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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**
**Washington, DC 20555-0001**

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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 **B** 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 **C** regulatory, research, and plant operations **C** 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, Director
Office of Nuclear Regulatory Research

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# 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

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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.

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[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.

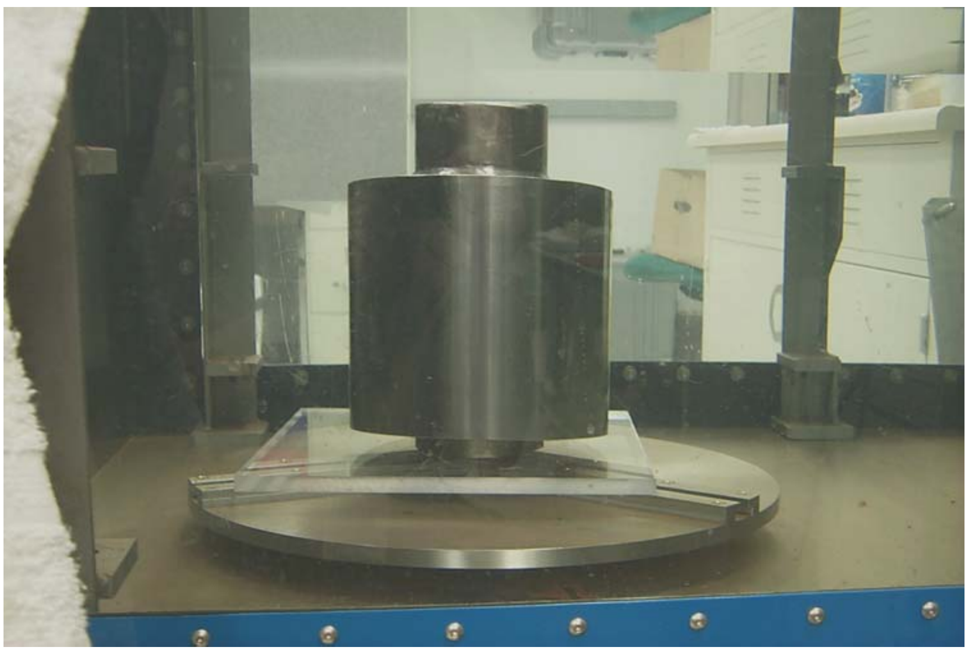

**Figure 1. Midland CRDM after machining and located in water immersion tank prepared for UT inspection.**

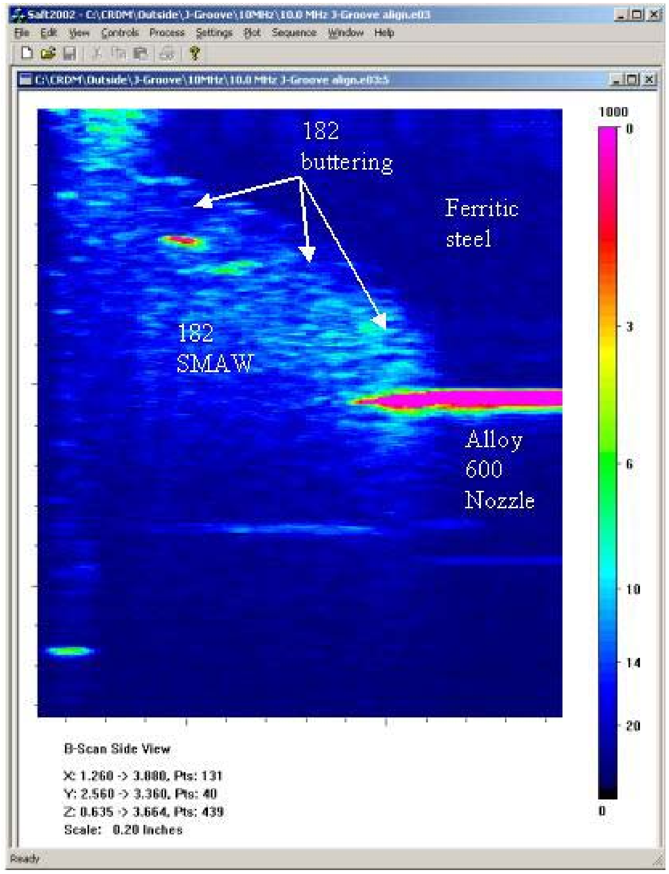


**Figure 2. UT results using normal incidence at 10 MHz spherically focused insonfication showing the four product forms.**

**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.

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 (≈650ºF), hot leg temperatures (≈600ºF), and cold leg temperatures (≈550ºF). 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): Tube and/or Weld Cracked | Tube Cracked | Weld Cracked | Notes |
|---|---|---|---|---|---|---|---|---|---|---|---|
| 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): Total | Due to Tube | Due to Weld | Repair Technique (Note 2) | Repair Method Would Likely Have Detected Significant Wastage? | Notes |
|---|---|---|---|---|---|---|---|---|---|---|---|
| 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 | | | |

