NEA/CSNI - UNIPEDE

SPECIALIST MEETING ON
OPERATING EXPERIENCE WITH

STEAM GENERATORS

BRUSSELS
BELGIUM
16-20 SEPTEMBER
1991

HOSTED BY THE
BELGIAN GOVERNMENT
SESSION 1  1  OPENING SESSION
SESSION 2  2  OPERATING EXPERIENCE - PART 1
SESSION 3  3  OPERATING EXPERIENCE - PART 2
SESSION 4  4  STRUCTURAL INTEGRITY AND LICENSING ISSUES
SESSION 5  5  ANALYSIS AND PREDICTION OF DEGRADATION MECHANISMS
SESSION 6  6  INSERVICE INSPECTION METHODS
SESSION 7  7  PREVENTIVE & CORRECTIVE ACTIONS - PART 1
SESSION 8  8  PREVENTIVE & CORRECTIVE ACTIONS - PART 2
SESSION 9  9  REPLACEMENT OF STEAM GENERATORS
NEA/CSNI-UNIPEDE
SPECIALIST MEETING ON
OPERATING EXPERIENCE WITH
STEAM GENERATORS

Brussels, Belgium
16th - 20th September, 1991

FINAL
PROGRAMME
MONDAY, SEPTEMBER 16th

08:30 - 09:30 Registration

09:30 - SESSION 1 - OPENING SESSION

Chairmen: P. Stallaert, Inspector General, SSTIN/DTVK, BELGIUM
          A. Jaumotte, Chairman of the Board, AVN, BELGIUM

1.1. Opening of the meeting by L. Vandenbrande,
     Minister of Labour and Employment.

1.2. Welcome address by P. Govaerts,
     Director of AIB-Vinçotte Nuclear (AVN).

1.3. Introductory remarks by J. Reig,
     NEA/CSNI representative.

1.4. Introductory remarks by L. Bertron,
     UNIPEDE representative.

10:30 - COFFEE

11:00 - SESSION 2 - OPERATING EXPERIENCE - PART 1

Chairman: L. Aye, EDF, FRANCE

2.1 OPERATING EXPERIENCE WITH STEAM GENERATORS

   A. Becquaert, Belgatom/Electrabel, Brussels, BELGIUM
   A. Timmermans, Belgatom/Electrabel, Brussels, BELGIUM
   P. Coenraets, Belgatom/Electrabel, Brussels, BELGIUM
   K. de Ranter, Belgatom/Electrabel, Brussels, BELGIUM
   C. Leblois, Belgatom/Tractebel, Brussels, BELGIUM

2.2 RECENT OPERATING EXPERIENCE WITH STEAM GENERATORS IN JAPAN

   S. Yashima, Japan Power Engineering and Inspection Co., Tokyo, JAPAN
2.3 STATUS OF STEAM GENERATORS IN SPAIN

J.M. Zamarron, Central Nuclear de Almaraz, Madrid, SPAIN

12:15 - LUNCH

14:00 - SESSION 2 - OPERATING EXPERIENCE - PART 1 (Cont’d)

2.4 LA SURETÉ D'EXPLOITATION DES GENERATEURS DE VAPEUR EN FRANCE - PREVENTION DES RISQUES ASSOCIES AU VIEILLISSEMENT EN SERVICE - LA POSITION DE L'AUTORITE DE SURETÉ FRANCAISE

P. Brossier, BCCN, Dijon, FRANCE
B. de l'Epinois, DSIN, Paris, FRANCE
J.L. Pierrey, CEA/IPSN, Fontenay-aux-Roses, FRANCE

2.5 OPERATING EXPERIENCE WITH HORIZONTAL TYPE STEAM GENERATORS OF THE VVER-440 PWR UNITS

S. Nagy, PAKS Nuclear Power Plant, HUNGARY

2.6 OPERATING EXPERIENCE WITH WESTINGHOUSE MODEL F STEAM GENERATORS

D.D. Malinowski, Westinghouse NSD, Pittsburgh, USA
R.M. Wilson, Westinghouse NSD, Pittsburgh, USA
M.J. Wooten, Westinghouse NSD, Pittsburgh, USA

15:15 - Chairman's closing remarks

15:20 - COFFEE

15:45 - SESSION 3 - OPERATING EXPERIENCE - PART 2

Chairman: P. De Meester, KUL, BELGIUM

3.1 OPERATING EXPERIENCE WITH STEAM GENERATORS

R. Bouecke, Siemens/KWU, Erlangen, GERMANY
3.2 AVT USING MORPHOLINE ALONE: A UNIQUE EXPERIENCE AT A CANDU-PHW PLANT IN CANADA

R. Gilbert, Institut de Recherche d'Hydro-Québec, Varennes, Québec, CANADA
Y. Dünder, Hydro-Québec, Centrale nucléaire Gentilly 2, Bécancour, Québec, CANADA
A. Marchand, Hydro-Québec, Centrale nucléaire Gentilly 2, Bécancour, Québec, CANADA

3.3 CHEMICAL CLEANING AS A MEASURE TO IMPROVE STEAM GENERATOR PERFORMANCE OF PWR-PLANTS

S. Oder, Siemens/KWU, Erlangen, GERMANY
K. Kuhnke, Siemens/KWU, Erlangen, GERMANY

3.4 OPERATING EXPERIENCE WITH STEAM GENERATOR WATER CHEMISTRY IN JAPANESE PWR PLANTS

K. Onimura, Mitsubishi Heavy Industries Ltd., Kobe, JAPAN
T. Hattori, Mitsubishi Heavy Industries Ltd., Kobe, JAPAN

3.5 OPERATING EXPERIENCE WITH STEAM GENERATORS IN KORI UNIT 1

Ul-Seong Hwang, Kori Nuclear Power Plant, KOREA

17:50 - Chairman's closing remarks

18:00 - 20:00 - C O C K T A I L at the "Palais des Congrès"

TUESDAY, SEPTEMBER 17th

09:00 - SESSION 4 - STRUCTURAL INTEGRITY AND LICENSING ISSUES

Chairman: J.L. Pierrey, CEA/IPSN, FRANCE

4.1 PWR STEAM GENERATOR TUBE AND TUBE SUPPORT PLATE PLUGGING CRITERIA

B. Cochet, Framatome, Paris La Défense, FRANCE
J. Engström, Vattenfall-Energisystem AB, Vallingby, SWEDEN
B. Flesch, EDF-SPT, Paris La Défense, FRANCE
4.2 LICENSING BASES FOR STEAM GENERATOR TUBE INTEGRITY AND RECENT OPERATING EXPERIENCE

B.D. Liaw, United States Nuclear Regulatory Commission, Washington D.C., USA
Emmet L. Murphy, United States Nuclear Regulatory Commission, Washington D.C., USA

4.3 TUBE PLUGGING CRITERIA FOR AXIAL AND CIRCUMFERENTIAL CRACKS IN THE TUBESHEET AREA

J. Van Vyve, Belgatom/Tractebel, Brussels, BELGIUM
P. Hernaisteen, Belgatom/Laborelec, Brussels, BELGIUM

4.4 SWEDISH EXPERIENCES OF A REVISED PLUGGING CRITERION FOR STEAM GENERATOR TUBES - A REGULATORY VIEW

J.E. Andersson, Swedish Nuclear Power Inspectorate, Stockholm, SWEDEN
P. Bystedt, Swedish Nuclear Power Inspectorate, Stockholm, SWEDEN
G. Hedner, Swedish Nuclear Power Inspectorate, Stockholm, SWEDEN
L. Skånberg, Swedish Nuclear Power Inspectorate, Stockholm, SWEDEN

10:40 - COFFEE

11:10 - 4.5 CSN REGULATORY VIEW ON STEAM GENERATOR TUBE DEGRADATION IN SPAIN

A. Esteban, Consejo de Seguridad Nuclear, Madrid, SPAIN
C. Mendoza, Consejo de Seguridad Nuclear, Madrid, SPAIN
J.F. Casillas, Consejo de Seguridad Nuclear, Madrid, SPAIN

4.6 SAFETY SIGNIFICANCE OF STEAM GENERATOR TUBE DEGRADATION MECHANISMS

G. Roussel, AIB-Vincotte Nuclear (AVN), Brussels, BELGIUM
P. Mignot, AIB-Vincotte Nuclear (AVN), Brussels, BELGIUM

4.7 STEAM GENERATOR TUBES RUPTURE PROBABILITY ESTIMATION-STUDY OF THE AXIALLY CRACKED TUBE CASE

B. Mavko, "Jožef Stefan" Institute, Ljubljana, Slovenia, YUGOSLAVIA
L. Cizelj, "Jožef Stefan" Institute, Ljubljana, Slovenia, YUGOSLAVIA
G. Roussel, AIB-Vincotte Nuclear (AVN), Brussels, BELGIUM

12:25 - Chairman’s closing remarks

12:30 - LUNCH
14:00  -  SESSION 5  -  ANALYSIS AND PREDICTION OF DEGRADATION MECHANISMS

Chairman:  J.M. Figueras, CSN, SPAIN

5.1 STUDY OF THE EFFECT OF MAINTENANCE ON THE SAFETY OF A MECHANICAL SYSTEM SUBJECT TO AGING - APPLICATIONS TO THE DEGRADATIONS OF STEAM GENERATOR TUBES

D. Dussarté, CEA/IPSN, Fontenay-aux-Roses, FRANCE

5.2 PREDICTION MODELS FOR THE PWSCC DEGRADATION PROCESS IN TUBE ROLL TRANSITIONS

P. Hernalsteenn, Belgatom/Laborelec, Brussels, BELGIUM

5.3 SAMPLING INSPECTION SCHEMES AND STEAM GENERATOR TUBE RUPTURE PROBABILITY

L. Cizelj, "Jožef Stefan" Institute, Ljubljana, Slovenia, YUGOSLAVIA
B. Mavko, "Jožef Stefan" Institute, Ljubljana, Slovenia, YUGOSLAVIA

5.4 PROBABILISTIC FRACTURE MECHANICS CODE FOR PWR STEAM GENERATOR TUBE MAINTENANCE

B. Granger, EDF - Direction de l'Equipement, FRANCE
P. Pitner, EDF - Direction des Etudes et Recherches, FRANCE
T. Riffard, EDF - Direction des Etudes et Recherches, FRANCE

15:40  -  COFFEE

16:00  -  5.5 THE EFFICIENCY OF PEENING ON INITIATION AND PROPAGATION OF PWSCC IN ROLL TRANSITION

J. Stubble, Belgatom/Laborelec, Brussels, BELGIUM
P. Hernalsteenn, Belgatom/Laborelec, Brussels, BELGIUM

5.6 AN ANALYSIS OF PRIMARY WATER STRESS CORROSION CRACKING IN PWR STEAM GENERATORS

P.M. Scott, Framatome, Paris La Défense, FRANCE

16:50  -  Chairman's closing remarks

16:55  -  BREAK
17:00 - PANEL 1 - OBSERVED DEGRADATION MECHANISMS AND LICENSING POSITIONS

Chairman: M. Hada, Nupec, JAPAN

Panelists: B.D. Liaw, NRC, USA
          D. Quéniant, CEA, FRANCE
          G. Roussel, AVN, BELGIUM
          G. Bollini, Tecnatom, SPAIN
          H. Schulz, GRS, GERMANY

In addition to discussing the main theme of the panel, supplemented by introductory statements by the panelists, the panel discussions are also intended to provide an opportunity for the audience to pose supplementary questions to the authors of related papers.

17:50 - Summing up by Panel Chairman

WEDNESDAY, SEPTEMBER 18th

09:00 - SESSION 6 - INSERVICE INSPECTION METHODS

Chairman: P. Hernalsteen, Laborelec, BELGIUM

6.1 ROUND ROBIN TESTS OF THE PISC III PROGRAMME ON DEFECTIVE STEAM GENERATORS TUBES

C. Birac, CEA/IPSN, Fontenay-aux-Roses, FRANCE
H. Herkenrath, CEC, Ispra, ITALY
S. Crutzen, CEC, Ispra, ITALY
Y. Miyake, Mitsubishi Heavy Industries Ltd., Kobe, JAPAN
G. Maciga, ENEL, Piacenza, ITALY

6.2 REGULATORY REQUIREMENTS ON INSERVICE INSPECTION FOR STEAM GENERATORS IN JAPAN

S. Yashima, Japan Power Engineering and Inspection Co., Tokyo, JAPAN

6.3 EDDY CURRENT INSPECTION METHODOLOGY

D. Dobbeni, Belgatom/Laborelec, Brussels, BELGIUM
6.4 AUTOMATIC EDDY CURRENT SIGNAL EVALUATION ENVIRONMENT (ENEAS)

J. Guerra, Tecnatom, Madrid, SPAIN
J. Vazques, Tecnatom, Madrid, SPAIN

10:40 - COFFEE

11:10 - 6.5 SPECIFIC ULTRASONIC INSPECTION METHODS FOR STEAM GENERATOR TUBES

F. Bodson, Framatome, Saint Marcel, FRANCE
B. Libin, Framatome, Paris La Défense, FRANCE
A. Thomas, Framatome, Lyon, FRANCE

6.6 ULTRASONIC INSPECTION METHODOLOGY

D. Dobbeni, Belgatom/Laborelec, Brussels, BELGIUM

6.7 ROSA III, A THIRD GENERATION STEAM GENERATOR SERVICE ROBOT TARGETED AT REDUCING STEAM GENERATOR MAINTENANCE EXPOSURE

P.J. Boone, Westinghouse NSD, Pittsburgh, USA

12:25 - Chairman's closing remarks

12:30 - LUNCH

14:00 - SESSION 7 - PREVENTIVE & CORRECTIVE ACTIONS - PART 1

Chairman: B.D. Liaw, NRC, USA

7.1 THE BELGIAN EXPERIENCE OF SG CHEMICAL CLEANING

R. Roofthooft, Belgatom/Laborelec, Brussels, BELGIUM
P. Havard, Belgatom/Electrabel, Brussels, BELGIUM
J. Maquinay, Belgatom/Electrabel, Brussels, BELGIUM
J. Brognez, Belgatom/Electrabel, Brussels, BELGIUM

7.2 CHEMICAL CLEANING FOR STEAM GENERATOR SECONDARY SIDE IN JAPANESE PWRS

S. Kimura, Kansai Electric Power Co. Inc., Osaka, JAPAN
7.3 SPANISH RESEARCH ACTIVITIES ON STEAM GENERATOR TUBES DEGRADATION

D. Gomez Briceno, Ciemat, Madrid, SPAIN
A.ME. Lancha Hernandez, Ciemat, Madrid, SPAIN
M.E.L. Castano Marin, Ciemat, Madrid, SPAIN

15:15 - COFFEE

15:45 - 7.4 AUTOMATIC MEASURES TO COPE WITH STEAM GENERATOR TUBE RUPTURES

K. Kotthoff, GRS, Cologne, GERMANY
A. Schütte, RWE Energie AG, Biblis, GERMANY

7.5 STEAM GENERATOR LIFE TIME EXTENSION BY MAINTENANCE AND REPAIR

H.F. Förch, ABB Reaktor GmbH, Mannheim, GERMANY
F. Pütz, ABB Reaktor GmbH, Mannheim, GERMANY

16:35 - Chairman's closing remarks

THURSDAY, SEPTEMBER 19th

09:00 - SESSION 8 - PREVENTIVE & CORRECTIVE ACTIONS - PART 2

Chairman: K. Kotthoff, GRS, GERMANY

8.1 CORRECTIVE AND PREVENTIVE MAINTENANCE TECHNIQUES FOR DEGRADATION OF STEAM GENERATOR TUBING

O. Takaba, Mitsubishi Heavy Industries Ltd., Kobe, JAPAN

8.2 STEAM GENERATOR FPAMATOME REMOVABLE PLUG

G. Morel, Framatome - Nuclear Services, FRANCE
J.C. Mounet, Framatome - Nuclear Services, FRANCE
J.P. Billoue, Framatome - Nuclear Eng. Primary Equip. Div., FRANCE
G. Poudroux, Framatome - Nuclear Eng. Primary Equip. Div., FRANCE
8.3 EXPERIENCE IN THE USE OF THE NICKEL PLATING PROCESS AS A PREVENTIVE OR A CORRECTIVE MEASURE

R. Houben, Belgatom/Electrabel, Brussels, BELGIUM
C. Verbeeck, Belgatom/Electrabel, Brussels, BELGIUM
C. Schinazi, Belgatom/Tractebel, Brussels, BELGIUM
J. Stubbé, Belgatom/Laborelec, Brussels, BELGIUM
C. Laire, Belgatom/Laborelec, Brussels, BELGIUM

8.4 PROGRAM FOR STEAM GENERATOR TUBING RELIABILITY TESTS IN JAPAN

F. Ishizuka, Japan Power Engineering and Inspection Co., Tokyo, JAPAN

10:40 - COFFEE

11:10 - 8.5 STEAM GENERATOR TUBE LASER SLEEVING

M. Batistoni, Framatome, FRANCE

8.6 DEVELOPMENT OF A STEAM GENERATOR SLEEVING SYSTEM USING FIBER OPTIC TRANSMISSION OF LASER LIGHT

T.R. Wagner, Westinghouse NSD, Pittsburgh, USA
L. Van Hulle, Westinghouse E.S.C., Nivelles, BELGIUM

8.7 QUALIFICATION AND FIELD EXPERIENCE OF SLEEVING REPAIR TECHNIQUES

J. Stubbé, Belgatom/Laborelec, Brussels, BELGIUM
J. Berthe, Belgatom/Tractebel, Brussels, BELGIUM
C. Verbeeck, Belgatom/Electrabel, Brussels, BELGIUM

12:25 - Chairman’s closing remarks

12:30 - LUNCH

14:00 - SESSION 9 - REPLACEMENT OF STEAM GENERATORS

Chairman: P. MIGNOT, AVN, BELGIUM

9.1 INFLUENCE OF MANUFACTURING PROCESSES ON STEAM GENERATORS BEHAVIOUR

J. Soberon, Equipos Nucleares, SPAIN
P. Veron, Equipos Nucleares, SPAIN
9.2 THE DAMIPIERRE 1 SGR

C. Verges, Framatome - Nuclear Services, Saint-Marcel, FRANCE

9.3 EXCHANGE OF STEAM GENERATORS IN PRESSURIZED WATER REACTOR POWER PLANTS

C. Hillrichs, Siemens/KWU, Erlangen, GERMANY

9.4 STEAM GENERATOR REPLACEMENT AS A PART OF A GENERAL PROBLEM MANAGEMENT PROCESS

Ph. Somville, Belgatom/Tractebel, Brussels, BELGIUM
P. Hernalsteen, Belgatom/Laborelec, Brussels, BELGIUM
R. Houben, Belgatom/Electrabel, Brussels, BELGIUM

15:40 - Chairman's closing remarks

15:45 - COFFEE

16:15 - PANEL 2 - INSPECTION, REPAIR AND REPLACEMENT STRATEGIES

Chairmen: W. De Roovere, General Manager, Electrabel, BELGIUM
J.P. Samain, Inspector General, SPRI/DBIS, BELGIUM

Panelists: One representative from each of:

* Framatome, France
* Regulatory Body, Sweden
* Westinghouse, United States
* Regulatory Body, Spain
* OECD/NEA Principal Working Group 3 on "Reactor Component Integrity".

17:05 - Summing up by Panel Chairman

17:15 - Closing of the meeting by Mrs. Miet Smet, Secretary of State for Environment.
9.2 THE DAMPIERRE 1 SGR

C. Verges, Framatome - Nuclear Services, Saint-Marcel, FRANCE

9.3 EXCHANGE OF STEAM GENERATORS IN PRESSURIZED WATER REACTOR POWER PLANTS

C. Hillrichs, Siemens/KWU, Erlangen, GERMANY

9.4 STEAM GENERATOR REPLACEMENT AS A PART OF A GENERAL PROBLEM MANAGEMENT PROCESS

Ph. Somville, Belgatom/Tractebel, Brussels, BELGIUM
P. Hernalsteen, Belgatom/Laborelec, Brussels, BELGIUM
R. Houben, Belgatom/Electrabel, Brussels, BELGIUM

15:40 - Chairman’s closing remarks

15:45 - COFFEE

16:15 - PANEL 2 - INSPECTION, REPAIR AND REPLACEMENT STRATEGIES

Chairmen: W. De Roovere, General Manager, Electrabel, BELGIUM
J.P. Samain, Inspector General, SPRI/DBIS, BELGIUM

Panelists: G. Slama, Framatome, FRANCE
G. Hedner, SKI, SWEDEN
J. Wootten, Westinghouse, USA
J. Reig, CSN, SPAIN
A. Tsuge, Mitsubishi, JAPAN

17:05 - Summing up by Panel Chairman

17:15 - Closing of the meeting by Mrs. Miet Smet, Secretary of State for Environment.
# LIST OF PARTICIPANTS

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Attached are the lists of participants to either DOEL NPP or TIHANGE NPP. Names have been transmitted to the management of the plants, in order to allow access: all participants are requested to give their passports or identity cards at the entrance.

A direct bus service to both plants has been arranged. Busses will leave at 8.30 on Friday, September 20th, in front of entrance 2 of the "Palais des Congrès" : they will be parked along the "Mont des Arts".

It takes about 1:30 bus travel from Brussels to any of these plants. Return to Brussels is foreseen around 17:00.

If you still want to join one of the two groups or, in the opposite, if you change your mind and do not intend to participate any more, please let P. MIGNOT or one of the secretary at the Reception Desk know about it, in order to facilitate the organization.
DOEL AND TIHANGE LOCATIONS

NUCLEAR POWER STATIONS IN OPERATION

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The Doel nuclear power station site.

La centrale nucléaire de Tihange: les unités 1, 2 et 3.
## TECHNICAL VISIT TO DOEL NPP

**SEPTEMBER 20th**

### LIST OF PARTICIPANTS

<table>
<thead>
<tr>
<th>NAME</th>
<th>AFFILIATION</th>
<th>NATION</th>
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<tbody>
<tr>
<td>Bouecke R.</td>
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<td>Van Britsom E.</td>
<td>Secr.Etat.Energie</td>
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<td>Englebert F.</td>
<td>Lovisa Pow. Plant</td>
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<td>Jokineva H.</td>
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<td>Japan</td>
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<tr>
<td>Hayashi Y.</td>
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<tr>
<td>Onozawa T.</td>
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<tr>
<td>Saito T.</td>
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<td>Tsuge A.</td>
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<tr>
<td>Yamamoto H.</td>
<td>Japan Power Eng.</td>
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<td>Yashima S.</td>
<td>Nucl. Safety Dept.</td>
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<tr>
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### TECHNICAL VISIT TO TIHANGE NPP

**SEPTEMBER 20th**

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<tr>
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<td>Casillas J.F.</td>
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<td>Esteban A.</td>
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<td>Figueras J.M.</td>
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<td>Kotthoff K.</td>
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<td>Takaba O.</td>
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<td>Hada M.</td>
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<td>Iwatsubo T.</td>
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<td>Chen Y.M.</td>
<td>MRL/ITRI</td>
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A GLANCE AT NEA’S PROGRAMME OF WORK IN THE NUCLEAR SAFETY AREA

OECD

In 1948, the United States offered Marshall Plan aid to Europe, provided the war-torn European countries worked together for their own recovery. This they did in the Organisation for European Economic Co-operation (OEEC).

In 1960, Europe's fortunes had been restored; her standard of living was higher than ever before. On both sides of the Atlantic the interdependence of the industrialised countries of the Western World was now widely recognised. Canada and the United States joined the European countries of the OEEC to create a new organisation, the Organisation for Economic Co-operation and Development. The Convention establishing the OECD was signed in Paris on 14th December 1960.

Pursuant to article 1 of the Convention, which came into force on 30th September 1961, the Organisation for Economic Co-operation and development shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and this to contribute to the development of the world economy;

- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and

- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Signatories of the Convention were Austria, Belgium, Canada, Denmark, France, the Federal Republic of Germany, Greece, Iceland, Ireland, Italy, Luxemburg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries acceded subsequently to the Convention (the dates are those on which the instruments of accession were deposited): Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971) and New Zealand (29th May 1973). The Socialist Federal Republic of Yugoslavia takes part in certain work of the OECD (agreement of 28th October 1961).
The OECD Nuclear Energy Agency (NEA) was established on 20th April 1972, replacing OECD's European Nuclear Energy Agency (ENEA, established on 20th December 1957) on the adhesion of Japan as a full member.

NEA now groups all the European Member countries of OECD and Australia, Canada, Japan and the United States. The Commission of the European Communities takes part in the work of the Agency.

The primary objectives of NEA are to promote co-operation between its Member governments on the safety and regulatory aspects of nuclear development, and on assessing the future role of nuclear energy as a contributor to economic progress.

This is achieved by:

- encouraging harmonisation of governments' regulatory policies and practices in the nuclear field, with particular reference to the safety of nuclear installations, protection of man against ionising radiation and preservation of the environment, radioactive waste management, and nuclear third party liability and insurance;

- keeping under review the technical and economic characteristics of nuclear power growth and of the nuclear fuel cycle, and assessing demand and supply for the different phases of the nuclear power to overall energy demand;

- developing exchanges of scientific and technical information on nuclear energy, particularly through participation in common services;

- setting up international research and development programmes and undertakings jointly organised and operated by OECD countries.

In these and related tasks, NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has concluded a Co-operation Agreement, as well as with other international organisations in the nuclear field.
The Committee on the Safety of Nuclear Installations (CSNI) of the OECD Nuclear Energy Agency (NEA), is an international committee made up of senior scientists and engineers. It was set up in 1973 to develop and coordinate the activities of the Nuclear Energy Agency concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster international cooperation in nuclear safety among the OECD Member Countries.

The CSNI constitutes a forum for the exchange of technical information and for collaboration between organisations which can contribute, from their respective backgrounds in research, development, engineering or regulation, to these activities and to the definition of its programme of work. It also reviews the state of knowledge on selected topics of nuclear safety technology and safety assessment, including operating experience. It initiates and conducts programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach international consensus on technical issues of common interest. It promotes the coordination of work in different Member Countries including the establishment of cooperative research projects and international standard problems, and assists in the feedback of the results to participating organisations. Full use is also made of traditional methods of cooperation, such as information exchanges, establishment of working groups, and organisation of conferences and specialist meetings.

The greater part of the CSNI's current programme of work is concerned with safety technology of water reactors. The principal areas covered are operating experience and the human factor, reactor coolant system behaviour, various aspects of reactor component integrity, the phenomenology of radioactive releases in reactor accidents and their confinement, containment performance, risk assessment, and severe accidents. The Committee also studies the safety of the nuclear fuel cycle, conducts periodic surveys of reactor safety research programmes and operates an international mechanism for exchanging reports on safety related nuclear power plant incidents.

In implementing its programme, the CSNI establishes cooperative mechanisms with NEA's Committee on Nuclear Regulatory Activities (CNRA), responsible for the activities of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also cooperates with NEA's Committee on Radiation Protection and Public Health and NEA's Radioactive Waste Management Committee on matters of common interest.
FOREWORD, prepared by the NEA Secretariat
for the Specialist Meeting on Operating Experience with Steam Generators

The long history of operating experience with pressurized water reactors has indicated that the steam generators are of primary importance in nuclear power plant design and operation; this is furthermore confirmed by analyzing the data of the Incident Reporting System (IRS). It is for this reason that the OECD/NEA Committee on the Safety of Nuclear Installations organizes, in cooperation with UNIPEDE, a Specialist Meeting on "Operating Experience with Steam Generators".

This Specialist Meeting, held in Brussels, Belgium, in September 1991, is hosted by the Belgian Government and AIB-Vinçotte Nuclear. In addition to being a follow-up to the October 1984 meeting (organized by the CSNI and UNIPEDE in Stockholm, Sweden), this Meeting reviews the current state-of-the-art of steam generator technology thus providing a forum for the exchange of related experience in operation, inspection, maintenance, repair, modifications, replacement, and licensing requirements pertaining to steam generators. Forty-seven papers are presented in eight sessions entitled: Operating Experience (two sessions), Structural Integrity and Licensing Issues, Analysis and Prediction of Degradation Mechanisms, Inservice Inspection Methods, Preventive and Corrective Actions (two sessions) and Replacement of Steam Generators. There are furthermore two panel sessions entitled "Observed Degradation Mechanisms and Licensing Positions", and "Inspection, Repair and Replacement Strategies".

These proceedings consist of a compilation of the papers presented at the Meeting, which is attended by more than one hundred and fifty participants from fifteen countries and several international organizations.
Palais des Congrès
Mont des Arts
Bruxelles

C. Palais des Congrès
   Entrées:
   1. Jardin
   2. Arcades
   3. Coudenberg
   P. Parking Albertine-Congrès

D. Salles Dynastie
   4. Rue des Sols
   5. Halle Dynastie
Au nom du gouvernement belge et en tant que ministre responsable de la sécurité des installations nucléaires ainsi qu'au nom de ma collègue, le secrétaire d'État à l'environnement, c'est une grande satisfaction pour moi d'accueillir l'Agence internationale pour l'énergie atomique et l'Union des producteurs d'électricité et de me faire votre hôte en vue de l'organisation de ce séminaire. L'expérience issue de la mise en service des générateurs de vapeur dans les réacteurs de puissance en est l'objet spécifique et il m'apparaît comme étant d'importance primordiale. Effectivement, la supposition tout-à-fait logique dans les stades de début de la construction des centrales nucléaires, que les générateurs de vapeur ne poseraient pas de problèmes techniques particuliers durant la durée de vie des centrales, cette supposition, ne s'est pas vérifiée complètement.
L'application des connaissances techniques a toujours pu faire en sorte que certaines déficiences n'ont pas eu de conséquences désastreuses, quoique plus d'une technique utilisée n'ait pas été couronnée de succès durable. Je constate dans le monde entier que dans pas moins d'une vingtaine d'installations nucléaires, les générateurs de vapeur ont, soit du être remplacés totalement ou partiellement soit ont fait l'objet de cette décision et d'études appropriées à l'exécution de celle-ci. Pour la Belgique, il a été décidé que, pour les installations nucléaires de Doel 3, il sera procédé durant l'année 1993 au remplacement des trois générateurs de vapeur.

Les tâches qui, à cet égard vous sont confiées font partie de la mission qui consiste à garantir et à accroître la sécurité technique des centrales nucléaires en exploitation, en particulier en vue d'assurer et d'améliorer la protection des travailleurs.

Ceci vaut autant pour les travailleurs exécutant leurs prestations dans leur entreprise que pour cette catégorie de travailleurs, pour laquelle la question se pose de manière aussi aiguë, je veux dire les travailleurs extérieurs qui n'appartiennent pas au personnel de la centrale mais qui sont chargés de travaux exceptionnels, le plus souvent à risques comme ceux effectués durant l'arrêt du réacteur.

Mesdames et Messieurs,

A l'occasion de ces assises, je me permets de présenter quelques idées qui, en quelque sorte, servent de point de départ à ma politique actuelle et future en matière de sécurité nucléaire.

Une première idée de base se rapporte à l'intrication étroite de la sécurité dite classique et de la sécurité nucléaire.
La construction des centrales de puissance issue de l'énergie nucléaire ainsi que leur exploitation ne sont pas simplement les retombées des connaissances acquises dans le domaine exclusif des sciences nucléaires. Sans vouloir sous-estimer les problèmes et les bases scientifiques qui leur sont liées et qui sont en particulier la physique des neutrons et le comportement des matériaux sous diverses conditions physiques, je constate - et le fait que vous soyez ici rassemblés en est une preuve - que beaucoup de problèmes étiquetés "de sécurité nucléaire" se situent en fait dans le secteur dit classique. La métallurgie, la mécanique des fluides, la résistance des matériaux, l'analyse probabiliste des accidents, le facteur humain et combien d'autres paramètres "non nucléaires" paraissent d'importance capitale lors de la conception, de l'exploitation et de la construction de centrales de puissance mûes par l'énergie nucléaire.

Surtout en matière de centrales nucléaires, l'utilisation répétée des mots "sécurité nucléaire" me paraît excessive et je pense qu'il vaut mieux mettre l'accent sur la sécurité tout court.

A cet égard je pense que, davantage que dans le passé, le "know how" en prévention acquis dans les autres secteurs industriels doit se propager au secteur nucléaire. Plus spécialement je pense ici au "know-how" du secteur de la chimie qui, en Belgique dans le domaine de la sécurité a démontré sa capacité de réalisation.

Mes compatriotes savent que, en Belgique, les projets en vue de confier à une Agence la sécurité dans le secteur nucléaire sont en voie d'exécution. Cette agence concue comme un organisme d'intérêt public devrait, dans l'optique de ma politique se préoccuper en ordre principal des aspects réellement spécialisés, alors que les aspects généraux dont les applications ne sont pas exclusives du secteur, restent attribués aux instances habituelles et aux organismes privés.
Très concrètement et s'appliquant parfaitement aux travaux de ce séminaire, le contrôle de la construction et l'utilisation des générateurs de vapeur doivent dans cette conception rester en dehors des attributions de l'Agence de sécurité nucléaire et doivent être confiés aux organes de contrôle qui, à cet égard ont acquis l'expérience et continuent de l'acquérir dans d'autres secteurs industriels.

La deuxième idée de base est l'interprétation par les autorités dans une société moderne du rôle qu'elles doivent essentiellement jouer.

Les installations nucléaires ne sont pas exclusivement des installations à très forte concentration d'énergie mais généralement des systèmes interactifs de forte complexité et du grande rigidité dans leur déroulement. L'accident ne se produit pas du dait d'une seule erreur mais résulte de la nature et de la forme du système en tant que tel. On va donc parler d'un accident systémique, d'un accident lié au système, d'un accident qui, heureusement, ne se produit que très rarement. Dans de tels systèmes, le rôle de la structure organisationnelle dans la prévention des accidents est parfaitement reconnue. Aussi faut-il être attentif à ces structures.

On peut avancer ici que dans le domaine qui nous préoccupe, les autorités peuvent jouer un rôle important.

Les autorités et plus concrètement pour la Belgique les services d'inspection de mon département sont par excellence les plus aptes à déterminer, grâce à la comparaison entre les secteurs et ce indépendamment de chaque technologie spécifique, ce qui est nécessaire au secteur nucléaire. Ceci est encore plus exact dans les situations qui font l'objet de ce séminaire et où pour une part importante des "tiers" sont impliqués dans les travaux. La mission des autorités se situe à mon sens pour une part importante dans l'orientation et la stimulation dans l'orientation et la stimulation de l'autorégulation des entreprises et beaucoup moins dans l'intervention directe et concrète dans les éléments techniques spécifiques des installations.
Cette vue générale sur l’action des autorités implique que le législateur et le pouvoir exécutif doivent se limiter aux obligations d’autorégulation plutôt que d’imposer des normes de sécurité détaillées. La suite logique en est que les contrôles de routine doivent être éliminés de plus en plus de l’ensemble des tâches et que de ce fait, il se libère un espace plus grand pour le contrôle de l’auto-régulation. De là, l’intérêt que les autorités doivent porter à un examen plus intensif de l’organisation dans les entreprises et en particulier des structures organisationnelles en matière de sécurité et d’hygiène.

Quand je parle de structures organisationnelles, je pense nécessairement à la concertation dans l’entreprise. Dans le contexte traditionnel de la Belgique j’inclus la participation des représentants des travailleurs qui est toujours essentiel.

Orientation et contrôle de qualité par des audits de l’organisation de la sécurité sont plus ici qu’ailleurs dans le collimateur des actions des autorités. Comme je l’ai déjà fait remarquer plus haut cette mission n’est pas propre au secteur nucléaire et ne fait donc pas partie de la tâche dévolue à l’Agence en voie de constitution.

Cette agence peut cependant dans une mesure restreinte s’occuper de formuler des normes techniques spécifiques pour autant que c’est une répétition – ceci comprenne une dimension spécialisée dans le nucléaire.

Une troisième idée de base est d’apporter en compte les caractéristique propres à chaque technologie de pointe.

Je veux me limiter à la seule caractéristique suivante: pour être appliquée la technologie de pointe suppose une concentration suffisante de "know-how" scientifique et actualisé. Cet arrière-fond de "know-how" est particulièrement nécessaire pour faire face aux problèmes de sécurité qui, en pratique, se présentent à l’improvisée. J’y ai fait allusion dans mon introduction.
Je puis y ajouter les problèmes créés par les déchets radioactifs et qui ne se sont jamais posés lors de la première exploitation des installations nucléaires de manière aussi nette que maintenant.

Je puis avancer avec fierté que, dès la phase initiale de développement de l'industrie nucléaire, la Belgique ne s'est jamais tenue à l'écart des progrès du "know-how". L'installation d'un centre d'études de l'énergie nucléaire à Mol, comportant un réacteur de recherches unique en son genre, en est l'exemple le mieux connu à l'étranger. L'institution et le développement d'organismes agréés spécialisés dans le contrôle nucléaire constituent les exemples connus en matière de sécurité. Le maintien de la sécurité de fonctionnement de sept réacteurs de puissance en est le résultat. Ces réacteurs confèrent à la Belgique au niveau mondial un rôle éminent dans la production d'électricité à partir de combustible nucléaire.

Il va de soi que tout ceci, installé dans un pays de superficie relativement faible mais à forte densité de population, implique une lourde responsabilité vis-à-vis de la population. A ce jour, les organismes agréés ont démontré qu'ils pouvaient sans discontinuer mobiliser le "know-how" nécessaire. Il est essentiel que ceci demeure et que dans l'avenir nous leur donnions les possibilities de la maintenir.

Mes compatriotes savent que je ne ménage aucun effort en vue d'assurer l'institution de l'Agence en harmonie avec l'essence même de l'existence de ces organismes agréés.

Dans ma vision politique, ces organismes remplissent un rôle irremplaçable dans le maintien de notre potentiel scientifique en matière de sécurité nucléaire et il n'est toujours pas démontré que ce rôle puisse être repris par une agence gouvernementale.
Mesdames et Messieurs,

Voilà, pour les considérations au sujet de ma politique. Je pense que, également dans d'autres petits pays le gouvernement va plus ou moins dans le même sens.

Ainsi que je l'ai déjà dit avec insistance, la sécurité des installations nucléaires est de nature multidisciplinaire et complexe. Je profite donc de l'occasion pour que l'on se garde de l'éparpillement des forces dans ce secteur. Je voudrais y ajouter que ceci vaut aussi bien pour d'autres problèmes de sécurité et d'environnement. Je m'adresse en particulier aux différentes organisations internationales actives dans le domaine nucléaire. J'ai l'impression qu'il est nécessaire que plusieurs d'entre elles devraient développer leur personnalité originelle et veiller avec la plus grande vigilance à l'occasion de l'extension de leurs activités à éviter les similitudes et doubles emplois dans les études de problèmes nucléaires. À défaut des effets néfastes sur les prises d'options finales se manifesteront immanquablement.

Tous nous devons assurer une utilisation efficiente du potentiel intellectuel, en particulier sur le plan de la sécurité des installations nucléaires ainsi que dans le domaine de la protection de nos travailleurs.

J'ai l'impression que ceci est déjà le cas avec le sujet qui fait l'objet de cette conférence. La diversité des nationalités des différents participants me renforce dans ma conviction que le problème posé se rencontre dans le monde entier et qu'il faut y travailler de matière coordonnée.

Je vous souhaite à tous une semaine de travail fructueux et déclare officiellement la réunion ouverte.
Geachte Vergadering,

Namens de Belgische regering, als Minister verantwoordelijk voor de technische veiligheid van de kerninstallaties, en mede namens mijn collega de Staatssecretaris voor Leefmilieu, is het voor mij een genoegen gastheer te mogen zijn voor het Agentschap voor Atoomenergie en de Unie van Electriciteitsproducenten, voor de organisatie van dit seminarie. Het specifieke onderwerp omtrent de operationele ervaring met stoomgeneratoren in vermogenreactoren lijkt mij van uitzonderlijk belang.

Inderdaad, daar waar men tijdens het beginstadium van de bouw van kerncentrales logischer wijze kon aannemen, dat stoomgeneratoren gedurende de levensduur van centrales geen bijzondere technische problemen zouden stellen heeft deze onderstelling zich niet volledig waargemaakt.
Uw technische kennis heeft steeds kunnen verhelpen dat bepaalde gebreken geen verregaande gevolgen moet zich meebrachten, maar menig gebruikte techniek werd niet altijd met blijvend succes beloond. Ik stel vast dat in de wereld reeds een bij twintigtal kerninstallaties de stoomgeneratoren geheel of gedeeltelijk werden vervangen of de nodige beslissingen werden genomen en de overeenkomstige studies werden ondernomen om tot deze realisatie over te gaan. Voor wat België betreft, werd beslist dat, dat voor de kerninstallatie Doel 3 gedurende het jaar 1993 tot de vervanging van de drie generatoren zal worden overgegaan.

De taken die U in dit verband zijn opgedragen maken deel uit van een opdracht die erin bestaat de technische veiligheid van de in exploitatie zijnde nucleaire centrales te garanderen en te verhogen, onder meer met als doelstelling de veiligheid van de arbeiders te verzekeren en te verbeteren. Dit geldt zowel voor arbeiders die hun taak verrichten binnen het bedrijf als voor de categorie van werknemers, waar deze problematiek zich even scherp stelt, namelijk de externe arbeiders, niet eigen aan het bedrijf, maar belast met uitzonderlijke en meestal risicodragende werkzaamheden zoals deze uitgevoerd tijdens stilstanden van reactoren.

Dames en Heren,
Naar aanleiding van deze vergadering wil ik mij veroorloven om enkele ideeën naar voren te brengen, die zowat als uitgangspunt voor mijn huidig en toekomstig beleid fungeren inzake veiligheid van de kerninstallaties.

**Een eerste kernidee houdt verband met de nauwe verwevenheid van de klassieke veiligheid met de nucleaire veiligheid**

De constructie van vermogencentrales met nucleaire energie en de exploitatie ervan is niet louter de neerslag van kennis verworven op zuiver nucleaire wetenschappgronden.
Zonder de problemen en de er mee gepaard gaande wetenschappelijke onderbouw van onder meer de neutronenhuishouding en het gedrag van materialen onder diverse fysische omstandigheden te onderschatten, stel ik vast - en uw bijeenkomst hiervan is daarvan eveneens een bewijs - dat vele problemen van wat men zo graag als "nucleaire veiligheid" bestempelt zich eerder situeren in de zogenaamde klassieke sector. Metallurgie, hydraulica, sterkte van materialen, waarschijnlijkheidsanalyse van ongevallen, menselijke factoren en nog zoveel andere "niet nucleaire" parameters blijken van kapitaal belang te zijn bij de conceptie, de exploitatie en de constructie van vermogencentrales met nucleaire energie.

Vooral wat de kerncentrales betreft lijkt het veelvuldig gebruik van de woorden "nucleaire veiligheid" overdreven en meen ik dat meer het accent dient gelegd op de veiligheid zonder meer.

In dat verband meen ik dat meer dan in het verleden de "know-how" inzake preventie in de andere sectoren van de industrie dient door te stromen naar de nucleaire sector. Meer speciaal denk ik hier aan de "know-how" uit de scheikundige sector die in België op vlak van de veiligheid bewezen heeft heel wat te kunnen realiseren.

Mijn landgenoten weten dat in België plannen in uitvoering zijn waarbij de veiligheid in de nucleaire sector aan een nuclear agentschap zou worden toevertrouwd. Dit agentschap geconcipieerd als een instelling van openbaar nut zou in het licht van mijn beleid zich eerder met de werkelijk specialistische aspecten dienen bezig te houden, terwijl de algemene aspecten, die de sector in hun toepasselijkheid overschrijden toegewezen dienen te blijven aan de normale gebruikelijke overheidsinstellingen en privé organismen.

Zeer concreet en zeer toepasselijk in het kader van dit seminarie de controle op de bouw en het gebruik van de stoomgenerator dient in dit concept buiten het werk domein te blijven van het nuclear agentschap en dient toevertrouwd aan controleinstellingen die terzake ook ervaring hebben opgedaan en blijven opdoen in andere nijverheidssectoren.
Een tweede kernidee is de vertolking van de essentiële rol van de overheid in een moderne maatschappij.

De kern-installaties zijn niet alleen installaties met een hoge energieconcentratie maar doorgaans ook systemen met een grote interactieve complexiteit en een grote strakheid van koppelingen. Een ongeval wordt hierbij niet veroorzaakt door één enkel mankement maar vloeit voort uit de aard en de vorm van het systeem als dusdanig.

Men spreekt dan ook van een systeemaccident, een systeemgebonden ongeval dat, gelukkig maar, hoogst zelden voorkomt. Bij dergelijke systemen is de rol van de organisatiestructuur bij het vermijden van ongevallen erkend. Vandaar, de aandacht die dient uit te gaan naar de organisatiestructuren.

Hierbij kan worden gesteld dat vooral op dit vlak de overheid een belangrijke rol kan spelen. Deze overheid en in België meer concreet de inspectiediensten van mijn departement is bij uitstek geschikt om door vergelijking tussen sectoren en los van elke specifieke technologie uit te maken wat hier noodzakelijk is.

Dit is des te meer waar in aangelegenheden die het onderwerp van dit seminarie uitmaken en waar ook in belangrijke mate "derden" bij de werkzaamheden betrokken worden.

De opdracht van de overheid in deze is naar mijn mening in een belangrijke mate gelegen in het begeleiden en stimuleren van de zelfwerkzaamheid van de ondernemingen en minder in directe concrete bemoeiing met specifieke technische installatiecomponenten.

Deze algemene visie op het functioneren van de overheid sluit ook in dat de wetgever, en de uitvoerende macht, zich meer en meer zal moeten beperken tot verplichtingen tot zelfwerkzaamheid in plaats van tot in detail veiligheidsnormen op te leggen. Als logisch gevolg vloeit hieruit voort dat routinecontroles meer en meer uit het takenpakket dienen geschrapt zodat er meer ruimte vrijkomt van de controle op de zelfwerkzaamheid.
Vandaar het belang dat door de overheid moet gehecht worden aan een intensiever bekijken van de organisatie in de ondernemingen, en meer bepaald van de organisatiestructuur inzake arbeidsveiligheid en -hygiëne.

Wanneer ik het heb over organisatiestructuur, bedoel ik tevens noodzakelijk het overleg binnen de onderneming. Vanuit de traditionele Belgische kontext is immers ook de betrokkenheid van de vertegenwoordigers van de werknemers essentieel.

Begeleiding en kwaliteitsbewaking van organisatorische safety audits staan hier meer dan elders in de focus van de overheidsactie.

Zoals ik hoger ook reeds opmerkte is deze opdracht niet typisch voor de nucleaire sector en bijgevolg geen opdracht voor het in oprichting zijnde nucleair agentschap.

Dit agentschap kan zich echter wel in een beperkte mate bezighouden met de specificatie technische normering voor zoverre dit, het weze andermaal herhaald, ook nog een specialistische dimensie inhoudt op vlak van het nucleaire.

Een derde kernidee brengt de kenmerken in rekening eigen aan elke spitstechnologie.

Ik wil me hierbij beperken tot één kenmerk nl. de omstandighed dat spitstechnologie een voldoende concentratie van "up to date" wetenschappelijke "know-how" onderstelt.

Deze "know-how" achtergrond is meer speciaal absoluut noodzakelijk om het hoofd te kunnen blijven bieden aan de veiligheidsproblemen die zich in de praktijk steeds weer op verrassende wijze voordoen. Ik maakte hierop een allusie in mijn inleiding. Ik kan ook verwijzen naar de problemen i.v.m. radioactief afval, die zich ten tijde van de eerste exploitatie van de kerninstallaties nooit zo pregnant hebben voorge- dan als thans het geval is.

Met fierheid kan ik stellen dat België sedert het beginstadium van de ontwikkeling van de nucleaire industrie op vlak van know-how uitbouw zich nooit terzijde heeft gehouden.

De oprichting van het studiecentrum voor kernenergie te Mol met een onderzoeksreactor die enig is in zijn aard, is hier-
aan het internationaal best bekende voorbeeld. De oprichting en de uitbouw van gespecialiseerde erkende organismen voor nucleaire controle zijn de bekende voorbeelden op vlak van veiligheid. Het veilig in werking houden en hebben van zeven vermogenreactoren zijn hiervan het resultaat. Deze reactoren bezorgen België trouwens op wereldvlak een vooraanstaande rol, qua produktie van elektriciteit uit nucleaire grondstof.

Het spreekt voor zich dat dit alles in een relatief klein maar uiterst dicht bevolkt land een zware verantwoordelijkheid meebrengt t.o.v. de bevolking. Tot op heden hebben onze erkende organismen bewezen de nodige "know-how" te kunnen blijven mobiliseren. Het is essentieel dat dit blijft en dat we hun de ruimte blijven geven om dit ook in de toekomst zo te houden.

Mijn landgenoten weten dat ik geen inspanning ongemoeid laat om de oprichting van het nucleair agentschap in harmonie te laten gebeuren met de essentie van het wezen van deze erkende organismen. In mijn beleidsvisie vervullen deze organisme een onvervangbare rol in het op peil houden van ons wetenschappelijk potentieel op vlak van de nucleaire veiligheid en het valt nog te bewijzen dat deze rol door een overheidsagentschap kan worden overgenomen.

Dames en Heren tot daar deze beschouwingen over mijn beleid. Ik meen dat ook in andere kleine landen de overheid in meer of mindere mate in een zelfde richting denkt.

Zoals ik reeds beklemtoonde draagt de veiligheid van kerninstallaties een multidisciplinair en complex karakter. Ik neem dan ook de gelegenheid te baat om te waarschuwen voor versnippering van krachten in deze sector. Ik zou er kunnen aan toevoegen dat dit tevens geldt voor andere problemen inzake veiligheid en milieu. Ik richt mij vooral tot de verschillende internationale organisaties die op nucleair gebied actief zijn.
Ik heb de indruk dat het nodig is dat meerdere ervan hun oorspronkelijke eigenheid dienen verder uit te bouwen en met de uitbreiding van hun activiteiten naar nieuwe domeinen een grote behoedzaamheid dienen na te streven om een ernstige overlapping van studies inzake nucleaire problemen te vermijden. Een dergelijke evolutie kan enkel maar een nefaste invloed hebben op de uiteindelijk te nemen opties.

We dienen allen een efficient gebruik van het intellectueel potentieel, meer bepaald op gebied van de veiligheid van kerninstallaties alsmede op het gebied van bescherming van onze arbeiders tegen ioniserende stralingen na te streven.

Ik heb de indruk dat dit alvast het geval is met het onderwerp dat het voorwerp uitmaakt van deze vergadering. De verscheidenheid in nationaliteit van de verschillende deelnemers sterkt mij in de overtuiging dat het gestelde probleem zich voordoet over de ganse wereld en dat er op gecoördineerde wijze aan wordt gewerkt. Ik wens U allen nog een vruchtbare werkweek en verklaar deze vergadering officieel voor geopend.
On behalf of the Belgian Nuclear Society, whose members belong to all sectors of nuclear energy (Universities and other education institutions, research centers, suppliers, electrical utilities, architect engineers, control organizations, etc.)

and

on behalf of AIB-Vinçotte Nuclear which is the Technical Support of the Belgian Ministries responsible for nuclear safety,

I have the pleasure to welcome you at this Specialist Meeting.

First of all, let me thank the Minister of Labour and Employment and the Minister of Public Health and Environment who gave financial support to this meeting and made its organization possible.

As the Minister of Labour and Employment told you a few minutes ago, about 60% of the electricity come from nuclear power plants in Belgium.

It implies that Belgian utilities, their architect engineer and their laboratories have given much attention to the problems of steam generators since the beginning of the Belgian nuclear program.

The number of Belgian papers presented at this conference is another proof of this interest.

The work covers all fields: prevention, measurement and diagnosis, maintenance, repair and, in the future, replacement.

This work was not undertaken for the benefit of pure science only: in fact Belgian steam generators have been plagued to various degrees by about all the illnesses experienced in the world on these equipments.

In the field of prevention, the four latest plants have undergone kiss rolling above the tube sheet; the steam generators of Doel 3 and Tihange 2 have been the first ones to be subjected to a shotpeening operation on an industrial scale, after a few years of operation; the steam generators of Doel 4 and Tihange 3 have been rotopoeened before being put in service and experience has now been accumulated on the benefits of these operations.
For the detection of cracks, many specific high performance detection methods have been developed and applied well before their current use in the world. For example, highly automatized eddy current methods for the detection of cracks, with a fast computer-assisted analysis of the results, and more recently specific ultrasonic probes, yielding very favourable results.

Maintenance activities have centered on secondary side cleaning, appropriate primary and secondary water chemistry, crevice cleaning, etc...

Repairs have consisted in the replacement of antivibration bars, in some cases in the replacement of condensors, in the installation of different types of sleeves, and in the development of the so-called kiss sleeving method in which a thin layer of Nickel is electro deposited on the inside surface of the tube, effectively bridging already existing cracks.

Different types of tube plugs have also been used, and the problems related to the long term integrity of such plugs are taken into account. It has also been decided to replace in 1993 the steam generators of Doel 3, and a careful analysis has been carried out for the choice of the material of the tubes.

What do we expect from steam generators?
We expect them to be safe and reliable.

Safety must be checked against principles, rules and criteria. In Belgium, the USSNRC rules have been applied. During the safety analysis a number of accidents, incidents and transients are assumed and investigated with pessimistic assumptions.

Steam generator tube rupture is a postulated accident. It was formerly considered as a class 4 accident, in the ANS standard terminology, i.e. something that has a very remote probability of occurrence. World experience however has shown up to now about 10 such events and as a consequence regulatory bodies have had the tendency to classify steam generator tube rupture as a class 3 accident and to ask for evaluations of the consequences of multiple tube ruptures in beyond design assessments.

Reclassification of an accident is not a purely formal exercise, as the acceptance criteria, like radiological consequences, may be affected. For the evaluation of radiological consequences, much work has still to be done to reach a consensus on the modelling of iodine spiking in the primary circuit and on the values of partition factors in the secondary side.

Such evaluations of radiological consequences actually limit the primary circuit concentration in radioactive products during normal operation. The need for the operator to stop safety injection after a steam generator tube rupture has been a subject of concern, due to the possibility of overfill of the steam generators, hence the requirement that steam lines can withstand being filled with water.
In order to mitigate the consequences of an incipient steam generator tube rupture, early warning systems have been developed, like Nl6 monitoring. The benefits of such systems are being assessed in some Belgian plants.

Other events considered in the safety analysis are the opening of a secondary side safety valve (class 2) and a steam line break (class 4) inside or outside containment. Evaluations of these events limit the secondary side contamination and as a consequence the primary to secondary leak.

In the steam line break scenario, a simultaneous steam generator tube rupture is not assumed. However, when many tubes have cracks, such an assumption needs to be supported by adequate evidence, based on experiments and on calculations. Such a work has been underway in Belgium, in order to demonstrate tube integrity during adverse conditions like steam line break, and also to develop tube plugging criteria allowing through wall cracks while retaining most of the conservatisms of Regulatory Guide 1.121.

In a defense in depth perspective one should also check the resistance and the tightness of the containment after a steam line break inside containment.

All the above mentioned postulated accidents, with their associated conservative assumptions, should be avoided by design in future reactors. It means demonstration that steam generator tube rupture are indeed much less probable with alloys better than inconel 600, and also provisions to render overfill impossible by a judicious choice of safety injection and secondary safety valves pressures.

Such design measures should also make the plants more reliable, as it is evident when better materials less sensitive to corrosion are chosen or when early warning detection systems are implemented, enabling the operator to take actions before the accident develops and to prevent it.

Steam generator tube ruptures have not only a technical impact on the plant but the Mihama-2 event last February has also shown us the media coverage they might stir. One may regret that tools like the International Nuclear Event Scale (INES) have not been used as soon as the accident was known, as INES was precisely developed to give the media a better understanding of the safety importance of events occurring in nuclear power plants.

According to the first press coverage emanating from Japan, the accident was compared to Three Mile Island or even to a near other Chernobyl. Such representations are certainly misleading for events which are taken into account in the design of the plants and during which safety injection has worked as intended and the operator has reacted wisely.

The Mihama event has thus shown the difficulty for the journalists to get objective information and technical explanations when their source of information is restricted to "concise" news agencies reports. As a result, the Belgian Nuclear Society, as a scientific association, contemplates offering the journalists the opportunity to get in touch with its members specialized in the different fields of nuclear energy. It is hoped this proposal will be seen by the members of the press as a positive one.
In his crusade for a better environment, Commandant J.Y. Cousteau wants "the rights of future generations" to be recognized. In the crowded world of next century the right to dispose of energy will certainly be a crucial one.

In these days where democracy becomes again fashionable, we should not forget that there is no democracy without the choice between alternate solutions. Even if this Specialist Meeting is devoted to water reactors, other types of reactors, like fast breeder reactors and high temperature gas reactors, have been developed in Europe as in other parts of the world and have reached technical maturity.

These alternate solutions should not be discarded by financial short-sightedness, and an energy policy should be elaborated for the next century using all the cards we have in our hands. Society will then have real choices at its disposal and will be able to select those solutions which suit best the wishes of the majority of people.

In this perspective, isn't the work that you will perform this week, which will hopefully improve the reliability of an energy system, a step forward towards democratic ideals and for the benefit of mankind?
CSNI/UNIFEDE SPECIALIST MEETING ON OPERATING EXPERIENCE WITH STEAM GENERATORS

INTRODUCTORY REMARKS

by

JAVIER REIG

PWG-1 CHAIRMAN

Good morning, ladies and gentlemen, I am very pleased to be here and to add my words of welcome to you at the beginning of the Specialist Meeting on Operating Experience with Steam Generators, sponsored by UNIFEDE and the OECD Nuclear Energy Agency, which I represent, and that I will briefly introduce to those of you who are not familiar with it.

The Nuclear Energy Agency (NEA) is one of fifteen bodies that make up the Organisation for Economic Cooperation and Development (OECD). Although the majority of the OECD's activities are oriented towards economics, the work of the NEA has a significant relation with the safety of nuclear power plants. 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...", "...the joint use of research installations...", and "...the dissemination of information... on the safety and regulation of nuclear activities".

The Committee on the Safety of Nuclear Installations (CSNI) is one of the eight technical committees that form the NEA. The CSNI is responsible for "...technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. It reviews the state of knowledge on selected topics of nuclear safety technology and safety assessment, including operating experience". "It initiates and conducts programmes so identified in order to overcome discrepancies, develop improvements and reach consensus on technical issues of common interest". "It promotes the coordination of work in different Member Countries including the establishment of joint undertakings, and assists in the feedback of the results of participating organisations".

The technical field of nuclear safety is so broad that the CSNI has established Principal Working Groups to perform the work. This Specialist meeting was promoted by two of them, PWG-1 dealing with operating experience and human factors, and PWG-3 concerned with reactor component integrity.

1.3-1/5
First and most important, steam generators constitute a special safety barrier since they function both as a reactor coolant boundary and as containment boundary, in some scenarios. In this sense a considerable amount of work have been accomplished in past years to better understand and describe the physical phenomena which may occur following a SGTR event and to improve operator guidelines for plant recovery. Recent events like North Anna have showed that much of this understanding has been achieved and that, if operator procedures are correctly followed, the plant may be taken to a safe shutdown situation in a reasonable time and with negligible radiological consequences.

Although recognising that positive fact, we must also admit that the probability of SG events is higher than the expected by designers and regulatory authorities, and that in some recent events, like Mihama and Maine Yankee, the primary to secondary leakage increased at rates that were significantly higher than would be predicted.

If we take also into consideration that in many countries leak-before-break criteria are being used, we should conclude the importance of adequate leak rate monitoring programs, which can provide for early detection and be an effective approach to minimize the frequency of SGTR events. I think that monitoring programs have been improved significantly in recent years, and that real-time information on leak rate and rate increase are more oftenly used by utilities. A good example are the Nitrogen-16 monitors, already incorporated in many designs or plant improvements. The use of this programs together with the procedures mentioned before may constitute an adititional safeguard to avoid safety problems.

Finally I also want to mention that steam generator problems are an important contributor to occupational radiation exposure, and this should be taken into account in ALARA programs and outage management.

Due to the safety and availability problems all parties involved, that is designers, manufacturers, research laboratories, utilities and regulatory bodies, have taken steam generator problems very seriously and dedicated a significant effort to understand and prevent them.

Probably you are aware that this meeting is in some way a continuation of the one held in Stockholm seven years ago on the same topic and sponsored by the same organisations. I remember Professor Jung, with more than 50 years in the power industry, making the summary of the meeting and bringing up more than a blush in the audience when he went through a long list of problems and pointed out the lack of transfer in heat exchanger experience from other industries like navy or thermal power plants.
Safe nuclear power can only be assured by responsible operators who are supervised and controlled by responsible governments. Both of them relied on the competence and quality of the work performed by designers and manufacturers. It is vital for the success of our industry that all parties cooperate to attain this goal. I am therefore very happy that we have specialists here from all these fields, which will allow us to broaden our discussions by adding the different perspectives.

In closing, Ladies and Gentlemen, I would not want to leave the floor without extending my thank to the Belgian Government for having invited us to hold our discussions here in Brussels and to the Association Vincotte and the other organisations and individuals who have contributed to the preparation of our meeting.

This is not the first time that NEA has sponsored a meeting in this beautiful country and I know from past experience that we can expect a perfect organisation and a graceful hospitality.

There now only remains for me to wish you a successful and informative meeting, and a pleasant stay in the lovely city of Brussels.
SESSION 2 - OPERATING EXPERIENCE - PART 1

2.1 OPERATING EXPERIENCE WITH STEAM GENERATORS

A. Becquaert, Belgatom/Electrabel, Brussels, BELGIUM
A. Timmermans, Belgatom/Electrabel, Brussels, BELGIUM
P. Coenraets, Belgatom/Electrabel, Brussels, BELGIUM
K. de Ranter, Belgatom/Electrabel, Brussels, BELGIUM
C. Leblois, Belgatom/Tractebel, Brussels, BELGIUM

2.2 RECENT OPERATING EXPERIENCE WITH STEAM GENERATORS IN JAPAN

S. Yashima, Japan Power Engineering and Inspection Co., Tokyo, JAPAN

2.3 STATUS OF STEAM GENERATORS IN SPAIN

J.M. Zamarron, Central Nuclear de Almaraz, Madrid, SPAIN

2.4 LA SURETE D'EXPLOITATION DES GENERATEURS DE VAPEUR EN FRANCE - PREVENTION DES RISQUES ASSOCIES AU VIEILLISSEMENT EN SERVICE - LA POSITION DE L'AUTORITE DE SURETE FRANCAISE

P. Brossier, BCCN, Dijon, FRANCE
B. de l'Epinois, DSIN, Paris, FRANCE
J.L. Pierrey, CEA/IPSN, Fontenay-aux-Roses, FRANCE

2.5 OPERATING EXPERIENCE WITH HORIZONTAL TYPE STEAM GENERATORS OF THE VVER-440 PWR UNITS

S. Nagy, PAKS Nuclear Power Plant, HUNGARY

2.6 OPERATING EXPERIENCE WITH WESTINGHOUSE MODEL F STEAM GENERATORS

D.D. Malinowski, Westinghouse NSD, Pittsburgh, USA
R.M. Wilson, Westinghouse NSD, Pittsburgh, USA
M.J. Wootten, Westinghouse NSD, Pittsburgh, USA
OPERATING EXPERIENCE WITH STEAM GENERATORS

A. Becquaert, A. Timmerman, P. Coenraets, K. de Ranter
Belgatom/Electrabel
Brussels (Belgium)

C. Leblois
Belgatom/Tractebel
Brussels (Belgium)

ABSTRACT

The Belgian utilities operate 7 PWR units, the steam generators of which suffer from different corrosion attacks. While PWSCC at the roll transition has long been the major difficulty, degradations of the external surface of the tubes were recently observed in different units, at the level of the top of tubesheet, at the tube support plates and in the sludge piles. Many of the observed cracks are through wall but do not reduce excessively the tube strength, what led to the development of specific plugging criteria, thus allowing most of the affected tubes to be kept in service.

The Belgian utilities have thus learned to operate imperfectly tight steam generators. They have improved the procedures for in-service leak monitoring and for detection of leaking tubes during outages, as well as the accuracy and efficiency of NDE tools. Many repair interventions were carried out, several of which were tests essentially aimed at assessing new techniques.

RESUME

La Belgique compte 7 centrales du type PWR, dont les générateurs de vapeur sont atteints de différents types de corrosion. Le problème majeur a longtemps été la corrosion sous tension du côté primaire, au niveau des transitions de mandrinage. Toutefois, des attaques externes des tubes ont récemment été observées dans différentes unités, au niveau du sommet de plaque tubulaire, des plaques supports et dans la zone des boues. Les fissures observées bien que souvent traversantes ne réduisent pas nécessairement la résistance des tubes de façon inacceptable. Dans plusieurs cas des critères de bouchage spécifiques ont été mis au point, lesquels autorisent le maintien en service d’une majorité des tubes affectés.

Les exploitants belges ont donc dû apprendre à gérer des générateurs de vapeur imparfaitement étanches. Ils ont mis au point des procédures de suivi en service des fuites et de détection, pendant les arrêts, des tubes qui fuient. Des améliorations en termes de précision et de vitesse ont été apportées aux techniques de contrôle non destructif. Beaucoup d’interventions de réparation ont eu lieu, certaines d’entre elles visant essentiellement à évaluer de nouvelles techniques.
1. STATUS OF THE BELGIAN NUCLEAR PLANTS

The Belgian utilities operate 7 PWR units, distributed over 2 sites: Tihange, on the Meuse river and Doel on the estuary of the Schelde river. They are equipped with different types of steam generators: model 44, model 51 and model E, the major characteristics of which are described in table I.

Since the beginning of operation, the maintenance of these steam generators raised a lot of problems, most of them related to corrosion of the tube bundle. Tube pulls and inspections by means of very sensitive NDE techniques allowed to follow precisely the evolution of degradation. The first major problem encountered has been PWSCC, which extended rapidly to a large number of tubes. As the usual plugging criterion, which sets the defect limit to 40% of the wall thickness, is overconservative for this type of defect, the Belgian utilities have very early departed from it and developed alternates which take into account the specific degradation mechanisms and the rules of RG 1.121. In parallel, the policy with respect to leaking tubes was progressively relaxed. Presently, small potential leakers are left into service as long as they do not cause a risk of overstepping the leak limit during the next operating cycle.

Many other defects have been observed in the Belgian steam generators. They affect both the inner and outer surfaces, at different locations along the tube, and as such require continuously growing inspection work. Not only the number of tubes to inspect per outage increases but also more inspections with different probes must be performed to detect and size all defects.

In the frame of the steam generators maintenance, a lot of R&D work has been done in Belgium, aimed at improving the precision of the in-service leak monitoring, at developing fast and accurate inspection tools, at defining specific plugging criteria for the different types of defects, at setting up predictive methods to assess the future degradation rates, at testing in lab and in situ repair techniques offered by vendors or developed in house.

In one plant, Doel 3, the degradation has reached such a level that replacement becomes a viable alternative to repair. A detailed cost/benefit assessment has concluded that the cumulated repair costs and the production losses will prove more expensive than replacing the steam generators, particularly when taking into account the benefit of a possible power upgrade. New steam generators have been ordered and a replacement operation will take place in 1993.

2. MAIN DEGRADATIONS OBSERVED

Table II summarizes the different degradations observed on Belgian SG’s.

PWSCC at transition zones

Primary Water Stress Corrosion Cracking (PWSCC) has been the major concern in the Belgian plants. Three units are severely affected: Doel 2, Doel 3 and Tihange 2. The two most recent plants, Doel 4 and Tihange 3, seem to have been
efficiently protected by the preventive roto-peening treatment applied prior to operation start-up. Doel 1 seems totally immune and Tihange 1 nearly so: only recently few and small PWSCC cracks have been observed.

The good performance of Tihange 1 is clearly related to the fact that the tubes were supplied in a sensitized condition, what is favorable to resist PWSCC. For Doel 1, the explanation is less obvious. The drastic difference of performance between Doel 1 and 2 is particularly impressive. These are twin units (2 x 400 MW) of the 2 loops PWR type. Both are provided with identical steam generators, of Westinghouse design (model 44) and Cockerill manufacture. The 7/8" tubing, in mill annealed Inconel 600, was supplied by Mannesmann according to the same specification. While unit 1 is still free from any PWSCC defect, unit 2 suffered from leaks 2 years after start-up and SG-A has presently 74% of all tubes affected by axial cracking, the other SG being somewhat less affected. No significant difference between Doel 1 & 2 SG's has been traced in either the chemical analysis or the mechanical properties of the tube material. However, a review of the micrographs showed a difference in grain size which correlates quite well with actual service behaviour. The larger grain size (lower ASTM number), probably resulting from a higher annealing temperature, is associated with the good performance of unit 1 SG's; the smaller grain size is associated with the generic cracking problem of unit 2 SG-A, while an intermediate behaviour was observed on SG-B.

The Doel 2 unit has the longest Belgian experience of operation with numerous cracks. All are longitudinal and located in the hot leg side although a few have been observed in the cold leg side. Most cracks are through wall or close to through wall. The Belgian authorities have accepted that operation continues without repairing or plugging the cracked tubes, as long as the leak remains below the limit of the technical specification (79 l/h). Their acceptance is based on the fact that all the cracks are confined within the tubesheet, as mechanical rolling is partial for this unit. Thus, unstable propagation of an axial crack is not possible. Even an hypothetical linkage of numerous axial cracks resulting in a fully circumferential rupture would not produce an excessive leak as the ruptured tube would be supported by the adjacent outer ones and would not disengage from the tubesheet.

The Doel 3 and Tihange 2 plants, constructed in parallel according to the same specifications appeared to be both very sensitive. Leakage was observed during the first cycle of Doel 3 and two tube pulls at the first outage confirmed PWSCC. The cracks are similar to those of Doel 2: longitudinal and nearly through wall. However, due to the fact that they are located above the tubesheet, they may result in a tube rupture if they grow up to a length at which unstable propagation occurs under the effect of pressure. For this problem a length based criterion was set up [1] which takes into account burst tests results, the pressure differential in normal and accidental conditions and the safety coefficients imposed by the RG 1.121. Furthermore, a fast RPC tool was developed to allow 100% inspections at each outage. This allowed to constitute a large data base of measurements and to derive from it prediction models for crack propagation [2] as well as an assessment of the efficiency of peening. [3] Figure 1 shows the evolution of cracking for the transition zones of Doel 3 and Tihange 2.

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PWSCC in the expanded zone

The units sensitive to PWSCC have also suffered from cracking in the region expanded into the tubesheet. The skipped rolls were particularly affected, even some which had been corrected by rerolling. This cracking appeared very rapidly; in fact before the one in the transition zone. But it did not extend to more than a few percents of the total number of tubes. Thus, after a few years it became negligible compared with the transition zone degradation.

These cracks in the rolled region are axial. They may be very long but are basically harmless except that in a few cases primary water may pass into the crevice and be trapped, what may produce, by thermal expansion, a local inner bulging ("Obrigheim effect"). Such an effect has been observed on a few tubes in Tihange 2 but remained very exceptional based on periodic inspections.

PWSCC at U-bends

As it could be expected, the plants sensitive to PWSCC have also experienced corrosion in the U-bends region. This cracking is more difficult to detect by NDE, so the precise cracking pattern is not known and preventive plugging or treatments are the only possible remedy. Two cases of leaks have been observed: in Doel 2, a rupture occurred during the start-up of the 4th cycle; in Tihange 2, leaks were observed during the 5th cycle.

No problem has been reported in the most recent units (Doel 4 and Tihange 3), probably thanks to the preventive U-bends heat treatment.

OD Corrosion in the tubesheet crevice

Three plants have a tubesheet crevice: Doel 1 & 2 and Tihange 1.

Only Doel 2 suffered from outside diameter corrosion in this region: axial ODSCC which appeared rapidly (6 years after start-up) and was the cause of forced outages for leakage.

This corrosion remained limited to a small number of tubes, maybe because of a periodic crevice cleaning operation (cf § 6).

OD Corrosion in the sludge pile

A significant OD corrosion has been observed in the sludgepile of Tihange 1. Detected after the 13th cycle, these degradations have extended rapidly and now affect several hundreds of tubes over heights up to 300 mm. The affected tubes are located near the centre of the tubesheet (R 25 C50), hot leg side. Examination of a tube pulled last year showed that defects are a complex mixture of IGA, multidirectional SCC and wastage. A plugging criterion for this type of defects was established on base of a correlation between the signal amplitude, measured with a bobbin coil, and the burst pressure of affected tubes.
OD Corrosion at the Tube Support Plates

Some axial SCC has been detected since 1985 in Tihange 1 at the level of the TSP. However the degradation signals progress very slowly.

The same phenomenon has been observed in Doel 4, during the 1990 outage (end of 5th cycle), at the level of TSP 2, 3 and 4. After an exhaustive inspection program four tubes were pulled the next year and showed deep IGA : up to 100 % at some locations; the corrosion affected, at different degrees, all intersections with TSP's. It might be mixed with some axial SCC, to an extent which is still under investigation.

Because of the TSP confinement there is no associated risk of tube axial rupture; the remaining risk of circumferential rupture is of lesser concern as only 25 % of the tube cross section is sufficient to provide a safety margin of 3 against rupture, taking into account the largest accidental pressure differential.

The plugging criterion developed for this type of defects is similar to the one used in the sludge pile and relies on a correlation between burst pressure and signal amplitude with the bobbin coil. This year, 47 tubes were plugged for OD corrosion at the TSP level.

OD Corrosion at the roll transition

OD Circumferential SCC has been observed in the roll transitions of Doel 4 tubes, at the end of the 6th cycle. Over four pulled tubes, all were cracked in the transition zone, to variable extents ranging from one case of through wall depth to another case of shallow isolated cracks, 1 to 2 mm long. The usual 100 % RPC inspection was supplemented by a sample UT inspection of 231 tubes, which detected more cases of shallow cracking, with no immediate concern.

This circumferential cracking is presently a major concern for the Doel 4 steam generators. The number of affected tubes is uncertain because only the significant cracks are detectable by the present ECT inspection technology. A plugging criterion for this type of defect was established on base of the crack length as measured by RPC eddy current inspections. It takes into account the fact that shallow cracking may exist and be undetected by this method. Eight tubes, of which 2 leakers, were plugged in 1991. Developments are underway to improve the detectability for this kind of cracking.

Vibrational wear under AVB

The most significant degradations of U-bends due to vibrational wear under the AVB have been observed in Tihange 1, where AVB's have been replaced. Also Tihange 2 suffered from such but only for very few cases. In this unit the AVB's have been replaced preventively, to avoid impairing the plugging margin.

More recently AVB wear has been observed in Doel 2, Doel 4 and Tihange 3. In the latter unit, it affected several tubes in 2 specific rows.
Denting

All Belgian units suffer from denting, except Doel 4 and Tihange 3 which have stainless steel plates and a tubesheet plated with Inconel on the secondary side.

Denting progresses steadily but until now has not been a major concern in any plant.

3. LEAK MONITORING

Our early deviation from the "40 % plugging limit", in favour of plugging criteria based on the margin with respect to burst pressure, has resulted in long term operation of the Belgian SG's with a large number of through wall defects, especially for the units Doel 2, Doel 3 and Tihange 2. Their operation remained nevertheless unaffected, with usually a low primary to secondary leakage (< 10 l/h). Table III lists the cases and reasons of forced outages of these units because of a leak rate exceeding the technical specifications. It appears clearly that having left consciously cracked tubes into service within a well defined plugging policy [1] had almost no impact on operation reliability.

A monitoring of the leak rate is however essential for several reasons :

- Insuring that the limit of the technical specifications, presently set to 79 l/h for all Belgian units, is met;
- Identifying the leaking steam generators and preparing the inspection and repair work to be done during the outages;
- Trying to correlate operation and leak;
- Giving warning of a fast propagating defect (LBB).

As no measurement method gives precise and reliable results in all circumstances, the leak must be determined by different methods and their results compared. Currently, three methods are used, each being susceptible to be applied to different isotopes carried along into the secondary circuit by the leak. Two of them give the leak per steam generator while the third gives a global measurement for all SG's.

Mass balance per steam generator

This method is based on a balance of the mass content of a chemical species in a steam generator. The time evolution of the concentrations in the in-and outgoing fluids allows to determine the leak rate (cf Figure 2). The following parameters are accounted for :

- flow and concentration of the feedwaterline (ingoing mass);
- concentrations in the primary water;
- flows and concentrations of the main steam and of the blowdown lines (outgoing masses);
- radioactive decay of the isotope in the steam generator.
The method is applicable to boron, F18, Na24 and N16. The major inaccuracy sources are the following:

- The results depend on usually imprecise steam flow measurements;
- It relies on a difference between two values of the same order of magnitude;
- The ratio of concentrations in steam and SG water must be known. While they probably depend on power, they are usually known only at 100% power and are therefore assumed constant;
- The concentrations in the steam generator may be uneven.

The different useful chemical species have their own specificities. Boron is useless at end of cycle due to its low concentration at that time in the primary circuit. In addition, the reliability of measurements based on that element decreases when the concentration in the secondary circuit is too high (> 1,5 ppm). Fluor has the advantage to be produced in considerable quantities in the primary circuit. But the precision is affected by the large fraction carried along by steam. At the opposite, sodium does not pass into the steam phase but its amount in the primary circuit is irregular and very low. Finally, N16 has a very short lifetime and consequently the measurements are strongly affected by the transit time between the primary circuit and the detectors on the steam lines.

**Reference model method**

This method is based on a mass balance per steam generator, like the previous one. But it relies on the difference between a tight steam generator and the leaking one, what suppresses different terms in the equations and permits to overcome the first two difficulties of the first method. Evidently, the method is not applicable when all SG's leak. In addition, a good precision is achieved only when the blowdown flows are close to each other.

**Measurement of the global leak**

This method is based on a mass balance of a chemical species in the entire secondary circuit. The ingoing masses depend this time on the flows and concentrations of the leak and of the make-up water; the outgoing ones on the condensor purge, the leak of the secondary circuit and sampling lines (cf Figure 2).

The method may be used with tritium and Ar 41. The difficulty with tritium is that its concentration in the primary circuit builds up slowly, making the method useless at the beginning of the cycle or after a reactor trip. With argon concentration remains low and must be measured in the primary circuit with the high pressure sampling system.

**Coherence of the leak measurement methods**

Figure 3 illustrates this by showing the evolution of records over a cycle in Doel 3. The tritium measurement is not directly comparable to the others as it gives a global result for the 3 SG's instead of SG-B alone. However, the leak on SG-B was
largely dominating the others. It appears that although there is a significant difference between results, all give a similar trend.

Figure 4 illustrates the coherence by plotting the frequencies of encountering a given difference between a single method and the mean value obtained by 5 methods, which is assumed to be the "right" result. This statistic was done over 3 cycles for the Doel 3 unit. Against the global tritium and argon methods are not directly comparable to the others. In addition argon has not been recorded all the time. In order to illustrate the result deviation in some circumstances, the figure includes tritium measurements in unfavourable conditions.

4. NON-DESTRUCTIVE EXAMINATION METHODS

The selection of tubes to plug for structural reasons is based on extensive NDE performed at each outage. Eddy currents are used, both with the bobbin coil and with the rotating pancake coil. UT probes are also used with different patterns of displacement.

The general philosophy for inspection is described in paper reference [4], while the NDE tools and their performances are outlined in references [5], [6].

5. LEAK DETECTION DURING OUTAGES

The plugging of the major leaking tubes during outages allow to maintain the leak rate at an acceptable level during the next cycle. When necessary, according to the leak rate during the previous cycle, a leak test is performed by pressurizing the secondary side of the steam generator filled with a mixture of auxiliary feedwater and about 50 ppm of fluoresceine. The test procedure is the following: steam generators are first emptied and then filled up with clean air and, if necessary to attain the initial pressure, with additional nitrogen. This initial pressure ranges between 7 and 12 bars. It is further increased to the test pressure, 25 to 40 bars, by pumping the mixture into the SG, what compresses the air cushion. After a delay of 8 hours (usual) to 20 hours (maximum sensitivity) a trained operator enters into the channel head, inspects the tubesheet with a UV lamp and marks the leakers. This test has been used for years and gives good results as well with tubes fully expanded in the tubesheet as with partially expanded tubes. The best time to perform the test is just after cooldown, because:

- the primary side is well dried by the SG residual heat;
- the lost time is minimum as the pressurisation is performed in parallel with the SG's opening;
- the information is obtained at the beginning of the outage.

Such a test is normally performed in less than a day.

During the summer 87, the Doel 3 unit suffered from successive outages due to a leak exceeding the limit, followed by a refueling outage and a few weeks later by an apparently power dependant "ghost leak". This gave an opportunity to compare the results of successive inspections by the fluoresceine method between them and

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with the result of a helium test. The crack lengths were also known thanks to eddy current inspections. The helium test was carried out at a pressure of 5 bar. The eddy current inspection was performed with the standard RPC probe.

Figure 5 gives the results of the successive leak tests inspection compared to the crack lengths. 64 tubes were detected as leakers either by one of the fluoresceine tests or by the helium test. As the sensitivity of this last test is extremely high, only the tubes measured with a leak rate higher than 5 cc/h are accounted as “leakers”. In addition to these, 271 very small leakers have been detected with the helium test (lower than 5 cc/h).

Several conclusions may be drawn from figure 5:
- all leakers detected by either fluoresceine or helium had cracks detected by RPC, but one (probably an error of location). The reverse is not true as 40 % of the tubes were cracked.
- very long cracks have a clear trend to leak: in category 1 the tube with a 22 mm long crack was clearly the major contributor to the total leak rate. However, there is no direct correlation between the leak rate and the crack length: while the steam generator contained at that time more than 50 cracks of 10 mm or more, only 7 were detected as small leakers. On the other hand, cracks as short as 2 mm have been detected repeatedly as leakers. This conclusion is also supported by the similarity between the crack lengths distribution measured by RPC and the leak rates measured by helium (figures 6 and 7)
- the reproducibility of the fluoresceine test is poor for small leaks
- a lot of tubes have been detected as leakers by the fluoresceine test and not by the helium test, even taking into account the tubes with very low helium leak rate. This is probably due to imperfect drying of the steam generator, although the contractors procedure and drying time had been respected.
- the reverse is also true: a lot of tubes have been detected as leakers by the helium test and not by fluoresceine. This seems quite normal: the sensitivity of the method is well known and a total of 335 leakers were detected so.

6. REPAIR INTERVENTIONS

The corrosion of the tube bundle has required a large number of repair interventions in the Belgian plants. They are listed below.

1. Change 1
1986: Antivibration bars replacement - When this operation was decided, 50 tubes had been plugged, due to vibrational wear at the level of the AVB's. The replacement changed the contact points and reduced the gaps.

1988/89: Chemical cleaning - In order to reduce OD corrosion observed at the level of tube support plates and in the sludges, a treatment was carried out according
to the EDF process.[7] Two SG’s were treated in 88 and the third in 89. A subsequent tube pull indicated that the treatment had probably been performed too late because deep IGA/SCC attack was found within the zone previously covered by the sludges.

**Doel 2**

1979: **Plugging** of overovalized tubes. This was a consequence of the occurrence of a rupture in the bend of such a tube.

1980 and 1981: **Reexpansion** of the tube in the tubesheet crevice, above the originally rolled section. The purpose of this operation was to provide a tight barrier between the secondary side and the cracked zone, at the roll transition. A total of 137 tubes were treated, applying first a hydraulic expansion over 5" and further, in the newly expanded zone, 2 steps of "hard" rolling (about 2" long, centrally located). Sporadic loss of tightness was detected during later service operation and further PWSCC was observed at both the hydraulic and the roll transitions. Nevertheless, some of the tubes repaired by that method are still in service today [8].

1982: Installation of 187 explosively welded "minisleeves". This operation was aimed at plating the transition zone of cracked tubes with a tight membrane. The sleeves were 40 mm long and 0.4 mm thick. Circumferential cracking of the parent tube at the limit of the sleeve was observed shortly after installation (6 to 9 months). All tubes were plugged [8].

1983, 1986 and 1989: **Crevice cleaning**. These operations were aimed at flushing the tubesheet crevice with boiling water in order to take out the impurities. They seem to have stopped or at least strongly slowed down the OD corrosion process. The reason of the process efficiency is not clearly known as only very low quantities of impurities were extracted.

1985: **Row 1 plugging**. This was a preventive countermeasure to PWSCC in the bends.

1985, 1986 and 1988: **Nickel plating** of a total of 124 tubes. The purpose of this repair technique was to plate the transition zone with a tight layer of electrodeposited nickel. Several procedures and nickel thicknesses have been used. They are described in the reference.[9]. Presently most of these tubes remain in service.

1990: Explosive **Reexpansion** of the tube in the tubesheet crevice and deposit of a nickel layer. This operation was similar to the expansion performed in 1980, except that it was carried out with a manipulator to reduce the radioactive doses, that the expansion was shorter (50 mm explosive and 27 mm mechanic) and that the inner face of the tube at the level of the new transition was protected against further PWSCC by a 50 microns layer of nickel. A total of 343 tubes were satisfactorily treated, including plugged tubes previously repaired with minisleeves.[10]

**Doel 3**

1985: **Shot peening** of the tubes, hot leg side. This treatment was the first shot peening operation in the world as a remedy for PWSCC and was carried out after an
extensive research program described in an EPRI Report [11]. A zone extending from
100 mm below to 50 mm above the transition was peened with Inconel beads. This
treatment performed after 3 years of operation slowed down the cracking but did not
stop it. The efficiency of the process and results of pulsed tubes expertises are
discussed in paper [3]

1987: **Row 1 plugging.** This was a preventive counter measure to PWSCC in
the bends.

1988: **Nickel plating** of 11 tubes. This operation, performed after the plating of
Doel 2, was aimed at completing the qualification of that repair method. Indeed, a
possible propagation of the cracks under the nickel, by OD corrosion, would be more
detrimental for a plant with roll transitions located above the tubesheet than for
Doel 2 which is part depth rolled. The result is favourable: no crack propagation has
been observed over the 3 cycles of operation already completed.

1988: Installation of **laser welded sleeves.** The purpose of this intervention
was to investigate, in the frame of a repair/replacement assessment, the installation
rates, the costs, the inspectability and the potential technical problems related to
sleeving. The advantages and inconveniences of a laser sleeving compared to nickel
plating are discussed in reference [12].

**Tihan 2**

1986: **Shot peening** of the tubes, hot leg side. This operation was performed
in the same condition as the Doel 3 (except that peening was performed for the full
height of the tubesheet), with the same results.

1988: **U-bends heat treatment of row 1 and row 2 tubes.** A large leak at the
level of U-bends in 1987 showed the sensitivity of the first rows to PWSCC. A
preventive treatment of 5 min at 760 °C was applied to relief the residual stresses and
possibly improve the corrosion resistance by precipitating carbides at the grain
boundaries.

1989: **Antivibration bars replacement.** This operation was essentially
preventive. It solved two problems: vibrational wear at contact points and potential
vibrational fatigue of unsupported bends. Indeed, the new design had lower gaps and
entered deeper into the tube bundle.

1989: **Chemical cleaning.** This treatment was applied after suppressing
copper from the secondary and replacing the condenser by a new one, equipped with
stainless steel tubes. The KWU process was applied satisfactorily. A description of the
treatment is given in reference [7].

**Doel 4**

1984: (Before plant start-up): **Roto-peening** of the tubes, hot leg side. This
treatment was performed to avoid the PWSCC problem evidenced by
Doel 3/Tihan 2. The preventive treatment proved to be very efficient but not
perfect as presently 2-3 % of the tubes are cracked while more than 60 % is affected in
a small sample left unpeened.
1984 : (Before plant start-up) : **Hydraulic expansion** of the tubes in the preheater TSP's in order to reduce possible vibrational wear.

1988 : **Shot peening** of the tubes, cold leg side. This purely preventive treatment was decided because the experience of Doel 3 and Tihange 2 had shown that the peening efficiency is low when performed after crack initiation.

1989 : **U-bends heat treatment.** This treatment was applied preventively on the row 1 and row 2 bends, in the same conditions as for Tihange 2.

**Tihange 3**

1985 (Before plant start-up) : **Roto-peening** of the tubes, hot leg and cold leg sides. The aim and results were the same as for Doel 4.

1985 : (Before plant start-up) : **Hydraulic expansion** of the tubes in the preheater TSP's, as in Doel 4.

1988 : **U-bends heat treatment.** This treatment was applied preventively on the row 1 and row 2 bends, in the same conditions as for Tihange 2.
7. CONCLUSIONS

In spite of the corrosion defects affecting the Belgian steam generators, good operating records in safe conditions have been achieved thanks to extensive R & D in NDE, innovative plugging criteria and large in-situ tests of repair techniques.

The major past concern was PWSCC at roll transition. In Doel 3 this has led to a replacement scheduled for 1993 because the expected repair costs, added to the production losses exceed the replacement costs. Recently, OD corrosion appeared to be another major threat. The future of the affected units will depend on the progression rate of these new defects, presently under assessment.
<table>
<thead>
<tr>
<th>Unit</th>
<th>Type/loops</th>
<th>Start-Up</th>
<th>SG model/supplier</th>
<th>Tubes/supplier</th>
<th>Expansion</th>
<th>Kiss-roll</th>
<th>Thot ΔT</th>
</tr>
</thead>
<tbody>
<tr>
<td>Doel 1</td>
<td>PWR/2 loops 400 MW</td>
<td>Jul. 74</td>
<td>Model 44 Westinghouse</td>
<td>7/8” MA Mannesmann</td>
<td>mech.roll 3 inches</td>
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<td>317 °C 30 °C</td>
</tr>
<tr>
<td>Tihange 1</td>
<td>PWR/3 loops 900 MW(CP0)</td>
<td>Feb. 75</td>
<td>Model 51 m Framatome</td>
<td>7/8” MA (1) Sandvick</td>
<td>mech.roll 3 inches</td>
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<td>320 °C 33 °C</td>
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<tr>
<td>Doel 2</td>
<td>PWR/2 loops 400 MW</td>
<td>Dec. 75</td>
<td>Model 44 Westinghouse</td>
<td>7/8” MA Mannesmann</td>
<td>mech.roll 3 inches</td>
<td>no</td>
<td>317 °C 30 °C</td>
</tr>
<tr>
<td>Doel 3</td>
<td>PWR/3 loops 900MW (CP1)</td>
<td>Aug. 82</td>
<td>Model 51 m Framatome</td>
<td>7/8” MA W-Blairsville</td>
<td>mech.roll full height</td>
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<tr>
<td>Tihange 2</td>
<td>PWR/3 loops 900MW(CP1)</td>
<td>Oct. 82</td>
<td>Model 51 m Framatome</td>
<td>7/8” MA W-Blairsville</td>
<td>mech.roll full height</td>
<td>yes</td>
<td>323 °C 37 °C</td>
</tr>
<tr>
<td>Doel 4</td>
<td>PWR/3 loops 1000 MW</td>
<td>Jun. 85</td>
<td>Model E Westinghouse</td>
<td>3/4” MA W-Blairsville</td>
<td>mech.roll full height</td>
<td>yes</td>
<td>326 °C(2) 36 °C</td>
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<tr>
<td>Tihange 3</td>
<td>PWR/3 loops 1000 MW</td>
<td>Jul. 85</td>
<td>Model E Westinghouse</td>
<td>3/4” MA W-Blairsville</td>
<td>mech.roll full height</td>
<td>yes</td>
<td>330 °C 36 °C</td>
</tr>
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</table>

(1) = in sensitized condition

(2) = 330 °C until 1990
<table>
<thead>
<tr>
<th>Unit</th>
<th>OD Corrosion</th>
<th>Vibration wear</th>
<th>AVB</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Preheater</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Support plates</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Sludge pile</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Transition zone</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Tubesheet crevice</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Transition zone</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Doel 1</td>
<td>axial SCC *</td>
<td>axial SCC *</td>
<td>IGA/SCC **</td>
</tr>
<tr>
<td>Thange 1</td>
<td>axial SCC *</td>
<td>axial SCC *</td>
<td>IGA/SCC</td>
</tr>
<tr>
<td>Doel 2</td>
<td>axial SCC *</td>
<td>axial SCC *</td>
<td>IGA/SCC</td>
</tr>
<tr>
<td>Thange 2</td>
<td>axial SCC *</td>
<td>axial SCC *</td>
<td>IGA/SCC</td>
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<tr>
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<td>axial SCC *</td>
<td>axial SCC *</td>
<td>IGA/SCC</td>
</tr>
<tr>
<td>Thange 3</td>
<td>axial SCC *</td>
<td>axial SCC *</td>
<td>IGA/SCC</td>
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<tr>
<td>Doel 4</td>
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<td>axial SCC *</td>
<td>IGA/SCC</td>
</tr>
<tr>
<td>Thange 4</td>
<td>axial SCC *</td>
<td>axial SCC *</td>
<td>IGA/SCC</td>
</tr>
</tbody>
</table>

* = with leak indication
** = tube rupture
(1) = before AVB replacement
(2) = to be confirmed
### TABLE III - FORCED OUTAGES DUE TO STEAM GENERATORS LEAKAGE

<table>
<thead>
<tr>
<th>Unit</th>
<th>Date</th>
<th>Duration</th>
<th>Cause</th>
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<tbody>
<tr>
<td>Tihange 1</td>
<td>SEPT 81</td>
<td>12 days</td>
<td>Loose part</td>
</tr>
<tr>
<td></td>
<td>NOV 83</td>
<td>15 days</td>
<td>AVB Wear</td>
</tr>
<tr>
<td></td>
<td>JUL 84</td>
<td>7 days</td>
<td>AVB Wear</td>
</tr>
<tr>
<td>Doel 1</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Doel 2</td>
<td>JUN 79</td>
<td>35 days</td>
<td>SGTR Row 1</td>
</tr>
<tr>
<td></td>
<td>AUG 82</td>
<td>7 days</td>
<td>Leaking plugs</td>
</tr>
<tr>
<td></td>
<td>OCT 82</td>
<td>7 days</td>
<td>SW SCC</td>
</tr>
<tr>
<td></td>
<td>NOV 82</td>
<td>Coupled with other operations</td>
<td>Circ. PWSCC at mini sleeves</td>
</tr>
<tr>
<td></td>
<td>JUN 86</td>
<td>8 days</td>
<td>Leaking plugs</td>
</tr>
<tr>
<td></td>
<td>JUL 86</td>
<td>7 days</td>
<td>Leaking plugs</td>
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<tr>
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<td>JUN 87</td>
<td>Outage extension 9 days</td>
<td>Leaking plugs</td>
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<td>JUL 87</td>
<td>9 days</td>
<td>Loose part</td>
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<tr>
<td></td>
<td>FEB 90</td>
<td>Outage extension 14 days</td>
<td>PWSCC at transition</td>
</tr>
<tr>
<td>Doel 3</td>
<td>MAY 87</td>
<td>13 days</td>
<td>PWSCC at transition</td>
</tr>
<tr>
<td></td>
<td>AUG 87</td>
<td>3 days</td>
<td>Ghost leak</td>
</tr>
<tr>
<td>Tihange 2</td>
<td>MAY 87</td>
<td>7 days</td>
<td>PWSCC U bend</td>
</tr>
<tr>
<td></td>
<td>FEB 90</td>
<td>7 days</td>
<td>PWSCC at transition</td>
</tr>
<tr>
<td>Doel 4</td>
<td>-</td>
<td>-</td>
<td>-</td>
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<tr>
<td>Tihange 3</td>
<td>-</td>
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</tbody>
</table>

**NOTE:** In addition the refueling outage was advanced twice in Tihange 2 due to a PWSCC leak:
- 1985: advance of 12 days
- 1989: advance of 8 days

2.1-16/23
Figure 2 - LEAK MEASUREMENT METHODS
FIG. 3 - COMPARISON OF DIFFERENT MEASUREMENTS OF THE IN-SERVICE LEAK.

- N 16 (SG-B)
- Na 24 (SG-B)
- F 18 (SG-B)
- H 3 (TOTAL OF 3 SG'S)

2.1-19/23

15 12 9 7 4 2 30 30 27 24 24 21
jul aug sep okt nov dec dec jan feb mrt apr 1990 1991
FIG. 4 - COHERENCE BETWEEN THE DIFFERENT LEAK MEASUREMENT METHODS (DOEL 3 UNIT)

FREQUENCY OF OBSERVATION OVER 3 CYCLES (%)

- N16 (SG - B)
- Na24 (SG - B)
- F18 (SG - B)
- Ar41 (TOTAL OF 3 SG'S)
- H3 (TOTAL OF 3 SG'S)

MEASUREMENT "ERROR" (SINGLE METHOD MEAN OF 5)
FIG 5 - COMPARISON BETWEEN RPC CRACK LENGTHS, SUCCESSIVE FLUORESCIN LEAK TESTS AND A HELIUM LEAK TEST.
FIG. 6 - DISTRIBUTION OF THE NUMBER OF CRACKS (RPC)

FIG. 7 - DISTRIBUTION OF HELIUM LEAKS (ALL LEAKS)
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RECENT OPERATING EXPERIENCE WITH
STEAM GENERATORS IN JAPAN

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Abstract

As of June, 1991, 18 PWR units are in operation in Japan, and additional 5 PWR units are under construction. Since Mihama-1, the first PWR unit in Japan was commissioned to service in 1970, the cumulative operating years of PWR units in Japan has reached approximately 190 reactor-years. During this period, various tube degradation troubles have been experienced with steam generators.

In this paper, the status of occurrence of such tube degradations, as well as in-service inspection, maintenance, research programs, etc. which are related to these troubles, are generally introduced.
1. Introduction

As of June, 1991, 18 PWR plants, with total capacity of 13,756 MWe, and 21 BWR plants, with total capacity of 18,137 MWe, plus 1 GCR plant having 16.6 MWe capacity, or a total of 40 nuclear power plants, with the combined capacity of 32,059 MWe, are in operation. In addition, 5 PWR plants with total capacity of 5,610 MWe, and 4 BWR plants with total capacity of 3,877 MWe, that is, the total of 9 nuclear power plants with aggregate capacity of 9,487 MWe are under construction.

Following the commissioning of a GCR plant, which is the first commercial nuclear plant, in Japan in 1966, a BWR plant and a PWR plant were commissioned to service in 1970, and the development of LWR plants has been aggressively pursued until the present time. During this period, in the early 1970's, the annual capacity factor of nuclear power plants has been reduced down to levels around 50%, due to tube wall thinning of PWR plants' steam generator tubing and the trouble in BWR plants caused by stress corrosion cracking and other factors. However, owing to various measures that have been taken to prevent these troubles, and together with design and manufacturing improvements, implementation of preventive maintenance, as well improved operating procedures, the annual capacity factor has exceeded 70% in 1983, and stayed at this level ever since.

When we look at the number of occurrence of incidents and failures, the number of events that have been reported to the Ministry for International Trade and Industry (MITI) in accordance to the legal provision has stayed at a level of 0.5 to 0.6 events/plant-year since 1984, as illustrated in Table 1. However, almost all incidents and failures that have been detected during plant shutdown (mainly for periodical inspections) are related to steam generators. These are the tube degradations which have been found by the steam generator tubing eddy current testing that is implemented during the periodical inspection. Currently, tube degradations are found in specific PWR plants, and shutdown period of these plants are longer, because corrective maintenance such as sleeving or plugging must be done in these plants.

In February 1991, the first steam generator tube rupture has been experienced in Japan at Mihama-2 Plant.

Under the situation as described above, maintaining integrity of steam generator tubing is currently the most relevant factor in assuring the reliability and safety of plants. The design features of steam generators of PWR plants which are in operation and under construction are presented in Tabla II for the purpose of reference.
2. History of Tube Degradation

Since the first wall thinning on the secondary side was experienced at Mihama–1 in 1972, tube degradation cases such as described below have occurred on the primary side and secondary side up to the present time.

(1) Tube Degradation of Primary Side

a. Stress Corrosion Cracking at Tight U–Bend

The stress corrosion cracking was first discovered on the row 1 tube of Takahama–1 in 1977, and the same phenomena were identified in Ohi–1 and Mihama–2 later.

b. Stress Corrosion Cracking at Roll Expansion Regions and Roll Transition Zones

The presence of stress corrosion cracking was discovered at the roll expansion regions of Ohi–1 in 1982, and also found at roll transition zones of Takahama–1. Later, the presence was also confirmed at Mihama–2 and –3, Ohi–2 and Ikata–1.

(2) Tube Degredation of Secondary Side

a. Wall Thinning

The occurrence of wall thinning was first experienced at Mihama–1 in 1972. Later, wall thinning was also identified in Mihama–2 and Takahama–1 plants in which the same phosphate treatment as Mihama–1 had been applied in secondary water chemistry.

b. Stress Corrosion Cracking in Tube Sheet Crevice

In 1977, the first stress corrosion cracking in tube sheet crevice was experienced in Takahama–1. Later, similar occurrence is experienced in Mihama–2.

2.2–3/10
c. Intergranular Attack at Tube Support Plate and Just Below Top of Tube Sheet

The intergranular attack was experienced in 1981 at Genkai-1. Later, the same is experienced at Mihama-2 and -3, Takahama-1 and -2, and Ohi-1.

d. Fretting Wear at Anti Vibration Bar

The fretting wear occurred at Ikata-1 in 1981. Later, the same phenomena is experienced at Mihama-2 and -3, Ohi-2 and Takahama-3 and -4.

e. Fatigue Cracking

A circumferential cracking, which cause was fatigue, was experienced on the upper end of top tube support plate at cold side of Mihama-2 in 1991.

The history of the number of tubes which has been plugged due to degradations as described above is illustrated in Figure 1. In the early years when PWR plants started operation in Japan, the major cause that compelled tube plugging was tube wall thinning. After the secondary water chemistry was switched from phosphate treatment to AVT in 1975, some number of tubes have been plugged due to stress corrosion cracking occurring at tube sheet crevice and tight U-bend parts. After 1981, the number of tubes that have been plugged due to stress corrosion cracking at tube expansion region and intergranular attack at tube support plate crevice has increased. In particular, the occurrence of degradation due to the latter cause turned out to be a serious problem for some plants, that is, Takahama-2, Ohi-1 and Genkai-1.

3. Inspection for Steam Generator Tubing

In Japan, it is mandatory by law that nuclear power plants are subjected to the inspection performed by the MITI to confirm their safety and integrity. In other words, the electric utilities are legally obliged to have their plants subjected to the "inspection before use" at the time of construction, and to the "periodical inspection" after they are commissioned to service. Naturally, steam generators are subjected to these inspection.
In the inspection before use, all the steam generator tubing are subjected to the eddy current testing for the full length of the tubes in order to confirm they are manufactured according to the design and also in order to obtain the base data for periodical inspections.

It is set forth that a periodical inspection on nuclear power plants are to be conducted during the period not exceed one (1) month before or after the date which is one (1) year after the completion of the previous periodical inspection. In the inspection on steam generator tubing, all tubes are subjected to the eddy current testing for the full length of tubes. The standard technique applied to the inspection is such that the data ara obtained by the multi-frequency eddy current instrumentation employing the differential bobbin coil probe, which are analyzed by a computer data screening analysis system to identify defects. In actual practice, additional probes, which are suitable to the degradation modes and locations to be detected, are used.

4. Remedial Action for Tube Degradation

(1) Investigation of Cause and Countermeasure

When a new kind of tube degradation is first discovered by eddy current testing which is performed in a periodical inspection, the usual procedure to be followed is to identify the defective tubes, clarify the cause and study the countermeasure for prevention of recurrence. That is, the defective tubes are pulled out from the steam generator, the degradation phenomena are confirmed, and investigations such as metallurgical examination, scale analysis and necessary examinations are performed, accompanied by the study and evaluation of the cause of occurrence to infer the real cause. In some cases, verification tests are conducted to identify the cause. The countermeasure for prevention of recurrence is formulated based on the results of such studies, tests, etc.

The clarification of cause and the formulation of countermeasures for prevention of recurrence, such as described above, are implemented at the primary responsibility of the electric utilities, and MITI, which is the regulatory authority, reviews the countermeasures in terms of their safety. In this procedure, the opinion of the Steam Generator Review Committee, which is subject to the Advisory Committee on Nuclear Power Generation, an advisory organization to the Ministry of MITI, is requested for advice, as appropriate.
(2) Preventive and Corrective Countermeasure for Defective Tubes

The technical standard which is set down in Japan is the standard that is referred to in design and manufacture, and at the same time it is the standard of the level at which the electric facilities are to be maintained and operated. According to this principle, a plant equipment on which a defect was found should not be operated. Therefore, all defective tubes on which defect is detected by eddy eddy current testing are repaired by plugging or sleeving, even if some of them are usable in terms of mechanical strength.

In the early years, explosive plug and welding plug were used for repair. Since 1981, mechanical plug was used in order to reduce the amount of radiation exposure as well as the working hours. Then, from 1989, plugs of Alloy 690 are used in order to reduce the sensitivity to stress corrosion cracking of mechanical plugs.

In 1980, the sleeving instead of plugging was first applied in order to avoid the reduction of allowable plugging margin of steam generator. At first, the welding sleeve was used for the tubes which stress corrosion cracking has occurred at the tube sheet crevice, and then the mechanical sleeve, with which the repair work is easier, was employed. Later, since 1984, brazing sleeve was used to repair the intergranular attack occurring in the tube support plate crevice. Then, laser welding sleeve was put into use since 1989.

(3) Preventive Maintenance

With the objective of preventing the stress corrosion cracking of the secondary side occurring at tube sheet crevice, all crevices on the hot leg side of steam generators in 4 nuclear power plants have been blocked by re-rolling during the period from 1981 to 1984.

In dealing with the stress corrosion cracking on the primary side occurring at tight U-bend, it was difficult to take effective countermeasure for prevention of recurrence as this was mainly due to the residual tensile stress produced during manufacturing. Therefore, zone plugging was implemented at Takahama-1 and Ohi-1.

In addition, in order to prevent the fretting wear occurring at antivibration bars, the antivibration bars were replaced at 2 plants of Takahama-3 and -4 in 1990 and 1991.
5. Research for Prevention of Intergranular Attack

Various research projects have been implemented as national projects and joint projects by electric utilities and manufacturers with the objective of maintaining and enhancing the integrity of steam generator tubing. In particular, the following major projects have been implemented or being carried out with the objective of preventing the intergranular attack which is currently the focus of attention.

- Research on prevention of intergranular attack on Alloy 600.
- Research on various chemical substances which affect the sensitivity to intergranular attack.
- Research on concentration and precipitation behaviors of trace impurities in water.
- Research on the effect of various oxides that affect the electric potential of Alloy 600 in alkaline water.
- Research on development of sleeving technique
- Research on verification of effectiveness of intergranular attack prevention measures.
- Research on chemical cleaning of the steam generator secondary side.

