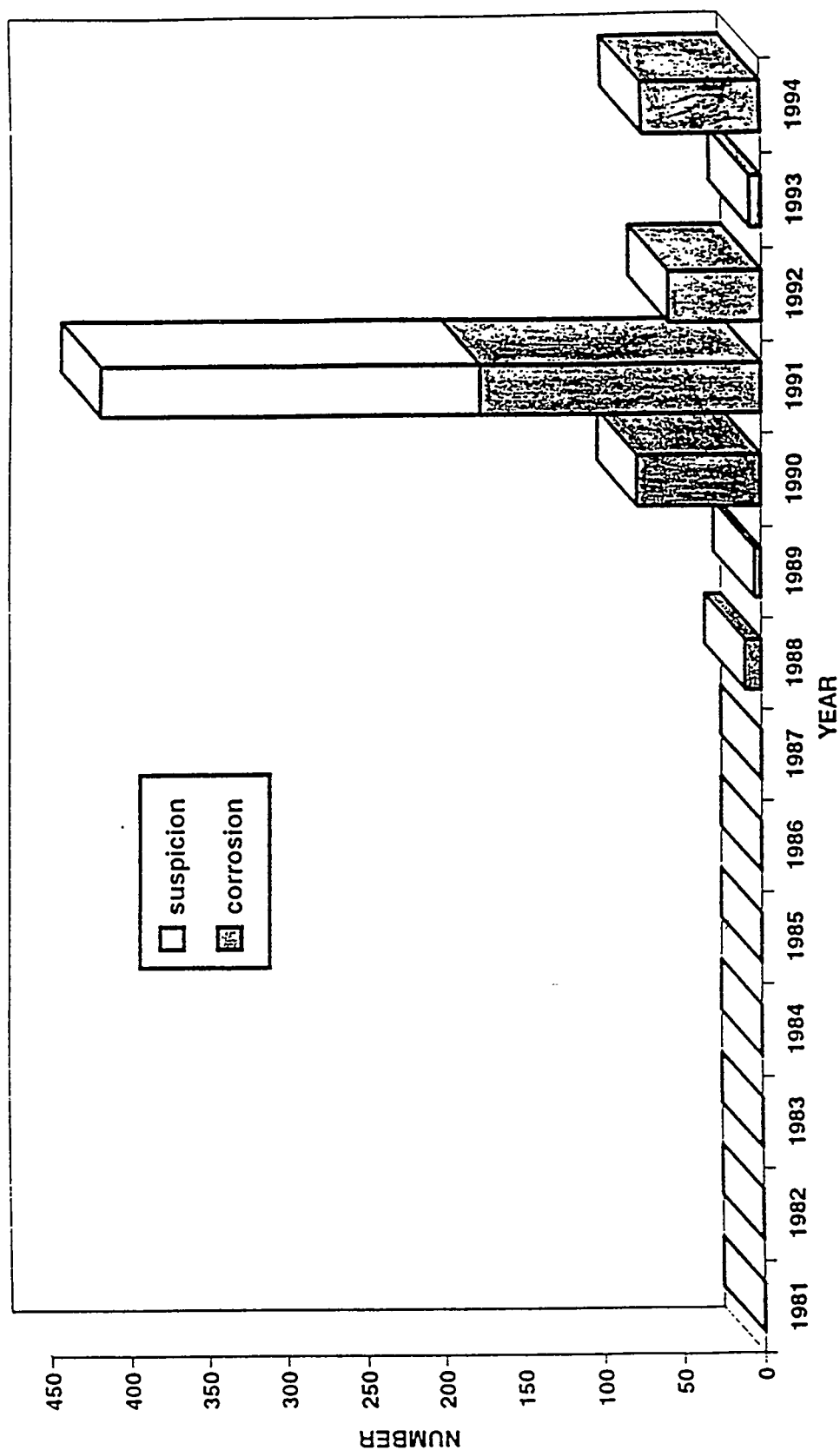
Total number of EDF SG tubes : about 800, 000

