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Proceedings of the

IAEA Specialists' Meeting on Cracking in LWR RPV Head Penetrations

Held at
ASTM Headquarters
Philadelphia, Pennsylvania
May 2-3, 1995

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

Proceedings prepared by
Oak Ridge National Laboratory

UNITED STATES NUCLEAR REGULATORY COMMISSION

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Proceedings of the

IAEA Specialists' Meeting on Cracking in LWR RPV Head Penetrations

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May 2–3, 1995

Manuscript Completed: February 1996
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ABSTRACT

This report contains 17 papers that were presented in four sessions at the IAEA Specialists' meeting on *Cracking in LWR RPV Head Penetrations* held at ASTM Headquarters in Philadelphia on May 2-3, 1995. The papers are compiled here in the order that presentations were made in the sessions, and they relate to operational observations, inspection techniques, analytical modeling, and regulatory control. The goal of the meeting was to allow international experts to review experience in the field of ensuring adequate performance of reactor pressure vessel (RPV) heads and penetrations. The emphasis was to allow a better understanding of RPV material behavior, to provide guidance supporting reliability and adequate performance, and to assist in defining directions for further investigations. The international nature of the meeting is illustrated by the fact that papers were presented by researchers from 10 countries. There were technical experts present from other countries who participated in discussions of the results presented. The IAEA issued a Working Material version of the meeting papers (IAEA IWG-LMNPP-95/1), and this present document incorporates the final version of the papers as received from the authors. The final chapter includes conclusions and recommendations.

DISCLAIMER

PROCEEDINGS OF THE
IAEA SPECIALISTS' MEETING ON
CRACKING IN LWR RPV HEAD PENETRATIONS

May 2-3, 1995

TABLE OF CONTENTS

FOREWORD

This report provides the proceedings of the Specialists' Meeting on Cracking in LWR RPV Head Penetrations that was held in Philadelphia, Pennsylvania, on May 2-3, 1995. The meeting was sponsored by the International Atomic Energy Agency's (IAEA) International Working Group (IWG) on Life Management of Nuclear Power Plants (LMNPP). The IWG is chaired by Acad. L. M. Davies of the U.K. and Dr. L. Ianko is its Scientific Secretary. The IWG placed this meeting on the 1995 calendar during its most recent meeting in Vienna when it examined priority topics for the next two years. The meeting was chaired by Dr. J. Strosnider from the U.S. and Mr. Davies.

The meeting was attended by 39 participants from 12 countries. It was designed to allow experts from these countries to share and review experience in the field for ensuring adequate performance of reactor pressure vessel (RPV) heads and penetrations. The scope included:

1. Overviews: Current issues and problem areas, their consequences and remedies
2. Operating Experience: Results of inspections, interpretations, and applications
3. Monitoring and inspection Methods: Techniques used, qualification of techniques, inspection reliability, and field experience.
4. Susceptibility Evaluations: Models for predicting susceptibility and operating experience
5. Structural Integrity Assessments: Flaw acceptance criteria, crack-growth rates, boric acid corrosion evaluations and risk/consequence evaluations
6. Repair and Mitigation: Techniques, qualification and field experience

The meeting was conducted under four sessions that emphasized:

1. Specific RPV head designs, materials and other characterizing parameters
2. Inspection requirements and practices
3. Crack-growth observations and modeling
4. Environment and behavior mechanisms

The Organizing Committee for this meeting was:

C. E. Pugh, Co-Chairman, USA (Oak Ridge National Laboratory)
M. B. McNeil, Co-Chairman, USA (U. S. Nuclear Regulatory Commission)
S. J. Ranney, Secretary, USA (Oak Ridge National Laboratory)
L. M. Davies, UK (IWG Chairman)
L. Ianko, IAEA (IWG Scientific Secretary)
J. R. Strosnider, USA (U. S. Nuclear Regulatory Commission)

A summary session produced a set of consensus conclusions and recommendations, which are expected to be useful in defining further investigations within IAEA member states. This report records the papers presented, and the Table of Contents shows the papers by session. Since specialists' meetings are aimed primarily at exchanging information between participants at the meeting, a few contributors were not in a position to provide full paper manuscripts. In some of those cases, copies of the visuals used in their presentation are included.

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ACKNOWLEDGMENTS

Claud E. Pugh
Oak Ridge National Laboratory

To open the meeting, the Organizing Committee welcomed everyone to the USA, to Philadelphia, and to the ASTM Headquarters. Appreciation was expressed to each contributor for taking the time to participate in the meeting and prepare a technical manuscript for the meeting proceedings. The attendees were advised that the proceedings would be published as a NUREG report by the Oak Ridge National Laboratory and the U. S. Nuclear Regulatory Commission. The final manuscript for each paper was solicited within three months from the date of the meeting. Dr. L. Ianko would also publish through the IAEA a Working Material version of the papers as available at the meeting. (This was accomplished in August 1995 when IAEA IWG-LMNPP-95/1 was issued.)

Special acknowledgments are highlighted for the following contributions:

1. Appreciation is expressed to each contributor and their employer for the time and effort to attend the meeting, prepare talks, and write manuscripts. These efforts made the meeting a success.

2. A special acknowledgment is given to the American Society for Testing and Materials (ASTM) who made facilities at their headquarters in Philadelphia available for this meeting. Mr. Ken Pearson and the ASTM staff showed extraordinary courtesy to the meeting, coordinated services needed to ensure that the meeting proceeded smoothly and had supporting information available in a timely manner.

3. The U. S. Nuclear Regulatory Commission who co-organized and underwrote much of the cost of the meeting. In this regard, M. B. McNeil served as co-chairman of the Organizing Committee, M. E. Mayfield's programs provided the support and J. R. Strosnider served as a technical co-chairman of the meeting.

4. The staff of the Oak Ridge National Laboratory provided assistance in many ways, and this included soliciting expert participants, coordinating meeting arrangements at ASTM, making hotel accommodations, preparing handouts, and preparing communications.

5. Thanks go to the IAEA IWG/LMNPP for having the insight to include this specialists' meeting on their calendar. The efforts of Leonid Ianko (IWG Scientific Secretary) and L. M. Davies (IWG Chairman) were instrumental in coordinating member-state participation. Mr. Davies also served as technical co-chairman of the meeting.

Specialists Meeting
on
Cracking in LWR RPV Head Penetrations
2-4 May 1995, Philadelphia, USA

WELCOMING ADDRESS

by L. Ianko

International Atomic Energy Agency

Mr. Chairman, Ladies and Gentlemen,

It is my pleasure to welcome you on behalf of the International Atomic Energy Agency to this Specialists' Meeting on "Cracking in LWR RPV Head Penetrations".

This meeting is being held within the framework of the IAEA International Working Group on Life Management of Nuclear Power Plants (IWG-LMNPP) and is convened with the support of the US Nuclear Regulatory Commission and the Oak Ridge National Laboratory.

Let me first express the Agency's gratitude to the Government of the USA for hosting this meeting, and for providing the opportunity to participants from all over the world to exchange information and experience.

The purpose of the meeting is to review experience in the field of ensuring adequate performance of reactor pressure vessel (RPV) heads and penetrations, including current issues and problem areas, their consequences and remedies, operating experience, monitoring and inspection methods, structural integrity assessments, repair and mitigation.

The main objective is to provide a forum for exchange of information among the participating experts from Member States through their interactions both at this meeting and later through the publication of the meeting's proceedings which will reach a much wider audience. I believe that the information exchange in the coming days will make an important contribution to reaching our common goal of achieving a high level of nuclear performance and safety. The results of this meeting should help to clarify the main issues for future work, both for you and for us in the IAEA.

Concluding, I wish to express our gratitude to the US Nuclear Regulatory Commission and the Oak Ridge National Laboratory for all arrangements which have been made, especially to the representative of the USA to the International Working Group on Life Management of Nuclear Power Plants Dr. C. Pugh, also to Dr. M. McNeil and Dr. J. Strosnider from the US NRC and other colleagues who did an excellent job in organizing the meeting.

welcome.usa

Opening Address

L.M. Davies

Ladies and Gentlemen:

The lifetime of nuclear power plants can be defined in many ways and these include depreciation, design, economic, technical regulatory, and political. I would suggest that while different criteria are used for different purposes, it is more appropriate to discuss NPP life in terms of its "technical life". This is the period that a plant continues to meet its safety and performance criteria. When discussing NPP lifetime, it is also worth considering whether a particular issue is of a generic plant nature or whether it relates to a specific plant.

At this particular meeting, I hope that my prejudiced view will be enforced, i.e., that the cracking of LWR RPV head penetrations is an issue which does not have direct implications for safety or plant lifetime. However, monitoring of the situation by inspection and mitigation by repair or replacement will be part of a maintenance strategy which will ensure the plant reliability for the generation of electricity.

It is also opportune to comment on the expression "Plant Lifetime Extension". This carries the implication, in human terms, that there is "life after death". So it is a misnomer and should be avoided. For many plants it is Plant Life Assurance (PLA) that is sought against a Plant Technical Life Assessment (PTLA). So, logically, Plant Lifetime Management in terms of PTL is a constructive approach.

2. Chairmen

Session Chairmen have been identified for this meeting, and it will be part of their task to prepare a short report of their session summarizing the papers and drawing out any major conclusions and recommendations. We will consider these reports and add any general conclusions and recommendations at the plenary session.

3. Agenda

There have been some small changes to the agenda and these will be incorporated and the agenda will be reissued. Papers are to be presented in 45 minutes which will include discussion.

4. Publications

The IAEA will publish the proceedings of this meeting on a timescale of about one- one and one-half months. It is therefore important that manuscripts are handed in before the end of this meeting.

5. Final

In thanking our hosts for this meeting, I wish you "a good meeting".

Thank you.

There are a number of "lives": depreciation; design; economic; technical' regulatory; political.

Technical Life refers to the period that a plant continues to meet its safety and performance criteria.

Economic Life results from a least cost comparison between different methods of electricity generation- or between the net present value of the future costs and benefits of a new capacity and the further operation of a current unit.

Regulatory Life refers to the license duration. Some countries have time-limited licenses; some do not.

Ageing has been defined as a continuous time dependent degradation of materials due to actual operating conditions, which include "normal" operation and transient conditions (but, excluding Design Basis Accidents and Beyond Design Basis Accidents).

Ageing leads eventually to deterioration of various features of the plant - as built and designed - in particular with regard to material properties - and is manifested in design margin reduction as time passes. (see schematic)

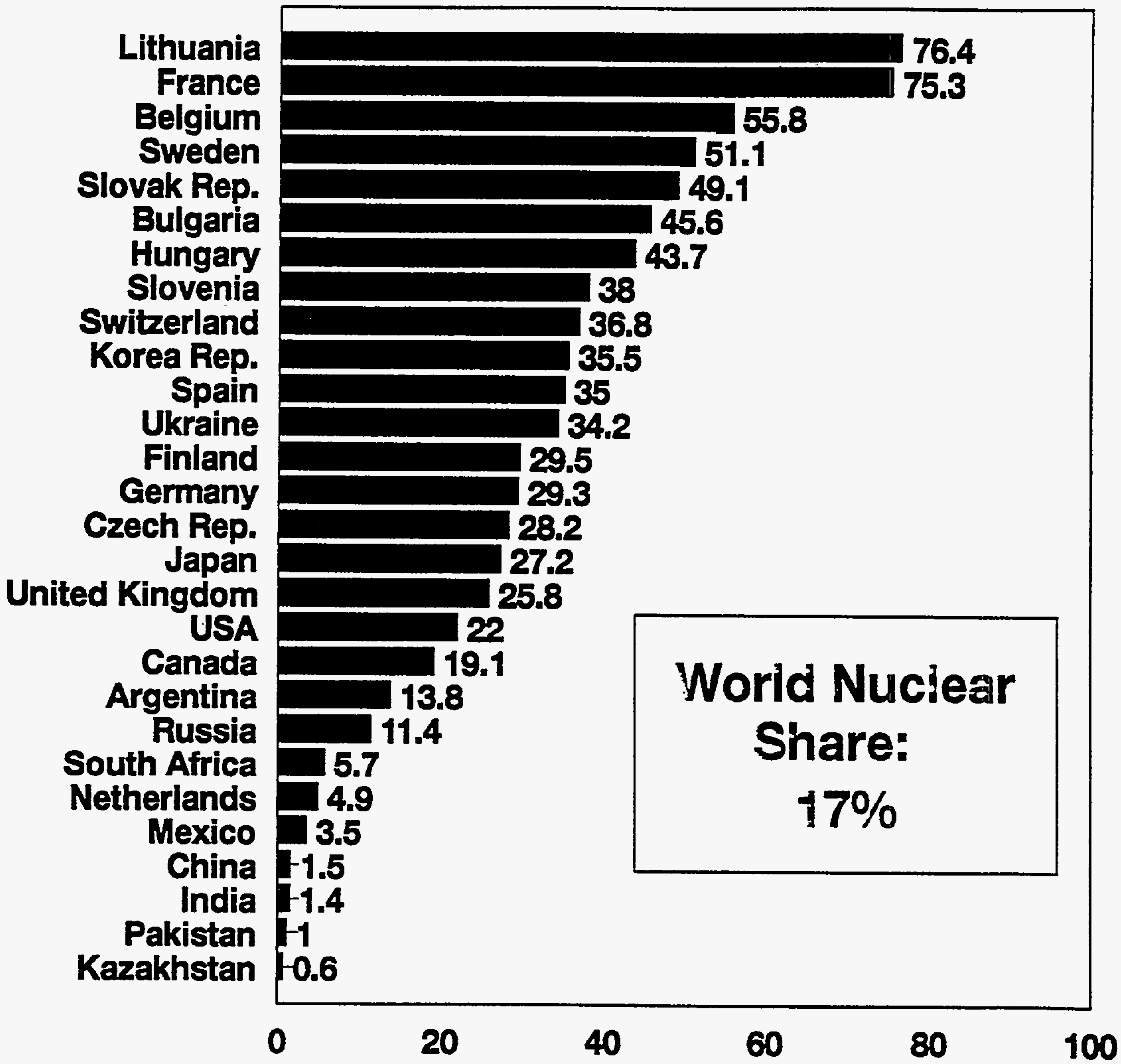
Depending on the degree of degradation the required safety margins (and reliability) could be reduced.

The objective of a plant life assurance or a plant life management programme is to achieve high availability and safe operation as long as the plant is economically viable.

Remember, the purpose of a nuclear power plant is to produce electricity!!

Further reading: M.E. Lapides, "Making Decisions About Plant Life Extensions" Nuc. Eng. Int., August 1989
L.H. Geraets, Nuclear Europe Worldscan 9-10/93
IWG-LMNPP-94/6 IAEA publication 1994 - L.M. Davies, A.D. Boothroyd, L. Ianko, "Aspects of Plant Life Management" IAEA Vienna, "Nuclear Power Option - Conference. Published IAEA March 1995

Nuclear Share of Electricity Generation in the World (as of 31 December 1994)



International Atomic Energy Agency
Vienna, Austria

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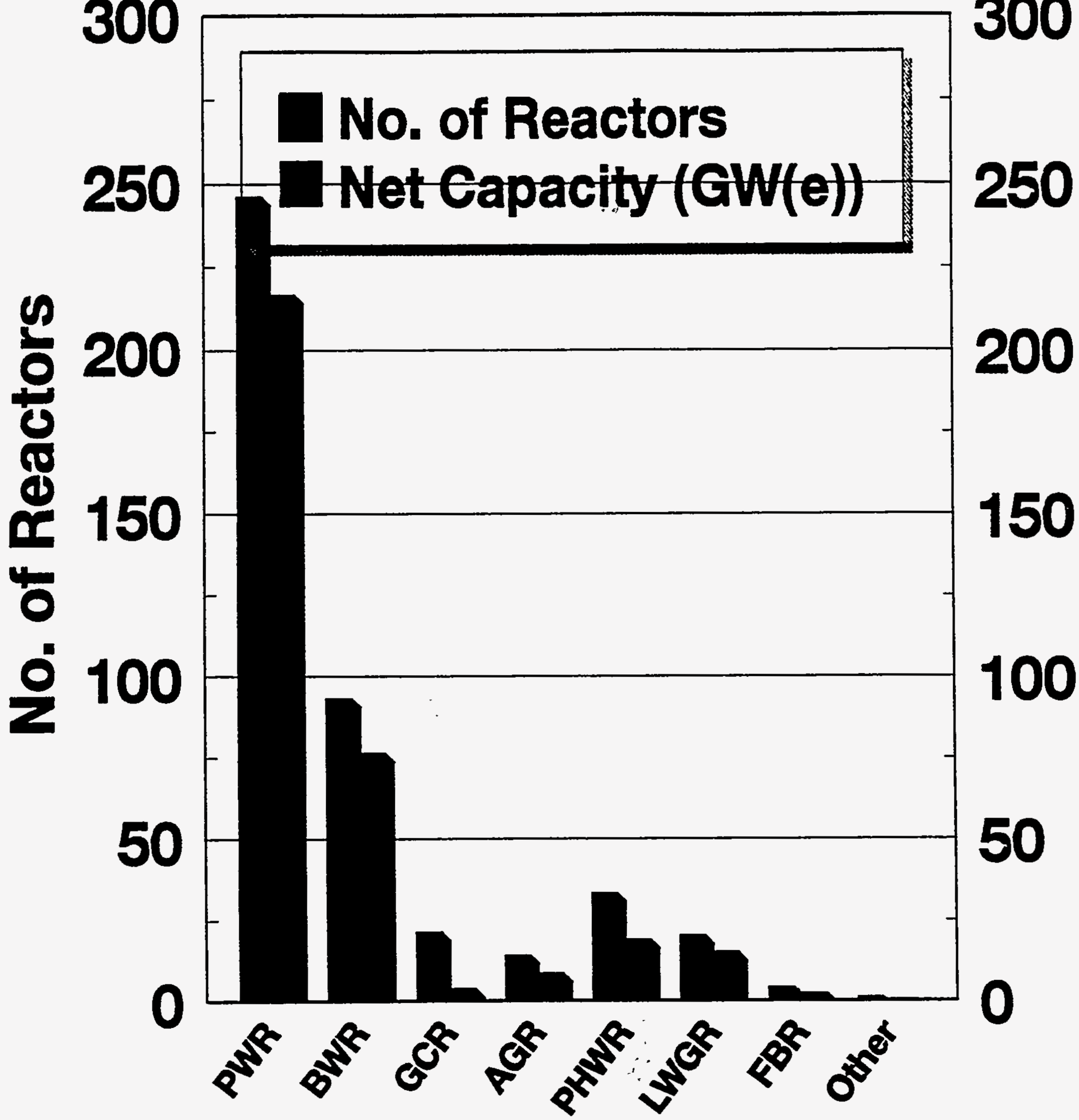
1995-05-17

Nuclear Power Plants in Operation as of 31 December 1994



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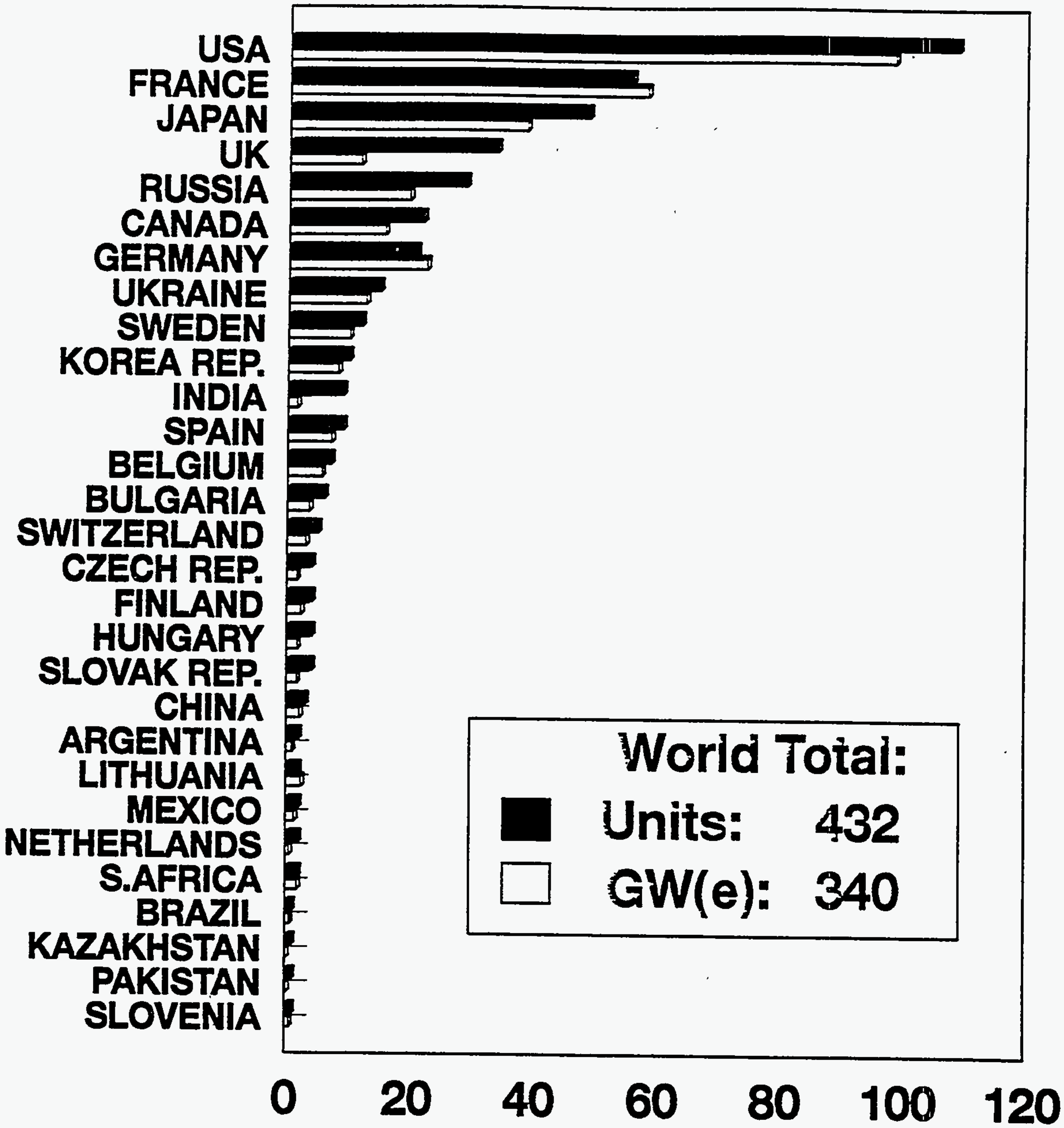
1995-05-17

Nuclear Power Plants in Operation in the World (as of 31 December 1994)



Note: Taiwan, China 6 units, 4.9 GW(e)

International Atomic Energy Agency
Vienna, Austria

1995-05-17

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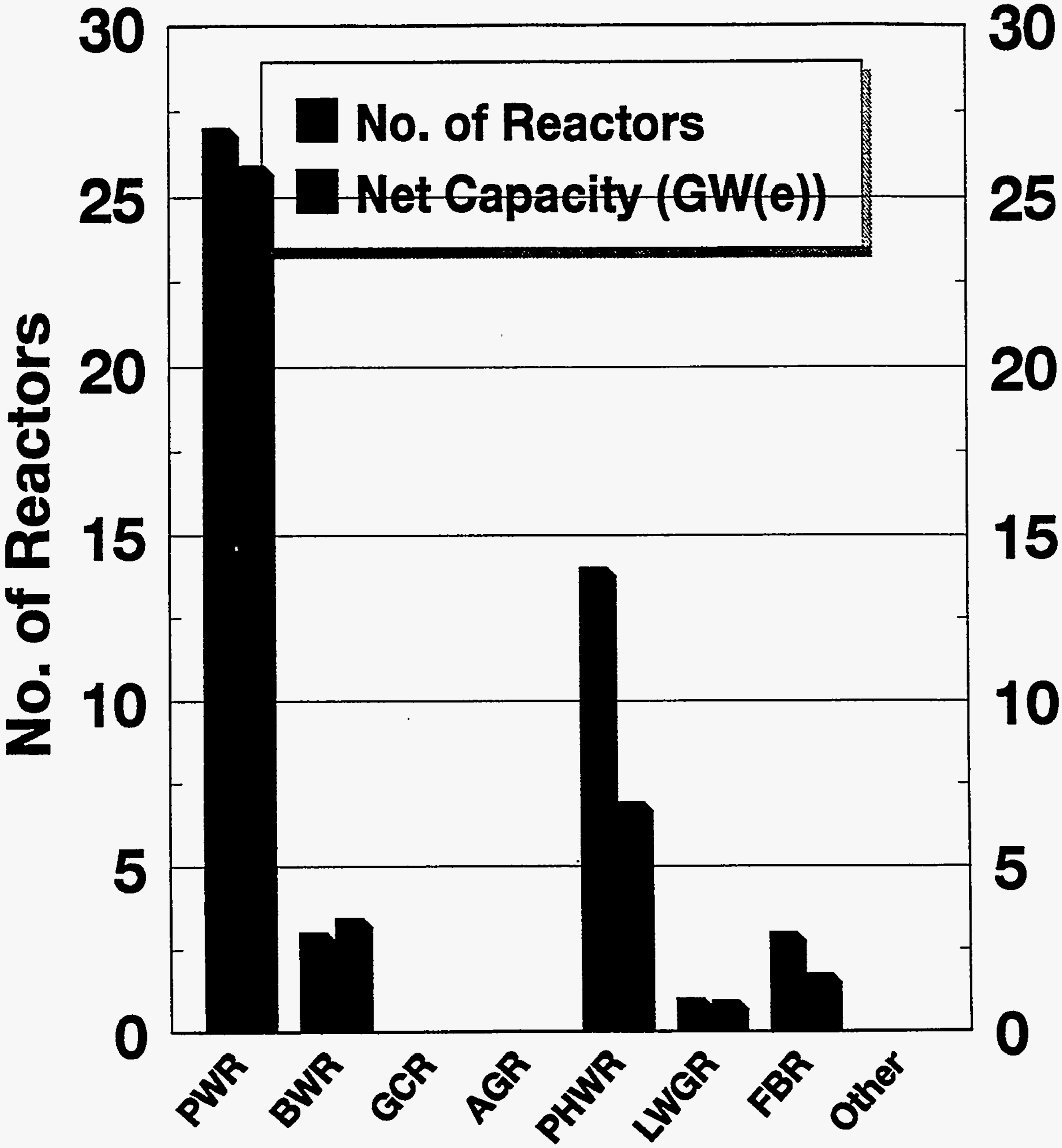
Nuclear Power Plants Under Construction as of 31 December 1994



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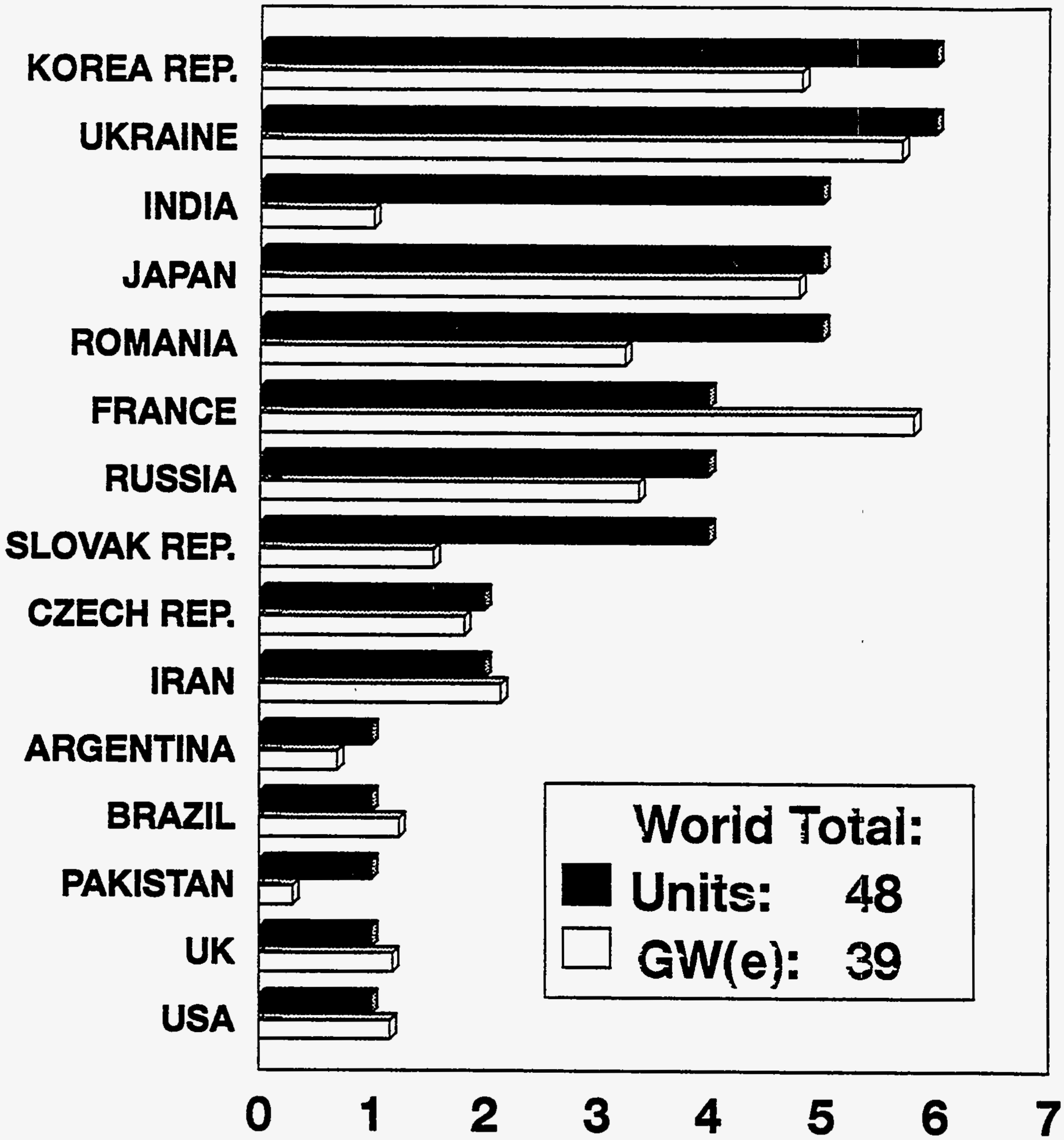
1995-05-17

Nuclear Power Plants Under Construction in the World (as of 31 December 1994)



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Vienna, Austria

1995-05-17

IAEA

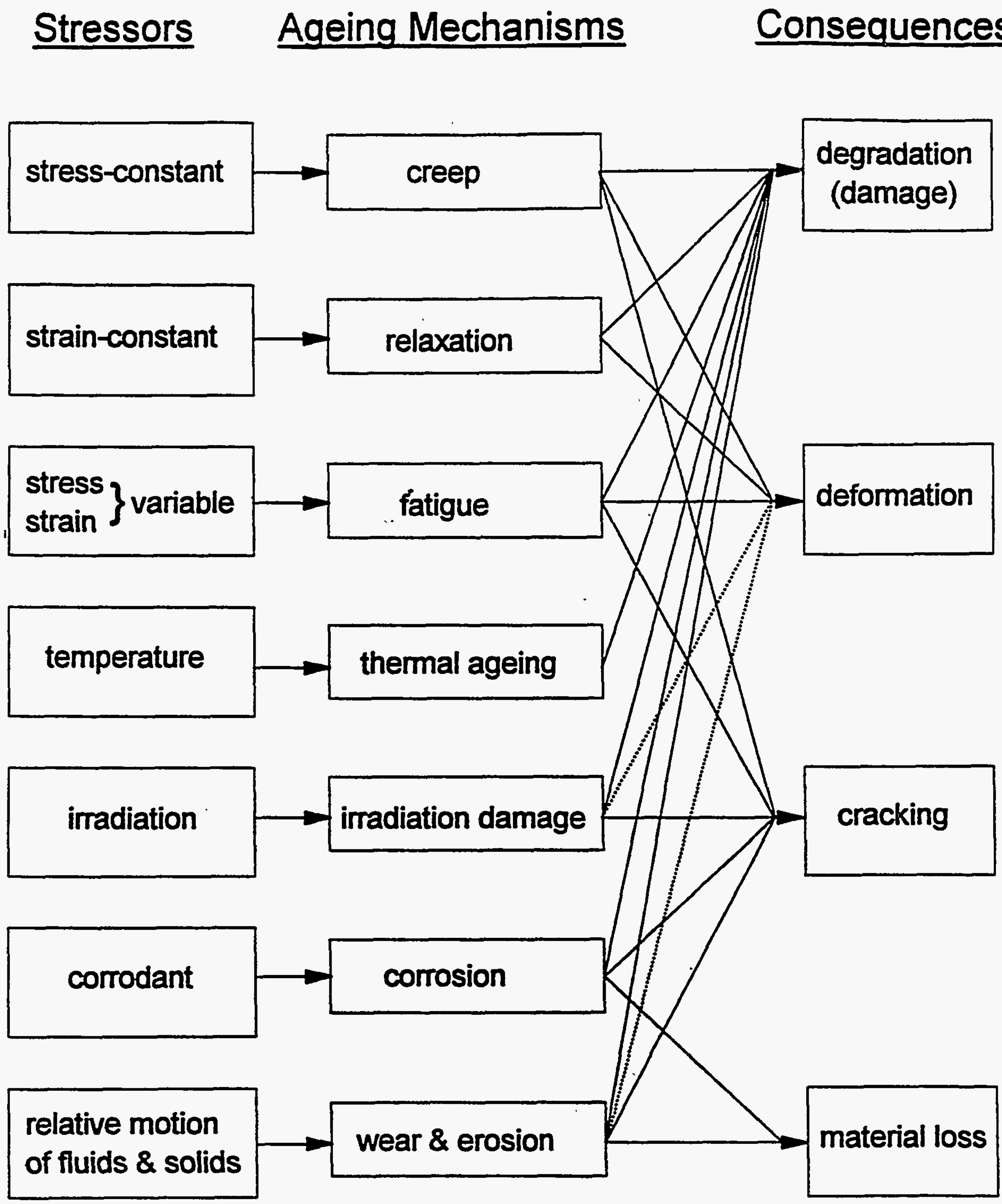
Understanding phenomena and degradation mechanisms is an extensive area of work- particulularly when there are interactive effects.

The initial focus of the IAEA programme has been the major degradationmechanism of neutron irradiation embrittlement of the RPV. The scope of the programme has been expanded

Thermal degradation, corrosion and fatigue are also very significant

NDE and FM together with the plant stress state are are also major contributors to the process of **assessing structural integrity**

Fig. 2: AGEING FACTORS, BASIC AGEING MECHANISMS AND CONSEQUENCES



Note: The links indicated by dashed lines may only apply in particular circumstances

AGEING MANAGEMENT consists of the following four elements:

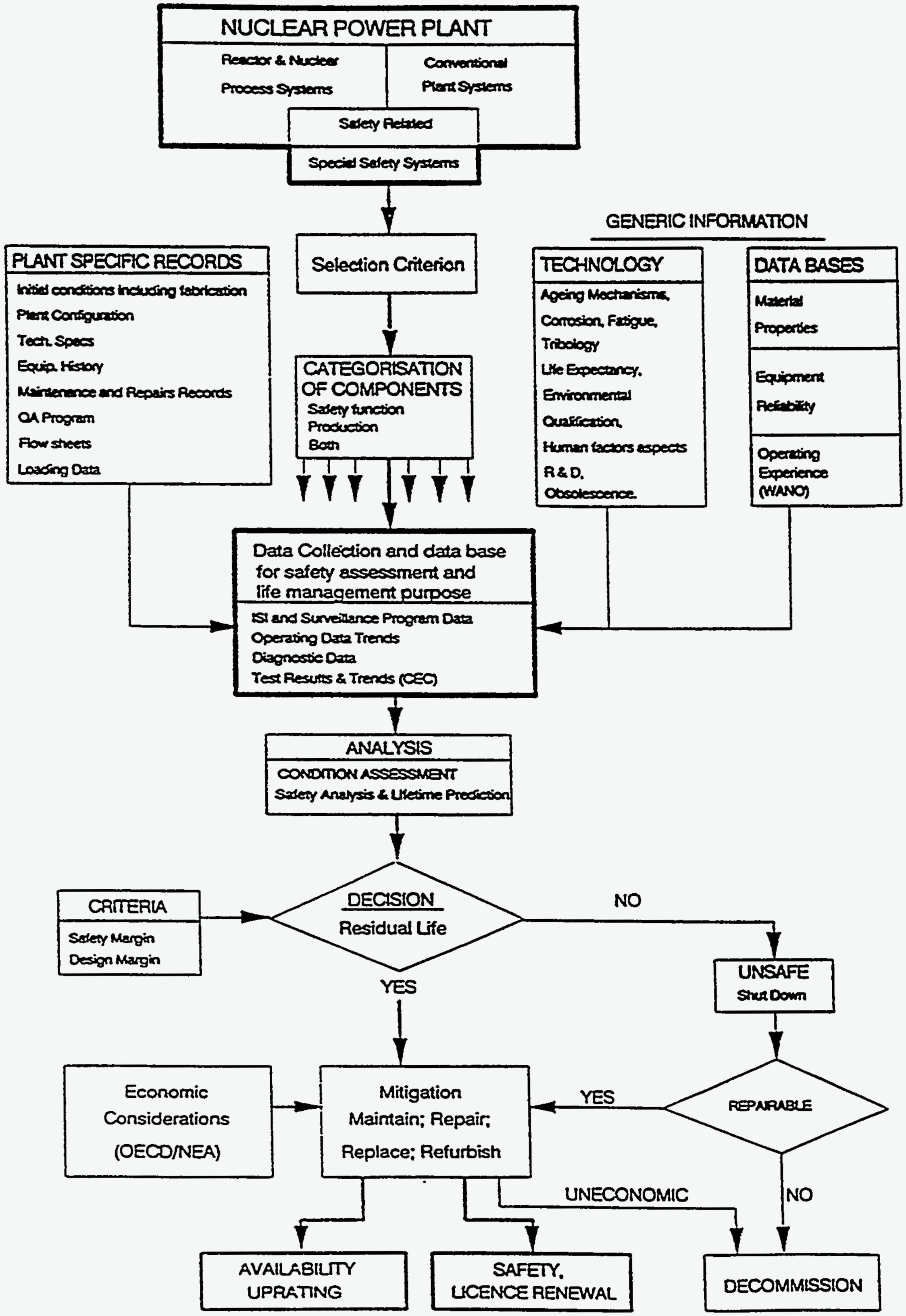
- Understanding and anticipating ageing effects
- Condition monitoring
- Assessment of remaining useful life.
- Mitigation or other action

DATA AVAILABILITY *is a key aspect and the quality and availability of relevant information is directly related to the quality of decisions on service life and the reliability of nuclear power plants.*

The **DATA REQUIRED** fall under the following headings:

- Baseline
- Operating history
- Maintenance
- Technology developments
- Material properties

N.P.P. LIFE MANAGEMENT PROCESS



Approach to NPP Life Management

The process of **PLANT LIFE MANAGEMENT** involves the **identification of key components** followed by the **collection and analysis of data** to determine the **condition and residual life** of these components

UNFAVOURABLE → Repair, Replacement or Decommissioning

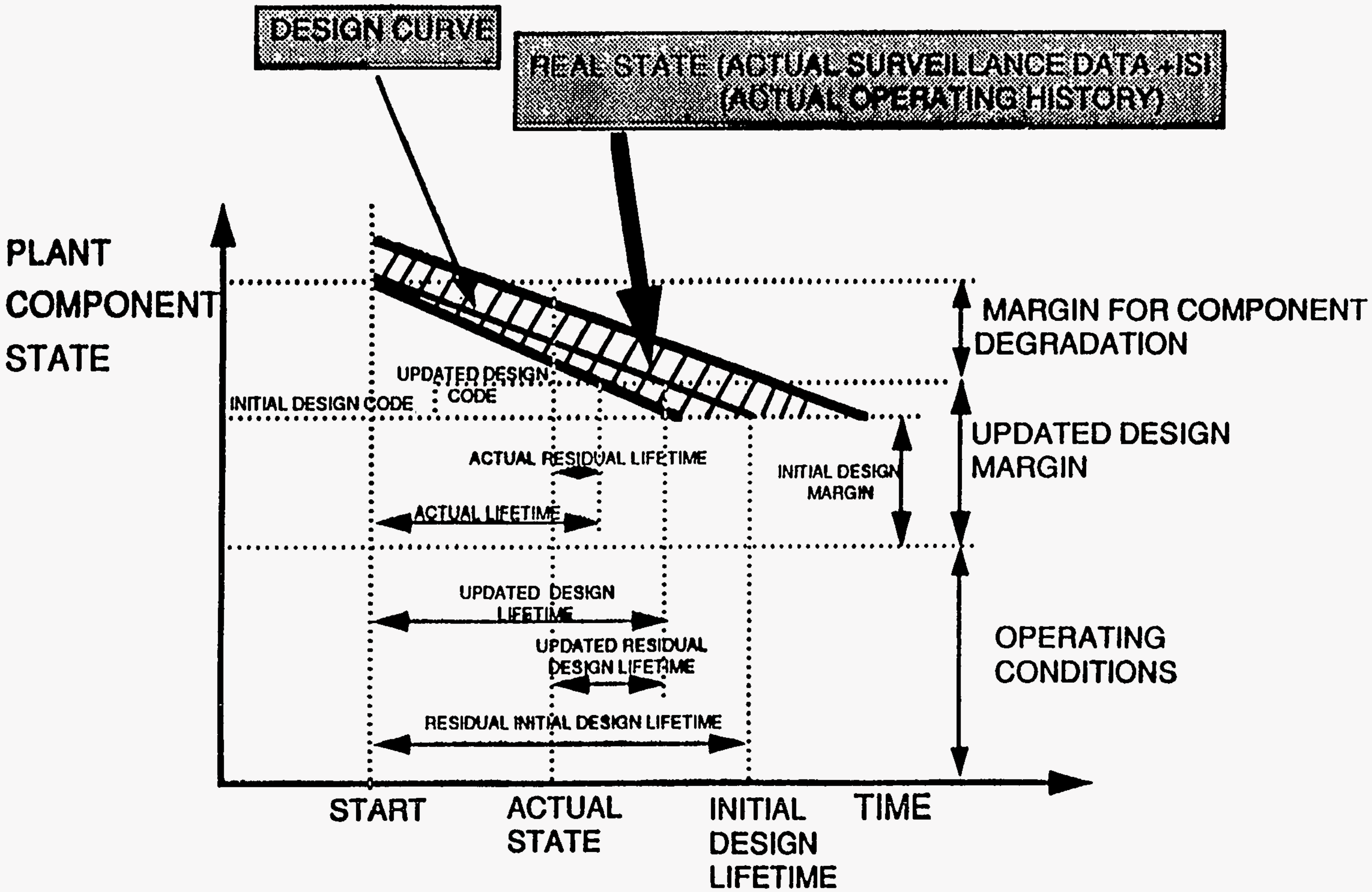
FAVOURABLE → can define a suitable maintenance programme

With the implementation of a pro-active or preventative maintenance programme continued, reliable, safe availability and economic production is achieved.

→→→→ (see schematic)

The process is the responsibilty of the plant Owner/operator and the safety aspects will be reviewed by the Regulator.

The objective of the IAEA programme in this area is to be supportive in the aim of achieving maximum productive service life and in particular to advance the understanding of degradation mechanisms encountered in NPP life



DESIGN CURVE
REAL STATE (ACTUAL SURVEILLANCE DATA +ISI
(ACTUAL OPERATING HISTORY)
PLANT
COMPONENT
STATE
MARGIN FOR COMPONENT
DEGRADATION
UPDATED DESIGN
CODE
INITIAL DESIGN CODE
UPDATED DESIGN
MARGIN
ACTUAL RESIDUAL LIFETIME
INITIAL DESIGN
MARGIN
ACTUAL LIFETIME
UPDATED DESIGN
LIFETIME
UPDATED RESIDUAL
DESIGN LIFETIME
OPERATING
CONDITIONS
RESIDUAL INITIAL DESIGN LIFETIME
START
ACTUAL
STATE
INITIAL
DESIGN
LIFETIME
TIME

IAEA SPECIALISTS' MEETING

on

CRACKING IN LWR RPV HEAD PENETRATIONS

MAY 2-3, 1995

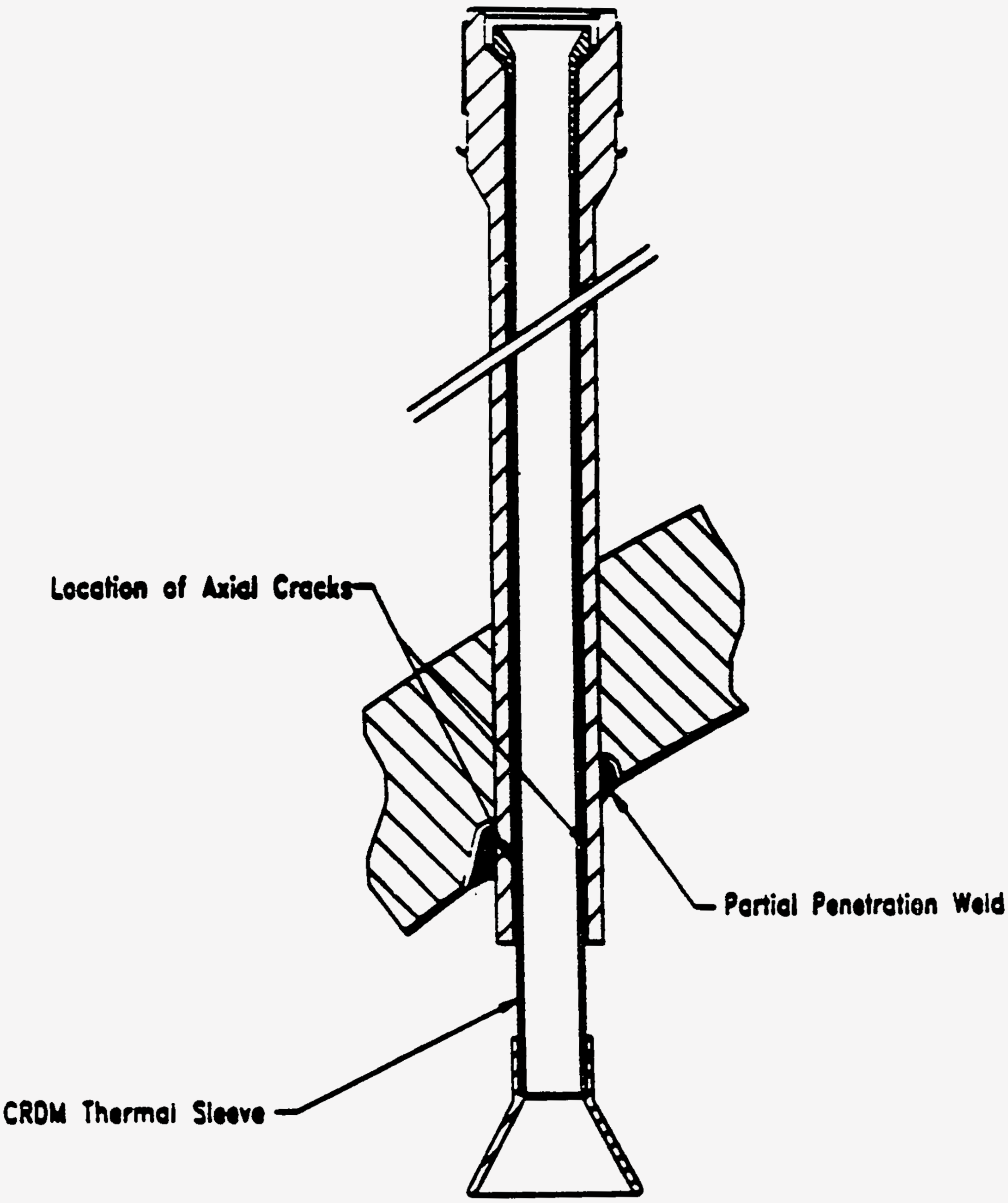
PHILADELPHIA, PENNSYLVANIA

OPENING COMMENTS

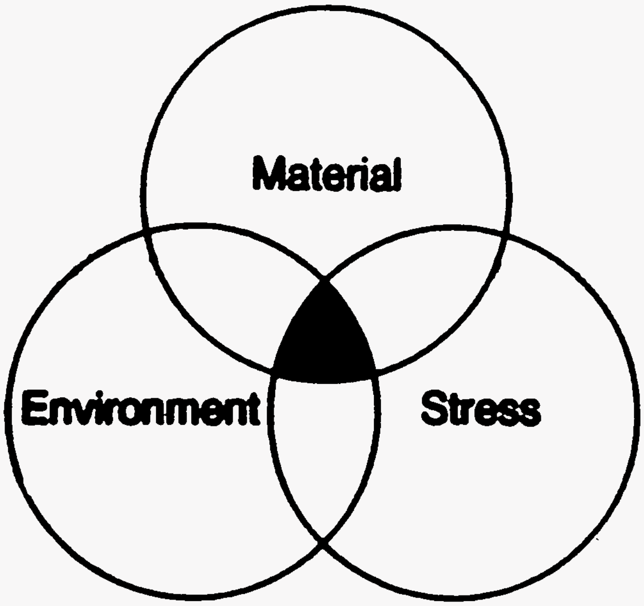
JACK STROSNIDER, CHIEF
MATERIALS AND CHEMICAL ENGINEERING BRANCH
U.S. NUCLEAR REGULATORY COMMISSION

R/V CLOSURE HEAD ALLOY 600 PENETRATION ISSUE

PENETRATION TUBE



INTERGRANULAR STRESS CORROSION CRACKING



TOPICS OF INTEREST

1) INSPECTION PROGRAM

2) INSPECTION METHODS

3) CRACK INITIATION AND GROWTH

4) REPAIR CRITERIA

5) LEAK DETECTION METHODS

6) REPAIR METHODS

7) MITIGATION METHODS

STAFF VIEW OF SAFETY SIGNIFICANCE

- PWSCC cracking of Inconel 600 has a minimal safety impact because all reported cracking has been short and axially oriented.
- Potential exists for cracking in a large number of CRD housing penetrations
- Concern about corrosion from boric acid deposits on head, early leak detection is important since corrosion rates may be high
- Unique stress distribution could cause circumferential cracking
- Prudence suggests an orderly inspection program
  - Defense in Depth
  - GDC 14 - Low Probability of RCPB leakage

PWSCC OF INCONEL 600 IN PWRs

* PWSCC IDENTIFIED TO NRC COMMISSIONERS AS AN EMERGING TECHNICAL ISSUE IN 1989

* COMPONENTS OTHER THAN CRDM PENETRATIONS ARE AFFECTED

The Activities of the IAEA International Working Group on Life Management of Nuclear Power Plants (IWG-LMNPP) in 1994 and Plans for 1995-1996

All components of nuclear power plants are subject to some form of ageing degradation, however, the rates of degradation and therefore component lifetimes vary considerably. Component deterioration due to ageing may significantly prejudice plant reliability and capacity factors unless failures are anticipated and prevented by timely maintenance, repair or replacement of components. On the other hand many components operate at a significant margin within design limit criteria, which are themselves significantly within safety limits. Components subject to ageing phenomena must be closely monitored if high performance in terms of reliability and availability, accompanied with low operating and maintenance costs, are to be achieved. It has to be ensured that the continued operation of, particularly, older plants does not pose an undue risk to public health and safety owing to obsolescence of equipment or of safety standards and requirements to which they were built.

The objective of the IAEA's International Working Group on Life Management of Nuclear Power Plants (IWG-LMNPP) is to provide the Member States with information and guidance on design aspects, material selection, testing, maintenance, monitoring and mitigation of degradation related to major components with the aim to assure high availability and safe operation of NPPs. Technical documents and reports on proceedings of specialists meetings on many of these topics have been produced or are in preparation.

Coordination of research aimed at understanding the phenomena which occur and the consequent degradation mechanisms is an extensive field of the IWG-LMNPP activities. Radiation embrittlement of reactor pressure vessel steels has been a major subject of concern. Thermal degradation, corrosion and fatigue are also considered to be very significant. In the monitoring field, non-destructive examination techniques and fracture mechanics are areas included in the IWG-LMNPP plans.

The IWG includes representatives from 23 countries and 2 international organizations (OECD/NEA and EC) - see Appendix 1.

According to the Terms of Reference approved by the IAEA Director General, the IWG-LMNPP should provide the Secretariat of the IAEA with advice and recommendations on the Agency's activities and forward programmes in this area by means of Specialists meetings, training courses, etc., when they have particular relevance to reliable plant life management, and specifically, on the priority, scope and content of publications in the form of guides and manuals and meetings to be organized and sponsored by the Agency. The scope of the IWG activities include the following aspects:

- Design
- Materials
- Fabrication
- Monitoring, testing, inspection and data bases of their results
- Information on service and test conditions
- Degradation mechanisms, their significance and mitigation
- Assessment and means of plant life management

The Terms of Reference are included as Appendix 2 and the list of IWG priorities as Appendix 3.

The major role that the International Working Group has been taking is in the provision through research programmes of generic information of Nuclear Power Plant Life Management. In the technology area this consists of a wide variety of subject areas. Proceedings of Specialists' Meetings on many of these topics have been produced or are in preparation. A list of meeting topics held are listed in Appendix 4.

In accordance with the recommendations made by participants at the meeting in February 1994, there were a number of meetings to be organised within the framework of the IWG-LMNPP activities.

- Specialists Meeting on "Erosion and Corrosion of Nuclear Power Plants Materials", 19-23 September 1994, Kiev, Ukraine. The objective was to review the various techniques applied, results obtained and the efficiency of the predictive models. Design criteria for repair and replacement policies as well as cost evaluation were considered. The meeting was a continuation of the 1988 specialists meeting on the same subject and aimed at technical personnel from utilities, service companies, regulatory bodies and appropriate research institutes. The proceedings were published by the IAEA.

- Specialists Meeting on "Technology for Lifetime Management of Nuclear Power Plants", 15-17 November 1994, Japan.

The purpose of the meeting was to provide an international forum for discussion on recent results in research and utility practice in the field of methodology for assessment methods and technology concepts for lifetime management of key components of nuclear power plants. Topics covered included analysis of main degradation processes in key components, modelling of properties degradation in key components, methods for residual life assessment and their limitations, mitigation measures and planned technologies for corrective actions, national approaches to NPP life management including the general strategy to specific components. The proceedings were published by the IAEA.

- The final Co-ordinated Research Programme (CRP) meeting on "Optimization of Reactor Pressure Vessel Surveillance Programmes and Their Analysis" (Phase III) was held in Vienna from 22 to 23 November 1993.

  Chief Scientific Investigators from Argentina, Austria, Belgium, Czech Republic, France, Finland, Germany, Hungary, India, Japan, Russia, Spain, and UK, and observers from Finland, France, Hungary and Spain attended the meeting and presented reports of the national contributions to the CRP. Final reports were provided and some data adjusted. The action plan for completing the programme was agreed. It was noted that this multi-million dollar effort has produced a wealth of unique data and a supply of special and representative steels both irradiated and unirradiated for further study if agreed. Several bilateral efforts will continue and data will be made available to the IAEA as a fulfillment of the condition that data from special steels will be provided. Further, it was agreed that the representative steel (typical of older steels) which is ware-housed in Switzerland will be surveyed for inventory and on agreement with the Agency will be reinstated in order to make this steel available to any IAEA member nation for new or added vessel surveillance as a standard reference material. The final document will be published soon.

- Joint CEC, OECD, IAEA Specialists Meeting on "Non-Destructive Practice and Results" State-of-the-Art and PISC III Results. (8-10 March 1994, Petten, The Netherlands).

  The meeting provided an opportunity for the discussion of recent results and of utility experience with non-destructive methods used for the inspection of steel components, and weldments.

  The meeting addressed, in terms of the state-of-the-art, the capability and reliability of NDE procedures applied to the major nuclear reactor components. Special emphasis was placed on NDE techniques to detect and size flaws in order to assure structural integrity during plant design life or beyond. The Proceedings were published by the EC.

- Specialists Meeting on "Advanced Structural Integrity Assessment Procedures, 14-18 March, 1994, San Carlos de Bariloche, Argentina.

  The purpose of the Specialists Meeting was to provide an international forum for discussion on recent results in research and utility practice in the field of methodology for the structural integrity assessment of components including relevant non-codified procedures. The scope of the meeting included deterministic as well as probabilistic approaches. The papers covered the following topics: leak-before-break concepts; non-destructive examination (NDE); statistical evaluation of NDE data; pressurized thermal shock (PTS) evaluation; fatigue effects (including vibration); verification qualification. The Proceedings were published by the IAEA.

The International Workshop on "WWER 440 Reactor Pressure Vessel Embrittlement and Annealing", Zavazna Poruba (Slovak Republic), 29-31 March 1994. It was organized by the Nuclear Regulatory authority of the Slovak Republic in cooperation with the International Atomic Energy Agency.

The purpose of the Workshop was to discuss the WWER 440 model 230 reactor pressure vessel integrity in terms of the measures already taken, current activities and future plans. The meeting was arranged in two parts, the Scientific programme followed by the review and revision of the IAEA Consultancy report on RPV Embrittlement and Annealing. The Proceedings of the Scientific Programme were published by the IAEA.

Another subject on the Agency's programme is the development of the International Data Base on NPP Life Management. The IAEA proposed the establishment of a data base on ageing management and life extension of NPP key components important to safety and productivity, will require great effort to be expended both from the participants and the IAEA. Because of this aspect it has been decided to build up the data base step by step and a first phase would be to elaborate the structure, the manual, data acquisition and management technology of the International Data Base on Ageing Management and Life Extension of Reactor Pressure Vessel Materials. Within the framework of the Co-ordinated Research Programme on "Optimization of RPV Surveillance Programmes" the IAEA has already established a data base which is being developed from the results derived from that programme. The IAEA is now expanding this activity by developing an "International Reactor Pressure Vessel Material Surveillance Data Base".

Further development of the data base will cover primary system piping, steel and concrete containment vessels and other concrete structures.

The benefits of the data base for ageing management are:

- The capability to make predictions of future performance and remaining service life of the reactor pressure vessel, and to provide a basis for timely implementation of mitigation measures such as flux reductions or annealing.

- The ability to identify emerging embrittlement and ageing effects before their impact on vessel safety, reliability and service life.

- To identify and evaluate the effects of operating conditions outside the range of existing data for the purpose of extrapolation and improved mechanistic modeling.

- To supplement or confirm reliability of prediction methods for vessels with inadequate or incomplete material surveillance programmes.

- To provide comprehensive information on plant operating experience of different plants and vessel materials as input for decisions concerning continued operation and license extensions of NPPs.

The IAEA will manage the International Data Base for Ageing Management and Life Extension of RPV Materials through a Custodian identified and appointed by the Agency. The Custodian acting as the agent for the IAEA will operate and maintain the data base and provide an effective interface for Member States/Participating Organizations.

The participants will include persons or organizations from Member States accredited by IAEA that provide data and are entitled to receive data base information. Each participant will be responsible for RPV embrittlement data gathering, as well as data validation and verification.

The details of the programme are given in the document "International Data Base on Ageing Management and Life Extension".

The meeting on "Guidelines for the Development of the IAEA International Reactor Materials Surveillance Database" held on 6-8 March 1995, in Moscow, considered the initiating steps for data gathering exercise, agree on a system for monitoring the progress of data acquisition and the utilization of data. This objective was appraoched by considering, in detail, the Working Draft of the Database Agreement and also the document IWG-LMNPP (94) entitled "IAEA International Database on Ageing Management and Life Extension - Database Specification". Most of the attenditon was given to considering the Working Draft of the Agreement and the revised version is now being provided for further consideration by the IAEA (Appendix 5).

- Coordinated Research Programme on Management of Ageing RPV Primary Nozzle (in the framework of the Pilot Studies of NPP Life Management which also includes studies of motor operated valves, cables, concrete structures).

The objectives of the CRP are:

- to exchange information on the state-of-the-art in assessing the remaining life of the RPV nozzles and mitigating effects of ageing,
- to perform a collaborative case study.

Organizations from Bulgaria, the Czech Republic, France, Germany, Hungary, Russia, Switzerland, the United Kingdom and the United States of America as well as the Commission of the European Communities - Institute of Advanced Materials/Joint Research Centre (IAM/JRC) are taking part in the programme. Participants are organized in a CRP network to facilitate co-operative work.

- The Advisory Group Meeting on "Maximizing Operational Lifetime - The Owner's Point of View" was held in Vienna, 20-23 February 1995.

The main purpose was to schedule and initiate actions in the preparation of a document (TRS) on the topic of NPP Lifetime Management - The Owner's Point of View.

The revised Chapter headings, scope for each Chapter and the schedule are given in Appendix 6. Suggested authors for the Chapters were considered and they would be approached by the IAEA for acceptance.

The list of meetings to be held in the framework of the IWG-LMNPP in 1995-1996 and the Information Sheets of the Specialists Meetings on Cracking in Head Penetrations and Irradiation Embrittlement and Mitigation are included as Appendices 7, 8 and 9 respectively.

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INTERNATIONAL ATOMIC ENERGY AGENCY

INTERNATIONAL WORKING GROUP ON LIFE MANAGEMENT OF NUCLEAR POWER PLANTS

ARGENTINA

AUSTRIA

BELGIUM

BRAZIL

CANADA

CZECH REP.

FINLAND

FRANCE

GERMANY

HUNGARY

INDIA

ITALY

JAPAN

KOREA, REP.OF

NETHERLANDS

RUSSIA

SLOVENIA

SPAIN

SWEDEN

SWITZERLAND

UKRAINE

UNITED KINGDOM

UNITED STATES

EU (EUROPEAN UNION)

OECD/NEA

Terms of Reference for the International Working Group on Life Management of Nuclear Power Plants (IWG-LMNPP)

These Terms of Reference were originally approved by the IWG-RRPC at its Meeting on 17-19 February 1975, modified on 14-16 March 1990 and on 7-9 February 1994 and serve as a basis for the work of the IWG. They will be reviewed on the request of working group members or the Scientific Secretary and recommendation will be made to the Director General of the IAEA on any modifications.

1. Objectives

1.1 To assist the International Atomic Energy Agency to provide its Member States with information and comments on design aspects, material selection, fabrication, testing, operational, maintenance, monitoring, and mitigation of degradation aspects related to key components with the aim to manage lives (design life as well as extended life) of nuclear power plants by having due regard for effects of components ageing and thus assuring their long-term reliable function.

1.2 To promote the exchange of information on national and international programmes and new developments and, if necessary, to stimulate co-ordinated research in the field of reactor plant and components in Member States and Organizations.

2. Scope of activities

The IWG-LMNPP should provide the Secretariat of the IAEA with advice and recommendations on the Agency's activities and forward programmes in this area by means of Specialists meetings, training courses, etc., when they have particular relevance to reliable plant life management, and, specifically, on the priority, scope and content of publications in the form of guides and manuals and meetings to be organized and sponsored by the Agency. The scope of the IWG activities include the following aspects:

2.1 Design
2.2 Materials
2.3 Fabrication
2.4 Monitoring, testing, inspection and data bases of their results
2.5 Information on service and test conditions
2.6 Degradation mechanisms, their significance and mitigation
2.7 Assessment and means of plant life management

3. Methods of work

The working group will determine its own methods of work, including frequency of regular and other meetings, preparation of Agenda, establishment of special groups, keeping of records and other procedures. The work of the working group between the regular meetings is carried out and coordinated by the Scientific Secretary taking into account the working group's recommendations and guidance. The working group normally meets at the IAEA Headquarters. It may meet from time to time away from the IAEA Headquarters to familiarize itself with activities in a member country. Special arrangements will be made to provide Secretariat services for such meetings in cooperation with the host country.

4. Organizational matters

4.1 Membership

In appointing the membership of this International Working Group the Director General will be guided by the following considerations:

1. The Working Group will include one member and not more than one alternate from each Member State which is an expert actively working in the field of life management of NPP and wishes to participate;

2. Each member and alternate will be appointed after consultation with the member's government; and

3. Members and alternates will normally serve on the Working Group for a period prescribed by their governments, preferably for a period of at least three years.

The Director General may from time to time co-opt members and invite observers from other Member States on an ad-hoc or continuing basis.

A limited number of advisers or specialists from member countries may be invited to attend regular meetings of the working group but the representation of a member country should include the member and/or his alternate.

International Organizations with interest in the same field could be invited as observers to the IWG meetings.

4.2 Chairmanship

A Chairman of the IWG is nominated by the Director General from the members of the Working Group. The chairmanship will be rotated among the members of the IWG periodically, not less frequently than every three years. The Chairman should with the assistance of the Scientific Secretary determine subjects of the meetings, chair the meetings, and conduct them along the lines of the subject. Reports on IWG activities should be reviewed before distribution.

4.3 Secretariat

The Agency provides the administrative and secretarial services required by the Working Group, including translation services, when necessary, meeting rooms, maintenance of records and the publication and distribution of documents. The Agency also provides the service of a permanent Scientific Secretary of the Working Group, who is to be in charge of the above mentioned matters.

4.4. Expenses

The respective Governments provide the Agency with experts for the IWG-LMNPP free of cost. Travel and subsistence expenses for experts are borne by the respective Governments or Organizations. Travel cost and subsistence for consultants invited to prepare a draft document or advise the Agency on special aspects of its programme will normally be borne by the Agency.

IWG-LMNPP Current Priorities

1. Radiation Damage and Annealing of RPV

   - surveillance data base
   - optimization of surveillance programmes
   - annealing

2. RPV Integrity Assessment

   - fracture mechanics
   - NDE
   - material data bases

3. Integrity Assessment of Primary Circuit

   - LBB
   - NDE
   - corrosion and water chemistry
   - monitoring (loads, water chemistry)

4. SG Life Management

   - corrosion and water chemistry
   - NDE
   - replacement and repair

5. Reactor Internals Integrity

   - radiation damage
   - corrosion (IASCC)

6. Concrete Structures Ageing

   - degradation
   - NDE

7. :Secondary Circuit Integrity

   - erosion corrosion
   - water chemistry (optimization and monitoring)

8. Plant Life Management of Other Components - Cables etc.

   Other Important Items

   - guidelines
   - codes and standards
   - quality (assurance)
   - economics

List of Meetings Held Within the IWG-LMNPP (former IWG-RRPC)

1.	3-7 October 1966	Panel on "Recurring Inspections of Nuclear Reactor Steel Pressure Vessels"
2.	2-4 October 1967, Austria	IWG on "Engineering Aspects of Irradiation Embrittlement of Reactor Pressure Vessel Steels"
3.	21-25 July 1969, Japan	"Development of Advanced Reactor Pressure Vessel Materials"
4.	9-13 February 1970	Panel on "Basic Structural Design Philosophy, Criteria and Safety of Concrete Reactor Pressure Vessels"
5.	10-12 May 1971, Austria	IWG on Reactor Pressure Vessels "Effect of Radiation and Other Time-Dependent Phenomena on Steel Pressure Vessel Integrity"
6.	29 Nov. - 3 Dec. 1971, Austria	Panel on "Non-Destructive Testing for Reactor Core Components and Pressure Vessels"
7.	2-4 May 1972, Germany	Specialists Meeting on "Assessment of Engineering Significance of Embrittlement Effects in Pressure Vessels"
8.	17-18 October 1972, Austria	IWG Consultancy on "Reliability of Reactor Pressure Components"
9.	27 Nov. - 1 Dec. 1972, Austria	Panel on "Experience and Techniques in Repair of Reactor Components"
10.	28 May - 1 June 1973, Austria	Panel on "Methods of Assessment for Assuring Nuclear Power Stations' Reliability"
11.	23-25 October 1974, Austria	CRP on "Irradiation Embrittlement of Reactor Pressure Vessel Steels"
12.	17-19 February 1975, Austria	Meeting of International Working Group on Reliability of Reactor Pressure Components (IWG-RRPC)
13.	3-5 December 1975, Switzerland	SPM on "Fracture Mechanics Applications: Implications of Detected Flaws"
14.	29-31 March 1976, USA	TCM on "Stress Corrosion Cracking Problems in the Primary System of Nuclear Power Plants

15.	17-18 May 1976, Czech Rep.	TCM on "Reactor Vessel Surveillance: Results of Programmes Conducted and Proposals for Revision"
16.	19-21 May 1976, Czech Rep.	IWG-RRPC Meeting
17.	25-27 April 1977, Japan	TCM on "Use of Non-Destructive Testing Techniques for In-Service Inspection of Reactor Pressure Components"
18.	20-22 June 1977, Sweden	TCM on :Operating Experience Relating to Reliability of LWR Pressure Components"
19.	14-15 October 1977, Austria	IWG on "Reliability of Reactor Pressure Components"
20.	13-15 September 1978, Denmark	TCM on "Repair Aspects and Procedures"
21.	21-25 October 1978, Sweden	SPM on "Periodic Inspection of Nuclear Reactor Steel Pressure Vessels"
22.	20-21 November 1978, Austria	TCM on "Time and Load Dependent Degradation of Reactor Pressure Bounding Materials"
23.	5-8 March 1979, Spain	Meeting on "Trends in Reactor Pressure Vessel Development"
24.	20-22 October 1980, Austria	SPM on "Environmental Factors Causing Pipe Cracks and Degradation in Primary System Components"
25.	1-3 December 1980, Austria	SPM on "Reliability Engineering and Lifetime Assessment of Primary System Components"
26.	13-15 May 1981, Germany	SPM on "Sub-Critical Crack Growth"
27.	19-21 October 1981, Austria	SPM on "Radiation Embrittlement and Surveillance of Reactor Pressure Vessel Steels"
28.	4-5 December 1981, Austria	IWG-RRPC Meeting
29.	14-16 September 1982, Denmark	SPM on "Repair Aspects and Procedures"
30.	22-26 November 1982, Austria	Int. Symp. on "Water Chemistry and Corrosion Problems of Nuclear Reactor Systems and Components"

31.	14-16 December 1982, Austria	IWG-RRPC Meeting
32.	21-25 March 1983, Germany	Int. Symp. on "Reliability of Reactor Pressure Components"
33.	12-15 May 1983, Italy	SPM on "Defect Detection and Sizing"
34.	October 1983, Finland	SPM on "Corrosion and Stress Corrosion of Pressure Boundary Components"
35.	February 1984, Austria	SPM on "Radiation Embrittlement and Surveillance of RPV Steels"
36.	May 1984, Italy	SPM on "Crack Initiation and Arrest Control during Thermal Transients"
37.	15-17 May 1985, Japan	SPM on "Sub-Critical Crack Growth"
38.	3-5 September 1985, Austria	IWG-RRPC Meeting
39.	25-28 November 1985, Austria	SPM on "Recent Trends in the Development of Primary Circuit Technology"
40.	27-29 January 1986, Hungary	SPM on "Time and Load Dependent Material Performance other than Irradiation Effects"
41.	27-30 May 1986, Czech. Rep.	SPM on "Reactor Pressure Vessel Behaviour under Transient Conditions Caused by Thermal Shock"
42.	February 1987, Austria	IWG-RRPC Meeting
43.	27-29 May 1987, USA	SPM on "Irradiation Embrittlement of RPV Steels and Ageing"
44.	25-27 May 1988, Germany	SPM on "Fracture Mechanics Verification by Large Scale Testing"
45.	27-29 June 1988, Finland	SPM on "Inspection of Austenitic Dissimilar Materials and Welds"
46.	12-14 September 1988, Austria	SPM on "Corrosion and Erosion Aspects of Pressure Boundary Components of LWR's"
47.	October 1988, Austria	IWG-RRPC Meeting
48.	5-9 June 1989, Czech. Rep.	SPM on "Experience and Further Improvement of In-Service Inspection methods

		and Programmes of NPPs with Particular Emphasis on On-Line Techniques"
49.	24-26 October 1989, Argentina	SPM on "Residual Stresses in Structural Materials and Components of NPPs"
50.	14-18 May 1990, Russia	SPM on "Sub-Critical Crack Growth"
51.	26-28 Sept. 1990, Hungary	SPM on "Radiation Embrittlement of Nuclear Reactor Pressure Vessel Steels"
52.	10-12 October 1990, Sweden	SPM on "Nuclear Power Plant Lifetime Assurance"
53.	24-25 September 1990, Hungary	CRP on "Optimization of Reactor Pressure Vessel Surveillance Programmes and Their Analysis"
54.	23-26 Sept. 1991, Spain	SPM on "Nuclear Power Plant Components Maintenance, Repair and Replacement for Plant Life Management"
55.	19-21 November 1991, UK	SPM on "Thermal and Mechanical Degradation"
56.	17-19 February 1992, Austria	IWG-LMNPP(International Working Group on Life Management of NPPs) Meeting
57.	25-29 May 1992, Hungary	SPM on "Integrity of Pressure Components of Reactor Systems"
58.	8-11 June 1992, Czech Rep.	SPM on "Experience in Monitoring Ageing Phenomena for Improving NPP Availability"
59.	26-29 October 1992, USA	SPM on "Fracture Mechanics Verification by Large Scale Testing"
60.	20-23 Sept. 1993, France	SPM on "Irradiation Embrittlement and Optimization of Annealing"
61.	18-22 October 1993, Spain	SPM on "Steam Generator Problems and Replacement"
62.	22-23 November 1993, Austria	CRP on "Optimization of Reactor Pressure Vessel Surveillance Programmes and Their Analysis"
63.	7-9 February 1994, Austria	IWG-LMNPP Meeting

64.	8-10 March 1994, The Netherlnds	SPM on "Non-Destructive Examination Practices and Results"
65.	14-18 March 1994, Argentina	SPM on "Advanced Structural Integrity Assessment Procedures"
66.	19-23 September 1994, Ukraine	SPM on "Erosion and Corrosion of Nucler Power Plant Materials"
67.	15-17 November 1994, Japan	SPM on "Technology for Lifetime Management of Nuclear Power Plants"

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THE IAEA INTERNATIONAL DATABASE ON AGEING MANAGEMENT AND LIFE EXTENSION OF REACTOR PRESSURE VESSEL

DRAFT

1. Establishment of International Database

1.1. In pursuit of Article III, paragraph A.3 of its Statute, the International Atomic Energy Agency (hereinafter referred to as the "Agency") has established the International Database on Ageing Management and Life Extension of Reactor Pressure Vessel (RPV) Materials (hereinafter referred to as "International Database") in collaboration with interested Agency Member States and organizations in Member States of the Agency.

1.2. The International Database shall include data for plants with PWR (WWER) BWR, gas-cooled reactors, pressurized heavy water reactors (PHWR) and test reactors.

1.3. The International Database shall compile data for unirradiated and irradiated reactor pressure vessel materials. The database shall also include data on materials irradiated and annealed and re-irradiated after annealing and include data on thermally aged unirradiated materials.

1.4. Database of the results from the IAEA Coordinated Research Programme (CRP) on Optimization of Reactor Pressure Vessel Surveillance Programmes (Phases 1, II, III) shall be incorporated into the International Database. The Agency has a right of access of its own to the CRP data for analysis and publication.

2. Membership

2.1 Membership in the International Database shall be restricted to States which are Members of the Agency, and organizations in Member States, recognized by those states, which are in possession of data relevant to the International Database.

2.2 To participate in the International Database an official request to this effect is to be made by the appropriate Member State or organizations to the Director General of the Agency. Such request shall include the information prescribed in section 2.3 below and a commitment to fulfil the obligations laid down in Section 4.1-4 below. Those States and organizations which are accepted by the Agency are hereinafter referred to as International Database Members.

2.3 An intended member of the International Database shall provide together with the application for membership to the Agency the following information from the NPPs to be included in the database:

- number of units
- types of reactors
- number of irradiated capsules/sets
- number of irradiated specimens tested

- type of specimens tested
- future trend of data provided to the Agency
- year - number of capsules/sets - number of specimens

2.4. Every Member State of the Agency which is a member of the International Database shall appoint a Liaison Officer to act as a focal point. An Alternate Liaison Officer may also be appointed.

2.5. Each International Database member fulfilling its responsibilities as laid down in Section 4 below shall have the same rights and privileges in the database.

3. Objectives of the International Database

3.1. The International Database shall facilitate:

- defining more closely the reactor pressure vessel lifetime by reducing the uncertainty of the existing assessments.

- making predictions of future performance and remaining service life of the reactor pressure vessel, and provide a basis for timely implementation of mitigation measures such as flux reduction or annealing.

- identifying emerging embrittlement and ageing effects before their impact on vessel safety, reliability and service life.

- identifying and evaluating the effects of operating conditions outside the range of existing data for the purpose of extrapolation and improved mechanistic modeling.

- supplementing or confirming reliability of prediction methods for vessels with inadequate or incomplete material surveillance programmes.

- providing comprehensive information on plant operating experience of different plants and vessel materials as input for decisions concerning continued operation and license extensions of NPPs.

- providing improved feedback for designers on the performance of reactor pressure vessel materials under actual service conditions and to give guidance in the design of future plants.

3.2 The International Database shall assist:

- safety authorities in preparing more advanced and precise design curves for reactor pressure vessels materials damage behaviour and decrease the conservatism while still ensuring the necessary and reliable safety margins as well as helping within licensing processes and safety reports and analysis.

- designers in receiving more generalized data on materials behaviour during reactor operation to be able to compare design and actual lifetimes as well as in supporting their proposals for necessary upgrading or mitigation activities.

- utilities in a more sophisticated and more supportive analysis of individual surveillance data or even in providing necessary data (based on principle of similarities) for reactor pressure vessel lifetime assessment and extension.

