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# Proceedings of the Conference on Vessel Penetration Inspection, Crack Growth and Repair

**Held at**
**Gaithersburg Marriott at Washington Center**
**Gaithersburg, MD**
**September 29 – October 2, 2003**

**Manuscripts**

**U.S. Nuclear Regulatory Commission**
**Office of Nuclear Regulatory Research**
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## ABSTRACT

These two volumes of proceedings contain the visual projections (in Volume I), and the contributed manuscripts (in Volume II) from the Conference on Vessel Head Penetration, Crack Growth and Repair, held at the Gaithersburg Marriott at Washingtonian Center on September 29 – October 2, 2003. The conference was co-sponsored by the U. S. Nuclear Regulatory Commission and Argonne National Laboratory. Over two hundred attendees were provided with 45 presentations, divided into five sessions: (I) Inspection Techniques, Results, and Future Developments, (II) Continued Plant Operation, (III) Structural Analysis and Fracture Mechanics Issues, (IV) Crack Growth Rate Studies for the Disposition of Flaws, and (V) Mitigation of Nickel-Base Alloy Degradation and Foreign Experience. The conference opened with a plenary session including presentations giving the overview from the NRC Office of Nuclear Regulatory Research, and an overview of nickel-base alloy cracking issues worldwide. The conference closed with a panel session consisting of industry representatives and NRC management discussing the prognosis for future issues in this area of concern.

## FOREWORD

Stress-corrosion cracking of nickel-base alloys used in both wrought and welded vessel penetration components has been an increasing and worldwide challenge for the nuclear industry and regulatory authorities since the mid-1980s. Cracks and resultant leaks were initially discovered in components fabricated from Alloys 600 and 182 exposed to higher temperatures, particularly in pressurizer heater sleeves and nozzles. Over time, cracks and leaks have also been discovered in components operating at lower temperatures, including vessel head and bottom-mounted instrumentation penetrations.

Given the safety-significance of this issue, the U.S. Nuclear Regulatory Commission (NRC) hosted a 4-day conference on September 29 – October 2, 2003, to provide a forum for presentations and discussions concerning inspection, stress analysis, flaw evaluation, and mitigation of stress-corrosion cracks in vessel penetrations. This conference also provided a valuable opportunity for participants from several venues — regulatory, research, and plant operations — to meet face-to-face to formally and informally exchange data and concepts with the individuals who are at the forefront of the cracking issue.

As such, the conference brought together much of the worldwide expertise in the area of nickel-base alloy cracking. More than 200 individuals attended the 4-day conference, which included 45 presentations that provided a wide-ranging perspective on the issue. Many of the presentations were prepared by researchers involved in crack growth rate studies and nondestructive inspection; those presentations described successes and difficulties in developing testing and inspection procedures. Several discussed the stochastic nature and statistical analysis of cracking incidents, predictive algorithms for this type of degradation, and the prognosis for the future, including head replacement strategies, mitigation of the cracking process, and the likelihood of increased resistance to cracking of the replacement materials (Alloys 690 and 152). Other presentations were prepared by reactor component vendors, utility representatives, and regulatory participants, who described plant responses to component degradation, structural integrity evaluation, or the repair and mitigation of cracking. Many of those presentations were marked by completeness and candor in the discussion of observed problems and the related solutions. In addition, several presentations described the experiences of non-domestic institutions, providing contrasts and alternative approaches to the same problem.

The complete proceedings package consists of all conference presentations and available manuscripts, in both printed and electronic formats. The broad, public distribution of the proceedings ensures that the presentations will be subject to the greatest possible scrutiny and accreditation. As a result, the conference organizers believe that these proceedings will give readers an overview of the current status of inspection technology and crack growth rate studies, as well as an understanding of reactor safety and the economic impact of the degradation of nickel-base alloys on plant operation.

Carl J. Paperiello

Carl J. Paperiello, Director
Office of Nuclear Regulatory Research

## CONTENTS

ABSTRACT ..... iii

FOREWORD ..... v

EXECUTIVE SUMMARY ..... xi

ACKNOWLEDGEMENTS ..... xv

Welcoming Address
A. C. Thadani, U.S. Nuclear Regulatory Commission, Rockville, MD ..... 1

Keynote Address: What are the Issues?
K. Gott, Swedish Nuclear Power Inspectorate (SKi), Stockholm, Sweden ..... 7

Session I: Inspection Techniques, Results, and Future Developments ..... 11
Chairpersons: S. Doctor, Pacific Northwest National Laboratory
D. Jackson, U.S. Nuclear Regulatory Commission

NDE of Austenitic Materials-A Review of Progress and Challenges ..... 13
F. Ammirato

Inspection Reliability of Reactor Vessel Head Penetrations ..... 15
S. R. Doctor and G. J. Schuster

The Evolution of Inspection and Repair Approaches for Reactor Vessel Head Penetrations ..... 25
M. Hacker, D. Schlader and D. Waskey

Inspection Techniques, Results and Future Developments: Summary of U.S. PWR Reactor Vessel Head Nozzle Inspection Results ..... 27
G. White L. Mathews, and C. King

Inspection Technology for In-Core Penetrations ..... 43
M. Lashley and S. W. Glass

EPRI MRP Alloy 600 RPV Head Penetration Inspection Demonstration Program ..... 49
T. Alley and K. Kietzman and F. Ammirato

Production of a Realistic Artificial Flaw in an Inconel 600 Safe-End ..... 51
M. Kemppainen, I. Virkkunen, J. Pitkänen, K. Hukkanen and H. Hänninen

Generic Guidance for an Effective Boric Acid Inspection Program for PWRs ..... 61
G. Rao, T. Satyan Sharma and D. Weakland

Risk-Informed Evaluation of PWR Reactor Vessel Head Penetration Inspection Intervals ..... 63
G. White, S. Hunt, and N. Nordmann

Reactor Vessel Head Penetration Inspection Technology, Past, Present and Future ..... 79
J. P. Lareau, D. C. Adamonis, S. C. McKinney, R. P. Vestovich, M. W. Kirby, R. S. Devlin and M. Melbi and F. D'Annucci

Session II: Continued Plant Operation ..... 81
Chairmen: A. Hiser, Jr., U.S. Nuclear Regulatory Commission
W. Bamford, Westinghouse Electric Company, LLC

Background and Technical Basis for the ASME Section XI Process for Evaluation of Upper Head Penetration Flaws ..... 83
W. H. Bamford and G. DeBoo

Maintenance Strategy of the Inconel Components in PWR Primary System in France ..... 85
F. Champigny, F. Chapelier and C. Amzallag

Strategic Planning for RPV Head Operation After Repair or Mitigation ..... 95
S. Hunt

Reactor Vessel Bottom-Mounted Instrumentation (BMI) Nozzle Repair Development and Implementation at South Texas Project ..... 103
S. Thomas and R. Payne, D. Schlader, and D. Waskey.

South Texas Project Experience with Alloy 600 Bottom-Mounted Instrument Penetration Cracking ..... 105
S. Thomas

Development and Justification of a Repair Process for Flaws in Reactor Vessel Upper Head Penetrations ..... 109
W. H. Bamford and P. Kreitman and P. R. Evans

Weld Overlay Deposit on Alloy 82/182 Butt Welds to Reduce ID Surface Stresses ..... 111
S. Hunt and J. Broussard and P. O'Regan

Session III: Structural Analysis and Fracture Mechanics Issues ..... 121
Chairman: G. Wilkowski, Engineering Mechanics Corporation of Columbus

Probabilistic Fracture Mechanics Analysis to Support Inspection Intervals for RPV Top Head Nozzles ..... 123
P. C. Riccardella, N. G. Cofie and G. A. Miessi

Parametric Studies of the Probability of Failure of CRDM Nozzles ..... 147
W. J. Shack

Analysis of Weld Residual Stresses and Circumferential Through-Wall Crack K-Solutions for CRDM Nozzles ..... 161
D. Rudland, Z. Feng, Y.-Y. Wang, R. Wolterman, and G. Wilkowski

Welding Residual and Operating Stress Analysis of RPV Top and Bottom Head Nozzles ..... 187
J. Broussard and D. Gross

Predicting the First Failure for LPSCC ..... 201
R. W. Staehle

Elevated Temperature Grain Boundary Embrittlement and Ductility Dip Cracking of Nickel-base Weld metals ..... 239
J. Lippold and A. J. Ramirez

Impact of PWSCC and Current Leak Detection on Leak-Before-Break ..... 249
D. Rudland, R. Wolterman, and G. Wilkowski

Session IV: Crack Growth Rate Studies for the Disposition of Flaws ..... 251
Chairpersons: K. Gott, SKI, Sweden
T. Shoji, Tohoku University, Japan

US Regulatory Experience and Prognosis with RPV Head Degradation and VHP Nozzle Cracking ..... 253
A. Hiser, Jr.

Stress Corrosion Crack Growth Rate (SCCGR) Data for Alloy 182 and 82 Welds ..... 267
S. Attanasio, J. V. Mullen, J. W. Wuthrich, W. W. Wilkening and D. S. Morton

Influence of Temperature on Primary Water Stress Corrosion Cracking of Alloy 600 Weld Metals ..... 297
Y. Nishikawa

Experimental and Numerical Approaches for Characterizing the Crack Growth Behavior of Alloy 600 in PWR Primary Water, and Lifetime Predictions for Welded Structures ..... 309
T. Shoji, Z. Lu and Q. Peng

HAF Model Predictions for the Sensitivities of Alloy 600 LPSCC to Stress, Temperature, Carbon Concentration, Cold Work and Crack Growth Orientation ..... 337
M. M. Hall, Jr. and W. C. Moshier

SCC Growth Rate of Nickel Based Alloy Weld Metals in PWR Primary Water ..... 351
K. Tsutsumi, H. Kanasaki, K. Yoshimoto, Y. Nomura, S. Asada and T. Yonezawa

The Stress Corrosion Crack Growth Rate of Alloy 600 Heat Affected Zones Exposed to High Purity Water ..... 371
G. A. Young, N. Lewis and D. S. Morton

In-situ Raman Spectroscopic Observation of Ni/NiO Equilibrium in PWR Primary Water Condition ... 387
J. H. Kim and I. S. Hwang

Results of Accelerated SCC Testing of Alloy 82, Alloy 182, and Alloy 52M Weld Metals ..... 413
J. Jacko, R. E. Gold, G. V. Rao K. Koyama and A. Kroes

Metallurgical Investigation and Root Cause Assessment of the Reactor Vessel Head Penetration Alloy 182/82 J-groove Weld Cracking in Rotterdam Vessels ..... 427
G. Rao J. Bennetch, G. Embring and W. Gahwiller

The French Regulatory Experience and Views on Nickel-Base Alloy PWSCC Prevention ..... 429
P. Chartier, D. Emond and G. Turluer

Session V: Mitigation of Nickel-base Alloy Degradation and Foreign Experience ..... 449
Chairmen: S. Fyfitch, Framatome ANP, USA
P. Scott, Framatome ANP, France

Belgian activities on alloys 600 and 182 issues ..... 451
R. Gérard and Ph. Daoust

German Experience with Vessel Head Penetrations ..... 465
R. Bouecke

Activities on Alloy 600 Cracking of PWRs in Japan ..... 477
M. Toyoda, S. Asada and N. Hori

RPV Heads: Inspection vs. Exchange Strategies for Ringhals PWR Units 3 and 4 ..... 485
P. Efsing

Alloy 600 PWSCC Mitigation: Past, Present, and Future ..... 493
S. Fyfitch

A Reactor Vessel Upper Head Temperature Reduction Program to Prevent and Mitigate Alloy 600 Cracking ..... 509
D. R. Forsyth and P. L. Strauch

Status Report on the Effect of Zinc Additions on Mitigation of PWSCC in Alloy 600 ..... 523
J. Hickling, R. Pathania and C. Wood

An Assessment of MSIP as a PWSCC Mitigative Technique by Instrumented Monitoring of ID Strains During the Full Scale Mock Up Demonstration Test of Pressurizer Nozzle Weld at Tihange Unit 2 Station ..... 533
G. Rao

## EXECUTIVE SUMMARY

The purpose for this conference was to examine the current state of technology for vessel head penetrations with respect to inspection, cracking, and repair. This subject is being examined because of penetration cracking which has been occurring for over a decade. The first reactor head penetrations to show signs of leakage occurred in France in the early 90's at Bugey 3. After this incident the French inspected a large number of their penetrations and reported that roughly 3% of their inspected nozzles had some type of indication. Because of the cracking in France, many power plants in the US and elsewhere started to examine penetrations and found ultimately that a large number were similarly cracked. The next leakage from a vessel head penetration occurred in the United States at Oconee 3 in 2000. Following Oconee there have been many other plants with cracked or leaking penetrations. This type of degradation led to one of the most serious nuclear incidents in the U.S. at Davis Besse. A crack in a vessel head penetration, possibly combined with the presence of substantial boric acid deposits, led to corrosion of the low-alloy steel, and the formation of a large cavity in the reactor head. Another significant event included the first leaking bottom mounted instrument penetrations discovered at the South Texas Project Plant in the United States. These instances of failure are a concern to the public, industry, and regulators. Knowledge gained from this conference will help reduce future incidents from occurring. The five sessions listed below were held at the conference and covered several topical areas.

Session I: Inspection Techniques, Results, and Future Developments
Session II: Continued Plant Operation
Session III: Structural Analysis and Fracture Mechanics Issues
Session IV: Crack Growth Rate Studies for the Disposition of Flaws
Session V: Mitigation of Nickel-base Alloy Degradation and Foreign Experience

The first session examined the area of inspection techniques for the vessel head penetrations. This is important research, since inspection capability is one of the first lines-of-defense against vessel head penetration leakage. A range of topics were discussed including how nondestructive examination (NDE) has evolved over time. Advancements in NDE were examined which included Phased Array Ultrasonic Testing and Eddy Current Testing Arrays. With regards to the area of NDE testing tools, cracked penetration mockups and performance demonstrations were discussed. This included examining new techniques for developing realistic flaws. The issue of reliability of NDE data was another topic of concern. This led into presentations about in-service inspections (ISI). One main area of discussion for ISI is the frequency of inspections. One question that was raised asked what should be the bases for determining the inspection frequency. Should the ISI be based on avoiding any leakage at a plant or should it be based on avoiding core damage? The discussion of inspection techniques carried over to the next session of Continued Plant Operation.

The second session examined Continued Plant Operation, and one of the first presentations examined the analytical and repair approaches for continued plant operation. Included in this session was a description of the cracking which occurred at South Texas Project in the bottom mounted instrument (BMI) nozzles. The repair techniques for these bottom mounted nozzles were discussed in detail. With regards to upper head penetrations, there is an understanding that evaluation methods are being developed and will be included in section XI of the ASME code sometime in 2004. The French discussed the initial leak at Bugey and investigations which followed. In France it was determined that the best choice of action was to replace the vessel heads with Alloy 690 nozzles and Alloy 152 weld material. The subject of how power plants in the United States have reacted to the nickel-based material cracking issues was similarly covered. In examining how to operate after repair or mitigation, taking into consideration cost and downtime, the optimum solution to this problem was to reduce the reactor vessel head temperatures. There were two repair techniques presented, which included embedded flaw repair and weld overlay

repair. In determining the acceptable usage for these two repair techniques, structural analysis must be taken into consideration, a discussion which provided a segue into the next session.

Structural Analysis and Fracture Mechanics Issues was the title of session three. The initial presentations focused on using probabilistic analysis to determine the probability that the head penetrations will either leak or be ejected. It seems that through this type of analysis, in conjunction with reasonable inspection plans, the top heads meet the safety limit for nozzle ejection. However, there are conservatisms still inherent in these calculations. The next topic focused on residual stresses present in the nozzle and how they may affect cracking. There are different variables that need to be considered to determine accurately the hoop and axial residual stresses. Some of these variables are nozzle thermal properties, welding procedure, joint configuration, and mechanical properties. The research presented suggests that residual hoop stresses are larger then the residual axial stresses. In the peripheral nozzles the stresses will depend upon the location in the nozzle, with respect to the downhill or uphill side. Specifically, as the weld height increased the axial stresses decreased while the hoop stresses increased. A logical consequence is that some type of medium weld height might be used in order to achieve a balance between both hoop and axial residual stresses. The session included discussion on the subject of ductile-dip-cracking, which seems to be a much larger problem for alloys 152/52 then it is for alloy 182/82. There was also some examination of the leak before break (LBB) concept. Initial LBB calculations utilized cracks which were more characteristic of fatigue cracks than PWSCC cracks. A reanalysis of LBB using PWSCC crack geometries leads to some new results. The presentation noted that it is difficult to satisfy LBB criteria using the PWSCC crack geometries. Another feature is that PWSCC could result in long circumferential surface cracks which may be more prone to failure than than the currently-utilized, simple, through-wall circumferential crack. The LBB screening criteria is not satisfied by this type of circumferential cracking. Finally, the last subject in this session examined the subject of predicting first failure by creating an all inclusive equation. This equation would predict failure by using past experience as a guide. Auxiliary equations would take into consideration variables such as temperature or stress, which affect failure. These small individual equations would be combined to create an overall cracking equation. However, this work is still in the beginning phase of development.

The fourth session of the conference was titled Crack Growth Rate Studies for the disposition of flaws. This is a very important subject because crack growth rates can be used to predict when an identified crack will lead to leakage of reactor coolant solution. A discussion of the history of Alloy 600 cracking at plants in the United States and France was followed by a description of new testing techniques for stress corrosion cracking growth rates (SCCGR). This description included the design details of compliant, self-loaded compact tension (CT) specimens and the conduct of accelerated crack growth tests with a clearly-defined acceleration factor. With regards to SCCGR evaluation procedures, there was discussion about using a maximum or average SCCGR. There was also discussion about the pros and cons of periodic unloading for more continuous crack tip activation. The next subject covered examined the SCCGRs for the materials such as Alloy 600, 182, 152, 132, 82, and 52. The conclusion is that SCCGR for alloy 182 is larger then alloy 82. Alloy 132 has a SCCGR on the order of Alloy 182 SCCGR. The crack growth rates in the heat affected zone (HAZ) in Alloy 600 may be 30 times larger then the non-HAZ material. Alloy 52M has been tested but no cracking was found in this material. In service, an alloy 182 weld with 5-10 effective full power years (EFPY) cracked. Alloy 600 showed cracking in a material with 6-13 EFPY. The participants discussed the effect of dissolved hydrogen on SCCGRs in this session. There was agreement that the SCCGRs are maximized when exposed to electrochemical conditions around the Ni/NiO equilibrium line on a Pourbaix diagram. Another subject covered was models for SCCGRs. The physical and mechanical-chemical models discussed are useful tools that can be utilized to examine the SCCGR inter-workings. The combination of models with the SCCGR data should provide a more accurate assessment of SCCGR curves.

The final session for the conference examined Mitigation of Nickel-Base Alloy Degradation and Foreign Experience. During previous sessions the experiences from both the United States and France had been presented. This session allowed other countries affected by the same degradation to present the issues occurring in their country. This foreign experience included presentations from Belgium, Germany, Sweden, and Japan. In Belgium, a proactive approach has been taken to repair, replace, or mitigate any alloy 600 cracking before leakage occurs. In Germany, the Obrigheim power plant is the only plant in that country which contains Alloy 600 in the reactor vessel head penetrations. As a result, Obrigheim uses leakage detection systems. In Japan, reactor heads were replaced in older plants, while newer plants have lowered the reactor vessel head temperature. Minor indications in the bottom mounted instrument nozzles have also been discovered in Japan. Sweden plants replaced the reactor vessel heads. The next subject of this session was mitigation techniques for nickel-based alloy degradation. One of the main directions industry is headed is to replace Alloy 600 parts with Alloy 690. Other then replacing the material, there are three ways to alleviate degradation. These mitigation strategies are mechanical surface enhancement, environmental barriers or coatings, or changes to the environment. The geometry of the component influences the choice of a particular strategy. One type of mitigation technique that has been employed is to reduce the head temperature of the vessel. This has the effect of reducing the rate of increase of effective degradation years. Another mitigation technique which is being tested is low-level zinc additions to the primary coolant. There has been some evidence that zinc reduces the initiation time for PWSCC. However, there is less evidence that zinc additions reduce the PWSCC crack growth rate. The last mitigation technique discussed was the mechanical stress improvement procedure (MSIP). MSIP has been demonstrated on thick walled PWR piping. The results from this demonstration show that compressive stresses are formed in the inner weld region and that the profile of the pipe after MSIP is still acceptable for in-service inspections.

## ACKNOWLEDGEMENTS

The bulk of the arrangements for this conference were handled by the Conference Office at Argonne National Laboratories. Ms. J. Brunsvold of that office negotiated with the lodging and conference staffs at the Marriott/Gaithersburg – Washingtonian Center, and arranged for a trouble-free four days of conferencing. The assistance of the staff at the Conference Center was managed by K. Natesan, Argonne National Laboratory, who is the program manager for several NRC-funded programs. Speaking for all the attendees, we appreciate the hard work of both Ms. Brunsvold, and Mr. Natesan.

Actually, Ms. Brunsvold and Mr. Natesan planned the logistics for this conference twice. The conference was originally scheduled for September 24, 2001, which turned out to be less than two weeks following the attacks of Sept. 11. The conference was successfully postponed, thanks to the efforts of the Argonne staff and the gracious understanding of the Marriott-Gaithersburg conference office staff. All participants were successfully contacted; not one person showed up at the venue expecting a conference to actually take place.

The organization of the sessions and contributing authors was handled by Ms. Karen Gott, Swedish Nuclear Power Inspectorate (SKi), who was participating in a rotational assignment at the NRC in the months preceding the original conference. We appreciate her hard work and diligence in this effort was extremely helpful in getting this conference off the ground successfully. At the rescheduled conference, Dr. T. Mintz was indispensable in his seemingly effortless handling of the audiovisual logistics during the course of the conference, and with his able assistance in the preparation of the proceedings. The registration and logistics table at the conference site was staffed by Ms. Brunsvold, together with Ms. C. Briggs and Ms. E. Miller, both of the NRC's Office of Nuclear Regulatory Research. Their contribution is much appreciated. T. Mintz also assisted in the compilation, tabulation and electronic linking of the written and electronic versions of the proceedings.

W. Cullen, NRC Program Manager

# WELCOMING ADDRESS

By. A. Thadani, Director[1]
Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission

## Importance and Status of Nickel-Based Alloy Research For Nuclear Reactor Safety

Good morning, ladies and gentlemen! It is my great pleasure to welcome you to this four-day conference on the inspection, crack growth and repair of nickel-based alloys used in vessel penetrations for pressurized water reactors. Events at several plants around the world over the last several years related to the cracking of key pressure boundary components have caused us to focus again on the potential for environmentally assisted cracking to challenge the safety of nuclear power plants. Many of us worked through the challenges brought about by Intergranular Stress Corrosion Cracking in the Boiling Water Reactors in the 1970's and 1980's, and we continue to deal with challenges with degradation of steam generator tubing. Recent cracking events in the PWR fleet have caused us to mobilize to, once again, understand and resolve a challenge to the safe operation of nuclear power plants.

Cracking of pressure boundary components challenges one of the key elements in the defense-in-depth philosophy. The defense-in-depth approach begins with the fuel cladding as the first barrier to the release of fission products to the environment, and builds on the integrity of the pressure boundary, and ultimately the containment building to protect the public and the environment. A challenge to any one of the barriers becomes a challenge to safety and is something that we have always worked hard to address. The number of scientists and engineers attending this conference attests to the emphasis being placed on addressing this new challenge by the international community.

Activities related to cracking of nickel-based alloys in U.S. light water reactors gained momentum with the V. C. Summer hot leg safe-end incident and the detection of axial and circumferential cracks in several reactor vessel head penetrations and associated J-groove welds. Currently, we are closely following the observations of leakage from bottom-mounted instrumentation nozzles as another, potential instance of Alloy 600 cracking.

Stress corrosion cracking of these alloys in pressure boundary applications was first reported in the middle 1980's and the French were the first to report control rod drive mechanism penetration cracking in 1991. Cracking of the control rod drive mechanism (CRDM) nozzles is important from two major aspects. First, axial or circumferential head penetration cracking, if not detected, may lead to leakage of primary coolant. This has in the past, and could again result in boric acid corrosion damage of the low alloy steel components of the pressure boundary. Secondly,

[1] At the time of the conference, Dr. Thadani was the Director of the Office of Research. In May, 2004, Dr. Paperiello became the head of this office.

extensive circumferential cracking may lead to control rod drive mechanism ejection, resulting in a more serious event.

Probabilistic risk analyses have shown that failure of a vessel head penetration does not pose undue risk of core damage. Conversely, failure of a bottom mounted instrumentation nozzle could more readily result in a core damage accident. However, in the U.S. we use the risk assessments to inform our decisions but we also take into account the need to preserve the integrity of the pressure boundary and the integrity of the defense-in-depth philosophy. This has led us to take an aggressive regulatory posture, where we have required inspections and repairs of vessel head degradation. Many U.S. plants are now replacing their pressure vessel heads to address this issue. However, nickel-based alloys are used in many other applications in nuclear power plants. Further, our experience with the replacement alloys is positive but, as yet, somewhat limited. Thus, it is important to continue to pursue a full understanding of the environmentally assisted cracking of these materials.

Environmentally assisted cracking is a multi-faceted problem, requiring a multi-faceted solution. Key elements of the solution include:

- inspection practices to detect and characterize cracking before it challenges component integrity;
- structural integrity assessments, which includes understanding how the cracks initiate and growth and having the validated analytical tools models necessary to predict critical crack sizes and inspection frequencies needed to preclude inservice failure; and
- mitigation and repair strategies that have been tested to ensure they are effective and do not create new challenges.

Of course, these are 'generic' elements in the solution to any cracking problem. So, not surprisingly, as we talk about specific problems with the cracking of nickel-based alloys and welds, they are the specific elements we are discussing at this conference. The presentations will also describe industry experience focused on vessel head penetration issues and results from in-service inspection programs from around the world.

Since over eighty percent of the world's nuclear power plants are based on light water reactor technology, full utilization of available operating experience is an vital factor in addressing the stress corrosion cracking issues. The Davis-Besse vessel head degradation incident emphasized the important role that operating experience plays in fulfilling our safety responsibilities. It is incumbent on us all to diligently review these matters and apply proper focus on any recurring material issues. As part of the lessons learned from Davis-Besse, the NRC is examining how it can more effectively utilize operating experience. Strong implementation of operating experience programs can be very beneficial in the resolution of material degradation issues before they lead to significant safety problems. As I mentioned previously, this conference is an excellent opportunity for us to exchange valuable experience from around the world.

The NRC's research program has been addressing environmentally assisted cracking in various forms for many years. These efforts have included issues related to crack detection, crack initiation, and crack growth for use in assessments of operating life and safety margins. We have

also conducted research relating to primary water stress corrosion cracking of steam generator tubes, and more recently, of thicker sections of nickel-based alloys. Since 1996, the latter efforts have been focused on control rod drive mechanism (CDRM) penetrations and other primary pressure boundary components. NRC research activities have also included nondestructive test methods and procedures, structural integrity analysis, fracture mechanics, corrosion, and probabilistic risk assessment. The discovery of cracks, including some instances of leakage, associated with control rod drive mechanism penetrations and safe-ends has necessitated the continuation and expansion of these research programs.

Although studies have been conducted on cracking and crack growth, our ability to integrate stress analysis into the prediction of crack growth in reactor components is limited. Ongoing research programs are expected to improve our ability to characterize the stress conditions, including residual stress and other parameters such as crack distributions. This should enable us to more accurately predict susceptibility to primary water stress corrosion cracking. Additionally, we need more information related to the component condition regarding such parameters such as strength and microstructure for predicting cracking in the components. This information could be used to improve the susceptibility algorithm addressing the frequency of inspections and the adequacy of mitigation techniques. Research is also needed on improved welding procedures to reduce residual stresses and thus the susceptibility to stress corrosion cracking in repaired or replacement components.

The number and severity of cracks in vessel penetrations and other pressure boundary components has emphasized the need for the NRC and the nuclear community to develop and exchange information on Alloy 600, Alloy 690, and their associated weld metals, and inspection practices. The scope and magnitude of this need was recently highlighted by the Davis-Besse event. As a consequence of this event, and the NRC's assessment of the event and our own performance, we have directed additional financial and human resources to the issue of degradation of nickel-based alloys. This conference is just one of many manifestations of the level of attention this issue commands.

The NRC research activities have, for the most part, been focused on generic issues and are intended primarily to provide the basis for future regulatory actions. Industry-sponsored research on these subjects, has often been developmental, innovative, and intended to address a specific plant or vendor issue. It is important to note that there has been extensive information exchange and collaboration through NRC-Industry coordinated research efforts, for example, with EPRI and the Materials Reliability Program (MRP). We understand that the vessel heads on at least 11 U.S. plants will be replaced during the next 3 years. The old head materials present an enormous potential for gathering additional information to help the understanding of the initiation and growth of cracks and leakage. It is vitally important that these productive collaborative efforts continue and are expanded to include even more of the international community. This should lead to a better understanding of crack growth rates, stress analysis, and identification of mitigation strategies for stress corrosion cracking in nickel-based alloys.

Increasing our understanding of the technical details of this degradation mechanism, and other potential degradation mechanisms yet to be observed, has become paramount in assuring safe continued operation of nuclear power plants. A better understanding of cracking and of the

factors that influence it needs to be developed from new research activities. A primary challenge is to establish and apply proven non-destructive methods to identify and characterize degradation in pressure boundary components in general, and stress corrosion cracks in nickel-based alloys as a specific current example. The outcome of ongoing and future research efforts should significantly enhance our ability to address identified and emerging degradation. A thorough understanding of the degradation mechanisms that can lead to failure will enhance our ability to effectively utilize risk-informed considerations. This will improve our ability to ensure primary pressure boundary integrity and thus, maintain public health and safety. Improving the collective ability of the nuclear community to deal effectively with this problem will also result in increased public confidence in the safety of nuclear power.

At this conference, we will hear a number of technical papers related to the cracking observed in nickel-based alloys and the adequacy of in-service inspections and mitigation programs as well as experiences from organizations outside the United States. In my previous remarks, I have pointed out that improving our understanding of the complex challenges presented by stress corrosion cracking in nickel-based alloys is important to all of us. For this conference, I would like to suggest that we focus our efforts on these questions:

1. How can we better monitor the condition of Alloy 600 components and Alloy 182 welds to ensure that primary pressure boundary integrity is maintained? This question includes consideration of inspection programs to detect cracks before they lead to coolant leakage or otherwise become a significant safety issue.
2. How can we most effectively model the degradation process, including crack initiation and growth, in order to predict the effects on the integrity of primary pressure boundary components?
3. How can we ensure that repair and post-repair monitoring activities are performed so that identified cracks in components are effectively addressed and integrity is maintained?
4. What technical issues need to be resolved to ensure that manufacturing processes are optimized to minimize the susceptibility of replacement components to stress corrosion cracking?

*I would like* to encourage all of you to participate in the technical discussions of these questions in the hope that we will make progress in identifying future research and collaboration possibilities. We have over 200 scientists and engineers from around the world attending this conference. The potential for progress through open and supportive discussion of this complex subject by this collection of broad and varied expertise is a truly unique opportunity. Let us take full advantage of it.

My hope is that this conference will provide a platform for gaining a better understanding of our knowledge of stress corrosion cracking in components of nickel-based alloys. This community of scientists and engineers is responsible for developing a better understanding of materials degradation, and the associated inspection practices for detecting and characterizing that degradation. I also hope that your interactions will lead to the identification of additional areas for collaborative research beneficial to the nuclear community and the public. Each of you brings relevant expertise that will contribute to this collective effort to improve our understanding of stress corrosion cracking in nickel-based alloys. This is a complex subject with

direct implications on safety. I wish you all a successful conference and look forward to working with you. To our out-of-town guests, enjoy your stay in the Washington metropolitan area and best wishes to you all.

Thank you.

## KEYNOTE ADDRESS[2]

What are the Issues?

K. Gott,
Swedish Nuclear Power Inspectorate (SKi), Stockholm, Sweden

Background

This conference, despite its title, is not intended to deal only with the observations of cracking found in vessel head penetrations, but is intended to be a more general forum for the discussion of all aspects of primary water stress corrosion cracking of nickel base alloys in thick section components. We should also remember that when these issues first saw the light of day, steam generator tube degradation had been studied for many years and has provided significant information which enabled us to get the plants back into operation in a timely manner. Perhaps because of their location and the related materials problems the vessel head penetrations have in recent time been in the international spotlight, but they should not be allowed to completely overshadow the problems associated with safe-end cracking and repair, and other components which could be affected in the future. Stress corrosion cracking in nickel base alloys in pressurized water reactors should be treated as a generic problem. This has been more evident in recent times with the discovery of cracking in the lower head penetrations and pressurizer in recent months.

All countries are concerned with public perception of the safe running of individual plants and the fleets as a whole. Events at one plant have an immediate impact on all other plants, in particular all similar plants, wherever they are in the world. This was clearly evident following the events in VC Summer and Davis-Besse when the international networks, both formal and informal, were quickly utilized to the full. No safety authorities or utilities could afford to assume that these events were one-off, a "French problem", or a "US problem". Many utilities in Europe initiated inspections or responses without specific formal requirements being issued, in other countries where such requirements are a natural part of the system the authorities were quick to respond. In Sweden and France for example the vessel head penetrations have been part of the volumetric inspection programmes since the early 1990's following the incident in Bugey 3. The United States Nuclear Regulatory Commission has issued an order requiring recurrent volumetric examination of the vessel head penetrations at inspection intervals depending on the susceptibility of the plant, and discussions are underway to introduce appropriate changes to the relevant Code cases.

[2] In lieu of the originally scheduled presentation by K. Gott, the presentation of the keynote paper at the time of the conference was made by W. H. Bamford, Westinghouse Electric Corp., who reprised a presentation made six weeks earlier at the 11th International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, August 10-14, 2003. That paper is copyrighted by the American Nuclear Society, and may be found in the proceedings from that conference. K. Gott, who could not attend the rescheduled conference, kindly provided her written contribution after the conference, and it is that contribution which follows.

## Historical highlights

Primary water stress corrosion cracking in nickel base alloys, particularly Alloy 600, is not a new problem, although in thick section components it is not as old as intergranular stress corrosion cracking in austenitic steels in the boiling water reactor fleet. The first reported incidents of primary water stress corrosion cracking outside steam generators were associated with pressurizer surge line nozzles. Subsequently the French reported cracking in the vessel head penetrations in the early 90's, the most famous event being the Bugey 3 through wall crack. The French reacted quickly and determined there to be a correlation between the vessel head temperature and the likelihood for cracking, and this concept is still being used worldwide as one of the most important indicators of susceptibility.

The extent of the problems with primary water stress corrosion cracking varies greatly from country to country mainly because of the choice of material for the various components in the different countries. For example, the French, who first recognized the problems with the vessel head penetrations, will not be having such problems with the safe ends since these are not made of Alloy 600, and therefore have not had to be welded with nickel base alloys, but with stainless steel. In Germany Alloy 800 has been used extensively in steam generators instead of Alloy 600, and it has been found to be much more resistant to stress corrosion cracking. The German plants have also performed post-weld heat treatments on the J-groove welds of the vessel head penetrations. At this conference there are contributions from several European countries as well as Japan with reports on their experience of stress corrosion cracking in thicker section components of nickel base alloys. In addition to those countries presenting their experience here, cracking has also been reported from Switzerland, South Africa and Korea.

## Current situation

Most of the reported cracking has been axial, located in the Alloy 600 penetrations below the J-groove weld, and is therefore of lesser structural concern. However there are now axial cracks extending above the J-groove weld, as well as more recently circumferential cracks being reported, and thus the structural concern has increased significantly. Cracking has also been reported in the Alloy 182 weld metal in the J-groove welds and in the safe-ends to the vessel nozzles. This has emphasized that this is a generic problem, and that we all need to be prepared to deal with it through reliable inspection programmes, mitigation measures and repair techniques.

The need to disposition flaws has led to experimental programmes in many countries to assess the susceptibility of nickel base materials to primary water stress corrosion cracking and also to predict crack growth rates.

A large amount of field data has become available from the French vessel head penetration inspection programme which has been in place since the early nineties. This information has been used by the French as an important portion of the data on which their crack growth rate relationships have been based. They also have one of the most extensive experimental

programmes in this area. In addition to these experimental programmes the French have a systematic programme for the replacement of their vessel heads using for the most part materials thought to be at least less susceptible to primary water stress corrosion cracking. Other countries very active in the field of crack growth rate measurement and the development of disposition lines are Japan, Sweden and the United States, all of which have collected and analyzed the available data, and there were presentations in this area in the third session.