The achievements gained in these researches are being applied to actual Plants, but the effectiveness of these measures have not been verified.

6. Conclusion

In this paper, the operating experience of steam generators in Japan has been generally discussed. The phenomenon which is of gravest interest at the present moment for maintaining the reliability of steam generators is the intergranular attack, and although various measures have been applied, we have not yet arrested this phenomenon completely. For this reason, the issue of replacing steam generators, on which the occurrence of intergranular attack is pronounced, is being considered by electric utilities.

The presumptive cause of steam generator tube rupture which occurred at Mihama-2 is the tube vibration caused by improper setting of anti-vibration bars, which lead to the fatigue crack.
Table 1  Number of Reported Incidents and Failures

(under the provision of laws)

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<th>During Operation</th>
<th>Discovered During Shutdown</th>
<th>Others</th>
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<td>Automatic Shutdown</td>
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<td>Except SG Tubing</td>
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<tr>
<td>Genkai -3</td>
<td></td>
<td></td>
<td>4</td>
<td>Under</td>
<td></td>
<td>Drill+ Chamfer</td>
<td>AVT</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Ikata -3</td>
<td></td>
<td></td>
<td>3</td>
<td>Under</td>
<td></td>
<td>Drill+ Chamfer</td>
<td>AVT</td>
<td></td>
<td></td>
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<tr>
<td>Genkai -4</td>
<td></td>
<td></td>
<td>4</td>
<td>Under</td>
<td></td>
<td>Drill+ Chamfer</td>
<td>AVT</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
Figure 1. Tube Plugging History
STATUS OF STEAM GENERATORS IN SPAIN

José Mª Zamarrón
Central Nuclear de Almaraz
Madrid (Spain)

There are a total of nine operating nuclear units in Spain, seven of the PWR type. To deal with the steam generators problems of these plants its owners are spending considerable investments to improve the secondary cycle (i.e. copper removal) and to maintain and inspect the units. There is also a considerable effort in research and development in: repair methods qualification (plugging, sleeving, chemical cleaning), improving reliability of inspection technics (E. Current, U.T., Laser), tube pulled examination, corrosion tests, plugging criteria development, and management tools (data base, degradation model). Due to the high operation and maintenance costs of maintaining the Steam Generators, four units are planning to replace them in the period 1995-1998.
1. INTRODUCTION

There are a total of nine operation nuclear plants in Spain totalling 7.350 Mwe. These units produced 54.265 x 10^6 Kwh in 1990, 36% of the total generation in Spain. Seven of these plants are of the PWR type.

The first plant in operation was Jose Cabrera (ZORITA) in 1968, one loop Westinghouse plant with a model 24 Steam Generator. Due to the design margin and careful operation of the Steam Generator of this plant its performance have been very good, with only 5% tubes plugged after 23 years of operation. This is one of the few units in the world that remains in fosfate chemistry.

During the period 1981-1985 a total of four units, two in Almaraz and two in Ascó entered in operation. These three loops Westinghouse units use model D-3 preheater Steam Generators. The poor design and manufacture of the Steam Generators of these units have caused a large number of problems: mechanical (Preheater and AVB's vibration), denting, and primary and secondary stress corrosion cracking. As a consequence, the owners of these units have had to expend considerable effort in: safety analysis, preventive and corrective maintenance and operation of these units. Some of the many activities performed have been; copper removal from secondary side, Reverse Osmosis in the make up plant, chemical cleaning of the Steam Generators, stress relief and shotpeening of Steam Generators tubes, sleeves installation, improved operation and chemistry, licensing new plugging criteria, etc.

Finally two units entered into operation in 1988: Vandellós 2, a three loops model F Steam Generators Westinghouse reactor, and Trillo, a three loops Siemens reactor.

Since its start up Vandellós 2 have experienced the problem of quite extensive AVB's freeting and a replacement of AVB's is scheduled for its next outage.

The Trillo plant has only experienced a small leak at one of its Steam Generators, due to wear caused by a loose part probably left during manufacturing.

To deal with all these Steam Generators problems all the Spanish plant owners formed the Steam Generators Comitee, that coordinates all the efforts in this field in Spain including: information exchange, participation in international projects, coordinated responses to the licensing authorities, and research and development.
A very important issue in the Steam Generators field is the international cooperation between countries with similar problems. The Spanish Steam Generators Comitee participates as a group in the EPRI's "Steam Generators Reliability Project". There is a lack of a similar European Program. In that absence we maintain many bilateral contacts with other European Countries.

Due to the importance of the problems presented and the number of plants affected there is a considerable effort in Research and Development in Steam Generators in Spain spending an average of $3 \times 10^6$ per year.

The Spanish Research and Development Program on Steam Generators is financed by the "Plan de Investigación Electrotécnico (PIE)" and is being developed by several Spanish companies and research centers (CIEMAT, ENSA and TECNATOM) and the Spanish utilities through the "Steam Generators Comitee".

Some of the most significant works performed to date are: Steam Generators management tools, improvements on Eddy Current inspection reliability, tube pulled examination, corrosion tests, plugging criteria (both for PWSCC at tube sheet and ODSCC at support plates) and repair techniques qualification (plugs, sleeves and chemical cleaning).

All the investments, inspection and maintenance, research and development and international cooperation performed have improved the situation of the Spanish Steam Generators. Nevertheless for some of them the efforts performed have not been sufficient to assure its design life. As a consequence the plants of Almaraz 1 and 2 and Asco 1 and 2 are, in this moment, in the process of evaluating bids for replacing Steam Generators during the period 1995-1998.
STATUS OF STEAM GENERATORS IN SPAIN
# Spanish PWR Nuclear Plants

<table>
<thead>
<tr>
<th>Generation</th>
<th>Plant</th>
<th>MW</th>
<th>NSSS</th>
<th>S.G. Model Constr.</th>
<th>S.G. Tube Material</th>
<th>Commercial Operating</th>
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<tbody>
<tr>
<td>1st.</td>
<td>Zorita</td>
<td>160</td>
<td>W</td>
<td>24</td>
<td>I-600 MA</td>
<td>1968</td>
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<tr>
<td>2nd.</td>
<td>Almaraz I</td>
<td>930</td>
<td>W</td>
<td>D-3</td>
<td>I-600 MA</td>
<td>1981</td>
</tr>
<tr>
<td></td>
<td>Almaraz II</td>
<td>930</td>
<td>W</td>
<td>D-3</td>
<td>I-600 MA</td>
<td>1983</td>
</tr>
<tr>
<td></td>
<td>asco I</td>
<td>930</td>
<td>W</td>
<td>D-3</td>
<td>I-600 MA</td>
<td>1984</td>
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<tr>
<td></td>
<td>asco II</td>
<td>930</td>
<td>W</td>
<td>D-3</td>
<td>I-600 MA</td>
<td>1985</td>
</tr>
<tr>
<td>3rd.</td>
<td>Trillo</td>
<td>1,030</td>
<td>KWU</td>
<td>54-GT</td>
<td>I-800</td>
<td>1988</td>
</tr>
<tr>
<td></td>
<td>Vandellos II</td>
<td>980</td>
<td>W</td>
<td>F</td>
<td>I-600 T.T.</td>
<td>1988</td>
</tr>
</tbody>
</table>
SPANISH
STEAM GENERATOR
RESEARCH PROJECT
MAIN TASKS

* S.G. DATA BASE AND DEGRADATION MODEL
  - PWSCC
  - IGA / ODSCC

* N.D.T. TECHNIQUES
  - E.C.
  - U.T.
  - LASER (PWSCC)

* PULLED-OUT TUBE EXAMINATION

* CORROSION TESTS
  - PRIMARY ENVIROMENTS
  - SECONDARY ENVIROMENTS
  - ALTERNATE S.G. TUBE MATERIALS

* PLUGGING CRITERIA
  - PWSCC
  - IGA / ODSCC

* EVALUATION OF MITIGATION AND REPAIR TECHNIQUES
  - CHEMICAL CLEANING
  - REMOVABLE PLUGS
  - SLEEVES
<table>
<thead>
<tr>
<th>ANALYSIS OF TUBE DEGRADATION EVOLUTION</th>
</tr>
</thead>
</table>

Fecha: SEP/91
ANALYSIS PERFORMED AS FUNCTION OF EFPY:

- TUBES WITH PWSCC INDICATIONS 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.
- ASCO, UNITS 1 AND 2.
Tubos afectados PWSCC

Mu (Media de la distribucion del neperiano) = 2.935;  
Sigma (Desviacion de la distribucion del neperiano) = 0.803  
Coeficiente de correlacion = 0.974;  Coeficiente de determinacion = 0.949
Tecnatom * Central Nuclear de Almaraz I * Generador de Vapor 1

Tubos afectados PWSCC

Papel Weibull (y:x=1:1)

Beta (Factor de forma) = 3.181 ; Lambda (Factor de escala) = 0.078 ; Vida caracteristica (Mu=1/lambda) = 12.866

Coeficiente de correlacion = 0.961 ; Coeficiente de determinacion = 0.923
Tecnatom * Central Nuclear de Almaraz I * Generador de Vapor 1

Tubos afectados PWSCC

Papel Gumbel (y:x=7:1)

Porcentaje acumulado

0.1 0.5 1.0 5.0 10.0 20.0 50.0

0. 2. 4. 6. 8. 10. 12. 14. 16. 18. 20. 22. 24. 26. 28. 30. 32. 34. 36. 38. 40. 42. 44. 46. 48. 50.

CEFPY (Cumulative Effective Full Power Year)

\[ a = 8.844; \quad b = 1.210 \]

Coeficiente de correlación = 0.916; Coeficiente de determinación = 0.840
N. D. T. TECHNIQUES

E. C. TECHNIQUE

- BOBBIN COIL
- RPC AND 3 x RPC

U. T. TECHNIQUE: (UNDER DEVELOPMENT)

- MIXED MODE CRACKING (AXIAL AND CIRCUNFERENTIAL)
- SLEEVE INSPECTION

LASER TECHNIQUE: (UNDER DEVELOPMENT)

- PWSCC MIXED MODE CRACKING
  * VERY HIGH RESOLUTION AND VERY LOW MEASUREMENT ERROR
**PRIMAY WATER STRESS CORROSION CRACKING TESTS**

**OBJECTIVE**

TO STABLISH THE INFLUENCE OF Li CONCENTRATION AND pH IN THE GENERATION AND PROPAGATION OF PWSCC.

**TEST CONDITIONS**

- OPERATING TEMPERATURE : 330 °C
- CONCENTRATIONS (p.p.m.):

<table>
<thead>
<tr>
<th></th>
<th>B (ppm)</th>
<th>1400</th>
<th>740</th>
<th>385</th>
<th>0</th>
<th>H₂ (cc kg/H₂O)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Li (ppm)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>3,5</td>
<td>O</td>
<td>*</td>
<td>O</td>
<td></td>
<td></td>
<td>23</td>
</tr>
<tr>
<td></td>
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<td>*</td>
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<td>O</td>
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<td>45</td>
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<td>0,7</td>
<td>O</td>
<td>O</td>
<td>O</td>
<td></td>
<td></td>
<td>23</td>
</tr>
</tbody>
</table>

* : COMPLETED
O : IN PROGRESS
* INTERGRANULAR ATTACK IN SECONDARY SIDE

. OBJECTIVE

TO STUDY THE EFFECT OF CONTAMINANTS PRESENT IN THE S.G. SECONDARY SIDE IN THE GENERATION OF IGA.

. TEST CONDITIONS:

- 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 ENVIROMENT (Ca)

KWU MODEL MOCK-UP

- AVT + ALKALINE ENVIROMENT (SULFUR)

DENTED MOCK-UP

- GENERATION OF ODSCC BY DENTING (AVT + Cu + Cl + O₂)
ALTERNATE S. G. TUBE MATERIAL

OBJECTIVE

TO STUDY COMPARATIVELY THE SUSCEPTIBILITY OF INCONEL 690 TT AND INCOLOY 800 Mod.
IN CAUSTIC AND ACIDIC ENVIROMENTS WITH SULFUR, COPPER AND LEAD

TESTS CONDITIONS

TEST SAMPLES : C-ring; 2 % STRAIN
TEMPERATURE : 350°C
ENVIROMENTS : CAUSTIC AND ACIDIC

TUBE MATERIALS

- INCONEL 690 TT
- INCOLOY 800 AND 800 SP
- INCONEL 600 MA (AS REFERENCE)
INCOLOY 800

MAXIMUM DEPTH
% WALL THICKNESS

INCONEL 690 TT

MAXIMUM DEPTH
% WALL THICKNESS
PLUGGING CRITERIA:

* PLUGGING CRITERIA APPLIED TO PWSCC IN R. T. Z. (L. B. B.)

OBJECTIVE: ANY RISK IN THE MOST SEVERE CONDITIONS OF A S.G. TUBE DUE TO PWSCC IN THE R. T. Z., IS ALWAYS PRECEDED BY AN ADMISSIBLE LEAK IN NORMAL OPERATION

HYPOTHESIS: ALL THE PRIMARY TO SECONDARY LEAK IN THE S.G. IS COMING FROM ONE CRACK

SPECIFIC EXPERIENCE: TYPE OF DEFECTS AND EVOLUTION

SPECIFIC STUDIES: THEORICAL ANALYSIS AND EXPERIMENTAL TESTS

SURVEILLANCE RULES: DURING PLANT OPERATION (LEAK RATE) AND REFUELING OUTAGE (CRACK LENGTH)
CRACK CONFIGURATION STUDIED

A: SINGLE CRACK
B: MULTIPLE PARALLEL CRACKS (N ≤ 20)
C: LONGITUDINAL CRACK IN CORRODED AREA
1.- CRITICAL CRACK LENGTH ANALYSIS

OBJECTIVE: TO DETERMINE THE MAXIMUM ADMISSIBLE CRACK LENGTH IN S.G. TUBES WITHOUT RISK OF RUPTURE

PHASES: EXPERIMENTAL AND THEORETICAL

BURST TESTS USED TO VALIDATE MECHANICAL RESISTANCE CRITERIA

ANALYSIS OF OPERATION TRANSIENTS. SELECTION OF MOST SEVERE CONDITIONS

SELECTION OF TUBE MECHANICAL PROPERTIES

ANALYSIS OF THE INFLUENCE OF TUBE GEOMETRY
2.- CALCULATION OF LEAK RATES

- CRACK OPENING AREA DETERMINATION
  
  * THECNICAL ANALYSIS (BY CALCULATION AND FINITE ELEMNTS)

- LEAK RATE DETERMINATION
  
  \[ Q = 3600 \times K \times S \sqrt{\frac{2 \Delta P}{\rho}} \]

  * K DETERMINE BY LEAK TESTS
RESULTS:

1. CRITICAL LENGTH FOR MOST UNFAVOURABLE GEOMETRY
   
   \[ L = 12.6 \text{ mm} \]

2. LEAK RATE DETERMINATION
   
   \[ K = 0.1 \]
   
   \[ Q_c = 30 \text{ l/h} \]

3. MAXIMUM ALLOWABLE CRACK LENGTH
   
   \[ L_A = L_C - L_G - L_E \]
   
   \[ L_A = 12.6 - 4.0 - 1.0 \]
   
   \[ L_A = 7.6 \text{ mm} \]
EVALUATION OF THE INFLUENCE OF THE PRESENCE OF CIRCUMFERENTIAL CRACKS IN THE APPLICATION OF THE PLUGGING LIMIT OF S. G. TUBES AXIAL PWSCC IN ROLL TRANSITION WITH.
OBJECTIVES:

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 PWS CC IN ROLL TRANSITION.
INFORMATION OF PULL TUBES IN SPAIN, REGARDING CIRCUMFERENTIAL CRACKS

. THE CIRCUMFERENTIAL CRACKS MAY BE ID OR OD INITIATED.

. IN ALL CASES, THE CIRCUMFERENTIAL CRACKS DETECTED BY E.C.T. WERE SMALLER THAN CRITICAL SIZE.

. IN ALL CASES THAT CIRCUMFERENTIAL CRACKS WERE DETECTED, IT WAS BY R.P.C.

. IN MOST OF THE CASES, THE CIRCUMFERENTIAL CRACK WAS THE ONLY TYPE OF CRACKING PRESENT IN THE ROLL TRANSITION.

. ONLY IN A SINGLE CASE, THERE WERE A CIRCUMFERENTIAL AND AXIAL CRACKS IN THE SAME ZONE, HOWEVER THEY DID NOT INTERCEPTED.

. IN ALL CASES, THE CIRCUMFERENTIAL CRACK WAS BELOW THE TOP OF TUBE SHEET.
## Burst Test Matrix (3/4” Tubing)

### Single and Mixed Mode Cracking

<table>
<thead>
<tr>
<th>L (mm)</th>
<th>α (degrees)</th>
<th>d/t (%)</th>
<th>e</th>
<th>P (bar)</th>
<th>Notes</th>
</tr>
</thead>
<tbody>
<tr>
<td>13</td>
<td>--</td>
<td>--</td>
<td>--</td>
<td>421</td>
<td>1 axial slot</td>
</tr>
<tr>
<td></td>
<td>270</td>
<td>54</td>
<td>--</td>
<td>650</td>
<td>1 circ. slot</td>
</tr>
<tr>
<td></td>
<td>360</td>
<td>49</td>
<td>--</td>
<td>820</td>
<td>1 circ. slot</td>
</tr>
<tr>
<td>9</td>
<td>360</td>
<td>63</td>
<td>--</td>
<td>404</td>
<td>1 axial crack</td>
</tr>
<tr>
<td>12</td>
<td>360</td>
<td>61</td>
<td>--</td>
<td>341</td>
<td>1 axial crack</td>
</tr>
<tr>
<td>13</td>
<td>270</td>
<td>30</td>
<td>3 mm.</td>
<td>355</td>
<td>axial burst</td>
</tr>
<tr>
<td>13</td>
<td>270</td>
<td>38</td>
<td>90°</td>
<td>305</td>
<td>axial burst</td>
</tr>
<tr>
<td>13</td>
<td>270</td>
<td>40</td>
<td>90°</td>
<td>310</td>
<td>axial burst</td>
</tr>
<tr>
<td>13</td>
<td>360</td>
<td>35</td>
<td>3 mm.</td>
<td>339</td>
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<tr>
<td>13</td>
<td>270</td>
<td>47</td>
<td>3 mm.</td>
<td>305</td>
<td>Flap, no axial burst</td>
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<tr>
<td>13</td>
<td>360</td>
<td>50</td>
<td>3 mm.</td>
<td>310</td>
<td>Flap, no axial burst</td>
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<tr>
<td>11</td>
<td>360</td>
<td>45</td>
<td>3 mm.</td>
<td>433</td>
<td>axial burst</td>
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<tr>
<td>11.8</td>
<td>360</td>
<td>80</td>
<td>3 mm.</td>
<td>333</td>
<td>axial burst</td>
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<tr>
<td>12.2</td>
<td>360</td>
<td>85</td>
<td>3 mm.</td>
<td>274</td>
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<td>13</td>
<td>360</td>
<td>76</td>
<td>3 mm.</td>
<td>390</td>
<td>Circ. inside T.S.</td>
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<tr>
<td>12.85</td>
<td>360</td>
<td>58</td>
<td>90°</td>
<td>380</td>
<td>Circ. inside T.S.</td>
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<tr>
<td>12.8</td>
<td>265</td>
<td>89</td>
<td>90°</td>
<td>226</td>
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<td>12.5</td>
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<td>94</td>
<td>90°</td>
<td>207</td>
<td>Flap, no axial burst</td>
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<tr>
<td>10.5 / 8.15</td>
<td>360</td>
<td>76</td>
<td>90°</td>
<td>360</td>
<td>Flap, no axial burst</td>
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<tr>
<td>11.1</td>
<td>360</td>
<td>57</td>
<td>90°</td>
<td>330</td>
<td>Flap, no axial burst</td>
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<tr>
<td>11 / 4.8</td>
<td>360</td>
<td>80</td>
<td>90°</td>
<td>330</td>
<td>Guillotine seizure</td>
</tr>
<tr>
<td>8</td>
<td>360</td>
<td>91</td>
<td>90°</td>
<td>250</td>
<td>Guillotine seizure</td>
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<tr>
<td>6</td>
<td>360</td>
<td>77</td>
<td>90°</td>
<td>390</td>
<td>Guillotine seizure</td>
</tr>
</tbody>
</table>
CONCLUSIONS:

- CIRCUMFERENTIAL CRACK WITHIN THE TUBE SHEET (1 mm. BELOW THE TOP OF TUBE SHEET).
  * THE MODE OF TUBE RUPTURE IN THIS CASE IS EQUIVALENT TO THE TUBE RUPTURE OF AXIALLY CRACKED TUBES.

- CIRCUMFERENTIAL CRACK OUT OF TUBE SHEET (1 mm. ABOVE THE TOP OF TUBE SHEET).
  * \( d/t \leq 40\% \): THE MODE OF TUBE RUPTURE IS EQUIVALENT TO THE TUBE RUPTURE OF AXIALLY CRACKED TUBES, WITH A SMALL REDUCTION IN THE RUPTURE PRESSURE.
  * \( d/t > 40\% \): IN GENERAL, A "FLAP" IS PRODUCED IN THE AFFECTED AREA. IF THE LENGTH OF THE AXIAL CRACK IS LONGER THAN 11 mm., THERE IS A SIGNIFICANT REDUCTION OF THE REINFORCEMENT EFFECT OF THE TUBE SHEET ON THE TUBE.
  * \( d/t \geq 80\% \) AND \( L(\text{AXIAL}) \leq 11 \text{ mm.} \): THE MODE OF TUBE RUPTURE IS IN GUILLOTINE SEIZURE.
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).

(FLOW DISTRIBUTION BAFFLE IS NOT INCLUDED)

MAIN TASKS:

- ANALYSIS OF ISI RESULTS
- ANALYSIS OF TUBE INTEGRITY:
  - BURST TESTS
- LEAK TESTS
- SAFETY ANALYSIS
- PLANT AVAILABILITY
- PLUGGING CRITERIA
ANALYSIS OF ISI RESULTS

- AXIAL CRACKS ARE ALWAYS LOCATED WITHIN THE TUBE SUPPORT PLATE WIDTH.

- AVERAGE CRACK LENGTH GROWTH IS ≈ 1.5 mm (BUT NOT IMPORTANT SINCE CRACKS ARE ALWAYS IN T.S.P WIDTH).

- AVERAGE CRACK DEPTH GROWTH IN ONE CYCLE IS ≈ 7% OF W.T.

- ERROR IN CRACK DEPTH SIZING IS 15% OF W.T.

CONCLUSIONS OF SAFETY ANALYSIS

S.G. TUBE WITH ONE OR MORE AXIAL THRU WALL CRACKS OF 19 mm WITHIN THE TUBE SUPPORT PLATE WIDTH DOES NOT PRESENT RISK OF BREAK.
. PLUGGING CRITERIA

- SG TUBES AFFECTED BY ODSCC AT T.S.P. ARE ALLOWED TO CONTINUE IN OPERATION

- TO AVOID LEAKS DURING OPERATION DUE TO THESE TYPE OF CRACKS, MAXIMUM DEPTH IS LIMITED TO:
  
  MAX. DEPTH = 100 % W.T. - ERROR - CRACK GROWTH RATE

- SIZING ERROR = 15 % W.T. (TO BE CONFIRMED UPON RESULTS OF ADDITIONAL PULLED OUT TUBES)

- CRACK GROWTH RATE IN ONE CYCLE = 7 % W.T.

- MAX. DEPTH = 78 % W.T.
EVALUATION OF MITIGATION AND REPAIR TECHNIQUES

* REMOVABLE PLUGS:

. OBJECTIVE:

TO EVALUATE THE PERFORMANCE OF SG. 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)
* **MAIN ACTIVITIES:**

. SINGLE TUBE AND MULTIPLE TUBE MOCK-UPS FABRICATION
. HELIUM LEAK TESTS
. HYDRAULIC TESTS
. FATIGUE TESTS (40 CYCLES)
. HELIUM LEAK AND HYDRAULIC TESTS AFTER FATIGUE
. I.D. TUBE PROFILOMETRY (AFTER PULLING - OUT OF PLUGS)
* BELGIUM, SWEDISH AND SPANISH SLEEVE QUALIFICATION PROGRAM

. OBJECTIVES:

THE EVALUATION WILL MAINLY ADDRESS THE LIFE EXPECTANCY OF THE UPPER SLEEVE JOINT AND FEATURES SUCH AS 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)
* 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 EXAMINATION ON SLEEVES (≈ 10) EXTRACTED FROM RINGHALS UNIT 2
TEST MATRIX

- VISUAL INSPECTION OF UPPER AND LOWER JOINT (STEREOMICROSCOPE + ENDOSCOPE)
- PROFILE MEASUREMENTS OF UPPER JOINT (OD AND ID) AND LOWER JOINT (ID)
- TESTING OF INTERGRANULAR CORROSION BEHAVIOUR IN UPPER JOINT (MODIFIED HUEY TEST)
- RESIDUAL STRESS DETERMINATION IN UPPER JOINT (X-RAY DIFF)
- AUTOCLAVE TESTING OF SLEEVE CAPSULES, UPPER JOINT (10% Na OH AT 350°C WITH AND WITHOUT Δp)
- NDE (ECT + UT) OF UPPER JOINT AFTER AUTOCLAVE TESTING
- METALLOGRAPHIC EXAMINATION. SAMPLES FROM CAUSTIC TEST AND SAMPLES OF THE LOWER JOINTS
TOP SIDE

Figura: 24 - 5B-04 Sample - Vol. -
Electropolished Etching 10% Ouate Acid
Figure 34 - FA-A3 Sample - X16 - Electrolytic Etching 10% Oxalic Acid

Figure 35 - FA-A3 Sample - X16 - Electrolytic Etching 10% Oxalic Acid
TOP SIDE

Electrolytic Etching 10% Oxalic Acid
Figure 50 - ABB-AC Sample - X30 -
Electrolitic Etching 10% Oxalic Acid
TOP SIDE

Figure: EB - BW-12  (Atol.-H) - C.C.
Electro-etched Etching in Oxalic Acid
Figure: 66 - 316L Sample - Red - Etchant used: Staining for Inclusions (Acid)
Figure:
30 - FA-Al Sample - X16 -
Electrolytic Etching 10% Oxalic Acid

Figure:
41 - FA-H Sample - X16 -
Electrolytic Etching 10% Oxalic Acid
Figure 1: EN - Cyl-42 Sample - X30 - Enamellic Etching 10% Oxalic Acid
Figure: 75 - MHI - 3A Sample - X30 - Electrolytic Etching 10% Oxalic Acid
Figure: PA - W. Plan - X30 - Electrolytic Etching by Oxalic Acid
TOP SIDE

Figure: 66 - W - Sample - X100 - Electrolytic Etching in 10% Oxalic Acid
Figure 1 - ABB-A1 Sample - X30 -
Electrolitic Etching 10% Oxalic Acid
Figure 1: B4 B3 Sample X16 -
Electrolytic Etching 16% Nitric Acid
Abstract

Since the early eighties, various types of degradation have been occurring on steam generator tubes of French pressurised water reactor thus creating safety concerns with respect to the likelihood of a steam generator tube rupture event.

Therefore, for the last 10 years, the French safety authorities have been devoting a large effort to assess the risk of steam generator tube rupture as a function of each kind of degradation, in relation with inservice inspections, preventive and corrective actions scheduled by the operator Electricité de France.

This paper presents the general analysis scheme and the requirements of the French safety authorities to several instances of the principal degradation causes observed into the more than 160 steam generators in operation in France in 1991.

Resume

Les tubes des générateurs de vapeur des réacteurs à eau sous pression ont présenté dès le début des années 1980 divers types de dégradations.

Un important travail d'évaluation a été effectué depuis 10 ans par l'autorité de sûreté française, lié à l'augmentation du risque de rupture de tubes de générateurs de vapeur qui peut résulter de chacune de ces dégradations, pour analyser les actions de maintenance préventive ou corrective proposées par l'exploitant Electricité de France et demander les compléments nécessaires pour éviter une rupture de tube de générateur de vapeur.

Ce papier présente la démarche générale suivie et la position adoptée par l'autorité de sûreté française sur quelques exemples des principales dégradations observées sur le parc de plus de 160 générateurs de vapeur en service en France en 1991.
INTRODUCTION

Depuis le début des années 1980, les tubes de générateurs de vapeur (G.V.) des tranches REP françaises sont atteints d'un nombre de dégradations croissant qui affectent une population toujours plus importante des tubes en service. Ces dégradations augmentent le risque de "rupture" de tube de générateur de vapeur (R.T.G.V.) pendant les situations normales et incidentelles, pour lesquelles le chargement de pression reste grossièrement constant et égal à 100 bars environ, et également pendant des situations accidentelles comme notamment celles correspondant à la perte complète de pression du circuit secondaire (rupture de tuyauterie eau ou vapeur) qui conduisent à soumettre les tubes à la seule pression primaire (172 bars maximum).

Depuis ces 10 dernières années, l'autorité de sûreté française a porté une attention toute particulière à ces dégradations et aux mesures proposées par l'exploitant, Electricité de France, pour y faire face et en particulier pour maintenir le risque de rupture des tubes de générateur de vapeur résultant de ces dégradations à des valeurs acceptables c'est-à-dire compatibles avec les probabilités d'accident prises en compte à la conception.

L'effort d'analyse a porté sur la prévention des ruptures de tubes qui fait l'objet essentiel du présent papier mais également sur la diminution des conséquences d'un tel accident, analyse qui comprend les études des séquences accidentelles, la diminution et la surveillance de l'activité du fluide primaire, les moyens de diagnostic du générateur de vapeur accidenté en cas de R.T.G.V. ainsi que la mise au point des procédures de conduite adaptées et la formation des opérateurs pour faire face à ces situations.

I) La prévention des ruptures de tubes de générateurs de vapeur en présence de dégradations en service : démarche générale

* La prévention du risque de rupture débute bien entendu par la prise en compte du retour d'expérience lors de la conception et de la réalisation des appareils ; ce point a fait l'objet de nombreuses publications pour l'ensemble des appareils aujourd'hui en service en France [1] [2]. Le tableau I résume les dispositions essentielles adoptées sur les différents types d'appareils des tranches de 900 MWe à 3 boucles et de 1300 MWe à 4 boucles au moment de leur mise en service, dispositions issues du retour d'expérience qui provenait essentiellement de l'étranger au moment où les choix de conception et de réalisation de ces appareils ont été effectués.

* Dès lors que les choix constructifs et d'exploitation ont été faits la prévention des ruptures dépend de la surveillance de l'état de appareils qui est mise en œuvre. Cette surveillance s'exerce à deux niveaux :
I.1. La surveillance permanente en fonctionnement qui s'appuie :

- sur les mesures de bruit, effectuées sur le circuit primaire, permettant de détecter des objets migrants essentiellement dans la boîte à eau des générateurs de vapeur côté primaire.

- sur la mesure des fuites, entre les circuits primaire et secondaire. Lors de la mise en service des tranches françaises il avait été admis dans les spécifications techniques que le fonctionnement des tranches ne devait pas être poursuivi lorsque le débit de fuite de 70 l/h était dépassé sur l'un des générateurs de vapeur. Cette limite est directement issue des valeurs adoptées à l'époque par les exploitants américains et correspond à un total de 1 gallon/mn réparti sur les trois générateurs de vapeur d'un réacteur de 900 MWe.

Initialement, cette valeur visait à définir une limite acceptable d'une pollution prolongée du circuit secondaire par le circuit primaire, limite compatible avec les moyens de détection et de mesure installés sur les tranches. Dans cette optique, on peut d'ailleurs rapprocher cette limite de celle des fuites non quantifiées du circuit primaire qui lui est d'ailleurs égale (230 l/h = 1 gallon/mn).

Ce n'est que postérieurement aux démarrages des premières tranches et à la mise en place de ces spécifications techniques, et après constat des premiers défauts sur les tubes de générateur de vapeur que la question de savoir si les tubes défectueux étaient ou non susceptibles de répondre à l'hypothèse "de fuite avant rupture" s'est posée et que l'on a cherché à quantifier les fuites qui peuvent naître des défauts apparaus en service pour les comparer au critère en vigueur sur les tranches.

Pour un certain nombre de dégradations, qui sont apparues depuis 10 ans en France, il n'est pas possible actuellement de démontrer que la valeur limite des premières spéciifications techniques correspond, à coup sûr, à la présence d'un défaut qui serait stable en toute circonstance et notamment en situation accidentelle. Ainsi, au fur et à mesure de l'avancement de ces analyses et de l'observation du retour d'expérience en France mais aussi à l'étranger puisqu'il semble que, dans la plupart des cas, les ruptures de tubes de générateur de vapeur ont été précédées d'une fuite, l'autorité de sûreté française a été progressivement amenée à demander à l'exploitant de limiter à des valeurs plus faibles, simplement compatibles avec les moyens de détection et de localisation des tubes fuyards, le domaine de fonctionnement en puissance des générateurs de vapeur (quelques l/h). Une telle mesure est appliquée actuellement pour des raisons diverses et avec des modalités adaptées aux
différents types de générateurs de vapeur à l'ensemble du parc français.

I.2. La surveillance à l'arrêt

I.2.1. La surveillance de base

Cette surveillance est fondée, depuis la mise en service des premières tranches REP françaises sur la mise en oeuvre de contrôles par courants de Foucault sur toute la longueur du tube au moyen d'une sonde axiale. Dès la mise en évidence de ces dégradations, elle est complétée par des examens spécifiques résultant des analyses objet du paragraphe I.2.2.

Le principe adopté lors de la mise en service des tranches est :

- de vérifier et de mémoriser avec ce moyen l'état de la totalité des tubes avant la mise en service.

- de s'assurer du bon état des tubes au cours de la vie de l'installation par le contrôle d'un échantillon limité de tubes, ayant pour objectif de mettre en évidence l'apparition d'éventuelles dégradations dont le développement serait progressif.

Le sondage actuellement retenu en France est de contrôler 1 tube sur 8 sur l'un des 3 générateurs de vapeur des tranches de 900 MWe ou sur 2 des 4 générateurs de vapeur des tranches de 1300 MWe à chaque arrêt pour rechargement. Des études simples de probabilité, faites tant par Electricité de France que par l'autorité de sûreté, ont montré qu'un sondage de ce type conduit à une probabilité élevée de mettre en évidence une dégradation, dès lors qu'elle affecte quelques dizaines de tubes du générateur de vapeur contrôlé. La figure 1 illustre les résultats obtenus avec ce type d'étude ; elle montre, en fonction du nombre de tubes contrôlés et du nombre de tubes défectueux supposés présents dans l'appareil, la probabilité de mettre en évidence l'un au moins de ces tubes donc de détecter l'apparition de la maladie.

Il est clair également qu'une telle politique de contrôle n'a que peu de chance de conduire à la mise en évidence d'un défaut isolé, qui surviendrait avec une cinétique rapide sur un tube pour des raisons particulières à ce tube ou à son environnement : l'exemple typique de ce genre de défaut est celui d'un défaut ponctuel issu de la fabrication ou du montage. On considère en France que ce type de défaut peut et doit être piégé lors du contrôle initial qui porte sur l'ensemble des tubes ; renouveler ce contrôle à échéance très rapprochée par exemple à chaque arrêt pour rechargement

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n'apparaît pas susceptible d'apporter de nouvelles informations sur ce défaut, sauf amélioration des performances des contrôles en matière d'acquisition des données.

En France, l'accent sera donc généralement mis davantage sur la qualité des interprétations ou l'engagement de nouveaux dépouillements pour interprétation complémentaire des enregistrements existants plutôt que sur la multiplication des contrôles, sauf nécessité démontrée bien entendu.

En outre, il est nécessaire de faire une mise à jour périodique du contrôle des tubes compte tenu notamment de l'évolution et de l'amélioration des méthodes de contrôles ; on a retenu en France une périodicité de 10 ans pour faire cette mise à jour.

Cet objectif est l'un de ceux qui conduisent à retenir les deux dispositions suivantes :

- le sondage 1/8 ne porte pas sur les mêmes tubes lors de chaque visite du générateur de vapeur concerné, mais il est décalé d'un contrôle à l'autre de façon à examiner le maximum de tubes à l'issue de l'ensemble des visites périodiques.

- le programme de contrôle mis en œuvre lors de la visite qui accompagne la réépreuve décennale du circuit primaire principal est complété pour que l'ensemble des tubes ait été recontrôlé au moins une fois depuis la première visite.

I.2.2. Démarche suivie en cas de mise en évidence d'une dégradation

Pour les différents types de dégradations mis en évidence sur les tubes de générateurs de vapeur en France, l'exploitant a cherché systématiquement à faire une évaluation des conséquences potentielles de ces dégradations de façon à proposer les mesures correctives les mieux adaptées pour assurer, le maintien d'un niveau de sûreté acceptable, c'est-à-dire compatible avec le risque de rupture de tube pris en compte à la conception. L'autorité de sûreté s'est attachée à procéder à l'analyse des dossiers ainsi présentés par l'exploitant en suivant la démarche illustrée sur la figure 2, qui appelle les commentaires suivants :

a/ Notion de défauts dangereux

On considère que l'apparition d'une fuite qui conduirait à la mise en service de l'injection de sécurité correspond de fait à la situation de rupture de tube de générateur de vapeur (RTGV) prise en compte à la conception. On appelle
donc défaut dangereux tout défaut susceptible de provoquer une fuite de l'ordre de 30 m³/h au moins.

b/ Compréhension du mécanisme de dégradation

C'est une phase essentielle de l'analyse des problèmes rencontrés, car elle doit permettre d'appréhender :

- quelles sont les zones où la dégradation est susceptible de se développer : liste des tubes concernés et partie des tubes concernés,

- quelle est la cinétique d'apparition de cette dégradation et quelle est sa cinétique d'évolution,

- quelle forme peut prendre la dégradation et quelles sont les morphologies de défauts qu'elle peut générer.

Les moyens mis en œuvre, pour tenter de répondre à toutes ces questions sont basés, tout d'abord, sur les résultats des investigations par les contrôles non destructifs, mais aussi sur les observations sur les tubes extraits (plus de 200 tubes extraits des différents GV français depuis 10 ans) et enfin sur de nombreuses études de laboratoire. La qualité et la fiabilité des données et des informations issues de cette phase de compréhension du phénomène conditionnent l'ensemble des autres tâches car il s'agit de données d'entrées indispensables :

- **morphologie des défauts**, dont dépendront les études de "fuite avant risque de rupture" et les performances à viser pour les méthodes de contrôles non destructifs,

- **cinétique d'évolution**, dont dépendra la politique d'inspection en service, notamment en matière de périodicité des contrôles,

- **mécanismes des dégradations**, qui pourront conditionner le choix des mesures préventives ou correctives autres que le bouchage des tubes.

c/ Etudes dites de "fuite avant risque de rupture"

Cette notion recouvre en France la démonstration que l'évolution naturelle en fonctionnement d'une dégradation affectant un tube doit nécessairement conduire à l'apparition d'une fuite détectable. Cette fuite doit être faible pour rester acceptable eu égard aux rejets qu'elle engendre (par exemple la valeur de 70 l/h des spécifications techniques retenues à l'origine). Le défaut doit présenter dans cet état un caractère de stabilité en
conditions normales et en conditions accidentelles (correspondant notamment à la situation de rupture de tuyauterie secondaire entraînant la dépressurisation du circuit secondaire ainsi qu'un refroidissement brutal) pendant toute la durée nécessaire pour ramener le réacteur à l'état d'arrêt à froid dépressurisé, si le débit est tel que les spécifications imposent un arrêt, ou pendant une durée indéterminée si elles ne l'imposent pas.

Cette position appelle un commentaire important sur la prise en compte du retour d'expérience en ce domaine. S'il est bien entendu tout à fait "agréable" de ne pas avoir observé de rupture de tubes après l'apparition d'une légère fuite décelable en service pendant une durée éventuellement importante, il est faux d'en déduire que les défauts correspondants répondent à l'hypothèse de "fuite avant risque de rupture", car heureusement à ce jour, aucun appareil concerné n'a été placé dans les conditions accidentelles d'une rupture de tuyauterie secondaire!

Seuls des essais d'éclatement après extraction des tubes ou les épreuves du circuit primaire (à la pression de 207 bars à froid, donc de l'ordre de grandeur de la pression subie en conditions accidentelles) permettent ce genre de vérification, à condition toutefois d'avoir pu évaluer la part de la fuite en service à attribuer au tube concerné.

d/ Surveillance et mesures préventives

Chaque type de dégradation mise en évidence sur les générateurs de vapeur nécessite, en fonction de l'analyse de ses causes et du comportement du tube vis-à-vis de cette dégradation, d'adopter des mesures préventives et/ou des contrôles de surveillance spécifique à une périodicité compatible avec la cinétique de développement des défauts. Ces mesures seront prises, soit parce qu'elles sont nécessaires pour maintenir le risque de rupture à un niveau suffisamment faible, et il s'agit alors de mesures prises au titre de la sûreté, soit parce qu'elles sont susceptibles de diminuer le risque d'apparition de fuite en service, et il s'agit dans ce cas de mesures prises en premier lieu au titre de la disponibilité des tranches mais qui contribuent aussi, de façon notable, à leur sûreté.

Par ailleurs, il faut souligner que pour une même dégradation, le choix d'une politique plutôt orientée vers la prévention ou plutôt orientée vers la surveillance périodique revient à l'exploitant seul, en fonction essentiellement de considérations économiques à court, à moyen et à long terme, dès lors que pour une tranche don-
née les dispositions retenues pour chacune de ces politiques satisfont aux objectifs de sûreté, en termes de prévention du risque de rupture pour le cycle de fonctionnement à venir jusqu'à l'arrêt suivant. Dans ce cas la position de l'autorité de sûreté tient compte d'un souci d'anticiper l'évolution des dégradations et des conséquences sur les appareils, de façon à disposer de solutions alternatives en termes de méthodes de contrôles, de réparations, de procédés qualifiés ou d'approvisionnements disponibles pour être prêt à les mettre en œuvre dès que nécessaire.

e/ Exemples d'applications

Nous illustrerons et approfondirons les considérations générales ci-dessus par quelques exemples :

* e.1 Lorsque la capacité de "fuite avant risque de rupture" d'une dégradation est démontrée de façon indubitable, la surveillance lors des arrêts pour rechargement intervient pour évaluer l'évolution de la dégradation et s'assurer que cette évolution est bien conforme aux prévisions du dossier de justification : à ce titre on pourrait dire que la surveillance à l'arrêt intervient comme seconde ligne de défense au titre de la défense en profondeur. Les critères de bouchage éventuellement adoptés pour des défauts non traversants peuvent également répondre au concept de seconde ligne de défense mais répondent surtout au souci de l'exploitant de ne pas mettre en jeu la disponibilité de l'installation en évitant l'apparition d'une fuite en exploitation qui entraînerait la mise à l'arrêt. L'exemple typique d'une dégradation répondant à cette optique est celui des usures de tubes par les barres antivibratoires.

* e.2 Pour des dégradations qui ne concernent qu'un nombre limité de tubes, la constitution d'un dossier complet de compréhension des phénomènes, qui permette d'étayer une démonstration de fuite avant risque de rupture, de spécifier et de qualifier les méthodes de contrôles non destructifs qu'il serait nécessaire de mettre en œuvre lors des arrêts pour quantifier le développement de la dégradation, peut ne pas apparaître économiquement valable par rapport à la mise hors service par bouchage préventif des tubes concernés (disposition adoptée pour les petits cintres avant la mise au point du traitement thermique de détensionnement).
Un certain nombre des dégradations qui apparaissent sur les tubes se situent en dehors des deux cas simples ci-dessus : d'une part, la capacité de fuite avant risque de rupture n'est pas systématiquement démontrée, et d'autre part, il n'existe pas de mesure préventive ou corrective réalisable garantissant une parade définitive au développement des défauts. Dans ce cas, la sûreté du fonctionnement va s'appuyer sur une série diversifiée de mesures complémentaires.

L'exemple typique d'une telle démarche de surveillance basée à la fois sur la surveillance en service et sur la surveillance à l'arrêt est celui du problème de fissuration par corrosion sous tension dans le milieu primaire au niveau des zones de fin de dudgeonnage.

La sûreté des appareils va reposer :

- sur une surveillance de fuite en fonctionnement avec des règles restrictives d'arrêt pour se donner les meilleures chances de détecter ceux des défauts qui pourraient présenter une fuite notable mais éventuellement très faible avant risque de rupture.

- sur une surveillance à l'arrêt adaptée à la caractérisation des défauts qui peut permettre éventuellement d'accepter le maintien en service de tubes dont la dégradation est bien identifiée et l'évolution suffisamment connue pour ne pas risquer qu'elle atteigne des dimensions rédhibitoires avant le prochain contrôle. Cette démarche suppose que la politique de surveillance, telle que l'autorité de sûreté française l'admet, comprenne :
  - un contrôle systématique à chaque arrêt de tous les tubes où des défauts ont déjà été caractérisés et laissés en l'état.
  - la définition d'un critère de bouchage tenant compte de l'évolution possible des défauts dans le cycle suivant.
  - la mise en place d'une surveillance régulière de la population des tubes sains susceptibles d'être atteints, à terme, par cette dégradation. La périodicité de contrôle de chaque tube de cette population doit être définie à partir d'une étude des cinétiques de propagation des dégradations à partir du seuil de détectabilité par les moyens de contrôle non destructifs.
Les incertitudes sur les paramètres dont va dépendre la cinétique de propagation étant souvent importantes, on peut avoir recours pour déterminer la périodicité des contrôles à des études probabilistes qui permettent d'évaluer le risque de rupture de tube de générateur de vapeur associé à la politique de contrôle adoptée pour la dégradation considérée. La communication [3] de la présente réunion de spécialistes est une illustration des travaux réalisés par l'Institut de Protection et de Sûreté Nucléaire pour porter avis sur les propositions faites par l'exploitant.

* e.4 Les différentes politiques de maintenance associées à chaque dégradation permettent de dégager, compte tenu de l'important retour d'expérience de 102 GV 900 MWe et de 60 GV 1300 MWe des prévisions dans le temps de l'évolution des dégradations, du nombre de tubes à boucher, et donc du terme auquel in fine le remplacement des appareils sera nécessaire. Toutefois, l'expérience a souvent montré que les prévisions peuvent être remises en cause par les faits, telle dégradation faisant son apparition alors qu'on espérait avoir pris les précautions nécessaires pour l'éviter, ou telle autre dégradation prenant tout à coup une extension imprévue. Une stratégie de remplacement de générateurs de vapeur basée sur des prévisions même considérées comme raisonnables à un instant donné, pourrait conduire l'exploitant à proposer la poursuite de l'exploitation de générateurs de vapeur ne présentant plus un niveau de sûreté satisfaisant dans l'hypothèse où les nécessités de remplacement s'avéreraient tout à coup supérieure aux prévisions : ce risque qui tient d'abord à la durée nécessaire à la fabrication d'appareils neufs et au délai nécessaire pour préparer l'intervention est d'autant plus accentué en France par l'importance du parc, sa standardisation et les durées de vie peu différentes d'un nombre élevé de tranches. Depuis 1984, l'autorité de sûreté française a donc vivement incité Electricité de France à anticiper les actions nécessaires pour être en mesure d'effectuer des remplacements d'appareils, en développant les études, les procédés, les outillages et les essais nécessaires à l'opération de remplacement et en engageant l'approvisionnement et les fabrications d'appareils neufs. Ainsi le remplacement des appareils sur la tranche de Dampierre 1 a pu intervenir assez tôt pour vérifier la capacité d'Electricité de France et du constructeur à mettre en œuvre une telle opération, alors que les moyens de surveillance mis en place permettaient encore
d'assurer un niveau de sûreté acceptable des appareils.

II. Moyens de surveillance

II.1. Surveillance en service

Il s'agit des moyens de détection et de mesure des fuites entre les circuits primaire et secondaire qui sont basés sur :

- la mesure d'activité des gaz extraits au niveau du condenseur.

- la mesure d'activité de l'eau du circuit de purge de chaque générateur de vapeur.

- la mesure d'activité de l'azote 16 sur la tuyauterie vapeur de chaque générateur de vapeur.

Des mesures de laboratoire par diverses méthodes peuvent compléter en tant que de besoin, ces instrumentations permanentes dont les indications sont reportées en salle de commande, où elles sont accompagnées d'un certain nombre d'alarmes. Tous ces moyens sont aujourd'hui opérationnels sur l'ensemble des tranches de 900 et 1300 MWe. En particulier l'autorité de sûreté française a insisté depuis de nombreuses années pour la mise en place des mesures d'activité de l'azote 16 sur l'ensemble des tranches ; ainsi l'effort d'anticipation qui a été demandé à l'exploitant dans ce domaine a permis de faire face de ce point de vue en 1989 au problème des déformations et des fissurations des tubes des générateurs de vapeur 1300 MWe (voir chapitre III.5).

II.2. Surveillance à l'arrêt

Les méthodes les plus utilisées sur les générateurs de vapeur en France sont :

- pour les tests d'étanchéité, le test à l'hélium qui permet de mettre en évidence les tubes présentant même une très faible fuite (de l'ordre du cm$^3$/h), dans le double but d'identifier les tubes dont la fuite nécessite une caractérisation pour déterminer le ou les défauts qui en sont à l'origine et de sélectionner, parmi les tubes dont les défauts présentent des caractéristiques qui permettraient d'en accepter temporairement le maintien en l'état, ceux qui présentent une fuite susceptible de conduire au dépassement des critères d'arrêt de la tranche au cours du cycle suivant.
pour les contrôles par courants de Foucault, la sonde axiale pour contrôler l'ensemble du tube, complétée par la sonde tournante pour l'examen détaillé des zones de fin de dudgeonnage.

Ces méthodes et leurs performances ont déjà fait l'objet de nombreuses publications de la part d'Electricité de France ou de ses prestataires telle par exemple [4] et nous ne les reprendrons pas ici.

L'usage de ces moyens s'est considérablement intensifié au cours de ces dernières années ainsi que le montrent les figures 3 et 4.

L'augmentation du volume des contrôles tient dans un premier temps à la mise en service progressive des différentes tranches du parc électronucléaire, puis essentiellement pour les dernières années, à la mise en place de la politique de contrôle des tubes en fonction de chaque dégradation ainsi qu'on en a décrit le principe au chapitre I et dont quelques exemples détaillés apparaissent au chapitre III.

Outre cette réponse quantitative au développement des dégradations sur le parc, Electricité de France est également sollicité par l'autorité de sûreté française pour apporter des améliorations dans les domaines suivants :


- développement de méthodes complémentaires d'expertise non destructives. Nous citons par exemple :

  . les ultrasons dans les zones de fin de dudgeonnage, qui peuvent apporter des compléments précieux au contrôle par courants de Foucault pour certaines configurations de défauts.

  . les courants de Foucault pour caractériser les défauts dans les autres parties du tube que la zone de fin de dudgeonnage (en particulier zone des plaques entretoises - zone de cintrage).
Dans ces domaines l'autorité de sûreté française est particulièremment soucieuse que l'exploitant anticipe les actions de développement, pour pouvoir traiter les problèmes sur les tranches de la façon la plus efficace, lorsqu'ils se font jour.

III. Programme de surveillance et de prévention adopté pour différentes dégradations

Ce chapitre est destiné à illustrer l'application des principes généraux exposés au chapitre I à différentes dégradations observées sur les tranches françaises.

On exposera successivement, dans chaque paragraphe :

- l'analyse de l'applicabilité de l'hypothèse de "fuite avant risque de rupture",
- la capacité des méthodes de surveillance à détecter et à caractériser les dégradations,
- l'étendue et la périodicité de la surveillance à l'arrêt,
- les critères de mise hors service des tubes.

III.1. Usure des tubes par des corps étrangers

Il est clair que la démonstration de "fuite avant risque de rupture" pour la dégradation d'usure par corps étrangers n'est pas possible. Par exemple, une usure uniforme sur une longueur de plusieurs centimètres du tube ne répond pas à l'hypothèse de "fuite avant risque de rupture". L'expérience étrangère (RTGV de GINNA le 25 janvier 1982 et de Prairie Island 1 le 2 octobre 1979) a d'ailleurs démontré la réalité de ce problème.

La sonde axiale est généralement bien adaptée à détecter les usures localisées importantes sur les tubes, mais la mesure de la profondeur du défaut par la phase du signal est à considérer avec prudence car la forme des défauts peut également modifier la phase du signal.

La surveillance consiste, systématiquement, en une inspection télévisuelle détaillée de la périphérie du faisceau tubulaire côté secondaire. La mise en évidence d'un objet susceptible de provoquer une usure des tubes entraîne l'examen à la sonde axiale de l'ensemble des tubes de la périphérie du faisceau. Cet examen est systématique si le générateur de vapeur fait l'objet d'un contrôle programmé à la sonde axiale. Les tubes présentant une usure supérieure à 40% sont obturés. L'extraction de l'objet est bien entendue systématiquement entreprise. Au cas où elle échoue, en général parce que le corps migrant est coincé entre les tubes,
tous les tubes qui pourraient se trouver au contact de l'objet, sont obturés, que ces tubes présentent ou non un signal d'usure.

La règle retenue conduit notamment à réexaminer, à chaque arrêt, les tubes voisins de tubes qui ont été antérieurement obturés en raison de la présence d'un corps étranger qui n'a pas pu être extrait et qui est donc susceptible de poursuivre la dégradation.

La figure 5 illustre l'impact de cette politique sur le nombre de tubes obturés dans les centrales REP françaises.

III.2. Usure au contact des barres antivibratoires

Le frottement des tubes sur les barres antivibratoires conduit à une usure des tubes suivant une géométrie bien définie dont la longueur est limitée à l'épaisseur de la barre antivibratoire et d'extension angulaire faible. Cette morphologie a permis de démontrer que, quelle que soit la profondeur de l'usure, les ouvertures qui se produisent au droit des usures restent toujours stables en conditions accidentelles. On considère donc que la démonstration de "fuite avant risque de rupture" est correctement établie pour cette dégradation.

La sonde axiale est bien adaptée à la détection de ces usures et l'évaluation de leur profondeur a été établie à partir de l'amplitude du signal par des essais reproduisant les usures par barres antivibratoires : l'incertitude de mesure de la profondeur affectée est donc réduite dans ce cas par rapport au cas des corps migrants.

La surveillance porte sur la zone où le retour d'expérience a mis en évidence cette dégradation (au-delà de la ligne 28 et entre les colonnes 30 et 65 des CV 900 MWe) à partir du 6ème arrêt pour rechargement. Le contrôle porte à chaque arrêt sur les tubes qui ont présenté un signal d'usure laissé en l'état à l'arrêt précédent et sur la moitié des tubes de la zone considérée : les tubes ne présentant pas de signal d'usure sont donc systématiquement revus tous les deux ans.

Le critère d'obturation retenu correspond à une usure supérieure à 40% qui tient compte :

- des profondeurs pouvant conduire à l'ouverture des défauts en situations normale (85%) et accidentelle (75%).

- de la cinétique d'usure observée par le retour d'expérience, dont la valeur maximum a été relevée à ce jour à 11% en un cycle (voir figure 6).

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de la périodicité de contrôle.

L'évolution du nombre de tubes obturés pour cette raison sur le parc REP français est illustrée sur la figure 7 : on y voit à la fois l'évolution de la dégradation et la mise en place progressive du programme de surveillance.

III.3. Fissuration dans les petits cintres

Cette dégradation, attribuée au phénomène de fissuration par corrosion sous contrainte, touche à ce jour les cintres des lignes 1 et 2 des générateurs de vapeur 900 MWe qui n'ont subi ni traitement thermique complémentaire ni traitement de détensionnement après cintrage en usine.

Le dossier d'étude de "fuite avant risque de rupture" est limité pour cette dégradation à l'évaluation des longueurs critiques de défauts traversants théoriques. Aucune expertise de défaut n'a été réalisée en France dans ces zones, c'est pourquoi la morphologie réelle des défauts n'est pas connue à ce jour (dimensions, orientation, profondeur, etc...). Cette morphologie semble d'ailleurs devoir être différente selon les fournisseurs de tubes sur lesquels de tels défauts ont été identifiés (Westinghouse, Vallourec, Sandvik) : l'utilisation d'un procédé de cintrage des tubes différent selon les fournisseurs est à l'origine d'un état de contraintes résiduelles différent et donc de risque de fissuration divers. La méconnaissance de la morphologie réelle des défauts constitue pour les autorités de sûreté françaises une lacune essentielle du dossier de "fuite avant risque de rupture".

En matière de contrôles non destructifs, on ne dispose de façon industrielle que de moyens de détection des défauts qui sont le test d'étanchéité à l'hélium et la sonde axiale à courants de Foucault et pas de moyen de caractérisation.

Chacun de ces moyens à une capacité de détection limitée : le test à l'hélium n'est bien sûr sensible qu'à des défauts traversants et le contrôle par sonde axiale peut être fortement perturbé par le problème d'excentrement de la sonde dans le tube au passage du cintre. Le retour d'expérience dans ce domaine laisse penser que le développement des fissures dans ces zones pourrait être rapide car il a été observé des fuites en cours de cycle sur des tubes qui ne présentaient pas de défauts au contrôle par courants de Foucault lors de l'arrêt précédent.

Dans ces conditions, constatant que la démonstration de "fuite avant risque de rupture" n'était pas apportée et que la surveillance en service était insuffisante pour garantir une détection suffisamment précoce des défauts, l'autorité
de sûreté française a considéré nécessaire la mise en oeuvre de mesures préventives ou correctives.

Deux types de mesures alternatives ont été adoptées :

- soit le bouchage préventif de l'ensemble des petits cintres,

- soit la mise en oeuvre d'un traitement thermique de relaxation des contraintes dans les parties cintrées (traitement thermique à une température supérieure à 700°C pendant 1 à 2 minutes) qui doit diminuer les contraintes résiduelles à un niveau suffisamment faible pour ne plus craindre de développement de fissures.

A la suite de ce traitement les cintres sont contrôlés par sonde axiale à chaque arrêt pour vérifier l'efficacité de la solution. À ce jour aucun nouveau défaut n'a été mis en évidence sur les tubes ainsi traités.

L'engagement de ces mesures a été appliqué aux différents types de cintres au fur et à mesure de l'apparition des premiers défauts :

- 1ère ligne des faisceaux Westinghouse (1985 : obturation),

- 1ère et 2ème ligne des faisceaux Vallourec (1988 : traitement thermique),

- 2ème ligne des faisceaux Westinghouse (1989 : traitement thermique),


III.4 Fissuration par corrosion sous tension en milieu primaire dans les zones de fin de dudgeonnage des GV 900 MW

- "Fuite avant risque de rupture"

Les études théoriques et les essais de laboratoire montrent que la présence d'une fissure traversante longitudinale ou transversale doit conduire à un débit de fuite notable sans présenter de risque d'instabilité sous chargement accidentel : typiquement, une fissure longitudinale de 16 mm, correspondant à la longueur critique sous 172 bars pour les conditions les plus pénalisantes (épaisseur et caractéristiques du matériau minimales), devrait conduire à une fuite en fonctionnement normal d'au moins 70 l/h.

Pratiquement, les observations effectivement faites sur les générateurs de vapeur ne confirment pas systématiquement ce
comportement. Plusieurs observations permettent d'illustrer ces écarts :

- les fissures simples et traversantes dans la zone de transition de dudgeonnage des tubes ne présentent pas toujours en service des débits de fuites conformes aux prévisions des modèles et aux résultats des essais de laboratoire, ceci a été constaté sur des tubes extraits ; plusieurs voies peuvent être approfondies pour tenter d'expliquer ces faits. Par exemple, la présente d'acide borique et de lithium dans l'eau du circuit primaire peut être à l'origine de la formation de composés solides dans les fissures qui en obstruent partiellement la section. On peut également supposer que la présence des dépôts solides puisse contrarié la fuite.

- la géométrie des fissures relevées dans les zones de fin de dudgeonnage de certains tubes est manifestement moins favorable à un comportement conforme à l'hypothèse de "fuite avant risque de rupture" que les fissures longitudinales ou circonférentielles traversantes prises en compte dans les premières évaluations.

Les exemples suivants, dont on pourra trouver plus de détail dans les publications [5], [6], [7] illustrent les difficultés posées par certaines configurations de défauts :

- certaines expertises ont montré la possibilité de développement de fissures alignées sur deux génératrices proches l'une de l'autre et séparées par un ligament de tube non fissuré. Cette morphologie de fissures conduit à un débit de fuite faible, car chacune des fissures, individuellement courte, conduit à un débit de fuite théorique faible, alors qu'une montée de pression accidentelle peut entraîner la rupture du ligament entre les deux fissures s'il est de taille réduite (de l'ordre du millimètre) et donc amener à la formation d'une fissure unique de grande longueur pouvant présenter un risque d'instabilité.

- les réseaux de petites fissures identifiées côté primaire des tubes des anciens générateurs de vapeur de la tranche de Dampierre 1 ont pour conséquence de réduire fortement la taille critique des défauts sans entraîner de variation notable des débits de fuite en conditions normales.

- l'expertise de certains défauts circonférentiels identifiés dans les zones de fin de dudgeonnage des tubes a montré qu'il était possible de développer des fissures circonférentielles affectant une proportion très importante de la section totale du tube avec présence de ligaments de métal non fissuré ayant pour conséquence de limiter fortement la section de passage du débit de fuite.
- Capacité des méthodes de contrôle par courants de Foucault

Les fissures se développant dans les zones de transition de dudgeonnage conduisent à une déformation du signal de la variation géométrique de la fin de dudgeonnage obtenu par la sonde axiale.

Ainsi la sonde axiale permet d'identifier les tubes qui présentent une anomalie des signaux de variations géométriques du dudgeonnage, sans permettre de caractériser la forme et le nombre des fissures présentes dans cette zone.

Cette caractérisation est assurée par la sonde tournante et son logiciel d'aide au dépouillement ESTELLE (voir [4]). Cette sonde permet de caractériser les fissures longitudinales et circonférentielles dès lors qu'elles ont une profondeur (environ 40%) et une longueur (3 à 7 mm) supérieures aux seuils de détection. La précision de mesure des fissures longitudinales est évaluée à ±2 mm. Des difficultés de caractérisation peuvent être rencontrées dans le cas de défauts complexes tels que ceux constitués par exemple de fissures circonférentielles et de fissures longitudinales nombreuses dont l'éloignement dépend de l'ordre de grandeur de la zone d'influence de la sonde (4 à 5 mm).

- Stratégie de surveillance

L'impossibilité d'apporter la démonstration systématique de l'applicabilité de l'hypothèse de "fuite avant risque de rupture" ne permet pas de faire reposer la démonstration de sûreté sur la seule surveillance en service des fuites. À l'inverse, la probabilité de détecter par les examens courants de Foucault toutes les fissures risquant de conduire à une instabilité n'est pas non plus suffisamment élevée pour faire reposer la sûreté sur les seuls examens non destructifs lors des arrêts. Plusieurs raisons peuvent en effet faire échouer le diagnostic fait à partir des examens par courants Foucault :

- présence de défauts complexes hors de portée des possibilités de caractérisation de la sonde,
- cinétique d'apparition et d'évolution d'un défaut sensiblement plus rapide que celle évaluée jusqu'à présent par l'observation du retour d'expérience.

Typiquement l'ensemble des générateurs de vapeur 900 MWe à tubes non traités thermiquement d'origine Westinghouse et Vallourec et avec DAM est susceptible de présenter ces
caractéristiques : ceci représente 59 générateurs de vapeur de 900 MWe.

La position de l'autorité de sûreté est de faire reposer la sûreté des tranches affectées de tels défauts sur les deux lignes de défense complémentaires que sont :

- une limitation à la valeur la plus basse possible des évolutions des fuites primaire-secondaire en fonctionnement : typiquement l'apparition d'une fuite évoluant de plus de 5 l/h en régime de puissance doit conduire à l'arrêt du réacteur.

- une surveillance à l'arrêt portant de façon exhaustive sur les tubes susceptibles d'être touchés par ces défauts complexes à une périodicité compatible avec la cinétique d'évolution observée par le retour d'expérience.

A ce jour la périodicité de contrôle par la sonde tournante des zones de fin de dudgeonnage en branche chaude est :

- à chaque arrêt pour rechargement pour tous les tubes sur lesquels des fissures ont été caractérisées et laissées en l'état,

- tous les deux cycles de fonctionnement pour tous les tubes susceptibles d'être atteints par cette dégradation. Pratiquement la surveillance de base détermine si le générateur de vapeur est affecté par ce problème et, dans ce cas, c'est la totalité des tubes de l'appareil qui fait l'objet de cette mesure.

- Critères de bouchages

Les critères de bouchages sont fixés compte tenu des performances de la méthode de caractérisation par la sonde tournante longue et de la connaissance, par le retour d'expérience et l'analyse de l'évolution du phénomène, de la cinétique de propagation des défauts pour en accepter éventuellement le maintien en service provisoire jusqu'à l'arrêt suivant. Les travaux réalisés par Electricité de France dans ce domaine et l'analyse qu'en a faite l'autorité de sûreté conduisent aujourd'hui :

- à boucher tout tube présentant une configuration de fissuration circonférentielle quelle que soit sa longueur,

- à obturer tout tube présentant une fissure longitudinale dont l'extension au-delà du dernier point de contact avec la plaque tubulaire dépasse 13 mm. Cette valeur tient compte de la taille critique minimale déterminée dans cette zone (16,5 mm) diminuée d'une valeur
de 3,5 mm comprenant l'incertitude sur la dimension issue du contrôle et une provision pour l'extension de la longueur fissurée pendant un cycle de fonctionnement supplémentaire.

III.5 Déformation et fissuration des tubes de générateurs de vapeur de 1300 MWe

Cette dégradation a été mise en évidence en 1989, lors de la première visite en service d'un réacteur de 1300 MW sur lequel une fuite était apparue en service au cours du premier cycle de fonctionnement (Nogent 1). La référence [7] donne de nombreuses informations sur cet incident.

Le mécanisme proposé pour l'explication des fissures observées est lié à la présence, lors de la mise en service, de certains résidus métalliques à base de fer (résidus de meulage, grenailles de traitement de surface) non extraits du générateur de vapeur ou des circuits secondaires lors de la fabrication et du montage.

Ces résidus étant rassemblés au centre de la plaque tubulaire, l'oxydation des particules ferreuses lors des mentées en température, provoque ensuite leur agglomération et leur gonflement par formation de magnétite Fe₃O₄ ; ce gonflement est responsable des déformations des tubes au voisinage immédiat de la zone de fin de dudgeonnage en sortie de plaque tubulaire. Ces déformations peuvent ensuite conduire à une fissuration par corrosion sous tension en milieu primaire.

Les analyses effectuées sur les tubes extraits ont montré que ces fissures étaient circonférentielles amorcées en peau interne [8].

Ces fissures peuvent présenter une extension angulaire importante sans traverser entièrement la paroi ; ainsi il a été identifié des tubes présentant des indications s'étendant sur près de 90 % de la circonférence au contrôle par courants de Foucault sans que le générateur de vapeur correspondant ne présente de fuite décelable en fonctionnement.

L'analyse de cette dégradation a porté sur plusieurs axes complémentaires.

- constats aussi précis que possible de l'état des appareils et de la nature des dégradations qui comprend :

- côté secondaire :
  - des mesures de hauteur de dépôts par un appareillage à ultra sons,
. l'établissement de cartographies précises des zones de dépôts par des vidéoscaméras et des endoscopes introduits entre les tubes,
. des prélèvements de dépôts aux fins d'analyses,

o côté primaire :
  . des contrôles systématiques de détection de fissures avec la sonde tournante sur les tubes entourés de dépôts côté secondaire.
  . un diagnostic des déformations présentes sur les tubes. Divers moyens ont été utilisés pour évaluer ces déformations. Un traitement spécifique du signal obtenu par la sonde tournante a finalement été retenu. Des empreintes de tubes par moulage ont été prises sur un échantillon de tubes pour quantifier les déformations réelles et fournir des éléments de qualification de la méthode d'examen par courants de Foucault.

- études et essais de compréhension des phénomènes. Un important programme a été lancé par l'exploitant comprenant :
  . des essais de reproduction du phénomène de gonflement des dépôts et de déformation des tubes,
  . des évaluations des contraintes correspondantes et des risques de fissuration associées.

- des actions de remise en propreté des appareils avec le développement et l'amélioration de la technique de langage par eau sous pression et la mise en œuvre d'un procédé de nettoyage chimique [9].

- le bouchage des tubes présentant une déformation notable (comprenant bien évidemment ceux présentant une fissuration).

- la réduction de la valeur des fuites primaire-secondaire entraînant la mise à l'arrêt du réacteur.

L'action de l'autorité de sûreté s'est exercée de façon continue sur cette affaire et leur position a tenu compte de l'état des connaissances acquises à chaque instant.

Ainsi, après les premiers constats faits courant 1989, l'autorité de sûreté a insisté sur la mise en place d'un programme cohérent de compréhension des phénomènes et de qualification des méthodes d'examen et de diagnostic utilisées.

Ceci a été associé à une évaluation exhaustive de l'état de tous les générateurs de vapeur 1300 MW qui s'est terminé en
juillet 1990 et qui a pu nécessiter l'arrêt exceptionnel de certaines tranches.

Il a été décidé de faire des investigations complémentaires au cours d'un arrêt programmé à cet effet, à la moitié du cycle des tranches les plus affectées par ces phénomènes, pour mieux évaluer la cinétique de la dégradation, vérifier l'étendue de la zone de dépôts, et bien entendu procéder sur ces tranches aux bouchages nécessaires et à des compléments de nettoyage.

En 1990, six tranches de 1300 MWe ont subi un tel examen intermédiaire. Sur ces tranches, il n'a pas été noté d'évolution importante des déformations ni d'apparition de fissures circonférentielles en grand nombre, apportant ainsi une confirmation des premiers résultats d'essais de laboratoire qui tendaient à montrer une évolution rapide, mais limitée dans le temps, aux premiers mois de fonctionnement des tranches.

Ainsi, début 1991, l'autorités de sûreté a admis que le fonctionnement des tranches en service de 1300 MWe pourrait à nouveau être engagé pour des cycles normaux moyennant le respect du critère d'arrêt sur fuite primaire-secondaire fixé à la plus basse valeur compatible avec les moyens de mesure et de détection des tubes fissurés soit un débit de 3 l/h.

Les actions d'études, d'essais et de qualification ainsi que l'amélioration des procédés de nettoyage dans l'objectif de restituer à terme un état de propreté correct autour de l'ensemble des tubes laissés en service se poursuivront dans les mois et les années à venir.

CONCLUSION

L'apparition progressive de plusieurs types de dégradations qui touchent un nombre croissant des quelques 700000 tubes de générateurs de vapeur aujourd'hui en service en France, exige de l'exploitant, Electricité de France, un effort important :

- pour comprendre les mécanismes de chaque dégradation et estimer son évolution a priori,
- pour développer et qualifier les méthodes de contrôle non destructif et les outillages d'intervention,
- pour établir l'état du parc pour chaque dégradation et tirer tous les enseignements de ce retour d'expérience,
- pour mettre en œuvre les mesures préventives et correctives et les examens nécessaires pour maintenir le niveau de sûreté des appareils à un niveau acceptable.

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L'autorité de sûreté française est particulièrement attentive aux dossiers que l'exploitant lui présente sur chacun de ces sujets et a été amenée à exiger de l'exploitant la mise en œuvre d'une stratégie de contrôle et de surveillance qui est industriellement lourde mais en rapport avec la taille du parc des générateurs de vapeur REP, 102 générateurs de vapeur de 900 MW d'âge moyen 8,5 ans et 60 générateurs de vapeur de 1300 MW d'âge moyen 3,5 ans et avec l'extension des dégradations qui les affectent. Sans que cela constitue une quelconque garantie pour l'avenir, on ne peut exclure que cette politique ait contribué au fait que l'on n'ait pas eu à déplorer d'accident de RTGV en France au jour où ces lignes sont écrites. Ces actions de contrôle et de surveillance devront, à court et à moyen termes, être complétées grâce au remplacement progressif et préventif des générateurs de vapeur, remplacement dont la faisabilité a été vérifiée à Dampierre 1 en 1991 dans des conditions tout à fait démonstratives.
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probabilité de détection d'au moins un tube défectueux parmi "n"

$P$ vs. proportion de tubes contrôlés

- $n=20$
- $n=14$
- $n=8$
- $n=4$
- $n=2$
FIGURE 3
2.4-28/32
FIGURE 5

TUBES OBTURES POUR USURE PAR OBJET MIGRANT

avec usure
à titre préventif

ANNEE

NOMBRE
0 20 40 60 80 100 120 140
PALIER 900 : Evolution des usures sous BAV durant le dernier cycle (577 usures réparties dans 25 GV)

Evolution de la profondeur en % de l'épaisseur
OPERATING EXPERIENCE WITH HORIZONTAL TYPE STEAM GENERATORS
OF THE VVER-440 PWR UNITS

Sandor Nagy
PAKS NUCLEAR POWER PLANT
PAKS, HUNGARY

ABSTRACT

The Paks Nuclear Power Plant consists of four VVER-440 PWR units, which have been placed into service in 1983, 1984, 1986 and 1987 respectively. After 26 reactor years operation the units have been running remarkably well. One of the most important components of the units are the steam generators which lifetime histories show no major events. Comparing to the other unit's steam generators tube integrity, the Paks' steam generators are running at the top. Among the 132,864 working tubes only 19 tubes have been plugged since their operation, which corresponds to 0.0143%. Up to the recent time the steam generator heat transmission degradation, which is induced by the secondary side magnetit sludge has been considered the most serious technical problem.
Steam Generators:

The steam generators in VVER-440's are horizontal units with submerged tube bundles and built-in separators. The steam generators produce dry saturated steam, with the capability for a slight (10 to 15 degrees C above the saturation temperature) initial superheating. The advantages of horizontal steam generators are the large evaporation surface. The large volume of water makes a cool-down of the reactor in an emergency shut-down much easier. The primary circuit has a natural circulation capability that can be used to cool the reactor during refueling.

The VVER-440 units have six horizontal steam generators with a diameter of 3,340 mm and a length of 12,000 mm and a mass of about 145 tonnes. 26 reactor years of operation experience shows that there are no major problems with the steam generators.

The main problems which have been occurred are:

- integrity problems
- heat transmission degradations

Integrity problems:

1. The collector flange leakage problems:

The horizontal steam generators primary side and secondary side are separated from each other by covers which are sealed with double Ni rings. For monitoring the flanges pressure gauges are installed. In case of leakage of the sealing rings there are regulations in the Technical Specifications.

Until now two leakages have occurred, in 1988 in the SG #4 on unit 1. At the end of the cycle during nominal operation, p 20 bars, high pressure signal of the slipping monitoring chamber of the cold leg collector of the SG #4 appeared in the control room. The pressure increased up to 123 bars, however it took 72 hrs. Water sampling was taken and the analyses revealed that the leakage had been occurred long before the signal. The slipping rate had been determined as low as 20 ml/hrs. Considering that fact and the commencing cycle end the authority gave permission to continue the operation for 28 days with additional technical measures. During the outage the sealing rings and the flange surfaces were inspected, and slight erosion signs were detected. During the unit restart the pressure in the monitoring chamber increased again. Due to this fact the unit was shut-downed and the flanges were resealed. Since the flanges and the sealing rings showed no deviations from the normal conditions, the sensor tube inside the collector was inspected and an improvised hydrotest was performed which later on turned out not to be proper enough to reveal the leakage. On the following cycle the steam generator operated

2.5-2/27
with leakage by the rate of 8 - 60 ml/hrs. On the 1989 outage the sensor line was covered by a shielding tube and according to this measure the sipping has ceased.

The second event took place this year on the unit 4. After the refueling outage during the unit heat-up to the required temperature for the criticality test, the high pressure signal of the sipping monitoring chamber of the hot leg collector of the SG #4 appeared in the control room. The pressure increased so rapidly - about 50 bars/hours - the unit shut-down was decided. The inspection found that the inner sealing ring was removed its position and lost its ductility and capability to seal the flanges. This flange was checked two years before that and during this outage it was not dismantled. It was supposed that the steam generator primary and secondary side hydro and tightness tests initiated the ring movements which was possible because of maintenance error. Taking corrective actions the practice, frequency and the level of the tightness tests have been reviewed to avoid any negative pressure differences.

2. Heat transfer tube leakages:

The Paks NPP unit’s steam generator tubes integrity are excellent. Among the 132,864 working tubes only 18 tubes have been plugged since their operation, which corresponds 0,0143 %. Until 1989 when the eddy-current test has been introduced there were no plugged tubes in the steam generators. Since then the EC test have been applying and the plugged tubes and their distribution can be seen in the table.

<table>
<thead>
<tr>
<th>Units</th>
<th>1989</th>
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<tr>
<td>Unit 1</td>
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<td>Unit 3</td>
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<td>Unit 4</td>
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Despite of the excellent condition of the steam generators there was an incident in 1990 on unit 2, when the unit was shut-down because of tube leakage. According to the Technical Specifications the steam generator and the unit can remain in operation unless either the sipping reaches the 5000 ml/hrs or the activity in the steam generator secondary side increases to 4000 Bq/l or the Tr concentration increases to 1000 Bq/l in the main condensate system. In this case the leakage in the steam generator #5 slowly increased from Nov. 1989 its first detection to 1000 ml/hrs until the beginning of March 1990. The Na-24 activity level was around 1200 Bq/l while K-42 was around 1000 Bq/l which were well below their operational limits. On 10 March the Tr activity exceeded the limit it was 1284/1295 Bq/l respectively and at that time the leakage rate was only 1494 ml/hrs. Due to the Technical Specification requirements
Technical Specification requirements the unit was shut-down and the damaged tube was plugged. The position of the slipping tube was detected by means of simple method and the surrounding tubes were inspected by EC test. No other leakage was discovered.

Heat transmission degradation and steam generator cleaning:

A heat transmission problem of the steam generator occurred in 1987 at the very first time. During the Unit 4 start-up tests the reactor inlet temperature was higher than expected and higher than the other unit's. At the level of 50% power it seemed that the reactor inlet temperature would reach the allowed 267 degrees C limit in case of load up to the nominal level. Later we learned that the inlet temperature was high but it decreased very slowly and according to that the limit was not broken.

The next event which drew attention was the unit 1 start following a reactor scram in February 1988 when the unit could be loaded up to the nominal power only with reduced secondary side pressure. Although there was some improvement to the end of that cycle the unit start-up following the annual outage took place within worse condition and almost the whole cycle can be characterised by reduced secondary side pressure. The visual inspection during the refueling time showed the continuously increasing amount of sludge on the heat transfer tubes was responsible for the heat transmission deterioration. The sludge was 99% almost pure magnetit, hard, flake like deposit baked to the tubes.

The suggested limit for the sludge is 150 g/m² and because of it was exceeded, the steam generator cleaning had been decided in 1988. The first solution was mechanical cleaning by means of water jet lanceing, which method was quite successful in removal of a pile of hard sludge from some vertical steam generators. The differences in the construction of the SG's and the lay-out of the sludge had lead to fall in removing last year. Since the sludge amount on Unit 1 was three times higher than the limit, the cleaning was very urgent. The Siemens KWU had been asked to fulfill this task last winter. Siemens KWU has had long experience in vertical SG's chemical cleaning and has been disposed of licensed procedures. Siemens KWU was carrying out the compatibility test, the possibly form of the standard procedure was consulted. Considering the VVER-440 unit properties and the requires of the chemical cleaning the following conditions were agreed:

- the execution would be in two steps each of them consists of 3 SG's
- the maximum temperature is 160 degrees C
- the emergency feedwater line would be used for reagent injection preheated up 80 degrees C
- after the chemical injection the unit should cool down for drainage of the SG's then heat up again to the required temperature level.
The above mentioned conditions and the compatibility tests resulted the EDTA application instead of the preagreed NTA, and since the huge amount of sludge, 150 g/l of the applied EDTA concentration. For the application of the procedure the Hungarian licensing authority required three additional examination and justification such as:

- influence of the relatively cold chemicals for the SG emergency feedwater nozzle
- SG safety relief valves operation and their contributions in the allowed cycles number
- the cooldown induced stresses for the closed SG's vessels

For the most appropriate procedures a simulation test was executed, which included the cooldown of the unit, injection and drainage of the steam generators. According to the received experiences and the computations the procedure was developed and approved by the authority. The performance of the chemical cleaning took place this year during the unit 1 shutdown for the outage. The execution took 60 hours and ran with no disturbances - according to the expectations. The removed sludge was as follows:

Steam generator # 1 ................................................................. 1427 kg
Steam generator # 2 ................................................................. 1450 kg
Steam generator # 3 ................................................................. 1474 kg
Steam generator # 4 ................................................................. 1447 kg
Steam generator # 5 ................................................................. 1480 kg
Steam generator # 6 ................................................................. 1443 kg

Total ..................................................................................... 8721 kg

The follow-up visual and endoscopic inspection showed that the steam generators were free of sludge the tubes were shiny, only couple of ten kilograms copper was remaining because the contract did not cover the copper removal.

The final verification of the steam generator chemical cleaning efficiency could have been - the heat transmission of the SG's. According to the preliminary estimation the KF factor should have been improved up to at least 13 MW/degrees C value. However on the first day of operation after the restart of the unit the KF factor was as low as 10 MW/degrees C less than its minimal value in 1988, and the nominal power level could not be reached even reducing the secondary side pressure till couple of days.
Two phenomena have been observed since the first occurring, the continuous deterioration until 1988; when the KF factor was 10.3 MW/degrees C after that it has been stabilised around 11 MW/degrees C level. The other one - the KF factor radical drop after the outages then slowly improvement. In our case the same phenomenon has been observed but the decrease has been 3 MW/degrees C instead of the usual 1 MW/degrees C. The heat transmission degradation problem has not been clarified yet.

The following possible causes were scrutinised:

☐ Dissolved gases concentration of the primary coolant
☐ Undissolved gases concentration of the heat sink
☐ Secondary side impurities (colloids)
☐ HPP's operation influence
☐ Measurement errors
☐ Built-in separator hydraulic resistance
☐ Different kind of boiling

Although there is no reason to presume any correlation between the chemical cleaning and the heat transmission deterioration Siemens KWU has been asked to investigate the problem. All the necessary data have been given and the studies started but not finished yet.
Type: GVG 213
Thermal power: 229 MW
Steam generation: 452 t/h
Steam pressure: 46 bar
Primary side pressure: 123 bar
Primary side pr. drop: 0.8 bar
Secondary side pr. drop: 0.12 bar
Moisture content: 0.025 %
Heat transport tube's size: 16 x 14 mm

Diameter of pipelines
- Primary loops: 500 mm
- Secondary inlet: 250 mm
- Secondary outlet: 400 mm

Number of coolant tubes: 5536
Heat exchange surface: 2510 m²
Inner collector diameter: 800 mm
Wall thickness at this point: 136 mm
Secondary side volume: 70 m³
Secondary side vol. at nominal level: 40 m³
Heat transmission coefficient: 4850 W/m² °C

Secondary coolant temperature
- At outlet: 258,9 °C
- At inlet: 226 °C

Primary coolant temperature
- At inlet: 296 °C
- At outlet: 265 °C

Log mean temperature difference: 22.3 °C
Mean heat flux: 108 KW/m²
Mean velocity of coolant in tubes: 2.37 m/s
Steam velocity in inlet of separator: 0.323 m/s
Steam generator blowdown: 0.5 % of generated steam
Dry weight: 185 t

Basic design characteristics

2.5-7/27
Normal conditions:

Normal start-up from cold state .............................................. 300
Start-up from half-hot state .................................................. 700
Normal shutdown ....................................................................... 1000
Shutdown initiated by reactor protection ..................................... 600
Start-up from hot-state after reactor protection initiation .......... 600
Load rejection up to household power level ............................... 90
Load-up from household power level to nominal power level ........ 90
Step-by-step power reducing from 100 % to 50 % and back .......... 200
Step-by-step power following with 10 % power ......................... 20000
Primary loop switch-off .............................................................. 100
Reactor coolant pump start ......................................................... 100
Hydrotest at 191 bar ................................................................ 20
Tightness test at 137 bar ............................................................. 130
Secondary side hydrotest at 76 bar ............................................ 20
Secondary side tightness test at 55 bar ....................................... 70
Safety relief valves test
  during unit start-up ................................................................. 70
  during operation ................................................................. 30
HPP's switch-off ...................................................................... 1096

Abnormal conditions:

CR's uncontrolled withdrawal .................................................... 10
Total loss of off-side power ....................................................... 10
Complete dry-out of secondary side .......................................... 10
Emergency feedwater supply with + 5 - 164 °C feedwater .......... 10
Steam-line rupture .................................................................... 1
Feedwater-line rupture .............................................................. 1
SG’s safety-relief valve opening and sticking ............................ 1
Pressuriser safety-relief valve opening and sticking ................. 1

Emergency conditions:

Large LOCA ........................................................................... 1
RCP rotor stuck ......................................................................... 1
Different size of primary pipes rupture .................................... 1 of each size

Basic design conditions

2.5-8/27
Average Corrosion of different C-Steel as a Function of Temperature

SG Chemical Cleaning
Steam Generator Chemical Cleaning

Test Results of 60g/l EDTA at different pH - Values

- pH = 9.5
- pH = 9.1
- pH = 8.6
- pH = 8.0
- pH = 7.5
- pH = 7.1
Steam Generator Chemical Cleaning

Test Results of 60g/l EDTA at different Temperatures

- **120°C 5h**
- **140°C 2.5h**
- **160°C 2h**
- **175°C 2h**
- **200°C 2h**
Material Compatibility Test (Paks)

Test Conditions:
Steam-off

Temp [°C]

Time [hrs]

Solvent:
- 100g/l NTA
  pH: 9.5
- 150g/l EDTA
  pH: 8.0

SG Chemical Cleaning
Steam Generator Chemical Cleaning

Test Results at different EDTA Concentrations

- Average corrosion
- Saturation carbon steel
- Part of dissolved sludge
- EDTA 60 g/l
- EDTA 90 g/l
- EDTA 120 g/l
- EDTA 180 g/l

μm

%
Saturation of Solvent B as a Function of Time at Different Temperatures

Saturation [%]

Time [min]

- 120°C
- 140°C
- 160°C
- 175°C
- 200°C

SG Chemical Cleaning
Average Corrosion of C-Steel as a Function of pH-value

Test Conditions
Temp. : 175°C
Solvent : 60g/l EDTA
Duration: 2hrs

SG Chemical Cleaning
The stress distribution of the SG's vessel induced by cool-down
The stress distribution of the SG's vessel induced by cooldown.
The deformation of the SG's
Steam generator chemical cleaning

- 1 Chemical injection
- 2 Cool-down
- 3 Filling-up the SG's
- 4 Cool-down
- 5 Drainage, rinsing filling-up
- 6 Heating-up
- 7 Chemical injection
- 8 Cool-down
- 9 Filling-up the SG's
- 10 Cool-down
- 11 Drainage, rinsing filling-up
Aus den DEs entfernte Ablagerungen
Paks Chemische DE-Reinigung

Abb. 3

The removed sludge from the SG's
The heat transmission coefficient of the unit 1 SG's after the unit restart
HEAT TRANSMISSION KOEFF.

Graph showing heat transmission coefficient over the period from 23-Jul-91 to 12-Aug-91.

X-axis: IDO dates from 23-Jul-91 to 12-Aug-91
Y-axis: Heat transmission coefficient values from 8 to 11
The heat transmission coefficient as function of time
OPERATING EXPERIENCE WITH WESTINGHOUSE
MODEL F STEAM GENERATORS

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ABSTRACT

In 1980 the first Model F steam generators designed by Westinghouse were put into operation. These steam generators were designed to address issues affecting the industry and included enhanced materials for tubing and tube supports, enhanced tube support plate hole design, and hydraulic tube-to-tubesheet expansion.

There are 84 Model F steam generators operating in 25 nuclear plants with a wide range of operating environments. This paper lists the operational conditions and describes the performance of the Model F steam generators over the past eleven years. To date, the Model F steam generators have accumulated over 450 SG-years of operation with less than 1 tube plugged per year of operation. The 25 Model F-type replacement steam generators have an even more impressive record with less than 1 tube plugged per steam generator for every 4 years of operation.

RESUME

Les premiers générateurs de vapeurs de type Modèle F conçus par Westinghouse ont été mis en exploitation en 1980. Ces générateurs de vapeur ont été conçus pour fournir un fonctionnement et une fiabilité accrus, grâce à toute une variété de matériaux et de changements de la conception. Parmi ceux-ci il y a les changements faits aux matériaux utilisés pour le tubage de transfert de chaleur et les plaques entretoises, une conception améliorée des trous de palques entretoises, et l'utilisation de l'expansion hydraulique tube-à-plaque tubulaire.

Actuellement il y a en exploitation 84 générateurs de vapeurs de type Modèle F dans 25 reacteurs nucléaires avec une large gamme d'environnements d'exploitation. Ce document énumère les conditions de fonctionnement et décrit la performance de ces générateurs de vapeur au cours des onze dernières années. Jusqu'à ce jour, plus de 450 années d'exploitation GV se sont accumulées avec moins de 1 tube obturé par année d'exploitation GV.
GENERAL BACKGROUND

Steam generators (SGs) designed and manufactured by Westinghouse first went into commercial operation in the early 1960's. The tube bundles of these early SGs represented heat transfer surface areas ranging from 2000 to 13000 ft² (186 to 1208 m²). The most recent Westinghouse designed steam generators, the Model F(1), have surface areas ranging from 44000 to 75000 ft² (4089 to 6970 m²). The first Westinghouse Model F steam generator came on line in 1980. At present, there are 84 Model F steam generators at 25 nuclear power stations in operation around the world.

Figure 1 presents a summary of all operating Westinghouse steam generators; Figures 2 and 3 list the population of Model F type plants and graphically represent their accumulated experience in SG-years of operation. All the Model F type steam generators have operated on All Volatile Treatment (AVT) secondary side water chemistry. Enhanced performance and reliability features were included in the design of the Model F SGs; these include such items as the use of thermally treated Alloy 600 or thermally treated Alloy 690 heat transfer tubing, full depth hydraulic expansion to close the tube-tubesheet crevices, and broached stainless steel tube support plates. A cutaway drawing of one of the latest Model F steam generators is shown in Figure 4.

A general description of some of the major features of Model F steam generators is provided below.

MODEL F DESIGN FEATURES

Full-depth Hydraulic Expansion. Tube-to-tubesheet crevices can boil dry, thereby permitting the concentration of impurities which can accelerate corrosion. Closing these crevices minimizes the potential for this type of corrosion. In Westinghouse Model F SGs the crevices are closed by hydraulic expansion; this process imparts lower residual stresses than other mechanical processes. Autoclave and model boiler tests, involving exposure of tubes hydraulically expanded in tubesheet simulants to aggressive chemical environments, have confirmed the benefits of this process. This laboratory performance has been verified in operating steam generators with hydraulically expanded tubing. No indication of tube corrosion associated with the hydraulic expansion transition region - either on the primary side or on the secondary side - has been detected. There have been no indications of corrosion within the tubesheet in the more than 800,000 hydraulic tubesheet joints in operation over the past eleven years.

(1) Throughout this paper, the term "Model F" will be used to describe those Westinghouse-designed SGs which contain Model F design features - i.e., thermally treated Alloy 600 or Alloy 690 tubing, hydraulically expanded into the tubesheet, and broached stainless steel tube support plates. This includes the Models 44F, 51F and 54F replacement SGs, the Model D5 preheat SGs, and the Model F SGs.

2.6-2/16
Broached Tube Support Plates. The Model F tube support plates, manufactured from Type 405 stainless steel, have broached holes which permit the secondary side water flow to pass through the plate next to the tubes. In all but the earliest SGs, flat contact surfaces support the tubes, limiting the accumulation of contaminants at the intersections between tubes and plates. Testing has shown that the concentration factor for a flat contact, broached hole is less than 1% of that for a cylindrical hole. Model boiler tests have also confirmed that local tube surface temperatures within the plate are much lower in the presence of the broached holes compared to those found in drilled hole crevices. The broached hole and stainless steel material were incorporated to address the corrosion phenomenon known as denting. Today, there are more than 5.5 million broached support plate-tube intersections in operation; not a single tube has been taken out of service because of denting at these intersections.

Thermally Treated Alloy 600 Tubing. In the mid-1970s, extensive laboratory testing indicated that thermal treatment of Alloy 600 in the temperature range near 1300°F (704°C) resulted in a microstructure, characterized by predominantly grain boundary carbide precipitation, with improved resistance to caustic stress corrosion cracking. Subsequent testing in accelerated pure and primary water environments indicated this microstructure also possessed greater resistance than mill annealed Alloy 600 to primary water stress corrosion cracking (PWSCC). Beginning with the manufacture of the Model F replacement SGs, thermally treated Alloy 600 was adopted as the heat transfer tubing material. Together with the use of hydraulic expansion, this combination of materials/manufacturing changes has resulted in excellent field performance to date.

RECENT ENHANCEMENTS

Westinghouse continues to refine and enhance the Model F steam generator design. Two of the most recent major enhancements of today's Model F steam generators are thermally treated Alloy 690 tubing and minimum-gap U-bend assembly. These features augment the design margins inherent in the Model F steam generator.

Alloy 690 Tube Material. This material has been under development since the early 1970's. Extensive industry-wide test programs have determined that it is the preferred material for steam generator applications. Perhaps the best overview of these programs is a recent EPRI report, "Alloy 690 for Steam Generator Tubing Applications" [1]. Figure 5 presents a comparative summary of the relative corrosion resistance of candidate SG tube alloys from this report, modified slightly to reflect the most recent data. In assembling the EPRI report, corrosion tests were surveyed for SG-relevant environments for which data were available for candidate alloys in addition to Alloy 690. This report concluded that Alloy 690 possesses physical characteristics and corrosion resistance that are superior to those of other materials which are candidates for steam generator tubing.
Data published for Alloy 800, an alloy sometimes considered as an alternate to Alloys 600 or 690, shows that this alloy exhibits susceptibility to corrosion in acid chloride and caustic environments. Results of a recent series of corrosion tests, performed by CIEMAT, were presented at an EPRI IGA/SCC Workshop in May 1991 [2]. These tests consisted of exposure of stressed c-rings of Alloys 690, 600 and 800 (nuclear grade) in pure caustic, caustic-plus-Cu or Pb or -thiosulfate, acid sulfate, and acid sulfate-plus-Pb showed that whereas 70 to 100% of the Alloy 800 samples cracked in 500 hour exposures, the only Alloy 690 cracking observed was in the caustic environments containing Pb or thiosulfate additions. Data like these have persuaded major utilities throughout the world to specify Alloy 690 as the material to be used in future steam generators. At present there are eight Westinghouse Model F steam generators in operation with thermally treated Alloy 690 tube material.

Minimum-Gap U-Bend Assembly. The tubes in the U-bend region of the tube bundle are supported by either two or three sets of anti-vibration bars (AVB's). The AVBs are oriented so that they are nearly perpendicular to the axes of the tubes.

The Model F U-bend support system is designed to provide margin against the potential for fluidelastic tube vibration. The assembled clearances between the tubes and their nearest anti-vibration bar supports are measured after assembly and have been confirmed to be less than 0.1 mm. This high precision construction is produced primarily by controlling the tube diameters and the thicknesses of the AVBs. All tubes and anti-vibration bars are measured after bending and confirmed to be acceptable before assembly. After installation of the AVBs, a flexible gap measuring tool equipped with strain gages is used to measure the as-built clearances between the tubes and bars within the tube bundle U-bend assembly. Westinghouse has manufactured sixteen (16) steam generators utilizing this method since 1986. Figure 6 demonstrates the advanced tube diameter dimensional control at the AVB intersection locations on the U-bends. Figure 7 shows the actual tube-AVB gap distribution measurement from recently manufactured Model F steam generators.

OPERATING HISTORY

Westinghouse-manufactured steam generators are presently in operation at 77 nuclear power plants with a total of 253 steam generators in service (as shown in Figure 1). Figure 8 summarizes the operating experience of all Model F type steam generators. Eighty-four steam generators are presently in operation with Model F features.

Westinghouse has furnished replacement steam generators for eight operating units. The Models 44F, 51F and 54F replacement steam generators all utilize features typical of the Model F steam generator. Other replacement designs are currently being offered and differ by tube surface area and, in some instances, by tube size and tube pitch. Of the 25 Westinghouse replacement steam generators in operation, 17 employ thermally treated Alloy 600 tube material. This population of steam

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generators has operated for an average of 8.8 years. In that time the total number of tubes that have been plugged is 46. This is equivalent to 1 plugged tube per steam generator every 3.2 years of operation. In addition, eight Westinghouse operating replacement steam generators are tubed with thermally treated Alloy 690. These steam generators have operated for an average of 2.2 years with no tubes plugged.

For all Model F steam generators, in the period from 1980 to the end of June 1991, a total of 591 tubes have been plugged; this corresponds to approximately 0.15% of the more than 400,000 tubes in operation, with more than 470 SG-years of operating experience. This is equivalent to approximately 1 tube plugged per steam generator for every 0.8 year of operation. Figure 8 shows graphically the Model F steam generator tube plugging by cause from 1980 through 1990.

The Model F SG corrosion performance is particularly impressive in view of the fact that all "full Model Fs" operate with inlet temperatures at/near 619°F (326°C). This temperature is approximately 12 to 15 Fahrenheit degrees (7 to 8 Celsius degrees) higher than the replacement SGs with Model F features. Both types of SGs in the Model F family, however, operate at temperatures above those at which tube degradation has been experienced in other SG models.

In the Model F experience, two forms of tube degradation have been observed; these are discussed briefly below.

Mechanical Wear. Two forms of mechanical wear have necessitated tube plugging in Model F SGs: wear from loose foreign objects and wear at AVB locations. At nine plants, tubes have been plugged for loose foreign object wear as identified by eddy current indications. Ten of the 25 Westinghouse operating plants with Model F steam generators have plugged tubes for AVB region tube degradation. The nature of this phenomenon is such that only a small percentage of the tube-to-AVB intersections are potentially affected. Close follow of the progression of this type of wear indicates that the issue is bounded and the conditions responsible for its occurrence are understood. At only one plant equipped with Model F steam generators has the wear experienced at the AVB region been exceptional and beyond the levels predicted by analysis. This situation is being addressed and replacement AVBs will be installed in this plant in 1992. All SG's manufactured since 1986 are manufactured using the minimum-gap AVB assembly process. No AVB region degradation has been observed in SGs with the minimum-gap AVB assembly.

Top-of-Tubesheet Denting. At one plant, 125 tubes were plugged because of denting above the top of the tubesheet. This denting has occurred within the sludge pile region but is located above the top of the tubesheet in what may be described as the "free length" of the tube. These tubes showed a reduction in diameter; in some cases this occurred as much as an inch above the secondary face of the tubesheet. This phenomenon is believed to be caused by corrosion of some foreign material that was introduced from outside the steam generator and was deposited on top of the tubesheet in the low-fluid velocity zone. This condition was detected by eddy current inspection. Retesting with surface riding detectors suggested the presence of cracks in some of these dent locations.
Metallurgical examination of this region of a tube removed from the steam
generator showed the presence of OD-initiated stress corrosion cracks
which were mainly axial in orientation. This occurrence is believed to be
due to factors independent of SG design or manufacture.

The two cases where tube plugging has been notable - one plant
with unusual AVB wear and one plant with free length tube denting - are
considered anomalies in that they clearly lie far beyond the bounds of the
behavior of the Model F SG population. If the plugging statistics for
these two cases are regarded separately from the overall data base, the
percent of tubes plugged in the 79 remaining Model F type steam generators
is 0.08% or equivalent to 2 tubes plugged per steam generator for every 3
years of operation.

In Figure 8, the "other" category for tubes plugged consists of a
variety of causes including foreign object wear as described above, random
eddy current signals not relevant to SG tube degradation, and indications
due to local geometric anomalies. In several instances, tube pulls have
been made to confirm the absence of degradation.

No tubes in Model F steam generators have been plugged due to any
of the major corrosion processes that have affected other model steam
generators. These processes include the following:

Thinning. No wall thinning in the sludge pile at the tube support
plates has been observed or detected.

Pitting. No pitting has been observed or detected.

Primary Water Stress Corrosion Cracking (PWSCC). No eddy current
indications have been attributed to PWSCC in Model F steam generators and
no plugging has been required for this process. The performance of
thermally treated Alloy 600 tubing, stress relieved after bending the
small-radius U-bends and expanded into the tubesheet using hydraulic tube
expansion, has been excellent. Based on the laboratory tests mentioned
above, the introduction of Alloy 690 is expected to continue this trend;
Alloy 690 is far superior to Alloy 600 in this regard.

OD Stress Corrosion Cracking/Intergranular Attack. With the
exception of the degradation observed in one plant in association with
free length denting, there has been no indication of OD intergranular
corrosion to date. While thermally treated Alloy 600 and Alloy 690 are
not immune to secondary side corrosion in the presence of sufficiently
consolidated faulted environments, their greater resistance, together with
the enhanced design and operating characteristics of the Model F steam
generators, has resulted in excellent performance.

Tube Support Plate Denting. There have been no eddy current
indications of denting at the tube support plate region. The type 405
stainless steel material and the broached tube support plate design are
credited with this excellent performance.

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EXPERIENCE SUMMARY

Eighty-four steam generators with the Model F type features are presently in operation; these include twenty-five replacement steam generators. Of the approximately 400,000 tubes in service only about 0.15 percent of the tubes have been plugged. The Model F performance is excellent and unequalled in the industry.
REFERENCES


<table>
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<tr>
<th>SG Model</th>
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<th>Number of SGs</th>
<th>First Year of Operation</th>
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<tr>
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<td>12</td>
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<td>44 Series</td>
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<td>8</td>
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</tr>
<tr>
<td>51 Series</td>
<td>21</td>
<td>73</td>
<td>1972</td>
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<td>Models D2, D3</td>
<td>11</td>
<td>35</td>
<td>1980</td>
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<tr>
<td>Models D4, D5, E</td>
<td>10</td>
<td>37</td>
<td>1981</td>
</tr>
<tr>
<td>Models 51F, 54F</td>
<td>3</td>
<td>10</td>
<td>1980</td>
</tr>
<tr>
<td>Model 44F</td>
<td>5</td>
<td>15</td>
<td>1982</td>
</tr>
<tr>
<td>Model F</td>
<td>14</td>
<td>47</td>
<td>1983</td>
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<tr>
<td><strong>All Models</strong></td>
<td><strong>77</strong></td>
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*Figure 1. Summary of Operating Westinghouse Steam Generators*
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<tr>
<th>25 Plants</th>
<th>Number of SGs</th>
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<tr>
<td>Surry 2</td>
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<td>Surry 1</td>
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<td>9.9</td>
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<td>Turkey Pt 3</td>
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<td>9.1</td>
</tr>
<tr>
<td>Turkey Pt 4</td>
<td>3</td>
<td>8.1</td>
</tr>
<tr>
<td>Kori 2</td>
<td>2</td>
<td>8.1</td>
</tr>
<tr>
<td>Maanshan 1</td>
<td>3</td>
<td>7.2</td>
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<tr>
<td>Pt Beach 1</td>
<td>2</td>
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<td>Callaway 1</td>
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<td>Robinson 2</td>
<td>3</td>
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<tr>
<td>Kori 3</td>
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<td>6.4</td>
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<td>Wolf Creek</td>
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<td><strong>Total or Average:</strong></td>
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<td><strong>5.7</strong></td>
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</table>

*Figure 2.*

*Model F Steam Generators*
Figure 3

Cumulative Operating Experience of Model F-Type Steam Generators
Secondary Separators

Steam Nozzle with Flow Restrictor

Secondary Side Manway

Sixteen 20 in (508 mm) Outside Diameter Primary Separators

Feedwater Inlet Nozzle

All Forged Shell

Stainless Steel AVBs and Tube Supports

Minimum-Gap U-Bend Supports

Thermally Treated Alloy 690 Tubes

Two 4 in (102 mm) U-Bend Access Ports (Rotated)

Broached Flat-Contact Tube Support Plates

Broached Flat-Contact Flow Distribution Baffle

Hydraulically Expanded Tube-to-Tubesheet Joints

Forged Channel Head

Six 6 in (152 mm) ID Handholes

2 Primary Side Manways

Figure 4.

Model F Steam Generator

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<table>
<thead>
<tr>
<th>Corrosion Issue</th>
<th>Alloy 600 MA</th>
<th>Alloy 600 TT</th>
<th>Alloy 800 Mod.</th>
<th>Stainless Steel (Nonstabil.)</th>
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<td>- Acid Chloride</td>
<td>1</td>
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<td>(2-3)</td>
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<td>- Caustic Below 6%</td>
<td>(3)</td>
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<td>- Caustic 10-50%</td>
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<td>- Pure, Primary and AVT Water with H₂</td>
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<td>1</td>
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<td>- Pure Water with O₂</td>
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<td>- Alkaline</td>
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<td>1</td>
<td>(1)</td>
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<td>- Acid</td>
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<td>- Plus Chloride</td>
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<td>1</td>
<td>U</td>
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<tr>
<td>- Neutral</td>
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<td>- Alkaline</td>
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<td>Lead (AVT Water):</td>
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<td>Intergranular Attack:</td>
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<td>- Alkaline</td>
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<td>Pitting (in Chlorides)</td>
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<td>(3)</td>
<td>(1)</td>
<td>(1)</td>
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**Rankings:** 1 - Best; 5 - Worst; ( ) - Estimates; U - Unknown

**MA** - Mill Annealed; **TT** - Thermally Treated; **Mod** - AS800 Nuclear Grade

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Figure 6.

Comparison of U-Bend Ovality Control
(Conventional vs. Model F SG)
Figure 7.