- researchers in obtaining a better understanding of damage mechanisms as well as in understanding the role of individual parameters.

4. Responsibilities of International Database members:

4.1. A member of the International Database shall be responsible for:

4.1.1. the collection, categorization, indexing, abstracting and related preparation of reactor pressure vessel material data on a "best efforts basis" as well as data validation at its own expense and without any financial or other obligations to the IAEA or other members;

4.1.2. providing the Agency with the full text of each item in Section 4.1.1 in a format to be jointly determined with the Agency and prepared to a standard specification, provided that there is no prohibition on the transfer of such information;

4.1.3. contributing advice and recommendations on matters relating to the maintenance, improvement and development of the International Database;

4.1.4. providing information services to and monitoring contact with, to the extent practicable, users of the database within its territory and representing user's news at meetings of the International Database;

4.1.5. obtaining clearance from the Agency International Database members before providing information derived from the International Database to non-members of the Database;

4.1.6 provision of a first set of data for inclusion in the International Database within six (6) months of membership.

5. Management of the International Database

5.1. Secretariat functions for the International Database

5.1.1. The Agency shall provide the secretariat functions for the International Database with most functions being carried out by the Division of Nuclear Power. The secretariat functions shall be carried out in accordance with the Agency's policies, procedures, channels of authority and Financial Regulations and Rules.

5.1.3. The principal functions of the secretariat are:

5.1.3.1. the management of the International Database to ensure that its rules and procedures are correctly and efficiently implemented and to take necessary actions for the efficient operation and continued improvement of the Database. The Secretariat shall take into account the interests of all members in managing the operation of the International Database;

5.1.3.2. the integration of input from members into the International Database file including identification and correction of errors;

5.1.3.3. the developments, in consultation with Database members, and the subsequent updating and maintenance of all the authorizations, standards, formats, definitions, rules, procedures and guidelines to be used for the preparation and processing of input and for the creation and utilization of output;

5.1.3.4. the training, upon request, of Members' personnel in the preparation of input and utilization of output;

5.1.3.5. the preparation of input of literature published by the Agency and other UN organizations;

5.1.3.6. arranging meetings of the Steering Committee of the International Database.

5.1.3.7. subject to the annual budgetary approvals of the principal organs financing the Database;

5.1.3.8. preparation and custody of a register of members. The register shall be updated regularly.

6. Access to the International Database

6.1. Members of the International Database who have fulfilled all the requirements specified herein and have supplied data shall have full access to all non-confidential information from the International Database. The Agency may release information from the Databases to non-members only after obtaining clearance from the data supplier.

6.2. Members who have access to the International Database shall use the data for their own analysis or evaluation and shall inform the Agency of such use by sending a draft report paper prior to its publication for clearance.

6.3. Each Database Member shall identify which data amongst those provided as input into the International Database shall be treated as confidential. If no express indication is made, the Agency shall treat data in shaded areas of a proposed data format as confidential.

6.4. The Agency shall prepare a coding system to protect confidential data. The coding/decoding list shall be retained by the Agency. A Database Member may, at its option, prepare the coding system for its confidential information.

6.5. Transfer of confidential data from a Member to the Agency shall be voluntary.

6.6. Access to confidential data shall be through the Agency. Such data can only be released with the express approval of the Member who originated it.

6.7. The Agency shall have the same rights of access to the International Database as a Member. The Agency may use non-confidential data in the International Database for elaborating guidance and recommendations for developing countries without releasing the data.

7. Steering Committee

7.1. There shall be a Steering Committee made up of one representative from each Members State of the International Database appointed by the appropriate national authority of the Member State.

7.2. Members of the Steering Committee shall elect from among themselves a chairman who shall be approved by the Director General of the Agency and hold office for three years.

7.3. Regular meetings of the Steering Committee shall be held once a year. The programme of the Committee shall be the:

- evaluation of an annual progress report prepared by the Agency;

- recommendations on the procedures for regulating the operation of the Database;

- preparation of a progress report for the Agency's International Working Group on Lifetime Management of Nuclear Power Plants (IWG-LMNPP) meetings; and

- discussion of proposals from Members of the International Database.

7.4. Prior to the take-off of the Steering Committee, the ad-hoc group nominated by the International Working Group on Life Management of Nuclear Power Plants shall perform the duties of the Steering Committee.

8. Technical Matters

8.1. Technical specifications and requirements are given in the Appendix 1 "Description of International Database on Ageing Management and Life Extension of Reactor Pressure Vessel Materials" in the IAEA Report IWG-LMNPP - 94/6 "International Database on Ageing Management and Life Extension - Database Specification".

8.2. Data Collection

Three different ways for data gathering are offered for the participants:

Method I To Gather Data from the Already Existing Databases

The Agency provides the already existing database in original format on diskette (E-mail, CD, diskette etc.) for the International Database.

The Agency checks whether that database can be used and converted into the International Database. The databases will be divided into three groups.

(a) Databases which can be used without any difficulty or supplementary information.
In this case the Agency converts the data by using converting software, and manual Agency's methods. The data put into the format will be sent back to the representative of the Member for verification and for local use.

(b) With supplementary information
In this case the Agency contacts the representative or the Member and they come to an agreement on supplying the supplementary data. Then the process is the same as in case (a).

(c) If the offered database cannot be converted into the International Database or the conversion is too difficult or time consuming, the Agency and the representative of the data owner will elaborate the suggestion how to solve the special unique difficulties.

Method II To Collect the Data in the International Database Format

The International Database will be written in dBase. The dBase format (DBF files) will be provided on diskettes to the Members, International Database manuals will be supplied too. The Members convert or type in their own data into the International Database and send back to the Agency. This method is recommended for those Members who still do not have their own database, or wish to use the Agency format.

Method III   To Collect the Data on Sheets

Data acquisition sheets are also provided. This method can be used by laboratories or utilities providing only limited number of data (e.g. new data from a certain surveillance programme or research programme). Those data will by typed in by the Agency and sent back to the Member for verification.

8.3. Design/Format of International Database

The database design shall be divided into three different categories:

1. Raw data (data received from the Members)
2. Qualified data (data which have been checked for error)
3. Processed data (data which are ready for use)

The data entries of each of these categories shall be stored separately. The identification of the source of raw data shall be protected.

8.4. Data Qualification

Following data compilation, an independent data qualification must be performed. This task requires that all data in the International Database be reviewed for adequacy and accuracy. Confidence levels should be established for each data set; low-confidence data should be identified and should undergo a more thorough review.

8.5. Data Processing and Validation

Data from the contributors shall be validated, restructured, statistically evaluated and provided according to the needs of the Database Members. Raw data shall be processed to conform to the needs of the Database Members and the database specifications. The data provided by the Database Members shall be qualified using criteria which will be developed having been approved by the Steering Committee.

8.6. International Database Maintenance

8.6.1. The task of International Database maintenance involves continuous updating to include new information into the database and providing support to the various Database Members. During the first three (3) years updated diskettes with all data shall be supplied yearly to all Database Members having the right for data access. After three (3) years the Steering Committee shall recommend the data distribution frequency. The Agency shall provide for special request of the Database Members.

8.6.2. The Database Members shall obtain the updated diskettes at any time from the Agency. Upon an agreement with the Agency sorted data can also be supplied to the Database Members.

8.7. Documentation

IAEA Report IWG-LMNPP-94/6, Appendix 1 shall serve as a main technical documentation for data collection and keeping as well as its maintenance. This documentation shall also be updated as a new data or database capabilities become available.

8.8. Quality Assurance

8.8.1. The data to be collected, the responsibilities for data collection, qualification and all other tasks involved have to be clearly defined and periodically reviewed and confirmed by the Steering Committee.

8.8.2. Data shall be screened for completeness, correctness and consistency of content. It is very important that Database Members interact on a regular basis with the Agency and other Database Members. Independent regular Quality Assurance audits of all aspects of the Database shall be undertaken.

8.8.3. For detailed specific quality assurance programmes and procedures, the IAEA Safety Guide 50-SG-QA2 shall be followed.

9. Change of Status

If the Agency, on the recommendation of the Steering Committee, decides that a Database Member is not adequately fulfilling its responsibilities as laid out in Section 4 above, discussions will be initiated and a concerted effort made to bring the Database Member into compliance. A Database Member that has not adequately fulfilled its responsibilities as laid out in Section 4 above for two years, shall forfeit its privileges as given in Section 6 above, except in cases where the Agency, on the recommendation of the Steering Committee, is satisfied that the failure is due to conditions beyond the control of the Database Member. The Database Member will retain its right as laid out in Section 4 above until a decision to the contrary is taken in accordance with this Section.

1995-03-27
a:\iaeadtbs.re3

TRS: NUCLEAR POWER PLANT LIFETIME
THE OWNER'S POINT OF VIEW

FOREWORD

outline methodology, but :
- to limit to water reactors of PWR(incl.WWER),BWR and CANDU, only
- to add : to whom the TRS is addressed
- relation between components integrity and NPP safety

1. OVERVIEW ON PLANT LIFETIME MANAGEMENT

1.1 Significance of PLIM to NPPs
including aspects of PLIM (technical,economical, safety, ecological, political/human relations)

1.2 State of the art = today's knowledge and understanding
incl.design, manufacturing and operational aspects

1.3 Definitions - see EPRI terminology
to add: lifetime (graph from Database manual)
- PLIM
- Categorisation of components

1.4 General approach and procedures
- see flow-chart in Database manual
- lists of key components
  Japan, US NRC, France, CANDU, Russia

1.5 International programmes and related activities
- IAEA: IWG LMNPP, pilot studies, safety margins reports
- OECD/EC
- EPRI
- references to international/main national reports/papers

1.6 Scope of the report
- short description of individual chapters
- why they were chosen
- links between chapters

2. SURVEY OF AGEING MECHANISMS RELATED TO NPPs

2.1 Definition of physical ageing phenomena
- graph : stressors/mechanisms/consequences

2.2 Manufacturing aspects of ageing
(e.g. effect of initial conditions)

2.3 Service (operational) conditions of NPPs

2.4 Definition of stressors
2.5 Ageing mechanisms specific to NPPs
(single, multiple - synergism)
2.6 Consequences of ageing mechanisms to components behaviour
2.7 Ageing analysis of typically key components
- flow-chart based on 2.3.-2.6.
- threshold values of stressors to be effective
- role of databases (NPP/ national/ international/ IAEA international)

3. METHODS OF IDENTIFYING AGEING IN SERVICE

3.1 Purpose
3.1.1 To detect changes from original plant operating parameters
3.1.2 To detect degradation in materials, equipments and systems and to assess rate of degradation
3.1.3 To detect existance of degradation mechanisms others than those visualised in design.
3.1.4 To assess safety margins.

3.2 Methods of monitoring.
3.2.1 Preoperational and in-service on-line monitoring
3.2.2 Pre-service and in-service inspection
(choice of base line measurements methodology)
3.2.3 Testing and surveillance programmes
(including archive materials)
3.2.4 Documentation of operating and maintanenance history
3.2.5 Data reduction and documentation
(International database and dissemination)
3.2.6 Description of monitoring techniques
3.2.7 Need for new techniques and advanced methods (incl.sampling etc.)

3.3 Life Assessment
3.3.1 Identification of ageing mechanisms during service
(based on table from Chapter 2)
3.3.2 Integration of damage and assessment of residual lifetime
3.3.3 Assessment of margins
- normal operation
- accident conditions (DBA, BDBA)
3.4. Role of R & D
Give some examples to illustrate the point being made

4. MITIGATION OF AGEING EFFECTS

4.1 Augmented surveillance
4.2 Reduction of rate of damage through changes in operational strategy or modifications
4.3 Changes in inspection and preventive maintenance schedule
4.4 Modification
- removal from service
- addition of new component

4.5 Repairs including annealing of damage (restoration)
4.6 Replacement of component
4.7 Safety assessment of modification
4.8 Qualification and requalifiction
4.9 Man-Rem reduction/optimisation

5. SAFETY ASPECTS OF AGEING

5.1 Safety perspective:
ageing in NPP must be managed to ensure the availability of required safety functions throughout the plant service
this requires: dealing with physical degradation of plant systems, structures and components (SSCs) as well as their obsolescence, both of which are likely to occur during plant life
5.2 Physical ageing and non-physical ageing:
explain meing of these terms and identify related safety aspects/concerns
5.3 Safety objectives :
identify safety olbjectives relating to both physical ageing and non-physical ageing
5.4 Responsibilities for achievement of safety objectives:
a) plant owners/operators have the primary responsibility
b) regulators are responsible for safety verification
5.5 Approach to managing safety aspects of ageing:
a) systematic ageing management programme
b) periodic safety review

6. ECONOMIC CONSIDERATIONS IN PLANT LIFE MANAGEMENT

1. Composition of nuclear electricity generation costs in different economic environments (capital charges, fixed and variable fuel and O&M costs)
   - levelized (lifetime) costs

- short-term marginal costs
- medium-term marginal costs

2. Comparison to alternative generation
   - current situation
   - medium term

3. Benefit/loss considerations
   - incentives to operate at a high capacity factor and to extend plant life
   - impacts of low availability and of premature shutdown.

4. Life management activities and costs
   - maintenance (normal, preventive, corrective)
   - mitigation
   - repair
   - replacement
   - outage cost
   - limit for total cost of refurbishment

7. LICENSING AND REGULATORY ASPECTS

7.1 Check for occurrence for a problem
7.2 Assessment of safety impact of an observed degradation
7.3 Safety assessment of a repair
7.4 Justification of life extension airising from a revised strategy

8. PLANT LIFE MANAGEMENT STRATEGIES FOR DIFFERENT REACTOR SYSTEMS

This chapter should be limited to the water reactors. The gas-cooled, RBMK, fast breeder and other reactors should be excluded due to their relative small number and specific technical features.

The chapter will be divided for four sections:

8.1 Introduction. Should include the common features of different reactors discussed below, and a list of common key components.
8.2 PWR-s (including the WWER-s)
8.3 BWR-s
8.4 PHWR-s (Candu and others)
Each section should include:
- general description of system and operational conditions

- categorization of components (could be different for each country)
- list of key components/systems
- ageing mechanism for key components
- systems for life management
- life management programmes, preventive maintenance
- case studies
- methods of monitoring specific components
- record keeping, data store
- specific actions for each component.
- cost benefit of different solutions
- type specific features
  * old reactors (question relicensing)
  * new generation (life management for long period)

Indicative list of key components to be considered

A. PWR-s

* Reactor pressure vessel
* RPV internals
* Reactor coolant system piping
* Reactor coolant pump
* Steam generator
* Pressurizer
* Safety injection accumulator
* Control rod drive mechanism (CRDM)
* Containment structure
* Containment penetrations
* RPV supports
* Spent fuel pool
* Control room and instrumentation control
* Main turbine and generator
* Emergency diesel generator
* Electrical cables
* Transformers
* Secondary piping

B. BWR-s

* Reactor pressure vessel and safe ends.
* RPV internals
* Reactor coolant system piping

* Reactor coolant pump
* Control rod drive housing
* Containment structure
* Containment penetrations
* RPV supports
* Spent fuel pool
* Control room and instrumentation control
* Main turbine and generator
* Emergency diesel generator
* Electrical cables
* Transformers
* Secondary piping
* Drywell metal shell

C. PHWR-s (Candu and others)

* Fuel channels
* Steam generator including internals
* Calandria vessel
* Primary piping
* Secondary piping
* Vacuum building
* Calandria Vault and end-shield cooling systems
* Cables (power, control, and instrument cables)
* Electromechanical systems
* Reactor building
* Turbines
* Generator
* Cooling water intake structure
* Refuelling machine
* Spent fuel pool

9. NATIONAL PERSPECTIVES

* general national NPP description (number of units, type, age)
* current scenario
* life management programs
* methodology
* preventive maintanence and mitigation
* operational experience
* significant events
* backfitting programmes (e.g. steam generator replacement, annealing etc.)

* international programmes
* licensing approach
* case studies

TIME SCHEDULE	F	M	A	M	J	J	A	S	O	N	D	J	F	M	A	M
Invitation of authors	X	X	X													
Acceptance by authors	X	X	X													
Editorial board meeting			o	o												
Writing				X	X	X	X	X	X	o	o	o				
Editorial board meeting											o	o				
Reviewing and re-iteration										X	X	X	X	X		
Final draft												X	X	X		
Printing															X	X

Appendix 7

List of Meetings 95/96

Title	Venue	Date
1. Consultancy on Guidelines for the Development of the International Reactor Materials Surveillance Database	Moscow	6-8 March 1995
2. SPM on Cracking in PWR, RPV Head Penetrations	USA	2-4 May 1995
3. SPM on Irradiation Embrittlement and Mitigation	Finland	23-26 Oct. 1995
4. SPM on Optimized In-Service Inspection Programmes and Reduction of Maintenance Costs - Utility Experience		1996
5. SPM on Steam Generator Repair and Replacement, Practices and Lessons Learned		1996
6. Consultancy on Maximizing Operational Lifetime - the Owners Point of View		May, 1995
7. AGM on Maximizing Operational Lifetime - the Owners Point of View	VIC	20-23.02. 1995
8.Consultancy on Optimizing Water Chemistry for Successful Corrosion Control (in co-op with NENF)		1996
9. Consultancy on Maximizing Operational Lifetime - the Owners Point of View		1996
10. Consultancy on Ageing Mechanisms, Degradation Monitoring and Management of Concrete Structures		1995
11. AGM on Ageing Mechanisms, Degradation Monitoring and Management of Concrete Structures		1996
12. Consultancy on Ageing Mechanisms, Degradation Monitoring and Management of Concrete Structures		1996
13. IWG meeting on Life Management in NPPs		1996
14. CRP on Management of Ageing of Reactor Pressure Vessel Primary Nozzle (jointly with NENS)		1996

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INTERNATIONAL ATOMIC ENERGY AGENCY

INTERNATIONAL WORKING GROUP
on
LIFE MANAGEMENT of NUCLEAR POWER PLANTS

SPECIALISTS MEETING
on

CRACKING IN LWR RPV HEAD PENETRATIONS

Philadelphia, USA

2 - 4 May 1995

INFORMATION SHEET

1. Introduction

The structural safety and integrity of nuclear power plants is largely dependent on the design of structures and materials which has been decided prior to the construction. Planning alone cannot guarantee the safety operation of the power plant throughout its life. The materials are deteriorated due to the mechanical and environmental (neutron irradiation, feedwater of primary and secondary circuits) effects. The safe use of mechanical components and structures requires continuous monitoring during the operational life of the plant.

Research on structures and structural materials aims to prevent accidents and unforeseen outages. and ensures the safe and reliable operation of equipment throughout its planned life. At the same time, ways are being sought to extend the lifetime of components.

2. Purpose of the Meeting

The Specialists Meeting on "Cracking in LWR RPV Head Penetrations" is organized by the IAEA International Working Group on Life Management of Nuclear Power Plants (IWG-LMNPP) and sponsored by the Oak Ridge National Laboratory and the US Nuclear Regulatory Commission.

The purpose of the meeting is to review experience in the field of ensuring adequate performance of reactor pressure vessel (RPV) heads and penetrations. The scope will include:

1. Overview(s): Current issues and problem areas, their consequences, and remedies.
2. Operating Experience: Results of inspections interpretations, and applications.
3. Monitoring and Inspection Methods: Techniques used, qualification of techniques, inspection reliability, and field experience.
4. Susceptibility Evaluations: Models for predicting susceptibility, operating experience.
5. Structural Integrity Assessments: Flaw acceptance criteria, crack-growth rates, boric acid corrosion evaluations, and risk/consequence evaluations.
6. Repair and Mitigation: Techniques, qualification and field experience.
7. Ongoing/Needed Research: Relative to Items 3 through 6 above.

Presentation should be aimed at better understanding of behaviour of reactor component materials, to provide guidance and recommendations assuring reliability adequate performance and directions for further investigations.

3. Participation

It is expected that the total number of participants will be approximately 50. Each person wishing to participate in the meeting should be officially designated by the relevant governmental authority (Ministry of Foreign Affairs or National Atomic Energy Authority)

Correspondence with regard to the meeting should be addressed to the Scientific Secretary of the meeting:

Mr. L. Ianko
Division of Nuclear Power
International Atomic Energy Agency
Wagramerstrasse 5
P.O. Box 100
A-1400 Vienna, Austria
Tel: 2360 ext. 2797, Fax: 43 1 23 45 64, E-Mail:ianko @ nepo1.iaea.or.at.

4. Organization of the Meeting

The meeting will be held from 2 to 4 May 1995 at the Headquarters of the American Society for Testing Materials (ASTM), Headquarters Building 1916, Race Street, Philadelphia, Pennsylvania 19103-1187, USA.

Organizing Committee

The following individuals are on the Organizing Committee for the meeting:

Dr. Claud E. Pugh
ORNL
P.O. Box 2009
Mail Stop 8063
Oak Ridge, TN 37831

Tel: (615) 574-0422
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Chairman of the IWG-LMNPP
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E-Mail:ianko @ nepo1. iaea. or. at.

Any correspondence to the Organizing Committee should be addressed to Dr. C. Pugh at:

ORNL
P.O.Box 2009
Oak Ridge, Mail Stop 8063
Tennessee 37831-8063, USA

Tel: 615-574-0717, Fax: 615-241-5005

5. <u>Abstracts and Papers</u>

Presentation of papers will be selected on the basis of abstracts which will be processed as received. Any individual requiring an early confirmation of acceptance of his presentation is encouraged to submit his abstract as soon as possible. The abstract must not exceed three single-spaced pages. All abstracts must be received not later than 1 April 1995. Two copies of the abstract should mailed to:

Mr. L. Ianko
Division of Nuclear Power
International Atomic Energy Agency
Wagramerstrasse 5
P.O. Box 100
A-1400 Vienna, Austria

Tel: 2360 ext. 2797
Fax: 43 1 23 45 64
E-Mail:ianko @ nepo1. iaea. or. at.

Those authors who are notified of acceptance and wish to publish their full-length papers in the proceedings should submit 1 original typed and two copies of the completed paper no later than 15 April 1995 to:

Mr. C. Pugh
ORNL
P.O.Box 2009
Mail Stop 8063
Oak Ridge, Tennessee 37831-8063
USA

Tel: 615-574-0717
Fax: 615-241-5005

Late manuscripts must be carried in fifty copies by the author together with the master copy for the production of proceedings.

The manuscripts will be published as received so they should be typed single spaced on A4 format or 8.5 by 11 inchs. paper and should include original illustrations and glossy prints of the photographs. Top, bottom, right and left margins should be 24 mm (1 inch).

INTERNATIONAL ATOMIC ENERGY AGENCY

INTERNATIONAL WORKING GROUP
on
LIFE MANAGEMENT of NUCLEAR POWER PLANTS

SPECIALISTS MEETING
on

IRRADIATION EMBRITTLEMENT AND MITIGATION

Finland, Espoo

23 - 26 October 1995

INFORMATION SHEET

1. Introduction

More than three-quarters of the world's nuclear power stations use steel pressure vessels to house the reactor core. Successful and safe performance of these power stations during their lifetime depends, to a large extent, on the reliability of the steel pressure vessels. Reactor steel pressure vessels are subjected to neutron radiation during operation. The effect of this radiation on the pressure vessel steel is manifested by an increase in yield strength, hardening, a shift in brittle-ductile transition temperature and a decrease in ductility. Safety assessments are carried out to safeguard against pressure vessel failure.

Amelioration of neutron irradiation effects has been implemented in some power plants. Utilities, research organizations and regulatory bodies in many countries are actively involved in assessing the effects of neutron radiation on the properties of pressure vessel steels.

2. Purpose of the Meeting

The Specialists' Meeting on "Irradiation Embrittlement and Mitigation" is organized by the IAEA International Working Group on Life Management of Nuclear Power Plants.

The purpose of the Meeting is to provide an international forum for discussion on recent results in research and utility experience on:

- radiation damage and its surveillance,
- annealing and re-embrittlement of PWR, WWER, and BWR reactor pressure vessel materials.

Papers are expected to focus on the following areas:

- mechanism of radiation damage,
- effects of operating parameters (flux, temperature, time etc.),
- results from surveillance programmes and their analysis,
- fracture mechanics testing and evaluation,
- annealing and optimization of the process,
- re-embrittlement after annealing.

Presentation should be aimed at better understanding of radiation damage, annealing and re-irradiation behaviour of reactor pressure vessels materials, to provide guidance and recommendations for optimization of annealing and surveillance programmes and directions for further investigations.

3. Participation

It is expected that the total number of participants will be approximately 50. Each person wishing to participate in the meeting should be officially designated by the relevant governmental authority (Ministry of Foreign Affairs or National Atomic Energy Authority).

Correspondence with regard to the meeting should be addressed to the Scientific Secretary of the meeting:

Mr. L. Ianko
Division of Nuclear Power
International Atomic Energy Agency
Wagramerstrasse 5
P.O. Box 100
A-1400 Vienna, Austria
Tel: 2360 - 2797
Fax: 43 1 23 45 64
E-mail: ianko @ nepo1. iaea.or.at

4. Organization of the Meeting

The meeting will be held from 23 to 26 October 1995 in Espoo, Finland.

Organizing Committee

The following individuals are on the Organizing Committee for the meeting:

Pr. K. Törrönen (Committee Chairman)
VTT Manufacturing Technology (Finland)

Acad. L.M. Davies
Chairman, IAEA International Working Group on Life Management of Nuclear Power Plants, (UK)

Mr. R. Rintamaa
VTT Manufacturing Technology (Finland)

Mr. K. Wallin
VTT Manufacturing Technology (Finland)

Mr. M. Valo
VTT Manufacturing Technology (Finland)

Mr. T. Planman
VTT Manufacturing Technology (Finland)

Mr. R. Ahlstrand
IVO International (Finland)

Mr. M. Ojanen
Finnish Centre for Radiation and Nuclear Safety (Finland)

Mr. L.Ianko
Scientific Secretary
(IAEA)

Any correspondence with the Organizing Committee should be addressed to:

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E-mail: kari. torronen @ vtt. fi

5. Abstracts and Papers

Presentation of papers will be selected on the basis of abstracts which will be processed as received. Any individual requiring an early confirmation of acceptance of his presentation is encouraged to submit his abstract as soon as possible. The abstract must not exceed three single-spaced pages. All abstracts must be received not later than 1 September 1995. Two copies of the abstract should be mailed to:

Mr. L. Ianko
Division of Nuclear Power
International Atomic Energy Agency
Wagramerstrasse 5
P.O. Box 100
A-1400 Vienna, Austria
Tel: 2360 - 2797
Fax: 43 1 23 45 64
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Those authors who are notified of acceptance and wish to publish their full-length papers in the proceedings should submit 1 original typed and two copies of the completed paper no later than 1 October 1995 to:

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P.O. Box 1704
FIN-02044 VTT
Finland
Tel: +358 0 456 6840 - Fax: +358 0 456 7002,

The manuscript should be typed single spaced on A4 format or 8.5 by 11 ins paper and should include original illustrations and glossy prints of the photographs. Top, bottom, right, and left margin should be 24 mm-es (1 inch).

The programme of the meeting will be distributed to the participants.

The optimum size of a paper is approximately 5000 words, or about 12 types pages including original illustrations and glossy prints. Sharp black and white photographs may be included..

On the first page the title of the paper, the author's name and address should be printed according to the following format:

TITLE OF THE PAPER

by
Mr. JOHN NO NAME*, GEORGE YOUNG**, etc.
*Company, ZIP code; town; street; country
**Institution, ZIP code; town; street; country

<u>ABSTRACT</u>. Should consist of 150 words, summarize the objectives and conclusions as specifically as possible.
Keywords:

Text supplied in ASCII format or edited by any internationally used text editor (Word Perfect, Microsoft Word etc.) and figures supplied PIC, PCX, or TIF format on diskettes are highly desired.

The proceedings including all of the papers which arrive before 10 August (final deadline) will be printed and offered to the participants free of charge.

Further information on proceedings and manuscript format requirements will be sent to the authors together with the acceptance of their contribution.

Late manuscripts must be carried in fifty copies by the author.

6. <u>Working language</u>

The working language of the meeting will be English.

INTERNATIONAL ATOMIC ENERGY AGENCY

INTERNATIONAL WORKING GROUP
ON
LIFE MANAGEMENT OF NUCLEAR POWER PLANTS

SPECIALIST MEETING ON
CRACKING IN LWR RPV
HEAD PENETRATIONS

PAPER: "SPANISH RPV HEAD PENETRATIONS. REGULATORY STATUS"

AUTHORS: JOSE M. FIGUERAS (CSN, Madrid, Spain)
JUAN R. COLINO (CSN, Madrid, Spain)

Philadelphia, USA
2 - 4 May 1995

"SPANISH RPV HEAD PENETRATIONS. REGULATORY STATUS"

by: JOSE M. FIGUERAS, JUAN R. COLINO
CONSEJO DE SEGURIDAD NUCLEAR
28040 Madrid; J.Dorado,11; SPAIN

ABSTRACT.

The paper presents the actual status of inspection results on the spanish PWR RPV CRD head penetrations (CRDH`s), after two years of a whole program of inspections in all affected plants.

Actual situation of penetrations pertaining to **ALMARAZ 1 and 2, ASCO 1 and 2 and VANDELLOS 2 NPP`s** show any damage in those CRDH`s inspected in 1993 and 1994 (roughly 20 out of 65 CRDH`s at each unit). The paper presents a summary of CRDH characteristics, inspection methods and results obtained in each plant.

TRILLO NPP has a different CRDH design (KWU-SIEMENS type) and for that reason is not considered an affected plant nor has conducted any inspection up to now.

JOSE CABRERA (ZORITA) NPP has shown extensive damage, both in the lower side (weldment to the vessel) and in the upper free span area, near bimetallic weldment to SS 304, in active and non-active penetrations and also in the vent nozzle. The paper comments extensively on the CRDH materials general data, root-cause analysis and structural analysis of degraded zones, inspection results, repair actions and other additional actions applied up to now.

Finally, the paper deals with the ***regulatory actions taken by CSN on this topic***, both for those NPP`s actually non affected by the IGSCC phenomenon in the RPV CRDH`s and for the specific safety case of ZORITA NPP.

1.- ALMARAZ 1/2, ASCO 1/2 AND VANDELLOS 2 RPV HEAD PENETRATIONS.

1.1.- GENERAL DATA.

* 65 PENETRATIONS: 57 OR 52, ACTIVE (CRDH) WITH THERMAL SLEEVE

  4, ACTIVE (THERMOCOUPLES) NOT SLEEVED

  4 OR 9, RESERVE (PLUGGED) NOT SLEEVED

* MATERIALS:

TUBE,	INCONEL 600 M.A. (SB-167)
UPPER PART,	S.S. 304
VESSEL HEAD,	C.S. 302 B
" CLADDING,	S.S. 308
WELD BUTTERING,	INCONEL 182
" MATERIAL,	INCONEL 182

* DIMENSIONS:

TUBE INNER DIAMETER,	69,85 MM.
TUBE THICKNESS,	15,875 MM.
TUBE LENGTH,	870 - 923 MM. (MIN) 1497 - 1668 MM. (MAX)

1.- ALMARAZ 1/2, ASCO 1/2 AND VANDELLOS 2 RPV HEAD PENETRATIONS.

1.2.- SUMMARY OF INSPECTION RESULTS.

* INSPECTION SCOPE AND TECHNIQUES:

ALL INSPECTIONS PERFORMED WITH ROBOTIZED EQUIPMENT:

- ALMARAZ 1 -> FRAMATOME/BW, SEPT. 93
- ASCO 1 -> ABB REAKTOR, JUNE 93
- ALMARAZ 2 -> PETAVA PROJECT, FEBR. 94
- ASCO 2 -> PETAVA PROJECT, MARCH 94
- VANDELLOS 2 -> PETAVA PROJECT, MAY 94

* RESERVE PENETRATIONS:

INSIDE ECT (MRPC); WELD AREA ± 2,0"

CRDH`S SELECTED FROM THREE OUTERMOST CIRCLES

SCOPE:
- ALMARAZ 1, 4 CRDH`S
- ASCO 1, 4 CRDH`S
- ALMARAZ 2, 4 CRDH`S
- ASCO 2, 9 CRDH`S
- VANDELLOS 2, 2 CRDH`S

RESULTS: NO DAMAGE

1.- ALMARAZ 1/2, ASCO 1/2 AND VANDELLOS 2 RPV HEAD PENETRATIONS.

1.2.- SUMMARY OF INSPECTION RESULTS (CONT`D).

* **ACTIVE PENETRATIONS (TC`S):**

  # INSIDE ECT (MRPC); WELD AREA ± 2,0"

  # TC`S ARE LOCATED IN OUTERMOST CIRCLES

  # SCOPE:
  - ALMARAZ 1, 0 TC`S
  - ASCO 1, 4 TC`S
  - ALMARAZ 2, 4 TC`S
  - ASCO 2, 4 TC`S
  - VANDELLOS 2, 2 TC`S

  # RESULTS: NO DAMAGE

* **ACTIVE PENETRATIONS (CRDH`S):**

  # INSIDE ECT (GAP SCANNER / "SABLE"); WELD AREA ± 2,0"

  # CRDH`S SELECTED FROM THREE OUTERMOST CIRCLES

  # SCOPE:
  - ALMARAZ 1, 16 CRDH`S
  - ASCO 1, 12 CRDH`S
  - ALMARAZ 2, 12 + 2 CRDH`S
  - ASCO 2, 5 + 2 CRDH`S
  - VANDELLOS 2, 16 CRDH`S

  # RESULTS: NO DAMAGE

2.- JOSE CABRERA RPV HEAD PENETRATIONS.

2.1.- GENERAL DATA.

* 37 PENETRATIONS: 17, ACTIVE (CRDH) WITH THERMAL SLEEVE

  2, ACTIVE (THERMOCOUPLES) NOT SLEEVED

  1, ACTIVE (S.I.S. NOZZLE) NOT SLEEVED

  17, RESERVE (PLUGGED) NOT SLEEVED

* MATERIALS:

TUBE,	INCONEL 600 M.A. (SB-167)
UPPER PART,	S.S. 304
VESSEL HEAD,	C.S. 302 B
" CLADDING,	S.S. 308
WELD BUTTERING,	INCONEL 182
" MATERIAL,	INCONEL 182

* DIMENSIONS:

TUBE INNER DIAMETER,	69,9 MM.
TUBE THICKNESS,	15,85 MM.
TUBE LENGTH,	395 (MIN) - 730 (MAX) MM.

2.- JOSE CABRERA RPV HEAD PENETRATIONS.

2.2.- SUMMARY OF INSPECTION RESULTS.

* INSPECTION SCOPE AND TECHNIQUES:

  # ALL INSPECTIONS PERFORMED WITH ROBOTIZED EQUIPMENT SPANISH DESIGNED, PETAVA PROJECT, JAN. TO MAY 94.

* RESERVE PENETRATIONS:

  # 17 INSPECTED FULL LENGTH, BOTTOM TO BIMETALLIC WELD

  # ALL INSIDE ECT (MRPC)

  # ALL OUTSIDE UT (P-SCAN AND MANUALLY) AND INSIDE UT (TOFD)

* ACTIVE PENETRATIONS (TC`S AND S.I.S. NOZZLE):

  # 3 INSPECTED FULL LENGTH, BOTTOM TO BIMETALLIC WELD

  # ALL INSIDE ECT (MRPC)

  # OUTSIDE UT (P-SCAN OR MANUALLY) -> Nº 23 (SIS), 30 (TC)

  # INSIDE UT (TOFD) -> Nº 30

* ACTIVE PENETRATIONS (CRDH`S):

  # 17 INSPECTED PART LENGTH, BOTTOM TO 60 - 225 MM.

  # ALL INSIDE ECT (GAP SCAN)

  # 12 OUTSIDE UT (P-SCAN); 5 CENTRAL UT INTERFERED

  # INSIDE UT (TOFD) -> Nº 10, 15, 17, 20.

2.2.- SUMMARY OF INSPECTION RESULTS (CONT'D).

* RESERVE PENETRATIONS:

16 PENETRATIONS DAMAGED IN VESSEL-TO-TUBE WELD AREA

- DEFECTS ARE AXIAL + CIRCUMFERENTIAL CRACKS NETWORK
- CIRCUMFERENTIAL CRACKS LOCATED ON UPPER BORDER OF WELD AREA
- CRACK LENGTHS, UP TO 48 MM. (AXIAL), 158º (CIRC)
- CRACK DEPTHS, UP TO WALLTHROUGH

12 PENETRATIONS ALSO DAMAGED IN FREE TUBE AREA

- DEFECTS ARE ISOLATED AXIAL CRACKS
- CRACK LENGTHS, UP TO 118 MM.
- CRACK DEPTHS, UP TO WALLTHROUGH

PENETRATIONS Nº 2, 3, 4 SHOW ONLY AXIAL CRACKING IN WELD AREA

PENETRATION Nº 5 IS FREE OF DEFECTS

MOST AFFECTED -> PERIPHERICAL PENETRATIONS

2.- JOSE CABRERA RPV HEAD PENETRATIONS.

2.2.- SUMMARY OF INSPECTION RESULTS (CONT`D 2).

* ACTIVE PENETRATIONS (TC`S AND SIS NOZZLE):

\# PENETRATIONS Nº 23 (SIS), 34 (TC) ARE FREE OF DEFECTS

\# PENETRATION Nº 30 (TC):

- 7 AXIAL CRACKS IN WELD AREA
- CRACK LENGTHS, 5 - 21 MM.
- CRACK DEPTHS, SHALLOW UP TO 4,4 MM.

* ACTIVE PENETRATIONS (CRDH`S):

\# 10 PENETRATIONS SHOW DAMAGE IN VESSEL-TO-TUBE WELD AREA

\# NO DAMAGE IN FREE TUBE AREA

\# DEFECTS ARE ISOLATED AXIAL OR CIRCUMFERENTIAL CRACKS

\# ONLY 3 CRDH`S (Nº 10, 15, 17) SHOW "SIGNIFICANT" CRACKS:

- ECT PHASE/AMPLITUDE SIGNAL DISCRIMINATION
  SOFTWARE ANALYZED =>
  . CRACK LENGTHS, 6 - 22 MM. (AXIAL), 24 - 80º (CIRC)
  . CRACK DEPTHS, 3 - 7,5 MM.

2.- JOSE CABRERA RPV HEAD PENETRATIONS.

2.3.- ROOT CAUSE ANALYSIS.

* METALLOGRAPHIC EXAMS ON 2 SAMPLES:

  # 1 FROM FREE AREA, CRDH Nº 36 (W STC, USA)

  # 1 FROM LOWER BORDER WELD AREA, CRDH Nº 37 (CIEMAT, SPAIN)

* EXAMINATION RESULTS:

  # PENETRATION ALLOY 600 MICROSTRUCTURE SIGNIFICANTLY SENSITIZED

  # GOOD GRAIN BOUNDARY CARBIDE COVERAGE

  # SULFUR PRESENCE INSIDE CRACKS FRACTURE SURFACES

* NPP CHEMISTRY RECORDS SUGGEST THAT SIGNIFICANT POTENTIAL FOR ACID SULPHATES AND OTHER SULFUR REDUCED SPECIES ENVIRONMENT EXISTED:

  # DURING SERVICE OPERATION DUE TO RESINS INTRUSIONS IN AUG.80 AND SEP.81 (CATIONIC DEMINERALIZER RETENTION MESH FAILURES)

  # FROM OCT.82 THROUGH DEC.83 NPP WAS IN PLANNED SHUTDOWN FOR MAJOR MODIFICATIONS =>

  - RPV HEAD SETTLED ON STAND IN AIR ENVIRONMENT (O_2)

  - POLITHIONATES (TIOSULPHATES + TETRATHIONATES) FORMED

2.- JOSE CABRERA RPV HEAD PENETRATIONS.

2.3.- ROOT CAUSE ANALYSIS (CONT'D).

* STRUCTURAL INTEGRITY ANALYSIS:

FEM ANALYSIS ON A PERIPHERAL PENETRATION WHOLE LENGTH

BOTH VESSEL-TO-TUBE WELD AND FREE TUBE AREAS SHOW HIGH AXIAL AND HOOP STRESSES OF ABOUT Y.S. ORDER, 60 KSI

CLASSIC LEFM AND LIMIT LOAD ANALYSIS:

- CRITICAL TW CRACK LENGTHS -> 166 MM. (AX), 241º (CIRC)
- CRITICAL CRACK DEPTH -> A/T < 0,75 (MAX. 12 MM.)
- CRACK GROWTH ESTIMATIONS (P. SCOTT CURVE) -> 3,5 MM/YEAR

* FINAL ROOT CAUSE CONCLUSIONS:

NO PWSCC MECHANISM

IGA + SCC, SULFUR REDUCED SPECIES INDUCED, MECHANISM

DIFFERENT DAMAGE DEPENDING ON THERMAL SLEEVE

- RESERVE PENETRATIONS MORE AFFECTED

16 RESERVE PENETRATIONS WILL EXCEED ASME IWB-3600 CRITERIA (A/T < 0.75, AX; < 0.56, CIRC) DURING NEXT CYCLE => REPAIR

4 ACTIVE PENETRATIONS WILL EXCEED ASME CRITERIA IN 1,0 TO 2,8 YEARS => REPAIR

REST (7) OF ACTIVE PENETRATIONS WILL BE UNACCEPTABLE IN 3,3 YEARS => NO REPAIR, BUT CONTINUOUS SURVEILLANCE

2.- JOSE CABRERA RPV HEAD PENETRATIONS.

2.4.- REPAIR METHODOLOGIES.

* REPAIR SCOPE AND TECHNIQUES:

  # ALL REPAIR TECHNIQUES PERFORMED WITH ROBOTIZED EQUIPMENT:

  - EDM -> SPANISH PETAVA PROJECT, JAN. TO MARCH 95.
  - CAPS -> WESTINGHOUSE, OCT. 94 TO JAN. 95

* RESERVE PENETRATIONS:

  # PLUGGED 16 CRDH`S, ALL BUT Nº 5

  # INSTALLED HEMISPHERICAL CAPS ATTACHED TO THE "OLD" WELDMENT

  # CAPS AND "NEW" WELD ACT AS PRIMARY PRESSURE BARRIER

  # CAP MATERIALS: INCONEL 690 TT (SB-166), INCONEL 182

  # FINAL HYDROSTATIC TEST AT 151 BARS

* ACTIVE PENETRATIONS:

  # REPAIRED 4 CRDH`S: Nº 10, 15, 17 (2 AREAS) AND 30

  # ELIMINATED CRACKS BY ELECTROEROSION (EDM)

  # DIMENSIONS OF "BATHTUB LIKE" EXCAVATIONS,

  - CRDH Nº 10: DEPTH 6,53 MM.; 49 MM.(AX.) BY 59º (ARC)
  - CRDH Nº 15: DEPTH 5,70 MM.; 59,5 MM. BY 109º
  - CRDH Nº 17: DEPTH 7,90 MM.; 41,6 MM. BY 44,5º
DEPTH 6,30 MM.; 47,7 MM. BY 102º
  - CRDH Nº 30: DEPTH 6,85 MM.; 86,6 MM. BY 69º

2.- JOSE CABRERA RPV HEAD PENETRATIONS.

2.5.- ADDITIONAL ACTIONS.

* ANALYSIS AND/OR INSPECTIONS SCOPE OF RCS COMPONENTS WITH POSSIBLE SULFUR INDUCED SCC:

COMPONENTS:
- RPV BOTTOM MOUNTED INSTRUM. NOZZLES
- RPV HEAD VENT NOZZLE
- RPV UPPER INTERNALS
- CRDH`S INTERNALS
- SG TUBES, TUBESHEET AND PARTITION PLATE
- PRESSURIZER PENETRATIONS
- FUEL ELEMENTS
- CONTROL RODS

MATERIALS:
- INCONEL 600, X-750, 718, 82, 182
- AUSTEN. SS. (FORGED, WELD MAT., DUPLEX)
- MARTENSITIC SS. 403, 410
- STELLITE-6, HAYNES-25

* ANALYSIS AND/OR INSPECTIONS RESULTS:

ALL COMPONENTS INSPECTED BUT ONE ARE FREE OF DEFECTS

RPV HEAD VENT NOZZLE DEFECTIVE (SAME ROOT CAUSE) =>

- REPAIRED BY INCONEL 690 TT PLUGGING (ACTUALLY IN PROCESS)
- VENT FUNCTION NOW THROUGH ONE ACTIVE CRDH

3.- REGULATORY ACTIONS TAKEN BY CSN.

3.1.- ALMARAZ 1/2, ASCO 1/2 AND VANDELLOS 2 RPV HEAD PENETRATIONS.

* ADDITIONAL ACTIONS REQUESTED:

ALMARAZ 1/2 NPP`S:

- RPV HEAD REPLACEMENT SCHEDULED BY UTILITY IN 1996/97
- DUE TO INSPECTION RESULTS AND SUSCEPTIBILITY STUDIES NO ADDITIONAL INSPECTION REQUESTED BY CSN

ASCO 1/2 NPP`S:

- NO RPV HEAD REPLACEMENT SCHEDULED BY UTILITY NOR REQUESTED BY CSN
- ADDITIONAL INSPECTION REQUESTED BY CSN IN 1995/96
- FUTURE ACTIONS DEPENDING ON INSPECTION RESULTS

VANDELLOS 2 NPP:

- NO RPV HEAD REPLACEMENT SCHEDULED BY UTILITY NOR REQUESTED BY CSN
- FUTURE ACTIONS (ADDITIONAL INSPECTION) IN STUDY BY CSN

3.- REGULATORY ACTIONS TAKEN BY CSN.

3.2.- TRILLO RPV HEAD PENETRATIONS.

* ANALYSIS:

    # KWU-SIEMENS HAS DIFFERENT CRDH`S DESIGN AND MATERIALS

    # NOT CONSIDERED UP TO NOW AS AFFECTED NPP

    # NO CRDH`S INSPECTION PERFORMED NOR REQUESTED BY CSN

3.3.- JOSE CABRERA RPV HEAD PENETRATIONS.

* ADDITIONAL ACTIONS REQUESTED:

    # IMPROVE RPV HEAD LEAK DETECTION SYSTEMS (NO N-13 METHOD)

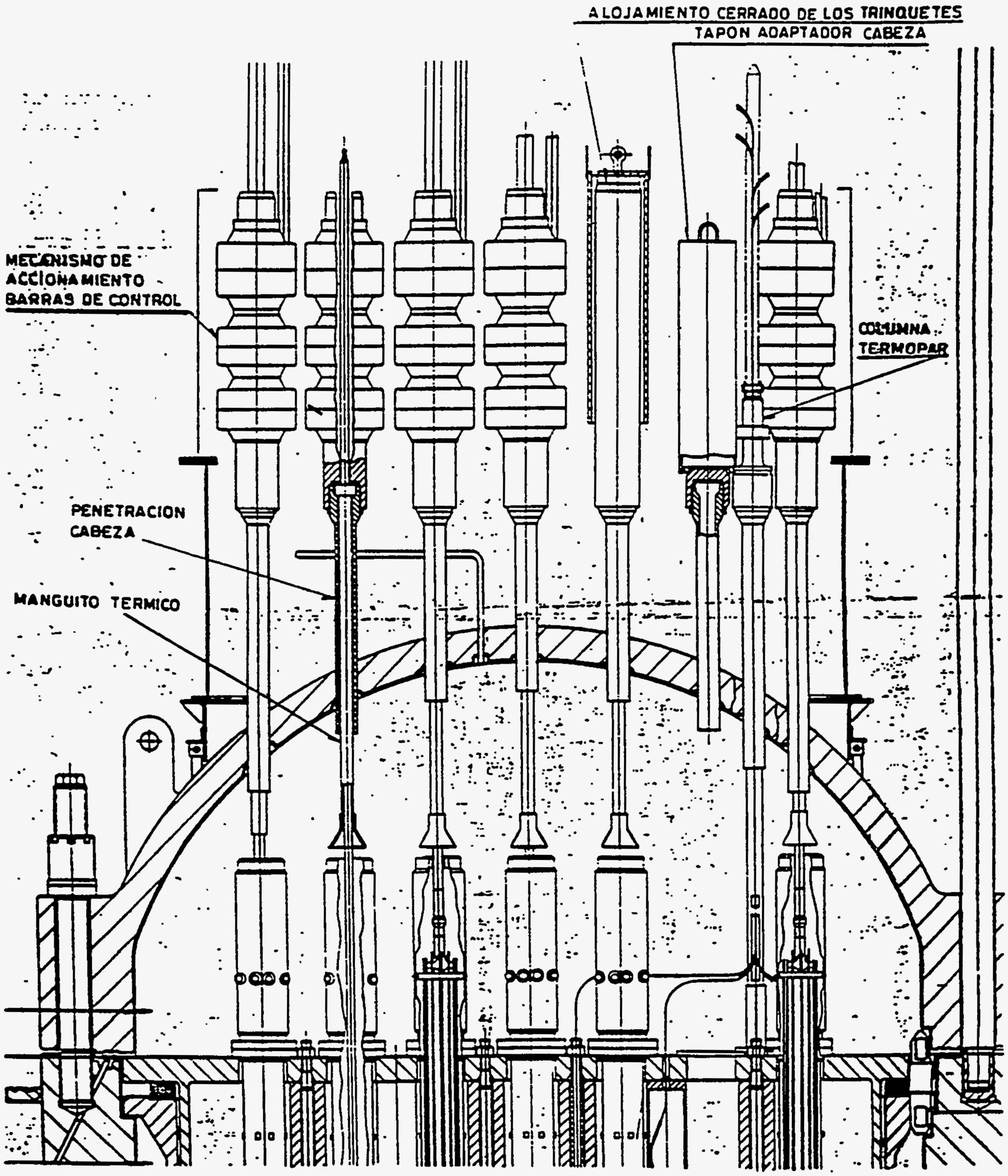
    # PSI (BASELINE) ON CAPS REPAIR AND EDM EXCAVATION REPAIR FOR FUTURE INSPECTIONS =>
    NEXT INSPECTION IN 1996 REFUELING SHUTDOWN

    # SPECIFIC SULFUR AND OTHER AGGRESIVE CHEMICAL SPECIES CONTROL DURING NEXT CYCLE OPERATION

    # SPECIFIC STUDY ON EDM EFFECTS FOR INCONEL MICROSTRUCTURAL PROPERTIES

    # COMPLETE DURING NEXT REFUELING SHUTDOWN INSPECTION ON REST OF RCS COMPONENTS (RPV BOTTOM MOUNTED INSTRUM. NOZZLES AND S.G. PARTS)

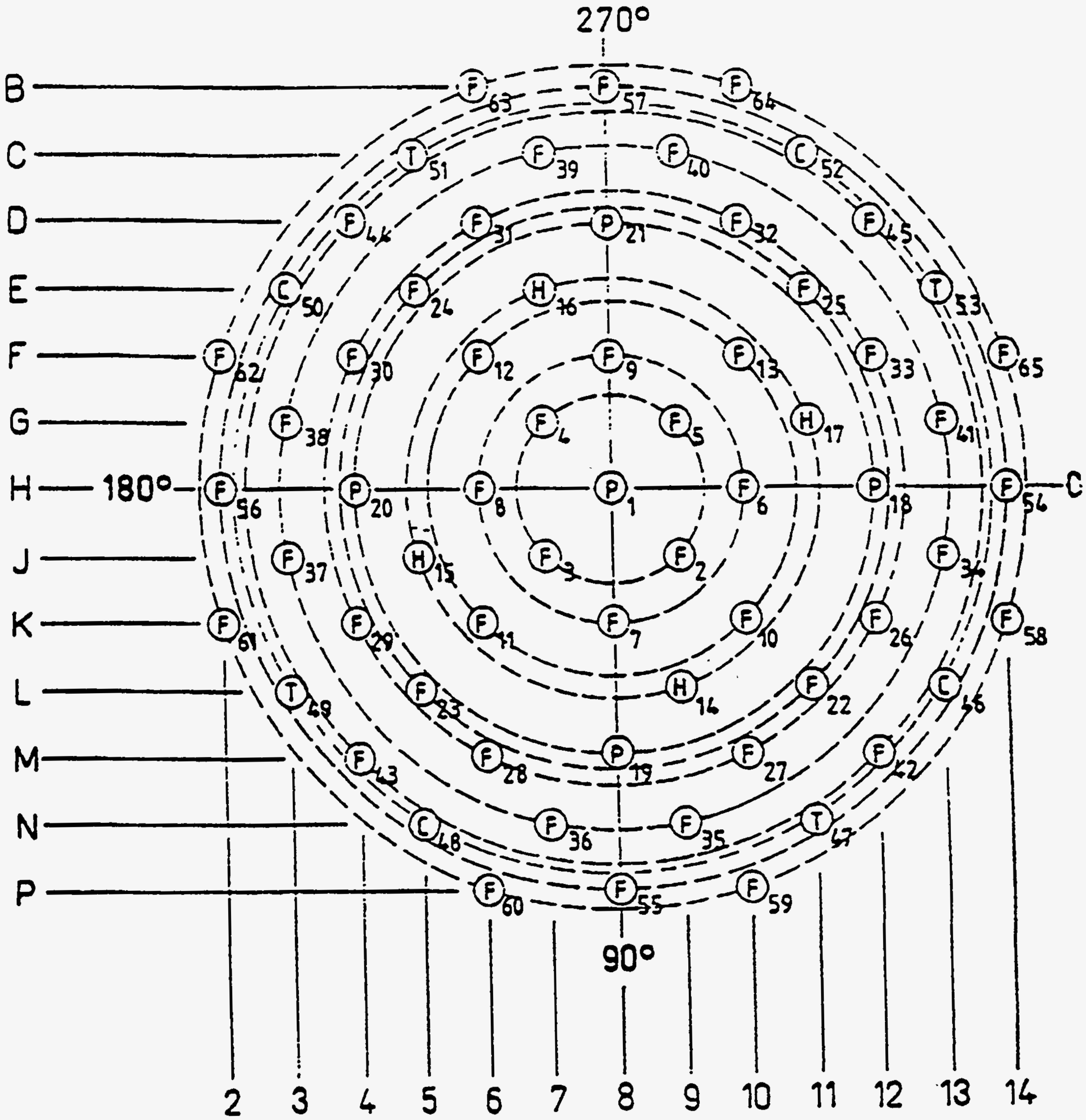
    # RPV HEAD, SHORT TO MEDIUM TERM, REPLACEMENT REQUESTED



DIRECCION PROYECTO

Fig. 1

FIGURE 1: ASCO & ALMARAZ RV HEAD PENETRATIONS



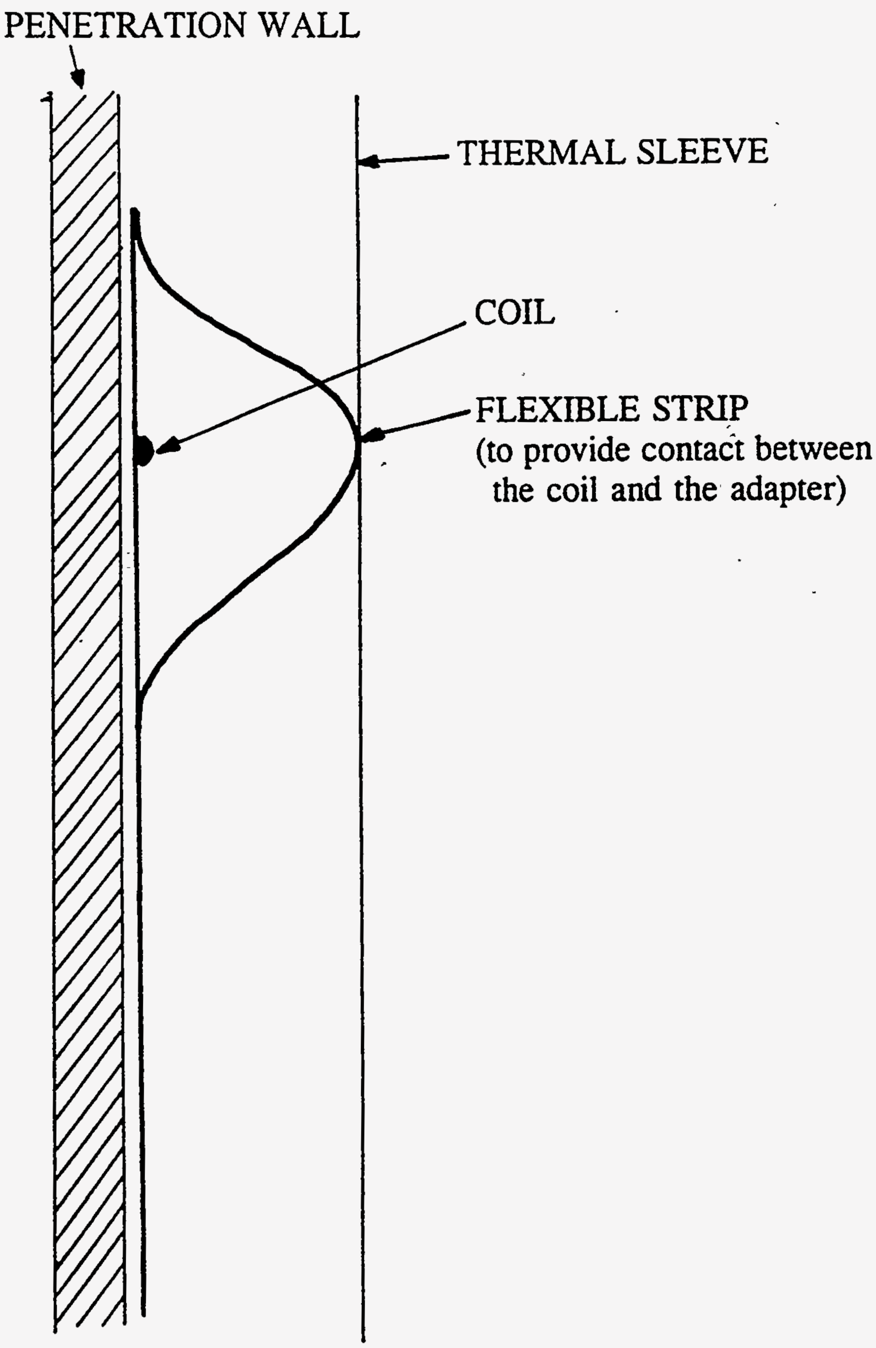


FIGURE 5: BLADE PROBE (ALMARAZ NPP)

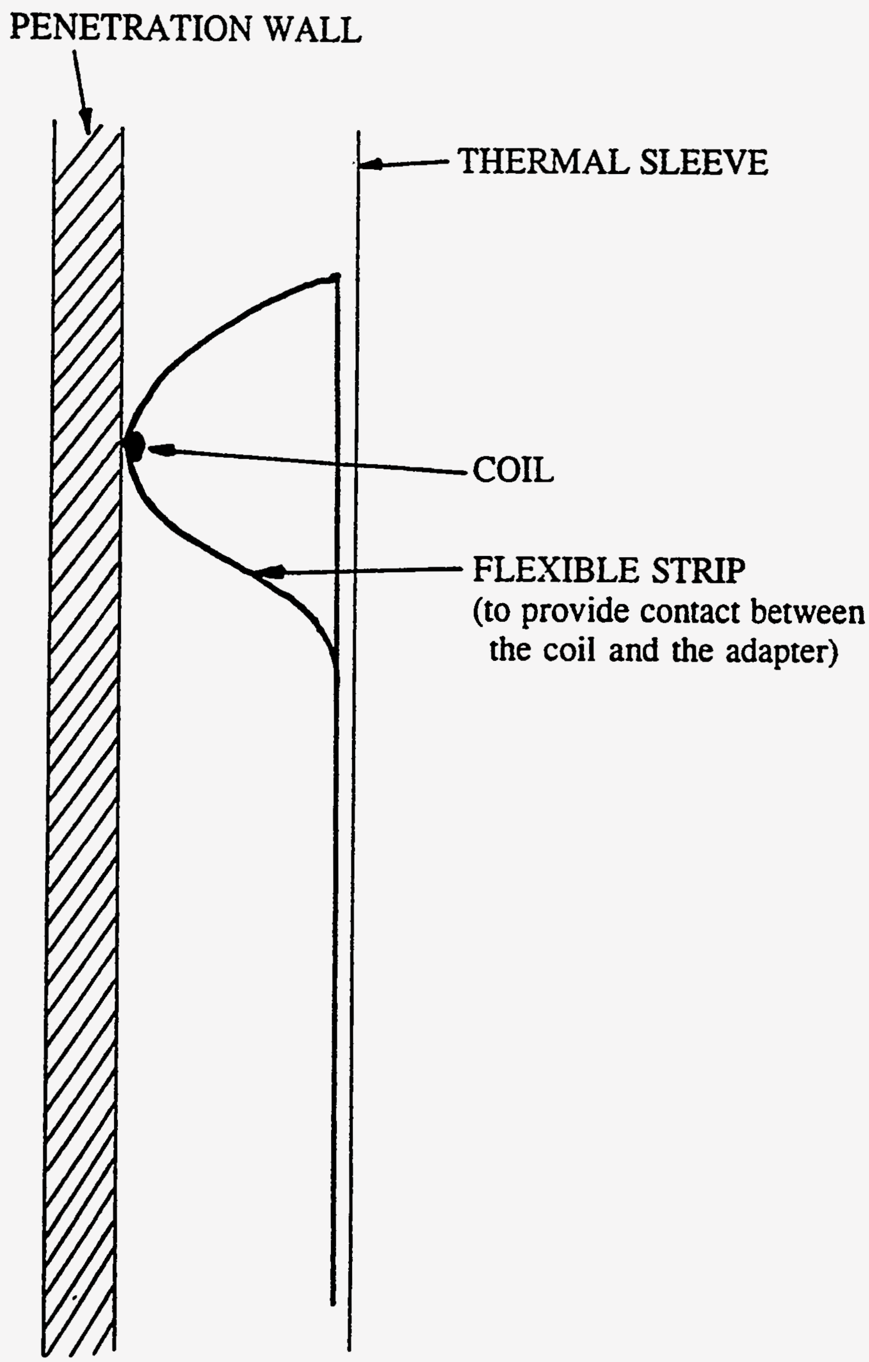
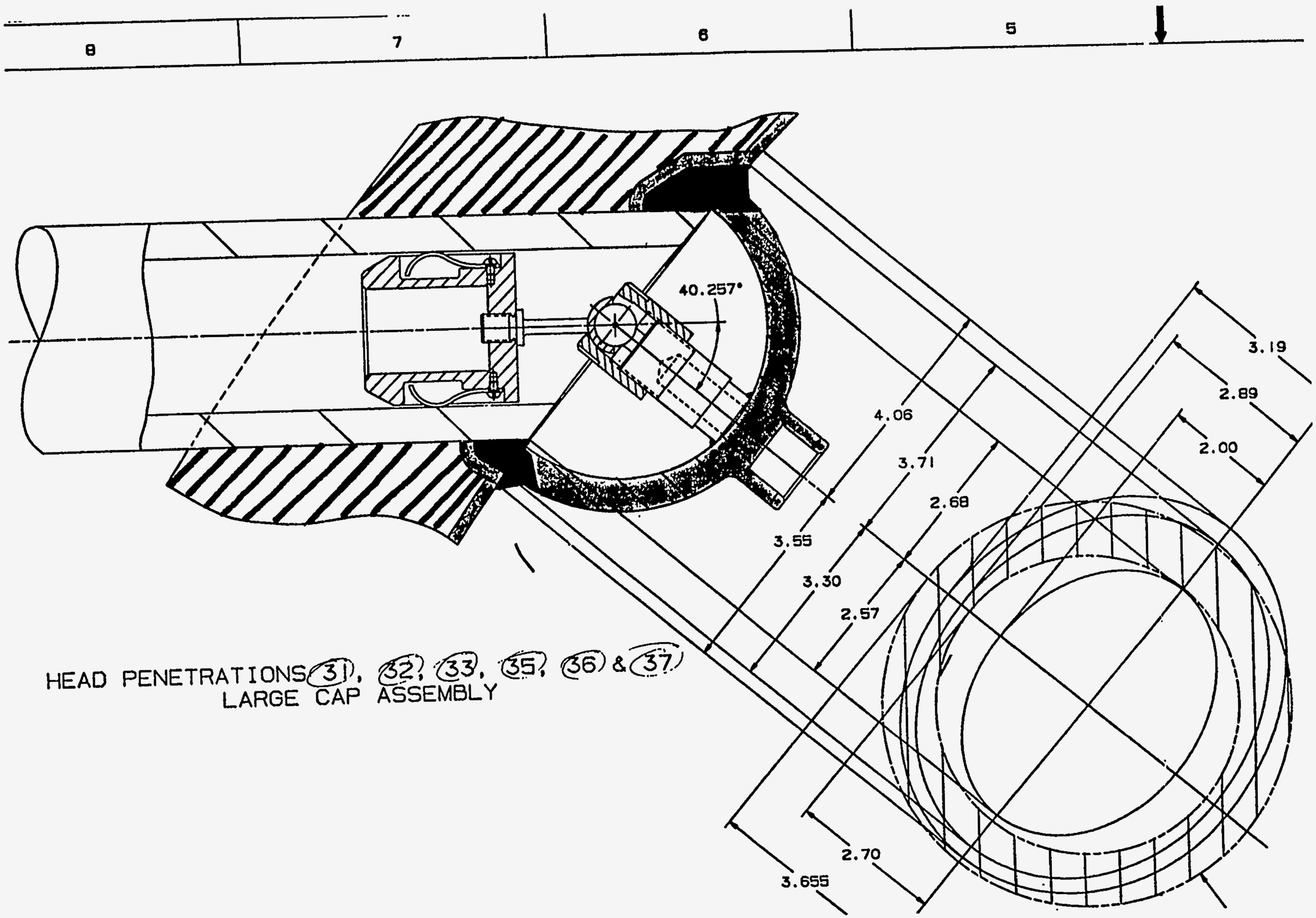
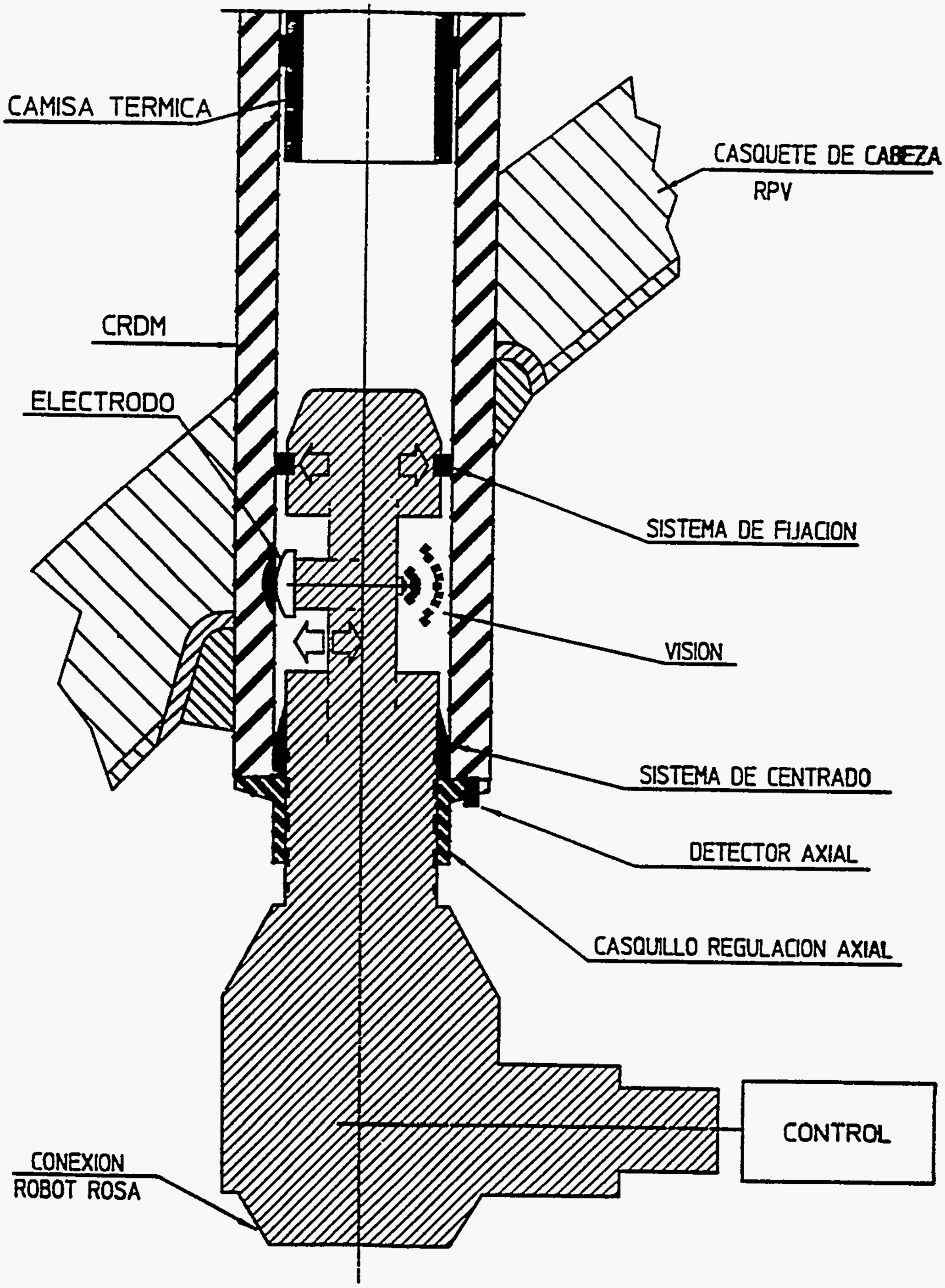


FIGURE 4 Bis: GAP SCANNER PROBE (ASCO NPP)



HEAD PENETRATIONS 31, 32, 33, 35, 36 & 37
LARGE CAP ASSEMBLY

HERRAMIENTA PARA RESANADO DE DEFECTOS



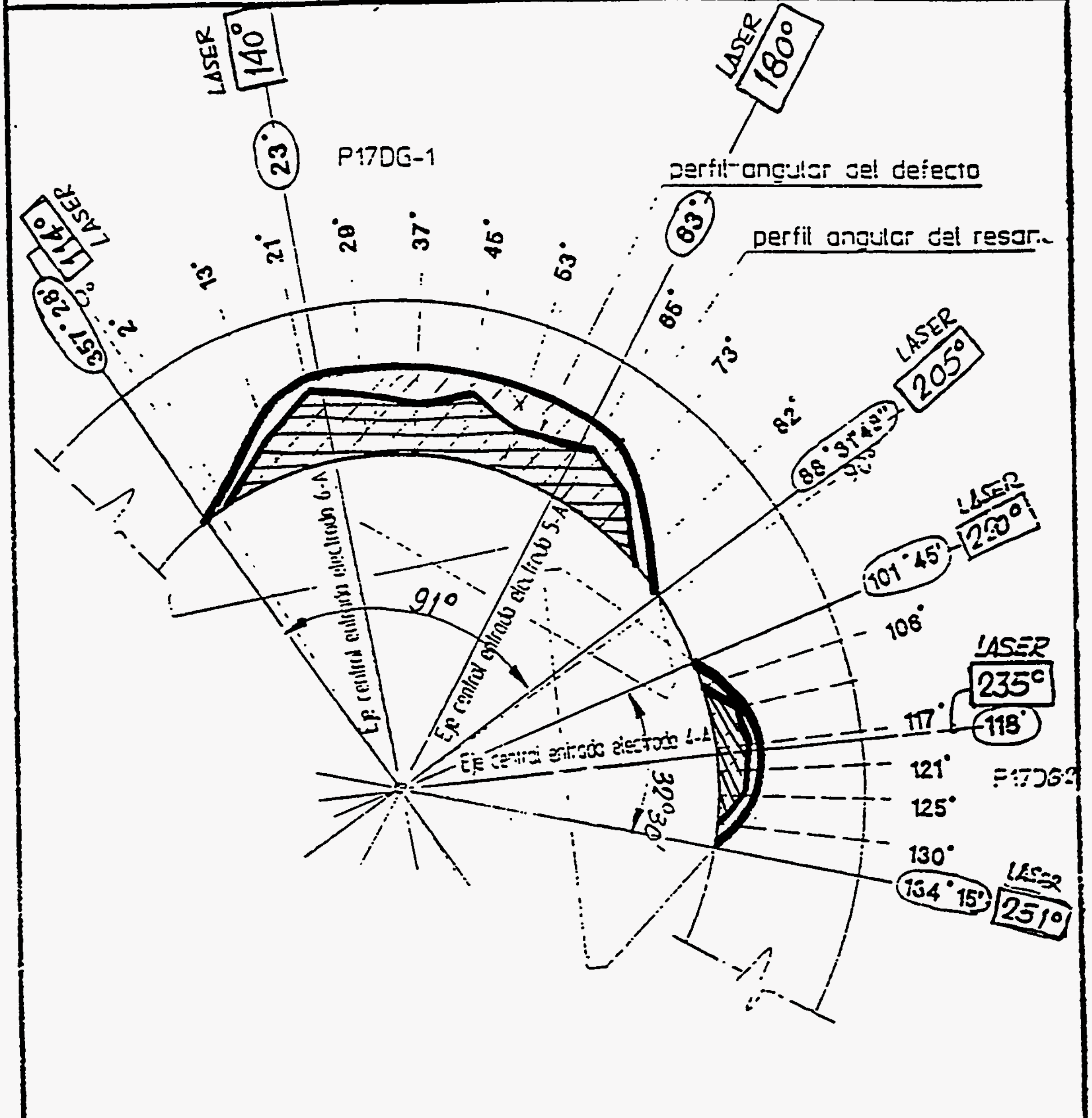
EQUIPOS NUCLEARES

ENSA

Dimensional Certificate

PENETRACION Nº 17

CONTRATO Job nº 4CM8	CLIENTE Customer UNION FENOSA	IPP.-4CM8-X17 REV.-1	OPERACION Operation 15 y 16
OBJETO Object: CONTROL DIMENSIONAL DEL NUEVO POSICIONAMIENTO DE EJES DE LOS ELECTRODOS 6-A, 5A y 4A (LASER)		PLANO Drawing 4CM8.010.103 REV.-4	HOJA Sheet 7 DE of 7



INSPECCIONES EXTERNAS			ENSA	
CLIENTE Customer	ENA Aut. Inspector	OTROS Others	PREPARADO Prepared by	APROBADO GESTION CALIDAD QA/QC Approved
FECHA Date 26/01/95	FECHA Date	FECHA Date	FECHA Date	FECHA Date 26/01/95

Fig. 10

PRIMARY WATER STRESS CORROSION CRACKING OF ALLOY 600

by
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ABSTRACT

As United States nuclear reactors have aged, a number of problems have arisen. Among these are primary water cracking (PWSCC) of Alloy 600 in PWRs. Since 1989, when PWSCC was identified to the Nuclear Regulatory Commission (NRC) as an emerging issue, it has been reported in several components, including control rod drive mechanism (CRDM) penetrations.

To address PWSCC of CRDM penetrations at U.S. plants, the industry developed a comprehensive inspection, evaluation, repair and mitigation program. Recent pilot inspections that revealed cracking at two of the three U.S. plants inspected indicate the problem is generic. Further, results of stress analyses indicate that an area of high stress exists that could cause cracking that would follow the J-groove weld. Such cracking was identified in a foreign reactor that had a resin intrusion. PWSCC of CRDMs remains an open issue.

Proactive NRC/Industry programs for inspection and repair or replacement of affected components are essential for continued operation of nuclear reactors and for license extensions.

Keywords: PWSCC, CRDM, penetrations, cracking, aging, Alloy 600, PWR, safety, inspection, materials problems, reactors

INTRODUCTION

Primary water stress corrosion cracking (PWSCC) was identified to the Nuclear Regulatory Commission (NRC), as an emerging issue in 1989, after leakage was reported from an Alloy 600 pressurizer heater sleeve penetration at Calvert Cliffs Unit 2. Other leaks have been occurring, since 1986, in several Alloy 600 pressurizer instrument nozzles at both domestic and foreign reactors from several different nuclear steam supplier vendors. In 1991 a leak was discovered in a Control Rod Drive Mechanism (CRDM) penetration at a European plant. Since this discovery, many European plants have conducted inspections and identified more cracked nozzles.

At meetings in 1992 with the Owners Groups, the staff discussed the significance of the CRDM leak at the European plant for domestic plants. Evaluations of CRDM nozzles in U.S. reactor vessels showed that they are not inherently less susceptible than European CRDM nozzles to PWSCC.

Subsequently, considering the generic implications of the cracking, the Nuclear Management and Resources Council (NUMARC), later NEI, coordinated the efforts of the PWR Owners Groups. The Owners Groups submitted safety analyses for their vessels supplied by Westinghouse, Babcock and Wilcox and Combustion Engineering. Upon reviewing them, the staff concluded that the cracking was not a significant safety issue. The basis for this conclusion was that the cracks, with perhaps one exception, were short and axial, leakage would occur before catastrophic failure and visual examination would find leaks. Degradation of the vessel head by borated water in a creviced area was predicted to occur very slowly and so an event such as ejection of a CRDM would be unlikely. Reduction of radiation exposure to personnel performing inspections and repairs was also desirable. Field experience in foreign countries has showed that occupational radiation exposures could be greatly reduced in a well-planned examination program, which would include the use of remotely controlled or automatic equipment.

CRDM INSPECTIONS/EVALUATIONS

To address PWSCC of CRDMs at U.S. plants, the industry responded by developing a comprehensive inspection, evaluation, repair and mitigation program.