NOTES:
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 |
| Already replaced | 2002 | Fall | 2 | North Anna 2 |
| Already replaced | 2003 | Spring | 3 | North Anna 1 |
| Already replaced | 2003 | Spring | 4 | Oconee 3 |
| Already replaced | 2003 | Spring | 5 | Surry 1 |
| Replacing next refueling outage | 2003 | Fall | 6 | Crystal River 3 |
| Replacing next refueling outage | 2003 | Fall | 7 | Ginna |
| Replacing next refueling outage | 2003 | Fall | 8 | Oconee 1 |
| Replacing next refueling outage | 2003 | Fall | 9 | Surry 2 |
| Replacing next refueling outage | 2003 | Fall | 10 | TMI 1 |
| Replacing next refueling outage | 2004 | Spring | 11 | Oconee 2 |
| Replacing next refueling outage | 2004 | Fall | 12 | Farley 1 |
| Replacing next refueling outage | 2004 | Fall | 13 | Kewaunee |
| Replacing next refueling outage | 2004 | Fall | 14 | Turkey Point 3 |
| Replacing after next refueling outage | 2005 | Spring | 15 | Millstone 2 |
| Replacing after next refueling outage | 2005 | Spring | 16 | Point Beach 2 |
| Replacing after next refueling outage | 2005 | Spring | 17 | Turkey Point 4 |
| Replacing after next refueling outage | 2005 | Fall | 18 | ANO 1 |
| Replacing after next refueling outage | 2005 | Fall | 19 | Farley 2 |
| Replacing after next refueling outage | 2005 | Fall | 20 | Point Beach 1 |
| Replacing after next refueling outage | 2005 | Fall | 21 | Robinson 2 |
| Replacing after next refueling outage | 2005 | Fall | 22 | St. Lucie 1 |
| Replacing after next refueling outage | 2006 | Spring | 23 | Beaver Valley 1 |
| Replacing after next refueling outage | 2006 | Spring | 24 | Calvert Cliffs 1 |
| Replacing after next refueling outage | 2006 | Spring | 25 | St. Lucie 2 |
| Replacing after next refueling outage | 2006 | Fall | 26 | Cook 1 |
| Replacing after next refueling outage | 2006 | Fall | 27 | Fort Calhoun |
| Replacing after next refueling outage | 2007 | Spring | 28 | Calvert Cliffs 2 |
| Replacing after next refueling outage | 2007 | Fall | 29 | Cook 2 |