Tube/AVB Gap Distribution
Figure 8.
Model F-Type Steam Generator Tube Plugging By Cause
3.1 OPERATING EXPERIENCE WITH STEAM GENERATORS

R. Bouecke, Siemens/KMU, Erlangen, GERMANY

3.2 AVT USING MORPHOLINE ALONE: A UNIQUE EXPERIENCE AT A CANDU-PHW PLANT IN CANADA

R. Gilbert, Institut de Recherche d’Hydro-Québec Varennes, Québec, CANADA
Y. Dünder, Hydro-Québec, Centrale nucléaire Gentilly 2 Bécancour, Québec, CANADA
A. Marchand, Hydro-Québec, Centrale nucléaire Gentilly 2 Bécancour, Québec, CANADA

3.3 CHEMICAL CLEANING AS A MEASURE TO IMPROVE STEAM GENERATOR PERFORMANCE OF PWR-PLANTS

S. Odar, Siemens/KMU, Erlangen, GERMANY
K. Kuhnke, Siemens/KMU, Erlangen, GERMANY

3.4 OPERATING EXPERIENCE WITH STEAM GENERATOR WATER CHEMISTRY IN JAPANESE PWR PLANTS

K. Onimura, Mitsubishi Heavy Industries Ltd., Kobe, JAPAN
T. Hattori, Mitsubishi Heavy Industries Ltd., Kobe, JAPAN

3.5 OPERATING EXPERIENCE WITH STEAM GENERATORS IN KORI UNIT 1

Uis-Seong Hwang, Kori Nuclear Power Plant, KOREA
OPERATING EXPERIENCE WITH
STEAM GENERATORS

R. Bouecke
Siemens AG
D-8520 Erlangen, Germany

ABSTRACT

In contrast to steam generator tube degradation problems that have been widely encountered worldwide, steam generators of the Siemens/KWU design have proven by operating experience that they are very efficient in minimizing tube corrosion or any other SG related problems. The paper will substantiate this statement by addressing the performance characteristics of nearly 20 years of operation experience. Emphasis is put on evaluations comparing the heat transfer capacity of Incoloy 800 with that of Inconel 690 TT. Various tube support designs are discussed with respect to hide-out behaviour. Recent evaluations confirm the superiority of grid type tube support designs compared to tube support plates. Anti vibration bars acc. to Siemens/KWU design allow proper support even of the innermost U-tubes, by which excessive vibration induced tube failures due to fatigue are ruled-out.
1. Introduction

Operating pressurized water reactors (PWRs) with U-tube steam generators (SGs) have encountered difficulties associated with either one or a combination of inadequate material selection, poor design and manufacturing and an unsufficient water chemistry control which resulted in excessive tube degradation. Tube degradation is related to corrosion phenomena, such as wastage, pitting, intergranular attack (IGA), primary water stress corrosion cracking (PWSCC), intergranular stress corrosion cracking (IGSCC) and others.

manufacturing and operational experience, it has been possible to specify optimum clearances between the tubing and both the straight-tube and U-bend supports.

The tube support grids in Siemens/KWU SGs are made of highly alloyed corrosion-resistant steel, which is not sensitive to excessive growth of magnetite, helping to minimize tube deformation in the region of tube-to-tube-support crevices because of denting, intergranular attack (IGA) and/or intergranular stress corrosion cracking (IGSCC).

The field experience comparing the hideout behaviour of the steam generators with respect to tube support design confirms the superiority of the eggcrate design (Fig. 4). Steam generators having eggcrate tube supports experience less hideout of impurities (i.e. concentrating of impurities in the SGs).

Flow Distribution Plate and Blowdown System

The blowdown system is a circular channel in the tubesheet with a drain and blowdown nozzle. To improve blowdown efficiency a flow distribution plate is used to direct the flow onto the tubesheet, keeping the corrosion products in suspension so that they can be removed through the blowdown line. This reduces accumulation of corrosion products on the tubesheet, thereby minimizing sludge pile corrosion and effects like wastage, pitting and denting.

The results of the tubesheet fibrescope inspections of old and new plants gained in the last 10 years show that old plants operating with low AVT (pH-value < 9,5) and having no flow distribution plate experience more corrosion product deposits on the tubesheet in comparison to new plants which operate with high AVT (pH-values > 9,8 %) and having flow distribution plates in their SGs.

2.1.3 Thermal Hydraulic Performance

Two parameters significantly influence the thermal/hydraulic performance of steam generators, firstly the heat transfer capacity of the tubing material and secondly long-term fouling effects due to deposition of crud on the tube surfaces. Both parameters have to be adequately specified in the design stage to ensure plant operation without loss of power.

The operating performance of the Siemens/KWU steam generators with respect to fouling is summarized in Figure 5.

In general a linear increase of SG fouling was experienced in the past, where AVT chemistry with pH-values lower than 9.5 was applied. This field experience is not SG specific; it has also been observed worldwide on SG's operating under similar chemistry conditions. Increasing of the pH-value was favourable with respect to SG fouling; thereby this phenomena could be reduced.

Recent experience with Inernal 690 TT indicates that the above statement is no longer adequate.
3. **Water Chemistry Characteristics**

Besides detailed design engineering, proper material selection and careful manufacturing, chemistry guidelines within rather close limits have to be specified and monitored. Nevertheless, considering the exceptional features of the Siemens/KWU steam generators, the possibility for relaxation of chemistry guidelines was considered in the actual specification, thus enabling the utilities to operate the plant more flexible.

3.1 **Primary Side**

The chemistry restrictions for the steam generators are much less demanding than those corresponding to the rest of the primary circuit. In all cases the limiting factors are found in other components (reactor, fuel elements) or in other objectives pursued (dose rate buildup minimization). Therefore, no steam generator related chemistry guidelines are specified for Siemens/KWU replacement steam generator.

3.2 **Secondary Side**

Important factors affecting the corrosion behaviour of steam generator tubes are corrosion products, impurities (salts and organic substances), oxygen and other oxidants. The corrosion products have the greatest influence. They are deposited particularly in regions of high heat flux and stagnant flow (SG tubes, tubesheet, crevices between tube and tube supports). It is on these deposits that salts and other non-volatile impurities accumulate. There are always impurities present even if only in very small quantities - as it is impossible to provide absolutely pure feedwater and make-up water.

Fig. 6 shows the development of the secondary water chemistry used in Siemens/KWU PWR nuclear power plant, illustrating the way in which changes have been made over the years:

- Phosphate conditioning of the steam generator water, low AVT for the rest of the steam/water cycles (pH levels between 9.0 and 9.5).
- Phosphate conditioning of the steam generator water, high AVT for the rest of the steam/water cycle (pH level above 9.8)
- Low AVT of the entire steam/water cycle
- High AVT of the entire steam/Water cycle.

The water chemistry specifications are summarized in the Tables 1 and 2.

The type of conditioning depends on the design of the secondary cycle. In the past the turbine condenser tubes in Siemens/KWU nuclear power plants were made of copper alloys. In these plants, which were commissioned before 1980, phosphate conditioning of the steam generator was applied. Towards the end of the seventies, more and more wastage occurred in SG tubes. In addition to short term remedies (e.g. lancing of the tubesheet), strategies were established for improvement of the overall integrity of the secondary side. Most utilities replaced their condenser tubes with new ones made out of stainless steel or titanium, thereby creating suitable conditions for increasing the pH level and considerably reducing the entrainment of corrosion products into the steam generator. Depending on the plant-specific circumstances, the
hardware modifications took varying periods of time, so it was ten years before full transition to this new chemistry was completed.

4. Provisions for Remedial Actions

4.1 Maintenance, Inspection- and Repair-Techniques

For inspection, maintenance and repairs, sufficient access to both chambers of the primary head is provided by manways. The accessibility of the manways and the design of the manway covers, studs and nuts allow use of stud tensioners. This shortens the opening and closing process and contributes to the ALARA man rem dose concept. The upper portion of the steam generators is equipped with a largely sized manway. The large size in combination with the utilization of a stud tensioner helps to reduce the exposure time for personnel during inspection, maintenance and repairs.

Four handholes are located above the tubesheet. These handholes allow inspection of the lower area of the tube bundle and the secondary surface of the tubesheet. The strategic location as well as the large size of the handholes facilitates sludge lancing and simplifies the necessary activities above the tubesheet and reduces the time needed for any inspection and maintenance activities in this area.

In order to ensure satisfactory performance of steam generators in-service inspections and maintenance work have to be performed regularly. In spite of the fact that steam generators equipped with Incoloy 800 mod. tubing show very little corrosion, a complete arsenal of equipment for testing and repairs of steam generator tubes has been developed. A robot was developed for all kinds of steam generator tube inspections and maintenance. The robot basically consists of a support structure and two arms by which every position of the tubesheet can be reached. The robot can be used for inspection of the steam generator tubes as well as for tube plugging, sleeving, welding, machining etc.

In addition to the standard multifrequency eddy-current technique, special eddy current and ultrasonic probes were developed. These probes are used to analyse tube defects and to obtain additional information on kind, size and depth of defects. A rotating combined ultrasonic and eddy-current probe is specially suited for the analysis of large volume defects resulting from wastage corrosion or fretting.

The most common repair technique is plugging. The former explosive plugs have meanwhile been completely replaced by superior removable mechanical plugs. They are placed in the defective tubes by manipulators such as finger walkers or robots as described above. The sleeving technique developed by Siemens/KWU is characterised by welding of the sleeves at both ends. Advantages of this technique are absolute leak tightness and inspectability of the welds.

4.2 Cleaning Achievements

Cleanliness is one of the most important prerequisites for a steam generator's long life. Therefore emphasis was put on the development of cleaning techniques:
A tubeshed lancing equipment serves to further reduce the risk of corrosion by removing the sludge pile on the tubeshed by water blasting. The equipment is designed to be installed in the tube lane of a steam generator through handholes above the tubeshed (Fig. 7). Several spray nozzles are arranged on a spray head, which moves along the tube lane. The spray head can also be moved up and down covering with its water jet the whole region of crud dropout. Because of the triangular pitch the whole circumference of the tubes can be reached by the water jets (Fig. 8).

Periodic tubeshed lancing in combination with measures such as replacing Cu-containing materials from the steam/feedwater cycle and thus maintaining a high pH to reduce the transport of corrosion products into the steam generators resulted in a minimum build-up of crud. After removal of usually less then 10 kg per steam generator the tubeshed is clean as was demonstrated by fiber optic inspections.

In case there are big amounts of crud in a steam generator at locations which cannot be reached by water jets, chemical cleaning of the tube bundle is an alternative for crud removal. The Siemens/KWU technique for chemical cleaning is able to remove copper as well as corrosion products from a steam generator without affecting the tubes or other structural materials. This method has been applied to the steam generators of numerous units in Germany and abroad. Up to 500 kg of crud could be removed per steam generator.

5. Operating Experience

As of the end of 1990 a total of 62 Siemens/KWU steam generators were in commercial operation worldwide. This number includes also 3 replacement steam generators for Ringhals Unit 2 with Inconel 690 TT tubes. Figure 6 provides an overview of the results of eddy current examinations and the measures which were taken in consequence.

As far as SG tubes made of Incoloy 800 mod. are concerned, phosphate wastage was in the past the only mechanism of significance experienced in the Siemens/KWU plants operating under phosphated treatment (especially Stade, Borselle and Biblis A). In addition to several secondary system improvements and the modifications to the water chemistry, annual cleaning of the steam generator (tubeshed lancing) during refuelling has proven a suitable means of restricting the progress of wastage. Since 1985 wastage corrosion is no more an issue for old Siemens/KWU SGs.

Apart from corrosion of this type within nearly 20 years there has been only one instance of a SG tube developing an intergranular crack within the sludge pile region on the secondary side of the tubeshed and two tubes with traces of pitting. With SG tubes made of Incoloy 800 acc. to Siemens/KWU specification there have been no indications of PWSCC, IGA, or chloride-induced SCC.

Operation experience with the stainless steel grid-type tube supports and vibration restraints in the U-bend region has verified that unacceptable tube vibration, which could lead to fretting, is effectively suppressed by this design. Extensive eddy current examinations carried out in all PWR nuclear power plants constructed by Siemens/KWU produced the following results:

- No fretting of SG tubes at the grid in the straight tube sections
Minimal fretting in the U-bend region in a few older plants
No fretting in the U-bend region in plants with standard vibration
restraints. This design is in operation since 1978.

Fretting damage to the extent discovered in preheater steam generators of
other vendors can be excluded on the grounds that Siemens/KWU - already from
the beginning - installed flow distribution boxes (Fig. 9) in order to achieve
uniform distribution of feedwater flow at a moderate velocity level which is
sufficiently below values inducing unacceptable tube vibrations. Eddy current
inspections showed neither indications of fretting attack nor any other kind
of tube wall degradation.

There were only eleven instances of SG tube leakage, of which half led to non-
scheduled outages. This resulted in a plant availability forfeit of less than
0.1 %.
Numerous publications deal with the effects of steam generator problems
on plant availability. The figures quoted range from just under 2 % for 1986
to over 4 % for the years between 1980 and 1984. These percentages do not
include outages for steam generator replacement. Comparison of the available-
ability figures confirms the effectiveness of the Siemens/KWU design and operat-
ing concept of steam generators in PWR nuclear power plants.

Due to the excellent inspection record, sleeving is not needed in Siemens/KWU
steam generators.

Neither at this time nor in the next future replacement of steam generators
have to be anticipated in Siemens/KWU designed nuclear power plants.

6. Conclusions

Steam generators are key components which can heavily effect plant
safety and availability. This was recognized by Siemens/KWU at a very early
stage of the SG design. A multi level concept was developed and consequently
applied, the characteristics of which can be highlighted as follows:

- Implementation of specific design features after careful experimental and/or
  analytical verification
- Material selection based on profound validation tests
- Stringent inspection requirements regarding control of manufacturing
- Deliberate specification and control of water chemistry guidelines
- Close feedback of operational problems to be readily considered in design
  improvements or remedial actions to be taken.

A strict application of this concept has reached a stage which allows the
following summarizing statements:

- All main design features of the Siemens/KWU SG are in use - basically
  unchanged - for roughly 20 years
- All of them are verified by excellent operating experience and available for
  implementation in any replacement or new SG design
- No need for replacement of Siemens/KWU SG is foreseen in the next future.
Fig. 1 Advanced Siemens/KWU Expansion Technique
Fig. 2 Steam Generator, Support of U-Tubes

**Corrugated strips at tube bends**
- No crud deposition
- Corrugated strips enable mutual support of the U-bends in the event of a main streamline break

**Eggcrates at straight portion of tubes**
- Little pressure loss (high circulation ratio)
- No crud deposition
- No denting (austenitic stainless steel)
- No fretting
Siemens/KWU Standard Support

Siemens/KWU Standard Support

SG Model 51

Fig. 3 Anti-Vibration Bar Arrangement in the U-Bend Region
Comparison between Ringhals 2 and 3 Steam Generator
Hideout of S-35

Fig. 4 Hideout - Characteristics
Fig. 5 Fouling - Behaviour
Fig. 6 Operating Experience with Siemens/KWU Steam Generators
Operation Periods and Inspection Times
Fig. 7  Tubesheet Lancing Equipment
Fig. 8  Tubesheet Lancung
Sludge Removal Possibilities of Triangular Versus
Square Pitch Design
• Split Flow Economizer
  Separate Feedwater Nozzles to Protect the Tubesheet Against Cold Feedwater Shocks

• Flow Limiters
  Reduce the Loads in Case of a FWL-Break

• Austenitic Stainless Steel
  Used for Baffle Material to Exclude the Risk of Denting

• Water Distribution Boxes
  Installed to Keep Flow Induced Excitations Low

• Baffles Connected by Tie Rods
  Thermal Expansion is not Restricted
  Stagnant Flow Areas are Minimized

• After More Than 9 Years of Operation:
  No Indication of Any Fretting Attack or Any Other Tube Corrosion Attack.

Fig. 9 Lower Part of the Siemens/KWU Steam Generator with Preheater

Status: 2/91

3.1-17/19
### CHEMISTRY GUIDELINES FOR REPLACEMENT STEAM GENERATORS 
OF THE SIEMENS/KWU TYPE 

#### POWER OPERATION

### BLOWDOWN SAMPLE CONTROL PARAMETERS

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Normal Frequency</th>
<th>Expected Value</th>
<th>Specified Value</th>
<th>Action Level</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cation Conductivity, µS/cm</td>
<td>continuous</td>
<td>&lt; 0,5</td>
<td>≤ 1,0&lt;sup&gt;c)&lt;/sup&gt;</td>
<td>&gt; 1&lt;sup&gt;c)&lt;/sup&gt; &gt; 2&lt;sup&gt;c)&lt;/sup&gt; &gt; 7&lt;sup&gt;c)&lt;/sup&gt;</td>
</tr>
<tr>
<td>Sodium, ppb</td>
<td>semi-continuous</td>
<td>&lt; 10</td>
<td>&lt; 50</td>
<td>&gt; 50 &gt; 100 &gt; 500</td>
</tr>
</tbody>
</table>

### DIAGNOSTIC PARAMETERS

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Expected Value</th>
<th>Frequency</th>
</tr>
</thead>
<tbody>
<tr>
<td>pH (25 °C)</td>
<td>&gt; 9,0</td>
<td>a</td>
</tr>
<tr>
<td>SO&lt;sub&gt;4&lt;/sub&gt;&lt;sup&gt;2-&lt;/sup&gt;</td>
<td>b</td>
<td>a</td>
</tr>
<tr>
<td>Cl&lt;sup&gt;-&lt;/sup&gt;</td>
<td>b</td>
<td>a</td>
</tr>
<tr>
<td>F&lt;sup&gt;-&lt;/sup&gt;</td>
<td>b</td>
<td>a</td>
</tr>
</tbody>
</table>

<sup>a)</sup> To be measured in case of overriding expected values  
<sup>b)</sup> Strong anions to be measured and their concentrations used to check the cationic conductivity specification 
<sup>c)</sup> Due to total strong anionic species 

The theoretical cation conductivity should be calculated and compared to the measured cation conductivity. Any inconsistencies should be investigated.

---

Table 1: Water Chemistry - Blowdown Water
### CHEMISTRY GUIDELINES FOR REPLACEMENT STEAM GENERATORS OF THE SIEMENS/KWU TYPE

**POWER OPERATION**

#### FEEDWATER SAMPLE CONTROL PARAMETERS

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Normal Frequency</th>
<th>Expected Value</th>
<th>Specified Value</th>
<th>Action Level</th>
</tr>
</thead>
<tbody>
<tr>
<td>pH</td>
<td>3/wk</td>
<td>≥ 9,3&lt;sup&gt;a)&lt;/sup&gt;</td>
<td>&lt; 9,3</td>
<td></td>
</tr>
<tr>
<td>Cation Conductivity, uS/cm</td>
<td>continuous</td>
<td>&lt; 0,12</td>
<td>≤ 0,2&lt;sup&gt;b)&lt;/sup&gt;</td>
<td>&gt; 0,2</td>
</tr>
<tr>
<td>Oxygen, ppb</td>
<td>continuous</td>
<td>&lt; 1</td>
<td>&lt; 5</td>
<td>&gt; 5 &gt; 20 &gt; 100</td>
</tr>
</tbody>
</table>

#### FEEDWATER SAMPLE DIAGNOSTIC PARAMETERS

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Normal Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Hydrazine, ppm</td>
<td>&gt; 0,02</td>
</tr>
<tr>
<td>Cation Conductivity, uS/cm</td>
<td>&gt; 5,5</td>
</tr>
</tbody>
</table>

<sup>a)</sup> as high as possible, compatible with BOP operation  
<sup>b)</sup> due to strong anions

Table 2: Water Chemistry - Feedwater

3.1-19/19
AVT USING MORPHOLINE ALONE: A UNIQUE EXPERIENCE AT A CANDU-PHW PLANT IN CANADA

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Bécancour, Québec, Canada

ABSTRACT

Gentilly 2 is a 685-MWe CANDU-PHWR (CANadian Deuterium Uranium - Pressurized Heavy Water Reactor) owned and operated by Hydro-Québec. The secondary cycle initially utilized an all-volatile treatment (AVT) with a combination of morpholine, as the volatile amine, and hydrazine, as the oxygen scavenger. Shortly after startup, it was decided to modify the AVT treatment by dropping the hydrazine addition and utilize the morpholine addition by itself. The system has been closely monitored to identify any problems that might develop due to the absence of the hydrazine. This paper reviews seven years of operating experience and includes the results of system inspections, steam generator tube and sludge analysis, and corrosion product transport studies. Both laboratory and field studies on morpholine stability and breakdown products are also discussed.

RÉSUMÉ

La centrale Gentilly 2, exploitée par Hydro-Québec, est un réacteur à eau pressurisée de 685 MWe de la filière CANDU-PHWR. Au début de l'exploitation, le conditionnement chimique de l'eau du cycle secondaire était entièrement effectué par injection de produits volatils: la morpholine pour le contrôle du pH et l'hydratzine pour le contrôle de l'oxygène. Peu de temps après le démarrage, les injections d'hydratzine ont été arrêtées, et pendant les sept années qui ont suivi, le traitement de l'eau a été réalisé uniquement par injection de morpholine. Un programme rigoureux de surveillance a été mis en place afin de détecter toute anomalie pouvant résulter de l'absence d'hydratzine. Cet article présente les données d'exploitation du système d'eau d'alimentation et des générateurs de vapeur. Il décrit brièvement les résultats d'inspections visuelles des systèmes, des analyses de dépôts, des études sur le transport de produits de corrosion ainsi que des études en laboratoire et en centrale sur les produits de décomposition thermique de la morpholine.
1.0 Introduction

Gentilly 2 is a 685-MWe CANDU-PHWR owned and operated by Hydro-Québec. The plant is located on the freshwater stretch of the St. Lawrence river, 150 km east of Montréal. Initial criticality occurred in September 1982 and the plant was declared in service in October 1983.

During the first 13 months of operation, the chemical treatment of the secondary cycle was an all-volatile treatment using a combination of morpholine and hydrazine. It was originally intended that the steam and feedwater systems would operate with 50 to 100 μg/kg hydrazine but pH control and hydrazine control soon proved incompatible [1]. The hydrazine was rapidly decomposing into ammonia: depending upon the hydrazine addition rate, the resulting solution could have a pH 9.5-9.8 with ammonia concentrations of 0.5-4.2 mg/kg.

This relatively high ammonia concentration and its possible deleterious effects on the copper alloy condenser tubes convinced the Gentilly 2 operating staff to modify the AVT program by dropping the hydrazine addition. This change was accompanied by extensive monitoring of both the feedwater and the steam generator systems, together with laboratory studies of morpholine, its effects and its breakdown products. Although the emphasis was on monitoring copper corrosion and transport, erosion-corrosion was also investigated.

Since March 1984, the plant has operated on morpholine alone. It is the only plant in North America to do so; all other Canadian and most American nuclear power stations use the conventional hydrazine-based treatment. This paper reviews Hydro-Québec's operating chemistry experience with the Gentilly 2 feedwater system and the steam generators. The results of system inspections, steam-generator tube and sludge sample analysis and corrosion product transport studies are presented along with laboratory and field studies on morpholine stability.

2.0 Plant Characteristics

2.1 General Description

A simplified description of the Gentilly 2 plant is shown in Figure 1. There are two heavy-water circuits: the moderator, which slows the neutrons to maintain the chain reaction, and the primary coolant, which transfers the heat from the fuel. In a CANDU, the fuel is in the form of 50-cm bundles of zircalloy-clad natural uranium oxide. Refuelling is performed on-power. As with other PWRs, the primary coolant passes through the tube side of the steam generator where the heat is transferred to "natural" water which boils to produce the steam.

A simplified flowsheet of the secondary circuit is shown in Figure 2. The Gentilly 2 turbine consists of one HP and two LP tandem turbines. Fresh water from the St. Lawrence River passes through the 54 000 condenser tubes to provide the cooling to condense the steam and maintain the system under vacuum. The condenser is designed to take up to 70% of the total steam flow and can be used in the event of a turbine trip.

The condensate is returned to the steam generators after advancing up the feedtrain in stages through two pairs of three LP heaters, a deaerator and two pairs of HP heaters. This gradual heating avoids thermal stresses in the steam generators. Of the three condensate extraction pumps, two offer 100% maximum continuous rated capacity, the other a 5% maximum continuous rated capacity. Five steam-generator feedwater-pumps are provided: three with 50% maximum continuous rated capacity and two with 4% maximum continuous rated capacity.
Figure 3 shows the design of the Gentilly 2 steam generators. There are four inverted "U" steam generators, each containing 3550 Incoloy 800™ tubes with 16mm OD. The Babcock & Wilcox Canada design has an integral steam separator and a high recirculation flow (>5). At full load, 1.0 Mg/s of steam is produced at 258°C with pressure of 4.5 MPa and a steam quality exceeding 99.75%.

2.2 System Materials

The chemical control specifications for the feedwater system and steam generators were based on the materials used to build the main components. Table I shows that the main secondary circuit components are carbon steel, copper alloy and stainless steel, which are not very compatible with regard to the pH of the water in contact. The corrosion rate of carbon steel is minimal at pH 10 to 10.5, whereas the corrosion of copper alloys increases rapidly at pH levels above 9.7. Consequently, a pH range of 9.2 to 9.5 was selected as the best compromise for chemical control of the condensate to ensure minimum tube corrosion in the main condenser and HP feedwater heaters, and throughout the secondary circuit.

<table>
<thead>
<tr>
<th>Component</th>
<th>Material</th>
</tr>
</thead>
<tbody>
<tr>
<td>Condenser:</td>
<td></td>
</tr>
<tr>
<td>Condenser tubes</td>
<td>Admiralty brass (Cu-Zn alloy)</td>
</tr>
<tr>
<td>Air extraction and high-velocity impingement zones</td>
<td>316 stainless steel</td>
</tr>
<tr>
<td>Tubesheet</td>
<td>Muntz (Cu-Zn alloy)</td>
</tr>
<tr>
<td>Feedwater system:</td>
<td></td>
</tr>
<tr>
<td>LP heater tubes</td>
<td>304 stainless steel</td>
</tr>
<tr>
<td>HP heater tubes</td>
<td>Carbon steel</td>
</tr>
<tr>
<td>Heater shells, tubesheets and bonnets</td>
<td>Carbon steel</td>
</tr>
<tr>
<td>Deaerator and storage tank</td>
<td>Carbon steel</td>
</tr>
<tr>
<td>Steam generators:</td>
<td></td>
</tr>
<tr>
<td>Steam-generator tubes</td>
<td>Incoloy 800™</td>
</tr>
<tr>
<td>Support plates</td>
<td>Tri-lobed broached plates, 410 stainless steel</td>
</tr>
<tr>
<td>U-bend supports</td>
<td>Staggered scallop bars, 410 stainless steel</td>
</tr>
<tr>
<td>Preheater baffle plates:</td>
<td>Drilled and coned:</td>
</tr>
<tr>
<td>bottom 3 plates</td>
<td>carbon steel</td>
</tr>
<tr>
<td>the rest</td>
<td>410 stainless steel</td>
</tr>
</tbody>
</table>

2.3 Secondary-Side Steam Generator Chemistry Specifications

The present steam generator and feedwater system chemistry specifications are given in Table II. Morpholine additions are controlled manually to keep the pH of the feedwater within the specified range; typical levels in the blowdown run from 10 to 15 mg/kg. The makeup specifications call for the water to contain <15 μg/kg sodium, <15 μg/kg chloride and <15 μg/kg total silica; the conductivity must not exceed 0.05 mS/m at 25°C. Most of the time,
contaminants are present in the makeup at levels below the detection limits (Na < 10 µg/kg, Cl<sup>-</sup> < 5 µg/kg and SiO<sub>2</sub> < 10 µg/kg).

Table II: Chemistry specifications for normal operating conditions

<table>
<thead>
<tr>
<th>Location</th>
<th>Parameters</th>
<th>Specifications</th>
</tr>
</thead>
<tbody>
<tr>
<td>Steam-generator blowdown:</td>
<td></td>
<td></td>
</tr>
<tr>
<td>pH at 25°C</td>
<td>mg/kg</td>
<td>9 to 10</td>
</tr>
<tr>
<td></td>
<td></td>
<td>&gt; 11 and &lt; 5**</td>
</tr>
<tr>
<td>Morpholine</td>
<td>µg/kg</td>
<td>10 to 15</td>
</tr>
<tr>
<td>Sodium</td>
<td>µg/kg</td>
<td>&lt; 70</td>
</tr>
<tr>
<td>Chloride</td>
<td>µg/kg</td>
<td>&lt; 100</td>
</tr>
<tr>
<td>Sulfate</td>
<td>µg/kg</td>
<td>&gt; 700**</td>
</tr>
<tr>
<td>Dissolved O&lt;sub&gt;2&lt;/sub&gt;</td>
<td>µg/kg</td>
<td>&lt; 1000**</td>
</tr>
<tr>
<td>Silica</td>
<td>µg/kg</td>
<td>&lt; 100</td>
</tr>
<tr>
<td>Spec cond at 25°C</td>
<td>mS/m</td>
<td>0.24 to 2.4</td>
</tr>
<tr>
<td>Iron/copper</td>
<td>µg/kg</td>
<td>ALARA****</td>
</tr>
</tbody>
</table>

Feedwater system:

| pH at 25°C              | mg/kg       | 9.2 to 9.5              |
|                        |             |                         |
| Morpholine             | mg/kg       | 10 to 15                |
| Ammonia                 | mg/kg       | < 0.8                   |
| Dissolved O<sub>2</sub> | µg/kg       | < 35                    |
| CEPD<sup>***</sup>      | µg/kg       | < 10                    |
| HPHO<sup>***</sup>      | µg/kg       | < 0.25                  |
| Sodium                  | µg/kg       | < 5                     |
| Iron/copper             | µg/kg       | < 15                    |
| Silica                  | µg/kg       | < 100                   |
| Spec cond at 25°C       | mS/m        | < 0.8                   |

* No hydrazine addition since March 1984
** Resulting in reactor shutdown
*** Measured in condensate extraction pump discharge (CEPD) and high-pressure heater outlets (HPHO)
**** As Low As Reasonably Achievable

3.0 Operational Experience

3.1 Feedwater System Chemistry

Figures 4 and 5 illustrate the chemical control of the feedwater system from April 1984 to May 1991. The pH, dissolved oxygen and sodium were measured daily and the iron, copper and crud (insoluble matter) were measured weekly. These graphs include all the raw data taken over the period.
The mean value of the pH measurements at the outlet of the HP feedwater heaters has been 9.35 (± 0.10 standard deviation) for the last seven years' operation. Almost all values lie between 9.2 and 9.5, which corresponds to the optimum pH range for the secondary-circuit materials (Fig. 4a).

The dissolved oxygen at the HP feedwater heater outlets meets the requirement of < 5 μg/kg, with an average close to 3.4 μg/kg for the entire period (Fig. 4b). Monitoring shows that dissolved oxygen at the deaerator inlet has typically been about 25 μg/kg. The deaerator removes the bulk of the dissolved oxygen and is able to maintain the level low enough that further addition of hydrazine is unnecessary, i.e. the dissolved-oxygen specification can be met.

The data for sodium (Fig. 4c) are only indicative. The sodium content was actually below the detection limit of the analyzer and the signal has been electronically enhanced in order to detect the trends of the sodium values in the practical range of the instrument. From measurements in the steam-generator blowdown, it appears that the real sodium content at the outlet of the condensate extraction pumps would be < 0.10 μg/kg. The few readings that largely exceed the specified maximum of 0.25 μg/kg usually coincide with reactor startups and power transients.

The mean value for iron has been 7.2 μg/kg over the last 12 months (Fig. 5a). This period coincides with a change in the analytical procedure used for iron and copper. Since the beginning of February 1990, these species have been determined from the insoluble matter filtered on a 0.45-μm Millipore filter rather than directly in water samples. This change was made following a mass-balance study showing that the amounts of iron and copper in water samples are negligible compared to their concentrations in crud [2]. The high values observed occasionally before February 1990 may have resulted from the accidental presence of insoluble matter in the water samples. Caution must therefore be exercised in any interpretation of the data.

The presence of admiralty brass (77% Cu/22% Zn) makes the monitoring of copper extremely important. The results over the years show that the concentration at the HP feedwater heater outlets (Fig. 5b) has been consistently below the 10 μg/kg maximum. The average value over the last year was 0.08 μg/kg. Higher values over the earlier years can be attributed to a less sensitive method for analysis. The present detection limit is 0.01 μg per kg of water.

The presence of insoluble matter in the feedwater system must be monitored very closely because these substances can be carried by the feedwater into the steam generators, where their accumulation would reduce the heat exchange efficiency of the tubes and promote corrosion phenomena. Results show that the crud level in the feedwater has dropped over the years (Fig. 5c). The content measured at the HP feedwater heater outlets averaged 9.3 μg/kg during the last 12 months' operation, including reactor startups and power transients. Typical steady-state full-power operation values range from 2 to 6 μg/kg, which is considered very acceptable.

3.2 Steam Generator Chemistry

Chemistry control in the steam generators is essential to avoid concentration of impurities under the sludge and deposits accumulated on the tubesheet, support plates and heat exchanger tubes. Inadequate control could easily accelerate the localized corrosion processes and eventually result in tube failures.

Figures 6 to 8 show the variations measured in the steam generator blowdown between April 1984 and May 1991. The pH, conductivity, dissolved oxygen, sodium and chloride were
measured daily and the iron, copper, silica and crud were measured weekly. These graphs include all the raw data taken over the period.

The steam generator pH has always been within the specified range of 9 to 10, with only a few readings apparently below 9 (Fig. 6a). An average value of 9.22 (± 0.14 standard deviation), slightly lower than in the feedwater, is noted over the full seven years' operation. Field measurements have shown that the morpholine distribution between the water and steam phases in the steam generators at Gentilly 2 is characterized by a relative volatility of approximately 1.15 [3, 4]. This would explain the slight difference in the pH between the blowdown and the feedwater (9.22 vs. 9.35).

The dissolved oxygen in the steam-generator blowdown has generally remained low and well below the specification throughout the period of interest with only a few readings apparently above 5 µg/kg (Fig. 6b). The average for the April 1984–May 1991 period is close to 1.1 µg/kg, with a slight upward trend noticeable in the third quarter of 1990. This trend was also observed at the HP feedwater heater outlets (Fig. 4b). The introduction of lower-temperature water, i.e. containing an higher level of oxygen, into the system at one of the recovery tanks appeared to be the cause; the dissolved oxygen was successfully reduced to about 0.2–0.5 µg/kg after the appropriate repairs.

The specific conductivity varied, but mostly within the specified range (Fig. 6c). An average of 0.43 mS/m (± 0.12 standard deviation) was calculated for the period of April 1984 to May 1991.

The sodium concentration remained well below the specified range, with only a few readings above 70 µg/kg (Fig. 7a). An average of 10 µg/kg was calculated, with typical values ranging from 2 to 5 µg/kg.

Some iron and copper transport was seen but normally the values were very low (Figs. 7b and 7c). The average value measured over the last 12 months' operation is about 180 µg/kg and 3 µg/kg respectively for iron and copper.

Chlorides were higher over the period April 1984 to October 1987 (Fig. 8a). In October 1987, modifications were made to the chlorination system at the water treatment plant to minimize the formation of halogenated organic compounds, and hence reduce the chloride content in the steam generators. Laboratory studies have shown that, once compounds such as trihalomethanes (chloroform, bromodichloromethane, dibromochloromethane and bromoform) have entered the cycle via the makeup water, they can decompose to give chloride and bromide ions [5]. In the past, chloroform concentrations of up to 85 µg/kg had been observed in the feedwater. The average for chloride during the last 12 months' operation is close to 20 µg/kg, which easily meets the < 100 µg/kg requirement.

The silica content remained within the specified range of < 1000 µg/kg (Fig. 8b). The average value measured over the entire period of operation was close to 60 µg/kg.

The insoluble matter measured in the steam generator blowdown was in the normal range, with a mean value near 290 µg/kg, which is well below 1000 µg/kg (Fig. 8c).

### 3.3 Reactor-Outlet-Header Temperature Measurements

Fouling in the secondary side of steam generator tubes would raise the temperature of the water in the primary heat transport system. The variations of the reactor inlet header (RIH) temperature over the years should therefore reflect the cleanliness of the steam generators. The
RIH temperature at Gentilly 2, as at other CANDU-600 plants, has shown an upward trend since initial startup, as seen in Figure 9.

The temperature increases appear to be related to outages. The data show a steady temperature within a limited range during reactor operation but an apparent increase occurs each outage. The new temperature level remains essentially stable until the next outage, when it again increases. This trend has now reached the point where the RIH temperature is causing restrictions in the operation of all CANDU-600 plants during on-power refueling of some of the fuel channels. It is interesting to note that the RIH temperature rise at Gentilly 2 and at Point Lepreau nuclear power plant, another CANDU-600 plant, has the same slope despite the fact that the latter uses a different chemical conditioning (hydrazine, phosphates and morpholine). Gentilly 2 operating personnel estimate that, at the current rate of rise, in less than four years nearly all channels will require derating of 2% to 10% full power in order to refuel the reactor.

All necessary efforts are being made at Gentilly 2 to determine the exact cause of the problem and, most important, to prevent the temperature rise at the reactor inlet headers. Among other possible mitigative actions, the operating staff intends first to limit the amount of crud entering the steam generators during normal operation by tightening chemistry control. The second measure will be to limit the amount of crud entering the steam generators during startups and power transients. This is most critical after extended outages when there has been maintenance work on the feedwater or steam systems.

4.0 System Inspections

Although Gentilly 2 has experienced no steam generator tube failures and the system chemistry control indicate that all the parameters are within acceptable limits, the operators can only determine whether a particular chemistry regime is functioning by inspecting the components. The first inspection, during the 1985 annual outage, was of the deaerator storage tank. The tank has a large drain pipe which has a 20-cm ridge running along the bottom. This arrangement allows the storage tank to act as a large settling tank for insoluble matter. During the inspection, only a few kilograms of such matter, the amount typically seen in a CANDU plant operating with hydrazine addition, were found and removed. During a subsequent outage, eddy-current testing of a large number of steam-generator tubes was performed but gave no indication of tube pitting or stress-corrosion cracking.

During the 1990 annual outage, an inspection of the tubesheet, selected rows of tubes and the first support plate of two of the four steam generators in operation was carried out. This inspection indicated that the tubes were relatively clean in the vicinity of the sludge pile up to the first support plate and its broached holes. The deposits on the tubesheet were measured using a fibre-optic support to guide a semi-rigid metal wire of calibrated length which was inserted between two rows of tubes, then lowered to the tubesheet or the deposits. Figure 10 shows the profile of the sludge pile measured at row 10 of the tube bundle. From the profile of the deposits measured in the steam generators at other CANDU plants, this row should correspond to the place where most deposits may be expected. The maximum height of the tubesheet sludge pile was about 10 cm, consisting of a hard base covered with a relatively soft top layer. A water-jetting trial proved that the sludge can easily be removed, and Hydro-Québec is planning full-scale cleaning in the near future.
5.0 Examination of Steam-Generator Tube and Sludge Samples

During an outage in 1987, samples of a steam-generator tube and tubesheet sludge were taken from one of the four steam generator units in operation. Visual inspection showed a thin black deposit on the tube, which can be removed easily by gentle scraping. The section of tube under the tubesheet sludge had a thicker, more adherent coating. Visual inspection could not detect any evidence of corrosion or tube damage. Metallurgical examinations of the tube showed no pitting nor any wall thinning. Energy Dispersive X-ray (EDX) examination showed that the deposit was largely magnetite.

The sludge sample consisted mainly of black powder with small chunks. Scanning electron microscope and EDX analysis revealed that the major components were iron 80.3 wt% and copper 10.3 wt%. Water leaching of the sludge sample produced a slightly alkaline solution (pH 7.7) with a chloride content of 0.4 mg/kg, which indicates that the chemical environment in the tubesheet sludge is not likely to cause significant corrosion problems. The total absence of corrosion on the tube samples corroborates the above conclusion.

Chemical-cleaning tests based on the generic process developed by the EPRI Steam Generator Owners' Group [6] were performed both on tubesheet sludge and on sections of the tube sample. This process comprises two steps: iron removal followed by copper removal. The efficiency for dissolving tubesheet sludge in the powdered state was 90% for two sequences of the iron and copper removal steps. The chemical-cleaning tests on the tube samples indicate that the EPRI process is very efficient for cleaning up tube deposits. As with sludge dissolution, almost all the iron and copper were removed in the first removal step. In fact, more than 50% of the iron and copper came off during the first hour and the additional amounts that were dissolved after four hours of the first iron removal step were negligible. The deposit thickness estimated on the basis of the iron content is of the order of 1 μm. There was no sign of corrosion attack by the solutions on the tube samples.

6.0 Corrosion Product Transport Survey

The extent of water-side corrosion during normal operation was evaluated by Babcock & Wilcox (Research and Development Alliance Division, Ohio) in April 1989 by sampling and measuring the level of corrosion products at key locations in the cycle [2]. At each sample point, particulate and colloidal forms of corrosion products were collected by passing the sample stream through a 0.45-μm Millipore filter. The filtrate was then passed through a stack of three cation resin-impregnated membranes to collect dissolved cationic forms of corrosion products. Samples collected on filters and resin membranes over the five-day test period were analyzed for iron, chromium, manganese, nickel, copper, zinc and lead by x-ray fluorescence spectroscopy. Table III shows a distribution vs. sample location giving average concentrations throughout the test period.

3.2-8/25
Table III: Average corrosion product concentrations at different locations of the Gentilly 2 secondary cycle

<table>
<thead>
<tr>
<th>Sample location</th>
<th>Corrosion product concentration µg/kg</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Iron</td>
</tr>
<tr>
<td>Moisture separator drain</td>
<td>3.16</td>
</tr>
<tr>
<td>Reheater drain</td>
<td>4.00</td>
</tr>
<tr>
<td>HP heater outlets</td>
<td>0.93</td>
</tr>
<tr>
<td>Steam generator feedpumps</td>
<td>0.46</td>
</tr>
<tr>
<td>Deaerator inlet</td>
<td>0.37</td>
</tr>
<tr>
<td>HP heater drain</td>
<td>3.61</td>
</tr>
<tr>
<td>Condensate pump discharge</td>
<td>0.37</td>
</tr>
<tr>
<td>Steam generator #4 blowdown</td>
<td>149.60</td>
</tr>
<tr>
<td>Steam generator #2 blowdown</td>
<td>115.71</td>
</tr>
</tbody>
</table>

Most of the iron present in the system is in filterable form. Apart from the blowdown streams which, as expected, proved to be the most concentrated, the next-highest iron concentrations were found in the three different drain lines from the moisture separator, the reheater and the HP heater. The lowest concentrations of iron were found in the condensate pump discharge and, at the other end of the LP feedwater heaters, at the deaerator inlet. Nickel followed the same pattern as iron, with the highest concentration occurring in the blowdown streams and the lowest in the condensate pump discharge. Intermediate values were found in the drain samples, where nickel concentrations showed the same variations as iron. Copper and zinc generally followed the pattern described for iron and nickel. The zinc concentration in the reheater drain was almost three times higher than the copper concentration, although both of these elements had a very low concentration at the outlet of the condenser. Negligible levels of lead, chromium and manganese were present at the HP feedwater heater outlets. All the above-mentioned elements were detected in the steam generator blowdown samples.

The mass balance of these corrosion products is shown in Table IV for some components. Totals were obtained by multiplying the average elemental concentrations (data of Table III) by the total flow rate at a particular sample location and assuming 100% full power 24 h/day, 365 days/yr. The net values obtained can be viewed as a measure of the accumulation or generation of corrosion products within a specific component.
Table IV: Elemental mass balance for some individual components

<table>
<thead>
<tr>
<th>Component</th>
<th>Iron</th>
<th>Nickel kg/yr</th>
<th>Copper</th>
<th>Zinc</th>
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<tr>
<td>LP feedwater heaters (1, 2, 3)</td>
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<td></td>
<td></td>
</tr>
<tr>
<td>in</td>
<td>9.33</td>
<td>0.76</td>
<td>0.76</td>
<td>1.26</td>
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<tr>
<td>out</td>
<td>9.33</td>
<td>2.02</td>
<td>1.51</td>
<td>0.76</td>
</tr>
<tr>
<td>net</td>
<td>0</td>
<td>-1.26</td>
<td>-0.75</td>
<td>+0.50</td>
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<tr>
<td>Deaerator</td>
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<td>in</td>
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<td>2.90</td>
<td>1.73</td>
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<td>out</td>
<td>14.87</td>
<td>2.59</td>
<td>1.62</td>
<td>0.32</td>
</tr>
<tr>
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<td>+5.84</td>
<td>+0.31</td>
<td>+0.11</td>
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<tr>
<td>HP feedwater heater</td>
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<td></td>
<td></td>
</tr>
<tr>
<td>in</td>
<td>14.87</td>
<td>2.59</td>
<td>1.62</td>
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<td>out</td>
<td>30.00</td>
<td>3.70</td>
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<td>net</td>
<td>-15.2</td>
<td>-1.19</td>
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<td>Each steam generator</td>
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<td></td>
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<tr>
<td>in</td>
<td>7.53</td>
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<td>0.48</td>
<td>0.41</td>
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<tr>
<td>out</td>
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<tr>
<td>net</td>
<td>+3.31</td>
<td>+0.79</td>
<td>+0.41</td>
<td>+0.17</td>
</tr>
</tbody>
</table>

It can be seen that neither accumulation nor generation occurred to any significant extent in the LP feedwater heaters but a net generation of corrosion products was evident in the HP feedwater heaters. The deaerator appears to be an efficient receptacle for crud. Lastly, the relatively low rate of crud accumulation in the steam generators testifies to the effectiveness of the simplified chemistry regime under which this plant operates. The estimated average sludge accumulation in all four steam generators, based on these 5-day results, is about 25 kg total metal oxides/yr. An estimate by the Gentilly 2 staff based on corrosion transport data gathered during seven years' operation is 50 to 100 kg/yr.

7.0 Laboratory Study

To further support the decision to implement the modified AVT program, Hydro-Québec established a research program in 1987 to study the thermal stability of morpholine. The program identified the decomposition products and measured the kinetics of their formation at temperatures and pressures within the CANDU secondary cycle. The laboratory study was compared with actual measurements made within the plant system.

The laboratory tests performed using an initial morpholine concentration of 150 mg/kg revealed that the disappearance of morpholine follows first-order kinetics with an activation energy of 131.9 kJ/mole and decomposition rate constants of 2.67, 8.73 and $21.25 \times 10^{-2}$ s$^{-1}$ at 260, 280 and 300°C [7]. The proposed reaction scheme is presented in Figure 11. The thermal decomposition of morpholine is a hydrolytic process following a C-N bond scission to give
2-(2-aminoethoxy) ethanol. The latter decomposes to give volatile amines (ethanolamine, ammonia, methylamine and ethylamine) together with ethylene glycol which then oxidizes, resulting in the formation of glycolic and formic acids. Hydrolysis of the ethylamine produces ethanol, while subsequent oxidation yields acetic acid.

Figure 12 presents the distribution of the morpholine decomposition products in the cycle [4]. The organic acids in the secondary cycle appear mainly in the steam-generator blowdown and the moisture-separator reheater drains. The concentrations observed suggest that these acids preferentially recirculate in a loop between the steam generator, the moisture-separator reheater drains, the deaerator and the HP feedwater heaters, largely bypassing the turbines and condensers. A mass flow rate/mass balance calculation shows that slightly more than half the acetate content of the main steam is directed towards the moisture-separator reheater drains. The results also reveal a fairly uniform distribution of morpholine in the different steam-condensate phases of the complete cycle, in contrast to ammonia, which tends to enrich the steam phase. While 2-(2-aminoethoxy) ethanol follows the same pattern as morpholine, ethanolamine collects mainly in the first condensate, which has a higher organic-acid content; methylamine has a similar distribution to that of ammonia. The preferential paths followed by these products, according to their relative volatility (RV), are illustrated in Figure 13.

Similar measurements at other CANDU plants using morpholine in combination with hydrazine show that, apart from the case of a plant equipped with a full-flow condensate polisher, these decomposition products are present in approximately the same amounts.

8.0 Conclusion

Certain precautions must be exercised in the interpretation of some of the data used in this paper. It is not restricted to steady-state operation; it also includes measurements taken during transient conditions (e.g. in the period following startup, before the system reaches its equilibrium state). The graphs used to display the data are scaled to include all values; this tends to put more weight upon the deviations than on the steady-state operation. Over a period of seven years, analytical methods have been improved. Nevertheless, all the operating data collected for the steam generators confirm, as in the case of the feedwater system control, that it is possible to meet the specifications with AVT treatment based on morpholine injections alone, without the addition of hydrazine. Since operation began, there has been no leakage from the steam generator tubes and no indication of pitting or stress-corrosion cracking. Experience indicates that a plant which has an all-ferrous feedwater system can be safely operated without hydrazine additions, although this does not imply that addition of hydrazine is entirely without benefit.

Hydro-Québec will continue to monitor plant performance at Gentilly 2 and will make every effort to determine and eliminate the cause for the increase in the reactor inlet header temperature. In the very near future, the utility plans a full-scale water-jetting operation to remove the sludge accumulated on tubesheet of the four steam generators in operation. If the trend continues, forcing restrictions of the reactor power output, chemical cleaning may have to be considered for the steam generators.

Acknowledgments

The authors thank Denis Brissette, Richard Laporte, Marcel Bergeron and Souheil E. Saheb who have had an active role in the followup of the Gentilly 2 physicochemical system parameters since the beginning of the operation. They also thank Marvin D. Silbert for useful
comments on the manuscript and many fruitful discussions on the subject. Thanks go to Leslie Kelley-Régnier for her editorial assistance.

References


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Figure 2: Simplified flowsheet of the Gentilly 2 secondary cycle
Figure 3: Gentilly 2 steam generator
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Figure 9: Average reactor inlet header temperature vs. effective full-power days
Figure 10: Radial profile of tubesheet sludge pile
Figure 11: Proposed reaction scheme for the thermal decomposition of morpholine
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Figure 13: Preferential circulation of morpholine, nonvolatile (RV < 1) and volatile (RV > 1) breakdown products.
CHEMICAL CLEANING AS A MEASURE TO IMPROVE
STEAM GENERATOR PERFORMANCE OF PWR-PLANTS

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Abstract

The steam generators of PWR nuclear plants show diverse
degradation mechanisms, affecting their performance (tube
fouling) as well as their integrity (corrosion).
SG tube corrosion problems are a consequence of the
concentration of impurities up to aggressive levels caused
by the heat flux on the tube surface in combination with
deposited corrosion products, it takes place even at low
impurity concentration levels in the SG bulk water.
The chemical cleaning of the steam generators, secondary
side, has been recognized as a countermeasure against these
SG degradation phenomena (corrosion, fouling) by eliminating
the corrosion products and salts accumulated in the SGs.
The present work summarizes the state of the technique and
the progress made by Siemens in the development and
application of SG chemical cleaning, discussing the
capability and features of the method as remedial action
against SG degradation.
Introduction

About 80 per cent of all plant outages involving steam generators are caused by corrosion problems. Such problems arise from the continuous ingress of non-volatile contaminants, i.e. corrosion products and salt impurities. These have their origin in the balance of plant systems. The corrosion products generally accumulate in the steam generators - forming both deposits in the flow restricted areas (such as on the tube sheet and tube support plate), and scales on the steam generator heating tubes.

The deposits are an essential prerequisite for the concentration of non-volatile salt impurities, which are always present in the feedwater in the \( \mu g/kg \) (ppb) range. Depending on which type of impurities are concentrated and where they are located in the steam generator tubes - different corrosion mechanism of SG-tubes - such as stress corrosion cracking, intergranular attack, denting, pitting and wastage may occur. On the other hand, scales on the tube surface cause a continuous reduction of the heat transfer from primary to secondary side, which can ultimately cause a reduction of power output.

In order to prevent the above mentioned SG problems (i.e. SG corrosion and fouling), the industry developed various types of cleaning techniques - as tube sheet lancing, pressure pulse cleaning, crevice flushing/soaking and chemical cleaning. Among these, especially chemical cleaning has been found to be very effective for removing products and concentrated impurities.

This paper presents the state of the experience gained by Siemens/KWU in the development and application of SG chemical cleaning process with respect to cleaning of tube surface, tube crevice areas and removal of impurities.
KWU's Cleaning Process

The KWU chemical cleaning process consists of two different steps: a high temperature, high speed iron process; and a copper process at ambient temperature. The iron and copper process can be applied in combination if necessary, but they use different solvents.

Iron Removal Process

The Siemens/KWU iron removal process is applied at high temperature under strongly reducing and alkaline conditions. The reaction time including the time for the injection of the chemicals is about four hours, the mixing of the chemicals in the steam generator is performed by steam-off operations.

The iron solvents have been extensively tested within the temperature range of 60 °C to 270 °C, resulting in an optimum application temperature range of 140 to 200 °C. The feasible application temperatures are usually limited by the operating design temperature of the residual heat removal system of the plant. Within this limited temperature range, the plant specific application temperature will be determined by the dissolution behaviour of the actual steam generator deposits. This again is influenced by the chemical composition and morphology of the deposits. Furthermore, the solvents don't require any inhibitor. There are therefore no after effects due to residuals containing hazardous impurities. Inhibitors left in the SG by insufficient flushing would decompose thermally during startup and aggressive products might thus be formed with possible negative effects for the SG tubing.

Copper Removal Process

The copper solvent is an ammonia-based chemical that contains a volatile amine as a selective complexing agent. The solvent performs its dissolution function at low temperatures and alkaline conditions. The application temperature is less than 70 °C and can be easily controlled, since no exothermic reactions are involved. The chelating agent used forms extremely stable complexes only with Cu-II, therefore oxidising conditions are essential.

Oxidation of metallic copper in the sludge is accomplished by compressed air in the presence of an ammonia-based catalyst. This catalyst allows sufficient and rapid oxidation to be performed without using the more common oxidizers such as hydrogen peroxide. The oxidation of the
metallic copper (and thus the oxygen consumption) is so fast that the process requires permanent air injection. The air injection will also be used for a proper mixing of the solvent. Air overpressure and slightly higher solvent temperatures increase the copper dissolution rate considerably.

Experience on the Effectiveness of the Processes

Both cleaning processes (i.e. copper and iron oxide removal) have been qualified in numerous plant-specific tests to optimize the deposit removal capabilities considering a variety of different cleaning objectives. Since its first application, in 1984 and up to date, they have been applied many times in different PWR steam generators (see table 1).

The experience gained during these qualification works as well as field application are the following:

1 Reduction of the Corrosion Product Inventory

The amount of the corrosion products removed by Siemens/KWU chemical cleaning process varied between 400 kg and 1500 kg per SG. The results are presented in detail in table 1. This amount of removed corrosion product deposits was always significantly higher than that removed by tube sheet lancing. The visual inspections by fiber scope and the results of pulled tube analyses indicated in many cases, corroborating the qualification test results, the efficiency of the chemical cleaning under different aspects:

* Cleaning of Tube Surface Scales to counteract fouling

The features of the iron cleaning process, which is characterized by having heat transfer through the tube walls while the cleaning proceeds, enables very effective removal of tube deposits. Clean, shiny tube surfaces were found as a rule during visual inspections after chemical cleaning. In fact, a significant percentage of the total amount of iron oxide removed was observed to come from the tube surfaces therefore chemical cleaning is a suitable method against fouling. An improvement of the heat transfer capacity is not always observed because as known, relative thick oxide layers on tube surfaces (>100 μm according to experience) have a noticeable detrimental effect. After cleaning of steam generators having such thick deposits, an improvement of the fouling situation was observed (see figure 1).
* Hard Caked Sludge

In several plants with deposits of hard caked sludge on the tubesheet, which could not be removed by tubesheet lancing operations, the application of chemical cleaning was an adequate tool to remove them (removal of 80% of the sludge in one cleaning operation and complete removal by a second application is considered to be a typical figure). See fig. 2.

* Crevice Cleaning

Many SG's have suffered in recent years from denting or IGA/IGSCC tube degradation caused by concentrated, aggressive impurities in the tube to tube support plate crevices. The cleaning of the crevices, i.e. removal of such aggressive environments, is the most effective countermeasure against tube crevice corrosion. The effectiveness of crevice cleaning depends mainly on an accelerated mass transport in the crevice area. This means that the loaded solvent should be refreshed, which is achieved in the KWU process by maintaining a small heat flux through the SG tube walls at 160 to 170 °C, thus causing some boiling at the surface of the SG tubes. Crevice cleaning tests were performed to optimize the crevice cleaning conditions. These efforts resulted in complete cleaning of narrow crevices within a couple of hours (see figure 3).

The cleaning technique recently developed with an improved qualified solvent resulted in improvements especially in the area of crevice cleaning and removal of hard caked sludge. This technique uses a small continous boiling during the injection and the soakink step to improve replenishment of the spent solvent.

2 Reduction of Impurity Inventory

Significant amount of impurities (salts) were removed by chemical cleaning compares to all other impurity removal techniques available. Some typical results are shown in figure 4.

Conclusions

One of the observed characteristics of PWR plant aging is the gradual deterioration of steam generator tubes by various corrosion processes. Chemical cleaning of SG's one procedure that can be used to increase the life expectancy of the SG's and can ultimately lead to greater plant
availability. In this respect, KWU iron and copper solvents have been proved to be very effective cleaning agents. Beside the efficiency of the Siemens/KWU process for the removal of corrosion products including hard caked sludge and salt impurities, significant features of the process are its inherent safety and minimum influence on critical path time during the outage as well as an extensive field experience.
# Table 1

REFERENCE LIST OF KWU CHEMICAL CLEANING FOR PWR STEAM GENERATORS

<table>
<thead>
<tr>
<th>YEAR</th>
<th>PLANT</th>
<th>MATERIAL REMOVED</th>
<th>QUANTITY REMOVED (kg)</th>
<th>SG,MFGL</th>
<th>COMBE OPER.</th>
<th>KWU</th>
</tr>
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<td>BIBLIS-B</td>
<td>Cu</td>
<td>0.6</td>
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<td>NECKARWESTHEIM</td>
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<td>998</td>
<td>923</td>
<td>W</td>
</tr>
<tr>
<td>1990</td>
<td>MIHAMA-1</td>
<td>Cu</td>
<td>25</td>
<td>24</td>
<td>35</td>
<td>W</td>
</tr>
<tr>
<td>1990</td>
<td>ALMARAZ-1</td>
<td>Fe3O4</td>
<td>195</td>
<td>202</td>
<td>-</td>
<td>CE</td>
</tr>
<tr>
<td>1990</td>
<td>OH-1</td>
<td>Cu</td>
<td>19.7</td>
<td>20.5</td>
<td>-</td>
<td>CE</td>
</tr>
<tr>
<td>1991</td>
<td>PAK-1</td>
<td>Fe3O4</td>
<td>1427</td>
<td>1450</td>
<td>1474</td>
<td>1447</td>
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<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>VVER</td>
</tr>
</tbody>
</table>

Legend:

- C: Coarse (Bar)
- CE: Combustion Engineering (US)
- KWU: Kraftwerk Union (FRG)
- W: Westinghouse (US)
- VVER: (SU)

3.3-7/11
Figure 1  NPP Stade – Fouling Measurements

Average Fouling Factors Measured in All SG's

Chemical cleaning
SG 2 and 3

* Fouling Measurements (average)
- Linear interpolation of measured values

Operation time (a)
Figure 2

Results of the First Chemical Cleaning at Plant R - Removal of Hard Caked Sludge

Visual Examination (schematic)

Before

Afterwards
Figure 3

Crevice cleaning process
chelate conc.: 6%
temperature: 175°C

Cleaning Efficiency

<table>
<thead>
<tr>
<th>Design gap size</th>
<th>2 hrs</th>
<th>3 hrs</th>
<th>4 hrs</th>
<th>6 hrs</th>
</tr>
</thead>
<tbody>
<tr>
<td>160 µm centric</td>
<td>---</td>
<td>---</td>
<td>66 %</td>
<td>100 %</td>
</tr>
<tr>
<td>160 µm excentric</td>
<td>---</td>
<td>---</td>
<td>100 %</td>
<td>---</td>
</tr>
<tr>
<td>220 µm centric</td>
<td>100 %</td>
<td>---</td>
<td>86 %</td>
<td>---</td>
</tr>
<tr>
<td>220 µm excentric</td>
<td>---</td>
<td>---</td>
<td>100 %</td>
<td>---</td>
</tr>
<tr>
<td>320 µm centric</td>
<td>100 %</td>
<td>---</td>
<td>---</td>
<td>100 %</td>
</tr>
</tbody>
</table>
OPERATING EXPERIENCE WITH STEAM GENERATOR WATER CHEMISTRY IN JAPANESE PWR PLANTS

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Mitsubishi Heavy Industries, Ltd.
Takasago R&D Center
Araicho, Takasago, Hyogo Pref.

ABSTRACT

Since the first PWR plant in Japan started its commercial operation in 1970, seventeen plants are operating as of the end of 1990. First three units initially applied phosphate treatment as secondary water chemistry control and then changed to all volatile treatment (AVT) due to phosphate induced wastage of steam generator tubing. The other fourteen units operate exclusively under AVT. In Japan, several corrosion phenomena of steam generator tubing, resulted from secondary water chemistry, have been experienced, but occurrence of those phenomena has decreased by means of improvement on impurity management, boric acid treatment and high hydrazine operation. Recently secondary water chemistry in Japanese plants are well maintained in every stage of operation.

This paper introduces brief summary of the present status of steam generators and secondary water chemistry in Japan and ongoing activities of investigation for future improvement of reliability of steam generator.
1. INTRODUCTION

Since the first PWR plant in Japan started its commercial operation in 1970, seventeen plants are operating as of the end of 1990. In the beginning phosphate treatment was applied as secondary water chemistry control, but it was replaced by all volatile treatment (AVT) due to phosphate induced thinning of Alloy 600 tubing of steam generator. After change of chemistry to AVT, a great deal of effort has been focused on the improvement of secondary water chemistry control and equipments. On the other hand, different corrosion problem resulting from secondary water chemistry has been experienced so far and it is called intergranular attack (IGA). In order to reveal causes and to establish remedies for IGA, much study were done and secondary water chemistry control has been improved further, and then occurrence of corrosion has been decreasing.\textsuperscript{1,2}

This paper describes briefly the history and present status of secondary water chemistry in Japanese PWRs and summarizes countermeasures to cope with corrosion problems, and provides some topics and ongoing activities of investigation for future improvement of reliability of steam generator.

2. HISTORY OF CHEMISTRY CONTROL IN JAPANESE PWRs

As mentioned above, there are seventeen plants running in Japan. The first unit is equipped with two Combustion Engineering's steam generators and the other sixteen have Westinghouse-type steam generators. Initially phosphate treatment was applied but concentrated phosphate salt in the crevices between tube and tube support plate caused thinning corrosion of Alloy 600. Therefore water chemistry was changed from phosphate treatment to AVT around 1975. After changeover secondary water chemistry had been maintained well and there had been no corrosion problems in the secondary side except for few occurrences of stress corrosion cracking in deep crevices due to free caustic generated from residual sodium phosphate.

![Figure 1 Water Chemistry Control in Japanese PWRs](image-url)
In the meantime a new type of corrosion problem, which is called intergranular attack (IGA), began to be observed in spite of the improved chemistry. From the results of laboratory studies and field surveys, it was concluded that free caustic and oxidizing species make Alloy 600 very susceptible against IGA. As a remedial actions for IGA, an increase of feedwater hydrazine concentration (high NH₂ operation) and boric acid treatment (BAT) are being applied to not only affected units but also some unaffected units as a preventive measure. BAT started at four units in 1985 and now seven units are applying BAT. By means of these remedies occurrence of corrosion problems is decreasing. Brief summary on history of water chemistry control in Japanese PWRs is shown in Figure 1.

3. SECONDARY WATER CHEMISTRY

3.1 Secondary System and Chemistry Control

3.1.1 Secondary System

Secondary system for PWR plant is essentially equal to that of conventional fossil fuel plant. In old units copper alloy are used not only for feedwater heater tubing but also condenser tubing and recent six units use titanium for condenser tubing. Furthermore, in one of recent units, stainless steel is used as tubing material for all feedwater heaters. Steam generator tubing is Alloy 600 and 690.

Improvement in secondary water chemistry is virtually achieved through the integrity of secondary system equipment and the following items shown in Figure 2 are considered to be essential.

- Leak tight condenser
- Condensate polisher
- Deaerator
- Clean-up facilities (clean-up line, SG blowdown recovery line and drain recovery line to condenser)

![Diagram of secondary system](image)

Figure 2 Facilities to Maintain Secondary Water Chemistry Integrity

Because all Japanese units are faced to sea and use seawater for condenser cooling, “Leak-tight” is very important for the prevention of impurity ingress. The most effective method of removing impurities from the system is installation and extensive usage of condensate polishing system. Table 1 shows design characteristics of condensate polisher in Japan. All PWR units are equipped with condensate polisher and thirteen of them are full flow.
type and the other four are partial flow type (10% of condensate). Basically resins are operated in hydrogen-hydroxide cycle and fixed throughput is between 120000–220000m³ which correspond to 10–30 days per cycle. The most important point to get high purity in polisher effluent is resin regeneration and for this purpose the improved regeneration system, which uses three towers for resin separation and regeneration, is applied. The third tower is used for storage of anion/cation interface removed to prevent cross contamination. Typical effluent level for sodium and chloride is less than 0.01ppb and conductivity is 0.06μS/cm.

Clean-up after long outage is important from the standpoint of removal of foreign materials and impurities. This clean-up is devided into two steps. One is whole system recirculation clean-up by use of condensate polisher just before start. The other is blowdown of initial heater drain, which contains much impurity, and successive recovery of drain to condenser. According to this procedure, residual impurities in the steam and drain line can be drained out and cleaned up effectively.

Steam generator blowdown recovery system can make it possible to increase blowdown rate as much as required without excessive consumption of make-up water. And this increase of blowdown rate lowers impurity level in the steam generator blowdown effectively.

Gaseous species such as oxygen are almost completely exhausted from a deaerator and the deaerator is important to prevent oxygen and oxidizing species from entering into steam generator.

<table>
<thead>
<tr>
<th>Table I</th>
<th>Design of Condensate Polisher</th>
</tr>
</thead>
<tbody>
<tr>
<td>Item</td>
<td>Characteristics</td>
</tr>
<tr>
<td>Type</td>
<td>Full flow deep bed (13 units)</td>
</tr>
<tr>
<td></td>
<td>Partial flow deep bed (4 units)</td>
</tr>
<tr>
<td>Operation</td>
<td>H-OH cycle</td>
</tr>
<tr>
<td>Fixed throughput</td>
<td>12,0000 ~ 22,0000 m³ (10 ~ 30 days)</td>
</tr>
<tr>
<td>Regenerant</td>
<td>HCl - NaOH</td>
</tr>
<tr>
<td>Regeneration</td>
<td>3 tower ( Anion / cation interface removal system )</td>
</tr>
<tr>
<td>Effluent quality (typical)</td>
<td>Conductivity : 0.06 μS/cm</td>
</tr>
<tr>
<td></td>
<td>Na : 0.01 ppb</td>
</tr>
<tr>
<td></td>
<td>Cl : 0.03 ppb</td>
</tr>
</tbody>
</table>

3.1.2 Secondary Water Chemistry Control

Final object of secondary water chemistry control is to prevent corrosion of secondary system equipment, especially steam generator and turbine. The basis of water chemistry control is summarized as follows.

(1) Prevention of impurity ingress into the system is achieved by the integrated secondary system equipments as described in 3.1.1.

(2) Removal of impurity introduced is done by the condensate polisher and extensive usage of it combined with clean-up line and blowdown recovery system.

(3) In order to prevent corrosion of system component, feedwater pH is controlled around 9.2 and hydrazine concentration in feedwater was kept around 0.2ppm. This hydrazine concentration was aimed at suppression of iron pick-up in the secondary system, especially in the moisture separator drain.[1]

In addition to the above chemistry control, two other remedial actions are applied. As mentioned in 2, causes of IGA are concluded to be free caustic and oxidizing environment. As a result of laboratory study and field survey, BAT was developed to moderate high pH environment and high NH₄⁺ operation to increase reducing environment.[3][4] Now in Japan seven units apply BAT which comprises high boric acid soaking (50–500ppm) during startup and successive continuous injection (5–10ppm) under normal power operation.

In order to increase reducing environment, 0.5–0.6ppm of hydrazine is maintained at feedwater in all units with full flow polisher and around 0.1ppm of hydrazine in units with partial flow polisher. High NH₄⁺ operation is applied for not only IGA affected units but also unaffected units as a preventive measure.
3.2 Secondary Water Chemistry Experiences

3.2.1 Causes of IGA

Previously the integrity of secondary system equipments was poor from the viewpoint of corrosion prevention. There were no condensate polishers nor sufficient clean-up systems. Then many kinds of impurities, especially welding slug residue which contains plenty of free caustic, introduced and left in the secondary system during construction had been ingressed into steam generator in the period of trial operation. Furthermore, after long outage and repair works, foreign materials and impurities could not be cleaned well without good facilities for chemistry control.

According to water chemistry records for affected units, a great deal of caustic seemed to be introduced into steam generator during trial operation and startup after initial inspection. This situation was common for all old units without condensate polisher but in those units which applied phosphate treatment, free caustic ingressed into steam generator had been neutralized by phosphate unexpectedly. Also, there were few units where sea water leakage had been experienced and free caustic in the crevice might be neutralized. These are considered to be the reasons why IGA occurrence were limited to some specific units.

3.2.2 Secondary Water Chemistry in Recent Days

In order to minimize the possibility of corrosion, much effort has been concentrated on improving secondary water chemistry. As a result of these efforts, secondary water chemistry has been improved very much and no significant chemistry excursion is observed in every stage of operation including startup and shutdown in these days.

Figure 3 shows an example of steam generator blowdown chemistry during startup. There is a little difference in behavior between unit with full flow condensate polisher (FFCP) and with partial flow condensate polisher (PFCP). Impurity excursions in the unit with PFCP is a little bit higher than those with FFCP. However, maximum (peak) values during startup even in the unit with PFCP are rather small and any impurity excursion can hardly be seen in the unit with FFCP. This figure shows that there are no significant impurity ingress during startup.

Typical steam generator blowdown chemistry is summarized in Table II which contains cycle averages for each chemistry parameter in recent cycles. Sodium level is less than 1ppb and chloride level is several ppb. Cation conductivity is 0.1μS/cm and 0.3μS/cm even in the unit under BAT. From this Table it is concluded that impurity level under normal power operation is low enough in recent cycles.

Figure 3  Steam Generator Blowdown Chemistry during Startup
Table II  Steam Generator Water Chemistry under Normal Operation

<table>
<thead>
<tr>
<th>Plant Cycle</th>
<th>a</th>
<th>b</th>
<th>c</th>
<th>d</th>
<th>e</th>
</tr>
</thead>
<tbody>
<tr>
<td>CW K, S (μS/cm)</td>
<td>0.1</td>
<td>0.1</td>
<td>0.1</td>
<td>0.1</td>
<td>0.1</td>
</tr>
<tr>
<td>DO (ppb)</td>
<td>&lt;5</td>
<td>&lt;5</td>
<td>&lt;5</td>
<td>&lt;5</td>
<td>&lt;5</td>
</tr>
<tr>
<td>pH (pH)</td>
<td>9.2</td>
<td>9.1</td>
<td>9.1</td>
<td>8.8</td>
<td>8.7</td>
</tr>
<tr>
<td>FW μS (μS/cm)</td>
<td>3.9</td>
<td>3.8</td>
<td>3.7</td>
<td>3.4</td>
<td>3.9</td>
</tr>
<tr>
<td>N₂H₄ (ppm)</td>
<td>0.01</td>
<td>0.09</td>
<td>0.13</td>
<td>0.09</td>
<td>0.12</td>
</tr>
<tr>
<td>Fe (ppb)</td>
<td>14</td>
<td>12</td>
<td>12</td>
<td>15</td>
<td>14</td>
</tr>
<tr>
<td>Cu (ppb)</td>
<td>&lt;2</td>
<td>&lt;2</td>
<td>&lt;2</td>
<td>&lt;2</td>
<td>&lt;2</td>
</tr>
<tr>
<td>SG K, S (μS/cm)</td>
<td>0.2</td>
<td>0.2</td>
<td>0.2</td>
<td>0.3</td>
<td>0.3</td>
</tr>
<tr>
<td>Na (ppb)</td>
<td>0.8</td>
<td>0.6</td>
<td>0.4</td>
<td>0.3</td>
<td>0.3</td>
</tr>
<tr>
<td>Cl (ppb)</td>
<td>2.8</td>
<td>2.7</td>
<td>2.5</td>
<td>1.1</td>
<td>0.6</td>
</tr>
</tbody>
</table>

(Not) K, S : Cation Conductivity, μS : Specific Conductivity

Impurities accumulated in the crevices during normal power operation are released to bulk water again during plant shutdown and this phenomenon is called hideout return (HOR). So the result of water chemistry control for the cycle is considered to be reflected to the degree of HOR. Figure 4 shows examples of HOR data. Maximum concentrations of return species are less than fifty ppb in any case and less than 10 ppb for sodium, potassium and chloride. From these results it could be said that impurity accumulation under power operation is small especially for soluble species such as sodium, potassium and chloride.

![Figure 4 Steam Generator Blowdown Chemistry during Shutdown](image-url)
3.2.3 Crevice Environment

Since HOR behavior is the indication of crevice chemistry, extensive survey on HOR at an operating plant is thought to provide much information about chemical environment in crevice and some results on HOR survey in Japanese PWRs have already been reported.\(^5\) As reported in ref.[5], crevice chemistry can be evaluated and pH of crevice simulated solution can be calculated by using HOR data.

Figure 5 shows results of pH calculation for crevice simulated solution evaluated from recent HOR data on several operating plants. Results range from 4 to 8 at 280°C and are thought to belong to neutral region because neutral point at 280°C is around 5.5. It was reported that IGA happens in the environment of pH above 10 under operating temperature from laboratory study.\(^5\) The estimated pH for present Japanese PWRs are much lower than 10 and crevice environment are considered to be rather well.

In the course of studying HOR behavior, it was found that there are differences with regard to apparent carry-over ratio for several ions.\(^5\) Since this may affect concentrating process and composition of crevice solution, distribution behaviors should be studied in more detail to properly evaluate crevice chemistry.

4. FUTURE PROGRAM

As described in 3.2, bulk water chemistry control is considered to be almost mature. Next steps to develop from now are summarized as follows:

- Crevice environment monitoring technique
- Water chemistry data management system with AI based diagnosis system

4.1 Crevice Environment Monitoring

Evaluation of crevice chemistry by using HOR data is useful but is just judgement from the results. Real-time monitoring for crevice chemistry is desired and essential to perform reliable chemistry control and to properly cope with abnormal chemistry. Two ways are conceivable. One is to calculate crevice concentration and chemistry equilibrium from data on bulk water chemistry parameters. From mass balance model through a crevice, an equation which describes concentration process is given and the combination of this model with high temperature chemistry equilibrium yields crevice chemistry environment. In this model, distribution coefficient of each species in the crevice solution is considered to play important role in determining the final concentration ratio. Detailed study on this technique, which is proceeding, is presented in different paper in this meeting.

The other way is to monitor crevice chemistry in the model boiler by using crevice pH electrode.\(^6\) The crevice pH monitoring in the model boiler, which is provided with steam generator blowdown water from an operating unit, shows that there was not significant excursion in crevice pH during start-up. \(^6\)
Recently a new technique has been developed. The crevice solution of model boiler is directly sampled under normal operating condition. Collected sample is analysed and the results obtained is used for evaluation of chemistry environment in the crevice. Design of the sampling device and some results are shown respectively in Figure 6 and 7[6]. This technique is thought to become powerful tool to monitor crevice chemistry though it might need further improvement.

![Sampling Device Diagram](image)

**Figure 6** Sampling Device

![Correlation Diagram](image)

**Figure 7** Correlation between Crevice Chemistry and Bulk Water Chemistry

### 4.2 Water Chemistry Data Management System

Water chemistry data management system with AI based diagnosis system has already been developed and effectiveness and reliability were established by field test and off-line test.[7] Figure 8 shows outline of the system which consists of monitoring, data acquisition and diagnosis. By combination of this data management system with crevice chemistry monitoring technique, further improvement in water chemistry control will be achieved.
5. SUMMARY

History and present status of secondary water chemistry in Japanese PWRs were introduced. In order to get improved water chemistry, the integrity of secondary system equipments is essential and the improvement in water chemistry has been achieved with the improvement in equipments and their usage. As a result of those efforts, present status of secondary water is excellent. However, further development for crevice chemistry monitoring technique and an advanced water chemistry data management system is desired for the purpose of future improvement of reliability of steam generator.

REFERENCES

SESSION 4 - STRUCTURAL INTEGRITY AND LICENSING ISSUES

4.1 PWR STEAM GENERATOR TUBE AND TUBE SUPPORT PLATE PLUGGING CRITERIA

B. Cochet, Framatome, Paris La Défense, FRANCE
J. Engström, Vattenfall-Energisystem AB, Vallingby, SWEDEN
B. Flesch, EDF-SPT, Paris La Défense, FRANCE

4.2 LICENSING BASES FOR STEAM GENERATOR TUBE INTEGRITY AND RECENT OPERATING EXPERIENCE

B.D. Liaw, United States Nuclear Regulatory Commission, Washington D.C., USA
Emmet L. Murphy, United States Nuclear Regulatory Commission, Washington D.C., USA

4.3 TUBE PLUGGING CRITERIA FOR AXIAL AND CIRCUMFERENTIAL CRACKS IN THE TUBESHEET AREA

J. Van Vyve, Belgatom/Tractebel, Brussels, BELGIUM
P. Hernalsteen, Belgatom/Laborelec, Brussels, BELGIUM

4.4 SWEDISH EXPERIENCES OF A REVISED PLUGGING CRITERION FOR STEAM GENERATOR TUBES - A REGULATORY VIEW

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P. Bystedt, Swedish Nuclear Power Inspectorate, Stockholm, SWEDEN
G. Hedner, Swedish Nuclear Power Inspectorate, Stockholm, SWEDEN
L. Skänberg, Swedish Nuclear Power Inspectorate, Stockholm, SWEDEN

4.5 CSN REGULATORY VIEW ON STEAM GENERATOR TUBE DEGRADATION IN SPAIN

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C. Mendoza, Consejo de Seguridad Nuclear, Madrid, SPAIN
J.F. Casílles, Consejo de Seguridad Nuclear, Madrid, SPAIN

4.6 SAFETY SIGNIFICANCE OF STEAM GENERATOR TUBE DEGRADATION MECHANISMS

G. Roussel, AIB-Vinçotte Nuclear (AVN), Brussels, BELGIUM
P. Mignot, AIB-Vinçotte Nucléaire (AVN), Brussels, BELGIUM

4.7 STEAM GENERATOR TUBES RUPTURE PROBABILITY ESTIMATION-STUDY OF THE AXIALLY CRACKED TUBE CASE

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L. Cizelj, "Jozef Stefan" Institute, Ljubljana, Slovenia, YUGOSLAVIA
G. Roussel, AIB-Vinçotte Nucléaire (AVN), Brussels, BELGIUM
PWR STEAM GENERATOR TUBE AND TUBE SUPPORT PLATE PLUGGING CRITERIA

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ABSTRACT

The operating experience of steam generators shows that the tubes in alloy 600 are affected by stress corrosion cracking in the roll transition zone at the top of the tubesheet and at the tube support plate levels. A large experimental program, analyses, in service inspections and pulled tube examinations from various steam generators have demonstrated that axial cracks in the roll transition areas which are smaller than the critical crack dimensions and axial cracks confined within the support plates can become through-wall in a cycle without constituting a safety or reliability problem. The critical crack dimensions determined using plastic instability rupture criteria depend on the pressure and the temperature, the mechanical properties of tubes, the tube dimensions and the location of the cracks. Results of these studies can be used by the plant operator to define the limits of tube degradation beyond which tubes should be repaired or removed from service. Tubes with circumferential cracks should be plugged or repaired.

RESUME

L'expérience d'exploitation des générateurs de vapeur montre que les tubes en alliage 600 sont affectés par la corrosion sous contrainte en zone de transition de dudgeonnage et au niveau des plaques entéroisées. Un programme expérimental important, analyses, inspections en service et expertises de tubes extraits de différents générateurs de vapeur ont permis de démontrer que les fissures longitudinales de longueur inférieure à la longueur critique en zone de transition de dudgeonnage ainsi que les fissures longitudinales localisées au niveau des plaques entéroisées, peuvent devenir traversantes en un cycle sans constituer un problème de sûreté ou de fiabilité. Les dimensions critiques des fissures, déterminées à partir des critères de rupture par instabilité plastique, dépendent de la pression et de la température, des propriétés mécaniques des tubes, des dimensions des tubes et de la localisation des fissures. Les résultats de ces études peuvent être utilisés par l'exploitant pour définir les limites des dégradations des tubes au-delà desquelles ils doivent être bouchés ou réparés. Les tubes avec des fissures circonférentielles doivent être réparés ou bouchés.
1. INTRODUCTION

The operating experience of pressurized water reactors shows that the tubes in alloy 600 are affected by stress corrosion cracking caused by the reactor coolant (PWS/CC) in the roll transition zone. The possibility of secondary side intergranular attack (IGA) and intergranular stress corrosion cracking (IGSCC) initiated at the outer surface of the tubes has also been revealed in the roll transition zone under sludge piles and at the tube support plate levels. In the roll transition zone, multiple longitudinal part-through or through-wall cracks of varying lengths, distributed over a part or all of the circumference of the tubes have been observed. In some cases, the cracks may be circumferential. When the degradation is located at the tube support plate levels, cracks are, in most cases, oriented longitudinally and crack lengths are confined within the support plate. The justification for allowing degraded tubes to stay in service is based on tube plugging criteria which take into consideration crack propagation kinetics and/or the concept of an allowable leak rate under normal operating conditions which preclude the risk of rupture in the event of accidental overpressure. Whatever the management approach to avoid excessive tube plugging, it is necessary to evaluate the critical crack dimensions for steam generator tubes subjected to internal pressure, thermal loadings and external loadings representative of mechanical loadings in accident conditions. Results of these studies performed by Framatome on 22.22 mm and 19.05 mm diameter tubes have been applied in France, in Spain and in Sweden.

2. TUBE PLUGGING CRITERIA FOR PRIMARY WATER STRESS CORROSION CRACKING NEAR THE TUBESHEET

2.1. Tubes with longitudinal cracks

Burst tests at room and elevated temperature (290°C ≤ θ ≤ 350°C) on tubes in alloy 600 with one or several longitudinal defects or cracks produced by stress corrosion cracking in the roll transition zone, have shown that the instability limit pressure of the longest through-wall crack is higher than that of a tube with the same type of damage located away from the tubesheet and support plates [1] to [5]. As experience shows, the instability of a longitudinal through-wall crack is, in all cases, preceded by very significant bulging which, in particular, depends on the length of the crack. The presence of the tubesheet prevents the tube from bulging freely up to a minimum of about 3V/ft above the tubesheet. This influence decreases progressively until, about 18 mm above the tubesheet when it becomes negligible. For a given crack length and pressure level, the bulging effect and therefore the stress magnification factor M are reduced. To obtain crack instability, the pressure must be increased inside the tube. From burst tests, it is demonstrated that a plastic instability rupture criterion can be used to predict reliably the rupture of cracked steam generator tubes in alloy 600. The rupture criterion based on the maximum shear theory according to Fresca or the application of the Mises criterion can be expressed as follow:

\[ P_{a} \frac{r}{t} \sigma = \frac{1}{M} (\lambda_{1P}) \]

where \( P_{a} \) is the instability limit pressure, \( r \) is the mean radius, \( t \) is the thickness. The best correlation between flow stress \( \sigma \) and material properties was obtained by \( \sigma = k (\sigma_{y} + \sigma_{u}) \) with \( k = 0.58 \pm 0.01 \) where \( \sigma_{y} \) is the yield strength and \( \sigma_{u} \) is the ultimate tensile strength. The bulging factor \( M \), which is a function of the geometry of the cracked tube section, is given by the following formula:

\[ M (\lambda_{1P}) = [1 + 1.61 a_{s}^{2}/rt_{1P}]^{1/2} \]

where \( 2a_{s} \) represents the crack length outside the tubesheet and \( t_{1P} \) represents the equivalent thickness which takes into account the interaction with the tubesheet [5] [6].

This criterion make it possible to determine the critical length of longitudinal through-wall cracks which depends on the pressure and the temperature, the mechanical properties of tubes, the tube dimensions as well as the tube to tubesheet last contact point with respect to the top of the tubesheet.
Additional burst tests on tubes with various crack configurations as a function of the location of the crack with respect to the top of the tubesheet as well as the location of the crack with respect to the last contact section between the tube and the tubesheet, were performed to verify the limit of validity of the rupture criterion in the roll transition zone. The comparison of theoretical and test results is presented in Figures 1 and 2. For all configurations examined it was confirmed that theoretical burst pressures for through-wall longitudinal cracks correlate well with test results while remaining conservative as long as the crack length outside the tubesheet is smaller than 18 mm and the X value (see Figure 1) is smaller than 7 mm. When these limits are exceeded, the critical crack length is given by applying the rupture criterion validated for cracks located away from the tubesheet and supports.

From burst tests it is also demonstrated that the type of rolling, i.e. full depth rolling or full depth rolling plus kiss rolling, does not significantly influence the instability limit pressure of a through-wall crack and the effect of interaction between several parallel longitudinal cracks distributed over the entire tube circumference remains small. The lower bound of the axial critical crack length whatever the crack location in the tube bundle is given by the following formula.

\[ L_{c} = 2 \left\{ \left[ \frac{t}{rP} \right]^2 - 1 \right\} \cdot \frac{rt}{1.61} \] \(^{1/2}\)

A knowledge of the critical crack length enables the plant operator to define a tube plugging criterion which takes into consideration the potential evolution of the crack during a cycle. The plugging criterion to be applied to preclude the risk of rupture in the event of accidental overpressure is given by the following formula:

\[ L_{\text{PLUGGING}} = L_{c} - \chi \]

Where:

- \( L_{c} \) is the critical longitudinal through-wall crack length corresponding to accident conditions.
- \( \chi \) is the potential evolution of the crack during the forthcoming cycle (this value must integrate the measurement error which depends on the type of non-destructive examination method used).

2.2. Tubes with circumferential cracks

By considering the most severe conditions likely to occur in a steam generator in the event of accidental overpressure, i.e. 182 bar and 350°C, the critical crack length of circumferential through-wall cracks can be estimated reliably with the criterion of maximum shear stress in the net section. The instability limit pressure \( P_{a} \) is given by the following formula (see Figure 3):

\[ P_{a} = \frac{2(\beta^2 - 1)(\pi - \alpha)\bar{\sigma}}{2\pi + (\pi - \alpha)(\beta^2 - 1)} \]

with \( \beta = \frac{R_{e}}{R_{i}} \)

where \( 2\alpha \) is the angle of the circumferential crack, \( R_{e} \) is the outer radius and \( R_{i} \) is the inner radius of the tube. Since the crack propagation kinetics of circumferential cracks are not well-characterized per operating cycle, tubes with circumferential cracks should be plugged or repaired.
3. TUBE SUPPORT PLATE PLUGGING CRITERIA ANALYSIS

To support the development of tube support plate plugging criteria, an experimental investigation of burst pressures of tubes degraded by IGSCC and various analyses were performed on 19.05 mm diameter tubes with axial part-through and through-wall cracks located at the tube support plate levels. Its purpose was to demonstrate that these part-through and through-wall cracks remain stable in normal operation or in the event of accidental overpressure (steam line or feedwater line break). Results of this study have been applied for Ringhals 3 and 4 steam generators [7].

3.1. Movement of support plates in the event of a steam line break

The maximum axial tube to tube support plate (TSP) relative movement in the case of rapid depressurization of the secondary side is such that the cracks may no longer be confined within the support plate and therefore the reinforcement effect of the tube support plate on the tube burst strength may be reduced.

The finite element model used to calculate the maximum TSP displacement under steam line break loads is shown in Figure 4. This model includes the tubes support plates, the tie rods and the divider plate separating the hot leg and the cold leg in the case of the steam generator with preheater.

The maximum deflection of TSPs depends on the design of the tube bundle support system (the location of tie rods and wedge supports, the boundary condition at the wedge support location) and the maximum differential pressure being simultaneously applied to each support plate. For safety purposes, calculations were performed without considering each individual tube/TSP interaction and therefore no friction effects were included in the model.

On the basis of data applicable for Ringhals 3 and 4 steam generators, the maximum TSP displacement for the top plate is about 13 mm and smaller than 8 mm for TSP levels 1 to 5.

3.2. Test program description

The tests were conducted at room and elevated temperatures (328°C ≤ θ ≤ 350°C) and a pressure of over 182 bar, on alloy 600 tubing with a nominal diameter of 19.05 mm, a nominal wall thickness of 1.09 mm and a minimum thickness of 0.99 mm based on fabrication tolerances. These burst tests were conducted on tubes with an axial part-through or through-wall defect and/or multiple secondary side stress corrosion cracks. The following tube specimens were tested:

- Tubes with a machined through-wall or part-through defect 19 and 25 mm long. The initial defect shape was chosen in such a way that the outer defect length L1 was at least as great as the inner length L2 (1 ≤ L2/L1 ≤ 1.3).

- Tubes heavily degraded by stress corrosion cracking in a caustic environment all round the outer surface and over a length between 18 to 24 mm. In addition, fatigue tests were used to produce a main through-wall crack with a length roughly equal to the 19 mm thick support plate.

- Tubes with a machined part-through defect 85% deep and 25 mm long located in the degraded region with numerous small longitudinal part-through corrosion cracks.

Burst tests were conducted on tubes with the defect or cracks located at the support plate level in such a way as to minimize the influence of the support plate in preventing the cracked tube from deforming freely, i.e.:

- The potentially unstable main crack was placed on the side with the maximum gap (tube off center in the support plate hole) : 0.89 mm

- A cracked region of unsupported length outside the support plate thickness was 7 and 11 mm. This free length outside the support plate was chosen so as to take into account the maximum movement of support plates in the case of a steam line break which is assumed occurring at the same time as the maximum differential pressure over the tube. In the case of an accidental depressurization of the secondary side, the support plate will partially uncover cracks in tube intersections if no crevice restraint exists.

4.1-4/12
The tubes were subjected to an internal pressure up to ligament rupture of the part-through defect or crack. When the rupture occurred without unstable propagation of the through-wall crack produced by plastic instability of the remaining ligament, the pressure was raised in steps up to tube rupture. Burst tests were also performed by applying an internal pressure representing the accidental overpressure likely to occur in the event of a secondary line break (182 bar, 350°C). The pressure was suddenly applied to the cracked tubes so as to obtain the rupture of the remaining ligament of the longest part-through defect or crack as well as instability of the longest through-wall crack and was maintained above 182 bar for some time. Burst tests were also performed by considering tube to TSP interaction when the locally damaged tube is subjected simultaneously to the maximum internal pressure and external loads likely to be exerted on tubes in the event of an earthquake or a line break. Burst tests at elevated temperature made it possible to take into account the following:

- The dynamic effect arising when the instability limit pressure $P_a$ of the through-wall crack resulting from rupture of the remaining ligament of the initial part-through crack is less than the rupture pressure $P_r$ of the ligament. Thus, if this effect becomes dominant ($P_a << P_r$) instability occurs before the gap between tube and TSP is filled by tube bulging following ligament rupture.
- The jet effect in the confined space within the support plate thickness. This effect may help to delay filling the gap.
- Maintaining a large leak rate through the potentially unstable crack ($Q > 2.5$ m³/H for a machined defect 25 mm long). The test conditions which do not employ a sealing device within the tube, allow the actual test pressure to be exerted on the sides of the crack, thus favoring instability.

3.3. Results and discussion

Based on the above considerations and by considering simultaneously the most unfavourable parameters likely to minimize the favourable influence of the support plate in preventing a cracked tube from deforming freely, the following general results were consistently observed:

- The tube support plate has no influence on rupture of the remaining ligament of the part-through defect or crack. Generally speaking, in the case of part-through defects, there is no bulging before breakthrough which usually occurs in the case of a through-wall crack. As a result, tube deformation is not sufficient to fill the tube/support plate gap. Since in this case the tube does not contact the support plate, the predicted ligament rupture is the same as for a defect located away from support plates [5].
- The experimental results showed that rupture pressures of the remaining ligament of a longitudinal part-through defect or crack are in good agreement with the plastic instability rupture criterion previously validated for cracks remote from support plates. Rupture pressures conform to theoretical predictions if the minimum defect length corresponding to the length of the remaining ligament is considered. When the maximum crack length (on the outside surface) is used in analysis, the ligament rupture pressure is then underestimated.
- After crack breakthrough of the tube wall thickness, the support plate limits bulging of the cracked tube once the gap between the tube and the support plate has been filled. This results in a considerable increase in the instability limit pressure of the through-wall crack. This increase depends on the total length of the crack, the tube-to-TSP gap and the length of the crack outside the support plate, see Table I.
- The shape of the defect has no significant effect on the instability limit pressure $P_r$ of the ligament.
- A through-wall crack having an initial length equal to the thickness of the support plate remains stable for a pressure equal to the burst pressure of a tube without defects (i.e. the case where the crack is totally confined within the support plate).

4.1-5/12
Tests show that the axial through-wall crack has to grow longer than 25 mm to become unstable at pressures exceeding 300 bar, see Table II. This propagation starts from the tip of the initial crack located outside the support plate. The same crack would have been unstable at about 150 bar if it had been located away from the support plate.

The presence of multiple stress corrosion cracks all round the tube circumference has no effect on the behavior of the potentially unstable main crack. The combination of the large leak rate from a defect within the tube support plate and a pressure which was kept higher than that producing instability after breakthrough in the case of a free span crack, was not sufficient to produce crack instability in this case.

When cracks in steam generator tubes are located at the support plate level (thickness 19 mm), the maximum allowable length of a longitudinal through-wall or part-through crack is 25 mm for all tubes. This result is valid as long as the maximum differential pressure over the tube remains limited to 182 bar and the free (unsupported) length of the crack outside the support plate does not exceed 11 mm. The unsupported length due to the tube-to-support plate relative movement during secondary-side depressurization shall be considered.

If there is no support plate, a through-wall crack 25 mm long produced by rupture of the remaining ligament of a part-through crack is unstable as predicted by the plastic instability rupture criterion, see Table I. As a result, provided the degraded region of the tube remains limited to the top thickness, i.e. 19 mm, as observed in steam generator tubes in operation, one or multiple axial part-through cracks confined to the thickness of support plate can become through-wall by IGSCC, fatigue or plastic instability of the remaining ligament without constituting a safety problem if it is possible to demonstrate that the maximum tube-to-TSP relative movement in the case of rapid depressurization of the secondary side is less than 11 mm.

4. SUMMARY AND CONCLUSION

To conclude, using methods which have been validated experimentally and are accurate to within less than 5 %, it is possible to determine the limits of tube degradation in the roll transition areas beyond which, tubes must be repaired or removed from service.

It is demonstrated that tubes degraded by IGSCC at the support plate levels do not constitute a safety problem under normal, upset or maximum postulated accident conditions when axial through-wall cracks up to 19 mm long are such that the free length outside the support plate do not exceed a limiting value of 11 mm during tube to support plate relative movement caused by secondary side depressurization. In the case of tubes heavily degraded by IGCA at the outer surface of tubes, it is noted that the plugging criteria must take into consideration the loss of mechanical strength which generate stresses in the net section likely to produce circumferential cracks.

Tubes with circumferential cracks should be plugged or repaired.

5. REFERENCES


Figure 1: Burst test results on tubes with a longitudinal through-wall defect in the roll transition zone.
Figure 2: Burst test results on tubes with multiple parallel defects in the roll transition zone.
Burst test results:
- No support conditions to simulate a tube support plate
- Restrained tubes - Tube bending prevented by presence of tube support plate
- Stable defects
- Burst pressure on tube without defect

Analytical results:
1. Unsupported tubes
2. $P_f = P$
3. $P_f = 0$

Where $P_f$ is the pressure applied on the sides of the crack.

Figure 3: Burst pressure on tubes with circumferential through-wall defects. Finite element model and beam plastic hinge model comparison between calculation and experimental data. Tubes in alloy 600 - Ø 22.22 mm
Figure 4: The finite element model perspective (1820 nodes - 2370 meshes) of the tube bundle support system
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<td>TEMPERATURE (°C)</td>
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**TABLE II**

**TUBE / TUBE SUPPORT PLATE INTERACTION: BURST TEST RESULTS AT ROOM TEMPERATURE**

- \( 2a_1 \): Initial inside length
- \( 2ae \): Initial outside length
- \( P_1 \): Pressure corresponding to initiation of tearing. The defect propagates from \( 2ae \) to \( 2a_{max} \) (final length) without instability
- \( P_{max} \): \( Pa \) (\( 2a_{max} \)) maximum pressure
LICENSING BASES FOR STEAM GENERATOR TUBE INTEGRITY
AND RECENT OPERATING EXPERIENCE

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ABSTRACT

The total heat transfer surfaces provided by the thin-walled tubing of the steam generators in pressurized water reactors (PWR) comprise well over 50% of the reactor coolant pressure boundary (RCPB). The integrity of the tubing during normal operating and accident conditions is important in that its failure (leaks and ruptures) would cause an escape of radioactive fission products directly to the environment, as well as complicate the system response if the failure occurs as a result of a loss-of-coolant accident (LOCA) or a main steam line break.

This paper presents a brief description of various regulatory instruments specifically developed for regulating the operation of steam generators to ensure that the regulations governing the integrity of the reactor coolant pressure boundary are met. Also presented is a summary discussion of recent operating experience with regard to various forms of tube degradation and the NRC approaches to deal with them. The forms of tube degradation that will be discussed include: primary water stress corrosion cracking (PWSCC) of tubes and tube plugs; intergranular attack (IGA)/stress corrosion cracking (SCC) of the tube outer surfaces; fretting/wear; and high cycle fatigue. A recently identified issue concerning collapse of tubing during combined safe-shutdown-earthquake (SSE) plus postulated LOCA loads is discussed. Finally, a brief status of steam generator replacement projects in the U.S. is presented.
1. INTRODUCTION

The thin-walled tubing of the steam generators is an integral part of the reactor coolant pressure boundary (RCPB), constituting well over 50% of the RCPB surface area. Failure of the tubing (i.e., leaks or ruptures) is of concern since it leads to a loss of primary coolant. In addition, such failures can permit the escape of radioactive fission products directly to the environment. A steam generator tube rupture (SGTR) is a complex event posing a significant challenge to the plant systems and to the operators. The system response is further complicated if the failure occurs as a result of a loss-of-coolant accident (LOCA) or main steam line break (MSLB).

To ensure that steam generators can be operated safely, the Nuclear Regulatory Commission (NRC) staff has established requirements for periodic tube inspections and leakage control to ensure adequate steam generator tube integrity, consistent with the governing NRC regulations concerning the integrity of the RCPB. This paper provides a brief description of the current licensing basis for ensuring adequate tube integrity. Recent operating experience trends, NRC staff responses to these trends, and other topics pertinent to the tube integrity issue are also presented.

Although outside the scope of the paper, the safe operation of steam generators is also assured through consideration of SGTR events as part of the design basis for each facility. Applicants for licenses to operate a pressurized water reactor (PWR) must demonstrate compliance with governing NRC regulations concerning the radiological consequences of such design basis events. In addition, plants are required to have emergency operating procedures for SGTR events and the operators are required to be trained in the use of these procedures.

2. REGULATORY REQUIREMENTS

NRC regulations (Title 10 of the Code of Federal Regulations (10 CFR)) establish the fundamental requirements pertaining to the RCPB that include steam generator tubing. Specifically, General Design Criterion 14 (GDC-14) of Appendix A to 10 CFR Part 50 requires that the RCPB have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture; GDC-31 requires that the RCPB be designed with sufficient margin to minimize the probability of rapidly propagating failure during normal operation and postulated accidents; GDC-32 requires that the RCPB be designed to permit periodic inspection and testing of critical areas to assess their structural and leaktight integrity; and GDC-30 requires that means be provided for detecting, and to the extent practical, identifying the source of reactor coolant leakage.

The regulatory approach to implement the requirements of relevant General Design Criteria, cited above, is to make sure that selection and fabrication of materials and design are in accordance with applicable ASME Section III Code Editions and Addenda endorsed by the regulations so that structures, systems, and components important to safety are designed, fabricated, erected, and tested to quality standards commensurate with the
importance of the safety functions to be performed. Once the plant is put in service, the regulatory approach to ensure that the integrity of systems, structures, and components are maintained is through in-service inspection and testing; and leakage monitoring and leakage limits.

Regulatory instruments specific to the maintenance of tube integrity in operating plants include the following:

- Plant Technical Specifications concerning the frequency of inspection, inspection sample sizes, tube plugging limits, and leakage limits;
- Regulatory Guide 1.83, "Inservice Inspection of PWR Steam Generator Tubes;" and
- Regulatory Guide 1.121, "Basis for Plugging Degraded PWR Steam Generator Tubes."

The explicit purpose of the periodic in-service inspections is to determine whether tube degradation is occurring in the steam generator, to assess the rate of tube degradation based on results of successive inspections, and to identify those tubes requiring plugging or repair. Criteria for establishing the tube plugging limits require that the plugging limit include margins for eddy current testing error and the projected rate of continued degradation.

The Technical Specifications also require an unscheduled steam generator in-service inspection when the plant Technical Specification leakage rate limit is exceeded. The Technical Specifications limit primary-to-secondary leakage through any one steam generator as well as all steam generators. These limits are based on two considerations: First, the total steam generator tube leakage limit for all steam generators ensures that the dosage contribution from tube leakage will be limited to a small fraction of 10 CFR Part 100 limits in the event of either a design basis SGTR or design basis MSLB. This limit is consistent with the assumptions used in the analysis of these accidents. Second, the leakage limit for any one steam generator ensures that steam generator tube integrity is maintained in the event of a MSLB or under LOCA. This leak rate is intended to correspond to a through-wall defect of a length that could not rupture the tube under normal or postulated accident conditions.

The NRC staff closely monitors SG operating experience and imposes additional requirements on a generic or case specific basis as needed to continue to ensure low risk and compliance with the NRC regulations concerning RCPB integrity.

3.0 RECENT OPERATING EXPERIENCE TRENDS

Pressurized water reactor (PWR) steam generator tubes in the U.S. have experienced widespread degradation by a variety of corrosion and mechanical mechanisms. These degradation problems have had a significant impact on steam generator reliability and, in addition, have

4.2-3/12
led to six SGTR events to date. These problems have necessitated extensive maintenance related activities throughout the industry resulting in significant costs and accumulated exposure to personnel.

3.1 Stress Corrosion Cracking

Various stress corrosion cracking (SCC) mechanisms have emerged in recent years as the dominant degradation mechanisms affecting steam generator tubes in the U.S. Susceptibility of the tubing to various SCC mechanisms is a function of the aggressiveness of the tubing environment, stress level within the tubing, and the inherent SCC susceptibility of the tubing material. SCC may initiate either from the inner diameter (ID) surface or outer diameter (OD) surface of the tubing.

ID initiated SCC, generally referred to as primary water stress corrosion cracking (PWSCC), generally occurs at tube locations of high residual tensile stresses associated with manufacturing process. This includes regions where the tubing has been expanded against the tubesheet by an explosive or mechanical rolling process, the transition region between the expanded and unexpanded portions of the tubing, and in the small radius U-bends of inner row tubes. Microstructural characteristics of the Inconel 600 tubing material, including lack of carbide precipitation in the grain boundary, appear to have been an important factor affecting many of these problems. Relatively high operating temperatures at some plants also appear to be a contributing factor to early PWSCC initiation. Early experience with PWSCC (through the mid-1980s) involved mainly cracks with an axial orientation. However, circumferentially oriented PWSCC has become widespread in recent years, particularly at expansion-transition locations of tubes which have been explosively or hard roll expanded against the tubesheet. Circumferentially oriented PWSCC has also recently begun to be observed at small radius U-bends.

OD initiated SCC, frequently associated with generalized intergranular attack (IGA), generally occurs in stagnant flow regions where impurities from the bulk water can concentrate. These include the sludge pile region on top of the tubesheet, the tubesheet crevice region (of steam generators where the tubes have been partial-depth expanded against the tubesheet), and in the tube-to-support-plate crevices. As is the case with PWSCC, circumferentially oriented OD SCC is becoming increasingly prevalent at expansion transition locations and in tube-to-support-plate crevices.

McGuire Unit 1 (with Westinghouse Model D-2 steam generators) experienced a steam generator tube rupture (SGTR) event on March 7, 1989, as a result of a 4-inch long OD initiated, axial stress corrosion crack. This crack initiated as a series of short microcracks under very unusual circumstances. The crack was located above the sludge pile region (and thus, was directly exposed to the bulk water) on the cold leg side. The crack occurred within a shallow groove which was believed to have been introduced sometime during tube or steam generator fabrication, although the precise cause of the groove has not been firmly established. Westinghouse theorizes that crack initiation may have been abetted by a contaminant within the groove, although no traces of such a contaminant were
observed in the groove subsequent to the SGTR event. No similar cracks were identified on other tubes during a 100% inspection with a bobbin coil probe performed immediately after the event and again during the next refueling outage.

3.2 Fretting/Wear

Fretting and wear type degradation continue to be important mechanisms affecting steam generator tubes in the U.S. These problems are usually the result of tube interaction with adjacent support structures due to tube or support vibration. The most noteworthy examples include wear at anti-vibration bar (AVB) supports of Westinghouse steam generators, at batwing supports of Combustion Engineering steam generators, at the baffle plate locations in the preheater regions of Westinghouse Model D and E steam generators, and at support plate locations within the economizer region of Combustion Engineering System 80 steam generators. These types of problems do not appear to pose significant tube integrity concerns, primarily because wear type flaws are relatively easy to detect and size, and because wear rates tend to be relatively low compared to the frequency of in-service inspection. The low wear rates reflect, in part, actions taken by utilities to mitigate wear. These actions include AVB replacement at a number of Westinghouse units with AVBs of an improved design, preheater modifications at Westinghouse Model D and E units to reduce tube vibration, and extensive preventive plugging of tubes at Combustion Engineering units which were anticipated to be subject to relatively high wear rates at the batwing supports and support plates.

3.3 High Cycle Fatigue

Since the late 1970s, once-through steam generators (OTSGs) supplied by Babcock & Wilcox have experienced numerous fatigue cracks and leaks at the fifteenth support plate and at the lower face of the upper tubesheet in tubes located near the open inspection lane. These failures have been attributed to flow induced vibration at corrosion initiated sites.

On July 15, 1987, North Anna Unit 1 (with Westinghouse Model 51 steam generators) experienced an SGTR event involving a complete circumferential failure of tube R9C51 just above the uppermost support plate as a result of high cycle fatigue. A post event assessment of air ejector radiation monitor and grab sample data indicated that the primary-to-secondary leakage had been present for 2 to 3 days prior to the event and had been trending upward. Leak rate monitoring procedures then in place did not take advantage of these data, and the operators were not aware of the precursor leakage trend until just before the rupture.