PLUGGING DUE TO IGA/SCC AT TSP ELEVATION

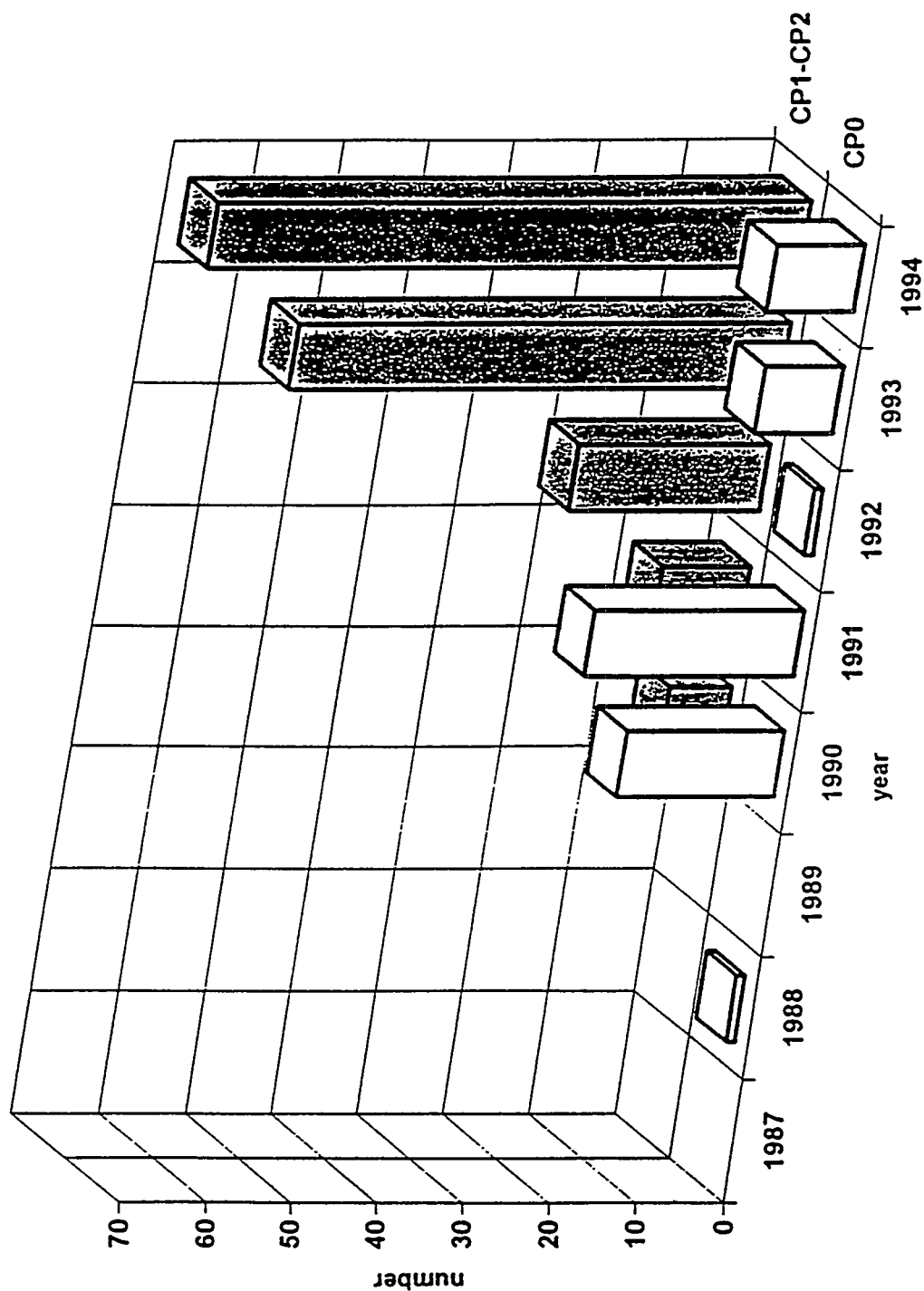


APPENDIX 4

PLUGGING DUE TO IGA/SCC ABOVE TUBESHEET



PLUGGING DUE TO OD CIRCUMFERENTIAL CRACKING



**NON DESTRUCTIVE EXAMINATION
DETECTION LEVEL**

APPENDIX 6

LOCATION	DEFECT	DETECTION	CHARACTERISATION	DETECTION THRESHOLD
Roll transition zone	AC PWSCC	BC, RPC	RPC	$l \geq 5 \text{ mm}$
	CC	RPC	RPC	$d \geq 600 \text{ microns (} t/2 \text{)}$ angular extension $\geq 50^\circ$
	Striction	RPC	RPC	$\Delta r \geq 70 \text{ to } 80 \text{ microns}$
Secondary side of the tubesheet	Bulging	BC, RPC	BC, RPC	$\Delta r \geq 150 \text{ microns}$
Loose parts	Wear	BC	BC	$\Delta t \geq 20\%$
Tube support plate	Denting, Deformation	BC	RPC	$\Delta \phi \geq 150 \text{ microns}$
Drilled tube support plates	IGA, IGSCC, AC	BC, RPC	RPC	$t/2$, sometimes better
Sludge deposit on tubesheet, tube support plate	IGSCC, AC	BC	RPC	
U-Bends	AC	BC	RPC	$l \geq 4 \text{ mm through-wall}$
	Wear	UB-RPC	UB-RPC	$l > 6 \text{ mm, } d \geq t/2$
	Wear	BC	BC	
Anti-vibration bars	Wear	BC	BC	$\Delta t \geq 20\%$

CC Circumferential Crack
 AC Axial Crack
 IGA Intergranular Attack
 t thickness
 IGSCC Intergranular Stress Corrosion Cracking
 PWSCC Primary Water Stress Corrosion Cracking
 BC Bobbin Coil probe
 d depth
 RPC Rotating Pancake Coil probe
 UB-RPC U-Bend Rotating Pancake Coil probe
 l length
 r radius

INSPECTIONS PERFORMED AT FRENCH PLANTS
SPECIFIC PROGRAM (2)

APPENDIX 7 BIS

Type of degradation	Non destructive testing (*)	Frequencies	Criterial size	Kinetics	Plugging criteria	Comments
Tube to tube support plate inter-sections - ODS/CC + IGA	BC	100 % every outage	6 V (50 % of tickness)	2 V	2 V 1 V	For TSP number 7 and 8
TSP with drilled holes						
Small radius U-Bend	FRPC	100 % every outage Row 1 or 2	?	?	All tubes affected	