Another important aspect of the disposition of flaws is an accurate knowledge of the stress fields. Work is for example underway to characterize these, in particular with respect to the vessel head penetrations. Residual welding stresses are particularly important in this respect but much more difficult to come to grips with since they are seriously affected by weld repairs. One of the problems associated with this is the completeness of the documentation available concerning the manufacturing of components and areas which have in fact been repaired, but which are not always formally designated as repairs.

Yield strength is known to have an influence on the susceptibility of Alloy 600 to primary water stress corrosion cracking, and it has for example been observed that in general the French plants have higher yield strength material than many of the plants in the United States. The steam generator programs have also clearly demonstrated the importance of microstructure, and for example that high or extensive carbide grain boundary coverage reduces the susceptibility to stress corrosion cracking in Alloy 600.

One of the characteristics of primary water stress corrosion cracks is that they can be very tight, almost undetectably tight at the inner surface, but that they are in fact connected to a more open crack in the body of the component. This poses enormous challenges to the inspection process, not least with respect to interpretation and qualification.

The weld metals in question are highly susceptible to hot cracking and there are similarities in morphology between hot cracks and stress corrosion cracks which can make it difficult to distinguish between the two. Some people believe that hot cracks can be used to simulate stress corrosion cracks for qualification purposes, or other studies, whilst others firmly believe that this is totally inappropriate. There are also questions as to whether hot cracks in fact constitute part of the crack path, and that all the propagation observed is in fact not due to stress corrosion cracking. How such questions should be reconciled in a safety evaluation must be addressed.

One interesting observation made by Ringhals was that some of the cracking in the vessel head penetrations had all the appearances of having been initiated by thermal fatigue. Other work in Sweden has shown that thermal fatigue has on a number of occasions been the precursor to stress corrosion crack propagation.

Repair methods need to be chosen with care so as not to introduce new defects or worsen existing defects. North Anna had problems after weld overlay repair with renewed cracking, and Ringhals has experienced problems with small cracks opening up after electro discharge machining repair. These examples illustrate the need for qualification of the repair techniques for the specific application.

**Future considerations**

Primary water stress corrosion cracking is not only an issue concerning penetrations, it is equally important to understand aspects which concern the safe-ends, and all the other components in the plants which contain Alloy 600 and welds of Alloy 182. A better understanding of the conditions under which primary water stress corrosion cracking can occur is essential for all parties in the community, and this can in part be obtained by detailed investigations of the root causes of the individual cases. For example in Sweden, SKI (the Swedish Nuclear Power Inspectorate) is collaborating with the utility (Ringhals) in more fundamental studies of the Alloy 182 cracking in the Ringhals safe-ends. Repair of safe-ends is also an important topic which is to be addressed at this conference, and Ringhals representatives will be describing a repair they have carried out which can also be classified as a mitigation procedure.

A prevalent hypothesis is that primary water stress corrosion cracking in Alloy 182 is associated with weld repairs. In the case of Ringhals 3 this has not been shown to be the case. There is no evidence, either documented or physical, of any repairs in the portions of the safe-ends which contained cracks. Even if regions containing repair welds might be expected to exhibit primary water stress corrosion cracking earlier than unrepaired regions because of the higher residual stresses associated with such regions, it does not eliminate the risk of primary water stress corrosion cracking developing later in lower stressed regions. It is therefore important to reassess inspection programmes to take into account known areas of weld repair, and also other weld regions which could be more likely to be susceptible to primary water stress corrosion cracking. In order to keep inspection within reasonable limits, and still as comprehensive as necessary to ensure the safety of the plant, information about the practical thresholds for primary water stress corrosion cracking, such as stress levels, susceptibility correlations and reliable crack growth data, are needed.

To answer these questions and improve our understanding of the issues, including those concerning the new materials, it is my opinion that extensive international co-operation will be needed. This conference is an example of the nuclear community at its best when all aspects of the industry are presenting their information and making it freely available. This is becoming more and more important as all of us suffer the reductions in resources, not least for research, and economic pressures of the twenty first century.

Last but not least I think it is of the greatest importance that we let recent events help us to attain an improved safety culture. It is easy to be complacent and say "How could they let that happen?" or even "That could never happen here", but we should possibly be asking "How can we ensure that nothing like this happens again, anywhere?" This is not just a question for the utilities but also for the regulators.

# Section 1:
# Session 1: Inspection Techniques, Results, and Future Developments

## NDE of Austenitic Materials-A Review of Progress and Challenges

**F. Ammirato, EPRI, Charlotte, NC**

Austenitic materials are used extensively in nuclear power plant construction because of their useful properties such as corrosion resistance and toughness. In both BWR and PWR applications, however, these materials, including weldments, have shown a susceptibility to various forms of stress corrosion cracking that has led to the implementation of augmented inservice inspection programs in the industry. Practical difficulties to inspection are imposed by the material properties, physical configuration, and in some cases, access limitations. Therefore, performing reliable periodic inservice examination that is required of these components has challenged the NDE community due to their particular characterizes and configurations. The high assumed crack growth rate of primary water stress corrosion cracking (PWSCC) in some PWR environments exacerbates the inspection challenge in that relatively small flaws are projected to grow significantly in a short time relative to practical inspection intervals.

This paper will review inspection austenitic materials and will describe advances in NDE technology, field practices, and personnel training and qualification initiatives in place to address these inspection challenges.

**Manuscript was not available for publication in the Proceedings**

# Inspection Reliability of Reactor Vessel Head Penetrations[1]

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## ABSTRACT

The United States Nuclear Regulatory Commission (USNRC) has conducted research in the areas of assessment and reliability of Non-Destructive Examination (NDE) and environmentally assisted cracking since 1977. Within the last three years occurrences of cracking in Inconel (Alloy 82/182) welds and Alloy 600 base metal at several domestic and foreign operating nuclear power plants have raised concern with the plant licensees and operators, industry groups and regulators. The occurrences of cracking have been identified through indirect means, specifically the discovery of boric acid deposits resulting from through-wall cracking in the primary system pressure boundary. Analyses indicate that the cracking has occurred due to primary water stress corrosion cracking (PWSCC) in Alloy 82/182 welds, in both hot leg nozzle-to-safe end welds and control rod drive mechanism (CRDM) nozzle welds. In addition, circumferential cracking of CRDM nozzles in Alloy 600 base metal originating from the outside diameter (OD) of the nozzle has been identified. The cracking associated with safe end welds is important due to the potential for a large loss of coolant inventory, and the cracking of CRDM nozzle welds and circumferential cracking of CRDM nozzle base metal is important due to the potential for control rod ejection and loss of coolant accident.

This paper reviews the most recent work being conducted for the USNRC by Pacific Northwest National Laboratory (PNNL) to compile information on failures that have occurred in Alloy 600/182/82 materials. The location of PWSCC has been found in a number of locations that has required refinement in the NDE inspections that are conducted. Some more recent failures have been caused by PWSCC in the J-groove weld and associated buttering. Some basic studies are being conducted on a weldment from the Midland head to understand the capability of conducting a volumetric inspection of the J-groove weld and buttering. Future studies are planned on CRDMs removed from service.

[1] This work was conducted for the U. S. Nuclear Regulatory Commission under DOE Contract DE-AC06-76RLO 1830: for JCN Y6534 with Carol Moyer, NRC program manager, JCN Y6604 with Debbie Jackson, NRC program manager and JCN Y6909 with Bill Cullen, NRC program manager.

## INTRODUCTION

Inconel Alloy 600 along with weld Alloys 182 and 82 were selected and employed in a variety of nuclear power plant components because of their attractive properties which include high strength, ductility and corrosion resistance. In pressurized water reactors (PWRs) these applications include the steam generators, pressurizer heater sleeves, instrumentation and sampling nozzles, control rod drive mechanism (CRDM) vessel head penetrations, and dissimilar metal piping weldments. Unfortunately these Inconel materials have been found to be susceptible to degradation by a mechanism known as primary water stress corrosion cracking (PWSCC). PWSCC has breached the reactor coolant pressure boundary resulting in degrading plant safety. Thus, the reliable detection of PWSCC in these materials before component structural integrity is challenged is important. This paper examines the nondestructive examination (NDE) programs being employed to reliably detect PWSCC in reactor vessel head penetrations.

Based on the preponderance of PWSCC occurrences in these Inconel materials, it has been concluded that this is a generic problem. A recent review paper by Bamford and Hall 2003 documents the history of cracking in these alloys. In addition, the NRC Davis Besse Lessons Learned Task Force Report 2002, has analyzed the licensee event reports (LERs) from 1986 through 2002 in Appendix E. There were a total of 89 LERs and leaks were found to occur in most locations where Inconel had been employed. Seventeen percent of the leaks, which was the highest frequency of occurrence, were associated with CRDMs and most of these leaks have taken place since 2000.

## HISTORICAL EVENTS

One of the earliest leaks occurred in the San Onofre Unit 3 nuclear power plant and was detected on February 27, 1986. The following account was obtained by interviewing the plant personnel that detected the leak. A plant engineer on a daily basis obtained radiation level measurements and manually plotted this data. He noted that there was a subtle rise in radiation levels in the zone where the pressurizer was located. A request was made to have a staff member enter this zone and look for a potential leak source. The staff member entered the zone and did not see any evidence of leakage or boric acid deposits. As he stood there perplexed as to what to do next, he noted an audible hissing sound. Upon investigation, he was able to locate a leak. This leak was on a pressurizer instrumentation nozzle and was estimated to be 0.15 – 0.2 gpm (0.57 – 0.76 lpm) in size. Basically this leak was first detected by subtle radiation level increases and then located and confirmed by audible acoustic emission.

The first significant leak in a CRDM occurred in September 1991 at the French plant Bugey 3 (Shah et al 1994, Buisine et al 1993). This through-wall failure was detected during a 10 year hydrotest and was detected with an acoustic emission technique. The leak rate was 0.003 gpm (0.7 l/h). There were two through wall ID axial cracks confirmed by destructive testing (DT) and determined to be PWSCC. There were also two circumferential cracks on the outside diameter (OD) of the penetration tube that were also confirmed by DT. One crack was a hot crack located in the weld that had been created during fabrication. The other was in the base metal and connected to the through wall axial crack on the down hill side of the nozzle and just above the J-groove weld. As a result the French made the decision to replace all of their reactor heads using more resistant materials. The French have conducted dye penetrant tests (PT) of the J-groove weld crowns and buttering from 11 replaced heads. The 754 PT inspections reported by Amzallag et al 2002 have found no cracks.

The vessel head penetration degradation became a significant problem in the U. S. with the PWSCC detected at Oconee. From 11/2000 to 2/2003 there have been ten plants that have detected PWSCC requiring repair. There were 79 CRDMs that required repair and 13 thermocouple nozzles. Twelve CRDMs had circumferential cracking and this is of concern because this cracking is above the J-groove weld where the potential of a CRDM ejection is increased. Of course, the worst problem was the severe wastage that was found in the vessel head at Davis Besse.

In April 2003, boric acid deposits were detected on the lower head of the reactor pressure vessel at the South Texas Project (STP) Unit 1. Two bottom mounted instrumentation penetrations had small boric acid deposits (3 mg and 150 mg) that was estimated to be 3 to 5 years old. NDE tests were conducted and confirmed the presence of axial cracking along the weld fusion zone between the penetration tube and the J-groove weld and extending into the penetration tube wall. The cracking was confirmed by ultrasonic (UT), eddy current (ET) and helium bubble testing. A boat sample was taken and documented the cracking with the DT results reported in a supplement to the LER 03-003. The surprising thing about this PWSCC was that it occurred in a zone where the temperature is low and was not expected to crack. The DT results show that a contributing factor was the presence of a lack of fusion flaw at this location.

Over time the problem of PWSCC and changed. The location of the PWSCC has moved from initially being found at the inside diameter (ID) of the penetration tube. Cracking was next found on the OD of the penetration tube at the fusion zone of the J-groove weld. This was followed by OD initiated circumferential cracking above the J-groove weld, cracking in the J-groove weld, cracking in the buttering and finally cavities in the ferritic steel resulting from through wall leakage. The natural question is what is next?

## NDE INSPECTION STRATEGY

The inspection of CRDMs for PWSCC can follow a number of strategies. The inspections being performed on CRDM penetration tubes have been driven in part by the requirements in ASME Section XI Code. The Code requires that visual tests (VTs) be conducted looking for the presence of leakage or boric acid deposits. Unfortunately, the Code does not require that the insulation be removed for conducting this inspection. As a consequence, small amounts of leakage that may occur, as has been found at STP, can be missed. There must be adequate access under the insulation to accommodate the VT equipment (normally a small robot with a TV camera and lights). Other possible sources of leakage such as from seals on the CRDMs above the head can obscure leaks of interest. A strategy based on use of VT will not prevent leaks from occurring but will only detect leakage. The goal of the NDE program should be to prevent leaks and VT should be used as a back up in case degradation is missed by other NDE inspections.

ET is used to inspect for the presence of surface breaking cracks on the surface being inspected. It is a very effective method for detecting surface breaking cracks and it also provides information about the crack length. It does not provide any information on crack depth. If the crack is near the surface, ET can still detect it. It does not require any coupling media and maximum sensitivity is obtained with the ET probe in contact with the surface being inspected. ET is very effective for inspecting the ID of the CRDM penetration tube because this is base metal that has machined surface conditions. ET is also used for the inspection of the J-groove weld crown and exposed surface of the buttering. This later inspection is somewhat more challenging because of the complex geometry of this region and the fact that the surface is manually ground (meaning that it is has been smoothed but can still be quite irregular).

UT has primarily been used for inspecting from the ID of the CRDM penetration tube for detecting and characterizing cracks on the ID, within the tube wall or on the OD. The most commonly employed UT implementation uses a time-of-flight diffraction (TOFD) method. The ID conditions of the CRDM are machined, the Inconel Alloy 600 tube materials are fine grained and adequate access exists for conducting an effective UT. Since PWSCC are cracks, they provide crack tip signals. TOFD creates a lateral wave that is useful for detecting near surface flaws and a back surface signal that is very effective for detecting OD initiating cracks. Hence, in this application TOFD works very well for the detection and characterization of PWSCC. It is equally effective for axial and circumferential cracks. The only location where detection capability may be limited is for the fusion zone between the J-groove weld and the OD of the penetration tube.

Dye penetrant testing (PT) is used for the detection or confirmation of surface breaking cracks that might be located on the OD of the penetration tube, in the crown of the j-groove weld or the exposed surface of the buttering. PT is basically an enhanced VT. It can be very effective but the quality of the surface conditions and the tightness of the cracks can degrade the inspection effectiveness. If the PT is performed manually, there will be a high radiation exposure to the inspection staff. If the PWSCC only break the surface in a limited number of locations, the indication may be misinterpreted as not being a crack or it may be called a number of small cracks.

Other technology has been used or is being developed for the detection and monitoring of PWSCC in CRDMs. This includes the use of acoustic emission technology for on line continuous monitoring. The use of phased arrays for detecting and characterizing wastage is being developed and evaluated but the effectiveness of this technology for this application is not known.

## PROGRAMS ADDRESSING NDE EFFECTIVENESS

Internationally there are programs underway at the Electricity de France, in Sweden and at the Joint Research Center in Petten, The Netherlands. In the U. S. all of the inspection vendors are working on improving their inspection process and the industry (EPRI NDE Center and Materials Reliability Program (MRP)) has a program that is developing mockups for NDE demonstrations and conducting other research activities. Other authors at this conference are addressing these programs and I would refer the reader to those papers. The remainder of this paper will focus on the work being performed at PNNL that is funded by the NRC. There are three different NRC programs involved and the information will be presented by program number.

### JCN Y6604

The objective of this work was to address the issue of volumetrically inspecting the J-groove weld and all of the buttering. A CRDM specimen was obtained that had been cut out of a head from the cancelled Midland nuclear power plant. This specimen was received from Oak Ridge National Laboratory who originally obtained it from Framatome. The specimen which was received had been flame cut from the head. PNNL chose to machine the ferritic material to obtain the largest cylinder of ferritic steel that could be machined and be concentric with the centerline of the penetration tube as shown in Figure 1. A typical result that was obtained by UT from the outside ferritic machined surface at 10 MHz using synthetic aperture focusing technique (SAFT) is shown in Figure 2. Four product forms are being imaged. The goal was to look for the presence of welding flaws and of course all welds are imperfect and contain fabrication flaws. Based on previous studies most of the welding flaws are located along the fusion zones and these have most often been found to be lack of fusion. In this case there were

a number of 1 to 2 mm flaws detected. The response of these weld flaws was about -35 dB while the response from the weld grains was typically -42 dB while that from the ferritic base metal was -57 dB when compared to the reference standard.

What we found in this study was that fusion zone fabrication flaws could be detected using normal incidence. However, these could only be detected for the near fusion zone. If the sound field had to pass through the weld, then it was not successful in detecting these small fabrication flaws on the far side fusion zone. Future work will evaluate the capability of UT to detect and to characterize fabrication flaws on all of the fusion zones as a function of the flaw size.

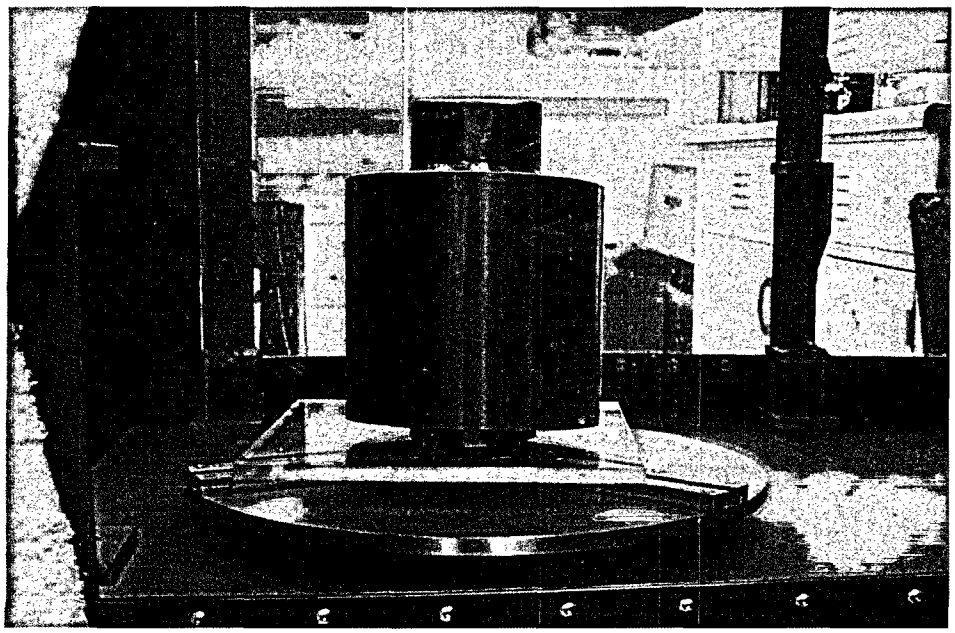

**Figure 1. Midland CRDM after machining and located in water immersion tank prepared for UT inspection.**

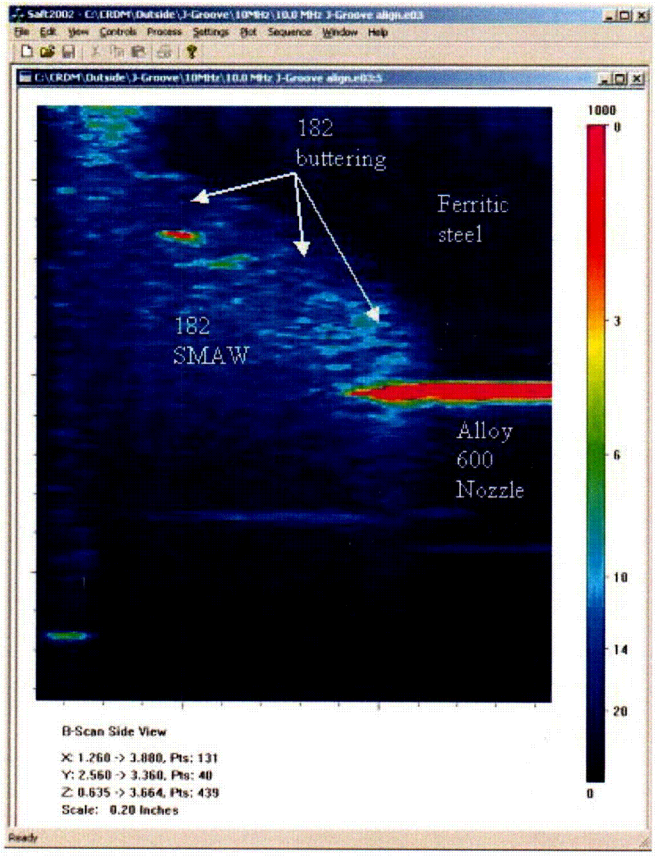


**Figure 2. UT results using normal incidence at 10 MHz spherically focused insonfication showing the four product forms.**

## JCN Y6534

One of the main objectives of this program is to develop an international cooperative to address the issues of PWSCC and NDE reliability of dissimilar metal welds (DMWs) and nickel based alloys. PNNL is providing support to the NRC in this effort. All interested parties both nationally and internationally are being contacted and offered the opportunity to participate. A presentation on this cooperative was made during the conference. An evening meeting was held to go into more details regarding ideas about the activities that might be addressed during this cooperative. A follow-up was proposed with a mailing to all interested parties to solicit interest and suggested priorities so that a proposed program could be developed based on this input for the cooperative. There are so many issues related to PWSCC in all of the Inconel applications that are not understood, thus, it is important to identify those of highest interest to the participants in the cooperative.



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One task that was proposed would involve producing an atlas of metallography documentation on PWSCC cracks and NDE responses. This information is needed to understand the variability as well as to understand how one might be able to simulate PWSCC with other flaws.

Another proposed task would be to organize and conduct a round robin study to assess NDE techniques for their reliability in detecting and characterizing PWSCC. The challenge here is identifying which of the many Inconel applications that should be studied so that appropriate mock ups can be designed and fabricated with realistic PWSCC.

Other activities that have been suggested for the cooperative include the assessment of NDE modeling for Inconel applications. This might be one way of extending the studies that are being proposed to those applications that were not studied. There are a number of conditions such as surface preparation that can impact NDE effectiveness and laboratory parametric studies that might be performed to address these issues.

The plan is to have the parties identified, agreements in place, and a plan refined to begin conducting the work by mid 2004.

**JCN Y6909**

Part of this work is a joint program with EPRI/MRP to decontaminate CRDMs that have been cut from the North Anna 2 head which has been removed from service. These CRDM nozzles were cut and shipped to PNNL where they are being decontaminated while maintaining as best we can, their pristine condition both for NDE studies and destructive characterization of the PWSCC including chemical analysis over the entire crack depth. The CRDMs will be decontaminated in early December 2003 in preparation for NDE vendor inspections. There will be four vendors that will conduct NDE inspections during December and January. Two of the CRDMs are scheduled for destructive testing.

Figure 3 shows a CRDM that has been decontaminated and is being readied for transit to a stand where it will reside during NDE inspections. Four CRDMs will be inspected by the commercial NDE vendors. The really fantastic opportunity that the North Anna 2 CRDMs offer is realistic NDE studies on service induced PWSCC coupled with flaw validation. This will allow us to fully understand how effective the NDE is in detecting PWSCC at all of the locations found in these samples. For the first time, we will also know what was found and what was not found. In addition we will for the first time understand how accurate the NDE techniques are in sizing the PWSCC.

PNNL plans to fabricate some research grade scanners and conduct NDE studies on the remaining North Anna 2 CRDMs that are not sent off for DT. The remaining NRC activity that is now in the planning stage is to conduct studies on material that was received at PNNL from Davis Besse. It is expected that there will probably be both NDE studies and DT studies conducted on this material over the next two years.

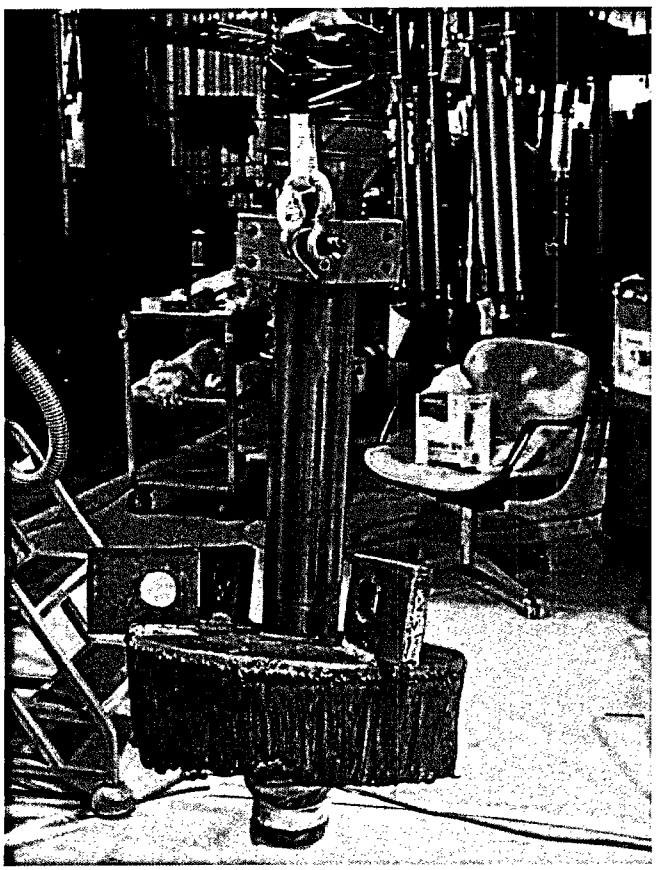

**Figure 3. North Anna 2 CRDM that has been decontaminated and is ready for moving to a stand for NDE inspections.**

## CONCLUSIONS

It is pretty well accepted in the nuclear community that the failure of Alloys 600/182/82 is a generic problem with the dominant failure mode being PWSCC. Because many of the PWSCC cracks do not occur until after many years of operation, there appears to be a long incubation period before PWSCC initiation. The strategy taken to date has focused on employing VT to find leakage. Because boric acid deposits result from leakage, the locations where leakage occurs is identified as long as there is good access for VT and there are no competing sources of leakage from sources such as seals. The shortcoming of this strategy is that it detects leaks but does not find the PWSCC before leakage occurs. There are many questions about the overall effectiveness of NDE to detect PWSCC in these Inconel materials. The J-groove weld and buttering have never received a volumetric examination and the results from STP indicate that fabrication flaws can be a significant contributing factor to cracking. At a minimum it would be useful to understand if the fusion zones and buttering of the J-grove welds can be effectively inspected to detect fabrication flaws of importance to structural integrity. Fortunately, there are a number of programs (NRC, industry, international and the proposed NRC cooperative) being conducted or being planned to address these issues. Hopefully, they will provide timely quantitative information needed to bring these issues to closure.

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**The Evolution of Inspection and Repair Approaches for Reactor Vessel Head Penetrations**

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Since the fall of 2000, hundreds of Reactor Vessel Head Penetrations have been inspected. Regulatory documents in the form of bulletins, and finally an order, have been issued to provide inspection guidelines for PWR licensees. Leaking penetrations and top of head degradation have lead to over 80 repairs and the rapid industry movement toward head replacement. Inspection approaches currently utilized by the industry have evolved through two rounds of Materials Reliability Program (MRP) demonstrations. In addition to bare head inspections, under head techniques for nozzles and the J-groove welds have seen extensive use in the plants most susceptible to primary water stress corrosion cracking (PWSCC). In a similar manner, repair approaches for penetrations exhibiting leakage or significant degradation have evolved from manual to remote welding techniques, decreasing dose, manpower and repair time. This paper will provide an overview of inspection and repair experience with details on the approaches used to address the NRC guidelines and the repair of degraded penetrations.

**Manuscript was not available for publication in the Proceedings**

## Summary of US PWR Reactor Vessel Head Nozzle Inspection Results

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Abstract: The results of reactor vessel head (RVH) nozzle inspections in the U.S. have tended to support the time-at-temperature model that has been used to prioritize inspections in the U.S. since the first evidence of RVH nozzle leakage was detected in late 2000. The time-at-temperature model, which ranks each plant on the basis of operating time scaled for differences in the RVH operating temperature, is based on voluminous laboratory and plant data showing that primary water stress corrosion cracking (PWSCC) of nickel-based alloys is a thermally-activated aging mechanism. As of August 2003, about 96% of the more than 5,000 RVH penetrations at the 69 U.S. PWR units have been inspected by bare-metal visual (BMV) examination, eddy current testing (ET) surface examination, and/or ultrasonic testing (UT) volumetric examination. The 51 leaking CRDM nozzles and all but 12 of the approximately 124 cracked nozzles detected have been from the 15 highest ranked units on the basis of time at temperature. However, the RVH nozzle inspection results also show that nozzle material processing and head fabrication differences are major factors affecting the cracking susceptibility of Alloy 600 RVH nozzles and their Alloy 182 attachment welds. Little or no cracking has been detected to date for plants having several combinations of Alloy 600 material supplier and RVH fabricator even though several of these plants are highly ranked in terms of time at temperature.

This paper presents work sponsored by the Electric Power Research Institute (EPRI) / Materials Reliability Program (MRP).

### USES OF INSPECTION SUMMARY STATISTICS

Inspection summary statistics are used for a number of purposes including:

- Verification of time-at-temperature (EDY) as a predictor of PWSCC susceptibility
- Revealing trends of cracking for subgroups of RPV heads including the head fabricator and the nozzle material supplier
- Providing input to safety assessments such as Weibull models of time to crack initiation or leakage, confirmation of laboratory crack growth test data, location and orientation of cracks, and low alloy steel wastage models
- Facilitating periodic evaluations of industry inspection plans

## INTRODUCTION

Figure 1 shows typical locations of thick-section Alloy 600 materials in PWR plants. These locations involve pressurizer temperatures (≈650°F), hot leg temperatures (≈600°F), and cold leg temperatures (≈550°F). PWSCC has resulted in leaks from penetrations at all three operating temperatures.

This paper focuses on PWSCC from reactor pressure vessel (RPV) head nozzles as shown in Figures 2 and 3. Figure 2 shows the locations of PWSCC that has been discovered in RPV head control rod drive mechanism (CRDM) nozzles in B&W and Westinghouse design plants and control element drive mechanism (CEDM) nozzles in Combustion Engineering design plants. The cracks have been located in the Alloy 600 nozzle tubes and in the Alloy 82/182 welds near the J-groove weld where high tensile residual stresses from welding combine with operating pressure and temperature stresses. Figure 3 shows a typical CRDM/CEDM nozzle, a typical head vent nozzle, and a typical incore instrument (ICI) nozzle in a Combustion Engineering design plant. To date, PWSCC has been detected in CRDM and CEDM nozzles only, with the exception of all 16 cracked small-bore thermocouple nozzles at two units.

The following summarizes the number of RPV head penetrations in the 69 operating PWR plants in the United States that have Alloy 600 nozzles attached to the head by J-groove welds:

- 3,871 CRDM nozzles (55 units)
- 1,090 CEDM nozzles (14 units)
- 94 in-core instrument (ICI) nozzles (11 units)
- 59 vent line nozzles (59 units)
- 16 small-bore thermocouple nozzles (2 units – currently replaced)
- 8 auxiliary head adapter nozzles (2 units)
- 2 de-gas line nozzles (2 units)

Several of the 69 PWR plants have Alloy 600 nozzles that are not attached to the head by J-groove welds. These are:

- 3 full-penetration weld vent nozzles (3 units)
- 6 internals support housing nozzles (2 units)
- 20 auxiliary head adapter nozzles (5 units)

## INSPECTION RESULTS

The MRP collects inspection results and updates the summary statistics each outage season. The data are collected on an individual flaw level and these data are processed to provide the desired statistical data. The MRP tracks the inspection results and summary statistics for all domestic PWR RVH nozzles in the context of the key design, fabrication, and operating parameters. The inspection techniques that have been applied are illustrated in Figure 4. These techniques include inspections for evidence of leakage, surface examination using eddy current techniques, and volumetric examination using ultrasonic testing.

Figure 5 is a graphical summary of the industry inspection results on the basis of temperature-adjusted operating time (effective degradation years—EDYs). Per standard practice, a thermal activation energy of 50 kcal/mole has been assumed for the temperature adjustment of operating

time. The inspection results show that cracks and leaks in RPV head nozzles have been concentrated in plants with the greatest number of EDYs. One purpose of this work has been to assess whether this experience shows that all plants with high EDYs are equally susceptible, or whether there are other significant factors such as the nozzle material supplier or head fabricator.

Table 1 provides a summary of plants with detected RPV head nozzle PWSCC. Table 2 provides a summary of plants with leakage, including the type of repair or replacement performed. Table 3 provides a summary of the orientations and locations of PWSCC cracks in nozzles. Table 4 provides a summary of the circumferential cracks that have been detected that are located above the weld in the nozzle tube or in the weld zone elevation of the nozzle tube.

The main conclusions from the inspections are as follows:

- About 51 CRDM nozzles have been found to be leaking. All of the leaks are in plants with greater than 12 EDY of operation.
- Forty of the leaks occurred in CRDM penetrations in the seven operating B&W design plants. This represents 8.3% of the nozzles in B&W design plants.
- Eleven of the leaks occurred in three heads fabricated by the Rotterdam Dockyard Company (RDY). These leaks were all associated with cracks in welds.
- Little or no wastage has been detected associated with the leaks except for the case of Davis-Besse. Forty-two of the leaking CRDM nozzles were repaired in a manner that would likely have shown significant boric acid corrosion had it occurred.
- As predicted by finite element analysis, the nozzle cracking has been predominantly axial. Only 35 of the 371 detected nozzle cracks have been circumferential and only two circumferential cracks above, or near, the top of the J-groove weld have been through-wall.

**SUBGROUP STATISTICS**

In addition to the overall summary, evaluations have been performed to assess several subgroups. As previously noted, one objective of this work is to determine if factors other than time at temperature (EDY) have a significant effect on PWSCC susceptibility. The assessments are shown in Figures 6 through 14. The data in these figures represent inspections performed from December 2000 through August 2003. Earlier inspections are not included given the limited awareness of the potential for PWSCC on the nozzle OD surfaces and welds prior to December 2000.

The data show the following:

- All 51 leaking CRDM nozzles and all but 12 of the 124 cracked penetrations are from the 15 highest ranked units based on time at temperature.
- The incidence of PWSCC in heads fabricated by Combustion Engineering is relatively low, and comparisons by EDY group show that these differences reflect more than just differences in temperature.
    - 0.7% of the penetrations in CE fabricated heads inspected nonvisually have shown cracks.
    - 13% of the penetrations in B&W fabricated heads inspected nonvisually have shown cracks.
    - 46% of the penetrations in RDY fabricated heads inspected nonvisually have shown cracks.

- The incidence of cracking in nozzles fabricated of materials supplied by Huntington Alloys or Standard Steel has been relatively low. Again, comparisons by EDY group show that these differences reflect more than just differences in temperature.
    - 0.5% of nozzles fabricated from Huntington Alloys or Standard Steel material inspected nonvisually have shown cracks.
    - 12% of nozzles fabricated from B&W Tubular Products material inspected nonvisually have shown cracks.
- Cracks in welds have been limited to vessels fabricated by Rotterdam Dockyard and B&W-designed units.

**PLANNED HEAD REPLACEMENTS AND INSPECTIONS**

Table 5 is a list of the 29 plants that have announced plans to replace their RPV heads. At least another two units currently have set replacement plans. It is expected that after the fall 2003 outage season, bare metal visual and/or nonvisual NDE examinations will have been performed on all RVH nozzles in operating original heads. In addition, it is expected that 28 of the 29 plants in the NRC's high susceptibility category (> 12 EDYs or detected cracking) will have completed baseline nonvisual examinations or head replacement. Finally, all 46 plants with > 8 EDYs are expected to have completed baseline nonvisual examinations or head replacement after the fall 2005 outage season.

**CONCLUSIONS**

The conclusions from this work are as follows:

- Time at temperature is an important susceptibility factor.
- The head fabricator and nozzle material supplier are also important factors.
- Relatively little cracking has been detected in heads fabricated by Combustion Engineering using nozzle material supplied by Huntington Alloys or Standard Steel.
- No weld cracking has been detected in heads fabricated by Combustion Engineering.
- The reasons for the better performance of Combustion Engineering fabricated heads with Huntington Alloys or Standard Steel nozzle material are not known, but are likely related to processing parameters such as annealing temperature, cooling rate, straightening practices, machining practices, welding procedure details, etc.