A failure investigation by Westinghouse identified three requisite conditions for a North Anna type failure. The first requisite condition is the buildup of hard magnetite corrosion product in the tube-to-support-plate crevice. This causes the tube to be fixed (locked-up) at that location, maximizing the tube bending stress at that location when the U-bend is subjected to flow induced vibration. The elimination of the crevice also
reduces vibrational fluid damping. Finally, the squeezing of the tube by
the hard magnetite produces a mean stress in the tube which reduces the
alternating stress level necessary to cause fatigue cracks. The second
requisite condition was the absence of an AVB support for the affected
tube. By design, the AVBs were installed into the tube bundle to at least
row 11; the inner rows (1 to 10) were nominally unsupported. The third
requisite condition was a fluid-elastic instability condition for the
affected tube, brought on by localized flow peaking affects (associated
with actual AVB insertion patterns) and by reduced damping in the
tube-to-support-plate crevice.

The staff issued NRC Bulletin 88-028 in response to the North Anna Unit 1
SGTR event, requesting specific actions at potentially susceptible plants
to preclude similar type failures in the future. The requested actions
included inspection of the steam generators for the presence of magnetite
at the tube-to-support-plate crevices, mapping the positions of the AVBs,
and analysis of the fluid-elastic response of the SG tubing.

In October 1988, Indian Point Unit 3 (with Westinghouse Model 44 steam
generators) experienced a 2 gpm primary-to-secondary leak which was the
subject of NRC Information Notice 88-999. This leak developed over a
period of 1 to 2 1/2 hours, but leveled off at 2 gpm and remained rela-
tively constant until plant shutdown was commenced. Subsequent inspection
identified a 240⁰ circumferential crack affecting tube R45C51, just above
the uppermost support plate. The tube was dented at this location due to
magnetite buildup in the tube-to-support-plate crevice. The tube was
removed from the field and examined in 1991. Preliminary information
indicates that high cycle fatigue was the failure mechanism. It is
interesting to note that the subject tube has the longest radius U-bend
and is supported by two rows of AVBs. Thus, this fatigue mechanism
appears to affect a different population of tubes than the North Anna
mechanism. The staff is monitoring the results of the ongoing investiga-
tion to determine if further regulatory action is warranted.

3.4 Cracking of Steam Generator Tube Plugs

In February 1989, North Anna Unit 1 experienced a "plug top release"
failure of a Westinghouse mechanical plug. The plug top was propelled up
the length of the tube, penetrating the tube at the U-bend and impacting
the adjacent tube. The failure mechanism for the plug was circumferen-
tially oriented PWSCC which caused the plug to become completely severed
from the plug body. A subsequent investigation revealed that the plug was
fabricated from a heat of Alloy 600 material exhibiting minimal
intergranular carbides which result from relatively low mill anneal
temperatures. The condition of minimal intergranular carbides is believed
to render the material highly susceptible to PWSCC. The high
stress/strain gradient associated with the plug design and the high hot
leg temperature at North Anna Unit 1 (620°F) are also believed to be
contributing factors. Investigations at other plants as a result of the
North Anna Unit 1 occurrence revealed that PWSCC of Westinghouse mecha-
nical plugs was extensive throughout the industry.
The NRC staff concluded that an industry wide systematic program to replace or repair Westinghouse plugs potentially subject to PWSCC was necessary to ensure the continued integrity of the reactor coolant system boundary and low risk. Accordingly, the NRC staff issued Bulletin 89-01\(^1\)\(^0\) requesting that licensees implement such a systematic program for plugs fabricated from the most susceptible heats of Inconel 600. Supplement 2 to Bulletin 89-01\(^1\)^1, issued in June 1991, applies to all Westinghouse mechanical plugs fabricated from Inconel 600.

The NRC staff issued Information Notice 89-65\(^1\)\(^2\) citing extensive PWSCC problems which have been experienced with B&W rolled plugs. The major contributors to the PWSCC in B&W rolled plugs are believed to be the same as for Westinghouse plugs. B&W rolled plugs are inspectable by eddy current test methods. The vast majority of PWSCC indications found to date by eddy current have been in the "heel" expansion transition region which is not part of pressure boundary of the plug. For this reason, cracks in B&W rolled plugs have been judged to be less urgent than those in Westinghouse mechanical plugs. However, the staff believes that heel cracks are a precursor to cracks at the "toe" transition which is part of the pressure boundary. Recently, a few plugs with heel transition cracks have been found to also contain cracks at the toe transition. Thus, the staff believes that identification (by eddy current) and removal of plugs with heel transition cracks are necessary to ensure the continued integrity of the B&W plugs. B&W has issued such recommendations to its customers. The staff continues to monitor experience with B&W plugs and the need for possible regulatory action.

3.5 Steam Generator Replacements

Table 1 summarizes steam generator replacement projects completed in the U.S. to date, and the associated radiological exposures to workers. The most recent project, completed in March 1991, was the replacement of the Combustion Engineering steam generators at the Palisades plant. Containment modifications at Palisades were necessary to accommodate entry of the replacement steam generators into containment. Replacement of the Combustion Engineering steam generators at Millstone Unit 2 is scheduled for 1992.

4. OTHER STEAM GENERATOR TOPICS

4.1 Inservice Inspection Issues

Eddy current testing (ECT) is the standard method for conducting inservice inspection of steam generator tubes. As evidenced by field experience, ECT has a long history of difficulties in the reliable detection and sizing of tubing flaws. These difficulties stem from limitations in ECT equipment, procedures, and personnel and from small inspection samples.
The industry has made a concerted effort to overcome these limitations and has succeeded in substantially upgrading ECT practices and capabilities in recent years. Many of these improvements have been embodied in the "Steam Generator Inspection Guidelines" developed by the Electric Power Research Institute (EPRI) sponsored Steam Generator Reliability Project (SGRP). However, the types of degradation being experienced have evolved over the years and SCC is now the dominant degradation mechanism affecting steam generator tubing. Detection and sizing of SCC flaws present a significant challenge to the capabilities of current ECT technology and equipment, procedures, and personnel due to small signal-to-noise characteristics of the SCC indications. This problem has been aggravated by the emergence of circumferential SCC as a significant degradation mode at several plants. Circumferential SCC can only be detected through the use of non-standard probes. Recent examples of problems with the detection of SCC, including circumferential SCC, were discussed in NRC Information Notice 90-4913.

The NRC staff believes that a key element to further improvements in ECT performance involves development of improved criteria for qualification and performance demonstration of ECT equipment, procedures, and personnel. The staff is participating in the ASME Code, Section XI, Special Working Group (SWG) on eddy current examination, and is actively promoting development of improved qualification and performance demonstration criteria for the Code. Revisions to Regulatory Guide 1.83 regarding inservice inspection of steam generator tubing are also under development by the staff to reflect recent advances in ECT technology and practice and to incorporate improved qualification and performance demonstration criteria and improved inspection sample criteria.

4.2 Primary-to-Secondary Leakage Monitoring/Limits

There is recent evidence from the field and from industry studies that the adoption of reduced limits (vs. the 500 gallons per day limit in the current Standard Technical Specifications) could increase the effectiveness of these limits in preventing tube ruptures for plants that are experiencing extensive SCC or are susceptible to fatigue. Indeed, many licensees of plants in these categories are already implementing such reduced limits voluntarily or at the request of the NRC staff.

However, the effectiveness of leak rate limits is also dependent on how well and quickly operators can monitor, trend, and respond to increases in primary-to-secondary leakage. The staff issued Information Notice 88-99 citing incidents of rapidly increasing primary-to-secondary leakage rates and shortcomings of plant equipment and procedures for monitoring and responding to such leakage. Improved capabilities in this area were cited as being potentially very beneficial in minimizing the likelihood of future SGTR events. More recent events have further underscored the urgency of this issue. At Mihama 2 in Japan, leakage developed only about one hour prior to the SGTR event which occurred in February 1991. SGTR events were narrowly averted at Three Mile Island in March 1990 and at Maine Yankee in December 1990 due to timely detection of, and quick response to, rapidly increasing leak rates. However, the effectiveness of methods and procedures for ensuring a timely response to rapidly increasing leakage appears to vary widely among U.S. plants.
The staff is continuing to examine leak rate limit and leak rate monitoring issues and the need for further regulatory actions in these areas. As part of this effort, the staff has initiated a program with Idaho National Engineering Laboratory to survey the effectiveness of leak rate monitoring programs at representative U.S. plants.

4.3 Steam Generator Tube Collapse Issue

In 1990, the staff first became aware of the potential for collapsing tubes during a combined safe-shutdown-earthquake (SSE) plus LOCA event, while reviewing a Westinghouse report supporting a proposed change to the Technical Specifications of a certain plant to permit up to 15% of the tubes to be plugged. Westinghouse analyses indicate that tube support plates may be deformed as a result of the dynamic loads associated with a combined SSE plus LOCA. The tube support plate deformation, in turn, was found to cause tube deformation. Following depressurization of the primary system (as a consequence of the LOCA), the resulting secondary-to-primary pressure differential was found to cause some of the deformed tubes to collapse, thus reducing the effective flow area through the steam generator. The reduced flow area increases the resistance to venting of steam generated in the core during the reflood phase of the LOCA, increasing the peak clad temperature (PCT). Total flow area reductions calculated by Westinghouse range between 0 and 7.5% depending on the magnitude of the seismic loads. Resulting PCT penalties (for affected plants) range between 0 and 50°F and no plants were found to exceed the 2200°F PCT limit in 10 CFR Part 50.46(b)(1).

The staff noted, during a meeting with Westinghouse on this matter, that many tubes at a number of Westinghouse plants have experienced extensive SCC degradation at the support plate locations. The staff expressed the concern that if tubes containing such cracks should collapse under SSE/LOCA conditions, such tubes could potentially rupture resulting in significant secondary-to-primary in-leakage during the LOCA event. Potential in-leakage of unborated secondary side water above 1 gpm during a LOCA event is not addressed in current LOCA/ECCS analyses. To address this concern, Westinghouse is issuing customer information letters during the first and second quarters of 1991 recommending that tubes susceptible to collapse (i.e., those located near the support plate wedge supports) be inspected for cracks at each scheduled inspection. This recommendation will only apply to plants with identified cracking at the support plates. The staff is continuing to assess this issue and the need for regulatory action.

5.0 Conclusions

Steam generator tube degradation problems continue to cause a significant economic burden to the industry by virtue of the extensive maintenance effort and attendant occupational exposures needed to ensure both operational reliability and tube integrity. Unscheduled outages due to SG tube leakage, including occasional SGTR events, and the need to replace steam generators (at the rate of one plant per year since 1980) are adding
significantly to this burden. There is no end in sight to these problems for plants operating with their original steam generators. However, in spite of the widespread tube degradation problems, the NRC staff concludes that the current regulatory approach for ensuring adequate SG tube integrity has been and continues to be effective in protecting public health and safety. A generic risk assessment performed as part of the NRC Unresolved Safety Issues Program and documented in NUREG-0844 indicates that risk from SGTR events is not a significant contributor to the total risk at a given site, nor to the total risk to which the public is routinely exposed.

Notwithstanding the effectiveness of the regulatory approach and the low risk, the NRC staff continues to closely monitor operating experience and to assess topical issues, such as the effectiveness of in-service inspection programs, the effectiveness of leak rate limits and monitoring programs, and the potential for tube collapse during SSE plus LOCA. As in the past, the staff may issue additional requirements on a generic or case-specific basis as necessary to continue to ensure low risk and compliance with the governing NRC regulations concerning RCPB integrity.
<table>
<thead>
<tr>
<th>Plant</th>
<th>No. of Loops</th>
<th>Steam Generator Manufacturer/Model</th>
<th>Completion Date</th>
<th>Radiological Exposure (man-rem)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Surry 2</td>
<td>3</td>
<td>W/51 W/51F</td>
<td>9/80</td>
<td>2141</td>
</tr>
<tr>
<td>Surry 1</td>
<td>3</td>
<td>W/51 W/51F</td>
<td>7/81</td>
<td>1759</td>
</tr>
<tr>
<td>Turkey Point 3</td>
<td>3</td>
<td>W/44 W/44F</td>
<td>4/82</td>
<td>2151</td>
</tr>
<tr>
<td>Turkey Point 4</td>
<td>3</td>
<td>W/44 W/44F</td>
<td>5/83</td>
<td>1305</td>
</tr>
<tr>
<td>Point Beach 1</td>
<td>2</td>
<td>W/44 W/44F</td>
<td>3/84</td>
<td>580</td>
</tr>
<tr>
<td>H.B. Robinson 2</td>
<td>3</td>
<td>W/44 W/44F</td>
<td>10/84</td>
<td>1206</td>
</tr>
<tr>
<td>D.C. Cook 2</td>
<td>4</td>
<td>W/51 W/51F</td>
<td>3/89</td>
<td>561</td>
</tr>
<tr>
<td>Indian Point 3</td>
<td>4</td>
<td>W/44 W/44F</td>
<td>6/89</td>
<td>541</td>
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<tr>
<td>Palisades</td>
<td>2</td>
<td>CE CE</td>
<td>3/91</td>
<td>484</td>
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<tr>
<td>Millstone</td>
<td>2</td>
<td>CE B&amp;W (Canada) Start 1992</td>
<td></td>
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</table>
REFERENCES


4.2-12/12
TUBE PLUGGING CRITERIA FOR AXIAL AND CIRCUMFERENTIAL CRACKS IN THE TUBESHEET AREA

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ABSTRACT

This paper presents the plugging criteria developed by Belgatom for steam generator tubes affected by primary water stress corrosion cracking (PWSCC). These criteria take advantage from the efficient ECT RPC inspection techniques developed by Laborelec and are backed up by an extensive experimental program.

The plugging limit is expressed as the length of a crack, conservatively assumed to be through wall; it is taken equal to the lowest of two calculated values, one based on a best estimate, the other on a most conservative estimate under accidental conditions.

The main benefit of using these criteria is to allow units to operate with numerous through-wall cracks without impairing safety and reliability.

RESUME

Des critères de bouchage spécifiques ont été développés par Belgatom pour les tubes de générateur de vapeur affectés par la corrosion sous tension en eau primaire (PWSCC). Ces critères tirent avantage des techniques d'inspection performantes par sonde CdF rotative mises au point par Laborelec et s'appuient sur un programme d'essai très étendu.

La limite de bouchage s'exprime sous forme d'une longueur mesurée de fissure, conservativement supposée traversante; elle est prise égale à la plus basse de deux valeurs calculées, l'une basée sur les valeurs nominales, l'autre sur les valeurs conservatives de tous les paramètres concernés.

L'avantage majeur de cette approche est de permettre le fonctionnement des unités dont les générateurs de vapeur ont de nombreux tubes fissurés, sans en affecter la sûreté ni la disponibilité.
1.0. INTRODUCTION

For a number of years, three Belgian nuclear power plants have experienced primary water stress corrosion cracking (PWSCC) in the expansion transition area on a very large number of tubes. [1]

The Belgian units operate with numerous through-wall cracks without impairing the safety and the reliability of the plants. This is achieved by a safety approach based on the extensive use of advanced non-destructive examination (NDE) techniques and the development of new plugging limits. These limits are derived from a realistic interpretation of NRC Regulatory Guide 1.121 and are backed up by a substantial experimental program.

The plugging limits apply to axial or circumferential stress corrosion cracks located between the weld to the tubesheet and the upper tube support plate.

2.0. HISTORICAL BACKGROUND

Basically, Belgian Utilities are committed to following the U.S. nuclear safety rules. However duly justified deviations (or interpretations) can be obtained on a case-by-case basis. In the plant Technical Specifications defining the SG inspection requirements, the usual "40 % plugging limit" is implemented but an allowance is made for alternative criteria based on Regulatory Guide 1.121.

It is useful to recall the origin of the 40 % plugging limit. This requirement originates from the first generic problem encountered with SG’s which was a uniform loss of thickness through corrosion, i.e. wastage in the sludge pile. Based on a factor of safety not less than 3 to be maintained under normal service conditions, the required minimum tube wall thickness was 40 %. This value was increased to 60 % in order to have an additional allowance to cover uncertainties regarding measurement of the flaw size and its growth between two consecutive inspections. The 60 % minimum wall thickness meant that tubes with thinning of 40 % or greater had to be plugged.

Generalizing this same criterion to other types of more local flaws (cracks in particular) can be excessively conservative. This is why alternate approaches were investigated in Belgium.

The Westinghouse P*, F* and L* approaches were reviewed but were not considered viable as they avoid the use of the 40 % limit within the tubesheet but maintain it unchanged for the roll transitions (and some depth below the secondary side of the tubesheet) where practically all of the PWSCC cracks are actually located. These latter cracks (above the top of the tubesheet) are also the only significant ones for safety and reliability.

The Leak Before Break (LBB) philosophy was also considered.

According to this approach : "a flaw that would be critical under accidental conditions would be reliably detected under normal service conditions by a leak exceeding the Technical Specification allowable limit".
Although it is certainly the usual behavior, this has not always been observed in actual rupture cases. Among the known exceptions, the following are worth mentioning:

- Occasional steam generator tube rupture without prior notice by any measurable leak.
- In-service low leak rate of relatively long axial cracks (possibly due to clogging by crud or precipitates).
- Aligned axial crack components, with an overall critical length, without detectable leakage through the components.

One could also imagine long and deep (but not through-wall) cracks in either axial or circumferential directions (possibly initiated by surface scratches).

While some of these exceptions may be dealt with through use of probabilistic assessments, there is an associated trend to lower the allowable in-service leak rate. This latter consequence is believed to be unduly constraining to the plant operator (possible increase of unscheduled shut-downs).

Moreover, even relatively large leaks cannot necessarily be located and removed. Such a case has been experienced in 1987 by a Belgian SG, where a 20 to 30 l/h (0.09 to 0.13 gpm) leak could not be located by using all possible detection methods, including the Helium leak test. Plant operation was eventually resumed and the leak remained practically unchanged during the three following operating cycles.

On the other hand, extensive hardware and software developments by Laborelec allowed eddy current inspection of all roll transitions without any penalty on plant down-time. [2] This capability of reliably sizing the cracks led to the decision to develop new plugging criteria.

3.0. CRACK SIZE MEASUREMENT PRACTICE

The "Rotating Pancake Coil" (RPC) ECT technique has the potential for accurate length sizing of individual axial cracks. A good knowledge of the sizing accuracy has been obtained by comparing the RPC ECT length measurements with the actual maximum (ID) length of about 60 PWSCC roll transition cracks from 6 pulled tubes.

The sizing accuracy margin has a maximum value of 1.5 mm.

As data acquisition and analysis of the roll transition area of 100% of the tubes of one steam generator can be performed by Laborelec in about 2 days, the common practice in Belgium is to inspect 100% of the tubes at each plant refuelling outage. This allows to enhance the knowledge on the length increase of defects between two inspections and to simplify the inspection, as it is not necessary to define a specific inspection sequence of tubes from the last inspection results.
4.0. CRITICAL SIZE CALCULATION PRINCIPLES

Discussion

Regulatory Guide 1.121 allows the establishment of acceptable flaw sizes based on the following safety factors (with respect to tube bursting):

- Three under normal service conditions.
- A value consistent with the limits set by the ASME III Code art. NB-3225, for accidental conditions.

R.G. 1.121 does not specify whether:

- Mechanical and geometrical characteristics of the tubes must be taken at their nominal or most unfavorable value. The frequent reference to the design code seems to imply this unfavorable combination.
- Additional margin must be applied to the dimension of the detected flaw (prior to comparison against the acceptable value) in order to account for:
  - Uncertainties relating to the NDE measurement method.
  - Flaw growth over the period of service until the next inspection.

The requirements of R.G. 1.83 and of ASME XI (together with the historical basis of the 40% criterion) suggest these should be taken into account.

Also the following interpretations are used in order to establish a concrete set of criteria.

Loads to be considered

Because of the high ductility of Inconel, tube rupture is preceded by considerable plastic deformation (high COD - Crack Opening Displacement - bulging, etc.) so that the "secondary" stresses are relieved and can be neglected (whereas they play a major part in the stress corrosion or fatigue processes). The only "primary" type stresses are those resulting from the differential pressure and, possibly, from the inertial effects induced by a steam line break (SLB) or a feed water line break (FWLB) concurrent with the Safe Shutdown Earthquake (SSE). But for the portion of tubes located just above the tubesheet, those inertial induced stresses are neglected because their value is comparatively low (at the tubesheet level), the current evolution of the ASME code tends to classify these also in the "secondary" type and the dynamic loads induce essentially tube bending with resulting stresses not significantly interacting with axial flaws.

However, for circumferential cracks the stresses induced by differential expansion between the hot and cold legs could not be negligible, despite the "secondary" character usually assigned to thermal stresses. Indeed, the tube axial deformation at the flaw level remains low compared with the displacement that would be needed to relieve stress. Nevertheless, these stresses were neglected, because they are low when compared against those resulting from pressure and because they are compressive in the hot leg, i.e. where practically all the stress corrosion cracks occur.

4.3-4/13
Safety factors

In compliance with the spirit of ASME III (and in strict conformity with article IWB 3612 of ASME XI) the safety factors are taken as:

- Three for normal and upset conditions.
- $\sqrt{2}$ for emergency and faulted conditions.

These factors are intended to apply to loads (in practice, the differential pressure). However, at pressures 3 times or $\sqrt{2}$ times the actual values, the margins in terms of flaw size (ratios of critical crack length to actual length) are higher than these values (3 and $\sqrt{2}$) for axial cracks, and considerably lower for circumferential cracks (Figure 1). This does not appear to be reasonable as the actual uncertainty is more related to size than to load.

Therefore, these safety factors have been applied to the flaw length rather than to the pressure.

Reinforcing effects

Based on results of the Laborelec experimental program, the reinforcing effect of the tubesheet (axial cracks) and of the nearby support plate (circumferential cracks) is taken into account.

Allowable crack size

The allowable value of the measured crack length is taken equal to the lowest of the two following evaluations:

First: **best estimate with safety factor**

The average critical length is divided by the R.G. 1.121 safety factor, for both the normal and accidental conditions. The lowest resulting value is retained.

The allowable length is obtained by deducting the average value of:

- Sizing inaccuracy resulting from the inspection method.
- The propagation (in length) of the flaw until the next inspection.

Second: **most conservative estimate, without safety factor**

The minimal critical length is considered and the allowable length is obtained by deducting the maximum effect of both sizing inaccuracy and crack propagation rate.

The plugging limit is taken equal to the (lowest) allowable length, rounded off to the next higher mm.

The entire procedure is summarized in Table I, and illustrated by an example in Table II for axial cracks in the roll transition area of 7/8" OD tubes.
5.0. EXPERIMENTAL WORK RESULTS

Axial cracks
The Laborelec experimental program was conducted to validate the general analytical model for defects in a free tube run and to establish an empirical reinforcing factor for defects located close to the tubesheet.

More than 200 specimens, with through-wall axial flaws, were burst tested, using a plastic bladder sealing system, locally reinforced by a thin metal patch (when required to reach the unstable rupture condition).

Circumferential cracks
The Laborelec experimental program was conducted to validate the general analytical model for both supported and unsupported tubes.

A major part of the program was devoted to the beneficial restraining effect of the flow distribution baffle (FDB) and/or first tube support plate (TSP) on the critical length of a through-wall circumferential crack located in roll transition.

A preliminary phase was aimed at a proper phenomenological understanding of the restraining effect from the nearby FDB.

In a later phase of the program, the lateral restraint (from FDB and/or TSP) was simulated by a special test rig, adjustable to the various geometries under consideration (7/8" or 3/4" OD, SG model 51 or D4).

More than 50 specimens, with through-wall circumferential flaws, were burst tested either in the unsupported condition or in the supported condition.

Combined cracks
The Laborelec program also addressed special defect morphologies such as

- Inclined flaws, up to 45 deg. from axial.
- Axial flaws, of various lengths, associated with a small (typically 1/4" long) circumferential component.

It was concluded that burst behavior can be safely evaluated on basis of the axial flaw component.

International comparison
For all tests conducted in Belgium, the reported burst pressure corresponds to unstable propagation. At this pressure, there is an instantaneous crack extension at both ends of the initial flaw. Crack arrest results from the quick pressure drop of the incompressible test medium; this would never be expected with gas pressurization or under actual SG conditions.
Experimental programs conducted by other countries have not always reported burst pressures corresponding to actual unstable conditions.

Within the framework of the "Mechanism Specific Defect Management Committee", conducted by EPRI in the U.S.A.[3], a comparison of various experimental programs indicated a large scatter, as illustrated by Figure 2. The lower curve (from the U.S.A.) was produced under different laboratory conditions (no metal reinforcement of the sealing system, not always propagation).

To clarify the situation, Laborelec recently conducted tests under more representative SG conditions, i.e. without any seal (thus also allowing the pressure effect on the flaw tips)[4]. This required a high pressure pump (up to 270 bar) with large flow rate (up to 100 m³/h).

The results obtained with this test rig fully validate the "Belgian curve" as illustrated by Figure 2.

6.0. CRACK GROWTH RATE

The RPC ECT inspections performed at Doel 3 and Tihange 2 since 1985 have yielded a very large database for the crack growth rate of PWSCC axial cracks in roll transitions (over 14000 crack growth data points from 2 cycles of 6 steam generators). This database is continuously expanding.

The systematic 100 % inspection of all tube bundles allows not only to derive average crack growth rates but also to differentiate them as a function of the initial crack length.

Figure 3 illustrates a typical distribution of the crack growth during one cycle for one steam generator of Tihange 2.

Now, for practical applications (i.e. definition of the allowable crack size) the reference data were limited to cracks having an initial crack size close to the allowable limit.

Because of the knowledge of the crack growth rate as a function of the initial crack size, it was possible to find the average and maximal values of the crack propagation until the next inspection:

- for axial cracks:
  \[ \delta_{\text{avg}} = 0.7 \text{ mm/cycle} \quad \text{and} \quad \delta_{\text{max}} = 3 \text{ mm/cycle} \]

- for circumferential cracks:
  \[ \delta_{\text{avg}} = 2.5 \text{ mm/cycle} \quad \text{and} \quad \delta_{\text{max}} = 6 \text{ mm/cycle} \]
7.0. CONCLUSIONS

Using the procedures and values outlined hereabove, a set of plugging criteria has been established for the Belgian steam generators that allows units to operate with numerous through-wall cracks without impairing the safety and reliability of the plants.

These plugging limits are larger, by 3 to 4 mm, in the circumferential than in the axial direction. The difference would have been still much larger if the RG 1.121 safety factors had been applied on the load (differential pressure) instead on the crack length.
<table>
<thead>
<tr>
<th>Critical length (thru-wall crack)</th>
<th>Best estimate with safety factor</th>
<th>Most conservative estimate without safety factor</th>
</tr>
</thead>
<tbody>
<tr>
<td>Average value, on basis of:</td>
<td>- nominal tube geometry (diameter, wall thickness)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>- average of actual mechanical properties (YS, UTS) (*)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>- service conditions</td>
<td></td>
</tr>
<tr>
<td>Safety factor (on length)</td>
<td>normal 3</td>
<td>accidental $\sqrt{2}$</td>
</tr>
<tr>
<td></td>
<td>lowest value</td>
<td></td>
</tr>
<tr>
<td>Additional margin</td>
<td>average</td>
<td>maximum</td>
</tr>
<tr>
<td>- sizing inaccuracy</td>
<td>average</td>
<td>maximum</td>
</tr>
<tr>
<td>- propagation until next inspection</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Plugging limit</td>
<td>lowest value of $\frac{\text{Critical length}}{\text{Safety factor}}$</td>
<td>additional margin</td>
</tr>
</tbody>
</table>

For cracks in ROLL TRANSITIONS, the plugging limit is increased to credit for the constraining effect
- of tubesheet, for axial cracks
- of support plates, for circumferential cracks

(*) Mechanical properties, measured at design temperature (343°C = 650°F) for all Inconel heats used in construction.
<table>
<thead>
<tr>
<th>Service condition</th>
<th>Best estimate</th>
<th>Most conservative estimate</th>
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</thead>
<tbody>
<tr>
<td>Critical crack length</td>
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<td>21.9 mm</td>
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<tr>
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<td>1.2 mm</td>
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<tr>
<td>Safety factor</td>
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<td>$\sqrt{2}$</td>
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<tr>
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<tr>
<td>Additional margins</td>
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<tr>
<td>- propagation (to be considered on one side)</td>
<td>0.7 mm</td>
<td>0.7 mm</td>
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<tr>
<td>- sizing inaccuracy</td>
<td>0 mm</td>
<td>0 mm</td>
</tr>
<tr>
<td>Allowable length</td>
<td>12.2 mm</td>
<td>15.6 mm</td>
</tr>
<tr>
<td>Plugging limit</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
$\frac{l_c}{l}$: ratio of crack critical length to actual crack length

$\frac{p_c}{p}$: ratio of critical pressure to service pressure

**AXIAL CRACK.**

Example:

At $\frac{p_c}{p} = 3$

$\frac{l_c}{l} = 4.2$

**CIRCUMFERENTIAL CRACK.**

Example:

At $\frac{p_c}{p} = 3$

$\frac{l_c}{l} \approx 1.4$

SAFETY MARGIN ON LENGTH

AS A FUNCTION OF

SAFETY MARGIN ON LOAD (pressure).

Figure 1

4.3-11/13
LABORELEC TEST RESULTS WITHOUT SEALING SYSTEM

Figure 2

Crack length (mm) for Ø 7/8"

Normalized Burst Pressure (P)

Normalized Crack Length (λ)

French
Belgian
Westinghouse
Lower Bound (EPRI)
+ Laborelec test results

TYPICAL CRACK GROWTH DISTRIBUTION DURING ONE CYCLE

Figure 3

%  
12  
18  
24  
30  
36  
42  
48  
54  
60

-1  0  1  2  3  4  5  6  
Length increase (mm)

4.3-12/13
REFERENCES


SWEDISH EXPERIENCES OF A REVISED PLUGGING CRITERION FOR STEAM GENERATOR TUBES – A REGULATORY VIEW

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Abstract

In 1989 the Swedish Nuclear Power Inspectorate (SKI) accepted revised plugging criteria for tube defects, based on tube loads following postulated Steam Line or Feedwater Line Break. The criteria accepts longitudinal through-wall stress corrosion cracks in the tube-sheet region, but puts limits on their length. The allowable defect length is obtained by subtracting from the critical length, $L_C$, the length growth under one cycle, $L_P$, and the accuracy of the length measurement, $E$. All tubes are to be measured each cycle. No limit was initially put on the number of defects of different sizes. The safety of the criterion relies heavily on the reliability of the estimates on $L_P$ and $E$, and past and present efforts have focussed on this point.

In 1991 SKI accepted a new plugging criterion for tubes affected by stress corrosion cracking at the tube-support plate region. This criterion also relies heavily on the reliability of the estimates of crack growth and crack sizing accuracy.

The paper will discuss the regulatory aspects of the plugging criteria.
**Background**

The Ringhals 3 and 4 PWR plants each have 3 Westinghouse model D3 steam generators with 4700 tubes made from Inconel 600 material. In 1989 the Swedish Nuclear Power Inspectorate (SKI) accepted a revised service acceptance criterion for steam generator tubes affected by primary side stress corrosion cracking in the tube sheet region. The normal action if the criterion is not met is to plug the tube, so it will be referred to as a plugging criterion, although other actions may be possible. The old criterion was that the depth of any defect was not allowed to exceed 50% of tube wall thickness. Due to difficulties to measure defect depths with sufficient accuracy, and prompted by the rather extensive cracking in the Ringhals 3 and 4 plants, a revised criterion was proposed by the plant operator, the Swedish State Power Board (SSPB), and was accepted by SKI. Since then the criterion has been subject to further discussion, and service experiences during two cycles have accumulated.

In 1990 SSPB also proposed a slightly revised plugging criterion for tubes affected by secondary side stress corrosion cracking at tube support plate intersections. In spite of the fact that the proposed change was small compared to the change for the tube sheet region SKI could not accept the proposed revision for general use. The reason for this was that the inspectorate's initial safety analysis raised questions about the support plate integrity during faulted condition such as Steam Line or Feed Water Line breaks. Further analysis during 1991 led however to an limited acceptance of the proposed criterion.

**Regulatory basis for new plugging criterion**

The steam generator tubes are an important part of the barrier between the primary and secondary side. Probabilistic risk assessment and other safety studies have unambiguously pointed out the safety significance of tube ruptures as initiating events in transient sequences involving high operator stress and possibilities to bypass the containment by blowing mixed primary/secondary steam into the open air. Continued operation must therefore fulfill the inspectorate's basic safety policy for handling known defects or degradation of such barriers, which states that

- the extent of any degradations or defects discovered, which are considered relevant to safety, must be determined with acceptable precision
- the degradation mechanism must be understood, to the extent that a prognosis of growth rates can be made
- necessary remedial and control measures – at least of a provisional nature – must be implemented on the basis of the estimated extent and growth rates. Some uncertainties in these estimates can be accepted if it can be shown that the probability of serious consequences with regard to safety is low.
In many situations when in-service inspection indicates defects in safety-related components that exceed commonly used acceptance standards (such as the "50% wall" tube plugging criterion of ASME Section XI, IWB 3500) the normal way is to proceed with the analysis procedure specified in ASME Section XI, IWB 3600. This is one way to demonstrate that the inspectorate's basic safety policy is satisfied. This evaluation procedure, however, turned out to put limits on both allowable crack depth and crack length which would lead to extensive plugging in Ringhals 3 and 4 steam generators.

The experiences from other European plants and knowledge of the measurement accuracy achievable with available technique indicated that an alternative approach might be considered. It was realized that leakage actually occurred even though no through-wall cracks were observed during the in-service inspection. (Notice that the old plugging criterion was based on the depth of the cracks, not on the leakage, so formally it may be argued that there is no contradiction.) Experience from a number of European steam generators affected by primary side stress corrosion cracking in the tube-sheet region had shown that the longitudinal cracks were almost always through wall when they become detectable by eddy current in-service inspection (2-3 mm in length). Tubes removed from the steam generators were found to have through-wall cracks which had not been identified during the in-service inspection. It was realized that the available eddy-current examination technique (with standard bobbin coils) was very unreliable for crack depth measurements.

By fracture mechanics arguments and supporting experiments it was demonstrated that the crack length rather than the crack depth was the relevant parameter for tube rupture, and hence for the safety of the primary to secondary side barrier. But also the crack length measurement (with standard bobbin coils) was unreliable.

The inspectorate's initial analysis of the safety aspects of the proposed plugging criterion led to the conclusion that longitudinal through-wall cracks of limited length could be accepted for tubes with primary side stress corrosion cracks in the tube-sheet region without violating the basic safety principle, provided that

- the annual in-service inspection program was extended to include all tubes
- it could be demonstrated that the acceptable length would be of an order which could be measured accurately
- SSPB could present a reliable estimate of the crack growth during one cycle.

If these conditions could be satisfied it could even be argued that the revised plugging criterion in combination with the extended in-service inspection program would lead to a higher degree of safety than would be achieved by the old criterion and in-service inspection program (3% of the tubes per year).
Criterion for tube sheet region

The plugging criterion accepted by SKI in 1989 for use in Ringhals plants 3 and 4, with their specific water chemistry, temperature, material and monitoring techniques, is applicable to the entire tube sheet region, although the specific requirements vary along the tube length. Basically it is required that the length of a crack shall be sufficiently small to avoid that the defect starts to propagate during a postulated Large Steam Line Break (SLB) or Feed Water Line Break (FLB). In terms of a formula the requirement is that the measured crack length must be less than or equal to an acceptable length, \( L_T \),

\[
L_T = L_C - L_P - E
\]

where \( L_C \) is the critical length which will cause tube rupture if the secondary pressure vanishes, \( L_P \) is growth of a crack over one period, and \( E \) is the accuracy of the eddy current measurement technique.

The value for \( L_C \) (11 and 13 mm) was based on most unfavourable combination of tube geometries (maximum diameter and minimum thickness), most unfavourable combination of mechanical properties at 343°C, maximum difference pressure (\( \Delta P = 182.5 \) bar at FLB and SLB) and crack location relative the top of the tube sheet (TTS). The chosen value can therefore be regard as conservative for the large majority of tubes.

The value for \( L_P \) (5 mm) was the maximum observed (measured) length growth in Ringhals 4 between 1987 and 1988. The measurement was done with an eddy current technique and scanner that were known to give a large scatter. An improved measurement technique introduced in 1989 and additional crack growth data for later cycles have verified that the growth rate is in fact conservative. Available data from other European steam generators were the crack growth estimates are based upon accurate eddy current measurement also indicates that the chosen value is conservative. For the Ringhals 3 plant the primary side temperature (T-hot) has been reduced from 343°C to 308°C, which should increase the conservatism in \( L_P \).

The value for \( E \) (±3 mm) corresponds to twice the standard deviation for all length measurements in a series of performance demonstration/qualification tests that were performed when the new multiple eddy current rotating pancake coils technique and scanner were introduced in 1989. The qualification tests included both tubes with laboratory grown cracks and pulled tubes from Ringhals 3 and 4. Over 1400 measurements were made with different set-ups, drive-shafts, probes and analysts. Detailed evaluation of the results showed that the inspection system has a tendency to underestimate short cracks and overestimate long cracks. For cracks longer than 4 mm the true length is overestimated by approximately 1 mm with a standard deviation of 0.9 mm. Chosen value can be regard as an relatively reliable estimate of the accuracy \( E \). This result was however somewhat obscured when measured crack growth rate from 1989 to 1990 initially showed a negative mean crack growth rate. This is obviously physically unrealistic and indicated the existence of systematic errors in the data evaluation. Plausible
reasons for the errors have been identified, and reevaluation of the data has given more realistic mean crack growth rate.

Further development of the criterion

Further verification of the measurement technique as well as experiences of crack growth during the cycle 1989 - 1990 has indicated that a development of the plugging criterion might be possible. Responding to a SKI request SSPB has presented a probabilistic type analysis relating the core damage frequency to tube crack lengths. The analysis shows that for a given low core damage frequency the allowable crack length may be increased if the number of tubes with long cracks is limited. In 1991 SSPB proposed a modification of the plugging criterion for the tube-sheet region. Instead of using maximum observed crack growth and measurement unaccuracy based on all crack length allowable length, $L_T$ should be determined from the expression

$$L_T = L_C - L_P - E - 2 \left( \sigma_{Lp}^2 + \sigma_E^2 \right)^{1/2}$$

where $L_P$ is the mean crack growth over the entire tube population during one period, $E$ is the mean value of the measurement accuracy, $\sigma_{Lp}$ is the standard deviation for true crack growth rate, and $\sigma_E$ is the standard deviation of the measurement accuracy for cracks longer than 4 mm. As before $L_C$ is the critical crack length determined for the most unfavourable combination of material and strength, 13 mm (or 11 mm), $L_P$ was proposed to be 1 mm and $E$ to be -1.1 mm, corresponding to a systematic overestimation of the crack length. The standard deviations are taken to be $\sigma_{Lp} = 1.5$ mm, $\sigma_E = 0.8$ mm. With this criterion the allowable crack length for a limited number of tubes should increase with a factor of approx. 2.

Although SSPB has shown that the chosen value for $L_P$ is conservative some uncertainties remain. One is the possibility that crack "jumps" occur by concatenation of separate cracks. Consider a main crack which grows slowly, but which has a small (undetectable) crack in front of it. When the cracks grow together, the apparent (observed) crack growth rate will be large, although other cracks appear to grow slowly. The proposed values and the standard deviations have been considered sufficiently conservative to allow the use of criterion for 10 tubes per unit.

Criterion for tube support plate region

During the past 2-3 years an increasing amount of observed secondary side stress corrosion cracks at the tube support plate intesections in Ringhals 3 has been observed. The same tendency has been reported from other affected steam generators, with many and deep cracks in tubes at the lower support plates. Based on similar experiments and analysis as for the cracks in the tube-sheet region SSPB 1990 therfore proposed a new plugging criterion for the tube support plate region, which in principle would allow
through-wall longitudinal cracks if they do not extend outside the support plate. The argument is that in this case a tube which ruptures will be prevented from opening up by the rather close fit to the hole in the tube support plate, and the primary to secondary flow will be limited. However, as SSPB have an ambition to avoid leakage in this region they propose a both a length and depth limit. The depth limit, 70 % of the wall thickness, was determined by subtracting the expected depth growth under one cycle, 10 % of the wall thickness, and the accuracy of the depth measurement, ±20 %. The length limit should correspond to the tube-support plate thickness.

The inspectorate’s safety analysis indicated that the length criterion was the most relevant from a safety point of view, but that it was necessary to demonstrate that the support plate would stay in place during SLB or FLB transient to support the damaged tube section. An analysis presented by SSPB indicated that the axial deflection of the lower tube support plates is limited, but that the upper support plates may undergo a significant deflection. Hence it could be expected that large openings and severe leakage from burst tubes could be prevented at the lower support plates but that this could not be guaranteed at the upper support plates. Relatively accurate crack length measurements can be achieved with multiple eddy current rotating pancake coils technique. Based on these arguments SKI has accepted that the measured depth of cracks may extend up to 70% of the wall thickness, except at the two upper tube support plates.

A matter which has emerged during the safety studies and which may require some further consideration in the future is the question whether the top tube support plate will stay in place in case of a SLB or FLB. Due to the rapid depressurization the water-steam mixture in the tube region will expand and subject the tube support plates to a lateral pressure. For the top support plate this causes the material to yield locally around some tie-rods, but according to current analyses general yield will not occur. The performed analysis is linear elastic and based on the initial nominal tube support plate dimensions and nominal weld dimensions, and no allowance is made for possible deviations from nominal dimensions.

Should the top support plate come loose and slide along the tubes, it would hit the tube bend region and possibly cause a large number of tube ruptures. If so, the integrity of individual tubes in the tube plate region becomes a matter of minor importance. It follows that a demonstration of the integrity of the top support plate should have a higher priority than that of individual tubes, when the loads during postulated SLB or FLB are considered.

**Leakage monitoring**

Leakage monitoring is considered useful for the monitoring of the overall steam generator condition. SSPB has carried out an ambitious program to improve the leakage monitoring in the Ringhals 3 and 4 plants. Still the present plugging criteria do not rely on leakage monitoring but only on a 100 % inspection of defects each outage. The reason is that in
SKI’s opinion the variation in leak rates from cracks of different sizes is too large to allow a reliable safety assessment.

It may be pointed out that although the number of defects is quite high, (Ringhals 3 has over 7500 cracks with a mean length of 3.9 mm, Ringhals 4 has 1300 cracks with a mean length of 5.1 mm), the observed leak rate for both units stays well below the reporting limit of 300 kg/day or 12.5 litres/h.

**Conclusions**

The experience gained from the revised plugging criteria so far are briefly

- the eddy current measurement technique for crack length measurement and its verification has developed significantly, and further development appears to be possible,

- the question of how corrosion cracks grow and the relevance of the measured mean growth rates are still not entirely clear,

- the importance of the integrity of the top support plate may require further analysis and possibly some kind of inspection,

- the leak rate from the comparatively large number of cracks appears to be small.
1. DEGRADATION TYPES OBSERVED IN SPAIN

In Spain four different models of steam generators are in service in seven nuclear power plants. These models are 24, D3 and F from Westinghouse and GT-54 from KWU-Siemens. The operating experience has been dependent on the various models, as well as the importance and nature of the degradations. This last due to the different features of each steam generator which are involved on the degradations, such as: tubing materials, tube expansion, chemistry of secondary water, etc.

The main causes of the degradation of the tubes has been loss of tube wall thickness by wear in tubes in contact with antivibration rods and at plate locations of the tubes in preheater sections; stress corrosion cracking from the primary and secondary sides, with preferent zones, as tube-support plate intersections and rolling transition; denting of the tubes due to corrosion of the carbon steel support plates. (although, this problem seems controled by boron addition in the affected generators) and finally, some problems involving a reduced number of plugged tubes that is atributed to a loose part or foreign object.

Generally speaking, the major contributor cause to leave out of service S.G. tubes is the stress corrosion cracking problem (SCC).

The 24 Model, the older steam generator in Spain with partial expansion tubes in the tubesheet and phosphates chemistry in the secondary water, suffers degradation at the top of the tubesheet by thinning. Also, this model which is the sole steam generator in the plant is affected by the SCC phenomenon. In the past, D3 Models were modified in their original design with respect to the feedwater inlet nozzle and manifold set up, blaming the tube fretting to a faulted design. At this moment, fretting by this reason is a forgotten problem. As above it is mentioned, the main cause to feel worried is the SCC problems from both primary and secondary sides.
The cracks are located at support plates and are axially oriented. Too, these cracks are present at the rolling expansion and transition zones. Moreover, the same mechanism of degradation has been the generator of circumferential cracks within expansion and transition regions. Some 24 Models have shown denting degradations but it is not a source of plugging, however specific inspections are still necessary to be carried out.

In the F Model, an important remark is the wearing of tubes by contacting with antivibration bars. Regarding the few operating cycles of this model severe degradation exists. However, antivibration bars substitution by optimized design should reduce sensitivity to this problem, in the near future.

Finally, steam generator model from KWU has been trouble free, since, it is not affected by the above mentioned SCC and only to point out degradation effects on few tubes due to a loose part or foreign object.

In summary, 19 steam generators are in service in Spain and their main characteristics appears on Table I.

Table II shows, in detail, the number of plugged tubes, as well as the corresponding plugging causes for each steam generator and by their respective nuclear power plants.

As one can see on this Table II, SCC problem is the first contributor cause upon the total number of the plugged tubes. In fact this phenomenon harped on the 24 and D3 models.

Figures 1 and 2 explain the percentage evolution of number of plugged tubes and the amount of plugging in each inspection outage since 1981.

An analysis of the plugging evolution versus time will take consideration of the following:

a) Scope of inspections and improved techniques which are used in Spain (inspections by ECT of 100% of the tubes and use of Rotating Probe Coil).

b) New plugging criteria (adapted P criterion and criteria based on leak before break considerations).
c) Remedies and mitigating actions (shotpeening, U bend heat treatment, sludge lancing, boron addition, sleeving, etc).

d) Design modifications (preheater, condenser materials, etc).

e) Operating incidents (primary-to-secondary leakages).

The high number of the tubes out-of-service is a large problem that is faced up with the efforts of Spanish Owner Group of PWR. In that way strong work is undergone through the same safety factors in order to evaluate tube structural integrity but remaining in service those tubes which should otherwise had been plugged.

2. THE TECHNICAL POSITION OF THE REGULATORY AUTHORITIES (C.S.N.)

In general, and from the licensing point of view, the structural criteria to be applied are dependent on the steam generator model and its specific problems. The criteria to face the structural integrity of tubes pledge three aspects, which are: inspection, plugging/repair criteria and limits for primary-to-secondary leakages.

Historically, the permitted loss of wall thickness was 40%, i.e. all tubes with any form of degradation, whether uniform degradation or cracking, should be plugged when depth of the trough-wall defect exceeds 40%, except for the 24 Model which is 50%.

Scope of inspections is according to Standard Technical Specifications and taking consideration of requeriments of the American Regulatory Guide 1.83, that is minimum of 3% of the tubes to be inspected and criteria for sample expansion.

The limit for primary-to-secondary leakages settled in Standard Technical Specifications is 78 litres/h per steam generator. Special mention is made to KWU, who limits leakages by quantifying the secondary activity.

During steam generators operation the observed degradation is the whole spectrum, as extensive as diverse. It compel to submit a set of technical aspects in relationship with tube integrity for their approval by CSN. The referred aspects were very particular and dependent on type and location of defects, and including involved characteristics of the implicated steam generator model.

In summary, at the time defects appeared in foreign or domestic plants, then, it became necessary to apply new plugging criteria in

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harmony with the extent of inspections and the limit for primary-to-secondary leakages.

However, an invariant for any plugging criteria has been the safety factors which in the American Regulatory Guide 1.121 are involved. In addition, all inspections have been carried out with larger scopes than required by the Standard Technical Specifications. It was also reduced the limit for primary-to-secondary leakages with respect to the mentioned specifications.

2.1. Current plugging criteria

2.1.1. For all steam generator models, except 24 Model, and as general application is the criterion based on the depth of defects exceeding 40% trough-wall. For 24 Model is applied 50% trough-wall because of the greatest thickness.

2.1.2. More specific is the P criterion application. This criterion allows continued operation of tubes containing defects which would otherwise necessitate plugging, providing that defects are located within a distance of 38 mm below the top of the tubesheet.

This criterion assumes that tubes will be supported by the tubesheet in spite of degradation, but only applies to circumferential defects. P criterion was approved under strong requirements; scope of the inspection is 100% tubes using bobbin coil. At the beginning was used 8 x 1 Pancake coil.

2.1.3. Another plugging criterion is based on the leak-before-break concept. It avoids the need to plug many tubes, but will accomplish some requeriments, which are:

a) The criterion can only be applied to the specific modes of degradation with recognized morphology, location and growth characteristics for which previous experience exists in nuclear reactors. From Spanish experience, one defect which meets this requeriment is primary water stress corrosion cracking. The cracks will be axially-oriented, so circumferential or complex arrays of flaws are not allowed. The maximum number of axial cracks in the tube must be \( \leq 20 \).

b) Axial cracks located within the roll transition zone and up to 18 mm. above top of tubesheet.
c) The critical crack length is 13 mm. This has been obtained by means of Folias "bulging factor" and was validated by burst testing carried out in Spain.

d) The tubes with these defects have shown, statistically speaking, crack growth rate of 4 mm, between inspections (ie. 12 months) as averaged.

e) Eddy current tests in tubes which were pulled out of Spanish steam generators have shown that crack ECT uncertainty of about 1 mm is suitable.

f) Then, considering that crack growth rate is 4 mm. and eddy current test uncertainties are about 1 mm, limit of crack length, from the point of view of plugging criterion application, is 8 mm. Hence, tubes with crack lengths ≤ 8 mm may remain in service.

g) Being the primary-to-secondary amount of leakage a tube integrity indicator, during plant operation, it is essential that leak detection methods have sensitivity, accuracy, adequate threshold, redundancy, coherence and fast response. So, in one hand, keeping in mind that predicted and measured leakages correlation is about 10, and in the other hand, that there is a lot of uncertainties in order to correlate leakages and crack length, hence suitable safety factors must then be applied and setup a value of 5 litres/hr per steam generator as limit that can be safely tolerated during operation. In some cases limit is settled to 15 litres/hr depending on the history and the extent and type of degradation.

2.1.4. It is clear that above point applies to specific defects; by that, a question is: When circumferential cracks exist at rolling transition region should a plugging criterion based on leak before break be accepted? Obviously, for application to axial cracks only, but with a modified scenario due to circumferential crack, the regulatory position was, for 24 and D3 Models, to consider the following,

a) To limit the maximum leakage that is permitted in operation for actual models which are more degraded: The limit is 5 litres/hr per steam generator above maximum stabilized leakage for 15 days, which is 5 litres/hr, too. And use of nitrogen-16 method for measuring leakages.

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b) To enlarge the scope inspections up to 100% of the tubes and using rotating probe coil.

c) All circumferential crack will be plugged without consideration of size and location at rolling transition zone.

3. THE LICENSING PROBLEMS FOR THE REGULATORY BODY

In the light of the above, licensing problems are presented in this paragraph:

3.1. The presence of circumferential cracks

Mainly this phenomenon has revealed on D3 models.

These cracks are located on the hot legs due to SCC on the outside diameter of the tubes and at or near the wall transition.

Previously, there was very poor experience in the world about that question. No indications or warning of a future occurrence was the reason for developing search of. However, latter, some indications during inspection outages were found. The genesis was not well explained and the knowledge that the cracking exists was only gained through the examination of pulled tubes. So, this degradation was not found via the eddy current inspection methods employed during former outages.

Before was written, that the circumferential cracks appear at the rolling transition, it is very important to remark that all cracks are within the tubesheet.

Under this situation, the most important problem to be dealt with was to harmonize the approved criterion for axial cracks on inside wall based on leak before break and the presence of the circumferential crack at the same zone. But it is necessary to say something more: Only a sole tube shows circumferential and axial cracks coexisting in the same place but within the tubesheet.

In consequence it was submitted a new plugging criteria to CSN for their approval. This criterion is developed by Westinghouse on the document titled "Justification for continued operation". From the analyses and experimental tests carried out by the vendor resulted that tubes with circumferential cracking at the top of the tubesheet (that is, 84% through wall and 360 degrees in extent or up to 270
degrees in extent and 100% through wall) are expected to remain stable under normal and accident conditions.

However, actual detection thres-holds for circumferential cracks, with respect to depth and length, are about 40% and 40 degrees. There-fore, any tube with axial cracking which is remaining in service using leak before break criterion could be coexisting, in addition, with circumferential cracks whose sizing is under the mentioned thres-holds.

A group of important facts, like:

a) In the world there is not data bank with re-liable circumferential crack growth rates,

b) Masking effects for detection when combined axial-circumferential cracks are present,

c) Poor experience about fatigue tube evaluations under-taking combined cracking, and

d) Missing correlation from complex cracks arrays and corresponding leakages,

are kept in mind to maintain the plugging criterion under circumferential cracks.

The plugging criterion based on leak before break concept will be applied for axial cracks as indicated but whether circumferential cracks do not exist out of the tubesheet.

On the contrary, if circumferentials cracks are within tubesheet, then, the $P^*$ criterion could be used. Moreover, for axial cracks the leak before break criterion can be applied safely.

3.2. **Fretting in AVB's: F Model**

The main problem in Spain of steam generator tubes of Westinghouse F Model is the degradation of tubes by wear caused by fretting of the tubes with antivibration bars in the U-bend zone.

This circumstance has forced to retire of service high number of tubes, by applying the criterion of 40% loss of wall thickness.
The licensee proposal to CSN has been to change initially the actual plugging limit of 40% in accordance with Technical Specifications, rising this limit up to 75% loss of the wall thickness.

This proposal should be applicable in a provisional way, since replacement of actual antivibration bar design will be performed in near future.

The new AVB's design, will optimize location so that tubes be stable if any one antivibration bar is inactive, to eliminate gaps between the tubes and antivibration bars and finally to reduce potential wall degradation rate by a factor greater or equal to 4.

The licensee submitted, for resolving the questions posed by CSN, a new criterion with a lower plugging limit. This limit should require to plug any tube with defect greater than 50% loss of wall thickness.

The regulatory authority doubts are the following:

d. By other hand, to maintain a 25% of remanent wall thickness, is as to propose a "LBB criterion" for not through-wall defects,
which is not very accepted, when the maximum leak primary-to-secondary according to Standard Technical Specifications, is 500 gallons/day.

3.3. **Axial cracks from the secondary side at support plates**

These cracks are axially oriented and are located in front of the support plates. Preferentially, these cracks were found between 2H and 5H support plates.

Flaws are complex, reaching as much as four by ECT indication.

This new defectology was detected in the ECT inspection performed, due to a leak in a D3 Model S.G. occurred in 1988, with the plant operating at 100%, which was caused by the presence of Na high concentration from BOP.

As is shown in Table 2, this type of defect which affects a high number of tubes, has caused a many tubes plugging as at present, Technical Specification related with plugging criteria in steam generator tubes with defects which are confined in support plates, compel to plug if tubewall depth penetration exceeds 40%.

Inservice inspections in Almaraz and Ascó plants (D3 Models) have revealed stress corrosion cracking on the outside diameter of tubes. These cracks have been associated mainly with support plates (TSP) or flow distribution baffles (FDB) on the hot leg, and are axially oriented. Ascó plant carried out tube removals in order to characterize the cracks and to calibrate ECT measurements. In addition, Ascó is affected by denting degradation, depth of which (90%) of them exceeds 40% plugging criterion. Hence, Asco and Almaraz, both of them, have submitted to CSN a new plugging criterion going beyond the 40% of depth and rising the value to 78%.

The new criterion, proposed by both licensees, is not free from questions by the Regulatory Authority, as are the followings:

a. The degradation that leads to a through-wall defect do not appear evident as a leak, because plastic deformation of cracks lips, being in front of support plate, obstructs the leaking path.

b. Effects from denting on tube with this type of cracks are unknown, as are effects of fatigue on the material integrity.
c. Transients and accidents effects, as LOCA and SSE, which cause relative movement of tube and support plate, can give a new setting in the presence of axial cracks.

d. As detection level is about 30% to 40% through-wall and the information is very poor (lack of data), there is not enough reliability about average depth grown rate.

e. Laboratory program results about leak measure from this type of defect with denting and accidental conditions are not conclusive.

However, in tests performed in tubes with through-thickness axial cracks (19 mm. length) and located in front of support plate, under accidental conditions like feed water line break, were obtained bursting pressures of 182.7 bar.

All these points justify that CSN is actually still evaluating this subject, being helped in the task by an USA engineering company.

3.4. **Loose-part or foreign objects in KWU steam generator**

The leak detection system located at the blowdown piping of one SG of a KWU designed plant reported a raise in secondary activity during the normal operation. So, it was supposed a leak from the primary to the secondary side was occurring.

Therefore, followings limits were fixed:

- Maximum activity on secondary side: $5 \times 10^{-3}$ Ci/m³.

- Maximum leak from primary to secondary side: 15 l/h.

The plant operated with limits above mentioned, during the rest of cycle (about 2 months).

When plant was shutdown for refuelling outage, it was realized from hydraulic tests that a tube had a small leak (pin-point type).

Eddy-current test were performed with bobbin coil and rotating probe coil (RPC), with a scope of, about 40% of tubes in SG2 (affected).

Results of eddy-current tests were as follows:
Tube R28-C102  Leaker tube with loss of thickness at 4.6 cm (hole type) above top of flow distribution baffle (01H)
Tube R29-C105  60% loss of thickness at 01H + 0,0 cm.
Tube R27-C105  76% loss of thickness at 01H + 0,25 cm.
Tube R28-C106  50% loss of thickness at 01H + 0,0 cm.

Losses of thickness were located outside of tubes, and also it was detected an outside indication in the R27-C103 tube, which was located 3,5 cm. above 01H(FDB). This indication showed a signal (metallic conduction) in contact with the outside wall tube, characteristic of ferritic material.

This gave evidence of existence of a loose-part or a foreign object with a geometry, given approximately by the configuration of the indications above mentioned.

Loose-parts monitoring system installed in the plant didn't gave any alarm during operation. Subsequently, during refuelling outage were plugged eight tubes, which envelopped the "hipothetic" piece.

The main cause of concern was her safe confinement, because during operation could make it more movable; for that reason it was analized the potencial damage that could cause to the plugged tubes.

Preservice inspection documented reports were reviewed, but any indication which could show if the piece was forgotten, during the fabrication process, were found; never-theless eddy-current inspection showed that piece is a ferritic material, therefore it is believed that is not a material coming from the steam generator.

Finally, CSN allowed the licensee for another cycle (1990-1991) with the piece left in the steam generator, provided that in next refuelling outage, the loose part or foreign object will be removed from the steam generator prior to the plant's return to service (scheduled in October 91).

3.5. **Leak determination from primary to secondary side.**

From 1989 up to now, there has been a worldwide concern for a solution about the problems of leak rates from primary to secondary side, through S.G.'s.
In this period so far, have been suggested different ways of solution by mean of methods and systems that tried to solve problems found with earlier methods.

By 1990, four spanish nuclear power plants with D3 Model S.G's., established the new method for detection of primary-secondary leakage. This method was based in the measure of N-16 formed in the primary circuit, with detectors located at the steam line outlet piping.

This is a consequence derived from the above mentioned plugging criterion based in the submitted which name is Leak before break concept.

This system allows by mean of algorithmic calculations, the determination of the possible primary-secondary leakage in four different locations: cold leg (BF), hot leg (MO), bottom of tubesheet plate (BC) and U-bend. In Spain, the application widely used is the measure at the steam generators tubesheet plate (BC).

This method gives a fundamental advantage in relation with other, (i.e. isotopic analysis of S.G's purges): is an on-line monitoring system, as any alteration and evolution in the measure is prompt and easy detected, allowing the operators to take the necessary actions.

During the system startup in some spanish nuclear power plants were detected problems that disturbed measures showedby detectors.

The problems encountered are, fundamentally, the following:

- Physical location of sensors
- Shielding of detectors
- Calibration of detectors

In relation with first of these points, and considering that nominal performance interval of the detector is between +10°C and + 55°C, the main problem has been to choose a place where the work temperature for detector was located in the range above mentioned. At the beginning this problem was more accused in one of plants with installed N-16 system.

The second problem arise when it was realized that magnitudes obtained with the N-16 detector were significantly greater than the ones obtained with other methods (i.e. isotopic analysis of S.G's purges).
Moreover, both methods didn't show any relation between their measures, this was the reason to shield the detectors. With this improvement it was obtained a significative reduction in the magnitudes of the N-16 measures.

Finally, relating with detector's calibration of N-16 and considering results showed by the calibrations performed, after to carry-out the shielding it was opted to use the same equipments for calibrations that the ones used by the supplier, in the original country (France).

Once were solved all these problems it was acceptable to license, as unique and reliable method of measure, the one based on N-16.

Figures 3 and 4, show for a period of time between December 1990 and May 1991, the results of leak measures by both methods. The values of measures obtained with N-16 method were always greater, and the relation between values remained constant (except in one SG).

4. CONCLUSIONS

a. The degradation in the Spanish steam generators is very diverse and depends on models of steam generator, fundamentally.

Actually, the more active cause and, therefore, the first contributor upon the total number of plugged tubes, is SCC from primary side and secondary side.

b. Due to above mentioned, have been licensed new plugging criteria as are $P^*$ criterion and the based on the leak-before-break concept. The new criteria have maintained the safety factors which were established in the original design.

c. New plugging criteria are still in evaluation by the Regulatory Authority due to technical aspects which are not fully demonstrated, yet.

Basically, circumferential cracks in the rolling transition and axial cracks in the tube-support plate intersection, as well fretting of the tubes against antivibration bars in the U-bend zone, are the causes that have motived the main regulatory concerns in aspects such as: detection limit, reliability of defects sizing, defects growth rates and structural integrity in the presence of fatigue and operating incidents, and finally,
reliability of leakage measure and their correlation with defects size.

d. At the other side, it is remarkable that E.C.T. scopes are in Spain, generally over those stipulated in Technical Specifications, reaching a 100% of tubes with standard bobbin coils and also, for critical zones, with rotating coils.

Moreover, limits to the primary to secondary leak rate, imposed by the Regulatory Authority as consequence of acceptance of plugging criteria (based in leak before break concept) are very low (15 lph, absolute value) when compared with those established in the original design, 78 lph.

e. It is worth to note a wide dispersion of technical criteria applied in order to guarantee the structural integrity of S.G. tubing, which is motivated by operation reasons faced up to the actual degradation level observed.
<table>
<thead>
<tr>
<th></th>
<th>J.CABRERA</th>
<th>ASCO I/II</th>
<th>ALMARAZ I/II</th>
<th>VANDELLOS II</th>
<th>TRILLO I</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>S.G. MODEL</strong></td>
<td>24 (W)</td>
<td>D3 (W)</td>
<td>D3 (W)</td>
<td>F (W)</td>
<td>KWU</td>
</tr>
<tr>
<td><strong>S.G.'s</strong></td>
<td>1</td>
<td>3</td>
<td>3</td>
<td>3</td>
<td>3</td>
</tr>
<tr>
<td><strong>TOTAL TUBES/S.G.</strong></td>
<td>2604</td>
<td>4674</td>
<td>4674</td>
<td>5626</td>
<td>4086</td>
</tr>
<tr>
<td><strong>MATERIAL</strong></td>
<td>INCONEL 600</td>
<td>INCONEL 600 MA</td>
<td>INCONEL 600 MA</td>
<td>INCONEL 600 INCOLOY 800 TT</td>
<td></td>
</tr>
<tr>
<td><strong>ROLLING</strong></td>
<td>PARTIAL</td>
<td>FULL DEPTH MECHANICAL</td>
<td>FULL DEPTH MECHANICAL</td>
<td>FULL DEPTH HIDRAULIC</td>
<td>PARTIAL MECHANICAL</td>
</tr>
<tr>
<td><strong>PREHEATER</strong></td>
<td>NO</td>
<td>YES</td>
<td>YES</td>
<td>NO</td>
<td>YES</td>
</tr>
<tr>
<td><strong>FEEDWATER INLET</strong></td>
<td>TOP</td>
<td>BOTTOM</td>
<td>BOTTOM</td>
<td>TOP</td>
<td>TOP &amp; BOTTOM</td>
</tr>
<tr>
<td><strong>2RY. CHEMISTRY</strong></td>
<td>PHOSPHATES</td>
<td>AVT</td>
<td>AVT</td>
<td>AVT</td>
<td>AVT</td>
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</table>

**TABLE I**
## DEGRADATIONS IN SPANISH S.G.'s

### NUMBER OF PLUGGED TUBES

<table>
<thead>
<tr>
<th>PLANT</th>
<th>S.G. THINNING</th>
<th>FRETTING</th>
<th>FRETTING</th>
<th>IDSCC</th>
<th>ODSCC</th>
<th>IGA</th>
<th>OTHERS</th>
<th>AVB's</th>
<th>PREHEAT</th>
<th>TOTAL</th>
<th>PERCENTAGE</th>
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<tr>
<td>J.Cabrera</td>
<td>19</td>
<td>0</td>
<td>75</td>
<td>3</td>
<td>0</td>
<td>25</td>
<td>141</td>
<td>(6.4%)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Almaraz I</td>
<td>1</td>
<td>-</td>
<td>6</td>
<td>21</td>
<td>55</td>
<td>103</td>
<td>-</td>
<td>3+1</td>
<td>219</td>
<td>(4.88%)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>-</td>
<td>5</td>
<td>-</td>
<td>171</td>
<td>174</td>
<td>-</td>
<td>11+1</td>
<td>382</td>
<td>(7.7%)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>3</td>
<td>-</td>
<td>1</td>
<td>2</td>
<td>92</td>
<td>131</td>
<td>-</td>
<td>3</td>
<td>230</td>
<td>(4.9%)</td>
<td></td>
</tr>
<tr>
<td>Almaraz II</td>
<td>1</td>
<td>0</td>
<td>1</td>
<td>7</td>
<td>130</td>
<td>190</td>
<td>0</td>
<td>10</td>
<td>335</td>
<td>(7.23%)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>0</td>
<td>3</td>
<td>1</td>
<td>103</td>
<td>30</td>
<td>0</td>
<td>1</td>
<td>135</td>
<td>(2.95%)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>3</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>22</td>
<td>15</td>
<td>1</td>
<td>2</td>
<td>49</td>
<td>(1.04%)</td>
<td></td>
</tr>
<tr>
<td>Asco I</td>
<td>1</td>
<td>0</td>
<td>1</td>
<td>1</td>
<td>153</td>
<td>187</td>
<td>0</td>
<td>7</td>
<td>349</td>
<td>(7.48%)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>2</td>
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<td>3</td>
<td>0</td>
<td>111</td>
<td>68</td>
<td>0</td>
<td>5</td>
<td>188</td>
<td>(3.9%)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>3</td>
<td>0</td>
<td>2</td>
<td>0</td>
<td>62</td>
<td>289</td>
<td>0</td>
<td>0</td>
<td>333</td>
<td>(7.1%)</td>
<td></td>
</tr>
<tr>
<td>Asco II</td>
<td>1</td>
<td>0</td>
<td>1</td>
<td>0</td>
<td>12</td>
<td>128</td>
<td>0</td>
<td>5</td>
<td>144</td>
<td>(3.08%)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>0</td>
<td>1</td>
<td>0</td>
<td>28</td>
<td>80</td>
<td>0</td>
<td>11</td>
<td>100</td>
<td>(2.14%)</td>
<td></td>
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<td>3</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>15</td>
<td>80</td>
<td>0</td>
<td>0</td>
<td>95</td>
<td>(2.03%)</td>
<td></td>
</tr>
<tr>
<td>Vandellos II</td>
<td>1</td>
<td>0</td>
<td>75</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>3</td>
<td>75</td>
<td>(1.38%)</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>0</td>
<td>69</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>6</td>
<td>66</td>
<td>(1.46%)</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>3</td>
<td>0</td>
<td>38</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>3</td>
<td>41</td>
<td>(0.83%)</td>
<td></td>
<td></td>
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<tr>
<td>Trillo I</td>
<td>1</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
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<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>2</td>
<td>2</td>
<td></td>
</tr>
</tbody>
</table>

- No tubes plugged by denting and pitting
- Tubes repaired(Sleeving) in Almaraz NPP: 103 tubes
Fig. 2

4.5-18/20
SAFETY SIGNIFICANCE OF STEAM GENERATOR TUBE DEGRADATION MECHANISMS

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AIB–Vinçotte Nuclear (AVN)
Brussels, Belgium

ABSTRACT

Steam generator (SG) tube bundle is a part of the Reactor Coolant Pressure Boundary (RCPB): this means that its integrity must be maintained. However, operating experience shows various types of tube degradation to occur in the SG tubing, which may lead to SG tube leaks or SG tube ruptures and create a loss of primary system coolant through the SG, therefore providing a direct path to the environment outside the primary containment structure. In this paper, the major types of known SG tube degradations are described and analyzed in order to assess their safety significance with regard to SG tube integrity.

RESUME

Le faisceau tubulaire des Générateurs de Vapeur (GV) fait partie de l’enveloppe de pression primaire. Ceci signifie que son intégrité doit être maintenue. Toutefois, l’expérience d’exploitation a mis en évidence divers types de dégradations de tubes GV pouvant amener des fuites ou conduire à leur rupture et provoquer à travers le faisceau tubulaire une perte de réfrigérant primaire, lequel trouve ainsi un chemin direct vers l’environnement extérieur, en-dehors de l’enceinte de confinement. Dans ce papier, les principaux types de dégradations de tubes GV connues sont décrits et analysés avec comme objectif l’évaluation de l’impact sur la sûreté d’une perte d’intégrité éventuelle des tubes affectés.
1. INTRODUCTION

Operating experience shows various types of degradation to occur in the tubing of the commercial PWR steam generators. The two dominant problems are corrosion and mechanical tube damage. Both lead at their extreme development to the failure (leak or rupture) of the tube.

The concept of "defense-in-depth" on which the whole framework of the nuclear safety regulations is built requires that the integrity of the reactor coolant boundary be maintained. The assurance thereof should be given with sufficient margin. In order that the degradation processes do not lead to reduction in safety levels below some stated values, corrective measures (plugging or repairing) are required. Cost-effective maintenance may also require plugging to prevent forced outages.

More specifically the following criteria shall be satisfied in order to ensure cost-effective maintenance (criterion 1) and safe operation (criteria 2, 3 and 4):
1. prevent leakage higher than the operational limit and subsequent forced shutdown during normal operation
2. avoid tube rupture during normal operation
3. avoid tube rupture during postulated accident conditions
4. avoid abnormal leakage during postulated accident conditions

Applying these criteria implicitly requires that the degradation mechanisms be understood with sufficient accuracy to provide the basis for developing and implementing defect management strategies.

The status of understanding of the major types of tube degradation encountered is summarized in this paper: for each degradation mode, a brief description emphasizing its main characteristics is made. It is then analyzed in order to assess the safety significance regarding steam generator tube integrity, i.e. the risk that this degradation leads to an excessive leakage, rapidly propagating failure or gross rupture. This analysis takes into account factors such as the nature of the degradation mechanism involved, the generic or isolated nature of the degradation, the degradation rate, the efficiency of the available inspection techniques.

2. DEGRADATION MECHANISMS AND THEIR SAFETY SIGNIFICANCE

Degradation mechanisms of concern in operating steam generator tubes include corrosion and mechanical damage. Although the following generally applies to all types of PWR steam generators, some particular points only address the recirculating U-bend tube steam generators.
2.1. **CORROSION**

Corrosion is the most common form of degradation affecting steam generator tubing. It includes pitting, uniform attack (wastage), intergranular attack and stress corrosion cracking. Denting, which is a degradation mechanism typical to steam generators tubes, is usually classified as a corrosion process.

**Pitting**

Pitting is a localized corrosion form resulting in small craters or holes in the inconel tubes. The most common cause of pitting is the concentration of metal ion or oxygen that sets up a local corrosion cell under a solid deposit (crevice corrosion - see intergranular attack). Another possible cause of pitting typical in heat exchangers is the mechanical rupture of the protective oxide surface layer due to an alternating stress in the tube. As the oxidized layer breaks away, pits on the tube surface are formed.

Pitting has been observed on the outside surface of the tubes in the sludge pile above the tubesheet. A tube rupture at the uppermost support plate (Indian point-3, October 1988) was suspected to be the result of pitting.

Pitting is difficult to detect with Eddy Current because of the small size of the pits and because the pits are often filled by corrosion products (copper).

The rate of penetration into the inconel by pitting may be greater by one or two orders of magnitude than for general corrosion.

**Safety significance:**

Pitting is potentially a serious degradation mechanism because it can lead rapidly to the perforation of the tube wall while inducing a small loss of material. Pitting may therefore require frequent unforeseen shutdowns due to the resulting leakage. However, the strength of the tube is not seriously weakened by pitting. Small thru-wall pits (just like short cracks - see Stress Corrosion Cracking) do not degrade the tube sufficiently to render it susceptible to rupture under normal or accident conditions.

Pitting plays an important role when combined with fatigue. The pits on the surface act as stress raisers which drastically reduce tube life under alternating stress. This is a typical case where mechanical damage and corrosion are synergistic.

**Uniform attack (Wastage)**

Wastage is a wall thinning due to the uniform attack of the outside surface. In the first PWR's, chemical treatment of the secondary water consisted in the use of sodium-phosphate to keep the pH sufficiently high to avoid generalized corrosion of the parts made of carbon steel. Acid phosphates developed by hydrolysis of the sodium phosphates can concentrate in crevice areas where they react with the Inconel to produce complex sodium-nickel phosphates.
Wastage generally occurs in the sludge pile area, above the tubesheet.

There is a general consensus that wastage can be detected and sized by bobbin probe when wall loss is higher than 10 to 20%.

Operating experience shows that propagation rate of wastage is compatible with the frequency of in-service inspection.

Safety significance:

Wastage results in a reduction in the structural capability of the tubes. Uniform thinning is the basis for the well known depth-based plugging limit (usually 40%).

Intergranular attack

Intergranular attack (IGA) is the preferential dissolution of the grain boundary regions with only slight or negligible attack of the grain matrix. The orientation of the cracks is not stress-related. IGA has been observed on the outside surface of tubes in locally restricted areas due to the presence of concentrated aggressive impurities (crevice corrosion). Bulk water impurities levels are normally low. However, in locally restricted regions the boiling process in the SG creates a thermal hydraulic mechanism for concentration of non-volatile impurities.

Concentration processes occur in crevice areas such as the gap between the tube support plate and tube or between the tubesheet and tube in some designs when the tube is only partially expanded. Concentration processes also occur within the porous sludge pile above the tubesheet.

Intergranular attack is difficult to detect and to characterize with Eddy Current testing. IGA results in a slow and progressive variation of the electric conductivity and magnetic permeability to which the bobbin coil probe with the differential technique is not very sensible. Moreover the Eddy Current probe is known to be sensitive to carbon steel tubesheet, tube support plates and deposits. Specialized Eddy Current probes such as the pancake coil probe or ultrasonic probes which are sensitive to axially or circumferentially oriented cracks and insensitive to geometrical or magnetic discontinuities seem to be inaccurate to detect IGA. The bobbin coil probe with the absolute technique allows however the detection and some estimation of the corrosion.

Due to the difficulties in characterizing IGA with Eddy Current probes, the propagation of IGA cannot be assessed accurately. However the evolution can be followed by the increase of the amplitude of the Eddy Current signal (absolute technique).

Safety significance:

IGA results in a reduction of the structural capability of the tube. The failure mode of the affected tube depends on the geometry of the corroded area and the location of the defect (fish mouth opening, bending failure mode, double-ended tube rupture). Some tests have shown that for an equal percentage of wall loss as measured by Eddy Current probe, the bursting pressure of a tube corroded by IGA is higher than for a tube affected by uniform wall thinning. Work is being performed to develop a constitutive equation relating the remaining tube integrity to the Eddy Current signal. There is no clear evidence that thru-wall IGA develops detectable leaks in normal operating condition.
Stress corrosion cracking

Stress corrosion cracking (SCC) is due to the synergistic action of a susceptible material, a chemically aggressive environment and a tensile stress field. The material fails by crack growth. The applied or residual stress required to induce SCC is usually below the yield strength.

Stress corrosion cracking of Inconel 600 may initiate either from the inner surface or outer surface of the tubing:

- Inside diameter initiated SCC (Primary Water Stress Corrosion Cracking-PWSCC) is observed at tube locations of high residual stresses resulting from the manufacturing processes. These locations include the tube portion that has been expanded against the tubesheet, the transition zone between the expanded and unexpanded part and the short radius U-bends of the innermost rows of tubes (at the tangent points between the straight portions and the bend). As SCC is a thermally activated process, cracks at roll areas have been observed so far on the hot leg side. Early experience involved mainly axially oriented cracks. However, cracks with a circumferential orientation have occurred recently, particularly at the rolling transition zone.

- Outside diameter initiated SCC (Secondary Side Stress Corrosion Cracking - SSSCC) occurs in crevice areas where impurities can concentrate, i.e. at locations where other types of corrosion (IGA or pitting) are also observed. SSSCC is often associated with IGA. SSSCC may also initiate in pits. Axially oriented cracks and more recently circumferential cracks have been detected in tube-to-support plates crevices (Hot leg side). Circumferential cracks have been observed at the expansion transition zone (Hot leg side).

Specialized Eddy Current probes (Rotating pancake coils) and ultrasonic probes have been developed in recent years to overcome the difficulties to detect cracks in the complex geometries and material combinations found in the tubesheet region. These probes have shown their ability to detect and size accurately axially or circumferentially oriented cracks. They are however less effective for the detection of SCC showing morphologies more complex than pure axial or circumferential cracking. Cracks in the U-bends are difficult to detect due to the eccentricity of the probe while travelling in the bend. The bobbin coil probe shows a capability to detect axially oriented cracks above some corrosion threshold (in terms of number of cracks and length of crack). Bobbin coil is also used to detect multidirectional SCC but cannot characterize the morphology.

The accuracy of the specialized probes and the availability of numerous data have allowed the crack propagation rate of PWSCC axial cracks in the roll transition zone to be estimated. The crack propagation kinetics of circumferential cracks are not well characterized so far. Operating experience suggests that the crack growth in the U-bends could be rapid (less than one fuel cycle): leaks have been observed in tubes which had not shown defects during the last inspection. The growth rate of multidirectional SSSCC can be estimated from the increase of the bobbin coil signal. It is believed that the growth rate of SSSCC is plant-specific.
Safety significance:

Bursting tests show that the failure mode of tubes with longitudinal or circumferential cracks is not crack propagation but plastic rupture. This demonstrates that the crack length rather than the crack depth is the relevant parameter for tube rupture and equation has been defined which describes the relationship between flaw size, tube geometry, material properties and tube bursting pressure. Tests and theory show that:

- tubes degraded by short cracks have a margin against bursting during normal and accident conditions.
- tubes with axial cracks have lower margin to burst than tubes with circumferential cracks for an identical crack length.
- the instability of a longitudinal thru-wall crack is preceded by a very significant bulging. This observation was at the basis of tests showing that for a tube in a confined space (like within the support plates) the axial crack remains stable at pressures where unstability should occur.
- the presence of multiple axial stress corrosion cracks all around the tube circumference was shown to have no effect on the behaviour of the potentially unstable main crack.
- circumferential cracks within the tube sheet sufficiently below the top face cannot result in tube severance out of the tubesheet.

When the morphology of SCC becomes complex (for instance the case of multidirectional short cracks), the use of the tube rupture equations defined for axial and circumferential cracks is no longer permitted. Work is being performed like for IGAS, to relate the remaining tube integrity to the amplitude of the Eddy Current signal.

Thru-wall stress corrosion cracks generally develop detectable leaks during normal operation. For a simple crack, tests and analytical models have allowed to develop equations predicting the leakage rate for a given crack geometry. However in the case of a widespread degradation of a steam generator by SCC, there is no reliable ratio between the leakage observed during service condition and the leakage during a design basis accident.

Denting

Denting is the plastic deformation of tubes resulting from the build-up of carbon steel support plate (or tubesheet) corrosion products in the tube-to-tube support plate (or tubesheet) crevice. These iron oxide corrosion products which have a bulk volume considerably larger than the volume of metal corroded squeeze and deform the tube. It must be noted that as denting proceeds, the tube does not deform symmetrically. As a result non-uniform strain is developed in tube wall. Denting may occur in hot leg as well as in cold leg.

Bobbin coil is usually employed to detect and size denting. However, specialized probes are also being used to plot dent profiles.

The denting growth can be considered as slow and its evolution as well controlled.
Safety significance:

Several major consequences can result from the uncontrolled progression of denting. Inside diameter cracks (PWSCC) occur at the point of highest strain. In addition denting leads in considerable distortion of the support plates and eventually to the closure of the flow slots: when this occurs at the uppermost support plate, it can induce bending and ovality at the apex of the inner row U-bends responsible for longitudinal crack (PWSCC). Denting at the top support plate was one of the combined causes having led to a tube rupture event in North Anna unit 1: denting produced a mean stress in the tube and changed the natural modes of the U-bends by imposing fixed-end conditions. Another (indirect) consequence of uncontrolled denting is that it can make impossible the travelling of the Eddy Current probe through the deformed section of the tube.

2.2. MECHANICAL DAMAGE

Mechanical damage to the secondary side of steam generator tubes can arise from different causes: fatigue, fretting-induced wear, loose parts.

Fatigue cracking

Fatigue cracking is a degradation process that occurs under cyclic loadings. The process of failure by fatigue consists of a crack propagation and a sudden fracture of the remaining ligament. Fatigue itself is a possible initiating mechanism for these cracks. Fatigue cracks initiate in regions of stress concentration. Other possible initiating mechanisms include localized corrosion and fretting. Mechanical damage and corrosion are often synergistic: pitting corrosion, for instance, can act as a stress raiser.

Circumferential tube cracking attributed to fatigue at the uppermost tube support plate recently occurred (North Anna unit 1 in July 87, Mihama-2 in February 91). The alternating load results from the out-of-plane deflection of the U-bends caused by flow-induced vibrations.

Safety significance:

Fatigue cracking is an insidious form of degradation. The initiating time can be very long followed by a rapid crack growth. The number of cycles to cause failure is a function of several variables such as mean stress level, alternating stress amplitude. The analyses of the two above mentioned rupture events suggests that in the early times of the crack propagation, the leak rate increases while remaining below the Technical Specification limit. Then, it increases very rapidly to values higher than the operational limit by several orders of magnitude.

Fretting induced wear

Fretting induced wear usually results from the interaction of the tubes with adjacent support structures due to tube vibration. Tube wear has been observed at the antivibration bars located in the U-bend area and at the baffle plate locations in the preheater sections of Westinghouse model D and E steam generators.
Wear is relatively easy to detect and size with bobbin coil.

Operating experience shows that wear rates tend to be low compared to the frequency of in service inspection. Figures of wear rate can be estimated from the non destructive inspection and the management of this degradation is successfully carried out.

Safety significance:

Wear results in the reduction of the structural strength of the tube. However the shape of the flaw is well known and its length is limited to the thickness of the supporting structure. Theory and tests suggest that in the case of wear with AVB (10mm thickness), the remaining tube ligament is susceptible to rupture under normal or accident conditions when some thinning is reached. However, the resulting thru-wall flaw does not become unstable. Moreover, wear is generally limited to some critical tubes, which renders the inspection much easier.

Loose parts

There have been several instances where mechanical damage have occurred as a result of foreign objects inadvertently left in the steam generators. Damage might also occur because previously plugged - but not stabilized - tubes break during operation and eventually rub against adjacent (non plugged) tubes. Tube wear due to foreign objects is limited to the peripheral tubes.

Loose parts induced wear is relatively easy to detect when it is suspected. The sizing of the affected depth is however less accurate than for fretting induced wear because the shape of the flaw is not known.

Safety significance:

Loose parts are an important concern. The severity of the damage depends on the impacted object: the length of the defect may be large and, in any case, is uncertain. The wear rate is similarly unpredictable. A long and rapidly progressing wear scar can lead to tube rupture (of the fish-mouth type) within a time less a fuel cycle.

3. CONCLUSIONS

The operational reliability and the safety of the PWR steam generators requires a sufficient knowledge of the degradation mechanisms to determine the amount of degradation that a tube can withstand and the time that it may remain in operation. They also require the availability of inspection techniques to accurately detect and characterize the various degradations.

The status of understanding of the major types of degradation summarized in this paper shows and justifies why efforts are being performed to improve the management of the steam generator tube defects.
STEAM GENERATOR TUBES RUPTURE PROBABILITY ESTIMATION -
STUDY OF THE AXIALLY CRACKED TUBE CASE

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ABSTRACT

The objective of the present study is to estimate the probability of a steam generator tube rupture due to the unstable propagation of axial through-wall cracks during a hypothetical accident. For this purpose the probabilistic fracture mechanics model was used taking into account statistical distributions of influencing parameters. A numerical example is presented, studying the change of rupture probability with different assumptions focusing mostly on tubesheet reinforcing factor, crack propagation rate and crack detection probability.
1 INTRODUCTION

Primary Water Stress Corrosion Cracking (PWSCC) is one of the main degradation mechanisms affecting mill annealed Inconel 600 steam generator tubes. Generally the most affected region is the roll transition zone where multiple axial cracks are initiated from the primary side and grow stable through the tube wall to propagate out of the transition zone as through-wall cracks.

To continue safe reactor operation with known (axial) through-wall cracks, utilities in some countries rely on adapted plugging criteria and extensive use of advanced NDE techniques. Criteria based on crack length recognize that a cracked tube has a margin against bursting and requires that all tubes with cracks greater than allowable length be detected and plugged (repaired).

The goal of the crack length criteria is to ensure with adequate safety margins that the loadings during operation will not initiate unstable crack propagation. These criteria derive from an analytical deterministic model taking into account the mean or extreme values of statistically distributed parameters, thus implicitly assuming that variabilities and uncertainties contribute to a small (but not evaluated) probability of the rupture.

When plants are operating with a large number of through-wall cracked tubes, the concern of Steam Generator Tube Rupture (SGTR) is raised. More specifically, the probability of a SGTR during hypothetical accidental conditions should be assessed. The results of the analysis may be used at the later stage to evaluate if the expected SGTR probability may be tolerated and to validate the crack length criteria.

One of the generally accepted techniques to evaluate failure probability of cracked components is probabilistic fracture mechanics (PFM). Some basic PFM relations applied to stress corrosion cracking in steam generator tubing [1] are used as the basis for the analysis presented in the paper.

2 MATHEMATICAL MODEL

Assuming that the steam generator tube rupture can be described by a Poisson's process, the probability $Q$ of having one or more tubes failed in a steam generator tube bundle containing $M$ cracks is given, for large values of $M$, by:

$$Q = 1 - \exp(-M \cdot Q_1)$$  (1)

$Q_1$ being rupture probability of a tube containing only one crack. General approach to determine the rupture probability of a component containing single crack [2] requires the solution of $k$-dimensional integral of probability density functions $f(x_1), \ldots, f(x_k)$:

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\[ Q_1 = \int \cdots \int_D f(x_1) \cdots f(x_k) \, dx_k \cdots dx_1 \] (2)

\[ D \] representing the rupture domain in \( k \)-dimensional space. In our case, \( Q_1 \) can be obtained by solving:

\[ Q_1 = \int_0^a f(a) \int_0 f_c(2a_c) \, d(2a_c) \, da \] (3)

\( f(a) \) and \( f_c(2a_c) \) being probability densities of observed and critical crack lengths, respectively. \( f(a) \) is defined from the in-service inspection of the tube bundle. \( f_c(2a_c) \) is determined through semi-empirical rupture model, from the probability densities of different parameters as described below. Equation (3) can be solved numerically using known techniques [2].

2.1 Critical Crack Length

The critical axial through-wall crack length calculation is based on the ductile fracture mechanics model, originally proposed in [3] and experimentally verified in [4]. By inverting the bulging factor correlation [3] and assuming the Poisson ratio to be 0.3, the critical crack length is obtained from:

\[ a_c = [-0.709 + 1.155 \cdot m - 7.056 \cdot \exp(-2.966 \cdot m) \cdot \sqrt{R \cdot t}] \] (4)

where

\[ m = \frac{\sigma_f}{\sigma_m} = \frac{K \cdot (\sigma_Y + \sigma_m)}{p \left( \frac{R}{t} - 0.5 \right)} \] (5)

The probability density \( f_c(2a_c) \) is then easily determined from probability densities of influencing parameters, considered as statistically independent variables (eq. (5)).

2.2 Tubesheet Reinforcement Effect

The tubesheet provides additional circumferential rigidity in the transition zone. This effect was observed in [5] and later quantified in [4] proposing the Tube Sheet Reinforcing Factor, defined as correction coefficient to the flow stress factor \( K \) (see eq. (5)):

\[ RF(a_c) = 1 + 10 \cdot \exp \left( -1.8 \cdot \frac{a_c}{\sqrt{R \cdot t}} \right) \]  (6)

4.7-3/11
RF can reach value of 1.3 for very short cracks and is vanishing with crack length. Unfortunately, its use is restricted only to the cracks tangent to the tube sheet [4] which may not be true for numerous cracks propagating from the tubesheet.

### 2.3 Observed Crack Length

When steam generator in-service inspection is completed, the probability density of measured crack lengths can be estimated [4], [6]. Tubes with cracks equal or exceeding the plugging limit PL are removed from service or repaired.

Plugging limits [4] are based on the observed mean or extreme values of the parameters including sizing accuracy and crack length propagation rate. The use of such limits should ensure that at the end of in-service inspection interval there is no crack longer or equal to PL. However there is a non-zero probability of having these parameters exceeding the observed values. Another consideration is the possibility of missing some cracks during in-service inspection. As a consequence, some of the cracks might be left in operation after the in-service inspection and actually exceed the plugging limit before the end of cycle. The model describing this behavior is summarized in:

\[
\begin{align*}
    a &= a_\varepsilon + \begin{cases} 
        a_m + a_\varepsilon, & a_m + a_\varepsilon < PL \\
        a_m + a_\varepsilon, & a_m + a_\varepsilon \geq PL \text{ and } \zeta > P_{od} \\
        0 & \text{otherwise}
    \end{cases} 
\end{align*}
\]

(7)

where \( P_{od} \) represents the probability of crack detection.

### 3 Numerical Example

Application of the proposed model is illustrated by a numerical example considering a typical steam generator seriously affected by axial stress corrosion cracking in the roll transition area. The probability of a steam generator tube rupture is calculated for the most unfavorable hypothetical conditions (Feedwater line break).

Two cases are considered. In case A, the parameters defining the assumed probability densities are calculated from the samples of the available test data [7]. They are coherent with the values used to define the plugging criteria [8]. This should give a point estimate of the probability \( Q_1 \) (see eq. (3)) consistent with the assumptions of the bases for plugging criteria [8].

For case B statistical re-analysis of data was performed in order to estimate with a given confidence level (95% in most cases) the most unfavorable values of parameters. Such choice of parameters should result in an upper bound for the probability \( Q_1 \) (see eq. (3)) consistent with the available data.
For each of two cases, the effect of the tubesheet reinforcing factor $RF$ (see eq. (6)) is studied by performing two analyses, differing only by considering the $RF$ or not.

### 3.1 Data Summary

The statistical properties of data used in analyses were summarized in Table I for case A and Table II for case B. The parameters which were taken as deterministic are given in Table III. In both cases, the same initial (as measured) crack length distribution was assumed (see Figure 1).

#### Table I Summary of data for case A

<table>
<thead>
<tr>
<th>Variable</th>
<th>Distribution</th>
<th>Parameters</th>
<th>Unit</th>
<th>Comment</th>
</tr>
</thead>
</table>
| $R_{out}$ | Normal | $\mu = 11.11$  
$\sigma = 0.0313$ | mm | tolerance $\pm 3\sigma$ |
| $t^1$ | Normal | $\mu = 1.195$  
$\sigma = 0.0423$ | mm | tolerance $\pm 3\sigma$ |
| $a_m$ | Gamma | $\alpha = 11.4$  
$\beta = 1.5$ | mm | assumed |
| $a_e$ | Normal | $\mu = 0$  
$\sigma = 0.75$ | mm | accuracy $\pm 2\sigma$ |
| $a_g$ | Gamma | $\alpha = 1.0$  
$\beta = 0.8$ | mm | Prediction model [4] |
| $K$ | Normal | $\mu = 0.51$  
$\sigma = 0.03$ | - | Experimental data [7] |
| $\sigma_Y + \sigma_M$ | Normal | $\mu = 945$  
$\sigma = 50$ | MPa | |

$^1$ Uniform thinning of 0.075 mm assumed.

Crack propagation law, used in case A does not consider the effect of initial crack length, although such dependence was observed in [4]. However, it predicted greater propagation rate than model used in case B for all propagation rates greater than 2 mm (see Figure 2).
### Table II Summary of data for case B

<table>
<thead>
<tr>
<th>Variables</th>
<th>Distribution</th>
<th>Type</th>
<th>Parameters</th>
<th>Unit</th>
<th>Comment</th>
</tr>
</thead>
<tbody>
<tr>
<td>( R_{out} )</td>
<td>Normal</td>
<td>( \mu = 11.11 ) &lt;br&gt; ( \sigma = 0.0313 )</td>
<td>mm</td>
<td>tolerance ( \pm 3\sigma )</td>
<td></td>
</tr>
<tr>
<td>( t^1 )</td>
<td>Normal</td>
<td>( \mu = 1.195 ) &lt;br&gt; ( \sigma = 0.0423 )</td>
<td>mm</td>
<td>tolerance ( \pm 3\sigma )</td>
<td></td>
</tr>
<tr>
<td>( a_m )</td>
<td>Gamma</td>
<td>( \alpha = 11.4 ) &lt;br&gt; ( \beta = 1.5 )</td>
<td>mm</td>
<td>assumed</td>
<td></td>
</tr>
<tr>
<td>( a_e )</td>
<td>Normal</td>
<td>( \mu = 0 ) &lt;br&gt; ( \sigma = 0.75 )</td>
<td>mm</td>
<td>accuracy ( \pm 2\sigma )</td>
<td></td>
</tr>
<tr>
<td>( a_g )</td>
<td>One-sided normal</td>
<td>( a_g ) ( \langle 1.5 ) &lt;br&gt; ( \sigma = 3.37 ) &lt;br&gt; ( 1.5 \leq a_g \leq 2.5 ) &lt;br&gt; ( \sigma = 2.36 ) &lt;br&gt; ( 2.5 \leq a_g \leq 3.5 ) &lt;br&gt; ( \sigma = 1.84 ) &lt;br&gt; ( 3.5 \leq a_g \leq 4.5 ) &lt;br&gt; ( \sigma = 1.47 ) &lt;br&gt; ( 4.5 \leq a_g ) &lt;br&gt; ( \sigma = 1.16 )</td>
<td>mm</td>
<td>operational data</td>
<td></td>
</tr>
<tr>
<td>( K )</td>
<td>Normal</td>
<td>( \mu = 0.47 ) &lt;br&gt; ( \sigma = 0.04 )</td>
<td>-</td>
<td>Experimental data [7]</td>
<td></td>
</tr>
<tr>
<td>( \sigma_Y + \sigma_M )</td>
<td>Normal</td>
<td>( \mu = 945 ) &lt;br&gt; ( \sigma = 50 )</td>
<td>MPa</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

1 Uniform thinning of 0.075 mm assumed.

### 3.2 Solution Method

A direct Monte Carlo method was used to solve eq. (3). To obtain \( Q_1 \) with resolution of the order of magnitude \( 10^4 \) it was necessary to perform \( 10^7 \) numerical experiments. \( Q_1 \) is then obtained by:

\[
Q_1 = \frac{N_{\text{all}}}{N_{\text{exper}}} \tag{8}
\]
Figure 1 Measured crack length distribution

Figure 2 Comparison of crack propagation laws
Table III Deterministic data

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Case A</th>
<th>Case B</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Value</td>
<td>Comment</td>
</tr>
<tr>
<td>p</td>
<td>178.5 bar</td>
<td>assumed</td>
</tr>
<tr>
<td>$P_{\text{od}}$</td>
<td>1.</td>
<td></td>
</tr>
<tr>
<td>RF</td>
<td>See eq. (6)</td>
<td></td>
</tr>
<tr>
<td>PL</td>
<td>14 mm</td>
<td></td>
</tr>
</tbody>
</table>

3.3 Results

The single tube rupture probabilities $Q_i$ for all four cases are presented in Table IV. Consideration of the number of cracks in the tube bundle is shown in Figure 3, as expressed by eq. (1).

![Rupture Probability](image)

Figure 3 Number of cracks and tube rupture probability

As expected, case B gives higher rupture probabilities than case A (see Table IV). However, the ratio between them remains below 1.17 and 2.31 with or without $RF$, respectively.

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This may be due to the fact that case B was more conservative than case A in treating flow stress coefficient and probability of crack detection but less in describing crack propagation rate.

<table>
<thead>
<tr>
<th>Table IV Probability of single tube rupture ($Q_r$)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Case A</td>
</tr>
<tr>
<td></td>
</tr>
<tr>
<td>With Reinforcing Factor</td>
</tr>
<tr>
<td>Without Reinforcing Factor</td>
</tr>
</tbody>
</table>

The reinforcing factor $RF$ decreases rupture probability by approximately an order of magnitude for both cases. Further consideration of the tubesheet reinforcing effect is thus of primary importance. The effect of the tubesheet reinforcement may be more accurately examined by means of sensitivity analysis, which is the suggested topic for further work.

Following the curves on Figure 3 and assuming case B without tubesheet reinforcement the rupture probability will exceed 50% (5%) if 730 (54) cracks were detected. Assuming case A with tubesheet reinforcement the same number of cracks will yield probability below 4.7% or 0.36% respectively. At this point it should be noted again that above mentioned rupture probabilities are conditional. A hypothetical Feedwater line break accident was assumed as initial event.

4 CONCLUSIONS

Rupture probabilities of axially cracked steam generator tubes can be estimated by means of probabilistic fracture mechanics model. Point and upper bound estimates have been calculated for a typical steam generator seriously affected by primary water stress corrosion cracking in the roll transition zone.

Additionally, the effects of tubesheet reinforcing factor and number of cracks on rupture probability were studied. The results showed safety significance of both parameters, suggesting the need for detailed sensitivity analysis.

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5 NOMENCLATURE

\(a\)  
predicted end of inspection interval crack length

\(a_c\)  
critical crack half-length (theoretical prediction)

\(a_e\)  
crack sizing error

\(a_g\)  
crack propagation between two consecutive inspections

\(a_m\)  
measured (in-service inspection) crack length

\(D\)  
tube rupture domain in \(k\)-dimensional space

\(f_r(2a_c)\)  
probability density of critical crack lengths

\(f(a)\)  
probability density of end of inspection interval crack lengths

\(f(x_i)\)  
probability density of \(i\)-th parameter \((i=1...k)\)

\(K\)  
flow stress factor

\(M\)  
number of cracks in the tube bundle

\(N_{\text{exper}}\)  
number of numerical experiments (Monte Carlo)

\(N_{\text{fail}}\)  
number of numerical tube ruptures (Monte Carlo)

\(p\)  
differential pressure

\(PL\)  
plugging limit

\(P_{od}\)  
probability of crack detection

\(Q\)  
probability of tube bundle failure (at least one tube rupture)

\(Q_t\)  
probability of single tube rupture

\(R\)  
tube mean radius \((R_{\text{out}} - 0.5t)\)

\(R_{\text{out}}\)  
tube outer radius

\(RF(a_c)\)  
tubesheet reinforcing factor

\(t\)  
tube wall thickness

\(\sigma_f\)  
flow stress

\(\sigma_m\)  
membrane stress

\(\sigma_y\)  
yield stress

\(\sigma_M\)  
ultimate tensile stress

\(\zeta\)  
uniformly distributed random variable

6 REFERENCES


SESSION 5 - ANALYSIS AND PREDICTION OF DEGRADATION MECHANISMS

5.1 STUDY OF THE EFFECT OF MAINTENANCE ON THE SAFETY OF A MECHANICAL SYSTEM SUBJECT TO AGING - APPLICATIONS TO THE DEGRADATIONS OF STEAM GENERATOR TUBES

D. Dussarté, CEA/IPSN, Fontenay-aux-Roses, FRANCE

5.2 PREDICTION MODELS FOR THE PWSCC DEGRADATION PROCESS IN TUBE ROLL TRANSITIONS

P. Hernalsteen, Belgatom/Laborelec, Brussels, BELGIUM

5.3 SAMPLING INSPECTION SCHEMES AND STEAM GENERATOR TUBE RUPTURE PROBABILITY

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B. Mavko, "Jožef Stefan" Institute, Ljubljana, Slovenia, YUGOSLAVIA

5.4 PROBABILISTIC FRACTURE MECHANICS CODE FOR PWR STEAM GENERATOR TUBE MAINTENANCE

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P. Pitner, EDF - Direction des Etudes et Recherches, FRANCE
T. Riffard, EDF - Direction des Etudes et Recherches, FRANCE

5.5 THE EFFICIENCY OF PEENING ON INITIATION AND PROPAGATION OF PWSCC IN ROLL TRANSITION

J. Stubbe, Belgatom/Laborelec, Brussels, BELGIUM
P. Hernalsteen, Belgatom/Laborelec, Brussels, BELGIUM

5.6 AN ANALYSIS OF PRIMARY WATER STRESS CORROSION CRACKING IN PWR STEAM GENERATORS

P.M. Scott, Framatome, Paris La Défense, FRANCE
STUDY OF THE EFFECT OF MAINTENANCE ON THE SAFETY OF A MECHANICAL SYSTEM SUBJECT TO AGING - APPLICATIONS TO THE DEGRADATIONS OF STEAM GENERATOR TUBES.

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ABSTRACT

The various degradations encountered in the exchanger tube bundles of Pressurized Water Reactors can lead to fear an increasing sensitivity to risk of fracture of tubes of steam Generator.

Preventing such an accident is based in particular on the implementation of inspections during the plant shutdown and on the application of criteria for the withdrawal of faulty tubes from service.

The definition of such a surveillance programme implies that the efficiency of adopted procedures can be quantified.

The aim of the present study is to propose a methodology of quantification for the effects of the preventive maintenance which was developed by us to solve this issue and consequently to have at our disposal some elements to assess the actions done by the utility.

RESUME

Les différentes dégradations observées sur les faisceaux tubulaires des réacteurs à eau pressurisée peuvent laisser craindre un accroissement sensible du risque de rupture de tube de générateur de vapeur.

La prévention d'un tel accident repose notamment sur la mise en œuvre de contrôles lors des arrêts des tranches et l'application de critères de mise hors service des tubes défectueux.

La définition d'un tel programme de surveillance suppose que l'on soit en mesure de chiffrer l'efficacité des actions entreprises.

L'objet de l'étude présentée dans cette note est de proposer une méthodologie de quantification des effets de la maintenance préventive que nous avons été amenés à développer pour répondre à ce problème et par voie de conséquence pour disposer d'éléments d'appréciation sur les actions entreprises par l'exploitant.
I. INTRODUCTION

Les données tirées de l'expérience d'exploitation des réacteurs à eau pressurisée font clairement apparaître une usure prématurée des faisceaux tubulaires de certains générateurs de vapeur due essentiellement à un phénomène de corrosion sous tension.

L'examen des différentes dégradations observées sur les tubes de générateur de vapeur révèle l'existence de tubes susceptibles de se rompre brutalment sans présenter auparavant de fuites décelables en temps voulu par la surveillance en service.

Pour de tels tubes qui sont typiquement ceux présentant des défauts échappant aux hypothèses des études dites de fuite avant risque de rupture, la prévention du risque repose entièrement sur la surveillance effectuée lors des arrêts.

La définition de ce programme de surveillance dont l'objectif est de maintenir le risque de rupture de tube de générateur de vapeur à un niveau requis (objectifs de sûreté) suppose que l'on soit en mesure de chiffrer l'efficacité des actions entreprises.

L'objet de l'étude présentée dans cette note est de proposer une méthodologie de quantification des effets de la maintenance que nous avons été amenés à développer pour répondre à ce problème.

Notons que ce modèle de prédiction peut être étendu à des matériels autres que les générateurs de vapeur et s'appliquer d'une façon générale à des systèmes mécaniques sujets à un phénomène de vieillissement.

II. DESCRIPTION GENERALE DE LA METHODOLOGIE

II.1 Hypothèses-données

Les systèmes mécaniques considérés sont constitués d'un nombre fini $N$ d'éléments fonctionnant simultanément.

Nous supposons que le phénomène de vieillissement est prématuré vis-à-vis de la fin projetée du système : les dégradations qui en résultent apparaissent nécessairement avant le terme de la durée de fonctionnement spécifiée $\theta$.

En l'absence de tout contrôle, ce phénomène conduit, en moyenne, à la défaillance d'une fraction $q$ des $N$ éléments du système, lesquels sont qualifiés de "sensibles" ; on pourra noter que la valeur $(1-q)$ représente la fiabilité intrinsèque de chacun des éléments du système à l'instant $\theta$.

Les défaillances prises en compte dans l'analyse sont de type progressives c'est à dire qu'elles sont précédées d'une période au cours de laquelle il est possible de suivre par le biais des contrôles l'évolution dans le temps de la dégradation.

Le temps à la défaillance, $t$, des éléments sensibles est supposé obéir à une loi de distribution normale de paramètres $m$ et $\sigma$.

Pour prendre en compte l'influence de la politique de surveillance, ce temps est décomposé en deux intervalles $t_d$ et $t_p$ ($t = t_d + t_p$) avec :

- $t_d$ : instant relatif au seuil de déetectabilité de l'endommagement,
- $t_p$ : intervalle de temps séparant cet instant de celui de la défaillance.

Etant donné que la variable aléatoire $t$ obéit à une loi de type Gaussien, les deux variables $t_d$ et $t_p$ supposées indépendantes obéissent à des lois de même nature de paramètres respectifs $(m_d, \sigma_d)$ et $(m_p, \sigma_p)$.

Tout contrôle exercé entre les instants $t_d$ et $t_d + t_p$ conduira par hypothèse à des actions correctives permettant de restaurer définitivement la fiabilité de tout élément sensible détecté.
Nous nous plaçons dans le cas de figure où les contrôles exercés dans le cadre de la politique de maintenance préventive sont effectués périodiquement, par rotation (sondages tournants), sur tout ou partie de l’équipement.