Wear by loose parts	BC	100 % every outage on peripheric tubes	75 % of tickness	15 % max/cycle	40 % All indications	When loose part can't be removed

(*) BC = Bobin Coil MRPC = Motor Rotating Pancake Coil FRPC = Flexible Rotating Pancake Coil

**INSPECTIONS PERFORMED AT FRENCH PLANTS
SPECIFIC PROGRAM (1)**

APPENDIX 7

Type of degradation	Non destructive testing (*)	Frequencies	Criterial size	Kinetics	Plugging criteria	Comments
Roll transition zone - IGSCC						
• Mill-annealed tubes without kiss-rolling	MRPC	100 % every outage	(1) 16,5 mm (2) 360° - 80 % t 270° - through wall	2,5 mm/cycle ?	13 mm All tubes affected	(1) axial cracks (2) circumferential cracks
• Tubes with kiss-rolling - mill-annealed	MRPC	100 % every outage	(1) 16,5 mm (2) 360° - 80 % t 280° - through wall	2,5 mm/cycle ?	13 mm All tubes	
- heat treated or Sandvik mill-annealed	BC + MRPC	25 % every outage 100 % sludge pile every outage (3)	(1) 16,5 mm (900 MWe) (1) 14,5 mm (1300 MWe) (2) 360° - 80 % t 280° - through wall	2,5 mm/cycle Lower than 2,5 mm/cycle ?	13 mm Length ≤ 9 mm in 1995 All tubes affected	(3) after 60,000 operating hours
Roll transition zone ODSCC						
- mill-annealed tubes	BC + MRPC	100 % every other outage	860 mV (70 % of tickness)	250 mV	500 mV 200 mV	If longitudinal crack close to 200 mV the ODSCC zone
- heat treated tubes	BC	25 % every outage				

**INSPECTIONS PERFORMED AT FRENCH PLANTS
BASIC PROGRAM**

APPENDIX 7 TER

	NON DESTRUCTIVE TESTING FOR DETECTION	FREQUENCIES	PLUGGING CRITERIA
SGs with Inconel 600 mill-annealed tubes	Bobin Coil	100 % at every other outage	See table for the specific program
SGs with Inconel 600 heat treated tubes	Bobin Coil	25 % at every outage	
SGs with Inconel 690 heat treated tubes	Bobin Coil	12,5 % on one SG 900 MW at every outage 12,5 % on two SG 1300 MW at every outage	