As a follow-up to the safety assessments, NUMARC submitted proposed generic acceptance criteria for flaws identified during inservice examinations of vessel head penetrations (VHPs) to the NRC in July of 1993. The NRC accepted the acceptance criteria for axial flaws above and below the J-groove weld (the weld that holds VHP to the vessel head and is part of the primary pressure boundary) and circumferential flaws below the J-groove weld, but rejected the criteria for circumferential flaws above the J-groove weld. Cracks below the J-groove weld do not violate the reactor vessel pressure boundary even if they are through wall, and axial and circumferential cracks below the J-groove weld were determined to be acceptable by the NRC staff. Axial cracks above the J-groove weld may result in a leak that would be detected by surveillance walkdowns before significant damage could occur. Circumferential cracks above the J-groove weld could result in the ejection of a control rod drive mechanism resulting a large break loss-of-coolant accident. Furthermore, the stress analyses conducted as part of the owners groups safety assessments predicted that it would be very unlikely that circumferential cracks would form due to the stress distributions in the VHPs. For these reasons, the NRC requested that circumferential crack-like indications above the J-groove weld be reported to the NRC for disposition.

In 1993 the industry developed remotely operated inservice inspection equipment and repair tools that reduced radiation exposure. Techniques and procedures developed by two vendors were successfully demonstrated in a blind Qualification Protocol developed and administered by the Electric Power Research Institute (EPRI) Nondestructive Evaluation (NDE) Center. In the demonstrations, examinations by rotating and saber eddy current and ultrasonics detected and sized all the flaws. Results of the qualification testing demonstrated that the vendors' inspection procedures and personnel would be highly likely to find any PWSCC in the CRDM nozzles.

Three licensees volunteered to conduct VHP inspections as part of the NUMARC program. In 1994 the licensees for Point Beach, Oconee, and D.C. Cook performed the first pilot inspections.

The eddy current inspection conducted by the Wisconsin Electric Power Company vendor (Westinghouse) at the Point Beach Nuclear Generating Station in April 1994, uncovered no crack-like indications in any of the 49 VHPs.

The eddy current inspection by the Duke Power Company vendor (Babcock & Wilcox) at the Oconee Nuclear Generating Station in October and November 1994, revealed 20 crack-like indications in one penetration. Ultrasonic testing (UT) could not quantify the depth of these indications, because they were shallow. (UT cannot accurately size defects that are less than one mil deep (0.03-mm)). These indications may be associated with the original fabrication. The licensee's analysis indicates that they will not exceed the acceptance criteria before the next outage. The licensee will reexamine and analyze the affected VHP during the next outage to see if the indications will exceed the acceptance criteria before the next outage. It will continue to reexamine until it finds no growth has occurred for two cycles, or until its analyses predict the cracks will exceed the acceptance criteria before the next inspection cycle. In the latter case, the licensee will repair or replace the VHP.

The examination by the Indiana & Michigan Electric Company vendor (Westinghouse) at D.C. Cook revealed three clustered crack-like indications in one penetration. The indications were 46-mm, 16-mm, and 6-8-mm in length and the deepest flaw was 6.8-mm deep. The tip of the 46-mm flaw was just below the J-groove weld. The acceptance criteria permits a through-wall, axial crack of any length below the J-groove weld since such a crack does not violate the primary pressure boundary. The licensee's analysis indicates that these flaws will not grow to exceed the acceptance criteria before the next outage at which time the licensee will reinspect. The licensee will continue to reinspect as described above.

The inspection results are consistent with the owners groups' analyses and the PWSCC found in the CRDMs in European reactors. The results observed during the these three VHP inspections do not pose a threat to safe plant operation. Based on the owners groups safety assessments, a leak in a VHP would be detected before significant damage could occur to the VHP or the reactor vessel. This would result in the deposition of boric acid crystals on the vessel head and surrounding area that would be detected during surveillance walkdowns.

However, the fact that cracking was found in 2 of 3 U.S. vessels indicates the problem is generic. Further, results of stress analyses indicate that an area of high stress exists that could cause cracking that would follow the J-groove weld. Such cracking was identified in a foreign reactor that had a resin intrusion. PWSCC of CRDMs remains an open issue.

The staff believes that continued interaction on this generic issue between the NRC and industry is warranted. The staff believes the industry should continue its proactive approach to this problem and develop an integrated inspection plan, determining the required inspection frequencies and repair techniques. Evaluating data from a foreign plant that had a resin intrusion and experienced circumferential cracking is warranted. Previous stress analyses should be reviewed, in particular to determine the potential for axial stresses to drive circumferential cracking.

The susceptibilities of Alloy 600 components in the primary pressure boundary should be ranked and current inspection programs should be reviewed to determine their adequacy in identifying potential degradation by PWSCC. Probabilistic risk and system assessments of potential degradation by PWSCC should be performed to confirm decisions on the extent and type of inspections that are adopted. These tasks, which require considerable effort, are probably best addressed by NEI or owners groups, rather than by individual licensees.

CONCLUSION

As the United States fleet of nuclear reactors has increased in age, a number of problems have arisen as a result of that aging. These problems include primary water cracking of Alloy 600 in PWRs. Proactive NRC/Industry programs for inspection and repair or replacement of affected components are essential for continued operation of these nuclear reactors. These programs are also essential as licensees consider license extensions for their facilities. These plants are licensed for 40 years and can be granted an extension for an additional 20 years of operation if all of the NRC rules and regulations are met. Proper handling of potential age related problems will be a key consideration in the granting of a license extension.

THE ALLOY 600 HEAD PENETRATION ISSUE
PAST EFFORTS AND FUTURE PLANS

IAEA SPECIALISTS' MEETING

May 2-4, 1995

W.E. Bamford

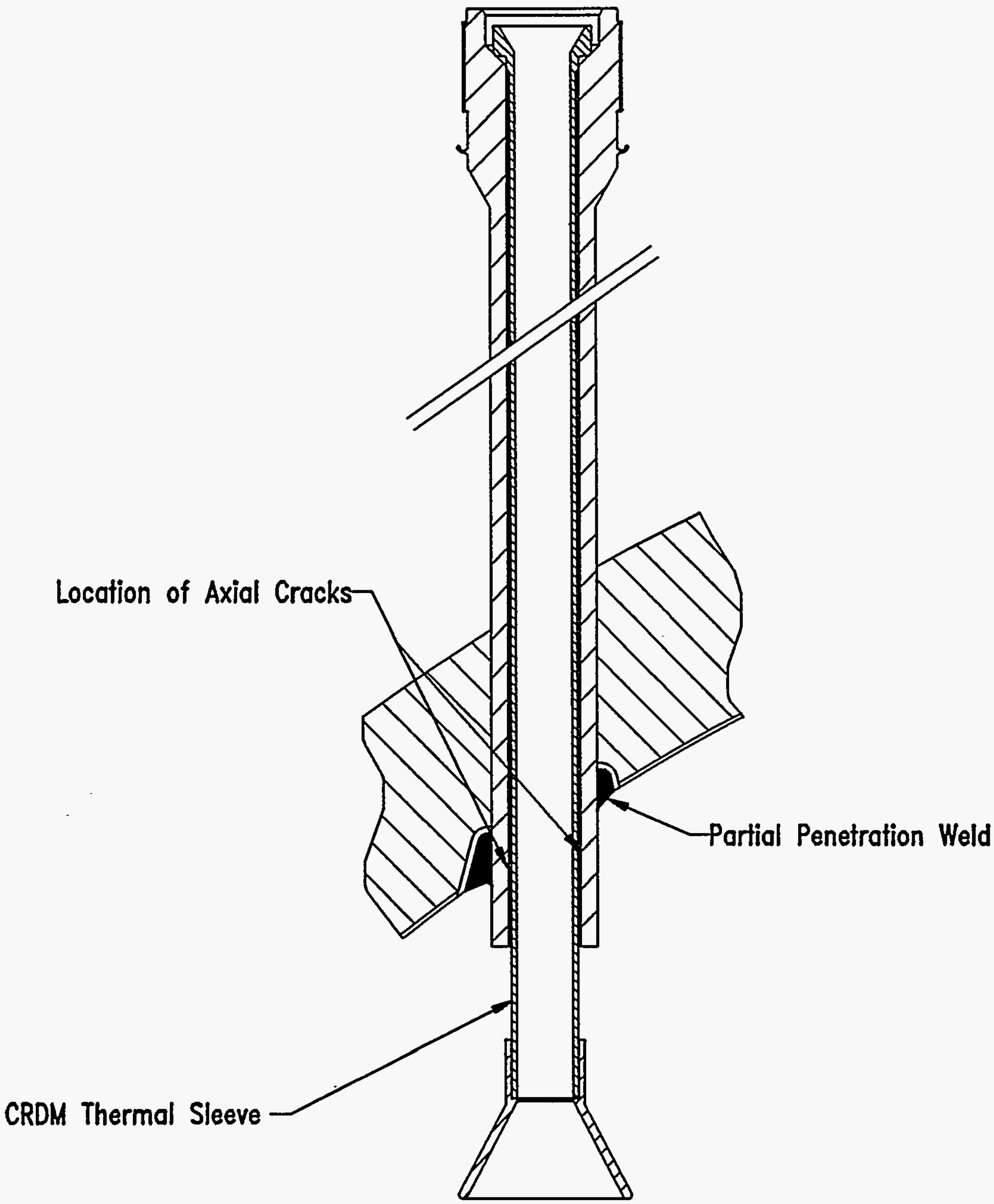
J.F. Duran

Nuclear Technology Division

Westinghouse Electric Corporation

TOPICS

- **Past Experience**
  - Plant Inspection Overview
  - Westinghouse Owners Group Program
- **Joint Owners Group Program**
  - Inspection Criteria
  - Evaluation Procedure
  - Inspection Demonstration
- **Key Accomplishments**
- **Future Plans**


Location of Axial Cracks
Partial Penetration Weld
CRDM Thermal Sleeve

PLANT INSPECTION OVERVIEW

Westinghouse Plant Inspection Summary

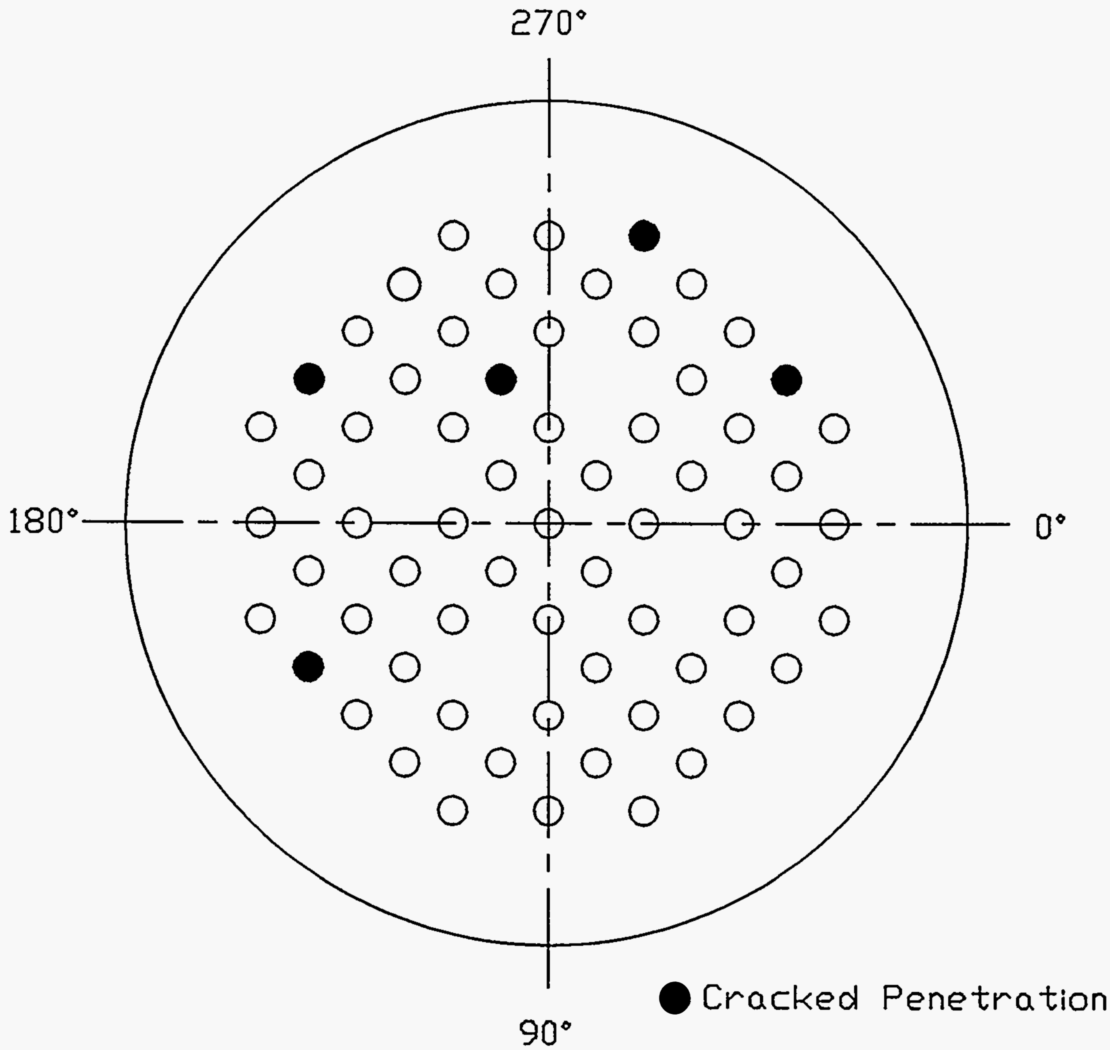
Country	Plant	Total Penetrations	Penetrations Inspected	Penetrations With Indications
Sweden	Ringhals 2	65	65	5
	Ringhals 3	65	60	0
	Ringhals 4	65	65	2
Switzerland	Beznau 1	36	36	2
	Beznau 2	36	36	0
Belgium	Tihange 1	65	65	0
	Tihange 3	65	65	0
	Doel 1	49	49	0
	Doel 2	49	49	0
Spain	Asco 1	65	20	0
	Asco 2	65	20	0
	Almaraz 1	65	20	0
	Almaraz 2	65	20	0
	Vandellos	65	20	0
Brazil	Angra	40	40	0
United States	Point Beach 1	49	49	0
	Cook 2	78	71	1
Totals	**17**	**987**	**750**	**10**

RINGHALS 2
PENETRATION TUBE CRACKING

- **1992 Inspection**
  - All 65 Penetration Tubes Inspected
  - 5 Penetrations With Indications

Penetration No.	Indication	
	Length (mm)	Depth (mm)
19	Zones	0.7
53	16	4
54	9	2
57	5	Very Shallow
68	5 & 15	Very Shallow

  - Indication in Penetration 53 Removed By EDM
- **1993 Inspection**
  - A Few Additional Indications Found
  - Propagation of Old Indications Negligible
    - Slight Increase In Length
    - No Growth In Depth
    - Differences Within Measurement Tolerances
- **1994 Inspection**
  - No Significant Changes In Indications



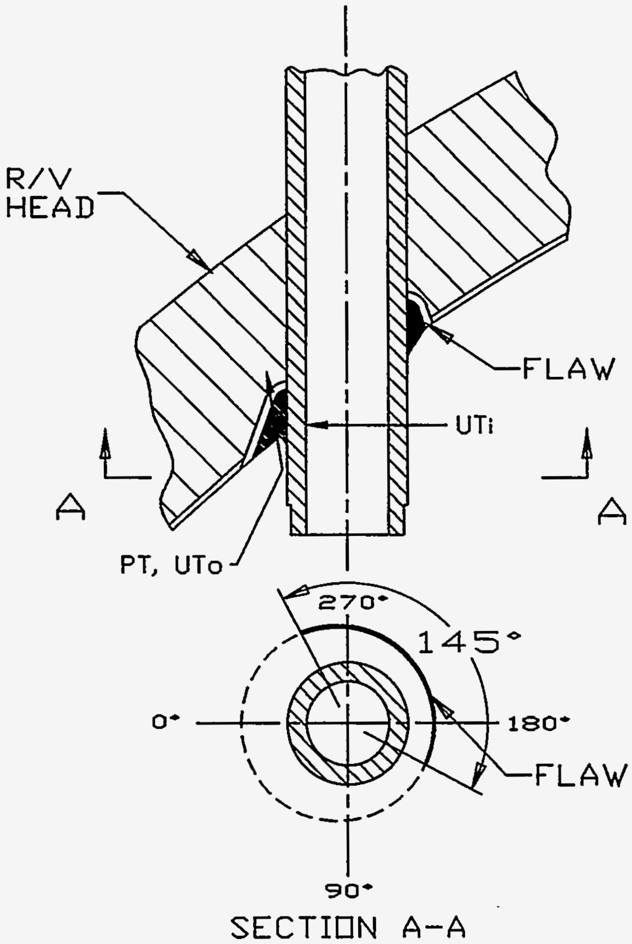
RINGHALS 2

RINGHALS 2

PENETRATION ATTACHMENT WELD CRACKING

- **Westinghouse 3 Loop Plant**
- **1992 Inspection**
  - ⇨ Liquid Penetrant Surface Examination of 3 Penetrations
  - ⇨ No Indications Found
- **1993 Inspection**
  - ⇨ Initial Liquid Penetrant Inspection Of 6 Penetrations
    - ➢ Surface Cracks Found In The Cladding Of One Of The Six - Penetration 62
  - ⇨ Shallow Boat Samples Removed From Penetration 62
    - ➢ Cracks Confirmed To Be Shallow
    - ➢ Extended Intermittently Through The Thickness Of The Cladding
  - ⇨ Ultrasonic Inspection Of Penetration 62 Indicated An Area Of Lack Of Fusion In Attachment Weld Just Above Surface Cracks
  - ⇨ Larger Boat Samples Removed Confirmed Lack Of Fusion
  - ⇨ Ultrasonic Inspection Of All 65 Penetrations Indicated Other Penetrations With Minor Weld Lack Of Fusion

RINGHALS 2
REACTOR VESSEL HEAD - PENETRATION WELDS
1993 INSPECTION RESULTS SUMMARY



	IND./TOTAL INSPECTED
Penetrant Inspection	1/8
UT_o -	20/65
UT_i - 1)	22/65
2)	6/65

UT_o: Fusion Zone Between Buttering & Base Metal

UT_i: 1) Inspection Of Interface Between Penetration & Weld
2) Inspection Of Interface Between Weld & Vessel Head

- **Only Three Penetrations With Lack Of Fusion >20% (Maximum 24%)**
- **Analysis Indicates That ASME Code Allowables Can Be Satisfied With Over 80% Of Fusion Zone Being Unfused**

RINGHALS 4

- **Westinghouse 3 Loop Plant**

- **1992 Inspection**
  - ⇨ All 65 Penetration Tubes Inspected
  - ⇨ 2 Penetrations With Indications
    - ➢ 5 mm and 7 mm long
    - ➢ Shallow In Depth
  - ⇨ No Repairs Required

- **1993 Inspection**
  - ⇨ Penetration Tube
    - ➢ No Significant Change In Indications
  - ⇨ Penetration Attachment Weld
    - ➢ Liquid Penetrant Examination Of 6 Welds - No Cracklike Defects Found
    - ➢ Ultrasonic Examination Of 10 Welds
      - o Low Amplitude Indications In Penetration/Weld Zone
      - o Number Of Indications And Extent Much Less Than Ringhals 2
      - o No Indications Detected In Head To Weld Zone

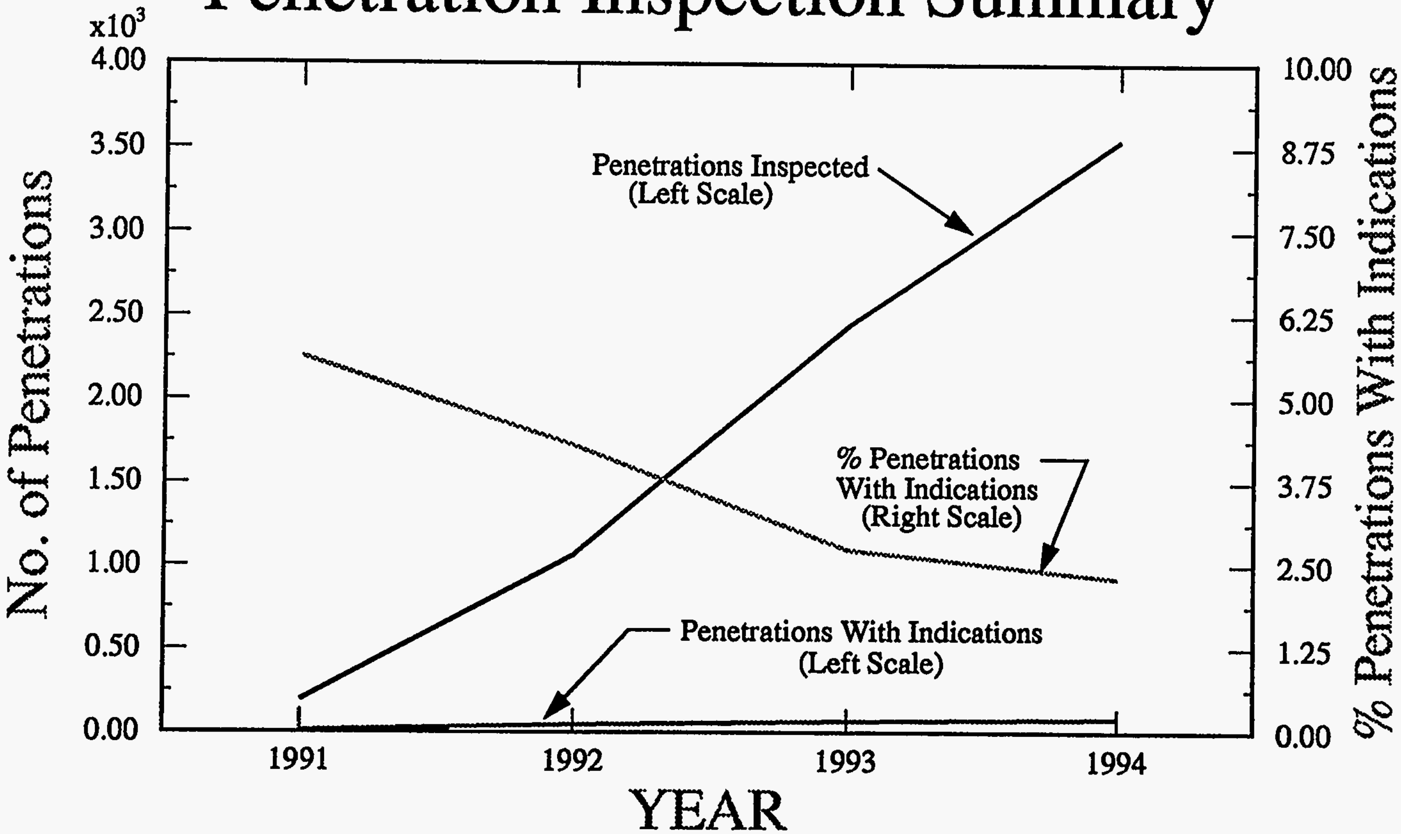
BEZNAU 1

- **Westinghouse 2 Loop Plant**
- **1992 Inspection**
  - ⇨ Two Penetrations Found With Small Indications
    - ➢ 3 mm Long; < 1 mm Deep
    - ➢ 28 mm Long; < 2 mm Deep
- **1993 Inspection**
  - ⇨ No Additional Cracks Found
  - ⇨ No Detectable Change In Existing Defects
- **Next Inspection Scheduled For 1997**

D. C. COOK UNIT 2

- **Westinghouse 4 Loop Plant**
- **Inspection Performed In September 1994**
  - ⇨ Eddy Current Examination of 71 of 78 Penetrations
    - ➢ Three Indications Found In One Outermost Penetration
    - ➢ Indications Axial
    - ➢ Located Below Weld Region
  - ⇨ Ultrasonic Examination Performed On Indications
    - ➢ Two Indications Below Ultrasonic Detection Limit (2 mm depth)
    - ➢ One Measurable Indication
      - o 45 mm Long
      - o 6.8 mm Deep
  - ⇨ Analysis Performed Justifying Operation For Next Fuel Cycle

Penetration Inspection Summary



Westinghouse Owners Group Programs 1992-1994

- **Safety Analysis Conclusions**
  - ⇨ PWSCC In Vessel Head Penetrations Does Not Represent A Safety Concern
  - ⇨ Cracking Will Not Propagate To A Size Where Penetration Stability Limits Are Exceeded
  - ⇨ Penetration Leakage, Should It Occur, Will Be Small And Structural Adequacy Is Of Head Will Continue To Comply With ASME Code Limits For A Period Of 6 Years After Leakage Has Occurred

- **Microstructure Correlation Developed**
  - ⇨ Based On Metallographic Evaluations Of Representative Materials Compared To Material Certifications

- **Predictive Damage Model**
  - ⇨ Probabilistic Methodology
  - ⇨ Tool To Establish When "Detectable" Cracks Initiate
  - ⇨ The Extent Of This Cracking In Compliment Of Penetrations In Vessel Head
  - ⇨ Crack Growth And Velocity In Each Penetration

Westinghouse Owners Group Programs 1992-1994

- **Repair Strategies**
  - Engineering Justification And Qualified Specifications Developed For 360° And Local Repairs

- **Crack Growth Studies**
  - Obtain PWSCC Data Quickly For Safety Evaluation Confirmation
  - Study Materials Effects: Heat To Heat
  - Key Parameters
    - Temperature
    - Stress Intensity Factors
    - Water Chemistry
    - Microstructure

- **Penetration Attachment Weld Evaluation**
  - Root Cause
  - Structural Assessment
  - Statistical Evaluation Of Possible Extent To All Vessels

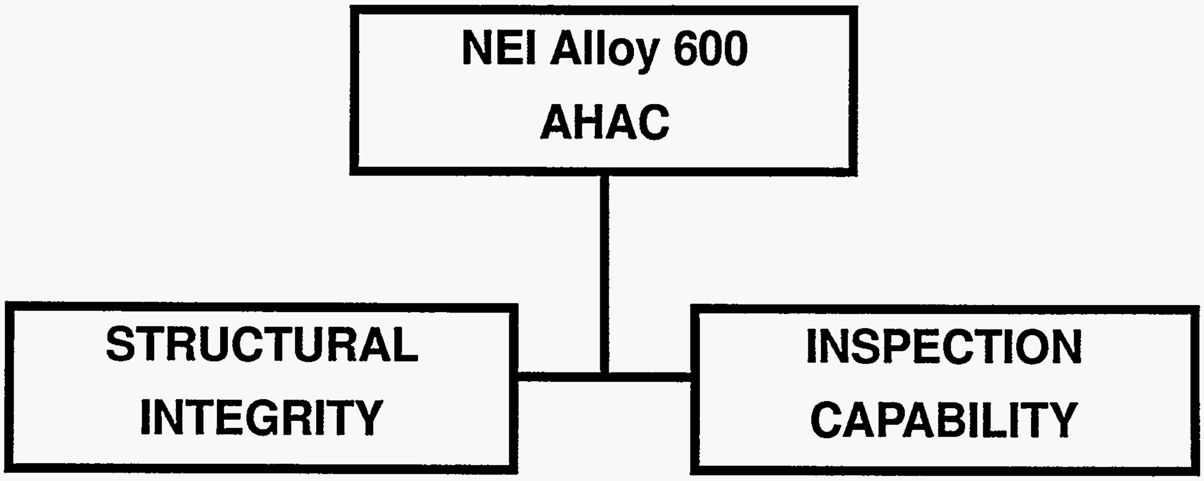
WESTINGHOUSE ACTIVITIES UPDATE

- **Remote Inspection Tooling Qualified To EPRI Standards**
- **Repair Tooling Developed**
  - RΘZ Positioner Tooling
  - ROSA Tooling
  - Repair Tooling Developed For:
    - Window In Thermal Sleeve
    - Removal And Rewelding Of Thermal Sleeve
- **Mitigation Methods**
  - Shot Peening
  - Zinc Addition
  - Sleeving
- **Replacement Head Designs**
  - Almaraz Replacement Heads
- **Support For Return To Power At Jose Cabrera (Sulfur Species Contamination)**

JOINT OWNERS GROUP

- Organization Of Group
- Safety Evaluation
- Flaw Acceptance Criteria
- Inspection Performance Demonstration

GROUP ORGANIZATION



GROUP GOALS

- Provide A Forum To Share Information On The Issue
- Develop Industry Consensus Approaches To The Following Areas
  - Safety Evaluation
  - Flaw Acceptance Criteria
  - NDE Performance Demonstration
- Provide A Focal Point For Industry Response To The Alloy 600 Head Penetration Issue

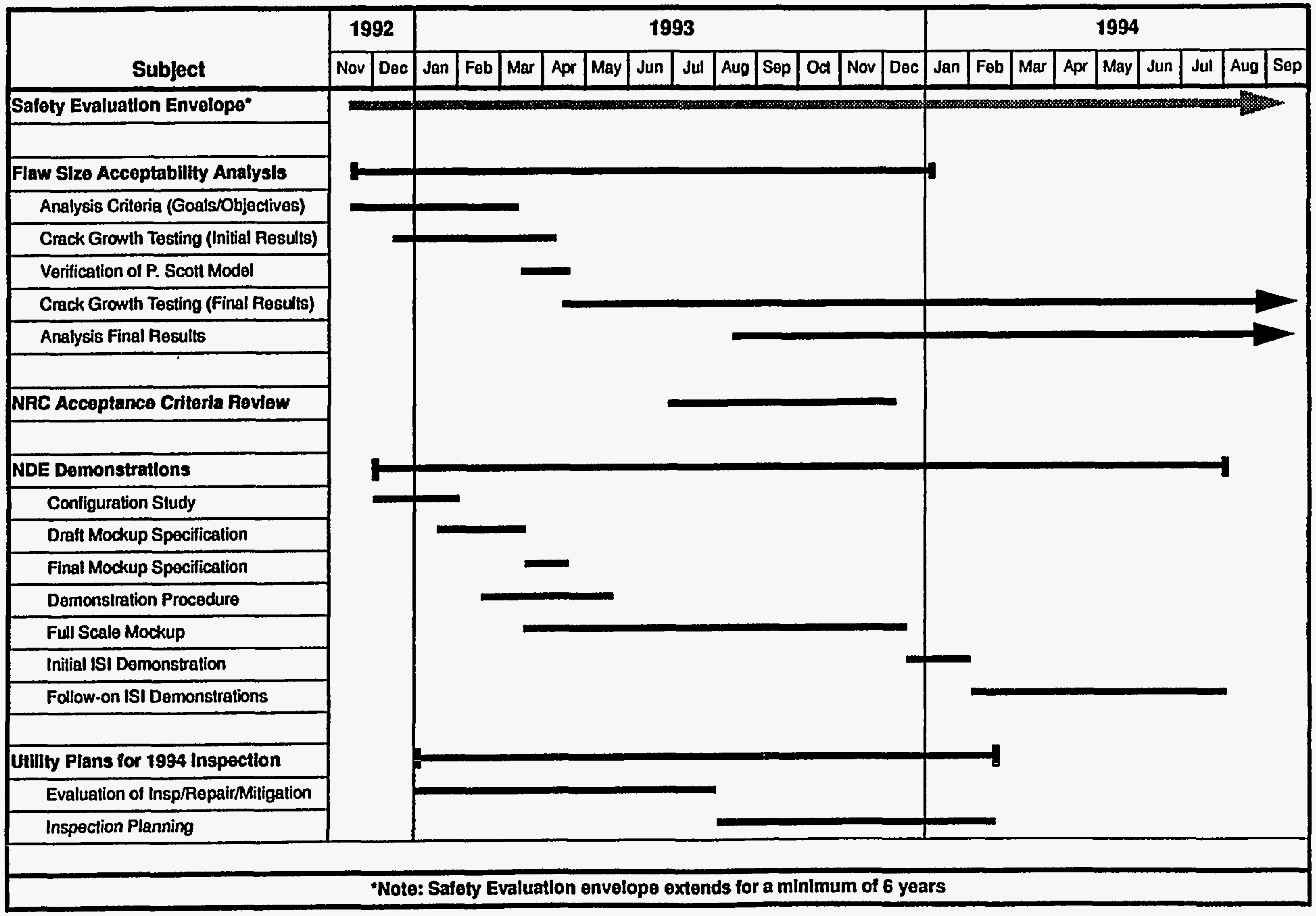
NEI ALLOY 600 AHAC MENBERS

- **US PWR Owners Groups**
  - BWOG
  - CEOG
  - WOG

- **US PWR Vendors**
  - B&W Nuclear Technology
  - ABB-Combustion Engineering
  - Westinghouse

- **Electric Power Research Institute**
  - EPRI Palo Alto
  - EPRI NDE Center
  - Dominion Engineering

NUMARC AHAC Planning Schedule for Industry Actions on CRDM Penetrations

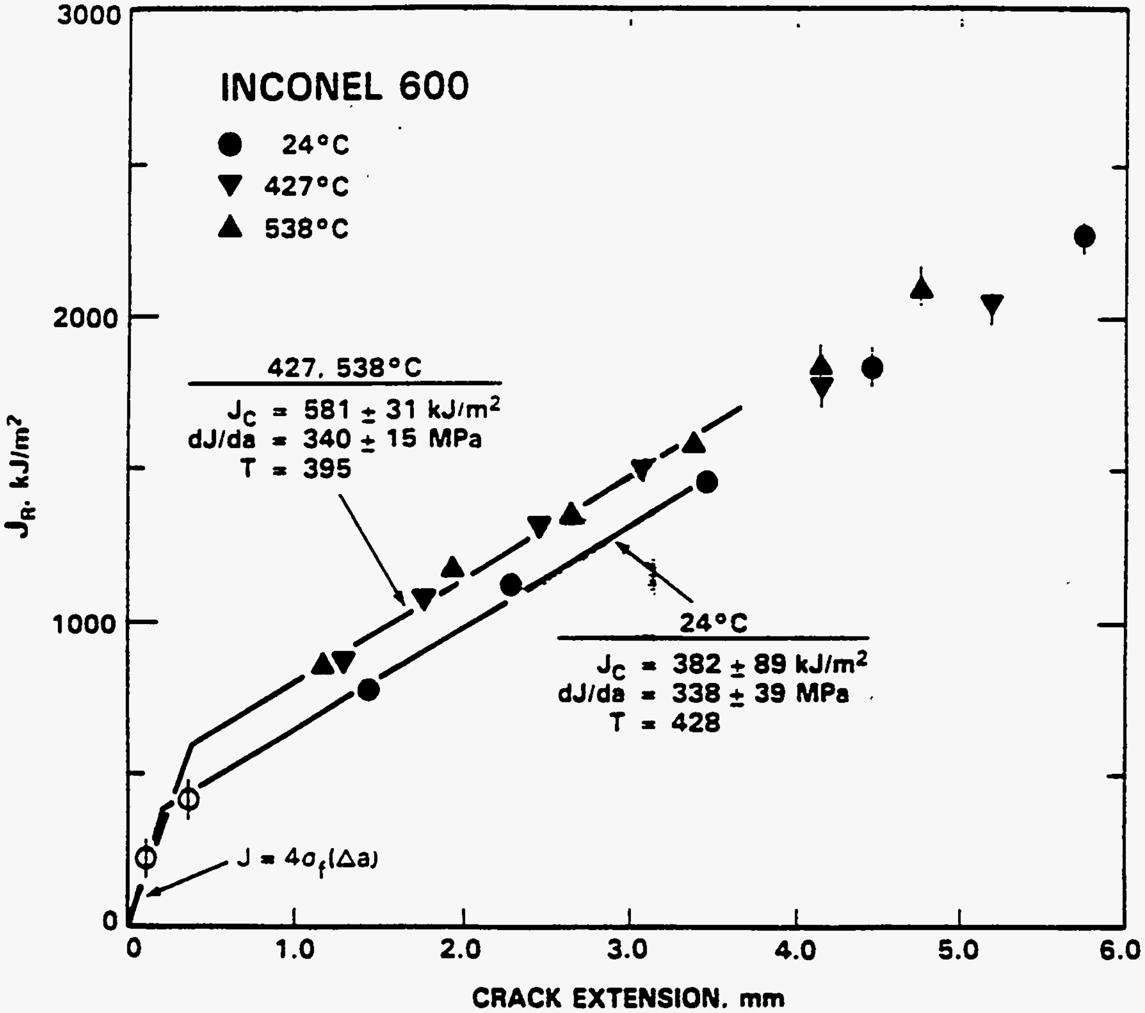


SAFETY EVALUATION

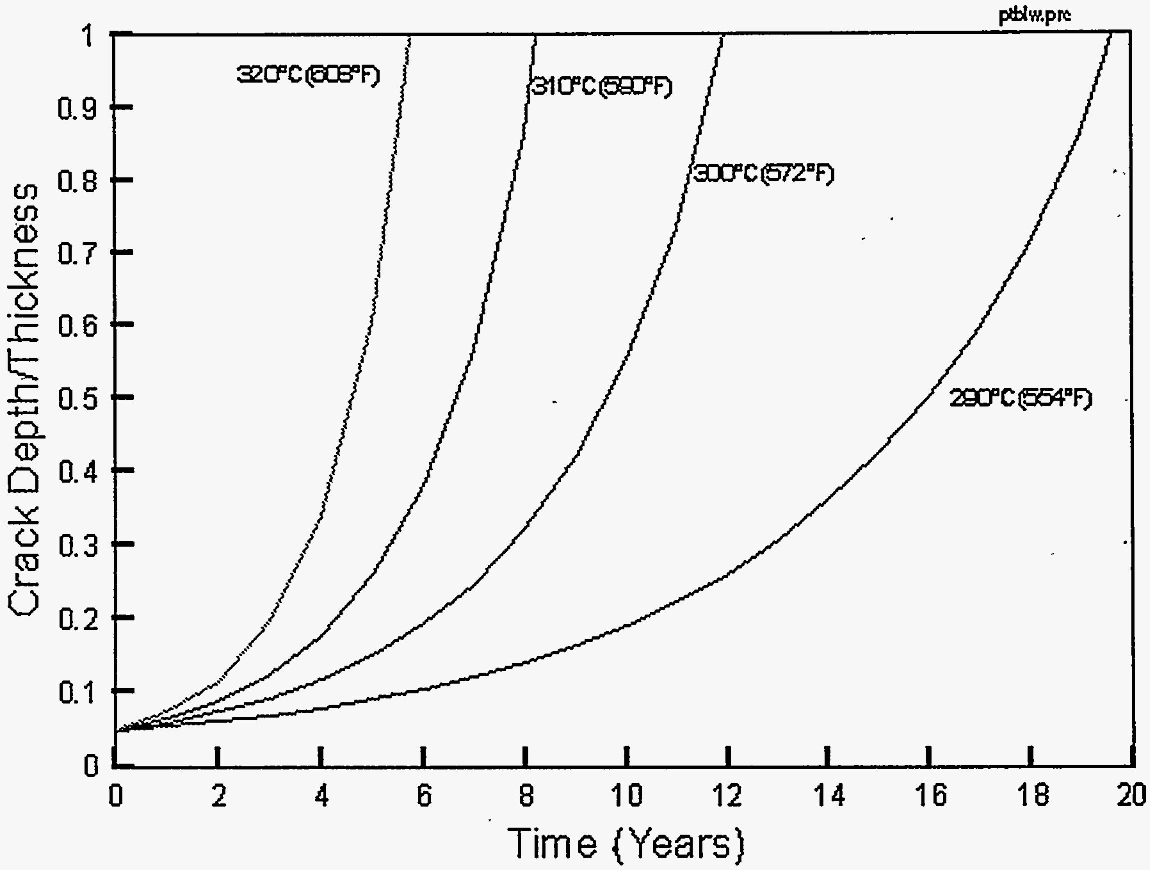
- Stress Analysis
- Fracture Analysis
- Crack Growth
- Results

KEY RESULTS

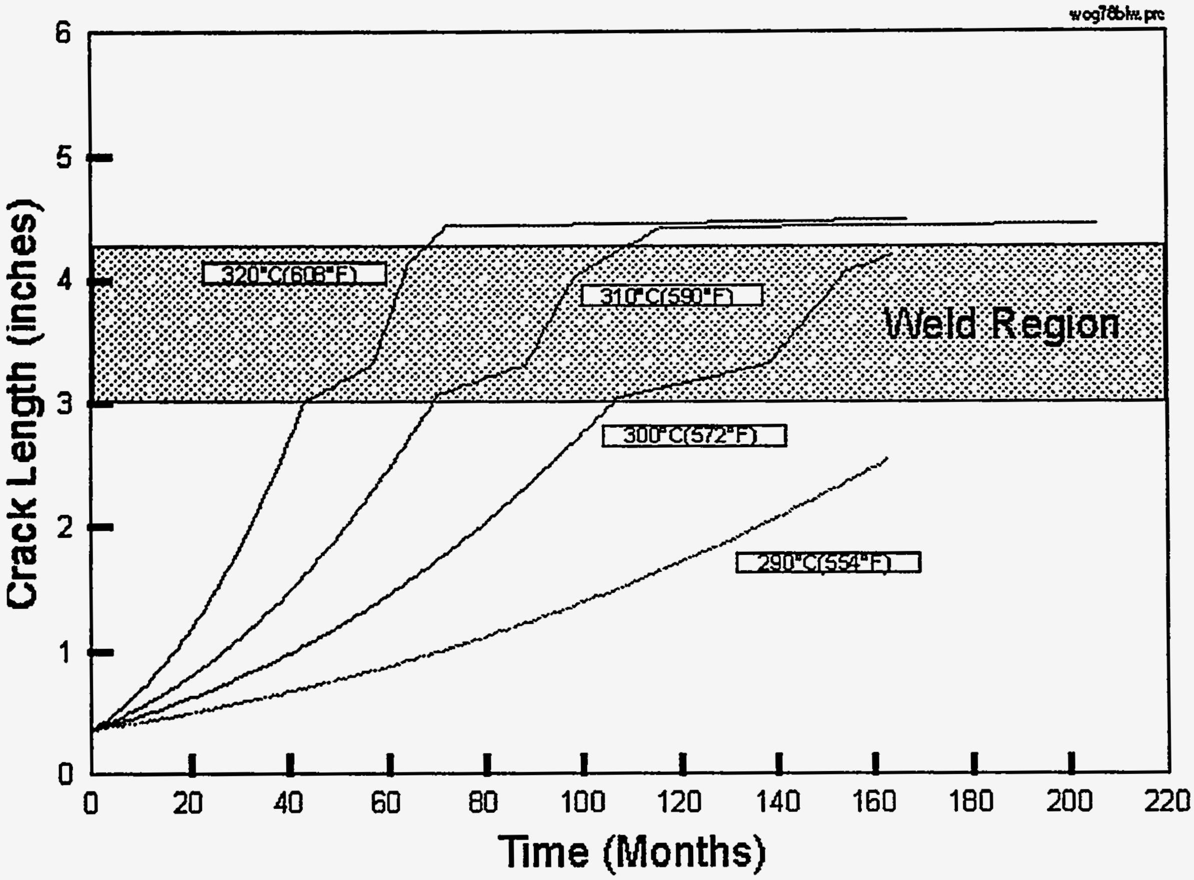
- **Hoop Stresses Are Greater Than Axial Stresses**
- **Locations Of Calculated Maximum Stresses Occur At Locations Where Cracks Were Observed**
- **Residual Stresses Due To Welding/Ovality Exceed Material Yield Point Locally**
- **Safety Evaluation Presented To NRC**
  - No Immediate Safety Concern
  - All Three Owners Groups Concurred
  - NRC Agreed



FRACTURE TOUGHNESS OF ALLOY 600 AT SEVERAL TEMPERATURES



Example of Crack Growth Calculations For Postulated Axial Flaws In A Head Penetration



Example of Crack Growth Calculations For Postulated Axial Surface Flaws In A Head Penetration

FLAW ACCEPTANCE CRITERIA

- ❑ Safety Assured (Safety Evaluation)
- ❑ Goal: Protection Against Leakage
- ❑ Flaw Characterization
- ❑ Flaw Acceptance

SUMMARY OF R.V. HEAD PENETRATION ACCEPTANCE CRITERIA

LOCATION	AXIAL		CIRC	
	a_f	1	a_f	1
Below Weld	t	no limit	t	.75 circ.
At and Above Weld	0.75t	no limit	0.75t	0.50 circ.*

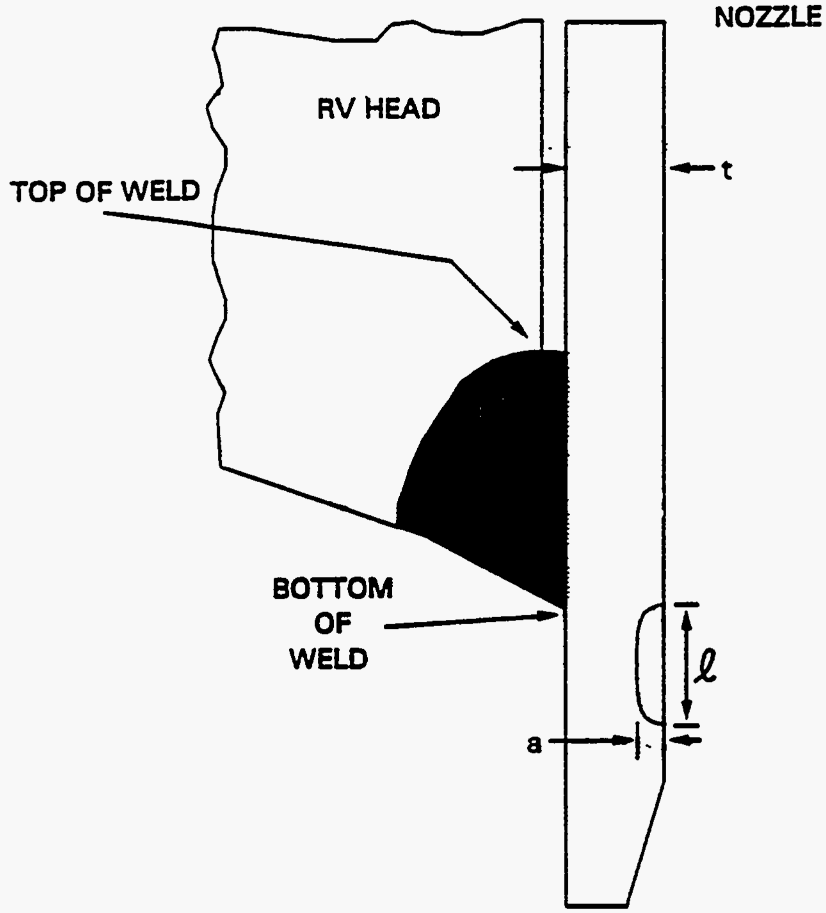
a_f = Flaw Depth as Defined in IWB 3600, Section XI

1 = Flaw Length

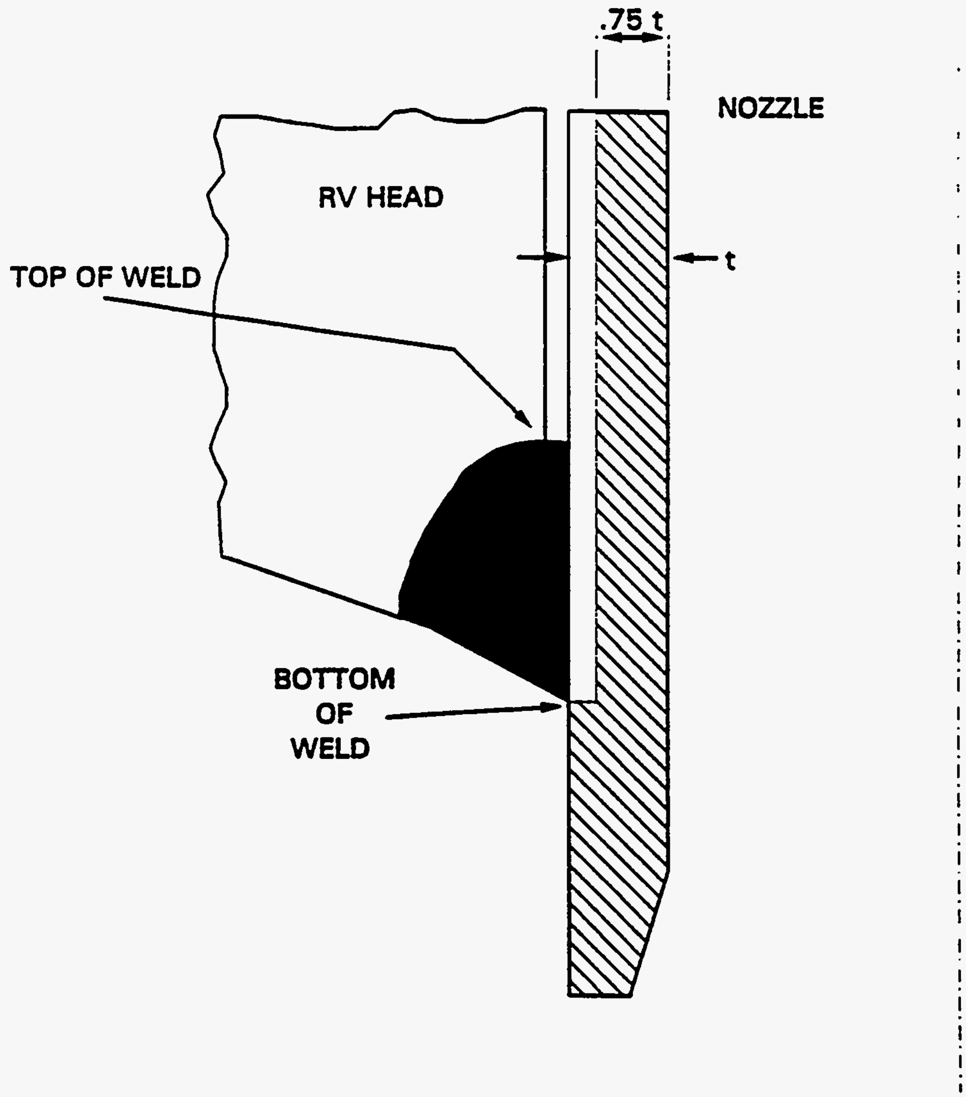
t = Thickness

* 0.10 circ. Stipulated By NRC For Point Beach

- **Flaws Which Exceed The Above Criteria Must Be Repaired Unless Analytically Justified (Analysis Shall Be Submitted To The Regulatory Authority Having Jurisdiction At The Plant Site)**
- **These Criteria Have Been Accepted By NRC**



TERMINOLOGY USED WITH ACCEPTANCE CRITERIA



LIMITING ALLOWABLE FLAW DEPTHS
CRDM HEAD PENETRATIONS

INSPECTION PERFORMANCE DEMONSTRATION

- Directed By EPRI NDE Center
- Mockups
- Flaws
- Comparison With Actual Flaws

STATUS OF CRDM PENETRATION NDE PROJECT

- **Industry Liaison**
  - ⇨ All Activities Have Been Coordinated through NEI
  - ⇨ PWR OEMs, Owners Groups, NEI, EPRI, & NDE Center Participate
  - ⇨ Design Principles for Mock-ups & Demonstration Protocol Discussed & Agreed upon
  - ⇨ NRC Briefed Several Times by NEI on NDE Demonstration Approach
  - ⇨ NRC Visited NDE Center Twice to Review Mock-up Development & to Discuss Protocol
  - ⇨ First Demonstration Completed 3/94 in Support of Point Beach
  - ⇨ Second Demonstration Completed 8/94 in Support of Duke Power

STATUS

- **Point Beach Unit 1 Inspected 1994**
  - ⇨ No Indications
- **D. C. Cook Unit 2 Inspected 1994**
  - ⇨ One Penetration With Indications
- **Oconee Unit 2 Inspected 1994**
  - ⇨ One Penetration With Very Shallow Indications

CONCLUSIONS

- A Small Percentage Of Inspected Penetrations Have Indications
- Most Indications Are Shallow
- Only A Few Have Required Repair
- Safety Evaluation Indicates No Immediate Safety Concern

KEY ACCOMPLISHMENTS

- Demonstrated Integrity: No Safety Issue
- Voluntary Inspections
- Developed Remote, Accurate Inspection Capability

FUTURE PLANS

- **Technical Issues**
  - ⇨ Crack Growth Rate Testing
  - ⇨ Economic Decision Model Development
  - ⇨ In-Plant Confirmatory Measurements
    - ➢ Follow-up Inspection At D.C. Cook

- **Methods For Prevention Or Mitigation**
  - ⇨ Replacement Reactor Vessel Heads
  - ⇨ RCS Chemistry Additives
  - ⇨ Mechanical Design For Local Repairs

- **Industry Technical Exchange Forums**
  - ⇨ Joint Owners Group Meetings
  - ⇨ NRC Informational Exchange Meetings
  - ⇨ Conferences: e.g. NACE/ASTM Meeting

INTERNATIONAL ATOMIC ENERGY AGENCY

INTERNATIONAL WORKING GROUP
on
LIFE MANAGEMENT of NUCLEAR POWER PLANTS

SPECIALISTS MEETING
on
CRACKING IN LWR RPV HEAD PENETRATIONS

Philadelphia, USA
2 - 4 May 1995
*****

ALLOY 600 HEAD PENETRATION CRACKING AT EDF : SHORT AND LONG TERM MAINTENANCE STRATEGY

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Presented by M R.P.SIOUFFI (EDF -UTO, 6 Avenue Montaigne, 93192 NOISY LE GRAND Cedex.)

ABSTRACT

The discovering of a small leak on BUGEY 3 NPP Reactor Vessel Head (RVH) during 10 years hydrotest (end of 1991), was the beginning of an industrial challenge in terms of technical and economical aspects.

Because of pressure vessel structural engineering and safety analyses, and overall, in-service inspection results on nearly 9/10 of the RVH in EDF plants (47), alloy 600 RVH penetration cracking is not a safety concern for the short term : only a few axial cracks (3%) on more than 3,200 penetrations inspected, and low probability to have an important circumferencial crack on ID.

Nevertheless, conservative measures have been taken for defence in depth : systematic first in-service inspection for diagnosis, repair criteria for deep crack, reliable and performant leak detection system on line, T cold conversion for 4 loops plants.

For the long term, as the phenomenon is evolutionary, cracks have to be repaired. The definitive maintenance strategy is a purely economic choice. EDF decided to replace cracked RVH on a several years schedule and to fit periodic in-service inspection, in connection with crack risk initiation and propagation kinetic.

SHORT AND LONG-TERM MAINTENANCE STRATEGY FOR VESSEL HEAD ADAPTERS

This paper presents facts based on :

- Analyses,
- In-service inspection,
- Theoretical studies,
- Safety analyses,
- Laboratory results,

resulting in options for short and long-term maintenance and strategy in the treatment of vessel head cracks.

STATE OF VESSEL HEADS

In September 1991, a minor leak was detected by acoustic means, then by visual inspection during the statutory hydraulic retest during the first ten-yearly outage (at 207 bars, i.e. 1.2 times design pressure/80°C), on a peripheral penetration (at a hillside location) (T54) on the BUGEY 3 vessel head after 84,000 hours of operation.

Analyses revealed numerous longitudinal cracks on the internal surface of the penetration boring made of Inconel 600 forged from a bar.

Metallurgical tests confirmed that it was a question of stress corrosion under pressure in a primary environment.

Because this problem is potentially generic, evolutive and dispersed, an extensive in-service inspection program was started using robotic inspection means (Sabre machine using eddy current technique) in the middle of 1992.

The objectives were :

- to confirm that there were no thru-wall cracks in the old vessel heads,

- to know the state of each vessel head in the nuclear power system so as to prepare a maintenance strategy and decide upon one of several possible options (leave as is, temporary repairs, apply preventive measures, replacement of penetrations or vessel heads, etc.).

The result of the in-service inspections showed that the phenomenon is generic but that it only concerns about 3% of vessel head penetrations inspected.

Fewer than 10 penetrations had cracks of more than 5 mm and on a longitudinal orientation. No circumferential cracks were found on the internal penetration surface. Moreover, inspections performed on the vent tube in the center of the vessel head revealed nothing.

Furthermore, liquid penetrant tests were performed on 104 adapter welds on four vessel heads (weld metal in alloy 182 - non stress relieved welds) and revealed no cracks from stress corrosion.

IN-SERVICE INSPECTION OF VESSEL HEADS/MAIN RESULTS
(at the end of 1994)

Units	Hours of operation (10^3)	Vessel head temperature (°C)	Vessel heads inspected	Vessel heads cracked	Adapters inspected	Adapters cracked
900 MWe CP0, 3 loops **(1)**	80-107	596-599	6	5	378	19
900 MWe CPY, 3 loops **(2)**	42-97	552	27	18	1755	66
1300 MWe PQY- DPY, 4 loops	32-51	597 589**(3)** 558**(4)**	14	9	1080	20
		TOTAL	**47**	**32**	**3213**	**105** (3,3%)

(1) 4 vessel heads already replaced
(2) 2 vessel heads already replaced, 3 penetrations repaired for deep cracks
(3) drop of average primary fluid temperature for steam generators
(4) conversion of vessel head hot dome for cold dome.

ANALYSES AND EXPERIENCE FEEDBACK

Metallurgical analyses performed on the T54 penetration of the BUGEY 3 vessel head revealed a longitudinal penetration crack of 50 mm long on the internal surface and 2 mm on the external surface. Oxidization of the habit prove that the crack had traversed for some time, before the hydrotest was performed. The corrosion of the ferritic steel at the interface was assessed at approximately 60 micrometers in depth.

Two circumferential defects were detected :

- one on the external surface of the penetration, 3 mm long and 2 mm deep, at an incline of 30° to the norm on the external surface. This defect was caused by corrosion under stress once the main longitudinal crack had traversed,

- the other had mixed intergranular and interdendritic habit, 3.5 mm deep maximum on almost 110° of the developed circumference, with propagation in the deposited weld and buttering metal.

Metallurgical inspections performed on other samples of the cracked penetration confirmed the mechanism of corrosion under stress in relation to the microstructural characteristics of Inconel 600 (precipitation of intergranular or mixed chromium carbide), residual stress related to the angle of penetration incline to the vessel head and the state of the surface (machined or lapped).

Experience feedback determined the following :

- Defects are localized at azimuth angles zero and 180° ± 45°,

- Their height depends on the angle of incline at the penetration in relation to weld dissymetry,

- Cracks seem to start first at 180° and then at 0°.

As regards weld metal in alloy 182 for non stress-relieved welds, no indication of a crack type has been detected by liquid penetrant testing, which contradicts laboratory results that show a susceptibility of this material to corrosion under primary environment stress, by tests that were performed with a substantial deformation rate (greater than 1%) and machined surface states (grinding influence).

ANALYSIS OF STRESS AND EXPERIMENTAL MEASURES

Calculation of stresses for the geometry of a vessel head penetration is very difficult becasue it combines residual stress from production (welding), the effects of the resistance tests and those due to operations.

Consequently, theoretical analyses were complemented by experimental measures on new vessel heads before commissioning and on models, in order to better define the level of superficial stresses that caused the cracks.

Finally the main results are the following :

- Circumferential stresses are substantially higher (400/500 MPa) than longitudinal stresses (200/300 MPa) for a peripheral penetration,

- The level of circumferential stresses decrease on central rather than peripheral penetrations,

- For the central penetration, the levels of circumferential and longitudinal stresses are of the same amplitude but relatively low

- The effect of the hydrotest relieves the residual stresses of manufacturing, particularly at 0°, which results in a higher level at 180°, so confirming experience feedback from in-service inspection.

MECHANICAL ANALYSIS OF DEFECTS

Mechanical analysis of rupture was performed on the basis of elastoplastic calculations designed for central and peripheral penetrations.

The main results revealed :

- Substantial margins as regards rupture for longitudinal cracks : the critical defect is 350 mm long above the vessel head,

- The critical circumferential defect represents more than 90% of the thickness for a non-traversing crack and over 90% of the circumference for a thru-wall crack,

- From a strict safety point of view, the only defect that could prove to be significant is the circumferential defect, starting above the weld.

Further studies were performed to define the margins involved in tearing of the vessel head : 27 mm for a semi-elliptical, longitudinal defect in normal and upset operating conditions (13 mm in accidental situations).

CONSEQUENCES OF A LONGITUDINAL THRU-WALL DEFECT

Different cases have been studied :

- If the shrunk-on assembly is tight, the leak is confined to the bore and the environment is identical to the primary environment. As a result, the corrosion kinetics of ferritic steel in the vessel head at a temperature of 300°C is very low (several tens of microns/year) and the behavior of the external surface of the alloy 600 penetration is no different from that in the inside surface.

- If the assembly is not tight, a thru-wall crack above the weld will result in a primary leak. In case of a low leakrate, and on the basis of the BUGEY 3 experience :

. pressure drop in nominal operating conditions is low in the crack compared to pressure drop in the bore, which explains why the leakrate is low,

. vaporization takes place in the bore at a level close to the vessel head surface,

. study of the chemical environment, resulting from the concentration of chemical elements present (boron, lithium) in the gap, shows that the environment evolves to develop a composition of 10% B_2O_3 and 2.2% of Li_2O for a pH of 8.7 at 300°C.

For such a chemical composition, the corrosion kinetic of Inconel 600, based on laboratory tests, is low (0.06 micron/hour for a stress field strength more or less 15 MPa $\sqrt{m}$),

. The generalized corrosion of ferritic steel on the external surface of the vessel head depends on the concentration of boron (from 1mm/year to several tens of mm/year).

These factors, corroborated by experience feedback in BUGEY 3, confirm that a traversing longitudinal crack has no immediate consequence in terms of safety.

Nevertheless, in the context of in-depth defence, the EDF has qualified and implemented an efficient system to detect leaks (detection by nitrogen 13 at a flowrate > 1l/hour), on each vessel head either whose state is not yet known, or with defects of a depth of more than 5 mm.

SAFETY STUDIES

Studies show that the presence of longitudinal cracks do not really cast doubt in any significant manner on the integrity of the pressurized containment.

However, further safety studies have been undertaken to check that the consequences of a hypothetical ejection of a penetration are acceptable.

Firstly, the antimissile device installed above the vessel head to protect the third barrier limits the consequences of a hypothetical ejection of a penetration.

Furthermore, in a very hypothetical situation where four control rods jam, there would be no risk of fuel deterioration, nor criticality accident for the reactor during cooling and accident management .

CRITERIA OF REPAIRS

The basis of repair criteria is to consolidate the integrity of the pressure vessel and to prevent the consequences of a penetrating crack in terms of leak risk and the propagation of the crack in the buttering or deposited weld metal.

As a result, all penetrations whose ligament of sound metal (from the top of the crack to the external surface of the adapter) is less than 4 mm (i.e. 1/4 of the thickness) must be repaired. This takes account of results available concerning the kinetics of propagation and the uncertainties in measuring the depth of cracks by ultrasonic means.

Repairs can be either a replacement of the penetration (as in BUGEY 3 T54), or removal of the defect by machining and refilling by automated GTAW welding. The latter repair is considered as temporary and acceptable for a limited number of operational cycles.

KINETICS OF PROPAGATION

Based on all results internationally available for Inconel 600, the entire range of values has been considered in our analyses. In addition, a scientific program with tests planned on representative materials from the various batches of forged bars is underway.

Moreover, a special in-service inspection program has been defined to know the in-depth evolution of real cracks found in vessel heads in operation. This program concerns twenty or so penetrations in vessel heads of 900 and 1300 MWe units. It requires a point by point analysis of ultrasonic inspection results to compare the development of the crack front.

To date, results obtained have proved that the in-depth evolution perceptible at the end of 8,000 hours for certain cracks is less than 4 mm, i.e. a maximum propagation kinetic of 0.5 micron/hour.

MAINTENANCE OPTIONS OPEN

The definitive treatment of the problem of penetration cracking in vessel heads is a major economic challenge, requiring a long-term strategy.

The parameters and stakes are now quite clear. Special dispositions have been taken to ensure in-depth defence for the short-term. For the long-term, the challenge involves optimization of industrial techniques for non-destructive testing, repair procedures, organization of maintenance and related strategy.

The following options have been assessed :

Inspection and repair when necessary

This option is suitable for the short-term while waiting for a definitive solution for the long-term. It does not allow anticipation and may prove to be very dear economically and industrially.

It is not competitive for a phenomenon that has already started and developed, especially because of the relative costs between in-service inspection, repairs and replacement of vessel heads.

Corrective and preventive process

The following are considered :

- Electrolytic coating of nickel,
- Machining,
- Shot-peening, etc.

They are eliminated because of :
- Their cost of development and implementation,
- The need to treat the entire vessel head during successive outages,
- Uncertainty as to the lasting solution,
- They do not remove the need to monitor treated vessel heads in service.

Vessel head replacement

Finally, this is the surest technical and most economic solution to definitively solve the problem by judiciously replacing vessel heads as soon as they are damaged and at the latest, when a major, costly repair is necessary.

By optimizing the tools necessary for the raising and reassembly of control rod mechanisms as well as the correct coordination of the various operations involved, replacement of a vessel head has no impact on the critical path of normal unit outage (more or less a two-week operation for a 900 MWe reactor).

Naturally, improvements have been made in the design of replacement vessel heads, including :

- Use of Inconel 690, specially designed to correct the weaknesses of Inconel 600 in a PWR primary environment.

- Fine-tuning of chemical compounds, metallurgical state and microstructure in the 690 alloy to obtain maximum resistance to corrosion under stress.

- Welding of the adapters using an Inconel 152 type weld metal that has been laboratory tested for insensitivity to corrosion under stress in a primary environment.

- Use of a lapping type surface treatment of the inside surface of adapters in order to reduce the level of residual tensile stresses.

SHORT-TERM STRATEGY

- Continued diagnosis of vessel head status in all units by means of a reference "zero point" in-service inspection.

By the end of 1994, this zero point had been achieved for 47 vessel heads, only the vessel heads of the six most recent 1300 MWe units remain to be inspected in 1995.

- In-service monitoring of cracks already revealed on certain vessel heads, so as to better understand the kinetics of how the phenomenon propagates and to programme, in advance, the replacement of the most affected vessel heads.

- In this respect, six vessel heads are programmed to be replaced in 1995 (four on 900 MWe units and two on 1300 MWe units).

- Inspection of non-affected vessel head adapter welds by liquid penetrant test.

- Removal by machining of minor cracks detected on certain less-affected vessel heads so as to prolong their service life and facilitate the industrial treatment of the phenomenon.

LONG-TERM STRATEGY

The choice has deliberately been taken to replace vessel heads, after considering the following :

- The progressive nature of the phenomenon that seems unavoidable in the medium-term unless corrective measures are taken,

- The need to implement maintenance operations that do not affect the critical path of unit outages,

- The lasting nature of the solution and the uncertainties in the various options open for potential maintenance,

- The potential risk of deposited weld metal (alloy 182) cracking under stress over the long-term,

- The aim of finding a definitive solution to the problem for the entire lifetime of nuclear power plants.

Consequently, EDF has ordered 29 new vessel heads from FRAMATOME, in order to solve the problem for those vessel heads already affected and to deal with any future development of the phenomenon.

CONCLUSION

The generic problem of stress corrosion on the Inconel 600 penetrations of vessel heads is a major challenge in maintenance.

After two difficult years, 1991 and 1992, as regards availability of units for lack of automated inspection means, 1993 enabled the EDF to find a solution thanks to industrial means and the strategy employed (only 0.2% of non-availability).

Measures taken in the short-term to consolidate the integrity of vessel heads, and the necessary elements for a long-term planning strategy, economically optimized, have been accomplished.
The efforts of the EDF are now concentrated on the analysis of risk of deterioration in other areas using Inconel 600 in the main primary system.

FORM TI-12(L)
8/94

WAPD-T-3056

ACCELERATED STEAM PLUS HYDROGEN TESTS FOR ALLOY 600 WROUGHT AND WELDED SPECIMENS

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G. Economy
R. Jacko
G. J. Powell
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U. S. Department of Energy
Contract DE-AC11-93PN38195

BETTIS ATOMIC POWER LABORATORY WEST MIFFLIN, PENNSYLVANIA 15122-0079

Operated for the U.S. Department of Energy
by WESTINGHOUSE ELECTRIC CORPORATION

Accelerated Steam Plus Hydrogen Tests for Alloy 600 Wrought and Welded Specimens

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Abstract

A chemical cracking test has been used to quickly obtain intergranular stress corrosion cracking (SCC) as it occurs in Alloy 600 wrought metal and EN82 weld metal in deaerated high temperature water environments. The test, referred to hereafter as the doped steam test, involves exposing the specimen surface of interest to 3000 psig (20.7 MPa), 750 °F (400 °C) superheated stagnant steam raised from water that contains 100 ppm each of chloride, fluoride, sulfate, and nitrate sodium salts and to 10 psia (69 KPa) hydrogen partial pressure. Alloy 600 and EN82 bent beam specimens loaded to various known stress levels were exposed to this doped steam environment for periods of one to eight weeks. Threshold behaviors were determined from this test series. For specimens loaded above the threshold stress, SCC occurred in less than one week. Welded specimens with partial penetration EN82 welds were also subjected to the doped steam environment in the built-in crevice associated with partial penetration welds. During this test, cracking occurred in both the weld and wrought materials. The weld cracks initiated at the root and grew through the entire thickness of the weld throat in two weeks. Metallographic sections in the crack region and fractographs of the weld crack surface confirmed the presence of the multiple branched intergranular cracking expected in SCC. The results clearly indicate that the superheated stagnant steam with hydrogen and these four dopants provides a useful environment to assess the tensile stress condition of Alloy 600 wrought metal and EN82 weld metal specimens.

Key terms: chemical cracking test, Alloy 600, EN82 weld metal, stress corrosion cracking

Introduction

A chemical cracking test using doped steam plus hydrogen has been used to quickly obtain intergranular SCC as it occurs in Alloy 600 wrought metal and EN82 weld metal in deaerated high temperature water environments. The test results can be used to compare welding procedures and design features and to locate component regions with tensile stresses greater than the threshold in doped steam. This Alloy 600 and EN82 test serves the same purpose as the stainless steel, boiling magnesium chloride test (ASTM G36-87).

Previous studies[1,2,3] have demonstrated that high pressure (3000 psig (20.7 MPa)) and temperature (750 °F (400 °C)), hydrogen containing (5 to 11 psia (34 to 76 KPa)) steam accelerates intergranular cracking in nickel based alloys. The cracking is further enhanced by adding 30 ppm each of chloride, fluoride, and sulfate (as sodium salts) to the water from which the steam was raised[4]. To further accelerate the cracking, the dopant concentration was increased to 100 ppm and nitrate salt was added. Once this accelerated environment was developed, it was used to assess long term SCC susceptibility of various mockup designs[5,6,7] and to evaluate the effectiveness of several design feature changes[5,8]. A possible mechanism that explains how these steam and hydrogen environments accelerate cracking has been proposed[9]. The dopant concentrations in the water are generally greater than the solubility limits for these dopants in the steam and are a function of steam pressure. At 3000 psig (20.7 MPa), the soluble concentrations of salts were measured to be Cl^- (~90 ppm), F^- (~6 ppm), SO_4^{-2}(~4 ppm), and NO_3^-(~9 ppm) in steam raised from water containing 100 ppm of each dopant. Keeping the salt concentration slightly supersaturated helps maintain a consistent test environment.

The latter test environment of 3000 psig (20.7 MPa), 750°F (400 °C) superheated stagnant steam raised from demineralized water containing 100 ppm each of chloride, fluoride, sulfate, and nitrate sodium salts and with 10 psia (69 KPa) hydrogen partial pressure was used for the testing reported herein and is referred to hereafter as the doped steam test.

Test Description

Threshold Testing

To investigate the threshold stress at which cracks would initiate in Alloy 600 wrought metal and EN82 weld metal in doped steam, sixteen Alloy 600 and sixteen EN82 preloaded, smooth surface beam specimens were loaded in four point bending and exposed to doped steam in an autoclave. The preload stress levels for each material were 0, 10, 20, and 30 ksi (0, 69, 138, and 207 MPa). After each of 1, 2, 3, and 8 weeks, four specimens of each material (one specimen for each stress level) were removed from the autoclave and destructively examined for the presence of SCC.

Welded Specimen Description

Two welded specimens with partial penetration welds, as shown in Figure 1, were also subjected to doped steam. A partial penetration weld is defined as one that has a built-in crevice as a result of the weld geometry. Both welded specimens were fabricated using the same materials and the same machining and welding procedures. Wrought Alloy 600 material was joined with an EN82 attachment weld using tungsten inert gas. As a result of weld geometry, root extensions occurred at the location shown in Figure 1. Stresses in the welded specimens during the doped steam test were welding-induced residual stresses and pressure stresses from the 3,000 psig (20.7 MPa) steam pressure. In this application the pressure stresses were small (approximately 2,000 psi (13.8 MPa)) whereas the welding-induced residual stresses in this highly restrained weld geometry were believed to be approximately equal to the room temperature yield stress of the weld metal. Only the built-in crevice was subjected to the doped steam environment.

Test Apparatus

Figure 2 is a schematic of the test set up. Since the welded specimens were too large to fit into available autoclaves, the heat to maintain the 750 °F (400 °C) steam temperature was provided by an oven which contained the welded specimens, silver palladium cells, and doped water inventory. The built-in crevice replaced the autoclave as the pressure boundary. Two silver palladium cells were used to monitor and control the hydrogen content in the steam. The cells, which display a high permeation rate for hydrogen at temperatures greater than 300 °F (150 °C), allowed hydrogen partial pressure to be measured with one cell and hydrogen to be backfilled with the other.

Test Execution

The test start-up sequence was to measure the pH and conductivity of the doped water, perform pretest chemical analysis, leak test the pressure boundary, evacuate the pressure boundary, add doped water, and slowly increase the oven temperature (less than 150 °F/hr. (65 °C/hr.)). The quantity of doped water added was approximately ten percent more than the amount necessary to create the desired steam pressure in the test volume. During the test the temperature was maintained at 750 ± 12 °F (400 ± 7 °C), the pressure was maintained at 3,000 ± 75 psig (20.7 ± 0.52 MPa), and the hydrogen was targeted for 11 ± 1 psia (76 ± 0.0.007 KPa). It was not always possible to maintain the hydrogen partial pressure within these tolerances due to diffusion of hydrogen through the specimen walls. Measured hydrogen partial pressure varied from 3 to 12 psia (21 to 83 KPa). For subsequent tests the hydrogen partial pressure was maintained within desirable limits using a servo controlled system. At preset intervals or at times when the test was interrupted due to loss of pressure, ultrasonic (UT) inspections were performed to determine crack initiation time and size. The UT inspection times are listed in Table 1. Conductivity, pH, and chemistry determinations were made each time the test was shut down. Typical start up and shut down values for pH were 6.5 and 8.0, for conductivity were 1100 μS/cm and 500 μS/cm, and for F, Cl, NO_3, SO_4, and PO_4 sodium salts were 96, 101, 97, 95, and <0.1 ppm and 57, 40, 52, 53, and <0.1 ppm.

Test Results

Threshold Testing

The EN82 weld metal specimens stressed to 8 ksi (55 MPa) did not experience SCC for the eight week test duration. Therefore, the threshold for eight weeks is greater than 8 ksi (55 MPa). The EN82 weld metal specimens experienced SCC after one week of exposure to 15 ksi (103 MPa). Consequently the threshold is less than 15 ksi (103 MPa).

The Alloy 600 wrought metal specimens with no applied stress experienced some shallow surface cracking (0.0015 inches (0.038 mm)) that was concluded to result from a surface condition not driven by the applied stress. Consequently, the zero stress specimens were considered uncracked for the purposes of determining a threshold. At a stress level of 10 ksi (69 MPa), the Alloy 600 specimens did not crack in one week but did crack for all time periods greater than one week. Consequently, for exposure times of two weeks and longer, the threshold for Alloy 600 wrought metal is less than 10 ksi (69 MPa).

Welded Specimen Crack Description

Through weld, multiple branched cracking that followed the large columnar grain boundaries occurred in both welded specimens. Cracking initiated early, as cracks were detected at each inspection after the start of doped testing. Intercolumnar SCC grew through the throat of the weld in 366 hours for Welded Specimen 1 and 252 hours for Welded Specimen 2. The presence of through weld cracks was confirmed by the inability of the built-in crevice to hold pressure and by bubble formation on the outside surface of the weld. Table 1 and Figure 3 show the crack depth for Welded Specimens 1 and 2. The weld length direction is perpendicular to the cross section of Figure 1. The variation in crack depth in the length direction resulted from a nonuniform distribution of residual stress along the weld length.

Metallographic sections showed the following: (1) intercolumnar cracking in the direction of the weld throat, (2) significant crack depth through weld throat and wrought metal, (3) significant variation in crack shape along the length of the weld, (4) multiple branched cracking, and (4) weld root extensions that occur when the weld metal is deposited. The crack morphology determined from

TABLE 1. INTERCOLUMNAR SCC HISTORY FROM ULTRASONIC TESTING		
WELDED SPECIMEN	CUMULATIVE TEST DURATION (HOURS)	MAXIMUM CRACK DEPTH (PERCENT OF CORRESPONDING WELD THROAT THICKNESS)
1	209	71 %
	347	86
	366	90
2	209	69
	252	96

metallographic sections agreed with that of the UT inspections defined above. The metallographic sections as in Figure 4 showed cracks in the weld metal that were ninety percent through the weld throat. Most of the locations investigated had intergranular SCC in both weld and wrought metal as in Figure 4. A few sections had cracking in the wrought metal only. The significant variation in crack path along the length of the weld is demonstrated by Figures 5a and b which show different crack path for two sections spaced only 0.050 inches (1.27 mm) apart. In the wrought Alloy 600, the intergranular SCC was thirty to fifty percent through the wall. The cracks in the weld metal almost always initiated at the root extension as in Figure 5a. The cracks in the wrought metal usually initiated at the root extension but sometimes originated at the smooth surfaces of the Alloy 600 wrought metal.