## Table 1 – Summary of Plants with Detected RPV Head Nozzle Cracking

| Number | Unit | EDYs thru Feb. 2001 (@ 600°F) (MRP-48) | Current Head Temp. (°F) | NSSS Supplier | Vessel Fabricator (Note 1) | Nozzle Material Supplier (Note 2) | No. of CRDM or CEDM Nozzles on Head | Number Cracked Penetrations Detected (Note 3) | | | Notes |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | Tube and/or Weld Cracked | Tube Cracked | Weld Cracked | |
| 1 | ANO 1 | 19.5 | 602.0 | B&W | BW | B/H | 69 | 8 | 7 | 2 | |
| 2 | Beaver Valley 1 | 12.4 | 595.0 | W | BW/CE | H/B | 65 | 4 | 4 | 0 | |
| 3 | Cook 2 | 13.0 | 600.7 | W | CBI | W | 78 | 3 | 3 | 0 | |
| 4 | Crystal River 3 | 15.6 | 601.0 | B&W | BW | B | 69 | 1 | 1 | 1 | |
| 5 | Davis-Besse | 17.9 | 605.0 | B&W | BW | B/H | 69 | 5 | 5 | 0 | |
| 6 | Millstone 2 | 10.5 | 593.9 | CE | CE | H | 69 | 3 | 3 | 0 | |
| 7 | North Anna 1 | 19.4 | 600.1 | W | RDM | S | 65 | 6 | 6 | 1 | |
| 8 | North Anna 2 | 18.3 | 600.1 | W | RDM | S | 65 | 42 | 8 | 42 | |
| 9 | Oconee 1 | 22.1 | 602.0 | B&W | BW | B | 69 | 3 | 3 | 2 | 4 |
| 10 | Oconee 2 | 22.0 | 602.0 | B&W | BW | B | 69 | 19 | 18 | 4 | |
| 11 | Oconee 3 | 21.7 | 602.0 | B&W | BW | B | 69 | 14 | 14 | 2 | |
| 12 | St. Lucie 2 | 12.3 | 595.6 | CE | CE | SS/H | 91 | 2 | 2 | 0 | 5 |
| 13 | Surry 1 | 18.6 | 597.8 | W | BW/RDM | H | 65 | 6 | 0 | 6 | |
| 14 | TMI 1 | 17.5 | 601.0 | B&W | BW | B | 69 | 8 | 7 | 4 | 4 |
| | | | | | | | *Unique Penetration Totals* | 124 | 81 | 64 | |

NOTES:

1. Key for Vessel Fabricators: BW = B&W, CBI = Chicago Bridge & Iron, CE = Combustion Engineering, RDM = Rotterdam Dockyard, CL = C.L. Imphy
2. Key for Material Suppliers: B = B&W Tubular Products, H = Huntington, S = Sandvik, SS = Standard Steel, W = Westinghouse, CL = C.L. Imphy, A = Aubert et Duval
3. The totals reflect nozzles that were found to have cracks requiring repairs.
   Other than the 16 small-diameter B&W thermocouple nozzles at two plants, all the cracked nozzles detected are either CRDM or CEDM nozzles.
4. Also all 8 small-diameter B&W thermocouple nozzles were found to be cracked.
5. The CEDM nozzle material at this plant was supplied by Standard Steel, and the ICI nozzle material was supplied by Huntington Alloys.

## Table 2 – Summary of RPV Head Nozzle Leakage

| Inspection Number | Unit | NSSS Supplier | Approx. EDYs at Insp. | Insp. Date | No. of CRDM Nozzles on Head | Number Leaking Penetrations (Note 1) | | | Repair Technique (Note 2) | Repair Method Would Likely Have Detected Significant Wastage? | Notes |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | Total | Due to Tube | Due to Weld | | | |
| 1 | ANO 1 | B&W | 19.6 | Mar-2001 | 69 | 1 | 1 | 0 | Embedded flaw | No | 3 |
| 2 | | | 21.1 | Oct-2002 | 69 | 1 | 1 | 0 | ID temper-bead | Yes | 4 |
| 3 | Crystal River 3 | B&W | 16.2 | Oct-2001 | 69 | 1 | 1 | 0 | ID temper-bead | Yes | |
| 4 | Davis-Besse | B&W | 19.2 | Apr-2002 | 69 | 3 | 3 | 0 | Replaced head | Yes | 5 |
| 5 | North Anna 1 | W | 21.4 | Mar-2003 | 65 | 1 | 0 | 1 | Replaced head | No | |
| 6 | North Anna 2 | W | 19.0 | Nov-2001 | 65 | 3 | 0 | 3 | Weld overlay | No | |
| 7 | | | 19.7 | Sep-2002 | 65 | 6 | 0 | 6 | Replaced head | See Note 7 | 6, 7 |
| 8 | Oconee 1 | B&W | 21.8 | Nov-2000 | 69 | 1 | 0 | 1 | Weld overlay | No | 8 |
| 9 | | | 23.2 | Mar-2002 | 69 | 1 | 0 | 1 | ID temper-bead | Yes | |
| 10 | Oconee 2 | B&W | 22.2 | Apr-2001 | 69 | 4 | 4 | 0 | ID temper-bead | Yes | |
| 11 | | | 23.7 | Oct-2002 | 69 | 10 | 7 | 3 | ID temper-bead | Yes | |
| 12 | Oconee 3 | B&W | 21.7 | Feb-2001 | 69 | 9 | 9 | 0 | ID temper-bead | Yes | |
| 13 | | | 22.5 | Nov-2001 | 69 | 5 | 5 | 0 | ID temper-bead | Yes | |
| 14 | Surry 1 | W | 19.1 | Oct-2001 | 65 | 2 | 0 | 2 | ID temper-bead | Yes | |
| 15 | TMI 1 | B&W | 18.1 | Oct-2001 | 69 | 5 | 1 | 4 | ID temper-bead | Yes | 9 |
| | | | | | *Unique Penetration Totals* | 51 | 31 | 20 | | | |

1. No CEDM, ICI, or other types of reactor vessel head nozzles have been found to be leaking (other than the B&W thermocouple nozzles at the two units that have this type of nozzle).
2. The "ID temper-bead" repair method for leaking nozzles involves cutting out the lower section of the nozzle, which makes the surface of the penetration hole in the head shell visible.
3. Although the 2001 repair of this nozzle would not have revealed the presence of low-alloy steel wastage, the subsequent repair in 2002 likely would have.
4. The leaking nozzle that was repaired in March 2001 was found to be leaking again in October 2002.
5. Detailed destructive examinations of the original Davis-Besse head have been performed to characterize the extent of wastage.
6. One of the leaking nozzles that was repaired in late 2001 was found to be leaking again in September 2002.
7. Several leaking nozzles have been extracted from the original North Anna 2 head and are expected to be examined for signs of wastage of the low-alloy steel shell material, among other tests.
8. Also 5 of the 8 small-diameter B&W thermocouple nozzles were found to be leaking.
9. Also all 8 small-diameter B&W thermocouple nozzles were found to be leaking.

Table 3
Orientation and Location for RPV Head Nozzle Cracks

| | | No. of Indications on the Nozzle ID | No. of Indications on the Nozzle OD | *Total* |
|---|---|---|---|---|
| **No. of Axial Tube Indications** | | 112 | 224 | *336* |
| **No. of Circumferential Tube Indications** | **Above Weld** | 0 | 7 | *7* |
| | **Weld Elevation** | 0 | 12 | *12* |
| | **Below Weld** | 6 | 10 | *16* |
| | ***Total*** | *118* | *253* | *371* |

Note: Craze cracking and other shallow indications with no depth detectable by UT are not included.

Table 4
Summary of Nozzle Circumferential Cracks Located Above the Weld or in the Weld Elevation Zone

| Unit | NSSS Design | Nozzle ID | Nozzle Angle (°) | Inspection Results | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | Date | Approx. EDYs | OD/ ID | Axial Location | Circ. Angle (°) | UH/DH Side | Depth (in) | TW Depth (%) |
| Crystal River 3 | B&W | 32 | 26.2 | Oct-01 | 16.2 | OD | above weld | 91 | DH | 0.29 | 47% |
| Davis-Besse | B&W | 2 | 8.0 | Mar-02 | 19.2 | OD | above weld | 34 | DH | 0.31 | 50% |
| North Anna 2 | W | 15 | 19.8 | Sep-02 | 19.7 | OD | ≥1.12" below root | 5 | DH | 0.23 | 36% |
| | | 41 | 33.1 | | | OD | ≥0.52" below root | 46 | DH | 0.10 | 16% |
| | | 54 | 38.6 | | | OD | ≥0.04" below root | 79 | UH | 0.23 | 36% |
| | | | | | | OD | ≥0.28" below root | 32 | DH | 0.16 | 25% |
| | | 59 | 40.0 | | | OD | ≥0.31" below root | 76 | DH | 0.15 | 24% |
| | | | | | | OD | ≥0.32" below root | 50 | UH | 0.15 | 24% |
| | | 65 | 42.6 | | | OD | ≥0.32" below root | 72 | DH | 0.15 | 24% |
| | | | | | | OD | ≥0.20" below root | 30 | UH | 0.08 | 12% |
| | | 67 | 42.6 | | | OD | ≥0.80" below root | 44 | DH | 0.09 | 15% |
| Oconee 2 | B&W | 18 | 18.2 | Apr-01 | 22.2 | OD | above weld | 36 | DH | 0.07 | 11% |
| Oconee 3 | B&W | 11 | 16.2 | Feb-01 | 21.7 | OD | over weld | 153 | DH | 0.36 | 57% |
| | | | | | | OD | over weld | 113 | UH | 0.25 | 40% |
| | | 23 | 23.2 | | | OD | above weld | 66 | DH | 0.22 | 35% |
| | | 50 | 35.1 | | | OD | above weld | 165 | UH | 0.62 | pin holes |
| | | 56 | 35.1 | | | OD | above weld | 165 | UH/DH | 0.62 | 100% |
| | | 2 | 8.0 | Nov-01 | 22.5 | OD | above weld | 48 | DH | 0.18 | 29% |
| | | 26 | 24.7 | | | OD | over weld | 44 | DH | 0.07 | 11% |

Table 5
Plants That Have Announced RPV Head Replacements

| Announced Head Replacement Plans as of September 2003 | | | | |
|---|---|---|---|---|
| **Status** | **Year** | **Season** | **No.** | **Unit Name** |
| Already replaced | 2002 | Fall | 1 | Davis-Besse |
| | | | 2 | North Anna 2 |
| | 2003 | Spring | 3 | North Anna 1 |
| | | | 4 | Oconee 3 |
| | | | 5 | Surry 1 |
| Replacing next refueling outage | | Fall | 6 | Crystal River 3 |
| | | | 7 | Ginna |
| | | | 8 | Oconee 1 |
| | | | 9 | Surry 2 |
| | | | 10 | TMI 1 |
| | 2004 | Spring | 11 | Oconee 2 |
| | | Fall | 12 | Farley 1 |
| | | | 13 | Kewaunee |
| | | | 14 | Turkey Point 3 |
| Replacing after next refueling outage | 2005 | Spring | 15 | Millstone 2 |
| | | | 16 | Point Beach 2 |
| | | | 17 | Turkey Point 4 |
| | | Fall | 18 | ANO 1 |
| | | | 19 | Farley 2 |
| | | | 20 | Point Beach 1 |
| | | | 21 | Robinson 2 |
| | | | 22 | St. Lucie 1 |
| | 2006 | Spring | 23 | Beaver Valley 1 |
| | | | 24 | Calvert Cliffs 1 |
| | | | 25 | St. Lucie 2 |
| | | Fall | 26 | Cook 1 |
| | | | 27 | Fort Calhoun |
| | 2007 | Spring | 28 | Calvert Cliffs 2 |
| | | Fall | 29 | Cook 2 |

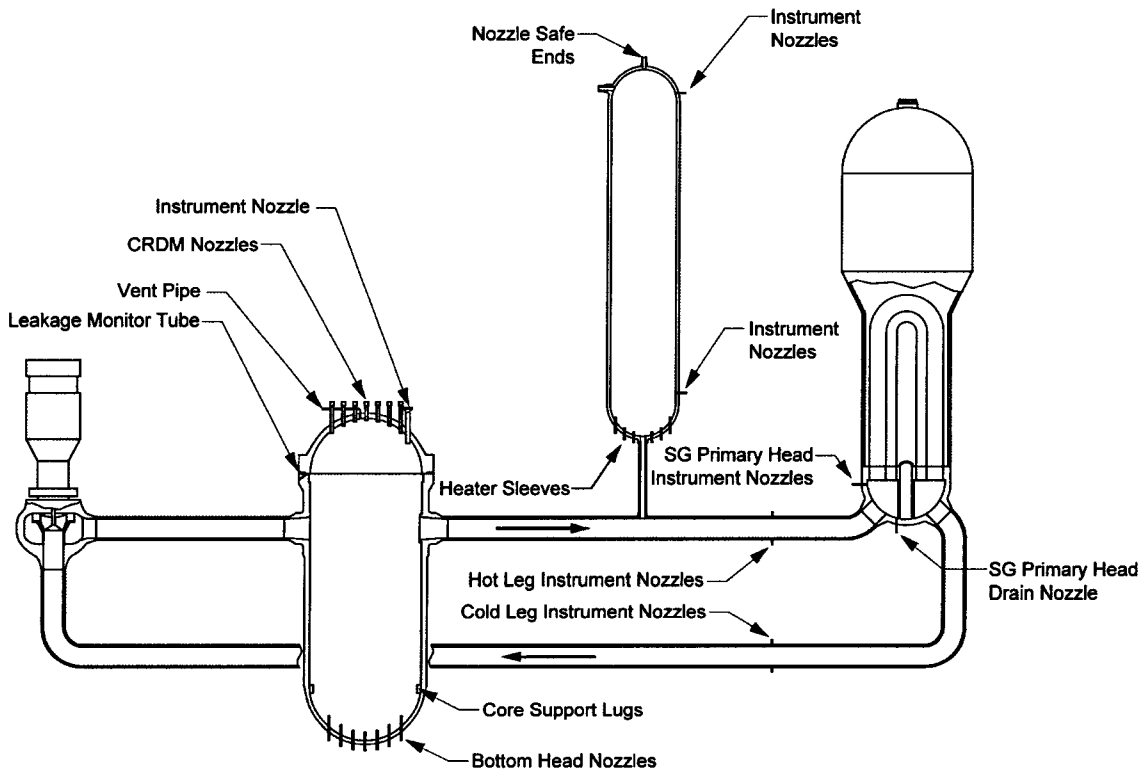


Figure 1
Locations of Thick-Section Alloy 600 Materials

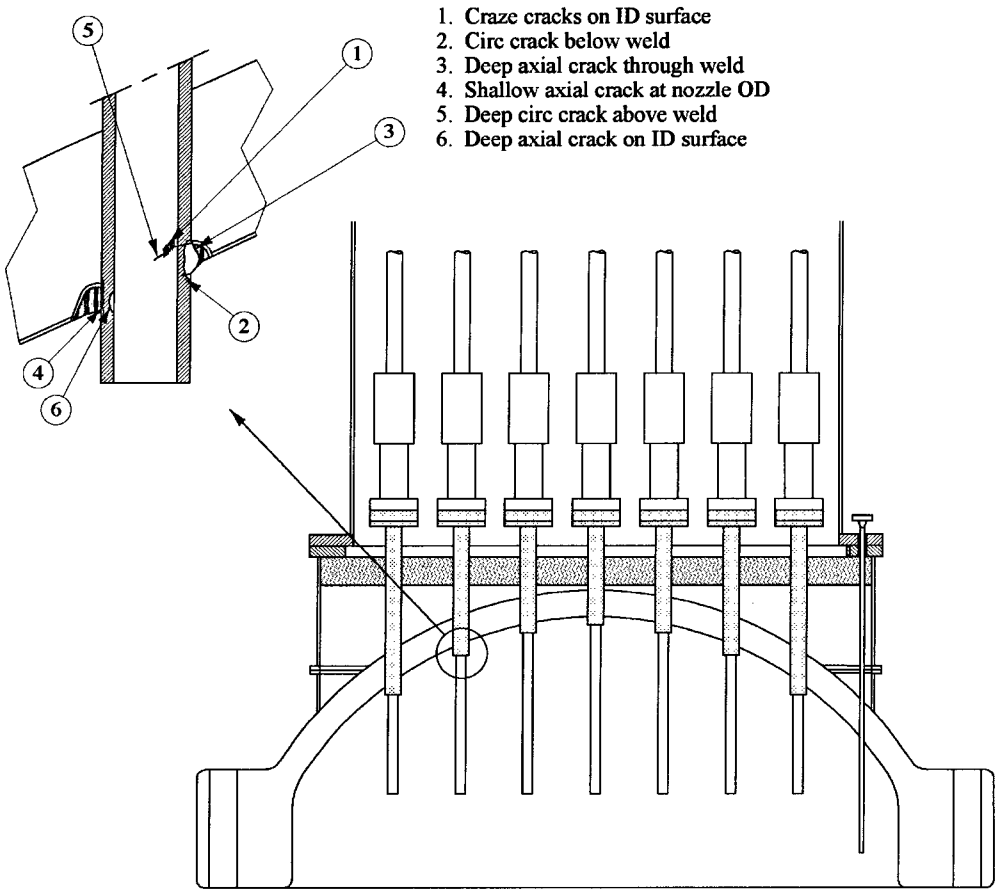


Figure 2
Location of Typical RPV Head PWSCC

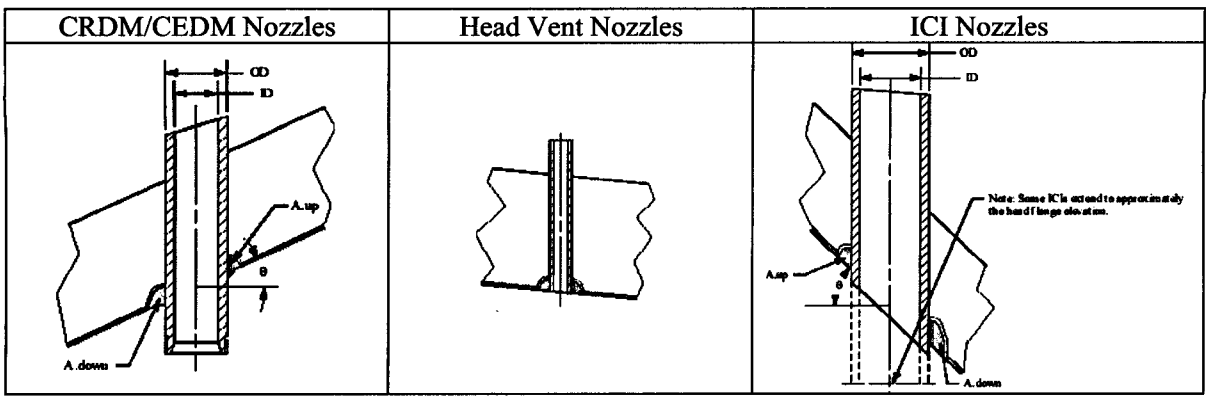


Figure 3
Typical RPV Head Penetration Types

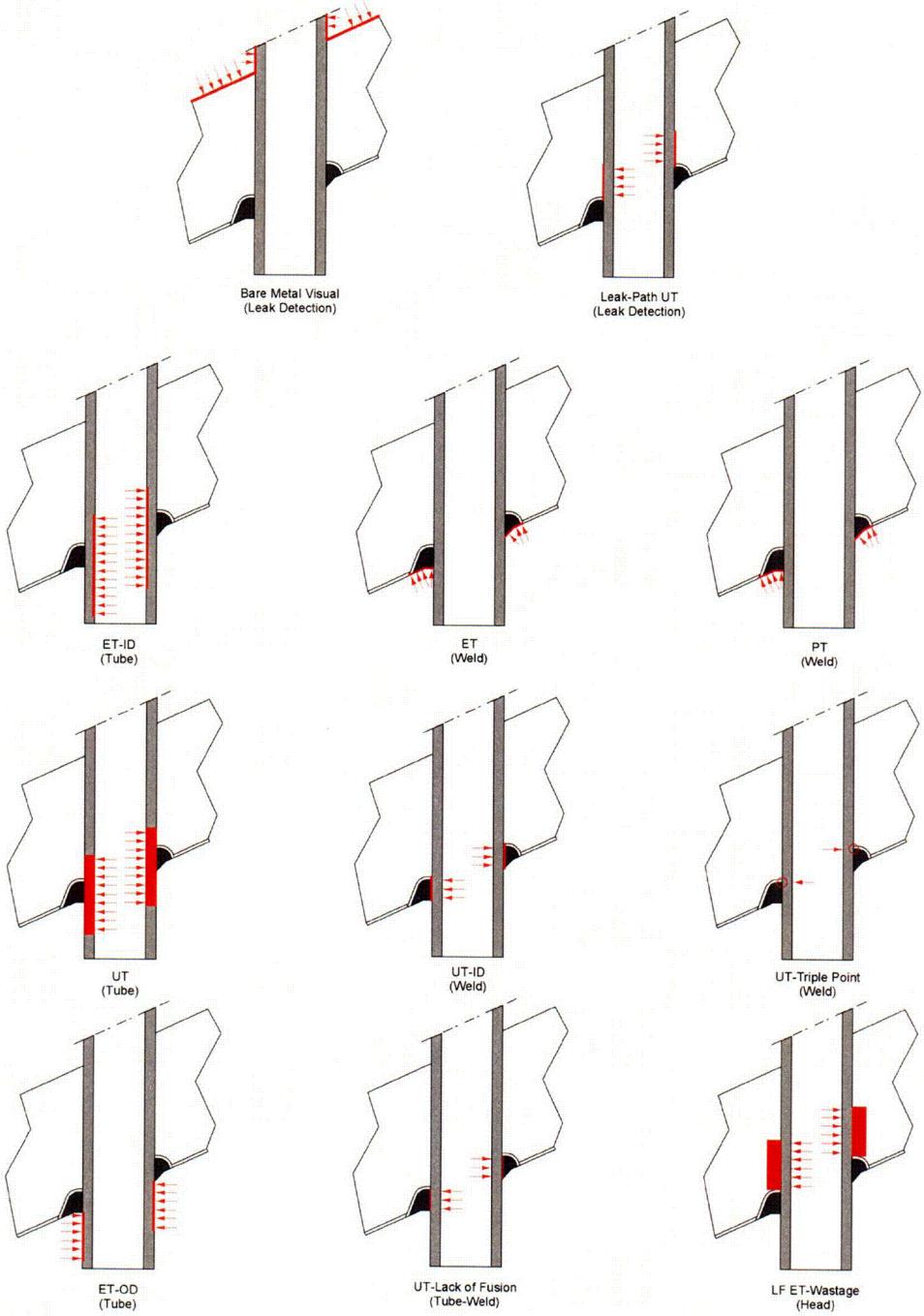


Figure 4
Summary of RPV Head Nozzle Inspection Techniques



C02

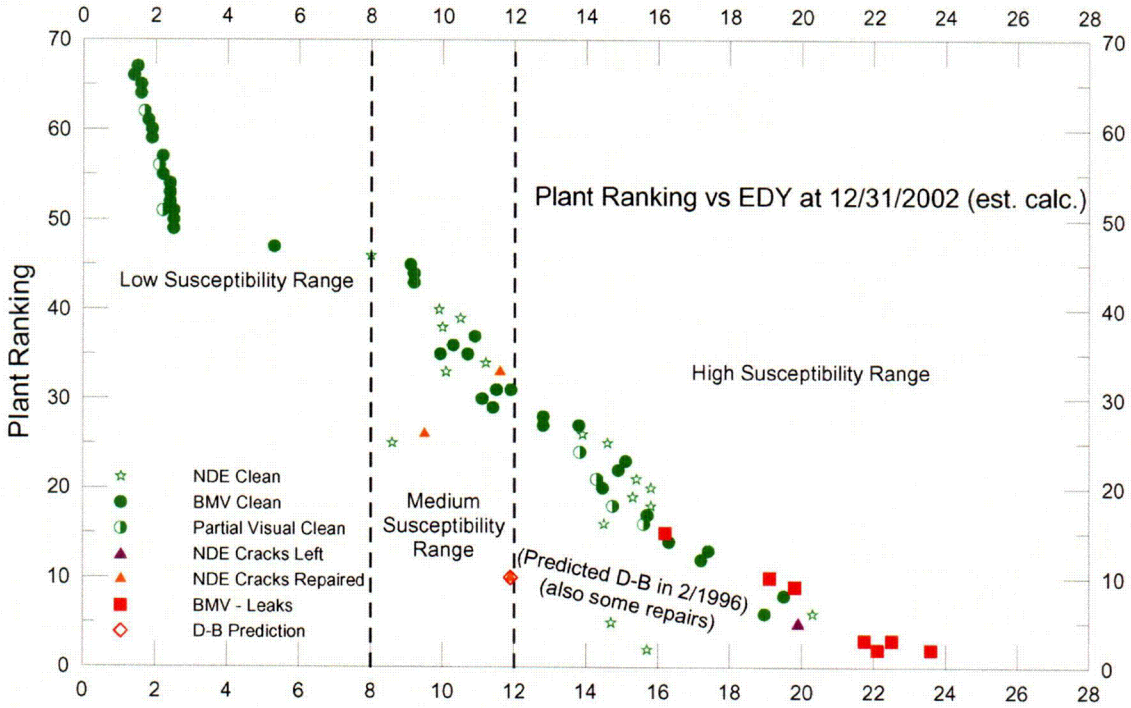


Figure 1. Ranking of domestic plants according to the EDY formula, showing results of inspections, evidence of leakage, and repairs. Many plants are shown with multiple symbols, indicating a "clean" inspection at inspection opportunity, followed by a different finding at a subsequent inspection (e.g., Oconee 2: clean NDE @ EDY=15.7, leaks and circ. flaws @ 22.1)

Figure 5
NRC Chart for Tracking Inspection Results[1]

[1] Attachment to Memo, L. Marsh, "Use of the 'EDY' Formula – Susceptibility Model Critique," NRR-2002-018 User Request & LLTF Recommendations Action Plan, July 14, 2003.



C03

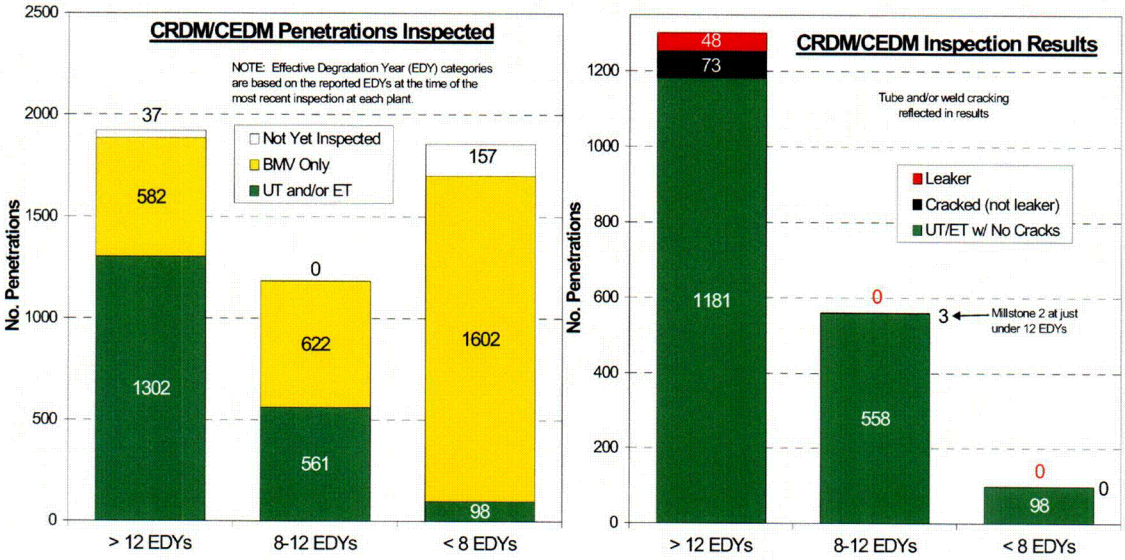


Figure 6
RPV Head Inspection Statistics – by EDY Group



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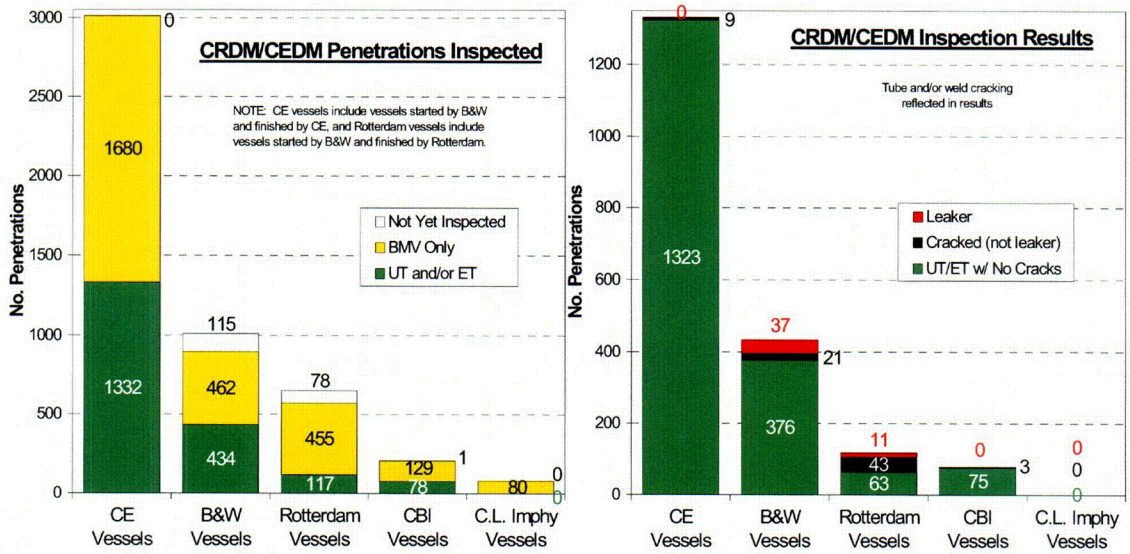


Figure 7
RPV Head Inspection Statistics – by Head Fabricator (All EDYs)

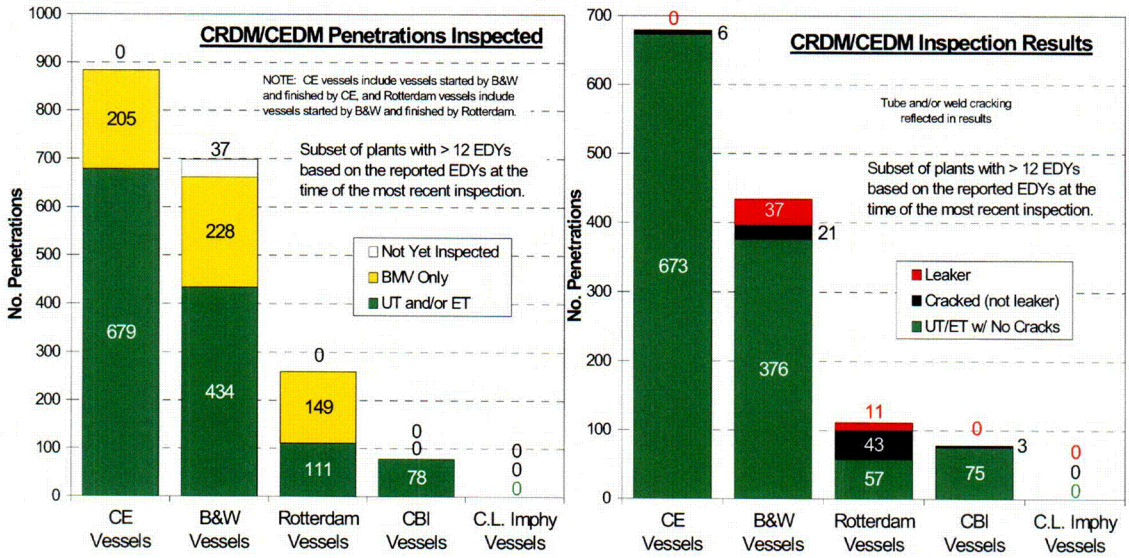


Figure 8
RPV Head Inspection Statistics – by Head Fabricator (>12 EDYs)



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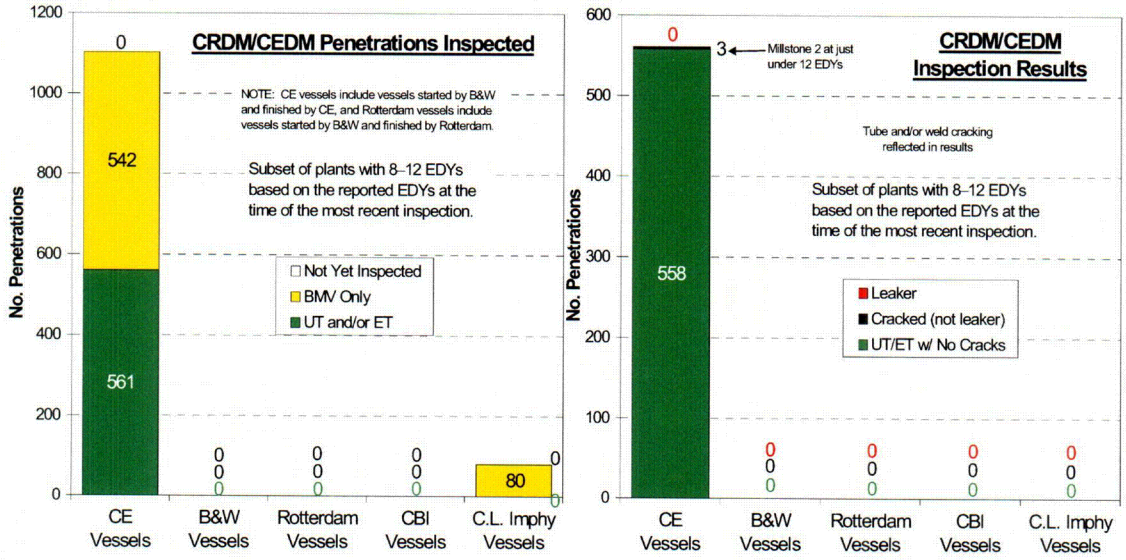


Figure 9
RPV Head Inspection Statistics – by Head Fabricator (8-12 EDYs)

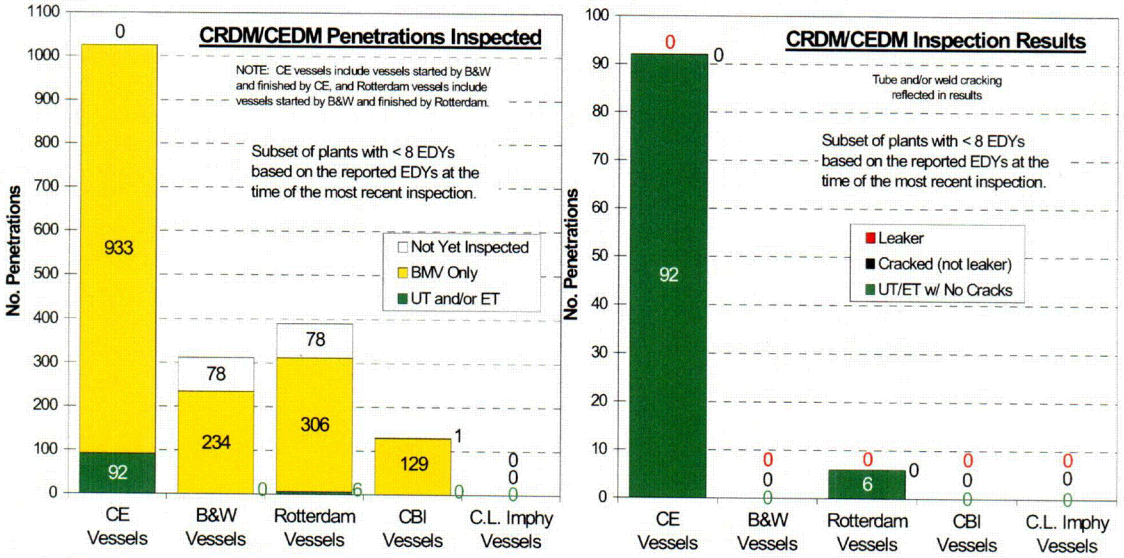


Figure 10
RPV Head Inspection Statistics – by Head Fabricator (<8 EDYs)