En notant $\Delta$ et $f$, respectivement la périodicité des contrôles et la fraction des éléments contrôlés à chaque visite, la période de rotation $T$ du système, c’est à dire l’intervalle de temps séparant deux contrôles successifs d’un élément quelconque de celui-ci, s’exprime par :

$$ T = \Delta / f. $$

II.2 Résultat fondamental

Notons $\tilde{f}(t)$ la densité de probabilité de la variable aléatoire temps de vie d’un élément sensible.

Si l’on n’exerce pas de contrôles préventifs, la fonction de répartition en fin de vie, $F$, d’un tel élément, c’est à dire sa probabilité de défaillance entre les instants $0$ et $\theta$, est égale à l’unité :

$$ F = \text{prob} \ (t < \theta) = \int_{0}^{\theta} f(t) \, dt = 1 $$

La mise en œuvre d’actions correctives aura pour effet de rabaisser cette valeur et nous avons donc étudié la dépendance de $F$ vis-à-vis des 6 paramètres de l’étude :

$$ F = F (m_d, \sigma_d, m_p, \sigma_p, \Delta, f) $$

Au terme d’une approche analytique, validée par un ensemble de calculs numériques (simulations de Monte-Carlo), nous avons pu établir les résultats suivants :

- la valeur de $F$ est insensible au choix des paramètres $m_d$ et $\sigma_d$ définissant le seuil de déetectabilité,

- la donnée des deux paramètres $\Delta$ et $f$ peut se ramener en fait à la donnée du seul paramètre $T = \Delta / f$ correspondant à la période de rotation,

- de façon plus précise, la fonction $F$ ne fait intervenir que les deux variables réduites $T/m_p$ et $\sigma_p/m_p$ notées respectivement $T^*$ et $C^*$.

Cette dépendance exprime un résultat essentiel que l’on pressentait a priori, à savoir que le phénomène est régi par une "compétition" entre la cinétique de l’endommagement et le temps séparant deux contrôles successifs d’une zone donnée du système.

En assimilant la loi Gaussienne à une fonction de type hyperbolique

$$ \left( \int \frac{1}{\sqrt{2\pi}} \exp(-x^2/2) \, dx - \int \frac{1}{2\sinh^2 x} \, dx \right). $$

$F$ peut être exprimée analytiquement sous la forme :

$$ F = F (T^*, C^*) = \left[ \text{th} \left( \frac{1}{C^*} \right) + \frac{C^*}{T^*} \ln \frac{\text{ch} \left( \frac{T^* - 1}{C^*} \right)}{\text{ch} \left( \frac{1}{C^*} \right)} \right] / \left[ 1 + \text{th} \left( \frac{1}{C^*} \right) \right] $$
Une étude détaillée des variations de cette fonction permet de dégager une règle pratique simple :

quelle que soit la valeur du paramètre C*, l'effet de la maintenance apparaît extrêmement sensible dès lors que T* devient inférieur à 1,5 environ ; dans la pratique on pourra retenir qu'un gain notable en fiabilité nécessitera la mise en œuvre d'une politique de contrôle telle que la période de rotation soit du même ordre de grandeur que le temps moyen séparant l'instant relatif au seuil de détectabilité de celui de la défaillance (T - m_p).

Le graphe 1 illustre pour des valeurs de C* comprises entre 1/6 et 1/2 l'ensemble de ces propriétés.

Afin de généraliser l'étude nous avons ensuite examiné les effets de la maintenance sur d'autres caractéristiques de sûreté telles que le taux de défaillance et la densité de défaillance.

Au terme d'une étude purement numérique où nous avons été amenés à évaluer l'effet de la maintenance pour tout instant courant, une propriété remarquable a pu être dégagée ; elle peut être énoncée comme suit : sous l'effet des contrôles, la loi de défaillance intrinsèque de type Gaussien est transformée en une loi de même nature, de paramètres quasiment identiques (écart type inchangé, moyenne très légèrement diminuée), mais dont la norme passe de l'unité à F(T*, C*).

Soit schématiquement :

\[
\tilde{f}(t) = \frac{1}{\sqrt{2\pi} \sigma} \exp\left(-\frac{(t - m)^2}{2\sigma^2}\right) \Rightarrow f(t) = \tilde{f}(t + \delta) F(T^*, C^*)
\]

Le décalage \( \delta \), estimé de façon empirique, s'exprime sous la forme :

\[
\delta = \min \left[ \frac{\sigma_p}{T^*}, 2\sigma_p \right]
\]

A noter que les fonctions de répartition "transformée" et "intrinsèque", notées respectivement F(t) et \( \tilde{F}(t) \), sont naturellement liées par la relation :

\[
F(t) = \tilde{F}(t + \delta) F(T^*, C^*)
\]

Il résulte de cette propriété que l'évaluation de l'effet de la maintenance sur toutes les caractéristiques de sûreté d'un système se réduit à l'étude de la seule fonction F(T*, C*) dont la forme analytique a pu être établie.

A titre d'exemple et pour illustrer la validité du modèle, nous avons joint les graphes 2 et 3 sur lesquels on peut observer une très bonne concordance entre les résultats obtenus par simulation et ceux établis à partir du modèle de transformée de la Gaussienne.
II.3 Étude de la sûreté d’un système mécanique

Le problème qui se pose dans la pratique est de définir le programme de maintenance préventive, caractérisé par un temps de rotation $T$, susceptible de maintenir à un niveau requis la sûreté d’un système entre l’instant de sa mise en service et le terme de sa durée de fonctionnement $\theta$.

Nous supposerons dans toute la suite de l’exposé que les paramètres $N$ et $q$ sont tels que l’utilisation de la loi de POISSON pour déterminer le nombre d’éléments sensibles est légitime ($N > 10$, $q \leq 0.1$) ; à noter que la méthode peut prendre en compte des situations autres que celles envisagées ici et pour lesquelles il est donc nécessaire d’utiliser la loi binomiale (cas général).

L’objectif de sûreté que l’on se fixe peut consister à :

- garantir un niveau minimal à la fiabilité du système en fin de vie (probabilité de n’observer aucune défaillance ou plus généralement moins de $k$ défaillances parmi les $N$ éléments),
- limiter la valeur maximale prise au cours du temps par la densité de défaillance ou le taux de défaillance etc...

Compte tenu des résultats établis précédemment et des hypothèses retenues, il est relativement aisé de vérifier que le problème posé se ramène à la résolution d’une équation du type :

$$\eta F(T^*, C^*) = F_0$$

Le coefficient $\eta$ vaut $Nq$ si la caractéristique de sûreté ne fait pas intervenir l’écart type $\sigma$ (fiabilité du système en fin de vie) et $\frac{Nq}{\sigma}$ dans le cas contraire (densité de défaillance, taux de défaillance, etc ...).

$F_0$ est directement lié à l’objectif de sûreté que l’on se fixe a priori.

L’expression analytique de la fonction $F$ étant connue, la solution du problème c’est à dire la détermination de la période de rotation $T$ est alors aisément obtenue au moyen d’une méthode numérique classique (résolution d’une équation transcendante).

Il importe de souligner que l’utilisation de cette méthodologie permet de traiter d’autres aspects que celui qui vient d’être évoqué ; dans ce qui suit nous nous limiterons à indiquer les résultats essentiels de quelques unes des analyses complémentaires qui peuvent être entreprises.

- Étude d’une association de systèmes mécaniques

Dans la pratique, la résistance au vieillissement n’est pas forcément identique en tous les points d’un système (chargements variables, cinématiques d’évolution des défauts différentes d’une zone à une autre, ...).

Tous les résultats établis précédemment sont facilement transposables au cas de tels systèmes qui peuvent toujours être décomposés en une association d’un nombre fini $K$ de systèmes homogènes ; la démarche est la suivante :

- dans le cas d’un système homogène, nous avons vu que le paramètre définissant une caractéristique de sûreté s’exprimait par le produit de la caractéristique intrinsèque par le terme correctif $F(T^*, C^*)$, soit $\eta F(T^*, C^*)$.
- On montre que dans le cas d’un système hétérogène, ce même paramètre sera simplement obtenu en sommant les paramètres $\eta_i F(T_i^*, C_i^*)$ relatifs à chacun des $K$ systèmes.

Pour de tels systèmes, des règles d’optimisation des contrôles ont pu être dégagées ; le problème posé consiste à déterminer $K$ temps de rotation rendant maximale la sûreté du système hétérogène étant donné un volume fixé de contrôles ou minimisant un volume de contrôles compte tenu d’un objectif de sûreté à atteindre.
Quel que soit le problème d'optimisation posé, on montre que les K inconnues, \((T_1^*, T_2^*, \ldots, T_K^*)\), sont liées entre elles par \((K - 1)\) relations simples du type: \(\omega_1 T_1^* = \omega_2 T_2^* = \ldots = \omega_K T_K^*\), où le coefficient \(\omega_i\) est un paramètre connu du système homogène (i). La solution au problème (existant toujours et unique) est alors obtenue en joignant ces \((K - 1)\) relations à celle définissant la contrainte.

- **Étude du prolongement du fonctionnement d’un système**

Dans l’étude d’un système mécanique homogène nous faisions l’hypothèse que les dégradations apparaissaient nécessairement avant la durée de vie spécifiée \(\theta\).

Lorsque l’on rejette cette hypothèse et que l’on considère donc les cas où les problèmes de vieillissement surviennent à une échéance de l’ordre de grandeur de la durée de fonctionnement escomptée (assimilée alors à une variable), la solution \(T^*\) peut être estimée à partir d’une forme analytique du type:

\[
T^* = \frac{\alpha}{\theta^* - \theta_0^*} + T_{\infty}^*, \text{ définie pour } \theta > \theta_0^*,
\]

avec:

- \(\theta^*\) : variable réduite \(\theta - \frac{m}{\sigma}\),
- \(T_{\infty}^*\) : valeur asymptotique mentionnée plus haut correspondant à un phénomène de vieillissement prématuré,
- \(\theta_0^*\) : durée de fonctionnement "seuil", pouvant être estimée analytiquement, qui dépend du niveau de sûreté visé, de la caractéristique de sûreté considérée et du nombre moyen \(N_q\) d’éléments sensibles,
- \(\alpha\) : coefficient dont la valeur estimée empiriquement est systématiquement faible (de l’ordre de 0.1 à 0.2).

**III. APPLICATIONS AUX DEGRADATIONS DES TUBES DES GENERATEURS DE VAPEUR**

**III.1 Définition du problème**

L’hypothèse de base de cette étude est que l’on ne peut pas exclure de façon catégorique, la présence éventuelle, au sein de faisceaux tubulaires de certaines tranches, de tubes échappant au concept de fuite avant rupture ; la seule façon de prévenir les défaillances de ces composants "sensibles" est de les identifier lors des opérations de maintenance préventive prévues dans le cadre des programmes de contrôle périodique.

Le problème qui se pose est de déterminer les caractéristiques des contrôles périodiques (périodicité des contrôles, taux de sondage) susceptibles de maintenir le risque de Rupture de Tube de Générateur de Vapeur (R.T.G.V.) en troisième catégorie ; au plan de la réglementation FRANÇAISE, cet objectif consiste à vérifier que les valeurs prises par la densité de défaillance, \(f_s(t)\), au cours de la vie de la tranche n’excèdent pas \(10^{-2}/(\text{réacteur } x \text{ an})\). Il importe de noter que ce choix correspond à la valeur maximale compatible avec la troisième catégorie et revient donc à n’autoriser que des valeurs négligeables des risques pour les autres causes évolutives de R.T.G.V., tel celui par exemple résultant d’une réaction inadaptée de l’opérateur à une situation de fuite primaire - secondaire.

Vis-à-vis du risque de rupture de tube de générateur de vapeur, la fiabilité d’un réacteur à l’instant \(t\), \(R_s (t)\), est définie comme la probabilité de n’observer sur celui-ci aucune rupture de tube dans l’intervalle de temps \([0, t]\), soit :

\[
R_s (t) = e^{-N_qF(t)}.
\]
Par définition, la probabilité instantanée de défaillance (ou densité de défaillance) s’exprime :

\[ f_d(t) = \frac{d}{dt} \left( 1 - R_s(t) \right) = Nq \frac{dF(t)}{dt} e^{-NqF(t)} \]

Le maximum pris au cours du temps par cette fonction est sous la seule dépendance des paramètres

\[ T^* = T/m_p, \ C^* = \sigma_p/m_p, \ \sigma = \sqrt{\sigma_d^2 + \sigma_p^2} \] et du nombre Nq correspondant au nombre moyen d’éléments sensibles peuplant la tranche.

Du fait de l’absence de données expérimentales se rapportant aux durées de vie des composants considérés dans cette analyse, nous ne pouvons pas appréhender d’une façon précise, ni même approchée, les valeurs des paramètres \((m_d, \sigma_d), (m_p, \sigma_p)\).

Compte tenu de ces incertitudes importantes, la démarche à suivre est de considérer ces paramètres, non pas comme des valeurs figées, mais comme des variables aléatoires ; il s’agit là d’une méthode classique de propagation des erreurs entachant les paramètres d’entrée.

Par ailleurs, la complexité du phénomène de vieillissement est telle qu’il est actuellement impossible d’établir une liste exhaustive des facteurs responsables de l’apparition de configurations échappant au concept de fuite avant rupture : le nombre moyen Nq d’éléments sensibles est donc un paramètre que nous ferons varier.

Pour des valeurs fixées des paramètres \(T, m_p, \sigma_p/m_p, \sigma \) et Nq, Max \( f_d(t) \) peut être estimé numériquement au moyen du modèle de prédiction décrit précédemment.

Compte tenu de la démarche entreprise (méthode de propagation des erreurs), le maximum de la densité de défaillance est en fait une variable aléatoire.

Ainsi, pour que l’accident rupture d’un tube puisse être classé en troisième catégorie (objectif à atteindre), il conviendra donc de vérifier que les valeurs pouvant être prises par Max \( f_d(t) \) n’excèdent pas \( 10^{-2} \) avec un niveau de confiance élevé, soit :

\[ \text{prob} \left[ \begin{array}{c}
\text{Max} \ f_d(t) < 10^{-2} \\
t
\end{array} \right] \geq 1 - \varepsilon , \]

où \( \varepsilon \) correspond à un risque résiduel.

La démarche adoptée pour répondre au problème posé consistera à rechercher, pour différentes valeurs de \(T\), la valeur maximale du nombre de tubes sensibles ou plus précisément la valeur maximale du paramètre Nq, pour laquelle les critères de troisième catégorie sont respectés.

Les lois de distribution associées aux variables aléatoires \((m_d, \sigma_d)\) et \((m_p, \sigma_p)\) ou plus précisément \((m_d, (m/\sigma)_d)\) et \((m_p, (m/\sigma)_p)\) sont de type Log - Normales (choix habituellement retenu dans ce type de démarche). L’algorithme de résolution se présentera sous la forme d’une simulation de Monte-Carlo (incontournable du fait de la méthode de propagation des erreurs) couplée à un calcul numérique.

Afin de s’assurer d’un taux de confiance convenable et de placer l’étude dans un cadre réaliste, nous avons été amenés à adopter pour ces lois des intervalles relativement larges.

Toutefois, il nous a paru nécessaire, afin d’augmenter la crédibilité de l’étude, de procéder à une étude de sensibilité sur les paramètres de ces différentes lois dans le but de connaître la réponse du modèle.

Comme on pouvait s’y attendre intuitivement, la réponse du modèle n’est sensible qu’au choix des paramètres relatifs à la loi de distribution de la variable la plus influente, à savoir \(m_p\).
Trois configurations ont été examinées :

- une configuration "optimiste" où en moyenne les valeurs prises par $m_p$ sont élevées (valeur moyenne = 9 cycles de fonctionnement [1 cycle = 1.5 ans]),
- une configuration "moyenne" où les valeurs de $m_p$ sont en moyenne de l'ordre de 6 cycles,
- une configuration "pessimiste" où en moyenne ces mêmes valeurs sont faibles (valeur moyenne = 3 cycles).

Les caractéristiques de ces 3 lois de distribution sont mentionnées dans le tableau I.

S'agissant enfin du niveau de confiance associé au respect des critères de troisième catégorie, nous avons considéré les cas où $(1 - \varepsilon)$ valait 90 %, 95 % et 99 %.

### III.2 Résultats

Sur les graphes 4, 5 et 6 nous avons consigné les variations du temps de rotation $T$ en fonction du paramètre $N_q$ ou plus précisément du logarithme décimal de $N_q$ ($\log(N_q)$) pour des niveaux de confiance, $1 - \varepsilon$, respectivement égaux à 90 %, 95 % et 99 % ; sur chacun de ces graphes, nous avons reporté les résultats relatifs aux différentes configurations étudiées, dites "pessimiste", "moyenne" et "optimiste".

De l'examen de l'ensemble de ces représentations graphiques, il ressort les tendances et propriétés suivantes :

- pour tous les cas de figure considérés, les courbes présentent le même type de comportement :
  
  . nous remarquons systématiquement l'existence d'une asymptote verticale ; en effet $T$ tend vers l'infini lorsque $N_q$ tend vers une certaine valeur "seuil" ($N_q_o$, qui semble dépendre de la configuration et du niveau de confiance retenu,
  
  . lorsque l'on s'éloigne de cette dernière valeur, les graphes présentent rapidement une forme aplatie ; on peut en effet observer une variation de $T$ par rapport à $N_q$ qui tend à se réduire très sensiblement lorsque ce dernier paramètre augmente. Il est important d'ajouter que lorsque $N_q$ croît indéfiniment, les calculs font apparaître - comme on pouvait s'y attendre intuitivement - que $T$ tend vers 0 ; à noter que cette propriété peut être facilement retrouvée de façon analytique,

- pour un niveau de confiance fixé, l'influence du choix de la configuration sur l'évolution de $T$ est d'autant plus importante que $N_q$ est faible ; dès lors que $N_q$ est supérieur à quelques unités, on ne relève en effet que de faibles décalages entre les 3 courbes,

- ce même type de tendance reste valable lorsque l'on s'intéresse à l'influence du niveau de confiance pour une configuration donnée.

Pour décrire au moyen d'une forme analytique ce comportement qui présente en fait les caractéristiques d'une décroissance de type hyperbolique, nous nous sommes naturellement attachés à une forme du type :

$$T\text{(cycles)} = \frac{\beta}{\log\left(\frac{N_q}{N_q_o}\right)}$$

où les deux coefficients $\beta$ et $(N_q)_o$ dépendent, a priori, du type de configuration étudié et du niveau de confiance retenu.

Pour tous les cas étudiés, ces deux coefficients ont été estimés d'après les résultats obtenus numériquement par une méthode d'ajustement par moindres carrés.
Le très bon accord quantitatif observé systématiquement entre les fonctions estimées et les résultats numériques nous permet de retenir ce schéma mathématique.

Dans les tableaux II, III et IV sont mentionnées les estimations des coefficients pour les différents cas étudiés.

Nous retiendrons essentiellement de cette étude une règle simple qui peut s'enoncer comme suit :

- si le paramètre Nq est inférieur à une valeur "seuil" (Nq)_o, dépendant du type de configuration adopté et du niveau de confiance retenu, le maintien en troisième catégorie du risque de R.T.G.V. ne nécessite pas la mise en œuvre d'une politique de maintenance préventive ; il est important de souligner que, en tout état de cause, (Nq)_o est très faible (toujours < à 10⁻¹),

- lorsque Nq est supérieur à (Nq)_o, le maintien en troisième catégorie de ce même risque nécessite la mise en place d'une politique de maintenance préventive dont la caractéristique T peut être estimée au moyen d'une formulation analytique simple et de la donnée de Nq.

III.3 Compléments d'analyse

Dans l'analyse présentée précédemment, la formule analytique liant le temps de rotation T au paramètre Nq est établie en faisant l'hypothèse que la maintenance préventive est exercée de façon homogène sur l'ensemble des faisceaux tubulaires : le temps de rotation T est supposé identique pour tous les tubes de générateur de vapeur.

D'une façon générale, on peut être amené dans la pratique à contrôler les différentes parties d'un système avec des temps de rotation différents : la décision de privilégier certaines zones d'un système peut résulter d'une crainte de rencontrer sur celles-ci une sensibilité au phénomène de vieillissement - et donc un risque de défaillance - bien supérieur à celle que l'on peut envisager sur les autres parties d'un tel système.

Dans de tels cas, nous nous trouvons en présence de systèmes mécaniques présentant des hétérogénéités vis-à-vis de la tenue au vieillissement et à la politique de maintenance exercée.

Nous nous proposons, dans ce qui suit, de rechercher des règles analogues à celles précisées en III.2, visant donc à dégager les caractéristiques de la maintenance préventive susceptibles de maintenir en troisième catégorie le risque de R.T.G.V., lorsque l'on s'intéresse à un système hétérogène.

Un tel système peut être décomposé en une association de K systèmes homogènes ; chaque système (i) est défini par les caractéristiques suivantes :

- Niqi : nombre moyen d'éléments sensibles (paramètre de POISSON),
- Ti : temps de rotation des contrôles.

Il est relativement aisé de montrer que dans le cas d'un tel système, les règles de maintien en troisième catégorie du risque de R.T.G.V. s'exprimeront sous une forme rigoureusement identique à celle précisée en III.2, le temps de rotation T devant être remplacé par un temps de rotation équivalent T_eq ; le paramètre Nq correspondra quant à lui à la somme des différents paramètres Niqi, soit :

\[
T_{eq} = \beta \frac{\beta}{\log (Nq)} \frac{(Nq)}{(Nq)_o}
\]
\[
\begin{align*}
N_q & = \sum_{i=1}^{K} N_i q_i \\
T_{eq} & = \frac{\sum_{i=1}^{K} N_i q_i T_i}{\sum_{i=1}^{K} N_i q_i}
\end{align*}
\]

A noter que les coefficients \((Nq)_o\) et \(\beta\) sont bien évidemment inchangés par rapport à ceux mentionnés en III.2.

Il importe enfin de souligner, à titre indicatif, que lorsque l'on étudie une association de systèmes (découpage du faisceau tubulaire en plusieurs zones) avec mise en œuvre d'une méthode de propagation des erreurs, on peut utiliser, sous certaines conditions, les résultats évoqués en II.3 relatifs aux règles d'optimisation des contrôles.

**IV. CONCLUSION**

L'étude développée dans cette note traite des effets de la maintenance préventive sur la sûreté d'un système mécanique atteint d'un phénomène de vieillissement.

Une application essentielle de ce modèle consiste à définir les caractéristiques du programme de maintenance préventive susceptibles de maintenir en troisième catégorie le risque de rupture d'un tube de générateur de vapeur d'un faisceau tubulaire d'un réacteur à eau pressurisée.

Nous rappelons que cet objectif consiste à vérifier que la probabilité de rupture d'un tube intégrée sur une année n'excède pas \(10^{-5}\) par réacteur.

Le résultat essentiel de cette analyse est l'établissement d'une forme analytique simple liant le temps de rotation des contrôles \(T\) (exprimé en cycles de fonctionnement [1 cycle \(- 1.5\) ans]) au nombre moyen \(Nq\) de tubes susceptibles de se rompre pour lequel les critères de troisième catégorie sont respectés ; cette forme analytique s'exprime :

\[
T = \frac{\beta}{\log (\frac{Nq}{(Nq)_o})}
\]

où \(\beta\) est un coefficient voisin de 4, et où \((Nq)_o\) de l'ordre de \(5 \times 10^{-2}\), est indicatif d'un "nombre moyen seuil". En d'autres termes cela signifie que si \(Nq\) est inférieur à \((Nq)_o\), il n'est pas nécessaire d'exercer une maintenance préventive \((T \sim \infty)\).
TABLE I

PARAMETRES DE LA LOI DE DISTRIBUTION DE $m_p$

<table>
<thead>
<tr>
<th>configuration</th>
<th>$m_p(5%)$ (cycles)</th>
<th>$m_p(95%)$ (cycles)</th>
<th>mode (cycles)</th>
<th>médiane (cycles)</th>
<th>moyenne (cycles)</th>
</tr>
</thead>
<tbody>
<tr>
<td>&quot;pessimiste&quot;</td>
<td>1.37</td>
<td>5.5</td>
<td>2.25</td>
<td>2.75</td>
<td>3.0</td>
</tr>
<tr>
<td>&quot;moyenne&quot;</td>
<td>2.75</td>
<td>11.0</td>
<td>4.5</td>
<td>5.5</td>
<td>6.0</td>
</tr>
<tr>
<td>&quot;optimiste&quot;</td>
<td>4.1</td>
<td>16.4</td>
<td>7.0</td>
<td>8.2</td>
<td>9.0</td>
</tr>
</tbody>
</table>
### TABLE II

**ESTIMATION DES COEFFICIENTS \( \beta \) ET \((Nq)_0\)**

\(1 - \varepsilon = 90\%\)

<table>
<thead>
<tr>
<th>Configuration</th>
<th>(\beta)</th>
<th>((Nq)_0)</th>
</tr>
</thead>
<tbody>
<tr>
<td>&quot;optimiste&quot;</td>
<td>4.57</td>
<td>(8.51 \times 10^{-2})</td>
</tr>
<tr>
<td>&quot;moyenne&quot;</td>
<td>4.05</td>
<td>(5.37 \times 10^{-2})</td>
</tr>
<tr>
<td>&quot;pessimiste&quot;</td>
<td>2.13</td>
<td>(4.79 \times 10^{-2})</td>
</tr>
</tbody>
</table>
TABLE III

ESTIMATION DES COEFFICIENTS $\beta$ ET $(Nq)_0$

$1 - \varepsilon = 95 \%$

<table>
<thead>
<tr>
<th>Configuration</th>
<th>$\beta$</th>
<th>$(Nq)_0$</th>
</tr>
</thead>
<tbody>
<tr>
<td>&quot;optimiste&quot;</td>
<td>4.42</td>
<td>$6.53 \times 10^{-2}$</td>
</tr>
<tr>
<td>&quot;moyenne&quot;</td>
<td>3.64</td>
<td>$4.79 \times 10^{-2}$</td>
</tr>
<tr>
<td>&quot;pessimiste&quot;</td>
<td>1.77</td>
<td>$4.47 \times 10^{-2}$</td>
</tr>
</tbody>
</table>
### TABLE IV

**ESTIMATION DES COEFFICIENTS $\beta$ ET $(Nq)_0$**

$1 - \varepsilon = 99\%$

<table>
<thead>
<tr>
<th>Configuration</th>
<th>$\beta$</th>
<th>$(Nq)_0$</th>
</tr>
</thead>
<tbody>
<tr>
<td>&quot;optimiste&quot;</td>
<td>3.88</td>
<td>$4.68 \times 10^{-2}$</td>
</tr>
<tr>
<td>&quot;moyenne&quot;</td>
<td>3.02</td>
<td>$3.63 \times 10^{-2}$</td>
</tr>
<tr>
<td>&quot;pessimiste&quot;</td>
<td>1.36</td>
<td>$3.55 \times 10^{-2}$</td>
</tr>
</tbody>
</table>
GRAPHE 1

Risque de défaillance (en fin de vie) en fonction du temps de bouclage adimensionnel - résultats analytiques.

$(mp/sp) = 2, 4, 6$
risque de défaillance (en fin de vie) en fonction du temps de bouclage adimensionnel.
 résultats numériques et analytiques (mp/sp)=4
GRAPHE 3

étude du maximum de la dérivée ; résultats numériques et analytiques ; $(mp/sp)=4$.

Résultats analytiques
ÉTUDE DU MAXIMUM DE LA DERIVEE AVEC PROPAGATION DES ERREURS (NB MOYEN DE TUBES "SENSIBLES") ;
DETERMINATION DU TEMPS DE BOUCLE T VERIFIAN UN LES CRITERES DE TROISIÈME CATÉGORIE .
NIVEAU DE CONFIANCE =0,90 - PARAMÈTRES DE PROPAGATION "OPTIMISTES", "MOYENS" ET "PESSIMISTES" .
ETUDE DU MAXIMUM DE LA DERIVEE AVEC PROPAGATION DES ERREURS (NOX NOYEN DE TUBES "SENSIBLES")
DETERMINATION DU TEMPS DE BOUCLEGE T VERIFIANT LES CRITERES DE TROISIEME CATEGORE.
NIVEAU DE CONFIANCE =0.95 - PARAMETRES DE PROPAGATION "OPTIMISTES", "MOYENS" ET "PESSIMISTES".

Diagramme: 3 courbes représentant différentes configurations optimiste, moyenne et pessimiste en fonction de l'erreur (NO) et du nombre de cycles (T).
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PREDICTION MODELS FOR THE PWSCC DEGRADATION PROCESS IN TUBE ROLL TRANSITIONS

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ABSTRACT

A huge data base has been generated in Belgium by the large scale Eddy Current Rotating Pancake Coil inspections performed since 1985 on several units affected by Primary Water Stress Corrosion Cracking in the tube roll transitions.

The statistical analysis of these data resulted in a prediction model of the steam generator degradation process, which is now used to support the maintenance policy and the longer term repair/replacement strategy.

The same model contributes also to the probabilistic assessment of the primary to secondary leak rate in case of accidental loading of the tube bundle.

The paper outlines the main features of the two prediction tools, which have been developed into the software programs LABOGROW and LABOLEAK.

RESUME

Une banque de données très importante a été constituée en Belgique suite aux inspections par sonde tournante à Courants de Foucault, effectuées à grande échelle depuis 1985 sur plusieurs unités affectées par des fissures de corrosion sous tension dans les transitions de mandrinage.

L'analyse statistique de ces données a généré un modèle prédictif du processus de dégradation des générateurs de vapeur ; ce modèle est utilisé dans le cadre de la maintenance préventive et en support de la stratégie de réparation/remplacement des générateurs de vapeur.

Il contribue également à l'évaluation probabiliste de la fuite primaire vers secondaire, en cas de sollicitation accidentelle du faisceau tubulaire.

La présente communication décrit les caractéristiques principales de ces deux outils prédictifs qui ont été intégrés dans les logiciels LABOGROW et LABOLEAK.

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1. INTRODUCTION

The tube bundle of Pressurized Water Reactor Steam Generators (SG) has been affected by numerous corrosion damages, all over the world. One of the main problems is Primary Water Stress Corrosion Cracking (PWSCC) in the roll transitions of Mill Annealed Inconel 600 tubes, mechanically expanded in the SG tubesheet (ref.[1]). Multiple axial cracks are initiated from the primary side and grow rapidly through the wall; they further grow in length and propagate outside of the roll transitions.

In most plants, both in Europe and in the USA, shot-peening has been performed on the inside diameter of the expanded section of susceptible tubing. While the compressive surface layer induced by peening is usually considered to be efficient in preventing crack initiation, field experience showed that it did not prevent preexisting cracks from further propagation (ref.[2]). For the usual case of SG peened after crack initiation, there is thus a remaining concern about the long term evolution of the population of cracked tubes.

Based upon the European practice (and the similar trend in the USA - ref.[3]), through wall deep (TWD) cracks are acceptable below a maximum length (called "plugging limit") which still provides adequate safety margins against tube bursting (ref.[4]). Tubes reaching the "plugging limit" must be either plugged or repaired (sleeving).

Another safety related limitation may result from the SG primary to secondary leak rate in case of a postulated large secondary side accident, such as Feed Water Line Break (FWLB). Because of the larger associated differential pressure (about 180 bar), the normal operation leak rate would considerably increase (mainly because of the widening of the throughwall cracks) and could lead to an excessive radioactivity release to the plant environment. This consideration may result in more tube plugging/repair than dictated by the structural "plugging limit".

Both limitations may lead to significant operational constraints; SG replacement may even be considered as an economical alternative to repair. It is thus of prime importance to be able to predict the SG degradation process and the accident leakage in order to support both the maintenance policy and the longer term repair/replacement strategy.

2. SG DEGRADATION PREDICTION MODEL

A SG degradation prediction tool should preferably be based on a sound physical model of PWSCC propagation in the tube material.

Physically based models have been proposed for crack initiation, such as the "damage model" where crack initiation is predicted to occur when the damage
\[ d = \int_0^t (\varepsilon(t))^n \, dt \]

expressed as a time function of the strain rate \( \varepsilon \) reaches a critical value \( d_{\text{crit}} \) characteristic of the material.

However there is no general agreement between authors and the inaccuracy of the proposed parameters results in unreliable predictions.

The situation is still worse for the case of crack propagation, where the lack of knowledge about the contributing factors results in an enormous scatter of the few available laboratory data.

Thus the development of a prediction model of SG degradation could not be based on physical principles but was aimed at an analytical approximation involving a minimum number of empirical parameters derived from a thorough analysis of the inspection data base.

3. **INSPECTION METHODOLOGY**

The length sizing of defects cannot be achieved by the conventional "bobbin coil", Eddy Current Testing (ECT) in service inspection of SG tubes but requires a more elaborate Rotating Pancake Coil (RPC) ECT technique. The RPC methodology developed by LABORELEC has been used, since 1984, for the systematic inspection of representative samples, of about 500 tubes, from all affected SGs (ref.[5]). When it became evident, in 1987, that shot-peening (performed in 1985 and 86) did not halt crack propagation and that the maximum length was approaching the plugging limit, the inspection hardware and software were improved to allow an accurate and efficient inspection of all tubes (over the 150 mm length of interest) without impairing the normal outage duration. Since 1988, such 100 % RPC inspections are now routinely performed by Laborelec on all SGs of two Belgian plants.

Typical data produced include:

- Crack length distribution curves
- Crack length increase distribution curves, by comparison of data from two successive inspections
- Crack propagation data, sorted as a function of initial crack length.

The extensive amount of accurate RPC data was the basis for the design of the statistical prediction model presented in the following paragraph.
4. **BASE STATISTICAL MODEL**

The prediction methodology was developed along the following principles:

- an available (reference) distribution of crack lengths is approximated by a simple analytical curve $y_n(x)$
- the distribution of crack length increases, between two prior consecutive inspections, is approximated by another analytical curve $z(\delta)$, assumed to be constant (as long as the operational parameters are kept unchanged)
- an assumption needs then to be made about the combination algorithm to be used for the propagation law over several operation cycles.

A stochastic combination, where the propagation rate of a crack during any cycle is independent of the rates observed during the previous cycles, appeared to fit the experimental data.

Hence the propagation law over 2 consecutive cycles is given by

$$\delta$$

$$z_2(\delta) = \int_0^\delta z(t) z(\delta-t) dt$$

and the distribution curve of crack length $x$ can be obtained by either

$$y_{n+2}(x) = \int_0^x y_n(t) z_2(x-t) dt$$

or the repetitive application of

$$y_{n+1}(x) = \int_0^x y_n(t) z(x-t) dt$$

The general shape of the observed histograms suggested curve fitting by a GAMMA distribution function (as used by statisticians)

$$f_\alpha(t) = \beta^{-\alpha} t^{\alpha-1} e^{-\beta t / \Gamma(\alpha)}$$
where $\alpha$ is a shape factor and $\beta$ is a scale factor.

With $y_n(x) = f_n(x)$ and $z(\delta) = \beta e^{-\beta \delta}$ a good fit was obtained for $\beta = 0.8 \text{ mm}^{-1}$, corresponding to an average propagation of 1.25 mm per cycle, and a family of curves (Figure 1) was produced, illustrating the progressive time distortion and shift of the crack length distribution curves.

However the limited number of cracked tubes (less than 200) available for the inspection data histograms did not allow an accurate parameter fitting until such limitation was eventually removed by the 1988 100% inspections.

For instance curve fitting (Figure 2) of the GAMMA distribution to the inspection histogram of one particular SG with a population of 1585 cracks, yielded the following values

$$\alpha = 9.09$$
$$\beta = 1.26$$

with an excellent correlation coefficient ($R^2 = 99\%$).

Also when several SGs were analysed in this way for successive operational cycles, the $\beta$ value appeared not to be constant but to cover a significant range, as evidenced by table I hereafter.

Thus, while the adequacy of the GAMMA curve was fully confirmed, a supplementary factor was needed to account for the variability in scale factor.

5. IMPROVED STATISTICAL MODELS

The inspection data clearly indicate that the longest cracks are likely to propagate less during the next operation cycle.

A first attempt at a simple analytical modelisation resulted in the following assumption.

For a population subset of initial length $x$, a fraction $(1 - e^{-kx})$ does not propagate during the next cycle, while the propagation of the remaining part $e^{-kx}$ follows the same propagation law as considered by the base model. This "improved model" has been presented in ref [6] and [7] and proved valuable for making predictions over one or two cycles.
However two limitations were soon identified:

- the shape of the growth distribution, for a given initial length, is dependent on that length, while the model assumed a constant shape
- the longer term trend of the length distribution parameters ($\alpha$ and $\beta$) did not appear consistent with that predicted by the model; the latter feature will be further discussed under paragraph 6.2.

The origin of these divergences was clearly related to the oversimplified assumption used to model the effect of initial length on crack growth. This oversimplification was the result of the desire for an analytical formulation of the model. Thus this latter constraint was removed, allowing for a numerical (computer based) approach.

However it was maintained to keep the model as simple as possible, with a minimum number of adjustable parameters at the level of the growth mechanism. From the now very large growth data base becoming available for each individual population of a given crack length (see table II), the following assumptions appeared reasonable:

- the average growth is given by $\delta_x = a + b e^{-kx}$
- the standard deviation $\sigma$ is constant (and taken equal to 1 mm)
- the distribution is approximated by a GAMMA curve, the parameters of which are thus defined by

$$\begin{align*}
\alpha &= \frac{2}{\delta_x} \\
\beta &= \delta_x
\end{align*}$$

From there on, the previously stochastic combination laws are kept unchanged and allow to calculate

- any crack length distribution
- any crack growth distribution

From the following input data:
- an initial crack length distribution defined as either a histogram or a GAMMA curve
- a set of growth parameters $a$, $b$ and $k$
- the desired number of cycles.
From a thorough analysis of all available data it appears that 2 growth parameters can be assumed constant

\[ b \approx 3 \text{ mm/cycle} \]

\[ k \approx 0.45 \text{ mm}^{-1} \]

while the third one (a) fluctuates within a range of 0.2 to 1.2 mm/cycle in accordance with the variability effectively observed in the field, between SG and between cycles, without any clear correlation with physical or operating differences. The value of "a" for a particular cycle can in fact be easily calculated from the field measured overall mean crack growth \( \delta_m \).

If the initial crack length distribution is defined by

\[ y(x) = \beta^\alpha x^{\alpha-1} e^{-\beta x} / \Gamma(\alpha) \]

\[ \delta_m = \int_0^\infty (a + b e^{-kx}) y(x) \, dx \]

\[ = a + b / (1 + k/\beta)^\alpha \]

The values of "a" calculated in this way are given in table III for 15 combinations of SG and cycle.

A remaining effect, not yet accounted for, is the occurrence of "new" cracks, in tubes previously inspected and found to be flawless.

When the distribution curve of these cracks is considered, it appears to be reasonably approximated by the assumed result of crack growth, when the initial length is taken equal to zero, i.e.: a gamma distribution curve

\[ a = (a + b)^2 \]

with \( b = a + b \)

The number of such additional new cracks is unfortunately much more unpredictable and has shown large field variations, between 1 and 10% of total tube bundle, without any clear systematic trend.
All of the previous considerations have been incorporated into a computer software, called LABOGROW, written in APL language and implemented on a personal computer.

6. LABOGROW PROGRAM VALIDATION

The LABOGROW program has been validated by comparison of calculated simulations against actual field measurements. Some examples are given hereafter to illustrate both the short term and long term performance.

6.1. Short term validation

The following examples are taken from the 1988-89 cycle of steam generator nr 3 from the Tihange 2 plant, starting from the 1988 crack length distribution illustrated by Figure 3 (with a GAMMA curve best fit for $\alpha = 4.61$ and $\beta = 0.96$) and applicable to 40.2% of the tube bundle.

Two values need be assumed for the variable parameters

- "a": growth parameter.
- "n": number of "new" cracks (in % of total tube bundle)

These values have been taken equal to the measurements obtained at the end of cycle (1989) i.e.

\[
\begin{align*}
  a &= 0.4 \text{ mm} \\
  n &= 9.7 \% 
\end{align*}
\]

Thus the following verifications bear more on the overall consistency of all features rather than on a capability of blind prediction (which involves the additional uncertainty of selecting the "a" and "n" values within an expected range).

The verifications are illustrated by the following 7 figures generated by the LABOGROW program and compared to the corresponding field data produced by the End Of Cycle (EOC) inspection.

Figure 4 depicts the EOC length distribution (GAMMA curve approximation) for the population of preexisting cracks.

Figure 5 depicts the EOC length distribution curve (using the "histogram" option, without GAMMA approximation) for the total crack population, thus including the expected "new cracks".

Figure 6 depicts the average crack growth as a function of the initial crack length.
Figure 7 depicts the GAMMA curve approximation for the population of "new" cracks.

Figure 8 depicts the distribution of crack growth for the population of preexisting cracks.

Figures 9 and 10 depict the same distribution when limited to a given initial crack length of respectively 3 and 8 mm. The two figures are drawn to the same scale in order to emphasize the difference in distribution shape.

In all cases, it can be seen that the agreement with the field data is quite good. This establishes the validity and robustness of the model, when recalling that only 4 parameters (3 for "growth" and 1 for "new cracks") are involved, of which 2 are kept constant and the 2 others are adjusted to the particular cycle under consideration.

### 6.2. Long term validation

Starting from the same initial conditions as defined hereabove, the prediction is now extended over 3 cycles, with the following parameter values for the 2 additional cycles:

<table>
<thead>
<tr>
<th>Cycle</th>
<th>(a)</th>
<th>(n)</th>
</tr>
</thead>
<tbody>
<tr>
<td>89-90</td>
<td>0.7 mm/cycle</td>
<td>7.6 %</td>
</tr>
<tr>
<td>90-91</td>
<td>0.3 mm/cycle</td>
<td>9.1 %</td>
</tr>
</tbody>
</table>

Figure 11 depicts the GAMMA curve approximation predicted by LABOGROW for the total crack population in 1991 and compares it with the actual inspection histogram. Again the agreement is pretty good.

Longer term predictions can be synthesized as plots of the \(\alpha\) and \(\beta\) parameters characterizing the expected crack length distributions.

It has been experimentally observed that such plots give a clear differentiation between the behaviour of several SGs and can be considered as a kind of characteristic "signature". They are considered hereafter for the predicted behaviour over a 10 cycles period (starting in 1988 as for all previous examples).

Figure 11 is the plot considering only the population of preexisting cracks (no allowance for "new" cracks) and assuming the same growth parameter (0.4 mm/cycle) as for the first cycle; the plot is marked up by the field data point observed after one cycle, with a perfect fit.
Figure 12 is the plot considering the total population of cracks, on basis of the following assumptions.

<table>
<thead>
<tr>
<th>Cycle</th>
<th>a (mm/cycle)</th>
<th>n (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>0.4</td>
<td>9.7</td>
</tr>
<tr>
<td>2</td>
<td>0.7</td>
<td>7.6</td>
</tr>
<tr>
<td>3</td>
<td>0.3</td>
<td>9.1</td>
</tr>
<tr>
<td>4 to 10</td>
<td>0.6</td>
<td>5</td>
</tr>
</tbody>
</table>

These values correspond to the field data for the first 3 cycles and have been kept at an arbitrary constant level for the remaining cycles.

Again the plot is marked-up by the 3 available field data points, with a good overall agreement in trend.

There is a striking difference in the general shape of this kind of plots depending on whether the population is kept constant or not; the location within the $(\alpha; \beta)$ plane may vary from SG to SG.

7. **PROBABILISTIC ASSESSMENT OF SG LEAK RATE UNDER ACCIDENT CONDITIONS**

Operation of a SG with a large number of (close to) through-wall cracks raises a potential concern of leakage during:

- Normal operation.
- Accident conditions (the highest differential pressure being associated with a FeedWater Line Break - FWLB).

A simplified deterministic model for leakage from an axial crack in a pressurized tube is given by:

$$Q = KS \sqrt{\Delta p}$$

This illustrates the three sources of the increased leak rate $Q$ under accidental conditions:

- the differential pressure $\Delta p$ increases from about 100 to 180 bar.
- the leakage area $S$ increases, both elastically and plastically, proportionally to the crack width which (depending on length) may grow from a few microns to a few tenths of a mm.

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the flow discharge coefficient $K$, which under normal operating conditions is governed by the "tortuosity" of the intergranular leakage path, may increase from a low value of less than 0.1 to a nominal value close to 0.6.

The combined effect of the three parameters may result in a wide range for the ratio of accident to service leak rates. In addition to this, other factors influence the service leak rate in an unpredictable way:

- A large number of PWSCC cracks are very deep (> 80% TWD) but not entirely through-wall. A thin remaining ligament (in the range of 5 to 10%, depending on crack length) is likely to break under the increased differential pressure of accident conditions.
- Some cracks are made up of aligned components which, even when TWD, leak very little if at all. Under accident conditions these ligaments might fail, resulting in a single "long" crack with a large leakage area.
- Some cracks, even when rather long and through-wall, may be prevented to leak by their outside environment (such as hard sludge); the loading imposed by accident conditions might break up these leak limiting barriers.
- Tight PWSCC cracks (with TWD penetration) are easily clogged by crud or precipitates and leak considerably less than under laboratory conditions; the substantial (elastic and plastic) widening of these cracks, under accident conditions, would restore a "clean" leakage path.

Based upon these considerations, it does not appear possible to establish a reliable ratio of accident to service leakage that could be further used by extrapolating the accident leak rate from the observed service leak rate. Thus a probabilistic model was developed by LABORELEC to provide a direct assessment of SG leak rate under accident conditions.

The methodology consists in a stochastic combination of the statistical distributions assumed for each of the contributing parameters:

- Mechanical properties (YS + UTS) at operating temperature.
- Correlation coefficient between "flow stress" (FS) and mechanical properties: $K = FS/(YS + UTS)$.
- Tube wall thickness.
- Sizing uncertainty on crack length (based on RPC inspection methodology).
- Expected crack lengths at the end of cycle.

The four first distributions are approximated by Gaussian curves while the last one results from the "propagation prediction model" described in § 5. Deterministic algorithms and discrete values are further used for:

- The tubesheet reinforcing effect (TSR).
- The crack shape factor $SF = OD$ length/ID length, on the basis of an assumed linear relationship between the OD and ID crack lengths.
. The plugging limit PL.
. The dependence of the flow discharge coefficient on the average crack width.

8. THE LABOLEAK SOFTWARE PROGRAM

The methodology described hereabove has been built into the LABOLEAK software program, using a Monte-Carlo type method, written in APL language and implemented on a personal computer (ref.[8]).

As illustrated by Figure 14, the program calculates resulting probability density functions for:

. Flow stress (FS);
. Crack critical length, with and without the tube sheet effect (cl, CL);
. The expected crack population at the end of the current operating cycle, with and without the effect of ECT uncertainty (ml, ML); this is based on beginning of cycle (BOC) inspection data and the previously described propagation model;
. The outer diameter (OD) crack length resulting from the assumed shape factor SF (ML2).

The program then establishes the crack average width, for any combination of actual (ML, ML2) and critical (CL) lengths, and the corresponding flow discharge coefficient (K) matrix. The final calculation step produces the distribution of tube leak rate; consideration of the number of cracks then yields the desired distribution curve of SG leak rate.

9. LABOLEAK PROGRAM VALIDATION

The validation process was centered on the following sensitive parameters.

. Leakage area

While the elastic component is well defined by available analytical studies, the plastic component needed to be established by an experimental test program; this allowed the leakage area to be expressed as a nonlinear function of the ratio of actual to critical length of the crack.


This coefficient is derived from experimental burst tests of flawed tubes, as a function of the flaw length. International comparison of test results from various programs (Belgium, France, UK, USA, ...) indicated a very large scatter; this was shown to result from the experimental conditions, mainly the sealing arrangement needed to allow pressurization of the test specimens with TWD flaws. Recent tests
performed by LABORELEC without any seal (with a high pressure pump allowing a flow rate in excess of 20 m³/h) allowed to identify the reliable results (ref. [9]).

"Flow discharge" coefficient K.

The K dependence on crack width has been modelled in a way to simulate the relatively scarce leak rate field data (or laboratory data, when performed under representative conditions); further refinement may be needed and is currently under investigation.

10. LABOGROW AND LABOLEAK PROGRAM USES

The LABOGROW program is used to address the following fields of practical interest

- Number of tubes to plug (next or successive outages)
- Time evolution of plugging margin
- Influence of:
  - plugging limit
  - Thöt reduction
  - preventive sleeving
- Comparison of repair versus replacement strategies
- Probabilistic evaluation of SG leakrate associated with a secondary side accident.

For the latter purpose the LABOLEAK program is currently used to perform parametric and sensitivity studies of the degraded SG under accident conditions.

Results are typically printed out as illustrated by Figure 15 combining:

- A summary of all input data.
- A plot of the leak rate distribution among the various crack length categories, with indication of the total leak rate.
- A comparison plot between the expected EOC crack length distribution and the critical crack length distribution, with indication of the resulting probability of having a SG Tube Rupture, in case of a FWLB.
- The plot of the total SG leak rate distribution, allowing to select a design value for any predefined confidence level.

11. CONCLUSIONS

On basis of the statistical analysis of a large amount of accurate field inspection data, complemented by extensive laboratory testing, two computer programs (LABOGROW and LABOLEAK) have been developed for the prediction of SG
degradation and the probabilistic assessment of SG leak rate under accident conditions.

The first tool is based on a simple crack propagation model, using only 4 adjustable parameters, and provides a close fit to all available field data; it is used to support both the SG predictive maintenance and the longer term repair/replacement strategy.

The second tool is used to perform safety evaluations, to verify the adequacy of the tube plugging limit and to support the repair/replacement strategy of steam generators. This methodology has drawn international interest and, in particular, has been adopted by the Mechanism Specific Defect Management (MSDM) Committee, conducted by the U.S.A. Electrical Power Research Institute (EPRI) to support a proposal for alternate tube plugging criteria (ref. [3]); within the latter context EPRI has taken a licence of the LABOLEAK program to make it available to U.S. Utilities operating PWR plants.
<table>
<thead>
<tr>
<th>SG</th>
<th>Year</th>
<th>Indications included in length distribution</th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>all</td>
<td>preexisting</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>α</td>
<td>β</td>
<td>α</td>
</tr>
<tr>
<td>R</td>
<td>1988</td>
<td>5.0</td>
<td>1.24</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td>1989</td>
<td>7.3</td>
<td>1.55</td>
<td>9.95</td>
</tr>
<tr>
<td></td>
<td>1990</td>
<td>8.85</td>
<td>1.73</td>
<td>12.45</td>
</tr>
<tr>
<td>G</td>
<td>1988</td>
<td>4.4</td>
<td>0.92</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td>1989</td>
<td>6.55</td>
<td>1.20</td>
<td>8.6</td>
</tr>
<tr>
<td></td>
<td>1990</td>
<td>7.1</td>
<td>1.20</td>
<td>9.0</td>
</tr>
<tr>
<td>B</td>
<td>1988</td>
<td>5.5</td>
<td>0.85</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td>1989</td>
<td>7.6</td>
<td>1.09</td>
<td>8.85</td>
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<td>10.85</td>
</tr>
<tr>
<td>1</td>
<td>1988</td>
<td>3.65</td>
<td>0.77</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td>1989</td>
<td>5.5</td>
<td>1.00</td>
<td>6.85</td>
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<td>13.2</td>
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<td>5.35</td>
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<td>-</td>
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</tr>
<tr>
<td></td>
<td>1991</td>
<td>7.6</td>
<td>1.27</td>
<td>13.3</td>
</tr>
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</table>
**TABLE II**

**DOE 3 and TIHANGE 2 - ISI 1989 RESULTS**

Crack growth \( \delta \) (mm/cycle) as a function of initial length \( x \)

<table>
<thead>
<tr>
<th>UNIT</th>
<th>CYCLE</th>
<th>( x = 1 )</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
<th>6</th>
<th>7</th>
<th>8</th>
<th>9</th>
<th>10</th>
<th>11 to 13</th>
<th>TOTAL</th>
</tr>
</thead>
<tbody>
<tr>
<td>ECD</td>
<td>1988 R</td>
<td>[AVG 2.06(66)]</td>
<td>[1.63(122)]</td>
<td>[1.22(200)]</td>
<td>[0.69(272)]</td>
<td>[0.61(227)]</td>
<td>[0.43(92)]</td>
<td>[0.39(38)]</td>
<td>[0.35(20)]</td>
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<td>[1.06]</td>
<td>[0.92]</td>
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<td>[0.76(194)]</td>
<td>[0.92(88)]</td>
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<td>[0.60]</td>
<td>[0.83]</td>
<td>[0.91]</td>
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TABLE III
CRACK GROWTH PARAMETER a

calculated from \( a = \delta_m - 3/ (1 + 0.45 / \beta)^\alpha \)

<table>
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<tr>
<th>SG</th>
<th>Cycle</th>
<th>( \alpha )</th>
<th>( \beta )</th>
<th>( \delta_m ) (mm)</th>
<th>( a ) (mm)</th>
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<tr>
<td></td>
<td>89-90</td>
<td>5.5</td>
<td>1.00</td>
<td>1.2</td>
<td>0.8</td>
</tr>
<tr>
<td></td>
<td>90-91</td>
<td>8.7</td>
<td>1.35</td>
<td>0.7</td>
<td>0.5</td>
</tr>
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<td>2</td>
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<td>5.35</td>
<td>1.18</td>
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<tr>
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<td>1.24</td>
<td>1.0</td>
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</tr>
<tr>
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<td>1.77</td>
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<td>0.9</td>
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<td>1.1</td>
<td>0.6</td>
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<tr>
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<td>0.7</td>
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<td>1.09</td>
<td>1.0</td>
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</tr>
</tbody>
</table>

5.2-17/32
Figure 1

Base Model

Figure 2

Frequency Histogram

1988 inspection

- gamma curve fitting
- inspection data (1585 tubes)
Figure 4

EOC distribution of preexisting cracks

Crack length (mm)

histogram
LABOGROW
Figure 5
EDC crack length distribution
(incl. "new" cracks)
Figure 6

Average growth as a function of initial length

Initial crack length (mm)

Average crack growth (mm)

LABOGROW

field data
Figure 7

Distribution of "new cracks" lengths

[Graph showing distribution of crack lengths with histogram and curve labeled LABOGROW]
Figure 10
Distribution of crack growth
(initial crack length \(= 8 \text{ mm}\))
Figure 11

Crack length distribution after 3 cycles

histogram

LABOGROW

Crack length (mm)

5.2-27/32
Figure 13

SG signature over 10 cycles
(all cracks included)

+ field data
LABOLEAK

LABORELEC LEAKRATE
PROBABILISTIC
ASSESSMENT PROGRAM
FOR PWSCC IN TUBE
ROLL TRANSITIONS

FLOW DIAGRAM

Fig. 14

5.2-30/32
Fig.15 - LABOLEAK program: typical print out of results
REFERENCES


SAMPLING INSPECTION SCHEMES AND STEAM GENERATOR TUBE RUPTURE PROBABILITY

L. Cizelj, B. Mavko
"Jožef Stefan" Institute, Ljubljana, Slovenia

ABSTRACT

Uncertainties which are affecting the steam generator tube inspection quality are defined and analyzed in the paper. Special attention is given to sampling inspection schemes and inspection reliability and sizing accuracy. A numerical example considering a typical steam generator with assumed defect distribution (stress corrosion cracks) is presented, relying on Monte Carlo technique simulation of inspection process. The effect of detection technique unreliability is shown in terms of probability of plugging a defective tube. Finally, the effect of inspection uncertainties on the tube failure probability is estimated by the means of probabilistic fracture mechanics.
1 INTRODUCTION

The most widely performed nondestructive examination of the steam generator tubing [1] is based on a three-stage sampling inspection scheme [2]. The first stage of this inspection requires a random sample selection (3% of all SG tubes) and evaluation against acceptance criteria. In general, if the acceptance criteria are not met within the first sample, additional samples should be selected, leading up to 100% inspection (The procedure outlined in [2] considers special attention rather than 100% examination!). The tubes found to be defective during inspection are removed from service, plugged or repaired. The applicable margin which defines the allowable extent of defect is usually called plugging limit (PL). In most cases, it is based on minimum acceptable wall thickness [1].

Fulfilled acceptance criteria after inspecting a sample of tubes will in deterministic terms declare the steam generator operable. In probabilistic terms, it will only limit the population of defective tubes below certain level with defined confidence. In other words, a limited population of defective tubes may still remain in the steam generator after the inspection process. The goal of this paper is to estimate this limited population and its effect on the steam generator tube rupture probability.

This estimate can be obtained through definition and analysis of uncertainties associated with the in-service inspection process. The outlined methodology is followed by a numerical example, which in this case serves for illustrative purposes. Numerical techniques such as Monte Carlo simulation were used for this analysis.

2 PROBABILITY OF PLUGGING A DEFECTIVE TUBE

The most obvious uncertainties which may be associated with the in-service inspection and plugging process are defined and analyzed in [3] and [4]. While extensive research activities were directed to the analysis of defect detecting uncertainties and sizing accuracy of standard eddy current in-service inspection techniques, one hardly find any comment on the US NRC Regulatory guide 1.83 [2] sampling inspection scheme. Therefore, this particular sampling scheme is analyzed first, followed by a brief summary of inspection technique reliability results, which define the overall uncertainty of the complete inspection process.

2.1 Probability of inspecting defective tube

Let us conservatively assume that the defects are randomly distributed throughout the steam generator. We are therefore giving no credit to the history of observed or any other similar steam generator. In addition, perfect detection and sizing capabilities of applied inspection technique are assumed in the derivation of inspection probabilities. Only one defect per tube is also assumed in all considerations.
2.1.1 Random sampling inspection

For the purpose of this analysis the affected tubes are ascribed as either degraded (with damage below the plugging limit PL) or defective, where the tube is affected beyond the PL. We shall assume that the fractions of defective and degraded tubes in the steam generator are \( p_1 \) and \( p_2 \), respectively. The probability of finding \( k_1 \) defective and \( k_2 \) degraded tubes in \( n \) random single tube inspections is then given by:

\[
P_{k_1, k_2, n} = \frac{n!}{k_1! \, k_2! \, (n-k_1-k_2)!} \, p_1 \, p_2 \, (1-p_1-p_2)^{n-k_1-k_2}
\]  (1)

To calculate the probability that the steam generator is acceptable in the first inspection step we only have to sum the probabilities of all possible states which are acceptable according to the procedure defined in [2]. Allowing for \( D_1 \) defective and \( d_1 \) degraded tubes in a random sample of \( n_f \) tubes (see Table II) will yield the total probability of accepting the steam generator in the first inspection step:

\[
P_1 = \sum_{i=0}^{D_1} \sum_{j=0}^{d_1} p_{i, j, n_f}
\]  (2)

If the first inspection step indicates rejection, additional sample of \( n_2 - n_f \) should be inspected. The acceptance probability is therefore conditional, because only certain outcomes of the first step will trigger and at the same time enable the acceptable outcome of the second step. Summing over all appropriate outcomes this gives:

\[
P_2 = \sum_i P(1_i^{-2}) \cdot P(2_i | 1_i^{-2})
\]  (3)

Rewriting eq. (3) in terms of allowable values yields:

\[
P_2 = \sum_{i=0}^{D_1} \sum_{j=0}^{d_1} P_{i, j, n_f} + \sum_{i=0}^{D_1} \sum_{j=0}^{d_1} P_{i, j, n_f} \sum_{k=0}^{D_2} \sum_{l=0}^{d_2} P_{k, l, n_f - n_1} + \sum_{i=0}^{D_1} \sum_{j=0}^{d_1} P_{i, j, n_f} \sum_{k=0}^{D_2} \sum_{l=0}^{d_2} P_{k, l, n_f - n_1}
\]  (4)

When analyzing the third inspection step the analogous approach should be used. Additionally, the possibility of bypassing the second step should be taken into account. This may happen if the allowable values for the second step are exceeded during the first step, but the outcome still enables acceptance of the third step. Summing over all appropriate outcomes again gives:

\[
P_3 = \sum_i P(1_i^{-2}) \sum_j P(2_j | 1_i^{-2}) \cdot P(3_j | 2_j^{-3}) + \sum_k P(1_k^{-3}) \cdot P(3_k | 1_k^{-3})
\]  (5)

Rewriting eq. (5) leads to:
\[ P_3 = \sum_{i=0}^{D_1} \sum_{j=0}^{D_2} p_{i,j,n_i} \]

\[
\left[ \sum_{k=0}^{D_3-i} \sum_{l=0}^{D_4-j} p_{k,l,n_2-n_1} \sum_{m=0}^{D_3} p_{m,n_3-n_2} + \sum_{k=0}^{D_3-i} \sum_{l=0}^{D_4-j} p_{k,l,n_2-n_1} \sum_{m=0}^{D_3} p_{m,n_3-n_2} \right] + \sum_{l=0}^{D_3} \sum_{j=0}^{D_4} p_{j,l,n_3} \]

\[
\left[ \sum_{k=0}^{D_3-i} \sum_{l=0}^{D_4-j} p_{k,l,n_2-n_1} \sum_{m=0}^{D_3} p_{m,n_3-n_2} + \sum_{k=0}^{D_3-i} \sum_{l=0}^{D_4-j} p_{k,l,n_2-n_1} \sum_{m=0}^{D_3} p_{m,n_3-n_2} \right] + \sum_{l=0}^{D_3} \sum_{j=0}^{D_4} p_{j,l,n_3} \]

\[
+ \sum_{l=0}^{D_3} \sum_{j=0}^{D_4} p_{j,l,n_3} \]

\[
+ \sum_{l=0}^{D_3} \sum_{j=0}^{D_4} p_{j,l,n_3} \]

Since the third step is the last one defined in [2] the total probability of accepting the steam generator is given by:

\[ P = P_1 + P_2 + P_3 \]  \hspace{1cm} (7)

with the complementary rejection probability:

\[ Q = 1 - P \]  \hspace{1cm} (8)

Some further useful expressions may be derived from the above equations. For example, the probability of plugging no more than \( D_1 \) tubes is clearly given by \( P_1 \).

### 2.2 Detection probability

In general, the detection probability is a function of parameters which influence the inspection process. A suitable model, proposed in [4], is based on extensive comparison of ECT signals and destructive metallographic analyses. This model given in Figure 1 does not consider the false call probability.

### 2.3 ECT sizing errors

An ECT sizing error model is proposed in [4], considering the systematic influence of real defect size \( X \) and random influence of other parameters (e.g. personnel proficiency) \( e \) on the ECT reading \( Y \):

\[ Y|X = a + b \cdot X + e \]  \hspace{1cm} (9)
Figure 1 Defect detection probability

Table I Parameters of ECT sizing model

<p>| | | | | | |</p>
<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
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</tr>
</thead>
<tbody>
<tr>
<td>$a$</td>
<td>$b$</td>
<td>var($a$)</td>
<td>var($b$)</td>
<td>cov($a$, $b$)</td>
<td>var($e$)</td>
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<tr>
<td>17.459</td>
<td>0.437</td>
<td>26.799</td>
<td>0.00845</td>
<td>-0.4335</td>
<td>261.14</td>
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</table>

2.4 Probability of exceeding the plugging limit

The defect size detected by ECT (eq. (9)) may vary significantly due to the random error $e$. Consequently, some tubes with detected defects exceeding plugging limit may not be plugged due to the sizing error. We should therefore define the probability of exceeding plugging limit if a non-zero ECT reading was obtained while inspecting a tube with a defect. An appropriate model is proposed in [4] for probability of exceeding the plugging limit:

$$P_d = 1 - F\left(\frac{PL - Y}{\sqrt{\text{var}(Y|X)}}\right) \tag{10}$$

with:

$$\text{var}(Y|X) = \text{var}(a) + X^2 \cdot \text{var}(b) + 2 \cdot \text{cov}(a, b) + \text{var}(e) \tag{11}$$
and $F$ being cumulative standard normal distribution. The probability of plugging a defective tube is plotted for some different plugging limits against defect size in Figure 2.

The parameters of eq. (9) and (11) are given in Table I.

![Figure 2 Probability of plugging a defective tube](image)

2.5 Probability of plugging a defective tube

Probabilities mentioned above result from different inspection uncertainties. Probability of plugging a defective tube is a conditional probability of all above mentioned events. It is beyond the scope of this paper to derive an analytical expression for this. However, definitions and relations stated above can help setting a Monte Carlo experimental scheme, taking into account all uncertainties and defining the final probability of operating with defective but not plugged tubes.

3 MONTE CARLO SIMULATION

A Monte Carlo simulation of steam generator inservice inspection was established according to the procedure outlined in [2]. The same conservative assumptions were retained: no credit is given for any information regarding the history of observed or any other steam generators. Additionally, no false calls are considered.
3.1 Simulation logic and data

Figure 3 Monte Carlo simulation setup

5.3-7/12
First, predefined number of defective and degraded tubes (Figure 4) is distributed on a random basis over the tube map. Defect distribution in Figure 4 corresponds to the recent 100% bobbin coil inspection in the Krško NPP. Second, an inspection procedure simulation is repeatedly performed over the same distribution of defects to evaluate the probability of accepting the steam generator (eq. (7)). Basically, if the defect is found within a sample, the decision regarding further processing is made on the basis of Figure 1. If this decision requires further processing, the defect size is calculated according to eq. (9). After the inspection of the initial sample is completed, the findings are evaluated in order to stop the inspection or to continue with the next inspection step (Figure 3). A typical Westinghouse D-4 steam generator as installed in Krško NPP is considered. The sampling inspection parameters are listed in Table II. Plugging limit is set to $PL=50\%$.

![Number of defects](image)

**Figure 4** Defect size distribution

### 3.2 Results

Probabilities of accepting the steam generator are listed in Table III. Additionally, average numbers of tubes found to be degraded and defective during the inspection simulation are listed, indicating another measure of inspection quality. The probability of accepting steam generator in first three steps when using perfect detection technique (see Table III) exceeds 20\% (see eq. (7)). Employing real inspection technique as described above increases this probability to over 60\%. In other words, we can conclude that a steam generator with 62 defective tubes (see Figure 4) will be declared operable in more than half of inspections.
### Table II Sampling inspection data

<table>
<thead>
<tr>
<th>Step</th>
<th>Cumulative sample size</th>
<th>Acceptable No. of defective tubes ($\geq PL$)</th>
<th>Acceptable No. of degraded tubes ($&lt;PL$)</th>
<th>Reference</th>
</tr>
</thead>
<tbody>
<tr>
<td>i</td>
<td>$n_i$</td>
<td>$D_i$</td>
<td>$d_i$</td>
<td></td>
</tr>
<tr>
<td>1</td>
<td>138</td>
<td>0</td>
<td>13</td>
<td>C.5.b</td>
</tr>
<tr>
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<td>276</td>
<td>1</td>
<td>27</td>
<td>C.5.c</td>
</tr>
<tr>
<td>3</td>
<td>552</td>
<td>3</td>
<td>55</td>
<td>C.7.d</td>
</tr>
<tr>
<td>all tubes</td>
<td>4578</td>
<td>N/A</td>
<td>N/A</td>
<td>N/A</td>
</tr>
</tbody>
</table>

1 defective tubes found during the inspection should be plugged afterwards

Further, it is clear (Table III) that defective tubes are governing the acceptance/rejection decision process. While perfect inspection technique will pick up all the tubes with any kind of degradation if 100% inspection was performed, we can only hope that half of defective tubes will be plugged using the real inspection technique. This fact may have considerable consequences in studying the steam generator tube rupture probability.

### Table III Sampling inspection scheme results

<table>
<thead>
<tr>
<th>Step $i$</th>
<th>Probab. of accept. $P_i$ [%]</th>
<th>Perfect detection Monte Carlo</th>
<th>Real detection Monte Carlo</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Probab. of accept. [%]</td>
<td>Average No. of plugged tubes</td>
</tr>
<tr>
<td>1</td>
<td>15.23</td>
<td>14.88</td>
<td>0.00</td>
</tr>
<tr>
<td>2</td>
<td>4.39</td>
<td>4.19</td>
<td>1.00</td>
</tr>
<tr>
<td>3</td>
<td>1.37</td>
<td>1.49</td>
<td>2.85</td>
</tr>
<tr>
<td>all tubes</td>
<td>N/A</td>
<td>79.44</td>
<td>62.00</td>
</tr>
</tbody>
</table>

1 probability of rejecting steam generator after three inspection steps

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Crack length distribution (Figure 5) was reported when the bobbin coil results (Figure 4) were verified by a rotating pancake coil during the recent in-service inspection in Krško NPP. This distribution will be used to evaluate the probability of having at least one tube out of this population failed. For this purpose the probabilistic fracture mechanics model described in [5] and [6] was used. Data listed for case A in [6] is adopted, considering 3/4 inch tubing with accidental pressure of 196.5 bar. No crack length based plugging limit was applied. The resulting single tube failure probability of $1.1 \times 10^{-3}$ was obtained for a tube randomly chosen from the population on Figure 5.

\[
\text{Weibull distribution} \\
\text{shape parameter} = 4.3 \\
\text{scale parameter} = 6.7
\]

![Probability density graph](image)

**Figure 5 Crack length distribution**

Using equation (1) in [6] we can evaluate the effect of number of defective tubes left in operation on the total steam generator failure probability. From Table III we can easily obtain number of defective tubes left in operation after accepting the steam generator in certain inspection step, which yields the steam generator failure probabilities for this number of affected tubes. This value are then multiplied by values of probability of acceptance (Table III) to calculate the conditional probability of failing after accepting steam generator in certain inspection stage. Probabilities listed in Table IV are conditional, of course. Accident (feed line break) is assumed to cause accidental pressure difference in steam generator.

Various conclusions may be drawn from the failure probabilities in Table IV. However, the most valuable conclusion may be the need for improving the inspection techniques together with performing 100% inspection of tubes.
Table IV Steam generator failure probability

<table>
<thead>
<tr>
<th>Step</th>
<th>Perfect detection</th>
<th>Real detection</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Average no. of defective tubes left</td>
<td>Failure probability $\cdot 10^3$</td>
</tr>
<tr>
<td>1</td>
<td>62.00</td>
<td>9.80</td>
</tr>
<tr>
<td>2</td>
<td>61.00</td>
<td>2.72</td>
</tr>
<tr>
<td>3</td>
<td>59.15</td>
<td>0.94</td>
</tr>
<tr>
<td>all tubes</td>
<td>0.00</td>
<td>N/A</td>
</tr>
</tbody>
</table>

5 CONCLUSIONS

Random sampling inspection scheme as defined in Regulatory Guide 1.83 [2] is analyzed in the paper, considering the uncertainties which are part of the in-service inspection process. The analysis results in probability of accepting steam generator with certain number of affected tubes and additionally, the estimate of number of defective tubes left in operation after the inspection was performed. The failure probability of given tube population is estimated. Importance of inspection technique quality was shown, especially when used in conjunction with sampling inspections.

The procedure outlined in the article may be a good basis for throughout analysis of sample sizes and inspection technique quality contributions to the steam generator safety. As a possible future work, a procedure to define an optimal inspection strategy should be considered.
6 NOMENCLATURE

- $a, b, e$: ECT sizing model parameters
- $D_i$: allowable number of defective tubes in i-th inspection step
- $d_i$: allowable number of degraded tubes in i-th inspection step
- $F$: cumulative standard normal distribution
- $i, j, k, l, m, n$: different counters
- $k_1$: number of defective tubes in a sample
- $k_2$: number of degraded tubes in a sample
- $n$: sample size (general)
- $n_i$: i-th inspection step sample size
- $P$: probability (general), probability of accepting the steam generator
- $P_{el}$: probability of exceeding plugging limit
- $P_i$: probability of SG acceptance after i-th inspection step
- $PL$: plugging limit
- $p(k_1, k_2, n)$: probability of finding $k_1$ defective and $k_2$ degraded tubes in a sample of $n$ tubes
- $Q$: probability of rejecting the steam generator
- $X$: real defect size
- $Y$: ECT defect size

7 REFERENCES


5.3-12/12
PROBABILISTIC FRACTURE MECHANICS CODE FOR PWR STEAM GENERATOR TUBE MAINTENANCE

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P. Pithier - EDF - Direction des Etudes et Recherches
T. Riffard - EDF - Direction des Etudes et Recherches

ABSTRACT

Steam generators of pressurized water reactors (PWR) have experienced worldwide various types of tube degradations from corrosion and mechanical mechanisms.

In order to optimize maintenance, Electricité de France has developed a computer code based on a probabilistic fracture mechanics methodology which quantifies the effects of in service inspection on the risk of tube failure, taking into account as random variables all significant parameters (defect size distribution, detection and sizing, defect initiation and propagation; leak detection probability and leak rate calculation, critical defect sizes).

It is possible to compare the effects of various inspection schemes on the risk of failure and lifetime, in order to optimize plugging limit. The code is also a powerful tool for various investigations (validation of input data, sensitivity to input uncertainties or specific statistical models,...). The paper will focus on the development of the methodology and an application to a real case.

UN CODE BASE SUR UNE APPROCHE MECANIQUE DE LA RUPTURE PROBABILISTE POUR LA MAINTENANCE DES GV PWR

Les générateurs de vapeur des centrales à eau pressurisée (PWR) ont été, partout dans le monde, affectés par divers types de dégradation de leurs tubes, ayant pour origine la corrosion ou des effets mécaniques.

Dans le but d'optimiser sa politique de maintenance, Electricité de France a développé un logiciel informatique basé sur une approche de mécanique de la rupture probabiliste, qui quantifie l'impact des contrôles en service sur le risque de rupture d'un tube en prenant en compte, comme variables aléatoires, tous les paramètres significatifs (distribution des tailles de défauts, capacité de détection et de dimensionnement, initiation et propagation, probabilité de fuite avant risque de rupture et calcul du débit de fuite, taille critique des défauts,...).

Le modèle permet de comparer l'impact de différents programmes d'inspection sur le niveau de sûreté et la durée de vie, et d'optimiser les critères de bouchage. Le code est aussi un outil puissant d'investigations et d'études de sensibilité (validation des données ou de la modélisation statistique,...). Le papier présente le développement de la méthodologie et un exemple d'application sur un cas réel.
1. INTRODUCTION

Le faisceau tubulaire des générateurs de vapeur (GV) est une des zones les plus sensibles du circuit primaire aux risques de dégradations.

La maintenance en service du faisceau doit permettre de garantir l'intégrité des tubes dans les différentes situations de fonctionnement en associant deux objectifs principaux :
- d'un point de vue sûreté, maintenir le risque de rupture d'un tube (RTGV) à niveau très bas imposé par les règles de conception,
- d'un point de vue disponibilité, limiter les arrêts non programmés dus à des fuites primaire-secondaire dépassant les spécifications.

Ces objectifs sont atteints par la mise en œuvre d'une politique d'inspection en service associée à des critères d'obturation des tubes. Ces contrôles et les critères d'obturation sont basés sur des analyses spécifiques à la nature de chaque dégradation rencontrée. Ceci permet de garantir les objectifs précédemment cités tout en évitant un raccourcissement exagéré de la durée de vie des appareils qui résulterait de critères exagérément conservatifs.

La démarche actuelle pour justifier les critères de bouchage reste cependant basée sur une approche déterministe qui consiste à combiner les valeurs les plus pénalisantes des différents facteurs d'influence intervenant dans l'analyse, sans pouvoir nécessairement quantifier les marges vis-à-vis du risque RTGV résultant d'une évolution du programme de contrôle.

Ces constatations ont conduit EDF à engager une étude probabiliste de maintenance (EPM) qui a abouti au développement d'un code de mécanique probabiliste, appelé COMPROMIS.

Il permet de quantifier l'effet des opérations de maintenance (programme de contrôle, critère d'obturation) sur la sûreté, la disponibilité et la durée de vie des GV vis-à-vis d'un type de dégradation donnée.

Compte tenu du retour d'expérience des GV du parc EDF, la première version du code a été adaptée à la prise en compte de la fissuration longitudinale en pied de tube par corrosion sous contraintes en milieu primaire. Cet endommagement représente encore aujourd'hui une des principales causes de bouchage et conditionne la durée de vie d'une grande partie des GV les plus anciens avec faisceau en Inconel 600.

2. DESCRIPTION GÉNÉRALE DU MODÈLE

Le modèle permet de prendre en compte la variabilité des paramètres qui interviennent sur la distribution des tailles réelles des défauts et des tailles des défauts critiques (qui entraîneraient la rupture du tube).


Lors d'une utilisation du code COMPROMIS, les 5 étapes suivantes (2.1 à 2.5) sont réalisées.

2.1. Estimation de la distribution initiale des tailles réelles des défauts

Le code permet d'utiliser directement en donnée d'entrée le résultat brut d'un contrôle.

A partir d'un histogramme de tailles de défauts mesurées obtenu à l'aide d'un contrôle aléatoire par Sonde Tournante Longue (STL) de l tube sur k et en tenant compte de la fiabilité des contrôles (erreurs de mesures et probabilité de détection d'un défaut en fonction de sa taille), le code fournit une
estimation des paramètres de la loi de distribution des tailles réelles des défauts.

Neuf types de loi peuvent être ajustés et le code fournit différents indicateurs permettant de mesurer la qualité de l’ajustement entre la loi théorique et l’histogramme observé. Un exemple d’ajustement des paramètres d’une loi de Weibull est présenté figure 2.

2.2. Prise en compte de l’obturation

Cette distribution des tailles réelles des défauts est ensuite modifiée par l’obturation avant redémarage de certains tubes présentant un défaut mesuré supérieur à un seuil. Le code permet de simuler l’obturation en tenant compte de toutes les opérations de maintenance complémentaires telles que :
- contrôle systématique par STL des tubes déjà vus fissurés lors d’un contrôle précédent,
- contrôle systématique de zones particulières "à risque" du GV (la zone des boues par exemple) alors que le reste du faisceau peut n’être contrôlé que par échantillonnage de 1 tube sur k,
- réalisation d’un test hélium afin de contrôler ultérieurement par STL les tubes fuyant à l’hélium et qui sont susceptibles de présenter des grands défauts.

La figure 3 présente la simulation de l’obturation avec un critère de 13 mm lors d’un contrôle de 100 % des tubes.

2.3. Evolution de la distribution des défauts

Le code permet de simuler :
- l’apparition en service de nouveaux défauts sur les tubes encore sains lors du dernier contrôle : pour cela, on utilise la loi d’amorçage qui fournit en fonction du temps le nombre de tubes affectés,
- l’évolution en longueur des défauts existant.

Le code utilise un modèle de cinématique de propagation basé sur une relation entre la vitesse de propagation $\frac{da}{dt}$ et le facteur intensité de contrainte, $K$, du type $\frac{da}{dt} = cK^n + \epsilon$ ou un des trois paramètres $c$, $n$ ou $\epsilon$ peut être probabilisé

[3].

Une représentation de l’évolution pendant un cycle de la distribution des défauts est donnée figure 4.

2.4. Estimation de la distribution des tailles critiques

Elle est obtenue à partir des équations de défaillances [4] en prenant en compte la variabilité des différents paramètres intervenant dans ces équations :
- $K$ : facteur adimensionnel reliant la contrainte à l’écoulement,
- $t$ : épaisseur du tube,
- $De$ : diamètre extérieur du tube,
- $\delta$ : position du point de contact du tube par rapport à la plaque tubulaire,
- $Re$ : résistance élastique,
- $R_m$ : résistance mécanique.

Ceci permet d’obtenir deux distributions des tailles critiques :
- une lors de conditions normales de fonctionnement (écart de 100 bar entre la pression primaire et la pression secondaire),
- l'autre lors d'une situation accidentelle de RTE (rupture d'une tuyauterie eau) ou de RTV (rupture de tuyauterie vapeur) entraînant un écart maximal de pression de 172 bar entre le primaire et le secondaire.

La figure 5 présente les deux distributions ainsi obtenues.

2.5. Calcul de la probabilité de RTGV

Cette probabilité est obtenue en comparant la distribution des tailles réelles à différents instants (cf. 2.3) à la distribution des tailles critiques (cf. 2.4). On obtient ainsi l'évolution de la probabilité de RTGV au cours de la période de simulation.

De plus, le code permet de tenir compte de la probabilité de fuite en service des grands défauts, ce qui entraînerait un arrêt de la centrale avant la rupture du tube (critère de fuite avant risque de rupture).

L'évolution prévisionnelle du débit de fuite en service est également fournie par le code.

La figure 6 présente l'évolution de la probabilité de RTGV au cours d'un cycle de fonctionnement, en situation normale (AP = 100 bar) pour différentes politiques de maintenance.

3. UTILISATION DU CODE

La version actuelle du code, opérationnelle depuis le deuxième semestre de 1990 permet, ainsi que précisé en introduction, de traiter la fissuration longitudinale par corrosion sous tension en milieu primaire et dans la zone de transition de dudgeonnage. Ces défauts représentent près de 80 % des défauts existants.

La résolution numérique des équations a été faite, tant que ceci a été possible, par des intégrations analytiques. Cependant, des méthodes de Monte Carlo stratifié ont été utilisées lorsque les intégrations analytiques n'étaient plus possibles (détermination de la distribution des tailles critiques par exemple).

Le code a été développé sur des ordinateurs IBM 3090. Une simulation de calcul de RTGV nécessite de 10 à 150 secondes CPU selon la complexité du calcul.

Un effort important a été fait pour que le code soit d'une utilisation conviviale tant au niveau de la définition des écrans de saisies de données que des écrans d'assistance.

Sa conception modulaire doit, par ailleurs, permettre une adaptation simple à d'autres types d'endommagement.

4. VALIDATION DU CODE

Des différentes sorties intermédiaires permettent de vérifier la cohérence des calculs. De plus, certaines simulations peuvent être réalisées.

Par exemple, on peut simuler le comportement du générateur de vapeur sur toute sa période d'exploitation en faisant intervenir la loi d'amorçage (évolution du % de tubes affectés en fonction du temps) et la loi de propagation (évolution de la taille des défauts). On obtient une distribution théorique des tailles de défauts simulés au moment du contrôle et que l'on peut comparer à la distribution réellement observée lors du contrôle. La figure 7 présente les deux distributions ainsi obtenues après 70 000 heures de fonctionnement. On constate un bon ajustement entre les deux distributions surtout pour les grands défauts (> 8 mm) qui sont ceux qui interviennent dans la probabilité de RTGV.
5. APPLICATION DU CODE

De nombreuses études de sensibilité ont été réalisées ; elles permettent de quantifier l'impact de différents paramètres sur le risque de RTGV. Ces études peuvent être classées en deux types :
- recherche de l'effet de variables physiques dans le but d'orienter les recherches futures vers une meilleure connaissance des paramètres qui influent le plus sur le risque de RTGV,
- comparaison de plusieurs politiques de maintenance.

Deux études de sensibilité concernant la politique de maintenance sont présentées ici.

5.1. Influence de la politique de contrôle des tubes

Dans cet exemple, quatre politiques de maintenance ont été testées :
- contrôle de 100 % des tubes par STL,
- contrôle de la moitié des tubes (échantillon aléatoire),
- contrôle de 1/8 des tubes (échantillon aléatoire),
- réalisation d'un test hélium et contrôle des tubes fuyant à l'hélium en supplément du contrôle de 1/8 des tubes de l'échantillon aléatoire.

Le code a été appliqué pour calculer l'évolution au cours du cycle qui a suivi le contrôle de la probabilité de RTGV. L'obturation de tous les tubes contrôlés et dont la taille du défaut dépassait 13 mm a été simulée.

La figure 8 présente l'évolution de la probabilité de rupture d'un tube de GV en condition normale de fonctionnement.

On constate que le fait de contrôler un tube sur 8 ou un tube sur 2 joue très peu sur le risque de RTGV. De plus, un test hélium réalisé en complément d'un contrôle de 1/8 des tubes ne permet pas compte tenu de la fiabilité du test hélium d'obtenir des résultats équivalents à un contrôle de 100 % des tubes.

5.2. Influence du critère d'obturation

Le code a été utilisé pour mesurer l'impact du critère d'obturation sur la probabilité de RTGV. 3 critères ont été appliqués après un contrôle à 100 % du générateur de vapeur (11 mm, 13 mm et 15 mm).

La simulation a été réalisée sur un cycle.

La figure 9 présente l'évolution comparative des probabilités de RTGV en condition normale de fonctionnement. On constate que les probabilités de RTGV sont très proches quel que soit le critère d'obturation. Ceci résulte du fait que les 3 critères testés sont très éloignés des tailles critiques.

La même simulation a été réalisée en faisant l'hypothèse que l'on avait une situation accidentelle (ΔP = 172 bar). Les résultats sont présentés figure 10.

On constate que les courbes obtenues pour un critère de 11 et 13 mm sont identiques. La courbe obtenue pour 15 mm ne diffère que très légèrement des courbes précédentes.

5.3. Estimation de la durée de vie résiduelle des GV

A partir de l'état d'un GV observé lors d'un contrôle, on peut utiliser le code pour obtenir la distribution des défauts après un cycle. En appliquant le critère d'obturation sur cette distribution, on peut estimer le nombre de tubes qui devront être bouchés à la fin de ce cycle. En répétant l'opération sur n cycles, on pourra prévoir le cycle à partir duquel le nombre de tubes à

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obturer serait trop important et pour lequel il faudrait procéder au remplacement du GV.

Le tableau I donne une estimation du nombre de tubes à obturer au cours des 5 cycles suivant le contrôle réalisé à 70 000 heures sur un appareil particulier en appliquant un critère d'obturation de 13 mm.

CONCLUSION

Le code COMPROMIS, grâce à sa démarche probabiliste, permet de répondre d'une façon quantitative à diverses questions que se pose l'exploitant vis-à-vis de la politique de maintenance qu'il met en œuvre sur les GV :
- justification de la périodicité de ces contrôles et des critères de bouchage,
- estimation améliorée de la durée de vie probable des GV.

Il permet par ailleurs de valider, par des simulations sur plusieurs cycles, des lois essentielles de comportement des tubes (loi de propagation par exemple) et constitue un cadre privilégié pour évaluer l'impact réel sur la prévision du comportement des résultats de R et D.

EDF poursuit actuellement ces travaux pour étendre à d'autres types de dégradations le champ d'application du code.

REFERENCES


Schematic diagram of various steps in the COMPROMIS model

Figure 1: Schéma de conception détaillée du modèle COMPROMIS
Figure 2 : Estimation de la distribution initiale des tailles réelles

Figure 3 : Prise en compte de l'obturation

Figure 4 : Evolution de la distribution initiale en fonction du temps

Figure 5 : Distribution des tailles critiques en fonctionnement normal et en situation accidentelle

Figure 6 : Probabilité cumulée de rupture en fonction du temps pour différents programmes de maintenance
Figure 7 : Comparaison des distributions observées et théoriques simulées par le code.

Figure 8 : Influence de la politique de contrôle des tubes sur le risque de RTGV.

Figure 9 : Influence du critère d'obturation sur le risque de RTGV et conditions normales de fonctionnement ($\Delta P = 100$ bar).

Figure 10 : Influence du critère d'obturation sur le risque de RTGV en situation accidentelle ($\Delta P = 172$ bar).

<table>
<thead>
<tr>
<th>Cycle $n_0$</th>
<th>Nombre simulé de nouveaux tubes à obturer</th>
<th>Nombre cumulé de tubes obturés après ce contrôle</th>
</tr>
</thead>
<tbody>
<tr>
<td>$n_0$</td>
<td>47</td>
<td>47</td>
</tr>
<tr>
<td>$n_0 + 1$</td>
<td>53</td>
<td>100</td>
</tr>
<tr>
<td>$n_0 + 2$</td>
<td>50</td>
<td>150</td>
</tr>
<tr>
<td>$n_0 + 3$</td>
<td>54</td>
<td>204</td>
</tr>
<tr>
<td>$n_0 + 4$</td>
<td>69</td>
<td>273</td>
</tr>
<tr>
<td>$n_0 + 5$</td>
<td>100</td>
<td>373</td>
</tr>
</tbody>
</table>

Tableau I : Evolution du nombre de tubes à obturer.
THE EFFICIENCY OF PEENING ON INITIATION AND PROPAGATION
OF PWSCC IN ROLL TRANSITION

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ABSTRACT

On base of promising results from a qualification program, peening was applied to the steam generators of four recent Belgian nuclear power plants, either before start up or after several cycles of service in order to prevent primary water stress corrosion cracking at the level of the roll expansion transition.

The Rotating Pancake Coil Eddy Current inspection, performed over a period of 5 cycles after peening gave very good results for the tubes peened before start up but evidenced an unexpected evolution for the tubes peened after several years of service.

The basic reasons for this evolution were searched in the expertise of pulled tubes.

RESUME

Sur base des résultats prometteurs d'un programme de qualification, le grenailleage a été appliqué aux générateurs de vapeur des quatre plus récentes centrales nucléaires belges, soit avant la mise en service, soit après quelques cycles dans le but d'empêcher la corrosion sous tension côté primaire aux transitions de dudgeonnage.

Le contrôle par courants de Foucault à la sonde tournante, réalisé sur une période de 5 cycles après le grenailleage, a donné d'excellents résultats pour les tubes grenailés avant démarrage mais il a mis en évidence une évolution inattendue pour les tubes grenailés après mise en service.

Les raisons de cette évolution ont été recherchées dans l'expertise de tubes extraits.
1. INTRODUCTION

Degradation of mill annealed Alloy 600 steam generator tubing in service due to Primary Water Stress Corrosion Cracking (PWSCC) has gained increasing attention in the last decade.

This phenomenon affects mainly the roll expanded zone, and is characterized by short, usually axial intergranular cracks which are initiated at the primary side surface, and which may propagate through wall.

In Belgium this phenomenon was detected for the first time 13 years ago on the roll transitions of Doel 2 steam generator tubes, partially rolled in the tube sheet. Later on, the Doel 3 and Tihange 2 steam generators, the tubes of which are expanded over the full depth of the tube sheet, with a kiss roll have experienced cracking not only in the roll transitions but also at roll overlaps after about 2 years of operation.

A program was then launched by the Belgian utilities and was further cosponsored by EPRI to develop generic preventive actions.

On the basis of a critical review of all possible solutions, the following techniques were considered for a near term application:

- stress relaxation and metallurgical improvement of the material by an in situ heat treatment of the whole tube sheet,
- introduction of residual compressive stresses on ID (Inner Diameter) by roto or shot-peening.

Finally, the last technique was chosen and applied, for the first time in the world, on steam generators already installed on site:

- in cold conditions : roto-peening on Doel 4 (Dec. 84) and Tihange 3 (Feb. 85) by Westinghouse, before start-up,
- in hot conditions : shot-peening on Doel 3 (Aug. 85) and Tihange 2 (Feb. 86) by Framatome, after 3 cycles of service.

In Doel 4 and Tihange 3, samples of 15 to 20 tubes per SG were kept unpeened for the purpose of comparing the behaviour of peened and unpeened tubes.

An Eddy current non destructive inspection with a Rotating Pancake Coil (RPC) is performed each year on a significant number of tubes (extending from 15% to 100%) so that the cracking evolution can be followed up.

Moreover, the peened areas have been directly investigated on tubes pulled out from Doel 3 in 1989, aiming at a complete characterization of the cracks and of the surface state.

All these information have been gathered in order to evaluate the efficiency of peening on crack initiation and propagation in the actual steam generator tubes.
2. **HOW DOES PEENING WORK?**

Stress corrosion cracking requires the contribution of the three following factors:

- aggressive environment,
- tensile stress,
- material susceptibility.

Peening acts on the second factor.

Indeed, in the as-built steam generators, the tensile stresses responsible for corrosion are mainly residual stresses which originate in the fabrication process.

The peening generates a thin work hardened layer on ID containing high residual compressive stresses (Figure 1), which counteracts the original tensile stresses and inhibit stress corrosion cracking.

The peening action is effected by the impingement of small beads on the surface; two processes are available (Figure 2):

- roto-peening: the tungsten carbide beads are captured in one or several rows on the surface of supple plastic flappers. The flappers are attached to a shaft rotating at a high speed, which produces the impact of the beads on the surface.

- shot-peening: beads are directly blasted on the inner wall of the tube through an injection nozzle.

3. **WHAT WAS EXPECTED FROM PEENING ON BASIS OF THE QUALIFICATION PROGRAM?**

An extensive qualification program performed on more than 1000 mock-ups had led to optimistic expectancies concerning the initiation and propagation of cracks.

3.1. **Initiation of cracks**

The protective barrier introduced by shot as well as roto-peening should be very efficient to avoid the initiation of cracks at the roll transition, with however a better efficiency for shot-peening. Indeed, the percentage of cracked test specimens after peening was 0% for shot-peened tubes whereas it amounted to about 15% for roto-peening in the corrosion tests performed during the qualification; however a better field efficiency could be expected, owing to a difference of cracking sensitivity between the test and actual materials.
It was assumed that the lower efficiency of roto-peening was due to the difficulty for the flappers to adequately follow the rolling discontinuities. Since Doel 4 and Tihange 3 applications, the roto-peening procedure has been improved by the use of split flappers to better cope with this difficulty.

The corrosion tests also showed that the roto-peening process was unable to suppress cracking at scratches while shot-peening succeeded to do so, probably because of the larger diameter of the beads for the roto-peening.

This lower efficiency of roto-peening at local discontinuities was also evidenced by X-ray residual stresses and microhardness measurements, which showed that the stress situation and the workhardening are nearly the same in roll transitions and unexpanded parts for shotpeened samples whereas they are different for the rotopeened samples (maximum stress and workhardening lower in the roll transition - Figure 3).

Moreover, the roto-peening effect is less homogeneous than the shot-peening, as regards surface residual stresses level as well as work hardened layer thickness (see next table).

<table>
<thead>
<tr>
<th>Peening process</th>
<th>Compressive stress at surface (MPa)</th>
<th>Thickness of work hardened layer (mm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Shot-peening</td>
<td>550 to 650</td>
<td>100 to 150</td>
</tr>
<tr>
<td>Roto-peening</td>
<td>200 to 500</td>
<td>0 to 150</td>
</tr>
</tbody>
</table>

This scatter probably originates from an incomplete coverage and/or a not random distribution of impacts on the surface.

3.2. Stability of the protection in service

It was demonstrated that a heat treatment of 240 h at 400°C does not alter the efficiency of the peening in stress corrosion tests.

3.3. Propagation of cracks

The laboratory tests had shown that an immediate effect of peening is to close the existing cracks (disappearance of cracks at dye check and increase of the pressure to get a leak - Figure 4).

During additional stress corrosion tests performed without inner pressure on precracked peened specimens, it appeared that cracks propagate in depth, but slowlier than for unpeened specimens, and not in length.
This beneficial effect was questioned for real steam generators where the service loads (especially the internal pressure) would tend to open the cracks.

3.4 Representativity of test specimens

The laboratory tests were performed on mock-ups representative of the real tube to tube sheet joints, using different types of materials adapted to the final purpose (surrogate materials such as sensitized Alloy 600 or stainless steel as well as representative Mill Annealed Alloy 600 tubes, and even spare material from the power plants).

None of these tests included the slight InterGranular Attack IGA (10 to 20 µm) which is always present at the inner surface of the SG tubes even after only a limited service duration. The influence of such a layer on the efficiency of peening is unknown. However, it was expected to be insignificant, as the compressive layer induced by peening is an order of magnitude larger.

4. FIELD INSPECTION RESULTS

Since the detection of the PWSCC problem in Doel 3, in 1983, the RPC Eddy Current technology, then already available as a prototypical expertise tool, has been developed by LABORELEC into an industrial method for performing (since 1985) large scale inspections and eventually (since 1988) 100% inspections of the tube bundle of each steam generator, at each outage, for all affected plants (the inspected length being limited to about 150 mm across the tube roll transition) [1].

This allowed to have an excellent knowledge of the SG's cracking status both at the time of peening and for all subsequent outages.

This knowledge is available in the following shape:

- number of affected tubes (at least one crack in the inspected length), expressed as a percentage of the total tube bundle
- location of the affected tubes, on a map of the tube sheet
- for each cracked cross-section
  - number of axial cracks
  - length of the longest crack
  - amplitude of the largest signal.

All of these features can be differentiated among

- roll transitions
- other locations (generally overlaps between adjacent hard roll steps) within the inspected length.
As most of the cracks are located in the roll transition, which is also the only location of actual concern for safety (tube rupture) or reliability (service leakage), the following discussion will be limited to the roll transition.

4.1. Inspection results of SG's peened after several cycles of service

4.1.1. Number and location of cracked tubes

The following table compares the number of cracked tubes at the time of peening and five years later (latest inspection results).

<table>
<thead>
<tr>
<th>Unit</th>
<th>SG</th>
<th>% cracked after</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>3 cycles (SP)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Tihange</td>
<td>1</td>
<td>27 %</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>25 %</td>
</tr>
<tr>
<td></td>
<td>3</td>
<td>29.5 %</td>
</tr>
<tr>
<td>Doel</td>
<td>R</td>
<td>1 %</td>
</tr>
<tr>
<td></td>
<td>G</td>
<td>2 %</td>
</tr>
<tr>
<td></td>
<td>B</td>
<td>24 %</td>
</tr>
</tbody>
</table>

Figure 5 also details the time evolution for one SG from each plant.

Clearly the shot-peening operation did not succeed in preventing cracking in a large number of additional tubes. However, this statement should be qualified in two ways:

1. there is a distinct slowing down in the rate of increase
2. it is not clear whether the "new cracks" have actually been initiated after peening or have grown from preexisting yet non detectable defects; however the latter assumption tends to look less plausible when the phenomenon does not come to an end with time!

As to the location of the cracked tubes, it is typically illustrated by Figure 6. While some clustering is apparent (the pattern of which may vary from SG to SG), there is no evidence of the worst defects (i.e. the longest cracks) having a preferential location (such as the central sludge pile).

4.1.2. Number of cracks per degraded roll transition

The average number of cracks per roll transition is between 3 and 5; a typical distribution curve is given by Figure 7. For tubes already known to be cracked, the
number of cracks has also increased since peening has performed, by an average rate of 0.2 to 0.5 per cycle.

This confirms the observation made under paragraph 4.1.1, subject to the same caution as to the proper interpretation.

4.1.3. Crack length

As already mentioned the crack length being considered is the maximum crack length in each of the defective roll transitions.

The following table compares the average crack length at the time of peening and five years later.

<table>
<thead>
<tr>
<th>Unit</th>
<th>SG</th>
<th>average crack length after</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>3 cycles (SP)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Tihange</td>
<td>1</td>
<td>(</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>(</td>
</tr>
<tr>
<td></td>
<td>3</td>
<td>(&lt; 2 mm</td>
</tr>
<tr>
<td>Doel</td>
<td>R</td>
<td>(</td>
</tr>
<tr>
<td></td>
<td>G</td>
<td>(</td>
</tr>
<tr>
<td></td>
<td>B</td>
<td>(</td>
</tr>
</tbody>
</table>

A typical crack length distribution is illustrated by Figure 8. Such distributions are progressively shifted to the right for successive outage inspections, as illustrated by Figure 9.

These length distribution curves have a characteristic "bell shape"; they can be closely approximated by GAMMA distribution curves [2]. This contrasts with

the early shape evidenced at the start of PWSCC in the same SG's (before peening) which is more like a decreasing exponential

the shape observed after even longer service times for SG's not subjected to peening (Doel 2 after 15 cycles; see Figure 10). It is however not clear if peening justifies this marked difference or if some other features contributed (time history, partial versus full depth rolling, kiss roll,...).

When crack length increases are measured for all known degraded roll transitions between two successive inspections, crack growth distribution curves are obtained as illustrated by Figure 11. The average crack growth varies somewhat from SG to SG and from cycle to cycle, in a range of 0.7 to about 2 mm/cycle.

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4.1.4. **New cracks characteristics**

For those roll transitions which are detected for the first time (after peening) to be cracked
- the average number of cracks is about 2, with maxima up to 10
- the average crack length is in the range of 3 to 4.5 mm, with maxima up to 10 mm.

A typical length distribution curve is illustrated by Figure 12. Again the "bell shape" contrasts with the "decreasing exponential shape" observed when PWSCC was first detected (before peening); also the number of cracks per roll transition was then rarely in excess of 1 or 2.

4.1.5. **Comparison with International experience**

Because of the early development and large scale implementation of the accurate Belgian RPC technology, all of the above trends were already evidenced since 1986 (one cycle after the first peening operation). However most of the results raised international controversial issues because they were much against all expectations and not yet detectable by the then more conventional inspection methods in use elsewhere. With time passing and the progressive generalization of the RPC technology, there is now an international consensus on all of the mentioned post-peening features.

4.2. **Inspection results from SG's peened before plant commissioning**

As mentioned earlier, the Doel 4 and Tihange 3 units were roto-peened before plant commissioning; however a reference population of 15 to 20 tubes per SG was left unpeened.
The following table compares the situation of the peened and unpeened populations (on basis of a 100 % RPC inspection performed after 5 to 6 operational cycles).

<table>
<thead>
<tr>
<th></th>
<th>number of cracked tubes</th>
<th>average crack length</th>
<th>average number of cracks/section</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>unpeened</td>
<td>peened</td>
<td>unpeened</td>
</tr>
<tr>
<td>Doel 4 R</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>G</td>
<td>67 %</td>
<td>1.9 %</td>
<td>4.1 mm</td>
</tr>
<tr>
<td>B (6 cycles)</td>
<td>67 %</td>
<td>2.1 %</td>
<td>3.6 mm</td>
</tr>
<tr>
<td>Tihange 3</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1</td>
<td>80 %</td>
<td>3.4 %</td>
<td>4.8 mm</td>
</tr>
<tr>
<td>2</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>3</td>
<td>60 %</td>
<td>1.6 %</td>
<td>4.2 mm</td>
</tr>
<tr>
<td>(5 cycles)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>2</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>3</td>
<td>50 %</td>
<td>1.7 %</td>
<td>5.1 mm</td>
</tr>
</tbody>
</table>

4.3. Synthesis from inspection results.

Peening performed before plant commissioning provides a high degree of protection against crack initiation; the limited proportion of cracked tubes might even be reduced by the further improved roto-peening technology (split flappers) or by the shot-peening technology.

For SG's peened after detectable crack initiation, the situation is far from looking as favourable. In fact, apart from a significant slow-down in the rate of increase in the number of defective tubes, there is hardly any benefit at all.

What inspection results cannot tell is why the expectations were not met, whether all of those cracks showing up were already initiated at the time of peening and what are the actual physics behind the observed variations in distribution curve shapes, growth averages,...

Destructive analysis of pulled tubes is one of the (not many !) ways to investigate and hopefully solve these puzzling problems, as discussed in the following section.
5. EXAMINATION OF PULLED TUBES

5.1. Investigated tubes

The investigation was performed on two tubes pulled out from Doel 3 in 1989. These tubes had been sleeved in 1988 so that the zone of interest had been exposed to primary water during 3 cycles after shot-peening.

The eddy current inspection of 1988 had showed indications of several cracks at the roll transition, with a rapid progression compared to the previous inspection.

The results were compared to those gained on other tubes of Doel 3 pulled out respectively in 1983, after 1 service cycle and in 1986, 1 year after shot-peening.

5.2. Objectives of the expertise

The expertise was aimed at determining if the unexpected progression (in number) of cracks was due to one of the following scenarios:

- the propagation of cracks already initiated before peening, but too small to be detected by Eddy Current inspection
- the initiation of new cracks.

From the in-service inspection experience gathered with peened tubes in many power plants, it makes nowadays no doubt, that there is no significant difference between peened and unpeened tubes concerning the propagation of cracks.

However the importance and the discontinuous nature of the evolution observed in Doel 3 seem to deviate from the propagation scenario.

The initiation scenario could happen if

- the peening had not induced the expected compression, either because of an incorrect application or because of an interaction with special features of the tubes not reproduced during the qualification phase, such as the superficial IGA.
- the protective compressive layer had been destroyed, by a corrosion damage such as pitting or IGA
- the compression had vanished in the work hardened layer, by thermal or mechanical effect, so that the tensile stresses generated by the inner pressure could initiate a crack. It could indeed be conceived that those stresses, harmless in an unpeened tube, could become dangerous in a peened tube, if the cracking process is governed by a combination of the principal stresses rather than by the maximum principal stress.
5.3. Aspect of the tubes. Position and shape of cracks

The roll transition inner surface is characterized by the typical aspect observed after peening ("orange skin").

Cracks are situated at the roll transition. Long (7 to 10 mm) through cracks as well as very short (< 1 mm) cracks are found, which could be the indication of a new initiation. However, this could also be explained by a stress relaxation effect caused by larger neighbour cracks.

The shape of the cracks is illustrated at Figure 13: they present a bulging under the inner wall, which is typical for the progression under a peened surface.

The inner surface is uniformly attacked by IGA (10 to 20 μm deep). This IGA does not seem to have progressed since it was already present on tubes pulled out after 1 cycle of service. At some places deeper penetrations (50 to 100 μm) are evidenced (Figure 14), which might act as initiation points.

5.4. Stresses

X-ray residual stresses measurements were performed on the inner surface of the as-pulled tubes and after step by step polishing.

In spite of some perturbing effects induced on the stress state by the sleeving operation in 1988 and by the pulling operation, the results evidenced that the compressive stresses at the level of the transition were lower than previously found during the qualification, not only at the surface (350 MPa) but also after polishing (450 MPa).

The reasons for that difference could be attributed to:
- a thermal relaxation (which can account for only 10 to 15% of decrease)
- the IGA of the inner surface (not tested during the qualification tests)
- a lower value from the beginning.

5.5. Postulated reasons for cracking progression

From the investigation of pulled tubes, it was concluded that the compressive barrier induced by shot-peening was still present in the roll transition and should prevent the initiation of new cracks, even if the compression level is lower than foreseen from the qualification tests. This lower level might be attributed partly to the thermal relaxation of stresses during service, and partly to the homogeneous intergranular attack present at the inner wall (10 - 15 μm) which is a usual feature of the SG tubes, whatever the material, and which was probably present before peening.
The upsetting progression of cracking, evidenced by non destructive inspections is thus probably the result of the propagation of preexisting defects. However the question of what is a "preexisting defect" may be raised.

Not only small stress corrosion cracks already initiated are concerned, but it can also be feared that local intergranular penetrations (50 to 100 μm) present randomly at the surface might be sufficient as initiation points, though they are smaller than the expected compressive layer.

So, even if this mechanism is not literally a stress corrosion cracking initiation, it exactly looks like it.

Whether these penetrations were present before shot-peening is still an unresolved issue. If they might appear after peening (by an intergranular attack independent of stress), some concerns would be raised about units peened before plant commissioning, such as Doel 4 and Tihange 3.

However, the available NDT data demonstrate an excellent behaviour of those tubes after six cycles, which suggests that such a mechanism does not occur.