CHANGES IN STEAM GENERATOR DESIGN

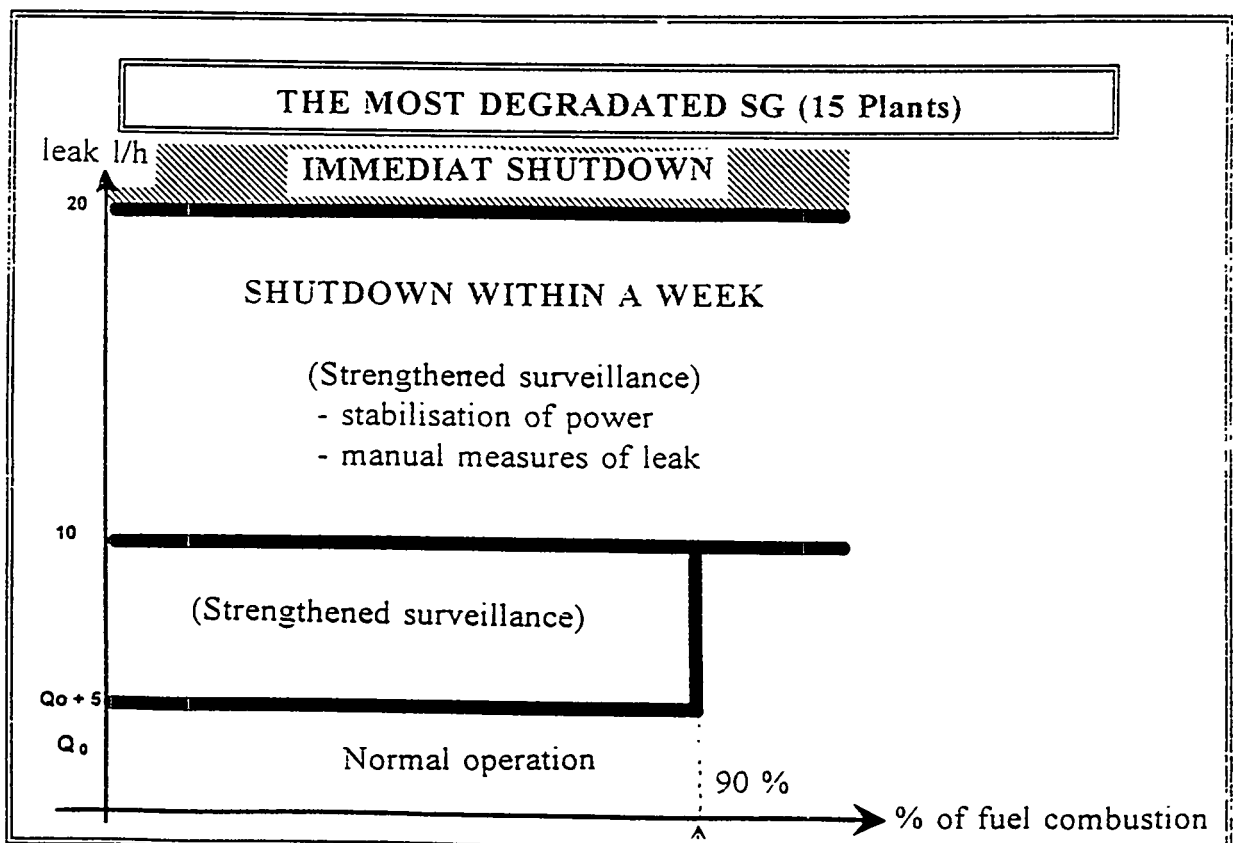
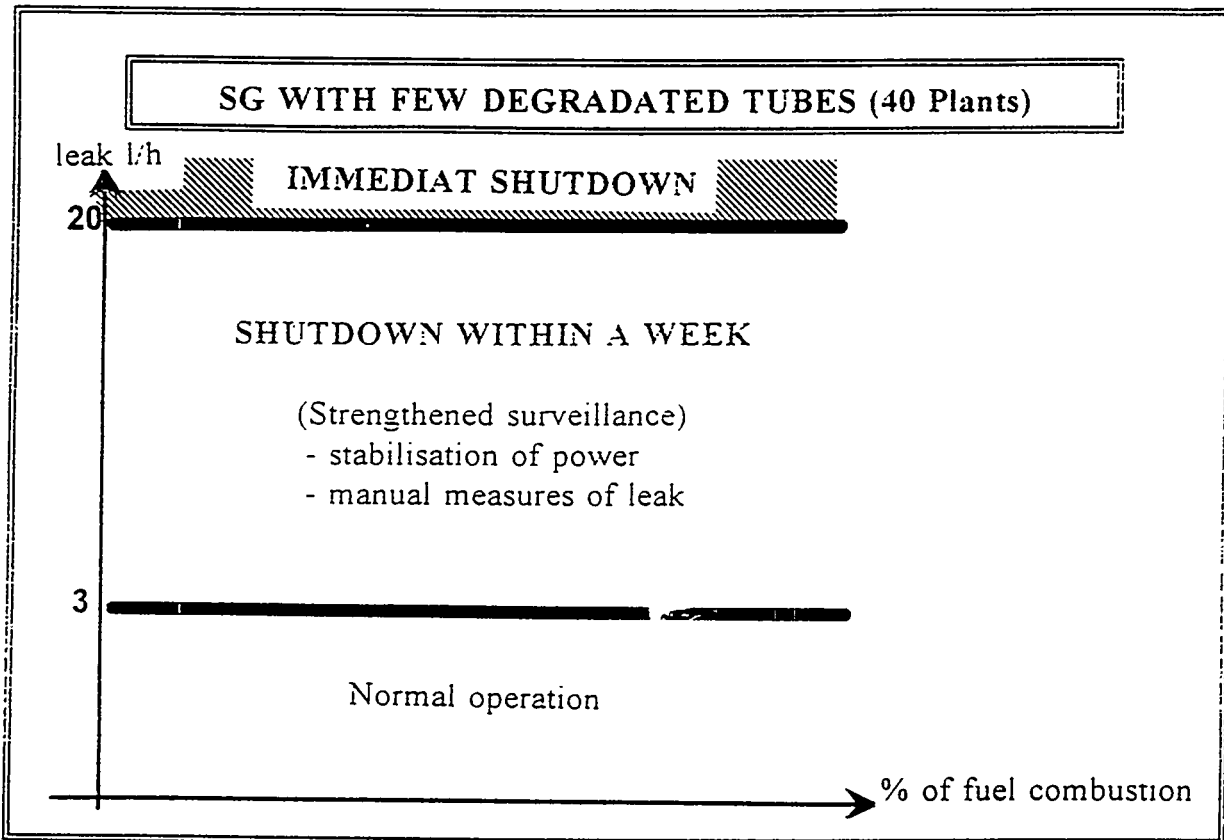
Exp	Int R	KISS ROLLING						ROLLING	
DRILLED		QUATTRIFOIL BROACHED holes						GEOMETRY OF TUBE SUPPORT PLATES	
A285 CARBON STEEL			STAINLESS FERRITIC STEEL					MATERIAL OF TUBE SUPPORT PLATES	
600 MA ALLOY (as received)					MA/TT	TT600	TT690	MATERIAL OF STEAM GENERATOR TUBES AND REACTORS AFFECTED	
FES1	FES2 BUG 2.3.4	TRI 1.2.3.4 GRA. 2.3.4 DAM 2.4 BLA.1 SLB.2	BLA 2.3 CHB1	CHB2	BLA4	BUG5 GRA5.6 CRU1.2.3.4 CHB3.4 PAL1.2.3.4 FLA1.2 SAL1.2 CAT1.2.3.4 BEL1.2 NOG1.2 PEN1 GOL1	PEN2 GOL2 CHZ BI.2 CIV1 CIV2 DAM1 GRA1 SLB1 DAM3		
DENTING									MAIN RISKS OF CORROSION DEPENDING ON STEAM GENERATOR DESIGN
YES ESPECIALLY SEA WATER		+	NO						
		-							
INTERGRANULAR CORROSION IN SECONDARY SIDE OF TUBE SUPPORT PLATE									
YES		+ or -			LOW	WERY LOW			
STRESS CRACKING IN PRIMARY SIDE (NO POLLUTANTS)									
YES				+or -	+ or -	NO			

Key : Exp. = Explosion Int R = Integral Rolling

Distribution of power plants units labelled by their trigram as a function of various design or fabrication features.

Figure 1

GENERAL SHUTDOWN CIRITERIA



SHUTDOWN CRITERIA LEAK INCREASING