C06

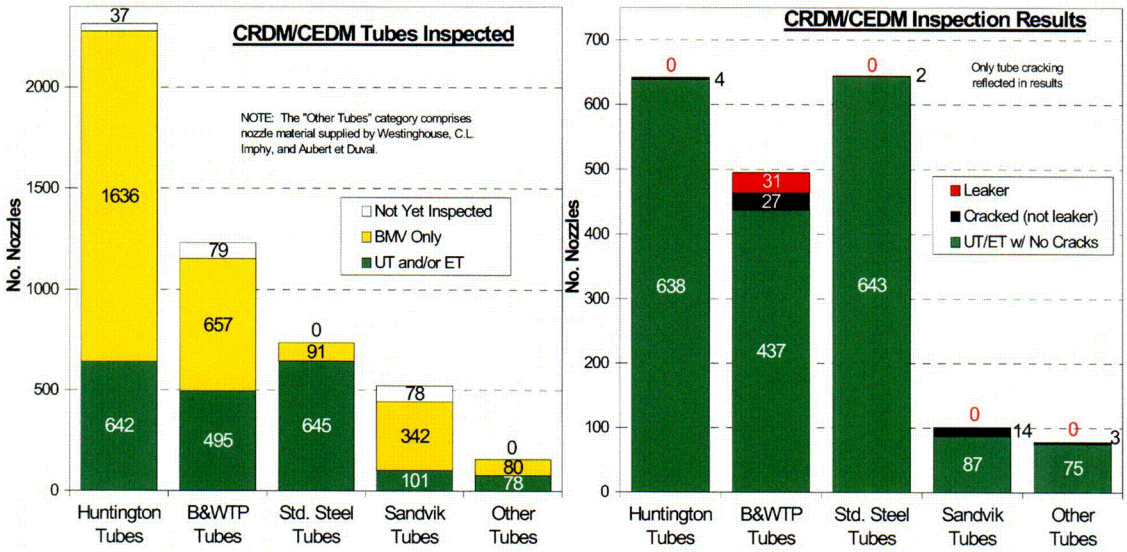


Figure 11
RPV Head Inspection Statistics – by Nozzle Material Supplier (All EDYs)

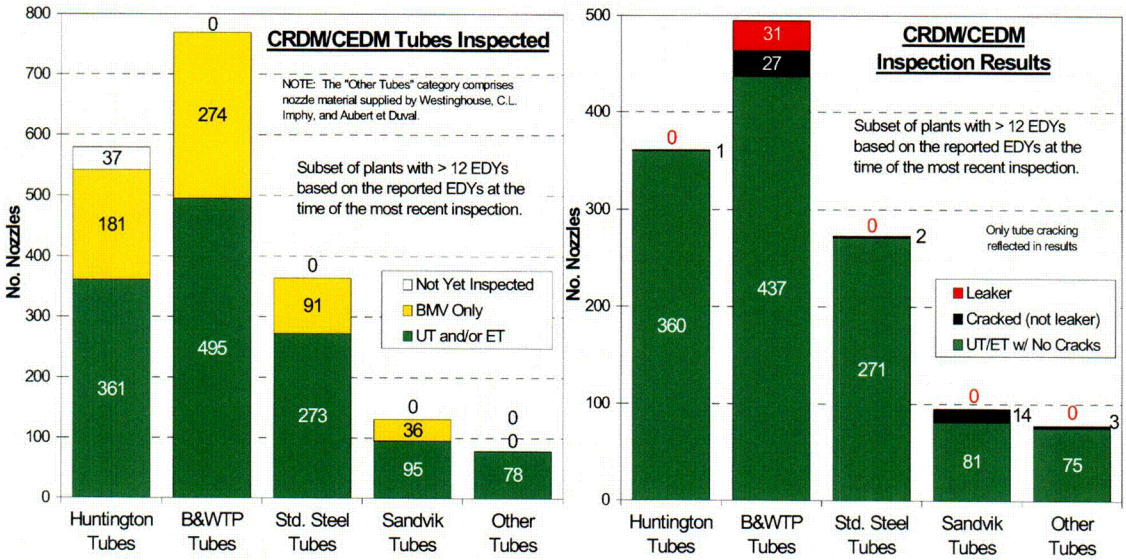


Figure 12
RPV Head Inspection Statistics – by Nozzle Material Supplier (>12 EDYs)



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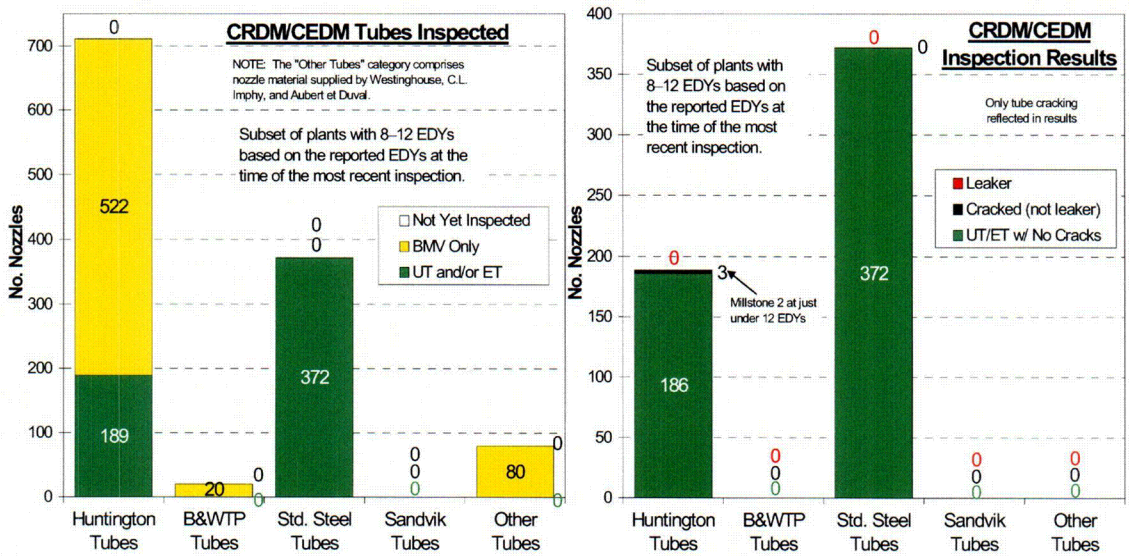


Figure 13
RPV Head Inspection Statistics – by Nozzle Material Supplier (8-12 EDYs)

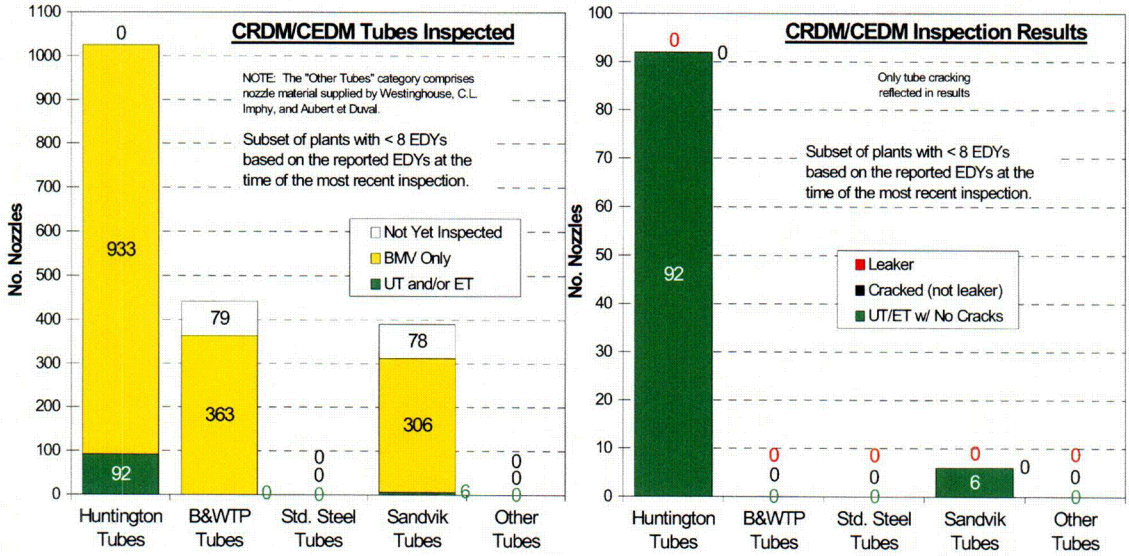


Figure 14
RPV Head Inspection Statistics – by Nozzle Material Supplier (<8 EDYs)



C08

# Inspection Technology for BMI Penetrations

**M.S. Lashley, South Texas Project, S. W. Glass and R.F. Cole, Framatome ANP Inc.**


## Abstract

Historically, United States (US) nuclear power plant inspections of the reactor vessel bottom-mounted-instrument (BMI) penetrations have been limited to visual verification via a combination of walk-downs and pressure tests. However in France, more than 18 campaigns had been performed to inspect the BMI penetration nozzles and welds since 1992 with no observed failures through 2002. In April 2003, South Texas Project Unit 1 discovered apparent leakage from two nozzles during a bare-metal examination. Based on the French inspection experience, Framatome ANP was selected for inspection and repair services to address the leaking nozzles. Inspection activities included ultrasonic examination (UT) of the tube, enhanced visual test (VT) and eddy current testing (ECT) of the J-groove weld, bobbin ECT and profilometry of the tube ID, helium leak test, phased-array UT, borescopic VT, and boat-sample removal with destructive metallurgical analysis. This presentation discusses BMI inspection technology particularly focused on the South Texas Project experience.


## BMI penetration description

The BMI penetrations at the bottom of the reactor vessel have similar characteristics as the vessel closure head penetrations. Although there are design perturbations among the different reactor designs, in general the construction is similar. An alloy 600 tube is welded to the vessel ID with a J-Groove weld. Although the tube ODs vary dramatically, the tube IDs range from approximately 0.390 to 0.75 inches. The gap between the vessel and the tube below the weld is not an interference fit. Some plants had the nozzles welded before thermal heat-treat of the vessel. This heat treat however was not directed to relieving the alloy 600 welds so even if the vessel were stress-relieved after welding, the stresses in the weld area would not be completely relieved. The under-vessel area is typically covered with insulation. Some plants have a comfortable gap between the insulation that facilitates under-vessel bare-metal visual examinations. Other plants have insulation designs that make effective bare-metal examinations impractical without removing some or all of the insulation.

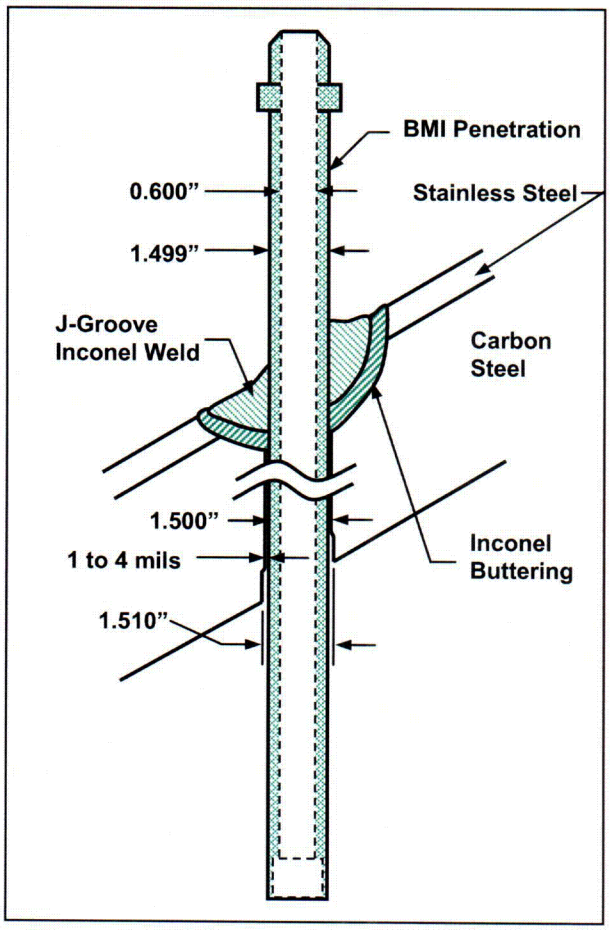


Figure 1: BMI general configuration

## Inspection History

Ever since the A-600 PWSCC susceptibility problem was recognized, bottom penetrations have been on the watch list for



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possible cracking as the nuclear fleet ages. In France, EDF commissioned inspection methodology to be developed and has conducted over 500 sample examinations since 1992. Framatome ANP was involved in most of those examinations. As a result of the increased incidence of head cracks, the MRP issued a recommendation encouraging US plants to perform bare-metal examinations of the heads and to extend the examination to also include the BMI nozzles (Reference 1). Following the South Texas Project experience, the NRC issued bulletin 2003-02 endorsing the MRP recommendation (Reference 2).

**South Texas Project Experience**

On April 12, 2003 during their planned bare-metal examination, South Texas Project observed 150 grams and 3 grams of boron deposit on nozzles 1 and 46 respectively (Figures 2 and 3). Framatome ANP was selected to support South Texas Project for the examination, analysis, and repair campaign based on the French experience. A comprehensive fast-paced program was conceived to analyze, understand, disposition, and repair the reactor vessel and return the unit to service. Ultimately inspection and analysis activities included:

- Bare-Metal examination
- Sample/Deposit Chemical Analysis
- Phased Array UT for wastage
- Enhanced VT examination of J-Weld
- Boat-Sample destructive examination
- Metallurgical Analysis of removed samples
- Eddy Current of Tube
- Eddy current of J-Weld
- UT volumetric Examination
- Helium Leak Test
- Profilometry
- Boroscope VT of tube ID

Each of these inspection and analysis technologies will be briefly discussed.

Figure 2: STP penetration 1

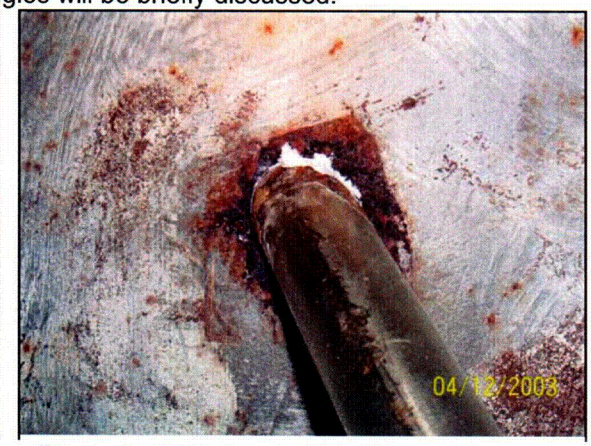


Figure 3: STP penetration 46

**Bare Metal BMI examination**

The bare metal examination is highly dependant on the under-head insulation design. Some plants have contour fitted metal plates with glass-wool packing that render a bare metal examination very difficult without removing most of the insulation. Other units like South Texas Project have observation ports and a comfortable plenum area between the insulation and the vessel that facilitate a bare-metal visual examination. South Texas Project has been performing bare-metal examinations of their BMIs every outage for more than 10 years. The examination typically takes less than an hour under vessel and the personnel receive 75 – 100 millirem. In previous years, no indication of leakage had been observed. .



C10

**Sample/Deposit Chemical Analysis**

Samples were collected and sent for laboratory analysis. The samples contained boron and elevated concentrations of lithium indicating fairly conclusively that the origin of the deposit was from inside the reactor vessel. The samples did not contain any iron thus there was no reason to suspect significant vessel erosion. Co-58 was also not present thus the deposits were more than 1 year old. Based on the ratio of Cs-134 to Cs 137, it had been 3 to 5 years since the material had been inside the reactor coolant system.

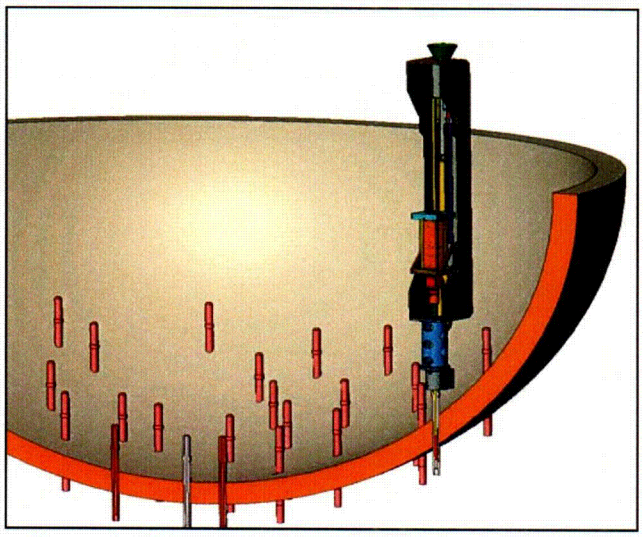

Figure 4: Representation of UT tool engaging with BMI penetrations

**UT of the penetration tube**

The French UT tool that had been used in more than 18 campaigns for EDF was engaged for the examination. Tool delivery required the bridge to be positioned over the target nozzle. The tool was then suspended from the bridge and lowered to cup the nozzles (Figure 4). Nozzle identification was aided with a laser positioning system to correlate the tool position with the nominal position of the penetration. The tool motions include clamping actuators to precisely align the probes with the nozzle, and probe motion control to generate a helical scan of the tube area of interest. The tool also includes an in-line calibration standard to assure transducer calibration and function without removing the probes.

Figure 5: UT probes for BMI examination included circumferential, axial & 0° configurations

The probe selections include an axial, a circumferential, and a zero degree probe. The zero degree probe focuses a narrow sound beam at the BMI tube wall OD to fully interrogate the wall material for anomalies and for weld profiling. The axial probe transmits a circumferentially oriented beam and a corresponding receiver "listens" for Time of Flight Diffraction (TOFD) to detect an axial crack. The circumferential probe uses a similar approach but the beam is directed axially to detect circumferential flaws. Although efforts are ongoing to combine these examinations into one probe, at the time of the South Texas Project campaign, the scan sequence was repeated three times – once for each probe type.

Before the examination, the technique was verified based on the previously performed French NDE qualification and on a blind South Texas Project flaw mockup. The mockup was fabricated with compressed EDM notches using the Cold-isostatic process (CIP) that was employed for the upper head mockups. All 58 penetrations were examined at South Texas Project.



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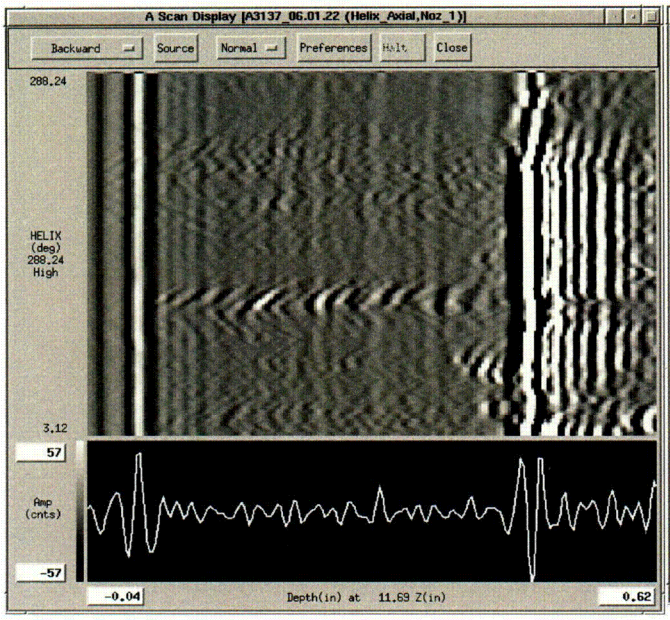


Figure 6: Penetration 1 axial probe with three axial indications and one full-penetration leak path

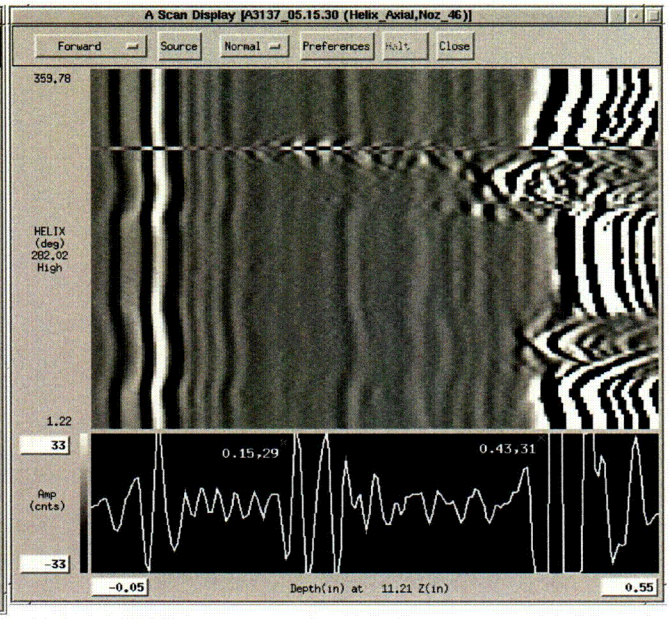


Figure 7: Penetration 46 with 2 axial indications and one leak path.

**Enhanced VT of J-Groove Weld Area**

An enhanced Visual Examination was performed using a high resolution camera to inspect the J-groove surface from the vessel ID. The Enhanced VT or EVT-1 focuses on a relatively small area such that a 0.5 mil diameter wire (0.0005 inches) can be seen. Normally in the French examinations, the EVT-1 examination is performed from the UT tool. Due to various logistic considerations at South Texas Project however, the examination was performed in parallel but with a separate tool. This tool included a water-jet to introduce clean water in front of the lens and to clean the viewing area as well as multiple lighting options.

**J-Groove Weld ET**

The J-Groove weld area of the BMI penetrations is very similar to the J-Groove welds of the head. The head examination tool pushes the probe against the shell and into the weld fillet. A rush program was undertaken to adapt a prototype upper-head probe for underwater service on the BMI welds. The tool was completed, demonstrated, and put into the field in 3 weeks. The South Texas Project tests performed the shell-side portion of the examination for 8 nozzles of interest including penetrations #1 and #46. These tubes were selected to confirm that none of the anomalous indications detected with UT had surface-breaking components that could subject the flaw to primary water and PWSCC. Although the inspection failed to detect the J-groove indication in tube # 1, subsequent correlation of the UT and bubble test and the boat sample analysis indicated that the probe did not scan far enough up the fillet region. Lessons learned from the initial deployment of this tool may serve the industry well for subsequent J-groove weld Eddy Current examinations.

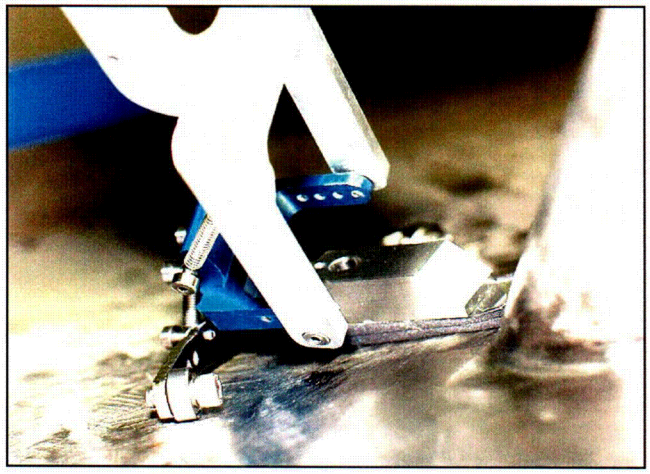

Figure 8: Compliant eddy current transducer pressed against shell and into weld fillet.

**Volumetric Phased Array Examination**



C12

One of the concerns with a boric acid leak was presence of wastage of the carbon steel material. Although difficult to understand how this can occur for the lower head configuration without being evident from the OD, Framatome ANP has observed limited wastage of the interior region of the upper head shell without significant wastage on either the ID or OD surface. A UT test was developed to examine the shell region where the penetration normally has no intimate contact with the shell. The technique uses a portable phased array instrument suitable for a manual scan of the shell area near the nozzle. The phased array transducer is manually moved radially around the penetration with the sector scan reflecting off the penetration bore. Any erosion or volumetric degradation is indicated as an enlarged void area between the tube and the shell bore. Sensitivity to this type of degradation was demonstrated with a calibration standard that included drilled holes (Figure 9).

**Helium Leak Test**

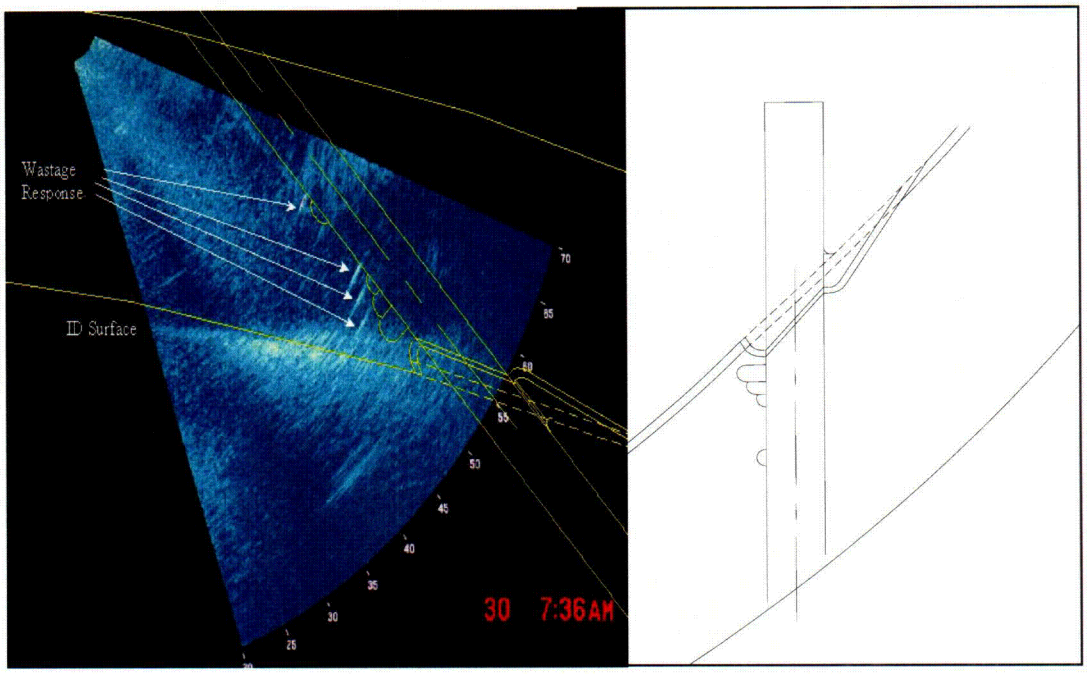


Figure 9: Phased array sector scan of shell volume calibration standard demonstrating sensitivity to wastage between surfaces.

A helium leak test was performed on BMI #1 and #46. A plenum was developed to seal against both the tube OD and the vessel shell. The plenum was pressurized to over 150 psi. This was actually only about 120 psi over the water head pressure under 70-ft of water. The BMIs were carefully monitored from inside the vessel. In BMI # 1, a small bubble was observed apparently coming from the J-Groove weld area approximately every few seconds. No bubbles were observed from BMI #46.

**Eddy Current and Profilometry of leaking penetrations**

A bobbin eddy current examination was performed on four tubes including the two tubes to be repaired. Tube #1 and #46 also received a multi-element eddy current profilometry examination. The bobbin probe was pulled to identify any cracks that broke the tube ID surface thereby subjecting the flaw to primary water. The large flaw detected with UT in the leaking tube #1 was confirmed with Eddy Current. This enhanced confidence in the examination. No other tubes had ID-surface-contacting-crack indications.



C13

In addition, the tube profile was characterized to establish a benchmark tube diameter and ovality or anomalous geometry prior to the repair.

**Borescope VT of severed penetration ID**

A borescopic VT of the severed tubes was performed to further confirm no cracks were present in the weld area. No cracks were observed.

**Boat-sample Extraction and Destructive Analysis**

A boat-sample was taken from both penetration locations. Only penetration 1 produced enough of a sample to destructively analyze. The boat-sample was prepared using an EDM scope designed to cut-out a section of both the tube and the J-Groove weld area in the vicinity of the crack indication. The samples were then taken to a hot-cell for destructive metallurgical analysis. The destructive tests corroborated the NDE assessment for BMI #1.

**Conclusion**

NDE technology exists to interrogate the BMI tubes. TOFD UT has been demonstrated to detect and size flaws in the BMI tube. Standard ultrasonic techniques provide meaningful insight into the tube to weld interface. Significant advancements have also been made to investigate the weld surface as well as the annular region surface. It is appropriate to continue refining the technology and tooling to incorporate lessons learned in an effort to improve the application for future BMI inspections.

**References**


1. Material Reliability Program (MRP) letter from Leslie Hart, MRP Senior Representative dated June 23, 2003 recommending Bare-Metal examinations during next or upcoming outages.
2. NRC Bulletin 2003-02, Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity, August 23, 2003, OMB control No: 3150-0012.


**Acknowledgements**


1. South Texas Project Nuclear Generating Station: Performed bare-metal examination, detected and collected the leakage indication, and lead the campaign to understand, characterize, and repair leaking BMI penetrations.
2. Framatome ANP: Provided technology and personnel to perform UT examinations, Eddy Current of J-groove and penetration ID, Profilometry, EVT-1, Volumetric Phased-Array wastage examination, borescope VT of severed penetration tube, helium leak tests, boat-sample extraction, engineering analysis, and repair technology.
3. BWXT: Performed hot-cell destructive analysis of boat-sample
4. SWRI & M&M Engr.: Performed chemical analysis of Residue.
5. EPRI: Fabrication of mockups and consultation on Demonstration program and examinations.
6. PCI: Boat sample extraction

**EPRI MRP Alloy 600 RPV Head Penetration Inspection Demonstration Program**

**T. Alley, Duke Energy, Charlotte, NC, and**
**K. Kietzman and F. Ammirato,**
**Electric Power Research Institute, Charlotte, NC**

MRP has developed and implemented a NDE demonstration inspection program that focuses on detection of safety-significant circumferential cracking on the OD of the penetration base material, weld flaws, and flaw location and sizing commensurate with the MRP RPV Head Inspection Plan. This program provides a means for evaluating and demonstrating NDE technologies and techniques to effectively inspect RV head penetrations for flaws that initiate from the surface of the weld and from the OD of the penetration. The NDE mock-ups are used by inspection vendors for procedure refinement and personnel training.

**Manuscript was not available for publication in the Proceedings**

# PRODUCTION OF A REALISTIC ARTIFICIAL FLAW IN AN INCONEL 600 SAFE-END

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The importance of NDT qualification has received significant attention during the recent years. Recent findings of cracks in Inconel 600 in different NPP components have also increased interest in the reliability of in-service inspections of this material. This, in turn, sets challenge for manufacturing of representative qualification specimens and flaws. A new, advanced flaw production technique has become available. The technique enables production of realistic cracks to ready-made mock-ups without implanting or welding.

This paper describes the advanced crack production technique and its application to Inconel 600. A realistic, controlled crack was produced to a core spray nozzle safe-end mock-up. The technique produces true fatigue cracks, which are representative of most real, service-induced cracks. The technique is applicable to any shape or size of component and results only in an intended crack without unwanted disturbances. The technique allows production of a single or separate cracks as well as different combinations of them.

In addition to the controlled crack production, the paper introduces studies of the effects of different thermal fatigue loading cycles on the ultrasonic response obtained from the crack in Inconel 600. Results of the study show the effect of different thermal fatigue loading cycles on the obtained ultrasonic response during dynamic loading of the artificially produced crack. Control of crack growth and relationship between loading parameters and ultrasonic response are discussed.

## Introduction

The last decade has brought new challenges for the nondestructive testing in the nuclear power field. Several through-the-wall leakages in components and structures that have not been covered by in-service inspection programs have gathered attention of the whole nuclear community. One of current concerns is the primary water stress corrosion cracking of Inconel 600 alloy and its weld metals in the pressure vessel head and bottom penetration nozzles. This type of degradation and crack growth was not originally considered in components in question.

The NDE qualification procedures are still under development all over the world. This includes development of better flaw production techniques producing representative flaws. There are certain factors that have to be taken into account when a flaw is used as a reflector for ultrasonic inspection. The ultrasonic response is affected by different crack characteristics, among others, location, orientation and size of a crack[1], the opening of a crack and crack tip[2,3,4], the remaining residual stresses in the material[5,6], fracture surface roughness[7,4], crack tip plastic zone[8] and filling of the crack with water[9]. These characteristics of cracks affect propagation, reflection, diffraction, transmission, attenuation and diffusion of ultrasonic energy[9,10].

Wüstenberg et al.[11] mentioned, that if the main interaction of a flaw used in qualification is based on the crack tip diffraction, the only possibility would be use of service-induced flaws as cut outs from real components and weld implant them to qualification mock-ups. This was based on the fact that there was no flaw production technique capable of producing realistic cracks or flaws which represent sufficient weak crack tip diffraction. Hence, there is a need to develop a flaw manufacturing technique that is capable of producing realistic flaws representative from all typical characteristics point of view.

A novel artificial flaw production technique and its applicability for Inconel 600 is introduced in this paper. The technique is used to produce a realistic crack in a core spray nozzle mock-up component of a BWR-type nuclear power plant. Furthermore, the ultrasonic response of the crack under dynamical thermal loading was studied in order to understand the relationship between ultrasonic response and different crack opening conditions.

**Materials and Methods**

The flaw production technique is based on thermal fatigue loading. Loading is applied by high frequency induction heating and water or air spray cooling. Produced flaws are representative of real, service-induced fatigue flaws in metallographic sense and hence they are supposed to be representative also in terms of NDE response. The technique allows production of realistic flaws with controlled location, orientation and size. Characteristics of flaws produced with the technique are introduced in more detail in references [12,13,14]. The technique is applicable to different materials and virtually any shape or size of a sample. The only requirement for the crack production is that the intended location must be accessible.

**Sample**

This paper introduces flaw production to a full-size core spray nozzle, safe-end mock-up (BWR-type nuclear power plant). Figure 1 shows the nozzle consisting of three different materials: A508 carbon steel, Inconel 600 and AISI 316 type austenitic stainless steel. There is a buttering and a joint weld between the carbon steel (with cladding on the inner surface) and Inconel 600 safe-end, and a butt weld between Inconel 600 safe-end and AISI 316 austenitic stainless steel pipe. Both welds were made with Inconel 182 filler material with Inconel 82 root pass. After welding the working allowances were machined away. The finishing machining removed the root pass so, that the welds of the ready-made mock-up are Inconel 182.

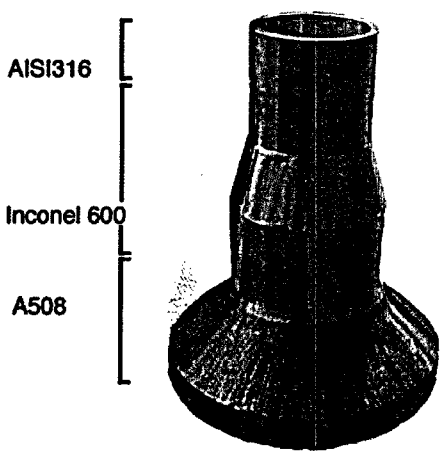


**Figure 1** Core spray nozzle mock-up with Inconel 600 safe-end.

Figure 2 shows the drawing of the nozzle mock-up and the intended location of the flaw production. The intended location is in Inconel 600 in the HAZ of the buttering weld. The wall thickness of the Inconel 600 safe-end in the intended location is 23 mm. Nozzle was received as ready-made and no machining or welding was allowed. Flaw was to be produced to the inner surface in as-received condition of the nozzle. The specimen was nondestructively tested after flaw production and no destructive tests were performed.