6. CONCLUSION

As expected from the qualification program, the extensive ECT inspections show that peening performed before plant commissioning provides a high degree of protection against crack initiation; the limited proportion (about 2 %) of cracked tubes might even be reduced (cancelled ?) by the further improved roto-peening technology (split flappers) or by the shot-peening technology.

For SG's peened after detectable crack initiation, peening does not bring the benefits expected from the qualification: the progression of cracking remains high after 5 cycles which could possibly result from the propagation of small intergranular penetrations randomly distributed at the roll transition and acting as initiation points, as evidenced by the expertise of pulled tubes.

However, there is a significant slow-down in the rate of increase of the number of defective tubes so that peening is at least "buying time", which may be an essential feature to help building an efficient repair/replacement strategy [3].
Figure 1

Distribution of stresses through the wall thickness of a peened tube
Figure 3

Distribution of the stresses from ID shot and roto-peened samples in the unexpanded area and in the roll transition.
**Fig. 4a**: Number of ID cracks at the roll transition by dye check versus by metallographic examination for peened and unpeened specimens.

**Fig. 4b**: Leak pressure increase after shot-peening.
Figure 7

TIHANGE 2 - SG 2 - Cracks/section distribution

nbr tubes = f [cracks/section] - 100% sample - 1988
Figure 8

Frequency Histogram

1988 inspection

- gamma curve fitting
- inspection data (1585 tubes)

Crack length (mm)
TIHANGE 2 – SG 2  - Length distribution

tubes (%) = f[长度 (mm)] - Total sample of 88, 89 & 90

Figure 9

CRACK LENGTH (mm)

1 2 3 4 5 6 7 8 9 10 11 12 13 14
DOEL 2 - SG A - Hot leg - Length distribution

Number of tubes (%)

Crack length (mm)
DOEL 3 – SG G - Length increase/year tubes (%) = f[length increase/year] - 89/90 total sample
Figure 12

Typical Frequency Histogram
for "NEW CRACKS"

frequency

0 100 200 300 400

1 2 3 4 5 6 7 8 9 10 11
crack length (mm)

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Figure 13: Open crack - Typical shape

Figure 14: Deep intergranular penetration at the inner wall (expansion transition) (x 500)
REFERENCES


AN ANALYSIS OF PRIMARY WATER STRESS CORROSION CRACKING
IN PWR STEAM GENERATORS

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Abstract

This paper discusses a fracture mechanics based model for estimating the progress of populations of longitudinal through-wall stress corrosion cracks in alloy 600 PWR steam generator tubes in the region of the roll transition.

The key role played by residual stresses and their relaxation by stress corrosion cracking is described so that the proportions of cracks arresting or continuing to propagate may be estimated. The same phenomenon is also shown to account for the characteristic reduction in growth rate with increasing crack length observed by non-destructive examination of operating steam generators.

Contributions to uncertainty in the results of non-destructive examinations influencing comparisons with these calculations are also examined.

Cet article présente un modèle fondé sur la mécanique de la rupture qui a pour but de calculer l'évolution des distributions de fissures longitudinales traversantes dues à la corrosion sous contrainte présentes dans de nombre de tubes de générateur de vapeur en alliage 600, en zone de transition de dudgeonnage.

On décrit l'importance du rôle du relâchement des contraintes résiduelles suite à la corrosion sous contrainte dans l'estimation de la proportion des fissures qui s'arrêtent ou continuent de se propager. Le même phénomène tend également à expliquer la diminution caractéristique de la vitesse de propagation en fonction de l'augmentation de la longueur des fissures observées par contrôles non-destructifs des générateurs de vapeur.

L'influence de la dispersion des résultats issus des contrôles non-destructifs est examinée et prise en compte dans les comparaisons avec les résultats des calculs.
Introduction

Primary water side initiated stress corrosion cracking of alloy 600 PWR steam generator tubing has proved to be a serious problem affecting the reliability and longevity of PWR steam generators. The phenomenon has been particularly severe in the roll transition region between the top of the tube sheet and the tube free span above the tube sheet.

In many plants, shot peening has been carried out to generate a compressive residual stress on the internal tube surfaces in the roll transition area with the objective of preventing further initiation of primary water stress corrosion cracks. Extensive non-destructive examinations over several fuel cycles have shown, as expected, that few if any new cracks have initiated since shot peening was carried out (1) although some doubts as to its total efficacy have been expressed (2). Clearly, however, any cracks already initiated and of the same order of size or greater than the thickness of the shot peened layer of \( \approx 100 \mu m \) are unlikely to be arrested. It is therefore highly desirable that their future progress be quantified in order to plan tube plugging campaigns, alternative remedial measures and, if necessary, steam generator replacement.

As with most time dependent degradation processes such as stress corrosion cracking which give rise to a distribution of cracks of different sizes, due mainly to inherent differences in initiation susceptibility, there are essentially two approaches which can be adopted to try and estimate future development. The first is a totally statistical approach where empirically based algorithms describing evolution of the phenomenon are derived mathematically from non-destructive examination data. In the context of primary water stress corrosion of roll transitions of PWR steam generator tubes, this has been developed by Hernalsteen (3) and is further elaborated at this conference (4). The second method is to try and describe each of the relevant processes by physically based equations together with their associated statistical distributions and then attempt to calculate the cumulative distribution for evolution of the phenomenon. Such an approach to steam generator tube cracking has been published by Granger et al (5) and is also extended at this conference (6). It also includes the broader issues of inspection reliability and estimation of the probabilities and limits for tube rupture as well as predictions of stress corrosion crack initiation and growth.

The purpose of this paper is to examine in detail a model for primary water stress corrosion crack propagation with the aim of being able to predict the evolution of distributions of longitudinally orientated cracks in roll transition zones; any circumferentially orientated cracks are immediately plugged on detection. The propagation model described here appears to have similarities with that described by Granger et al (5, 6) since both are based on a fracture mechanics description of the phenomenon but, as will become apparent, there are important differences of detail. Inevitably too, it is essential to examine to what extent uncertainties in the quantitative results of non-destructive examination influence the evolution of crack length distributions which the models initially try to reproduce and then try to forecast the future course of events. It is also assumed that shot peening has been effective in preventing further crack initiation, an assumption which is also examined later in association with uncertainties arising from non-destructive examinations.
The elements of the model; a description of crack growth rates as a function of the crack tip stress intensity factor, a detailed description of the stresses responsible for extending longitudinal through-wall cracks and a criterion for crack arrest are described in more detail below. The predictions of the model are then compared with some available data on specific steam generators.

**Description of the model**

(i) Crack growth rates.

Our analysis of primary water stress corrosion crack propagation kinetics in alloy 600 is based on the data published by Smialowska et al. Several different water chemistries were studied at 330°C most of which were outside the primary water specification envelope for lithium hydroxide, boric acid and dissolved hydrogen (for the purpose of investigating trends in crack growth rates as a function of water chemistry). The following equation was however fitted to the data for a standard primary water chemistry of 2ppm Li, 1200 ppm B, pH = 7.3:

\[
\frac{da}{dt} = 2.8 \times 10^{-11} \ (K-9)^{1.16} \text{ m/sec (K in MPa} \sqrt{m})
\]

A value of \( K_{isc} = 9 \ \text{MPa} \sqrt{m} \) is thus assumed. This equation is plotted in Figure 1.

Since the specimen used by Smialowska et al for crack growth measurements was machined from a flattened half of a short length of steam generator tubing, the degree of cold work was judged to be significantly more severe than is relevant to the roll transition region (maximum of 2%). Independently, Cassagne et al have demonstrated in hydrogenated steam tests at 400°C that 5% prior cold work leads to growth rates between 5 and 10 times faster than those observed in non-cold worked material. Accordingly a correction was made to the above crack growth rate equation to take this effect into account together with the difference in temperature between the tests and typical primary water hot leg operating temperatures. Generally, the rates given by the equation above were divided by 5 for primary water temperatures of \( \approx 320°C \) and by 10 for \( \approx 310°C \) as shown in Figure 1.

One important difference between the definition of crack growth rates above and that of Granger et al, is the implicit assumption in the latter case that crack growth rates are related fundamentally to crack tip strain rate which in turn is calculated as a function of \( K \) using the equation proposed by Ford and Andreason for sensitized austenitic stainless steels in BWR type environments. A similar though empirically derived dependence on strain rate has been used by Garud in his correlations with stress corrosion data from laboratory tests on alloy 600 in PWR primary water. Such a strain rate dependence arises fundamentally from the anodic dissolution model of stress corrosion cracking which is quite well proven for sensitized stainless steel in BWR water but is not a universally accepted explanation for primary water stress corrosion in alloy 600. A consequence which is plainly obvious in the case of data for sensitized stainless steel is that growth rates are not a single
valued function of K and vary over several orders of magnitude depending strongly on the externally applied loading rate during transients for example as well as on K. Such a dependence, for alloy 600 in constant water chemistry, is not particularly strongly apparent in the available data. However, some dependence on applied strain rate can be expected in laboratory slow strain rate tests since the rates of intergranular cracking and ductile failure are not likely to be the same.

We have thus used a single valued function of K for alloy 600 in the absence of strong evidence to the contrary.

In order to make a comparison between longitudinal crack growth rates expressed as a function of K with those derived statistically by Hernalsteen (3, 4) and others as a function of crack length, it is necessary also to estimate tube hoop stresses in the roll transition region as follows.

(ii) Tube hoop stresses

The starting point for estimating the stresses responsible for longitudinal crack propagation in the roll transition region is a recent publication by Flesch et al (11). Residual stresses in these studies were derived from x-ray measurements, MgCl₂ tests on stainless steel mock-ups, caustic soda tests on alloy 600 mock-ups and computer simulations. Operating stresses were derived from finite element modelling.

The highest total of circumferential residual plus operating stresses reported for kiss roll transitions from full depth rolling is 410 MPa. Although in practice such high stresses probably only apply to a relatively thin layer on the internal tube surface, this value was nevertheless adopted here as an upper bound average hoop stress in this region. (In all cases it is assumed that only the average of the through-wall bending stresses plus the membrane stress is relevant to the propagation of through-wall cracks). A lower bound estimate of the total through-thickness average hoop stresses for the kiss roll transition region used in our calculations is 230 MPa.

It is known that when cracks initiate in the roll transition region and then propagate up the tube, a high proportion of them arrest (3, 4, 12). Thus it is clear that the cracks are propagating into a sharply decreasing stress field. The decrease in crack growth rates with increasing length observed by Hernalsteen (3, 4) and also shown by additional data given here is further evidence supporting this conclusion. Clearly, if all the residual stresses are relaxed by cracking, the minimum stress which remains is the operating stress. Upper and lower bound estimates of the average operating hoop stress in a steam generator tube adopted here are 120 MPa (from reference 11) and 100 MPa (since it is noted that the differential pressure stress must be at least 95 MPa).

The key question which must now be addressed is how quickly the driving hoop stress decreases from 410 to 120 MPa (upper bound) or 230 to 100 MPa (lower bound) with distance up the tube measured from the roll transition. It is clear from reference 11 that this decrease in the residual stress component of the hoop stresses can occur within a few millimetres of the roll transition. The initial strategy adopted here was to test various functions describing this

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decrease in hoop stress until a good fit was obtained between the growth rates calculated as a function of crack length from our fracture mechanics analysis and the results reported by Hernalsteen(3.4.13).

The upper and lower bound stress profiles finally selected are shown in Figure 2. In effect, the selected functions allowed the total stress to decrease rapidly from 410 to 120 MPa between 2 and 10 mm measured from the start of the roll transition (figure 2). The starting point of 2 mm was chosen on the basis that the minimum length of through-thickness or near through-thickness cracks observed is about 2 mm. The end point for the stress decrease was chosen on the basis of the number of cracks of a given length observed in any one tube (see next section). However, this was not a particularity sensitive parameter in our calculations.

The resulting calculated crack growth rates as a function of crack length using the stress relaxation functions of Figure 2 above and the crack growth rate equations of Figure 1 are shown in Figure 3. Apart from the generally good agreement with the steam generator inspection results also shown in Figure 3, two other features stand out. The first is a maximum in growth rates at a particular value of crack length which arises from the dependence of $K$ on $\sqrt{a}$ and of stress on a stronger function of crack half length 'a'. The second is a slow rise in growth rates for long cracks which arises from a constant assumed driving hoop stress at that point (the operating stress) and a slight increase in growth rates with increasing $K$ rather than an idealised assumption of constant stage II stress corrosion crack growth rates. Both features described above of the crack growth rate curves derived by fracture mechanics have some important consequences when predicting the future distribution of crack sizes.

(iii) Criterion for crack arrest

As mentioned earlier, it is clear that many cracks which initiate at the roll transition subsequently arrest as they grow up the tube into a decreasing stress field. It has been reported from destructive examinations of tubes extracted from steam generators of French 900 MWe plants that ≤ 20 cracks are found with lengths ≤ 5 mm and ≤ 2 cracks with lengths ≥ 10 mm (12). If it is assumed that each crack releases an equal amount of elastic strain when 18 cracks arrest between stress levels of 410 and 120 MPa, then the opening of each crack on a 19 mm diameter tube is equal to about 5 μm. Significantly, this is very close to the effective value of crack width derived for through-wall cracks in this size range for the purpose of primary to secondary leak rate estimation. In stress terms this means that each crack relieves about 16 MPa.

Given the above assumption that each crack relieves an equal amount of elastic strain, then the number of cracks which continue to propagate as the cracks grow into a decreasing stress field will be proportional to the stress (or elastic strain). This is the criterion for crack arrest adopted in these calculations and is indicated on the right hand axis of Figure 2 but in this case with the stress relaxation per crack rounded to 20 MPa. Thus, between 16 and 17 cracks propagating can be supported by a stress of 410 MPa, 8 or 9 by 250 MPa, 2 by 120 MPa and 1 by 100 MPa.
Another interesting point to note in figure 2 is the calculated threshold stress as a function of crack length based on a value of $K_{isc}$ of 9 MPa$\sqrt{m}$. For cracks between 5 and 10 mm long, this threshold varies from 100 to 70 MPa which is always 10 to 30 MPa below the stress available as a crack driving force.

From all these considerations described above it is deduced that 100% of cracks between zero and 2 mm long and those greater than 10 mm long continue to propagate. Between 2 and 10 mm long, a significant fraction arrest.

An immediate consequence of the proposed crack arrest criterion is that it is possible to calculate the fractional distribution of arrested cracks and the total fraction of cracks which propagate beyond 10 mm, provided that no new cracks enter the population after shot peening. The results for the stress distributions given in Figure 2 shown in Table I.

The results given in Table I are used directly in our calculation of the evolution of crack size distributions to determine the fraction of cracks in a given size range which can propagate each reactor cycle. This fraction is necessarily re-calculated after each crack growth integration step. However, it must be emphasised that all tubes which were cracked prior to shot peening are, within the scope of this model, likely to have one crack which will eventually reach the plugging limit. It is interesting to compare this approach to the method used by Hernalsteen derived from statistical analysis of Belgium PWR steam generator inspection data (3,4). In his analysis the fraction of cracks propagating in a size range 'c' is given by:

$$F_c = e^{-kc} \text{ where } k = 0.11 \text{ mm}^{-1}.$$  

The results from this function are also shown in Table I from which it can be seen that numerically similar data are obtained.

**Model results**

After some trial calculations it was decided to integrate the crack growth rate equation analytically, reactor cycle by reactor cycle, assuming no significant change of stress intensity for a given initial crack length during any one cycle. In this way for any particular stress distribution, crack growth increments such as those represented by the curves in figure 3 could be applied directly to crack growth distributions given in the form of histograms. Algorithms were also developed to apply the crack arrest criterion described earlier.

Examples of calculations reproducing the evolution of crack length distributions are shown in Figures 4 and 5 for two Swedish PWR steam generators. Figure 4 shows good agreement between observed and calculated crack distributions for two successive years while Figure 5 shows a five year prediction. In this case histograms of all cracks found by non-destructive examination are represented by smooth curve fits. Thus it was necessary to apply the crack arrest criterion in these calculations. An inevitable consequence seen in Figure 5 is that a small proportion of cracks continue to grow while a large fraction arrest thus producing a characteristically double humped distribution curve as also found by
Granger et al (5, 6). In some cases, it is noted that published crack length distributions represent only the longest cracks in each tube which by definition in our model are the ones which continue to propagate.

Discussion

The initial results of this relatively simple fracture mechanics model of longitudinal crack propagation in the roll transition has quite successfully reproduced in the cases examined so far both the average crack growth rates as a function of crack length and the evolution of crack size distributions in successive reactor cycles. It can thus be used to predict the near-term future. Since there is no crack growth data to indicate a strong influence of microstructure on growth rates measured in fracture mechanics type specimens of alloy 600 (as distinct from initiation), then differences in rates observed between different reactor systems are most likely due to different residual stresses in the roll transition and/or the primary water operating temperature.

A related aspect to emphasise is the fact that the curves representing crack growth rate as a function of crack length derived from non-destructive examinations are themselves generally fitted to the mean growth rates observed. In fact these measurements have large associated standard deviations (which is quantified directly in the predictive model of Hemalsteen(5, 6)) and is due at least in part to a random sizing error of order ± 1.0 mm.

A complication affecting the accuracy of prediction of crack length distributions and judgements regarding the efficacy of shot peening in preventing further crack initiation arises from cracks whose sizes are below the non-destructive detection limit one year and above it the next year. As noted by Granger and Pitner(5, 6) it is necessary to make due allowance for this effect in calculations.

Careful analysis of the raw inspection data in the few cases we have examined in detail also reveals another important contribution to the evolution of crack distributions caused by cracks of quite long length appearing in the inspection data for the first time. Figure 6 shows an example deduced from the distribution of the longest cracks in each cracked tube from a particular steam generator. At this time we believe this may be a general phenomenon. This does not imply that the rotating pancake coil eddy current method used here is inefficient but that from a relatively large population of cracks a small proportion of long ones will be missed one year and found the following year. Furthermore, small co-linear cracks below the detection limit may join to form a large one above the detection limit between two inspections.

This type of detailed examination of the raw results of steam generator inspection is being actively pursued. In this way it is hoped to build up a better picture of true primary side stress corrosion cracks growth as an aid to the development of the physical model described earlier.

Conclusions

A fracture mechanics model can describe successfully the evolution from reactor
cycle to reactor cycle of distributions of longitudinally orientated primary water stress corrosion cracks in steam generator tube transition zones.

An important feature of the model described is the relaxation of residual stresses by stress corrosion crack growth. It allows the maximum number of cracks per tube and the fracture of a given length arresting or propagating to be estimated.

The method of analysis of random error in detection and sizing of cracks by non-destructive examination can have an important effect on perceived crack propagation rates and influence judgement of the efficacy of shot peening in preventing new cracks initiating.

Acknowledgement

The author thanks Mr R. Comby of EdF, Mr L-E Bjerke of Vattenfall Ringalsverket, Mr P. Hernalsteen of Laborelec and Mr A. McIlree of EPRI for permission to use previously unpublished data in this paper.

Table I

The fraction of longitudinal cracks which continue to propagate as a function of distance from the roll transition.

<table>
<thead>
<tr>
<th>Crack size range mm</th>
<th>Fraction of all cracks propagating</th>
<th>Compare $e^{-k}$, $k = 0.11 \text{ mm}^{-1}$ (Hernalsteen)</th>
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<tbody>
<tr>
<td></td>
<td>upper bound stresses</td>
<td>lower bound stresses</td>
</tr>
<tr>
<td>0 - 1</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td>1 - 2</td>
<td>1.0</td>
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<tr>
<td>2 - 3</td>
<td>0.72</td>
<td>0.74</td>
</tr>
<tr>
<td>3 - 4</td>
<td>0.53</td>
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<tr>
<td>4 - 5</td>
<td>0.40</td>
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<td>5 - 6</td>
<td>0.30</td>
<td>0.34</td>
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<td>6 - 7</td>
<td>0.23</td>
<td>0.27</td>
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<td>7 - 8</td>
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<td>8 - 9</td>
<td>0.14</td>
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<tr>
<td>9 - 10</td>
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<tr>
<td>&gt; 10</td>
<td>0.12</td>
<td>0.17</td>
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</table>

5.6-8/16
References


4 Hernalsteen, P., "A predictive model for the PWSCC degradation process in tube roll transitions", Paper 5.2 This conference


6 Granger, B., Pitner, P., Flesch, B., "Probabilistic fracture mechanics code for steam generator maintenance", Paper 5.4 This conference


5.6-9/16

Crack Growth Equations for Alloy 600 Steam Generator Tubing

\[ \dot{a} = 2.8 \times 10^{-18} (K - 9)^{1.16} \]

\[ \dot{a} = 2.8 \times 10^{-12} (K - 9)^{1.16} \]

FIG. 1

Crack Velocity (m/s) vs. Stress Intensity (KMPa m)
FIG. 2  Hoop Stress Distributions along the Roll Transition

Stress (MPa)

Distance from Roll Transition (mm)

HIGH ESTIMATE OF STRESSES

THRESHOLD STRESS FOR $K_{thc} = 9$ MPa$\sqrt{m}$

LOW ESTIMATE OF STRESSES

Number of Cracks

1 - 17
2 - 16
3 - 15
4 - 14
5 - 13
6 - 12
7 - 11
8 - 10
9 - 9
10 - 8
11 - 7
12 - 6
13 - 5
14 - 4
15 - 3
16 - 2
17 - 1
FIG. 3 Crack Growth as a Function of Crack Length

- Plant A + B
- Plant C
- Plant D (310°)

INCREASE PER REACTOR CYCLE (mm)

CRACK LENGTH (mm)
FIG. 4 Calculated distribution of cracks for 1990 from 1989 Data and comparison with 1990 inspection results (310°)

Fraction of all cracks

Crack size (mm)

--- 1989 RESULTS    --- 1990 CALCULATED    --- 1990 RESULTS
FIG. 6 CRACK LENGTH INCREASES BETWEEN TWO SUCCESSIVE CYCLES

Number of tubes

Crack length increase 1989-1990 (mm)

-3 -2 -1 0 1 2 3 4 5 6 7 8

- New cracks 1990
- All apparent growth of longest cracks
SESSION 6 - INSERVICE INSPECTION METHODS

6.1 ROUND ROBIN TESTS OF THE PISC III PROGRAMME ON DEFECTIVE STEAM GENERATORS TUBES

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6.2 REGULATORY REQUIREMENTS ON INSERVICE INSPECTION FOR STEAM GENERATORS IN JAPAN

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6.3 EDDY CURRENT INSPECTION METHODOLOGY

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6.4 AUTOMATIC EDDY CURRENT SIGNAL EVALUATION ENVIRONMENT (ENEAS)

J. Guerra, Tecnatom, Madrid, SPAIN
J. Vazques, Tecnatom, Madrid, SPAIN

6.5 SPECIFIC ULTRASONIC INSPECTION METHODS FOR STEAM GENERATOR TUBES

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B. Libin, Framatome, Paris La Défense, FRANCE
A. Thomas, Framatome, Lyon, FRANCE

6.6 ULTRASONIC INSPECTION METHODOLOGY

D. Dobbeni, Belgotam/Laborelec, Brussels, BELGIUM

6.7 ROSA III, A THIRD GENERATION STEAM GENERATOR SERVICE ROBOT TARGETED AT REDUCING STEAM GENERATOR MAINTENANCE EXPOSURE

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ROUND ROBIN TESTS OF THE PISC III PROGRAMME
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ABSTRACT

The PISC III Actions are intended to extend the results and methodologies of the previous PISC exercises, i.e. the assessment of the capabilities of the various examination techniques when used on real or realistic flaws in real components under real conditions of inspection. Being aware of the industrial problems that the degradation of steam generator tubes can create, the PISC III Management Board decided to include in the PISC III Programme a special Action on Steam Generator Tubes Testing (SGT).

RESUME

Les actions du programme PISC III ont pour but d'étendre à d'autres composants que la cuve les méthodes d'évaluation des exercices PISC précédents: la vérification dans les conditions industrielles d'inspection de la capacité de différentes techniques à détecter et à caractériser des défauts réels ou réalistes.

Etant conscient de l'importance des problèmes industriels et de sûreté posés par les dégradations des tubes des générateurs de vapeur, le Groupe de Gestion du Programme PISC a décidé d'inclure dans le PISC III une Action spécifique à l'inspection des tubes Générateurs de Vapeur.
1. INTRODUCTION

The SGT programme differs from those of the other PISC III Actions [1] where heavy steel components are considered. Significant distinctions are material (INCONEL 600), geometry (diameter 7/8" = 22.22 mm, wall thickness 1.27 mm), the technique mainly applied (Eddy Current), and the number of flaw types which can occur on steam generator tubes. The definition of the programme depended on the interest of the participating countries in the large variety of failure mechanisms. An inquiry inside of the countries helped to understand their priority of interest in the various flaws.

In the meantime, results from international activities in the field (SURRY Project) became available and are taken into account together with recommendations from international workshops. The programme (samples matrix and test schedule) of what is considered a first phase, capability tests on loose tubes, is formulated and the Round Robin tests started in early 1990 with the circulation of Training Tubes Boxes.

2. PRELIMINARY INQUIRY

Differences in the applied material, in design and operating conditions of steam generators, utilized in the PISC member countries can cause a large variety of failure mechanisms on the steam generator tubes. Therefore, the first step undertaken for establishing a test matrix was an inquiry [2] on the specific technical aspects and especially on the priority given to the different kinds of flaws in the member countries.

The inquiry contained four categories of questions concerning

- tube design characteristics,
- steam generator operating conditions,
- existence of failure mechanisms and degree of interest,
- availability of tubes and mock-ups.

Answers to the questionnaire arrived from eleven countries and showed the following trends:

- Wear, IGSCC and IGA are the failure mechanisms most frequently observed or of most potential interest.
- There are many proposals of tubes or mock-ups but they were corresponding to flaws sparsely represented in the answers to the inquiry or not fully available.
- Service induced flaws are only represented by the SURRY steam generator tubes, but containing sparsely IGSCC and IGA.

The SGT programme, proposed to the PISC Management Board in Spring 1988, was based in large part on the information from this inquiry and got a positive response of 27 teams from 10 countries interested in the participation in related Round Robin Tests (RRTs). Actually 30 teams are participating.
3. IMPLICATION OF THE SURRY WORK

In the light of extensive information that became available from the USNRC Steam Generator Group Project (SURRY Project) there was a strong requirement to review the PISC programme on NDE of steam generators tubes [3]. A workshop on this matter, held at Paris in April 1988 developed programme recommendations:

- There is a further need for studies to demonstrate the capability of detection, sizing and characterization of flaw types as IGSCC and IGA. (In the SURRY steam generator tubes the mainly observed flaws were denting, pitting and wastage).
- complicated situations (combinations between different flaws, between flaw and deposit, between flaw and tube geometry or environment).

- There is a need of performance demonstration of improved methods of data processing and signal analysis.

These recommendations became implemented in the Steam Generator Tubes Testing (SGT) programme of PISC.

4. CSNI WORKSHOP RECOMMENDATIONS

In October 1988 the CSNI-Principal Working Group No. 3 organized a "Workshop on Steam Generator Integrity" at Staatliche Materialprüfungsanstalt (MPA) at Stuttgart [4].

The main objective of the Workshop was to discuss steam generator tube plugging criteria as applied in practice in different Member Countries and to prepare recommendations and guidance for the PWG3 and the PISC Management Board on general integrity requirements and objectives. It was an updating of the general understanding of practices in the different countries and an aid to the evaluation of results on NDE testing of steam generator tubes in the PISC Round Robin Test programme.

The discussions led to the formulation of the following recommendations to the PISC Management Board:

- a strong emphasis has to be put on IGSCC and IGA failure mechanisms;
- not only the flaw depth has to be retained, but also other flaw characteristics like the nature (crack or not crack), the length, the orientation, the location;
- key values or "bench marks" have been indicated such as
  - 20% wall thickness as reporting level,
  - 40% wall thickness as plugging criterion
  - about 16 mm as plugging criterion in the case of longitudinal cracks.

The last recommendations will be taken into account for the evaluation of the programme results.
5. THE PISC STEAM GENERATOR TUBES TEST PROGRAMME

5.1. Objective and Content of the Programme [5,6]

The objective, relatively close to that of the PISC III Actions on heavy structures, is the experimental evaluation of the performance of test procedures used for steam generator tubes in nuclear power plants during in-service or pre-service inspection.

A proposal of a SGT programme in three phases has been presented:

PHASE 1 is a capability exercise consisting of Round Robin Tests (RRTs) on individual tubes including calibration, training and blind test tubes.
PHASE 2 includes capability and reliability Round Robin Tests on uncontaminated mockups adequate for automatic tools inspection.
PHASE 3 also consists of capability and reliability tests but on a contaminated mock-up, simulating real access conditions to the steam generator tubing. This mock-up will contain tubes with service induced flaws removed from the SURRY steam generator.

The resources presently made available by the Operating Agent of PISC III (JRC of CEC) in its multiannual programme 1988-1991 only permit to consider the first phase. It is hoped, in 1992 to find the financial support necessary for the organization of the two last phases when Phase 1 is completed in 1993.

5.2. Artificial flaw validation exercise

Although the SGT programme was accepted in general in Spring 1988, the PISC III SGT group and the Management Board decided that no RRT sample would be manufactured and proposed for circulation until the flaws to be introduced in tubes and mock-ups would be certified as realistic flaws by a group of specialists.

The Reference Laboratory of the PISC Programme, the NDE Service of JRC-Ispri (responsible of the preparation of the test samples for the Round Robin tests) organized a validation exercise including destructive examination of all flaw types acquired or made available by Member Countries. This action was actively supported by specialized laboratories of Belgium, France, Germany, Italy, Spain, United Kingdom, USA and Japan. The flaws were mainly provided by Japan and the Netherlands, mechanical flaws and simulations were manufactured by JRC Ispri [7].

As a conclusion, in September 1989, it has been recognized that realistic flaws are possible to be manufactured of the following categories: denting, wastage, fatigue crack simulations (EDM), pitting, deposit, wear, IGA, SWSCC, PWSCC.

Taking into account the recommendations from the activities mentioned in Chapters 3 and 4, the SGT group accepted part of a new samples matrix in terms of flaw realism, nature, orientation and dimensions.

5.3. Major flaws considered

From a fabrication point of view, or with regard to the origin of the sample, four categories of tubes are now involved in the SGT Round Robin tests:
- blank tubes with different pilgrim noises, transition zones, environment (antivibration bar, tube support plate, tube sheet, etc.);
- tubes with machined simple flaws but also combinations of flaws, reproducing various detection and characterization situations and appearing as realistic flaws for ET;
- tubes with in-service induced flaws, e.g. pitting, wastage, light denting and deposit-flaw combinations;
- tubes with artificial chemically induced flaws certified as realistic simulations of PWSCC, SWSCC and IGA.

Artificial flaws fabrication methods were selected by the PISC SGT Group as a result of the certification exercise. Besides the contribution of Japan and France described here under, several flaws were ordered in UK and the Netherlands. Many spark eroded flaws were made by JRC Ispra, to simulate realistically wear, pitting, wastage, fretting and cracks (Figures 1 and 2).

The major corrosion flaws are IGA, SWSCC and PWSCC.

a. SWIGA and SWSCC - According to the various investigation results up to date in Japan [8], it is considered that IGA can be caused by the concentration of NaOH in the crevice region of TSPs or of the TS. The corrosion is not always only a general attack type IGA but often it produces SCC with various orientations depending on the strain directions.

Figure 3 shows the microstructure of cross sections on the pulled out tubes from those typical SWSCC and IGA at the TSP level. To introduce chemically SWIGA, autoclave equipments are used. 40% NaOH solution with little amount of oxidizer (CuO) is filled in secondary side. Primary side is water and the pressure of both side is controlled to have 100 kg/cm² higher primary pressure at elevated temperature of 280°C. The through wall depth of IGA/SCC is controlled by the holding time in the autoclave. The control of surface area to be attacked is achieved by the masking with teflon seal tape.

b. PWSCC - This kind of flaw has been observed in the tight U-bend region and roll expanded or roll transition region.

Examination results of the pulled tubes from operating plants reveal that PWSCC is mainly axially oriented and very tight. Some plants in several countries have shown circumferential PWSCC.

To introduce chemically PWSCC at the roll transition, the following procedure is used [8]:
- Heat treatment to sensitize the material: 1050°C solution treatment and 675°C sensitizing,
- Installation of sensitized tube to tube sheet mock-up by mechanical rolling,
- Installation of test sample to SCC stand,
- Filling polythionic acid into the tube,
- Pressurizing by N₂ gas,
- Holding in a refrigerator for specific period,
- Holding temperature is 5°C +/- 0.5°C and pressure is kept at 100 - 150 kg/cm² by N₂ gas.

Through wall depth of SCC is controlled by holding time, but normally
shows some extent of distribution. Approximately 50% through wall depth SCC needs 20 to 30 hours holding time. Figure 4 shows the cross-sections of a sample tube with PWS MCC introduced by the procedure mentioned above. These artificially introduced PWS MCCs on roll transition are not too far from PWS MCC on pulled tubes from operating plants. Artificially introduced PWS MCC is more like IGA type degradation.

c. Artificial production of Real Corrosion flaws - The CEA, at La Hague has been asked by the Reference Laboratory to introduce, using its installation like a Mini Steam Generator, conditions as corrosion of the TSP which would induce denting and real corrosion (SWS MCC) of the tube.

The installation of CEA at La Hague can also be used to produce in the roll transition zone some SWS MCCs and PWS MCCs, longitudinal and circumferential, that can be considered as real flaws.

5.4. Organization of the Round Robin Tests

The capability tests on loose tubes involve about 100 tubes (1 m long) which are placed in sealed boxes. Special devices developed for these boxes ensure

- blind test conditions during circulation;
- identical test conditions also after transportation of the boxes.
Three kinds of samples are in circulation:

- Calibration tubes: ASME calibration tubes and blank tubes made available by the Reference Laboratory;
- Training tubes in 3 boxes with typical examples of simple flaws, rarely composite flaws; these boxes illustrated in Figure 5 and 6 are fully documented: flaw type, flaw position, flaw characteristics, ET type indications. This documentation on about 25 training tubes with a total of 60 flawed areas has been provided to all participating teams. Table 1 represents the content of the documentation of one flaw on one training tube.
- Blind test tubes in 8 boxes containing straight or curved tubes with simulation of structural elements. Each box contains 6 to 9 tubes of 1 m length with two or three possible flaw zones each. For the circulation the 8 boxes are divided into two batches (Figure 7).

Present plans consider 4 batches: Training Tubes Box 1, Training Tubes Boxes 2 and 3, Blind Test Boxes 1-4 and Blind Test Boxes 5-6.

It has been agreed that each team has one week time available for testing one batch applying one procedure: ET or UT.

The circulation schedule is based on the number of 30 teams which confirmed participation (9 countries). The RRT started with the circulation of Training Tubes in February 1990 (Table 2 and 3). The circulation of the first batch of Blind Test Tubes started in June 1991 and taking into account the necessary transportation time between teams and countries, the RRT will be concluded in the first half of 1993 (Table 4).
5.5. Evaluation of Results

5.5.1 Organization of the evaluation

The Reference Laboratory (RL, JRC Ispra) is responsible for the collection of inspection data. The Reference Laboratory is also responsible for the certification of flaws and for conducting or directing all destructive examinations. The analysis and evaluation of results is performed by the Reference Laboratory under the guidance of a Data Analysis Group (DAG) following methodologies approved by the Evaluation Task Force. All DAGs report to the PISC III Management Board. The PISC Referee Group (RG) established by the Operating Agent has the responsibility to ensure the confidentiality of team inspection results. The few members of this RG (JRC staff members) are the only ones to have direct contact with the inspections teams.

5.5.2 Bench-marks for the evaluation of the RRT results

For the evaluation of the RRT results, the Management Board of the PISC programme approved a strategy based on a set of bench-marks for important parameters, such as:

- minimum size of flaws for destructive examination (5% of wall thickness);
- reporting level for indications (20% of wall thickness);
- rejection level for indications (40% of wall thickness);
- differentiation between cracks and other flaws;
- rejection sizes in length for cracks.

On this basis, discussions are still going on to finalize the methodologies to be followed.

6. CONCLUSIONS

The objective of PISC III Action No. 5: Steam Generator Tubes Testing (SGT), is the assessment of capability and reliability of procedures as applied for the in-service inspection of steam generator tubes. The programme, proposed in Spring 1988, was based on the results of an international enquiry regarding the interest of each country in the different failure mechanisms which may occur on steam generator tubes. This programme has been revised taking into account the results of the US-NRC Steam Generator Group Project (SURRY Project) and the recommendations of CSNI Workshops.

Specialists meetings contributed to the definition of the flaws and tube specimens to be used for the Round Robin tests of the SGT programme. Phase 1 of the programme, capability tests on loose tubes, started in the beginning of 1990 and should be concluded in June 1993.
The circulation of the first three boxes of tubes: the Training Tubes Boxes, revealed already the validity of such performance demonstration samples for SG tubes inspection techniques and indicate that the flaws and "situations" chosen for the exercise are of effective industrial interest.

REFERENCES


1. TUBE SUPPORT PLATE: CIRCULAR HOLE

Position:

\[ \begin{align*}
    X_1 &= 271.5 \text{ mm} \\
    X_2 &= 290.5 \text{ mm} \\
    \Delta X &= 19 \text{ mm}
\end{align*} \]

TSP hole diameter: 22.6 mm

Material: Carbon steel

2. ENVELOPE OF FLAW AS INTENDED

\[ \begin{align*}
    X_1 &= 271 \text{ mm} \\
    X_2 &= 291 \text{ mm} \\
    \Delta X &= 20 \text{ mm} \\
    Y_1 &= 0 \text{ mm} \\
    Y_2 &= 0.05 \text{ mm} \\
    \Delta Y &= 0.05 \text{ mm} \\
    Z_1 &= 0.36 \text{ mm} \\
    Z_2 &= 1.27 \text{ mm} \\
    \Delta Z &= 0.91 \text{ mm}
\end{align*} \]

Simulation of external longitudinal crack under TSP

EXAMPLE OF EDDY CURRENT TESTING

Flaw before placing of TSP

Flaw with TSP placed
<table>
<thead>
<tr>
<th>Country</th>
<th>Team</th>
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<th>Period (weeks)</th>
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<tr>
<td>Transport to Spain</td>
<td>E</td>
<td></td>
<td>8 (1990)</td>
</tr>
<tr>
<td></td>
<td>Tecnatom</td>
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</tr>
<tr>
<td></td>
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Status: May 10, 1991

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Status: May 10, 1991

Table 4. Circulation schedule of the Blind Test Boxes (First batch of four boxes)
TUBE: No 2
DFCT: No 3

Figure 1. Volumetric defect (EDM)

TUBE: No 5
DFCT: No 6

Figure 2. Crack simulations (EDM)

6.1-13/17
Example of SWSCC at TSP
(multiple cracks)

Example of IGA at TSP

Figure 3. IGA and SWSCC at TSP
Figure 4. PWSCC at rolling transition

Figure 5. Training Tubes Boxes and Calibration Tube
Figure 7. Blind Test Boxes of the SGT-RRT
REGULATORY REQUIREMENTS ON INSERVICE INSPECTION

FOR STEAM GENERATORS IN JAPAN

Seiji Yashima

Japan Power Engineering and Inspection Corporation

Tokyo, Japan

Abstract

In Japan, it is set forth in the Electric Utility Industry Law that nuclear power plants must be subjected to the periodical inspection of the government, and the inservice inspection of steam generator tubing is defined as one of the item of this periodical inspection.

The eddy current test technique is adopted as the standard method of inspection for tubing, in addition, a superior detecting performance is realized by unique technique according to the defect mode. And then, when defects are found, all defective tubes are dealt with by either plugging or sleeving.
1. Introduction

Since the steam generator tubing accounts for approximately 50% of the total area of the primary system pressure retaining boundary, it is very important to maintain integrity of the tubing in assuring the safety of PWR plants. It is for this reason that sufficient considerations are provided in order to maintain the tubing integrity in design, manufacturing and operation of steam generators.

In Japan, the technical matters which are relevant to the assurance of safety in construction, maintenance and operation of nuclear power plants are defined in the Technical Standard of the Electric Utility Industry Law. This Technical Standard is used as the criterion in application and approval of construction plan of a nuclear power plant, and it is also referred to as the standard in the inspection before use. Also, the electric utilities have legal obligation to maintain their plant facilities in conditions that conform this Technical Standard.

Therefore, the inservice inspection on steam generator tubing is being conducted as an action of verifying that there is no defect in tubing. It is not allowed to use tubing on which defect is detected, and electric utilities are legally obliged to repair all defective tubing by plugging or sleeving. Operating a nuclear power plant with its steam generator leaking the primary coolant to the secondary side is not permitted in Japan.

2. Legal Enforcement of Periodical Inspection

In Japan, an nuclear power plant which has been commissioned to service is periodically shut-down by legal requirement to have its facilities and structures inspected. The inspection items consist of those legally set forth and those which are voluntarily conducted by the utilities (utility inspection).

It is stipulated in the Electric Utility Industry Law that the electric utilities shall have their nuclear reactors and their incidental facilities, and the steam turbines subjected to the inspections to be conducted by the Ministry of International Trade and Industry (MITI). The specific facilities subjected to this MITI inspection, and the timing of the inspection, are provided in the Enforcement Regulations of Electric Utility Industry Law.
(1) Facilities Subjected to Inspection

a. Nuclear Reactor and Its Incidental Facilities
   
   Main body of nuclear reactor
   Reactor cooling system
   Instrumentation control system
   Fuel system
   Radiation control system
   Waste disposal system
   Reactor container
   Auxiliary boiler
   Emergency power generating system stand-by

b. Steam Turbine

(2) Timing of Inspection

a. For a nuclear reactor and its incidental facilities, within one (1) month before or after the date which is one (1) year after the previous inspection has been completed.

b. For a steam turbine, within one (1) month before or after the date which is two (2) year after the previous inspection has been completed.

The Nuclear Power Safety Administration Division of the Agency of Natural Resources and Energy has set down the Enforcement Regulations of Periodical Inspection that controls the implementation of a periodical inspection. This Enforcement Regulation provides the objective, timing, items and methods of inspection, and the in-service inspection of steam generator tubing it is set forth as one of the item in this Enforcement Regulation.
3. Actual Status of Steam Generator Tubing Inservice Inspection

(1) Organization for Implementation of Inservice Inspection and Its Roles

The inspection of steam generator tubing is performed by the electric utilities, and it is witnessed by Japan Power Engineering and Inspection Corporation (JAPEIC), a third party inspection institution, who issues the inspection certificate, and the inspection procedure is completed when MITI approves this inspection certificate.

That is, the electric utilities conduct the inspection as a part of voluntary safety management, award a contract to a specialized inspection company and the inspection is performed by the employees of the electric utilities. The inspectors of JAPEIC witness this inspection, confirm the testing condition, review all acquired data, and verify the data analysis results, to issue the inspection certificate. In this process, the certificate is issued only for the tubes on which no defect was detected. The inspection is completed when the MITI inspector inspects and confirms these data on the inspection certificate.

The defect tubes, which have been excluded form the inspection certificate, are subjected to a separately stipulated procedure, and then repaired.

(2) Inspection Method

The eddy current testing is adopted for inspection of steam generator tubing. The scope of testing, and the testing technique such as probe and frequency have been improved according to the occurrence of degradations and degradation modes that have been experienced. The technical progress of eddy current testing in Japan is presented in Table 1.

In the early years when PWR plants were operated in Japan, the sampling inspection method was used in the utility inspection. Since a steam generator tube leakage occurred at Mihama-1 in 1972, the steam generator tubing inspection was defined as the MITI inspection in 1973 based on the principle of Technical Standard such that the use of degraded tube is not to be used, and the scope of inspection was expanded to cover all tubes and full lengths. Later, the application of sampling inspection on the steam generators having no tube degradation history had been studied, but the all tube, full length inspection by MITI is being carried out even today, in view of the fact that tube degradations due to intergranular attack, stress corrosion cracking, etc. are actually occurring.
Concerning the probe used in eddy current testing, the optimal probe types for the part of tube to be inspected and the degradation modes are selectively used.

a. Dual Function Probe

This is a complex type probe, in which a large pitch coil, having the same function with standard bobbin coil probe, and a small pitch coil that focused magnetic flux which is not affected by the change of inside diameter, are combined.

This probe is used in conventional inspection for probing the full tube length, and in particular, it can detect a flaw at roll expansion regions.

b. U-Bend Probe

This probe is fundamentally the same with the standard bobbin coil probe, but the probe diameter is small so that it can be easily inserted to the bend section, and it is provided with guides that prevents wobbling of the probe to reduce the unwanted signal.

This probe is used for inspection of U-bend part of Raw-1 to Raw-3 tubes.

c. Single Coil Probe

In this probe, one of a pair of coil is fixed and another is moved.

This probe is used to detect the stress corrosion cracking which is long in the longitudinal direction occurring at tube sheet crevice.

d. Brazed Sleeve Probe

This probe contains a permanent magnet in order to reduce the permeability change. This probe is used to inspect the part of tube at which the brazed sleeving maintenance has been implemented.

e. Pancake Coil Probe

In this probe, four (4) pancake coils are positioned in two stages along the circumferential
direction, or eight (8) pancake coils are positioned on the same circumference.

This probe is particularly used to detect the stress corrosion crack along the circumferential direction.

The multi-frequency signal processing technology method is employed in the testing, and generally 100 kHz – 400 kHz is used. In addition, a multiple frequency system of 400 kHz – 100 kHz – 600 kHz is used with the objective of reducing or eliminating the effect of signal generated by the deposits on the secondary side surface of tubing, and this system is applied to confirm the presence of defect in tubes for which the deposit signal has been detected in the standard testing by the dual function probe.

Therefore, in conducting the MITI inspection, the type of probe to be used in the eddy current testing is determined for each plant based on the past experience of tube degradations. This probe selection is illustrated in Table II.

3) Acceptance Criteria

The acceptance criteria currently applied in evaluating results of the steam generator tubing eddy current testing conducted in a periodical inspection is presented in Table III.

That is, in principle, tubes for which depth of flaw can be quantitatively evaluated are accepted if the depth is 20% or less, and those for which quantitative evaluation is not possible are accepted if the meaningful signal change is not observed during operation. The definition of the meaningful signal differs a little depending on the mode and location of the degradation, but fundamentally, the signal which amplitude is apparently larger than the noise signal amplitude is regarded as the meaningful signal. This method is applied to the testing for such degradations as stress corrosion cracking and intergranular attack.

4. Conclusion

In Japan, operation of a nuclear power plant is not permitted when the primary coolant is leaking to the secondary side through steam generator tubing. Therefore, eddy current testing is applied to all tubes for full length in a periodical inspection, and corrective maintenance measure, such as plugging or sleeving, is implemented on the tube on which a defect is detected, thereby preventing unscheduled outage of a plant in operation due to primary coolant leakage.
However, in the case of Mihama-2, the tube rupture has occurred despite the fact that the plant had been operated only approximately 7 months after the previous inspection, resulting in actuation of the emergency core cooling system, thereby degrading the reliance on the current inspection procedure.

For this reason, it would be necessary to develop the detection technology which can deal with unexperienced tube degradation mode, and further improve the flaw detection performance and accuracy.
<table>
<thead>
<tr>
<th>Year</th>
<th>ECT Techniques</th>
<th>Reasons</th>
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<tr>
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<td>Sampling inspection</td>
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<td>1973</td>
<td>All tubes, fall length inspection</td>
<td>SG leakage occurred at Mihama-1</td>
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<tr>
<td>1975</td>
<td>Data evaluation by complex signal catalogue</td>
<td>Wastage occurred at tight U-bend</td>
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<td>1977</td>
<td>U-bend probe</td>
<td>SCC occurred at tight U-bend</td>
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<tr>
<td>1982</td>
<td>Four segment probe</td>
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<td>1982</td>
<td>COM evaluation technique</td>
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<td>Single coil probe</td>
<td>Long length SCC occurred at T.S.crevice</td>
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<td>Brazed sleeve probe</td>
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<td>1987</td>
<td>Computerized data evaluation system</td>
<td>Shorten evaluation time and improve accuracy</td>
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<td>1988</td>
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Table II  Probes Using Eddy Current Testing
Table III(1/2)  Concept of Eddy Current Signal Evaluation Technique

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<td>Y</td>
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<td>(1) Signal greater than noise level is rejectable</td>
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<td>crack</td>
<td>X</td>
<td>(2) Lessojuass figure</td>
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<td></td>
<td>full expanded region</td>
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<td>(3) evaluate significant</td>
<td>(2) evaluate significant</td>
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<td>(400KHz)</td>
<td>signal change</td>
<td>signal change (DF probe large pitch coil)</td>
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<td>Secondary side</td>
<td>Y</td>
<td>- do -</td>
<td>- do -</td>
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<td>crack</td>
<td>X</td>
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<td>primary side</td>
<td>(400KHz)</td>
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<td>(1) evaluate significant</td>
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<td>X</td>
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<td>signal change</td>
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<td>full expanded</td>
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<td></td>
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<td>400KHz)</td>
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<td>Portion</td>
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<td>EC Signal</td>
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<td>----------------------------------------------------------------</td>
<td>----------------------------------------------------------</td>
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</table>
| TSPL             | wastage             | (100KHz)        | (1) Lissajous figure  
(2) Y amplitude of mixing  
signal (100KHz-400KHz)  
(3) evaluate significant  
signal change            | (1) <20% thru wall depth  
indication is acceptable  
(2) evaluate significant  
signal change            |
| TSPL crevice (include top of TS) |                     | (400KHz)        | - do -                                                          | (1) TSPL mix residue signal greater than  
noise level (1.5V) is rejectable  
(2) evaluate significant  
signal change            |
| U bend (Row 1-3) | crack               | (100KHz)        | (1) X,Y chart  
(2) Lissajous figure  
(3) evaluate significant  
signal change            | (1) Any change of signal  
from last inspection is rejectable |
| U bend           | 7th TSPL            | (100KHz)        |                                                                 |                                                          |
| Anti Vibration Bar | cross section       | Y (100KHz-400KHz) | (1) Lissajous figure  
(2) X,Y chart  
(3) Evaluate significant  
signal change            | (1) <20% thru wall depth  
indication is acceptable |
EDDY CURRENT INSPECTION METHODOLOGY

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Belgatom/Laborelec
Brussels (Belgium)

ABSTRACT

World-wide operating experience with nuclear pressurized water reactors shows that most steam generators show evidence of major degradation phenomena. Laborelec has designed several automated inspection systems using eddy current (bobbin coil, pancake coils and rotating coil) and ultrasound techniques. The bobbin coil is mostly used as a fast and global detection method while dedicated sensors, like the rotating pancake coil, are applied for an improved defect characterization. Experience has demonstrated that significant advantages in accuracy, reliability and cost are intimately related to the automation process.

RESUME

L'expérience mondiale concernant l'exploitation des centrales nucléaires à eau pressurisée indique que la plupart des tubes de générateur de vapeur sont atteints par plusieurs phénomènes de dégradation. Laborelec a conçu plusieurs systèmes de contrôle automatisés et basés respectivement sur les méthodes par courants de Foucault (sonde axiale, sonde multibobine, sonde rotative) et par ultrasons. La sonde axiale est utilisée principalement pour un examen rapide et global alors que des capteurs plus spécialisés comme la sonde ponctuelle rotative, sont appliqués pour une meilleure caractérisation des défauts. L'expérience a démontré que les avantages observés en terme de précision, de reproductibilité et de coût étaient intimement liés à l'automatisation des unités de contrôle.
1. INTRODUCTION

Experience with nuclear pressurized water reactors indicates that steam generators are among the most critical components of the primary loop, resulting in unscheduled or extended outages, costly repair or replacement operations. To assist utilities in coping with these problems, non destructive examinations are applied during the refuelling outages to detect the possible degradations and to help predict their evolution. Besides the mandatory tests, which must comply with the current national regulations (in each country), dedicated methodologies are continuously developed to improve the detectability and the sizing accuracy.

For steam generators, various corrosive mechanisms are responsible for the degradation observed on the tube bundle (Figure 1): wastage, denting, pitting, secondary side corrosion (intergranular attack and stress corrosion cracking) and primary side stress corrosion [1]. Different repair techniques are currently available: peening, nickel coating, sleeving and thermal stress relief. These techniques add new inspection challenges while the concept of the multipurpose inspection system helps to reduce the research and development costs. Indeed, it was soon envisioned that different methods would be needed to characterize the multiple degradation mechanisms. The design methodology, used by Laborelec, is based on the adequate matching between the defect characteristics (from destructive analysis), the sensor and the instrumentation. For faster operations, the error prone manual operations inherent to large inspection programs were systematically replaced with computer controlled operations.

The design of this multipurpose inspection system, where dedicated tools and techniques can be quickly and easily added, showed improved diagnoses. As an added benefit, lower costs were soon demonstrated as being the consequence of higher inspection speed (three or ten times faster than provided by other service companies) and smaller inspection teams.

Special inspection systems remain complementary needed for dedicated applications. They have been developed with the same objective of automation and efficiency.

2. MULTIPURPOSE INSPECTION SYSTEM

2.1. Methodology

Eddy current using a bobbin coil sensor constitutes usually the baseline inspection technique for the tube bundle. The rationale behind this choice lies in the high speed at which data can be acquired and the potential for remote control and automated signal analysis.
The eddy current technique is based on the influence of a defect on the electrical and electromagnetic properties of a conducting material placed in an induction field. This field is obtained by injecting a sinusoidal current in a coil. A loss of material or a dimensional variation in the tube is revealed by a signal of typical form, phase and amplitude. The phase can provide information as to the depth of the flaw and its internal or external origin, whereas the amplitude depicts the amount of material with disturbed current flow. As any conducting material that surrounds the test material may perturb the induction field, appropriate data processing is needed to minimize their influence. This method is known as the "multifrequency eddy current"[2] and is applied in Belgium since 1976.

Different coil configurations can be selected to enhance some specific defect characteristics: rotating pin-point coils (called rotating pancake coil) or multicoil probes of different geometries (called pancake probes). Send - receive configurations have also been applied but to a lesser extent.

As the steam generator environment was not designed for large and sophisticated inspection programs, innovative approaches based on field experience were needed to reduce the personal exposure (ALARA objective).

2.2. Data acquisition

The multipurpose inspection system is installed in mobile trailers. Fixed electrical wirings link each steam generator platform to the trailer instrumentation (Figure 2). Each Belgian nuclear plant is equipped with such a set of wirings that accounts for a large part for the short installation time (usually less than 8 hours).

The probe positioning requires two devices: the manipulator and the pusher-puller. The manipulator moves the probe injection device in front of the tube to inspect while the pusher-puller moves the probe within the tube.

Laborelec manipulators are based on a RØ design equipped with one rotary arm and a trolley. Two units were designed; a full size unit for a 100% inspection and a small unit for special operations (small samples for expertise or unplanned shut-down). The installation in the channel head of the full size manipulator is easily achieved (one minute) and does not have to be repeated for the entire inspection. For both units, the displacements were optimized and require less than 2 seconds between two neighbour tubes. Absolute angle encoders ensure the accuracy to the millimetre, and the entire system is remote controlled.

The pusher-puller design depends on the selected method (eddy current or ultrasound), the displacement pattern (rectangular or helicoidal) and the maximum tube elevation where the examinations are to be performed. These units are computer driven and their sequencing is coupled to the manipulator. To date, five dedicated pusher-pullers have been designed and are regularly operated within the Belgian units.
The main features of the measurement system are a high dynamic range, a very fast data acquisition and the use, since 1985, of a laser recording system. The whole inspection system is computer driven and does not require any human action with the exception of the initial calibration and the probe replacements, if required. Several tasks run concurrently so as to optimize the inspection speed (i.e., the data acquisition, the quality control and the laser recording). The on-line quality control verifies the probe speed and the calibration of each measurement channel with reference defects that are acquired at the end of each tube. For the bobbin coil, the average rate is as high as 50 tubes an hour (full length tubing). It is close to 120 tubes an hour for the inspection with a rotating pancake coil of tube sheet area, thus allowing 100% inspection of a steam generator in two days. In each circumstance, only one technician is needed for the supervision of a data acquisition unit.

2.3. Data analysis

The interpretation of the recorded test signals includes three consecutive tasks: the detection of abnormal signals, the identification of their origin and the final defect sizing. In 1985, a computer-assisted interpretation was taken into service after three years of extensive testing. This analysis software combines the recorded signals with the expert knowledge, triggers the detection rules, identifies the most probable defect origin and finally issues a preliminary diagnostic. Besides higher reproducibility and detectability than available from the human analyst, it allows a five to ten times speed improvement and smaller inspection teams [3].

To ensure an even better diagnosis capability, Laborelec originated a three years Esprit II project (project 2167 - AITRAS) funded by the Commission of the European Communities. Its aim is the design of a real time expert system for the analysis of instrumentation signals that can be applied to numerous non-destructive inspection and monitoring techniques including the predictive maintenance of rotating machinery [3]. The integration of the resulting software within the Laborelec inspection system has already started and has demonstrated improved diagnosis for the bobbin coil analysis. Complete integration is foreseen in 1992.

2.4. Reporting and evolution study

This inspection task, vital though fastidious, involves the structuring of all the results emerging from the analyses. Its aim is to document the formatted diagnoses and to evidence, in the shortest time possible, the evolution of the various types of degradation. Several dedicated softwares were developed and account for a large part for the total system efficiency. By matter of example, the site report for a 100% inspection of the top of the tube sheet by means of a rotating pancake coil (or approx. 5000 diagnoses) is issued only half an hour after the last inspected tube. This report details not only each individual diagnosis but includes also the evolution study of all known degradations. For each known phenomenon tube sheet maps and histograms are produced at the user request. Figure 3 shows a tube sheet map with the
localization of the SCC cracks longer than 10 mm. The evolution of the crack length after one fuel cycle can be observed at figure 4 while the crack length distribution after three inspections can be analysed on figure 5. These plots easily demonstrates the continuous trend for longer axial SCC lengths with a fairly constant yearly propagation and without a preferential location on the tube sheet. The increasing need for improved defect sizing led to the design of more specialized eddy current sensors.

3. SPECIAL INSPECTION SYSTEMS

3.1. Eddy current profilometry

An original measurement system of the inner profile of the tubes was developed for the pre-service inspection of the four last PWR units. This system is based on a set of eight contactless pancake coils. Each coil provides a measure of the distance between the inner tube wall and the outer probe surface. Calibration allows four diameters, spaced 45 degrees apart, to be deduced on the same cross section. The complete inner tube profile is recorded with a diameter accuracy of +/- 0.02 mm. A typical record of a fully expanded tube with one missing roll is shown in figure 6. The inspection of the expanded area in the tube sheet was achieved at a rate of 350 tubes an hour. This technique has also been applied for the accurate sizing of dents and the quality control of repair operations located in the tube sheet area.

3.2. Eddy current rotating pancake probe

The sensitivity of Inconel 600 to primary water stress corrosion cracking was confirmed for the Belgian plants since 1977. These small volume cracks are hardly detected by the bobbin coil inspection method. A rotating pancake coil technique is needed to improve both the detection sensitivity and the sizing accuracy. When a plugging criterion based on the crack length was proposed, a fast inspection technique had to be selected for the 100% inspection of the affected steam generators. A special rotating pancake probe was developed for this purpose [4]. The sensor scans the inner tube surface according to a helical scan. The corresponding eddy current signals allow the measurement of the length and the number of cracks. Comparison with real stress corrosion cracks, measured by destructive means, showed the accuracy of the length sizing to be better than +/- 1.5 mm. Figure 4 shows a typical record where one longitudinal stress corrosion crack can be easily recognized.

The large field experience of the design team was a major factor in achieving system speed and accuracy. As an example, the inspection of three steam generators with 4864 tubes each was performed within 8 days including the equipment installation and removal. This timing could be further reduced if the three steam generators had been inspected simultaneously.
Recently, indications of circumferential secondary side stress corrosion cracking were observed on a few tubes of one unit. A dedicated software routine allowed the identification of the defects which required preventive plugging. As an example, figure 8 shows the largest crack observed during the latest in-service inspection. However, the rotating pancake coil is not sufficient to detect at the earliest stage this kind of circumferential cracking. A rotating ultrasound system provides a complementary tool for the study of this degradation phenomenon [5].

4. CONCLUSION

The concept of a multipurpose inspection system satisfies the safety goal and the utility requirements for reliability, accuracy and lower costs. An optimum was reached by equipment modularity, computer control and integrated design of sensors and instrumentation. What previously required several man-weeks of work is now achieved within a few hours time. With a presently unparalleled field performance, Laborelec has set the basis of new industrial standards.
Figure 1: Sketch of recirculating steam generator with possible problem areas
Figure 2: Scheme of trailer/plant link for the steam generator inspection
Figure 3: Example of a tube sheet map with the localization of the longest axial cracks at the top of the tube sheet (length ≥ 11 mm).

| < 25 tubes | 11 - 11 mm |
| 16 tubes   | 12 - 12 mm |
| 11 tubes   | 13 - 13 mm |
| 2 tubes    | 14 - 14 mm |
Figure 4: Example of an histogram plot with the evolution in axial crack length at the top of the tube sheet after one fuel cycle
TIHANGE 2 - SG 3-04 - Length increase/year for DAM/21

$\text{tubes (\%) = f(length increase/year)} - 89/90$ total sample

Figure 5: Example of an histogram plot with the number of tubes in function of the maximum axial crack length at the top of the tube sheet after two consecutive fuel cycles
Figure 6: Profilometry: Example of a plot from a full depth expansion with one skipped roll

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Figure 7: Rotating pancake coil: Example of a signal plot with a single axial stress corrosion crack in the roll transition
Figure 8: Rotating pancake coil: Example of a signal plot with a circumferential stress corrosion crack in the roll transition.
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ENEA

AUTOMATIC EDDY CURRENT SIGNAL EVALUATION ENVIRONMENT

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ABSTRACT

The Guidelines recommend that analysis of Steam Generator Eddy Current inspections be duplicated. Automatic analysis systems are gradually taking over from manual methods.

Tecnatom has developed an automatic system named EVA which can be used for one of the two analyses required, the system being both compatible with and independent from other commercially available systems.

The EVA system is integrated within an environment known as ENEAS, which provides a capacity for other applications and for data bases.
1 INTRODUCTION

A series of recommendations govern Steam Generator (SG) inspections using eddy current (EC) techniques. These recommendations are included in the EPRI document "PWR Steam Generator Examination Guidelines" [1].

One part of these recommendations refers to the characteristics of the analysis of data provided by the acquisition system.

These Guidelines recommend that acquisition of the final inspection results be based on a dual data analysis process, each part of which may be carried out manually or automatically.

Given that multifrequency equipment has to be used for these inspections and that this equipment provides large volumes of information, and given also the large numbers of tubes to be inspected it is advisable to maximize to the extent possible automation of all processes related to inspection, particularly the processes of analysis.

Consequently, it is advisable to perform the two necessary analyses automatically, for which two independent systems will be required.

2 OBJECTIVES

The objective of this article is to underline the importance of automatic analysis systems, and to describe the experience gained in this field in Spain.

In this respect, the article will present an analysis system developed by Tecnatom which allows the processes to be carried out automatically and independently.

3 DATA ANALYSIS

Eddy current SG tube inspections provide data which actually consist of graphic signals in two dimensions (lissajous type), these signals having two components: horizontal and vertical.

The analysis process consists of recognizing those signals which are produced by defects, and differentiating them from those others which are generated by the internal structures of the steam generator, and which are not harmful as regards tube integrity.

Many of the signals generated by alterations to the tube wall may occur simultaneously with others produced by the SG structures, this making detection more difficult. To this problem should be
added the number of signals provided by each tube and the number of tubes inspected, all of which gives an idea of the difficulty inherent in analysis and of the need for considerable experience for manual analysis or for signal processing for automatic analysis.

4 ANALYSIS SYSTEM

The two analyses recommended by the Guidelines may be any combination of manual or automatic processes.


Until recently, two manual processes were used for inspection result acquisition.

Although these two processes were independent one from the other, the fact that the human factor was involved meant that certain errors could occur. Also, this method implied the use of considerable resources due to the lack of reliable automatic processes.

Manual-Automatic

To date, this is the methodology which has been used in Spain, based on software supplied by Zetec and including manual and automatic processes (CDS).

The two processes are carried out in parallel using different teams of analysts.

The automatic process uses thresholds and rules for classification and dimensioning.

Automatic-Automatic

This methodology, which includes two automatic processes, has not yet been implemented definitively, but is possible since an automatic system independent from CDS and known as EVA is now available.

EVA is an analysis program developed by Tecnatom and embedded in an environment known as ENEAS, which also provides applications other than standard inspection analysis.

Consequently, EVA allows circular coil EC inspections to be manually or automatically analyzed.
CHARACTERISTICS OF AUTOMATIC ANALYSIS

Generally speaking, the automatic signal analysis systems used for this type of inspections are made up of two fundamental parts: detection and classification.

Detection is usually carried out by means of more or less complicated thresholds on channels of a given frequency, or as a result of processing the information on one or more channels (filters, mixes, etc).

Classification is performed following detection, and may be based on a series of different techniques, the most common being based on thresholds, pattern recognition, or expert systems.

All of these techniques use more or less complex rules ranging from amplitude thresholds and distance determination functions to rules similar to those used by the analysts.

TRENDS IN SPAIN

The data acquisition systems used in Spain are of the MIZ-18 type supplied by Zetec, along with manual and automatic analysis software.

In order to allow the two processes of analysis to be carried out automatically, Tecnatom developed the compatible and independent analysis software EVA, which was integrated within an environment known as ENEAS.

The current trend is thus to address both analysis processes automatically, one using EVA and the other with CDS.

ENEAS

ENEAS is a computer environment which brings together a series of applications all of which are oriented toward eddy current inspection analysis [2].

Each of these applications covers different aspects of use of EC techniques in the field. The applications included within this environment are as follows: manual circular coil inspection evaluation, automatic circular coil inspection evaluation, automatic tube profile parametrization, semi-automatic rotating coil inspection analysis, and data base.

All of these applications share a common environment, and in some cases are interrelated. This provides a real analysis environment which uses acquisition data, information provided by previous inspections, a system providing a learning function for the

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discrimination of signals, and subsequent processing of data obtained from the analysis for the drawing up of reports.

The user interface has been designed to ensure suitable ergonomics and user friendliness, and ensures easy access to the major functions. The interface also incorporates a methodology allowing macro-functions to be used for rapid, efficient and uniform performance of certain of the operations.

The system is developed on the basis of the UNIX operating system, and uses a window-based on-screen display system. Various programmes may be run simultaneously, with distribution of the windows established on user criteria.

The hardware requirements for this environment are those associated with the computers which may be used, i.e., HP series 300, 400 or 700 machines, or other similar systems compatible with the UNIX operating system.

8 EVA APPLICATION

EVA is one of the applications included within the ENEAS environment. It also constitutes the basis for all the other applications, since a part of the programs and tools comprising EVA are shared by these applications.

EVA may be used for the analysis of inspections performed using circular coils, in either the manual or automatic mode.

The core of this application is the program known as EVA, the system including also a screen for display of tube registers and function keys for the user.

The EVA program is backed up by others allowing additional information to be processed or displayed. For example, a program for display of tube index, another for display of results reports, another for display of additional lissajous, and another allowing data time registers to be displays for eight components. A brief description of the windows presented by these programs is given below.

Figure 1 shows the main EVA window, which is broken down into different zones each of which has a specific application. Subwindow A shows a time register for two of the tube components. B includes an amplification of what is displayed between the A subwindow cursors. Subwindow C shows two lissajous for two channels corresponding to the zones located between the subwindow B cursors. Subwindow D shows the value of rotation, amplitude, etc., of the channels and mixes, and optionally the calibration curve for any channel. Finally, subwindow E includes a menu with all the options and functions provided by the program; this menu is also used for activation of other programs.
Figure 1.

The INDEX program is activated from the main EVA window, and shows the tube index and messages contained on a tape, along with the data acquired. As in the previous case, this window (Fig. 2) is made up of several subwindows, each of which has a specific function. For example, subwindow A shows a part of the index, and subwindow B indicates the function keys to be used for selecting items from the index.

As in the previous case, the REPORT program may be activated from the EVA program, and shows the data inserted as a result of analysis of the different SG tubes. This display also has several subwindows (Fig. 3), such as A which displays data alphanumeric information, and B which includes the function keys for operation of the previous subwindow.
The LISSAJOUS program may be activated from EVA. By selecting the number and configuration of the lissajous to be displayed, the same number of rectangles as lissajous selected is displayed on the screen. The user may select the channels to be represented in each case (Fig. 4).

Figure 4.

The 8 COMPONENTS program, which as in the previous cases may be activated from EVA, shows 8 subwindows displaying 8 time components of the tube registers. The component to be displayed may be selected in each of these subwindows.

Apart from the programs described above, the EVA application includes a series of auxiliary programs allowing automatic processes to be defined. This application has been implemented on the basis of a specific language which, by means of keywords, allows signals and data to be handled depending on the criterion established.

MANUAL ANALYSIS

For manual analysis, the EVA application provides the analyst with a single screen containing all the information required. This implies presentation of the different windows described above (Fig. 5), as well as the possibility of displaying on screen the historical data for the tube to be analyzed. This allows the analyst to have all the information he requires systematically, which contributes to reducing the possibility of error and the time required for analysis.
The application also provides automatic operations such as calibration, location of all structures for each tube, and a generation of report lines by simply pressing a key, etc.

10 AUTOMATIC ANALYSIS

In our case, the EVA program includes development of a series of functions which may be configured in macros, thus allowing signals to be classified by means of pattern recognition techniques, and different threshold criteria [3].

Parallel to the above, a prototype automatic system with a learning function has been developed on the basis of a philosophy similar to that underlying expert systems.

In order to perform automatic analysis using the EVA program, the macro programming language is used to determine the criteria for analysis. In this way, the system will include a series of macros which when executed will perform all the processes stipulated.

The criteria to be established relate to the following: signal detection, signal parametrization, signal discrimination criteria, and execution of appropriate actions (such as generation of report lines, graphic printer outputs, etc).

11 NEW CONTRIBUTIONS MADE BY THE ENEAS ENVIRONMENT

Eddy current inspection analysis systems are few in number, and are in most cases affected by certain restrictions.

Consequently, the ENEAS environment constitutes an alternative to these systems and provides easy use, simultaneous display of all the information required by the analyst, a reduction in the number of tasks to be carried out by the analyst due to the use of automatic processes, simple and straightforward information management, etc.

Other contributions made by this environment relate to automatic analysis. In this respect, a new language providing an infinite number of possibilities for signal recognition and classification has been incorporated. The philosophy of the learning system, which will allow the efficiency of this type of system to be considerably increased, is also to be underlined.

Finally, the modular structure of the environment will make it possible to incorporate new applications required for the inspection analysis without problems.
12 FUTURE DEVELOPMENTS

Although analysis systems have evolved considerably, finally leading to the environment described herein, considerable progress is to be expected in the near future.

More specifically, the developments foreseen include the following:

- Network integration of the entire inspection system. This implies linking the data acquisition system to the ENEAS environment.

- Further development of the learning function analysis system, which will open up important vistas for automatic analysis processes.

- Centralization of analysis and report acquisition tasks, with connection to the data acquisition systems located in the field by means of telephone networks.

13 CONCLUSIONS

As has been shown throughout this article, the ENEAS environment is an alternative to the few eddy current analysis systems currently available. The advantages offered by the environment are as follows:

- The time required for analysis and for inspection itself is reduced, this implying cost savings.

- Increased reliability and repeatability.

- Intervention by the analysts is both reduced and made more straightforward.

- The environment is a totally modular concept, which allows new applications to be easily incorporated.

14 REFERENCES


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<td>![Graph 1](3.51V 30° 58% 10C)</td>
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<td>![Graph 3](4.70V 385° 82% 18C)</td>
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6.4-15/16
SPECIFIC ULTRASONIC INSPECTION METHODS
FOR STEAM GENERATOR TUBES

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ABSTRACT
Framatome has developed a computerized equipment for inspecting PWR steam generator tubes using a rotating ultrasonic probe. Firstly devoted to the examination of the roll transition zone at the tube sheet secondary side level, the testing system can also operate now for inspections at the tube support plate levels. It is used independently for specific tube inspection, or it can be integrated into a broader-purpose system for sleeve weld testing, etc. The testing results are displayed in real time by means of two eight-level coded colored maps. Some applications, ranging from mockup testing to on-site inspection, are presented below.

RESUME
FRAMATOME a développé une chaine informatisée pour l'inspection des tubes de générateurs de vapeur à l'aide d'une sonde tournante ultrasons. D'abord prévu pour l'examen de la zone transition de dudgeonnage au niveau de la face secondaire de la plaque tubulaire, l'équipement peut maintenant être également mis en oeuvre pour des inspections au niveau des plaques entretoises. Il est utilisé pour la réalisation de contrôles d'expertises ou peut être intégré dans des ensembles plus complets pour le contrôle des soudures de manchonnage. Les résultats de contrôle sont affichés en temps réel sous la forme de cartographies a huit niveaux de couleurs. Les applications, allant du contrôle de maquettes aux interventions sur site sont présentées.
Steam generator (SG) tube bundles are one of the most sensitive items in pressurized water reactor (PWR) coolant systems. In particular the roll transition zone at the tubesheet secondary side level is critical to the integrity of the second confinement barrier. More recently, degradations were also found at the tube support plate levels.

These zones generally undergo nondestructive examination during periodic in-service inspections, using either eddy current testing methods or leak tests. Other methods, such as camera endoscopy or ultrasonic testing, can be also applied.

Utilities rely on accurate and reliable data from these inspections, which support their decisions to leave the situation as it is, perform preventive repairs, or plug the defective tubes.

It is to meet this need that Framatome started a program in 1987 to develop SG tube ultrasonic examination equipment, with the following two goals:

1) To develop tools and methods enabling an overall examination of a SG tube section some 50 mm long, located within the roll transition zone, in less than two minutes, including the following inspections:
   - detection and sizing of circumferential and longitudinal cracks,
   - internal profilometry, and
   - tube wall thickness measurement.

   The testing results would appear in real time as eight-level coded colored maps, with automatic issue of a diagnostic report.

2) To apply these methods to the in-service inspection and follow-up repair operations (such as steam generator tube sleeving), and finally to create multipurpose ultrasonic testing equipment, within the scope of tests carried out in its laboratories.

STEAM GENERATOR TUBE ULTRASONIC INSPECTION SYSTEM

A sketch of the ultrasonic inspection system is shown in Figure 1.

The system's small size enables its fast installation in the reactor building annulus (1 hour) and the probe drive unit is the only part that needs to be inserted into the steam generator bunker.

The system controls all the operations required for the ultrasonic examination, and thus can work in a rather autonomous manner, or it can be incorporated into an integrated system intended for SG tube repairs (sleeving, electrolytic nickel plating, etc.).
The two main parts of the system are:

1) The motion control device (MCD), which coordinates all the inspection sequences and can simultaneously control the rotating unit, the "pusher-puller", and the associated devices (water pump, etc). It synchronizes the ultrasonic data acquisition and also ensures a dialog between the ultrasonic examination system and the central control unit of a total repair system.

2) The ultrasonic probe assembly, which consists of two separate parts:

- A part that ensures the transmission and transformation of the rotating motion into a helical displacement of the probe. Various type of equipment were developed to adapt the accuracy of the probe motion to the application.
  The reliability of this system has been tested several times during on-site inspections.

- The probe head, which is specific to each application and ensures one or several functions accordingly.

Furthermore, an up-grading of the system has been under-taken in order to perform inspections at the tube support plate levels.

ACQUISITION AND PROCESSING OF ULTRASONIC IMAGES

The ultrasonic inspection is based on the principle of rotation of a single or multi-function probe, previously inserted into the tube to be inspected. The probe travels by helical motion with a pitch chosen to be appropriate to the application (0.1 to 0.8 mm) and at a speed of 3 revolutions/second.

The ultrasonic signals are picked up, then processed by a flawdetector. The selected information is transmitted as analog or digital data to a microcomputer, which ensures its acquisition, processing, and storage.

All the results are displayed in real time on two eight-level coded colored maps (C-Scan), showing the developed surface of the tube. Depending on the application, the displayed data may correspond to:

. the time of flight, as measured between ultrasonic reflections (for thickness measurements, radial profilometry, etc)
. the amplitude of an ultrasonic echo selected in a detection gate of the flaw detector (for crack detection, sleeve weld testing, etc.).

In both cases, a color scale enables the operator to appreciate:

. the sound path difference between echoes,
. the relative amplitude of a reflection, expressed for a fixed gain, in percent of the screen height.
Several applications for this new ultrasonic inspection system are described below.

APPLICATIONS

THICKNESS MEASUREMENTS

Optimization of tube service life vis-à-vis the plugging criteria requires highly accurate knowledge of both crack length and tube-wall thickness at the same position.

To meet this need, a method was developed to determine the effective wall thickness in the straight portions of the steam generator tubes, close to the roll transition zone, with a measurement accuracy of ± 0.01 mm. Detection and length sizing of longitudinal stress corrosion cracks is to be carried out at the same time.

For example, the color-coded map in Figure 2 show the wall thickness zones observed in sections of Inconel 600 SG tubes having outer diameters of 3/4", with a cross-section representation Figure 3. The wall thickness map indicate greater variations on the circumferences than along the tube walls in the longitudinal direction.

RADIAL PROFILOMETER AND CIRCUMFERENTIAL CRACK DETECTION

The simultaneous acquisition of the internal profile of the tube, in addition to the detection and sizing of circumferential cracks, enables the location of these defects with respect to the tube distortions observed. As an example, some results of an on-site inspection performed on a set of 15 steam generator tubes are presented in Figures 4 and 5.

The figures show the ultrasonic recordings for the radial profilometry and circumferential internal crack detection, ejected in the zone just above the tube sheet.

Moreover, the UT examination indicates the presence of a circumferential crack with an angular width of 140° located 4 mm above the tube-sheet secondary side at the level of the maximum distortion observed on the tube (Figure 5). This ultrasonic indication was detected by eddy current testing with a 160° circonferential extension.

The radial profilometry recording (Figure 6) indicates a localized tube deformation above the secondary side of the tube sheet, which is characterized by tube denting zones (violet-colored area), as well as a lateral displacement of the tube with respect to its initial axis (orange-colored area).

An analysis of the radial profilometry ultrasonic recordings enables identifying the region involved as a tube cross section (Figures 7 and 8).

The axial profile calculation, after averaging the radius measurements over the circumference, shows good correlation between the ultrasonic profilometry results and those obtained by eddy current testing using an axial probe designed by Framatome.
DETECTION OF LONGITUDINAL CRACKS

The roll transition zone of steam generator tubes is particularly sensitive to the stress corrosion phenomenon that may lead to longitudinal and circumferential cracks. Ultrasonic inspection enables fine detection of these internal or external cracks (Figures 9 and 10) even after electrolytic nickel plating as well as complex external longitudinal crack lattices.

Good correlation is observed between the size of these ultrasonic indications and those from other NDE methods, such as the microfocus X-ray technique (for internal cracks) or liquid penetrant testing (for external cracks).

ULTRASONIC EXAMINATION OF SMALL DIAMETER TUBES

The system described above is presently used for tube inspection with diameters down to 12 mm within the scope of the European Fast Reactor project for radial profilometry and thickness measurements. It has also been employed for the on-site UT inspection of top thimble guide tubes for wear detection and sizing.

CONCLUSIONS

An automated ultrasonic real-time imaging system for tube-wall thickness measurement, internal profilometry, and flaw detection, has been developed.

This system has been successfully applied for on-site specific inspections within an industrial environment. In all cases the ultrasonic acquisition time was less than two minutes per tube.

It should be pointed out that analysis of this new set of ultrasonic inspection results should also improve understanding of the in-service behaviour of these materials and components.

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FIGURE 1: On-site ultrasonic equipment set-up

FIGURE 2: Ultrasonic tube-wall thickness measurement

FIGURE 3: Tube thickness cross section
FIGURE 4: Ultrasonic radial profilometry

FIGURE 5: Circumferential crack detection
FIGURE 6: Ultrasonic radial profilometry
Detection of a local deformation

FIGURES 7 and 8: Tube profile cross-section display
FIGURE 9: Ultrasonic detection and location of longitudinal cracks
FIGURE 10: Ultrasonic detection and location of circumferential cracks
ULTRASONIC INSPECTION METHODOLOGY

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ABSTRACT

Steam generator tubes are known to be susceptible to stress corrosion cracking (SCC). Primary water SCC (PWSCC) and more recently secondary water SCC (SWSCC) have been observed in some Belgian plants. To help dealing with these problems, Laborelec developed an ultrasonic (UT) inspection system. It has been used for the last two years on a sampling basis in several plants. The field and laboratory measurements confirmed the advantage of using UT for the early detection of small circumferential cracks while an excellent correlation was demonstrated between eddy current RPC and UT for axial PWSCC.

RESUME

Les tubes de nombreux générateurs de vapeur présentent une sensibilité à la corrosion sous tension. Ce phénomène de fissuration en eau primaire et plus récemment en eau secondaire a été observé dans certaines centrales belges. À cet effet, Laborelec a développé un système d'inspection par ultrasons. Cette unité a été utilisée dans plusieurs centrales sur un échantillon de tubes. Les mesures réalisées sur site et en laboratoire confirment l'avantage des ultrasons pour la détection des petites fissures circonférentielles. Une excellente corrélation entre les méthodes à la sonde rotative par ultrasons et par courants de Foucault a été mise en évidence pour le dimensionnement des fissures axiales.
1. **INTRODUCTION**

Although axial primary water stress corrosion cracking (PWSCC) is a well known problem for the Belgian plants, no circumferential stress corrosion cracking was reported by the Rotating Pancake Coil (RPC) eddy current in service inspection. The first evidence of circumferential PWSCC was found on a tube pulled for axial PWSCC (in 1986 from the steam generator B of Doel 3). A small circumferential crack (length : 3.5 mm - depth : 65 %) was found by destructive examination within a shallow groove in the roll transition. No evidence of these cracks could be found on the Laborelec eddy current RPC results. The reason could be identified since the circumferential crack was located at the bottom of several "large" axial cracks.

This early experience identified the limitations of the eddy current techniques for circumferential SCC. As the focal spot of an ultrasonic beam could be made significantly smaller than the sensitive area of an eddy current coil, it was expected that an improved detectability would be obtained with a rotating ultrasonic probe.

A first commercially available UT system (NUCON) was tested on field for the measurement of small fretting wear located at the support plates in the preheater area of model E steam generators. This inspection system used a probe with a rotating mirror. It was also applied during the initial laboratory phase for the inspection of the inner surface of nickel plated tubes [1]. The significant inspection time and the lack of digitization of the RF waveform were the main reasons for the development of the LABORELEC UT system.

The first in-service inspection with the new UT system was performed at Doel 3 in September 1989. The high inspection rate and an improved signal to noise ratio confirmed that UT could be used as an industrial inspection method for the tube bundle. The UT system was further improved, and the present capabilities allow the inspection of 60 mm located at the top of the tube sheet in less than 60 seconds.

While the development was made to detect and size PWSCC circumferential cracking, very few occurrences have yet been evidenced; instead, less expected secondary side circumferential SCC was recently detected and sized by the UT system.

2. **DESCRIPTION OF THE METHOD**

The UT method is based on a sound beam that is directed towards the tube material following a specific incident angle. When the sound beam hits the water to tube interface, a part of the sound beam enters the material following one or two refracted angles. A material discontinuity placed on the beam trajectory will reflect some proportion of the beam energy in function of the defect inclination and surface. The transducer frequency, focus and angle will be a function of the requested detection and sizing capabilities.
The LABORELEC UT system is based on a probe design with interchangeable heads for either wall thickness measurement, axial or circumferential crack detection. A three transducers head is also available for the simultaneous inspection following the three search paths. The transducers were selected with a central frequency between 12 MHz and 18 MHz. A UT computer aided design (UT-CAD) program developed by the NDE laboratory of the Electrical Power Research Institute (EPRI) was used to select the transducer angles. It allowed the visualization of the sound paths for different defect localizations and dimensions and to study the influence of the roll transition area. The UT-CAD program was at the origin of significant improvements in detection capabilities and the signal to noise ratio. Also, a Laborelec original design enables the same transducer (either for axial or circumferential cracks) to simultaneously measure the inner tube profile. This capability provides two advantages:

- it ensures the perfect localization of the cracks in function of the tube profile,
- it cancels the influence of the probe to tube spacing through an automated synchronization of the detection windows.

A new pusher-puller was also developed for the UT system. It is fully programmable and is able to translate the probe following different scanning patterns.

The probe for axial or circumferential PWSCC detection is usually moved following an helicoidal path with a pitch of 0.7 mm. UT data are acquired up to one point every 0.2 mm on the circumference. During the data acquisition, the UT repetition rate is synchronized with the angular and axial locations of the transducer(s).

The ultrasonic instrument is based on a TOMOSCAN system (from RDTECH - Quebec). It is controlled from a HP 9000 desktop computer with the help of a VME bus interface. The UT data (amplitude, time of flight and/or digitized RF waveform) are transferred with a 1 Mbytes/s link designed jointly by RDTECH and Laborelec. This capability provides the high acquisition rate requested by the probe rotational speed.

3. QUALIFICATION

The UT system was qualified for the detection and sizing of axial PWSCC using pulled tubes and laboratory induced defects. The qualification for the circumferential cracks could only be achieved on EDM notches. The method was optimized for the length measurement as required by the plugging criteria established for the Belgian plants.

Figure 1 shows the relative amplitude obtained for external axial and circumferential EDM notches of increasing depths. In comparison, the amplitude of real axial PWSCC assumed to be 100 % through wall is shown on the same figure.
The qualification results demonstrated the high sensitivity for small volume cracks like FWSCC.

The laboratory experiments confirmed that the smaller focal spot of the UT beam could significantly improve the detection of small circumferential corrosion cracks even in the presence of multiple axial cracks. Circumferential cracks of 2 mm length and 100 % penetration depth could be identified and sized in length. Although no pulled tubes could be used to confirm these measurements, recent field evidence of secondary side circumferential SCC demonstrated a detection capability far beyond the eddy current rotating pancake coil. This advantage is somewhat reduced by the simultaneous higher sensitivity for spurious indications from inner surface scratches or grooves. To this aim, special filtering algorithms were conceived. They use a combination of the time of flight and the signal amplitude patterns to differentiate between spurious and real indications.

The length measurement were demonstrated within +/- 1.8 mm for 100 % through wall axial cracks [2]. A complementary qualification program is currently going on for cracks of smaller depth originated either from inside or outside of the tube surface. This accuracy is strongly related to the analysis technique. Indeed, the UT technique on small thickness material suffers from the multiple reflections and refractions that lead to a large number of indications. Fortunately, an "expert system" approach combining the different available information was able to transform this disadvantage into a powerful screening technique.

4. IMPLEMENTATION

The UT system is divided into two main parts: the data acquisition and the data analysis. A trailer contains the analog and digital equipment required to perform remotely the inspection. With the exception of the new pusher-puller, UT probe and instrument, all the equipment are identical with the ones needed for the eddy current bobbin coil and RPC inspection systems [3].

The UT data are verified during the acquisition process by a "quality control" computer before being recorded on a laser disc (Figure 2). During the measurement sequence, the operator is provided with real time A and C scans. Simultaneously, the tube profile is displayed for an easy probe localization.

For the analysis, the laser disc is shared with the data acquisition computer and a high speed colour plotter provides the hard copy printouts for the final report (Figure 3). Figure 4 shows a typical colour printout where the tube profile, the UT signal amplitude and the time of flight are easily identified. The two upper displays use the screening technique to improve the defect identification and sizing. Although the analysis uses different signal processing algorithms, the defect identification and the length sizing are still performed by a human analyst. Full automation of the analysis process is currently in development.
5. CONCLUSION

The latest in-service inspections demonstrated an average measurement cycle of 50 to 60 seconds per tube including the manipulator displacement from tube to tube. A sample of about 250 tubes with axial cracks was measured on-site with the eddy current RPC technique and the new UT system. All the UT measured lengths were within +/- 1.5 mm of the RPC results. The same results were obtained from the inspection of the tubes repaired with the nickel process (Doel 3). Some indications of circumferential secondary side stress corrosion cracking were recently observed. A comparison with the eddy current rotating pancake coil confirmed the improved detectability of the UT system.

The field and laboratory results obtained with this UT inspection system demonstrated the advantage of applying the ultrasonic technique for the detection and sizing of small volume cracks like SCC. Also, the small focal spot of the UT beam provided a clear advantage for the detection of circumferential PWSCC in the presence of multiple axial cracks. With an average rate close to 60 tubes per hour for the top of the tubesheet area, this UT system can be considered as an industrial tool for the inspection of steam generator tubes.
FIGURE 1: RELATIVE SCREEN AMPLITUDE (SA) FOR AXIAL AND CIRCUMFERENTIAL EDM NOTCHES AND AXIAL PWSCC FROM FIELD RESULTS
FIGURE 2 : ACQUISITION SCHEME
FIGURE 3: ANALYSIS SCHEME
FIGURE 4: COLOUR PRINTOUT OF CIRCUMFERENTIAL OD STRESS CORROSION CRACKS
REFERENCES


[3] Dobbeni D., "Five years experience with a computer controlled measurement system for the inspection of PWR steam generator tubes", 1st Congress COFREND on Non Destructive Testing - Nice, November 1990
ROSA III, A THIRD GENERATION STEAM GENERATOR SERVICE ROBOT TARGETED AT REDUCING STEAM GENERATOR MAINTENANCE EXPOSURE

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ABSTRACT

ROSA III is the third generation of maintenance and inspection robots designed, manufactured and operated by Westinghouse. An integrated system approach built around a network architecture has led to many areas of improvement. The single 16 mm digital network cable replaces the bulky analog cables, reducing setup time and containment penetration requirements. The robot arm was configured specifically for steam generator service and has the capability of remote, no entry arm and tool loading. End effector control has been integrated with robot control to provide a single point source for data acquisition and maintenance services. A power distribution system adds a consistent ground plane and filtered power for both in and out of containment components. A graphics based user interface ties it all together for point source control.
BACKGROUND

The Westinghouse Nuclear Service Division has employed two delivery robots for the past eight years. The simplest is a two degree of freedom robot (WL-2) that has a design goal of delivering Eddy Current Acquisition and Mechanical Plugging services. The delivery capability of this robot is 111 N at a reach of 2.36 M. The robot is somewhat limited because two degrees of freedom cannot provide general end point approach or orientation alignments for maintenance tools which require camlocks. But for delivery of the above two services the design goal is very much satisfied. The second robot is ROSA I, its design goal is to provide the heavy duty maintenance operations on steam generators and reactor vessels. ROSA I has six degrees of freedom, has a reach of 2.36 M, and a load capacity of 222 N. The actuators of ROSA I are electric motor driven through a 200/1 harmonic drive. There are 677 N-M actuators at axes 1, 2 and 3 and 338 N-M actuators at axes 4, 5 and 6. These are arranged in a elbow configuration with axes 2, 3 and 4 providing the elbow shape. The services provided by ROSA I include Eddy Current, Mechanical Plugging, Sleevings, U-bend and Support Plate Heat Treating, Plug Removal and Tube Removal. ROSA II, having six degrees of freedom, is capable of generalized tool placement and orientation to any point in space within its reach envelope. ROSA II is a extension of ROSA I. A mast, carriage and rotating base were added to provide inspection and maintenance services on reactor vessel shells and nozzles. The new system, ROSA III, has a goal of replacing all of the steam generator services provided by the WL-2 and ROSA I robots plus provide a single platform for data acquisition and tool control.

SYSTEM LEVEL DESIGN

ROSA III is our new integrated service system, it provides all the delivery services of the WL-2 and ROSA I robots and in addition adds a stand alone power distribution and tool control system. The block diagram for the system is shown in Figure 1. The basic components are: a user interface workstation, robot and tool Input/Output (I/O) control system, a robot arm, power distribution system and interconnecting cabling. The system is transported by truck to the power plant site in three 1.9x3.8x1.4 M boxes. Once at site it is setup in three locations; the user interface workstation is placed outside the primary containment boundary in a four station control trailer in the parking lot somewhere within 474 M of the robot, the robot and I/O control hardware is placed inside the primary containment building behind the biological protection wall within 94 M of the robot, and the arm and end effectors are in the Steam Generator channel head.

The architectural focus for the system was to provide a platform that could expand and contract to the maintenance and inspection needs of a local site. This objective was accomplished by building the system around a 10Base5 Ethernet network whereby smart nodes could be added or removed upon demand. The robot and I/O control network as seen in Figure 1, contains two nodes, one
for the user interface and the other for the robot and I/O controller. This single 0.25 CM network cable represents a significant reduction in the number of containment wall penetrations needed for control cables. The previous design required four robot control cables plus one control cable for each end effector. At this writing there are three additional smart nodes, a laser weld sleeving node, a shot peening node and the system maintenance and diagnostics node. A second network, a data analysis network, connects the four user interface workstations to the data analysis trailer.

**SYSTEM HARDWARE DESIGN**

The User Interface is a Silicon Graphics 4D25G RISC based workstation. This workstation is considered the top of the line for graphics, and in addition is capable of providing 16 MIPS (million instructions per second) for the number crunching duties. It provides the focus for all the maintenance and inspection tasks conducted on the steam generator. A typical control station setup is shown in Figure 2. Here we see the workstation, keyboard, optical mouse, Emergency Stop pendant, a four monitor video dis-
Figure 2. **ROSA III Control Station Setup**

play and an audio communications setup. The video monitors are setup at site to provide the operator with the following views: A global view of operations inside the steam generator channel head coming from a pan, tilt and zoom camera that is attached to the manway, A tool end point view, which is a point in space about which the robot joystick positions and, Two camlock tip views for docking the tool camlocks. Audio communications at site are established between the operator, containment support worker, Health Physics, Quality Assurance and the site coordinator. The E-Stop pendant provides power on/off and computer reset for the robot and I/O control system along with the analog kill for all tool and robot functions.

The control hardware, shown as the top two boxes in Figure 3, are 0.8x0.8x0.7 M aluminum boxes, one containing the robot and I/O computer hardware and the other the robot amplifiers. The boxes are sealed, latched and wrapped in polyethylene for contamination protection and are free released at the job completion. They weigh 284 and 240 N respectively and are carried into and out of the containment building by two people. Internal cooling for the control boxes is provided by vortex cooling tubes, one in each box.
Figure 3. ROSA III Control Hardware Setup

The cooling system consumes 0.75 CMM (Cubic-Meters/Minute) air when the control boxes are located in the design environment of 38 degree Centigrade. The robot and I/O control functions are performed by a VME, 68030 based single board computer running under a real time operating system. The additional I/O computer hardware includes a sixteen channel analog output board, a thirty-two channel analog input board and a one hundred twenty eight channel digital I/O board. The robot control hardware includes three dual axes control boards, each with a piggy back board containing four Resolver to Digital (R/D) converters. The auxiliary hardware shown in Figure 2 are; The hydraulic power supply and, The universal I/O
interface. The hydraulic power provides high pressure hydraulics to the robot base plate camlocks, which secure the robot to the tubesheet and to other process end effectors requiring high pressure camlocks. The universal I/O interface in conjunction with the I/O computer hardware represents a significant step toward reducing the equipment inventory needed for maintenance and data acquisition operations. The process and tool control functions of all our previous stand alone controllers, which do not have dedicated nodes, have been integrated into this single controller and all maintenance and data acquisition end effectors have been wired to interface to the single universal interface box.

The newly designed robot arm is shown in Figure 4. The arm is shown attached at the base to a test fixture with the Mechanical Plugging System End Effector mounted on the tool end. The arm has been designed specifically for operations inside a steam generator channel head. The configuration shown is the normal operational configuration for tubesheet operations inside the steam generator. It contains six Degrees Of Freedom (DOF); Axes 1, 2 and 6 provide the X and Y planer positioning as well as the roll orientation and tube approach angle, Axes 3 and 5 provide the elevation and pitch

Figure 4. ROSA III Delivery Arm

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orientation and axis 4 provides the yaw orientation. It is a two piece arm, with axes 1,2 and 3 forming one half and axes 4,5 and 6 the other. This feature allowed us to stiffen the links against mechanical deflections and use high torque actuators at axes 4,5 and 6 while maintaining a two person handling weight. The actuators are designed for a five year maintenance free life, which will reduce our radiation budget for maintenance. They are electric motor driven through a set of 200/1 harmonic gear reducers. The motors are current limited to produce 789 N-M actuator output. They have a torque rating of 2030 N-M but will ratchet the harmonics at 981 N-M. When in operation with the design 355 N tool load, the motors run at about a 20 percent duty cycle, relative to the 789 N-M. The operational duty cycle is low for two reasons; 1) The high torque "lift" actuator (#3) is placed at one half the full reach which reduces the operational and gravity torque requirements and, 2) Axes #1 and 2 are placed in an orientation where moments not torques need to be transmitted. Axis #3 is air cooled during normal operations and axes 6 is air cool for loading, if the arm is loading is to a Series 80 Combustion Engineering generator or larger. The arm is equipped with linear potentiometers on the base for docking and levelometers on the tool end for pitch and yaw tool orientation corrections. Each actuator is also equipped with tandem resolvers, one resolver is attached to the actuator output and the other is attached to the motor. These are utilized in closed, dual control loop to provide simultaneous smooth joint motion and positional accuracy. The arm is designed for "no entry" loading and tubesheet mounting. The forward and inverse kinematics are solved to provide end point positioning with either the base end fixed or the tool end fixed. The no entry loading is accomplished by first; attaching the tool end to a worm gear driven hing loaded fixture that is bolted onto the manway then; executing a robot motion file that directs the base end through a series of preplanned end point motions while simultaneously rotating the loading fixture. This sequence finishes with the base two inches down from the tubesheet, at which time the operator switches the robot control into a teleoperated joystick mode and with the aid of the manway camera inserts the base camlocks into the tubesheet and hydraulically expands them to secure the arm to the tubesheet. Once the base is secured the tool end of the arm is released from the loading fixture and is ready to accept an end effector. The entire loading operation at site on the platform with full anticontamination clothing, plastics and bubble hood takes about thirty minutes. The same loading operation takes ten minutes in our development laboratory. The containment support worker is out of the manway radiation shine for all but five minutes of the loading time, which is typically the time required to place the arm onto the loading fixture. The high torque actuators at axis 4,5 and 6 were specifically included to provide the lift needed for this operation. The arm has a design capability of 355 N at 3.3 M.

The power distribution system, shown in block format in Figure 1, services four systems. The system interfaces site power at the filter skid which is located somewhere in the vicinity of the con-
control trailer in the parking lot. The power filter provides transient protection and the location of the single point ground plane for the tool and robot control system. The main distribution box, located inside containment provides breaker protection as well as distribution to local robot and tool power distribution boxes. The containment power distribution boxes are all sized for a single person carry.

System Software Design

The software design is centered around a integrated system concept utilizing a distributed control network. This is implemented with the user interface workstation providing a single point of control for all operations and the attached network nodes providing the real time control of processes, end effectors and robots.

The workstation provides the high level robot path planning, tool control and data base management. The software is partitioned

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Figure 5. ROSA III User Interface With One Step Drill Displayed

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into separate executable routines, there is; one for system initialization, one for robot control, one for data base management, one for maintenance and trouble shooting, one for network management and one for each application. Data is passed between executable routines through shared memory. The user interfaces for the routines share the computer screen space. This is illustrated in Figure 5. This Figure shows the interface with robot control displayed on the right section on the screen, tool control for the Tube Recovery Drill End Effector on the lower left section of the screen and a display of the data base showing which tube locations can be reached from the selected approach angle and arm configuration in the upper right section.

The robot control section of the computer screen displays, joint angles, reference coordinates, tool point pitch and yaw levelometer bar graphs, base linear potentiometer bar graphs and joint torques on a 200 ms (millisecond) update rate. The available robot delivery controls include 1) Individual axes jog, 2) Tool and base Local or Global X, Y, Z, Roll, Pitch and Yaw motions through joystick control 3) Automated tubesheet motions and, 4) Teach and repeat. All motion selections are mouse picked and directed. The jog and joystick motions are based on end point velocity control giving the user a much better mental feeling of robot motion control when operating in the teleoperation mode. The tubesheet motion is automated and qualified safe, prior to field operations. The safe tube to tube motion paths are calculated by a set of routines that access the data base for environmental geometry, robot geometry, end effector geometry, and robot limits, checking all body to all body collisions and robot limits to generate a list of tubes that can be safely reached for all available tool approach angles. These list are stored in the data base and are accessed appropriately when the operator selects a tube for positioning. Tube selection is done from the Tubesheet menu, (See Figure 6, lower right corner) tubes can be input either by incrementing or decrementing the row/column picks with the mouse, input with the keyboard, picked with the mouse off the tubesheet screen display or input as a preprogrammed sequence. Once the tube is selected the workstation builds a motion file composed of global end point coordinates and orientations along with some other control parameters and sends this file over the network to the robot controller for execution.

Generalized preprogrammed motions are executed through the robot repeat menu shown in Figure 5 at the lower right hand section of this Figure. A selection from this menu will generate a file of motion records which are sent over the network to the robot controller that will execute under the command and control of the operator. Some of the major features of the robot control from this section include; 1) Automated docking of the base utilizing the base linear potentiometers, 2) Calibration of the base with the tool point at a known location, 3) Calibration of the tool position with the base at a known location, 4) Calibration of the manway location with the base fixed and the tool end mounted to the loading fixture, 5) Automatic generation of tool load files.
Figure 6. ROSA III User Interface With Plugging Shown

from the manway to the tubesheet with a known manway calibration and data base geometry information, 6) Auto return to the manway from any position on the tubesheet for consumable change outs, 7) Execution of any motion paths generated by the jog and joystick motions, input through the keyboard, or input from the robot simulation routines.

Tool control from the workstation is standardized which allows quick integration of a new end effector control. A set of generic C language based routines are modified to tailor the control interface to the application. The tool control user interface for the Mechanical Plug End Effector is shown on Figure 6 at the lower left section of the screen. The standard interface includes a title and version bar, a set of process specific icons, an analog display and control, a automatic process execution and a manual execution. Scanning the above Figure we see first the version bar which shows the name of the process and when it was last modified. Below that is a set of selectable icons specific to an Mechanical Plug End effector which can be paged if more selections are needed. Below that are three boxes, the one on the left is the
Figure 7. ROSA III User Interface With Eddy Current Shown

analog Input control and analog output display, the one in the middle is the auto cycle selection, which allows selection of auto tool control with 200 ms display of status, the one on the right is the digital control panel, from here the operator can select a function and turn it on or off. For comparison, Figure 5 shows the same architecture as above only with the One Step Drill end effector control interface. The upper left section of the user interface can also serve to display time histories of process data as illustrated in Figure 5, where we see pressure and expansion displacement parameters for the Mechanical Plug Process.

Data acquisition is a separate process, it occupies all the left section of the user interface. Figure 7 illustrates the data display screen for Eddy Current Acquisition which is one of eight screens needed for this process. Moving though all of data acquisition is beyond the scope of this paper, but the features that are unique to this control are; The data is acquired with a real time controller, Transmitted over the Ethernet network for review by the operator at the interface workstation for data quality, Then sent over the second Ethernet network for analysis and sto-

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7.1 THE BELGIAN EXPERIENCE OF SG CHEMICAL CLEANING

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P. Havard, Belgatom/Electrabel, Brussels, BELGIUM
J. Maquinay, Belgatom/Electrabel, Brussels, BELGIUM
J. Brognez, Belgatom/Electrabel, Brussels, BELGIUM

7.2 CHEMICAL CLEANING FOR STEAM GENERATOR SECONDARY SIDE IN JAPANESE PWRs

S. Kimura, Kansai Electric Power Co. Inc., Osaka, JAPAN

7.3 SPANISH RESEARCH ACTIVITIES ON STEAM GENERATOR TUBES DEGRADATION

D. Gomez Briceno, Ciemat, Madrid, SPAIN
A.Mª. Lancha Hernandez, Ciemat, Madrid, SPAIN
Mª.L. Castano Marin, Ciemat, Madrid, SPAIN

7.4 AUTOMATIC MEASURES TO COPE WITH STEAM GENERATOR TUBE RUPTURES

K. Kotthoff, GRS, Cologne, GERMANY
A. Schütte, RWE Energie AG, Biblis, GERMANY

7.5 STEAM GENERATOR LIFE TIME EXTENSION BY MAINTENANCE AND REPAIR

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F. Pötz, ABB Reaktor GmbH, Mannheim, GERMANY
THE BELGIAN EXPERIENCE OF SG CHEMICAL CLEANING

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ABSTRACT

Chemical cleaning of the steam generators has been implemented at Tihange 1 and Tihange 2 nuclear power stations, using respectively the EDF and the Siemens-KWU processes.

In both cases extensive qualification programs have been carried out on plant specific materials in order to demonstrate the innocuity and the effectiveness of the cleaning procedure.

The cleaning operations have proven successful. In addition to iron and copper removal, which results are given hereafter, other metals and salts have also been removed. Corrosion monitoring during the application confirmed the low corrosion rates.

Some unexpected effects have been observed after the first cleaning of Tihange 1, but not in Tihange 2.

The radioactive contamination of the effluents has been extremely low which simplified the waste treatment. The eddy current tests show a beneficial effect.

RESUME

Les générateurs de vapeur des unités nucléaires Tihange 1 et Tihange 2 ont été nettoyés par voie chimique, respectivement par les procédés EDF et Siemens-KWU.

Dans les deux cas un programme de qualification important a été réalisé pour démontrer la non agressivité et l'efficacité de la procédure de nettoyage.

Les nettoyages ont été couronnés de succès. Les résultats d'élimination de fer et cuivre sont donnés. D'autres métaux et sels ont également été éliminés. Les mesures de corrosion pendant l'application confirment les faibles vitesses de corrosion.

Après le premier nettoyage de Tihange 1 quelques problèmes chimiques ont été rencontrés au redémarrage. Aucun effet n'a été constaté à Tihange 2.

La contamination radioactive des effluents était extrêmement faible ce qui a simplifié le traitement des effluents.

Les examens aux courants de Foucault ont démontré un effet bénéfique.
1. INTRODUCTION

Seven PWR's are in operation in Belgium. After some years of operation all the SG's of the first generation units (Doel 1/2 and Tihange 1) evidenced indications of minor denting at the tubesheet and the tube support plates.

The evolution however is rather low and not a serious concern in relation with the predicted lifetime of the steam generators. In 1984 however some indications of IGA were found in Tihange 1. As a 10-year revision was approaching the chemical cleaning of the steam generators was taken into consideration and a qualification program started in 1986.