After evaluating the metallographic sections, the weld pieces were broken out of the mounts, pulled apart to expose the crack surface, and evaluated using scanning electron microscopy (SEM). The fractograph of Figure 6 shows long columnar grain boundaries exposed by intercolumnar SCC and ductile tearing regions caused by pulling apart the weld piece. The ductile tearing occurred at the ligament near the surface of the weld that was not cracked by SCC and at small islands of weld metal surrounded by intercolumnar SCC. These latter features result from the discontinuous cracking characteristic of intercolumnar SCC. These two types of ductile tearing regions are labeled in Figure 6. Near the weld root where intercolumnar SCC initiates, higher magnification fractographs show surface deposits that completely cover the grain boundaries. Near the outside surface of the weld, the surface deposits do not cover the entire surface and are very sparse, since the time of exposure of these grain boundaries is shorter.

Interpretation of Results

The crack patterns in Welded Specimens 1 and 2 define locations of tensile residual stresses greater than the threshold stress for cracking in doped steam. The maximum tensile stresses in the partial penetration weld basically follow a forty-five degree plane starting at the weld root. The tensile stresses in the partial penetration weld are large enough through the full throat thickness to cause cracking through the full partial penetration weld throat. The tensile stresses in the wrought Alloy 600 are large enough to cause cracking though thirty to fifty percent of thickness of this material within the test duration. In locations where there was no intergranular SCC during the doped steam test, it can be concluded that either the combination of pressure and welding-induced residual stresses were compressive or small tensile (i.e., less than threshold stress levels) or that the stresses were relieved as the intergranular SCC progresses in the cracked regions. Intergranular SCC from doped steam tests is very similar to that observed in Alloy 600 studies[10].

Conclusions

A chemical cracking test using doped steam plus hydrogen has been used to quickly obtain intergranular stress corrosion cracking as it occurs in Alloy 600 wrought metal and EN82 weld metal in deaerated high temperature water environments. The test results can be used to compare welding procedures and design features and to locate specimen regions with tensile stresses greater than the threshold in doped steam.

Acknowledgments

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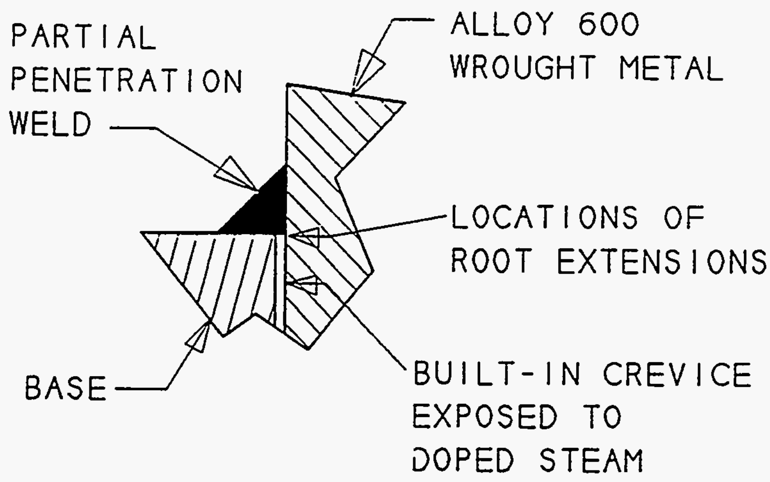

FIGURE 1. PARTIAL PENETRATION WELD GEOMETRY

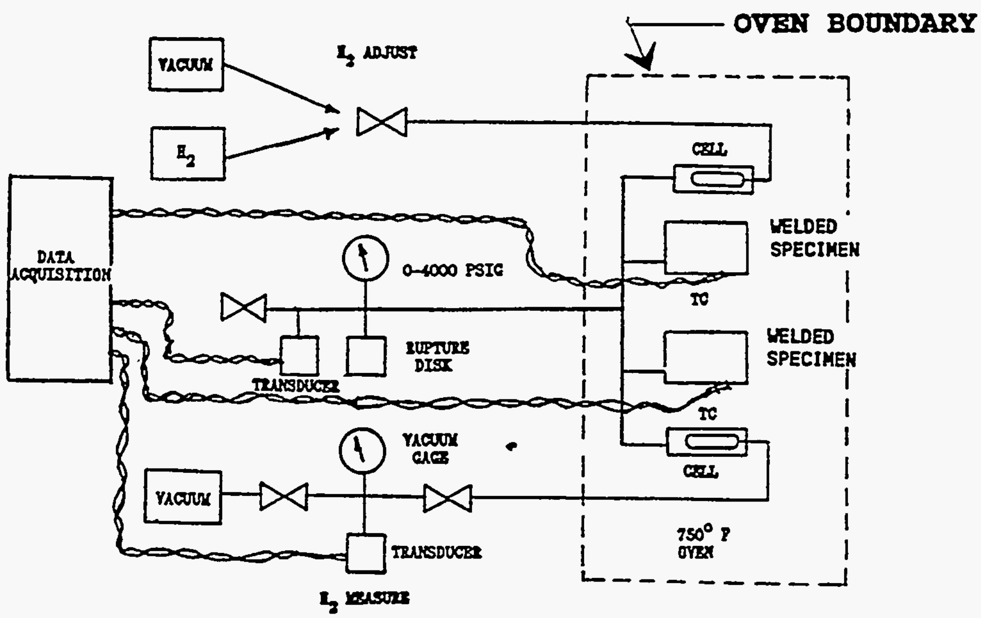

FIGURE 2. SCHEMATIC OF DOPED STEAM TEST
(Note: This figure will be redrawn for final copy)

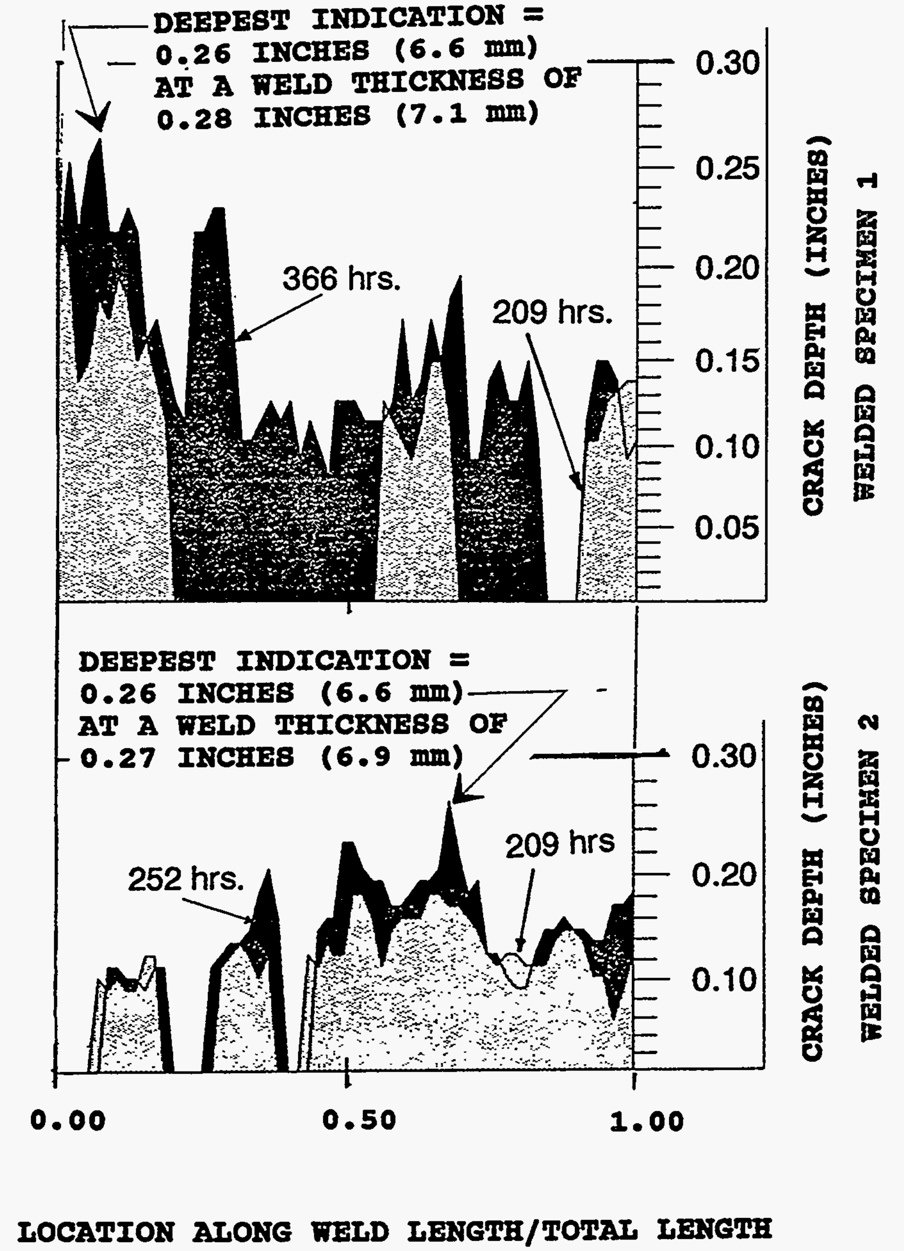

FIGURE 3. UT INDICATION OF CRACK SIZES IN WELD METAL OF WELDED SPECIMENS 1 AND 2 (1 INCH = 25.4 mm)

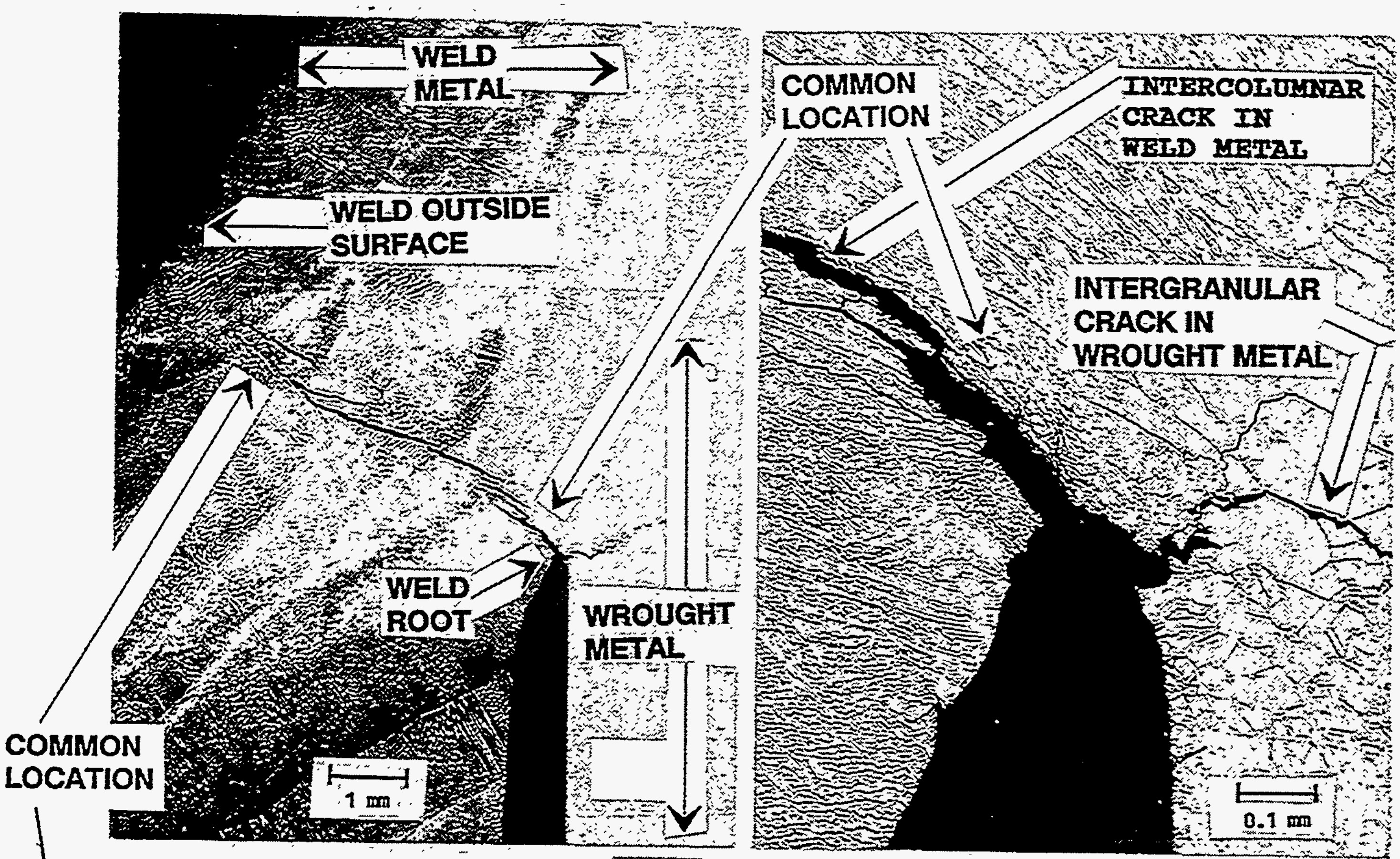

FIGURE 4a. CRACK OVERVIEW

FIGURE 4b. WELD ROOT

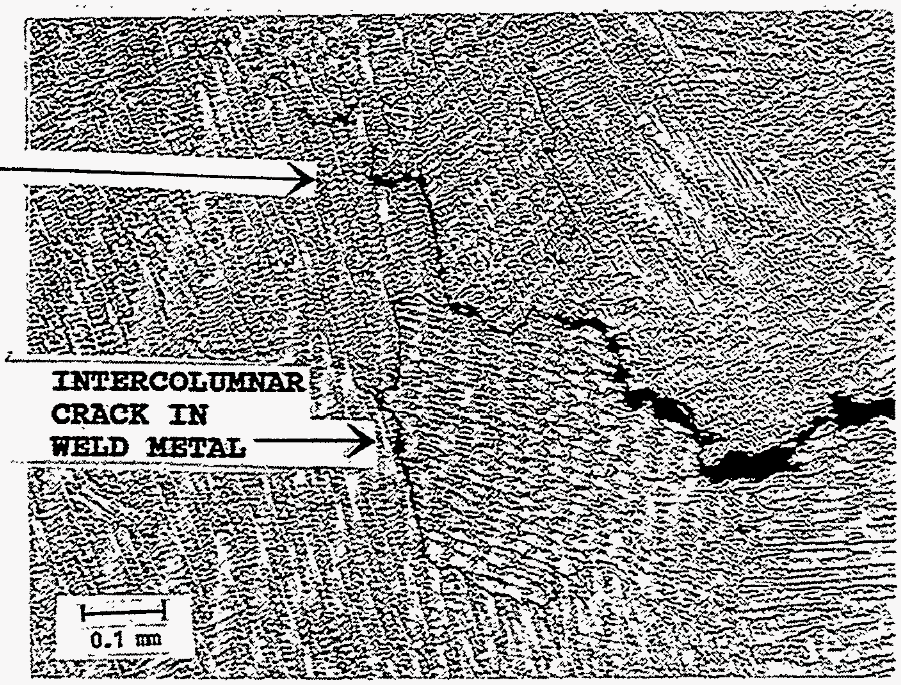

FIGURE 4c. CRACK TIP IN WELD METAL

FIGURE 4. PHOTOMICROGRAPHS OF WELDED SPECIMEN 2 AFTER 252 HOURS OF EXPOSURE TO DOPED STEAM (PHOSPHORIC ETCH)

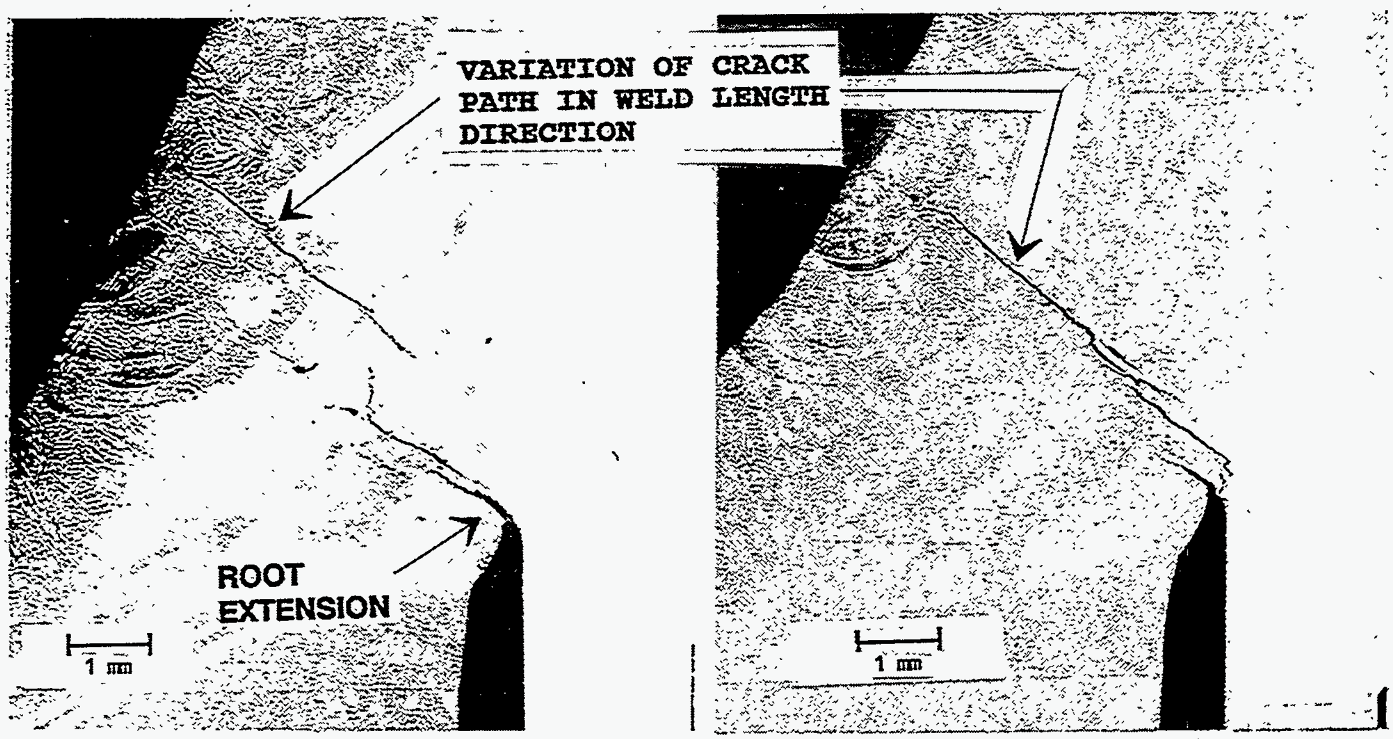

FIGURE 5a. PLANE 1

FIGURE 5b. PLANE 2

FIGURE 5. PHOTOMICROGRAPHS AT TWO PLANES OF POLISH SPACED 0.050 INCHES (1.27 mm) APART

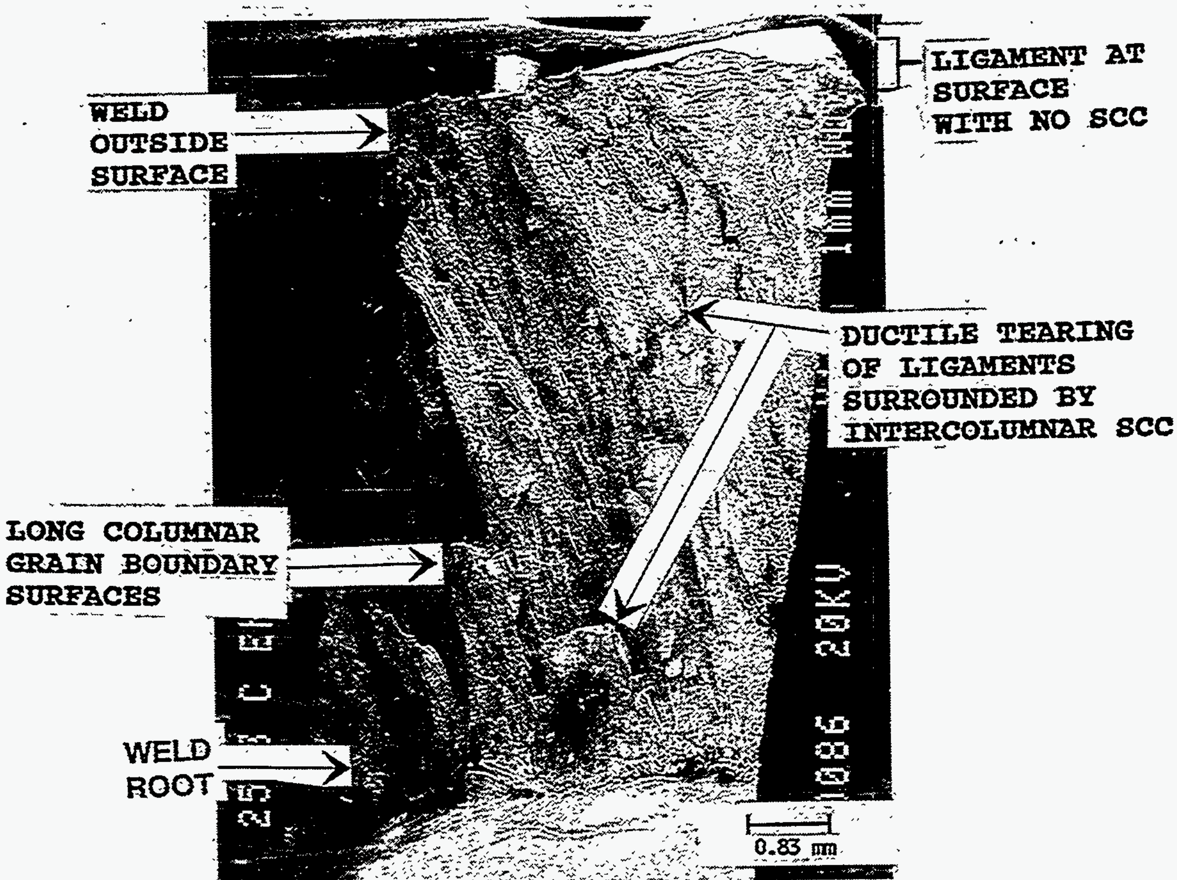

FIGURE 6. FRACTOGRAPH OF WELDED SPECIMEN 2

EFFECTS OF GRAIN BOUNDARY STRUCTURES ON STRESS CORROSION CRACKING OF FACE CENTERED CUBIC ALLOYS, AND IMPLICATIONS FOR CRACKING OF NICKEL ALLOY PRESSURIZED WATER REACTOR COMPONENTS

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ABSTRACT

Extensive studies of the effects of grain boundary structures on susceptibility of face centered cubic alloys to stress corrosion cracking (as well as other degradation mechanisms) have led to the development of an extensive capability for predicting the resistance of nickel (and other materials) to intergranular stress corrosion cracking (SCC). Since the effects of thermomechanical history on austenitic alloys are also well understood, these analyses can be used to predict dependence of SCC vulnerability on thermomechanical history. In particular, this work may offer a vehicle for better understanding variability in head penetration cracking behaviour between different reactors.

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LICENSING EXPERIENCE ON THE ISSUE OF CRACKING IN LWR RPV PENETRATIONS

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Abstract

The crack detection in control rod drive mechanism (CRDM) nozzles at the Bugey-3 plant in France in 1991 was the motive for the Korean Regulatory Body to pay attention to the primary circuit penetrations, especially those made of In-600. As a result, primary water stress corrosion cracking (PWSCC) became one of the licensing issues for Young Gwang (YGN) Units 3 and 4 in 1991-1993 which were under safety review for the issuance of operating licenses.

Based on the evaluation of PWSCC susceptibility, the In-600 components that were estimated to be cracking before end of life have been replaced by In-690. Because of difficulties in replacing the CRDM nozzles, the operating temperature was lowered instead by increasing the bypass flow rate going to the head plenum.

This paper presents the licensing experiences in Korea that were drawn from the YGN Units 3 and 4 to upgrade the safety of primary circuit penetrations.

1 Status of Korean Nuclear Power Plants

The Korea Electric Power Corporation (KEPCO), which is a government invested electric utility (the one and only nuclear power plant operator), owns and operates ten nuclear power plants. The last one, Young Gwang (YGN) Unit 3, was put into commercial operation only one month ago.

Nuclear power represents about 36 percent of the installed electrical capacity in Korea, and it currently produces nearly 50 percent of the total electricity. Korean nuclear power plants achieved an average capacity factor of 87.2 percent in 1993.

Nuclear power is expected to play an important role in the future in the production of electricity until and beyond 2000. KEPCO is now constructing 6 nuclear units and is planning to build 11 more units by 2006.

The long-range electricity development plan drawn up by KEPCO forecasts that nuclear power will ultimately share 40 percent of the total installed capacity to produce electricity.

The construction permit for YGN Units 3 and 4, fabricated in Korea under the license of ABB-CE, was issued in 1989. The operation license of YGN Unit 3 was granted in 1994, and YGN Unit 4 is in a preoperational stage and is scheduled to load fuel in a couple of months.

2. Background

The significance of the control element drive mechanism (CEDM) nozzle cracking due to primary water stress corrosion cracking (PWSCC) began to be recognized in Korea when a leakage through control rod drive mechanism (CRDM) nozzle cracks at Bugey nuclear power plant had been reported at the end of 1991. In Korea, there are two nuclear power plants supplied from Framatome, called Ulchin (UCN) Units 1 and 2. These are

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known as the CP1 series of French reactors. Efforts have been made on these reactors to determine if such cracking is possible. The investigation revealed that the reactors of type CP1 had relatively low head temperature, estimated at 552°F. Inasmuch as temperature is known as a governing parameter to developing PWSCC in In-600 material, there seemed not to be an urgent safety problem. The Korea Institute of Nuclear Safety (KINS), took a position of "wait and see" for the reactors in service until similar reactors were inspected.

In 1992, KEPCO requested the replacement of sleeves and nozzles in the reactor coolant system (RCS) boundary from In-600 to In-690 for YGN Units 3 and 4, which were likely to develop PWSCC within their design lifetime due to relatively high temperatures. The CEDM nozzles are made of In-600 as well. However, at that time the decision was not made to replace them. A little later, in 1993 and 1994, during the licensing safety review for the issuance of an operation license for YGN Units 3 and 4, an effort had been made to find a solution for CEDM nozzles as described here.

3. Regulatory Approach to the Reactors in Operation

As mentioned above, there were eight pressurized water reactors (PWRs) in operation when this issue was opened. The oldest reactor is Kori Unit 1 which has been in service since 1978. Kori Unit 1 has a core bypass flow of 4.5 percent, one fourth of which is estimated to be used for head cooling. That is, head temperature is around 1/4 $T_{coldleg}$ + 3/4 T_{hotleg}, which is 590°F. Since the latest inspection results of American reactors showed that CRDM nozzles were free of significant flaws, the priority was given to UCN Units 1 and 2.

Specific attention to identify the occurrence of such cracking was paid to UCN Units 1 and 2 because such problems were revealed in Europe rather than in the United States. UCN Units 1 and 2 turned out to be cold domes with core bypass flow rates of 6.5 percent. The core bypass flow was initially designed to 4.5 percent and later it was increased to 6.5 percent to reduce the water temperature in the core upper head. As a result, it was estimated to be less susceptible to PWSCC considering the operation period. In 1993, the inspection results of the reference plant of UCN Units 1 and 2 revealed the presence of cracks in CRDM nozzles even for cold domes. Therefore, an inspection of UCN Units 1 and 2 was recommended by KINS. During the latest overhaul of UCN Unit 1, a visual inspection was performed and no trace of boric acid was identified. The nondestructive examination, such as eddy-current testing (ECT) and ultrasonic testing (UT), is planned to be carried out for the first time at UCN Unit 2 during the upcoming refueling outage in December 1995. The appropriate measures for the other reactors will be determined based on the inspection results of UCN Unit 2.

4. Licensing Experience on the Reactors Under Construction

4.1 PWSCC Evaluation

Since 1986 it has been reported that a number of instrumentation nozzles and pressurizer heater sleeves made of In-600 experienced PWSCC. The first PWSCC susceptibility classification for In-600 nozzles was developed as part of a Combustion Engineering's Owners Group (CEOG) task in 1990 in response to concerns regarding the presence of high yield strength material in ABB-CE-supplied RCS nozzles. The objective of this initial program is to provide a classification of high, medium, and low PWSCC susceptibility based on adverse combinations of material yield strength, environment temperature, and final annealing temperature. The reported PWSCC attack in low yield strength pressurizer nozzles in 1992 necessitated a reassessment of the susceptibility of all RCS nozzles fabricated from In-600. This was timely and appropriate for YGN Units 3 and 4 projects since all possible replacement and remedial activities could take place under uncontaminated conditions. A quantitative life evaluation was performed and its preliminary results were presented to KINS in June 1992. Since all of the confirmed PWSCC data in ABB-CE plants occurred at pressurizer temperature with the exception of hotleg nozzle failure, a very conservative approach was selected against time-to-failure.

The environmental temperature effects for this estimation were obtained using an Arrhenius correction factor approach to correct the predominantly pressurizer-based data for other RCS conditions. The most widely used activation energy value for predictions is 48.5 kcal/mole, which leads to a decrease in time-to-failure by a factor of two for each 18°F increase. As a general approach for PWSCC in In-600, a value of 50 kcal/mole was used for the life evaluation [1]. Taking the lower bound of the pressurizer instrumentation nozzle failure data, an acceptance criteria was established as shown in Fig. 1. The curve of an initiation model for PWSCC at 653°F is given by [2]:

$$t_i = 1.68 \times 10^{-4} \times (87 - \sigma_y)^5 \qquad (1)$$

where ti = initiation time for PWSCC (EFPH), σ_y = yield strength (ksi).

This is based upon the data of through-wall cracks in pressurizer instrumentation nozzles. Since an immediate leakage is not expected as soon as the crack penetrates the wall, it was conservatively assumed that a detectable leakage is produced when a crack length at the inside surface is 3/4 in. for pressurizer nozzles. This is the case for Plant A presented in Ref. [2]. The crack is assumed to propagate in a semicircular manner so that the crack depth at leakage is assumed to be about 3/8 in. Consequently, KINS conservatively considers Eq. (1) as a time-to-failure criteria at 653°F at a depth of 3/8 in. whereas the applicant considers it a time-to-initiation criteria at a depth of 0.003 in.

An additional correction is required for nozzles that operate at different temperatures. The adjustment of temperature effects in conjunction with material yield strength is made using an Arrhenius equation as follows:

$$t_f = A \exp(Q / RT) \qquad (2)$$

where t_f = time-to-failure (to the depth of 0.375 in.

Q = activation energy (50 kcal/mole)

R = gas constant (0.001986 kcal/mole K)

T = absolute temperature (K)

A = proportionality constant = $t_i / (4.925 \times 10^{17})$ where t_i is given in Eq. (1)

Equation (2) yields a time-to-failure expressed by Eq. (1) in case of a temperature of 653°F. The acceptance criteria generated from Eq. (2) is illustrated in Fig. 2.

Based on this evaluation, the disposition of each nozzle for YGN Units 3 and 4 was made, and those nozzles that cannot meet the acceptance criteria will be replaced as In-690, as shown in Table 1, with the exception of CEDM nozzles.

When the results showed that the CEDM nozzles could not meet the conservative acceptance criteria without corrective action, ABB-CE evaluated a number of design modifications to determine the best method of assuring that the nozzles will not experience PWSCC during design life time. The mechanical modifications— such as shot peening to generate some compressive stress in the CEDM inner surface, heat treating to lower residual stress, weld overlay, and nickel plating and sleeving to separate the CEDM surface from primary water— were investigated, and the thermal hydraulic solutions— such as suppressing up-flow by closing the flow path in the guide structure support system (GSSS) and increasing core bypass flow by modifying the alignment key— were examined as well. Of those solutions, the alignment key modification was selected as the most appropriate method. This modification could decrease the temperature in the reactor head region to as low as 587°F. This lowered temperature was regarded as acceptable based on the following evaluation.

Considering a value of 0.25 in. for the minimum required design thickness of CEDM nozzles, an interim criteria was determined such that a flaw depth should be less than 75 percent of the nozzle wall thickness. When a crack has initiated and propagated to a depth of 3/8 in., the time for the crack to arrive at 0.75t (t = thickness of CEDM nozzle) is estimated using the crack growth rate provided in Ref. [3]. The assumed configuration for radial flaw propagation is an initial flaw located above the weld at the downside of the peripheral CEDM nozzle. The flaw is assumed to be axially oriented and to be a semicircular surface flaw to minimize the time-to-through-wall propagation. In addition, the highest stress value through the wall is assumed as a further conservative assumption.

From the calculation results [3], the crack growth rate could be linearized in the crack length range of 0.375 to 0.69 in. The resultant value of crack growth rate is of 1.26×10^{-5} in. / h for the case of 617°F. Since the crack growth rate is strongly affected by temperature [4,5,6], once again a temperature adjustment is necessary. The temperature effect on crack growth rate is computed using an Arrhenius rate law with an activation energy of 33 kcal/mole. A time-to-0.75t is obtained from the initiation time plus the time necessary for a crack to grow from 0.375 to 0.75 in.

The initiation time is strongly affected by yield strength, whereas the crack growth rate is dependent on stress intensity and temperature but not dependent on yield strength. From the material data in the certified material test report (CMTR) of YGN Units 3 and 4 CEDM nozzles, a yield strength of 40 ksi is taken because almost all of the CEDM nozzles have yield strength less than 40 ksi. Table 2 shows the crack growth rate and the time-to-0.75t for different temperatures. Based on this evaluation, reducing the temperature in the reactor head region to lower than 590°F is thought acceptable from the point of view that even for the PWSCC cracking in CEDM nozzles, the crack would not propagate to a depth exceeding acceptance criteria within the design life time.

4.2 Consideration of the Bypass Flow Rate

As it was decided to decrease the reactor head temperature by increasing the core bypass flow rate, modification of the alignment key and its consequence were assessed. The increase of core bypass flow rate could affect the FSAR safety analysis because a design value of 3 percent was used as the core bypass flow rate. The Core Operating Limit Supervisory System (COLSS) and the Core Protection Calculators (CPCs) algorithm constants are determined based on 3 percent as well. Therefore, the total core bypass flow rate should be limited by 3 percent.

The initial analysis demonstrated that the increase in leakage flow through the key holes was from 0.23 to 0.9 percent and, as a result, the total bypass flow increased from 2.18 to 2.84 percent. The schematic view of flow paths to the reactor head region is illustrated in Fig. 3. This modification resulted in reducing the reactor head temperature to 587°F, which is lower than 590°F; therefore it was considered acceptable.

In the beginning of 1993, it was identified that ABB-CE underestimated the friction coefficient of the fuel space grid (KGRID). A reassessment of flow distribution in the core was required . As a result, the core bypass flow decreases from 2.84 to 2.71 percent, and the leakage through the key path is lowered to 0.58 percent as shown in Table 3. The decrease in bypass flow consequently increases reactor head temperature from 587° to 606°F; this requires a reevaluation on PWSCC. This change of head temperature brings about a reduction in time-to-0.75t from 57 EFPY to 31 EFPY, which is unacceptable. Therefore, the regulatory position was to require the applicant to take an additional measure such as enforcement of inservice inspection.

Summary

The significance of CEDM nozzle cracking due to PWSCC began to be recognized in Korea at the end of 1991. Since this issue came from France, attention was paid to French reactors (i.e. Ulchin Units 1 and 2). Ulchin Unit 1 was visually inspected this year and did not reveal any trace of leakage. As the first inspection with a mechanized tool, Ulchin Unit 2 will be subjected to inspection during the next refueling outage, and measures for the other reactors will be determined depending on those inspection results.

In the meantime, the PWSCC susceptibility classification for In-600 nozzles was developed as part of a CEOG task in 1990. This was opportune for the YGN Units 3 and 4 projects since all possible replacement and remedial activities could take place under uncontaminated conditions. The acceptance criteria were developed using pressurizer nozzle failure data. The items that could not meet these criteria were to be replaced by In-690 with the exception of CEDM nozzles. Of the available methods to minimize the probability of PWSCC occurring within design lifetime, the modification of the alignment key was selected to increase core bypass flow. This results in reducing the reactor head temperature from 621° to 606°F.

The appropriateness of lowering the reactor head temperature is evaluated, and the conclusion is that the reduction of temperature is not sufficient to satisfy the acceptance criteria. Therefore, some additional measures, such as enforcement of inservice inspection, are required.

References

1. J. A. Gorman, R. A. Ogren, and J. P. N. Paine, "Correlation of Temperature with Steam Generator Tube Corrosion Experience," Proceedings, 5th International Symposium on Environmental Degradation of Material in Nuclear Power System Water Reactor, Monterey, August, 1991.

2. J. F. Hall, T. P. Magee, B. W. Woodman, and M. A. Melton, "Evaluation of Leaking Alloy 600 Nozzles and Remaining Life Prediction for Similar Nozzles in PWR Primary System Application," PVP-Vol. 288, Service Experience and Reliability Improvement, ASME 1994.

3. Safety Evaluation of the Potential for and Consequence of Reactor Vessel Head Penetration Alloy 600 ID Initiated Nozzle Cracking, CEN-607, prepared for the CEOG Group, May 1993.

4. R. B. Rebak, A. R. MacIllree, and Z. S. Smialowska, "Effect of pH and Stress Intensity on Crack Growth Rate in Alloy 600 in Lithiated + Borated Water at High Temperature," Proceedings, 5th International Symposium on Environmental Degradation of Material in Nuclear Power System Water Reactor, Monterey, August 1991.

5. T. B. Cassagne, "Crack Growth Measurements on Alloy 600 Steam Generator Tubes in Steam and Primary Water," Proceedings, 5th International Symposium on Environmental Degradation of Material in Nuclear Power System Water Reactor, Monterey, August 1991.

Table 1: Disposition of In-600 nozzles for YGN Units 3 and 4

Components	Number	Estimated Temp. (°F)	Disposition
Reactor			
CEDM Nozzle	83	> 590°	Lower Temp.
Vent Pipe	1	>590°	"
Monitor Tube	1	-	--
BMI tube	45	545°	Accept
Pressurizer			
Instrument Nozzle	7	>590°	Replace
Heater Sleeve	36	>590°	Replace
S/G			
Instrument Nozzle	4	545°	Accept
Drain Nozzle	2	>590°	Replace
Piping			
RTD Nozzle			
Coldleg	12	545°	Accept
Hotleg	10	>590°	Replace
Sampling Nozzle	9	>590°	Replace

Table 2. Estimated time-to-0.75t penetration in CEDM nozzles

Temperature (°F)	EFPH for initiation	Crack growth rate (in./hr)	Time-to-0.75t (EFPH)	Time-to-0.75t (EFPY)
621	130000	1.3965E-05	156852	17.91
617	150000	1.26E-05	179762	20.52
610	199000	1.0505E-05	234698	26.79
600	285000	8.0673E-06	331484	37.84
590	442000	6.1644E-06	502833	57.4

Table 3. Bypass flow to cool reactor head

	Original (%)	1st Modification (%)	2nd Modification (%)	Temperature (°F)
Key Gap & Key Holes (path #1)	0.07	0.76	0.58	568
Surveillance Access Hole (path #2)	0.16	0.14	0.21	568

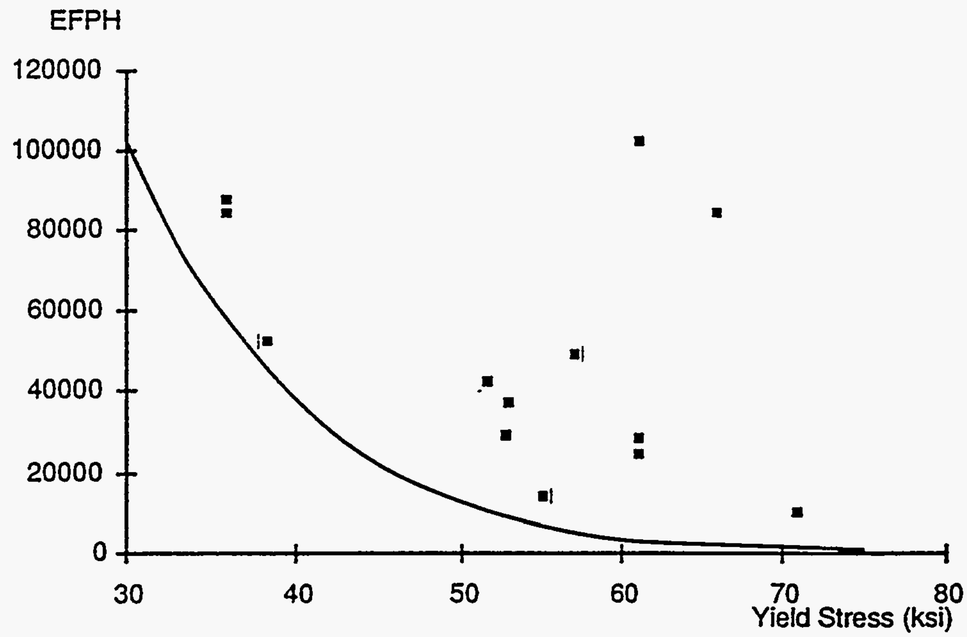


Fig. 1 Acceptance criteria for time-to-failure to a crack depth of 3/8 inch for In-600 forged nozzles at 653°F.

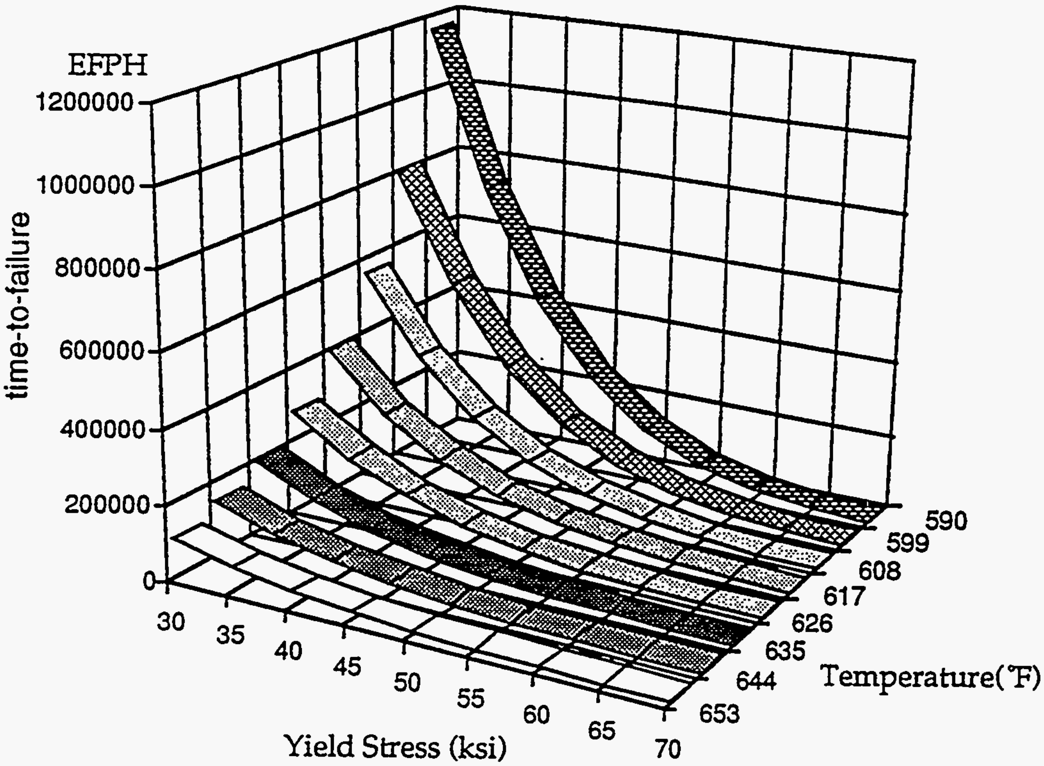


Fig. 2 Acceptance criteria for time to failure for In-600 forged nozzles.

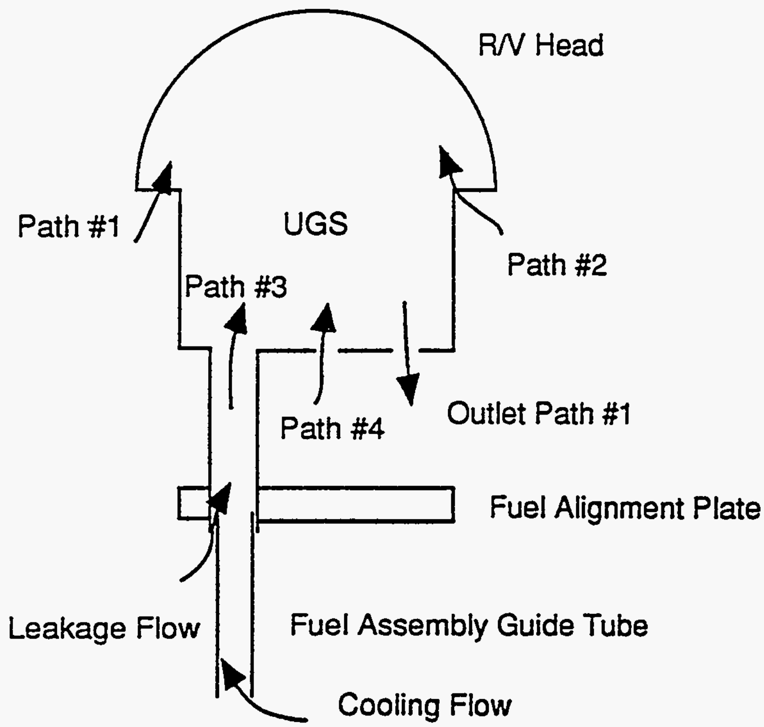
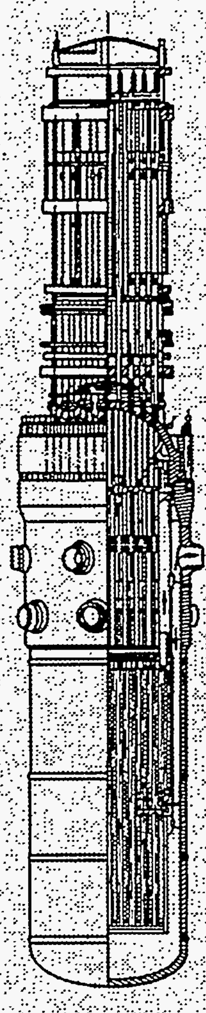


Fig. 3 Schematic View of Flow Paths to Reactor Head

THE INSERVICE INSPECTION OF THE REACTOR VESSEL HEAD PENETRATIONS - VVER 440



INETEC

TABLE OF CONTENTS

1. INTRODUCTION

VVER-440 Pressurized Water Reactor Vessel Heads (RPVH) are equipped with carbon steel penetration tubes (PT) which are fitted into RPVH penetration. The lower parts of the PT are welded to the inner clad surface and base material of the RPVH. Penetration tubes are equipped with corrosion protecting tube made of stainless steel (SS). Another stainless steel tube (heat protecting tube) is inserted into PT and centered into it. The designed distance (gap) between SS corrosion protecting tube and heat protecting tube is three millimeters.

In order to detect any abnormality in the areas of PT and RPVH intersection as well as inside surface of the heat protecting tube the five different end effectors and examination techniques have been developed.

2. SCOPE OF INSPECTION

For the volumetric examination of PT circumferential welds at RPVH intersection, a system which employs contact ultrasonic technique has been developed. Ultrasonic Data Recording and Processing System (UDRPS 2) is used to record, enhance and analyze ultrasonic (UT) data. UDRPS is fully integrated with the Inetec Reactor Vessel Head Inspection Tool to record all ultrasonic and position data as well as to assess indications.

The eddy current technique is applied to examine the surface of the weld and adjacent clad material in order to detect surface abnormalities on the inner part of a RPVH.

Another application of the eddy current method is examination of the inner surface of the corrosion protecting stainless steel tubes. The surfaces of a corrosion protecting stainless steel tubes placed into PT are examined by specially designed eddy current probe inserted into a gap. A new end effector is engineered in order to drive inspection probe on an examination required path. A MIZ-18 system is used as frequency generator and data acquisition station for eddy current examinations while data analysis is performed on the HP series 700 computer(s).

Visual testing (VT) of the whole interior of the RPVH and heat protecting tube is performed by using specially designed camera and software which accompanies video signal from the camera and tool positions from the tool controller. The composite signal obtained from the processing board (TARGA+) is recorded on a standard video recorder tapes.
Similarly, visual testing of circumferential PT weld is performed from inside of the RPVH.

3. POSITIONING MANIPULATOR AND INSPECTION PRINCIPLES

3.1. POSITIONING MANIPULATOR DESCRIPTION

The positioning manipulator is based on concept of SM-22, ZETEC's eddy current manipulator. This manipulator consist of arm and pole which are attached to the spacer over the base frame. The pole and arm provides rotation around two vertical axes. The end of the arm is equipped with vertical carriage carrying end effectors. The carriage provides vertical movements (up - down) and rotation around its axis. FIG.1. presents the positioning manipulator and main axes.

3.2. UT SERVICES DESCRIPTION

The Inetec's access to inspection of the PT circumferential weld has been determined in accordance with all mandatory NDE technical requirements and ALARA principles. The examination techniques are defined to provide reliable inspection in the reasonable time interval.

In order to improve and verify designed examination technique and tool the research and experimental works were accomplished on the RPVH mockup situated at Inetec lab.

The main goals of these research works were as follows:

- to detect, locate, characterize and accurately size defects
- to minimize personnel radiation exposure
- to minimize inspection time

The volumetric examination of PT circumferential welds at RPVH intersection is performed by use of simplified ultrasonic equipment designed for reactor vessel inspections which is fully integrated with positioning manipulator and end-effector (EE) controller.

Penetration Tube Weld Ultrasonic Examination

For the PT weld ultrasonic examination a system which employs contact ultrasonic technique has been developed.

Due to the acoustic properties of these welds ultrasonic examination is performed by use of longitudinal wave transducers. The primary detecting and sizing transducers to be applied will be 70 degree and creeping wave transducer. The ultrasound beams will be oriented in the three directions - two directions parallel to the weld (clockwise and counterclockwise) and one direction perpendicular to the weld (radially towards the penetration center). A creeping wave, dual element transducer is applied for examination of the material volume close to the weld surface while a dual element, transmit - receive, focused, 70 degree transducer will provide examination coverage of weld root and material between root and surface. The principle of creeping wave probe and examination volume covered by use of these transducers is presented on FIG.2.

With intent to direct ultrasound beam perpendicularly to the crack, transducers are equipped with rotating probe holder. In this way it is enabled that, by using proper skew angle, ultrasound beam attacks particular examination area perpendicularly. The idea of differently positioned defect detection is presented on FIG.3..

UT End Effector

The PT ultrasonic EE is designed in the way that its mounting and disassembling on the positioning manipulator is performed remotely. Thus personnel radiation exposure is reduced to a minimum level.

The UT EE is designed to ensure centering into a PT and scanning motions parallel and perpendicular to the weld as well as follow the shape of the inspection surface. The transducer rotary systems are integrated into a vertical linear motion system together with water supply tubing and transducers wiring. The UT EE examination position is presented on FIG.4.

Ultrasonic Data Recording and Processing System (UDRPS 2) is used to record, enhance and analyze ultrasonic (UT) data. UDRPS is fully integrated with the Inetec Reactor Vessel Head Inspection Tool to record ultrasonic and position data as well as to assess indications.

FIG.5.presents equipment positioning and interconnection schematic.

3.3. EDDY CURRENT TESTING (ECT) SERVICES DESCRIPTION

3.3.1. INSPECTION OF THE STAINLESS STEEL CORROSION PROTECTING TUBE I.D. SURFACE

ECT inspection of the stainless steel corrosion protecting tube inner diameter (I.D.) surface is performed by using rotating eddy current probe in a following manner : probe body is designed in shape of cylinder (FIG.6.), so probe body can be inserted into gap between heat and corrosion protecting tubes. Three ECT 'pancake' coils are mounted on a probe body. Inside tube's gap probe body rotates around tubes axis. Corrosion protecting tube I.D. surface can be examined 290 mm from bottom of the inner tube. After probe is inserted into gap, it is rotated and pulled out. Inspection is provided using 3-coil rotating probe i.e. 3 'pancake' coils are mounted on probe body, separated 120° in circumference. Coils are designed as low-frequency probe. This gap-scanner rotation probe provides required sensitivity and resolution so it is possible to detect, locate and measure both circumferential and axial cracks.

Applied probe has efficient frequency range 30-200 kHz. A standard depth of penetration while operating at frequency of 30 kHz is 2.5 mm. If operating frequency is 200 kHz, depth of penetration is approx. 1 mm, so surface breaking defects are easily detected.
Four surface breaking notches are presented on FIG.7. All displayed notches are 0.5 mm deep and 1.27 mm wide. The length of notches varies between 3 - 10 mm as it is visible from C-scan display.

Experience with steam generators tube inspections has showed that eddy current method quickly provides reliable information about existence of even the smallest defects. For surface defects detection an eddy current inspection is preferable in comparison to other nondestructive examination techniques.

INSPECTION LIMITATIONS :

Possible limitations during inspection may be deposits inside tube gap and ovality of tubes resulting in locally decreased gap width. The final result is resistance to probe motion in both axial and circumferential direction. If gap is restricted to probe motion due to deposit existence, tube is cleaned by use of specially designed cleaner while in the case of tube ovality "blade" probe will be used.

INSPECTION SCHEME :

Using positioning manipulator, probe is positioned to area of interest. Eddy current data are generated using "MIZ-18" Remote Data Acquisition Unit and recorded on an optical data disks where also data evaluation results will be permanently stored.
Data evaluation is provided using appropriate analysis software for rotating probe data, i.e. C-scan.
Analysis results data base are generated in Data Management System.
The simplified scheme of inspection organization and data flow is given on FIG.8..

3.3.2. INSPECTION OF THE PENETRATION WELD SURFACE

The penetration weld surface inspection is performed using eddy current probe designed for weld testing. Probe operates in differential mode and it is designed to minimize undesired features affecting this scan area, as probe lift-off, permeability variations and conductivity changes. Probe frequency range of interest is 30-200 kHz. Standard depth of penetration while operating at frequency of 30 kHz is approx. 2.5 mm. Weld is scanned on it's whole circumference starting on a minimum radius, than radius is incremented and scanning is repeated until whole weld surface is tested. FIG.9. represents end effector position while testing weld surface.

The two surface breaking notches detected by weld scan probe are shown on FIG.10. The notches are 1 and 0.5 mm deep.

INSPECTION SCHEME :

Inspection scheme and data flow are the same as during inspection of the stainless steel corrosion protecting tube ID surface, chapter 3.3.1. with differences in specific routine of data collection and data evaluation.

3.4. VISUAL INSPECTION

VT SERVICE DESCRIPTION

Visual testing of RPVH will be performed on three areas of interest:

- interior of RPVH
- interior of inner stainless steel heat protection tubes
- PT circumferential welds

Remote visual testing technique will be applied as is presented on FIG.11.- Sample VT inspection drawing.

HEAT PROTECTION TUBES VISUAL TESTING

The inner surface of heat protection tubes will be visually tested by camera equipped with radial viewing head and integral light.

The RPVH positioning manipulator with attached visual testing end effector will be positioned under the penetration to be inspected. The camera, mounted on vertical carriage, will be driven into penetration up to the top of flange. The scan will be performed by camera rotation from 0 to 360 degrees. When the scan is finished, vertical carriage will be incremented down for next scan and camera will be rotated back to initial position. After that, camera will be incremented again.

Described steps will be repeated until the whole surface of heat protection tube will be visually tested. The increment value will be set to provide 10% overlap.

PT CIRCUMFERENTIAL WELDS AND RPVH VISUAL TESTING

The surface of PT circumferential welds will be visually tested by camera equipped with prismatic viewing head and integral light.

The RPVH positioning manipulator with attached visual testing end effector will be positioned under the penetration to be inspected. The camera will be driven by vertical carriage near the PT circumferential weld. The scan will be performed by means of prism rotation in camera viewing head. The increment will be performed by means of camera rotation around PT axis (camera will rotate around center line of penetration in horizontal plane).

Described steps will be repeated until the whole surface of PT circumferential weld will be visually tested. The angle of increment will be set to provide 10% overlap.

VISUAL TESTING EQUIPMENT

Visual testing equipment consists of:

- REES R93 Mk3 B/W camera
- REES radial viewing head with light
- REES prismatic viewing head with light
- REES camera control unit
- PC with TARGA+ video processor board
- VHS video cassette recorder
- B/W video monitor
- visual standard 1/32" black lines on 18% natural gray card

B/W camera is used because it has better life time in the high irradiated areas and it has better resolution. To make sure that video system with camera and light provide required resolution visual standard is used.

The camera control unit provides remote control of following camera functions: focus, iris, light intensity and prism rotation.

In order to record position data and video signal accompanied, the video composite signal from camera control unit is overlapped with real time coordinates by PC with TARGA+ board. The real time coordinates are being obtained from the tool control system. Overlapped picture is recorded on the video cassette recorder in PAL video standard.

Before inspecting of RPVH penetration the header picture containing all significant data will be used for announcing. The operators voice could be recorded on the tape as well as video signal.

4. EQUIPMENT SETUP

The equipment is set up after the RPVH is positioned on its stand. Assembling of the manipulator is provided through a manway placed in the floor. The manipulator is connected to remote acquisition system. Electrical and mechanical functions are tested afterwards.

Following actions with tooling such as changing the end effector or other maintenance activities during the inspection take place from outside.

Therefore there is need to enter under the RPVH only during the assembling and disassembling the manipulator.

5. INSPECTION SEQUENCE

Reactor vessel head inspection consists of visual testing, eddy current testing and ultrasonic testing. The inspection sequence is as follows:

1) The camera for inside tube VT is mounted to manipulator carriage. The manipulator is aligned with the penetration to be inspected. The camera is driven up and entered the tube. When camera is reached the PT flange the scanning sequence is started. The camera is rotated around its axis for a one whole turn 360° scans the completely tube.

2) The camera for VT of welding surface is mounted to vertical carriage. The manipulator is aligned with the penetration to be inspected. Scanning is performed with rotating the camera around the vertical axis.

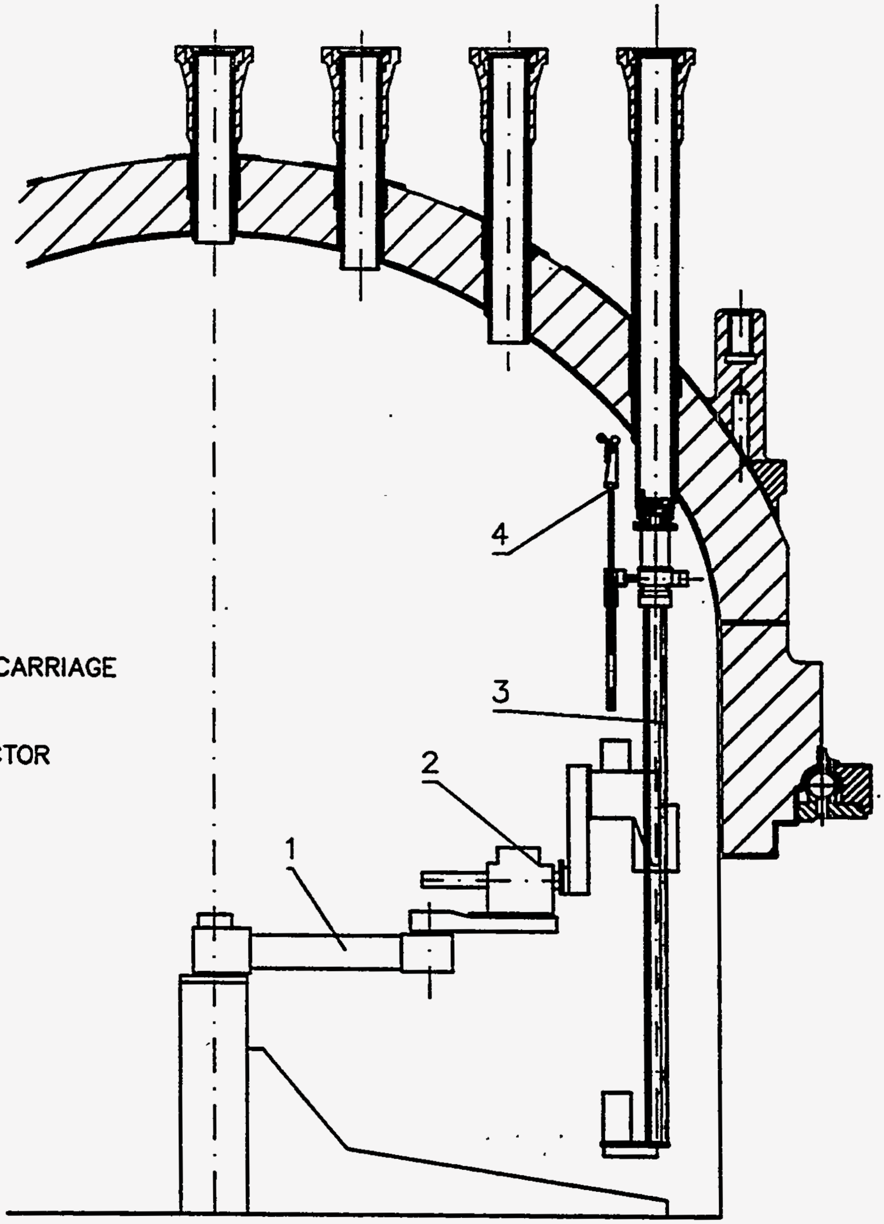
3) The MRPC probe is mounted to manipulator carriage. The manipulator is aligned with the penetration to be

3) The MRPC probe is mounted to manipulator carriage. The manipulator is aligned with the penetration to be inspected. The probe is driven up to the maximum inspection height (about 300 mm above the tube end). After probe is placed at required position, it starts to rotate and moves down at the same time. During performance of this scanning motions EC data recording is performed.

4) The weld surface EC inspection probe with its drive system is mounted to manipulator carriage. The manipulator is aligned with a penetration to be inspected and the probe is driven up to the surface to be tested. The EC data are collected during rotation of the probe around PT. After complete cycle, a small horizontal movement of the probe drive system is performed.
The sequence started again until inspection plan for this penetration is completed.

5) After positioning manipulator is reached desired penetration tube position, the centering device of UT EE is inserted into a penetration tube. Entrance of the centering device into a PT is monitored via video system applied. After centering in the tube, transducers are placed into a contact with an examination surface. The information about contact between transducers and examination surface is controlled by non-contact sensors. The couplant recovery system controlled by sensors starts to operate and supplies transducers with water. At this moment the preprogrammed scanning sequence is started. When scanning sequence is finished all activities are stopped automatically. An operator is lifted down the UT EE and examination sequence may be started again to finish the inspection plan.

FIGURE 1.



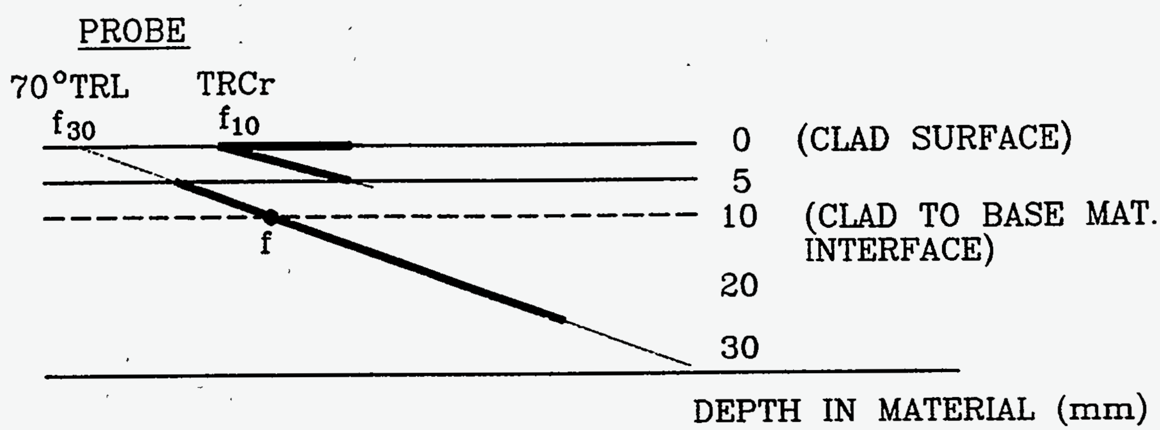
1 – POLE

2 – ARM

3 – VERTICAL CARRIAGE

4 – END EFFECTOR

FIGURE 2.



CREEPING WAVE PROBE

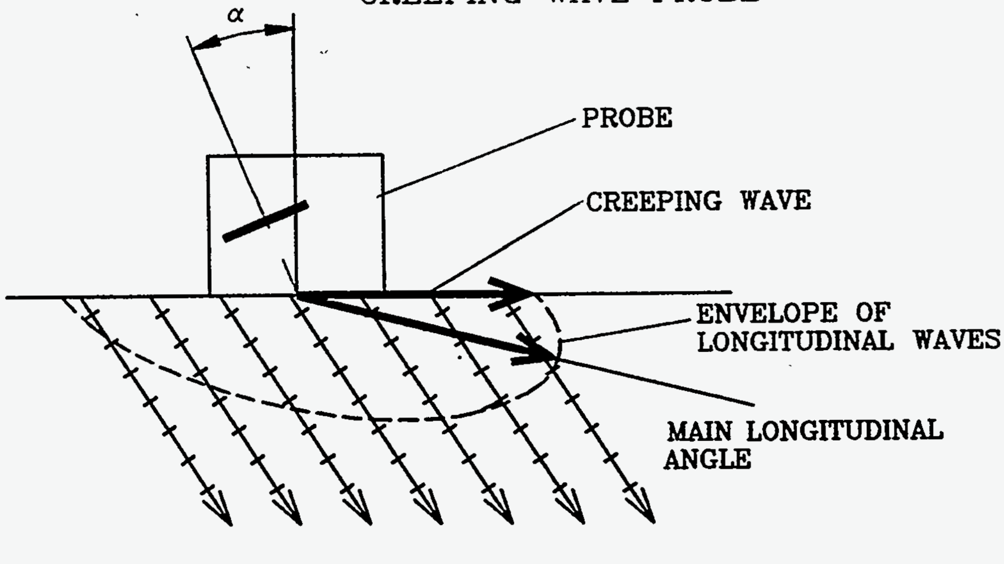
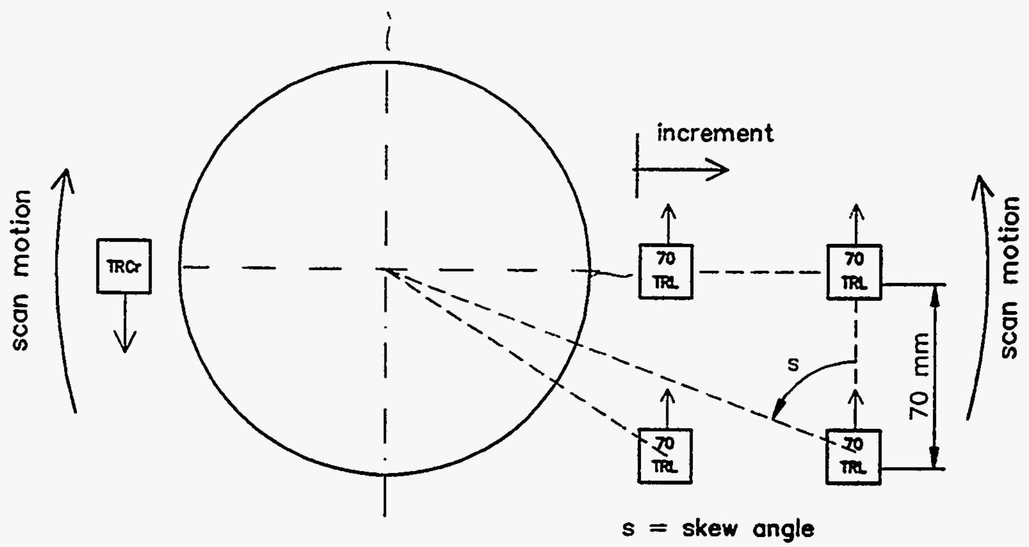


FIGURE 3. CIRCUMFERENTIAL SCANNING
(RADIAL DEFECTS DETECTION)



RADIAL SCANNING
(CIRCUMFERENTIAL DEFECTS DETECTION)

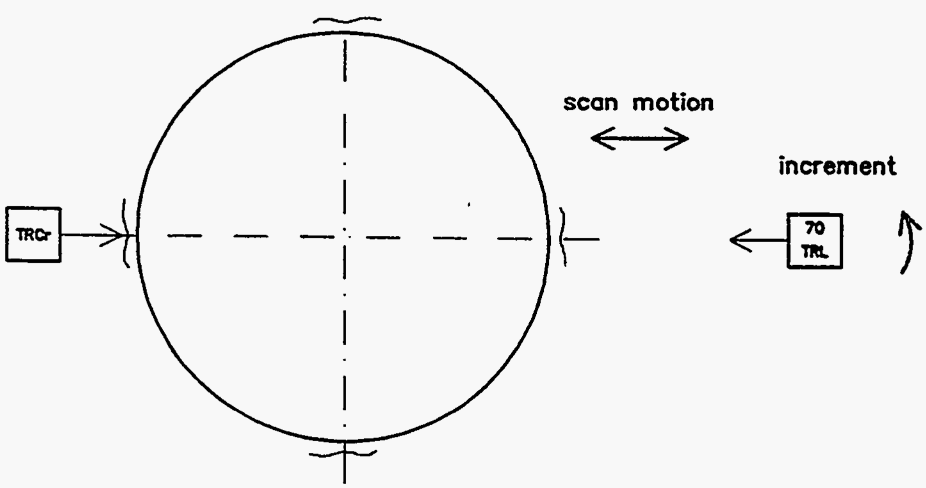
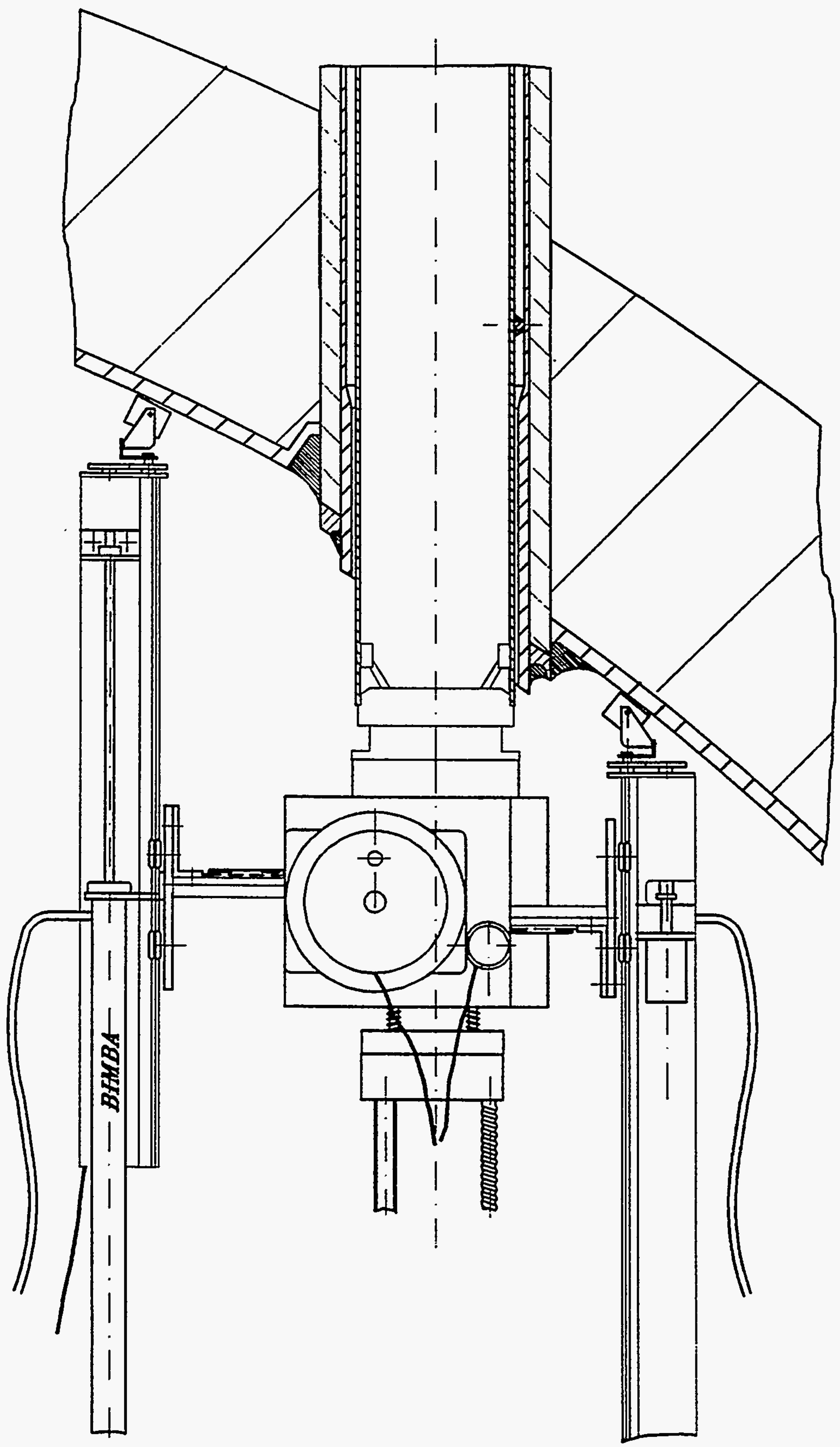


FIGURE 4



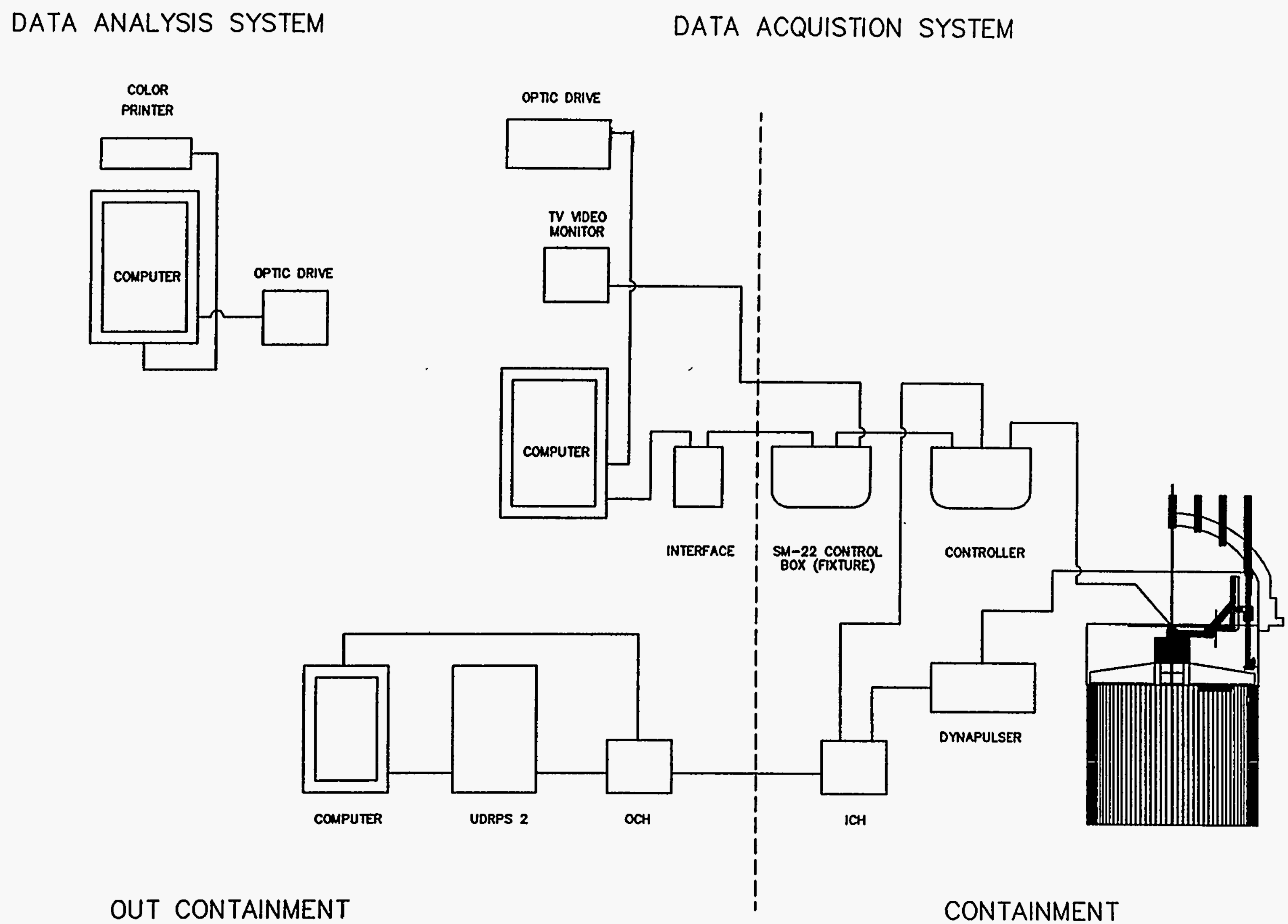


FIGURE 5

FIGURE 5.