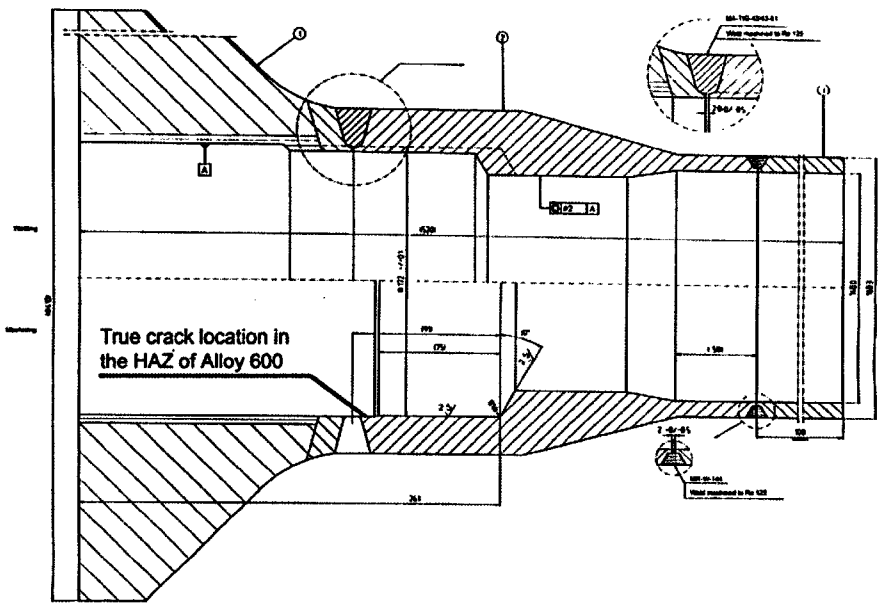


**Figure 2** Drawing of the core spray nozzle and location of the produced crack in Inconel 600 safe-end in the heat affected zone of the buttering weld.

## NDT set-up

A pulse-echo shear wave probe (41°, 1.5 MHz) was used when performing the inspection of the nozzle after crack production. The same probe was used during the studies of the relationship between ultrasonic response and crack loading. These studies were performed with a ready-made crack. The probe was attached on the outside surface of the mock-up and the surface breaking crack in the inner surface was monitored through the wall, in front of the weld. Ultrasonic signals were gathered in-situ during continued thermal fatigue cycling of the crack. Details about the NDT measurement system are given in reference[15].

## Applied loads

In order to study the effect of different loadings, two different thermal fatigue loading cycles were applied. Temperature curves of applied cycles are shown in Figure 3 as measured from the sample surface. The first cycle (B1) had high heating rate and short cooling time with heating and cooling times of 10 and 15 s, respectively. The second cycle (B2) had lower heating rate and longer cooling time with heating and cooling times 20 and 25 s, respectively. Water spray cooling was applied for both cycles. The first cycle reached higher temperature than the second cycle. In order to see the effect of the stabilised cycles, B1 loading was applied as 20 and B2 as 16 successive cycles.

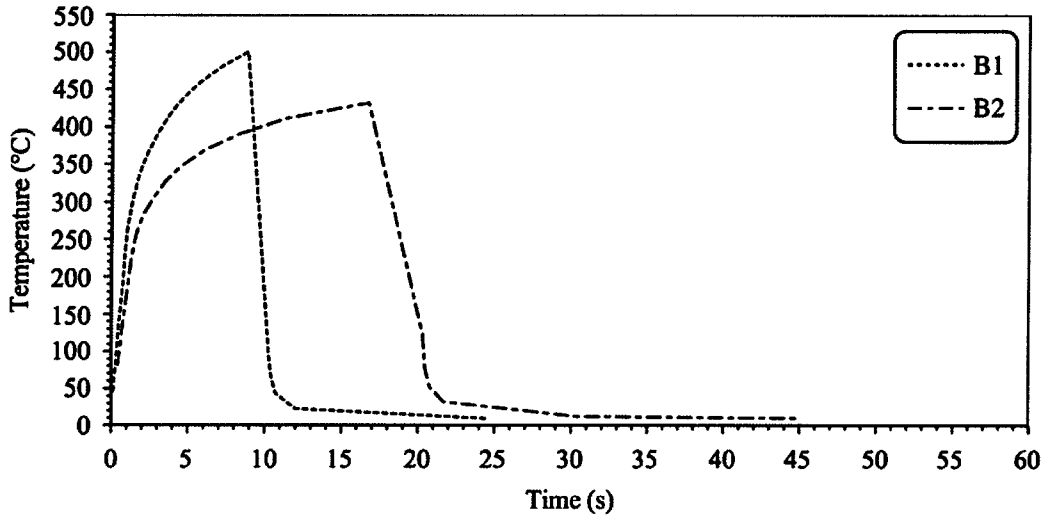


**Figure 3** Two different temperature loading cycles used in the studies.

**FEM-analysis**

Applied cycles were analysed by finite element modeling (FEM) giving results of temperature and strain distributions through the material thickness during dynamical loading. Used finite element model is presented in more detail in reference[16].

**Results**

A realistic crack was produced in the inner surface of the nozzle. Figure 4 shows the dye penetrant indication of the produced single crack in Inconel 600 safe-end in the heat affected zone of the buttering weld. The weld is located in the upper part and Inconel 600 base material in the lower part of the figure. The length of the crack is 14.2 mm and the depth is 5 mm, thus being about 22% through the wall. The maximum surface opening of the crack varies locally between 30 – 45 μm. In the figure, there is also a very small (less than 1 mm deep) secondary indication in the corner of the shoulder visible in the lower part of the figure. The initiation of the secondary crack was caused by the stress rising effect of the shoulder. Without vicinity ofsuch a stress riser, there would have been no secondary cracking. The secondary indication does not affect the performance of ultrasonic testing as it is located about 7 mm away from the actual crack.

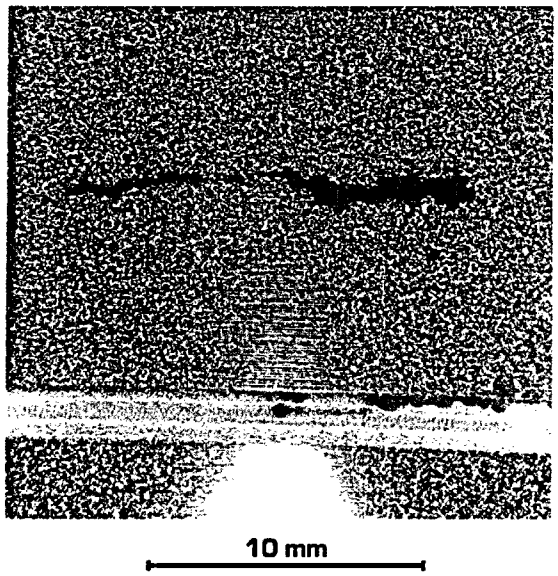


**Figure 4** Dye penetrant indication of the produced realistic crack in Inconel 600 safe-end in the heat affected zone of the buttering weld.

The size of the crack was controlled by process control during the production and confirmed by ultrasonic testing. The obtained signal from the crack at room temperature is shown in Figure 5. The reflections from crack opening corner and subsurface parts of the crack are visible in the figure. The ultrasonic inspection sized the crack to be 18 mm long and 6 mm deep. The measured length by ultrasonic testing is clearly bigger than the actual value as seen from Figure 4. Also the measured depth differs from the given process value, but it lies inside the production tolerances (±1 mm).

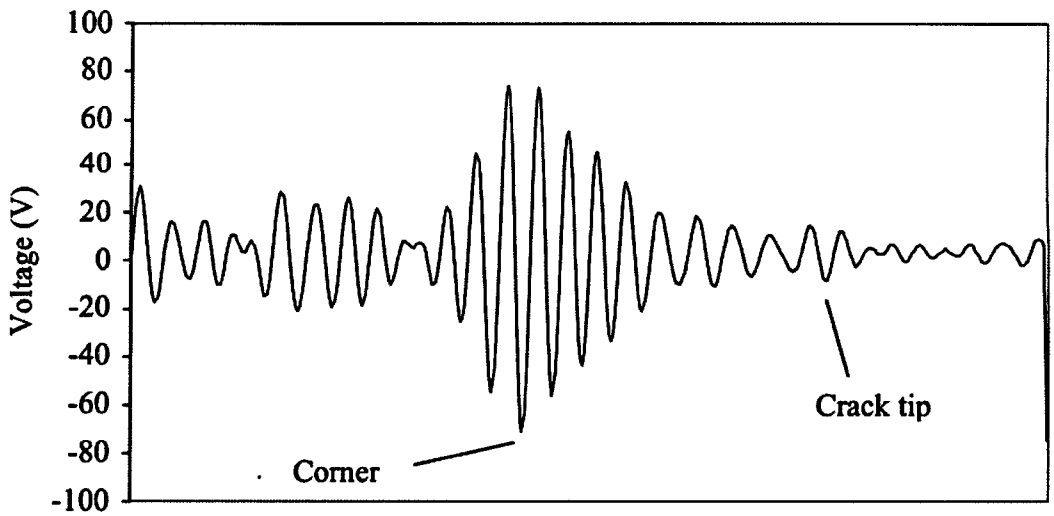


**Figure 5** A-scan obtained from the crack at room temperature (41°, 1.5 MHz, shear wave probe).

The studies of ultrasonic response versus dynamical thermal loading resulted in a large amount of ultrasonic data. Figure 6 shows the ultrasonic signal obtained from the crack in the end of cooling and heating phases of cycle B2. The figure clearly shows the differences between different crack opening states. Results shown in the figure have been obtained in the turning points of surface temperature cycles.

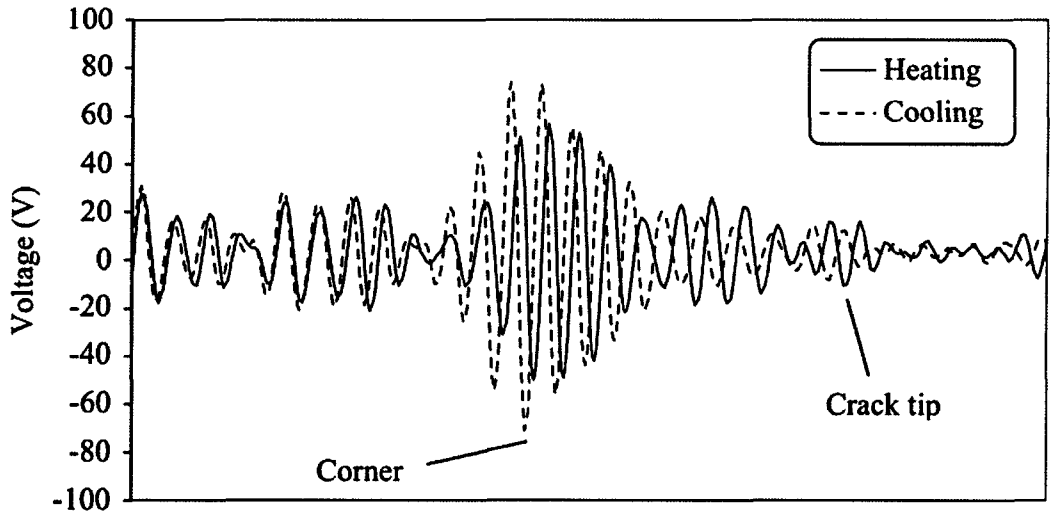


**Figure 6** A-scans from the crack in the end of cooling and in the end of heating of thermal fatigue loading cycle. Differences in ultrasonic response are related to the crack opening and closing behaviour.

Results of finite element modeling gave temperature and strain distributions through the wall thickness. Figures 7 and 8 show solved strain distributions for analysed cycles B1 and B2, respectively. Nozzle ID is in the left side and OD on the right side of both figures. The results clearly show the difference between the faster and slower loading rates.

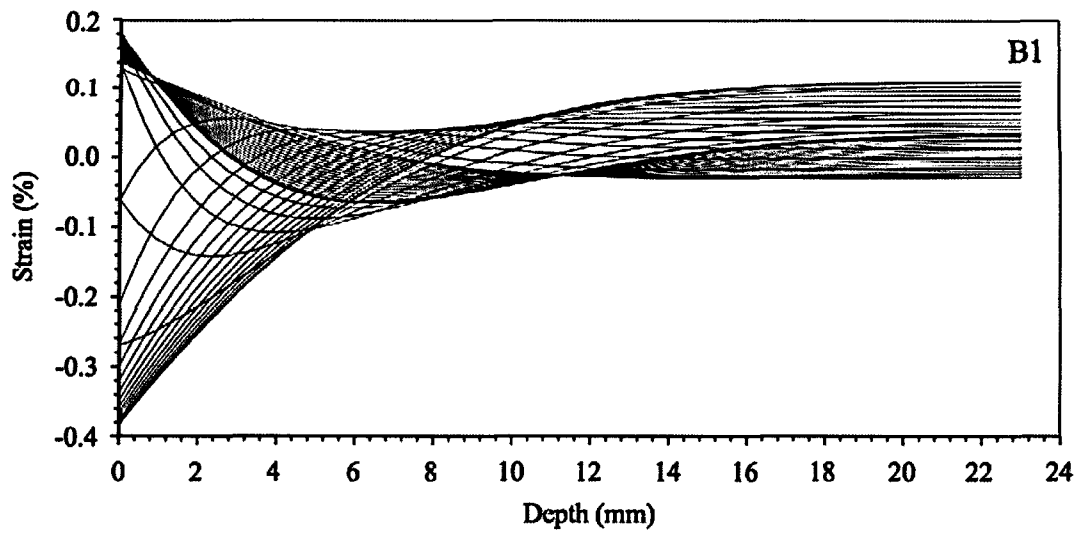


**Figure 7** Strain distribution for loading cycle B1. Nozzle ID on the left and OD on the right side of the figure.

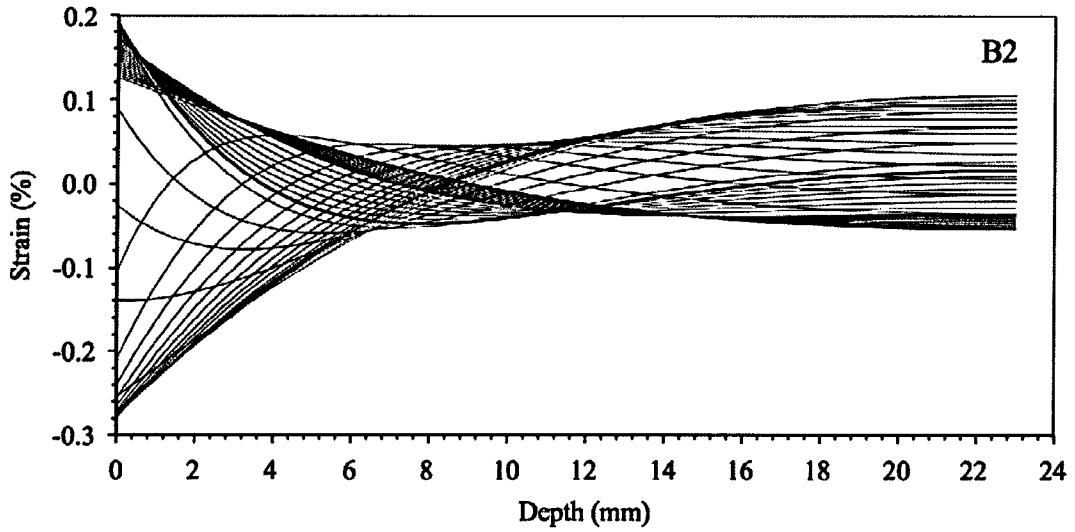


**Figure 8** Strain distribution for loading cycle B2. Nozzle ID on the left and OD on the right side of the figure.

The obtained ultrasonic signal amplitudes from corner reflection and crack tip varied during the loading. These variations are related to the opening behaviour of the different parts of the crack. Figures 9 and 10 show the combined results of strain variations from modeling and measured changes of ultrasonic amplitudes from corner reflection and crack tip for cycles B1 and B2, respectively.

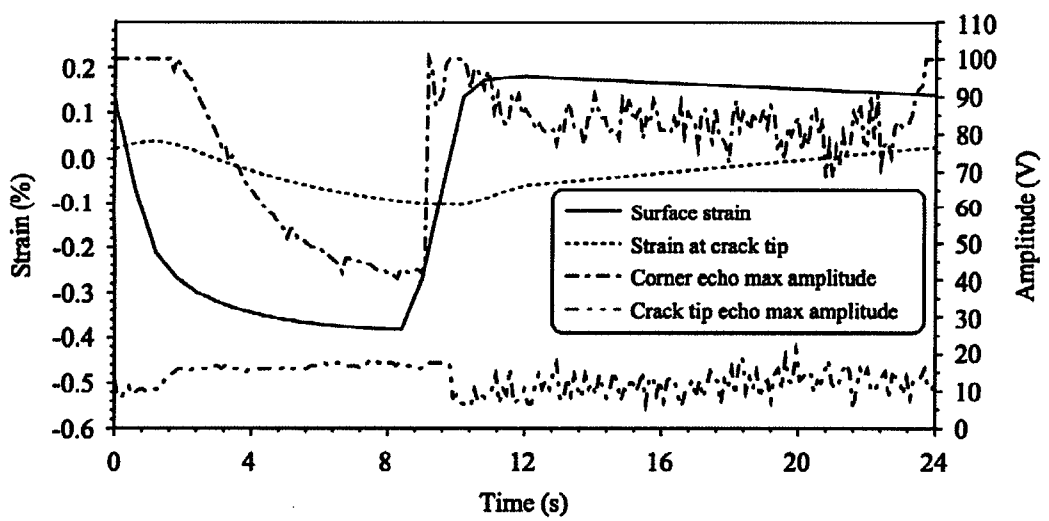


**Figure 9** Combined results of strain and ultrasonic amplitude variations from crack corner and tip caused by loading cycle B1.

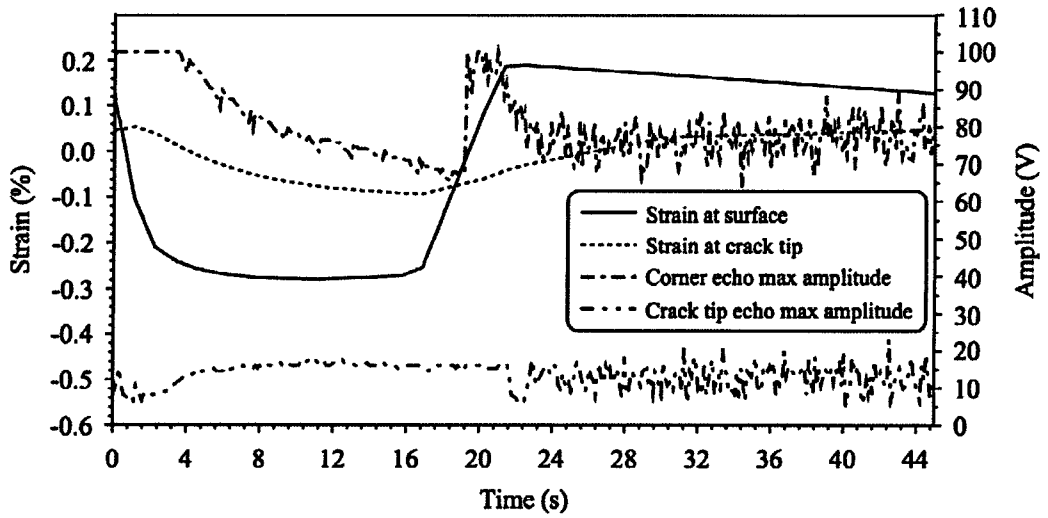


**Figure 10** Combined results of strain and ultrasonic amplitude variations from crack corner and tip caused by loading cycle B2.

**Discussion**

The results show that a realistic crack was produced in the heat affected zone of the buttering weld in Inconel 600 safe-end, as intended. The flaw location and size were accurately controlled. The dye penetrant indication shows a single, tortuous crack, which has a natural propagation in the heat affected zone of the buttering weld. The crack is narrow and its opening varies in different parts of the crack.

The ultrasonic response is determined to be a crack-like indication. Similarly, the amplitudes from corner, face and crack tip are representative and set realistic challenge for the inspection. Ultrasonically the produced crack represents a difficult reflector caused by its realistic characteristics. The realistic crack causes unhomogeneous reflections affecting the detection. The tight crack tip and small crack tip radius make the sizing of the crack challenging.

It was shown that the technique is applicable to ready-made mock-up without causing any alterations to the component. The results show, that the technique fulfills the important factors to be taken into account when performance demonstration is designed and an artificial flaw is used as a reflector. These factors include correspondence of reflector dimensions and dynamic range of echo amplitude, representativeness of position, orientation, fracture surface roughness and reproducibility of the artificial reflector both metallografically and echodynamically[1,11].

The results of ultrasonic response versus thermal fatigue loading show how different parts of the crack are opening and closing at different time moments. For example, the corner amplitude decreased during heating and increased during cooling. While the crack tip amplitude increased during heating and decreased during cooling. That is, crack tip amplitude changes were opposite to the corner amplitude.

Amplitude decrease is caused by crack closure and increase by opening of the crack. It is known that the surface breaking part of the crack is closed during heating and opened during cooling as described, e.g., in reference[17]. However, the ultrasonic results of the crack tip

amplitude show, that the tip is openend during heating and closed during cooling. This is caused by temperature cycling inducing stress gradients in the specimen. During heating the surface layer of the material is heated up and experiencing increased compressive stresses. At the same time, subsurface parts of the crack are at lower temperature and may be under tensile stress. The increase of crack tip amplitude during heating clearly indicate that the crack tip is opened, i.e. under tensile stress.

The finite element modeling, however, shows different results for the strain variations in the depth of the crack. For both analysed cycles the model shows decreasing strains during heating and increasing strains during cooling at the crack tip. This is explained by the fact that the model was made for solid material and does not take into account the flaw in the material.

## Conclusions

The novel artificial flaw production technique is available for different materials including Inconel 600. The technique is applicable to full size mock-ups with challenging multi-material structures. Flaw production does not cause any unwanted alterations and is applied to ready-made, finished surfaces. The produced flaws are realistic thermal fatigue cracks. Cracks are tortuous, tight, narrow and have a small crack tip radius. Hence, the reflection properties of produced cracks are realistic.

Flaws produced with the new technique can be used in NDE training and qualification purposes. The accurate positioning, control of crack size and reproducibility offer an opportunity to have realistic reflectors in testing, training or qualification specimens. The production process does not set any requirements for the specimen and, hence, also specimens with existing flaws can be used.

## Acknowledgements

This work was performed in a research and development project funded by Technology Agency Finland (Tekes), Trueflaw Ltd., Pacific Northwest National Laboratory (PNNL, USA), TVO Oy and Fortum Nuclear Services Ltd. The participants are acknowledged for giving the funding, delivery of test materials and technical support.

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## Generic Guidance for an Effective Boric Acid Inspection Program for Pressurized Water Reactors

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**T. Satyan Sharma, American Electric Power Company and**
**D. Weakland, First Energy Nuclear Operating Company**

This paper summarizes the scope and key elements of the Generic Guidance document that has been developed recently (Ref.1) to aid utilities in developing plant-specific Boric Acid Corrosion Control Programs (BACCPs) and procedures for PWR plants. The document is the result of an effort undertaken by Westinghouse Owners Group as part of the industry initiative to address the primary water stress corrosion cracking (PWSCC) and subsequent leakage of Alloy 600 reactor vessel head penetration tubes, and the resulting reactor vessel head wastage at the Davis-Besse nuclear plant (Ref.2). The guidance provides a structured approach for the inspection and mitigation of boric acid leakage and corrosion wastage in the ASME Class 1, 2, and 3 systems and components which, when integrated into existing plant programs, can improve program effectiveness. The guidance addresses potential leaks with wastage significance from components both inside and outside the containment containing borated water. The inspection of the reactor vessel head is not covered under the scope of the guidance since it is addressed separately in the MRP-75 document (Ref.3) currently under preparation as well as in the recent NRC interim Order (Ref.4).

Included in the guidance are key elements such as: basis for identifying inspection locations, methods of inspection and data collection, damage assessment and corrective actions, program ownership and management oversight, personnel training, and continuous improvement by self assessment. Coordination of data from related parallel programs and utilization of critical early-warning indicators to detect the occurrence and location of a leak are also considered. Inspection of inaccessible locations and criteria for removal of insulation are also discussed.

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3. EPRI PWR Reactor Pressure Vessel (RPV) Upper Head Penetrations Inspection Plan Report No. 1007337, MRP-75, Draft under revision.

4. Order Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors, NRC Order EA-03-009, February 11, 2003.

**Manuscript was not available for publication in the Proceedings**

# Risk-Informed Evaluation of PWR Reactor Vessel Head Penetration Inspection Intervals

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Abstract: A risk-informed evaluation provides a rational basis for setting the re-inspection interval for non-visual nondestructive examinations of reactor vessel closure head penetrations in pressurized water reactor (PWR) plants. The main potential safety concerns addressed by such inspections are nozzle ejection due to circumferential nozzle cracking and rupture of the head due to extensive boric acid wastage caused by leakage from through-wall penetration cracks. An integrated, whole-head Monte Carlo probabilistic fracture mechanics model has been developed to calculate the nozzle ejection accident initiating event frequency given periodic eddy current or ultrasonic examinations of the nozzles and their attachment welds, as applicable. The model includes explicit treatment of axial cracking initiating on the nozzle ID and nozzle OD below the J-groove attachment weld, as well as cracking initiating on the wetted surface of the J-groove weld. This facilitates calculation of the probability of leakage with appropriate credit for the applicable periodic inspections. The probability of leakage may be used as an input to a probabilistic assessment of wastage of the low-alloy steel head shell, for example based on the Materials Reliability Program methodology documented in report MRP-75, typically assuming a bare metal visual examination of the top head surface during every refueling outage. In combination with the conditional core damage probabilities for the appropriate analyzed loss of coolant accident scenarios, the probabilistic results are used to determine the core damage frequency associated with the assumed re-inspection interval for comparison to the risk-informed criterion provided by U.S. NRC Regulatory Guide 1.174. This comparison in combination with the corresponding deterministic evaluations and related assessments—such as evaluations of the relevant material and fabrication processes and evaluations of the most relevant industry inspection results—completes the plant-specific risk-informed evaluation.

## PURPOSE

The purpose of this document is to provide a rational basis for setting the re-inspection interval for nonvisual examinations of reactor pressure vessel (RPV) closure head penetrations. Two types of analyses will be described. First is a deterministic approach to demonstrate that nozzle ejection and significant head wastage are unlikely to occur given the assumed re-inspection interval. Second, a probabilistic approach is described where Monte Carlo analyses are performed to demonstrate that the calculated increase in core damage frequency (CDF) due to potential nozzle ejection and boric

acid wastage is within acceptable limits. Acceptable limits for the probabilistic analysis are defined based on the criterion of NRC Reg. Guide 1.174, which sets an acceptable increase in core damage frequency as $1\times10^{-6}$ per year. The probabilistic assessments are also used to verify that the inspection strategy used results in a low probability of penetration leakage.

## EVALUATION ELEMENTS

There are seven key elements to establishing a risk-informed RPV head nozzle inspection interval. These are:

1. Flaw and wastage tolerance calculations. It is important to know the size of flaws and the volume of wastage of the RPV head that can be tolerated without risk of a small- or medium-break LOCA.
2. Plant design, materials, fabrication, and operating parameters including time at temperature. These parameters are key inputs to the risk assessments.
3. Results of previous visual and non-visual inspections at the subject plant.
4. Evaluation of the expected inspection detectability limits and probability of detection curves. Obviously, knowledge of the plant condition after an inspection is a function of the inspection sensitivity and probability of detection. In the final analysis, this information sets the size defects that must be assumed to exist after completion of a non-destructive examination.
5. Evaluation of industry inspection results including results for plants with the most similar materials and fabrication. As shown in a previous paper,[1] there appear to be significant differences based on the nozzle material supplier and the vessel head fabricator.
6. Nozzle ejection and boric acid wastage calculations. These calculations are based on the largest flaw that can escape detection by the inspections performed, the predicted growth rate for these flaws, primary coolant leak rates as a function of the flaw size, and boric acid corrosion calculations.
7. Assessments of the risk of ejection or significant wastage, consequential damage, and the risk posed by potential loose parts.

## FLAW TOLERANCE

Figure 1 shows the tolerance of a typical CRDM nozzle to axial flaws above the J-groove weld, circumferential flaws above the J-groove weld, and axial-circumferential (lack of fusion type) flaws between the nozzle and J-groove weld. These calculations show that axial flaws of about 5 inches length and circumferential through-wall or loss-of-fusion type cracks of about 270° total arc can be tolerated with a safety factor of 2.7 on the pressure load. This factor of safety is based on the level specified by the ASME Boiler & Pressure Vessel Code for continued service with actual flaws.

Figure 2 shows that, for corrosion progressing from the outside inward such as might have occurred at Davis-Besse, large volumes of low-alloy steel material in the RPV head can be lost without exceeding ASME Code allowable stress values in the remaining ligament. For the case evaluated, about 150 $in^3$ of material can be lost without exceeding ASME Code allowable stresses.

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[1] G. White, L. Mathews, and C. King, "Summary of US PWR Reactor Vessel Head Nozzle Inspection Results," *Vessel Head Penetration Inspection, Cracking and Repairs Conference*, U.S. NRC and ANL, September 29 – October 2, 2003.

**DETERMINISTIC NOZZLE EJECTION AND BORIC ACID WASTAGE ASSESSMENTS**

The main concern regarding nozzle ejection is with a circumferential crack through the nozzle wall above the J-groove weld. There is less concern with ejection due to a "lack-of-fusion" type flaw since this flaw would have to be perfectly concentric for the nozzle to eject. Any deviation from a pure cylinder would create protrusions that would tend to "pin" the nozzle and prevent ejection.

Based on the above, it is conservatively assumed that a 30° through-wall circumferential crack exists in the nozzle above the J-groove weld immediately after restart following the nondestructive (nonvisual) examination. The stress intensity factor is calculated for the circumferential though-wall crack using a fracture mechanics model that accounts for relaxation of the welding residual stresses as the crack grows in length. Figure 3 shows the stresses perpendicular to the assumed circumferential crack plane above the J-groove weld for a typical CRDM nozzle. Figure 4 shows the fracture mechanics model used for through-wall circumferential flaws. Further details of the model are provided in another DEI paper.[2] The calculations are performed assuming that the internal pressure in the vessel acts across the entire crack face, thereby increasing the axial force as the crack grows in length.

Finally, the time for the through-wall circumferential crack to grow to the limiting flaws size (≈270°) is calculated using the stress intensity factors and the crack growth rate model for Alloy 600 base material described in report MRP-55.[3] The nominal MRP-55 crack growth rate curve is increased by a factor of 2× to reflect the potential for accelerated PWSCC in the presence of the annulus environment, although MRP-55 concludes that significant acceleration is unlikely for usual leak rates. Figure 5 shows the results of typical growth calculations for through-wall circumferential cracks above the J-groove weld.

A deterministic approach to assessing the risk of significant boric acid corrosion is presented in Appendices C, D, and E of MRP-75.[4] This model is based on the time for a crack to grow from the length that has resulted in small amounts of boric acid deposits on the head to the length that is capable of producing a crack with sufficient crack opening displacement that the resultant leakage cools the head to the point where liquid boric acid can concentrate and cause significant corrosion. Figure 6 illustrates this approach.

**PROBABILISTIC NOZZLE EJECTION AND BORIC ACID WASTAGE ASSESSMENTS**

While a deterministic analysis is capable of providing assurance that nozzle ejection and significant wastage are unlikely given an inspection program, it does not quantify the level of risk involved. Quantification of the risk requires a probabilistic analysis.

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[2] J. Broussard and D. Gross, "Welding Residual and Operating Stress Analysis of RPV Top and Bottom Head Nozzles," *Vessel Head Penetration Inspection, Cracking and Repairs Conference*, U.S. NRC and ANL, September 29 – October 2, 2003.

[3] *Materials Reliability Program (MRP) Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Thick-Wall Alloy 600 Materials (MRP-55) Revision 1*, EPRI, Palo Alto, CA: 2002, 1006695.

[4] *PWR Reactor Pressure Vessel (RPV) Upper Head Penetrations Inspection Plan (MRP-75): Revision 1*, EPRI, Palo Alto, CA: 2002. 1007337.

Figure 7 is a simplified flowchart for the Monte Carlo probabilistic model used by DEI to determine the risk of nozzle ejection or significant wastage resulting from boric acid corrosion. This model provides for cracks in the base metal and weld metal up to the limit of detectability for the NDE method used at the time the examination is performed. The limit of detectability depends on the inspection method and location and is randomly generated from assumed probability of detection curves. Figure 8 shows the types of cracks assumed. The number of newly cracked penetrations as a function of time is assumed to follow a Weibull distribution as shown in Figure 9. The Weibull slope is assumed to be a distributed function that encompasses the range established by industry experience.

The probabilistic analysis allows the cracks to grow through the nozzle and weld to the point where a leak occurs. At the point when leakage is predicted to occur, it is conservatively assumed that there is a 30° through-wall circumferential crack that grows around the nozzle. Crack growth is modeled using the equation developed by the Materials Reliability Program (MRP) and reported in MRP-55. The crack growth rate distribution that reflects heat-to-heat material differences for Alloy 600 base metal is shown in Figure 10.

The probabilistic model for boric acid wastage is based on the MRP methodology documented in Appendices C, D, and E of MRP-75. The analysis approach and typical results are illustrated in Figures 11 and 12. Relatively wide tolerance bands are used for the key model parameters in order to reflect the uncertainties in current understanding of the wastage process. The key parameters include:

- The point within the operating cycle that leakage begins
- The stress intensity factor driving crack growth
- The crack growth rate distribution
- Leak rates as a function of axial crack length
- Wastage rates as a function of leak rate
- Sensitivity of the bare metal visual inspection

The probability of core damage per year (frequency) is obtained by multiplying the probability of nozzle ejection or wastage exceeding the allowable volume of 150 $in^3$ during a year times the conditional core damage probability (CCDP) for the appropriately sized loss of coolant accident (LOCA). The base result is then compared to the $1.0\times10^{-6}$ per year criterion from NRC Reg. Guide 1.174. Sensitivity cases are also run to show that the results are not too dependent upon the input assumptions and parameter distributions. Variables addressed in the sensitivity analyses include:

- Probability of detection curves
- Crack geometry and location
- Weibull crack initiation reference
- Base metal and weld metal crack growth rate assumptions
- Credit for bare metal visual (BMV) inspections to detect small leaks

## CONCLUSIONS

After consideration of additional factors such as the potential effects of loose parts, consequential damage, and the effect of large early release frequency (LERF), the methodology forms a rational basis for setting the re-inspection interval. Because RPV heads are quite flaw tolerant, typical results

show that re-inspection every second or third operating cycle maintains the requisite level of nuclear safety.

The probabilistic model of nozzle ejection is also used to verify that the probability of leakage is appropriately low given the program of repeat inspections. The calculation of the probability of leakage is facilitated by the explicit treatment of axial nozzle cracking and weld cracking, with appropriate credit for the applicable periodic inspections to detect cracks before they produce a leak path to the annulus on the nozzle OD above the J-groove weld.