The first years of operation at Tihange 2 were characterized by several condenser leakages and relatively large amounts of copper in the condensate and feedwater. Minor denting at the tube support plates developed progressively and reached values of about 200-300 μ in 1988.

In 1989, it was decided to replace the condenser with a stainless steel tubed one. This was also the right time to consider a complete removal of all Copper deposits from the system and to bring the steam generators in a copper-free environment. Chemical cleaning was taken into account. It was further contemplated to remove iron oxides to the maximum possible extent.

This chemical cleaning had to be incorporated in the normal refuelling outage period and the whole operation was postulated to be completed within 80 h.

2. CHEMICAL CLEANING PROCESSES

Tihange 1 is owned by both Belgium (Electrabel) and France (EDF). In 1985 EDF developed a chemical cleaning procedure based on the following chemicals and parameters.

- iron removal: gluconic acid 7,0 %
citric acid 4,0 %
ammonia 2,0 %
inhibitor P6 0,8 - 1,2 %
pH 3,3
T 85°C
Time 170 hours

- copper removal: increasing pH of solution to 9,5 and addition of an oxidizing agent (initially sodiumbromate, later changed to air bubbling through the solution). Copper dissolution is achieved at low temperature (about 50°C).
- finishing step to remove sulphur residues from Inconel 600 by citric acid, monoethanolamine to increase pH to 3, inhibitor with sulphur, later monoethanolamine to pH 9,5 - 10 and air bubbling.

The temperature required for the iron oxide step is obtained through an external heating system. The chemicals are recirculated through the steam generators by an external loop.

Tihange 2 selected the KWU chemical cleaning process consisting of two different single-step processes for sludge removal, i.e. a high-temperature iron process and a copper process at ambient temperature. The chemicals and parameters are as follows:

- iron removal: NTA 4,4 %
  NH4OH to increase pH to 9,6
  N2H4 50 g/l
  T°C 170°C
  time 1-2 hours

- copper removal: EDA 1 %
  pH 9,5 - 11,5
  catalyst 0,5 %
  T°C 50°C

The temperature required for the iron oxide step is obtained using the primary system as a heat source. The chemicals injected into the steam generator are mixed by a short steam-off. No external heatup or recirculation systems are used.

The copper cleaning step can be applied at any time during the refuelling outage.

3. QUALIFICATION PROGRAMS

At the time the procedures for chemical cleaning were selected both EDF and KWU had already gone through an extensive qualification program. Specific testing for local conditions was done on some materials.

Especially for Tihange 1 the possible effects of residual sulphur on Inconel 600 were assessed as Tihange 1 had known some adverse effects of sulphur in its first cycle.

For Tihange 2 the plant specific programme aimed at establishing the application procedure and to carry out sludge dissolution tests as well as material compatibility tests (main structural materials and combinations such as weldings and couplings). In both cases the general corrosion of the structural materials was
considered low enough to avoid problems. The steam generator tubing material (Alloy 600) showed neither selective corrosion nor surface roughening.

Bare carbon specimens generally show uniform general corrosion to a small extent (max 60 μ) but specimens covered with sludge have a weight loss corresponding to 10-15 μ only.
4. PROCESS APPLICATION

4.1. Tihange 1

The EDF procedure needs an external recirculating loop for the chemicals. This loop was constructed during the decennial revision of the unit. It was tested in November 1986, with hot demineralised water and ammonia to simulate chemical injection. The actual chemical cleaning of two steam generators was performed in 1988. The third SG has been cleaned in 1989. The first cleaning (1988) operation was completed within 30 days as shown in Figure 1. The intervention on the third steam generator in 1989 took 22 days.

4.2. Tihange 2

The process was planned as to minimize the external equipment and time-consuming preparatory works. The only external equipment consisted of portable dosing tanks, charging pumps and valves. All temporary piping used flexible connections. This work was carried out prior to plant shutdown and took five days to complete (see Figure 2). During the shutdown operation the cooldown was interrupted at 175°C for 4.5 hours. The temperature was controlled by the primary system. The actual time schedule is shown in Figure 3. After filling the SG's, evaporation was performed by opening the steam relief valves by 25-30 % and monitored by observing the SG water level. About 1.5 hours after dosing the iron oxide solvent, the sample analysis showed solvent exhaustion. Then the cooldown procedure was restarted at about 27°C/h until 130°C was achieved.

The copper removal was initiated after complete draining and rinsing of the SG's. It took 33 hours to fill SG 04 because a leaking tube had to be plugged. All Cu-removal operations have been completed in 3 days, including the 33 hours delay. The process was monitored by sampling and analyzing the solvent for Cu during the application.

5. RESULTS

The main results of both chemical cleaning processes are presented in Table I. Figure 4 shows a comparison of the evolution of Fe concentration during application and autoclave tests. A good correlation is found. In Tihange 2, two fiberscope inspections were performed, before and after the application of the copper step. Before the copper step, the tubes showed a clean, slightly shiny appearance. The tubesheet was also clean except some particles and spots.

After the copper step and tubesheet lancing the SG's were inspected at four levels in the cold and hot leg side. Again the tubes looked clean and slightly shiny.
There were no deposits of any kind so far as could be inspected. Corrosion monitoring on Tihange 1 confirmed the data of the qualification tests.

On Tihange 2 monitoring by coupons was not possible but assuming that the pressure above the saturation pressure is entirely due to corrosion hydrogen from C-steel surfaces would give a 7 μ thinning.

6. WASTE TREATMENT

The waste treatment for both processes was studied extensively. The most important study was done before the first application in 1988 of the EDF-process at Tihange 1. Different alternatives such as evaporation, lyophilisation, atomisation at low or high temperature, incineration and ion exchange were considered from a theoretical and a practical point of view. In conclusion of laboratory studies the use of chelating ion exchangers was recommended.

For Tihange 2 tests were performed on simulated and real waste solvent from the Almaraz plant in Spain. Once contaminated iron solvent possibly requires treatment but in this instance there were no chemically related restrictions for release. However the copper solvent has to be treated to remove the copper before discharge.

A mobile unit for demineralisation has been built. It operated at about 9 m³/h. It consists of 1 filter and 2 demineralisers of 2 m3 resins each, operating in series. The average results in Tihange 2 are given in Table II.

7. CHEMISTRY AT STARTUP

After cleaning 2 SG's at Tihange 1 in 1988 some problems occurred in chemistry control during the startup phase:

- pH remained low in cleaned SG's (~ 5.5)
- the conductivity was high with important amounts of sulphate and fluoride

<table>
<thead>
<tr>
<th>Conductivity</th>
<th>9 μs/cm</th>
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<tbody>
<tr>
<td>SO4</td>
<td>600-2800 ppb</td>
</tr>
<tr>
<td>F</td>
<td>&gt; 100 ppb</td>
</tr>
<tr>
<td>Acetate</td>
<td>300-400 ppb</td>
</tr>
</tbody>
</table>

After 72 hours of maximum blowdown the conductivity decreased to 2 μs/cm. The anion species mentioned above could be detected in the SG blowdown for 1 month. The amount of suspended solids was also significantly higher in the cleaned SG's.
These observations were not found after the 1989 cleaning when the secondary side of the SG was rinsed continuously from the moment the preheaters were available.

At Tihange 2 the chemistry at startup did not differ from the previous ones and a good steam generator water chemistry was soon obtained.

8. EDDY CURRENT TESTING IN LATER CYCLES

Chemical cleaning is expected to also clean, at least partially, the crevices of the tubesheet and the gap between tubes and tube support plates. This could have a beneficial effect on denting rates which might decrease.

The eddy current testing has not provided an exhaustive answer to this hypothesis but statistically the results seem favourable in Tihange 2 where the average rate of 40 ì per year before cleaning has been reduced to 10 ì per year after cleaning.

This decrease probably results from the elimination of sludge from the tube support plates and of salts which concentrate in these places.

9. CONCLUSIONS

Chemical cleaning has successfully been applied in both Tihange 1 and Tihange 2. Utilities show growing interest in chemical cleaning, since they believe that it can be beneficial and presents no associated degradation effects. Future efforts should be directed to reduce the duration of application and minimize wastes.
### TABLE I
RESULTS OF CHEMICAL CLEANING

<table>
<thead>
<tr>
<th></th>
<th>Ng Fe304</th>
<th>Sn ZnC</th>
<th>Kg Cu</th>
<th>Sludge Lancing</th>
<th>Other metals (mainly Zn)</th>
<th>Salts (1)</th>
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<td></td>
<td>SG 1</td>
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<td>SG 3</td>
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<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>2.3</td>
<td>2.4</td>
<td>2.3</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

(1) Salt balance  
| SC4 | 4011 g |
| F   | 1392   |
| Cl  | 812    |
| Na  | 596    |
| Ca  | 470    |
| K   | 142    |
| Mg  | 134    |

### TABLE II
WASTE TREATMENT IN TINHANGE 2

<table>
<thead>
<tr>
<th></th>
<th>Inlet</th>
<th>Outlet</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cu</td>
<td>190 ppm</td>
<td>140 ppm</td>
</tr>
<tr>
<td>g-activity</td>
<td>5.6 MBq/m³</td>
<td>&lt; 1.6 MBq/m³</td>
</tr>
</tbody>
</table>

7.1-8/12
Figure 1

Process application at Tilange 1

1988

- Assembling
- Cleaning SG 1
- Cleaning SG 3
- Disassembling on SG 1
- Disassembling on SG 3
- Complete removal

1989

- Preparation
- Assembling
- Cleaning SG 2
- Disassembling
- Complete removal
Figure 2: Tihange 2: Schematic flowsheet of the chemical cleaning installation
Figure 3

Process application at Tihange 2 (iron removal step)

- \( \text{NH}_4 \) injection
- Cooling down to 170°C
- SG level stabilisation
- NTA-dosing
- Feedwater injection
- Reaction Time
- Steaming SG 2/3
- Steaming SG 4
- Sampling
- Cool-down
- Draining SG 2
- Draining SG 3/4
Figure A: Tihange 2 SG Chemical cleaning - Evolution of Fe concentration during application and autoclave tests

7.1-12/12
CHEMICAL CLEANING FOR STEAM GENERATOR SECONDARY SIDE IN JAPANESE PWRS

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Nakanoshima, Kita-ku, Osaka

ABSTRACT

In order to prevent corrosion problems due to sludge accumulation and to improve reliability of steam generator, chemical cleaning had been carried out at Mihama Unit 1 and Ohi Unit 1, and KWU process which has many experiences in Europe was applied. This paper describes results on qualification test performed in advance and results on plant applications.
1. INTRODUCTION

In the secondary side of PWR steam generator, iron or copper compounds, which are corrosion products from secondary system components, is accumulated as power operation proceeds and these corrosion products form sludge (accumulation on the tube sheet) or scale (deposition on the tube surface). Though these sludge and scale themselves neither cause any corrosion problems on steam generator tube nor harm reliability of tubes, they might supply the site (place) where other impurity species can concentrate and result in the initiation and propagation of corrosion problems of steam generator.

In order to prevent corrosion problems and to improve reliability of steam generators, it is thought effective to remove such kind of accumulated sludge and scale. For the purpose of this improvement in chemical environment, several remedies such as removal of sludge itself by sludge lancing and removal of dissolved impurities by crevice flushing have been applied so far and those remedies are considered to be fairly effective.

On the other hand, some alternate methods to remove chemically sludge and scale including impurities, which is called chemical cleaning, have been newly developed in addition to traditional physical methods such as sludge lancing and there are already several field experiences. As one of new countermeasures for improving chemical environment of steam generator secondary side, chemical cleaning was carried out at Mihama unit 1 and Ohi unit 1.

This paper describes results on qualification tests performed in advance and results on applications at Mihama unit 1 and Ohi unit 1.

2. QUALIFICATION TEST

KWU process was selected because it has rather many experiences in Europe and some qualification tests had been performed prior to plant applications. Main results on this qualification tests are as follows.

2.1 Sludge Dissolution and Cleaning Conditions

2.1.1 Magnetite (Fe₃O₄)

In case of magnetite dissolution, nitrilo tri-acetic acid (NTA) is used and magnetite is dissolved under reducing condition resulting in a stable NTA-chelate. The effects of cleaning temperature and NTA concentration on magnetite dissolution are shown in Figure 1. Following results were obtained.

1) Relative dissolution rate of magnetite in NTA solution is increasing with temperature and this increasing tendency starts to saturate over 160°C.

2) Amount of dissolved magnetite depends on the NTA concentration. Relative dissolution rate is almost independent from NTA concentration.

Figure 1 Effects of Temperature and NTA Concentration on Magnetite Dissolution
Based on these results and sludge inventory estimation, cleaning conditions on plant applications were settled as follows.

Temperature : 160°C
NTA concentration : 6%

2.1.2 Copper (Cu)
Ethylene di-amine (EDA) is used as copper cleaning solvent and metallic copper is dissolved in the form of EDA chelate under oxidizing condition by means of pressurized air injection. The effects of EDA concentration and air injection on copper dissolution are summarized as follows.
(1) EDA concentration has no significant effect up to 6%.
(2) Copper dissolution rate increases with dissolved oxygen increase by pressurization (0~3kgf/cm²G) in the air injection operation.
Based on the above-mentioned informations, plant application conditions were settled as follows.
Temperature : 60°C
EDA concentration : 1.5~6% (Depending on Cu inventory)
Pressurized air injection : 0~3kgf/cm²G-Air

2.2 Adverse Effect of Chemical Cleaning
With regard to adverse effects of chemical cleaning on the integrity of secondary components, extensive evaluation on the following components were carried out and it was confirmed there were no significant adverse effects.
(i) During cleaning
  • Surface where cleaning solution contacts.
  (⇒ SG tube, tube sheet, downcomer, etc.)
  • Surface where cleaning solution does not contact.
  (⇒ upper support plate, steam pipe, etc.)
(ii) After cleaning
  • Cleaning solution residue
Some results on material compatibility tests are shown in Table I and corrosion rates of base metal were within acceptable range.

<table>
<thead>
<tr>
<th>Test piece</th>
<th>Result</th>
<th>Observation</th>
</tr>
</thead>
<tbody>
<tr>
<td>Carbon steel base metal</td>
<td>SB 42</td>
<td>70~76</td>
</tr>
<tr>
<td>Low alloy/low alloy weldment</td>
<td>SA-508Cl2/SA-533Type B</td>
<td>74~82</td>
</tr>
<tr>
<td>SG tube</td>
<td>JIS NCF600TB</td>
<td>2</td>
</tr>
</tbody>
</table>

(1) No local corrosions such as pitting in any specimen
(2) Surface roughness : 5~10μ

3. APPLICATION RESULTS

3.1 Mihama Unit 1

3.1.1 Plant Outline and Operating History
Outline of Mihama 1 is described in Table II. Mihama 1 is a plant that has two steam generators manufactured by Combustion Engineering and started its commercial operation on November in 1970. Accumulated operating time by this chemical cleaning is approximately
Table II Plant Outline of Mihama 1

<table>
<thead>
<tr>
<th>Item</th>
<th>Outline</th>
</tr>
</thead>
<tbody>
<tr>
<td>Date of commercial ope.</td>
<td>28th November 1970</td>
</tr>
<tr>
<td>Output (MWe)</td>
<td>340</td>
</tr>
<tr>
<td>Manufacturer</td>
<td>Combustion Engineering</td>
</tr>
<tr>
<td>SG Specification</td>
<td></td>
</tr>
<tr>
<td>No. of SG</td>
<td>2</td>
</tr>
<tr>
<td>Tube</td>
<td>Alloy 600MA</td>
</tr>
<tr>
<td>Tube support plate</td>
<td>Grate</td>
</tr>
<tr>
<td>Secondary water chemistry</td>
<td>PO₄ (Initial 4 cycles) ⟷ AVT</td>
</tr>
</tbody>
</table>

79000 hours.

3.1.2 Chemical Cleaning Procedure

Figure 2 shows result of application schedule. Chemical cleaning at Mihama 1 took seven days from pre-cleaning inspection to final rinsing. Area to be cleaned at Mihama 1 was up to 80cm on the tube sheet and volume of cleaning solution was about 3.1m³/

![Application Schedule of Mihama 1 Chemical Cleaning SG.](image)

3.1.3 Sludge Dissolution

(1) Copper Removal Step

Results on two copper removal steps are shown in Figure 3 and as follows.

(i) Rather fast copper dissolution was seen in the first copper step and this first step was completed under 6~7g/l copper concentration after seven depressurizations.

(ii) No significant additional dissolution of copper was observed in the second copper step.

(2) Iron Removal Step

Results on four iron removal steps are shown in Figure 4 and as follows.

(i) Four iron removal steps were

![Results on Copper Removal Step at Mihama 1](image)
carried out and almost 70–80% of the final concentration of each step was obtained in 2–3 hours after cleaning solution injection.

(ii) Since iron dissolution rate in the beginning of the fourth step was smaller than those in former three steps, heating was stopped at around 150°C and this step was completed.

(3) Sludge Removal Quantity
Sludge quantity removed through the chemical cleaning and the subsequent sludge lancing operation are summarized in Table III and as follows.

(i) Sludge quantity removed through the chemical cleaning and the subsequent sludge lancing was 304 kg/SG. (231 kg by chemical cleaning and 73 kg by sludge lancing)

(ii) The value is more than ten times average quantity removed by previous sludge lancing and so this chemical cleaning seems to have been much more effective than usual sludge lancing.

(iii) Taking into consideration that preliminary evaluation for sludge inventory was 309 kg/SG, almost all sludge to be cleaned is thought to be removed.

Table III Sludge Quantity Removed through Chemical Cleaning and Subsequent Sludge Lancing

<table>
<thead>
<tr>
<th>Item</th>
<th>Chemical Cleaning</th>
<th>Sludge Lancing</th>
<th>Total</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fe_3 O_4</td>
<td>199</td>
<td>49</td>
<td>248</td>
</tr>
<tr>
<td>Cu</td>
<td>20</td>
<td>18</td>
<td>38</td>
</tr>
<tr>
<td>Others</td>
<td>12</td>
<td>6</td>
<td>18</td>
</tr>
<tr>
<td>Total</td>
<td>231</td>
<td>73</td>
<td>304</td>
</tr>
</tbody>
</table>

Unit: kg
3.1.4 Impurity Removal

Quantity of dissolved impurity removed through this chemical cleaning is shown in Figure 5 and the ratio of phosphate to magnetite removed by the chemical cleaning is described in Figure 6, compared with those in the past sludge lancing. Results on impurity removal are as follows.

(i) Impurity species removed were mainly phosphate, calcium, magnesium and chloride. Phosphate might be residue of old water chemistry.

(ii) Considering the history of water chemistry, chloride is thought to be a residual component of past seawater leakage. 24g/SG of chloride removed is twenty times that of the latest crevice flushing and similar to those of hide out return.

(iii) Phosphate content in the sludge removed by sludge lancing has been decreasing after change of phosphate treatment to AVT and it is hardly detected in recent lancing. Extensive appearance of phosphate in the fourth iron step of the chemical cleaning suggests that iron cleaning proceeded into the lower layer of piled sludge and dissolved old sludge containing phosphate.

3.1.5 Visual Inspection

(i) Grayish black scale deposited commonly on the tube surface before cleaning was almost removed except for a little residue on hot leg side.

(ii) Before cleaning a great deal of sludge had piled up in almost all region of tube sheet, but after cleaning fairly much sludge was removed through chemical cleaning and subsequent sludge lancing except for residue of hard-caked sludge in some limited area.

(iii) Neither significant corrosion on the surface of SG tubes, tube sheet and blowdown pipes nor anomalous change was observed.

3.2 Ohi Unit 1

3.2.1 Plant Outline and Operating History

Outline of Ohi 1 is described in Table IV. Ohi 1 is a plant with four Westinghouse steam generators and started its commercial operation on March in 1979. Accumulated operating time by this chemical cleaning is approximately 62000 hours.
3.2.2 Chemical Cleaning Procedure

Figure 7 shows results of plant application schedule. Chemical cleaning at Ohi 1 took six days from pre-cleaning inspection to final rinsing. Area to be cleaned at Ohi 1 was up to 150cm on the tube sheet, including the first tube support plates, and volume of cleaning solution was about 8.6m³/SG.

3.2.3 Sludge Dissolution

(1) Iron Removal Step

Results on iron removal step are shown in Figure 8 and as follows.

(i) 80% and 90% of the final concentration were obtained respectively in 2 and 3 hours after cleaning solution injection.

(ii) The reason why iron step had been completed rather fast is considered that almost of all sludge was located on the tube surface and surface area of sludge contacted to cleaning solution was much bigger.

(2) Copper Removal Step

Results on copper removal step are shown in Figure 9 and as follows.

(i) Copper dissolution was completed rather fast by reaching to saturation after 10 hours.

(ii) This fast dissolution at Ohi 1 is

Figure 7 Application Schedule of Ohi 1 Chemical Cleaning

Figure 8 Results on Iron Removal Step at Ohi 1
attributed to wide surface area of sludge contacted to the solution and low copper content in sludge.

(3) Sludge Removal Quantity
Sludge quantity removed through the chemical cleaning and the subsequent sludge lancing are summarized in Table V and as follows.
(i) Sludge quantity removed through the chemical cleaning and the subsequent sludge lancing was 207kg/SG (191kg by chemical cleaning, 16kg by sludge lancing)
(ii) The value is more than ten times average quantity removed by previous sludge lancing and so this chemical cleaning seems to have been much more effective than usual lancing.
(iii) Taking into consideration that preliminary evaluation for sludge inventory was 183kg/SG, almost all sludge to be cleaned is thought to be removed.

3.2.4 Impurity Removal
Impurity removal through this chemical cleaning is summarized as follows.
(i) Dissolution of calcium, magnesium or sulphate which are contained in crevice sludge was observed and this suggests that there were some sludge dissolusion from crevices.
(ii) 29g/SG of sodium removed is almost 100 times that by crevice flushing and several tens times that of hide out return.

3.2.5 Visual Inspection
(i) Grayish black scale on the tube surface was almost removed both on hot and cold leg side.
(ii) Some sludge, which had been on the central region of tube sheet (banana region), was not observed after chemical cleaning.
(iii) Some crevices, whose contour lines had not been seen before cleaning, were observed to open their mouths.
(iv) Neither significant corrosion on the surface of SG tubes, tube sheet, first tube support plates and blowdown pipes, nor anomalous change was observed.

3.2.6 Pulled Tube Examination
2 (two) tubes were pulled and surface condition was investigated.
(1) Almost all scale on tube surface except crevice region was removed comparing with it of non-cleaning portion.
(2) Concerning deposit in crevice region accumulated during plant operation, some of crevice opening region was removed but in center region hard deposit was still observed remaining.

Figure 9 Results on Copper Removal Step at Ohi 1

Table V Sludge Quantity Removed through Chemical Cleaning and Subsequent Sludge Lancing

<table>
<thead>
<tr>
<th>Item</th>
<th>Chemical Cleaning</th>
<th>Sludge Lancing</th>
<th>Total</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fe₂O₄</td>
<td>191</td>
<td>14</td>
<td>195</td>
</tr>
<tr>
<td>Cu</td>
<td>6</td>
<td>1</td>
<td>7</td>
</tr>
<tr>
<td>Others</td>
<td>4</td>
<td>1</td>
<td>5</td>
</tr>
<tr>
<td>Total</td>
<td>191</td>
<td>16</td>
<td>207</td>
</tr>
</tbody>
</table>
4. SUMMARY

As one of new countermeasures for improving chemical environment of steam generator secondary side, chemical cleaning was carried out at Mihama 1 and Ohi 1 respectively. According to obtained results as follows, these chemical cleaning are considered to have contributed to improvement in secondary side chemical environment through the removal of sludge and impurities.

(i) Sludge quantity removed by the chemical cleaning and the subsequent sludge lancing are as follows and these values are more than ten times those by conventional sludge lancing. Almost of all sludge to be cleaned is thought to have been removed.

<table>
<thead>
<tr>
<th>Plant</th>
<th>Sludge removed by Chemical Cleaning (kg/SG)</th>
<th>Average sludge removed by conventional sludge lancing (kg/SG)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Mihama 1</td>
<td>304</td>
<td>26</td>
</tr>
<tr>
<td>Ohi 1</td>
<td>207</td>
<td>16</td>
</tr>
</tbody>
</table>

(ii) Removed quantity of impurities by the chemical cleaning were much more than those by crevice flushing performed during annual inspection.

(iii) The cleaning processes were applied in a relatively short period of time, considering the magnitude of the operations. The effective time employed were:

- Mihama 1: 7 days
- Ohi 1: 6 days

(iv) Corrosion of structural materials were in conformance with the qualification results.

(v) The amounts of waste produced were as follows.

- Mihama 1: 36 m³
- Ohi 1: 69 m³
SPANISH RESEARCH ACTIVITIES ON STEAM
GENERATOR TUBES DEGRADATION

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Mª. L. Castaño Marín
Ciemat
Madrid (Spain)

ABSTRACT

Inconel 600 MA steam generator tubes have undergone a severe degradation in Spanish nuclear power plants during the last years. This degradation can have important implications for the availability of the plant. This paper presents the activities of the Spanish Research and Development Project on Steam generators relating to steam generator tubing materials degradation. These activities are carried out in two complementary forms. On the one hand, the behaviour of Inconel 600 MA, Inconel 690 TT and Incoloy 800 has been studied under primary and secondary simulated conditions, and on the other hand, the destructive examination of steam generators tubes from commercial plants was carried out.
1. INTRODUCTION

Inconel 600 MA steam generator tubes have undergone a severe degradation in Spanish nuclear power plants during the last years. This degradation can have important implications for the plant availability. New defect types, such as OD and ID circumferential cracks, have been found and also, new secondary environment, non alkaline environments, have begun to be considered.

Furthermore, the effect of boric acid as an inhibitor of intergranular attack (IGA) and intergranular stress corrosion cracking of alloy 600 is not clear.

This paper presents the activities of the Spanish Research and Development Project on steam generators related to the steam generator tubing materials degradation. These activities are carried out in two complementary forms. On the one hand, the behaviour of Inconel 600 MA, Inconel 690 TT and Incoloy 800 has been studied under primary and secondary simulated conditions and the other hand, the destructive examination of steam generators tubes from commercial plants was carried out.

2. SCOPE

This experimental work aims to obtain a better understanding of the behaviour of Inconel 600 MA steam generator tubing and other nickel alloys such as Inconel 690 TT and Incoloy 800.

The effects of changes in primary water chemistry on the primary water stress corrosion cracking (PWSCC) of steam generator tube materials have been studied.

Comparative tests of Inconel 690 TT and Incoloy 800 are being carried out in caustic or acidic environments with S, Pb and Cu. Furthermore, steam generator tubes from Spanish nuclear power plant have been examined destructively.

Soon, tests of Inconel 600 MA, Inconel 600 TT, Inconel 690 TT and Incoloy 800 in caustic or acidic environment with or without boric acid will start.

3. EXPERIMENTAL FACILITIES

Corrosion tests in both primary and secondary chemistry conditions have been carried out in facilities suitable to simulate operating conditions of steam generators. These facilities have been designed and built at Ciemat. A block diagram of the facilities is shown in Fig. 1.
Primary tests have been made in six refreshed stainless steel autoclaves each fed by a separate high pressure/high temperature loop. The loops have instrumentation to measure oxygen, hydrogen, conductivity and pH. Secondary tests can be carried out in five model boilers with seven tubes each fed by a common primary loop and five separate secondary loops.

Tubing materials and support plate types used in each model boiler are shown in Table I. Support plates have thermocouples so that temperature during the test can be known.

There is also another model boiler with nine tubes located as in model D3 Westinghouse steam generator used for "denting" testing.

Aggressive environment tests are performed in static nickel autoclaves, caustic environments, and in Hastelloy autoclave, acidic environments.

4. TEST PROGRAMME

4.1. Primary water stress corrosion cracking tests.

The objectives of these tests are to determine the effects of pH changes in primary water chemistry on the PWSCC behaviour of Inconel 600 MA steam generator tubing and to examine the influence of dissolved hydrogen concentration on PWSCC behaviour over the range of hydrogen concentrations relevant to PWR's.

Six simulated primary water chemistry environments were defined. Table II shows the test conditions and the resultant high temperature pH calculated at 330°C, a typical PWR primary circuit maximum temperature. The test temperature and the system pressure were controlled at 330°C and 150 kg/cm² respectively.

The RUB specimens used in this work were made without spring back during bending. Three heats of Inconel 600 MA were tested. Heat 96834 L of Inconel 600 MA was supplied by EPRI. Specimens made by other laboratories were also tested. One heat of Inconel 690 TT and another one of Incoloy 800 were tested in some environments.

The chemical composition of the test materials are given in Table III and the mechanical properties are listed in Table IV.

4.2. Inconel 690 TT and Incoloy 800 in very aggressive environments.

The objectives of these tests are to contribute to the selection of one of them as an alternative material to Inconel 600 MA, and to know their susceptibility in environments that are very aggressive to Inconel 600 MA.
Table V shows the caustic or acidic environments used in the tests. The test temperature is controlled at 350°C.

C-Ring specimens were prepared according to ASTM G38-73 and strained to 2% strain. The strain were measured during calibration tests for each tubing heat. Table VI shows the heats of Inconel 690 TT and Incoloy 800 used in these tests. One heat of Inconel 600 MA was included as a reference.

4.3. Examination of steam generator/tubes.

The objectives of this work are, firstly, to determine the existence, depth and morphology of the present degradation to obtain information about the environmental conditions in the degraded zones, and secondly, to contrast it with non destructive inspection.

Up until now, nine tubes from Spanish nuclear power plants at Zorita, Almaraz and Asco have been examined. The destructive examination consist of:

- Visual inspection, dimension measurement, X-ray radiography.
- Optical microscopy examination, deposit analysis and scanning electron microscopy.
- Microstructure, Huey test, tensile test and hardness test for characterization of the tubing.

CIEMAT has recently acquired an Auger spectrometer and a XPS spectrometer that will be used to analyze some samples from the tubes examined.

5. RESULTS AND DISCUSSION

5.1. PWSCC test

The results obtained on heat 96834L Inconel 600 MA RUB are shown in the table VII. Fig. 2 shows the 50% PWSCC initiation time for serie 1 heat 96834L RUB vs environments conditions after 5,500 h. of exposure. The results have been fitted to Weibull distribution.

Table VIII and Fig 3 show similar information for heat 1450 Inconel 600 MA. On the other hand, only one RUB specimen from heat 752491 of 42 specimen tested was cracked. Furthermore no cracked specimens from Inconel 690 TT and Incoloy 800 were detected for 5500 h. test.

The test results for specimen RUB from heat 96834L show no statistically significant differences in PWSCC initiation time over
the six primary water environments examined with a probability of 0.05, but the differences are significant for some tests with a probability of 0.20. However, the results from heat 1450 show no statistically significant differences even with a probability of 0.20. Therefore, we only used the results from heat 96834L to infer the possible effects of primary water chemistry in PWSCC initiation time.

Comparing tests 1, 3 and 5 with 2, 4 and 6 respectively, the results seem to indicate that the initiation time in the low hydrogen levels (11.6 kpa) is higher than in high hydrogen levels (23.3 kpa). This tendency has been shown in test performed by Westinghouse at 310°C (1).

Tests made by Studsvik show a small reverse effect of hidrogen at 330°C (2). Nevertheless, the range of parcial hydrogen pressures used in STUDSVIK test (7.0 and 13.6 Kpa) and in CIEMAT test (11.6 an 23.3 kpa) can be considered complementary. (3)

On the other hand, the results from tests with different pHs suggest that the initiation times are lower in test with a higher pH.

However, the lack of statistically significance may point to the fact that either more specimens should be tested in each environment, or the influence of tested environments is not really significant.

These no statistically significant differences in PWSCC initiation time in several test environments have also been found by Westinghouse (4).

5.2. Inconel 690 TT and Incoloy 800 in very agressive environments.

Table IX shows the results obtained in several heats of Inconel 690 TT and Incoloy 800 in different environments, for a 500 hour test.

All specimens were inspected with a stereoscopic binocular microscope and some of them were examined by metallography to determine depth and morphology of cracks.

Tests 1, 2 and 5 were continued for up to 1000 h for Inconel 690 TT and Inconel 600 MA specimens. Fig. 4 and 5 show the results obtained in 10% NaOH and in 10% NaOH+0.01% 0Cu.

No cracked specimens of Inconel 690 TT were found in 0,75M Na₂SO₄+ 0.25M FeSO₄ after 1000 h testing.

Fig. 6-9 show the aspect of cracks detected on Incoloy 800 specimens in caustic and acidic environments with or without lead. Fig. 10-11 show degradation obtained on Inconel 690 TT specimens in lead caustic and acidic environments. Similar information is shown in Fig 12-14 for Inconel 600 MA.
SCC resistance of alloy 690 TT in the environments used is superior to that of Incoloy 800 and Inconel 600 MA. However, it is important to emphasize the susceptibility of Inconel 690 TT in lead caustic environments.

The effect of shot peening on SCC resistance Incoloy 800 is not clear. Furthermore, the results obtained on Incoloy 800 show the importance of steam generator design in tubing materials behaviour.

5.3. Examination of steam generator tubes.

Different types of degradation have been found in seven of the nine steam generator tubes examined (6-8).

The tube from Almaraz I had two axial through-wall cracks and another crack 92% depth in the flow distribution baffle region. All the cracks were intergranular, initiated by the secondary. In this tube, there was also one ID through-wall axial crack at the roll transition zone.

Three of the six tubes from Asco have been examined up to support plate 11 or 12. The others have been examined only up to the flow distribution baffle.

The tubes from ASCO had axial cracks, small pitting and shallow intergranular attack, initiated on the external surface at the elevation support plates, Fig 15. Also, three of them had OD circumferential cracks at the roll transition zone, Fig. 16. Besides, ID circumferential cracks were found at the roll transition zone of two tubes. In another tube ID axial cracks have been detected at the same zone. None of the tubes had OD and ID cracks together at the roll transition zone.

Energy dispersive spectrometry (EDS) was performed on surface deposits at the degraded zones. The results obtained on the tubes from ASCO point to a Cr enrichment at these zones. Also, Pb was found in some analysis performed at degraded zones.

Fig 17 shows a photomicrograph of OD circumferential cracks at the roll transition zone, and Cr, Fe, Ni X-ray mapping that were performed on this zone. Cr enrichment can be seen at the bottom layer of the deposits.

The data supports the absence of a strongly caustic environment.
SUMMARY

- The results obtained from primary water test point to the fact that initiation time of primary water stress corrosion cracking are lower in tests with less hydrogen, and in tests with higher pH. More tests with more samples in each case are necessary.

- SCC resistance of Inconel 690 TT in the environments used is superior to that of Incoloy 800 and Inconel 600 MA. Inconel 690 TT is susceptible to lead caustic environments.

- The effect of shot peening on SCC resistance Incoloy 800 is not clear.

- OD axial cracks, shallow pitting and intergranular attack have been detected at the support plates of steam generator tubes

- OD and ID circumferential cracks have been found at the roll transition zone. These cracks were not found together.

BIBLIOGRAPHIA


### TABLE I. TUBING MATERIAL AND SUPPORT PLATE TYPES IN MODEL BOILERS

<table>
<thead>
<tr>
<th>MODE BOILER TYPE</th>
<th>TUBING MATERIAL</th>
<th>SUPPORT PLATE</th>
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</thead>
<tbody>
<tr>
<td>24</td>
<td>INCONEL 600 MA</td>
<td>NORMAL</td>
</tr>
<tr>
<td>D3</td>
<td>INCONEL 600 MA</td>
<td>NORMAL</td>
</tr>
<tr>
<td>D3</td>
<td>INCONEL 600 MA</td>
<td>NORMAL</td>
</tr>
<tr>
<td>F</td>
<td>INCONEL 600 TT</td>
<td>QUATREFOIL</td>
</tr>
<tr>
<td></td>
<td>INCONEL 690 TT</td>
<td></td>
</tr>
<tr>
<td>GT-54</td>
<td>INCOLOY 800</td>
<td>NORMAL</td>
</tr>
<tr>
<td></td>
<td>INCONEL 690</td>
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### TABLE II. TEST CONDITIONS

<table>
<thead>
<tr>
<th>TEST</th>
<th>B ppm</th>
<th>Li ppm</th>
<th>pH(309°C)/pH(330°C)</th>
<th>B₂ ppm</th>
<th>mlH₂(STP) KgH₂O</th>
<th>PH₂(330°C) (Kpa/psia)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>1400</td>
<td>2</td>
<td>6,9/7,24</td>
<td>4</td>
<td>45</td>
<td>23,3/3,4</td>
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<tr>
<td>2</td>
<td>1400</td>
<td>2</td>
<td>6,9/7,24</td>
<td>2</td>
<td>23</td>
<td>11,6/1,7</td>
</tr>
<tr>
<td>3</td>
<td>740</td>
<td>3,5</td>
<td>7,4/7,73</td>
<td>4</td>
<td>45</td>
<td>23,3/3,4</td>
</tr>
<tr>
<td>4</td>
<td>740</td>
<td>3,5</td>
<td>7,4/7,73</td>
<td>2</td>
<td>23</td>
<td>11,6/1,7</td>
</tr>
<tr>
<td>5</td>
<td>385</td>
<td>2</td>
<td>7,4/7,73</td>
<td>4</td>
<td>45</td>
<td>23,3/3,4</td>
</tr>
<tr>
<td>6</td>
<td>385</td>
<td>2</td>
<td>7,4/7,73</td>
<td>2</td>
<td>23</td>
<td>11,6/1,7</td>
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</table>

7.3-8/20
### TABLE III. - CHEMICAL COMPOSITION (% Wt)

<table>
<thead>
<tr>
<th>MATERIAL (HEAT)</th>
<th>C</th>
<th>Mn</th>
<th>Fe</th>
<th>S</th>
<th>Si</th>
<th>Cu</th>
<th>Ni</th>
<th>Cr</th>
<th>Ti</th>
<th>Co</th>
<th>P</th>
</tr>
</thead>
<tbody>
<tr>
<td>1600MA(1450)</td>
<td>0.045</td>
<td>0.18</td>
<td>9.34</td>
<td>0.002</td>
<td>0.18</td>
<td>0.26</td>
<td>73.84</td>
<td>16.16</td>
<td>0.21</td>
<td>0.05</td>
<td>0.07</td>
</tr>
<tr>
<td>1600MA(96834L)</td>
<td>0.038</td>
<td>0.26</td>
<td>8.03</td>
<td>0.001</td>
<td>0.32</td>
<td>0.01</td>
<td>74.79</td>
<td>15.84</td>
<td>0.022</td>
<td></td>
<td></td>
</tr>
<tr>
<td>1600MA(752491)</td>
<td>0.015</td>
<td>0.83</td>
<td>9.28</td>
<td>0.003</td>
<td>0.31</td>
<td>0.01</td>
<td>72.4</td>
<td>16.5</td>
<td>0.022</td>
<td></td>
<td></td>
</tr>
<tr>
<td>I800 (81373)</td>
<td>0.021</td>
<td>0.73</td>
<td>bal</td>
<td>0.002</td>
<td>0.46</td>
<td>0.08</td>
<td>33.02</td>
<td>20.48</td>
<td>0.42</td>
<td>0.056</td>
<td>0.016</td>
</tr>
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</table>

### TABLE IV. MECHANICAL PROPERTIES OF ALLOY 600

<table>
<thead>
<tr>
<th>HEAT</th>
<th>YS MPa/Ksi</th>
<th>UTS MPa/Ksi</th>
<th>ELONG %</th>
<th>MA Temperature °C</th>
</tr>
</thead>
<tbody>
<tr>
<td>1450</td>
<td>386/56</td>
<td>793/115</td>
<td>36</td>
<td>*</td>
</tr>
<tr>
<td>96834L</td>
<td>389/56</td>
<td>737/101</td>
<td>37</td>
<td>927 (3-5 minutes)</td>
</tr>
<tr>
<td>752491</td>
<td>308/45</td>
<td>687/99</td>
<td>46</td>
<td>1040 (15 minutes)</td>
</tr>
</tbody>
</table>

* Unknown. Microstructure suggests low temperature
TABLE V. TEST CONDITIONS, INCONEL 690 TT AND INCOLOY 800

### CAUSTIC
- (R) * 10% NaOH
- (R) * 10% NaOH + 0.01% Cu
- (R) * 10% NaOH + 0.1 M Pb
- 10% NaOH + 0.01 M Pb
- 4% NaOH + 0.01 M Pb
- 50% NaOH
- (R) * 50% NaOH + 5% Na₂S₂O₃
- 50% NaOH + Na₂SO₄
- 50% NaOH + Na₂S
- 50% NaOH + Na₂CO₃

### ACIDIC
- (R) * 0.75 M Na₂SO₄ + 0.25 M FeSO₄
- (R) * 0.75 M Na₂SO₄ + 0.25 M FeSO₄ + 0.1 M Pb
- 0.2 M NaHSO₄ + 0.4 M FeSO₄ + 0.4 M Na₂SO₄
- 50 ppm CuCl₂ + 50 ppm NaCl

(*) ENVIROMENTS ALREADY TESTED AND UNDER TEST
(R) RESULTS PRESENTED

---

TABLE VI. MATERIALS

<table>
<thead>
<tr>
<th>MATERIAL</th>
<th>HEAT</th>
<th>% C</th>
<th>TUBE</th>
<th>TOTAL SAMPLES</th>
</tr>
</thead>
<tbody>
<tr>
<td>INCONEL 690 TT</td>
<td>764408</td>
<td>0.019</td>
<td>11/16&quot;</td>
<td>15</td>
</tr>
<tr>
<td>INCONEL 690 TT</td>
<td>WF 816 T</td>
<td>0.022</td>
<td>3/4&quot;</td>
<td>15</td>
</tr>
<tr>
<td>INCOLOY 800 SP</td>
<td>81373</td>
<td>0.021</td>
<td>7/8&quot;</td>
<td>15</td>
</tr>
<tr>
<td>INCOLOY 800</td>
<td>7-73243</td>
<td>0.017</td>
<td>7/8&quot;</td>
<td>4</td>
</tr>
<tr>
<td>INCONEL 600 MA</td>
<td>1450</td>
<td>0.045</td>
<td>3/4&quot;</td>
<td>9</td>
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</tbody>
</table>

7.3-10/20
### TABLE VII. CRACKED SPECIMENS/TESTED SPECIMENS. TEST 1, 2 AND 3.

<table>
<thead>
<tr>
<th>Autoclave</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
<th>6</th>
</tr>
</thead>
<tbody>
<tr>
<td>8 (ppm)</td>
<td>1400</td>
<td>1400</td>
<td>740</td>
<td>740</td>
<td>385</td>
<td>385</td>
</tr>
<tr>
<td>Li (ppm)</td>
<td>2</td>
<td>2</td>
<td>3.5</td>
<td>3.5</td>
<td>2</td>
<td>2</td>
</tr>
<tr>
<td>H₂CC/kg H₂O</td>
<td>45</td>
<td>23</td>
<td>45</td>
<td>23</td>
<td>45</td>
<td>23</td>
</tr>
<tr>
<td>pH a 309 °C</td>
<td>6.9</td>
<td>6.9</td>
<td>7.4</td>
<td>7.4</td>
<td>7.4</td>
<td>7.4</td>
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</table>

<table>
<thead>
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<th>3</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>1</th>
<th>2</th>
</tr>
</thead>
<tbody>
<tr>
<td>500 horas</td>
<td>0/6 0/5</td>
<td>0/6 0/8</td>
<td>0/6 0/5</td>
<td>0/6 0/7</td>
<td>0/6 0/7</td>
<td>0/6 0/9</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1000 horas</td>
<td>0/6 0/4</td>
<td>0/6 0/4</td>
<td>0/6 0/4</td>
<td>0/6 0/2</td>
<td>0/6 0/2</td>
<td>0/6</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1500 horas</td>
<td>0/5 1/4</td>
<td>1/8</td>
<td>0/5 0/4</td>
<td>1/7</td>
<td>0/7 0/2</td>
<td>0/9</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>2000 horas</td>
<td>2/6 1/5</td>
<td>5/6 1/8</td>
<td>2/6 1/5</td>
<td>5/6 2/7</td>
<td>4/6 0/7</td>
<td>6/5 0/9</td>
<td></td>
<td></td>
<td></td>
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</tr>
<tr>
<td>2500 horas</td>
<td>3/6 3/4</td>
<td>5/6</td>
<td>3/6 1/4</td>
<td>6/6</td>
<td>4/6 0/2</td>
<td>-</td>
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</tr>
<tr>
<td>3000 horas</td>
<td>3/5 3/4</td>
<td>1/8</td>
<td>1/5 2/4</td>
<td>4/7</td>
<td>3/7 1/2</td>
<td>5/9</td>
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<td></td>
<td></td>
<td></td>
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</tr>
<tr>
<td>3500 horas</td>
<td>4/6 3/5</td>
<td>5/6 4/8</td>
<td>5/6 3/5</td>
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<td>- 4/7</td>
<td>- 8/9</td>
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<td></td>
<td></td>
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</tr>
<tr>
<td>4000 horas</td>
<td>4/5 5/6</td>
<td>5/6</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td></td>
<td></td>
<td></td>
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</tr>
<tr>
<td>5000 horas</td>
<td>4/6 6/6</td>
<td>6/6</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td></td>
<td></td>
<td></td>
<td></td>
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</tr>
<tr>
<td>5500 horas</td>
<td>5/6</td>
<td>-</td>
<td>5/6</td>
<td>-</td>
<td>-</td>
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INCONEL 600 MA. TUBES 3/4". HEAT 96834L

### TABLE VIII. CRACKED SPECIMENS/TESTED SPECIMENS. TEST 1 AND 2.

<table>
<thead>
<tr>
<th>Autoclave</th>
<th>1</th>
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<th>3</th>
<th>4</th>
<th>5</th>
<th>6</th>
</tr>
</thead>
<tbody>
<tr>
<td>8 (ppm)</td>
<td>1400</td>
<td>1400</td>
<td>740</td>
<td>740</td>
<td>385</td>
<td>385</td>
</tr>
<tr>
<td>Li (ppm)</td>
<td>2</td>
<td>2</td>
<td>3.5</td>
<td>3.5</td>
<td>2</td>
<td>2</td>
</tr>
<tr>
<td>H₂CC/kg H₂O (STP)</td>
<td>45</td>
<td>23</td>
<td>45</td>
<td>23</td>
<td>45</td>
<td>23</td>
</tr>
<tr>
<td>pH a 309 °C</td>
<td>6.9</td>
<td>6.9</td>
<td>7.4</td>
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<tbody>
<tr>
<td>500 horas</td>
<td>0/7 0/5</td>
<td>0/7 0/5</td>
<td>0/7 0/5</td>
<td>0/7 0/5</td>
<td>0/7 0/5</td>
<td>0/7 0/5</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1000 horas</td>
<td>0/7</td>
<td>0/7</td>
<td>0/7</td>
<td>0/7</td>
<td>0/7</td>
<td>0/7</td>
<td></td>
<td></td>
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<tr>
<td>1500 horas</td>
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<td>0/5</td>
<td>0/5</td>
<td>0/5</td>
<td>0/5</td>
<td></td>
<td></td>
<td></td>
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</tr>
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<td>2000 horas</td>
<td>0/7 0/5</td>
<td>0/7 0/5</td>
<td>1/7 0/5</td>
<td>1/7 0/5</td>
<td>2/7 0/5</td>
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<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>2500 horas</td>
<td>2/7</td>
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<td>0/7</td>
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<td>2/7</td>
<td>3/7</td>
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<tr>
<td>3000 horas</td>
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<td>0/5</td>
<td>0/5</td>
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<td>0/5</td>
<td>0/5</td>
<td></td>
<td></td>
<td></td>
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</tr>
<tr>
<td>3500 horas</td>
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<td>4/7 0/5</td>
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<td>6/7 2/5</td>
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<td></td>
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</tr>
<tr>
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<td>4/7 5/7</td>
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<td>7/7</td>
<td>7/7</td>
<td>5/7</td>
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<tr>
<td>5000 horas</td>
<td>4/7 5/7</td>
<td>0/7</td>
<td>7/7</td>
<td>7/7</td>
<td>6/7</td>
<td>7/7</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>5500 horas</td>
<td>4/7 6/7</td>
<td>1/7</td>
<td>--</td>
<td>--</td>
<td>7/7</td>
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INCONEL 600 MA. TUBE 3/4". HEAT 1450

7.3-11/20
TABLE IX. Results of Metallographic examination. 500 h of exposure

<table>
<thead>
<tr>
<th>MATERIAL</th>
<th>1 - 800</th>
<th>I - 800 SP</th>
<th>1 - 690 TT</th>
<th>1 - 690 TT</th>
<th>1 - 600 MA</th>
</tr>
</thead>
<tbody>
<tr>
<td>ENVIRONMENT</td>
<td>7 - 73243</td>
<td>81373</td>
<td>WF 816 T</td>
<td>764408</td>
<td>1450</td>
</tr>
<tr>
<td>10 % NaOH</td>
<td>70 %</td>
<td>70 %</td>
<td>0 %</td>
<td>0 %</td>
<td>45 %</td>
</tr>
<tr>
<td></td>
<td>IGSCC</td>
<td>IGSCC</td>
<td></td>
<td></td>
<td>IGSCC</td>
</tr>
<tr>
<td>10 % NaOH + 0.01 % OCu</td>
<td>93 %</td>
<td>80 %</td>
<td>0 %</td>
<td>0 %</td>
<td>40 %</td>
</tr>
<tr>
<td></td>
<td>IGSCC</td>
<td>IGSCC</td>
<td></td>
<td></td>
<td>IGSCC</td>
</tr>
<tr>
<td>10 % NaOH + 0.1 M O\text{Pb}</td>
<td>100 %</td>
<td>100 %</td>
<td>100 %</td>
<td>100 %</td>
<td>10 %</td>
</tr>
<tr>
<td></td>
<td>TGSCC</td>
<td>TGSCC</td>
<td>TGSCC</td>
<td>TGSCC</td>
<td>IGSCC</td>
</tr>
<tr>
<td>50 % NaOH + 5 % Na\text{2} S\text{2} O\text{3}</td>
<td>100 %</td>
<td>100 %</td>
<td>100 %</td>
<td>100 %</td>
<td>100 %</td>
</tr>
<tr>
<td></td>
<td>IGSCC</td>
<td>IGSCC</td>
<td>IGSCC</td>
<td>IGSCC</td>
<td>IGSCC</td>
</tr>
<tr>
<td>0.75 M Na\text{2} \text{SO}_4 + 0.25 M Fe\text{SO}_4</td>
<td>63 %</td>
<td>93 %</td>
<td>0 %</td>
<td>0 %</td>
<td>82 %</td>
</tr>
<tr>
<td></td>
<td>IGSCC</td>
<td>IGSCC</td>
<td></td>
<td></td>
<td>IGSCC</td>
</tr>
<tr>
<td>0.75 M Na\text{2} \text{SO}_4 + 0.25 M Fe\text{SO}_4 + 0.1 M O\text{Pb}</td>
<td>18 %</td>
<td>68 %</td>
<td>0 %</td>
<td>0 %</td>
<td>100 %</td>
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<td>IGSCC</td>
<td></td>
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<td>IGSCC</td>
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</table>

CRACK DEPTH ( % WALL THICKNESS)

7.3-12/20
Fig. 1.- Block diagram of the experimental facilities
<table>
<thead>
<tr>
<th>TEST</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
<th>6</th>
</tr>
</thead>
<tbody>
<tr>
<td>B (ppm)</td>
<td>1400</td>
<td>1400</td>
<td>740</td>
<td>740</td>
<td>385</td>
<td>385</td>
</tr>
<tr>
<td>Li (ppm)</td>
<td>2</td>
<td>2</td>
<td>3.5</td>
<td>3.5</td>
<td>2</td>
<td>2</td>
</tr>
<tr>
<td>$H_2$ (ppm)</td>
<td>4</td>
<td>2</td>
<td>4</td>
<td>2</td>
<td>4</td>
<td>2</td>
</tr>
</tbody>
</table>

Fig. 2.- 50% PWSCC initiation time. Heat 96834L, Inconel 600 MA

<table>
<thead>
<tr>
<th>TEST</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
<th>6</th>
</tr>
</thead>
<tbody>
<tr>
<td>B (ppm)</td>
<td>1400</td>
<td>1400</td>
<td>740</td>
<td>740</td>
<td>385</td>
<td>385</td>
</tr>
<tr>
<td>Li (ppm)</td>
<td>2</td>
<td>2</td>
<td>3.5</td>
<td>3.5</td>
<td>2</td>
<td>2</td>
</tr>
<tr>
<td>$H_2$ (ppm)</td>
<td>4</td>
<td>2</td>
<td>4</td>
<td>2</td>
<td>4</td>
<td>2</td>
</tr>
</tbody>
</table>

Fig. 3.- 50% PWSCC initiation time. Heat 1450, Inconel 600 MA

7.3-14/20
Fig. 6. - Incoloy 800 SP. 10% NaOH (X50)

Fig. 7. - Incoloy 800 SP. 10% NaOH + 0.01M Pb (X50)

Fig. 8. - Incoloy 800 SP. 0.75M Na₂SO₄ + 0.25M FeSO₄ (X50)

Fig. 9. - Incoloy 800 SP (X50)  
0.7M Na₂SO₄ + 0.25M FeSO₄ + 0.1M Pb.
Fig. 12. - Inconel 600 MA
0.75M Na₂SO₄+0.25M FeSO₄+0.1M OPb (X50)

Fig. 13. - Inconel 600 MA (X200)
0.75M Na₂SO₄+0.25M FeSO₄+0.1M OPb

Fig. 14. - Inconel 600, 10% NaOH+0.1M OPb (X200)
Fig. 15.- Metallographic cross section at the 2nd S.P. 
Tubo R4C58. SGA Ascó II.

Fig. 16.- O.D. Surface at the roll transition zone. Circunferential cracks can be observed. Tube R8C57. SGA. Ascó II.
Fig. 17.– X-Ray mapping for Fe, Cr, Ni at the roll transition zone.
Tube R4C68. SGA. Ascó II.
AUTOMATIC MEASURES TO COPE WITH STEAM GENERATOR TUBE RUPTURES

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ABSTRACT

The paper describes operating experiences with the steam generators of both units of Biblis NPP and actions taken to improve the procedures to cope with a steam generator tube rupture event. After some years of operation the Biblis NPP experienced some problems with steam generator tube degradation. The paper provides a short review of the degradation mechanisms observed and of the precautions taken to stop steam generator tube degradation. Though, no major leakage or rupture of a steam generator tube occurred at Biblis NPP, lessons learned from foreign operating experience as well as from probabilistic evaluations within the German Risk Study, Phase B identified some weakness of the procedure to cope with a steam generator tube rupture. Short-term and long-term actions taken by RWE Energie AG to remedy these deficiencies are presented in detail in this paper.
1 Introduction

The paper describes the operating experiences with the eight steam generators (SG) of both Biblis units and actions taken to improve the procedures to cope with a steam generator tube rupture event.

After first years of operation the Biblis NPP experienced some problems with SG tube degradation. The paper provides a short review of the degradation mechanisms observed and of the precautions taken to stop SG tube degradation. These measures have been effective until now, as results of the surveillance tests show.

Though, no major leakage or rupture of a SG tube occurred at Biblis NPP during it's operating history, lessons learned from the evaluation of foreign SG tube rupture events as well as from probabilistic evaluations within the German Risk Study - both performed by GRS on behalf of the Federal Minister of Environment and the Federal Minister of Research and Technology respectively - identified some weakness of the procedures implemented at Biblis NPP to cope with a SG tube rupture. The corrective actions taken by RWE Energie AG to remedy these deficiencies are presented in detail in this paper.

Even in the case of a SG tube rupture exceeding the double ended break of one tube, the technical solution realized at Biblis NPP fully complies with the "30 minute rule", i.e. it does not require manual actions by the operators during the incident within the first 30 minutes.

2 General Information on the Biblis Nuclear Power Plant

The Biblis NPP has two units and is located 10 km north of the city of Worms on the east bank of the Rhine River. The plant is operated by the RWE Energie AG.

The two units have a rated output of

- unit A: 1204 MWe and
- unit B: 1300 MWe.
The units are equipped with 4-loop pressurized water reactors and were delivered as turnkey units by Kraftwerk Union (KWU). First power to the grid was produced on:

- unit A: August 25, 1974 and
- unit B: April 25, 1976.

Gross power production since commissioning until July 31, 1991:

- unit A: 119655.2 GWh and
- unit B: 104974.5 GWh.

3 Operating Experiences

3.1 Steam Generator Design

The SGs of the Biblis NPP are vertical U-tube SGs manufactured by Babcock (fig. 1) and GHH respectively. The four SGs of unit A were the first of this size in a German NPP. The four SGs of unit B are similar to those of unit A with some improvements.

Shell and tube sheet of the SGs are manufactured of 22NiMoCr37, i.e. the same material as the reactor pressure vessel (RPV) and the reactor coolant loops. Inlet and outlet chambers of the SGs as well as the primary side of the tube sheet are weld cladded with the austenitic material 1.4550. The SGs have about 4,000 U-tubes each, made of Incoloy 800, which is highly resistant against stress corrosion cracking. The vertical part of the U-tubes is fixed by steel bar lattices (fig. 2). The bend part is supported by anti vibration bars (AVB) of the same material (fig. 3). The steam generators are designed for a power of 885 MWth (unit A) and 938 MWth (unit B) with a 30 % margin in the heat transfer surface. The main design data of the Biblis SGs are summarized in table one.

Till 1987/88 the secondary side water of the SGs was treated with phosphate. Due to wastage problems especially in unit A, the secondary water chemistry has been changed to high AVT in 1987/88 after replacement of the main condensor brass tubing with stainless steel tubes (1.4439).
3.2 SG Tube Degradation Mechanisms Observed in Biblis NPP

Substantial SG tube degradation has been observed in unit A of Biblis NPP only. Up to now unit B experienced merely minor tube degradation problems. This may be explained by the improved design of unit B SGs, the more rigorous quality control during manufacturing and the lesson learned from the operating experience with the unit A SGs.

In unit A of Biblis NPP three major areas of degradation have been identified, which are illustrated in fig. 4:

- wastage above the tube sheet,
- wastage at the inner bend part of the U-tube bundle,
- fretting at the outer bend part of the U-tube bundle.

Wastage above the tube sheet

Like other NPPs with phosphate treatment of secondary side water, Biblis unit A experienced some wastage problems just above the tube sheet. To resolve the problems the following corrective actions have been taken at both units:

- lancing of the tube sheet during each refueling outage since 1980
- reduction of corrosion product sedimentation in the SGs by replacing erosion corrosion sensitive material in the steam / water circuit with more erosion corrosion resistant material
- changing secondary side water chemistry to high AVT water chemistry after replacement of the main condensor tubing (unit A since 1987 and unit B since 1988).

Table two contains the mean wastage rate per cycle observed in unit A since 1985. After changing to high AVT the wastage rate decreased by more than 50 % to about 1 %/a. The wastage rate measured in unit B is significantly lower compared to unit A.

Today about 15 kg corrosion products per SG are lanced out of the SGs during the annual refueling outage.
Wastage at the inner bend part of the U-tube bundle

As can be seen from fig. 5, the AVBs supporting the bend part of the U-tube bundle are connected to segment plates, which are fixed by a supporting tube in the inner part of the bend bundle. Since these segment plates have reduced the circulation of water in the area of bend part of the innermost of 5 tube rows sedimentation of corrosion products similar to that above the tube sheet occurred.

After a considerable increase of the wastage rate at the bend part of the innermost 5 tube rows in 1985 and 1986, as a precautionary measure all degraded tubes of these rows have been plugged and the secondary water chemistry has been changed to high AVT.

Fretting at the outer bend part of the U-tube bundle

To fix the bend part of the U-tube bundle the AVBs are connected by steel bars steel welded to the AVBs. Due to failure of a few of these welds in 1983 some tubes of outer rows experienced vibration induced fretting.

All tubes with wall thickness degradation higher than 40 % have been plugged. Since 1984, those parts of the tube bundle affected by vibration induced fretting have been fixed by 71 special comb like devices. This corrective action stopped fretting in the bend part of the U-tube bundle completely.

Though the design of AVBs in unit B differs from that of unit A, some AVB degradation in the outer bend part of the U-tube bundle has been observed in unit B as well. As a precautionary measure damping devices have been installed and proved effective so far.

3.3 Plugging of Degraded SG Tubes

In general SG tubes are plugged if the wall thickness of a tube is degraded by more than 40 %. As a precautionary measure some tubes have been plugged before this threshold has been exceeded. Fig. 6 summarizes the number of tubes plugged in both units.
During first years of operation explosive plugs have been used. Later on plugging has been performed by removable plugs to keep the option of tube repair. Since 1988 in general welded plugs are installed because of their leaktightness.

In total, there are good operating experiences with plugs except that some of the explosive plugs suffered minor leakage. The affected plugs have been replaced with welded plugs.

3.4 Surveillance Testing of SG Tube Integrity

Due to the problems experienced with SG tubes especially in unit A the scope of the surveillance programme has been extended significantly in the past. In the early years of operation during each refueling outage eddy current testing of 1300 tubes at the hot side of one SG had to be performed.

In 1980 the eddy current testing of the bend part of the outer eight tube rows (at unit A also the not plugged tubes of inner five rows) and visual inspection of the bend part of the tube bundle from the secondary side has been added to the inservice inspection programme.

Finally in 1988 1300 tubes at the cold side of the SGs have been included in the inservice inspection programme.

4 Automatic Measures to Cope with a Steam Generator Tube Rupture

Irrespective of the actions taken to reduce the degradation of SG tubes and thus to reduce the probability of a SG tube rupture, detailed investigations had been performed to evaluate the effectiveness of hardware and procedures to cope with a SG tube rupture taking into account the operating experience. The results obtained and the improvements implemented will be discussed in the following paragraphs.
4.1 General Concept to Cope with a SG Tube Rupture

In German KWU PWR's such as both Biblis units the automatic measures to cope with a SG tube rupture are designed to limit the leakrate from the reactor coolant system into the affected SG by a fast pressure reduction in the reactor coolant system below the setpoint of the SG safety valves to control the incident without actuation of the emergency core cooling system. The automatic actions are initiated by the reactor protection system on high radiation level in the main steam lines. Three N 16 detectors per steam line monitor the radiation level at the Biblis units. The original automatic measures taken in the event of a SG tube rupture are:

- reactor trip
- delayed turbine trip
- auxiliary spray in the pressurizer
- switching-off of all pressurizer heaters
- reduction of turbine bypass valve setpoint

Additionally, the letdown rate is reduced to minimum value and the make up rate to the reactor coolant system by the chemical and volume control system (CVCS) is increased by the pressurizer level control system. Auxiliary spray in the pressurizer is automatically stopped when the reactor coolant system pressure drops below the setpoint of the main steam safety valves. With all measures completed successfully, the leak rate into the affected SG decreases below the make up rate of the CVCS and the operators can identify and isolate the affected SG and cool down the plant.

4.2 Results of Investigation Regarding SG Tube Rupture Events

On behalf of the Federal Minister of Interior GRS started in 1982 an in-depth investigation of SG tube rupture events based on the international available information on significant SG tube ruptures. The objective of the investigation was to identify lessons which could be learned from foreign operating experience.

One important finding of this study was that reduction of reactor coolant system pressure in sufficient short time is essential to prevent overfill of the affected SG. In
the original design reduction of reactor coolant system pressure in the event of a SG tube rupture had to be performed by the auxiliary spray. Normal spraying with hot water from the loops is not efficient enough and pressure reduction via the pressurizer relief valves was not provided by design.

Thermohydraulic calculations performed for Biblis unit B as a reference plant, showed that pressurizer level would decrease to the low level setpoint in case of failure of automatic measures, especially failure of the auxiliary spray. By level in the pressurizer low the second necessary criterion to actuate the emergency core cooling system (ECCS) and containment isolation (CI) will be fulfilled. Actuation of ECCS and CI leads to loss of RCP's, normal and auxiliary spray, and charging pumps as well.

In such a situation with no pressure reduction capabilities in the pressurizer available it would not had been possible to restore the pressurizer level and to switch off the high pressure safety injection (HPSI) pumps because of the limited zero head of the HPSI pumps of about 11.0 MPa. Consequently there would be a continous high leak rate into the affected SG and finally an overfill of the SG as well as a flooding of the main steam line which could result in a significant loss of coolant to the environment via the main steam safety valve. To cope with such an incident unplanned operator actions would have been necessary which were not included in the procedures at that time.

Thorough investigations of SG tube rupture event sequences performed in the German Risk Study, Phase B for Biblis unit B as the reference plant evaluated SG tube ruptures to be a main contributor to the overall risk of the plant due to the low reliability of the auxiliary spray system.

4.3 Improvements Implemented to Cope with SG Tube Ruptures

4.3.1 Actions Taken in the Past

Due to the insights gained from operating experience and theoretical investigations substantial improvements in both procedures as well as engineered safety features
have been implemented at the Biblis plant during the last decade. Major backfittings are:

- In 1980/81 procedures for two additional SG tube rupture scenarios had been provided:
  - SG tube rupture assuming simultaneous loss of offsite power, i.e. actuation of the high pressure safety injection system
  - and SG tube rupture during low power operation, since actuation of the automatic measures by N16 - activity can not be guaranteed during low power operation depending on size and location of the leak.

- Following a SG tube leak in unit A in 1983 and based on results of thermohydraulic calculations the automatic measures have been extended and optimized. In addition, the actuation trains of the automatic measures and part of the auxiliary spray system have been provided with some redundancy and diversity to avoid actuation of the high pressure safety injection system.

- In 1989 as a consequence of the findings of the German Risk Study, Phase B, immediate short term actions have been taken in both units to improve the pressure suppression capabilities in the pressurizer. These measures, which were preliminary in part, include
  - the possibility to reset the automatic isolation of the reactor coolant system on low pressurizer level which disables spray and steam relief of the pressurizer and
  - procedures as well as hardware provisions to depressurize the reactor coolant system via the pressurizer vent line or relief valves.

4.3.2 Long-Term Solution to Cope with SG Tube Ruptures

Although the measures implemented in 1989 significantly reduced the risk that a SG tube rupture with additional failure of the auxiliary spray system would result in a serious accident, those measures did not comply with the "30 minutes concept", which requires no operator actions to be necessary within the first 30 minutes of a design basic accident.
Table 1: Steam Generator Design Data Biblis Unit A

<table>
<thead>
<tr>
<th>Description</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Number of Steam Generators</td>
<td>4</td>
</tr>
<tr>
<td>Design Pressure, Reactor Coolant / Steam</td>
<td>175.5 / 82.4 bar</td>
</tr>
<tr>
<td>Design Temperature, Reactor Coolant / Steam</td>
<td>350 / 350 °C</td>
</tr>
<tr>
<td>Reactor Coolant Flow</td>
<td>18 000 t/h</td>
</tr>
<tr>
<td>Total Heat Transfer Surface Area</td>
<td>4 510 m²</td>
</tr>
<tr>
<td>Heat Transferred at Full Load</td>
<td>885 MW</td>
</tr>
<tr>
<td>Steam Conditions at Full Load:</td>
<td></td>
</tr>
<tr>
<td>Steam Flow</td>
<td>1 670 t/h</td>
</tr>
<tr>
<td>Steam Temperature</td>
<td>265.2 °C</td>
</tr>
<tr>
<td>Steam Pressure</td>
<td>52 bar</td>
</tr>
<tr>
<td>Maximum Moisture Carryover</td>
<td>0.25 %</td>
</tr>
<tr>
<td>Feedwater Temperature</td>
<td>207 °C</td>
</tr>
<tr>
<td>Number of U-tubes</td>
<td>4 060</td>
</tr>
<tr>
<td>U-tube outer Diameter / Wall Thickness</td>
<td>22 x 1.2 mm</td>
</tr>
<tr>
<td>Reactor coolant volume</td>
<td>32.5 m³</td>
</tr>
<tr>
<td>Secondary Side Volume</td>
<td>174.7 m³</td>
</tr>
<tr>
<td>Steam Generator Shell Material</td>
<td>22 NiMoCr 37</td>
</tr>
<tr>
<td>Tube Sheet Material</td>
<td>22 NiMoCr 37</td>
</tr>
<tr>
<td>U-tube Material</td>
<td>Incoloy 800</td>
</tr>
</tbody>
</table>
Table 2: Mean Wastage Rate per Year Observed in Biblis Unit A since 1985

<table>
<thead>
<tr>
<th>Year</th>
<th>Mean Wastage Rate [ % ]</th>
</tr>
</thead>
<tbody>
<tr>
<td>1985</td>
<td>2.5</td>
</tr>
<tr>
<td>1986</td>
<td>2.4</td>
</tr>
<tr>
<td>1987</td>
<td>2.8</td>
</tr>
<tr>
<td>1988</td>
<td>2.1</td>
</tr>
<tr>
<td>1989</td>
<td>1.1</td>
</tr>
<tr>
<td>1990</td>
<td>1.0</td>
</tr>
</tbody>
</table>

Phosphate

AVT
Figure 1 Steam Generator Biblis Unit A

Figure 2 Tube Support Lattice
Figure 3 Tube Support Structure in the U-bend Part of Tube Bundle Biblis Unit A

Figure 4 Main Steam Generator Tube Degradation Mechanisms Observed in Biblis Unit A
Figure 5 Anti Vibration Bars at the U-bend Part of Tube Bundle

Figure 6 Plugged Tubes in the Steam Generators of Biblis NPP
ECC-Criteria (2 out of 3)

RPV Level < MIN3

Level of 1 out of 4 SG > 13.5 m

\[ \Delta p \text{ Containment/Atmosphere} > 30 \text{ mbar} \]

HPSI-Actuation

Figure 7 HPSI-Actuation Logic

Instrumentation Pipe

Upper Support Plate

Measuring Probe

Reactor Coolant Loop

Figure 8 RPV Level Measurement
STEAM GENERATOR LIFE TIME EXTENSION
BY MAINTENANCE AND REPAIR

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ABSTRACT
Steam Generators (SG) in PWR's are highly safety related and very important components for the operation of the plant. Worldwide experience has shown that tube degredations might occur which need short or long term repair measures. For this application ABB has developed various SG tube repair methods and applied also plant specific remedies. How the systematic analysis of a problem followed by main-tenance or repair methods can improve the operation of a plant will be discribed on the basis of the work ABB has performed since 1985 in KORI Unit 1, a 2 loop plant built by Westinghouse which started its commercial operation in 1978.
1. INTRODUCTION

ABB is involved in steam generator maintenance and repair work in South Korea since 1985. Service Work has been performed in KORI Unit 1, 2, 3 and 4, Yeonggwang 1 and Wolsung 1. Service equipment and technologie has been supplied for all operating nuclear units.

How the systematic analysis of a problem followed by maintenance or repair measures have improved the operation of a plant will be demonstrated for the work ABB performed during the last years in KORI Unit 1.

KORI Unit 1 is a 2-loop plant built by Westinghouse. It has a power output of 587 MWe. The start of the commercial operation was in 1978.

In 1985 KEPCO experienced an unexpected high number of steam generator tubes which were plugged due to defect indications. As an alternative to a possible steam generator replacement, ABB started discussions about a steam generator lifetime extension program with KEPCO and offered steam generator maintenance and repair work.

The work performed since then by ABB in KORI Unit 1 is summarized in [Table 1].

<table>
<thead>
<tr>
<th>Year of Service</th>
<th>Service Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>1985</td>
<td>General evaluation of the steam generator situation and discussion of possible improvements</td>
</tr>
<tr>
<td>1986</td>
<td>Reanalysis of previous results from eddy current inspections of steam generator tubes</td>
</tr>
<tr>
<td></td>
<td>Performance of eddy current inspection of both steam generators</td>
</tr>
<tr>
<td></td>
<td>Installation of 103 ABB mechanical plugs</td>
</tr>
<tr>
<td>1987</td>
<td>Award of contract for steam generator reconstitution and technology transfer</td>
</tr>
<tr>
<td></td>
<td>Qualification and preparation for steam generator reconstitution</td>
</tr>
<tr>
<td>1988</td>
<td>Reanalysis of eddy current inspection results from 1987</td>
</tr>
<tr>
<td></td>
<td>Eddy Current Inspection and Analysis of both steam generator</td>
</tr>
<tr>
<td></td>
<td>Removal of 1380 Westinghouse plugs</td>
</tr>
<tr>
<td></td>
<td>Removal of 103 ABB mechanical plugs</td>
</tr>
<tr>
<td></td>
<td>Extraction of 4 tubes</td>
</tr>
<tr>
<td></td>
<td>Installation of 558 ABB TIG-Sleeves</td>
</tr>
<tr>
<td></td>
<td>Installation of 106 ABB welded plugs</td>
</tr>
<tr>
<td></td>
<td>Recommendation for chemical cleaning of SG</td>
</tr>
<tr>
<td>1989</td>
<td>Installation of 193 ABB TIG-Sleeves</td>
</tr>
<tr>
<td></td>
<td>Installation of 16 ABB mechanical plugs</td>
</tr>
<tr>
<td></td>
<td>Installation of 42 ABB welded plugs</td>
</tr>
<tr>
<td></td>
<td>Recommendation for chemical cleaning of SG</td>
</tr>
<tr>
<td>1990</td>
<td>Chemical Cleaning of both steam generators</td>
</tr>
<tr>
<td></td>
<td>Removal of remaining 34 Westinghouse plugs</td>
</tr>
<tr>
<td></td>
<td>Installation of 40 ABB welded plugs</td>
</tr>
<tr>
<td></td>
<td>Installation of 88 ABB mechanical plugs</td>
</tr>
<tr>
<td></td>
<td>Installation of 415 ABB ATS-Sleeves</td>
</tr>
</tbody>
</table>

Table I
Maintenance and Repair performed by ABB in KORI Unit 1

2. MAIN ACTIVITIES

Since 1986 ABB is performing steam generator service work in KORI-Unit 1, starting with eddy current inspection and technologie transfer to KEPCO. The orders for the various work were always awarded to ABB against heavy international competition.

In 1986 ABB introduced the ABB mechanical plug to the Korean utilities [Fig. 1]. This plug is characterized by a leak tight microlock fit. It can be installed and removed easily without damage to the SG-tube. The plug can also be inspected by NDE methods, such as eddy current inspection. Plugging equipment and technologie have been supplied to the customer, so that in case of tube failures KEPCO can plug the tube fast and easily with own personnel and restore the tube at a later date by sleeving.

In 1988 ABB has reactivated over 750 tubes which had been plugged before. The work was presented in detail during the 3rd KAIF/KNS Annual Conference 1988 in Seoul and included the removal of plugs by hydraulic pulling devices and mechanical machining and the installation of TIG welded sleeves.
Finally 1990 the steam generators have been chemically cleaned and the hard sludge with high copper content has been removed from the tube sheet to over 95%. [Fig. 4] A low temperature tube sheet cleaning process was applied using EDTA/EDA solvents. External equipment for temperature adjustment and recirculation was used to allow parallel maintenance work by the utility such as main cooling pump maintenance. After the chemical cleaning an eddy current inspection was performed. No remaining sludge signal or copper signal were detected.

Also in 1990 ABB introduced the advanced tube sheet sleeve (ATS-Sleeve) to KEPCO and installed 415 of those sleeves. [Fig. 2] This sleeve is characterized by only a TIG weld on the upper end. The lower end has a leak tight mechanical microlock, similar to the one of the ABB mechanical plug. The installation of the ATS-Sleeve is limited to applications with the lower joint within the tube sheet. The elimination of the lower weld is the first step towards a leak tight tube support plate sleeve without any weld or other metallic bonding. This kind of sleeve could be installed into any position in the straight tube section and is currently under development at ABB. It should be available for field installation by 1992.

As the chemical cleaning as well as the ATS-Sleeve are new ABB service products, they will be described in more details.

Fig. 1: Removable Plug, General Design

Fig. 2: ABB-ATS-Sleeve

3. CHEMICAL CLEANING

The analysis of sludge samples showed a copper content as high as 80% during the first years of operation with a rapid decrease after the replacement of the condenser tubes in 1988. The average was estimated to be above 50%. The thickness of the sludge pile was measured by eddy current to be up to 450 mm. This required a multiple step cleaning process to assure the complete removal of the sludge.

Extensive KORI-1 specific qualifications were performed before the final process was applied. The qualification program included the materials evaluation, bench scale and pilot scale testing including actual sludge evaluation from the last KORI-1 sludge lancing.
3.1 PROCESS DESCRIPTION

A low temperature process was selected with application temperatures of 95 °C for the iron components and 90 °C for the copper coil. Operating temperatures are 90 °C and 60 °C.

![Diagram of INCONEL 690]

**Fig. 10** ABB Standard SG-Sleeve

![Diagram of Design 1 and Design 2]

**Fig. 11** ABB-ATS-Sleeve

The first step to a leak tight sleeve without welds was the elimination of the lower weld. As the joint between the lower sleeve end and the tube looks very similar to the one of the mechanical plug joint [Fig. 11] only a limited no of additional testing was necessary to qualify this joint. The only item which had to be verified was the axial load due to the thermal differential expansion of the sleeve (Inconel 690) and SG-tube (Inconel 600) and the different tensions and strains due to the installation conditions.

The installation of the ATS-Sleeve is limited to applications with the lower end still within the tube sheet.

The installation steps and equipment were basically the same as for the welded sleeve, except that the preparation for the lower weld was eliminated.

5. SUMMARY

ABB is performing steam generator maintenance work in KORI-1 since 1986. Tube sleeving instead of tube plugging is performed wherever possible to keep the reduction of heat transfer area as low as possible.

During the 1990 outage in KORI Unit-1 the first chemical cleaning in a Korean NPP was applied, removing the hard deposits form the tube sheet and the tubes. This cleaning eliminated the collected impurities and possible reasons for tube degradations. The introduction of the ATS-Sleeve in 1990 was a further step to keep tube plugging to an absolute minimum and thus improving plant operation. All these activities, plus the modifications performed by KEPCO as recommended by ABB - such as replacement of copper containing components, improvement of water chemistry control - have lead to an improvement in the plant conditions and safety.
8.1 CORRECTIVE AND PREVENTIVE MAINTENANCE TECHNIQUES FOR DEGRADATION OF STEAM GENERATOR TUBING

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8.2 STEAM GENERATOR FRAMATOME REMOVABLE PLUG

G. Morel, Framatome - Nuclear Services, FRANCE
J.C. Mounet, Framatome - Nuclear Services, FRANCE
J.P. Billoue, Framatome - Nuclear Eng. Primary Equipment Division, FRANCE
G. Poudroux, Framatome - Nuclear Eng. Primary Equipment Division, FRANCE

8.3 EXPERIENCE IN THE USE OF THE NICKEL PLATING PROCESS AS A PREVENTIVE OR A CORRECTIVE MEASURE

R. Houben, Belgatom/Electrabel, Brussels, BELGIUM
C. Verbeeck, Belgatom/Electrabel, Brussels, BELGIUM
C. Schinazi, Belgatom/Tractebel, Brussels, BELGIUM
J. Stubbe, Belgatom/Laborelec, Brussels, BELGIUM
C. Laire, Belgatom/Laborelec, Brussels, BELGIUM

8.4 PROGRAM FOR STEAM GENERATOR TUBING RELIABILITY TESTS IN JAPAN

F. Ishizuka, Japan Power Engineering and Inspection Co., Tokyo, JAPAN

8.5 STEAM GENERATOR TUBE LASER SLEEVING

M. Batistoni, Framatome, FRANCE

8.6 DEVELOPMENT OF A STEAM GENERATOR SLEEVING SYSTEM USING FIBER OPTIC TRANSMISSION OF LASER LIGHT

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L. Van Hulle, Westinghouse European Service Center, Nivelles, BELGIUM

8.7 QUALIFICATION AND FIELD EXPERIENCE OF SLEEVING REPAIR TECHNIQUES

J. Stubbe, Belgatom/Laborelec, Brussels, BELGIUM
J. Berthe, Belgatom/Tractebel, Brussels, BELGIUM
C. Verbeeck, Belgatom/Electrabel, Brussels, BELGIUM
CORRECTIVE AND PREVENTIVE MAINTENANCE TECHNIQUES
FOR
DEGRADATION OF STEAM GENERATOR TUBING

Osamu Takaba
Mitsubishi Heavy Industries Ltd.

Abstract
In order to maintain steam generator tube integrity, we applied many types of techniques. The followings are our major activities.

1. Sleeving
Now sleeving technique is used not only repair of degraded tube but also reuse of previously plugged tube and prevention of tube degradation.


4. TT600 Mechanical plug Replacement.
1. Introduction

In order to maintain the tube integrity of operating Steam Generators, it is important to implement the preventive and corrective action against the recurrence of tube degradation as well as the determent of propagation of that. During the year of operation, various types of tube degradation has been experienced. For many years since then, strenuous efforts to suppress the degradation have been exerted.

This report introduces corrective and preventive maintenance techniques experienced in Japan.
2. Slewing

If tube degradation is detected by eddy current test at annual inspection, the tube needs plugging or sleeving. Originally, all tubes with defect were repaired by plugging. However, as the rate of plugging increased, the sleeving technique was developed as a means of allowing continued use of tubes once they had been repaired. Now sleeving technique is used not only repair of degraded tube but also reuse of previously plugged tube and prevention of tube degradation.

Types of the sleeving repair technique are as follows.

(1) Mechanical sleeve

The mechanical sleeve technique was developed in 1980 and has been applied to the repair of secondary side intergranular stress corrosion cracking into tubesheet crevice and primary water stress corrosion cracking in roll transition region. The typical example of this method is shown in Fig. 1.

(2) Brazing Sleeve

The brazing sleeve technique was developed in 1984 and has been applied to the repair of secondary side intergranular attack at the tube support plates and at the secondary surface of tube sheet. So far about 5,000 tubes had been sleeved by this technique. The concept of this sleeve is shown in Fig. 1.

(3) Laser welding sleeve

Recently the laser welding sleeve was developed, and from 1989 this new technique started to be applied to the actual plants. This technique is applicable to both tube support plate region and tube sheet region and has improved repair efficiency as shown in Fig. 2. Therefore preventive sleeving became possible in order to prevent the secondary side intergranular attack. So far about 5,000 tubes had been sleeved by this method.
Fig. 3 shows the typical location of degradation (IGA, PWSCC) having been experienced in the Steam Generator tubes. According to the degradation location in the tube, the sleeve is fitted to the tube support plate area or tube sheet area.

Fig. 4 and 5 show the outline of the laser welding sleeve system. The YAG laser oscillator is placed at the outside of the containment vessel (C/V). The incident optical system condensed the laser beam to the long optical fibers of a small diameter, and the laser beam is transmitted to the Steam Generator in the C/V. Then, the laser beam is led through the sleeve of about 16mm I.D. having been inserted and fixed at the damaged position. The laser beam that is focused by small focusing lens system, reflected by the mirror, is irradiated at the surface of the sleeve. From inside the sleeve, a part of these optical systems rotates, and the sleeve and tube are lapwelded together. This welding was carried out in three passes to increase the welding width. The cross section of three path lapwelded joint is depicted in Fig. 6.

Procedure of laser welding sleeve mainly consists of five steps as shown in Fig. 7. Hydraulic expansion is the process to fit the sleeve to the tube surface and to hold the sleeve at the right position. Visual inspection is the process of measuring the width of bead at the sleeve surface of three path lapwelded joint, instead of measuring the welded length between sleeve and tube by ultrasonic testing, that is impossible because of roughness of welded bead surface. ECT is the process to get the reference data to compare with that of future in-service inspection.
Figure 1 History of applied sleeving repair techniques.

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T.S : TUBE SHEET
T.S.PL : TUBE SUPPORT PLATE
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<tr>
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Figure 2 Comparison of sleeving technique.
Figure 3  Location of damages and concept of sleeve.
Figure 4 Outline of laser welding sleeve technique.
Figure 5  S/G laser welding sleeve repair system.
Weld joint appearance

Cross section

Figure 6 Tube with sleeve laser-welded joint.

8.1-10/18
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Figure 7: Laser welding procedure at tube support plate.
3. Anti-Vibration Bar (AVB) Replacement.

At some 5F series Steam Generator, tube wear was occurred at Anti-Vibration Bar in U-bend.

The cause of tube wear is the tube vibration due to secondary two phase fluid flow at U-bend for the tube with inadequate gaps between tube and AVB. To evaluate availability and effectiveness of countermeasures, full scale model (above #7 tube support plate) mock up test was carried out. Gap between tube and AVB are measured by tube vibration measurement method as shown in Fig. 8, and it is concluded that the AVB replacement is the most reliable remedial method to prevent the tube wear due to flow induced vibration at U-bend.

New AVB is the special design of V-shape with hinge to access easily from man way through U-bend region as shown in Fig. 9.

And to minimize the clearance between tube and AVB, combination of slide type expandable AVB and flexible AVB with slot is adopted.

After insertion of expandable AVB and flexible AVB, expandable AVB is expanded by means of sliding the half side of the bar to reduce the gap between tube and AVB.

In that time, flexible AVB will bend and absorb the force to prevent the tube deformation.

Careful preparation for AVB replacement is necessary and replacement work is carried out under water as shown in Fig. 10.

AVB replacement was carried out for 13 Steam Generators in Japan.
Figure 8  Steam Generator tube vibration measurement method.
Figure 9 New AVB Assembly.

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Figure 10 Preparation for AVB replacement.
4. Shotpeening

As the preventive maintenance work, crevice blockage for partially expanded tubes in tube sheet regions was executed. The purpose of crevice blockage was to make new hydraulic expanded transition to take out the susceptibility of PWSCC at roll transition region and close the secondary side tube to the tube sheet crevice to prevent the secondary side IGSCC. The tube to tube sheet crevice was closed by hydraulic and mechanical roll expansion as shown in Fig. 11. Because mechanical roll expanded region has some possibility of PWSCC, shot peening after mechanical roll expansion was applied.

Shotpeening is a method to shot the beads (small particle) to insidetube surface in order to produce the residual compression stress at the material surface. Beads are fed by beads feeder and transfered by air to inside tube and reflect at the nozzle surface and shot inside surface of tube as shown in Fig. 11.

Shot peening applied to 4 units in Japan.
Figure 11 Procedure of crevice blockage expansion.
5. Mechanical Plug Replacement

In early years, the explosive plug and weld plug were used. However since 1981 TT600 mechanical plug has been adopted to reduce the amount of radiation exposure and to shorten the repair time.

When the TT600 mechanical plug failure occurred at North Anna unit 1, Over 13,000 mechanical plugs have been installed at 11 units in 17 operating units in Japan.
About 3,500 of 13,000 plugs were imported and the remainders were domestic product.

Based on the information such as NRC Bulletin, Action plan was established. "2 or more plugs of each lots shall be removed and examined at the next scheduled refueling outage."

Total 25 different heat lots (96 plugs) were removed and inspected.
Some plugs were identified as axial PWSCC below expander.
Laboratory tests results and removed plug examination results in Japan and U.S. shows that TT600 mechanical plug has potential to experience PWSCC.

To keep the integrity of Steam Generator, all TT600 mechanical plug in hot leg side are decided to be removed and replaced with TT690 mechanical plug. Plug replacing schedule are developed plant by plant. Nowadays, only about 450 TT600 plugs in hot leg side are waiting for removal.
TT690 mechanical plugs are installed in model boiler and corrosion test in primary side condition is being continued with no defect.

8.1-18/18
STEAM GENERATOR FRAMATOME REMOVABLE PLUG

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JP. Billoue - G. Poudroux (Framatome - Nuclear Engineering Primary Equipment Division).

ABSTRACT

Numerous degradations of steam generator tubes necessitate their plugging either definitively or temporarily because it is often advantageous to temporarily plug a few leaking tubes while waiting for either a scheduled extended outage to perform the necessary repairs or a new repair process development.

To meet these two objectives Framatome has developed an easily removable plug allowing a better scheduling of repair campaigns.

This plug is easy to install and remove, have good inservice behaviour and corrosion resistance and is suitable for all expanded tube diameters within manufacturing tolerances.

A huge qualification program including accidental conditions was performed to confirm the performances of this removable plug.

The presentation describes all the design and justification steps which led to the qualification and the tools and procedure used for installation and removal.

BOUCHON DEMONTABLE

Les types de dégradation rencontrés sur des tubes de GV nécessitent leur bouchage définitif ou temporaire. Il est avantageux d’obtenir provisoirement ces tubes en attendant un arrêt programmé pour effectuer les réparations nécessaires ou le développement d’un nouveau procédé de réparation. Pour ce faire, Framatome a développé un bouchon facilement démontable, permettant d’optimiser et de planifier les opérations de maintenance des faisceaux tubulaires. Ce bouchon facile à poser et à déposer, a un bon comportement en service et une excellente résistance à la corrosion. Il convient pour tous les diamètres de tubes expansés dans les tolérances de fabrication. Un programme de qualification très important dans les conditions de fonctionnement y compris les conditions accidentelles a été réalisé pour confirmer les performances de ce bouchon. La présentation décrit toutes les phases de conception et de justification qui ont conduit à la qualification du produit, des outils et des modes opératoires utilisés pour la pose et la dépose de ce bouchon.
A number of damaged steam generator (SG) tubes necessitate either a definitive or a temporary plugging. Indeed, it is often advantageous to temporarily plug a few leaking tubes, while waiting for either a scheduled extended outage to perform the necessary repairs, or for a new process development.

To meet these two objectives—temporary as well as definitive plugging, FRAMATOME Nuclear Services has developed an easily removable plug.

The possibility of rapid plug insertion and removal permits greater flexibility in scheduling tube bundle repair operations and allows the establishment of a maintenance strategy. The possibility of periodic inspection of plugged tubes to get a better understanding of their inservice behaviour is also one of the advantages of the easily removable plug.

In designing its new plug, FRAMATOME has taken into account operating information (experience) such as that concerning the possibility of cracking leading to breaks, the loss of leaktightness, and the loss of mechanical strength causing the creation of loose parts.

Analysis of these problems and the examination of the damage observed permitted finalizing the design and defining an unprecedented qualification program to test:

- the leaktightness and mechanical strength of the plug under all steady state and transient operating conditions, regardless of its location on the tube sheet (taking into account the loadings induced by the tube sheet behaviour in service conditions).

- the leaktightness and mechanical strength of the plug during hydrotests.

- the inservice corrosion resistance.

- the effect of the plug on the corrosion resistance of the tube after it has been removed.

- the mechanical and leaktightness strength of the plug under faulted conditions.
PLUG DESCRIPTION

The removable plug must be easy to install and remove, have good inservice behaviour, resist corrosion, and be suitable for all expanded tube diameters, within manufacturing tolerances, and for Inconel 600 and Inconel 690 tubes.

These requirements have led to the design shown in figure 1. The original design Framatome removable plug is based on a shell with two sealing lands which are put into contact with the inner surface of the tube by a permanent expander, via an insert.

The materials used were chosen for their mechanical properties and their high corrosion resistance to both the reactor coolant and secondary fluid: Inconel 690TT for the shell, 316 L stainless steel for the expander, and nickel 201 for the insert.

Plug installation doesn't necessitate any prior preparation of the tube (no diameter measurement, no cleaning). After rework of the tube inlet, if necessary, the installation is performed in one single operation, by traction on the expander. This phase and the tool design were optimized to obtain a very low residual stress level in the shell. Installation criteria allow verifying that the plug is installed in accordance with the qualification limits. Plug installation is characterized by the five phases shown in the curves in Figure 2. These curves are used to define an installation criterion which ensures that the plug has been inserted in a tube whose diameter is within the qualification range.

The plug, in the "inserted" position is represented in Figure 3.

The removal operation is also performed in only one step, by pushing the expander against the shell top. The reduction of the shell diameter allows eliminating the tube/plug contact, and the plug can then be withdrawn easily, without efforts and without damage on the tube wall. After removal, it is possible to install a new plug on the same tube, but it must be 2 mm higher or lower.

The FRAMATOME plug removal sequences are represented in Figure 4.
EXTENSIVE QUALIFICATION

Prior to qualification, the plug was evaluated in a test program that took into account in-service behaviour under all conditions (including accident conditions), corrosion resistance, materials acceptability, and its ability to be effectively removed.

In addition, a theoretical analysis has permitted understanding how the plug operates, and simulating its behaviour in order to evaluate the effects of parameters not tested during the experimental program.

In-service behaviour was demonstrated based on the testing and analysis programs discussed below:

- **Functional aspects**

  - The first part of the test program consisted in verifying the plug performances on single tube mockups during static tests for different configurations (cleaned and oxidized tubes, scratched tubes, skiprolls, etc ....).

The following tests were performed:

- Rotation tests to evaluate the tube-to-plug interface pressure,
- Primary and secondary side hydrotests to verify leaktightness and the mechanical strength.
- Helium pressure test to evaluate leaktightness. The criteria to be respected are the same used for tube/tubesheet welds.
- Expulsion tests to evaluate mechanical strength margins.
- Installation under water followed by heating to demonstrate the absence of any risk due to pressure buildup.
- Measurement of marks left in the tube by the plug to evaluate any possible damage to the tube.

All these tests are performed either after or before and after thermal cyclings.

- The second part consisted in tests performed on specific facilities to simulate the in-service behaviour. Leaktightness and mechanical and fatigue resistance were demonstrated for the most severe parameters under normal, upset, emergency, faulted, and test conditions. Three-D modeling (finite element analysis) of the lower part of the SG allowed calculating the deformation of the tubesheet holes for the different service conditions. The results showed two different types of deformations, and then two different plug behaviours, depending on the plug location.
The plug was therefore qualified for any tube location in the tubesheet by means of two facilities equipped with multi-tube mockups:

- "SEFTRAN" mockup: central zone with bi-axial hole deformations (see figure 5).
- "PERI" mock-up: peripheral zone (between 500 and 600 tubes) with monoaxial hole deformations (ovalization) see figures 6 and 7.

Dynamic test program

Three experimental programs were undertaken.

The tests performed on the "SEFTRAN" test loop (figure 5) served to demonstrate the leaktightness and mechanical and fatigue resistance of the plugs installed in the tubesheet, away from the periphery, for normal, upset, emergency and test conditions.

To obtain this tubesheet hole deformation, cyclings were simulated on the mock-up and the plugs were continuously monitored to see if displacement or leakage developed. Helium and water leaktests, thermal cyclings, mechanical tests and plug expulsion tests were carried out before and after fatigue cycles and water leak tests, mechanical tests, and expulsion tests were performed during the cycles with maximum tensile or compressive stresses in the tubesheet ligaments.

The above tests did not permit simulating the deformation of holes close to the outer edge of the tube sheet. "Peri" mockups (Figure 7) were developed to test plug performance in this region for normal, upset, emergency and test conditions.

The same kind of leak and mechanical tests as for the "SEFTRAN" test program before and after fatigue cycles and during cycles with maximum tensile and compressive stresses in the tubesheet ligaments were performed.

Faulted conditions

The hole deformations during faulted conditions are much more important than the maximum simulation capacity of the above-mentioned test facilities. So the qualification in faulted conditions was performed with a third specific mockup ("LOCA"). The results obtained during the test program allowed to verify the mechanical resistance and the leaktightness of the plug for maximum ID tube in faulted conditions.
For all the test programs mentioned above (static and dynamic tests) the ability of the plug to be effectively easily removed was verified.

- **Theoretical analysis program**

  The model developed permitted simulating plug installation, thermal cycling, envelope transients to determine the orders of magnitude of the tube-to-tube sheet interface pressure, the tube-to-plug interface pressure, and the depth and width of the plug tooth marks in the tube.

  These analyses permitted evaluating the effect of varying certain parameters, such as the mechanical characteristics, on plug performance, and to verify the severity of the different test conditions.

- **Metallurgical and corrosion aspects**

  Numerous examination were performed on mockup cross sections (microhardness measurements) to verify the acceptability of each material used and that the tubesheet ligaments were not affected.

  In addition, sections were taken from the plugs inserted in the tubesheet simulators. They showed that the ligaments were not affected by insertion of the plugs, as well as the excellent design of the plug with respect to the tube: the effect on tube hardness due to the teeth was less than that observed from the overlapping of successive rolling steps.

  Corrosion studies dealt with:

  - The choice of materials noted for their resistance to corrosion (Inconel 690 TT for the shell, nickel 201 for the insert, and 316 L stainless steel for the expander).
  - The design of an installation process that limits the tensile stresses in the shell during insertion.
  - The design of a plug that minimizes the risk of tube damage from the shell teeth, thereby limiting the risk of tube corrosion after the plug has been removed.

  The results of numerous tests in water containing 10% NaOH confirmed this choice of plug design, since no cracking was observed.
In addition, others tests (10% Na OH) permitted verifying that there was no increase in the risk of tube cracking after the plug was removed (no cracking was observed in the region of the contact area between plug and tube).

Material reports were prepared for the Inconel 690 TT, the nickel 201, and the cold-drawn 316 L stainless steel used for the various parts of the plug.

EASE OF USE

Installation and removal are each performed in a single operation using the same tool. The tool is designed to minimize the overall dosimetry.

Depending on the number of tubes to be plugged, the operation is performed manually, with the insertion or removal tool being operated by "jumpers" or automatically, with the insertion or removal tool fed from an automatic magazine, thus avoiding the need for personnel to enter the channel head, except to install and remove the spider (Figure 8).

Regardless of whether the operation is manual or automatic, the insertion curve (force/displacement of expander core) is recorded; it permits comparison with insertion criteria to check that the plug is correctly installed in a tube whose diameter is in the qualification range.

Calibration kits permit setting and checking the parameters during the operation, as well as validating the tools.

PROCEDURES

The procedures are applicable both for the manually operated tool and the automatic tool. In describing this operation, it is assumed that the tool is in place in the channel head.

Prior to insertion, the plug is either held by the operator from the manway or by the pneumatically loaded magazine. The plug expander core is screwed to the end of the tool. It is then inserted into the tube and positioned either flush with the bottom of tubeshell or further into the tube sheet. By means of a hydraulic cylinder with restricted travel, the expander is pulled down, and by compressing the insert, pushes the teeth of the plug against the tube.

During this operation, the recorder indicates the pulling force as a function of the displacement of the expander. Checking certain characteristic points on the curve, as well as expander travel and insertion force, permit validating the plugging operation.
To remove the plug, the threaded end of the tool is screwed to the plug shell. The hydraulic cylinder, fitted with a withdrawal head, pushes the expander and the insert upwards, thus releasing the teeth; then it stretches the shell, causing the pulling in of its walls in the area around the teeth, permitting the plug to be easily removed.

The tooth marks left on the tube are quite shallow, only a few microns, and have no effect on the mechanical strength or corrosion resistance of the tube.

OPERATION

The maintenance operation includes site access formalities, installation of the maintenance equipment, jumper training, tube plugging, removal of the maintenance equipment, and site departure formalities.

The rate of automatic installation or removal of the plugs is approximately 100 plugs per shift of 8 hours (tool installation and removal not included).

The time necessary for manual installation and removal of this new Framatome plug is the same as the time needed for installation (only) of a classical mechanical plug.

CONCLUSION

The Framatome removable plug design permits obtaining an excellent mechanical performance without sacrificing corrosion resistance. This fact was demonstrated throughout the qualification program by:

- The large margins observed with respect to inservice leaktightness and mechanical strength, even under accident conditions, regardless of where the plugged tube was located.

- The large margins with respect to the risk of corrosion (no cracking observed after 1500 hours in water containing 10% NaOH).

- The absence of any impact on the corrosion resistance of the tube following removal of the plug (no cracking observed after 500 hours in water containing 10% NaOH).

The speed with which the plug can be installed and removed permits greater flexibility in scheduling tube bundle maintenance operations.

From now on, Framatome recommends the use of this new removable plug, be it for permanent or temporary plugging.
SEFTRAN TESTS
"TRANSIENT BEHAVIOUR"

Figure 5
AUTOMATIC REMOVABLE PLUG TOOLING

FIGURE 8
EXPERIENCE IN THE USE OF THE NICKEL PLATING PROCESS
AS A PREVENTIVE OR A CORRECTIVE MEASURE

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ABSTRACT

Preventive and repair methods were very early investigated by the Belgian utilities to fight against primary water stress corrosion cracking in Steam Generator (SG) tubes.

Among the different techniques, a very promising method is the nickel electroplating of the SG tube roll transition zones.

This paper gives the main characteristics of this process and reviews its applications in the Belgian NPP’s since 1985 as a corrective as well as a preventive technique.

RESUME

Les exploitants belges ont développé d’importants efforts afin de combattre la corrosion sous tension en milieu primaire apparaissant dans les tubes de Générateurs de Vapeur (GV).

Parmi les différentes techniques mises en œuvre, le nickelage électrolytique des zones de transition de dudgeonnage des tubes de GV semble être promis à un bel avenir.

Ce papier donne les caractéristiques propres à ce procédé et passe en revue les différentes applications aux centrales belges depuis 1985 en tant que méthode corrective et préventive.
1. INTRODUCTION

A lot of different methods were investigated by the Belgian utilities to fight against Primary Water Stress Corrosion Cracking (PWSCC) in the roll transition of SG tubes, ranging from preventive techniques in undamaged or slightly damaged SG's, like global heat treatment or shot/roto peening during preservice operations or an outage, to curative and/or preventive techniques on already damaged SG's, like sleeving and nickel plating.

These two last techniques can be considered for in service SG's as very promising for long term operation of PWSCC affected tubes.

This paper will consider only the Ni plating process, developed jointly by Laborelec and Framatome since 1984. The sleeving technique will be reviewed in a companion paper [1].

2. NICKEL ELECTROPLATING DESCRIPTION

2.1 Basic principle (figure 1)

The nickel plating consists in depositing on the PWSCC affected areas a thin pure nickel coating, usually 50 to 200 microns thick, by an electrolytic process. The nickel plugs the existing cracks, protects the plated area from primary water contact and as a consequence restores the leaktightness, prevents any further propagation from the primary side of the plugged cracks, as well as appearance of new cracks. The process is fully adapted as a corrective or a preventive method to cope with PWSCC.

2.2 Main characteristics of the Ni plating

Compared with other "heavier" repair methods nickel electroplating offers the following advantages:

2.2.1 Functional characteristics

- does almost not introduce geometrical restrictions.
- does not affect the thermal transfer.

2.2.2 Expected service behaviour, safety, Inspectability

- does not introduce new residual stresses.
- preserves the same "leak before break" behaviour as for unplated tubes (safety concern).
- the nickel plated areas remain inspectable by ultrasonic inspection (UT).
2.2.3 Possible secondary detrimental effects

- no possibility of an "Obrigheim effect".
- no modifications of the parent tube structure.

2.2.4 Process flexibility and versatility

- applicable to any tube of the SG at any elevation in the straight portion of the tubes.
- fully reversible.
- allowing the implementation of other repair methods if needed.
- the process is monitored in "real time" by plating samples outside the SG during the implementation. Deviations can be detected very soon and corrective actions can be taken rapidly, limiting the amount of failed tubes.

2.3 Possible limitations

Nickel plating does not restore the mechanical resistance of tubes, as does sleeving. Except for partially rolled tubes, the process is thus limited in its corrective application to tubes with axial cracks smaller than their critical length.

Additionally, the experience has shown that for cracks wider than 20 to 30 microns the following problems could occur:

- incomplete plugging of the cracks
- secondary side attack of the nickel layer in case of partially rolled tubes due to a mixture of process products (like sulfamic acid) and secondary side impurities trapped in the crack and in the gap (only for through cracks).

However since such wide cracks are not frequent, this concern does not represent a practical limitation.

Finally, the potential propagation in length of through wall cracks from secondary side which was earlier feared does not seem to be a concern anymore since UT inspections in Doel 3 performed before plating and after plating during two successive cycles did not reveal any crack propagation.

2.4 Process description

The process includes the following steps:

- mechanical cleaning (honing)
- chemical surface preparation (pickling)
- rinsing
- precoating nickel deposit ("strike")
• rinsing
• nickel plating
• rinsing

Except for the mechanical cleaning, all the steps are performed with the same tool. The whole process is carried out remotely.

The quality of the plating solutions is controlled permanently by pH measurement and plating potential monitoring. Baths are changed whenever deviation of these control parameters occurs and in any case after a limited number of plated tubes. Additionally, samples are plated during the process in order to check the quality of the deposited nickel.

3. **ON SITE APPLICATIONS**

3.1 **Nickel plating as corrective measure against PWSCC**

Corrective implementation of the nickel plating technique was first investigated during three applications in Doel 2 SG equipped with partially rolled tubes without kiss roll:

• 1985 : 10 tubes
• 1986 : 81 tubes

These applications confirmed the great sensitivity of nickel plating to process parameters and bath compositions as previously highlighted by the laboratory tests.

Concerning the service behaviour, the different inspections (endoscopic inspection in the SG tubes, visual and destructive examination of pulled tubes) also evidenced some degradation mechanisms such as pitting, erosion or in case of long (> 10 mm) and wide cracks (> 20 microns) secondary side corrosive attack, and cracking of the nickel layer.

The experience gained during these applications allowed Laborelec and Framatome to develop jointly an improved procedure with fully reproducible results which was applied in 1988 in Doel 2 and Doel 3:

• plating of 33 tubes chosen among the most severely damaged in Doel 2.

• plating of 11 tubes with a very well known crack pattern in Doel 3 in order to investigate the potential crack growth by UT inspection (Eddy Current Testing, or ECT, can not be applied for nickel plated tubes because of the magnetic properties of the nickel).

8.3-4/11
For Doel 2 tubes, the endoscopic inspection and the samples examination did not reveal any defects, while pitting was observed in some tubes of Doel 3 resulting from installation of the equipment (elevation of the different parts) and probe geometry. These problems are now completely solved.

The further UT inspection of the Doel 3 crack patterns under nickel layer gave satisfying results as no defect propagation could be detected.

Finally these two applications performed during normal shut-down with remote control robot arm demonstrated the potential suitability for large scale industrial campaigns

3.2 Nickel plating as preventive measure against PWSCC

3.2.1 Context

In February 1990 Doel 2 unit was brought to a forced shut-down due to excessive primary to secondary leakage in SG A tubes (70 l/h).

After ECT inspection and pressure test some 140 tubes were found leaking and 85 were plugged.

The plugging rate of SG A increased from 12 to 16.9% while it remained unchanged in SG B (3%).

After start-up of the plant a leak rate of 25 l/h remained.

This situation together with the unbalance in flow induced by uneven plugging of the two SG's and a very small remaining plugging margin into SG A made it necessary to undertake a large scale repair campaign.