FIGURE 6.

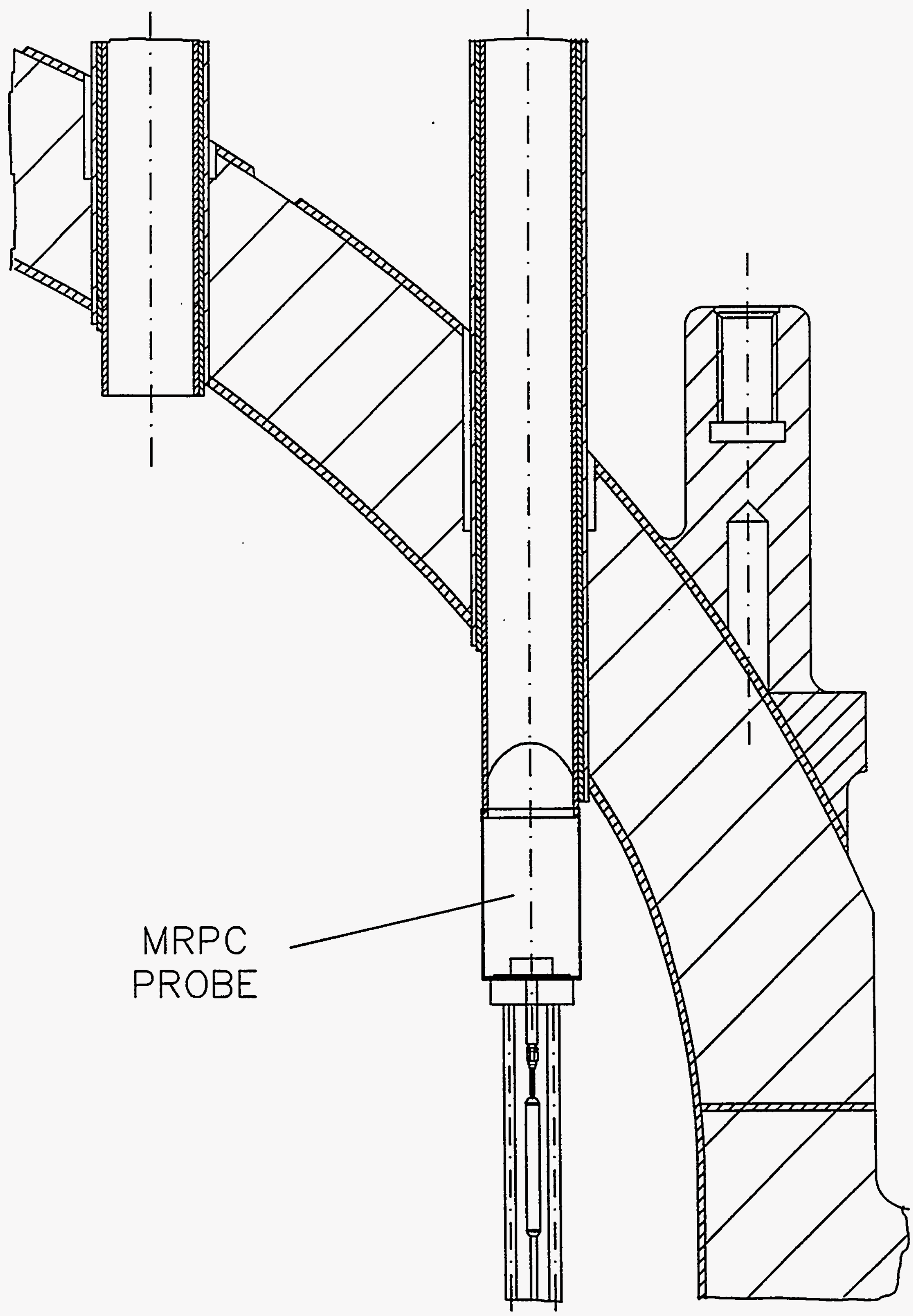
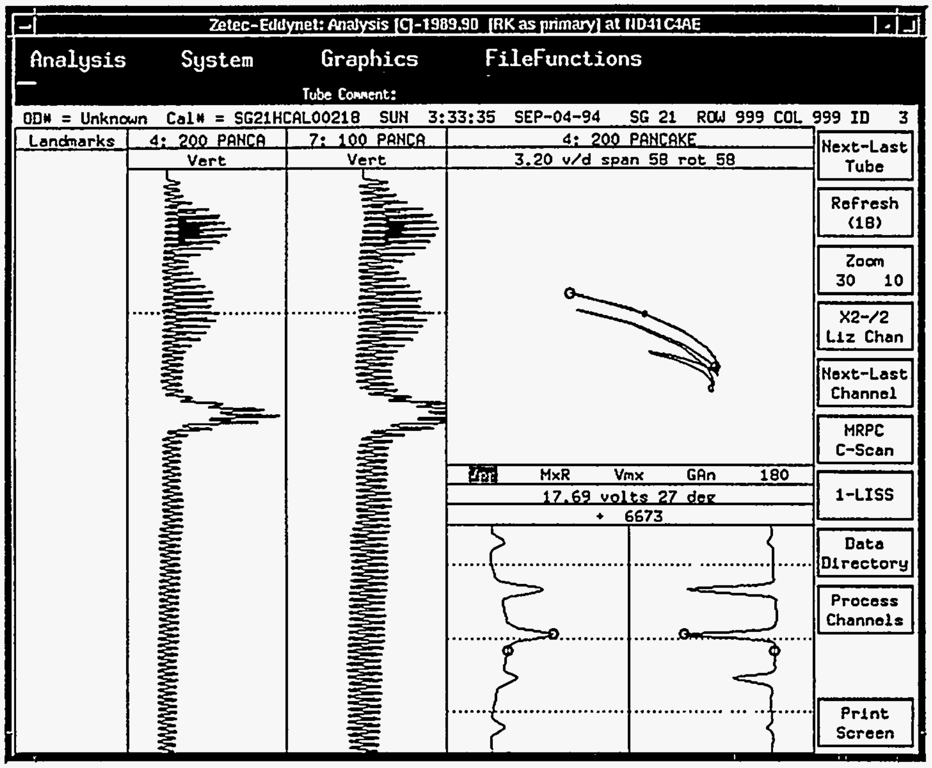


FIGURE 7.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7: 100 PANCA
Vert
Vert
4: 200 PANCAKE
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+ 6673
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Zoom 30 10
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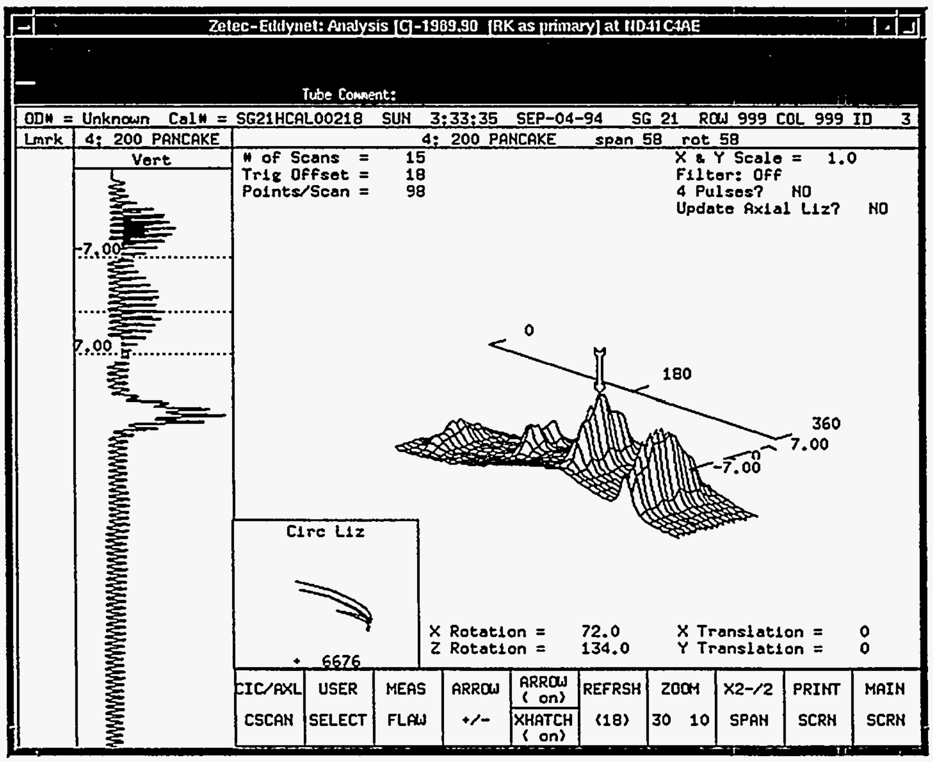
Next-Last Channel
MRPC C-Scan
1-LISS
Data Directory
Process Channels
Print Screen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Lmrk
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Vert
4: 200 PANCAKE span 58 rot 58
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Trig Offset = 18
Points/Scan = 98
X & Y Scale = 1.0
Filter: Off
4 Pulses? NO
Update Axial Liz? NO
-7.00
7.00
0
180
360
7.00
-7.00
Circ Liz
+ 6676
X Rotation = 72.0
Z Rotation = 134.0
X Translation = 0
Y Translation = 0
CIC/AXL CSCAN
USER SELECT
MEAS FLAW
ARROW +/-
ARROW (on)
XHATCH (on)
REFRSH (18)
ZOOM 30 10
X2-/2 SPAN
PRINT SCRN
MAIN SCRN

FIGURE 8.

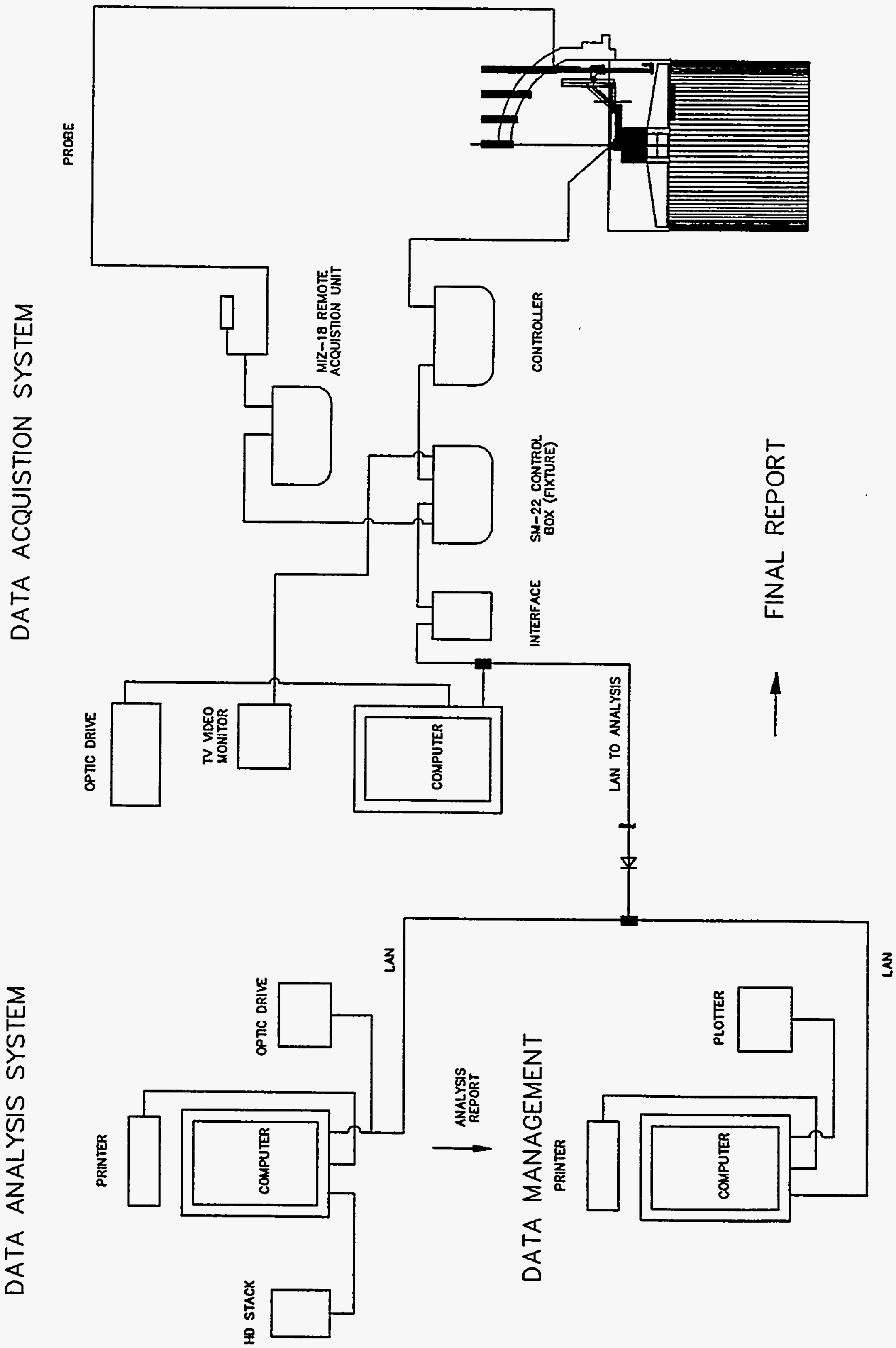

DATA ACQUISTION SYSTEM
PROBE
MIZ-18 REMOTE ACQUISTION UNIT
CONTROLLER
SM-22 CONTROL BOX (FIXTURE)
INTERFACE
OPTIC DRIVE
TV VIDEO MONITOR
COMPUTER
LAN TO ANALYSIS
FINAL REPORT
DATA ANALYSIS SYSTEM
PRINTER
COMPUTER
OPTIC DRIVE
HD STACK
LAN
ANALYSIS REPORT
DATA MANAGEMENT
PRINTER
COMPUTER
PLOTTER
LAN

FIGURE 9.

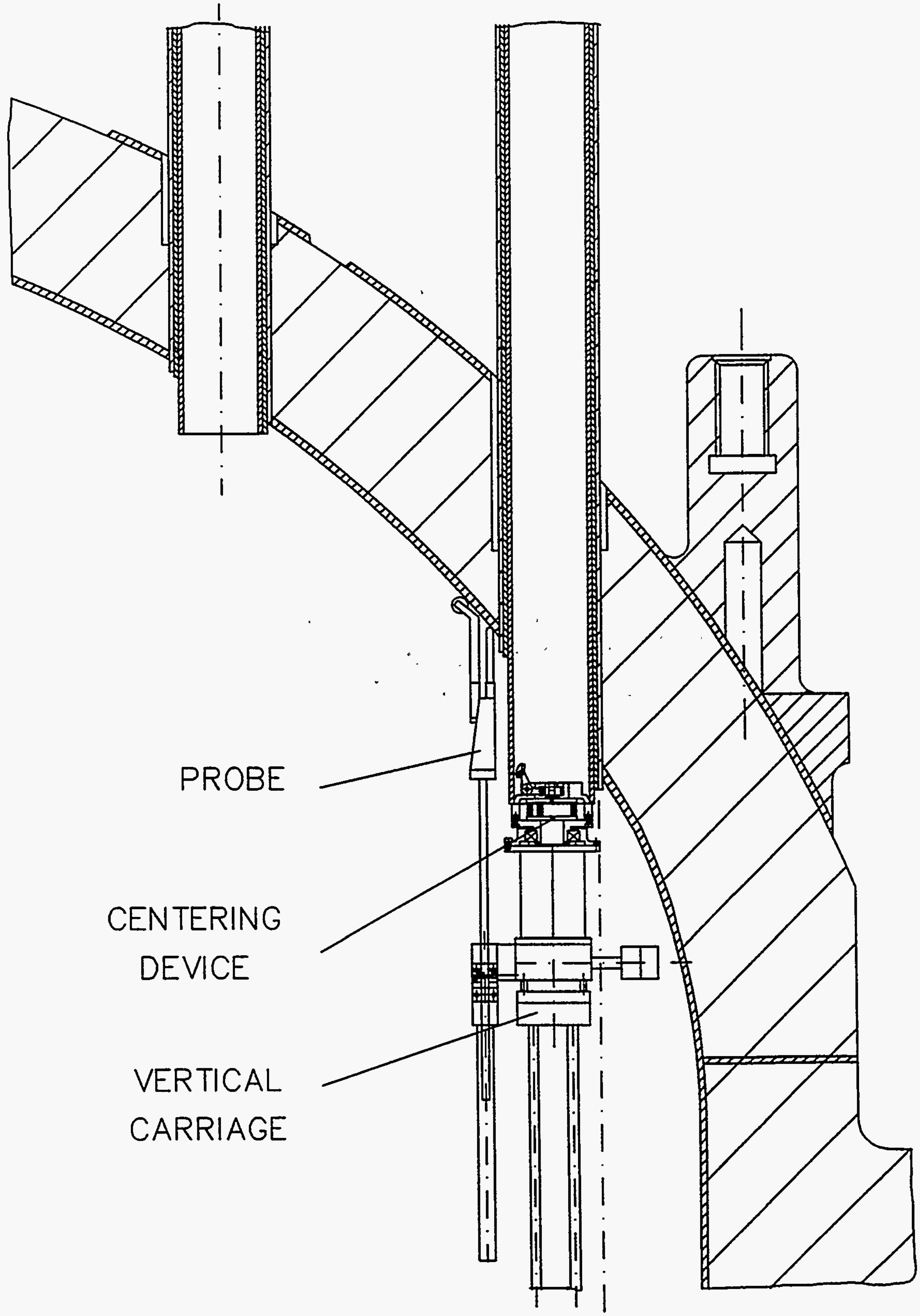
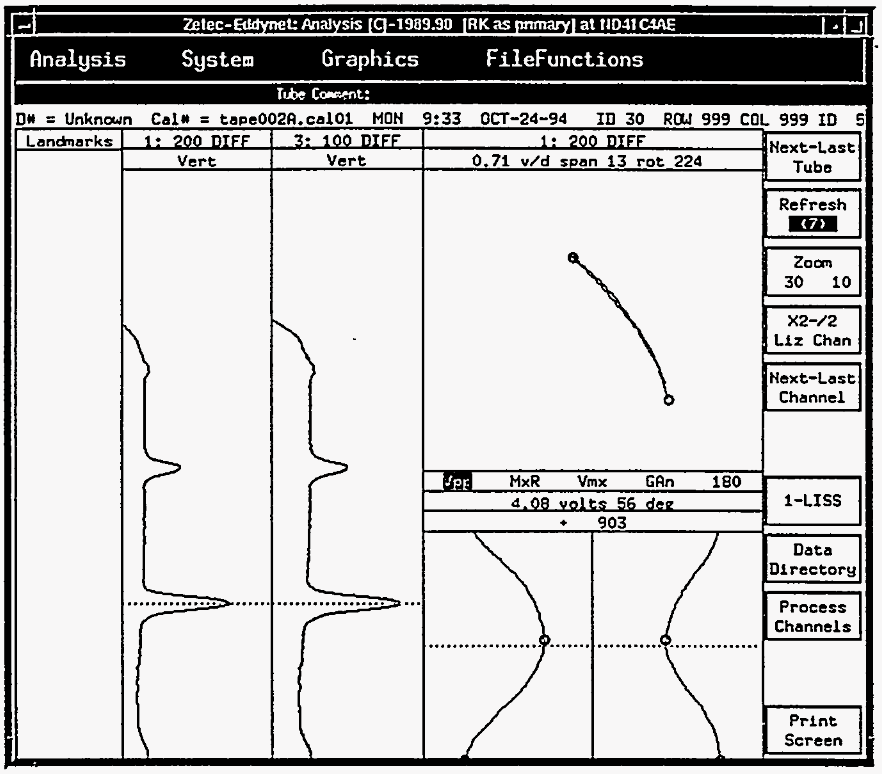


FIGURE 10.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Vert
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4.08 volts 56 deg
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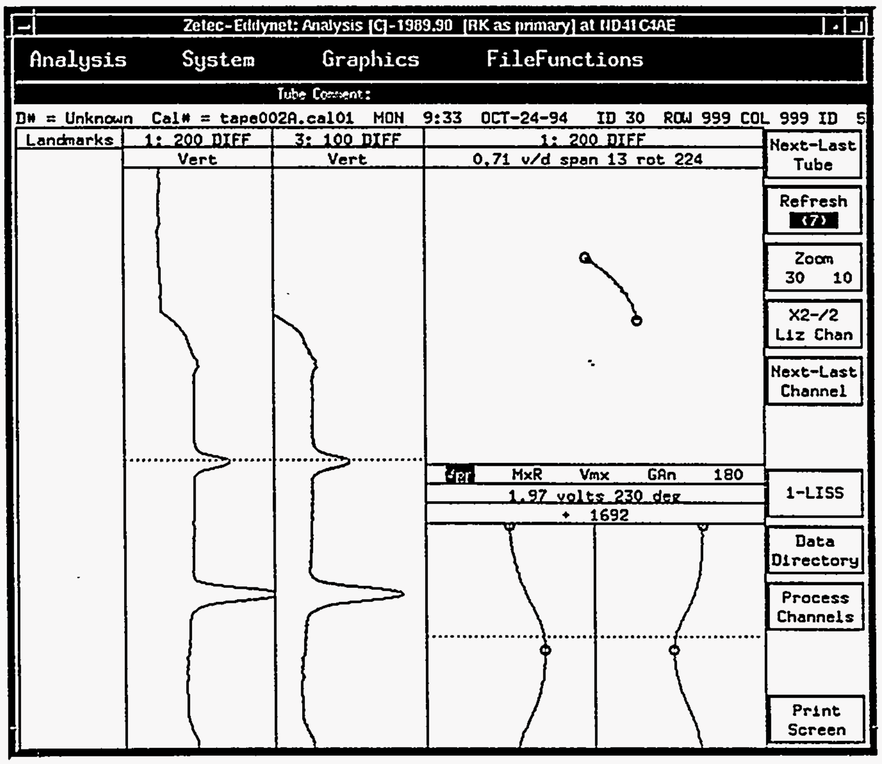
X2-/2 Liz Chan
Next-Last Channel
1-LISS
Data Directory
Process Channels
Print Screen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Vert
1: 200 DIFF
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1.97 volts 230 deg
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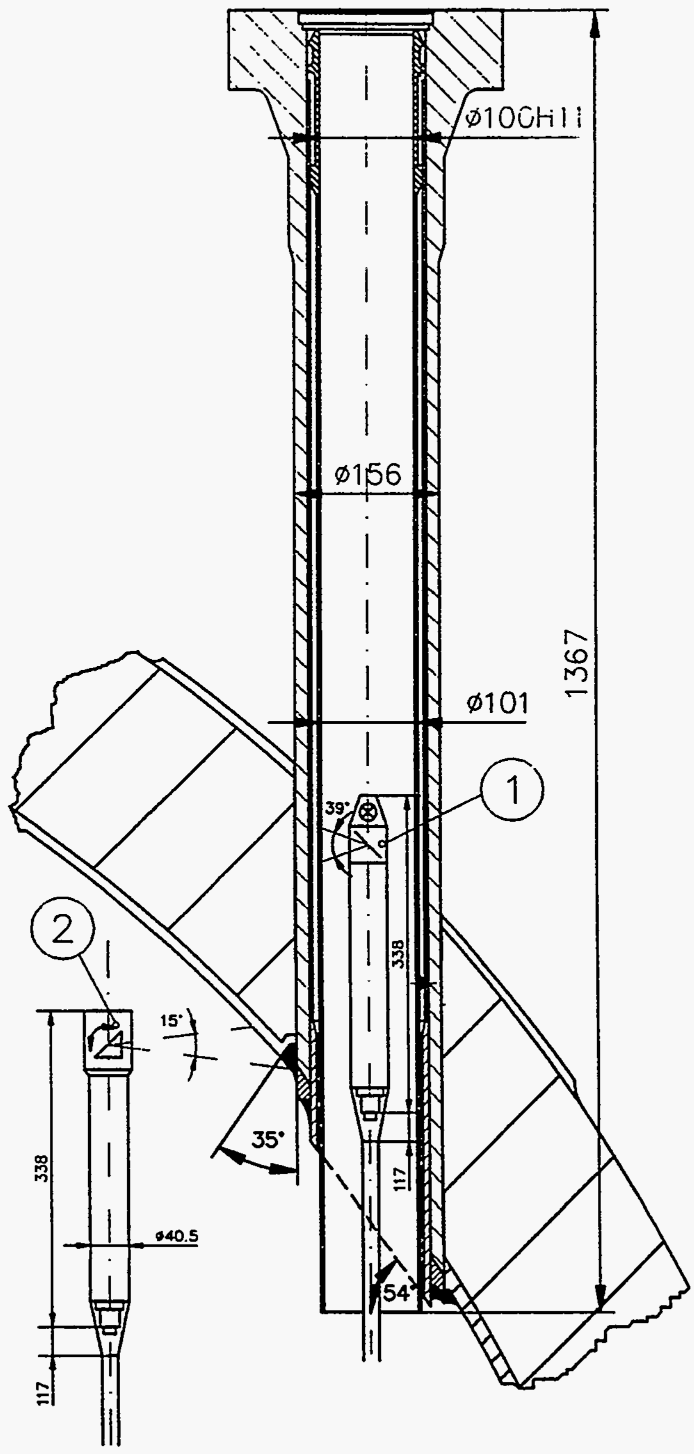
Next-Last Tube
Refresh (7)
Zoom 30 10
X2-/2 Liz Chan
Next-Last Channel
1-LISS
Data Directory
Process Channels
Print Screen

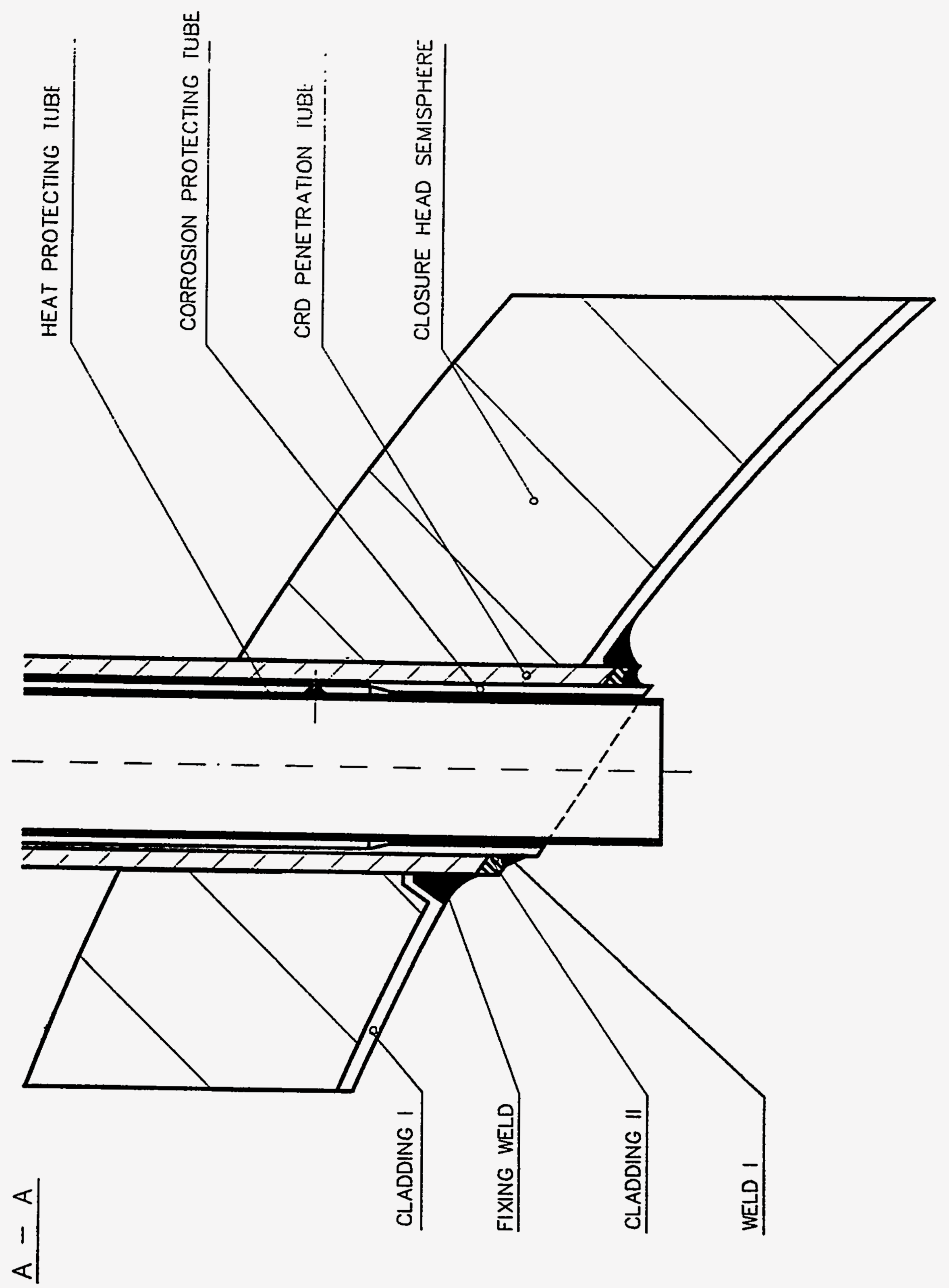
FIGURE 11



REES R93 Mk3 CAMERA WITH:

① – REES RADIAL VIEWING HEAD
② – REES PRISMATIC VIEWING HEAD

FIGUPE 12.



HEAT PROTECTING TUBE
CORROSION PROTECTING TUBE
CRD PENETRATION TUBE
CLOSURE HEAD SEMISPHERE
A – A
CLADDING I
FIXING WELD
CLADDING II
WELD I

CRDM NOZZLE INSPECTION/REPAIR

K.W. FLEMING
B&W NUCLEAR TECHNOLOGIES
3315 OLD FOREST ROAD
P.O. BOX 10935
LYNCHBURG, VA 24506-0935

S. FYFITCH
B&W NUCLEAR TECHNOLOGIES
3315 OLD FOREST ROAD
P.O. BOX 10935
LYNCHBURG, VA 24506-0935

ABSTRACT

The susceptibility of Alloy 600 to Primary Water Stress Corrosion Cracking (PWSCC) continues to plague nuclear power plants. Recently, the problem of PWSCC has manifested itself in Control Rod Drive Mechanism (CRDM) head penetrations in nuclear plants in Europe and the U.S. B&W Nuclear Technologies (BWNT) has been involved in responding to this potentially industry-wide concern through the B&W Owners Group (B&WOG). The program also includes work performed in cooperation with the Nuclear Energy Institute (NEI) Task Force and other Owners Groups. BWNT, building on our parent company's (Framatome) technology, has also developed a fully integrated service package and robotic manipulator to inspect and repair CRDM head penetrations for U.S. utilities. This paper describes the work performed by the B&WOG to address this issue globally, and the overall range of inspection and repair tooling and processes developed by Framatome and BWNT to address this issue.

INTRODUCTION

In September 1991, the first Control Rod Drive Mechanism (CRDM) penetration cracks in a reactor vessel (RV) closure head were discovered during the 10-year hydro test of the Bugey 3 plant in France. The cause was determined to be stress corrosion cracking (SCC) of Alloy 600, a problem that has been previously encountered in other primary plant components, most notably pressurizer nozzles and steam generator tubes.

Evaluations showed that the potential for this problem to occur in CRDM penetrations in U.S. reactor vessel heads existed. As a result, B&W Nuclear Technologies (BWNT), through the B&W Owners Group, undertook a comprehensive program to evaluate the susceptibility and consequences of PWSCC in B&W-design CRDM penetrations. This work was performed in cooperation with the Nuclear Energy Institute (NEI) Task Force that was specifically formed to address this issue on an industry-wide basis.

To be prepared should this problem occur in the U.S., BWNT developed a fully integrated service package and robotic manipulator to inspect and repair CRDM head penetrations in the U.S. This tooling is based on inspection/repair tooling developed by our parent company, Framatome, who has been extensively involved in both inspecting and repairing of CRDM head penetrations at Electricité de France (EdF) plants.

OVERALL APPROACH

The overall approach to address this potential problem involves the following actions:

- Perform eddy-current testing to initially locate defects
- Utilize ultrasonic testing methods to depth characterize the defects
- Perform a local repair of the affected CRDM penetrations that is simple and quick to implement to enable the restart of the unit as soon as possible
- Perform analysis and develop justification to allow the restart of the unit with defects without making immediate repairs.

The development of CRDM nozzle inspection and repair tooling presented a number of significant challenges.

Due to their configuration, the majority of the CRDM nozzles are difficult to access for inspection. The majority of nozzles are physically accessible only from the underside of the reactor vessel head. Furthermore, because of the presence of thermal sleeves that are not readily removable, inspections on the majority of nozzles from below the head must be done in the 1/8-inch gap between the CRDM nozzle ID and the OD of the thermal sleeve. Since the thermal sleeves of most CRDMs are not readily removed, inspection with the sleeve in place is highly desirable. Inspecting the nozzle ID is made more difficult due to the ovalization and deflection of the CRDM nozzle caused by weld distortion during fabrication. In addition, the presence of boron crystals on the ID of the nozzle can negatively impact inspection results.

Furthermore, the radiation levels under the head are very high, usually in the range of 3-5 Rem, thus necessitating the development of robotic tooling to deliver CRDM nozzle inspection and repair tooling.

The final challenge involves time. Since the head is normally on the head stand for a limited duration during a typical outage, robotic tooling was considered necessary for minimizing the impact on the outage schedule.

Facing these challenges, BWNT developed RΘMAN (Remotely Operated Manipulator to Access Nozzles), a manipulator specifically designed to deliver the inspection and repair end effectors to CRDM nozzles without human intervention under the RV head. A total of 13 end effectors have been developed to address this issue.

INSPECTION TECHNIQUES

Eddy-current testing (ECT) techniques are used to initially locate defects. If defects are detected by eddy-current, other non-destructive examination (NDE) techniques such as ultrasonic testing (UT), dye-penetrant testing (PT), and visual inspection (VT) are used to further characterize the defect. In particular, UT techniques are used to measure the depth of the cracks detected.

Detecting cracks by ECT

A blade-type ECT probe was specially developed for penetrations that are fitted with thermal sleeves or leadscrew support tubes. This probe consists of a head, mounted on the upper end of a flexible blade that is inserted into the narrow gap between the CRDM penetration and the thermal sleeve.

The probe head (as shown in Figure 1) includes two ferrite coils, which have a 45° orientation with respect to the penetration axis, to enable high-sensitivity detection of axial and circumferential cracks. A spring ensures the inspection head is kept continuously in contact with the surface to be inspected. The total thickness of the ECT blade probe inspection head is less than 2 mm (.080 in.).

The delivery tooling establishes a nominal gap between the thermal sleeve and the CRDM penetration to allow the blade probe to be inserted. The blade is then withdrawn from the gap, and as the coils move down through the weld region, the penetration surface is inspected. When the blade is completely removed from the gap, it is incrementally rotated 2° around the CRDM nozzle circumference. An in-line calibration after each increment ensures that accurate data is obtained. This procedure is followed until the entire circumference is inspected.

For penetrations without thermal sleeves or leadscrew support tubes, a motorized rotating pancake coil (MRPC) ECT probe is utilized. This probe has three coils designed to detect both circumferential and axial cracking. It revolves helically as the probe is being withdrawn.

Both inspection techniques employ multi-frequency signals in both differential and absolute modes to ensure accurate defect detection and avoid false calls. For analysis, dedicated software converts the line-by-line signals into 3-D representations.

Characterizing cracks using ultrasonics

Specific ultrasonic probes were designed to measure the depth of the cracks precisely, as well as determine the position of the cracks with respect to the nozzle/RV closure head weld. This information is used along with ECT inspection data to determine if a repair is required, and if so, what type.

Crack depth is determined by using time-of-flight diffraction (TOFD) techniques. In addition, a 0° transducer is used, to determine the location of the flaw with respect to the partial penetration weld.

Similar to ECT, two types of ultrasonic probes are used: a rotating probe for penetrations without thermal sleeves (or leadscrew support tubes) and a blade probe for penetrations with thermal sleeves.

The rotating ultrasonic probe (shown in Figure 2) has several sets of dual-element transducers, optimally designed to size different types of cracks: isolated cracks or clustered cracks of minimal depth to through-wall. These transducers are also used to determine the flaw location with respect to the penetration/RV closure head interface. The forward and back scatter techniques are used to ensure accurate characterization.

For penetrations that are fitted with thermal sleeves or leadscrew support tubes, three types of ultrasonic blade probes were developed by Framatome to size the different types of cracks expected and indicate their locations with respect to the weld. Each of these types of ultrasonic blade probes has a set of dual-element transducers. The separation between elements for each type of probe is designed so that the overall wall thickness can be examined.

With the exception of blade UT, the above inspection techniques were successfully demonstrated on three mockups developed by the Electric Power Research Institute's (EPRI) NDE Center as a result of coordinated industry effort to address this issue.

REPAIR TECHNIQUES

The finding in January 1992 of near through-wall defects on the Bugey 4 Unit in the 0° orientation led to the development of repair techniques. A repair technique was developed by Framatome that involved locally excavating the defect by machining to a limited depth, followed by laying a weld deposit on the excavated defect to restore leaktightness. Dye-penetrant testing (PT) is utilized to determine whether a defect exists after excavation. This local temporary repair technique allowed restarting the plant as soon as possible without performing the more time-consuming CRDM nozzle replacement.

The operations at Bugey-4 were primarily performed manually and involved removing the CRDM and thermal sleeve to access the nozzle ID for repairs. In addition, the lower ends of the repaired CRDM nozzles were cut off using a water-jet process to allow access of manual welding operations. Biological shielding was installed under the RV head while it was on the head stand during the repair operations.

Since the initial repairs of Bugey 4, repair tooling has been developed so that both BWNT and Framatome could use fully automated processes to repair CRDM nozzles. This repair tooling includes:

- Remotely operated tooling to cut off and reweld the lower part of the thermal sleeve, thus avoiding the need to remove the CRDM to access for repairs (shown in Figure 3)

- An automated excavation tool that removes defects in any orientation up to approximately 75% throughwall

- Automated welding, which lays down axial weld beads to minimize circumferential stresses. This technique eliminated the need to cut off the lower part of the CRDM nozzle to allow access for manual welding operations (shown in Figure 4)

- Additional welding qualifications, which allow the area to be repaired to be enlarged

- An automated boring machine to machine the entire inner surface

This tooling has been used by Framatome to repair 17 CRDM nozzles at EdF plants. This tooling has significantly reduced the time and dose needed to perform repairs, as well as increased the service life expectancy of the repair before reinspection is required.

DELIVERING INSPECTION/REPAIR TOOLING

The high radiation level anticipated during the performance of inspection and repair from underneath the reactor vessel head necessitated automation to minimize the dose.

To perform inspection and repairs, BWNT developed a robotic tooling system, named RΘMAN (Remotely Operated Manipulator to Access Nozzles), that is capable of delivering all inspection and repair without human intervention underneath the vessel head.

RΘMAN has five independent motions and was designed to deliver large payloads. The robot moves in a polar coordinate reference frame (R and θ) with dual vertical slides (Z1 and Z2), so that the end effectors can access 100% of all pressurized water reactor (PWR) reactor vessel head nozzles (thermocouple, capped housing, CRDM, and center vent nozzles). The fifth motion (β) enables the end effectors to be positioned from vertical to horizontal and sent out from underneath the vessel head by the manipulator for maintenance or changeout with other end effectors. This changeout process can occur very quickly because the β interchange plate utilizes integral air ports and robocouplings for rapid power and signal cable mating. Changeover from an inspection task to a repair task takes only a matter of minutes, thus minimizing the impact to the outage. Since the end effectors are changed quickly outside of the vessel head stand, radiation exposure is minimized. Figure 5 shows a schematic of RΘMAN. Figure 6 shows RΘMAN delivering the excavation repair tooling to a CRDM nozzle.

The use of robotics has allowed BWNT to lower the overall average impact on outage cost, dose, and critical path time associated with CRDM nozzle inspections and repairs.

Because CRDMs in B&W-designed plants are flanged, removing the CRDMs, to gain access to the nozzle ID, is easier than in other plant designs. Since these CRDMs are removed from time to time due to other problems, BWNT has also developed tooling to deliver inspection end effectors from above the RV head. Similar tooling will also be used to inspect the in-core instrumentation (ICI) nozzles from above the RV head at Palisades during their 1995 refueling outage.

FIELD IMPLEMENTATION

To date, Framatome has performed 100% ECT inspections at 13 EDF plants. In addition, they have repaired 17 CRDM nozzles at EDF plants, as well as replaced the Bugey 3 CRDM nozzle.

BWNT performed the blade ECT inspection, using RθMAN, of 20 CRDM nozzles in the outer three circles at Almaraz Unit 1 in September 1993. BWNT also recently completed the inspection of the Oconee 2 CRDM nozzles in October 1994, which was one of the first three plants inspected in the U.S. RθMAN was utilized to clean the nozzles and perform the blade ECT inspection of all 69 nozzles. Additional non-destructive techniques (MRPC ECT , PT, and UT) were also utilized to further characterize eddy-current results on several nozzles. In addition to the inspections, BWNT was prepared to perform any repairs that may have become necessary.

RV HEAD AND CRDM NOZZLE REPLACEMENTS

CRDM Nozzle Replacement
Clearly, from an economic or shutdown duration standpoint, replacing damaged nozzles rather than the entire RV closure head should be considered when there are a few highly crack-affected nozzles.

Framatome's first experience with a nozzle replacement of this kind was in May 1992 during the replacement of the nozzle at Bugey-3, which experienced a leak in September 1991. The particular condition of the plant (all of the CRDMs and thermal sleeves had been removed) along with the selected metallurgical goals led to a unique design for the replaced nozzle and for special tools and welding procedures. Framatome has since completed the design and development of faster and more versatile methods and tools (for machining, welding, and inspection) to replace a CRDM penetration on site during a routine outage.

RV Head Replacement
In specific cases, it may be preferable to replace the entire RV closure head rather than undertake repetitive inspections, local repairs, specific replacements, or preventative actions.

For new closure heads, Alloy 600 has been replaced by Alloy 690, which has been demonstrated internationally as being significantly less susceptible to PWSCC. In addition, the SCC sensitivity of the weld materials has been improved by adopting new weld materials - Alloys 52 and 152.

The on-site RV closure head changeout is now a well-proven operation. In 1992, Framatome replaced the Bugey-4 head with one that had been destined for the Lemoniz-2 plant in Spain. The goal of performing this changeout - to do the job during a routine scheduled outage including disassembly and reassembly of all the CRDMs - was accomplished.

RV closure head replacements began in 1994 with Bugey-5 and Le Blayais-1, which were performed simultaneously. A total of six RV heads were replaced in 1994. More replacements are proposed for this year as well as in the future.

OVERALL PLAN TO ADDRESS CRDM NOZZLE PWSCC

As a result of the potential for this problem to occur in U.S. plants, the materials committee of the B&W Owners Group (B&WOG) initiated a comprehensive program to evaluate the susceptibility of B&W-designed CRDM nozzles. This program included the following activities:

- Preparation of a CRDM Nozzle Safety Evaluation
- Development of Flaw Acceptance Criteria
- Participation in the Demonstration of Vendor Inspection Performance
- Verification of Crack-Growth Curves.

B&WOG activities were performed in coordination with the NEI Task Force, which was formed to address this issue on an industry-wide basis.

The safety evaluation included the performance of stress and fracture analysis, crack growth analysis, leakage assessment, and wastage assessment for the B&W-designed CRDM nozzles. The stress analysis performed indicated that if cracking should occur, it would be axially oriented. Furthermore, it was estimated by fracture mechanics analysis that it would take a minimum of six years for a crack to grow throughwall.

The evaluation concluded that there was no immediate safety concern and that if throughwall cracking was to occur, sufficient time would be available to identify leakage and develop a strategy for repair or replacement of the leaking nozzles.

In cooperation with the NEI Task Force, BWNT participated in developing simplified flaw-acceptance criteria for CRDM nozzle cracking. These criteria have been reviewed by the NRC and the criteria for axial cracking has been accepted. Circumferential indications, however, are to be dispositioned on a case-by-case basis.

Another aspect of the overall program to address this issue was the verification of the inspection techniques utilized by the vendors to perform CRDM nozzle inspections. Three mockups that contained flaws representative of what has been seen in the field were fabricated by the EPRI NDE Center. Prior to the Oconee 2 CRDM nozzle inspection, BWNT successfully demonstrated our blade and MRPC ECT, and Rotational UT techniques on the EPRI NDE Center mockups. All of the implanted defects were detected using both UT and ECT. Length sizing, flaw location, and depth sizing accuracy were well within industry standards.

BWNT through our B&WOG programs has also supported the performance of crack growth studies to verify the safety evaluation results.

BWNT through its B&WOG program continues to maintain a proactive approach to this problem. In 1995, the B&WOG will be performing replication on CRDM nozzles located on the reactor vessel head periphery to better characterize material microstructures. The B&WOG is also developing a long-range planning/economic model to manage from a utility perspective the CRDM PWSCC issue. Additionally, the B&WOG continues to support NEI Task Force activities.

SUMMARY& CONCLUSIONS

The complete treatment of the problem for the cracking of reactor vessel closure head CRDM nozzles has posed a real challenge. The dedicated development of full-service processes and tooling has enabled Framatome to meet this challenge effectively in French plants and BWNT to be prepared for evaluating similar cracking as it occurs in domestic plants. Response to this issue on an industry-wide basis has allowed utilities to address this issue in an economical manner. Future plans in the U.S. will depend entirely on the extent of the problem.

Biography of Authors

Kevin W. Fleming / B&W Nuclear Technologies - Currently holds the position of Business Development Manger, Specialized Plant Services of the Special Products and Integrated Services (SPIS) division of BWNT. He is responsible for the development of BWNT's CRDM head penetration service package. He has a B.S. in Physics from the University of Maryland and an M.B.A. from Lynchburg College. He has worked with BWNT for 13 years.

Steve Fyfitch / B&W Nuclear Technologies - Currently holds the position of Senior Supervisory Engineer of the Materials Group in the Engineering and Project Services (EPS) Division of BWNT. He has a M.S. in Materials Science and Metallurgical Engineering from Carnegie-Mellon University. He has worked with BWNT for 8 years.

Blade ECT Probe

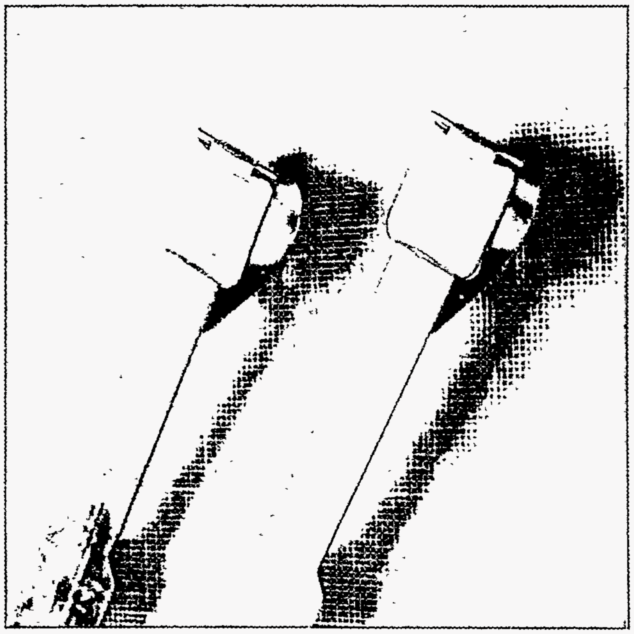


Figure 1

UT Transducer Array

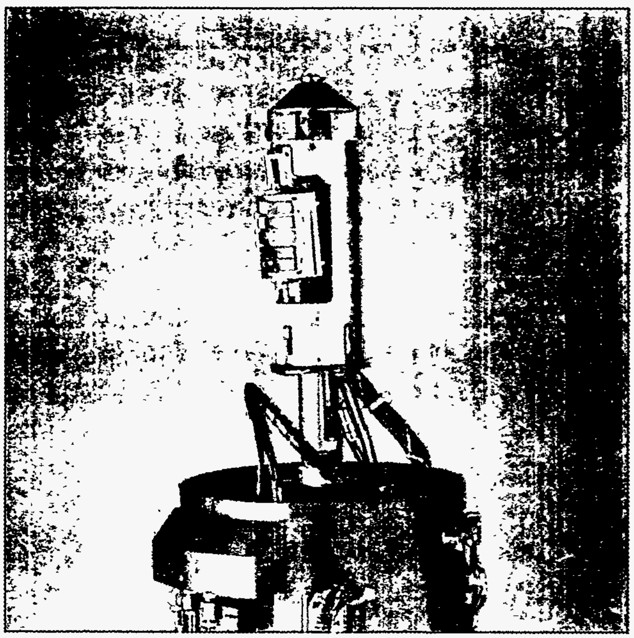


Figure 2

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Thermal Sleeve Cutting

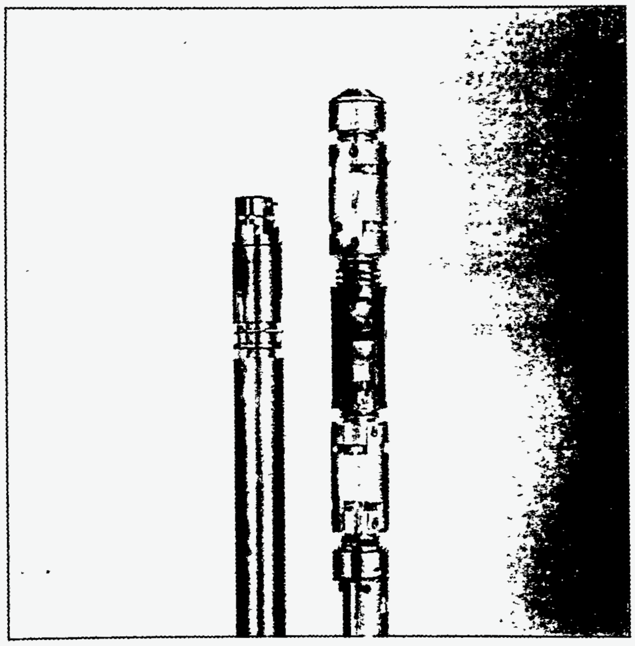

Figure 3

Local Cavity Weld Repair

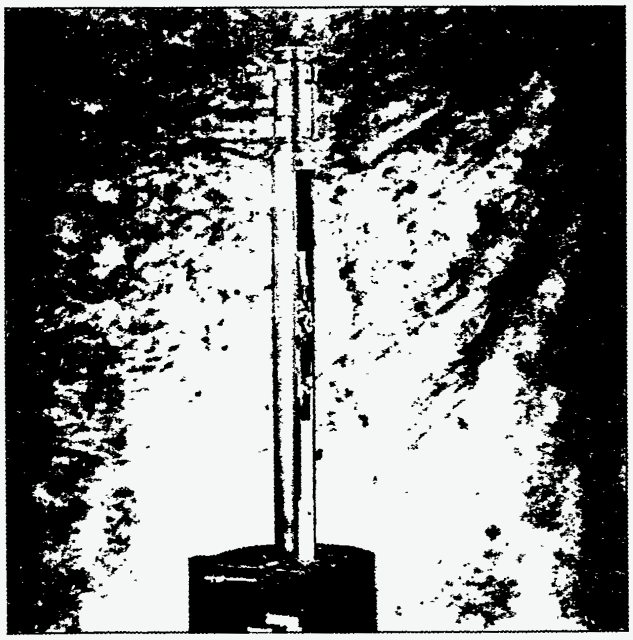

Figure 4

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RθMAN

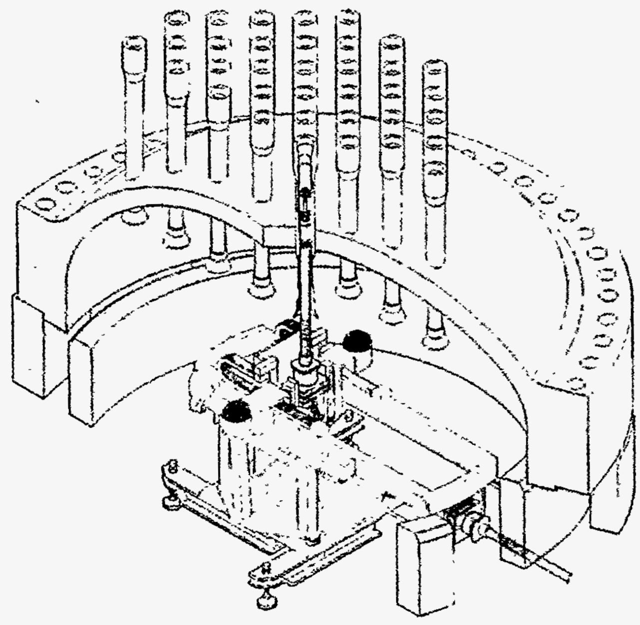

Figure 5

Excavation Tool on RθMAN

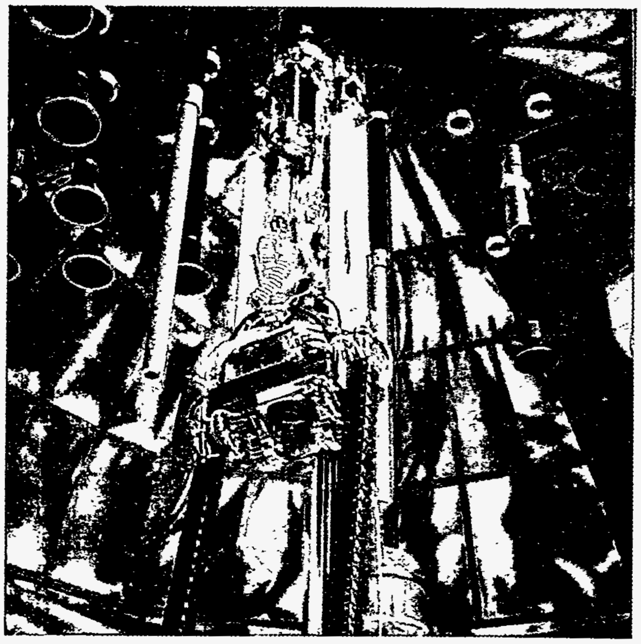

Figure 6

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RESIDUAL AND OPERATING STRESSES IN WELDED ALLOY 600 PENETRATIONS

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ABSTRACT

An elastic-plastic finite element model has been developed for calculating residual and operating stresses in Alloy 600 penetrations which are installed in pressure vessel shells by J-groove welds. The welding process is simulated by multiple passes of heat input with heat transfer into the adjacent parts during welding and cooling.

Analysis results are presented for CRDM nozzles, pressurizer instrument nozzles and pressurizer heater sleeves. The effect of several key variables such as nozzle material yield strength, angle of the nozzle relative to the vessel shell, weld size, presence of counterbores, etc. are explored. Results of the modeling are correlated with field and laboratory data. Application of the stress analysis results to PWSCC predictive modeling is discussed.

1. INTRODUCTION

Primary water stress corrosion cracking (PWSCC) of Alloy 600 primary loop penetrations in PWR plants was first reported in 1982 in a pressurizer heater sleeve at Arkansas Nuclear One Unit 2. PWSCC has subsequently been reported in several other types of primary loop penetrations in PWR plants including pressurizer instrument nozzles, steam generator drains, and reactor vessel head CRDM nozzles (1). Figure 1 shows typical penetration arrangements and crack locations. It is noted that the cracks in all of the penetrations are located near the J-groove welds which attach the penetrations to the vessel shells.

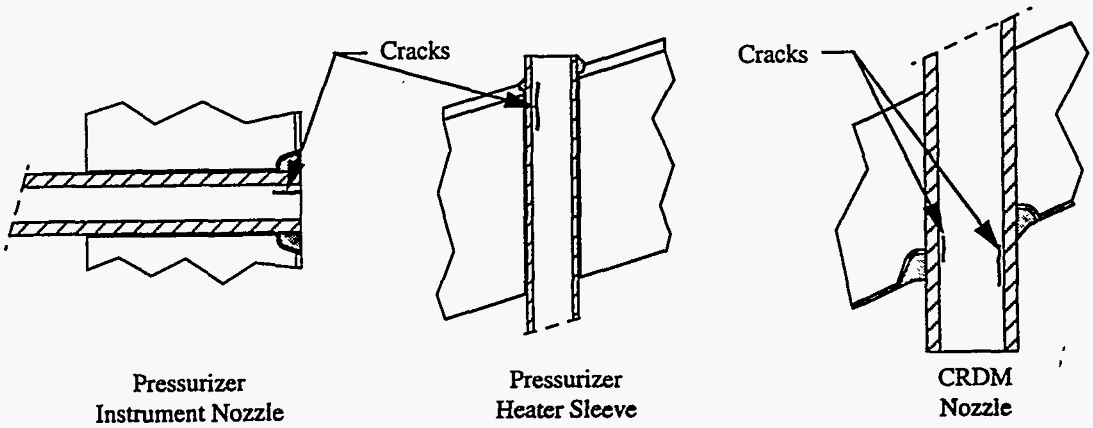

Figure 1– Typical Alloy 600 Penetrations and Crack Locations

Tensile stresses are necessary for PWSCC to occur in Alloy 600 materials. Tensile stresses can be induced by internal pressure and differential thermal expansion under operating conditions. However, these stresses are limited by design code requirements and the stresses generally do not exceed 100–200 MPa. These pressure and thermal stresses are less than the stress threshold for PWSCC of Alloy 600 material in normal PWR environments. However, much higher tensile stresses can be created by shrinkage forces when penetrations are welded into vessel shells. The combination of operating stresses superimposed on high residual stresses produced by welding is believed to be a major contributing cause of the PWSCC which has been reported in Alloy 600 PWR penetrations.

2. ANALYSIS METHOD

Stresses in CRDM nozzles, pressurizer instrument nozzles and pressurizer heater sleeves have been calculated using the elastic-plastic finite element method and the ANSYS finite element analysis program. All nozzles are analyzed as 3D penetrations in spherical shells. As shown in Figures 2 and 3, the models include a conical sector of the vessel shell, and the boundary conditions are selected to permit only radial deflections in the spherical coordinate system. These boundary conditions simulate the stiffness and pressure stresses in the vessel shells remote from the penetrations.

Four materials are considered in the analyses. The vessel shells are assumed to be low alloy steel, the penetrations are assumed to be Alloy 600, the J-groove welds and buttering are assumed to be Inconel weld metal, and the cladding on the inside surfaces of the vessel shells is assumed to be either 300 series stainless steel, or Inconel as appropriate. Materials are assumed to have temperature dependent material properties. Figure 4 shows typical yield strengths of the materials as a function of temperature. High temperature properties for the Inconel materials were developed by EdF (2) based on work performed in their mockup testing program.

The analyses of the penetrations involved four loading steps:

Nozzle Installation – Pressurizer instrument nozzles and heater sleeves are installed in clearance holes in the vessel shells. The clearance holes are simulated in the finite element models by gap elements. CRDM nozzles are installed in the vessel heads using either a metal-to-metal fit or a small interference. CRDM nozzle analyses were all performed for the case of a metal-to-metal fit since separate effects studies showed that this condition produces the highest tensile stresses on the inside surface of the nozzles. Gap elements with zero initial clearance are used to simulate the metal-to-metal fit between the CRDM nozzles and holes in the vessel heads, and to permit some displacement of the nozzle wall as the pressure stresses in the vessel shell cause the hole to dilate.

Welding – All of the nozzles evaluated are joined to the vessel shells by J-groove welds. The J-groove welds are simulated as rings of weld metal in which one ring is applied at a time. Non-axisymmetric effects of progressively deposited welds are neglected. Parametric analyses showed that two weld passes provide a good approximation of the stress distributions on the inside surface of the nozzles predicted using larger numbers of passes. The welding process is simulated by combined thermal and structural analyses. Thermal analyses are used to generate temperature distributions throughout the joint at several points in time during the welding process. These temperature distributions are then used as input conditions to the structural analyses. The sequence of thermal analyses followed by structural analyses was duplicated for each of the two simulated welding passes. Figure 5 shows the temperature distribution during the first pass of a typical CRDM nozzle weld.

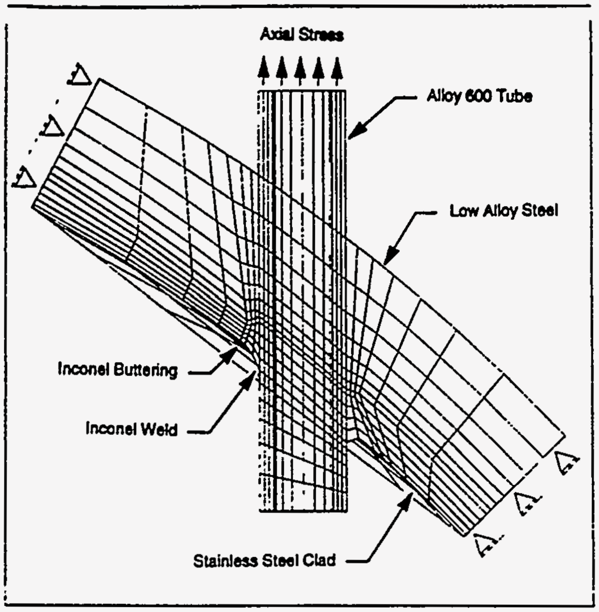

Figure 2 – Finite Element Model of Typical CRDM Nozzle

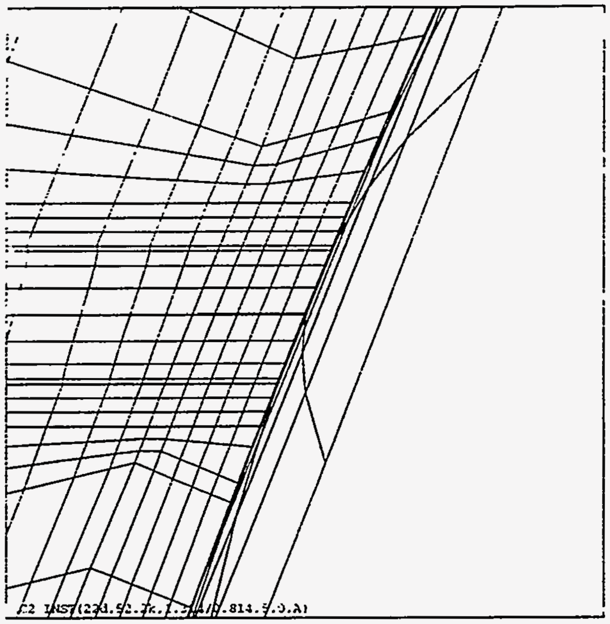

Figure 3 – Finite Element Model of Typical Pressurizer Instrument Nozzle

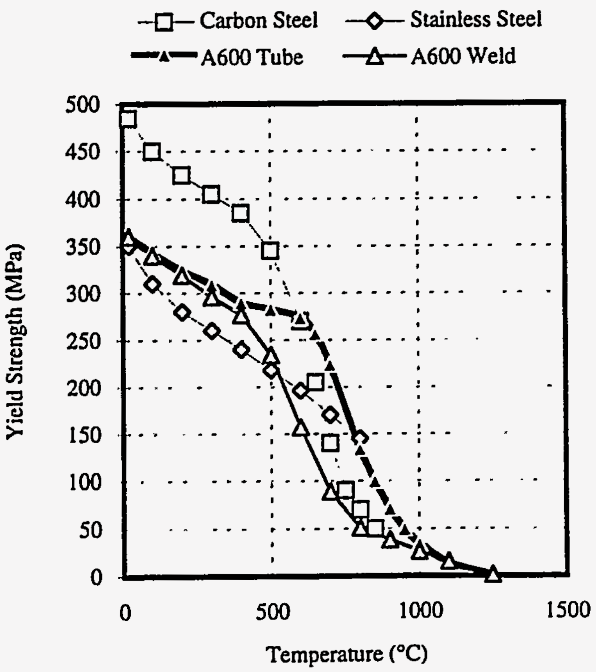

Figure 4 – Typical High Temperature Yield Strengths

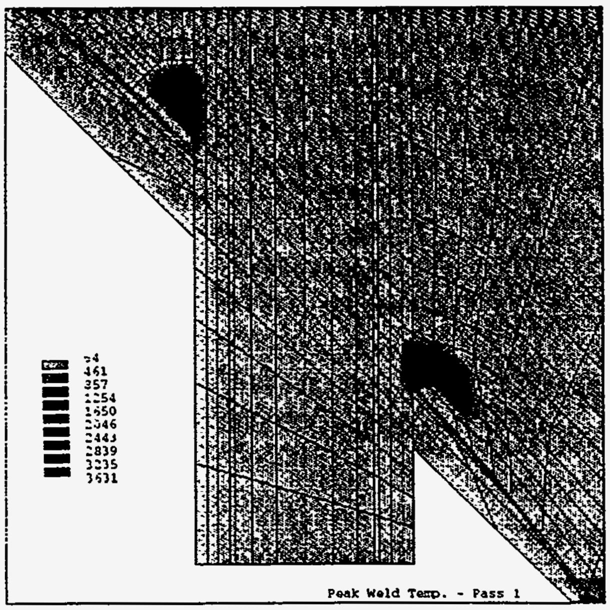

Figure 5 – Typical First Pass Weld Temperature Distribution

Hydrostatic Testing – The pressure vessels and nozzles are hydrostatically tested to a pressure of about 215 bar prior to plant operation. This operation is simulated by applying an internal pressure to the inside surface of the vessel shell and nozzle and an axial load on the end of the nozzle.

Operating Conditions – The final loading step is to apply an internal pressure of 155 bar, a uniform temperature of 315°C, and an axial load on the end of the nozzle.

3. ANALYSIS CASES AND RESULTS

Analyses have been performed for a wide range of different penetration and mockup designs. Several cases of general interest and key analysis results are given in Table 1.

Figure 6 shows the deformed shape of typical central and outer row CRDM nozzles after welding (Note: The displacement scales are exaggerated.) The weld metal is not shown in these figures since it is highly distorted at the selected scale. The figure of the central nozzle shows the nozzle wall being pulled radially outward by the weld shrinkage. This deflection produces high tensile hoop stresses and somewhat lower axial stresses. The figure of the outer nozzle shows that the bottom of the nozzle is deflected laterally away from the center of the vessel head and that the nozzle becomes ovalized by weld shrinkage.

Figure 7 is a plot of the hoop stresses along the inside surface of a typical outer row CRDM nozzle on the downhill side of the nozzle (side farthest from the center of the vessel head). This figure shows high residual stresses after welding, reduced residual stresses after a cold hydrostatic test, and operating condition stresses between these two extremes. This figure shows the beneficial effect of hydrostatic testing in reducing residual stresses. The cause of the stress reduction is as follows: 1) maximum stresses in the nozzle wall are close to the material yield strength after welding, 2) the pressure induced dilation of the vessel head causes yielding of the portion of the nozzle near the weld, and 3) the material relaxes to a residual stress less than the material yield strength after the internal pressure is removed. The absence of a hydrostatic test after replacing some pressurizer nozzles may have contributed to the replacement nozzles developing through-wall cracks in a shorter time than the original nozzles.

Figure 8 shows the hoop and axial stresses along the inside surface of a typical outer row CRDM nozzle. These data show that the hoop stresses exceed the axial stresses at all high stress locations. This fact is confirmed by the data in Table 1 for several other nozzle configurations and locations.

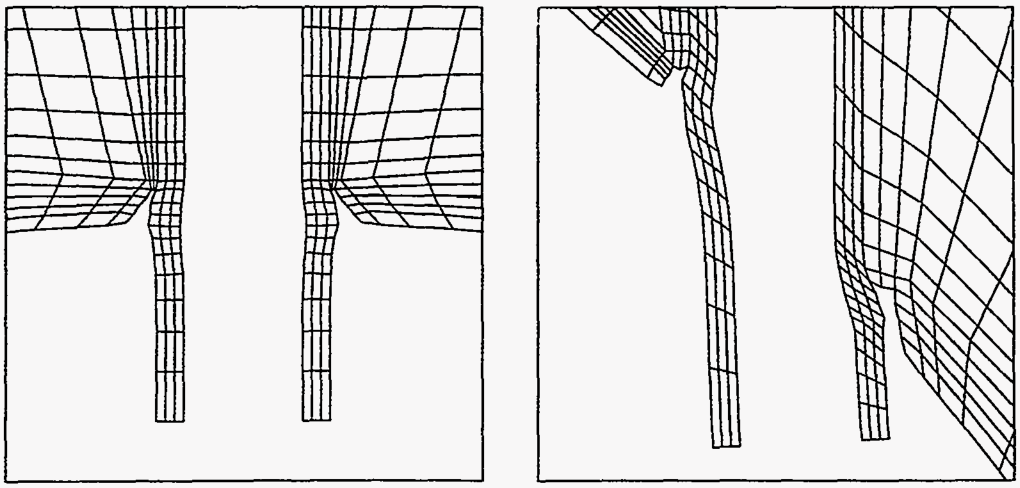

Figure 6 – Deflections After Welding in Typical Central and Outer Row CRDM Nozzles

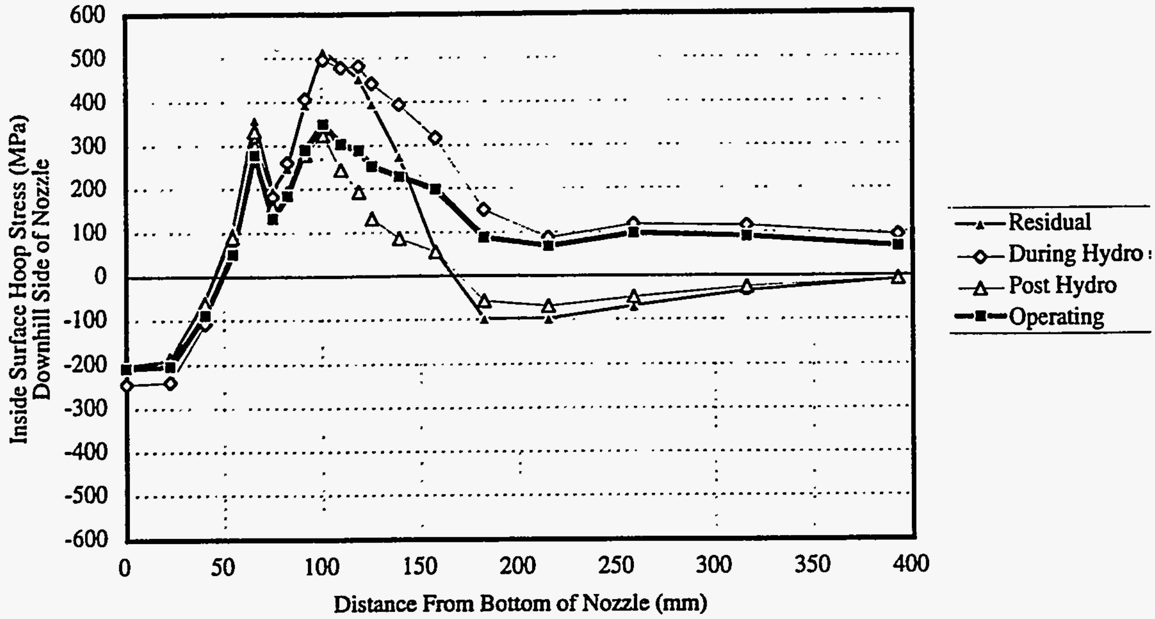

Figure 7 – Hoop Stress on Inside Surface of Typical Outer Row CRDM Nozzle

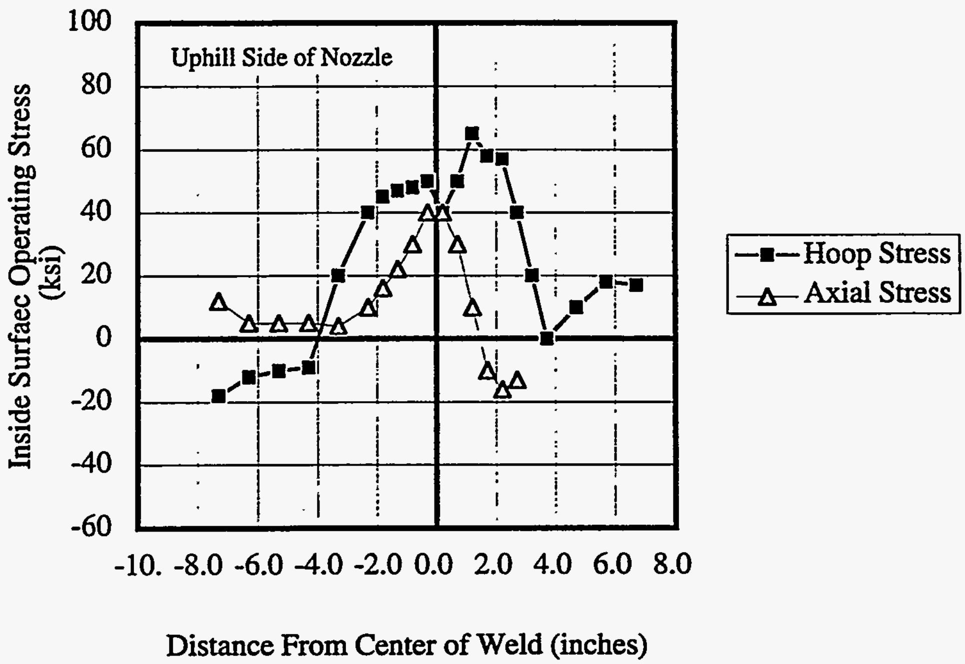

Figure 8 – Operating Hoop and Axial Stresses on Inside Surface of Typical Outer Row CRDM Nozzle

Table 1 Analysis Cases and Selected Results

Parameter	Units	Pressurizer Nozzles			EdF CRDM Nozzles*				
		Instrument	Heater Sleeve		Central	Intermediate Row		Outer Row	
Counterbore on Nozzle Inside Surface	---	No	No	No	No	No	Cylind	No	Taper
Material Yield Strength	MPa	359	359	435	359	359	359	359	359
Nozzle Incidence Angle	deg	22.5	45	45	0	39	39	47	47
Nozzle Geometry									
– Nozzle Outside Diameter, OD	mm	33.4	29.4	29.4	101.6	101.6	101.6	101.6	101.6
– Nozzle Inside Diameter, ID	mm	20.7	23.2	23.2	69.9	69.9	69.9	69.9	69.9
– Nozzle Wall Thickness, t	mm	6.4	3.1	3.1	15.9	15.9	15.9	15.9	15.9
– OD/t	---	5.26	9.51	9.51	6.40	6.40	6.40	6.40	6.40
MaxID Surface Deflections (Residual)									
– Diametral Increase	mm	---	---	---	0.28	---	---	---	---
– Lateral Deflection	mm	---	---	---	---	0.66	0.66	0.97	0.97
– Ovality	mm	---	---	---	---	0.89	0.84	1.19	1.32
Max ID Hoop Stresses (Operating)									
– Axisymmetric Nozzles	MPa	---	---	---	302	---	---	---	---
– Oblique Nozzles – Uphill	MPa	401	423	497	---	396	395	384	381
– Oblique Nozzles – Downhill	MPa	408	358	410	---	280	298	308	348
Max ID Axial Stresses (Operating)									
– Axisymmetric Nozzles	MPa	---	---	---	239	---	---	---	---
– Oblique Nozzles – Uphill	MPa	264	341	399	---	331	331	329	324
– Oblique Nozzles – Downhill	MPa	333	239	270	---	220	230	219	258
Ratio of Axial Stresses to Hoop Stresses									
- Axisymmetric Nozzles	---	---	---	---	0.79	---	---	---	---
– Oblique Nozzles – Uphill	---	0.66	0.81	0.80	---	0.84	0.84	0.86	0.85
– Oblique Nozzles – Downhill	---	0.82	0.67	0.66	---	0.79	0.77	0.71	0.74

* Dimensions of EdF nozzles taken from EdF specification (2).

4. CORRELATION WITH FIELD AND EXPERIMENTAL DATA

Results of the finite element analyses have been correlated with available experimental and field data. These correlations are described in a recent EPRI summary report on Alloy 600 PWSCC (1). Key conclusions are as follows:

Correlations With Reported Crack Locations and Orientations – There is good correlation between the locations of high calculated tensile stresses in each nozzle type and the locations of reported cracking from the field.

- Cracking reported in the field has typically been located near the welds. Analyses show that the maximum stresses for all nozzle designs are located near the welds at locations correlating with the reported cracking.

- Cracking on the inside surfaces of CRDM and pressurizer nozzles has been predominantly axial. The only exceptions for plants with normal water chemistry have been shallow non-axial fissures in a few nozzles which have many predominantly axial fissures, and shallow circumferential cracks in roll expanded EdF pressurizer instrument nozzles. This experience is consistent with analysis results which show that the hoop stresses exceed the axial stresses at all high stress locations.

- Cracking in oblique CRDM nozzles has been axial and located on the uphill and downhill sides of the nozzles. The analyses show that the hoop stresses in oblique nozzles are highest on the uphill and downhill sides of the nozzles. High stresses at these locations result from ovality induced by weld shrinkage.