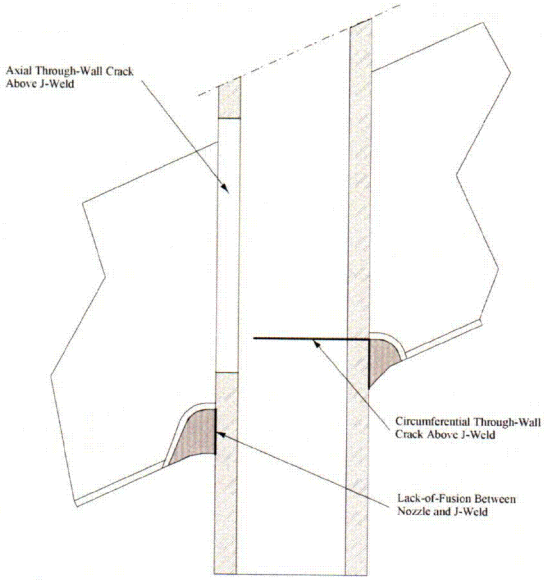
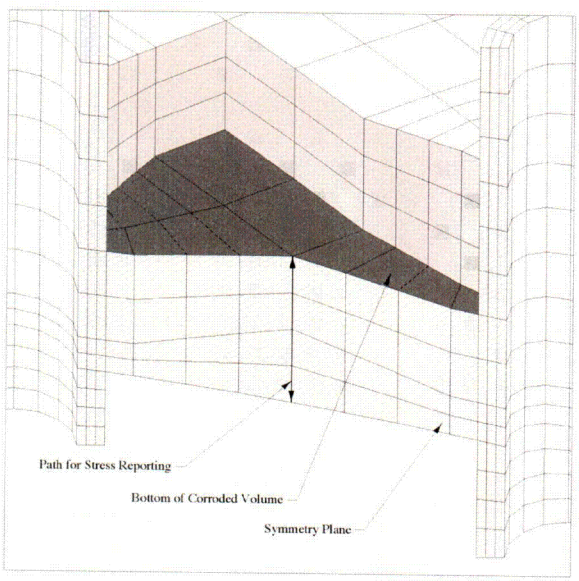


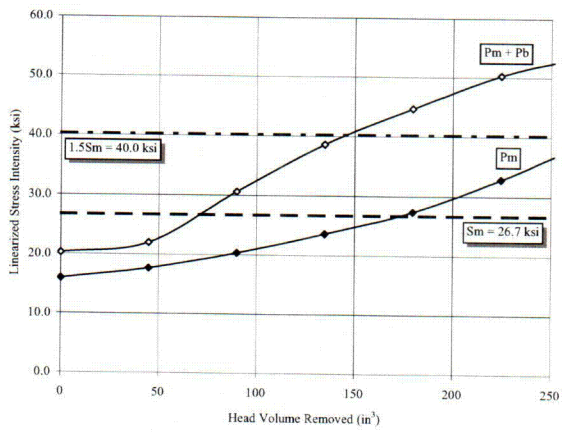

Figure 1
Tolerance to Axial and Circumferential Cracks

**Typical Results for CRDM Nozzle**

| | 2500 psi | 6750 psi |
|---|---|---|
| Axial through-wall flaw in nozzle above J-weld | 14.3 inches | 5.3 inches |
| Circ. through-wall flaw above J-weld | 330° | 284° |
| Lack of fusion between nozzle and weld | 327° | 271° |




**Finite Element Model of Representative Head**

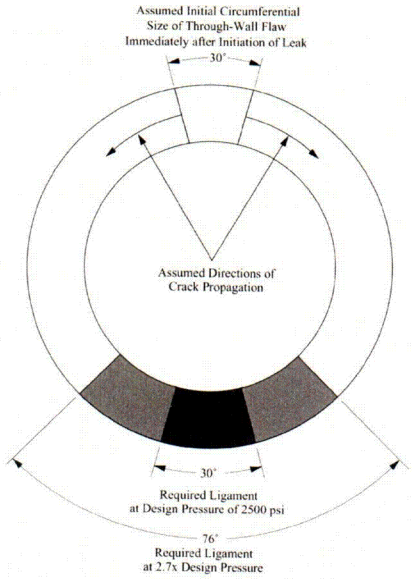



**Primary Membrane and Membrane Plus Bending Stress as Function of Wastage Volume**

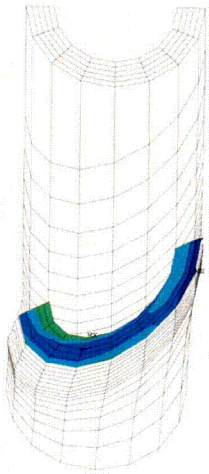
Figure 2
Tolerance to Boric Acid Wastage



C14

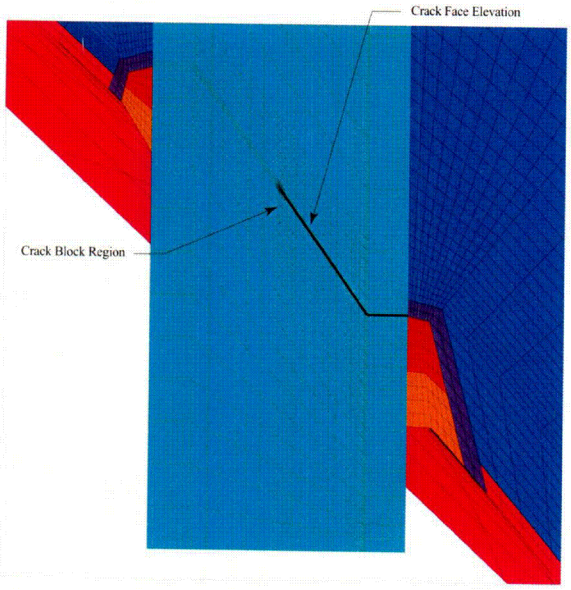



**Typical Critical Flaw Size of 330°**

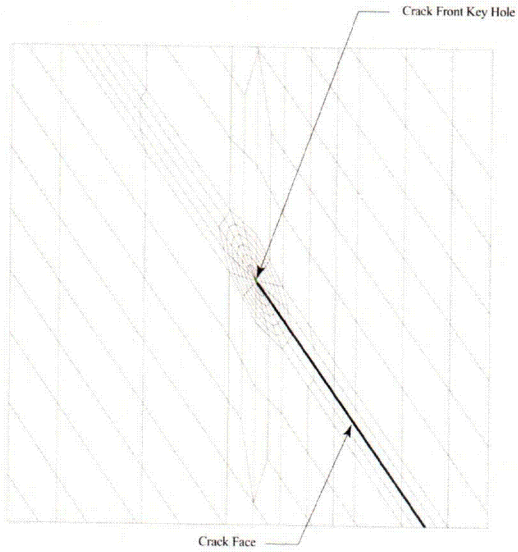


**Example of Operating Stress Perpendicular to Circ Crack Plane**

Figure 3
Crack Growth Around Nozzle Circumference




**180° Downhill-Centered Crack**




**Crack Mesh Detail**

Figure 4
Example Fracture Mechanics Model for Nozzle Circumferential Cracks



C15

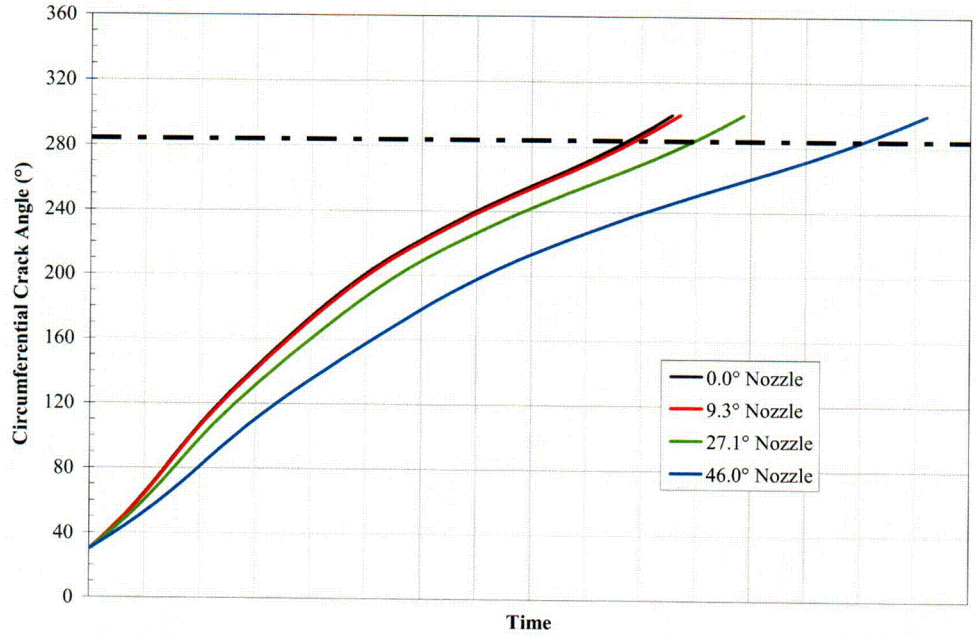
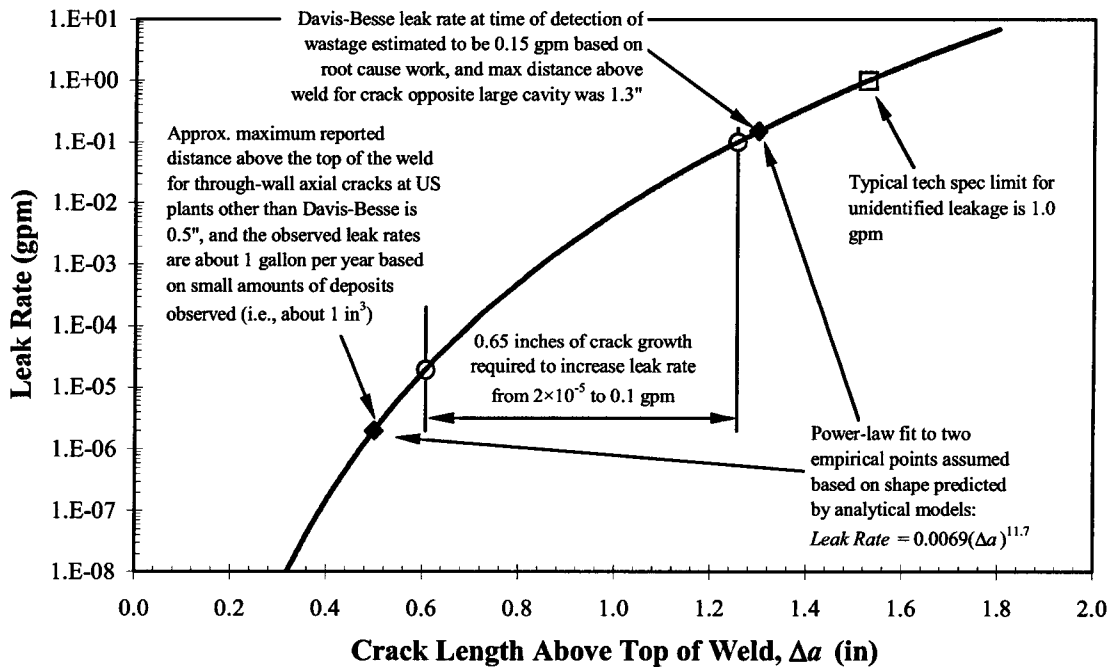



Figure 5
Example Results for Growth of Circumferential Through-Wall Cracks



C16



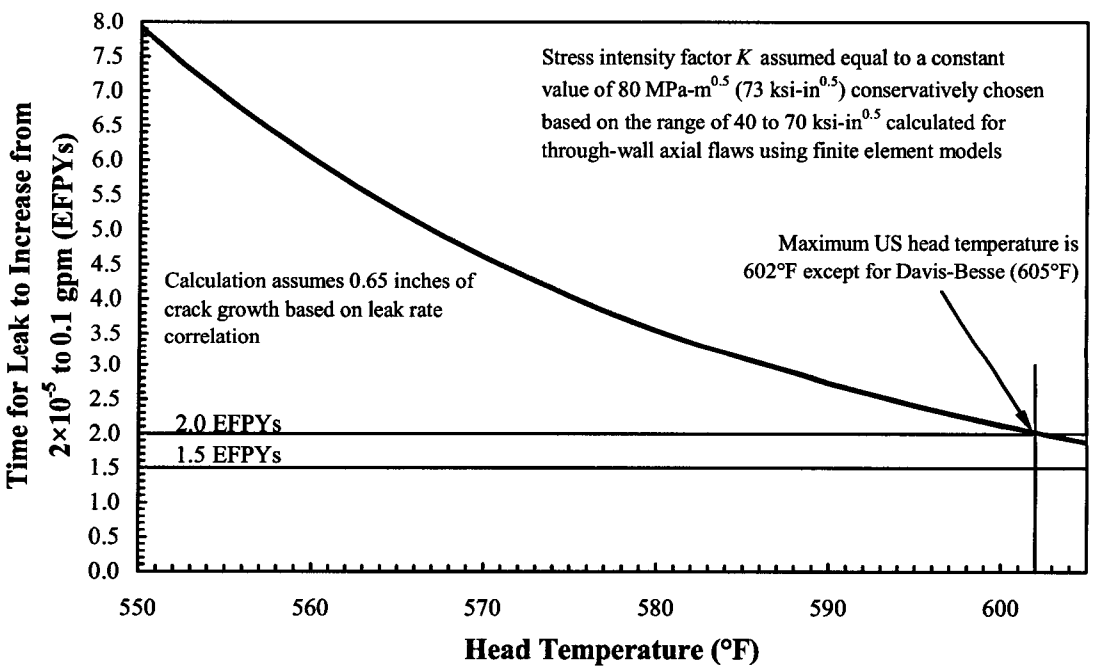




Figure 6
Deterministic Model for Boric Acid Wastage

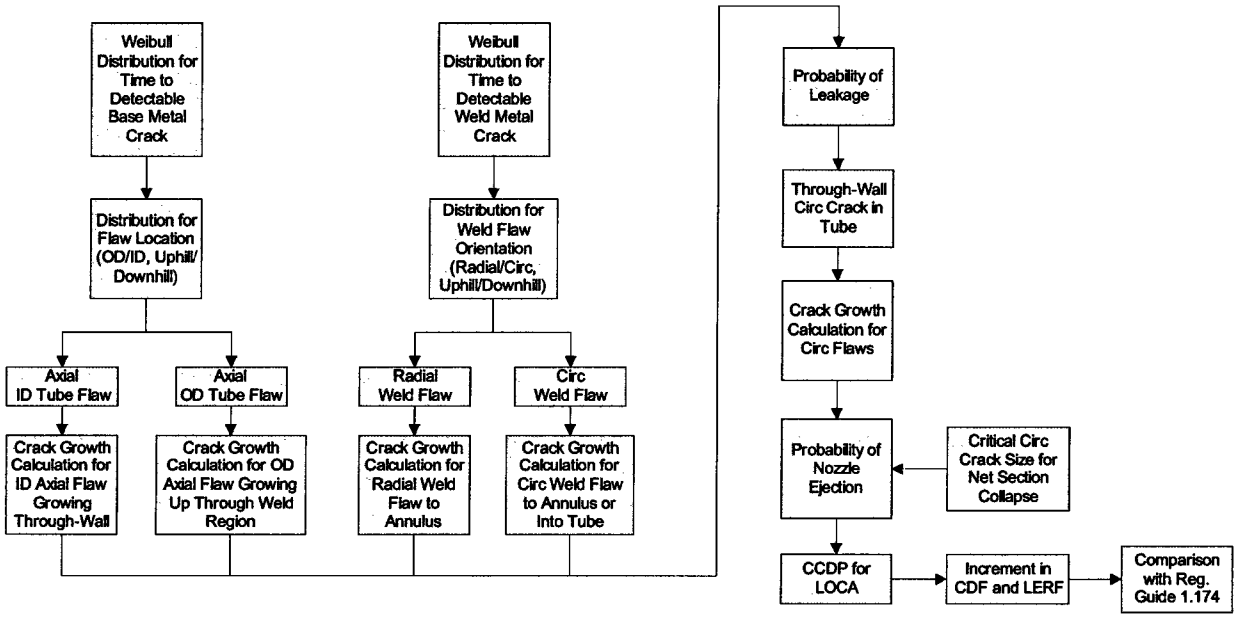



Figure 7
Simplified Flowchart of Monte Carlo Risk Assessment Model

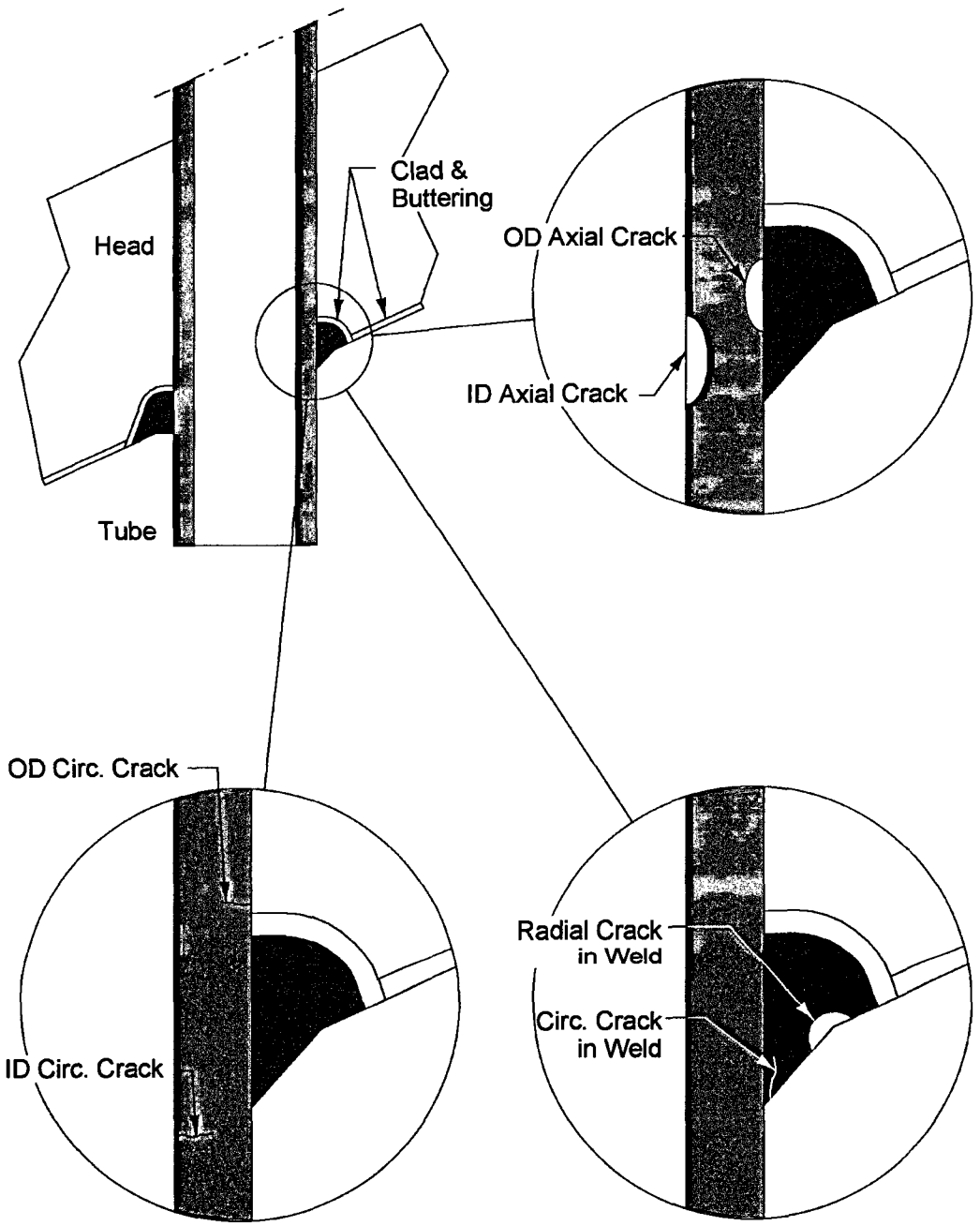



Figure 8
Modeled Flaw Geometries

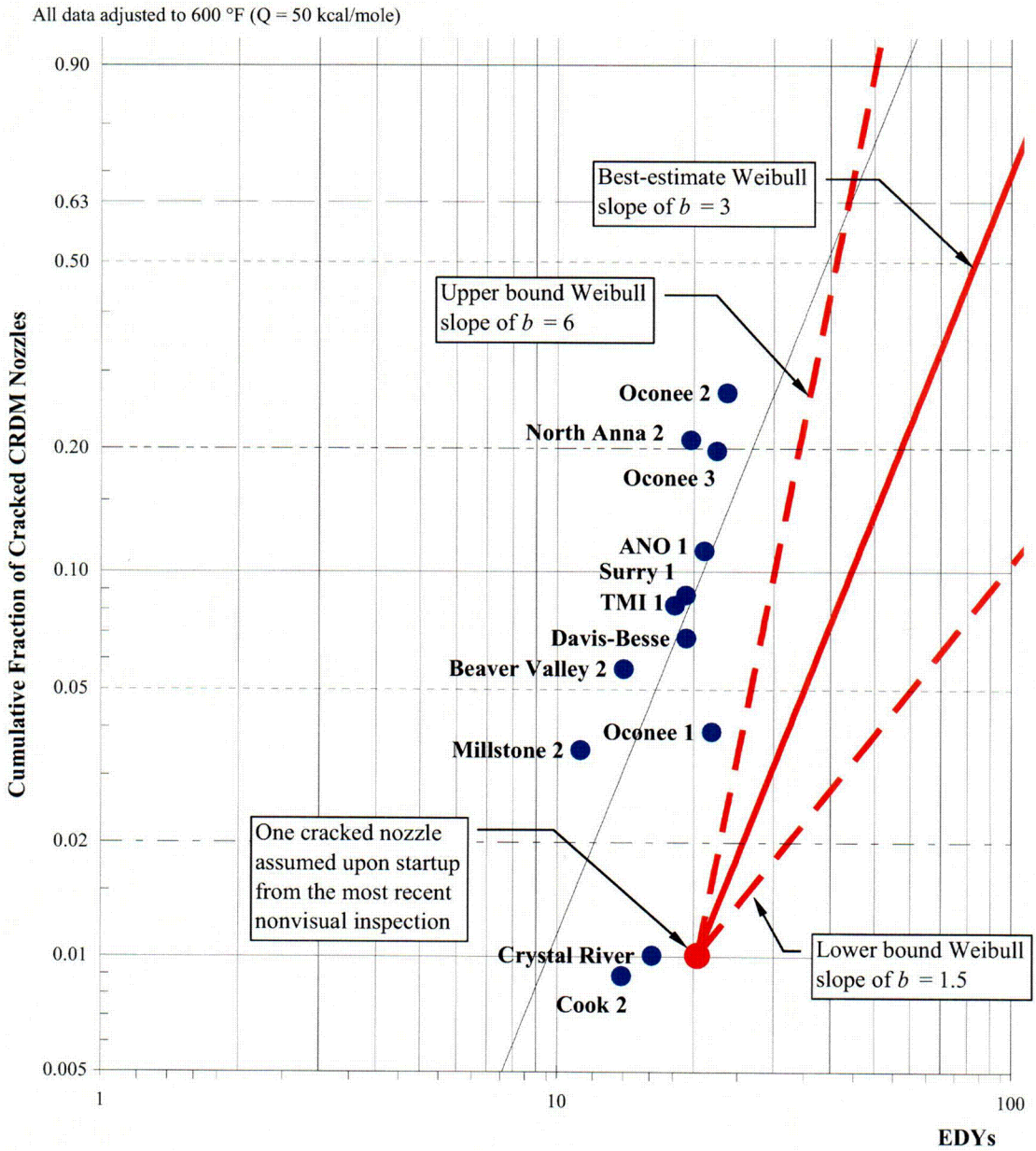



Figure 9
Weibull Statistical Model for Crack Initiation



C17

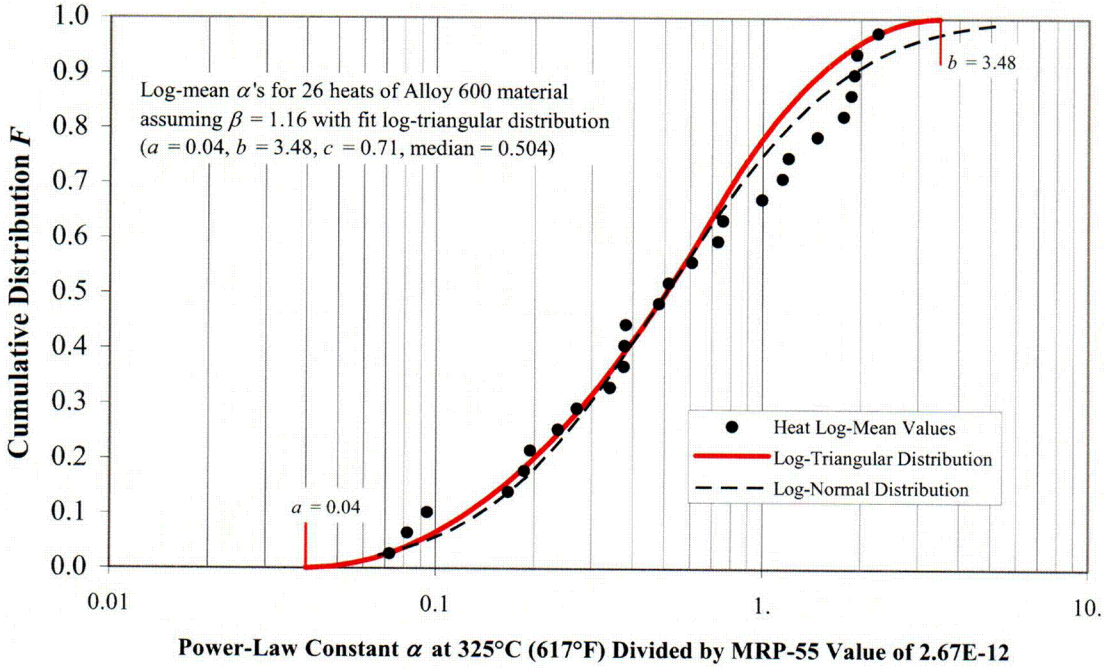



Figure 10
Crack Growth Rate for Alloy 600 Base Metal (MRP-55)



C18

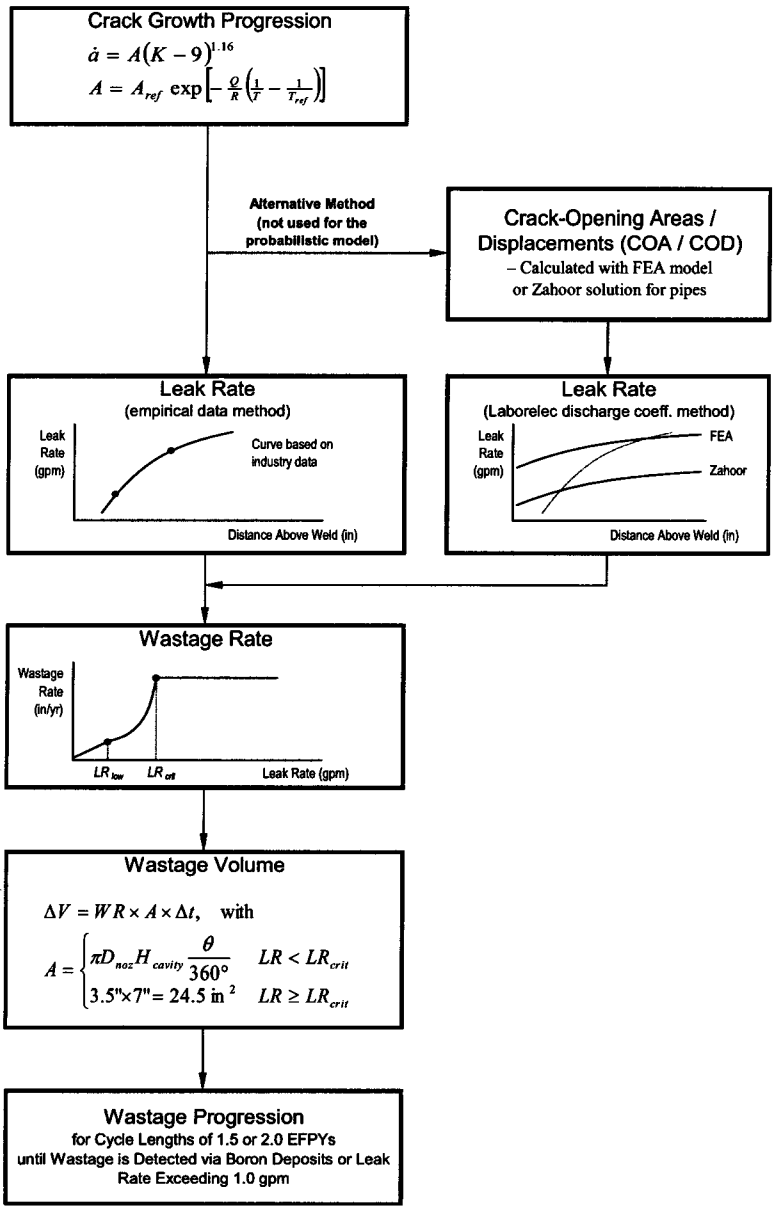
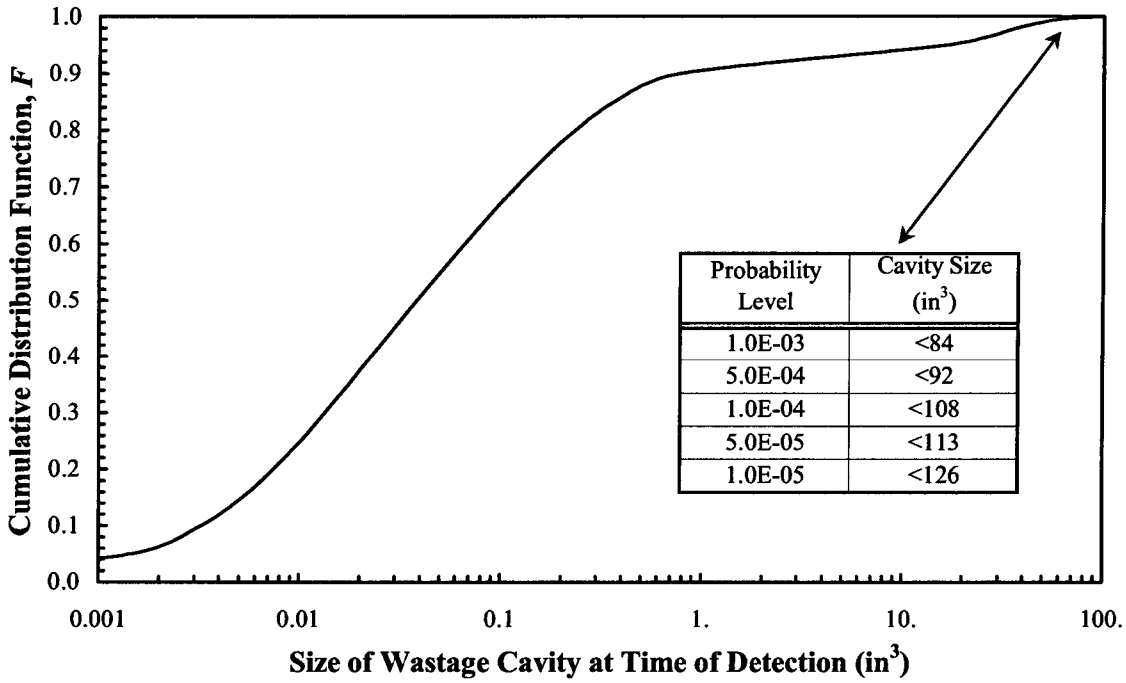



Figure 11
Probabilistic Model for Boric Acid Wastage (MRP-75)




| Probability Level | Cavity Size ($in^3$) |
|---|---|
| 1.0E-03 | <84 |
| 5.0E-04 | <92 |
| 1.0E-04 | <108 |
| 5.0E-05 | <113 |
| 1.0E-05 | <126 |

Figure 12
Typical Results of Probabilistic Wastage Analysis for a Single Leaking Nozzle (Assuming Bare Metal Visual Inspections are Performed During Each Refueling Outage)

**Reactor Vessel Head Penetration Inspection Technology**
**Past, Present and Future**

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Several pressurized water reactors have experienced primary coolant leaks as a result of cracking in the tubes and J-Groove welds in reactor vessel head penetrations. The cracks have been attributed to primary water stress corrosion cracking (PWSCC) of the Alloy-600 nozzle material and Alloy-182 weld material. Westinghouse has been actively involved in investigating the root cause, establishing a safety position, and developing inspection and repair strategies to address the reactor-vessel-head penetration cracking issue.

Westinghouse resources from Germany, Sweden and the United States have cooperatively developed equipment and nondestructive examination technology for providing "under the head" eddy current and ultrasonic inspection capabilities for identification and characterization of degradation that might exist in the penetration tube OD and ID surfaces and the J-Groove welds. These developments represent significant advancements to technologies and equipment developed originally ten years ago. Since 1992, Westinghouse has performed over 150 reactor vessel head penetration nondestructive examinations of various scopes at pressurized water reactors all over the world.

This paper will describe the inspection capabilities that Westinghouse has available to support our comprehensive Reactor Vessel Head Penetration Degradation Management Program, field experience with those inspection technologies, and the status of ongoing NDE development efforts to enhance reactor vessel head inspection programs in the future.

**Manuscript was not available for publication in the Proceedings**

# Section 2:
# Session 2: Continued Plant Operation

**Background and Technical Basis for the ASME Section XI Process for Evaluation of Upper Head Penetration Flaws**

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**Exelon Corporation, Chicago, IL**

The recent findings of cracking in the reactor vessel head penetrations at a number of plants have prompted Section XI to develop requirements for inservice inspections of these regions. An Alloy 600 Task Group has been in operation since spring of 2002, and its goal is to keep abreast of industry inspection findings, and to develop inspection requirements for these regions. When inspections are required, a methodology is needed to deal with any indication that may be found, to determine whether a repair is required.

This paper will provide the details of the flaw evaluation rules that are now being proposed for Section XI, along with their technical basis. In addition, the status of the approval process within Section XI will be discussed.

**Manuscript was not available for publication in the Proceedings**

# MAINTENANCE STRATEGY OF INCONEL COMPONENTS IN PWR PRIMARY SYSTEM IN FRANCE

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Keywords: nickel - alloy - Inconel – primary system – RPV – steam generator – NDE – corrosion - maintenance

## ABSTRACT

Base nickel alloys like Inconel 600 or 182 can be sensitive to high temperature primary water stress corrosion cracking. This fact is known since Corriou's works at the beginning of the sixties and its applications to the steam generator tubes in the seventies. For the RP vessel heads, the major fact of the nineties was the leak that occurred during ten years hydro test on one penetration in 1991 in the French NPP unit of Bugey. Several important decisions were taken after discovery of this leak. First of them was to understand why it appeared so quickly, then test repairs for the Bugey case, then decide to replace all cracked vessel heads considering that the repair solutions was too high cost. This is an example of EDF anticipation approach in relation to optimum safety and competitive standardized plants. In parallel, many developments were launched to establish laws for PWSCC and develop non-destructive methods to inspect the head penetrations. The conclusions obtained show the decision was good and no new leak happened on the VH penetrations. A large investigation program of understanding was launched in order to establish laws for PWSCC and predict the risk of cracking in the Inconel components. The analysis of these areas shows that the risk is less important in the other components than for VH penetrations.

## INTRODUCTION

On September 1991, an important event occurred in France at the Bugey unit 3 NPP; a leak in the vessel head was discovered during the hydro test at 207 bars. It appeared that a control rod drive mechanism (CRDM) penetration had failed. Few investigations showed that the leak was produced by longitudinal cracks coming from inside the penetrations. After a destructive examination, the damage mechanism, which produced the cracks, was clearly identified and attributed to primary water stress corrosion cracking (PWSCC). In fact, for what concerns alloy 600, this problem was not a new one. Since few years, cracks had been discovered in the steam generators tubes fabricated with the same material and PWSCC in such material was already known since 1981. Later, in 1989, pressurizer alloy 600 nozzles had cracked (cracks were also discovered during ten years hydro test) and had been replaced

by stainless steel ones. Nevertheless, this cracking was not expected so early in the VH penetrations as all previous studies showed an initiation after more than $10^5$ hours operating.

Effects of stress, material susceptibility, and welding, and manufacturing process had been underestimated. Then Electricité de France (EDF) has decided to launch an investigation program to better understand the phenomena and has applied it on different primary system Inconel components. In parallel, due to the important cracking in VH penetrations, EDF decided at the beginning of 1993 to replace all cracked vessel heads with alloy 600 by new vessel head with alloy 690. Both aspects of the program are developed hereafter.

## PHENOMENON

900 MW VH are equipped with 65 CRDM penetrations and 1300 MW VH with 78 of them. Just after discovery of the leak, many works were engaged to understand what had happened. First of them were based upon the expertise of the Bugey 4 T65 VH penetration and after this from the Bugey 3 T54 where the leak had been discovered. Crack morphology was fully analyzed and first explanations could be given [1]. Clearly, the leak was attributed to a PWSCC (see figure 1). After VH removal and CRDM dismantling, several examinations were performed. Firstly, few non-destructive examinations, eddy current testing, visual testing, leak tests, ..., revealed longitudinal cracks in the lower part of the penetration near the weld area.

Secondly, after a destructive examination of the T54 VH penetration, the crack was initiated from inside and had a through wall extent that explained the leak.

Consequently, in 1992-1993, the analysis of the phenomenon concluded to effects of:

- Temperature and water chemistry,
- Stress level particularly for the peripheral penetrations due to the angle between the penetration and the VH and the fact the weld is not symmetrical,
- Possibly, material susceptibility.

Then it was decided to orientate research and developments in three main directions:

- Develop methods to repair the VH penetrations and study the possibility to replace vessel heads,
- Develop non-destructive methods for detection and characterization of the surface breaking cracks in order to know as soon as possible the crack initiation and measure the crack propagation,
- Launch an important investigation program in laboratory to better understand PWSCC of both alloy 600 for base metal and alloy 182 for weld metal.