3.2.2 Repair technique

Nickel plating of the cracked area as repair method like in Doel 3 was not suitable for the following reasons:

- a large number of tubes to be repaired had cracks longer than 10 mm.
- for some tubes (more or less 180) such a repair was not possible due to the presence across the upper transition of explosively expanded mini sleeves installed in 1982. Additionally circumferential cracks at the end of these mini sleeves in the parent tube, due to the explosive expansion required mechanical resistance to be restored before putting back the tube into service.

Instead, the specific design of the tube to tube sheet joint (tube rolled on 72 mm from the bottom of the tube sheet) allowed the application of an original repair method consisting in a new expansion above the original cracked roll expansion followed by a preventive nickel plating on the total new expansion length.
The repair method aimed at restoring the leaktightness and the mechanical resistance of the tube with the expansion and at preventing initiation of PWSCC in the new expansion transitions with nickel plating.

The expansion consisted in an explosive expansion over 50 mm followed by a roll expansion over 28 mm centered in the first expansion (fig. 2).

Nickel plating was performed on a length of 110 mm covering the four new transitions. A thickness of 50 microns was enough for a preventive application.

3.2.3 Qualification of the repair method

Since nickel plating used the process qualified for the Doel 3 application in 1988, only the expansion phase had to be qualified. This was performed by laboratory tests including:

- secondary side and primary side corrosion test on samples expanded with the final process parameters.
- leak tests with helium and water on samples expanded with and without the presence of sludges.
- pull-out tests for checking the mechanical resistance of the expansion.
- profilometry by ECT of the expansion

Additionally a demonstration of the expansion process on 67 tubes was performed during a planned shut-down of the plant in July 90. Tubes selection was the following:

- in service tubes requiring plugging because of leakage (19).
- tubes plugged because of cracks (37)
- tubes with mini sleeve plugged because of circumferential cracks (11).

This demonstration allowed to improve tooling and adjust some process parameters.

For the nickel plating the process parameters were adjusted to the planned thickness using test samples and a modified probe was designed for the tubes with a restricted diameter due to the presence of mini sleeve. This probe was later used for all the tubes.

3.2.4 Choice of the tubes to be repaired

The aim was to recover as many plugged tubes as possible during the October 90 outage. The list of tubes to be expanded and plated was fixed after the water leaktest and ECT inspection and included:

a) in service tubes which required plugging (89 tubes):
   - tubes found leaking during leak test.
tubes with cracks > 13 mm (large leakage risk in case of secondary side
depressurisation).

b) plugged tubes:
- tubes with mini sleeves plugged with removable plugs (119).
- tubes plugged with removable plugs because of cracks (73).

Additionally, the 67 tubes expanded in July had to be plated.

The total number of tubes to be repaired amounted to 348 of which 240 were
plugged tubes.

3.2.5 Industrial application

3.2.5.1 Expansion
The expansion took place from October 01 to October 21 including equipment
installation and withdrawal.

The main phases of this process were the following:

- unplugging using a TIG shrinking tool and a hydraulic pulling device.
- brushing of the area to be expanded with a stainless steel brush.
- rolling of the tube end to make easier the explosive cartridge insertion.
- enlarging the mini sleeve (if any) with a rolling device, for the same
  reason.
- expansion with explosive cartridge.
- one step rolling.
- profilometry of the expanded tubes.

Except for the tool change (in the channel head), all the steps were performed
remotely with the same robot manipulator.

All the 348 tubes were expanded with no major problem.

The total dosimetry was to 15 mSV (1.5 Rem) for the Contractor's personnel
and 150 mSV (15 Rem) for the jumpers.

3.2.5.2 Nickel plating
The nickel plating phase took place as follows:

- installation of equipment: October 22
- brushing of the 348 tubes: October 23 and 24
- electroplating of 347 tubes: from October 25 to November 12
• equipment withdrawal: November 13

The total work period on SG floor was 23 days. The average rate of nickel plating, including brushing, was 16.5 tubes a day. The nickel plating phase only reached an average rate of 18 tubes a day and a maximum rate of 27 tubes a day.

Total dosimetry for the nickel plating phase was 90 mSV (9 Rem).

3.2.6 Inspections
An endoscopic examination was performed before leak test on more or less 10% of the plated tubes with satisfactory results.

The fluoresceine leak test did not reveal any leakage among the repaired tubes.

3.2.7 Conclusion of the 1990 application
The plant restarted with a leak rate of 4 l/h (leak rate just before plant shut-down had increased to more than 50 l/h).

A total of 240 previously plugged tubes were put back into service thus reducing the plugging status down to 9.7% and restoring a more comfortable plugging margin.

4. CONCLUSIONS

The different applications of the nickel plating technique in the Doel power plant demonstrated the potential ability of this process to be implemented as a corrective method for fighting against existing PWSCC in the roll transitions as well as a preventive method combined with others techniques.

The 1990 application moreover demonstrated that high production rate can be achieved, thus allowing successful use of large scale nickel plating during normal plant shut-down.
DOEL 2 SGA 1990
CROSS SECTION OF THE REPAIR

FIG. 2
REFERENCES

PROGRAM FOR STEAM GENERATOR TUBING RELIABILITY TESTS IN JAPAN

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ABSTRACT

Secondary side of steam generator tubing of PWR plant has been suffering from various types of degradations which are wastage, denting, stress corrosion cracking, intergranular attack (IGA) and so on. Almost of all degradations have already been overcome by means of improvement in design, material, water chemistry and operational procedures.

In Japan IGA occurred in some steam generators since 1981 and countermeasures dealing with structure, material, environment etc. have been employed to cope with this problem. As a result of these countermeasures, IGA has been reducing in its initiation and propagation.

But, further investigation to clarify mechanism of corrosion and to verify those countermeasures is considered to be necessary in order to improve effectiveness of countermeasures.

This paper describes a five year program to assess reliability of steam generator tubing which started in 1989.
1. INTRODUCTION

Presently, Japan has 17 PWR nuclear power plants in operation. Secondary side intergranular attack (IGA) has occurred since 1981 in steam generators of certain plants which use mill annealed (MA) 600 Alloy tubing. Design of steam generator and the place where IGA occurred are shown in Figure 1.

From results of laboratory studies and field surveys including inspection of water chemistry records, causes of IGA were concluded as follows.

1. Free caustic derived from welding slug residue at the time of construction as well as impurities such as sodium coming from make up water were so concentrated at crevices of tube support plate in hot leg side that IGA resulted.
2. Sodium phosphate used in the past as a chemical for auxiliary boiler water treatment was carried over into steam generator and free caustic which was formed from residual phosphate caused IGA.
3. Oxidizing species such as cupric oxide (CuO) and dissolved oxygen accelerated initiation and propagation of IGA.

Therefore, countermeasures for environmental improvements such as impurity ingress control and boric acid injection, design and material improvement such as application of broached egg crate tube support and alloy 690 and so on, have been employed to reduce extent of IGA. Also internal sleeve repairs are performed on steam generator tubing affected from IGA. However, sleeve repair implementation needs the extension of outage and might be a rather big factor that lowers plant availability.

In order to verify the effectiveness of present countermeasures and find out further possibility of improved countermeasure, Japanese government had decided to carry out a five year program of steam generator tubing reliability tests which started in 1989. Japan Power Engineering and Inspection Co. (JAPIC) had been designated to perform this reliability tests through Ministry of International Trade and Industry (MITI).

This paper describes objects, methodology and present status of this program.
2. STEAM GENERATOR TURBING DEGRADATION

Steam generators have experienced tube degradation since nuclear power stations were put into operation in the early 1970s. In early operation, main damage were tube thinning at secondary side crevices and tube denting as a result of corrosion of tube support plates. "Steam Generator Reliability Test" sponsored by MITI were performed from 1975 to 1980 to resolve these problems. After then new type of tubing degradation had begun to proceed in early 1980s at the crevice region between tube and tube support plate. It is a kind of intergranular corrosion and is called intergranular attack (IGA). Based on wide variety of laboratory studies and field surveys, various countermeasures have been applied. However, effectiveness of them has not been necessarily quantified even though they reduce the extent of IGA. This is the reason why MITI decided to become a sponsor for steam generator tubing reliability tests in order to validate their effectiveness.

3. TEST PROGRAM

The program is mainly divided into three areas.

- Identification of IGA generating mechanisms
- Verification of reliability improving countermeasures
- Verification of reliability on continuous operation of affected tubing

General methodology and flow diagram of this program is shown in Figure 2 and test schedule is summarized in Table I. Objects and procedures for each test are briefly described in the following section.

![Flow Diagram of Reliability Tests](image)

Figure 2 Flow Diagram of Reliability Tests
Table I Schedule for Steam Generator Tubing Reliability Test

<table>
<thead>
<tr>
<th></th>
<th>Fiscal Year</th>
</tr>
</thead>
<tbody>
<tr>
<td>Identification of IGA generating mechanism.</td>
<td>Planning</td>
</tr>
<tr>
<td>Verification of reliability improving countermeasures.</td>
<td>Planning</td>
</tr>
<tr>
<td>Burst tests</td>
<td>Planning</td>
</tr>
<tr>
<td>Evaluation</td>
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</tr>
</tbody>
</table>

3.1 Identification of IGA Generating Mechanism

(1) Identification of Corrosion Generating Condition

Susceptibility of alloy 600 against IGA is strongly affected by pH of environment and potential of material, and some works have been already done. Examples of those

Figure 3 Effects of pH and Potential on IGA[1],[2]

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work are shown in Figure 3[1], [2].

In this program, electrochemical techniques such as a constant extension rate technique (CERT) under potentiostatic control, are employed and the behavior of alloy 600 (MA) is studied specially in acidic region. Results are finally summarized into “corrosion generating condition map” combined with existing data in alkaline region. Also, in order to complete a general map, not only IGA but also pitting and denting (corrosion of tube support plate material) are investigated. Table II shows the test conditions. Effects of environmental factors, that is to say, pH, potential, temperature etc., on corrosion such as IGA, pitting and denting are studied by electrochemical techniques under operating temperature.

Informations obtained here will be basis by which secondary side corrosion phenomena of SG material can be avoided.

<table>
<thead>
<tr>
<th>Table II Condition of Electrochemical Test</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Object</strong></td>
</tr>
<tr>
<td></td>
</tr>
<tr>
<td>Verification of Material Improvement</td>
</tr>
<tr>
<td></td>
</tr>
</tbody>
</table>

Note. (1) CERT: Constant extension rate technique
(2) E: potential

(2) Development of Techniques for Crevice Environment Evaluation
Tubing degradations exclusively occur at crevice region between tube and tube support plate because rather low level of impurity in the bulk water concentrates in the crevices due to hydro-thermal effects and results in formation of concentrated solution which can be harmful environment. There is no direct method to know crevice condition which is very important for corrosion protection.

In this item, it is the object to develop a technique for monitoring and deducing crevice environment from bulk water chemistry. Steps of study is as follows.

(i) Assuming that vapour liquid distribution coefficient of each compound in the crevice solution plays important role in the concentration process, distribution coefficients of concerned species are summarized. Chemical species to be concerned are selected based on hide out return studies and distribution coefficients of some species are measured if coefficients data are not available.

(ii) Concentration model in crevice is studied. Conceptional mechanism of crevice concentration is shown in Figure 4 and mass balance of each specie is described as follows.

\[ \rho V \frac{dC_i}{dt} = C_{w\nu\lambda} m_i - \sum_j D_{ij\nu\lambda} C_i \alpha_j - C_i (m_i - m_0) \]

(3-1)
\[ V \quad : \quad \text{Crevice volume} \]
\[ \rho \quad : \quad \text{Density} \]
\[ t \quad : \quad \text{time} \]
\[ C_{i}^{\text{b}} \quad : \quad \text{Bulk concentration \textit{i}-species} \]
\[ C_{i}^{\text{c}} \quad : \quad \text{Crevice concentration \textit{i}-species} \]
\[ m_{1} \quad : \quad \text{Water entering into crevice} \]
\[ m_{o} \quad : \quad \text{Steam departing from crevice} \]
\[ D_{ij} \quad : \quad \text{Distribution coefficient \textit{j}-compound} \]
\[ \alpha_{ij} \quad : \quad \text{Molar ratio of \textit{j}-compound for \textit{i}-species} \]

Since only molecular species such as NaCl can vaporize, calculation of chemical equilibrium under high temperature is always necessary to get the extent of undissociation molecule.

(iii) Technique obtained by combination of concentration and chemical equilibrium calculation code is verified in comparison with hide out return data from operating plants and laboratory model boiler.

3.2 Verification of Reliability Improving Countermeasures

(1) Environmental Countermeasure

Boric acid treatment and increase of hydrazine in feedwater are applied as part of countermeasures against IGA.\textsuperscript{[14]} In this item, effectiveness of boric acid and hydrazine is assessed first and then possibility of alternate chemicals as environmental countermeasures is studied.

(i) Physical and chemical properties, and informations on the field application are collected and evaluated first. Then alternate chemicals are surveyed under some criteria and candidates are selected (amines, phosphates, borates etc.). High temperature properties measurements such as pH, solubility, distribution coefficient and corrosiveness, are performed if necessary.

(ii) Buffering capacity of selected candidates against free caustic and acid forming substance (e.g. sea water) is checked by measuring pH of simulated solutions.

(iii) Potentiality of selected candidates is finally confirmed by model boiler tests.

(2) Improved Design (Laboratory Test)

Improved configuration has been applied in the tube support plate (TSP) design of recent steam generator to reduce the extent of concentration in its crevice region. This new design is called broached egg crate (BEC) type and shown in Figure 5.

\[ \Sigma_{j} D_{ij} m_{o} c_{i} \quad \alpha_{ij} \]

Figure 4 Mechanism of Crevice Concentration

\[ C_{b1} \quad : \quad \text{Bulk concentration \textit{i}-species} \]
\[ C_{i} \quad : \quad \text{Crevice concentration \textit{i}-species} \]
\[ m_{1} \quad : \quad \text{Water entering into crevice} \]
\[ m_{o} \quad : \quad \text{Steam departing from crevice} \]
\[ D_{ij} \quad : \quad \text{Distribution coefficient \textit{j}-compound} \]
\[ \alpha_{ij} \quad : \quad \text{Molar ratio of \textit{j}-compound for \textit{i}-species} \]

Figure 5 BEC Type Tube Support Plate

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Effectiveness of this improved design is verified by laboratory test in which concentration factor is measured. Test TSP with model crevices is settled in the model boiler and test is performed under operating condition. A tracer chemical is used to estimate concentration in the crevices.

(3) Improved Material
In recent design material for tubing and tube support plate has been improved to respectively alloy 600 (TT) and alloy 690 (TT), and stainless steel. Corrosion susceptibility of improved material is examined through the same electrochemical test as in 3.1.(1) and test conditions are summarized Table II also.

In this item, improvement in alloy 690 (TT) is investigated not only by electorochemical technique but also by new metallurgical approach in which improvement is verified by means of evaluation of relationship among hydrogen behavior, carbide precipitation along grain boundary and susceptibility. Hydrogen behavior is measured by autoradiography.

(4) Combination Effect of Improved Material and Design
Model boiler test to verify the combination effect of improved material and design is being carried. Figure 6 shows examples of simulated tube support models which are equipped in the model boiler and testing condition is summarized in Table III. Test sections are inspected periodically by eddy current testing and other new technique to confirm whether any deficiencies initiate and propagate or not.

![Figure 6 Simulated Tube Support Models](image_url)
(5) Reliability Test on Repaired Portions

IGA suffered portions are basically repaired by sleeves which are blazed and recently laser-welded. Reliability of repaired portion under operating condition is being monitored by model boiler test in which sleeved TSP models are exposed to primary and secondary operating environment. Examples of simulated sleeved model are shown in Figure 7 and test condition is summarized in Table IV.

![Diagram](image)

**(a) Brazed**

**(b) Laser-welded**

Figure 7  Simulated Sleeved Models
Table IV  Model Boiler Test Condition for Repaired Portion Reliability

<table>
<thead>
<tr>
<th>Item</th>
<th>Target</th>
</tr>
</thead>
<tbody>
<tr>
<td>Temperature (°C)</td>
<td>320 ± 5</td>
</tr>
<tr>
<td>Pressure (kg/cm²)</td>
<td>157 ± 3</td>
</tr>
<tr>
<td>Temperature (°C)</td>
<td>272 ± 5</td>
</tr>
<tr>
<td>Pressure (kg/cm²)</td>
<td>57 ± 3</td>
</tr>
<tr>
<td>pH (at 25°C)</td>
<td>9.0 ± 0.2</td>
</tr>
<tr>
<td>NH₃ (ppm)</td>
<td>0.3 ± 0.1</td>
</tr>
<tr>
<td>DO (ppb)</td>
<td>&lt; 5</td>
</tr>
<tr>
<td>Cl (ppm)</td>
<td>&lt; 0.1</td>
</tr>
</tbody>
</table>

3.3 Verification of Reliability on Continuous Operation of Affected Tube

In order to verify the safe operation of steam generator tubing which has small undetectable defects, burst pressure test and leak rate test are carried out on tubes with simulated IGA. Relationships among defect shape, burst pressure and leak are investigated and analysed by fractomechanics. An example of test specimens is shown Figure 8.

![Diagram of test specimens]

Figure 8 Test Specimen for Burst Pressure and Leak Rate Measurement

4. CONCLUSION

A five year program on "Steam Generator Tubing Reliability Tests" sponsored by Japanese government had started in 1989. The objects of this program are to evaluate the effectiveness of countermeasures against corrosion, specially IGA, which have been employed in the field and to improve the reliability of steam generator tubing. The program proceeds according to schedule, that is to say planning and preparation of tests have almost finished, and results obtained in this program will gradually open in near future.
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ON OPERATING EXPERIENCE WITH STEAM GENERATORS

STEAM GENERATOR TUBE LASER SLEEVING

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FRAMATOME
France
I. PRESENTATION

II. SLEEVED UNIT FEATURES

III. THE LASER WELDING PROCESS

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IV. AVANTAGES DU PROCÉDE LASER PAR RAPPORT AU PROCÉDE TIG

V. GAMME OPERATOIRE

VI. MOYENS INDUSTRIELS
I. PRESENTATION

For many years, FRAMATOME has been used with different techniques and means to perform the steam generator tube sleeving operation, such as the "mechanical" either GTAW welded or kinetic welding process. As soon as first power laser units appeared on the market we felt right interested in applying this process to the sleeving operation.

I. INTRODUCTION

FRAMATOME dispose depuis plusieurs années de différentes techniques et moyens de manchonnage des tubes de générateurs de vapeur ("mécanique", soudé TIG, explosif). Dès l'apparition sur le marché des premiers lasers de puissance, nous avons ressenti l'intérêt de ce procédé pour une application au manchonnage.
After comparison between all the processes and equipments existing at that time (that is to say CO₂ and YAG laser units), we chose the YAG and bought a 1.2 KW NEC laser unit in 1988.

As it was installed in our Welding Center of Le Creusot, this equipment enabled us carrying out a preliminary test programm which targets were:

- getting the mastery of the equipment and associated technologies,
- implementing this process for the sleeve welding operation by improvement of the welding-pen.

Après comparaison des procédés et matériels disponibles (lasers CO₂ et YAG), nous avons opté dès 1988 pour le YAG en acquérant un laser NEC de 1.2 KW.

Cet équipement installé dans notre Centre de Soudage du Creusot nous a permis de mener un programme d’essais préliminaires. Les objectifs de celui-ci étaient doubles :

- il s’agissait, d’une part, de maîtriser l’équipement et les technologies connexes,
- d’autre part, d’aborder l’application au soudage de manchettes, avec notamment la mise au point du stylo de soudage.
The NEC laser unit became afterwards transferred to our workshop in Chalon-sur-Saône (June 1990), where we achieved the final tests of the process at the same time we were investigating the development of industrial operation means.

The actual program is mainly focused on 7/8" tube steam generator repair process at tubesheet outlet. Yet made sure that our methods and means apply as well to 3/4" tubes up to second tube support-plate level.

Cette première phase étant achevée en juin 1990, le laser NEC a été transféré à Chalon-sur-Saône où nous avons effectué la mise au point finale du procédé, parallèlement au développement des moyens industriels d'intervention.

Le programme actuel est principalement centré sur la réparation de GV à tubes 7/8" en sortie de plaque tubulaire. Nous nous sommes cependant assurés de la convertibilité des procédés et moyens pour des interventions sur tubes 3/4" et ce, jusqu'au niveau de la deuxième plaque entretoise.
II. SLEEVED UNIT DESIGN

Sleeves are made of heat-treated Inconel 690. The sleeved unit has been designed to provide the same breaking strength and leaktightness as the tube. The upper part of sleeve consists of an anti-popout length which ensures some locking-up in case the tube breaks in upper transition expansion area.

II. CONCEPTION DE L’ASSEMBLAGE MANCHONNE

Les manchettes sont en Inconel 690 traité thermiquement. La conception de l’assemblage manchonné a été définie pour assurer les fonctions de résistance mécanique et d’étanchéité en lieu et place du tube. Dans leur partie supérieure, les manchettes comportent une zone anti-débattement qui garantit une relative immobilisation dans l’hypothèse d’une rupture du tube dans la zone de transition supérieure de l’expansion.
III. THE LASER WELDING PROCESS

a) **Process preliminary tests**:

These tests dealt with the various parameters which may exert an influence on geometry and quality of the weld bead, as:

- laser beam power (for continuous and pulsed modes),
- welding speed,
- focal spot size and location from the surface to be welded,
- protective gas.

III. PROCEDE DE SOUDAGE PAR LASER

a) **Essais préliminaires de procédé**:

Ces essais ont porté sur les différents paramètres qui influent sur la géométrie et la qualité du cordon de soudage, soit:

- la puissance du rayonnement (mode continu et pulsé),
- la vitesse de soudage,
- la dimension et la position de la tache focale par rapport à la surface de la pièce à souder,
- le gaz de protection.
Weld beam geometry obtained by laser process looks as follows:

The compliance with $l$ and $p$ dimensions ensures the weld quality. $p/l$ ratio is a welding reliability factor which exerts an influence on both volume and welding power.

La géométrie des cordons obtenus par laser se caractérise de la façon suivante:

Les cotes $l$ et $p$ conditionnent la qualité de la soudure. Le rapport $p/l$ est un facteur d'efficacité du soudage qui joue sur le volume du bain et donc sur la puissance de soudage.
b) **Process chosen by FRAMATOME:**

After performance of preliminary tests on many thousands of weld beads we decided to use the process according to following criteria:

- **Weld quality:**
  
  * metallurgical aspect,
  * geometry (penetration, bulge, width),
  * volume (non porosity),
  * reproductiveness;

- **Weld inspectability through UT examination;**
- **Easy and reliable process;**
- **Long lifetime of optical equipment (mirrors, lenses, fibers).**

---

b) **Procédé retenu par FRAMATOME:**

Suite aux essais préliminaires qui ont porté sur plusieurs milliers de cordons de soudure, nous avons retenu le procédé sur les principaux critères suivants:

- **Qualité de la soudure:**
  
  * métallurgique,
  * géométrique (pénétration, largeur, bombé),
  * volumique (absence de porosités),
  * répétitivité de cette qualité;

- **Contrôlabilité de la soudure par ultra-sons;**
- **Simplicité du procédé;**
- **Tenue des équipements optiques (miroirs, lentilles, fibres).**
Considering these criteria we have chosen:

- working in pseudo-pulsed mode with 50 hz frequency,
- using nitrogen as protective gas,
- a welding power of about 600 W,
- 30 cm/mn advance speed.

Au travers de ces critères, notre choix s'est porté sur:

- un travail en mode pseudo-pulsé à une fréquence de 50 hz,
- l'azote comme gaz de protection,
- une puissance voisine de 600 W,
- une vitesse d'avance de 30 cm/mn.
IV. ADVANTAGES OF LASER PROCESS COMPARED TO GTAW PROCESS

The laser welding process brings two main additional advantages if we compare it to the GTAW one:

- Reproductibility of the penetration and resisting cross-section and this, independantly of surrounding conditions (fitup strength, tube/plate hard-rolling tightening,...) ;

- Flexibility and performances which enable a remote-controlled work with great output.

IV. AVANTAGES DU PROCEDE LASER PAR RAPPORT AU SOUDAGE TIG

Le soudage par laser apporte deux avantages majeurs par rapport au soudage TIG :

- Répétitivité de la pénétration et de la section résistante, indépendamment des conditions d'environnement (serrage de l'accostage, serrage du dudgeonnage tube/plaque,...) ;

- Souplesse et performances qui permettent le travail à distance et à forte cadence.
V. PROCESS IMPLEMENTATION SCHEDULE

It consists of five main operations to which two optional ones can be added. These main operations are:

a) **Tube cleaning**:

This operation aims at making sure of tube cleanliness before welding because boric acid wastes may induce some weld degradation (porosity). It has to be performed using a stainless steel wire brush (tube dry scrubbing).

V. GAMME OPERATOIRE

La gamme opératoire comporte cinq opérations principales auxquelles s'ajoutent deux opérations optionnelles.

**Opérations principales**:

a) **Nettoyage du tube**:

Cette opération est destinée à assurer la propreté du tube avant soudage : des résidus d'acide borique peuvent provoquer des dégradations de la soudure (soufflures). Elle est effectuée par brossage à sec avec une Brosse en acier inoxydable.
b) **Sleeve insertion/expansion**:

As fastened on expansion mandrel the sleeve has to be inserted into the tube. The mandrel is provided with two expansible seals which simultaneously make possible both upper and lower expansions. The hydraulic pressure is monitored by a computer in order to get a minimal deformation of the tube.

b) **Introduction/expansion de la manchette**:

La manchette est introduite emmanchée sur le mandrin d’expansion. Celui-ci comporte deux membranes expansibles qui assurent simultanément les expansions hautes et basses. La pression hydraulique est pilotée par un ordinateur qui assure une déformation minimale du tube.
c) **Upper and lower joint welding**:

Each of both joints is made of two circular and shifted weld beads. The joints have to be achieved successively. Welding parameters are continually inspected and recorded so as to guarantee the weld quality.

c) **Soudage des joints supérieur et inférieur**:

Les deux joints constitués chacun de deux cordons circulaires décalés sont réalisés successivement. Les paramètres de soudage sont contrôlés et enregistrés de manière continue pour garantir la qualité de la soudure.
d) **Weld televisual inspection**:  

Every joint is inspected by use of a special camera. Have to be examined:

- weld surface condition,
- right achievement of both weld beads,
- lack of protruding defects,
- weld dimensional features.

---

d) **Contrôle télévisuel des soudures**:  

Tous les joints sont contrôlés grâce à une caméra spécialement développée. On vérifie:

- l'état de surface de la soudure,
- la bonne exécution des deux cordons,
- l'absence de défauts débouchants,
- les caractéristiques dimensionnelles de la soudure.
e) **Upper assembly stress-relieving process**:

Upper assemblies undergo a stress-relieving operation by means of an electric heater, in order to reduce residual strains in weld area as well as in upper transition zone of the hydraulic expanded part. Corrosion loop tests made possible checking the reliability of this heat treatment.

e) **Traitement de détensionnement des assemblages supérieurs**:

Les assemblages supérieurs sont détensionnés grâce à une canne électrique afin de réduire les contraintes résiduelles, aussi bien dans la zone de la soudure que dans la zone de transition supérieure de l'expansion hydraulique. Des tests de corrosion en boucle ont permis de vérifier l'efficacité de ce traitement thermique.
Optional operations:

a) Tube inlet rework:

Tube fitting conditions on tubesheet may induce a necessary rework of tube inlets. This operation has to be carried out using the hard-rolling process which ensures to get the minimum diameter required for a right sleeve insertion.

Operations optionnelles:

a) Reconformage de l'entrée des tubes:

Suivant les conditions d'assemblage des tubes sur la plaque tubulaire, il peut être nécessaire de reconformer les entrées de tube. Nous effectuons cette opération par dudgeonnage et garantissons un diamètre minimal nécessaire à l'introduction sans aléa des manchettes.
b) **Weld UT-inspection**:

Our Non Destructive Examination Development Center designed an ultrasonic examination method for checking laser weld resisting cross-sections.

If we consider laser process reproductiveness, this inspection does not seem to be obligatory. As a consequence, it is considered as optional but useful in case the weld quality file has to be documented by further detailed informations.

b) **Contrôle ultrasonique des soudures**:

Notre Centre de Développement en Contrôle Non Destructifs a mis au point une méthode de contrôle ultrasonique, qui permet notamment de vérifier la largeur de la section résistante des soudures laser.

Compte-tenu de la répétitivité du procédé laser, ce contrôle n'est pas obligatoire. C'est pourquoi nous le présentons comme opération optionnelle permettant de renforcer si nécessaire le dossier de qualité des soudures.
VI. INDUSTRIAL MEANS

The following three main concerns determined the design and installation of industrial laser sleeving means:

- the reliability,
- the most reduced doserate,
- the output.

VI. MOYENS INDUSTRIELS

La conception et la mise en place des moyens industriels de manchonnage laser ont été orientés par trois soucis principaux:

- la fiabilité,
- la dosimétrie la plus faible,
- la productivité.
The reliability

Every equipment used has been chosen for its reliability criteria. The most sensitive among them, especially the welding equipment, have undergone acute reliability analyses which enabled improving the design as well as choosing the right components to get a better accuracy and lifetime of equipment.

La fiabilité

Tous les équipements mis en œuvre ont été choisis en fonction de critères de fiabilité. Les plus sensibles d'entre eux, notamment l'équipement de soudage, ont fait l'objet d'analyses fiabilistes complètes. Ces analyses ont permis de corriger la conception ainsi que de guider le choix des composants pour aller à la fois dans le sens d'une plus grande fiabilité et d'une meilleure maintenabilité.
The lowest doserate

Equipment development aimed first at reducing the doserate. In channel-head no operation is necessary, robotic arm and its properties fitting-up excepted.

La dosimétrie la plus faible

La réduction de la dosimétrie a été un objectif prioritaire pour le développement des équipements. Aucune intervention en boîte à eau n'est nécessaire en dehors de celles de mise en place du bras porteur et de ses accessoires.
Operations in steam generator bunker have been reduced to the minimum. For this, we have developed some multi-purpose equipments which enable performing all operations of the process using a few tooling permutations and refittings.

As the control unit is located outside reactor building and remote-operated, a few people have to work during short times in high-irradiated areas, so we can say that we gain a noticeable reduction of integrated doserates.

Les interventions en casemate GV ont été réduites au minimum. Pour cela, nous avons développé des équipements polyvalents qui permettent d'effectuer toutes les opérations du procédé avec un nombre réduit de permutations d'outillages et de reconnexions.

Le poste de pilotage est extérieur au bâtiment réacteur. Les durées et le nombre des présences en zones actives sont donc réduits, avec pour conséquence une diminution correspondante des doses associées.
The output

Our purpose was to get a channel-head occupancy duration as short as possible. Yet we were not taking any risks linked to the quality of treatment regarding the suppression of any operation which might be essential for the process (tube cleaning before welding, upper assembly stress-relief heat treatment), but we have been investigating each of the operations and associated tool means so as to reduce technological times.

At the same time the equipment polyvalence and connection simplification made possible reducing as well conversion times between two operations. These efforts shall allow us setting more than twenty sleeves a day.

La productivité

Nous avions pour objectif de limiter les durées d'occupation de la boîte à eau. Nous avons refusé de risquer de diminuer la qualité du traitement par la suppression d'opérations nécessaires au procédé (nettoyage des tubes avant soudage, traitement thermique de détensionnement des assemblages supérieurs). Par contre, nous avons travaillé chacune des opérations et les moyens correspondants pour limiter les temps technologiques.

La polyvalence des équipements et la simplification des connexions nous a permis, en parallèle, de réduire les temps de conversion d'une opération à l'autre. Ces efforts nous permettront de poser plus de vingt manchettes dans une journée.

8.5-23/25
Equipment location

- **Outside reactor building**:

  You have three containers for:

  - The control-unit consisting of drive and supervision computers as well as video means;
  - The laser unit itself (connected to the welding equipment in steam generator bunker by an optical fiber spreading the power) and its control cabinet;
  - Auxiliary equipments, as:
    * laser coolant system,
    * main electric power supply,
    * dry air-compressor,
    * gas-distributing station,
    * cable storage coils.

Implantation des matériels

- **A l'extérieur du bâtiment réacteur**:

  Trois containers sont placés à l'extérieur du BR ; ils renferment:

  - Le poste de pilotage qui comprend les ordinateurs de commande et de supervision, les moyens vidéo,
  - Le laser et ses armoires électriques (il est relié à l'équipement soudage, placé dans la casemate, par une fibre optique dans laquelle est transmise la puissance);
  - Les équipement auxiliaires comprenant notamment:
    * le dispositif réfrigérant du laser,
    * l'alimentation électrique de l'ensemble,
    * le compresseur d'air sec,
    * une station de distribution de gaz,
    * les tourets de stockage des liaisons.
- Inside reactor building:

You will find the equipments specific to each process (welding-device, brushing-unit, expansion hydraulic generator,...) as well as the electric cabinets connected to control unit. Data flow from outside to inside reactor and vice-versa through an optical fiber.

Investigations about these industrial means are not achieved up to now but will be qualified (with EDF collaboration) in the CETIC afterwards and shall be used on a French power plant at the beginning of year 1992.

- Dans le bâtiment réacteur:

Les équipements spécifiques à chacun des procédés (machine de soudage, machine de brossage, groupe hydraulique d'expansion,...) ainsi que les armoires électriques nécessaires au dialogue avec le poste de pilotage. Le transfert d'informations entre l'extérieur et l'intérieur du BR est réalisé par une fibre optique.

Ces moyens industriels sont en cours de mise au point. Ils feront l'objet d'une qualification au CETIC (en liaison avec EDF) et permettront une intervention sur un site français début 92.
DEVELOPMENT OF A STEAM GENERATOR SLEEving SYSTEM
USING FIBER OPTIC TRANSMISSION OF LASER LIGHT

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Westinghouse has designed and qualified a new laser welding system for sleeve installation in steam generator tubing. The processes used permit welded sleeves at elevations up to 8 meters from the tubesheet face. Over 230 meters of fiber optics can be used to transmit light energy from a pulsed Nd:YAG laser positioned outside of containment to the weldhead in the tube. By selecting a fiber optic transmission system, the amount of equipment inside containment is minimized and associated radiation exposure for equipment setup and maintenance will be reduced.

Westinghouse a élaboré et qualifié un nouveau système de soudage au laser pour l'installation des manchons dans les tubes des générateurs de vapeur. Les procédés mis en œuvre permettent d'installer des manchons en altitude jusqu'à 8 mètres à partir de la face primaire de la plaque tubulaire. L'énergie lumineuse fournie par un laser industriel de type pulsé Nd:YAG situé à l'extérieur de l'enceinte de confinement du réacteur, est transmise à la tête de soudage dans le tube/manchon par l'intermédiaire de fibres optiques dont la longueur peut atteindre 230 mètres. Par la sélection judicieuse d'un système adéquat de transmission optique, le volume d'équipement utilisé à l'intérieur des bâtiments réacteur est réduit au minimum et par conséquent l'exposition du personnel aux radiations pendant les opérations d'installation et de maintenance est diminuée.
Introduction

Westinghouse has qualified a process for welding steam generator sleeves at support plate locations in 22 mm diameter steam generator tubing, with qualification for 19 mm diameter tubing being completed. This system uses fiber optics to transmit light energy from a pulsed Nd:YAG laser positioned outside of containment. By using a fiber optic transmission system, the amount of equipment inside containment is minimized and associated radiation exposure for equipment setup and exposure will be reduced.

In addition to fiber optic delivery of weld energy, a new delivery system and effector was developed. This system, which is controlled from a trailer outside containment, uses a common multi-function end effector. This end effector is used for the installation of sleeves and for the welding, heat treatment, and ultrasonic inspection of the installed sleeve. Use of this common end effector, which is designed to mate with both the Vermaat Flexivera or Westinghouse ROSA type delivery systems, minimizes the need to change tools. This will reduce the subsequent radiation exposure associated with end effector changes.

Background

Periodic inspection of steam generator tubes is performed using non-destructive examination techniques to evaluate tube degradation. Should the degradation exceed established criteria, the tube is removed from service by plugging or otherwise modified to bring it back into compliance. One such modification technique is to secure a sleeve into the tube. The sleeve, which is a tube of slightly smaller diameter than the parent tube, is secured to the tube at each of its ends, effectively forming a bridge over the degraded zone.

Sleeving was initially applied for secondary side tube corrosion associated with sludge deposition on the top of the tubesheet, both in the sludge pile and in the tube-to-tubesheet crevice. Between 1980 and 1991 Westinghouse has installed over 23,600 tubesheet sleeves, the majority using mechanical joining techniques. Of this count, over 400 tubes were recently returned to service after plugging. However the need for sleeving for secondary side degradation has been somewhat mitigated by enhanced chemistry control of the secondary side water.

More recently degradation characterized as Primary Water Stress Corrosion Cracking (PWSCC) has been exhibited at the roll transition and both primary and secondary side degradation has been exhibited at some support plate locations. In response to the identification of PWSCC, Westinghouse began the development of a laser welded sleeve in 1984. This system was based around a developmental fast axial flow CO₂ laser and a beam toss (mirror) system. This effort culminated with the installation of 55 laser welds in a Belgian Power Plant. However reliability issues with the laser (as well as its removal from the market by its manufacturer) prompted Westinghouse to research the state of laser technology again in 1989.
At the end of 1989 we determined that fiber optic transmission of Nd:YAG laser energy was feasible on the scale we required (in 1984 fiber optic transmission of Nd:YAG energy was only in its infancy). We also determined that a number of production quality lasers were available in the worldwide market. Thus our Nd:YAG laser system development effort was begun.

**Configuration Philosophy**

The laser welded sleeving system consists of a number of processes which are performed sequentially in the field. In general, the following sequence of operations is used:

1. The tube is inspected to locate degradation and evaluated for applicability of the sleeving process.
2. The tube is honed to remove frangible oxides.
3. The sleeve is installed onto an expander, moved into position in the tube, expanded to hold it in place, and the expander removed.
4. The weld head is positioned and the upper weld of the uppermost sleeve performed.
5. Next the weld head is moved down, making the lower sleeve weld. If multiple sleeves are installed, the weld head is moved down to the next sleeve and step 4 repeated. For tubesheet sleeves the lower weld can either be structural or a seal weld (when used in conjunction with a lower hard mechanical roll).
6. Typically, the welds are stress relieved using a radiant heater to enhance PWSCC resistance. Welds are stress relieved in a similar sequence to welding.
7. An ultrasonic examination (UT) of the welds is then made, following the same general sequence as welding. The UT data is transformed into a C-scan, and the cross section of the weld evaluated.
8. A baseline eddy current test (ECT) is performed for use in subsequent inspection of the sleeve.

All of these processes are performed using remotely operated tooling based on a steam generator service robot, either the Vermaat Flexivera or Westinghouse ROSA. Tool control is from outside containment, with steam generator platform workers feeding consumables to the tools in the channelhead when required.

The basis for the system design is "probe" type tooling; that is the processes are performed by small, short, tools or "probes" which are secured to the end of polyethylene or nylon conduits. These probes are driven into position using either or both a Platform Drive Assembly (PDA), situated on the steam generator platform, or the Select And Locate
End Effector (SALEE), located on the delivery robot [Figure 1]. Both the PDA and SALEE feature driven rollers coupled with an encoder to allow for course but rapid positioning of the probes. Fine positioning is obtained by eddy current coils which are built into the probes (which detect the support plates or sleeve ends). In addition the SALEE has external gripping bladders which are used for short fixed distance translations. Their use will be discussed later.

Tube cleaning is performed by honing. A hone is secured to a flexshaft which is driven from the steam generator platform. From the platform the conduit drops into a thin storage box and emerges to pass into a PDA. From there it is guided by a conduit to an end effector on the end of the delivery robot [Figure 2]. To provide rinsing of the tube, water is injected at the flexshaft and emerges from the top of the hone. This water, along with the oxides removed by honing, is captured at the tube mouth by the end effector and removed from the channelhead through the flexshaft guidance conduit. A wiper assembly diverts the waste water to a filter bank; filtered waste is then pumped to the plant radioactive sump.

Since tube cleaning generates the most radiological waste, dedicated equipment is used for honing. Typically all potential slewing candidates are honed altogether, then the system is removed and the platform cleaned prior to performing all other slewing processes. All other slewing processes use the same delivery system; a conduit storage drum below the steam generator platform, a PDA on the platform, and a SALEE on the delivery robot [Figure 3]. This simple approach minimizes the amount of equipment on the platform and allows for quick changeout of processes.

Sleeves are installed by putting them on an expansion mandrel which is then positioned using SALEE. Final positioning is determined using the ECT coil on the expansion mandrel; once the coil is nulled on the support plate, SALEE’s grippers pull the conduit back to center the sleeve on the support plate. Once in position, the mandrel bladders are pressurized with water, expanding the sleeve into contact with the tube. By using a computer to monitor the time-pressure relationship, expansion of the sleeve is stopped just following contact with the tube. This minimizes tube deformation yet allows the process to accommodate the full range of tube yield strengths.

Once sleeves are installed, the laser weld head conduit is installed in the cable management system. The weld head is driven by SALEE and the PDA up to the top welding position. An eddy current coil on the weld head senses the bottom of the sleeve to set the head in the upper weld position. Once in place the weld head seals the tube with an inflatable bladder, cover gas is injected, and the weld performed. Upon the completion of the weld, the head is translated downward a short fixed distance by SALEE, and the process repeated for the lower weld.

The energy source for welding is a 1000 Watt, Lumonics JK706 pulsed Nd:YAG laser. This laser is shipped in a trailer [Figures 4 and 5], which is used as an equipment room for the laser outside containment. Energy
from the laser is transmitted through a Lumonics designed, four segment, fiber optic delivery system. Each fiber segment is 50 meters long [Figure 6], although longer lengths could be applied. The segmentation was designed to allow rapid containment isolation if a fuel assembly were to be damaged during refueling.

Each "joint" in the system is made in a protected box. After the third segment an optical switch is included in the system [Figure 7]. This switch allows one laser and trunk fiber to serve multiple weld heads on a time share basis. This feature further minimizes the impact on the power plant by minimizing the amount of containment penetrations required. Following the optical switch, each platform has a junction box (typically sited below the platform next to the conduit storage drum). At this box cover gas is injected into the weld head conduit and gas flow is monitored. Also at this point laser power is continuously monitored and weld head rotation speed is controlled.

In addition, the laser system utilizes two programmable logic controllers to continuously monitor the control and integrity of the fiber optic network. In order to allow Class I laser operation, continuous safety barriers and monitors for the laser radiation must exist. This was accomplished by interlocking all connections, placing the fiber optic in an armored conduit, and adding special safety circuits to the fiber optic itself. At the termination of the system, the conduit delivery system and steam generator itself serve as the safety barrier for the weld head and its conduit. Integrity of the weld head fiber is monitored to within centimeters of the weld head itself.

Heat treatment, if desired, is performed using a radiant heat source. Prior to heat treatment, a series of tests are performed throughout the region of the steam generator to be sleeved to determine the thermal emissivity of the tubes. Once complete, a calculation is performed to determine the appropriate power level for stress relief to prevent out of process range temperature conditions.

The radiant heat treat probe is installed in the PDA and SALEE, and a sequential operation similar to welding is performed. Currently only one weld per tube is stress relieved at a given time to prevent excessive loading on the tube should support plate denting (fixity of the tubes) be present. A power controller controls and records the heating cycle.

Several options for inspection exist, and the choice depends on national and utility requirements. The most rigorous technique is volumetric examination by ultrasonics. With this process the tube to sleeve weld interface is examined. A second technique is visual examination of the weld surface, coupled with the periodic creation of workmanship samples. A third inspection method, though used by Westinghouse only for future in-service examinations, is ECT using a crosswound probe. With this technique a signature is created which is used for comparison to future examinations.

To perform the ultrasonic examination, special equipment was designed. Specifically a small, probe type UT head was developed for use with the
conduit delivery system. Contained within the UT head is a highly focussed crystal, a rotational drive with an encoder, a translational drive with an encoder, and an inflatable barrel seal. Ports are also included to inject and extract the UT couplant, deionized water. The head has the ability to perform circumferential and axial scans of the weld, however to date helical scans have been used.

For the UT process a highly focussed transducer is focussed perpendicular to the weld inner surface, with its focus at the weld interface. Gates are set monitor the reflected signal of the weld surface (a measure of the coupling efficiency of the weld surface), of the weld interface (an absence of reflectors indicates a good weld), and the tube backwall (confirms signal passed through the weld and returned) [Figure 8]. The tube-sleeve interface signal is processed and displayed as a C-scan. From this the effective weld width can be verified, as well as inspecting for inclusions. The surface scan and tube backwall reflection are used only to verify proper coupling of the UT signal.

**Design and Qualification**

The weld design chosen was a partial penetration, autogenous weld contained within a hydraulic expansion region. From this goal the following weld objectives were established:

1. Weld width (at faying surface) to meet ASME Code guidelines
2. Weld free of cracks and inclusions
3. Weld depth less than 85% of the tube wall thickness
4. Weld surface conducive to UT inspection

The weld [Figure 9] has been designed to meet ASME Code guideline calculations for plant operating and postulated accident conditions. A proprietary Westinghouse analysis program, WECAN, was used to calculate the required weld interface width based on ASME Code allowable stress limits. This was further verified by tensile, fatigue, and thermal cycling testing of weld specimens. Additional conservatisms were applied to account for the accuracy in UT inspection (for verifying width). Weld depth (penetration into the parent tube) is only controlled to prevent secondary side contaminants from entering the weld pool.

The weld process development was a collaborative effort between Westinghouse and Luminumics, with Luminumics performing initial development and Westinghouse performing process refinement and final qualification. This division of development allowed process development to proceed in parallel with the fiber and weld hardware design.

A JK701 (400 watt Nd:YAG) laser with 5 meters of fiber optics and a Westinghouse designed weld head was used for initial process development. First flat plate samples and sectioned tubes/sleeves were welded using commercial weld heads to determine basic operational parameters. Next short tube/sleeve sections were welded using the Westinghouse weld head.
Finally the Westinghouse designed, self contained field weld head [Figure 10 and 11] was applied, first at Lumonics' Rugby UK facility, then at Westinghouse's Madison PA facility, and now at the Westinghouse European Service Center in Nivelles, Belgium. Final weld qualification was performed in a full scale mockup, with samples inserted up to and including the next to the last support plate location.

Variables considered during process development were weld mode (continuous molten pool or solidification between pulses), pulse width, pulse energy, pulse frequency, pulse shape, weld head speed, spot size, weld head aperture, focal distance and focus location, cover gas compound, cover gas flow volume, cover gas flow direction, weld orientation, tube oxidation, tube secondary side thermal effects, moisture, single or multiple passes, weld width, weld depth, porosity, microfissuring, and surface finish. Once past the initial flat plate and split tube program, over 700 welds were sectioned and examined.

Results of the Weld Process Development Program

With the range of variables listed above, it is not practical to discuss every variable and its inter-relationship. However some general conclusions can be summarized with respect to the dominant variables.

a. Weld Mode -- With the wide control range of the JK706 laser, it is possible to maintain a Continuously Molten weld Pool (CMP) or to allow the weld to solidify between pulses of the laser. We selected to operate with solidification between pulses since the CMP process was sensitive to environmental conditions. In addition the CMP weld pool exists long enough to heat neighboring material, giving it a large heat affected zone (HAZ), which has the appearance and negative characteristics of a Gas Tungsten Arc Weld (GTAW).

When solidification between pulses is used, less total energy is put into the material. This results in a small heat affected zone around the weld itself. Environmental effects are minimized since most energy is absorbed for melting of the material, as opposed to heating the weld pool and neighboring material, as in the CMP weld. In addition, the weld is more uniform in width since preheating of the sleeve is minimized as the weld circles the tube.

b. Weld width, average power, and weld speed -- Skipping over variables such as frequency of the laser pulses (although pulse frequency is what determines if a weld pool remains molten or not), focal point, aperture, and pulse shape; the key inter-related variables are weld width, average power, and weld speed. If too wide of a weld is attempted, microfissuring can appear in the weld. The material of the tubing in the steam generator is fixed (Alloy 600), and the sleeve material (Alloy 690) is also somewhat set since the physical and chemical properties need to be compatible with both the tube metallurgy and steam generator environment. Thus one key variable is picking a parameter range in which microfissuring is not present. Microfissuring can begin to appear in single pass welds which are forced to be wide (too high of an energy input).
A second bounding condition is the power-speed relationship which causes the weld width to fall below the design objective. A third variable which occurs is surface condition; too slow of a weld speed might initiate undulations in the surface which do not permit adequate coupling with UT. Given that the gearbox in the weld head has certain limitations, as well as the laser having power limitations, the range of acceptable process parameters can be accurately identified [Figure 12].

c. Surface condition -- This variable is driven by the ability of the UT system to adequate coupling for inspection. In terms of mechanical measurement or visual comparison, the weld surface is acceptable over a wide range of variables. However UT coupling is determined solely through attempted measurement with the UT system. Surface condition is very dependent on the specific weld regime and cover gas flow. When cover gas is used for multiple functions as in our system (weld pool shielding, weld head cooling, and weld spatter control on mirrors and optics), the flow must be controlled as it passes over the weld pool. As seen in the process diagram, the welds which remain molten longest are most susceptible.

The process development program was concluded in late 1990, when a qualification was performed using ASME code guidelines. This was performed in a steam generator mockup using specimens located at the first through sixth support plates. This program, performed for 22mm outer diameter tubes and their associated sleeves, is currently being followed by finalization of a program for 19 mm outer diameter tubes and their associated sleeves. Weld processes and equipment for both 19 and 22 mm tubesheet and tube support plate sleeves (Series 44, 51, and the Model D and E series of steam generators) will be available for application in fall 1991.

**Summary**

Westinghouse has installed over 23,600 sleeves using brazing, mechanical, and CO₂ laser welding techniques. Knowledge learned from this experience has been incorporated into our new laser welding system. Tooling modularity, minimization on penetration demands, packaging of subsystems, and full robotic delivery will result in high field productivity at low radiation exposures. An extensive ongoing process development program has yielded a weld process which is insensitive to it environment, can be applied at support plate levels in the steam generator, and is of high quality. Extensive analytical, physical, and corrosion testing has exhibited excellent weld performance.
Figure 1. (right) SALEE being positioned by a ROSA robot for sleeve and expansion mandrel loading.

Figure 2. (below) Schematic of honing system showing relative placement of equipment on and below the steam generator platform.
Figure 3. (above) Schematic of the conduit delivery system used for sleeve insertion and expansion, welding, heat treatment, and UT inspection.

FIGURE 4. (right) Exterior of laser trailer. Only two cables (one fiber and one electrical) cable pass into containment. They can be seen at the front of the trailer ahead of the power cord. Additional cables (not related to the weld equipment) are run into containment from the robot control trailer.
Figure 5. (right) Interior of laser trailer. Spare parts, or in this case a complete spare laser, are stored in the trailer. The laser and its control equipment are shock mounted in place in the trailer, minimizing transport damage and site setup time.

Figure 6. (below) Schematic of fiber optic delivery system showing general arrangement of components.
Figure 7. (right) Four output optical switch used to allow one trunk fiber to service up to four steam generator platforms on a time share basis. This housing is typical of all portaged laser equipment.

Figure 8. (below) Gating scheme used for ultrasonic inspection of weld interface.
Figure 9. (above) Cross section of a typical sleeve to tube weld.

Figure 10. (above) The weld head contains a motor, gearbox, and encoder, focussing optics, fiber termination, sealing bladder, gas flow channels, easily replaceable turning mirror, centering device, and eddy current coil for axial positioning.
Figure 11. (above) The weld head exterior is clean and simple to facilitate handling. The rigid portion of the weld head is 30 centimeters long.

Figure 12. (above) Process diagram illustrating relationship of welding parameters.
QUALIFICATION AND FIELD EXPERIENCE OF SLEEVEING REPAIR TECHNIQUES.

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ABSTRACT

Sleeving has been evaluated as repair method for tubes cracked at the roll transition from the primary side.

A qualification program was performed on sleeves provided by different manufacturers with the aim of

- ranking (by efficiency) the different sleeves
- estimating their expected life compared with expansion transitions.

Moreover a field application was performed in Doel 3 in 1988 and pulled out sleeved tubes were destructively investigated.

RESUME

Le manchonnage a été évalué comme méthode de réparation des tubes fissurés à la transition de l'expansion par corrosion primaire.

Un programme de qualification a été réalisé sur les manchons fournis par différents fabricants, dans le but de

- classer les différents manchons, suivant leur efficacité
- estimer leur durée de vie par comparaison avec les transitions de l'expansion.

De plus, une campagne de manchonnage a été menée à Doel 3 en 1988 et des tubes manchonnés extraits de centrales ont été soumis à des examens destructifs.
1. INTRODUCTION

Up to now, the Belgian utilities have developed a tremendous effort in order to flight against Primary Water Stress Corrosion Cracking (PWSCC) at the roll expansion transition of Steam Generator (SG)tubes.

The problem was approached in two ways:

• At first, when potential cracking was identified for new steam generators, preventive repair methods were considered. They were aimed at preventing the initiation of cracks and stopping the propagation of already existing cracks.

The investigated methods were:
- Peening (either by rotating brushes or by projection of beads)
- Global heat treatment at low temperature.
Shot-peening was applied on the SG of two units after 3 cycles of service (Tihange 2 - Doel 3) and roto-peening on the SG of two units before service (Tihange 3 - Doel 4). This experience is described in a companion paper [1].

• Afterwards, when it appeared that cracking was increasing in spite of the peening treatment, curative methods were regarded with the purpose of restoring a leaktight barrier across the existing cracks and preventing the initiation of new cracks.

The considered methods were:
- Nickel electroplating
- Sleeving.

Nickel electroplating is discussed in another companion paper [2], whereas sleeving is evaluated in this paper.

The evaluation is based on:
- qualification laboratory tests
- field experience
- pulled tubes expertise.
2. MAIN FEATURES OF SLEEVING

2.1. Fundamentals - main requirements

Sleeving consists in fixing another tube in the parent tube, so that the damaged area is "bridged" by the sleeve and the structural integrity is thus restored. Sleeving has been used for years essentially in the USA and Japan to repair SG tubes damaged mainly from secondary side in sludge piles or at the tube support plate intersections. These sleeves are fixed either by TIG welding, brazing or mechanical expansion.

Sleeving was considered in Belgium as repair method for primary stress corrosion cracked tubes.

The first Belgian experience with sleeving (tubes partially rolled in the tube sheet) was the installation in Doel 2 in 1982, of about two hundred mini sleeves by Babcock and Willcox. These short (40 mm long) and thin (0.4 mm) sleeves were explosively expanded and welded over the cracked area, the sleeve acting as a leaktight membrane without any structural function.

This was not a success because some months later, more than 50 % of the sleeved tubes were circumferentially cracked, at the level of the weld limits, just below both ends of the mini-sleeve.

The lesson learned was that it was mandatory to keep stresses induced by the sleeving process low enough to avoid cracking at new locations. This condition should particularly be emphasized because such cracks are not detectable by currently available non destructive inspection techniques.

Moreover, achieve tightness seemed very important as a large number of tubes would be sleeved.

At a first glance, only welded sleeves with postweld heat treatment of the upper joint appeared to fulfil those requirements. Several manufacturers presently offer such sleeves.

2.2. Sleeve design

The sleeve material proposed by all manufacturers is Alloy 690, which has proved to be much more resistant to corrosion than the previously used Alloy 600.

The sleeve is a tube typically 75 mm long and 1 mm thick which is fixed in the parent tube by an upper and a lower welded joint (Figure 1).

However, due to the difficulty of welding at the lower joint (in the tube sheet), some manufacturers use a roll expansion.
The upper fixation includes an expansion of the sleeve in order to ensure contact with the parent tube prior to welding. The final touch is the postweld heat treatment. Laser, as well as TIG or explosive welding may be used. For explosive welding, there is no prior expansion of sleeve. Expansion may be hydraulic, roll or explosive type, or a combination of these.

The upper joint is normally situated beneath the end of the sleeve (overlap weld) so that the protruding end prevents the dislocation of the tube in case of cracks at the level of the weld.

One manufacturer offers a weld situated at the end of the sleeve (fillet weld).

The different designs are illustrated on Figure 2.

2.3. Advantages and drawbacks of sleeving

Sleeting restores the structural integrity of the tube so that this process appears well suited to repair heavily damaged tubes (very long and large cracks, circumferential cracks, ...).

However, it raises several drawbacks:

- it introduces a geometrical restriction which inhibits further repair and non destructive inspection of the tube
- it impairs the heat transfer.

Moreover, several types of degradations might happen in service:

- the existence of a gap between sleeve and tube makes the "OBRIGHEIM effect" possible in case of through cracks in parent tubes (water trapped in the gap during a shut-down may not escape rapidly enough during start up, which produces a deformation of the tube and/or the sleeve). Under particular chemical conditions this gap could also favour the development of Intergranular Attack (IGA) or Intergranular Stress Corrosion Cracking (IGSCC) in the sleeve.
- the welding process may generate a sensitized structure at the outer surface of the parent tube, which could detrimentally interfere with secondary pollutants.
- new residual stresses are introduced, which make the tube potentially susceptible to PWSCC at new areas namely
  - weld and heat affected zone
  - transition of the expanded area.

The first concern could be avoided by drilling a hole in the parent tube before sleeving.
The next two concerns depend on the sleeving parameters. The qualification program was therefore aimed at verifying that these degradations are not likely to occur in service.

2.4. Inspection of sleeved tubes

During plant operation, the inspection of the welds is performed by UT in order to check the continuity and the penetration of the beads (a minimum width of penetration is required to ensure structural integrity). A visual inspection with an endoscope is also proposed by some manufacturers.

In service inspections may presently give valuable results for the sleeve but not for the parent tube at the level of the weld, which is the most critical area.

3. QUALIFICATION OF THE PROCESS

3.1. Context

The test program started in 1988 and was focused on the upper joints as these are conditioning the structural integrity.

A first program concerning the 7/8" tubing was initiated by the Belgian utilities whereas later on another program on 3/4" tubes was undertaken as a joint venture between Spanish, Swedish and Belgian utilities[3].

These tests were aimed at:
- ranking (by efficiency) the different sleeve designs
- estimating the expected life of sleeves compared with expansion transitions.

An additional limited separate program was performed on Westinghouse laser welded sleeves prior to on-site application at Doel 3 in 1988. This topic is handled in paragraph 4.

3.2. Mock-ups for comparative test program

The mock-ups were fabricated by seven manufacturers from the same parent tube:
- Vallourec Mill Annealed (M.A.) Alloy 600 for 7/8" sleeves
- Westinghouse M.A. Alloy 600 for 3/4" sleeves.

These materials had been selected because of their high sensitivity towards stress corrosion cracking in 10 % NaOH.
For the 7/8" program only isolated upper joints were fabricated whereas for the 3/4" program, sleeves were installed in a complete mock-up including the tube sheet and the first tube support plate, so that both upper and lower joints were made. The parent tube had been blocked by welding at the level of the support plate in order to simulate denting at this location. The upper joints were then cut from the complete mock-ups and tested individually.

The postweld heat treatment was performed only on a part of the mock-ups so that a comparison could be made in the tests between heat treated and as welded conditions.

Those conditions were compared to an "ideal" situation simulated by upper joints which were separately heat treated in a laboratory furnace with a controlled homogeneous temperature and a slow cooling down.

The purpose was indeed to evaluate the efficiency of the industrial heat treatment to relax the residual stresses.

3.3. Comparative test program
This program includes:
- a characterization of the sleeves through mechanical and structural investigation
- an evaluation of the cracking resistance through residual stress measurements and corrosion tests.

The latter is particularly important because of the potential risk of new cracks developing in the heat affected zone and in the expansion transitions.

3.3.1. Characterization of the sleeves
The following tests were performed:
- ultrasonic inspection (when possible) in order to check the continuity of the weld and compare with the data of the manufacturer, when available
- dimensional measurements inside and outside the sleeved tube in order to quantify the expansion deformation and the protrusion of the weld
- visual inspection aiming at detecting weld defects
- metallographic examination of samples through the upper joint in order to check the weld (defects, width,...) and the heat affected zone
- modified Huey and EPR (Electrochemical Potentiodynamic Reactivation method) to estimate the sensitization of the Alloy 600 parent tube generated by welding and postweld heat treatment
- hardness measurement through weld and heat affected zone
- tensile test on strips aiming at determining the resistance of the bonding (effort applied on parent tube on one side and on sleeve on the other side).
3.3.2. Evaluation of stress corrosion cracking resistance

The following tests were performed:

- X-ray residual stress measurements at the outer surface of the sleeved tube, and after electropolishing. These measures were performed in heat treated and as welded conditions (including "ideal" furnace heat treatment). As those measurements are performed at the OD surface, they are not directly relevant to the service behaviour (cracks are expected at the ID surface of the parent tube). However, they may give useful information concerning the efficiency of the heat treatment.
  - stress corrosion test in deaerated 10% NaOH solution at 350 °C on
    . upper joints without inner pressure (to look at residual stresses only)
    . pressurized capsules (180 bar) either without counter pressure or with counter pressure (90 bar).

The tests were focused on the ID parent tube behaviour at the level of the weld heat affected zone and the expansion transitions. They were performed on as welded and on heat treated upper joints.

3.4. Main results

3.4.1. Characterization

. All the welds of one type (TIG - LASER - EXPLOSIVE) are very similar, whatever the manufacturer, but the type to type differences are important. The explosive weld largely differs from the others.
. For a series of mock-ups from one manufacturer, the welds are more or less reproducible. They most often show no defect. The penetration is normally wide enough and not through wall for TIG and laser welds with however some systematic exceptions such as an obvious lack of penetration.
. The welding and/or heat treatment may induce a sensitization which can reach the outer surface of the parent tube, depending on the process
. The mechanical properties of the joint are sufficient in any case.

3.4.2. Stress corrosion cracking resistance

. In the as-welded condition, the stress corrosion behaviour looks very similar for all welds, whatever the manufacturer: circumferential cracks appear rather rapidly in the parent tube at the limit of the weld (Figure 3). The higher the differential pressure during the test, the quicker the cracking is.
  . Cracks may also show up in the sleeve, either in the weld bead or occasionally in the heat affected zone, or in the parent tube at the expansion transitions.

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The cracking resistance of the upper joints is in any case increased by the postweld heat treatment, the effect being very different from one manufacturer to another, varying from nearly no influence to a dramatic difference.

A scatter may exist in a set of specimens produced by one manufacturer. 7/8" and 3/4" series produced by one manufacturer may give very different results, even if the same nominal parameters are used.

Industrial heat treatment performed by an internal probe may reach about the same efficiency as the "ideal" heat treatment in a furnace.

3.4.3. Final expectations

From these laboratory results, it is concluded that sleeves may be produced which are likely to withstand a long service duration without cracking, but that the heat treatment is a critical step which sometimes fails.

The difference between the manufacturers mainly arises from this step.

4. FIELD APPLICATION

4.1. On site experience

A structural sleeving operation was conducted by Westinghouse in Doel 3 during the 1988 outage. It was the first trial of laser welded sleeving.

The main characteristics of the process are:

- sleeve material: Alloy 690 TT
- both upper and lower joints are laser welded
- the upper joint was made in successive steps:
  . parent tube mechanical cleaning
  . introduction of sleeve and hydraulic expansion
  . CO₂ laser weld - 3 passes
  . thermal heat treatment

Initially, the number of tubes to be sleeved amounted to 200, all of them showing eddy current indications of long cracks, and located in one working zone of the robot arm. However, because of some initial difficulties with the first batch of 20 tubes the utility authorized only 35 supplementary tubes to be sleeved in the second batch instead of the 180 tubes initially foreseen. The residual available time was then too short to allow further sleeving.

Indeed, the UT check as well as the boroscope examination of the upper weld of the first batch evidenced a disturbance, appearing as a "wavy" profile on the weld which perturbates the UT control of the weld (Figure 4). This defect could result from
the tooling installation (laser alignment) and from a deviation of some very sensitive process parameters.

Consequently, the process parameters were slightly modified and a satisfactory production rate was achieved.

Both upper and lower welds were inspected by boroscope and the continuity of the upper welds was checked by two independent UT methods (Westinghouse and Laborelec-Nucon). This led to the plugging of one tube, the upper weld being suspected to have a leak path.

As there remained some concern about the health of the "wavy" welds, it was decided to pull out two tubes at the next outage (see paragraph 5).

4.2. Qualification program

Prior to this on site application, the process was qualified by a laboratory test program divided in two phases which were aimed at:

**FIRST PHASE:**
characterization of the upper joints and evaluation of the influence of sleeving conditions such as
- the amplitude of the parent tube deformation induced by the hydraulic expansion
- the sleeve/tube gap
- the presence of oxide at the tube inner wall
- the presence of humidity inside or outside the tube
- the utilization ratio of the laser mirror

**SECOND PHASE:**
evaluation of stress corrosion resistance by corrosion tests.

The first phase was completed before the field application whereas the second was planned for the following year.

The latter was cancelled because of the decision to replace the SG's of Doel 3 and because the CO2 laser process had been abandoned by Westinghouse.

The first phase results demonstrated that
- laser welds are not influenced by the sleeving conditions: they always show a very regular aspect with a very limited heat affected zone; they contain porosities
- a variation of the process parameters may induce very different results concerning the penetration and the amount of porosities, which may disturb the UT inspectability.
No "wavy" profile was met during this qualification program, which points out the problem of the reproducibility of such a process.

From this program, the process parameters had been definitely set for the field application.

5. PULLED TUBES EXPERTISE

5.1. Investigated tubes

Two laser welded sleeved tubes pulled out from Doel 3 after one service cycle were investigated in order to

- characterize the "wavy" welds and check the correspondence between real state of joints and UT inspection performed on site just after sleeving
- evaluate the service behaviour of the sleeved tubes, more particularly the stress corrosion cracking resistance and the possible occurrence of an "Obrigheim effect"
- compare the real tubes with the qualification samples.

Several other sleeved tubes pulled out from the old SG's of Ringhals 2 have also been investigated in the frame of the Swedish - Spanish - Belgian joint venture. They included:

- Combustion Engineering TIG welded sleeves
- Westinghouse brazed sleeves.

The investigation was aimed at evaluating the service behaviour and more particularly the stress corrosion cracking resistance after several service cycles which could give a lower bound for the lifetime to be expected from such a repair process, as those sleeves had not been stress relieved.

5.2. Main results

For the laser welded sleeves it appeared that the "wavy" welds show a good penetration and a limited amount of porosities, so that the structural integrity of such joints may not be questioned.

No service degradation was observed for any of the investigated sleeves. However, these favourable results may not be extrapolated to a longer term.

The investigation demonstrated a good agreement between the UT measurements of weld width penetration and the real state.
6. CONCLUSIONS

Sleev ing appears as a very interesting repair method as it restores the structural integrity which allows to repair heavily cracked tubes.

This advantage compensates somewhat the drawbacks such as geometrical restriction (reduced inspectability and access for further repair) and heat transfer disturbance, especially if plugged tubes are recovered.

The main concern raised by sleev ing has always been the life time of the repair, i.e. the risk of formation of new cracks in the parent tube at the level of the upper joint. Qualification tests as well as the expertise on pulled tubes show that correct heat treatment of the upper joint gives confidence for a long service life of the assembly. However, this is not achieved by all manufacturers.

The other concerns, the Obrigheim effect and an increased risk of secondary attack because of the sensitization induced by welding and heat treatment do not appear critical.

The field experience with laser welded sleeves may be considered as successful in spite of some disturbances; it however points out that the process may be very sensitive to some parameters so that the results are not always reproducible. This is in agreement with the qualification program.
Figure 1: Schematic representation of a sleeved tube
Figure 2: Different types of upper joints
Figure 3: Stress corrosion cracks at the weld limit in the parent tube

Figure 4: "Wavy" profile of laser weld at upper point
Figure 5: Metallographic sample through "wavy" weld
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SESSION 9 - REPLACEMENT OF STEAM GENERATORS

9.1 INFLUENCE OF MANUFACTURING PROCESSES ON STEAM GENERATORS BEHAVIOUR

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9.2 THE DAMPIERRE 1 SGR

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9.3 EXCHANGE OF STEAM GENERATORS IN PRESSURIZED WATER REACTOR POWER PLANTS

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9.4 STEAM GENERATOR REPLACEMENT AS A PART OF A GENERAL PROBLEM MANAGEMENT PROCESS

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INFLUENCE OF MANUFACTURING PROCESSES IN
STEAM GENERATORS BEHAVIOUR

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1.- INTRODUCTION

The recent industrial experience has showed that the Steam Generator's behaviour may affect severely the operativity and economic efficiency of the Light Water Nuclear Power Stations.

Actually, the service has promptly damaged the Steam Generators with an Inconel 600 MA tube bundle, in such an extent that it has been necessary to remove and to replace them with new components.

As expected, the design of the replacing steam generators incorporates a certain amount of changes in comparison with the initial ones.

Although the main purpose of these design changes is to avoid the damaging phenomenons already experienced, other additional aspects are considered in order to obtain an better component in reliability and maintenance.

Based on this and taking advantage of the state of the art, the different designers have developed several designs for the replacing Steam Generators.

From our point of view of components manufacture, we can see similar tendencies among the new different designs in comparison to the original component.

In this paper, we analize such tendencies based on our shop experiences and service maintenance. Nevertheless, our special point of view should be
considered: we cannot compare many important details of the different designs (for instance, tube bundle support, blow down system, moisture separation system). In fact our only comment is a positive one: all of them are feasible by means of a good shop practice.

This is not the case with other areas of the component, where homologation and production test coupons and NDE supply a lot of relevant information.

2.- THE STEAM GENERATOR FOR REPLACEMENT VERSUS THE ORIGINAL ONE

The main improvements in the design for replacement steam generators can be classified in accordance with the corresponding part of the component, namely, pressure boundary tube-sheet, tube to tube-sheet joint and miscellaneous.

2.1.- Pressure Boundary

The designs of steam generator for replacement specify a ring forging for the transition cone and forgings with integrated nozzles for the upper and lower heads.

The corresponding advantages over the original design (plate for the transition cone, while plate or cast steel for the bottom head and plate for the upper head with forged nozzles attached by welding) are at lower price, shorter manufacturing time and simple and cheaper in service inspection (ISI).

The other shell courses are specified either forging or plate. Sometimes, the lower ISI requirements of the former (due to the elimination of the longitudinal weld) balances its higher price, and in the case of the course next to the tube-plate, the forging can be often justified by the manufacturing schedule.

Other interesting feature of the new design is the quality of the new forgings: the specification ASME SA 508 Class 2 is no longer used. Instead ASME SA 508 Class 3, no susceptible to reheat cracking, is specified.

Another change in the replacement components is the design of the primary inner/outer nozzle to main pipe joints. Formerly, in the original component, this weld was through an austenite stainless steel buttering. Thus the austenite / low alloy steel interface of the joint, pressure retaining, suffered during manufacturing heat treatment (600 °C) carbon migration and thermal stresses leading to lower toughness, high residual stresses and lower corrosion resistant properties along it.
Now, these joints are done through NiCrFe-3 buttering which avoids those inconveniences.

2.2.- Tube-sheet

The Ni-Cr-Fe3 weld overlay of the tube-sheet is performed by an automatic welding process like plasma, or strip submerged arc welding, the shield metal arc welding being restricted to the small central zone.

The plasma process is more popular among manufacturers because of its higher deposition rate, but it is argued that the strip submerged arc welding overlay shows in one particular design a higher weldability when joining the tubes to the tube-sheet.

Our shop experience with both processes has been quite satisfactory.

2.3.- Tube to tube-sheet joint

The most relevant parameter of the tube to tube-sheet joint is the tube expansion (INCONEL 690 thermally treated or INCOLOY 800 modified tubes against the former INCONEL 600).

Formely the tube expansion was by mechanical rolling pilgrim pace sequence, often only on a partial length of the whole thickness of tube-sheet.

Now, all the new designs try to close any deep crevice between tube and tube-sheet by expanding the tube all along the tube-sheet.

The actual expansion processes specified by the designers are two, namely, the hydraulic and a mixture of the hydraulic and mechanical rolling, that intend to get a smooth expansion transition.

In addition to this improvement, the qualification procedure for the tube expansion is now more complex with a thorough study of the tube inner diameter and profile, tube pull out strength and, in one design, stress corrosion cracking tests also.

Our laboratory experiences with the hydraulic and mechanical rolling have confirmed its high capacity to close any crevice between tube and tube-sheet, as stated in bibliography.

Finally another interesting new features of the replacing steam generator
are the customer's requirement of production monitoring coupons for tube to tube-sheet welding and expansion at the beginning of each shop shift, and the determination of the inner profile of the actual component expanded tubes by eddy current.

2.4.- **Miscellaneous**

Other modifications have been done in replacing steam generators that improve their maintenance and are oriented to reduce the radiation buildup and personnel exposure, as those indicated below:

- Cladding surface finish improvement in channel head,

- Change of bolts by studs, and

- Special requirements on dimensional tolerances and flatness of the nozzles dam rings.

3.- **CONCLUSIONS**

Our review of steam generator's replacement design from the manufacturing and maintenance point of view has shown an improvement in the quality and reliability of these components.
TACK-ROLLING

MECHANICAL EXPANSION

HYDRAULIC EXPANSION

TEST COUPON OF EXPANSION PROCESS

HYDRAULIC + MECHANICAL ROLLING

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CNSI/UNIPEDE
SPECIALIST MEETING
ON OPERATING EXPERIENCE WITH STEAM GENERATORS
BRUXELLES, SEPTEMBRE 1991

THE DAMPIERRE 1 SGR

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9.2-1/7
ABSTRACT

In 1990, FRAMATOME carried out replacement of the three steam generators of the DAMPIERRE 1 nuclear power unit in France, for the owner, Electricité de FRANCE (EDF).

Meticulous preparation and skillful execution of this operation enabled attaining a collective radiation dosage far below that recorded for any other steam generator replacement (SGR) operation to date.

After having summarized the context for the DAMPIERRE 1 SGR, the author analyses the main technologies applied during this operation and the results obtained, from both the technical and dosage standpoints.

He then discusses how FRAMATOME intends to integrate the feedback of experience from DAMPIERRE 1 for its future SGRs.
SUMMARY

1. INTRODUCTION

The DAMPIERRE 1 (DA 1) SGR operation was performed in 1990 at the occasion of the 10-year in service inspection outage.

A comprehensive preparation program was launched in 1985, with particular emphasis on decontamination, cutting, bevelling, fit-up and welding processes and equipment.

FRAMATOME, the NSSS supplier, was designated by EDF (the French national utility) as "main contractor", responsible for reactor coolant circuit (RCS) repair vis-à-vis the French safety authorities.

The basic techniques selected were:

- SGR removal and installation in one piece
- RCS piping cutting only at the SG nozzles (2 CUT SCENARIO).

At Dampierre, moreover, it was decided to implement different competing techniques (topometry versus templates, soft chemical decontamination versus electrodecontamination).

The following focuses on the Framtome scope of work related to the primary loops during the Dampierre 1 SGR operation.

2. DESCRIPTION OF THE DA 1 STEAM GENERATOR REPLACEMENT

2.1 Assignment of the replacement Steam Generator

Topometry measurements of the bunkers carried out in 1986 and of the as-built RSGs enabled assigning each RSG a particular SG bunker before starting the replacement operation.

2.2 Decontamination: Two processes were used

2.2.1 RCS piping elbow electrodecontamination (loops 1 and 2)

It was carried out after the removal of the old SGs by means of two leaktight suction cups, through which the electrolyte is circulated applied to the RCS piping wall.
A carrier provided continuous motion of the suction cups.

The operation was remotely monitored.

The process and procedure were jointly developed by EDF, Framatome and STMI.

2.2.2 RCS piping elbows and channel head soft chemical decontamination (loop 3)

The SG 3 having been selected for an expertise program to follow its removal from the unit, a soft chemical decontamination was performed using the LOMI process. Expansion of special inflatable plugs inside the RCS piping provided leaktightness at the boundary of the portion of circuit to be decontaminated.

Decontamination was performed by circulating alternatively oxidating and reducing solutions through the channel head manway.

Most of the decontamination equipment was installed in the fuel building.

2.2.3 The results obtained by both methods were comparable, with a reduction in the dose rates by a factor between 8 and 13.

2.3 Clamping of the RCS piping

Clamping devices were installed in order to provide the means for:

- locking in their original position both the primary pipes, the primary pump and SG support legs
- moving the primary pipes to the fitup position.

Three clamping devices were installed respectively on the hot and cold legs and between the RC pump and SG support legs.

Hydraulic jacks fitted on the cold and hot leg clamping assemblies permitted to perform and monitor the pipe movements required for fitup in accordance with values given by the topometry measurements and within authorized limits given by the stress analysis calculations.
2.4 Cutting of RCS piping

A plasma cutting process was selected because of its high speed of execution.

An internal protection of the pipe against incandescent slag was installed inside the pipes. The waste gases produced were extracted and filtered.

2.5 Removal of the old SGs

The old SGs were raised to the service floor level and up to the tilting position by means of the polar crane. A tilting shell fitted to the SG lower head and a handling carriage moving along a track in line with the equipment hatch were employed for the tilting operation.

Prior to the containment building, the external surface of the SG was checked for contamination and the SG nozzles were sealed.

Each SG was moved out of the reactor building through the equipment hatch and finally lowered onto the transportation trailer by means of the existing external gantry crane.

2.6 Measurements : two methods were used for one of the most critical operation

2.6.1 Topometry

For the "measurement" function, two electronic theodolithes connected to a computer and controlled by means of a special software program were used.

"BEST FIT" a software developed by FRAMATOME, was used to combine the measurement results with the fitup requirements.

Such measurements and calculations were necessary for :

- determining the RSG piping fitup positions
- monitoring the movement of the RCS piping to the fitup position
- determining the RCS piping level positions
- installing the "reference ring".

2.6.2 Template (redundancy)

The template solution was also used as an alternative solution to topometry.

2.6.3 Both solutions were found to be satisfactory. Topometry appeared to be the more accurate method.
2.7 Revelling of the RCS ends of the RCS piping

Framatome developed a "reference ring" adjusted to the pipe end to locate the bevel position using the results of the topometric measurements and associated calculations.

The machine developed for the pipe end machining operation is centered inside the pipe and adjusted to the reference ring. It is equipped with a numerical control system and is remotely monitored.

2.8 Installation of the new SGs

These operations proceeded in reverse order to the removal operations. To prevent any contact between the SG nozzles and RCS piping ends during the positioning of the new SGs, the RCS piping was moved back away from its fitup position.

2.9 Final fitup...The Moment of Truth

The fitup is completed when the RCS piping is moved back to its fitup position. After this movement, the gap, concentricity and alignment of the two weld preparations were found within the welding specification tolerances. Then the piping was locked onto its temporary supports.

2.10 Welding

In order to provide for larger fitup tolerances, it was chosen for the DA 1 operation to weld using a mixed manual with coated electrodes and automatic with GTAW welding process.

Root passes were manually welded.

Filling passes were performed with a Framatome developed equipment using the GTAW process. The original features of this equipment are:

- a remote control system including a custom designed video monitoring system and a digital programmer associated to a microprocessor-based operator assistance;

- an automatic device for driving, guiding and storing the cable bundles.

Six such equipments were used to simultaneously weld the hot and cold legs of the three primary loops.
3. **LESSONS OF THE DA 1 SGR OPERATIONS FOR THE FUTURE**

The 212 man-rem total accumulated exposure for the DA 1 SGR operation is the lowest figure ever recorded for such a job.

This was achieved because of a constant control over (1) the preparation work including extensive training of the crews, and full size mock-up qualification of the techniques, (2) the on-site integrated organization, and (3) daily monitoring of the exposure associated with the installation of additional shielding, where and when needed. Future operations will be planned along with same proven approach.

The innovative choice of a two-cut scenario demonstrated to be a good one. Thanks to the very high accuracy achieved for the fitup of the RCS piping to the SG, the use of narrow-gap welding is considered for future SGR operations.

Full size demonstration of these techniques are scheduled to be performed during the construction of new plants being currently underway.

The use of the techniques during future SGR operations will mean additional savings in terms of cost, duration and dose exposure as well as the achievement of the highest possible quality.
EXCHANGE OF STEAM GENERATORS IN PRESSURIZED WATER
REACTOR POWER PLANTS

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ABSTRACT

As a result of problems experienced in particular in connection with the tubing of the
steam generators, several utilities are having to replace their steam generators.
Approximately 4 to 5 years before the scheduled replacement outage a feasibility
study has to be conducted. In this study all site specific conditions have to be
considered. As a result the optimum replacement method will be selected.

Following the study a detailed engineering and planning has to be done with focus on
a step by step work sequence plan and in a two hours split time schedule. Before
starting the replacement work extensive training and qualification of personnel and
equipment is unavoidable. Special technical features as
* optical survey
* mechanized cutting and machining methods
* remote controlled narrow gap welding
* specially developed decontamination methods
make sure that steam generator replacements can be performed in high quality within
a short time schedule and with low dose radiation exposure.
As a result of problems experienced, in particular in connection with the tubing of the steam generators, several utilities are having to exchange their steam generators.

Besides the design and supply of complete replacement Steam Generators Siemens KWU Group performs all activities necessary for the replacement of these components. These activities are, essentially:

- Performance of studies, in particular of feasibility studies
- Performance of detailed engineering and execution planning
- Performance of personnel training and qualification
- Design, supply and qualification of special tools and equipment
- Procurement of the necessary hardware
- Performance of steam generator replacement on site.

All these activities can be performed as separate tasks or as a turnkey package.

1. **Feasibility Study**

   The feasibility study, as a rule carried out with a long lead time before steam generator exchange proper, investigates the following points:

   - Transportation of the steam generators out of and into the containment
   - Transportation and handling of the steam generators within the containment
   - Examination of the various replacement options possible
     
     • Replacement of a complete SG
     • Replacement of a split SG (steam dome cut, channel head cut)
     • Evaluation of possible reactor coolant pipe cut methods as they are essentially 2 cut, 3 cut, 4 cut method
   - Check of available crane capacity and of potential for modification
   - Identification of decontamination work possible
   - Comparison of options in terms of scheduling and economy

2. **Detailed Planning**

   Highly detailed planning is essential for completion of component exchange to schedule.

This requires an engineering schedule planning procedure to establish service activities such as:

- On-site walk through, including details as built templating and field surveying, during a plant refueling shutdown before commencement of exchange
- Identification of potential subcontractors
- Definition of deadline for ordering of all necessary hardware
- Development of a detailed engineering time schedule
- Preparation of review documents for hardware and for on-site implementation
- Preparation of a “general work sequence plan” for on-site implementation
- Personnel deployment planning
- Equipment deployment planning
- Design and qualification of special-purpose machines and devices as necessary
- Performance of calculations and analyses as necessary
- Performance of safety evaluations
- Performance of dose rate calculations
- Development of shielding plans according to ALARA concepts
- Preparation of on-site time schedule
- Performance of on-site organization planning
- General management and follow up of all these activities

All these engineering and preparative activities are performed by a centralized planning team comprising specialists from the various engineering departments, as well as subcontractors working in close proximity. This makes for a short, efficient and rational engineering lead time.

3. 

Hardware

The hardware required for an SG exchange consists for instance of:

- SG hoisting equipment outside containment
- SG hoisting equipment inside containment
- SG transport facilities
- Reactor coolant piping components, secondary side pipes, valves, pumps, supports, etc.
- Insulation
- Adaptation of existing machines and manipulators for machining, welding, decontamination, etc.
- Models and personnel training and qualification facilities
- General and special duty shielding
- Infrastructure facilities like storage building; temporary access building; offices, etc.

All the above mentioned hardware has to be scheduled, designed, calculated as required, fabricated and delivered in such a way that it will be on site in due time before commencement of SG replacement.

4. 

Transportation and Lifting-In of the SG’s

Removal of the old and installation of the new steam generators is essentially divided into three major phases:

- Transportation and hoisting outside the containment
- Transportation inside the containment
- Hoisting and turning operations within the containment

Special hydraulic cable jacks are used for this work. For work inside the containment, these special facilities can be used in conjunction with the reactor building crane or on the polar crane girders but independent of the reactor building crane.

The steam generators are transported into and out of the containment
- through an equipment airlock
- through the opening left when a material airlock has been removed (as it was conducted by Siemens KWU, when they replaced the steam generators at Obrigheim, West Germany, in 1983)
or via a transfer opening specially made to permit SG exchange and which must be closed again upon completion of SG exchange; (as it was conducted by Siemens KWU when they replaced the steam generators at Ringhals 2, Sweden, in 1989), and which of these transfer routes will be used will depend on local plant conditions and the exchange option selected.

5. Measuring Technique

Two techniques are normally used to record exact measurements above all in the area of steam generator supports and below the steam generators in the area of the reactor coolant piping:

- Photogrammetry and
- Optical measurement

The as is installed condition is normally measured by means of photogrammetry during a scheduled refueling outage and the results used as a basis for further planning.

Optical measurement is used for previous measurement on the new components (steam generators, reactor coolant line elbows) and for measurements during replacement operations. It will be used, for example, to establish the location of weld edges already prepared on the steam generator nozzles and on the remaining sections of the reactor coolant piping. These results will be then used as the basis for machining the elbow ends.

6. Machining and Welding

Machines specifically designed for local space conditions and application requirements are used for machining (cutting piping, weld-edge preparation, rounding-off pipe inside surface, outside weld surfaces, etc.).

These machines and devices can be suitably adapted for specific plant requirements.

An automatic GTA narrow gap welding process allowing relatively speedy, low-repair and fully remote controlled welding is used for the primary coolant piping. Here too, suitable equipment and the necessary expertise are available and can be adapted as required by the specific site conditions.

7. Decontamination

In order to keep the dose received during exchange work as low as possible, the parts to be exchanged, and especially the reactor coolant piping, must be decontaminated.

Depending on customer specification, this decontamination can take the form of decontamination of the entire primary system or of local decontamination of the SG primary heads and close-in parts of the reactor coolant piping before commencement of the first cutting work.
A further possibility - and this was used by Siemens KWU in their already performed SG replacements - is decontamination of the cut locations by electro-polishing or by mechanical blasting after cutting of the piping.

8. Personnel Training and Qualification

In order to ensure that work on site is performed smoothly and on schedule, major attention must be paid to personnel training and qualification.

A model designed specifically for this purpose is used to give the operating personnel hands-on experience, particularly in the following activities:

- Machining
- Welding
- Operation of the decontamination manipulators
- Optical/electronic surveying

A subsequent qualification program ensures that only well-trained personnel is used for component exchange in the power plant.

Prior to the personnel training and qualification program the same 1:1 mock-up will also be used for equipment qualification in order to make sure that all special tools and designed equipment fits according to the specific plant geometric requirements.

9. Siemens References in Steam Generator Replacement Worldwide

Feasibility Study

Beznau 1
Ringhals 2
Angra 1
Doel 3
Genkai 1
Almaraz I and II
Ascó I and II

(364MW, W)  (1989 Transport Study of 2 SG’s)
(860MW, W)  (1987 Replacement of 3 SG’s)
(650MW, W)  (1989 Replacement of 2 SG’s)
(392MW, ACEC)  (1989 Replacement of 3 SG’s)
(559MW, MHI)  (1990 Transport Study of 2 SG’s)
(930MW, W)  (1990 Replacement of 3 SG’s)
(930MW, W)  (1990 Replacement of 3 SG’s)

Detail planning

Obrigheim
Ringhals 2
Beznau 1

(345MW, KWU)  (1982 Replacement of 2 SG’s)
(860MW, W)  (1988/89 Replacement of 3 SG’s)
(364MW, W)  (Soft- and hardware planning for the Replacement of 2 SG’s, each in two pieces, executed in 1985)

with Bechtel-KWU-Alliance:

North Anna 1
Palisades
Farley 2
V.C. Summer
Ginna

(915MW, W)  (1990 Replacement of 3 SG’s)
(750MW, CE)  (1990 Replacement of 2 SG’s)
(828MW, W)  (1990 Replacement of 3 SG’s)
(885MW, W)  (1990 Replacement of 3 SG’s)
(470MW, W)  (1990 Replacement of 2 SG’s)

9.3-5/9
SG Replacement Execution
Obrigheim (345MW, KWU) (1983 Replacement of 2 SG's)
Ringhals 2 (860MW, W) (1989 Replacement of 3 SG's in only 72 days)

Current projects
Beznau 1 (364MW, W) (Execution in 1993)
Doel 3 (392MW, ACEC) (Execution in 1993)
Ringhals 2 NPP, Sweden
SG Replacement
DE Austausch

View “a”

Hoist
Hubeinrichtung

Auxiliary Hoisting Unit
Hilfshubeinrichtung

Connecting Links
Verbindungslaschen

Lifting Attachment
Anschlagvorrichtung

Tilting Support
Kippbook

Hoist
Hubeinrichtung

Transport Saddle
Verschubsattel

Rail
Verschubbahn

Transporter with 12 Axes
12 Achser Straßenroller

Rigging of Steam Generators
Hebe- und Transportbewegungen der Dampferzeuger
Beznau Unit I, Steam Generator Replacement

Transportation of Steam Generator into the Reactor Building
STEAM GENERATOR REPLACEMENT AS A PART OF A GENERAL PROBLEM MANAGEMENT PROCESS

Ph. Somville
Belgatom/Tractebel
Brussels (Belgium)

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ABSTRACT

The integrated Belgian approach to steam generator problem management is presented. It focuses on the real safety issues and operational concerns based on a step by step process.

Applying this approach to the particular problem of Primary Water Stress Corrosion Cracking (PWSCC) in roll transitions of the Doel 3 plant required a number of original developments and led to the conclusion that the immediate replacement of the steam generators was the most cost-effective strategy. This replacement will be combined with a power uprating study to increase plant output to 107-110 %. Finally the most important choice facing the Utility after the replacement has been decided, i.e. selection of the tube material, is addressed.

RESUME

L’approche belge intégrée en matière de gestion des problèmes de générateurs de vapeur (GV) est présentée. Elle est focalisée sur les réels problèmes de sûreté et les préoccupations liées au fonctionnement satisfaisant des GV au moyen d’une démarche par étapes successives.

L’application de cette approche au cas particulier de la corrosion sous tension d’origine primaire dans les transitions de dudgeonnage des GV de Doel 3 a nécessité plusieurs développements originaux et conduit à la conclusion que le remplacement rapide des GV était la stratégie la plus économique. Ce remplacement sera combiné avec une étude d’augmentation de puissance pour porter la puissance de la centrale à 107-110 % de la valeur actuelle. Enfin, le choix le plus important à effectuer par l’Exploitant après avoir décidé de remplacer les GV, c’est-à-dire la sélection du matériau des tubes, est abordé.
1. INTRODUCTION

Steam generators (SG) in the Belgian power plants have been affected by a large variety of problems, leading to the development of an integrated approach to steam generator problem management [1]. This approach focuses on the real safety issues and operational concerns, through the following steps:

- problem detection;
- dedicated sample monitoring;
- implementation of preventive methods;
- development of specific plugging criteria;
- dedicated 100% inspection;
- implementation of repair methods;
- adjusted sample monitoring;
- repair versus replacement strategy.
2. THE INTEGRATED BELGIAN APPROACH

Each step is illustrated by the particular case of Primary Water Stress Corrosion Cracking (PWSCC) in the tube roll transitions, which is presently the main problem in Doel 3 and Tihange 2.

2.1. Problem detection (Step 1)

The problem is usually detected by leakage observation or by routine Eddy Current Testing (ECT) with the bobbin coil. Once detected, the problem needs further characterization as to its type, extent and cause; this may require extension of the inspection sample and/or use of other destructive or non-destructive methods.

PWSCC case: The PWSCC problem was detected in 1983 by an in-service leak during the first operational cycle of Doel 3 and confirmed by ECT inspection during the next outage. The direction, length and depth of the cracks were assessed by a prototypical rotating ECT probe (RPC = rotating pancake coil) [2] and their location was related to roll transitions (between overlapping steps) by ECT profilometry of the expanded section of the tubes; both techniques were original developments of LABORELEC. Destructive examination of two pulled tubes confirmed the PWSCC diagnostic while laboratory tests established the generic nature of the problem.

2.2. Dedicated sample inspection (Step 2)

After being detected, the problem needs to be monitored by a dedicated sample inspection which complements the routine ECT.

This dedicated inspection often requires the use of a specific examination method and the sample size must be sufficient to yield statistically meaningful distribution curves of:

- Relevant defect size;
- Defect growth rate.

The examination may be limited to the areas which are likely to be affected and to give a safety or operational concern.

PWSCC case: Owing to the poor performance of "bobbin coil" ECT for PWSCC detection and sizing, the problem has been monitored since 1984 by a modified version of the early RPC probe, on samples of a few hundred tubes with a coverage of the entire tubesheet but an inspection length limited to about 150 mm across the roll transition [2]. This allowed collection of a large amount of data and establishment of reliable distribution curves of crack length (depth assumed to be throughwall) and crack length increase between successive inspections.
2.3. Implementation of preventive methods (Step 3)

After the generic nature of a problem has been established, and in parallel with step 2, it is appropriate to perform a comparative evaluation of the possible preventive methods, if available, or to develop (R & D) an adequate industrial process, if time permits!

The objective of a preventive method is to “freeze” the disease status (prevent damage initiation or avoid damage propagation for tubes already affected). To be efficient, it should thus be implemented as soon as possible.

PWSCC case: A large program was launched in 1984 by the Belgian Utilities and further cosponsored by EPRI [3], to develop peening techniques and global in-situ heat treatment of the whole tubesheet.

Eventually, the peening techniques were selected and implemented, for the first time in the world, on steam generators already installed on site.

2.4. Development of specific plugging criteria (Step 4)

The usual criteria, requiring to plug any tube with a defect deeper than 40% of the wall thickness, are unduly conservative for some types or locations of defects.

Alternate plugging criteria should be established, on a mechanism-specific basis, to assure adequate safety margins against a SG Tube Rupture (SGTR) under all design conditions.

These criteria allow operation of the SG with through wall cracks while maintaining the required structural integrity. However, in such a case, the primary to secondary leakage requires to address the corresponding safety issue of radioactive release to the environment.

PWSCC case: It was soon realized by the Utilities, and accepted by the Belgian safety authorities that the usual “40% depth” plugging limit was not realistic for PWSCC. Alternate plugging criteria were developed by Tractebel and Laborelec, on basis of a more realistic interpretation of the Regulatory Guide (RG) 1.121 issued by the USNRC.

The selected approach allows the SG to continue operation with a large number of TWD cracks [4], while still meeting the leak limits of the technical specifications. In case of accidental conditions resulting in a larger differential pressure on the tube, the leak rate could however increase significantly. This has been addressed by a probabilistic approach.

2.5. Dedicated 100% Inspection (Step 5)

If and when the distribution curves of defect size indicate a risk of exceeding the plugging limit, it is necessary to switch from a sample monitoring (step 2) to a
100% inspection (all tubes within the degraded area), using the same dedicated method that allows a reliable defect sizing. Such a constraint may require to improve the time efficiency of the inspection method in order to avoid costly extension of plant downtime.

**PWSCC case**: In 1987, it became obvious that shot-peening did not prevent further propagation of preexisting axial cracks. The potential for some of them to reach the plugging limit was high enough to justify the development by Laborelec of an optimized RFC technique allowing an efficient 100% inspection of the roll transitions. Such large scale inspection is implemented since 1988 for both the Tihange 2 and Doel 3 units.

### 2.6. Implementation of repair methods (Step 6)

Selection or development of repair methods is necessary to the extent that the implemented preventive measures do not prove to be sufficiently successful.

Repair methods, such as sleeving, are an alternative to plugging; they need only be implemented when the plugging limit is reached, but preventive repair may prove economically preferable.

**PWSCC case**: In 1988 two repair methods were tentatively applied to Doel 3 steam generators, on tubes not yet requiring plugging [5]:

- Nickel electroplating, an original process jointly developed by LABORELEC and FRAMATOME;
- Laser welded sleeves.

Further non-destructive (UT) and destructive (pulled tubes) examinations allowed to verify the validity of these methods and to confirm their respective merits.

### 2.7. Adjusted sample inspection (Step 7)

100% inspection is not necessarily required after each cycle. Reinspection may be limited to those tubes with a defect size such that the maximum increase rate would allow them to reach the plugging limit; for tubes previously found to be defect free, this involves consideration of the maximum initial size (as observed at detection time). This dedicated monitoring inspection is complementary to the general inspections:

- Base inspection ("bobbin coil" ECT);
- Repaired tubes (examination, by an appropriate method, of the repaired area of a sample population).

**PWSCC case**: While this principle would have allowed to reduce the extent of inspections in 1989 and 1990, it was not applied because the high speed performance of a 100% inspection (2 days/SG) did not warrant the trouble of reviewing the scope;
meanwhile special inspection techniques, including Ultrasonic Testing (UT) are developed to allow a meaningful examination of the repaired tubes.

2.8. Repair versus replacement strategy (Step 8)

Depending on the extent of repair which might be needed in the long term, it is worthwhile to weigh the repair approach against the SG replacement. This step is detailed in the next chapter.
3. REPAIR VERSUS REPLACEMENT STRATEGY - APPLICATION TO DOEL 3

When the degradation rates become fairly high, the ultimate step of steam generator replacement comes into play as a possible solution. To decide which strategy to adopt for the Doel 3 steam generators, a long term study has been performed[6].

3.1. Approach adopted for the study

The study is based on a deterministic approach (Figure 1), where each parameter is given a fixed value except when the uncertainty concerning one parameter is judged too large, in which case a sensitivity study is performed.

The first step of the study consists in analysing all present and potential future problems of the steam generators. In parallel, technical and financial assumptions are necessary to assess the impact of each problem.

The second step is the selection of a series of scenarios. For each, the evolution of the plugging rate is studied as well as the risks and the cost, to allow selection of the best strategy taking all factors into account.

3.2. Prediction of degradation rates

3.2.1. Present degradation modes

The major degradation mode of the tube bundle in Doel 3 steam generators is PWSCC in roll transitions and U-bends.

Based on extensive inspections performed every year, Laborelec has developed a model for the prediction of the future rate of degradation for PWSCC in the roll transitions[8]. Sleevings and nickel plating are assessed as possible remedies.

For the U-bends, all row 1 tubes are presently plugged but plug removal is possible. A heat treatment is however required before reusing the tubes. As this is not fully effective for tubes already cracked or for tubes with large ovalities, a limited benefit is accounted in the assessment.

3.2.2. Possible future problems

This exercise is more difficult. It includes a review of all possible problems and for each one an evaluation of its probability, extent and date of occurrence in Doel 3. The evaluation is based on a thorough survey of international experience combined with plant-specific features like chemistry, operational parameters and tube sensitivity. Here again the goal is to define the remedies and associated costs.
Conclusions for Doel 3 in 1987 were that tube wear against the antivibration bars (AVB) was predicted as nearly certain to occur within 2-4 years with AVB replacement required before year 2000. In addition moderate secondary side stress corrosion cracking (ODSCC) could appear after 10 years of operation. Incidentally the first prediction proved quite good as AVB wear was detected in 1989 for the first time.

Globally the present situation corresponds to a degradation rate comprised between the lower and the average rates used in the analysis.

3.3. Cost analysis

3.3.1. Technical assumptions

Several technical assumptions must be performed before proceeding with the cost analysis. The most important concern:

- Plant availability factor;
- Frequency of unscheduled outages, which is obviously different if the steam generators are replaced quickly or if the plant operates with degraded steam generators;
- Time available each year to perform repairs of the steam generators;
- Repair techniques (speed of application, number of defective repairs and more important of all: lifetime of the repairs);
- Radiation doses (ALARA policy).

3.3.2. Financial assumptions

In parallel with the technical aspects, various financial parameters have to be evaluated so as to select an appropriate value for each one. These are mainly:

- Interest rates for discount calculations;
- Replacement power costs per day of scheduled or unscheduled outage;
- Inspection costs;
- Other cost elements (unit costs for tube repair...).

3.3.3. Repair costs

Costs associated with repair operations are evaluated on the basis of proposals submitted by several vendors and of the prediction model for PWSCC.

3.4. Scenarios and repair strategy

Each scenario is based on a degradation rate and on a strategy regarding the management of the steam generator problems.
Several strategies can be adopted:

- Immediate replacement of the steam generators;
- Plugging of degraded tubes to delay the replacement while minimizing the maintenance costs (no repairs);
- Repair of degraded tubes;
- Repair of degraded tubes combined with preventive actions (like AVB replacement for instance) to extend as far as possible the lifetime of the steam generators.

For the last two strategies, replacement of the steam generators might still prove necessary if the repairs are not fully successful.

The objectives of the repair strategy is to extend the lifetime of the steam generators while minimizing the impact of the repair work on the duration of the refueling outages and minimizing the global cost.

Selection of the best strategy is based on:

- Plugging margin;
- Plugging criteria;
- Extent of repair expected for each degradation mechanism;
- Available time period for repair work during each outage;
- Minimum number of tubes to repair during a campaign for cost effectiveness. It may be more economical to plug tubes temporarily so as to proceed with large repair campaigns only rather than to repair a small number of tubes each year.

3.5. Power uprating

Uprating may prove to be a determining factor in the decision-making process. This aspect is detailed in chapter 4.

3.6. Conclusions

Figure 2 shows an example of the evolution of the plugging rate for different cases analysed.

As for the risks or uncertainties, these are clearly more important for the repair strategy, especially regarding the long term reliability of the repair techniques.

Finally the cost comparison is presented in Figure 3 which has been drawn with the set of assumptions most favourable to the repair, i.e. the slowest evolution of the corrosion and considering only the uprating associated with the larger heat exchange area of the new steam generators without increasing the core power. From this figure, it may be concluded that the repair is only cost effective compared to the immediate replacement if applied to delay the replacement by about 5 years or if the
repair techniques allow to keep the present steam generators until the end of the plant lifetime, which is not very probable.

So with more risks for the repair and no benefit expected whatever the assumptions used in the analysis, the replacement has been selected for Doel 3 and programmed in 1993 together with the tenth year outage.

From a short term evaluation, the natural tendency would have been to go on with the present generators. Only a complete long term study could identify that immediate replacement was the most cost effective strategy.
4. PLANT UPRATING ASSOCIATED WITH SG REPLACEMENT

4.1. Preliminary uprating evaluation

Replacement of SG provides a unique opportunity to uprate the plant. This uprating may also influence the design of the replacement SG. For Doel 3, uprating studies are under way with Framatome to increase the power output to 104.3% with the present SG.

For the new SG, a preliminary uprating evaluation led to the recommendation to change the tube diameter from 7/8" to 3/4" and the tube pitch from square to triangular to increase the heat exchange capacity of the SG as far as reasonably achievable. This would allow to build up the electric power by 1.1% (from 104.3 to 105.4%) without raising the core power, thanks to the better turbine efficiency associated with the higher steam pressure delivered by the new SG.

4.2. Detailed uprating study

A detailed uprating study to increase the core power is now in progress in parallel with the construction of the new SG. This study comprises 3 successive phases.

Phase 1 is a feasibility study to evaluate the maximum possible power, combined if possible with an increase in flexibility for fuel management and plant operation. This phase is further subdivided into phases 1A and 1B. Phase 1A uses thermal hydraulic considerations to tentatively determine a new operating point. Phase 1B includes analysis of the most limiting accidents, like large break Loss Of Coolant Accident (LOCA) or DNB-limiting events, to confirm or adjust the operating point selected at the end of phase 1A. The goal of phase 1 is to determine a reliable set of parameters describing the new operating conditions and associated safety margins for which the complete safety analysis has to be performed.

Phase 2 encompasses the safety analysis of all events influenced by the new operating condition.

Phase 3 includes the work necessary to update the technical specifications, safety analysis report and setpoints.

Based on the results presently available, it is expected that the power of the Doel 3 plant will be increased to 107-110% of the present power. This uprating can be achieved with only minor modifications to the turbine. No problems are expected either with the electrical equipment designed from the start for 110% power.

9.4-11/17
5. SELECTION OF THE TUBE MATERIAL FOR THE NEW SG

When SG replacement has been decided, the most important choice facing the Utility is probably the material selection for the tubes: Inconel alloy 690 or Incoloy alloy 800?

A comparative evaluation was performed for Doel 3 [7] where up to now the only corrosion problem is PWSCC.

For this type of corrosion numerous laboratory tests have shown both candidate materials to be immune.

As far as ODSCC is concerned, both alloys are generally good and superior to alloy 600.

In chlorides alloy 690 is excellent whereas unpeened alloy 800 may crack if high oxygen or copper is present. However the concentrations required for cracking are not likely to occur in a SG.

In caustic environment, alloy 690 is good for low (< 5 %) and high concentrations (= 50 %). Alloy 800 is less good for high concentrations (> 40 %) but here again such concentrations are not expected in the SG.

On the other hand recent unfavourable laboratory results have been obtained for alloy 690 in caustic environment contaminated with lead.

Globally alloy 690 is slightly superior to unpeened alloy 800 for SSCC, based on laboratory tests.

Peening alloy 800 will remove all concerns for SSCC except to some extent in the roll transition areas where rolling reduces the effectiveness of the peening operation.

As far as field experience is concerned, it is very limited for alloy 690. Alloy 800 has good operating experience, the only significant problem being wastage associated with a secondary side phosphate chemistry.

Some other aspects have also to be taken into account:

- the higher thermal expansion of alloy 800 which requires special precautions during manufacture;
- the somewhat lower than expected heat exchange capacity shown by replacement SG tubed with alloy 690, a point that is still under investigation.

As a conclusion, both materials are acceptable. Peened alloy 800 was selected for Doel 3 based on a larger field experience and commercial considerations.
6. CONCLUSION

The integrated Belgian approach to SG defect management results in a global optimization of the plant operation in terms of safety, reliability and economy.

Applying this approach to the particular problem of PWSCC required a number of original developments in such fields as non destructive testing and probabilistic modelling of SG behaviour. It allowed after performance of a long term study to conclude that the most cost-effective strategy for Doel 3 was the immediate replacement of the SG combined with an uprating of the plant.

This study included a thorough comparison of the 2 candidate alloys for the tube material of the new SG: Inconel alloy 690 and Incoloy alloy 800. Finally the latter was selected in the peened condition because of its larger service experience and of favourable economic arguments.
Deterministic Approach

- Analysis of present and potential problems
- Technical and financial assumptions

Selection of scenarios

- Evaluation of the plugging rate
- Risk analysis
- Cost analysis

Choice of best strategy

Figure 1
Figure 3
REFERENCES


9.4-17/17