- Cracking in oblique nozzles has occurred predominantly below the welds on the uphill sides of the nozzles. Analyses for most outer row CRDM nozzles show that the maximum hoop stresses are typically highest below the weld on the uphill sides of the nozzles. However, this is not true in all cases. Figure 9 shows a plot of hoop stresses in Oconee 2 nozzle No. 23 and the very shallow indications which were found in this nozzle. Note that the indications are centered within the highest stress location, and that this location lies above the weld on the downhill side of the nozzle.

- Finally, cracking in CRDM nozzles occurs most frequently in the outer row nozzles. Analyses of typical central and outer row nozzles typically show that stresses are higher in the outer row nozzles. However, this is not always the case as discussed in Section 5.

Correlations With Measured Deflections – The maximum lateral deflection and ovality have been measured on several CRDM nozzle mockups. Table 2 provides a comparison of the measured and calculated deflections and ovality for mockups fabricated by EdF (2) and Ringhals (6). These results show that the measured and calculated deflections are in reasonably good agreement.

Table 2 – Comparison of Measured and Calculated Deflections and Ovalities

Mockup Design and Type of Deflection	Measured	Calculated
EdF CRDM Nozzle Mockup (47°)		
- Lateral Deflection (mm)	1.73	0.97
- Ovality (mm)	1.62	1.32
Ringhals CRDM Nozzle Mockup (43°)		
- Ovality (mm)	1.15	1.25

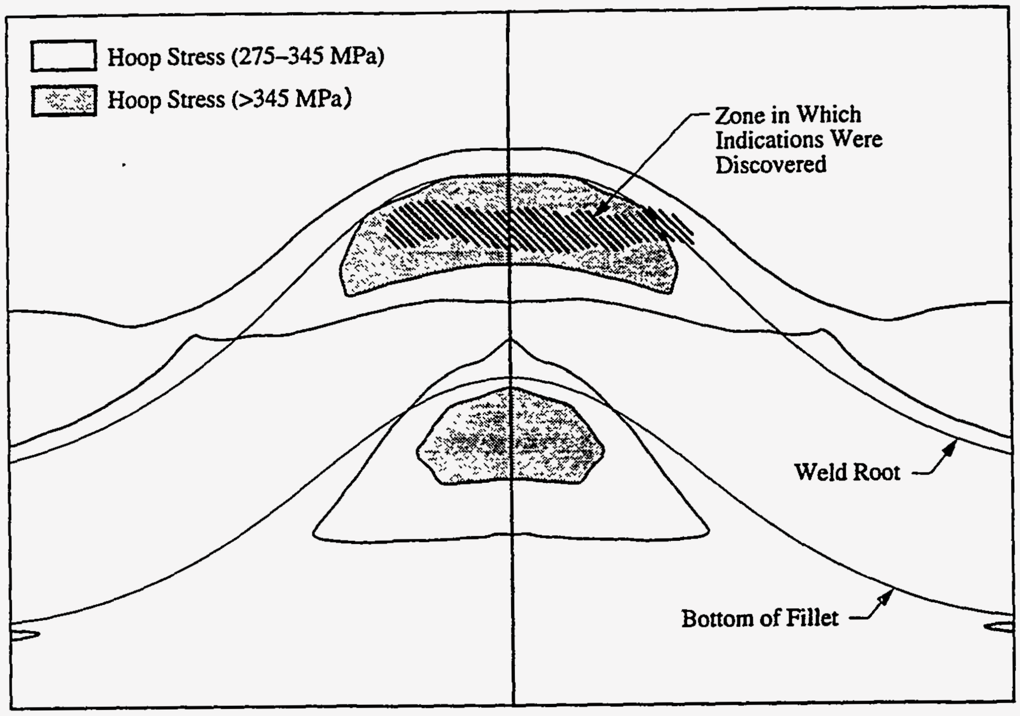

Figure 9 – Correlation Between High Stresses and Location of Axial Indications in Oconee 2 CRDM Nozzle No. 23

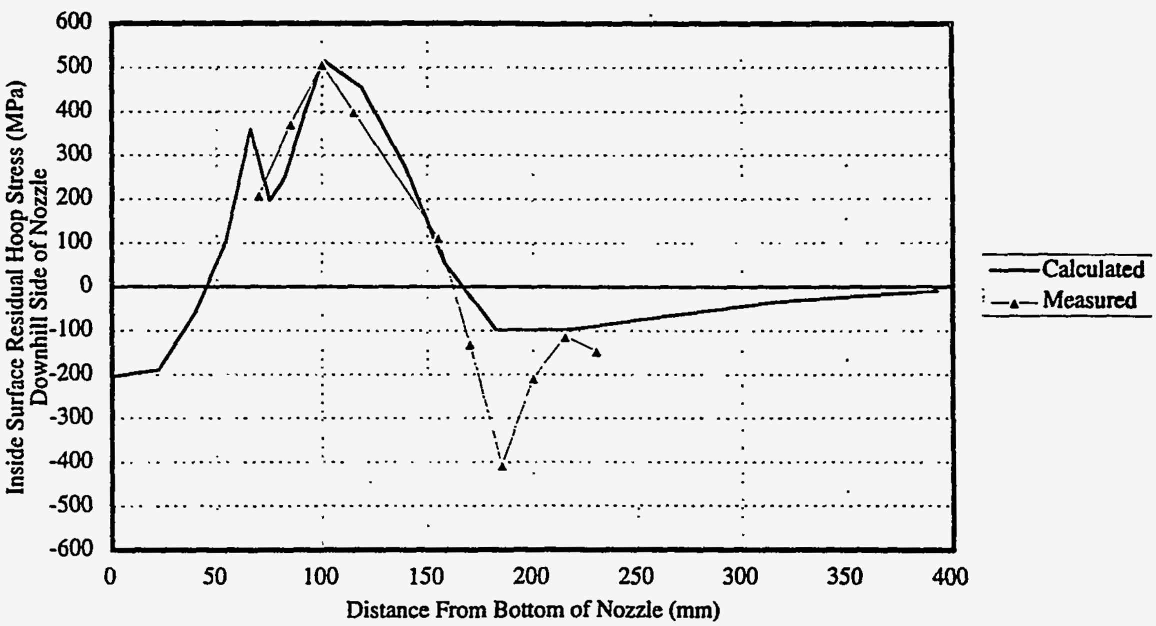

Figure 10 – Correlation Between Calculated and Measured Hoop Stresses in Typical Outer Row CRDM Nozzles

Correlations With Measured Residual Stresses – Combustion Engineering has performed x-ray residual stress measurements of pressurizer heater sleeve mockups for EPRI. A comparison of the experimental and analytical stress analyses is included in a recent EPRI report (5). The main conclusion from this work is that the analytical results provide an upper bound to the experimental results.

EdF has performed strain-gage hole drilling residual stress measurements on several CRDM nozzle mockups. Figure 10 shows that there is good agreement between the measured and calculated inside surface stresses for the downhill side of an outer row CRDM nozzle. More complete results are included in a recent EPRI report (1).

Comparison With Corrosion Test Experience – EdF and Framatome have performed corrosion tests on Alloy 600 CRDM nozzle mockups to determine crack orientation. The results of these tests showed that: (1) cracks occurred in outer row mockups but not in central mockups, (2) all cracks were axial, and (3) all cracks were located on the uphill and downhill sides of the nozzles. These findings are in agreement with the stress analysis results.

In summary, analysis results are considered to be in reasonably good agreement with reported field experience and laboratory testing.

5. EFFECT OF WELD DESIGN VARIATIONS ON STRESSES

A cursory review of drawings of CRDM nozzles from different vendors suggests that they are all essentially the same. On this basis it would be expected that all of the nozzles should have essentially the same maximum calculated stresses if analyzed using the same modeling assumptions, material properties, and computer program. In order to check this assumption analyses were performed for several different CRDM nozzle designs using the same assumptions, material properties and computer program described in Section 2 of this paper. The main differences between these nozzles were small differences in nozzle diameters and more significant differences in weld sizes.

Figure 11 shows the maximum hoop stresses in several different central CRDM nozzle designs as a function of the ratio of the length of the weld in contact with the nozzle to the nozzle wall thickness. The conclusion from this work is that the weld size is a significant variable in predicting the maximum operating condition stress.

Figure 12 shows the maximum inside surface hoop stresses in nozzles from several different plants for a range of nozzle incidence angles. These results were calculated assuming that all of the nozzles have a yield strength of 360 MPa. These data also show that weld size is a significant factor in predicting hoop stresses.

6. USE OF FINITE ELEMENT MODELING FOR STRATEGIC PLANNING

Work described in this paper has shown that there is a correlation between locations of high predicted tensile stresses and locations where cracking has been reported from the field. This information can be used in developing strategic plans to predict when cracks are likely to occur in a specific plant and which nozzles are likely to be affected by the cracking first. This information can be used by utilities to target inspections towards locations of highest predicted PWSCC susceptibility.

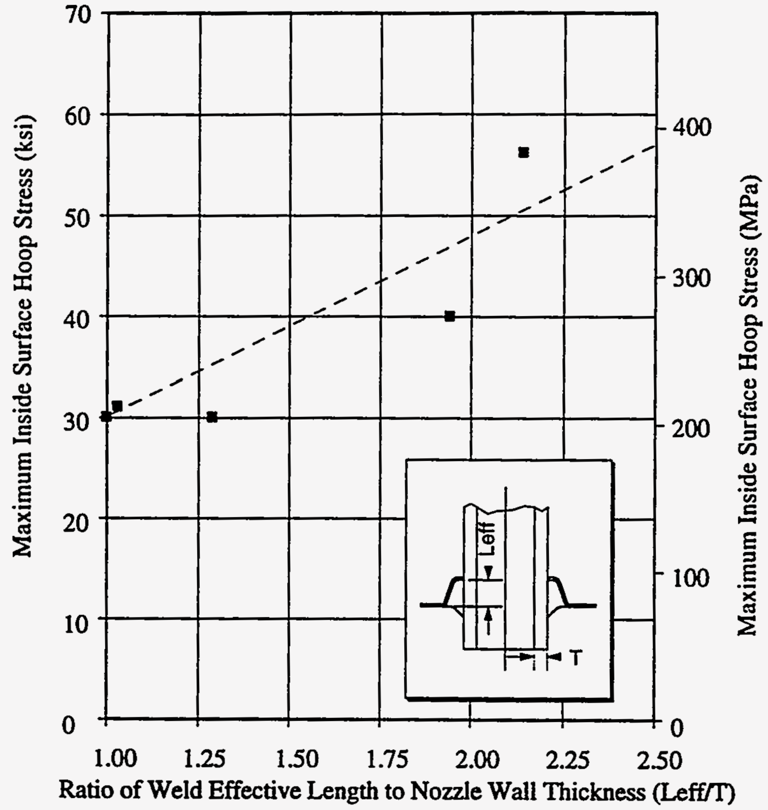

Figure 11 – Effect of Weld Size on Maximum Hoop Stress in Central CRDM Nozzles

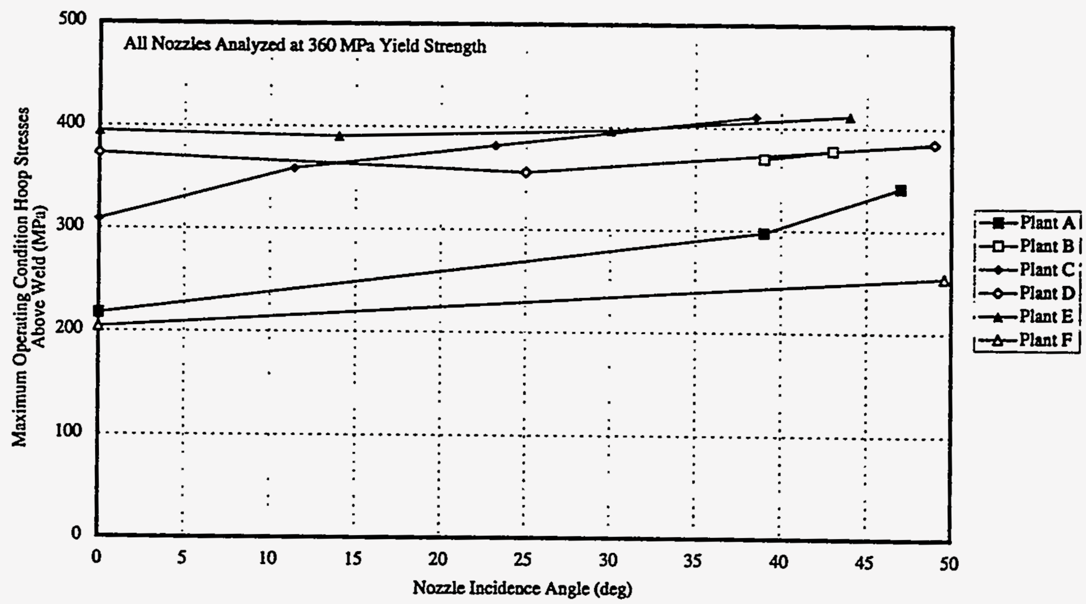

Figure 12 – Effect of Vessel Design on Hoop Stresses in CRDM Nozzles

7. CONCLUSIONS

The main conclusions from the finite element analysis work were as follows:

- The finite element method provides a reasonable approach for evaluating stress levels in welded nozzles.
- Welding a nozzle into a vessel shell with a J-groove weld produces a high restraint condition that can cause high tensile residual stresses.
- Weld heat input causes the nozzle wall to expand radially outward. Some of this thermal displacement is "locked-in" by the weld as it cools, contracts and strengthens.
- For outer row nozzles, the highest stresses are hoop stresses resulting from ovalization.
- Hoop stresses exceed axial stresses at all high stress locations such that cracking is expected to be predominantly axial.
- For the case of oblique nozzles, the cracks are likely to be skewed axial cracks, oriented in a direction 90° to that of the maximum principal stress, and centered on the uphill and downhill sides of the nozzles.
- It should be assumed that all Alloy 600 nozzles attached to vessel shells by J-groove welds have operating condition stresses high enough to cause PWSCC unless it can be shown by analysis, or other means, that the stresses are below a stress threshold of about 250 MPa.

8. ACKNOWLEDGMENTS

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INTEGRITY EVALUATION OF ALLOY 600 RV HEAD PENETRATION TUBES IN KOREAN PWR PLANTS

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Abstract

The structural integrity assessment of Alloy 600 reactor vessel head (RVH) penetration tubes has been an important issue for the economical and reliable operation of power plants. In this paper, an overview of the integrity evaluation program for the RVH penetration tubes in Korean nuclear power plants is presented. Since the crack growth mechanism of the penetration tube is due to the primary water stress corrosion cracking (PWSCC), which is mainly related to the stress at the tube, this paper consists of three primary activities: the stress evaluation, the flaw evaluation, and data generation through material and mechanical tests.

1 Introduction

Since the leaking crack was discovered on the Bugey-3 reactor vessel head (RVH) penetration tube during its 10-year hydrotest (September 1991), much work has been done both in the United States and in Europe to develop analytical approaches and maintenance strategies for this issue. Based on their work, it was demonstrated that the cracking of the reactor vessel head penetration tube was not an immediate safety issue. However, the results of the work indicated that the tube cracking has to be managed as a long-term project to develop an optimal maintenance strategy.

The nuclear industries in Korea have given great attention to the structural integrity of the RVH penetration tubes since the first indication of cracking was reported at the Bugey 3 unit. The present work, from August 1994 to August 1996, is to provide a quantitative measure of the integrity of the RVH penetration tubes of Korean nuclear power plants for the continuation of safe operation.

Stress evaluations, flaw evaluations, and material tests for selected plants are carried out in this evaluation program to evaluate the structural integrity of the RVH penetration tubes.

Stress evaluation focuses on the analysis procedure and the simulation methodology of input loads such as thermal effect due to welding, hydrotest pressure, and operating pressure and temperature. For flaw evaluation, some assumptions are established, and the related literature for the primary water stress corrosion cracking (PWSCC)crack growth assessment is surveyed. Tests for obtaining the material properties are being done. The inspection program of the head penetration tubes of Korean pressurized water reactor (PWR) power plants is presented briefly.

2. Stress Evaluation

Stress analysis is required to evaluate the structural integrity for the RVH penetration tubes. The stress analysis for the RVH penetration tubes is very complex because all tubes are geometrically nonsymmetrical except center one. Thus 3D finite element analysis should be employed for the stress analysis. Also, the residual stress resulting from the nonsymmetric weld procedures at the RVH area should be determined, because that is

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one of the key stress components for the RVH penetration tubes. The magnitude and distribution of residual stress resulting from welding can be determined analytically by simulating the welding procedure. The weld pass, weld volume, and weld heat input may be the main parameters for the determination of stress values. However, the plant-specific weld procedures that can give the above information are not available. Thus, the parametric study with a 2D finite element model is carried out to develop the conservative and simplified analysis procedure, which is directly applied to the 3D finite element stress analysis.

The following are our basic approaches and procedures to calculate the stress at the RVH penetration tubes.

The loads included for stress analysis are thermal effect due to welding, hydrotest pressure, and operating load. The reliability of the stress analysis results is entirely dependent on the simulation method of the loads. The simulation methodology used in this work is described in the following sections.

Finite Element Modeling

For stress evaluation, 2D and 3D finite element models for three kinds of penetration tubes at center, medium, and outermost locations are constructed. The elastic-plastic stress analysis for those models is performed by using the ANSYS code. Since the central penetration tube and surrounding vessel head is axisymmetric through the tube centerline, the two-dimensional model is enough. This 2D axisymmetric model of the central penetration tube is used to determine the detailed analysis procedures, such as the number of load steps for transient heat transfer analysis and elastic-plastic stress analysis by lots of trial and error. Also, the main parameters for load simulation are determined by various sensitivity studies with the 2D model. Since most of cracks are known to be detected at outermost tubes, more precise modeling and analysis are required for an outermost tube. Half the outermost tube is modeled by taking advantage of symmetry through the vessel and tube centerline.

The reactor head, cladding, and tube are modeled as carbon steel, stainless steel, and Inconel, respectively. The materials of weld and weld buttering are also Inconel.

Simulation of Weld Residual Stress

One of the governing stress components at the head penetration tube is the residual stress that results from thermal effects due to welding. To simulate the weld procedure by finite element analysis, the transient heat transfer analysis and the elastic-plastic thermal stress analysis are carried out. The weld residual stress is affected by the number of weld passes and the combination of heat-input rate and time. Those parameters are examined by various sensitivity studies applied to the 2D central penetration tube model. The stress analyses are performed independently for 1, 2, and 4 weld pass models to determine the number of simulated weld passes to conservatively predict the stress. Heat input rate and time as a heat source of the weld elements are determined based on the diameter of the filler metal and weld speed. In welding, the peak temperature used is about 3500°F for the weld elements and about 2000°F for the adjacent base metal elements.

Simulation of Hydrotest Load and Operating Load

The manufacturing and the installation hydrotest pressure are applied through a series of loading and unloading procedures. The operating load is also applied by full loading and unloading to check the variation in the stress distribution between the final unloading state and before applying the operating load. In this case the effects of temperature as well as pressure are considered. Both the hydrotest pressure and the operating load are applied incrementally for elastic-plastic stress analysis.

Stress Evaluation for Crack Growth

Since most of the detected cracks at the penetration tube are in an axial direction, the critical stress component is the circumferential stress at the outermost tube wall during operation. This stress distribution is approximated by a cubic polynominal function for the flaw evaluation. Therefore, the mesh density at tube wall thickness becomes another important parameter. If the stress results vary seriously at the tube wall, a finer mesh will be adopted.

Even though fatigue is known to have minor influence on the crack growth of penetration tubes, stress analyses for assessing the effect of the fatigue crack growth are carried out by applying the transient load due to plant heatup and cooldown.

Present Status

The elastic-plastic stress analysis procedure established in this paper is shown in Fig. 1 assuming two weld passes. Sensitivity studies on the main parameters for simulating input loads with 2D models are almost completed. Two weld passes are determined in view of conservatism. Heat input to the weld element is as follows:

1) Linear increase of heat input rate from 0 to 100 BTU/in.^{2}/rad during 0.7 s.
2) Maximum heat input rate maintained uniformly for 1.1 s.
3) Linear decrease of heat input rate to 0 during 0.5 s.

These parameters will be used for stress analysis of the outermost penetration tube.

3. Flaw Evaluation

There are two objectives of the penetration tube flaw evaluation to predict the time required for a crack to propagate to the acceptance criteria. The first objective is to perform the parametric evaluation for a postulated crack. The second objective is to develop the flaw evaluation program for the cracks detected during the inspection.

Assumptions

Since it has been reported that most of the cracks in the RVH penetration tubes resulted from PWSCC in the Alloy 600 base metal, there is a high probability for a crack to be initiated from the surface in contact with the primary water. One of the geometric assumptions is the location and orientation of the postulated cracks. Experience with European plants has shown that the cracks are located above and below the tube J-weld and primarily in the axial direction. In addition, cracks have occurred predominantly in outermost tubes.

For the parametric evaluation, it is assumed that an ID-initiated axial crack propagates through the penetration tube above and below the tube J-weld. Although not expected, it is further assumed that a circumferential crack exists potentially. The effect of the fatigue crack growth rate obtained from experimental results is also evaluated.

Methodology

The crack growth evaluation for the part-through surface crack is based on the stress distribution through the tube wall thickness at the highest stress locations along the inner surface of the tube. The stress profile along the tube thickness is approximated by a cubic polynomial to calculate the stress intensity factor. For the throughwall crack, the crack is assumed to be subjected to the average stress through the tube wall thickness.

To perform the parametric study for a postulated crack and develop the flaw evaluation program for the cracks detected during the inspection, the stress intensity factor expression for a wide range of the surface crack configuration is required. The stress intensity factor expression presented by Raju & Newman [1] is used for the axial surface crack, while the expression by Poette and Albaladejo [2] is used for the circumferencial surface crack. These expressions include a relatively wide range of the surface crack configuration. The stress intensity factor for crack growth prediction of a throughwall crack is calculated using the standard expression for a throughwall crack in a plate. The crack propagation rates are calculated by integrating a stress intensity factor vs. crack growth curve. The formula for the crack growth rate is to be obtained from a survey of the available literature, considering the temperature effect on the growth rate based on a collection of crack growth information from both laboratory and field data [3,4].

Design data for stress analysis and flaw evaluation were reviewed, and the mechanical property data were provided by a series of experiments on the Alloy 600 specimens that had been similarly heat-treated to the penetration material of specific power plants.

4. Characterization of Material Properties of Alloy 600

Data for integrity evaluation of penetration tubes will be provided by a combination of experimental tests with a literature survey of available laboratory and field data. Tensile tests were performed to specify the stress-strain relationship, and fatigue crack growth tests were performed to estimate the extent of crack growth during transient loads. The materials used in these tests were similarly heat treated to the actual PWR penetration tubes of the Ulchin unit in Korea. The microstructure will be reviewed to compare it with the metallographic characteristics of Alloy 600 in the foreign power plant's penetration materials. Data for PWSCC crack growth were surveyed and accumulated in a data base of Alloy 600 material properties. The purpose of this study is to provide data that similarly represent the properties of PWR power plants in Korea. The data will be used to analyze the stress distribution around the penetration tubes. The PWSCC data will be used for integrity evaluation of crack growth rate for the penetration tubes.

Material Tests

Alloy 600 materials that satisfy the specification of ASME-SB-168 were used in these tests. The Alloy 600 materials were heat treated to simulate and investigate the carbide distribution of the actual materials used in head penetration tubes in Ulchin units. Solution treatments of the as-received Alloy 600 for tensile and fatigue crack growth tests were performed at 1150°C for 30 min. Then the alloys were water quenched. After water quenching, the specimens were annealed to simulate the condition of heat treatments of head penetration materials of Ulchin units.

Tensile tests were performed on the as-received and heat-treated materials at room temperature and 300°C in air to investigate the stress-strain characteristics of the materials. Tensile strength decreases as the test temperature increases. The tensile strengths of head penetration tubes that were heat treated similarly to Ulchin units were relatively lower than those of as-received materials. One of the reasons for the decrease in tensile properties of heat treated Alloy 600 is grain growth during heat treatments. So, the effects of grain size should be accounted for when determining the stress-strain characteristics.

Fatigue crack growth tests were carried out on the as-received and heat-treated materials in air to assess the effects on crack growth during transient loads. The specimens for these tests were prepared according to ASTM E 647. The width of specimens is 50 mm and the thickness is 6 mm. The fatigue crack growth tests were carried out at a frequency of 10 Hz and at a load ratio of $R = 0.2$. The test results show that the number of total cycles for the crack in the heat-treated specimens to grow from the length of 13 mm to the length of 23 mm is larger than that for the as-received materials. The crack growth rate data that are shown in Fig. 2 are similar to the data of other researchers [5].

The PWSCC data for Alloy 600 will be collected from open literature and field data. The accumulated data will be arranged in a materials property data base for Alloy 600. We will rearrange the collected data into a data base system according to material properties, stress intensity factors, and environmental conditions. Finally, a formula for crack growth rate for PWSCC of Alloy 600, which accounts for various environments, materials properties, and stress intensity factors, will be obtained by analysis of the data base system.

The microstructures of the specimen that were heat-treated similarly to Ulchin units will be reviewed to compare them with the metallographic properties of foreign unit's penetration materials. The carbide distribution of the materials will be carefully studied to estimate the susceptibility of the materials of the Ulchin units' penetration tubes to PWSCC.

5. Inspection Plan

The inspection program lasting from December 1995 and to April 1996 (shown in Table 1) was planned to determine the existence, shape, and size of cracks in head penetration tubes of PWRs in Korea. The target reactors of inspection were Ulchin Units 1 and 2, which were supplied by Framatome and the oldest one in Korea, Kori Unit 1, supplied by Westinghouse. The number of head penetration tubes is 57 for Ulchin units and 40 for Kori Unit 1. This first inspection program will cover all penetrations at the Kori and Ulchin units.

Eddy-current testing (ECT) inspection will be used to investigate the existence of cracks in penetration tubes. Ultrasonic testing (UT) reinspection for crack sizing will be applied if any cracks exist in the tubes. The information about the shape and size of cracks in penetration tubes from UT inspection will be provided as input data to the final integrity evaluation program.

Further inspection plans will be determined based on the results of the inspections of the three power plants in Korea.

6. Future Work

The final integrity evaluation of the penetration tubes will be carried out in parallel with the inspection plan of penetration tubes for power plants in Korea lasting from December 1995 to April 1996. A further R&D program will follow the present study.

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Table 1. RVHP Inspection Program

Plant	Type & Capacity (MWe)	Commercial Operation Date	Head Temperature (° F)	Total No. of Penetrations	Material of Penetrations	Expected Inspection Date
Kori-1	PWR 587	Apr. 29, 1978	554 ~ 607	40	Inconel 600	Jan., 1996
Ulchin-1	PWR 950	Sep. 10, 1988	548 ~ 613	57	"	Apr., 1996
Ulchin-2	PWR 950	Sep. 30, 1989	548 ~ 613	57	"	Dec., 1995

- KFPCO's Activities Coordination Not Fixed.

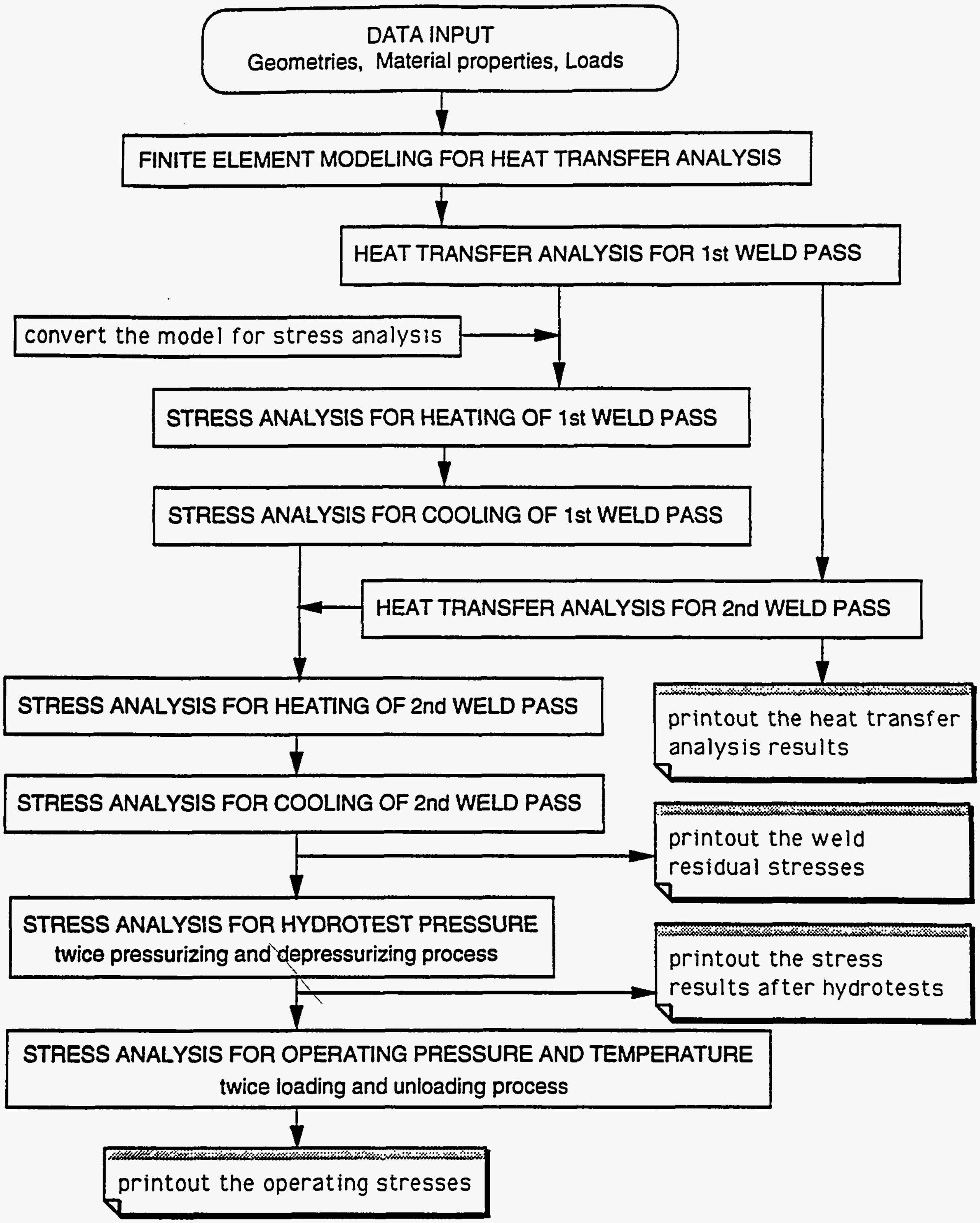

Figure 1 Flow Chart for Stress Analysis Procedure

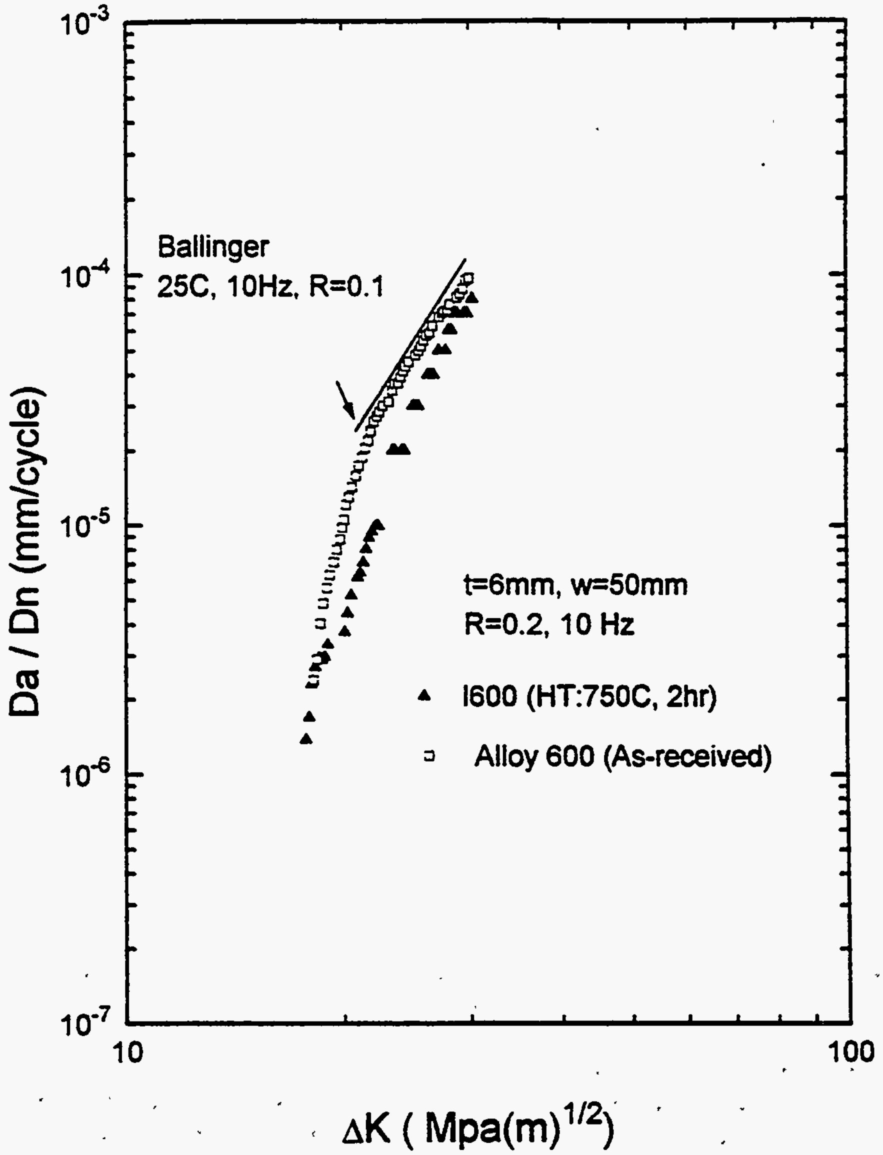

Fig.2 Fatigue Cracking Properties of Inconel 600 Alloys

WAPD-T-3055

Stress Corrosion Cracking Behavior of Alloy 600 in High Temperature Water

G. L. Webb and M. G. Burke

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Abstract

The stress corrosion cracking (SCC) susceptibility of Alloy 600 in deaerated water at 360°C, as measured with statically-loaded U-bend specimens, is dependent upon microstructure and whether the material was cold-worked and annealed (CWA) or hot-worked and annealed (HWA). All cracking was intergranular, and materials lacking grain boundary carbides were most susceptible to SCC initiation. CWA tubing materials are more susceptible to SCC initiation than HWA ring-rolled forging materials with similar microstructures, as determined by light optical metallography (LOM). In CWA tubing materials one crack dominated and grew to a large size that was observable by visual inspection. HWA materials with a low hot-working finishing temperature (below 925°C) and final anneals at temperatures ranging from 1010°C to 1065°C developed both large cracks, similar to those found in CWA materials, and also small intergranular *microcracks*, which are detectable only by destructive metallographic examination. HWA materials with a high hot-working finishing temperature (above 980°C) and a high-temperature final anneal (above 1040°C), with grain boundaries that are fully decorated, developed only microcracks, which were observed in all specimens examined. These materials developed no large, visually detectable cracks, even after more than 300 weeks exposure. A low-temperature thermal treatment (610°C for 7h), which reduces or eliminates SCC in Alloy 600, did not eliminate microcrack formation in the high temperature processed HWA materials. Detailed microstructural characterization using conventional metallographic and analytical electron microscopy (AEM) techniques was performed on selected materials to identify the factors responsible for the observed differences in cracking behavior. The major difference between the high-temperature HWA material and the low-temperature HWA and CWA materials was that the high temperature processing and final annealing treatment produced predominantly "semi-continuous" dendritic M_7C_3 carbides along grain boundaries with a minimal amount of intragranular carbides. Lower temperature processing produced significant proportions of intragranular M_7C_3 carbides, with less intergranular M_7C_3.

Introduction

Alloy 600 is susceptible to SCC in deaerated aqueous environments, most noticeably at temperatures above 316°C. SCC of Alloy 600 has been studied extensively since the observation of cracking in laboratory tests by Coriou[1], and there is a vast amount of literature on the effect of environment, material condition, temperature, and stress on SCC initiation[1-4]. Previous SCC initiation test data[5] from two very susceptible heats of tubing showed a temperature dependence corresponding to an activation energy of 50.6 kcal/mole, and a stress dependence of $\sigma^{-5.7}$. Previous studies showed that SCC susceptibility in Alloy 600 is related to the intergranular precipitation of Cr-rich carbides[2-4]. The formation of a "continuous" intergranular carbide network is associated with good SCC performance in a deaerated water environment, whereas materials with "clean" grain boundaries, or those which have few carbides, are more susceptible to SCC. The differentiation between "good" and "poor" microstructures does not depend on the extent or presence of a Cr-depleted zone near the intergranular carbides, particularly as the more resistant microstructures generally exhibit more pronounced Cr-depletion[6]. Furthermore, materials that receive a low-temperature thermal treatment at 610°C, which promotes grain boundary Cr-depletion, are more resistant to SCC, although very little change in microstructure may be observed by LOM examination.

Studies were conducted in high temperature deaerated water to determine if the working method, ring forging and annealing vs. cold drawing and annealing, had an effect on SCC initiation. At the completion of SCC testing of the ring forged materials, three ring forgings from two heats, with both high and low finishing temperatures, were selected for examination by AEM to attempt to determine the microstructural characteristics that could lead to the observed SCC differences between these two materials.

Test Materials

CWA tubing materials came from more than 40 heats of Alloy 600 tubing with an outer diameter (OD) of 12.7 or 14.3 mm and a minimum wall thickness of 1.07 mm or 1.35 mm, respectively. The tubing heats have carbon contents between 0.02% and 0.05%[1]. The tubing was produced by a normal tube reduction process. In this process, the final operations are tube sinking with more than 50% reduction, final annealing in hydrogen while traveling through the hot zone of a furnace where the material is at temperature for several minutes, rotary straightening, and belt polishing of the OD surface. Tubing from heat A, the most susceptible tubing heat tested[5], which has between 10% and 50% grain boundary decoration (GBD) with carbides as estimated from LOM examination, had been final-annealed at a temperature of 925°C, while all other heats of tubing were annealed at temperatures between 1010°C and 1093°C to achieve a microstructure with carbide-decorated grain boundaries. High final annealing temperatures, which put more carbides into solution, result in more GBD than low final annealing temperatures. The amount of GBD is very dependent on the cooling rate from the final anneal, and for thin-wall tubing it is easy to have a cooling rate that is fast enough to leave carbon in solution, resulting in a less desirable microstructure.

In addition to the tests conducted on CWA tubing, SCC tests were also conducted on specimens from 59 ring forgings from 10 heats of material. The carbon contents for all the ring forging heats varied from 0.03% to 0.06%. Ring forging is a three step process, starting with a 25 cm diameter billet reduced from a 50 cm diameter ingot. The billet is upset forged to produce a "pancake" from a temperature of 1150°C. A hole is punched in the pancake, which is then given an intermediate ring roll, starting at a temperature of 1150°C and finishing at about 925°C. Final ring rolling is done at a starting temperature of 1150°C and finishing at 925 - 1000°C. The finished ring forging has a 71 cm outside diameter and a 5 cm ring thickness. Final annealing occurs in the 1010-1065°C temperature range, with a cooling rate that is controlled at 220 C-deg per hour to a temperature of 650°C, followed by air cooling. The final annealing temperature is selected, based on trial anneals, to produce carbide-decorated grain boundaries while maintaining the room temperature yield strength above 204 MPa.

As-processed Alloy 600 has a tendency to develop "*bands*" that consist of fine-grained regions containing many intragranular carbides. "Banding" is discussed in more detail in the Results Section. Optimizing the thermo-mechanical processing of Alloy 600 ring forgings by employing high-temperature processing treatments (completing ring rolling above 980°C with 28% final rolling reduction) tends to minimize the banded microstructure, as opposed to the low-temperature processing (completing ring rolling at below 925°C). The effect of ring rolling finishing temperature on the SCC susceptibility was included as a variable in this program. The processing parameters for the three ring forgings selected for the AEM examination are listed in Table 1.

Experimental Procedures

SCC Testing

Specimen types included split tubing U-bends (STUBs) tested with the inside diameter (ID) surface in tension, double U-bends (DUBs), and a small number of single U-bends (SUBs). STUBs were used to test CWA tubing materials, and DUBs and SUBs were used to test HWA ring forging materials. The ID surface of tubing is as-annealed in hydrogen, with no additional machining operations, and the ID surface is < 1.6 μm (RMS) following tubing manufacture. Double U-bend and single U-bend specimens have a machined surface with a finish of < 3 μm (RMS).

SCC tests were conducted in a stainless steel autoclave at a temperature of 360°C. The SCC test environment was flowing (1-2 cc/s) deaerated pressurized water ($\approx$10 ppb dissolved O_2) with a pH of 10.1-10.3 at room temperature and a hydrogen concentration of 10-60 cc H_2/kg H_2O at a pressure of $\approx$ 20 MPa. Specimens were removed from the autoclave periodically for visual inspections at a magnification of 30X, and uncracked specimens were returned to the autoclave for further exposure.

Light Optical Metallography

Specimens for microstructural examination were prepared using conventional metallographic preparation techniques, followed by electrolytic etching witn 8:1 orthophosphoric acid to water. The degree of grain boundary carbide coverage was estimated from photomicrographs taken at 500X.

[1] All compositions are in weight percent.

Analytical Electron Microscopy

Three ring forgings were selected for detailed microstructural examination. Two ring forgings from the same heat (Heat B), but with high or low temperature processing, were selected to investigate the effect of processing treatment while eliminating heat-to-heat effects. A third ring forging (Heat C), with low temperature processing but a higher temperature final anneal that was close to the final annealing temperature of the high temperature processed ring forging, was selected to investigate the effect of final annealing treatment.

Both thin-foil and carbon extraction replica specimens were obtained from the two Heat B ring forgings. For comparison, AEM specimens were also prepared from the Heat C ring forging. Electron-transparent thin-foil specimens were prepared using conventional electropolishing techniques with an electrolyte of 20% $HClO_4$ - 80% CH_3OH at ≈-40°C. The extraction replica specimens were prepared using a solution of 5% Br in CH_3OH. Microstructural characterization was performed using a Philips CM12 and a Philips EM420 analytical electron microscope operated at 120kV and equipped with Link Analytical ultra-thin window energy dispersive x-ray spectrometers (EDS). Precipitates were identified by electron diffraction techniques and microchemical analysis using the scanning transmission electron microscopy (STEM) mode of the AEM. STEM-EDS x-ray maps were obtained from the carbon extraction replica specimens to complement the thin-foil analyses.

Results

SCC Test Results

The SCC data were separated into two major groups based on the working treatment that the base material received, either hot work or cold work. Within each of these groups, the data were grouped based on the amount of grain boundary carbide decoration. The groupings correspond to 10-50%, 50-75%, 75-90%, and 90-100% decoration. Cumulative failure[2] plots were made of the SCC data using the procedure described by Nelson[7]. Previous results[5] indicated that the failure times were distributed lognormally (i.e., the logarithms of the failure times are normally distributed). The data are characterized by two parameters, the median failure time, which is the intercept at 50% failures, and the slope of the curve, which is a graphical estimate of the standard deviation.

The raw SCC data separate into groups that are statistically different and consistent with the degree of grain boundary carbide decoration, indicating that the observed SCC behavior is dependent on the optical microstructure. Tubing and ring-forging materials have different yield strengths, resulting in different initial stresses being imposed by the constant deflection U-bend specimens. Because stress has such a strong effect on SCC initiation, it is necessary to account for this effect in the initiation data to determine the true effect of material condition on SCC initiation. Strain measurements on STUBs and DUBs indicate (true) plastic strains of 18% and 14%, respectively. Stresses were calculated for the various groups of materials tested using a power law mathematical approximation of a true stress-true strain curve developed from multiple heats of Alloy 600. In this approximation, the power law exponent, n, is a function of the yield strength, σ_y.

$$\sigma = 1535\ (\ 0.0185 + \epsilon_p\ )^n$$
$$n = 0.256 \ln\left(\frac{1535}{\sigma_y}\right)$$

where n = *strain hardening exponent,*
ϵ_p = *plastic (true) strain, and*
σ_y = *yield strength (MPa)*

The SCC test data, normalized to a common applied stress, are shown in Figure 1. All the SCC test data were normalized to a common test stress of 815 MPa, which is the stress determined for the STUB specimens taken from tubing from Heat A[5]. Estimates of the as-tested stresses on the various groups of specimens (i.e., ring forgings, tubing) were made based on the yield strength of the material and the measured specimen strain. A yield strength of 204 MPa, which is consistent with the yield strength values measured for these materials, was used for determining stresses in all materials that were annealed at > 1010°C. These estimated stresses are about 580 MPa for ring forging DUB specimens and 680 MPa for STUB specimens. The failure times were adjusted by using the previously determined stress exponent for SCC initiation of -5.7. For example, the failure times for ring forging DUB specimens were reduced by a factor of $(815/580)^{5.7} \approx 7$. No failures were observed in tubing with 90-100% grain boundary decoration, although tests were

[2] Visual observation of SCC initiation at 30X magnification.

discontinued after a short time (42 weeks), or in ring forging material with either 75-90% or > 90% grain boundary decoration after 300 weeks. These data are indicated by arrows in the figure.

The results plotted in Figure 1 show that there is a difference in SCC susceptibility between CWA and HWA materials, even after normalizing for the differences in stresses between the materials and specimen types. For a given microstructure and stress, HWA materials are more resistant to SCC initiation than CWA materials.

Destructive Examination Results

Destructive examinations (DE) were undertaken on some of the DUB specimens from ring forgings to determine if small cracks were occurring that were not visible during the normal visual examinations. Based on extensive previous experience with testing of STUB specimens of tubing, no small cracks were anticipated. Specimens were selected at random from those remaining in test at the completion of 255, 273, and 296 weeks. The specimens were sectioned longitudinally and mounted so that both an edge and centerline could be examined in the as-polished condition at a magnification of 200X or greater. When many small cracks of a uniform size were found, especially in the high temperature processed ring forging materials, they were called microcracks to differentiate them from the large cracks detectable by 30X visual examination. A typical microcrack is shown in Figure 2.

The DE results are shown in Table 2. Forty-one ring forging specimens with low-temperature processing were randomly selected for DE, and a majority (31) did not have microcracks. Thirteen ring forging specimens with high-temperature processing were randomly selected for DE, and all had at least one microcrack, although a scanning electron microscope (SEM) examination was required to find cracks on one specimen. In contrast, no microcracks were found during an SEM examination of an uncracked low-temperature processed specimen. The final group of specimens selected for DE also included specimens that had received a low-temperature thermal treatment at 610°C for 7h, and three of these six specimens had microcracks.

Light Optical Microstructural Characterization Results

The grain size of the high-temperature processed forging material sample, final-annealed at 1060°C for 2h, was approximately 200 μm, as compared to the ≈ 100 μm grain size of the low-temperature processed forging Heat B sample which was final-annealed for 1h at 1024°C. The low-temperature processed forging material from Heat C, which received a final anneal at 1050°C for 2h, also had an average grain size of ≈ 100 μm. The yield strengths for these three material conditions, although not strictly related to the grain size and microstructure, were significantly different and are listed in Table 1. The general microstructure of these materials is presented in Figure 3.

Examination of the metallographic specimens from the low-temperature processed materials revealed the presence of a duplex grain structure, with many intragranular carbides (most likely MC-type inclusions) associated with the fine-grained regions or bands. Ring forgings frequently develop patterns of banding, obvious in metallographic mounts and low magnification photomicrographs, Figure 4, with dark-appearing regions containing extensive, coarse, intragranular carbides and little intergranular precipitation, alternating with light-appearing regions with no intragranular precipitation and significant amounts of intergranular precipitation.

Analytical Electron Microscope Characterization Results

The results of the AEM evaluation are summarized in Table 3, while the details are included in the following paragraphs.

<u>Heat B - High Temperature Processing</u>

The high-temperature processed Heat B ring forging was final-annealed at 1060°C for 2h. The LOM metallographic evaluation showed nearly complete grain boundary carbide decoration (>90% GBD), with a very small amount of intragranular carbide precipitation. AEM characterization of this material revealed extensive intergranular Cr-rich M_7C_3 carbides, Figure 5, with about 85% GBD. These carbides generally appeared in thin-foil specimens as "semi-continuous" precipitates, ≈ 0.2 μm in thickness and varying in length up to ≈ 7 μm. Several fine, discrete $M_{23}C_6$ carbides were also observed at some boundaries, Figure 6. More detailed examination revealed that the morphology of the intergranular M_7C_3 was a complex dendritic type, Figure 7. These secondary electron images were obtained from an inclined grain boundary region after a light electrolytic etch that show the 3-dimensional morphology of the intergranular M_7C_3. Although the predominant carbide configuration was intergranular, isolated regions contained a high proportion of intragranular M_7C_3 carbides that may be due to incomplete removal of carbide banding. An example of such a region with both the intergranular and intragranular carbides is shown in Figure 8.

Heat B - Low Temperature Processing

The low-temperature processed Heat B ring forging was final-annealed at 1024°C for 1h. The LOM metallographic evaluation indicated about 50-75% GBD, while the AEM analysis indicated about 50-60% GBD. Intergranular M_7C_3 and $M_{23}C_6$ were observed at many high-angle grain boundaries, Figure 9. The coarse intergranular M_7C_3 had a dendritic morphology similar to that observed in the 1060°C specimens from Heat C. The $M_{23}C_6$ were discrete intergranular carbides, approximately 0.1 μm in length.

STEM-EDS x-ray maps confirmed that the majority of the carbides are Cr-rich. However, a few Ni-rich precipitates, frequently associated with the Cr-rich carbides, were also detected. The x-ray maps, Figure 10, indicate that a central portion of the carbide is Ni-rich. Electron diffraction confirmed that these precipitates had a face centered cubic (fcc) structure, with a lattice parameter ≈3 times that of the matrix, and exhibited a cube-cube orientation relationship with the matrix. Such Ni-rich intergranular precipitates have been observed in other studies of NiCrFe alloys[9,10], and have been identified as $Ni_{23}(B,C)_6$. The electron diffraction and STEM-EDS data indicate that the fine precipitates in the ring-forged specimens are Ni-rich $M_{23}X_6$, where X represents B and C.

This material also has extensive intragranular M_7C_3 precipitation. Two carbide morphologies were observed for the intragranular precipitates: globular and facetted, Figure 11. Examination of the facetted carbides indicates that there are multiple habit planes with the matrix, and the crystallographic morphology suggests that this carbide formed upon cooling from the final anneal temperature. The globular M_7C_3 precipitates appear to have formed during prior thermo-mechanical processing, and were not fully dissolved during the final anneal. Both carbide morphologies are shown in the STEM micrographs of Figure 12.

Heat C - Low Temperature Processing

The low-temperature processed Heat C ring forging was final-annealed at 1050°C for 2h. The LOM evaluation indicated 50-75% GBD, similar to the Heat B forging annealed at 1024°C for 1h. The grain sizes of these two materials were also comparable (≈100 μm). The estimated grain boundary coverage from AEM examination is 50-60%. The AEM examination revealed a microstructure with extensive regions of intragranular M_7C_3 carbides, coarse globular-type M_7C_3 intergranular carbides, and fine intergranular $M_{23}C_6$ carbides. Examples of the various carbides are presented in Figure 13. Although this material exhibited a similar extent of inter-granular carbide precipitation to that observed in the Heat B ring forging final-annealed at 1024°C for 1h, no semi-continuous or dendritic-type M_7C_3 carbides were detected. The discrete globular M_7C_3 morphology may be related to the cooling rate from the forging and final annealing temperatures.

Discussion

The SCC susceptibility of Alloy 600 has been correlated with the optical microstructure as determined with the 8:1 ortho-phosphoric acid electrolytic etch, which primarily attacks chromium-containing precipitates. Materials that do not have a high proportion of intergranular chromium carbides are more susceptible to cracking than materials with highly-decorated grain boundaries[2,4]. The AEM examination indicated that the microstructures with the highest resistance to SCC initiation had a high amount of intergranular M_7C_3 precipitates, which precipitated during cooling from the high-temperature final anneal.

This study also found that the SCC susceptibility of Alloy 600 is dependent upon how the material was processed, whether by cold-work and annealing or hot-work and annealing, and upon the hot-working parameters, especially the working temperature. CWA tubing materials were found to be more susceptible to SCC initiation than HWA ring forging materials with similar LOM microstructures. Even when corrected for differences in the as-tested stress levels, CWA materials are more susceptible to SCC initiation, with the differences being statistically significant at the 95% confidence level. When corrected for strength differences, the HWA materials with 50-75% GBD have failure times that are a factor of 5 longer than the failure times for CWA tubing materials.

The difference is not believed to be due to differences in stress relaxation between the two specimens types, STUB vs. DUB or to residual stress differences. The STUB specimens have a higher measured strain than the DUB specimens, and at the same yield strength levels, the STUB specimens will have a higher stress. Stress relaxation would be expected to be higher in the more highly strained specimens, resulting in longer crack initiation times for the STUB specimens, but the difference in failure times is in the opposite direction. Stress relaxation would have to be about 34% higher in the lower stressed DUB specimen than in the STUB specimen to account for the difference in SCC initiation times. Residual stresses would not be responsible for the differences in SCC susceptibility. Although the machined surfaces of double U-bend specimens will have higher residual stresses than the final annealed surfaces of tubing specimens, the high strains imposed during bending, 14-18%, will minimize this effect because the strain associated with a 300 MPa residual stress is only 0.15%.

There are two significant differences between the processing of hot-worked forging materials and cold worked tubing materials, other than the obvious differences in working temperature. First, the microstructural results may be misleading. The tubing materials have very thin walls, less than 1.2 mm, and cooling rates that are too fast to allow for carbide precipitation are easily achieved. The orthophosphoric acid etch attacks both the carbides and the matrix near the carbides, so the presence of unprecipitated carbon at the grain boundary could also contribute to the grain boundary decoration in the LOM microstructure. Previous work has shown that the low temperature thermal treatment at 610°C is beneficial in improving the SCC resistance of Alloy 600 tubing, and has the greatest effect in materials with the poorest microstructures, with less visible grain boundary carbide precipitation[5]. Because of the low temperature involved in this thermal treatment, no significant amount of diffusion occurs, so only the carbon already present at the grain boundary is given a chance to precipitate during this thermal treatment. This results in an improvment in the SCC resistance, sometimes without a significant change in the LOM microstructure. In the forging rings, which have a heavier section size and a slower cooling rate, carbides have a greater opportunity to precipitate during cooling from the final anneal, and the bulk of the grain boundary decoration comes from actual precipitates rather than from supersaturated carbon in solution at the grain boundary.

The second major difference between tubing and ring forgings is in the final annealing atmosphere. Tubing is final annealed in a dry hydrogen atmosphere, and the final tested surface is the as-hydrogen-annealed surface, whereas forgings are final annealed in an inert atmosphere, but the tested surface is machined. It can be inferred from two other phenomena, which show that hydrogen has a detrimental effect on SCC, that final hydrogen annealing may be detrimental to SCC resistance. First, Airey[11] indicated that SCC initiation times were much longer for specimens tested in an autoclave containing pure water without a hydrogen overpressure than with a hydrogen overpressure. The presence of an oxide film on the surface also affected the SCC resistance by interfering with the hydrogen transport to the metal. Second, pickling, which introduces hydrogen into the metal, has a detrimental effect on the SCC resistance of Alloy 600[2]. The difference in the susceptibility to SCC initiation between the CWA tubing and HWA ring forgings is attributed mainly to the hydrogen final-annealing atmosphere used in tubing manufacture.

In addition to the differences between CWA and HWA materials in the initiation of visually detectable SCC, a phenomenon called ***microcracking*** was observed in HWA materials. Microcracks are shallow intergranular cracks, typically on the order 125 μm in depth. HWA materials that were processed at temperatures below 925°C developed both large cracks and microcracks, but HWA materials from 3 heats (35 forgings), which were processed at temperatures above 980°C to eliminate banding, developed only microcracks. No microcracking has been observed in specimens from CWA tubing materials, including specimens that were destructively examined following completion of long term testing (as much as 500 weeks at 288°C) of tubing materials that were previously reported[5].

The processing of ring forgings was investigated to find a way to eliminate banding. When the major part of the hot-working operation was completed above the carbide solvus temperature ($\approx$ 1030°C for 0.040% carbon[8]) and when the annealing treatments were carried out at temperatures about 15 - 40 C-deg above the carbide solvus, the banding was minimized or eliminated. On the other hand, when hot-working and annealing took place considerably below the carbide solvus temperature, such as below 925°C, or when the annealed material is slowly cooled through the carbide precipitation region, the banding occurred or reoccurred depending on the material's tendency toward banding.

Banding is a concern because the microstructure within the dark bands contains little of the intergranular Cr-rich carbide precipitates necessary for SCC resistance. Elimination of banding is desirable for reducing the susceptibility to SCC initiation since it has provided an initiation site for SCC. Once bands form, they are persistent, remaining even after high-temperature reannealing. Electron microprobe examination indicated that these dark bands are regions with enrichments of Cr, Mn, and Ti, and depletion of Fe, which occur during solidification. The preferential intragranular carbide precipitation within these bands is consistent with the affinity of C for Cr and Ti. The persistence of the bands, despite attempts at eliminating them by reannealing, is due to the sluggish diffusion rates of Cr, Mn, and Ti. High-temperature annealing, above the carbide solvus, will homogenize the interstitial carbon in solution, but not the substitutional elements, and slow cooling through the carbide precipitation range allows the bands to re-form during cooling. Later studies, which focused on using high-temperature processing to encourage homogenization of the Cr, Mn, and Ti-enriched regions, resulted in microstructures with significantly reduced banding. This high-temperature processing was only utilized on ring forgings from 3 heats of material.

As anticipated, based on the LOM microstructures with fully decorated grain boundaries, the specimens from these high-temperature processed ring forgings did not develop visually detectable cracks during testing. A destructive metallographic examination of selected specimens indicated very shallow intergranular cracking, not detectable by 30X visual inspections. These materials developed either single or multiple (up to 10 have been observed) microcracks.

The occurrence of microcracking is not related to only the temperature of the final anneal; low-temperature processed HWA materials with final anneals of 1065°C did not develop microcracks to the same extent as the high-temperature processed HWA materials with lower final annealing temperatures. Microcracking has thus far been the only form of SCC observed in high-temperature processed and high-temperature annealed ring forgings.

Based on pre-test examinations of several specimens, these microcracks do not form during bending, but require high-temperature exposure to the environment to initiate. The exposure time required for the initiation has not been determined, but they were only initially looked for in specimens with exposures longer than 250 weeks. Likewise, the growth rate of microcracks has not been determined, but it appears to be slow, with no obvious changes in number or depth of cracks in specimens examined at 250 or 300 weeks. A low-temperature thermal treatment (610°C for 7h) has been found to greatly reduce or eliminate SCC in both CWA and HWA with low hot-working finishing temperatures. However, microcracks were observed in three of six low-temperature thermally-treated specimens from HWA materials with a high hot-working finishing temperature.

It is not clear whether microcracking is the same as SCC, or a different phenomenon. Microcracks are intergranular, occur after exposure to high temperature water, and occur in a material that has long been known to be susceptible to SCC. If microcracking is the same as SCC, its presence depends on a slow crack growth rate. Since the formation of a large crack will relieve stresses at the tensile surface of the specimen, so that other cracks will not be likely to form, multiple cracks may have a chance to form in materials that do not quickly develop a large crack. Microcrack formation may be a different phenomenon than SCC. The conventional wisdom with SCC is that high temperature annealing and slow cooling, which produce grain boundaries with abundant chromium carbide precipitation, result in a material that is more resistant to SCC formation. This is clearly not the case for microcracks, for they formed in 100% of the high-temperature processed HWA materials, which have the most completely decorated grain boundaries. There may be a relationship between the Cr-carbide morphology and the susceptibility to microcracking. In the high-temperature processed ring forgings, most susceptible to microcracking, the predominant Cr-carbide is M_7C_3 with a dendritic morphology. In the low-temperature processed ring forgings, which are less susceptible to microcracking, both globular and dendritic M_7C_3 were observed.

The mechanism for SCC formation in Alloy 600 has not been unequivocally established, and at least 5 different mechanisms under consideration (dissolution-oxidation, hydrogen embrittlement, corrosion-enhanced plasticity, creep, and internal oxidation). Therefore, with microcracks observed only in statically loaded specimens from hot-worked and annealed materials after long term exposures to a single high temperature water environment, it is difficult to ascertain whether the same mechanism is responsible for SCC and microcracking.

Microcrack formation is a problem, even though it has not been detected under normal SCC testing procedures where visual examinations at 30X are utilized to detect cracking. Under conditions involving cyclic loading, these microcracks may result in a degradation in the environmentally assisted cracking behavior because of the presence of a sharp crack tip. The strain levels required to induce microcracking have not been determined, and thus far they have been observed in DUB with 14% strain. If microcracks develop in Alloy 600 at strain levels of $< 5\%$, which can be induced in wrought Alloy 600 as a result of weld shrinkage, the results from statically-loaded specimens would not conservatively predict the behavior of actively-loaded structures.

The AEM microstructural characterization confirms that the presence of intergranular carbides enhances the resistance of the material to the formation of large SCC cracks. In the material most resistant to the initiation of large cracks, high-temperature processed Heat B, these carbides are Cr-rich M_7C_3 with a dendritic morphology. These carbides were also observed, although to a lesser degree, in the low temperature processed Heat B ring forging. As indicated above, materials with this dendritic carbide morphology may have a greater tendency to microcrack formation.

Based on the microstructural examination of the 1024°C and 1060°C final anneal forgings, a high-temperature final anneal is essential for carbide dissolution. The two intragranular carbide morphologies suggest that other factors, such as banding, may be responsible for the increased proportion of intragranular carbide precipitation in the materials that were annealed at 1024°C.

The estimates of the degree of grain boundary carbide decoration are lower based on AEM examination than the estimates from the LOM because the orthophosphoric acid etchant attacks both the carbides and matrix near the carbides, thus yielding an overestimate of the proportion of grain boundary carbides. The AEM observations also serve to demonstrate that materials which appear to be very similar by optical metallographic analysis, such as the materials final-annealed at 1024°C and 1050°C, can exhibit pronounced microstructural differences in terms of extent and nature of carbide precipitation.

Conclusions

Material with a microstructure consisting of carbide-decorated grain boundaries is resistant to SCC initiation in high-temperature deaerated water. This microstructure may be developed with a high-temperature final anneal which can dissolve the pre-existing carbides and promote extensive intergranular M_7C_3 precipitation upon cooling. The difference in susceptibilities of CWA and HWA Alloy 600 materials with similar LOM microstructures, however, is an indication that more sophisticated techniques are necessary to provide a true indication of SCC susceptibility.

CWA tubing materials are more susceptible to SCC initiation than HWA materials with microstructures that have similar degrees of grain boundary carbide decoration based on LOM examination. This difference remains even when yield strength differences between the materials are taken into account. The difference in susceptibilities is attributed to the hydrogen final-annealing atmosphere used in tubing manufacture, and to the effect of the thin wall of the tubing on carbide precipitation during cooling.

Although resistant to general SCC, HWA materials with high-temperature processing, which was used to eliminate the banded microstructure, have the greatest tendency to develop microcracks. The shallow microcracks that have been observed in these materials are associated with "good" microstructure, i.e., extensive precipitation of semi-continuous intergranular Cr-rich M_7C_3 carbides. In the high-temperature processed ring forgings, these carbides have a dendritic morphology. Despite the fact that these microcracks have not developed into visually detectable cracks in statically-loaded specimens, their presence may lead to a degradation in SCC resistance if they also occur in actively-loaded applications.

The estimates of the degree of grain boundary carbide decoration are lower based on AEM examination than the estimates from the LOM because the orthophosphoric acid etchant attacks both the carbides and matrix near the carbides, thus yielding an overestimate of the proportion of grain boundary carbides.

The AEM microstructural characterization confirms that the presence of intergranular carbides enhances the resistance of the material to the formation of large SCC cracks. In the material most resistant to the initiation of large cracks, high-temperature processed Heat B, these carbides are Cr-rich M_7C_3 with a dendritic morphology. These carbides were also observed, although to a lesser degree, in the low temperature processed Heat B ring forging. Although no correlation has been established, materials with this dendritic carbide morphology may have a greater tendency to the formation of microcracks. Based on the microstructural examination of the ring forgings with either a 1024°C or 1060°C final anneal, a high-temperature final anneal is essential for carbide dissolution. The two intragranular carbide morphologies suggest that other factors, such as banding, may be responsible for the increased proportion of intragranular carbide precipitation in the materials that were annealed at 1024°C.

The AEM observations also serve to demonstrate that materials which appear to be very similar by optical metallographic analysis, such as the materials final-annealed at 1024°C and 1050°C, can exhibit pronounced microstructural differences in terms of extent and nature of carbide precipitation.

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Table 1: Summary of Processing, Grain Size, and Yield Strength for Materials Referenced				
Heat	Ring Rolling Finishing Temperature	Final Anneal	Grain Size	Yield Strength
Heat A	CWA Tubing	925°C	20 μm	340 MPa
Heat B	High Temperature	1060°C	200 μm	175 MPa
Heat B	Low Temperature	1024°C	100 μm	204 MPa
Heat C	Low Temperature	1050°C	100 μm	230 MPa

Table 2: Results of Destructive Examination of Double U-Bend Ring Forging Specimens		
Ring Forging Material Heat Treatment	No. Examined	No. with Microcracks
Low Temperature Processing	41	10
High Temperature Processing	13	12[†]
High Temperature Processing + LTTT[‡]	6	3

† A microcrack was observed in the "uncracked" specimen during SEM examination.
‡ Low-Temperature Thermal Treatment, 610°C for 7h

Table 3: Summary of AEM Results				
Heat	Hot Work Finishing Temperature	Grain Boundary Decoration, % Visual	Grain Boundary Decoration, % AEM	Comments
B	High Temp + High Temp Anneal	90-100	85	Intergranular, semi-continuous, dendritic Cr-rich M_7C_3 Some fine intergranular $M_{23}C_6$ Almost no intragranular precipitation
B	Low Temp + Mid Temp Anneal	50-75	50-60	Intergranular, coarse, dendritic Cr-rich M_7C_3 Some intergranular Ni-rich, discrete, $M_{23}C_6$ Extensive intragranular globular and facetted M_7C_3 (Globular precipitate from prior processing)
C	Low Temp + High Temp Anneal	50-75	50-60	Coarse intergranular globular M_7C_3 and fine intergranular $M_{23}C_6$ (No intergranular dendritic M_7C_3) Coarse intragranular M_7C_3

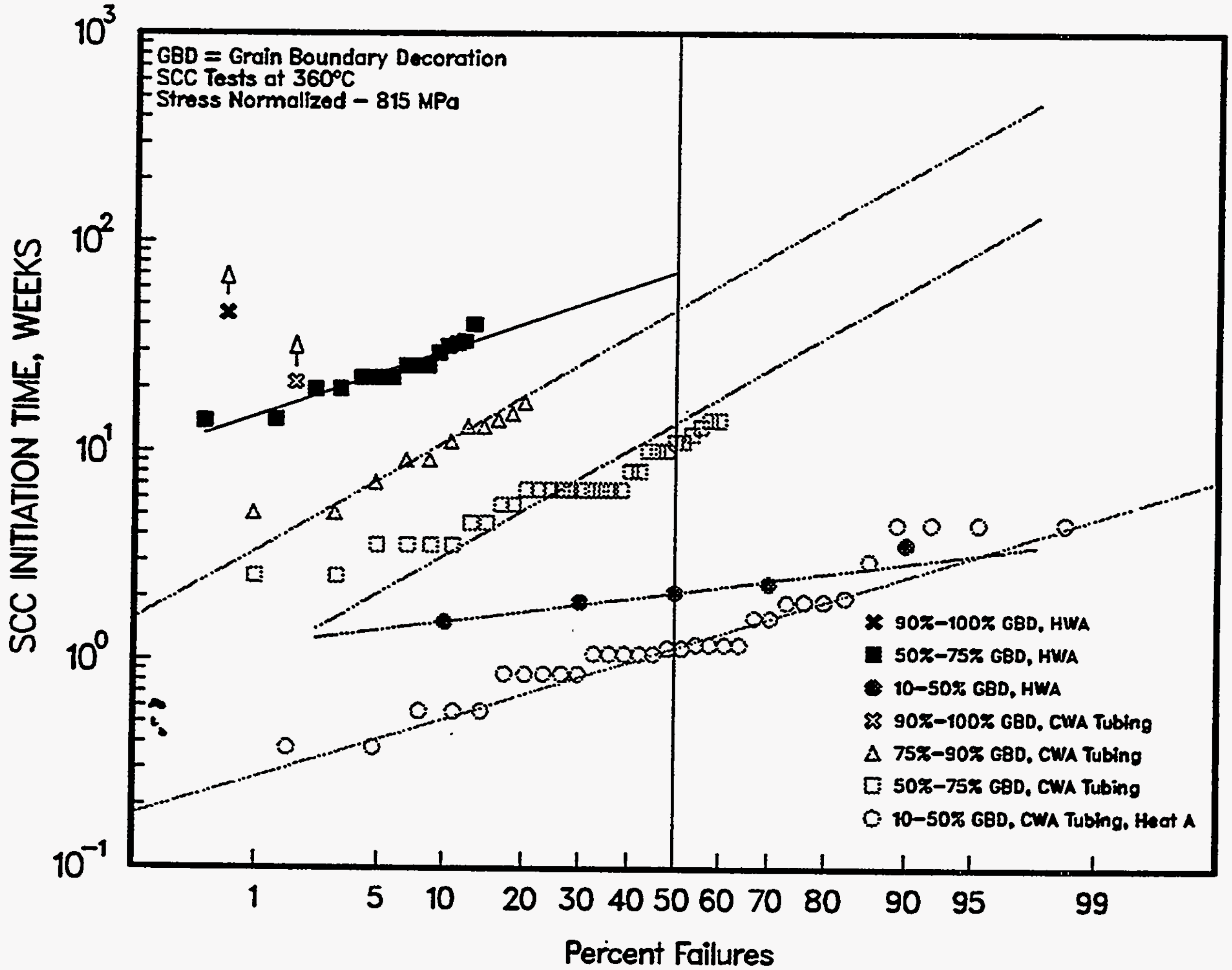
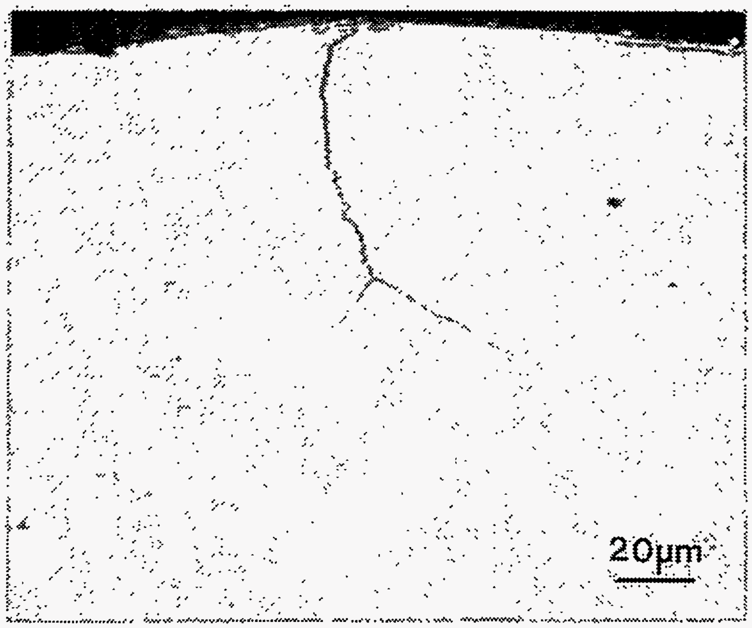


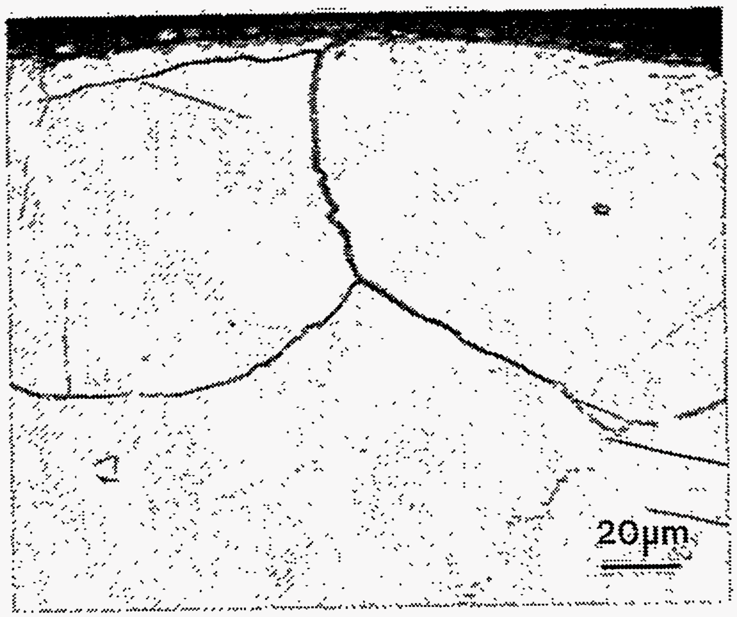
Figure 1: Cumulative Distribution Function failure plot of Alloy 600 SCC data, corrected for stress

Figure 2:

A microcrack in as-polished specimen.



The same microcrack following electrolytic etching in 8:1 orthophosphoric acid to show grain boundary locations.



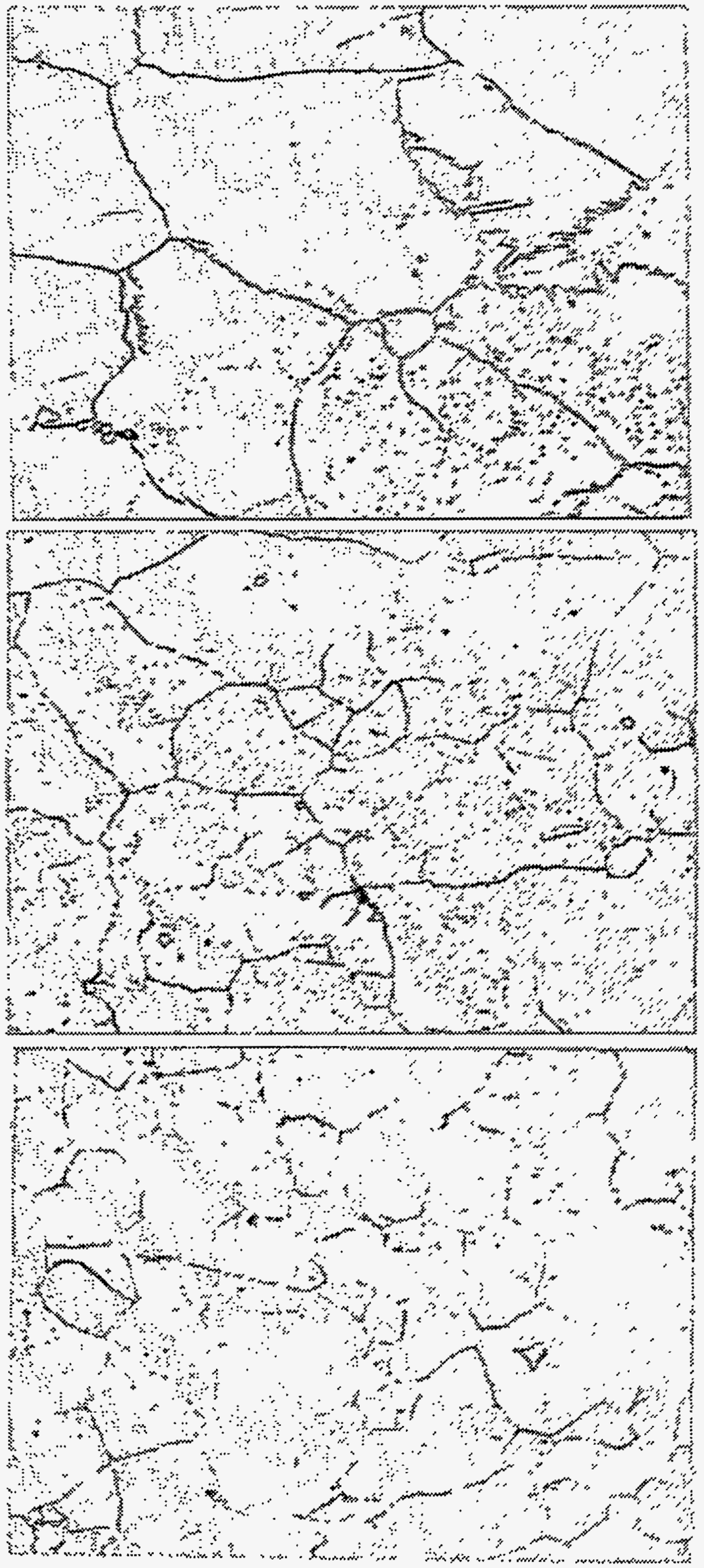


Figure 3:

General Microstructures representative alloy 600 materials electrolytically etched in 8:1 ortho-phosphoric acid to water.

(a) Heat B in high temperature processed condition.

(b) Heat B in conventional processed condition.

(c) Heat C in conventional processed condition.

Figure 4:

Example of banding in a ring rolled forging.