### *SAFETY CONCERNS*

For vessel head penetrations, a maintenance policy has been established and based upon a safety criterion in relation to the minimum residual ligament for a longitudinal crack initiated from inside the penetration. The NDE follow up of cracking is driven on a basis of a 3 mm / cycle crack propagation. Then for 12 months operating cycles, a 900 MW VH replacement has to be performed before reaching a value of 4 mm for residual ligament. For 18 months operating cycles this value is fixed at 6 mm.

## INDUSTRIAL DEVELOPMENTS

### *VESSEL HEAD REPAIRS*

Proposals for repairs were made immediately but due to the shape of the VH and the number of penetrations, the high dosimetry, the difficulties were important and only few tests and repairs were performed during more than one year. In fact this solution was abandoned for the benefit of a replacement solution. For EDF it was obvious that with the first results of the non-destructive inspections all VH would be concerned and then to be replaced.

### *VESSEL HEAD NON-DESTRUCTIVE INSPECTION METHODS*

Few NDE methods were studied but the effectiveness of some of them was proved after many tests. For crack initiation, visual testing, dye penetrant testing and eddy current testing were studied. Due to the thermal sleeve inside the penetration, all three NDE methods and particularly dye penetrant testing were very difficult to implement. They needed to be removed before the inspection. The solution came with the development of "eddy current sword probes" that could be used with the sleeve. It was easy to perform the examination from inside inserting this probe between the sleeve and the penetration. Many tests were performed in EDF and CETIC facilities [1].

For crack propagation, the challenge was also very difficult. Then special "ultrasonic sword probes" were developed (maximum thickness 2 mm) and qualified using the "Time Of Flight Diffraction Technique" (TOFDT). It revealed to be the best for such flaw to be sized. The performance obtained showed a 3 mm surface-breaking crack was sized with a good accuracy on mock-ups with artificial flaws, realistic and real cracks. All results of the qualification were discussed at length [2], [3].

### *COMPLEMENTARY LEAK DETECTION SYSTEM*

In order to prevent any new leak from another VH penetration, EDF decided to install, on the most sensitive vessel heads, $N^{13}$ leak detection systems on the top of vessel heads.

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[1] CETIC is a common EDF and FRAMATOME facility located in Châlon sur Saone in France.

## INVESTIGATION PROGRAM ON ALLOY 600 / ALLOY 182

From the experience gained with the alloy 600 steam generator tubes, many results about PWSCC were acquired. The sensitivity of the material (for instance carbon content) was known and many expertises of pulled tubes helped to understand this type of cracking. Nevertheless, the situation of VH penetration was different by the fact the level of stress can be very important due to welding process. Particularly, peripheral penetrations are subject to constrains because of the dissymmetrical welds that introduce high level of stress. This is confirmed by the NDE results (the peripheral penetrations are more cracked).

Many studies were performed and a methodology has been developed to assess the stress corrosion life of Alloy 600 components.

### *CRACK INITIATION*

The studies undertaken to characterize the PWSCC behavior of alloy 600 have identified the factors that play the most important roles in the occurrence of the crack initiation.

Three factors appear to be essential to the susceptibility of a given part to PWSCC initiation:

1- the material properties (microstructure, composition ...),

2- the temperature,

3- the mechanical loading.

As these factors can be quantified, it has been possible to combine them in a procedure enabling the utilities to deem the risk of crack initiation by PWSCC in the components of the PWR plant primary loops made from alloy 600.

The influence of these parameters can be quantified by indexes of susceptibility: $i_m$, $i_\theta$, $i_\sigma$.

The global index of susceptibility is the product of the three indexes.

The configuration of reference (global index of 1) corresponds to a tube of a "sensitive alloy 600", submitted to a mechanical loading of 450 MPa, at the temperature of 325°C.

In these conditions, the minimum time for crack initiation is about 10 000 hours.

While being arbitrary, these conditions must nevertheless match together, namely any material rated $i_m = 1$, exposed to primary water at 325°C under a stress of 450 MPa should crack in 10 000 hours or more in the laboratory or in the plant.

When the model was developed in 1992, these conditions were based on the limited experimental results and operating experience available at that time. Since then, much experimental work has been performed on steam generator tubes, vessel bottom penetrations, steam generator partition plates and reactor support pads and also much operating experience has become available. The great majority of these recent data, used to check the above reference conditions, is in good agreement with them.

Therefore, the relation between the minimum time for crack initiation and the indexes is:

$$t_i \text{ in hours} = 10000 / i_\theta \cdot i_\sigma \cdot i_m$$

In the application of the method to components, we have cumulated the conservatisms in the following manner:

. Material index, maximum, for a given class of material,

. Stress index based on maximal stresses determined from calculations based on models which have been checked with envelop values of measurements and results of mock-up tests,

. Temperature index based on the best knowledge of the temperature of components and on activation energy of the phenomenon derived from laboratory tests and compared to the field experience.

In these conditions, the crack initiation time corresponds to the time, minimum, for initiating a macroscopic crack. This crack is defined, in accordance to the performance of in service inspections, as a crack for which the depth is the lowest of the values of 2 mm or 5% of the thickness wall.

The crack initiation method has been used, firstly, to rank the susceptibility of stress corrosion cracking of components made in alloy 600.

The method has then been applied to these different classes of components in order to define a maintenance strategy.

### *CRACK PROPAGATION*

Between 1991 and 1996, 52 vessel heads have been inspected by contact eddy current.

A program of determination of in service crack propagation has been conducted on 27 penetrations from 15 vessel heads: 2 or 3 successive inspections were performed on 180 cracks [5].

The temperature of the inner surface of the penetration, in front of the weld of the penetration to the vessel head, in all configurations (cold dome, hot dome, penetration in central, intermediate or peripheral position) has been determined by calculations, studies on mock-ups and measurements on site.

Best estimates of the temperatures in the inner surface of the penetration in front of the weld are:

- In cold dome: 290°C,
- In hot dome: 300°C.

The difference of temperature between the T-hot dome and T-cold dome is close to 10°C.

With the accuracy of the methods, it is not possible to singularize a particular plant or the position of a penetration in the plant

Figure 2 presents the variation of the crack growth rate da / dt as a function of the stress intensity factor K for the T-hot and T-cold domes. In this representation, the points corresponding to the T-hot dome do not differ from those corresponding to the T-cold dome.

The activation energy, for crack propagation, has been deduced from laboratory data obtained between 290°C and 360°C (results EDF, Framatome, CEA and ETH for which a satisfactory agreement has been obtained (Q = 130 KJ/mol) [5].

It is not possible to deduce activation energy from available field data since the scatter of data, for a given configuration (T-hot and T-cold), is greater than the difference of data converted with the activation energy of 130 KJ/mol (figure 2).

2 laws have been derived from field and laboratory data:

- The first law corresponds to the mean curve of field measurements:

Its expression at 290°C is:

$da / dt = 0{,}03 \times (K - 9)^{0{,}52}$ (da / dt in µm/h; K in MPa√m)

- The second law is the upper bound of results measured in field and in laboratory.

Its expression at 290°C is:

$da / dt = 0{,}3 \times (K - 9)^{0{,}10}$ (da / dt in µm/h; K in MPa√m).

Figure 3 shows a comparison of field data and laboratory data:

- Black curve representing the most sensitive heat tested in laboratory; maximum CGR is about 4 mm/year.

- Blue curve representing the average CGR coming from field measurements (VH penetrations). All values are in accordance and the most sensitive value from laboratory experiment remains above field measurement.

## OTHER INCONEL COMPONENTS

Other Inconel components in the primary system were reviewed in the frame of the anticipation but taking into account results of the investigation program. Those, for which a potential risk of PWSCC was estimated, are (see figure 4):

- VH penetration J groove welds,
- Steam generator partition plate stub weld,
- Bottom head instrument penetration (BHIP),
- RPV nozzle repaired areas.

For these components, it was decided to initiate a complementary NDE program on a sample of them but taking into account material susceptibility and on the basis of previous laboratory results.

Safety evaluations were performed for each component. For instance, defects in the SG partition plate have been studied in all cases; a through wall extent crack doesn't modify SG operating conditions. For BHI penetrations, a fracture toughness study has been done considering longitudinal defect initiated from inside or outside tube penetration. For RPV nozzles, mechanical fast fracture studies show the acceptability of large defects.

Non-destructive examinations are performed during periodic outages or ten year visits (exceptionally during short outages) but using judiciously resources. For the two first ones, dye penetrant testing (PT) method was developed using an automatic tool to inspect the weld. For the second one, same techniques used for vessel head are used but adapted to the smaller diameter of the BHI penetration. A specific tool is implemented. For RPV nozzle repaired areas, detection of potential cracks is performed using eddy current probes and characterization using ultrasonic narrow beam focusing transducers. For the last one, video inspection is also performed at each ten years visit.

## FEEDBACK EXPERIENCE FROM THE FIELD

### A) ALLOY 600

As mentioned here above, many non destructive examinations were performed on all 54 French vessel heads with alloy 600 CRDM penetrations since 1994-1995 when ET and UT inspection procedures were finalized and stabilized. From this important set of data, few conclusions are now established and show clearly evidences. For the other areas, numerous inspections were also performed. Results are detailed hereafter.

#### *VESSEL HEAD INSPECTION PROGRAM AND RESULTS*

The inspection program is based upon the maximum CGR measured on the penetrations of a VH considered as the most sensitive. As example for 900 MW VH penetrations and 12 months cycles; the inspection policy is crack height measurement dependant: every 3 years for cracks below to 3 mm, every 2 years for cracks between 3 and 5 mm, every year for cracks above 5 mm.

In all cases, VH is replaced before reaching the safety criterion (4 mm remaining ligament for 900 MWe).

#### *MAIN CONCLUSIONS FROM VH PENETRATIONS*

Two important conclusions have to be mentioned using the experience feedback from the NDE field:

- Firstly, it is obvious that the level of stress has a major role for the initiation and propagation cracking. The main fact concerns the number of cracked VH penetrations and the circle where it belongs. Figure 5 shows that peripheral penetrations are mostly cracked compared with central ones.

- Secondly, the effect of the material susceptibility is particularly obvious for those that are classified sensitive. All VH penetrations were ranked in four categories from the less sensitive A to the most one D taking into account the material

properties of each penetration. Figure 6 shows the effect of this susceptibility. Type B and C heat number are more often cracked than type A heat number but A type VH penetrations heat are more numerous. So in percentage, the value is undervalued. The effect also exists for type D but is less obvious due to the small amount of type D VH penetrations.

### *VHP REPLACEMENT PROGRAM*

Replacement program started at the beginning of 1994. Up to now 41 VH have been replaced and 3 new ones will be installed in the next 2 years (see figure 7). As the CGR is low; it is expected to finish replacement around 2009-2010.

### *BHI PENETRATIONS*

Fifteen RPV have been inspected up to the end of 2002. No crack indications have been discovered in the BHI penetrations at the end of 2002. Also, acoustic emission is performed during ten years visit hydro test pressure. Probes are very close to BHIP and could detect a noise due to a leak.

## B) ALLOY 182

### *VESSEL HEAD*

For J groove welds, PT examination on 754 welds from removed vessel heads was performed between 1996 and 1999. No cracks were detected in these welds.

### *SG PARTITION PLATES*

SG partition plate (hot branch) is considered as possible precursor due to higher temperature and no stress relief from a heat treatment. Nevertheless, pressure hydro test is very favorable and decreases significantly the stress level.

Forty-two stub welds in the hot branch and twenty ix in the cold branch have been inspected. No cracks were found except in the case of a SG water box hammered by a loose part. A small crack was discovered during the SG expertise after removal as shown on figure 8.

### *RPV NOZZLE REPAIRED AREAS*

Seven 900 MWe RPV have repaired nozzles with alloy 182. The inspections performed during the ten years visit have not revealed any cracks in the nozzles.

## ALLOY 600 - VH PENETRATIONS BASE METAL ALLOY 182 - WELD METAL

Another risk is the weld metal cracking concern. In fact the Bugey 3 leak did not concern weld metal as the crack developed fully in the base metal. Taking into account international information and after the discover of a welding defect in the J groove weld on Ringhals VH, EDF decided to perform dye penetrant examinations of the J weld on several removed VH.

Eleven vessel heads (754 welds) were inspected from 1994 to 1999 and none of them were found cracked.

## NEW PROGRAM

A new program has been proposed for years 2001 to 2008 following French regulator's request. This program requires inspection of:

- 26 SG partition plates (precursors) + 9 randomly chosen,
- 12 RPV bottom head penetrations

Also qualified NDE will have to be developed at the next ten years visit for RPV nozzles repaired zones and core support lugs.

## CONCLUSIONS

In 1991, after discover of a leak on a vessel head penetration of Bugey NPP unit 3, numerous studies to understand the phenomenon were implemented. Several of them were in the field of expertise, others in the field of non destructive examinations in order to have all necessary tools to perform inspections The most important works were laboratory studies on both alloys 600 and 182. One of the major results concerned crack growth rates laws that can predict crack propagation.

In parallel, the maintenance policy was based upon the following key points:

- Very early, replacement of all VH equipped with alloy 600 by alloy 690. At the end of 2002, 41 of them have been replaced.

- Importance of the NDE inspections to measure cracks and determine the best instant for replacement (safety criterion).

- Importance of 2 parameters, penetration angle in the VH and material susceptibility.

For what concerns J groove welds, many PT inspections did not reveal any surface breaking cracks.

For the other Inconel components, up to now no cracks were discovered except for a very specific case in a hammered water box. For BHI penetrations, no cracking is expected before the end of life.

In conclusion, this maintenance policy was very fruitful, as no new leak has appeared up to now on vessel heads and bottom head instrumentation nozzles. It confirmed the need to anticipate and arrange strategy for the other Inconel components at the benefit of safety and competitiveness of the nuclear standardized series. EDF had a real proactive position about alloy 600/alloy 182 and more generally about Inconel components.

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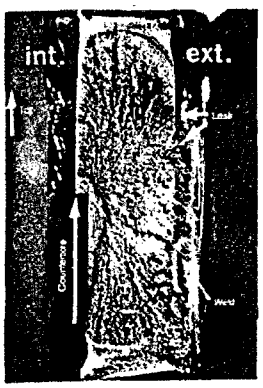


Fig. 1: TWE crack in T54 VH penetration

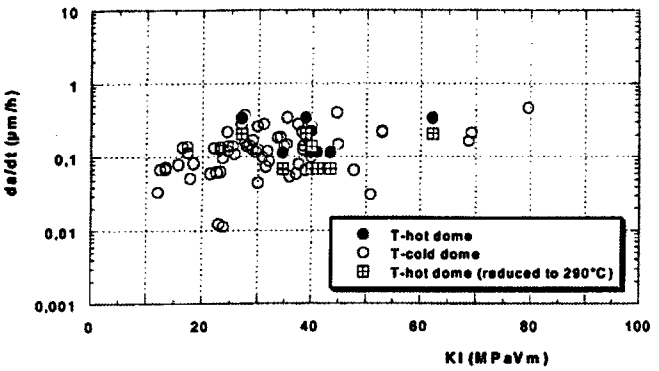


Fig 2 : Field data - T-hot dome data (300°C) reduced to 290°C with Q = 130 KJ/mol and T-cold dome data (290°C)

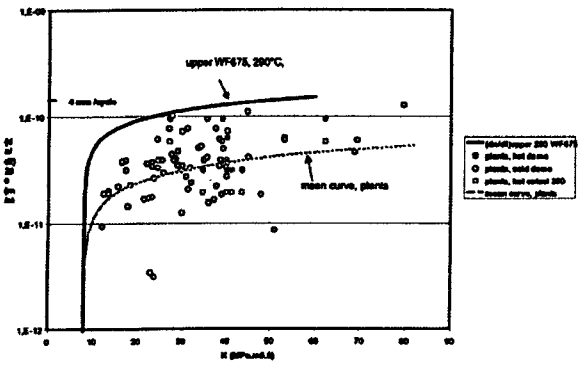


Fig. 3 : Comparison between laboratory and field experience

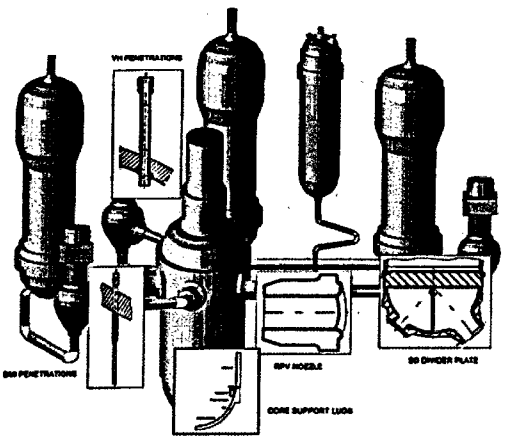

Fig 4 : Inconel components in primary system

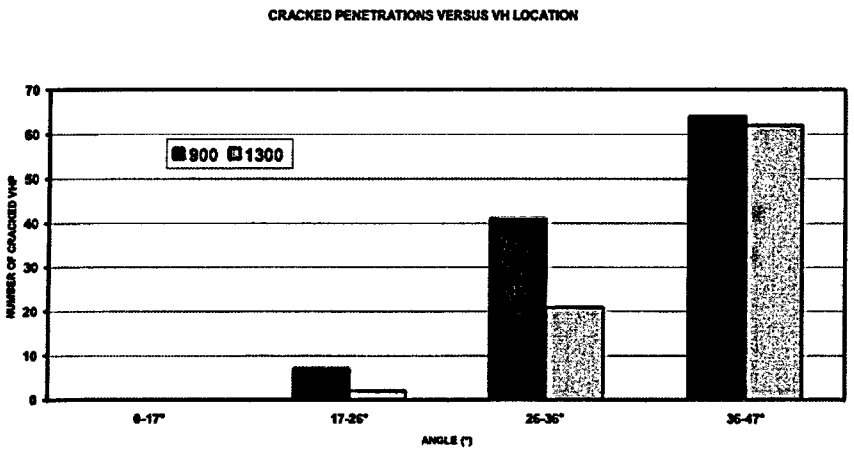


Fig. 5 : Importance of the "angle effect" versus VH penetrations cracking

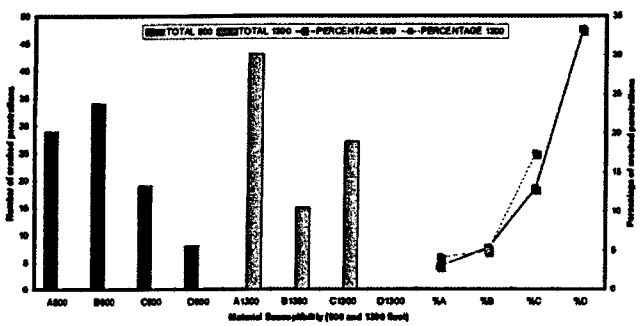

Fig 6 : Importance of the material susceptibility versus VH penetrations cracking (percentage values are given versus the total number of penetrations in standardized plant series 900 or 1300)

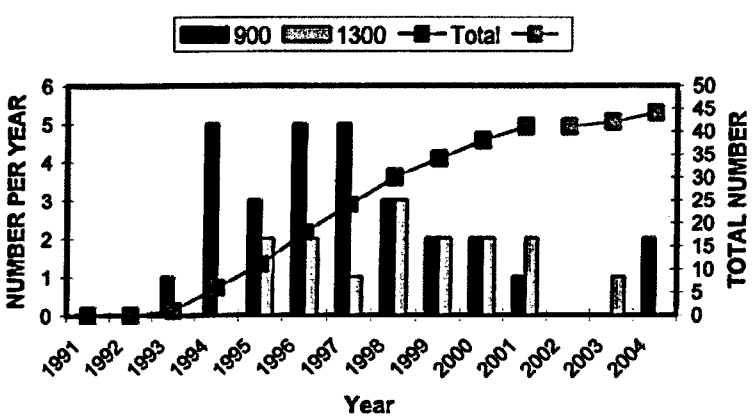


Fig. 7 : Vessel head replacement since 1993

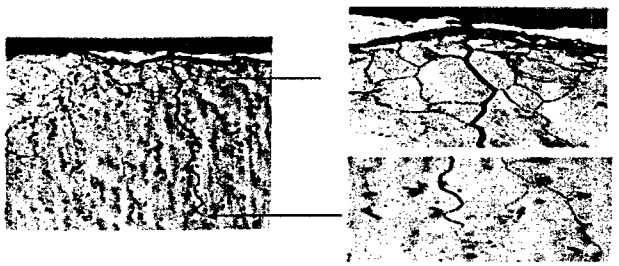

**Fig. 8 : Expertise from a removed part of a hammered steam generator channel.**

# Strategic Planning for RPV Head Operation

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Abstract: Utilities are faced with difficult economic choices in managing PWSCC of RPV head penetrations. Inspections are expensive, especially if required on a frequent basis; if leaks are discovered during an outage when no inspections are performed, and no provision to perform repairs has been made, repairs can lead to significant lost production; repairs and remedial measures are expensive and, depending upon conditions, may result in future cracks or leaks; and head replacements are expensive. Life cycle management planning has been performed for a moderate susceptibility plant to determine the most attractive long-term strategy. This work has been based on predictive modeling and net present value economic analyses. The approach described can help a utility determine the best management strategy for its plant.

## BACKGROUND

The economic consequences of managing RPV head nozzle PWSCC can be significant. For example:

- EdF is replacing all of its RPV heads.
- Cracks and leaks in nozzles in several domestic plants have resulted in significant outage extensions and repair costs.
- Boric acid corrosion resulting from a PWSCC leak at the Davis-Besse plant has led to over a 20 month outage.
- Cracks in large numbers of welds at North Anna 2 led to about a four month outage while the head was replaced.
- Industry findings and NRC guidelines/requirements have led to many expensive inspections.
- 29 plants in the US have announced plans to replace heads as of September 2003.

The purpose of this paper is to review issues associated with developing a strategic plan for managing RPV head PWSCC at a moderate susceptibility plant. The strategic plan must:

- Ensure an extremely low risk of core damage. For purposes of this paper, it is assumed that this objective can be met by meeting the core damage risk criterion of NRC Regulatory Guide 1.174. Analyses to determine the risk of core damage are discussed in another DEI paper[1] presented during the same NRC-ANL conference during which this paper was presented.
- Ensure a low risk of leakage.
- Select a strategic plan that results in the lowest practical net present value (NPV) cost.

[1] G. White, S. Hunt, and N. Nordmann, "Risk-Informed Evaluation of PWR Reactor Vessel Head Penetration Inspection Intervals." *Vessel Head Penetration Inspection, Cracking and Repairs Conference*, U.S. NRC and ANL, September 29 – October 2, 2003.

### INPUTS AND CONSTRAINTS

There are a number of constraints that apply to establishing a cost effective strategic plan. Issues that should be addressed include:

- The current condition of the vessel head must be established by non-destructive examination. The head must be free of cracking to remain in the moderate susceptibility category.
- The rate of future PWSCC initiation and growth should be predicted based on industry experience and modeling that accounts for differences between the subject plant and relevant industry peers.
- Non-destructive examination intervals should be selected such that there is a low risk of leakage and extremely low risk of core damage.
- Planned refueling outage durations can have a significant effect on establishing a strategic plan. For example, plants with short refueling outages will have little time for inspections or repairs without extending the outage, and plants in a long outage, such as for steam generator replacement or 10 year ISI, may have longer time available for inspections or repairs.
- The time and cost for nozzle inspections and repairs.
- The time and cost for replacement head procurement and installation. For example, inspections may show the need for immediate replacement while there may be a long delay to obtain a suitable replacement head.
- Potential remedial measures, including an assessment of their cost and effectiveness.
- Special attention should be given to the possibility of discovering leaks from a nozzle at an inopportune time such as during a mid-cycle outage conducted for another purpose, during system leak checks at the end of a refueling outage, or during a regular outage when a leak is discovered but no provisions have been made in advance for inspections or repairs.

NRC inspection requirements such as EA-03-009, *Issuance of Order Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors*, are obviously significant inputs to strategic planning. It is necessary to comply with the order or to obtain relaxation from the requirements based on appropriate technical assessments.

### ALTERNATIVE MANAGEMENT APPROACHES

There are four main alternative management approaches for RPV heads. These are:

- Continue to inspect and make repairs as necessary to ensure a low risk of leakage and an extremely low risk of core damage.
- Perform remedial measures to reduce the risk of PWSCC and possibly to increase the inspection intervals.
- Replace the head as soon as possible after discovery of the first PWSCC.
- Replace the vessel head as quickly as possible and perform NDE at increased intervals based on materials in the new head.

Comments on each of these approaches are as follows:

#### Inspect and Repair as Necessary

Inspection and repair methods are currently available to support the first alternative. The main nondestructive examination methods are volumetric NDE (UT) of nozzles plus volumetric examination for leakage paths through the annulus, or eddy current examination of the entire wetted

surface of the nozzle and weld to show absence of cracks. Repair methods that have been used include removing shallow surface flaws, embedding deeper flaws, and removing the lower part of the nozzle and rewelding the bottom of the remaining nozzle to the head using a temper bead technique.

**Remedial Measures**

A number of remedial measures have been proposed for RPV head nozzles including:

- Modification of the internals flanges to increase bypass flow and thereby reduce the temperature of the vessel head. Some utilities have already performed this modification.
- Surface treatment of the nozzle and weld surfaces by shot peening or water jet conditioning to reduce the tensile stresses and, in the case of water jet conditioning, to remove small preexisting flaws.
- Nickel plating the nozzle and weld surfaces to keep the primary water coolant from contacting the Alloy 600 materials.
- Applying Alloy 152 weld overlays on the nozzles and welds.
- Roll expanding the nozzles into the vessel head to provide a redundant load path above the J-groove weld, and then conditioning the rolled surface to reduce the potential for new cracks.
- Application of a new structural weld between the nozzle and low-alloy steel vessel head, either on the top surface of the head or after boring out the lower part of the nozzle.
- Mechanical stress improvement.
- Zinc injection.

EPRI has sponsored testing of many of these remedial measures.[2] Upon completion of the testing, the remedial measures were ranked in terms of effectiveness. The three main categories were:

- Most Effective
  - Water jet conditioning
  - Electro mechanical nickel brush plating
  - Shot peening
- Intermediate Effectiveness
  - Electroless nickel plating
  - GTAW weld repair
  - Laser weld repair
- Least Effective
  - EDM skim cutting
  - Laser cladding
  - Flapper wheel surface polishing

To date, the main remedial measure applied in the field has been modification of the internals to increase the bypass flow and thereby reduce the head temperature. The lower head temperature should reduce the rates of crack initiation and growth based on the thermal activation energy. However, experience in France suggests that PWSCC may occur at head temperatures close to the

[2] *Materials Reliability Program: An Assessment of the Control Rod Drive Mechanism (CRDM) Alloy 600 Reactor Vessel Head Penetration PWSCC Remedial Techniques (MRP-61)*, EPRI, Palo Alto, CA: 2003, 1008901.

reactor cold leg temperature. This is especially significant given the PWSCC at the South Texas Project Unit 1 bottom mounted instrument nozzles at a temperature of about 565°F. The South Texas Project experience shows that poor material properties and weld defects can result in PWSCC at temperatures lower than otherwise expected.

Finally, while remedial measures may reduce the rates of PWSCC initiation and growth, and thereby reduce the cost of future repairs, it may be difficult to take credit for the improvement in the form of increased inspection intervals.

**Head Replacement**

Installation of a new RPV head with improved nozzle and weld materials is a clear success path that has been taken in France and has been announced by many plants in the United States. The Alloy 690 nozzles and Alloy 52 J-groove welds in these heads should provide better service life than the original heads with Alloy 600 nozzle base material and Alloy 182 welds. In addition to the cost of the new head, consideration must be given to:

- Access provisions for getting the new head into containment.
- Whether the head will be installed with new CRDM drives.
- Disposal of the old head.

One variation on head replacement is to use this as an opportunity to replace the original design reactor head service structure with an improved service structure that requires less effort to disassemble and reassemble every outage. Figure 1 shows a typical original design head service structure and one possible configuration for an improved head service structure. As shown, the original design requires disassembly and removal of the following parts at the beginning of every refueling outage:

- Head insulation
- Head cooling ductwork
- CRDM cables
- Head cooling fans
- Head missile shield

An integrated head service structure, such as shown in Figure 1, can be developed for most plants. In this arrangement, most of the above listed components are integrated in such a manner that only one main lift is required after disconnecting the electrical cables to the CRDM drives.

While an integrated head service structure can reduce the required manpower, it may not result in a reduction of refueling outage critical path time since other constraints can establish the point in time at which the head can be removed and replaced. Nevertheless, there a number of significant benefits of an integral head service structure including:

- Freeing up labor and crane time inside containment during normal refueling outages.
- Reducing the risk of personnel injury by eliminating many crane lifts.
- Cutting several days off of the time required to perform a rapid head disassembly and reassembly such as for a leaking RPV flange o-ring seal or internals inspection after a slow rod drop test, etc.

## ECONOMIC EVALUATIONS

In most cases, strategic plans will include some level of economic evaluation. A deterministic "best-estimate" approach can be used for these evaluations, provided the analysis includes sufficient detail and includes costs over the remaining plant life. In some cases, utilities may elect to perform a Monte Carlo type probabilistic analysis to provide better information of the range and probability of possible costs.

Regardless of the type of economic evaluation, the analysis should include the following:

- The risk of future cracks and leaks for each alternative considered.
- The cost of performing planned (preventive maintenance) work.
- The cost of making repairs (corrective maintenance).
- The value of lost production.
- The value of consequential risks.
- The potential risk that leaks will be discovered at inopportune times such as during a mid-cycle outage or during a system leak check at the end of an outage.
- The planned operating life, including life extension.
- The discount and inflation rates.

Guidance on developing strategic plans is provided in EPRI report 1000806, *Demonstration of Life Cycle Management Planning for Systems, Structures, and Components with Pilot Applications at Oconee and Prairie Island Nuclear Stations*. Deterministic type economic analyses can be performed using the LcmVALUE version 1.0 software prepared as part of the EPRI LCM demonstration program.

Figure 2 shows typical results of net present value calculations for a moderate susceptibility plant. The figure shows the discounted net present value cost over the remaining plant life including preventive maintenance, corrective maintenance, value of lost production and consequential costs.

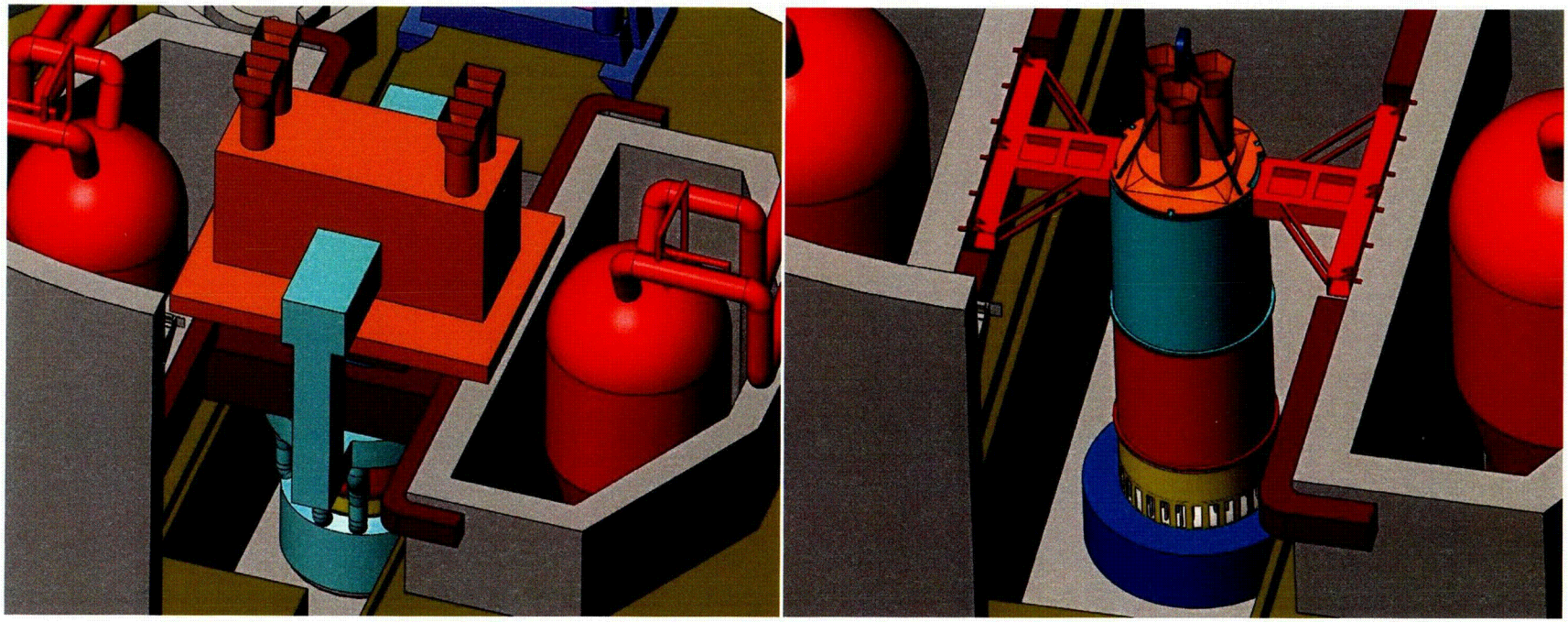
## CONCLUSIONS

For the sample case presented, the optimum alternative appears to be bare metal visual inspections of the RPV head every refueling outage with nondestructive examinations of the nozzles and J-groove welds every second refueling outage.

- As future inspection data become available and predictive models are refined, there may be a technical basis for retaining inspections every second outage when the plant enters the high susceptibility category based on EDYs.
- Volumetric examination every outage and immediate head replacement appear significantly more expensive.
- Remedial measures such as reducing head temperature or waterjet conditioning may be attractive provided inspection intervals can be increased as a result of the effort.
- A reasonable longer term plan is to replace the vessel head the second outage after identifying PWSCC.

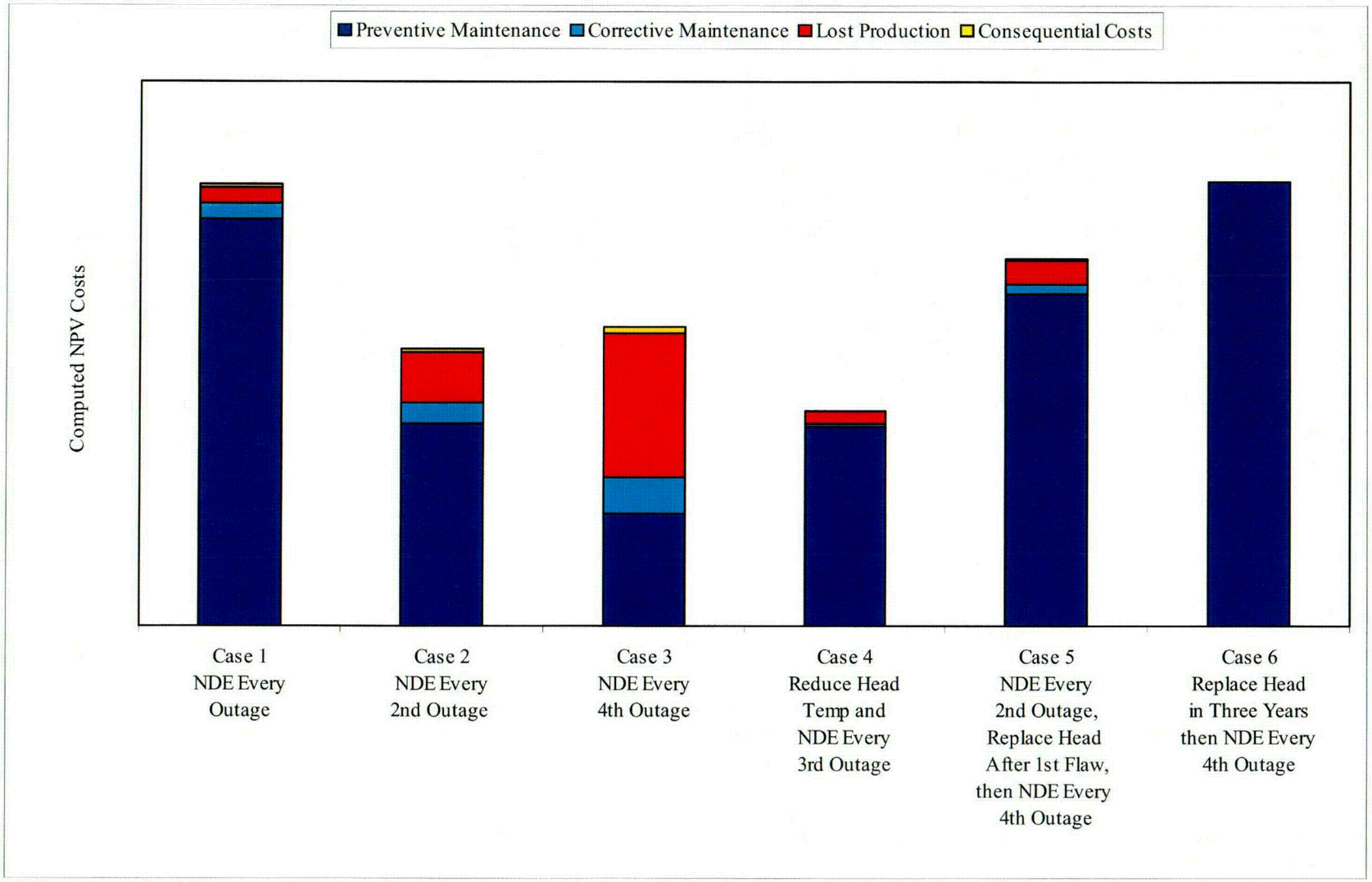
These results are plant specific, and other plants may have different constraints that would affect the optimum solution.