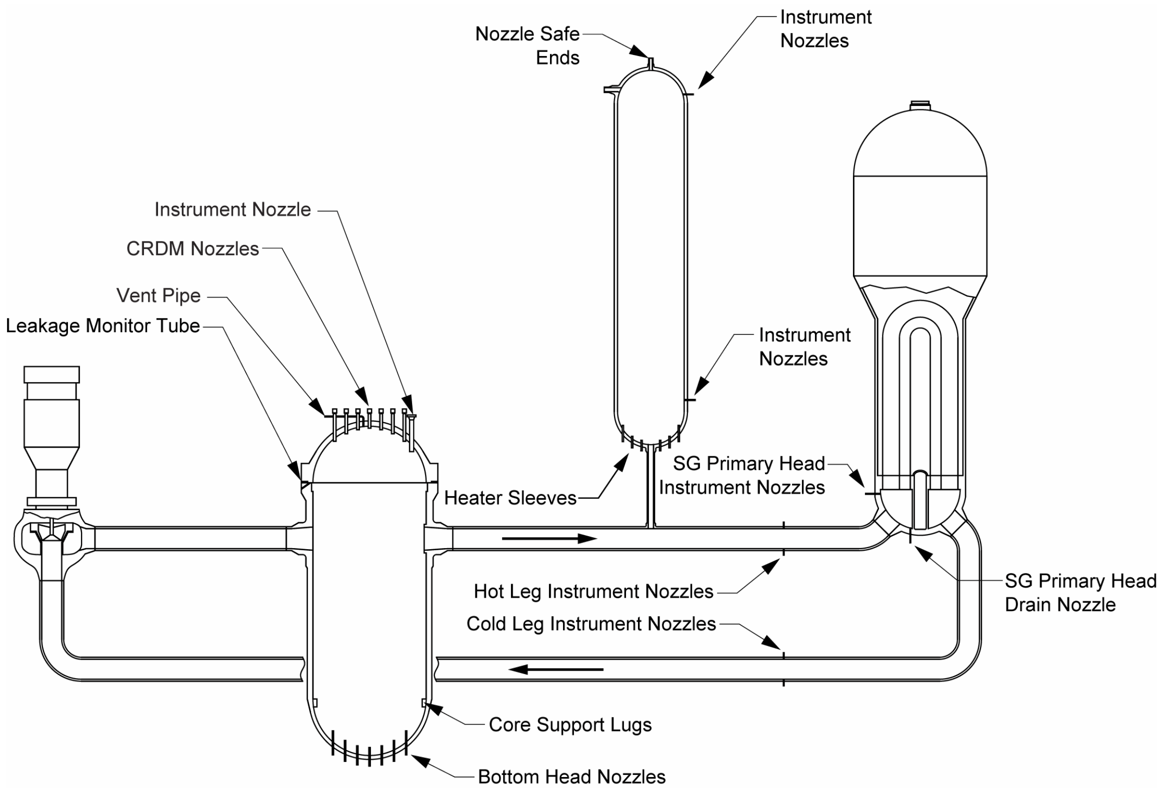



Figure 1
Locations of Thick-Section Alloy 600 Materials

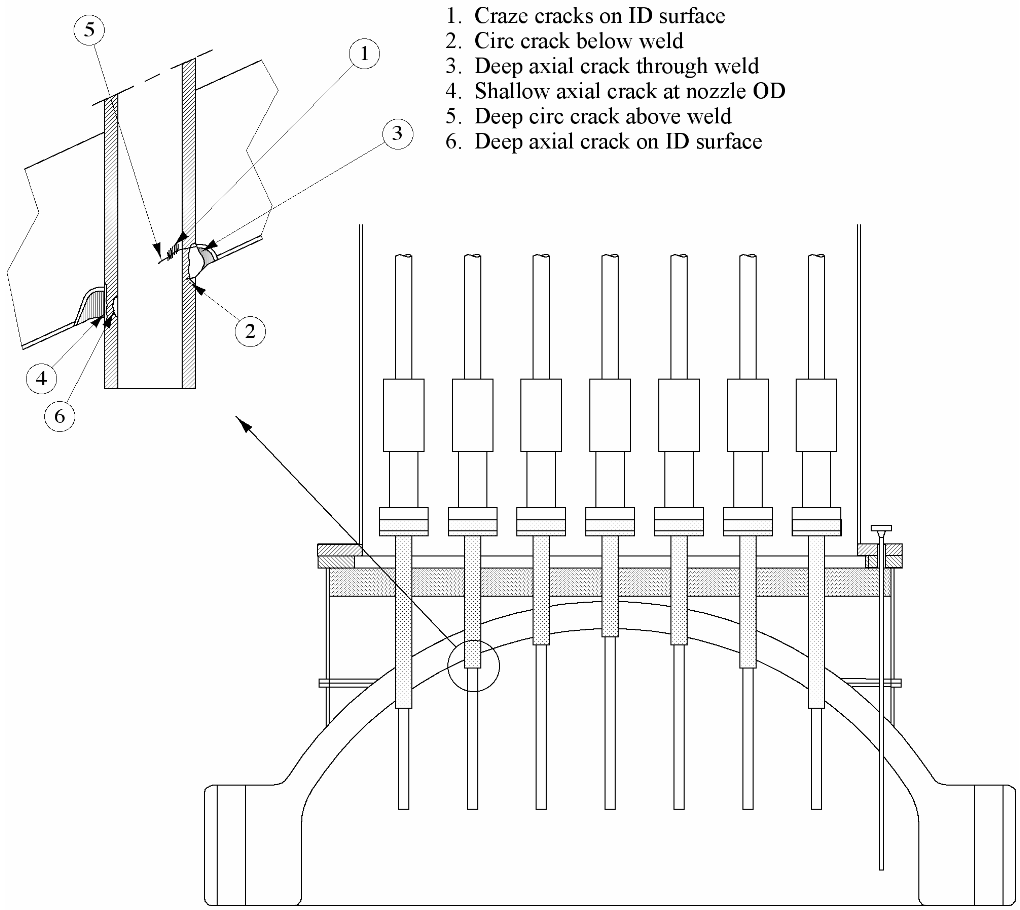



Figure 2
Location of Typical RPV Head PWSCC

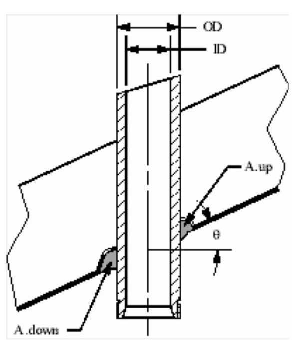
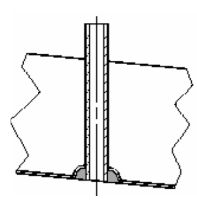
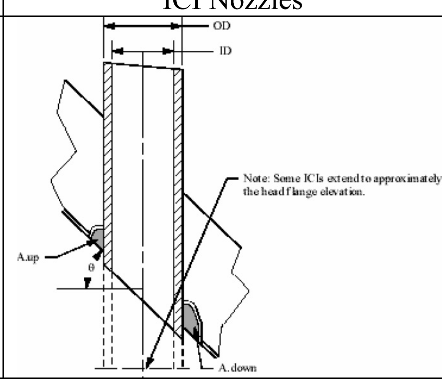
| CRDM/CEDM Nozzles | Head Vent Nozzles | ICI Nozzles |
| --- | --- | --- |
|   |  |   |