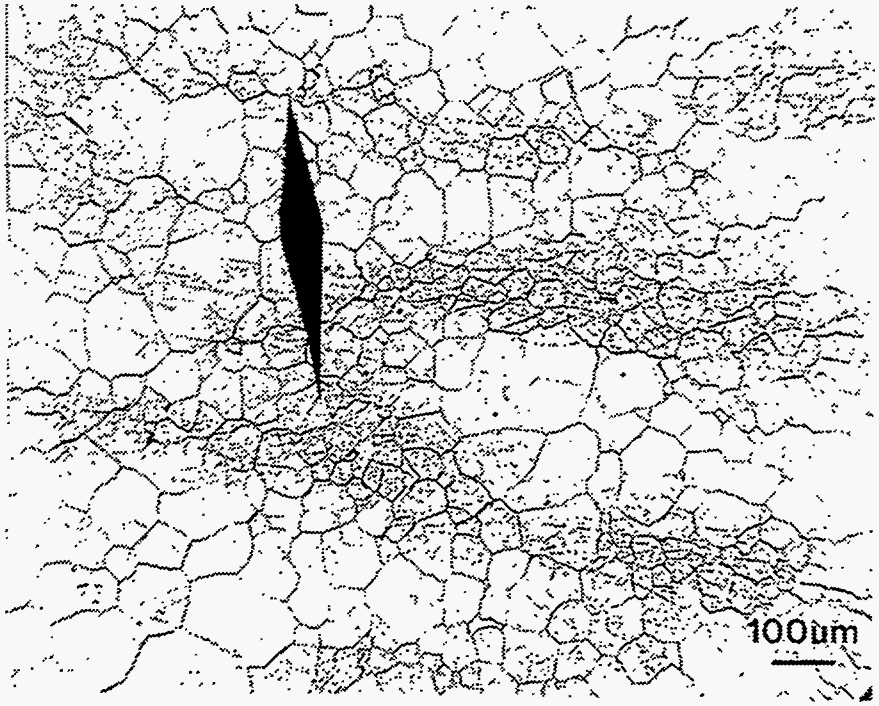


Figure 5:

Transmission electron micrograph of the predominant M_7C_3 carbides observed in the forged material final-annealed at 1060°C for 2 h.

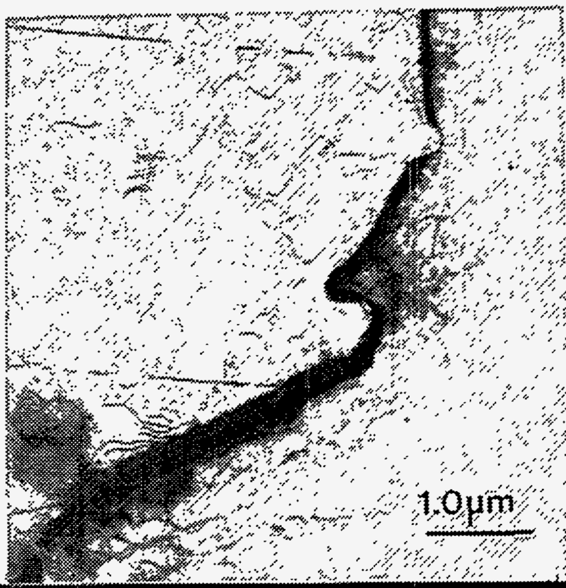
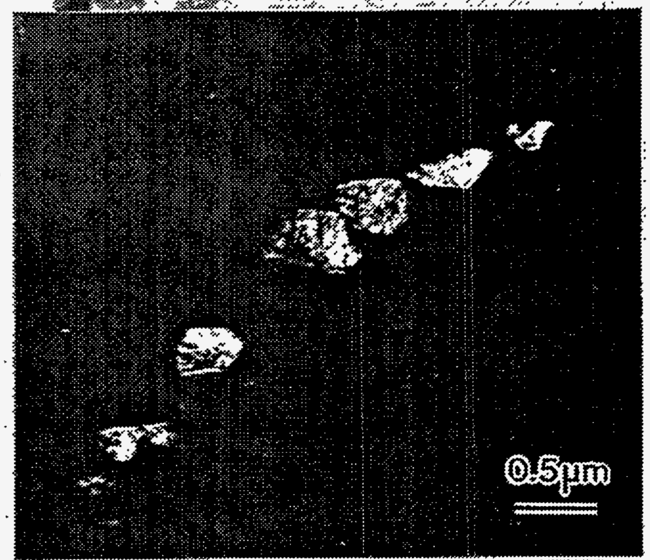
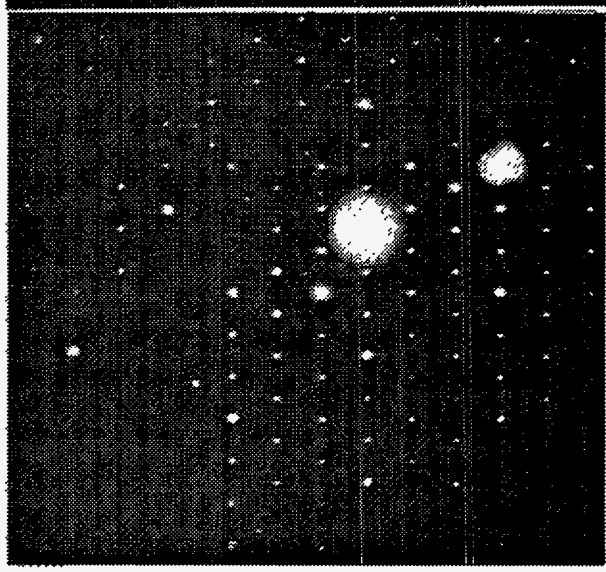


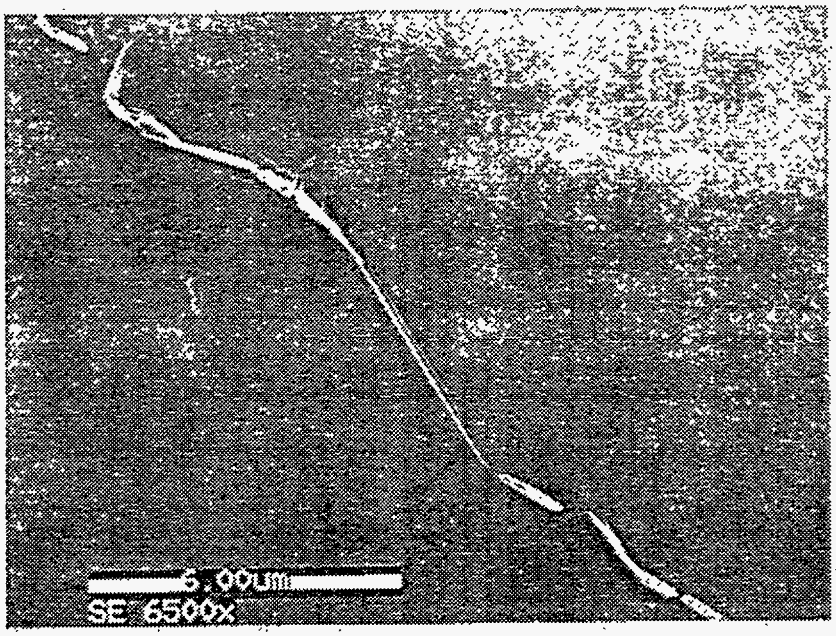
Figure 6:

(a) Dark-field transmission electron micrograph of fine $M_{23}C_6$ carbides observed at some grain boundaries in material final-annealed at 1060°C.

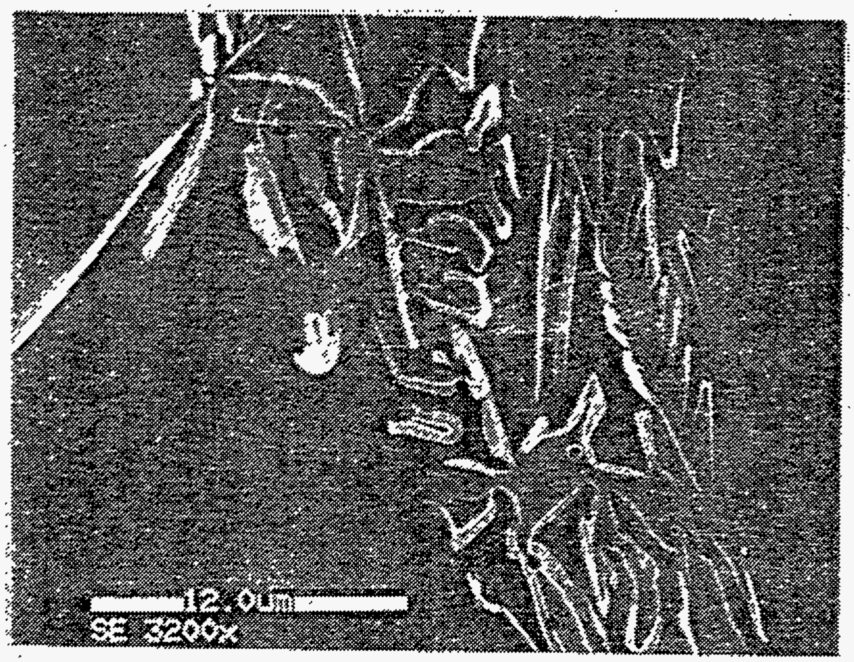


(b) Associated selected area diffraction pattern of fine $M_{23}C_6$ carbides observed at some grain boundaries in material final-annealed at 1060°C.



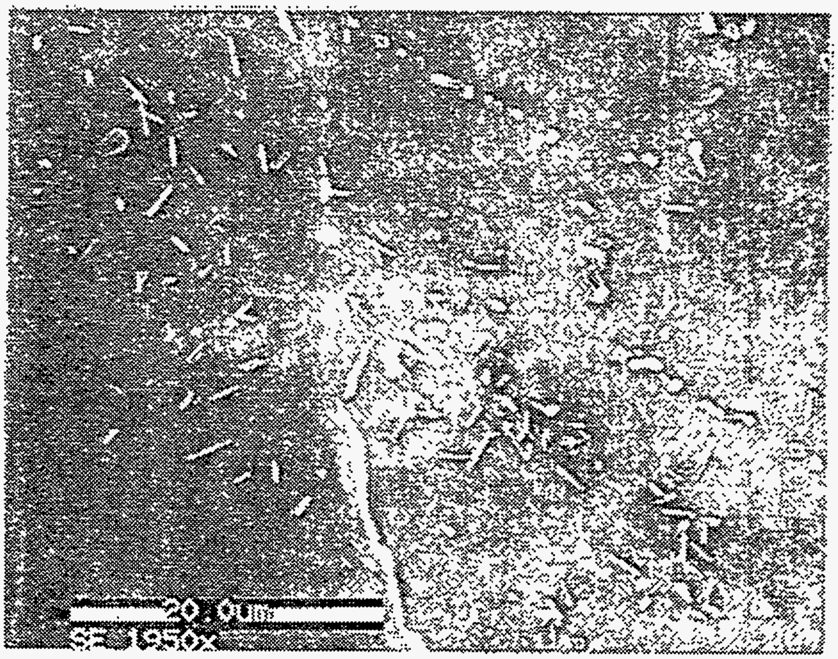


Secondary electron images of the high-temperature processed forging material (final anneal at 1060°C) showing (a) the "semi-continuous" morphology in the material final-annealed at 1060°C for 2 h.



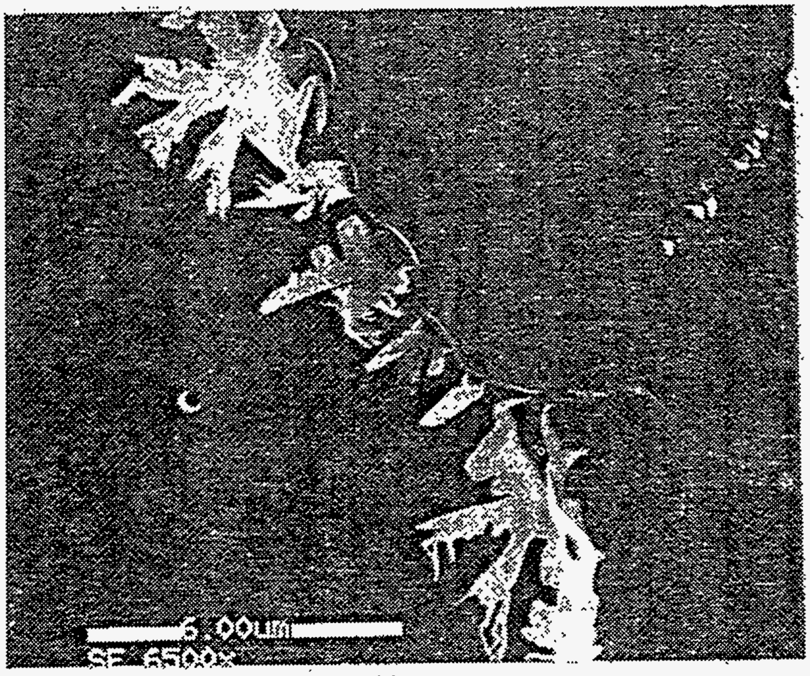
Inclined angle secondary electron images of the high-temperature processed forging material (final anneal at 1060°C) showing (b) the dendritic-type morphology of the M_7C_3 carbides observed in the material final-annealed at 1060°C for 2 h.

Figure 7

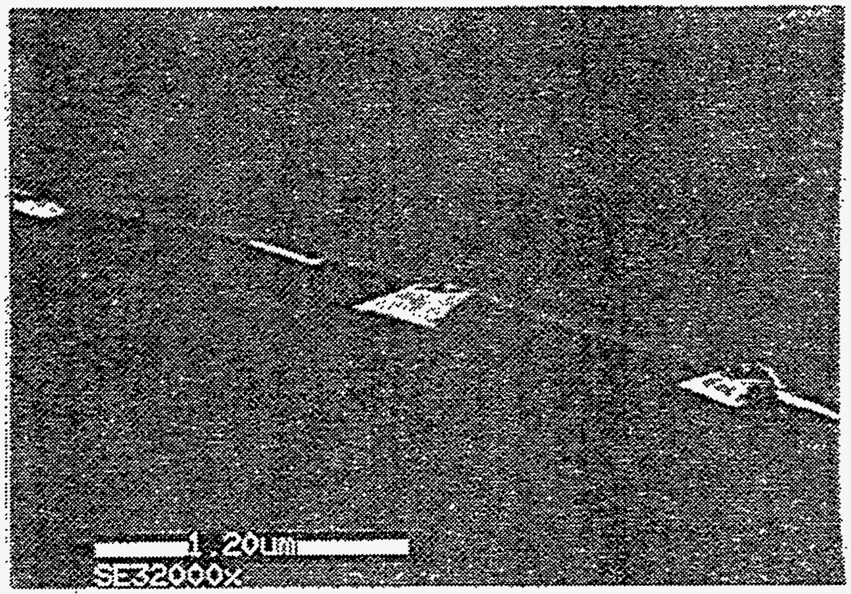


Secondary electron image showing one of the few regions containing both intergranular and intragranular carbides associated with residual carbide banding in material final-annealed at 1060°C for 2 h.

Figure 8



(a)



(b)

Secondary electron micrographs showing the presence of both (a) coarse, dendritic M_7C_3 carbides and (b) the fine discrete $M_{23}C_6$ carbides in specimens final-annealed at 1024°C for 1 h.

Figure 9

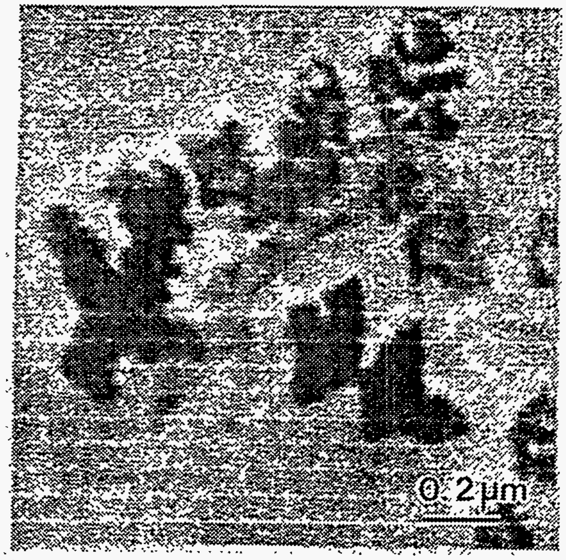
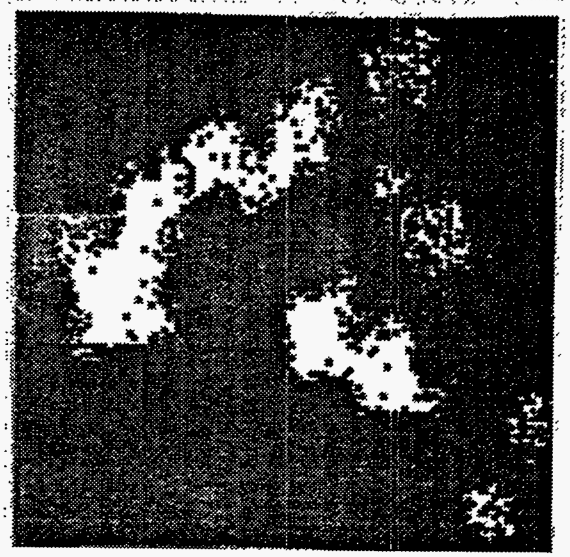
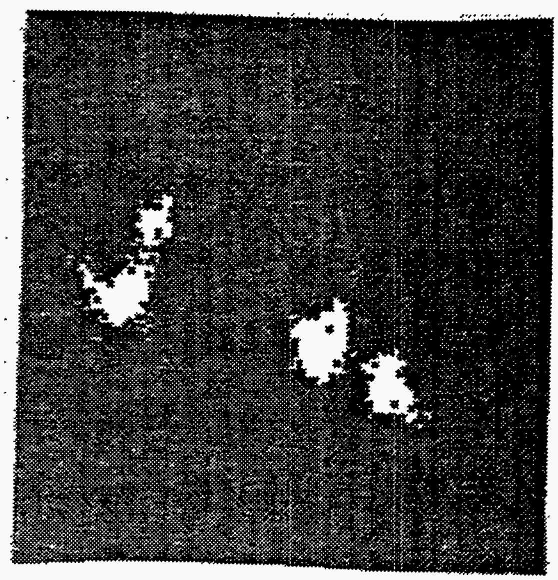


Figure 10: Digitized STEM image, and corresponding x-ray maps for Cr and Ni that show the association of the Ni-rich precipitates with the Cr-rich carbides in material final-annealed at 1024°C for 1 h.

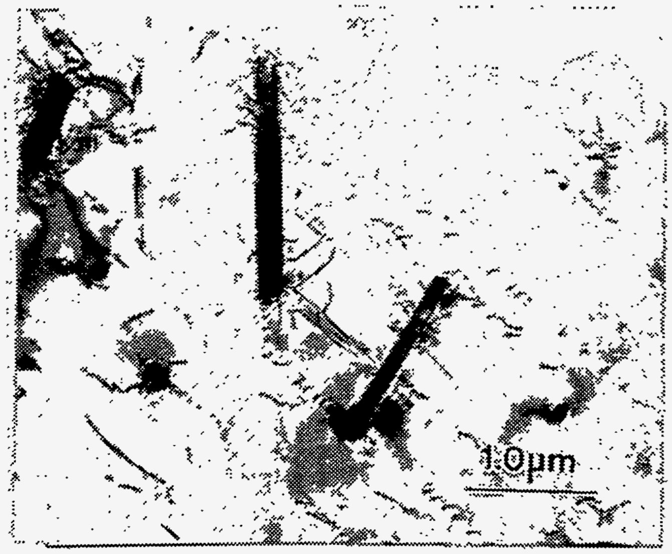


X-ray map for Cr that shows the association of the Ni-rich precipitates with the Cr-rich carbides in material final-annealed at 1024°C for 1 h.



X-ray map for Ni that shows the association of the Ni-rich precipitates with the Cr-rich carbides in material final-annealed at 1024°C for 1 h.

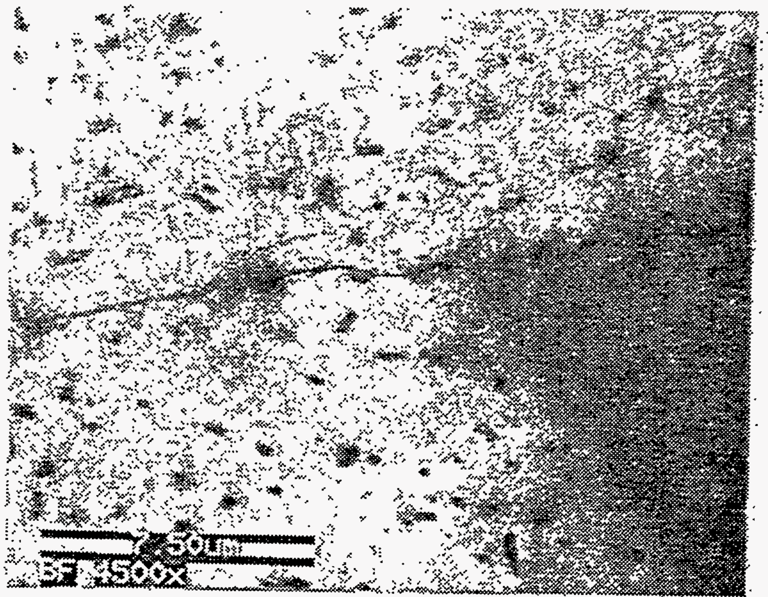
Globular carbide



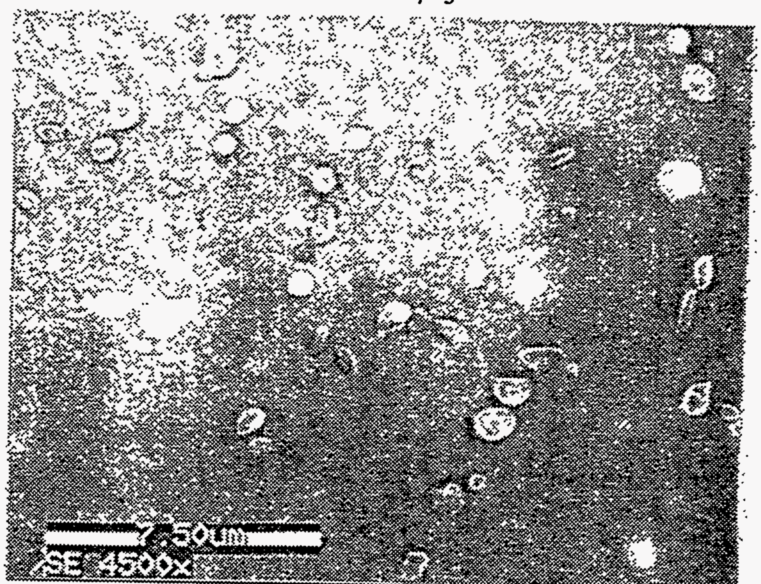
Faceted carbide

Transmission electron micrographs showing the two intragranular M_7C_3 carbide morphologies (globular and faceted) observed in the samples final-annealed at 1024°C for 1 h.

Figure 11



(a) STEM SEM images obtained from material final-annealed at 1024°C for 1 h. Note the high proportion of intragranular M_7C_3 carbides.



(b) SEM images obtained from material final-annealed at 1024°C for 1 h. Note the high proportion of intragranular M_7C_3 carbides.

Figure 12

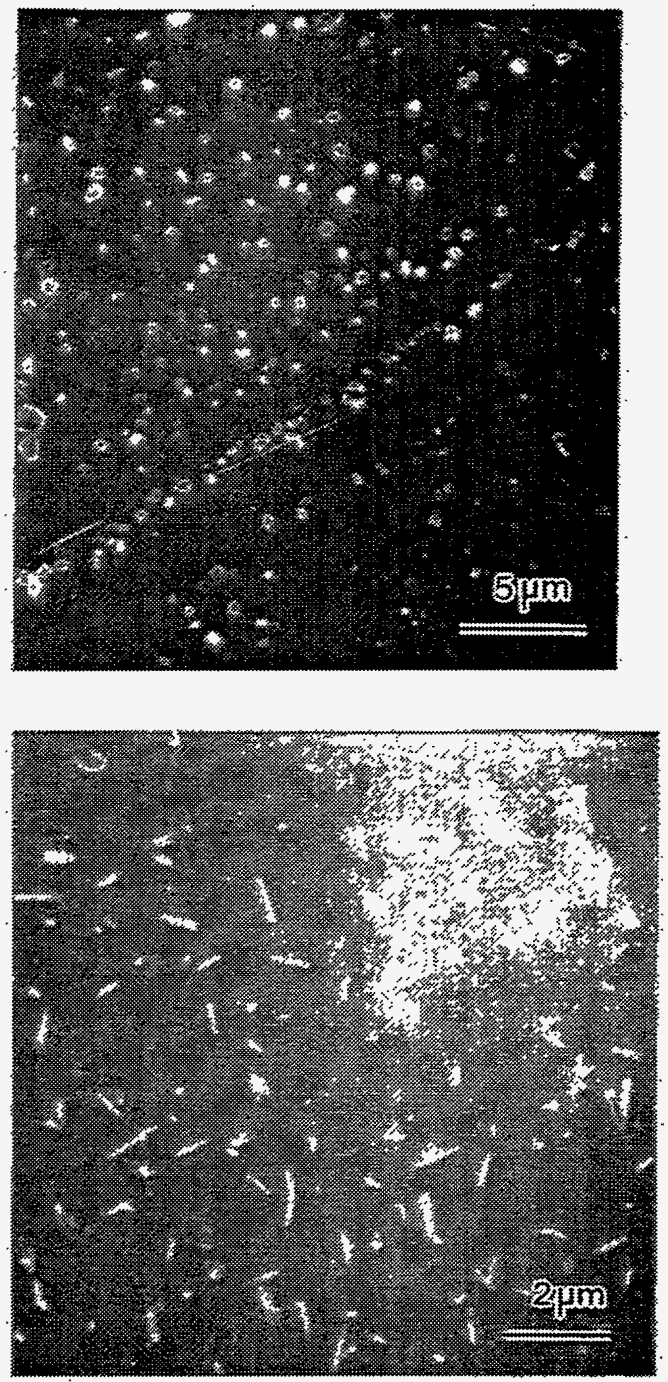


Figure 13: Secondary electron micrographs showing the extent of intergranular and intragranular carbide precipitation in conventionally-processed A600 forging with a final anneal at 1050°C for 2 h.

IAEA IWG -LMNPP
Specialist Meeting
on
Cracking in LWR RPV Head Penetrations

RECENT INFORMATION ON TOP HEAD REPLACEMENT IN JAPAN

Philadelphia, USA

2 - 4 May 1995

Takao Takahashi

Japan Power Engineering and Inspection Corporation

1. STATUS ABROAD

(1) BACKGROUND

In September, 1991, a damage was discovered on the tube base of reactor vessel head of Bugey-3.

67 plants in France and other countries were inspected, and damage cases were discovered in 38 plants.

(2) Cause of Damage

3 Factors

(1) Material

(2) Stress

(3) Environment

(reactor top temperature,

operating hours)

(3) Countermeasures

Vessel Head Replace in 6 Plants.

2. STATUS IN JAPAN

INSPECTION CONDUCTED

In view of the situations abroad, the electric utility companies implemented inspections on 17 plants having relatively long operating hours.

No anomaly was found in any of 17 plants inspected.

Table
Results of ECT for the Reactor Vessel Head Penetration Tubes (as of March, 1995)

'lant ype	Plants inspected	Operating time *1 (x1000hrs)	Total penetrations	Penetrations inspected	Damage Found
Loop	7	34 - 108	283	250	0
Loop	7	62 - 109	462	379	0
Loop	3	46 - 92	232	196	0
otal	17	-	977	825	0

*1 : Total operation time at the ECT inspection.

3. CONCLUSION

It is difficult to conceive that a particular safety problem would occur in Japan concerning the reactor vessel head.

Even if a damage should occur, it can be dealt with by repair. For this reason, inspections corresponding to the plant conditions should be done by electric utility companies.

Table
Replacement of RV Head

Plant	(Loop)
Mihama-3	(3)
Takahama-1	(3)
Takahama-2	(3)

FORM TI-12
8/94

WAPD-T-3045

THERMALLY ACTIVATED DISLOCATION CREEP MODEL FOR PRIMARY WATER STRESS CORROSION CRACKING OF NiCrFe ALLOYS

M. M. Hall, Jr.

U. S. Department of Energy Contract DE-AC11-93PN38195

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Operated for the U.S. Department of Energy
by WESTINGHOUSE ELECTRIC CORPORATION

THERMALLY ACTIVATED DISLOCATION CREEP MODEL FOR PRIMARY WATER STRESS CORROSION CRACKING OF NiCrFe ALLOYS

M. M. Hall, Jr.
Bettis Laboratory
Westinghouse Electric Corporation
West Mifflin, Pennsylvania 15122

ABSTRACT

There is a growing awareness that environmentally assisted creep plays an important role in intergranular stress corrosion cracking (IGSCC) of NiCrFe alloys in the primary coolant water environment of a pressurized water reactor (PWR). The expected creep mechanism is the thermally activated glide of dislocations. This mode of deformation is favored by the relatively low temperature of PWR operation combined with the large residual stresses that are most often identified as responsible for the SCC failure of plant components. Stress corrosion crack growth rate (CGR) equations that properly reflect the influence of this mechanism of crack tip deformation are required for accurate component life predictions.

A phenomenological IGSCC-CGR model, which is based on an *apriori* assumption that the IGSCC-CGR is controlled by a low temperature dislocation creep mechanism, is developed in this report. Obstacles to dislocation creep include solute atoms such as carbon, which increase the lattice friction force, and forest dislocations, which can be introduced by cold prestrain. Dislocation creep also may be environmentally assisted due to hydrogen absorption at the crack tip. The IGSCC-CGR model developed here is based on an assumption that crack growth occurs by repeated fracture events occurring within an advancing crack-tip creep-fracture zone. Thermal activation parameters for stress corrosion cracking are obtained by fitting the CGR model to IGSCC-CGR data obtained on NiCrFe alloys, Alloy X-750 and Alloy 600. These IGSCC-CGR activation parameters are compared to activation parameters obtained from creep and stress relaxation tests. Recently reported CGR data, which exhibit an activation energy that depends on yield stress and the applied stress intensity factor, are used to benchmark the model. Finally, the effects of matrix carbon concentration, grain boundary carbides and absorbed hydrogen concentration are discussed within context of the model.

Keywords: stress corrosion cracking, crack growth rate, models, creep, NiCrFe alloys

INTRODUCTION

Recently discovered intergranular stress corrosion cracking (IGSCC) of Alloy 600 components in the primary water circuits of commercial nuclear power reactors[1] has renewed interest in obtaining a more fundamental understanding of the IGSCC of NiCrFe alloys. An improved understanding of IGSCC and the availability of a physically based quantitative IGSCC crack growth rate (CGR) model could potentially prevent significant economic losses for electric-power utilities that operate nuclear reactors. Unfortunately, a thorough scientific understanding of stress corrosion mechanisms does not exist today and cannot reasonably be expected to exist for some time.

However, reliable models having engineering utility often can be developed for complex physical processes using phenomenological modelling methods. A phenomenological model of environmentally assisted crack growth must be functionally descriptive of the physical processes that are considered to be controlling the crack growth rate. Mechanistic models and the results of controlled laboratory experiments provide insight into the selection of the appropriate mathematical relationships. In developing the phenomenological model, all model variables and parameters are chosen so as to be relatable to engineering parameters. The model parameters are obtained by fitting the model to the available experimental data. The range of each parameter may have to be statistically broadened to obtain satisfactory comparisons between predictions of the model and field experience. To gain acceptance for use, the model must describe all features of the data considered important and no assumption or prediction of the model can be in contradiction with other known experimental facts and established physical understanding. When successfully applied, this phenomenological approach results in engineering methods of component failure prevention that are logical, internally consistent and are less reliant on engineering judgement.

Evidence for the Influence of Dislocation Creep in SCC

There is a growing awareness that environmentally assisted creep plays an important role in primary water stress corrosion cracking of NiCrFe alloys. Was and co-workers[2-4] have reported on the influence of microstructure on the IGSCC and creep deformation of controlled impurity Alloy 600-type alloys in high purity water. Their results show that creep modes of fracture play an important role in IGSCC of these alloys. They conclude that dislocation-controlled creep is the operative creep mechanism based on their observation of large stress exponents in their creep rate-stress correlations. Bousier et al.[5] have reported on the influence of mechanical parameters and the water environment on the IGSCC and creep deformation of Alloy 600 and the creep deformation of Alloy 690 in pressurized water reactor (PWR) primary water. Based on their constant extension rate and stress controlled creep tests, they conclude that strain rate is the mechanical parameter that controls the IGSCC crack propagation rate. Their creep test results show that the high-stress low-temperature creep rates of Alloy 600 and Alloy 690 in water are two to three times the rates achieved in air.

Thermally Activated Dislocation Creep Model

Due to the relatively low temperature of PWR operation and the large residual stresses that are most often identified as responsible for IGSCC failures of reactor components,[6] the expected creep mechanism is thermally activated glide of dislocations. This deformation mechanism is enabled by the thermal agitation of the metal lattice and can operate at temperatures less than about 0.35 of the melting temperature. The lattice friction stress and short range obstacles combine to establish a threshold stress for dislocation slip. As the temperature is reduced, the thermal energy available to overcome obstacles to dislocation glide is reduced and the yield stress increases. This is illustrated by the temperature dependence of the Alloy 600 yield stress as shown by the data of Mulford and Kocks[7] in Fig. 1. Indicated in this figure is the so-called athermal temperature, T_o, at which thermal activation is sufficient to overcome short range obstacles but not long range obstacles. Analysis of these data yields an athermal temperature of 668 K. The yield stress at this temperature is due to long range obstacles. It is termed the athermal stress and is designated here as the threshold stress, σ_{th}.

As shown by Kocks, et al.[8] in their treatise on the thermodynamics and kinetics of dislocation slip, the rate equation that relates creep strain rate, $\dot{\varepsilon}$, to temperature, T, and the applied stress, σ, can be written as:

$$\dot{\varepsilon} = \dot{\varepsilon}_s \exp - \frac{\Delta G(\sigma)}{RT}. \qquad (1)$$

In this equation ΔG is the Gibbs free energy and R is the universal gas constant. The pre-exponential factor, $\dot{\varepsilon}_s$, is an intrinsic strain rate that is proportional to the density of mobile dislocations and the thermal-activation frequency.

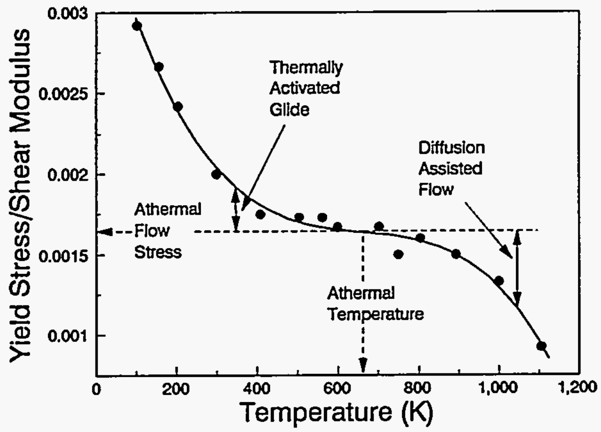

Figure 1. Alloy 600 Yield Stress Normalized by the Shear Modulus. Data From Ref. 7.

As shown in the Appendix, Eq.(1) can be specialized to represent creep due to the thermal component of the flow stress only. The thermal component of the flow stress is commonly called the "effective" stress. A normalized effective stress is defined by

$$\frac{\sigma^*}{\sigma_s^*} \equiv \frac{\sigma - \sigma_{th}}{\sigma_s - \sigma_{th}}. \qquad (2)$$

The subscript s refers to the maximum or "saturation" internal shear stress generated in overcoming the obstacles to dislocation glide at absolute zero. For stresses above the saturation stress, dislocations can find no equilibrium positions. Consequently, large scale slip, followed by a rapid approach to mechanical instability, will occur.

As shown in the Appendix, the deformation rate equation can be expressed in terms of the effective stress and energy parameters:

$$\dot{\varepsilon} = \dot{\varepsilon}_s \exp\left\{ - \frac{\Delta H_o^*}{RT^*}\left[1 - \left(\frac{\sigma^*}{\sigma_s^*}\right)^p\right]\right\}. \qquad (3)$$

The asterisk indicates effective parameters as defined in the

Appendix. The stress exponent p in Eq.(3) has a value less than one and greater than zero. The effective activation enthalpy, ΔH_o^*, is the activation energy associated with the short-range obstacles only. The effective temperature, T^*, is given by

$$\frac{1}{T^*} = \frac{1}{T} - \frac{1}{T_o}. \tag{4}$$

IGSCC - CGR MODEL DEVELOPMENT

We assume that stress corrosion crack advance occurs by a process of creep fracture of a crack tip region having a radius r_c. Repetition of this process results in sustained crack growth. If a critical creep strain, ε_c, is required for fracture of this zone, the crack growth rate is

$$\dot{a} = \frac{2 r_c \dot{\varepsilon}_{ct}}{\varepsilon_c}. \tag{5}$$

The size, r_c, of this crack-tip creep-fracture zone (CFZ), depends on the micromechanism of crack advance[9] and may be influenced by dimensions of microstructural features, local stress as determined by deformation characteristics of the material within and surrounding the CFZ, and environmental effects.

The Alloy 600 creep data of Boursier, et al.[5] show that the creep strain rate decreases with time, which is consistent with the expectation that strain hardening will dominate any recovery mechanism that may operate at the low temperatures assumed for a thermally activated creep mechanism of IGSCC. Therefore the crack tip strain rate, $\dot{\varepsilon}_{ct}$, in the above equation is taken to represent the average rate during the time between CFZ fracture events.

Crack Tip Stress

We assume that during the time interval between CFZ fracture events the crack is stationary and that the creep deformation is confined within the CFZ. Furthermore, the CFZ is assumed to be small compared to the dimensions of the crack tip plastic zone. The stress within the crack tip CFZ will decrease somewhat due to creep relaxation during the interval between fracture events. However, we assume that the CFZ stress is constant and is equal to the time-independent stress that exists at the boundary of the CFZ and the plastic zone into which the IGSCC crack advances when the CFZ fractures.

The time-independent stress within the crack tip plastic zone has been derived by Hutchinson,[10] Rice and Rosengren[11] and is given by

$$\sigma = \left(\frac{\alpha(n)}{r}\right)^{\frac{1}{n+1}} \sigma_o^{\frac{n-1}{n+1}} K^{\frac{2}{n+1}}. \tag{6}$$

In this equation σ_o is the flow stress, which we take here to be the yield stress, K is the applied stress intensity factor, r is the distance from the crack tip, n is the inverse of the strain hardening exponent and α is a slowly varying function of n having a value of $1/4\pi$ for plane strain in the elastic limit (n = 1). Using Eq. (6) and substituting normalized effective stress parameters for the yield stress and stress intensity factor, the normalized effective stress, Eq.(2), evaluated at r_c, can be written as

$$\frac{\sigma^*(r_c)}{\sigma_s^*} = \left(\frac{\alpha(n)}{r_c}\right)^{\frac{1}{n+1}} \left(\frac{\sigma_o-\sigma_{th}}{\sigma_s-\sigma_{th}}\right)^{\frac{n-1}{n+1}} \left(\frac{K-K_{th}}{K_s-K_{th}}\right)^{\frac{2}{n+1}}. \tag{7}$$

In this equation we have introduced a threshold stress intensity factor, K_{th}, and a saturation stress intensity factor, K_s, corresponding respectively to the threshold stress, σ_{th}, and the saturation stress, σ_s. The saturation stress intensity factor is a mechanical instability parameter that, like σ_s, has a maximum value at absolute zero.

Eq.(7) can now be substituted into Eq.(3) to obtain the crack tip strain rate, which can in turn be substituted into Eq.(5) to obtain an expression for the IGSCC crack growth rate. The resulting equation was used to fit IGSCC data obtained on Alloy 600 materials having a range of yield stress due to application of a range of cold prestrain. Using these data, it was discovered that the product of the first two factors on the right hand side of Eq.(7) is essentially constant over the range of yield stress and stress intensity factors represented in the data. This result requires that the characteristic CFZ size, r_c, decrease as the yield stress increases roughly as

$$r_c \sim \frac{1}{(\sigma_o-\sigma_{th})^{n-1}}, \tag{8}$$

as α is a relatively weak function of n. Using this result, Eq.(7) can be simplified to:

$$\frac{\sigma^*(r_c)}{\sigma_s^*} = b\left(\frac{K-K_{th}}{K_s-K_{th}}\right)^{\frac{2}{n+1}}, \tag{9}$$

where b is a constant. Note that this equation retains an implicit dependence on yield stress through the inverse of the strain hardening exponent, n.

IGSCC Crack Growth Rate Equation and Apparent Activation Energy

Now the crack growth rate equation, Eq.(5), becomes

$$\dot{a} = \frac{2r_c\dot{\varepsilon}_s}{\varepsilon_c}\exp\left\{-\frac{\Delta H_o^*}{RT^*}\left[1-b\left(\frac{K-K_{th}}{K_s-K_{th}}\right)^p\right]\right\}. \quad (10)$$

The apparent activation energy, Q, is given by

$$Q = \Delta H_o^*\left[1-b\left(\frac{K-K_{th}}{K_s-K_{th}}\right)^p\right]. \quad (11)$$

Fitting of Alloy 600 IGSCC data to Eq.(10) confirmed that the stress exponent p is equal to 2/(n+1) for values of n less than about 4. However, an alternate expression for p, which has 2/(n+1) as a limit for small values of n, is required to obtain a good fit for materials having large n (large yield strength). This is discussed in more detail below.

IGSCC CRACK GROWTH RATE DATA AND DATA CORRELATIONS

There are few primary water IGSCC crack growth rate data, obtained on NiCrFe alloys, that are suitable for benchmarking the IGSCC-CGR model given by Eq.(10) above. The best available data are from the work by Shen and Shewmon[12] and Speidel and Magdowski[13]. However, there are limitations to each of these data sets. Both sets of data were obtained on displacement controlled specimens, which results in a decreasing stress intensity factor as crack growth occurs. Crack growth rates were determined by Shen and Shewmon using in-situ measurements of crack length but their tests were conducted in steam, instead of water. Speidel and Magdowski obtained crack growth rates from measurements of average crack extension divided by the total exposure time. Their rates do not, therefore, account for the time to the onset of IGSCC crack growth. The Shen-Shewmon data obtained on Alloy X-750 and Alloy 600 are shown in Fig. 2 and Fig. 4. The Speidel-Magdowski data obtained on Alloy 600 are shown in Fig. 3, Fig. 5 and Fig. 6. Note that the best fit Arrhenius curves for each data set in Fig. 2 and Fig. 3 were adjusted slightly to intersect at a common athermal temperature, T_o. The Shen-Shewmon data are best fit using an athermal temperature of 682 K while the Speidel-Magdowski data could be fitted reasonably well using

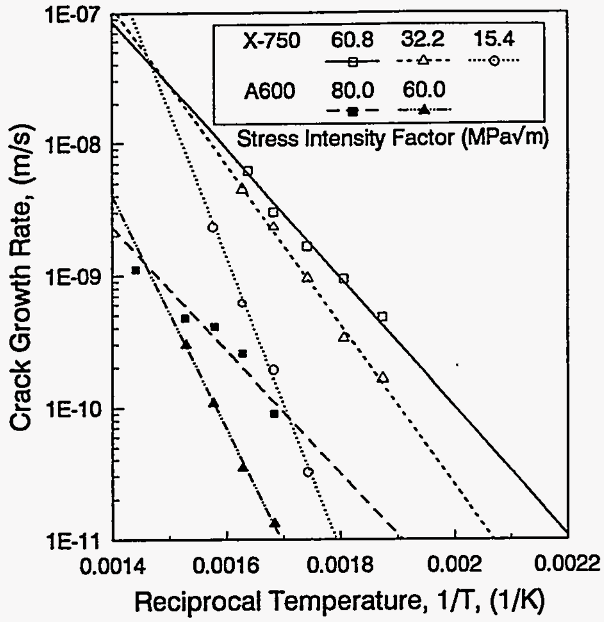

Figure 2. Effect of Temperature and Stress Intensity Factor on Crack Growth Rate. Data From Ref. 12.

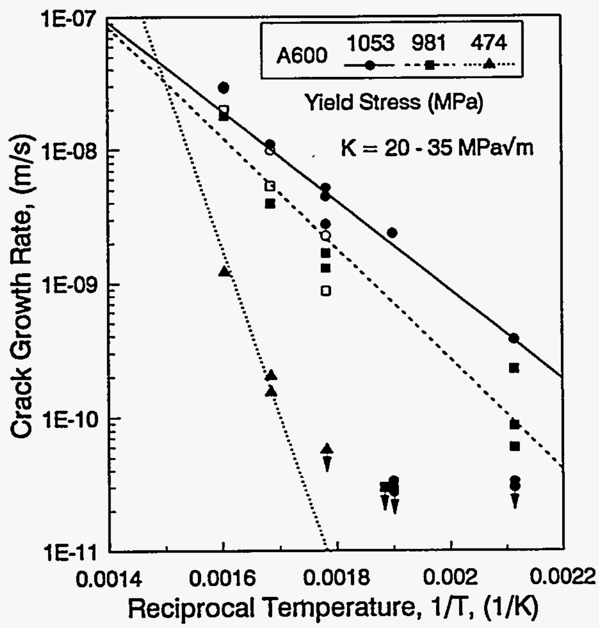

Figure 3. Effect of Temperature and Yield Stress on Crack Growth Rate. The solid symbols represent data taken from Ref. 13. The open symbols represent values obtained by cross-plotting the data correlations shown in Figure 5 for K = 30 MPa√m.

the value of 668 K obtained from the data in Fig. 1. Results of the data fits to Eq.(10) are given in Table I.

	$$\dot{a} = \dot{a}_o \exp\left\{-\frac{\Delta H_o^*}{RT^*}\left[1 - b\left(\frac{K-K_{th}}{K_s-K_{th}}\right)^p\right]\right\}$$			
Data Source	Shen-Shewmon Ref. 12		Speidel-Magdowski Ref. 13	
Temperature Range	593 K - 693 K		473 K - 623 K	561 K - 623 K
Material	X750	A600	A600	A600
a_o (m/s)	4.3E-8	1.2E-9	4.0E-8	1.0E-7
ΔH^*_o (kJ/mol)	445.3	Not Analyzed	432.2	442.2
T_o (K)	682	682	668	657.5
K_{th} (MPa√m)	15.35	Not Analyzed	6.37	7.94
K_s (MPa√m)	219.8	Not Analyzed	99.9	96.6
b	0.876	Not Analyzed	0.891	0.748
p	0.0623	$p = 1 + 0.4281 \ln(1/n)$ (13)		
n	≈ 6	$1/n = 0.256 \ln(1559.3/S_y)$ (14)		

Table I. Crack Growth Rate Equation Parameters. There are insufficient Alloy 600 data from Ref. 12 to obtain most parameters. The second correlation for Ref. 13 data provides a better data fit for the temperature range 561 K - 623 K. This correlation is used for the data comparisons in Fig. 5 and Fig. 6.

The expression for the inverse hardening exponent, n in Eq.(14), was obtained from the stress-strain data reported by Webb and Hall[14]. In this equation, S_y is the yield stress.

DISCUSSION

The data in Fig. 2 through Fig. 6 have several interesting features that are correctly predicted by our IGSCC-CGR model. First, consistent with the model predictions, the negative slopes of each of the Arrehnius curves (which are proportional to the apparent activation energies, Q) in Fig. 2 and Fig. 3 decrease with both increasing applied stress intensity factor and increasing yield stress. Second, these data can be fitted well with curves that intersect, as the model requires, at the athermal temperature, T_o. Third, there is an apparent threshold stress intensity factor evident in Fig. 4 and Fig. 5. Finally, the slopes of the log-crack growth rate versus log-stress intensity factor curves in Fig. 6 decrease with increasing temperature and yield stress and increase slightly with increasing stress intensity factor consistent with the model predictions. The magnitude of these slopes are equivalent to the stress exponent in a more conventional "power law" SCC rate equation.

The activation enthalpies, ΔH_o^*, in Table 1 are all nearly equal, being about 440 kJ/mole (4.6 eV). Sirois and Birnbaum[15] found activation enthalpies of 1.9 eV and 2.9 eV in stress relaxation and differential temperature tests of pure Ni and Ni-C, respectively. The larger value found for the more complex engineering alloys modelled here is not surprising given the potential for a greater variety of microstructural barriers to thermally activated slip. Moreover, the carbon content of the simple Ni-C alloy used in Sirois and Birnbaum's study was 0.025 weight percent (w/o) as compared to 0.04 w/o to 0.10 w/o, respectively in the Shen-Shewmon and Speidel-Magdowski A600 materials. Bandy and van Rooyen[16] found that the apparent activation energy for initiation of primary water IGSCC in Alloy 600 tubing increased as the carbon concentration increased. Their data show an increase in Q of about 126 kJ/mol (1.3 eV) as carbon increases from 0.02 w/o to 0.06 w/o. Therefore, the value of 4.6 eV found here for the activation enthalpy compares favorably to the values of Sirois and Birnbaum considering the higher carbon content of the alloys that were used to obtain the data used in this study.

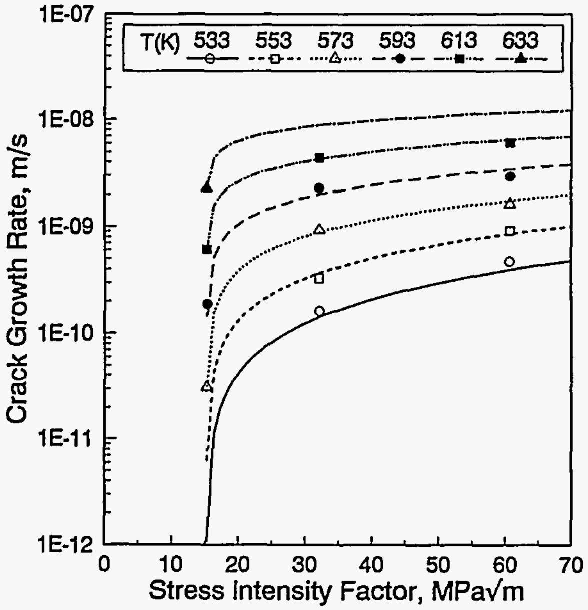

Figure 4. Alloy X-750 Crack Growth Rate as Function of the Stress Intensity Factor. Data From Ref. 12.

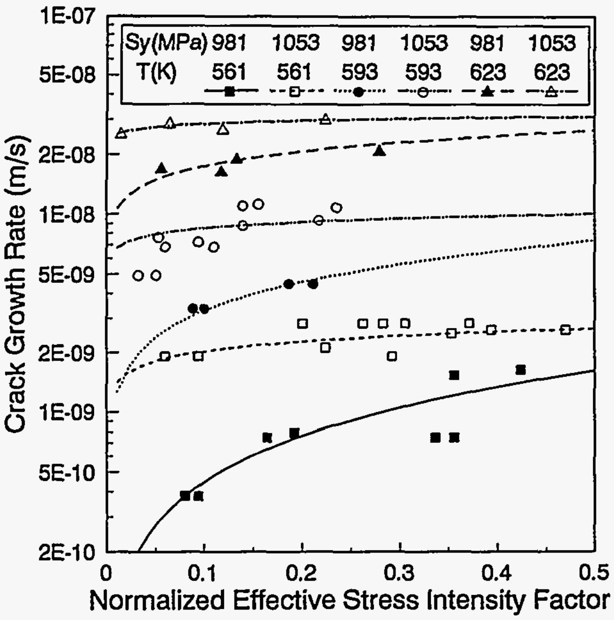

Figure 5. Alloy 600 Crack Growth Rate as Function of the Normalized Effective Stress Intensity Factor. Data From Ref. 13.

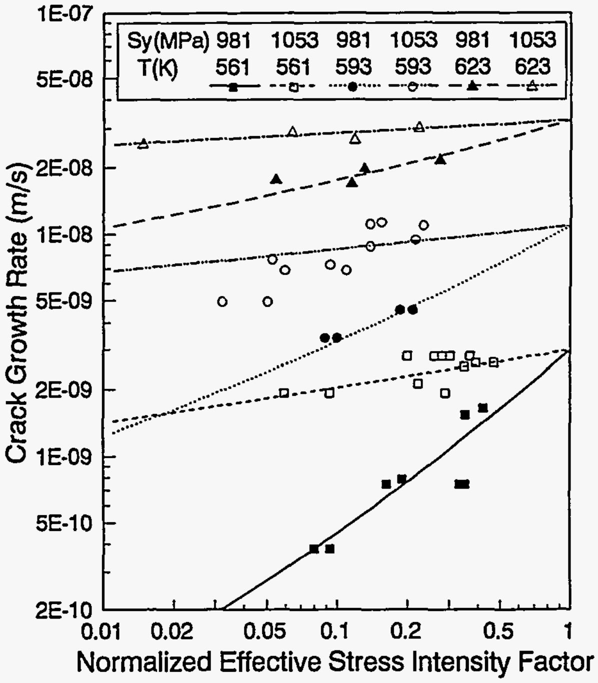

Figure 6. Alloy 600 Crack Growth Rate as Function of the Normalized Effective Stress Intensity Factor. Data From Ref. 13.

Figure 7 shows the effect of stress intensity factor and yield stress on the apparent activation energy, Q, as predicted by Eq.(11). The Q values derived from Figure 2 and Figure 3 are shown for comparison. Note that Q values obtained from the Shen-Shewmon A600 data are in good agreement with the predictions based on the Eq.(11) correlation of the Speidel-Magdowski data. Note also that the Q correlations for Alloy X-750 and A600 are very similar for similar yield stresses.

The model predicts that an increase in the activation enthalpy leads to a reduction in the creep rate, which results in a reduced crack growth rate. This prediction is supported by the results of Was et al.[3] who showed that an increase in solute carbon concentration from 0.002 w/o to 0.032 w/o results in about 3 orders of magnitude decrease in the creep rate of controlled purity Alloy 600-type alloys. They furthermore showed that, in a controlled extension rate IGSCC test of these alloys, higher solute carbon concentration results in a significant reduction in the relative amount of IGSCC.

In addition to demonstrating that solute carbon increases the activation enthalpy for dislocation creep, Sirois and Birnbaum showed that hydrogen, in their Ni-H and Ni-C-H alloys, causes a reduction in the activation enthalpy. Their results show that an addition of about 11 weight parts per million (wppm) hydrogen to the Ni-C-H alloy results in a reduction in the activation enthalpy by 0.9 eV (87 kJ/mole). At this rate, an increase of 4 wppm, which is the hydrogen solubility in unstressed Alloy 600 at 633 K, would increase the creep rate by about a factor of three at low stress and assuming an initial activation enthalpy of about 440 kJ/mole. This prediction compares favorably to the creep test results reported by Boursier et al.[5] who report a factor of two increase in creep rate for Alloy 600 and Alloy 690 when tested in water. They also report increased crack growth rates due to an increase of the hydrogen overpressure in their controlled extension rate tests.

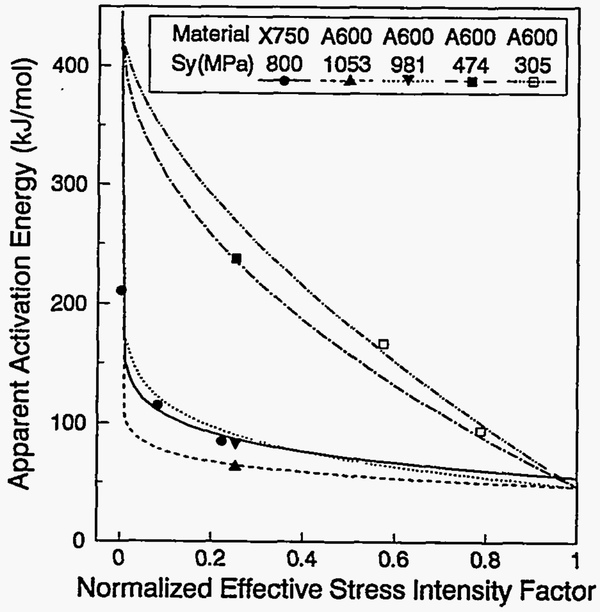

Figure 7. Apparent Activation Energy as Function of the Normalized Effective Stress Intensity Factor Showing the Effect of Yield Stress. The A600 curves are derived from the A600 data from Ref. 13 and the open symbols represent the Q values derived from the A600 data from Ref. 12.

Note that the IGSCC-CGR model developed here does not predict the observation by Boursier et al. that susceptibility to IGSCC first increases then decreases with increased hydrogen overpressure. This effect may be related to corrosion film stability, which is not a feature explicitly considered in the model development.

A final observation is that there is more than an order of magnitude difference between the pre-exponential terms in the A600 CGR equations. This difference is not understood. It is speculated that the difference is brought about by differences in heat treatment. The heat treatment histories of these materials and their resulting microstructures are not detailed by the authors. Both materials are reported as received in the hot formed condition. However, the as-received strength and carbon concentration of the Speidel-Magdowski material are high, as compared to that of the Shen-Shewmon material; 474 MPa versus 305 MPa, and 0.10 w/o versus 0.06 w/o, respectively.

The CGR equation preexponential has two factors that, within the context of the model, influence the crack growth rate but are not developed here. These are the critical strain, ε_c, and the characteristic distance, r_c, in Eq.(5). Both are expected to be influenced by the heat treatment and the carbon concentration.

The critical creep strain for advancing the crack is expected to be a function of the local crack tip hydrogen concentration. This in turn is expected to be dependent upon the balance between the rates of hydrogen production and absorption at the crack tip with the rate of hydrogen diffusion away from the creep fracture zone. Grain boundary carbides may act simply to retard hydrogen production and absorption by limiting exposure of a less corrosion resistant matrix to the water. The rate of hydrogen loss is controlled by the hydrogen diffusion rate, which may be influenced by microstructural and strain induced trapping of hydrogen within the creep fracture zone, potentially at grain boundary carbide-matrix interfaces. Finally, since they are a dominant grain boundary feature, grain boundary carbides, and perhaps the grain size, may influence the characteristic distance over which the creep fracture process operates.

CONCLUSIONS AND SIGNIFICANCE

A primary water stress corrosion crack growth rate model has been developed for NiCrFe alloys X-750 and Alloy 600 based on an assumption that the IGSCC growth rate is controlled by the kinetics and thermodynamics of a thermally activated low temperature creep mechanism. Phenomenological features of the best available stress corrosion crack growth rate data are well correlated using this model. The model correctly predicts an effect of both stress intensity factor and yield stress on the apparent activation energy and an effect of both yield stress and temperature on the stress intensity factor dependence of the crack growth rate.

The thermal-activation enthalpies obtained by analysis of IGSCC crack growth rate data are consistent with independent measures of these parameters. The potential effects of the corrosive environment, matrix carbon and grain boundary carbides on the IGSCC crack growth rate can be rationalized within context of the model. This is an area for additional research.

Crack growth rate equations of the type developed here have application to engineering concerns for reactor power plant component performance. Application of the model allows interpolation and extrapolation of the available data with increased confidence and provides direction for additional testing.

ACKNOWLEDGMENTS

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APPENDIX

Strain Rate Equation For Thermally Activated Glide With Two Stage Obstacles

Environmentally assisted cracking of NiCrFe alloys X-750 and Alloy 600 has occurred in nuclear reactor components under conditions of high stress and temperatures less than about 0.35 T_m, the melting temperature. The probable mechanism for crack tip deformation under these conditions is thermally activated glide of dislocations. Details of the thermodynamics and kinetics of dislocation glide are discussed in the References.[8, 17-19]

The following development follows that of Kocks et al.[8] We consider thermally activated glide in the presence of both short and long range obstacles. Examples of long range obstacles include dislocations on parallel slip planes and large precipitate particles. Examples of short range obstacles include dislocations threading the glide plane ("forest" dislocations) and the Peierls-Nabarro stress. See Conrad[18] for a discussion of obstacles to dislocation glide.

We choose a reference state defined by a reference temperature, T_0. Above T_0 the short range obstacles to dislocation glide are overcome by thermal fluctuations without the need for an applied stress greater than a threshold stress, σ_{th}, called the "athermal" stress. Below T_0 thermal fluctuations are no longer sufficient acting alone to overcome short range obstacles. An additional "effective" stress, σ^*, is required. The effective stress is defined by

$$\sigma^* \equiv \sigma - \sigma_{th}. \tag{A.1}$$

For temperatures below T_0 the strain rate is given by

$$\dot{\varepsilon} = \dot{\varepsilon}_s \exp-\frac{\Delta G(\sigma)}{kT}. \tag{A.2}$$

The pre-exponential factor, $\dot{\varepsilon}_s$, is a material parameter that is proportional to the density of mobile dislocations and the thermal-activation frequency. The activation enthalpy, ΔG, is given by

$$\Delta G(\sigma) = \int_{\sigma}^{\sigma_s} b\Delta a\, \delta\sigma. \tag{A.3}$$

In this equation, $b\Delta a$ is an "activation volume" where b is the Burgers vector and Δa is the activation area. In terms of the effective stress

$$\Delta G(\sigma^*) = \int_{\sigma^*}^{\sigma_s^*} b\Delta a^*\, \delta\sigma^*, \tag{A.4}$$

where

$$\sigma_s^* = \sigma_s - \sigma_{th}, \qquad \sigma^* = \sigma - \sigma_{th}. \tag{A.5}$$

Utilizing a phenomenological glide resistance profile described by Kocks et al. for short range obstacles,

$$\Delta G(\sigma^*) = F_0^*\left[1-\left(\frac{\sigma^*}{\sigma_s^*}\right)^p\right]^q, \tag{A.6}$$

where F_0^* is the free energy necessary to overcome the short range obstacles without the aid of an external stress. In terms of the activation free enthalpy (Gibbs free energy), ΔG,

$$F_0^* \equiv \Delta G(\sigma^* = 0) = \int_0^{\sigma_s^*} b\Delta a\, \delta\sigma^*. \tag{A.7}$$

Now

$$\Delta H \equiv \Delta G + T\Delta S, \tag{A.8}$$

where ΔH is the activation enthalpy and ΔS is the activation entropy. If we assume that both ΔH and ΔS are independent of temperature, and if we choose their values at T_0, we have for all temperatures, $\Delta H = \Delta H_0^*$ and $\Delta S = \Delta S_0^*$. Then

$$F_0^* \equiv \Delta G = \Delta H_0^* - T\Delta S_0^*. \tag{A.9}$$

And since

$$\Delta S_0^* = \frac{\Delta H_0^*}{T_0}, \tag{A.10}$$

$$\Delta F_0^* = \Delta H_0^*(1 - T/T_0). \tag{A.11}$$

Then, using Eq.(A.6),

$$\begin{aligned}\frac{\Delta G}{kT} &= \frac{\Delta H_0^*}{kT}(1-T/T_0)\left[1-\left(\frac{\sigma^*}{\sigma_s^*}\right)^p\right]^q,\\ &= \frac{\Delta H_0^*}{kT^*}\left[1-\left(\frac{\sigma^*}{\sigma_s^*}\right)^p\right]^q,\end{aligned} \tag{A.12}$$

where

$$\frac{1}{T^*} = \frac{1}{T} - \frac{1}{T_0}. \qquad (A.13)$$

Then Eq. (A.2) becomes

$$\dot{\varepsilon} = \dot{\varepsilon}_s \exp\left\{-\frac{\Delta H_0^*}{kT^*}\left[1-\left(\frac{\sigma^*}{\sigma_s^*}\right)^p\right]^q\right\}. \qquad (A.14)$$

In the analysis of the Shen-Shewmon and Speidel-Magdowski IGSCC crack growth data we find satisfactory data fits if we assume that q; which can have values from 1 to 2, has a value of 1.

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Prediction of PWSCC in Nickel Base Alloys Using Crack Growth Rate Models

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Abstract

The Ford/Andresen slip dissolution SCC model, originally developed for stainless steel components in BWR environments, has been applied to Alloy 600 and Alloy X-750 tested in deaerated pure water chemistry. A method is described whereby the crack growth rates measured in compact tension specimens can be used to estimate crack growth in a component. Good agreement was found between model prediction and measured SCC in X-750 threaded fasteners over a wide range of temperatures, stresses, and material condition. Most data support the basic assumption of this model that cracks initiate early in life.

The evidence supporting a particular SCC mechanism is mixed. Electrochemical repassivation data and estimates of oxide fracture strain indicate that the slip dissolution model can account for the observed crack growth rates, provided primary rather than secondary creep rates are used. However, approximately 100 cross-sectional TEM foils of SCC cracks including crack tips reveal no evidence of enhanced plasticity or unique dislocation patterns at the crack tip or along the crack to support a classic slip dissolution mechanism. No voids, hydrides, or microcracks are found in the vicinity of the crack tips creating doubt about classic hydrogen related mechanisms. The bulk oxide films exhibit a surface oxide which is often different than the oxides found within a crack. Although bulk chromium concentration affects the rate of SCC, analytical data indicates the mechanism does not result from chromium depletion at the grain boundaries. The overall findings support a corrosion/dissolution mechanism but not one necessarily related to slip at the crack tip.

Key terms: Alloy 600, Alloy X-750, predictive models, mechanism, stress corrosion cracking

Introduction

This paper describes the application of the slip-dissolution model to the pure water stress corrosion cracking (PWSCC) of nickel base alloys. This model was first developed by D. Vermilyea[1] and later applied to stress corrosion cracking of stainless steel in boiling water reactor conditions by P.Ford and P. Andresen[2,3]. The paper covers four aspects of SCC modeling: demonstration that the model fits existing Alloy 600 crack growth data, demonstration that the model gives reasonable estimates for A600 crack growth rates based solely on fundamental inputs, application of the model to predict component SCC, and mechanistic considerations.

The conceptual basis for the slip dissolution model assumes a step-wise process of crack advance that can be summarized as follows:

- Oxide film rupture at the crack tip due to time dependent strain (creep).
- Bare metal (anodic) dissolution at the crack tip.
- Repassivation at the crack tip due to reformation of the protective oxide.
- Continued deformation at the crack tip resulting in a new oxide rupture event that repeats the sequence.

Creep strain is the primary variable proposed to control the periodicity of the film rupture events. The rate of anodic dissolution and repassivation is determined by the alloy, the nature of its protective oxide, and the environment at the crack tip.

Mathematically, the rate of crack advance can be derived from Faraday's law as the depth "a" of metal that corrodes electrochemically within an increment of time:

$$da = \frac{M}{zpF}\, dQ_f = \frac{M}{zpF}\, i dt \qquad (1)$$

where: da = depth of corrosion (cm)
M = molecular weight (g/mole)
p = density of metal (g/cm^3)
z = charge on the dissolving metal (equivalents/mole)
F = Faraday constant (96,500 coulombs/mole)
Q_f = charge density per film rupture event (coulombs/cm^2)
i = current density at the crack tip (amperes/cm^2)
t = time (seconds)

The repassivation current typically follows a power law:

$$i = i_o\left(\frac{t}{t_o}\right)^{-n} \qquad for\ t_o < t < t_f \qquad (2)$$

where: i_o = bare metal current density (amps/cm^2)
t_f = time between rupture events
t_o = time at beginning of repassivation
n = repassivation rate parameter

The incremental growth for each rupture event (i.e., average crack growth rate) is:

$$V_{CT} = \frac{M}{zpF} \int_{t_o}^{t_f} i_o\left(\frac{t_o}{t_f}\right)^{-n} dt = \frac{M}{zpF}\,\frac{Q_f}{t_f} = \frac{M}{zpF}\,\frac{Q_f}{\epsilon_F}\,\dot{\epsilon}_{CT} \qquad (3)$$

where: ϵ_f = oxide fracture strain
$\dot{\epsilon}_{CT}$ = crack tip strain rate

Andresen also developed an empirical relationship which showed that $\dot{\epsilon}_{CT}$ is proportional to the stress intensity factor[2]. The average crack growth rate can be expressed generally as a function of n, ϵ_f, i_o, and stress intensity factor K_I:

$$V_{CT} = A\ \dot{\epsilon}_{CT}^{n} = A\ K_I^{4n} \qquad (4)$$

where A and n are parameters which are specific functions of the crack tip material and environment combinations.

Crack Growth Rates

To assess the applicability of the slip dissolution model to PWSCC of nickel alloys, a model was evaluated against an Alloy 600 primary water CGR database. Figure 1 provides a compilation of Alloy 600 CGR data from several sources[4,5,6,7], including KAPL generated data. This data is edited to include only actively loaded specimens to avoid the issue of stress relaxation in constant displacement specimens. Also shown is the model prediction for CGR vs. K_I using repassivation ratio (n) with values of 0.5 and 0.7. The functionality expressed by the Andresen model is in reasonable agreement with the observed trends.

On the basis of this fit of the data from the slip dissolution model, it was decided to examine some of the fundamental input parameters of the model as further evidence that the model is viable for PWSCC application of nickel base alloys.

Model Fundamental Parameters

The purpose of this section is to examine the slip dissolution model prediction of Alloy 600 crack growth rates based on the fundamental inputs. The four fundamental inputs (measured parameters) to the slip dissolution model are maximum bare metal dissolution current density (i_o), repassivation rate (n), creep rate (as a function of applied load), and oxide fracture strain.

Figure 2 shows a plot of log current vs. time for the repassivation of Alloy 600 wire at 288°C (550°F) in boric acid/sodium hydroxide with pH_{288} approximately 7.2. These data were obtained by stepping the applied potential to -0.711 V_{SHE} after 15 minutes at -1.5V_{SHE}. The plot indicates a lower bound bare metal current density of 3.5 mA/cm^2 and a repassivation parameter, "n", of 0.7. Similar results have been obtained using the drop weight method at the same potential, indicating that the Alloy 600 oxide found at the test condition is in fact reduced at the test conditions. This repassivation current is equivalent to those reported by Soji[8] at 288°C in 0.01M Na_2MoO_4, but the repassivation time is somewhat faster.

Crack growth rate data for Alloy 600 at 288°C (550°F) is not available. However, based on the data of Figure 1, a nominal value of 0.3 mils/day at 338°C (640°F) and at a K_I of 25 ksi$\sqrt{in}$ is a reasonable baseline value to extrapolate to lower temperatures. Apparent activation energies for PWSCC of A600 range from 15 to 54 Kcal/mol depending on stress intensity factor and degree of cold work[9,10]. Extrapolation of 0.3 mils/day at 338°C to 288°C (550°F) with this range of activation energies yields crack growth rates of 0.01 mils/day (Q = 54) to 0.1 mils/day (Q = 15).

Figure 3 presents the predicted crack growth rate of A600 from the slip dissolution model as a function of the periodicity of the film rupture. A periodic rupture time (t_f) of about 550 seconds would be required for a crack growth rate of 0.1 mils/day, whereas a t_f of 26,000 results in a calculated rate of 0.01 mils/day.

An oxide fracture strain of 0.003, approximately equal to the base metal yield strain, represents an order of magnitude estimate for mixed spinel oxides at 80°F, and it is assumed that this value is independent of temperature up to 550°F. For a crack tip strain rate $\dot{\epsilon}_{CT} = \epsilon_f/t_f$, and ϵ_f = 0.003, the resulting creep strain rate would be calculated to be between 4.2 x 10^{-6} to 1.1 x 10^{-7} (sec^{-1}). These strain rates are 2-3 orders of magnitude higher than the steady state (secondary) strain rates predicted by Garud for Alloy 600[11]. However, with the t_f being between 9.2 minutes and 7.2 hours, primary creep appears to be dominant with predicted strain rates between 10^{-6} and 10^{-8} (sec^{-1}), which is in agreement with the required crack tip strain rate to maintain the required crack growth rates predicted by an anodic dissolution mechanism.

The general conclusion from this analysis is that the slip dissolution model yields reasonable estimates of Alloy 600 crack growth rates based on fundamental input parameters provided primary creep rates are considered dominant rather than secondary creep rates.

Prediction of Component SCC Endurance

KAPL has developed an engineering method to relate actual component SCC endurance to laboratory crack growth rates. The needed input data consists of measured crack growth rates as a function of stress intensity factor for the material of interest at the temperature of interest. It is assumed that crack growth initiates at time zero in a very small initial flaw. We have generally assumed an initial flaw size of 0.0001" (0.00025 cm) and have found that the calculated crack lengths are fairly insensitive to the initial flaw size assumptions. The general process consists of the following steps:

- Assume initial small flaw size.
- Calculate K_1 for initial flaw and given load.
- Calculate corresponding crack growth rate.
- Calculate amount of crack growth for an incremental period of time at the K_1.
- Advance the crack by the given amount.
- Recalculate K_I and repeat the process.

SCC data is necessary on both laboratory compact tension specimens and full size components for matching material/environment conditions. Such data has been obtained by KAPL for X-750 condition AH. Figure (4) presents KAPL crack growth rate data on Condition AH X-750 obtained from 0.4T CT specimens under constant load at 315-360°C (600-680°F). Correlating predictions are also plotted against the data. A K_I^{2n} correlation was found to be better than K_I^{4n} for this material/environment combination.

The laboratory crack growth rates and correlating model has been used to assess the SCC of threaded fasteners. This model prediction is shown in Figure (5) and is superimposed on the measured crack depth data for specimens tested at 338°C (640°F). Another prediction and matching fastener data for 282°C (540°F) is shown in Figure (6). Crack lengths in the fasteners were determined by destructive metallographic examination. Figure (7) compares the predicted crack lengths to the observed crack length for 88 specimens over a wide range of test conditions with applied stresses from 28 to 103 ksi and test temperatures from 282 to 360°C (540 to 680°F). Good predictive agreement is obtained over a wide range of measured crack lengths (from about 3 to 123 mils). The general agreement is within 2X and it is concluded that the CGR modelling process can provide reasonable predictions of SCC in plant components provided quality CGR data is available.

Physical Evidence

Figure 8 shows an AEM cross-sectional image of an Alloy 600 specimen tested at 338°C (640°F). The foil preparation process is described in Reference (12). The significant observation is lack of indication of any deformation associated with the crack process. Figure 9 shows an SCC crack tip on a grain boundary without carbides with no evidence of voids or grain boundary sliding often associated with creep. To date, more than 100 AEM foils from 50 separate A600 specimens have been analyzed. We have found no voids, hydrides, or microcracks ahead of the crack tip, commonly cited as indications of a hydrogen related SCC mechanism. The lack of unique deformation associated with the crack tip casts doubt on slip or creep as a participant in the SCC mechanism.

The bulk oxide films exhibit a surface oxide which is often different than the oxides found within a crack. Surface oxides are Cr and Ni rich near the metal surface and contain NiO and spinels in the outer layers. In a number of instances, NiO is found within the crack adjacent to the metal which is not observed at the bulk specimen oxide/metal interface (Figure 10).

The presence of oxides at the crack tip is an indication that oxidation is taking place as a result of exposure to the environment. The differing oxides inside the crack as compared to the outside surface suggests a difference in the environment and/or material (grain boundary) at the tip. The changes in the oxide composition inside the crack suggests non-uniformity in solubility and probably repassivation kinetics. These observations taken as a whole suggest an oxide fracture/dissolution mechanism for crack propagation, but not one associated with slip at the crack tip.

Conclusions

- The slip-dissolution model provides an adequate engineering fit to the measured crack growth rates for Alloy 600 in pure/primary water.
- Based solely on fundamental inputs, the slip dissolution model gives reasonable estimates for A600 crack growth rates, provided primary creep rate values are inputed.
- An engineering model (independent of mechanisms) that uses laboratory crack growth rate data provides a sound basis to predict the SCC endurance of actual components.
- Issues pertaining to the application of the slip dissolution model to PWSCC of nickel alloys include lack of evidence of plastic deformation uniquely associated with the crack tip, understanding oxide changes as a function of grain boundary microstructure and chemistry, and the rates of creep deformation.

Acknowledgements

This work was performed at Knolls Atomic Power Laboratory, operated for the U. S. Department of Energy by KAPL, Inc., Martin Marietta Corporation, under contract DE-AC12-76SN00052. The repassivation studies were performed by subcontract at General Electric Corporate Research & Development.

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11. YS Garud, "Development of a Model for Predicting Intergranular Stress Corrosion Cracking of Alloy 600 Tubes in PWR Primary Water", EPRI NP-3791, 1985.

12. N Lewis, B Bussert, M Bunch, "Cross Sectional AEM Analysis of SCC Cracks", EPRI Steam Generator IGA/SCC Workshop, December 8-10, San Antonio, Texas, 1992.