Figure 1
Head Replacement – Integrated Head Service Structure Option



**Typical Original Structure** **Conceptual Integrated Structure**

Figure 2
Typical Strategic Planning Results for Moderate Susceptibility Plant With No PWSCC






C20

**Reactor Vessel Bottom Mounted Instrumentation (BMI) Nozzle Repair Development and Implementation at South Texas Project**

**S. Thomas, South Texas Project, and**
**R. Payne, D. Schlader, and D. Waskey, Framatome ANP, Inc.**

Following a visual inspection that revealed a small amount of a powdery substance around two penetrations in the lower reactor pressure vessel (RPV) during a regular refueling and maintenance outage, Framatome ANP was contracted to perform inspection and repair of the South Texas Project (STP) Unit 1 RPV. This was the world's first indication of the possibility of a similar issue with the lower RPV that has affected the RPV heads for several years.

After extensive inspection of all 58 BMI nozzles, two nozzles were identified for repair. Repair technology implemented on other similar nozzle designs was modified for the two specific STP nozzle locations. Tooling was designed, fabricated, tested, and qualified over a short time period to respond to the emergent need at STP. The teaming between STP and Framatome ANP led to a successful first time implementation of repair on BMI nozzles.

The repair approach, associated tooling, and processes utilized at South Texas will be presented.

**Manuscript was not available for publication in the Proceedings**

# South Texas Project Experience with Alloy 600 Bottom Mounted Instrument Penetration Cracking

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## Abstract

The PWSCC susceptibility of the two PWR units at the South Texas Project has been considered low. Although the bottom mounted instrument (BMI) penetrations are Alloy 600 base metal with Alloy 82/182 welds, PWSCC was not expected at $T_{cold}$ temperatures of the bottom head. Nevertheless, on April 12, 2003, routine visual inspections in Unit 1 discovered boron deposits at two BMI penetrations. Subsequent examinations revealed five axial cracks in the Alloy 600 tubes of the two leaking penetrations and no cracks in the tubes of any of the other 56 penetrations. The two leaking penetrations were repaired utilizing a "half-nozzle" design concept. Although ID initiated PWSCC was originally suspected as the initiating mechanism, several facts are inconsistent with a classical PWSCC scenario. Analysis of a boat sample confirms the presence of welding defects at the tube/J-groove weld which could be responsible for initiating the cracks and explain why the extent of cracking was limited to the two nozzles.

## Background

On April 12, 2003, STP performed a scheduled visual inspection of the Unit 1 reactor vessel bottom head during the closing stages of 1RE11. The inspection, which has been performed routinely since startup, revealed deposits on two of the 58 bottom mounted instrument (BMI) penetrations. Chemical and spectrographic analysis determined the deposits to contain of boron and elevated concentrations of lithium consistent with reactor coolant system (RCS) leakage. Isotopic analysis detected no Co-58, indicating the deposits were at least one year old, and the ratio of Cs-134 to Cs-137 showed the deposits to be about 3-5 years old.

## Evaluation and Repair

An extensive NDE campaign involving UT examination of all 58 BMI nozzles detected 5 axial cracks in the two leaking penetrations, three in Penetration #1 and two in Penetration #46. One crack in Penetration #1 penetrated the ID of the nozzle and extended from just above to just below the J-groove weld. The other two cracks were small and just penetrated the OD of the nozzle. Neither of the cracks in Penetration #46 penetrated the ID of the tube, as verified by a supplemental ET examination. One crack extended from just above to just below the J-groove weld. The UT examination also discovered a number of "discontinuities" at the tube/J-groove weld interface in the same general area of the cracks in Penetrations #1 and #46. Several other penetrations exhibited similar discontinuities, but no cracks.

In an effort to positively identify the leak path, a helium leak test was performed on the two leaking penetrations by pressurizing the annulus between the nozzle and the vessel. No bubbles were observed in Penetration #46. In Penetration #1 a small helium bubble was observed about every two seconds rising from a location outside the nozzle in the J-groove weld fillet at the tube interface.

Penetrations #1 and #46 were repaired utilizing a "half-nozzle" repair which relocated the pressure boundary to the exterior of the vessel with an ambient temperbead weld pad, new J-groove weld, and new Alloy 690 nozzle.

The details of deposit analyses, the NDE campaign, and nozzle repairs have all been previously discussed in public presentations. The remainder of the discussion in this paper will focus on the cause of the cracks.

## Metallurgical Sampling

To facilitate metallurgical analysis of the actual cracks, boat samples were removed from Penetrations #1 and #46 employing an Electric Discharge Machining (EDM) cutting technique. In the case of the BMI nozzles inside the reactor pressure vessel, the boat sample excavations could not be repaired. The desire for the largest possible boat sample was balanced against conservative structural limitations. A number of mockups were constructed to test the remotely operated equipment.

The boat sample from Penetration #46 was designed to capture as much tube material as possible in an attempt to harvest a portion of a crack not connected to the ID of the nozzle. The margins for error associated with positioning the EDM equipment through 70 feet of water resulted in a shallow cut in Penetration #46. The resulting undersized sample was either inadvertently discarded or completely consumed in the margins of the EDM cutting tool. The boat sample from Penetration #1 captured material and defects from the J-groove weld and J-groove/tube interface, as designed.

## Boat Sample Results

The boat sample from Penetration #1 contained a portion of the large through-wall axial crack in the tube, three "discontinuities" which were confirmed to be lack of fusion resulting from slag inclusions, and one crack at the helium bubble location which connects the surface of the J-groove weld to the largest area of lack of fusion.

## Axial Crack in Penetration Wall

Earlier UT results identified an axial crack in Penetration #1 which penetrated the ID of the nozzle and extended from just above to just below the J-groove weld. The boat sample from Penetration #1 captured a part of the upper portion of this crack in the region of the tube/J-groove weld interface. The intergranular nature of this crack exhibits classical PWSCC characteristics. The extent of the crack was examined by progressively grinding away thin layers of the section of the boat sample. The orientation of the ground surface was such that more weld material and less tube material was exposed at each successive grind. The initial exposed surface consisting of nearly all tube material exhibits this crack which extends through the tube material and just into the weld material where the crack ends. As successive layers are ground away, exposing more weld and less tube, the crack becomes smaller and smaller. The final ground surface, which

consists almost entirely of weld material, reveals no crack at all in the weld and a small vestige of crack in the remaining small bit of tube.

The axial crack in the tube appears to grow from the EDM surface out toward the tube/J-groove interface since it branches and connects two of the three voids. Crack growth in the tube toward the defects at the OD of the tube could indicate ID initiated PWSCC. However, neither of the two cracks in Penetration #46 connects to the ID of the tube. A supplemental ET examination of the ID surface was performed specifically to confirm the UT results that the flaws did not penetrate the ID. ET established that the cracks did not connect to the ID. Based on this fact STPNOC has concluded that the PWSCC axial crack in the tube is OD initiated. The crack most likely originated on the OD of the tube in the highly stressed region of the flooded weld defects.

Unless the cracks in Penetration #46 resulted from a different mechanism than the large crack in Penetration #1, the cracks must be OD initiated, since the flaws in Penetration #46 do not connect to the ID. The fact that the leakage in both penetrations appeared at the same time and are about the same age, suggests a single mechanism, and it seems too coincidental that two separate mechanisms could produce such similar results and be so closely connected in time.

### Three Discontinuities

X-ray examination of the boat sample revealed three discontinuities or voids located at the tube/J-groove weld interface. Material found in the voids contained elements found in weld electrode coatings indicating that the voids were areas of lack of fusion resulting from weld slag inclusions. The peripheries of the LOF flaws contain a number of short cracks to a depth of 1 or 2 grains. Although hot cracking in the weld material is a likely possibility, these intergranular cracks also appear in the nozzle, where hot cracking is not possible. Therefore, STPNOC has concluded that this cracking is PWSCC resulting from flooding of the LOF voids.

### J-Groove Weld Crack

The crack in the weld that connects the surface of the J-groove weld to the largest area of lack of fusion is singular and unique. The 0.2-inch long crack spans an 80 mil-thick ligament separating the lack of fusion void from the surface of the J-groove weld in the ground fillet transition at the tube/J-groove weld interface. The length of the crack spans and is limited to the width of the lack of fusion void. The section of the boat sample containing this crack was broken in the laboratory to expose the crack face for examination. Tenacious deposits obscured the crack face, and gradually more aggressive attempts to remove the deposits also attacked and distressed the metal surface. The crack exhibits some intergranular characteristics. The nature of the oxide deposits could be indicative of hot cracking. Fatigue and stress corrosion could also be factors in the development of this crack. In the final analysis, the precise mechanism responsible for initiating and propagating this crack could not be determined from an examination of the crack surface.

### Conclusions

The root cause is the use of Alloy 600 combined with nozzle manufacturing and installation methods that further increased the susceptibility of the metal to stress corrosion cracking when in contact with primary water.

The following discussion outlines the most likely sequence of events. The SMAW process used to construct the J-groove welds produced slag inclusions on the interface between the Alloy 600

tube and the weld. Already located in highly stressed areas on the OD of the penetration, these weld defects acted as stress risers. Early in the life of the vessel, a solitary crack developed that connected a lack-of-fusion or slag inclusion weld defect to the surface of the weld and primary water. Once the 80 mil-thick ligament was cracked and the lack-of-fusion void became flooded with primary water, all of the requisite conditions to support stress corrosion cracking existed. Minute cracks developed in both the tube and weld material around the edges of these flooded defects. One of these cracks propagated in the tube material, but not the weld. Consistent with the analytically predicted residual stresses, the crack was axially oriented.

Although there are other possible theories regarding the crack development scenario, several points are very clear regardless of the specific sequence of events.

1. The Alloy 600 BMI nozzles are susceptible to PWSCC and will crack under the right conditions. Even at Tcold, PWSCC is possible.

2. The SMAW process used to construct the J-groove welds is prone to leaving weld defects in service and creating high residual stresses.

Note: The South Texas Project did not identify any materials or fabrication techniques unique to the construction of Unit 1 that contributed to the cracking.

3. Visual examination of bare metal BMI penetrations is an effective mechanism for detecting leakage long before flaws become structurally significant.

**Development and Justification of a Repair Process for Flaws in Reactor Vessel Upper Head Penetrations**

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In preparation for the inspections of reactor vessel upper-head penetrations, which began in 1992, a repair process was developed for embedding the flawed region behind an Alloy 52 weld, to ensure that the flaw was isolated from the water environment. The weld itself is composed of a material that is immune to stress corrosion cracking, and the weld is just thick enough to qualify the flaw as an embedded flaw according to the rules of Section XI of the ASME Code.

The initial concept was intended for application to the inside surface of the head penetration tubes, and the NRC approved the process in 1996. In the year 2000, as degradation was found on the tube outer diameter, and in the attachment welds, the process was revised to cover these regions as well.

This paper will discuss the development of the repair process, as well as its application for several repairs that have been completed in operating plants. The service experience for these repairs has been excellent. The steps that have been taken to assure the quality and integrity of the weld repair will also be discussed.

**Manuscript was not available for publication in the Proceedings**

# Weld Overlay Deposit on Alloy 82/182 Butt Welds to Reduce ID Surface Stresses

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D. Covill, *Progress Energy, Raleigh, NC*

Abstract: PWSCC has been detected in Alloy 82/182 butt welds in plants in the US and Europe. The most significant incident to date resulted in leakage from a reactor vessel hot leg outlet nozzle at VC Summer during the fall of 2001. A potentially attractive remedial measure for Alloy 82/182 butt welds is to apply weld overlay on the outside of the butt weld to reduce the tensile stresses on the inside surface. Finite element analyses have been performed of pressurizer-surge-nozzle butt welds in the as-designed condition and with inside surface repairs ranging from a 30° partial arc to a full 360° weld repair. The effect of weld overlay cladding on inside surface stresses has been assessed.

This paper presents work sponsored by the Electric Power Research Institute (EPRI), Materials Reliability Program (MRP), Alloy 600 Issue Task Group.

## BACKGROUND

PWSCC cracks have been discovered in reactor pressure vessel (RPV) inlet and outlet nozzle to primary coolant pipe butt welds at VC Summer and Ringhals. Figure 1 shows the VC Summer nozzle and weld configuration with the main PWSCC cracks superimposed. An axial crack propagated completely through the weld and arrested at the low-alloy steel nozzle and the stainless steel pipe. This crack led to a leak that was discovered during a refueling outage. A small circumferential crack in the Alloy 182 cladding blunted when it reached the low-alloy steel nozzle material.

The root cause investigation at VC Summer concluded that the subject weld had been subjected to several repairs, including a repair weld to the inside surface after completion of the entire weld. Figure 2 shows results of stress analyses performed for EPRI of the VC Summer weld. The left side figure shows the operating condition hoop stress, including the effect of welding residual stresses, for the as-designed case without repairs to the ID surface. The right side figure shows that making a weld repair to the ID surface after completing the entire weld, significantly increases the inside surface hoop stress and therefore increases the potential for PWSCC. Figure 2 also shows the stress contours used for all of the stress plots in this paper.

It should be noted that the operating condition stresses in this paper represent the combined effect of welding residual stresses, hydrostatic testing (provides some mechanical stress relief) and operating pressure and temperature. The operating condition results do not include the effect of piping forces and moments. These were not included since they vary significantly from plant to plant.

### WELD OVERLAY TO REDUCE TENSILE STRESSES

A weld overlay applied to the outside surface of a butt weld will reduce the tensile stresses on the inside surface of the weld. Specifically weld shrinkage causes a reduction in diameter at the weld and a resultant reduction in tensile stress. Reducing tensile stresses will delay the time to PWSCC crack initiation and the growth rate of any preexisting cracks.

While weld overlays can also be used to provide a redundant load path around a cracked weld, the subject work was not focused upon creating a redundant load path.

Figure 3 shows a typical pressurizer surge nozzle assumed for this evaluation and Figure 4 shows a weld overlay applied over the weld to reduce ID surface stresses. The overlay assumed for this study had a thickness of 17% of the nominal pipe wall thickness and a length 5.5 times the nominal pipe wall thickness.

### FINITE ELEMENT MODEL

Finite element analyses of the surge nozzle were performed using the ANSYS finite element software. Figure 5 shows the overall finite element model, Figure 6 shows the mesh in the area of the weld and overlay and Figure 7 shows the weld passes assumed for the calculation. The area assumed for the weld repair is outlined in this figure.

The finite element modeling was performed using the following basic methodology:

- The weld was simulated using ten passes in layers from the ID to the OD.
- Thermal analyses were performed for each pass to determine the welding temperatures and these temperatures were input to the stress analysis as a function of time to determine the welding residual stresses.
- After completion of the main weld, the backgouge weld repair was simulated by analytically removing from the inside surface already completed weld elements.
- The backgouged area was repaired assuming that four passes were applied to the inside surface.
- The completed weld joint was subjected to hydrostatic test conditions that act to reduce peak stresses.
- Finally, the operating pressure and temperature were applied.

### WELDING RESIDUAL AND OPERATING CONDITION STRESSES

Figure 8 shows the welding residual stresses, operating condition stresses for the as-designed case with no ID weld repair, and operating condition stresses with the assumed ID weld repair. Axial stresses are plotted on the left and hoop stresses are plotted on the right. Also reported for each case are the maximum axial and hoop stresses on the ID surface for the operating condition cases. These results show that the maximum hoop stresses exceed the maximum axial stresses and that a weld repair to the ID surface after completing the main weld significantly increases both the axial and hoop stresses on the ID surface. These results are for a 2D axisymmetric model.

Figure 9 shows results for a 3D model with the weld repairs performed for partial arcs of 30°, 60°, and 90°. These results show similar stresses to the case for a 360° repair, although the axial stresses tend to be slightly higher. On this basis it is assumed that the effectiveness of the overlay can be evaluated using the 2D axisymmetric model.

Figure 10 shows the effect of applying weld overlay for both the as-designed case and the case with a 360° ID weld repair. In both cases the axial and hoop stresses are significantly reduced. This should have a beneficial effect on both crack initiation and growth.

## SUMMARY OF RESULTS

Table 1 summarizes the analysis results.

## CONCLUSIONS

The conclusions of the analysis are as follows:

- Repairs to the ID surface of a butt weld after completing the through-wall weld increases both the axial and hoop stresses on the ID surface.
- Partial-arc ID repairs also produce higher hoop and axial stresses in the inside surface.
- Weld overlay applied to the outside surface of the butt weld reduces the hoop and axial stresses on the ID surface and should therefore reduce the susceptibility to PWSCC initiation and growth.
- Weld overlay dimensions (thickness and length) can be selected to produce the desired stress reduction over the area of potentially high ID stresses.
- The axial length of the overlay deposits must be selected such that any increase in axial ID stress due to bending occurs at a location where the material is not susceptible to PWSCC.

Finally, it should be noted that the subject analysis is not intended to represent an actual weld overlay design. The purpose of the work was to demonstrate the beneficial effect of weld overlays.

Table 1
Summary of Analysis Results

- As-Designed (no weld repair)

| Direction | No Overlay | Overlay |
|---|---|---|
| Hoop | 9.0 ksi | -23.2 ksi |
| Axial | -2.7 ksi | -8.6 ksi |

- With 360° ID Weld Repair

| Direction | No Overlay | Overlay |
|---|---|---|
| Hoop | 52.8 ksi | 19.9 ksi |
| Axial | 32.5 ksi | 1.7 ksi |

Figure 1
Locations of Cracks in VC Summer Hot Leg Outlet Nozzle Weld

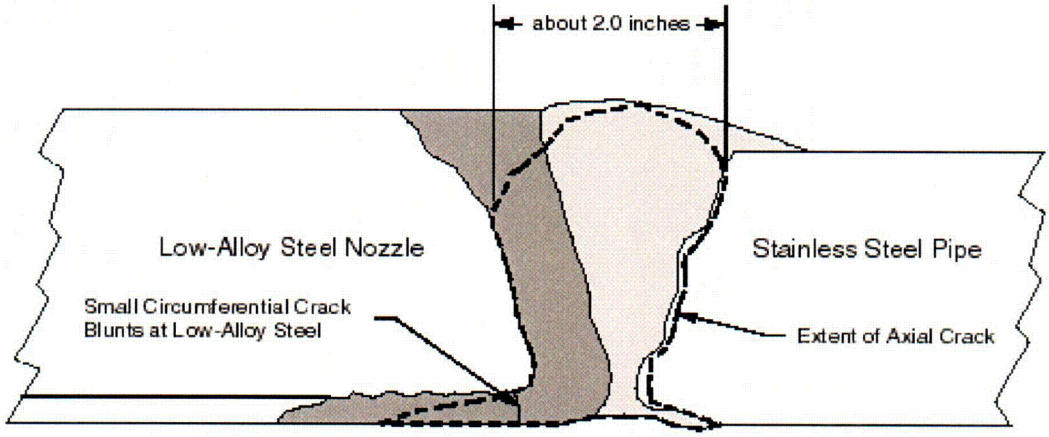
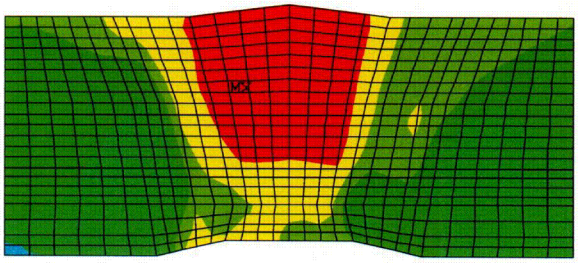
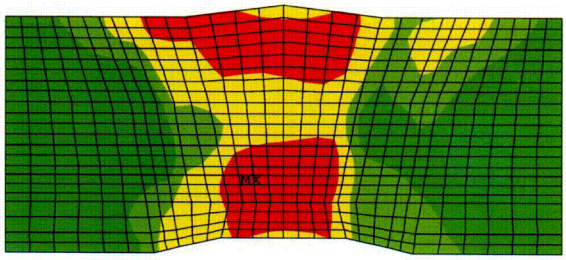



Figure 2
Effect of ID Weld Repairs on Butt Weld Operating Hoop Stress

| As-Designed Condition | With ID Repair to Completed Weld |
|---|---|
|  |  |



C21

Figure 3
Pressurizer Surge Nozzle Cross Section

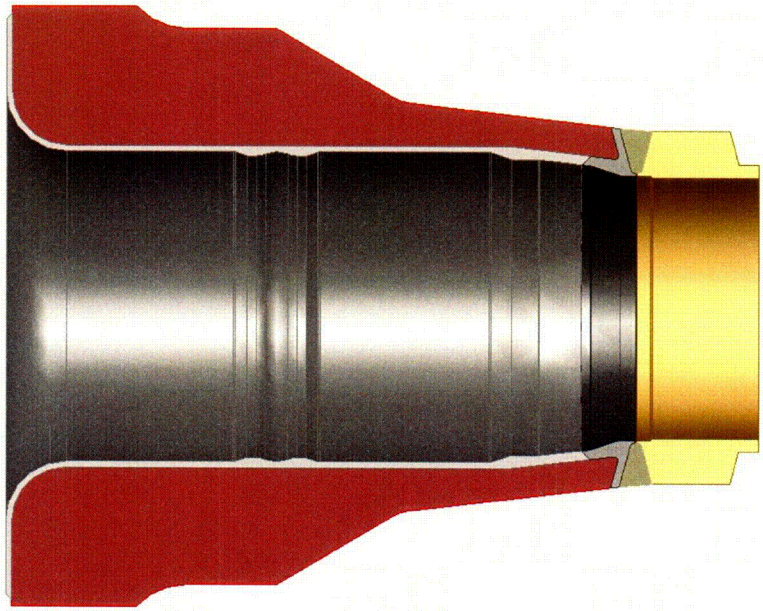
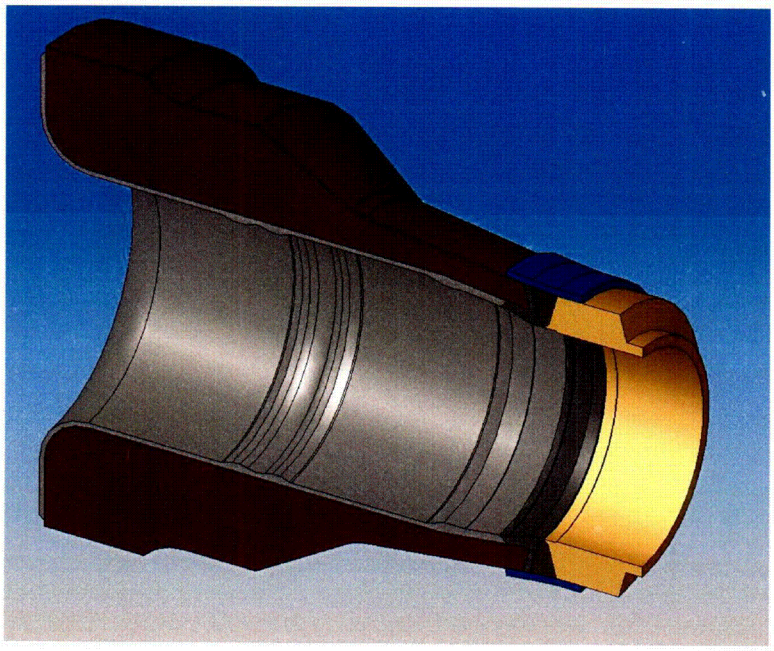


Figure 4
Pressurizer Surge Nozzle with Weld Overlay to Reduce Tensile Stress





C22

Figure 5
Pressurizer Surge Nozzle – Overall Finite Element Model

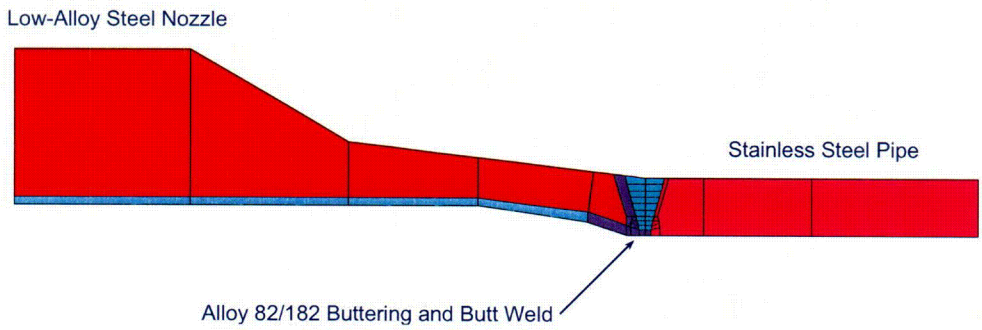



Figure 6
Pressurizer Surge Nozzle – Mesh in Area of Weld and Overlay

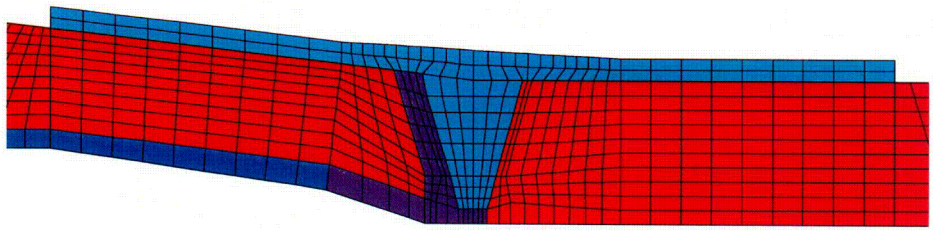
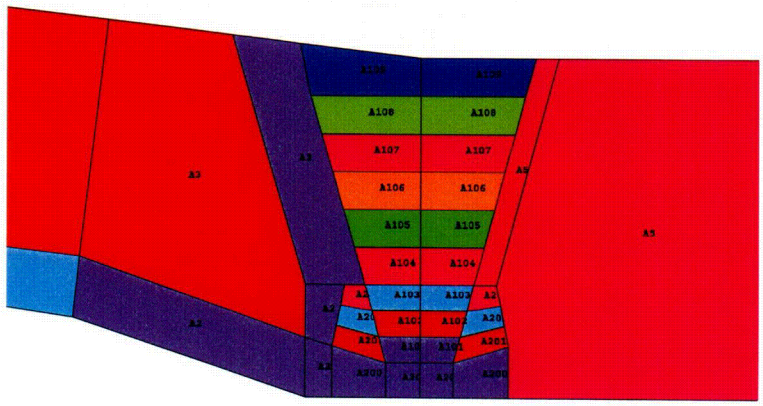


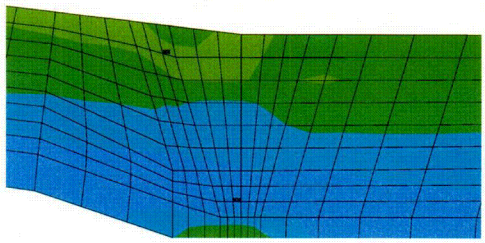
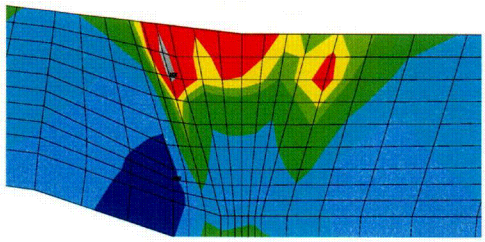
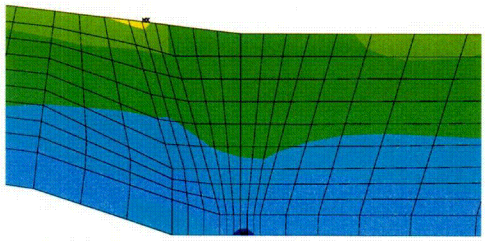
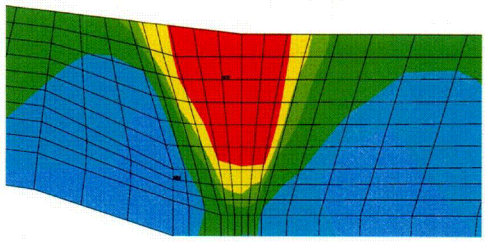
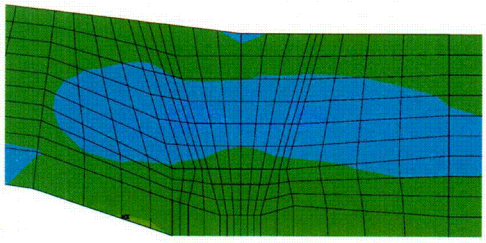
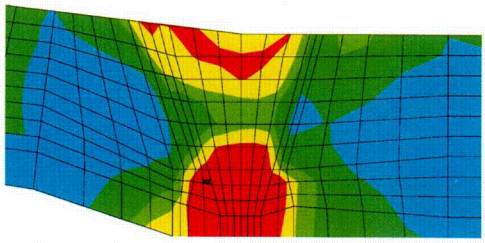
Figure 7
Pressurizer Surge Nozzle – Weld Passes Modeled






C23

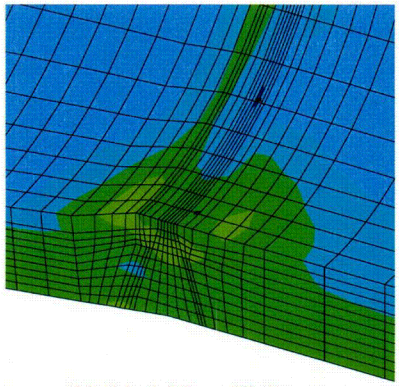
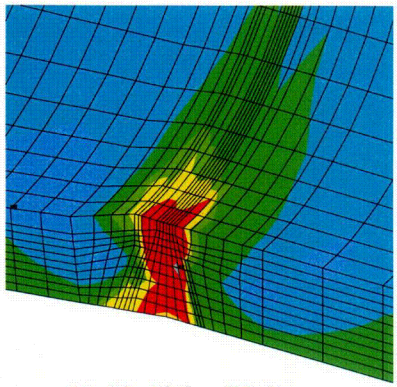
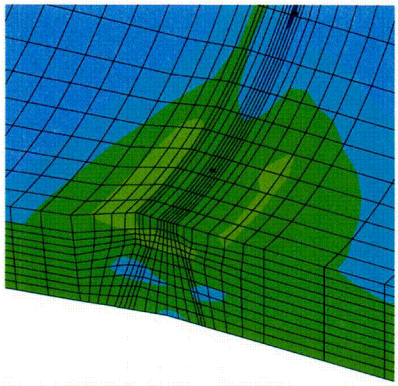
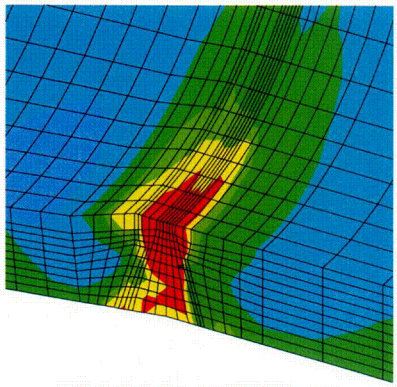
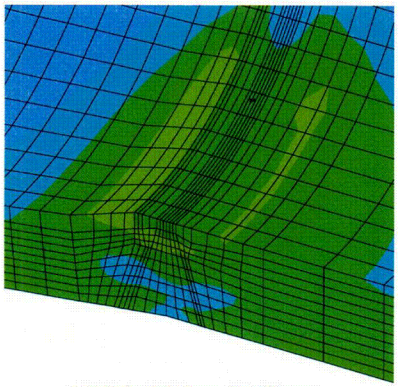
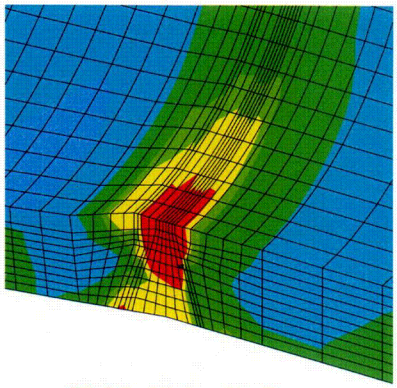
Figure 8
Welding Residual and Operating Stresses With and Without 360° ID Weld Repair

| Condition | Axial Stress | Hoop Stress |
|---|---|---|
| Welding Residual Stresses |  |  |
| Welding Residual and Operating Stresses – As Designed |  -2.7 ksi Max ID Stress |  9.0 ksi Max ID Stress |
| Welding Residual and Operating Stresses – With 360° ID Repair |  32.5 ksi Max ID Stress |  52.8 ksi Max ID Stress |



C24

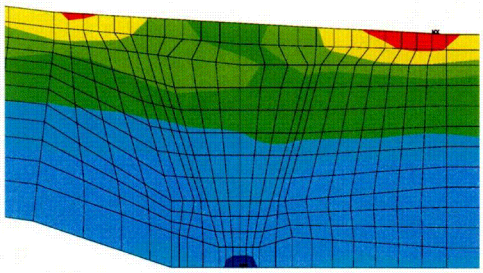
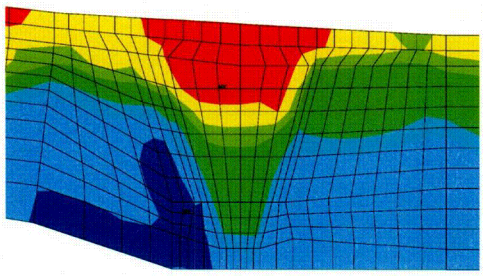
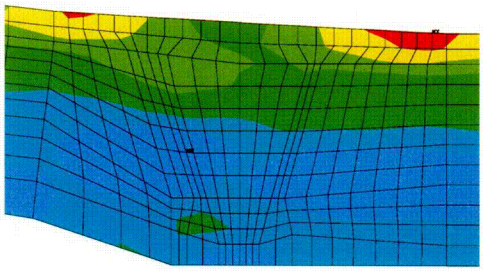
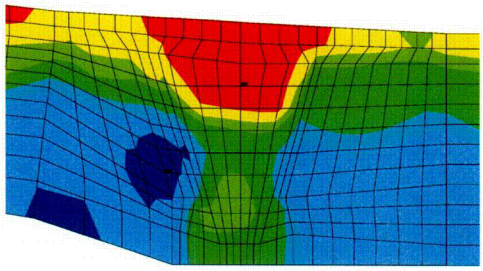
Figure 9
Welding Residual and Operating Stresses With Partial-Arc ID Weld Repairs

| Condition | Axial Stress | Hoop Stress |
|---|---|---|
| Welding Residual and Operating Stress 30° Arc ID Repair |  39.9 ksi Max ID Stress |  54.6 ksi Max ID Stress |
| Welding Residual and Operating Stress 60° Arc ID Repair |  38.6 ksi Max ID Stress |  52.6 ksi Max ID STress |
| Welding Residual and Operating Stress 90° Arc ID Repair |  40.2 ksi Max ID Stress |  52.0 ksi Max ID Stress |



C25

Figure 10
Welding Residual and Operating Stresses With Overlay Deposit

| Condition | Axial Stress | Hoop Stress |
|---|---|---|
| Welding Residual and Operating Stresses With Overlay – As Designed |  -8.6 ksi Max ID Stress |  -23.2 ksi Max ID Stress |
| Welding Residual and Operating Stresses With Overlay - 360° ID Weld Repair - |  1.7 ksi Max ID Stress |  19.9 ksi Max ID Stress |



C26