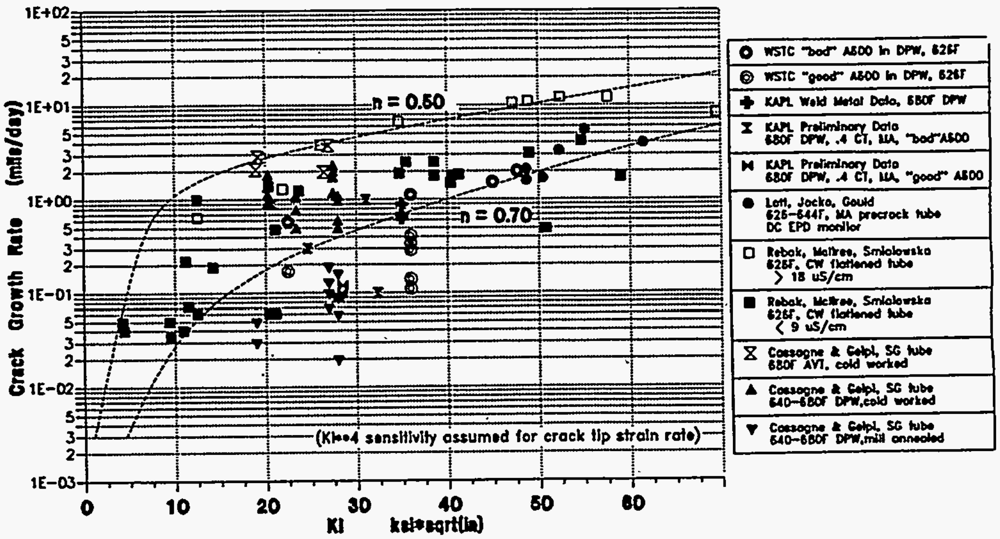


Figure 1. Compilation of Alloy 600 crack growth rate data in high temperature deaerated pure water and Ford/Andresen model correlation for crack growth rate vs. K_I for 'n' values of 0.5 and 0.7.

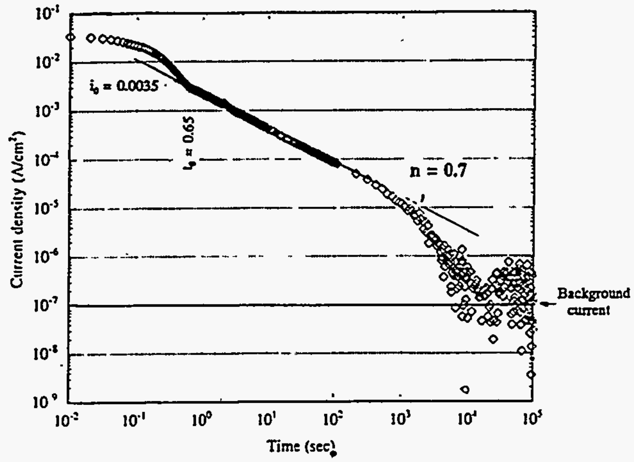


Figure 2. KAPL repassivation data on Alloy 600, potential pulse from $-1500mV_{SHE}$ to $-711mV_{SHE}$ at 288°C, deaerated 0.1M boric acid titrated with NaOH to a cell resistance of 105 ohms and pH=7.6 at 25°C.

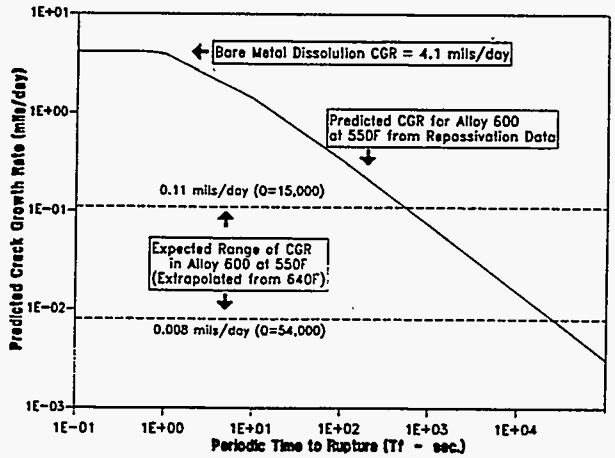


Figure 3. Predicted crack growth rate from repassivation data as a function of the periodic oxide rupture time.

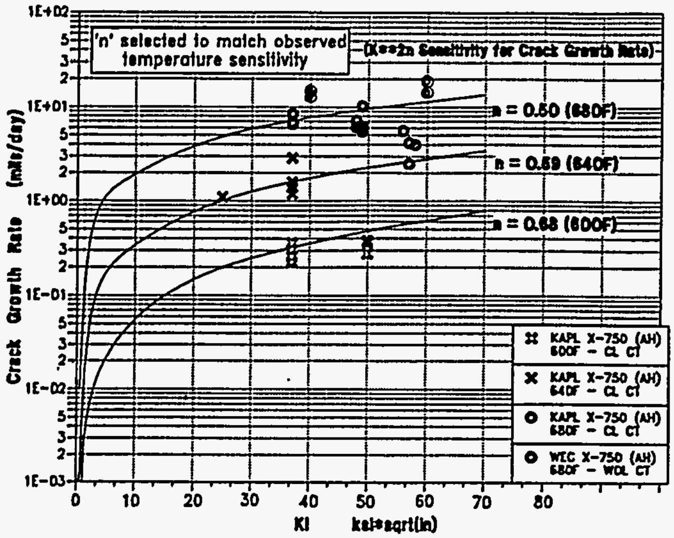


Figure 4. Crack growth rate data for X-750, Condition AH in high temperature deaerated pure water, constant load 0.4T CT specimens.

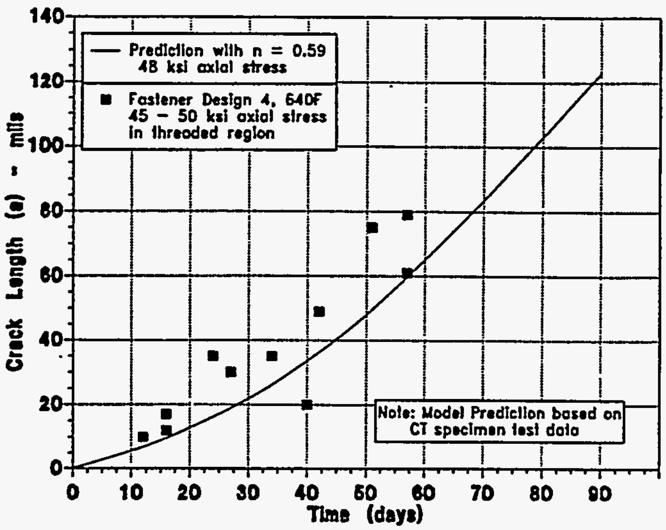


Figure 5. Model prediction of crack depth based on CT specimen data and corresponding observations for X-750, Condition AH threaded fasteners tested in DPW at 640°F.

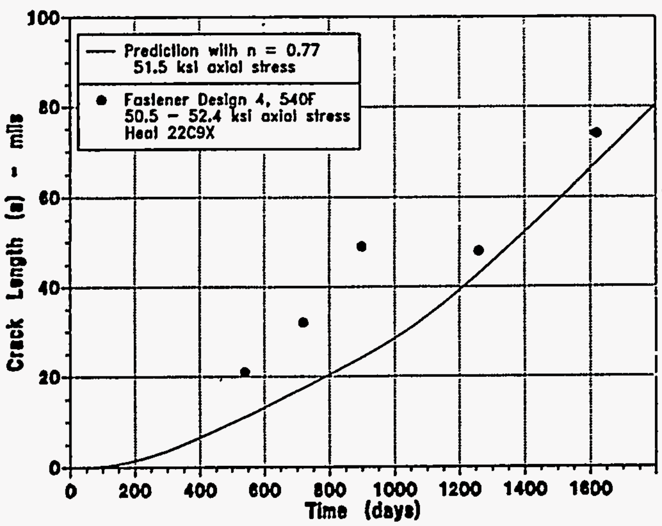


Figure 6. Model prediction of crack depth and corresponding for X-750, Condition AH threaded fasteners tested in DPW at 540°F.

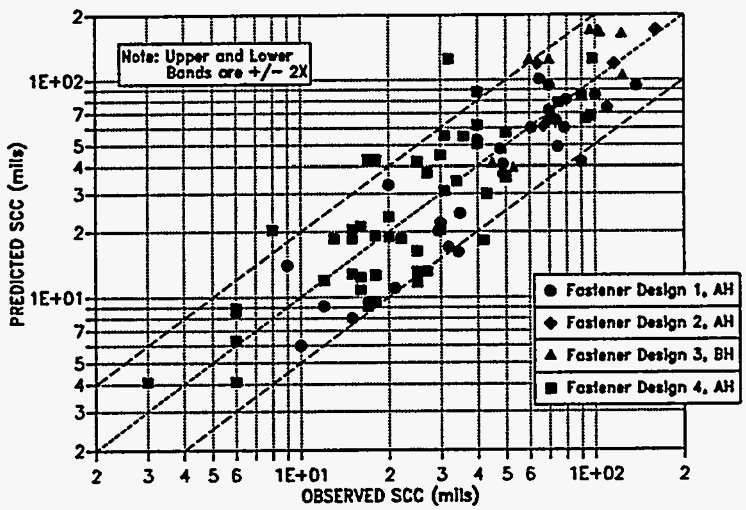
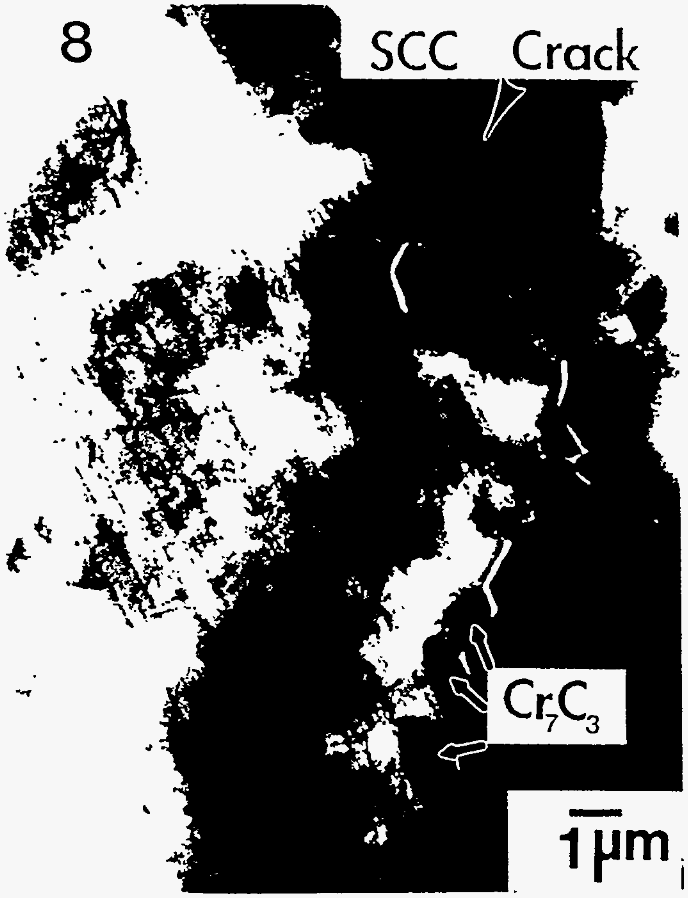
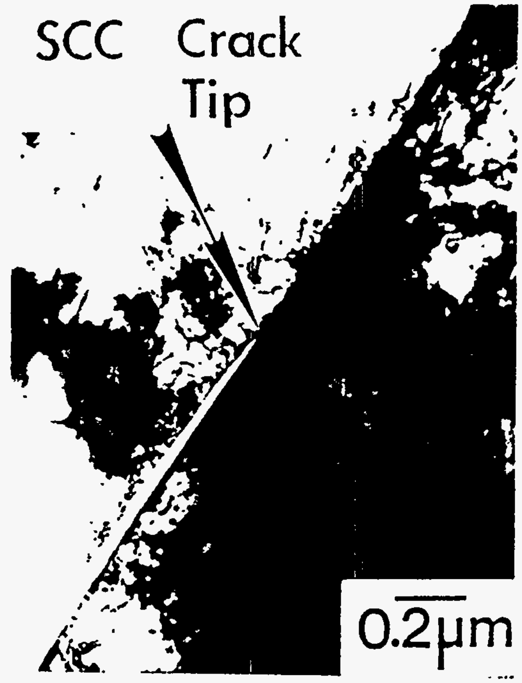
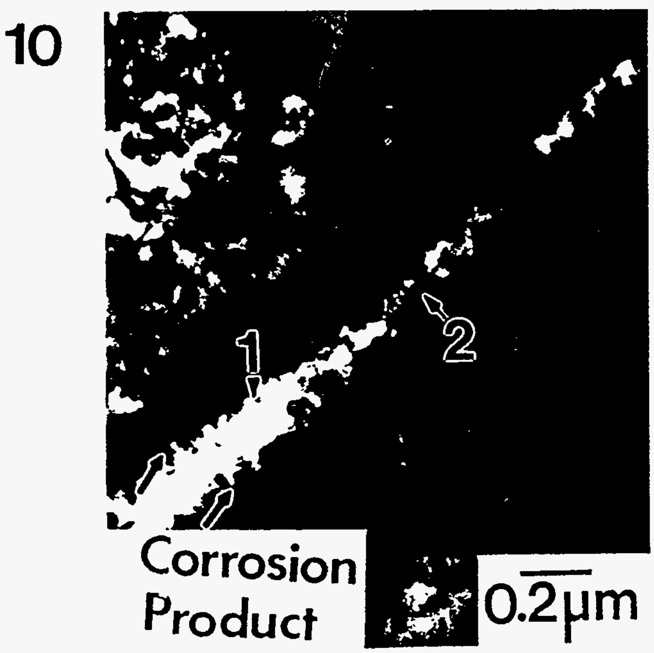


Figure 7. Predicted crack length based on CT specimen data versus observed crack length for wide range of fastener geometries, applied stress, and temperature.







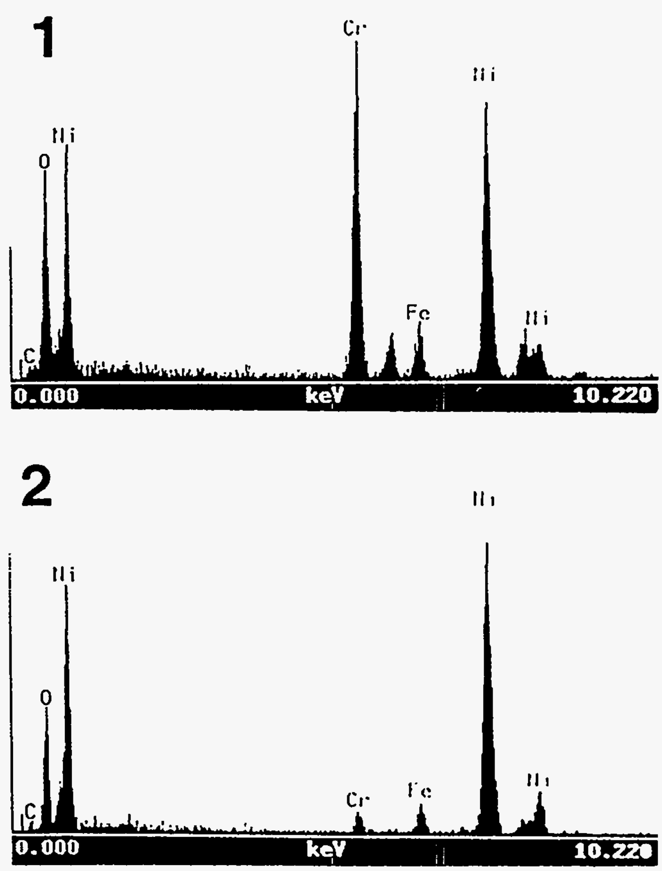


FIGURE 8. AEM CROSS-SECTIONAL IMAGE SHOWING SCC CRACK PROPAGATING AROUND COMPLEX GRAIN BOUNDARY CONTAINING Cr_7C_3 CARBIDES. NOTE AS CRACK CHANGES DIRECTION DISLOCATION STRUCTURE REMAINS THE SAME. FIGURE 9. SCC CRACK TIP. NO CARBIDES, VOIDS OR MICROCRACKS OBSERVED AHEAD OF CRACK. FIGURE 10. CORROSION PRODUCT IN SCC CRACK CONTAINING BOTH CR AND NI RICH OXIDE AND NiO.

Stress Corrosion Cracking Initiation

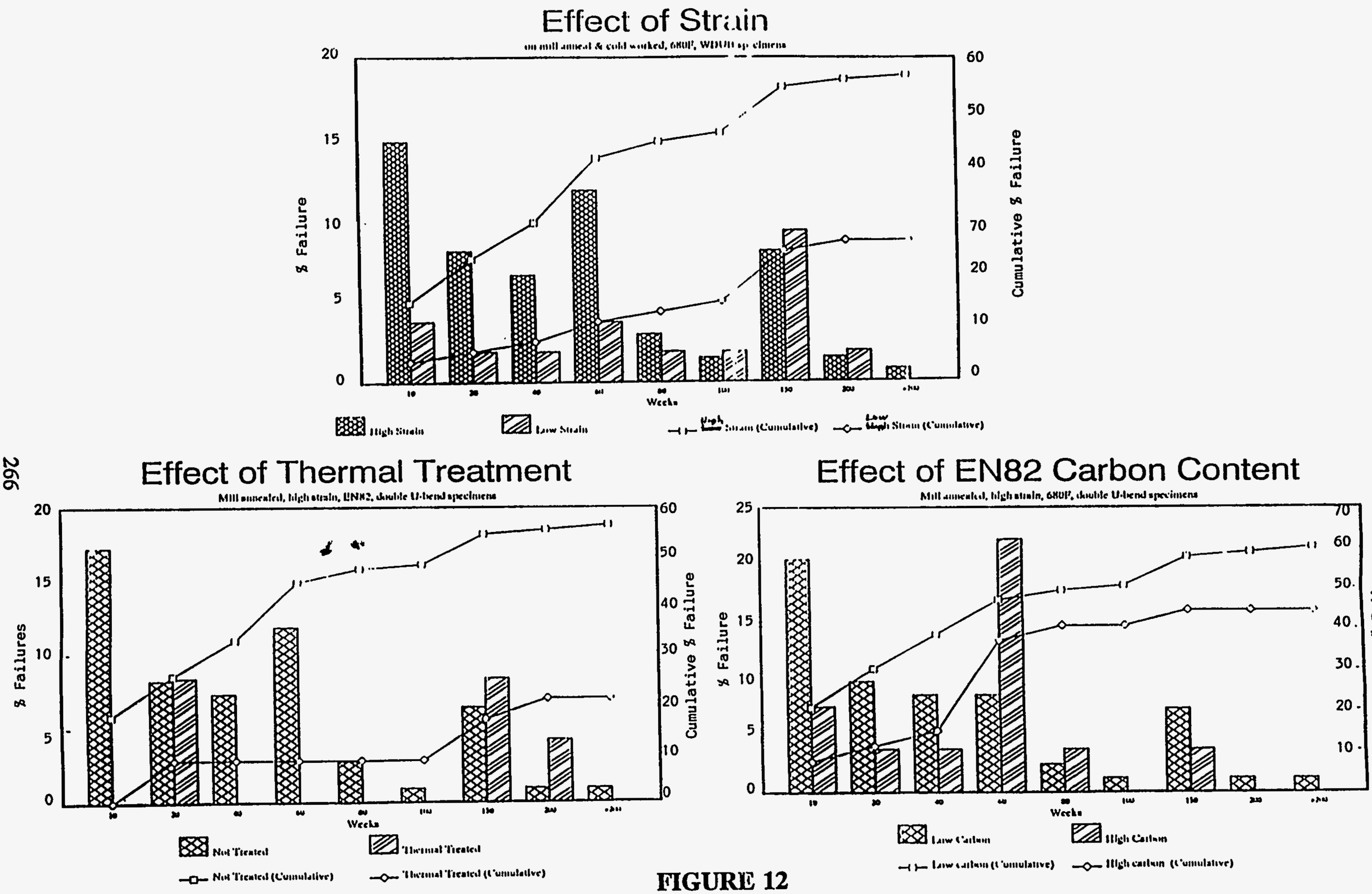
- 502 Double U-bend & 112 C-ring specimens
- EN62, EN82 weld metal
- Tested at 600°F, 640°F, 680°F (315, 338, 360°C) for up to 249 weeks
- Four strain levels;
  - 10% and 16% for U-bends
  - .45% and 4% for C-rings

SCC Initiation - Results

Thermal treatment of weld metal (1125°F (607°C) for 7 hours)

- Beneficial in reducing SCC
- Statistically - 95% confidence of a significant difference
- 56% of non-thermal treated specimens failed
- 21% of thermally treated specimens failed

FIGURE 11



Effect of Strain
% Failure
Cumulative % Failure
Weeks
High Strain
Low Strain
High Strain (Cumulative)
Low Strain (Cumulative)
Effect of Thermal Treatment
Mill annealed, high strain, EN82, double U-bend specimens
% Failures
Cumulative % Failure
Weeks
Not Treated
Thermal Treated
Not Treated (Cumulative)
Thermal Treated (Cumulative)
Effect of EN82 Carbon Content
Mill annealed, high strain, 680P, double U-bend specimens
% Failure
Cumulative % Failure
Weeks
Low Carbon
High Carbon
Low carbon (Cumulative)
High carbon (Cumulative)

FIGURE 12

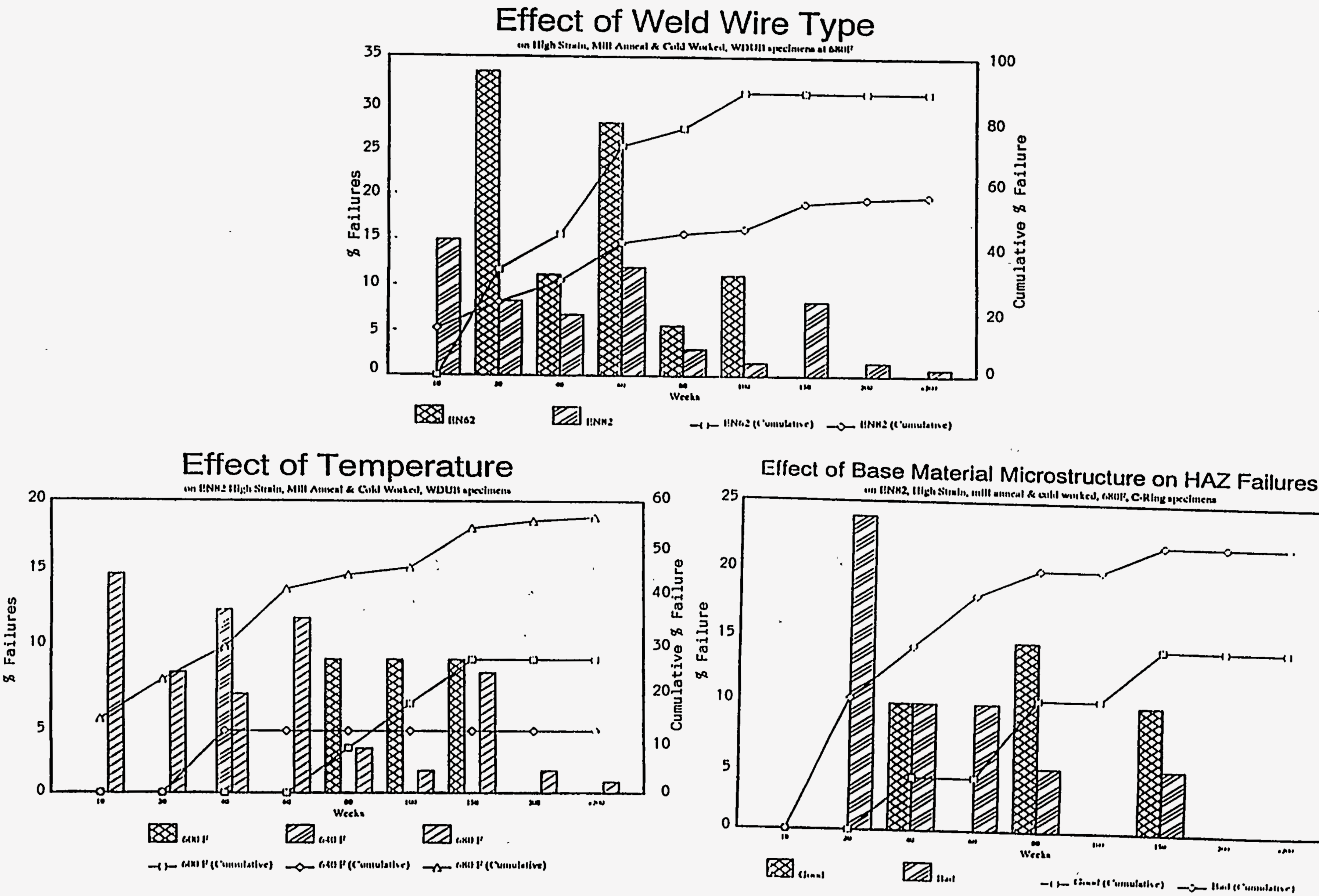

Effect of Weld Wire Type
% Failures
Cumulative % Failure
Weeks
Effect of Temperature
% Failures
Cumulative % Failure
Weeks
Effect of Base Material Microstructure on HAZ Failures
% Failure
Cumulative % Failure
Weeks

FIGURE 13

STRESS CORROSION CRACK GROWTH RATE TESTING

- 3 - .4T precracked compact tension specimens
- EN82 weld metal
- Constant load - stress intensity of 35 ksi$\sqrt{in}$ (38 MPa$\sqrt{m}$)
- 680°F (360°C) primary water for 108 days

Results/Conclusions

- Specimens showed a range of growth rates of 0.62 to 1.08 mils/day (1.8×10^{-10} to 3.2×10^{-10} m/s)
- Best estimate SCC growth rate of 0.80 mils/day (2.4×10^{-10} m/s)
- SCC initiated in less than 16 days

FIGURE 14

Characteristics of Weld Metal Used for Crack Growth Rate Specimens

Characteristic	
Filler Metal ID Carbon Level	EN82 761668S 0.009 w/o
Welding Process	Hot Wire Gas Tungsten
Welding Parameters Shielding gas Current Volts	 Argon 300 amps 12.5 v

0.4T Compact Tension Specimens were oriented such that the notch is perpendicular to the welding passes.

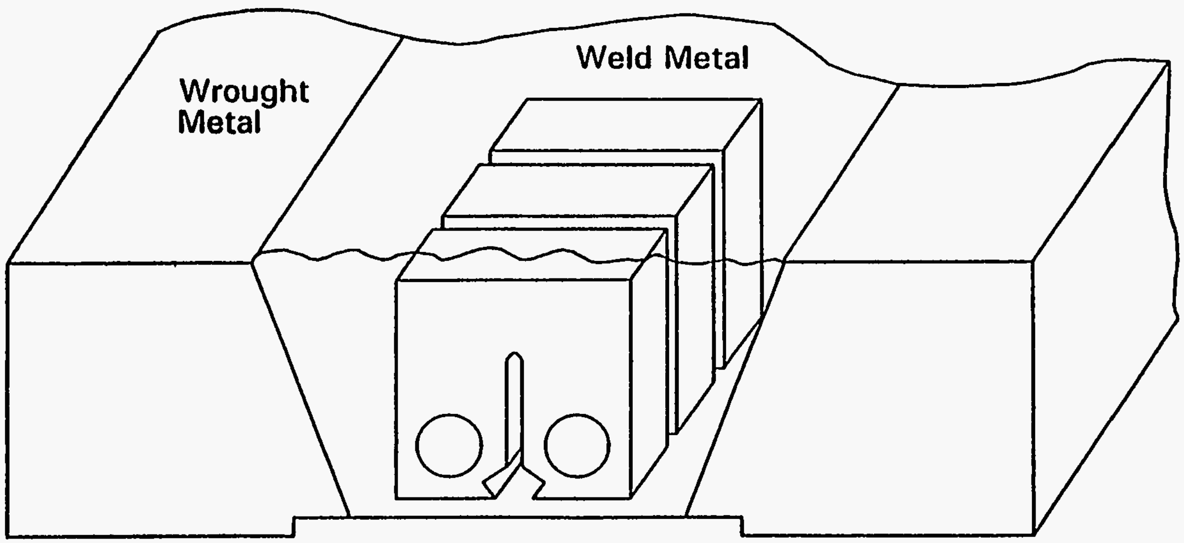


FIGURE 15

COMPARISON OF SCCGRs FOR EN82 WELD METAL SPECIMENS

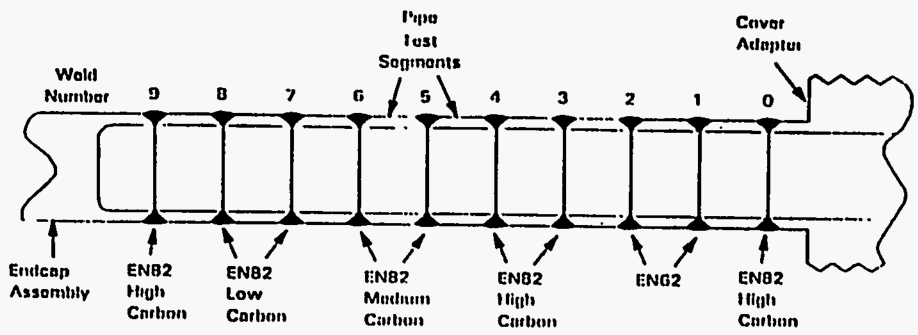
Specimen I.D.	Test Time (days)	Crack Detection Time (days)	Cracking Time (days)	Maximum Crack Depth (mils)	SCCGR[(1)] (mils/day)	SCCGR [(2)] (mils/day)
7741 Side 1	108	3.60	104.40	112.3	1.08	0.95
7741 Side 2	108	3.60	104.40	95.9	0.92	0.83
0726 Side 1	108	3.40	104.60	67.8	0.65	0.56
0726 Side 2	108	3.40	104.60	79.8	0.76	0.69
7742 Side 1	108	15.90	92.10	67.8	0.74	0.74
7742 Side 2	108	15.90	92.10	56.8	0.62	0.62
				Overall Average	$\Sigma = 0.80$	$\Sigma = 0.74$

Notes:

(1) SCCGR = (Maximum Crack Depth - Crack Depth at Detection)/Cracking Time
For this calculation, the crack depth at detection was assumed to be zero.

(2) SCCGR = (Maximum Crack Depth - Crack Depth at Detection)/Cracking Time
For this calculation, the crack depth at detection for specimens 7741 and 0726 were determined by correlating the total SCC area to the LVDT compliance. Due to a broken LVDT wire, a crack depth at detection could not be calculated for 7742 and was assumed to be zero.

FIGURE 16



Cross sectional (schematic) view of test pipe assembly

Test Parameters

• Autoclave pressure	2850 psi (20 MPa)
• Water temperature	680°F (360°C)
• Hydrogen overpressure	40-60 cc/kg
• Axial load plus pressure	20.3 ksi (140 MPa)
• Axial tensile residual stress	30 ksi (207 MPa)
• Test duration	78 weeks

<u>ALLOY 600 PIPE WELDMENT STRESS CORROSION TEST</u>

- Axially loaded pipe specimen
- Nine 3" diameter Schedule 160 Alloy 600 pipe test segments
- Two heats of piping (good and bad microstructure)
- Total of 10 welds
  - 8 EN82 welds (2 low, 2 medium, 4 high carbon content)
  - 2 EN62 welds

FIGURE 17

Summary of Post-Test Evaluation of Pipe Welds

Weld No.	Met. Sect.	Penetrant Test Results	Ultrasonic Test Results	Weld Metal	Heat Affected Zone	Base Metal
1	A	1 circumferential linear indication, 1.0" long, root (U)	No indications	No cracks	2 cracks > 0.100" long, 0.015" deep (U) 0.013" deep (U)	No cracks
1	B	8 longitudinal linear indications: (5) 0.125" long, (3) 0.063" long, weld metal	No indications	1 longitudinal crack 0.020" long, 0.0075" deep	No cracks	No cracks
1	C	6 longitudinal linear indications: (4) 0.031" long, (2) 0.125" long, weld metal	No indications	No cracks	No cracks	No cracks
1	D	1 circumferential linear indication, 1.0" long; 3 longitudinal linear indications, 0.125" long, weld metal	No indications	No cracks	1 crack - < 0.001" deep (U)	No cracks
2	A	Lack of fusion, 0.700" long (A), scattered porosity	No indications	No cracks	1 crack - > 0.080" long, 0.024" deep (U)	No cracks
2	B	Scattered porosity	No indications	2 cracks > 0.060" long, 0.004" deep > 0.050" long, 0.008" deep	1 crack - 0.020" long, 0.005" deep (U)	No cracks
2	C	Scattered porosity	No indications	1 crack - 0.030" long, 0.009" deep	2 cracks > 0.030" long, 0.002" deep (U) > 0.010" long, 0.001" deep (U)	No cracks
2	E	Scattered porosity	1 indication weld metal	No cracks	No cracks	No cracks
3	B	Scattered porosity	No indications	No cracks	No cracks	No cracks
3	C	Scattered porosity	No indications	No cracks	No cracks	1 crack - 0.002" deep (A)
3	D	Scattered porosity	No indications	No cracks	2 cracks 0.001" deep (A) 0.007" deep (U)	1 crack - 0.002" deep (A), counterbore
4	B	Scattered porosity	No indications	3 cracks > 0.020" long, 0.003" deep > 0.020" long, 0.002" deep 0.002" deep	2 cracks > 0.060" long, 0.001" deep (U) > 0.040" long, 0.001" deep (A)	1 crack - > 0.060" long, 0.016" deep (A), counterbore

FIGURE 18

SUMMARY OF POST-TEST EVALUATION OF PIPE WELDS (CONTINUED)

Weld No.	Met. Sect.	Penetrant Test Results	Ultrasonic Test Results	Weld Metal	Heat Affected Zone	Base Metal
4	D	Scattered porosity	1 indication weld metal	No cracks	2 cracks > 0.020" long, 0.001" deep (U) 0.001" deep (A)	No cracks
5	B	Scattered porosity	No indications	No cracks	1 crack - > 0.020" long, 0.003" deep (U)	No cracks
6	A	Scattered porosity	No indications	No cracks	1 crack - 0.001" deep (U)	1 crack - 0.001" deep (A)
6	C	Scattered porosity	1 indication HAZ	1 crack - 0.0006" deep (A)	1 crack - > 0.04" long, 0.006" deep (U)	No cracks
6	D	Scattered porosity	1 indication HAZ	No cracks	1 crack - > 0.060" long, 0.007" deep (U)	1 crack - > 0.080" long, 0.007" deep (A), counterbore
7	A	Lack of fusion, 4.1" long (A). Linear porosoty, 2.1" long	No indication	No cracks	1 crack - > 0.060" long, 0.009" deep (U)	No cracks
7	B	5 diagonal linear indications, 0.060"-0.120" long, weld metal	No indications	2 cracks 0.002" deep 0.002" deep	1 crack - > 0.060" long, 0.024" deep (U)	No cracks
7	C	1 diagonal linear indication, 0.125" long, weld metal. Linear porosity, 0.700" long	1 indication base metal	1 crack - 0.001" deep	2 cracks > 0.060" long, 0.012" deep (U) > 0.060" long, 0.007" deep (A)	No cracks
8	B	Scattered porosity	1 indication HAZ	2 cracks 0.003" deep 0.001" deep	2 cracks > 0.080" long, 0.012" deep (U) > 0.080" long, 0.006" deep (A)	1 crack - > 0.080" long, 0.011" deep (U), counterbore
9	A	No indications	No indications	No cracks	1 crack - 0.020" long, 0.001" deep (A)	No cracks
0	B	No indications	No indications	No cracks	No cracks	No cracks

NOTES: Crack length was established by grinding and polishing incrementally until the crack disappeared.
If no crack length is shown, the crack had disappeared after the first increment (usually 0.020") was removed.
All cracks are circumferential unless otherwise indicated.

(A) Side of weld with acceptable microstructure base metal - NX8908
(U) Side of weld with unacceptable microstructure base metal - NX8913

Pipe wall thickness = 0.437" nominal

FIGURE 18A

Alloy 600 Pipe Weldment Stress Corrosion Test - Continued

Results

- SCC initiation observed in weld metal and HAZ of each weld

Location	Maximum Crack Depth (mils/mm)
Weld Metal	8.8/.22
HAZ	23.8/.60
Base Metal	16.0/.41

- EN62 and low carbon EN82 had highest propensity for weld SCC
- Worst HAZ SCC occurred in base metal with bad microstructure
- HAZ and base metal cracks initiated from stress concentrators

Alloy 600 Pipe Weldment Stress Corrosion Test
Longitudinal Section of Pipe Weld

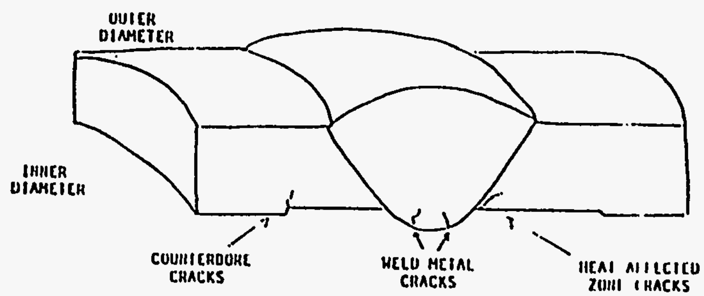


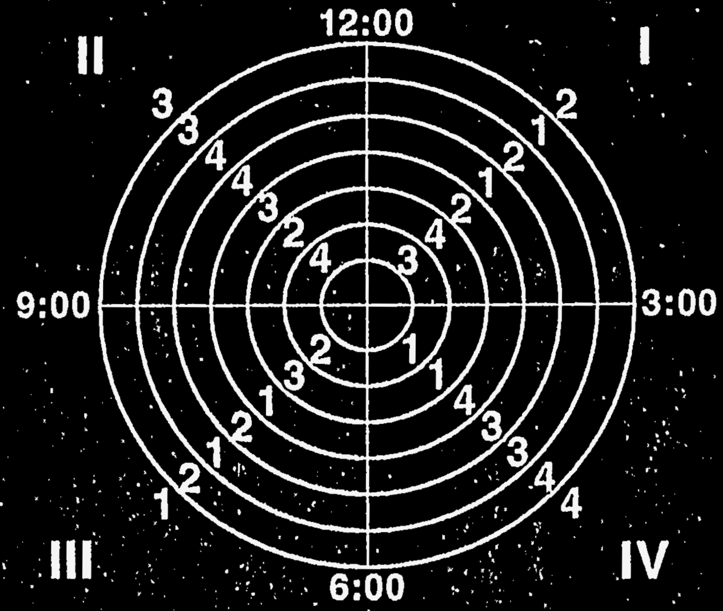
FIGURE 19

WELDING SEQUENCE

	Quadrant of Each Bead			
	Bead 1	Bead 2	Bead 3	Bead 4
Pass 1	IV	III	I	II
Pass 2	IV	II	III	I
Pass 3	III	I	II	IV
Pass 4	I	III	IV	II
Pass 5	III	I	IV	II
Pass 6	I	III	II	IV
Pass 7	III	I	II	IV

Section A-1

GIRTH WELD 4 LAYERS; 7 PASSES TOTAL



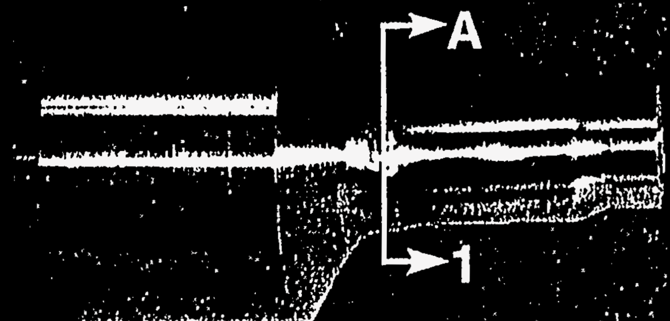


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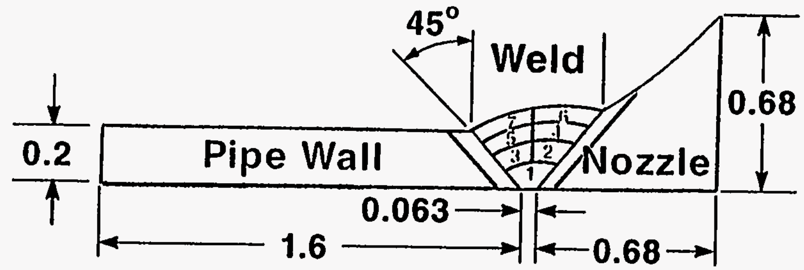
FIGURE 20

THERMAL BOUNDARY CONDITIONS



2.15" Weld Outer Dia
1.75" Weld Inner Dia

- Air with NC Heat Transfer
- Exterior Surface of Weld Elements Insulated After Birth
- Monitor Depth of Molten Zone (40-50 Mils Max)
- Thermal Constraint Equations for Uniform Weld Metal Cooling
- Internal Heat Generation in Weld Passes
- Timed Weld Element Heat Generation to Model Arc Travel Speed
- Adjusted Heat Generation Function to Match Thermocouple Data

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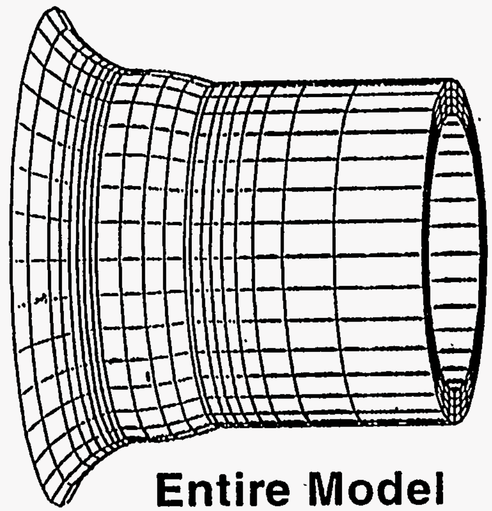
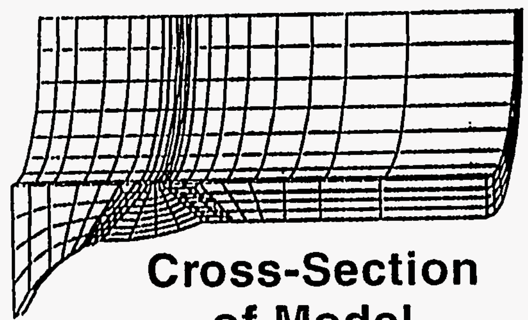
FIGURE 21

STRUCTURAL MODEL & BOUNDARY CONDITIONS

- 5056 Nodes, 4227 Hex/8 Elements
- Fixed Supports at Base
- Planar But Not Parallel at Pipe End
- Elastic/Plastic Temperature Dependent Material Properties
- Bi-Linear Stress-Strain Curves [70°F to 2100°F]
- Kinematic Material Hardening
- Root Gap Set Using 3 Tack Welds
- Deposited 7 Full 360° Passes



Entire Model



Cross-Section of Model

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FIGURE 22

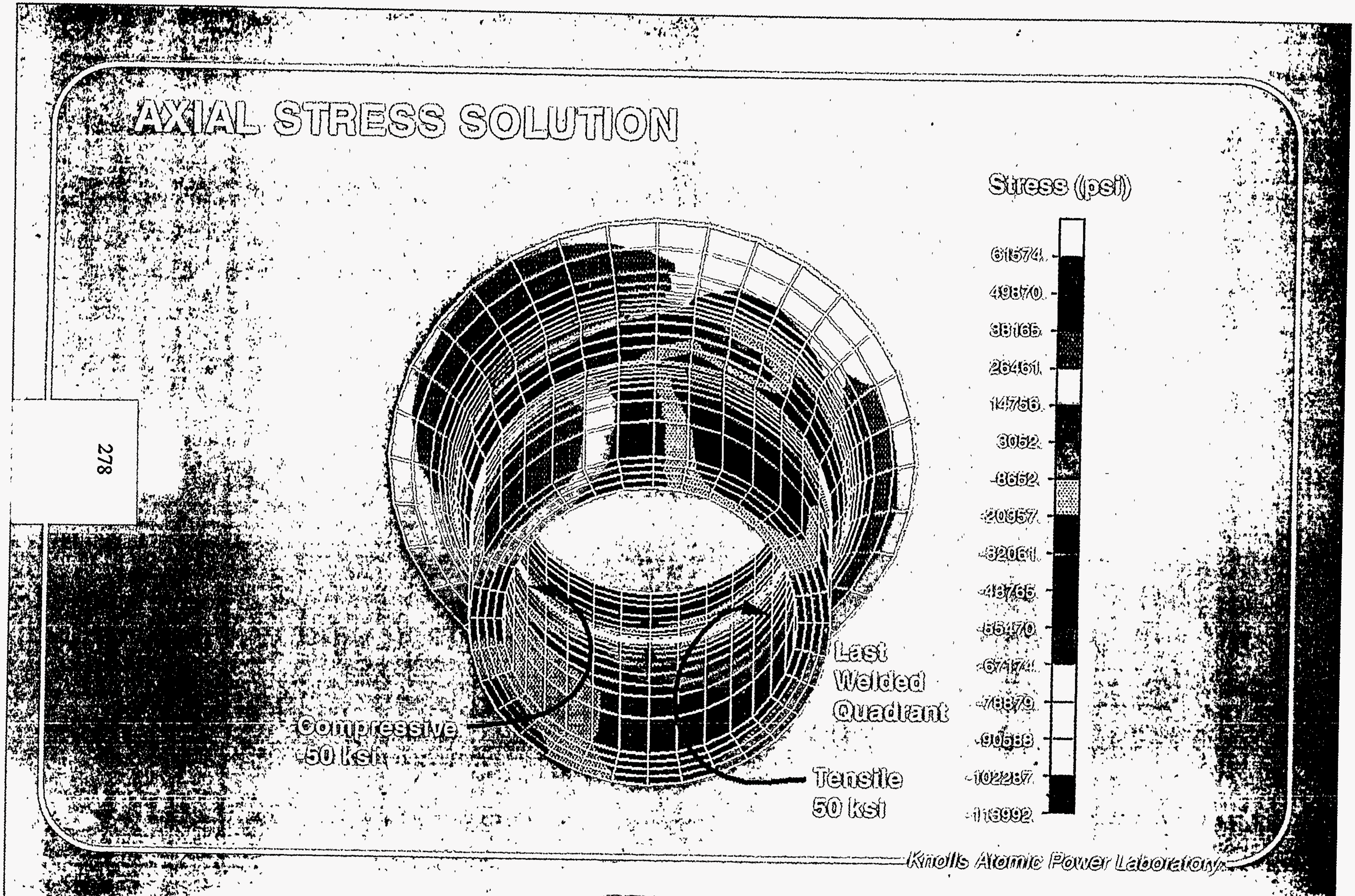
AXIAL STRESS SOLUTION
Stress (psi)
61574.
49870.
38165.
26461.
14756.
3052.
-8652.
-20357.
-32061.
-43765.
-55470.
-67174.
-78879.
-90583.
-102287.
-113992.
Last
Welded
Quadrant
Compressive
50 ksi
Tensile
50 ksi
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FIGURE 23

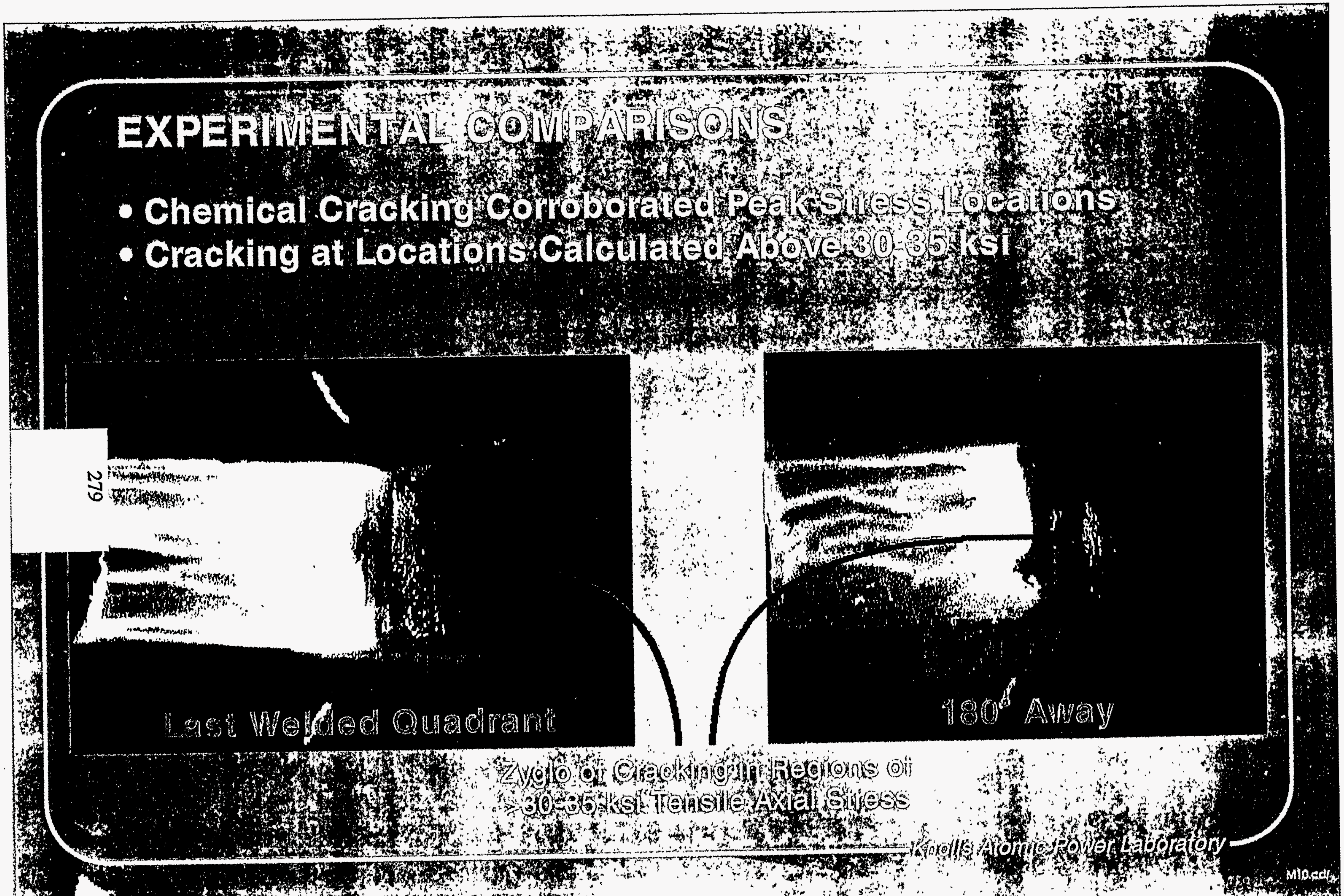
EXPERIMENTAL COMPARISONS
• Chemical Cracking Corroborated Peak Stress Locations
• Cracking at Locations Calculated Above 30-35 ksi
Last Welded Quadrant
180° Away
Zyglo of Cracking in Regions of
>30-35 ksi Tensile Axial Stress
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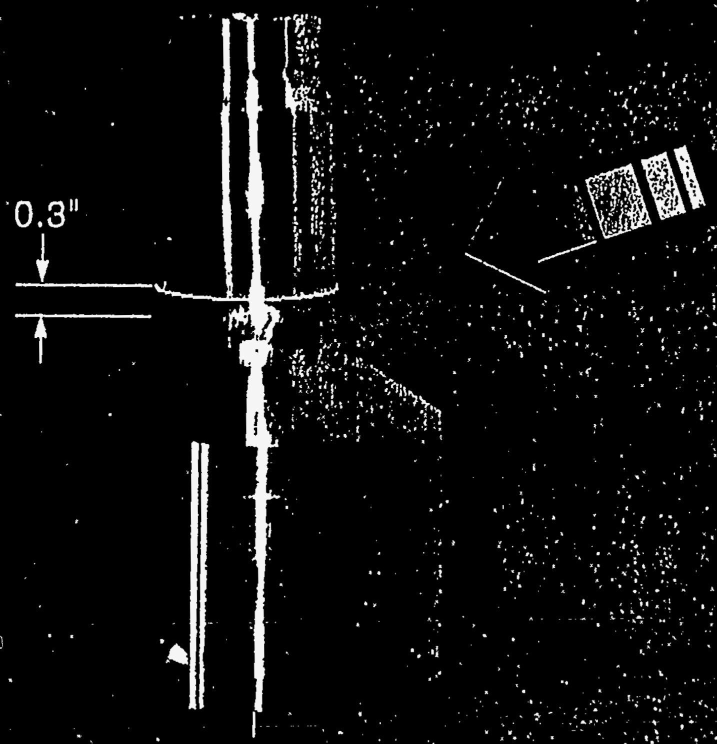
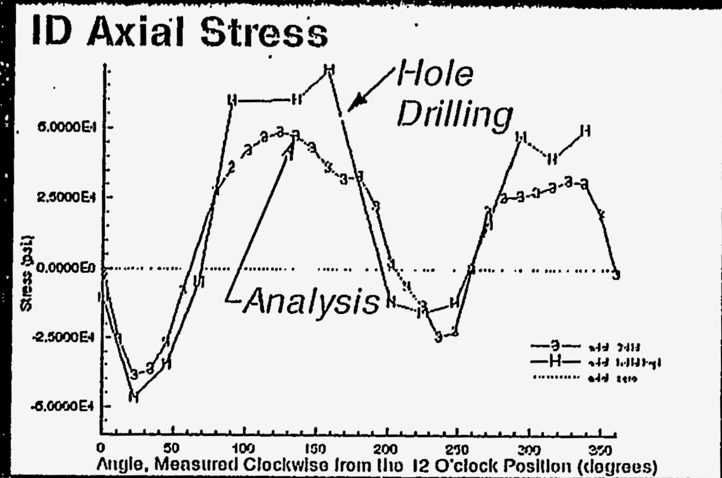
FIGURE 24

EXPERIMENTAL COMPARISONS (CONTINUED)

Hole Drill 16 ID & OD Locations:
- **Matched ID Cyclic Stress Variation**
- **Calculated Stress Magnitudes Well**



Hardware Mock-up



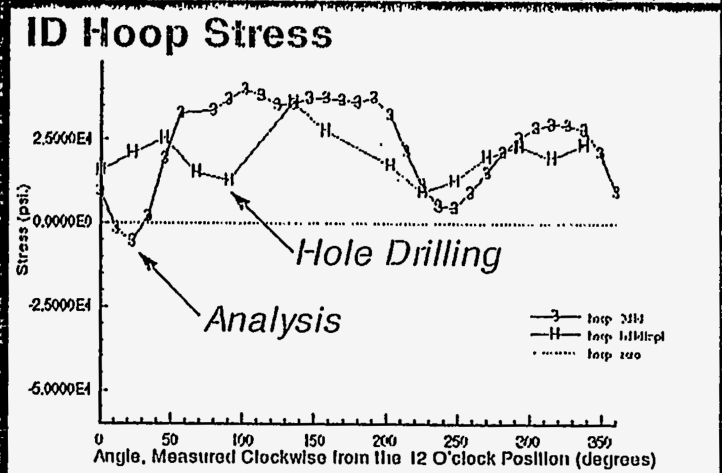


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FIGURE 25

COMPARE AXISYMMETRIC AND 3D SOLUTIONS

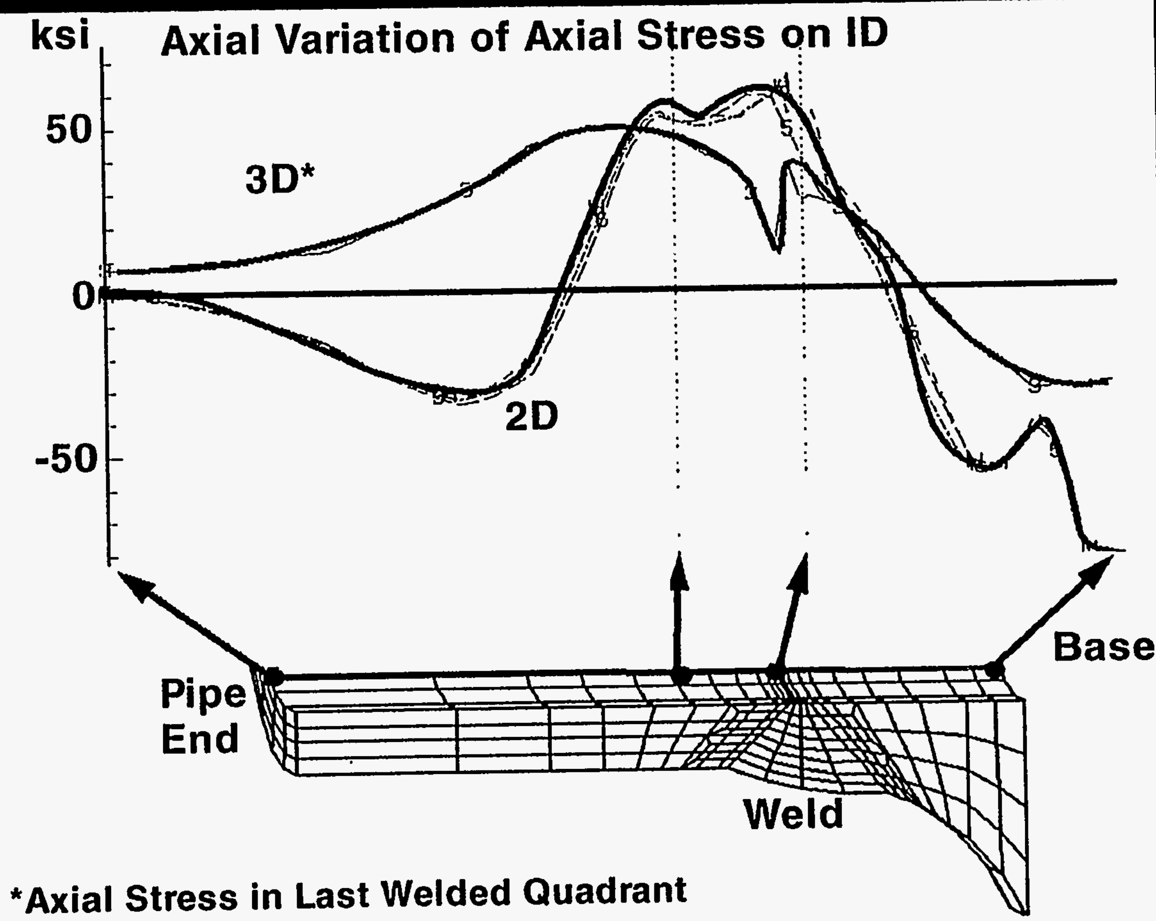

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FIGURE 26

CONCLUSIONS

- Good First Order Effects of Welding Process
- 3D Analysis Necessary for True Weld-Induced Stress Response
- Expect Tensile Stresses in Last Welded Quadrant for Thin Wall Weld
- Axisymmetric Analysis Overpredicts Peak 3D Stresses Near Weld, Less Agreement Away from Weld
- Through Wall Bending Assumption Not Valid for Stress Measurement Techniques

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FIGURE 27

HEAD PENETRATION CRACKING - COMPARISON OF THE BWR TO PWR EXPERIENCE

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San Jose, California

Head Penetration Cracking
Comparison of the BWR to the PWR Experience

This is a brief technical note intended to document the Boiling Water Reactor (BWR) perspective on light water reactor (LWR) vessel head penetration cracking that was informally presented at the IAEA Specialists Meeting on Cracking in LWR RPV Head Penetrations, Philadelphia, May 2-3, 1995. The intent of the presentation was to clarify the conditions and experience of BWRs relative to a phenomenon that has predominantly been associated with Pressurized Water Reactors (PWRs).

In general arrangement, there is considerable similarity between a PWR top head control rod drive penetration and a BWR bottom head control rod drive penetration. However, there are a number of differences in the details that lead to different experience and different conclusions regarding potential for and consequences of cracking of nickel base Alloy 600. First, the typical BWR bottom head penetration (see Figure 1) is installed such that cracking of Alloy 600 resulting in ejection of a control rod drive (CRD) housing is essentially impossible. As can be seen from the figure, the Alloy 600 is welded to the bottom head in the form of either a "set in" or "set on" stub tube. A stainless steel control rod drive housing is then welded to the Alloy 600 stub tube. With this arrangement even a full 360° circumferential through-wall crack in the Alloy 600 could not result in ejection because the stub tube is fully restrained by the bottom head. In addition, to back up this feature, BWRs have a permanently installed restraint structure underneath the bottom head that prevents both lateral and vertical movement of the CRD housings. Therefore, while leakage could occur in the event of crack, failure resulting in ejection of a housing cannot.

A second important point is that there are marked differences in the environment of a BWR bottom head relative to a PWR top head. Operating temperature is 288°C. The water is essentially deionized with a neutral pH having a conductivity typically less than 0.2 μS/cm and dissolved oxygen of approximately 300 ppb. The electrochemical potential in the BWR bottom head region is about +100 mV (SHE). These differences in environment relative to the PWR top head condition lead to considerably different behavior of Alloy 600 and its associated weld metals, both with respect to laboratory data and field experience.

It has been observed that in the presence of a highly stressed crevice in the BWR environment, Alloy 600 will suffer intergranular stress corrosion cracking (IGSCC).[1] However, in the absence of a crevice, Alloy 600 appears to be quite resistant to cracking regardless of heat treatment condition. In fact no cracking or leakage of Alloy 600 BWR bottom head penetrations has been observed to date. Likewise, no other instances of uncreviced Alloy 600 cracking in the BWR have been reported. The only bottom head penetration leakage reported for BWRs was related to IGSCC of either furnace sensitized Type 304 stainless stub tubes (there are only two units of this type) or one case of a weld sensitized in-core monitor housing (also Type 304) for which the stress corrosion susceptibility in the BWR is well documented.

With respect to Alloy 182 weld metal, the concern in the BWR is also somewhat the reverse of the PWR, again because of environmental conditions. While no cracking of Alloy 182 has been seen in PWRs, this material is known to be susceptible to stress corrosion in the BWR environment.[2] While no cracking of head penetrations has been observed to date, other Alloy 182 applications, e.g. reactor vessel nozzle weld butters, have been observed to be susceptible to cracking. However, again referring to Figure 1, cracking of Alloy 182 around a CRD stub tube, if it occurred, would at worst lead to leakage.

There is a general consensus that stress corrosion cracking of top head penetrations does not represent a reactor safety issue for PWRs. Owing to the differences in design and materials behavior cited above, this appears to be even more the case for BWR bottom head penetrations.

References:

1. K.S. Brown and G.M. Gordon, "Effects of BWR Coolant Chemistry on the Propensity for IGSCC Initiation and Growth in Creviced Reactor Internals Components", Third International Symposium on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, Traverse City, Michigan, August 30-September 3, 1987.

2. A. McMinn and R. A. Page, "Stress Corrosion Cracking of Inconel Weldments in a Simulated BWR Environment", International Symposium on Environmental Degradation of Materials and Nuclear Power Systems - Water Reactors, p. 108, ANS 1986.

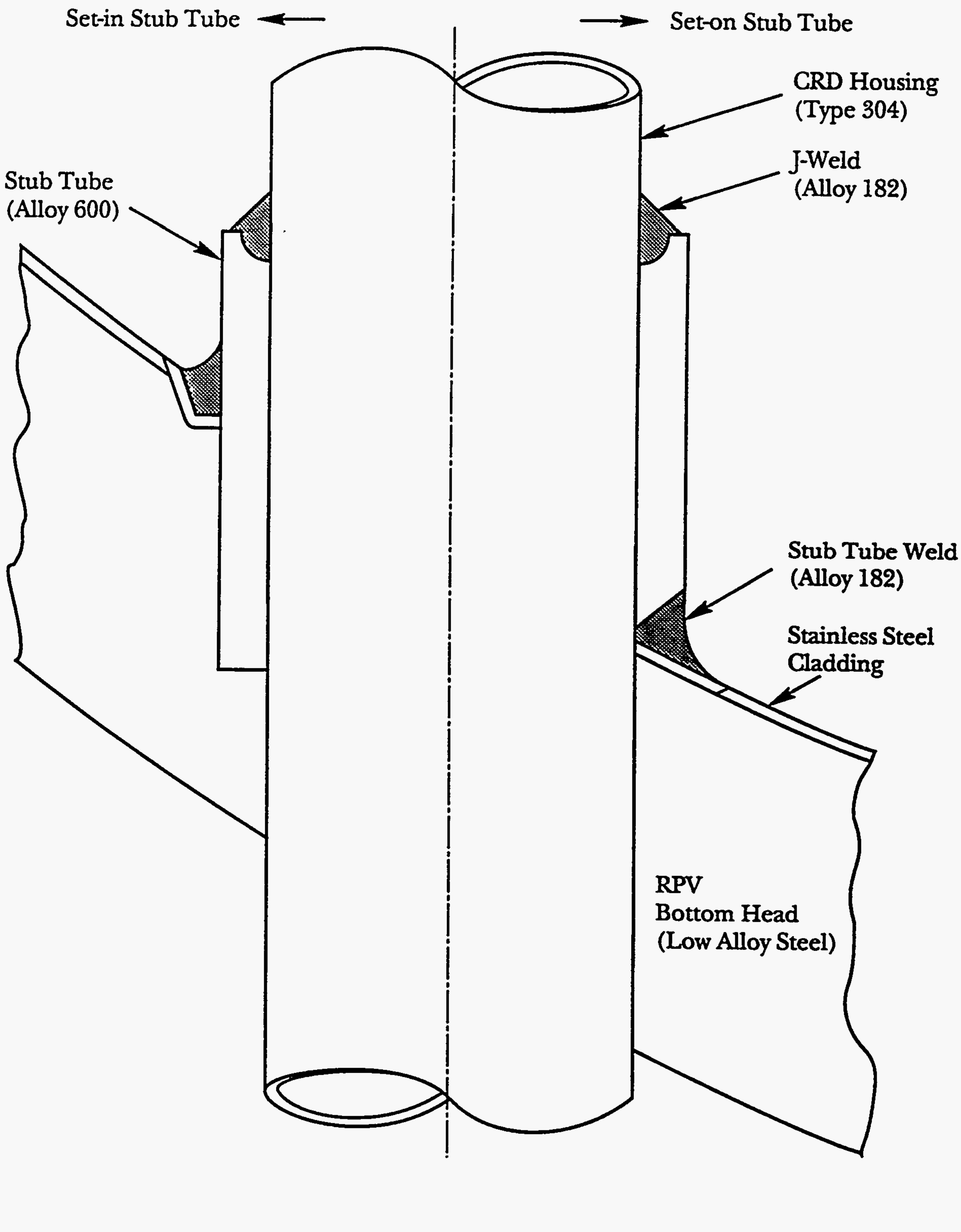

Figure 1. BWR CRD Penetration

CONCLUSIONS

AND

RECOMMENDATIONS

OPENING CEREMONY AND SESSION I

Chairmen: L.M. Davies and C.E. Pugh
Rapporteur: G.B. Heys

Summaries

Dr. Strosnider of USNRC briefly described the RPV head penetration cracking issue in PWRs. Topics of interest such as inspection programmes and methods, crack initiation and growth, repair criteria and mitigation were discussed. The current NRC view is that RPV head penetration cracking is not an immediate safety concern as cracks observed so far are axially oriented. However, it is prudent to implement an inspection programme.

Academician Davies, Consultant, discussed plant life management and suggested the use of the term "technical life" was most appropriate. This was defined as the period that a plant continues to meet its safety and performance criteria. Plant life extension was a most inappropriate descriptor. He also stressed the significance of separating plant specific and generic issues.

The activities of the IAEA International Working Group on Life Management of Nuclear Power Plants in 1994 and plans for 1995/1996 were described by Mr. Ianko of the IAEA. IAEA initiatives on the development of international databases on ageing were described.

Dr. Figueras of the Consejo de Seguridad Nuclear of Spain, presented the regulatory position on RPV head penetration cracking in Spanish PWRs. The results of RPV head penetration inspections at ALMARAZ, ASCO and VANDELLOS plants, where approximately 30 percent of the head penetrations were inspected showed no cracking. In contrast, at ZORITA-1 axial and circumferential cracks in Alloy 600 penetrations were observed. Cracking was related to the accidental introduction of sulphur species in the primary water due to resin intrusions in 1980 and 1981. The strategy for repair and surveillance was described.

Dr. Hermann described the USNRC approach to address the issue of PWSCC of Alloy 600 components in PWRs. PWSCC of Alloy 600 was identified to the Commission in 1989 as an emerging technical issue. The complex interacting variables in PWSCC results in difficulties in predictive modeling in Alloy 600 components. Dr. Hermann identified the need for continuing proactive NRC/industry programmes for inspection and repair or replacement of affected components.

Conclusions

1. The IAEA has a rolling programme of Specialists' meetings on ageing and plant life management. Initiatives on the development of international databases are progressing.

2. Nuclear Power Plant life may be defined in many ways. Mr. Davies suggested

that "technical life" - the period a plant continues to meet its safety and performance criteria - is an appropriate generic definition.

3. Cracking of RPV head Alloy 600 penetrations is not an immediate safety concern or plant life limiting, as cracks observed so far are axially oriented. However, it is prudent to implement inspection programmes to confirm this position.

4. Sulphur contamination of the primary circuit of PWRs can lead to enhanced SCC of Alloy 600. This was clearly demonstrated from the results from ZORITA in Spain where extensive axial and circumferential cracking were reported.

5. PWSCC of Alloy 600 components is an emerging complex technical issue for which predictive modelling is difficult. PWSCC of CRDM penetrations remains an open issue.

6. The nuclear industry has been proactive on this issue. The development of integrated inspection plans to determine inspection frequencies and repair techniques is essential.

7. There is a need to rank the susceptibility of primary circuit Alloy 600 components to PWSCC and to review inspection programmes to determine their adequacy in identifying potential degradation by PWSCC.

Recommendations

1. We recommend that interaction between regulator and licensee continues on this generic issue and that integrated inspection plans, which determine inspection frequencies and repair techniques, should continue to be developed and improved.

2. We suggest that the implications of the circumferential cracking observed at ZORITA be fully evaluated. Previous analyses should be reviewed to determine the potential for circumferential cracking in well controlled primary water chemistry.

3. There is a need to develop a strategy to improve mechanistically based probabilistic models to predict potential degradation by PWSCC of components manufactured from nickel-based austenitic materials. This should be used to confirm the strategies developed for these components. We suggest that this could best be achieved by Owners Groups or by international collaborative programmes.

4. We recommend that the scope of future IAEA Specialists' meetings on this issue should be broadened to cover SCC of nickel-based austenitic materials (excluding SG tubing).

SESSION II

Chairmen: M.B. McNeil and T. Takahashi

Some cracking problems in head penetrations are due to special problems (whether fabrication flaws or water chemistry), but there is a body of data on SCC of these penetrations. Cracking initiates and begins to propagate intergranularly and appears to be less common and less severe in penetrations fabricated from tubular products than in those fabricated from forgings.

A method has been developed (using H_2 doped steam) to test for vulnerability to this type of cracking. Testing and research activities are under way which may demonstrate why the tubular penetrations appear to be more resistant. One reasonable conjecture is that the different thermomechanical history of the tubular product produces a grain structure less vulnerable to intergranular SCC. This reasoning is supported by the observation that alloy 690 (which has a lower stacking fault energy and more twin boundaries) seems more resistant to crack initiation than alloy 600.

These analyses do not affect the short-term inspection/repair/replacement strategies, which appear to be dominated by economic and regulatory issues and consequently are very country-specific. Most strategies described involve either replacement, or welding or milling repair; little has been said about mitigation by plating and other surface treatments.

SESSION III

Chairmen: J. Strosnider and M. Brumovsky

The papers addressed the following areas related to cracking of reactor pressure vessel (RPV) head penetrations:

- Inspection Methods and Equipment
- Susceptibility Evaluations Considering Stress, Environment and Material
- Integrity Evaluations Including Crack Growth Rate, Acceptance Criteria
- Repair Methods.

The following major observations and conclusions were made for these areas:

Inspection Methods and Equipment

- Methods, e.g., Eddy Current, Ultrasonic, Dye Penetrant, Visual, for performing inspections are well developed.
    - Methods have been qualified through performance demonstration.
    - Detection is considered reliable and sizing accuracy is adequate to support integrity evaluation.
    - Methods have been field demonstrated (many plants/penetrations have been successfully inspected).
- Inspections can now be performed with due consideration of personnel exposure and without impacting critical path (even for a large number of penetrations).

Susceptibility Evaluations

- The factors affecting susceptibility; i.e., material, stress, environment are complex
    - Material product form and fabrication history can produce a wide variability in susceptibility (bar vs tube, heat treatment, etc.)
    - Stresses are complex (location in head, geometry, vendor/plant specific, cold working, repairs, fit up stresses etc.)
    - Environment can be critical (temperature, contaminants can reduce IGSCC stress threshold, etc.).

- Susceptibility factors compete with each other

    - Simple rules regarding which penetrations are most susceptible must be viewed with caution; although, the outer most penetrations appear to be generally more susceptible.

- Susceptibility and statistical models are important in developing inspection programmes and economic decision models

    - Need to be tempered by inspection experience

    - Better for assessing relative susceptibilities than predicting where or when cracking will occur.a

Integrity Evaluations

- Cracking in CRDM penetrations does not represent an immediate safety concern

- Repair criteria are easily established

    - Significant margins exist to structural integrity limits

    - Uncertainty in NDE and crack growth rates are understood well enough to support CRDM integrity evaluations (nonetheless continued evaluations of crack growth rate are valuable to assess conservatisms and for other applications)

Repair Methods

- Repair methods have been developed (removal of defect, capping of inactive penetrations, sleeving, head replacement, etc.)

- Are achievable remotely thereby reducing exposure.

Recommendations

1. Incorporate enlarged inspection database into susceptibility model as data become available.

2. Continue work to compare and understand differences in design, material, installation, environment and fabrication in U.S., French, German, Japanese and other designs for understanding susceptibility factors.

SESSION IV

Chairmen: W.H. Bamford and C. Faidy

There were two presentations on modelling of the PWSCC crack growth process, one from the point of view of a creep model and the second from the view of a global model of crack growth after the Andreson/Ford work. Both models seem promising, but are in the early stage of development for engineering application.

EdF presented an excellent summary and interpretation of the results of follow-up inspections of about 100 penetrations with cracks. There were some very interesting trends observed, after one re-inspection, with the growth rate tending to be higher for higher temperatures and larger initial cracks, but not very sensitive to set up angle or the location in the penetration. An important observation was that many of the flaws showed no additional propagation in follow-up inspections.

Recommendations

1. Continued international programmes to study the mechanisms of PWSCC are clearly needed, with the hope that a better understanding of the fundamentals would allow more confident predictions in the future.

2. It is important to continually be assessing the service experience in head penetrations, to refine crack growth models.

3. We must be aware of cases where cracking has been observed, as well as cases where cracking has not been found, or has been difficult or impossible to produce in the laboratory. A balanced approach to this issue is essential to avoid excess conservatism.

Overall Conclusion

CRDM cracking is not an immediate safety or technical plant life limiting concern. However, the problem needs to be monitored and remedied through well planned inspection and maintenance programmes. The inspection and remedial actions to support these programmes have been developed. Susceptibility models can provide valuable insights to establishing inspection and maintenance programmes based on both economic and safety considerations. However, their use should be tempered considering operating experience and the complexity of predicting susceptibility of specific penetrations. It is important to continually assess service experience and its implications for models used to assess IGSCC in CRDM penetrations. Their best application is for assessing relative susceptibilities and allocating resources.

The table below was an attempt to compare French, US Plant (Westinghouse B&W) and perhaps Japanese head penetration.

Recommendation

Agenda for a Future Meeting on the Subject

The initiation and early growth of stress corrosion cracks in Alloy 600 and related alloys. Topics to be examined should include:

1. Experimental data both on initiation and growth
2. Effects of alloy chemistry
3. Effects of microstructure
4. Relation of microstructure to thermochemical history.

A purpose of the meeting should be to identify sources of material - especially archival material from heats and to encourage metallurgical examination and the crosswalking both of microchemical/metallographic observations and cracking data.

Table 4.1 Comparison of Characterizing Parameters for RPV Head Penetrations in French, U.S. and Japanese LWR Plants

	French	US*	Japanese
Material	Forged bar stock with machined hole	Mostly tubular product/machined OD/IC	Tubular product machined OD/ID
Yield Strength	50-65 Ksi	30-64 Ksi	32-45 KSI
Heat Treatment	Lower annealing temperatures, with second heat treatment for some heats, based on yield strength	Various annealing temperatures - higher	Various annealing temperature - Higher than French
Surface preparation	Reaming/Machining	Some grinding performed for highly ovalized tubes	Some grinding performed after welding
Geometry	Some counterbores	Few counterbores	No counterbores
Plant operation	Load follow	Base load for all plants	Base load for all plants
Water chemistry	Nominally the same for French, U.S. and Japanese, but small variations may exist. This should be further checked.		
Weld design	per ASME Code Section III	per ASME Code Section III	per MITI Code (same with ASME Sec. III)

* Note that two US reactor vessels were fabricated in France.

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10. SUPPLEMENTARY NOTES

M.G. Vassilaros, NRC Project Manager

11. ABSTRACT (200 words or less)

This report contains 17 papers that were presented in four sessions at the IAEA Specialists' Meeting on Cracking in LWR RPV Head Penetrations held at ASTM Headquarters in Philadelphia on May 2-3, 1995. The papers are compiled here in the order they were presented in the sessions, and they relate to operational observations, inspection techniques, analytical modeling, and regulatory control. The goal of the meeting was to allow international experts to review experience in the field of ensuring adequate performance of reactor pressure vessel (RPV) heads and penetrations. The emphasis was aimed at better understanding of behavior of reactor component materials, to provide guidance and recommendations assuring reliability, adequate performance, and directions for further investigations. The international nature of the meeting is illustrated by the fact that papers were presented by researchers from 10 countries. There were technical experts present from other countries who participated in discussions of the results presented. The IAEA issued a Working Material version of the meeting papers (IAEA IWG-LMNPP-95/1), and this present document incorporates the final version of the papers as received from the authors.

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