Figure 3
Typical RPV Head Penetration Types

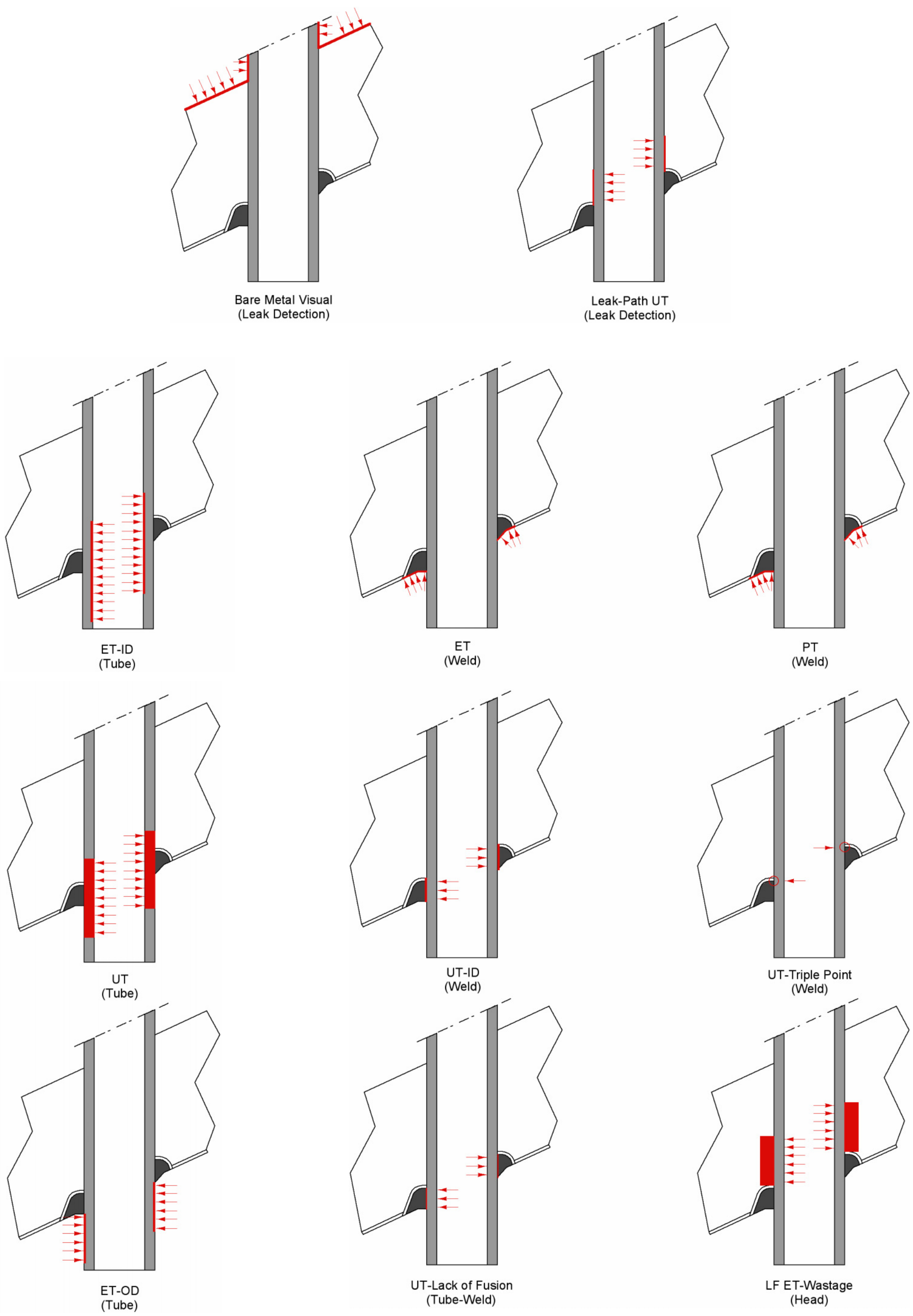



Figure 4
Summary of RPV Head Nozzle Inspection Techniques

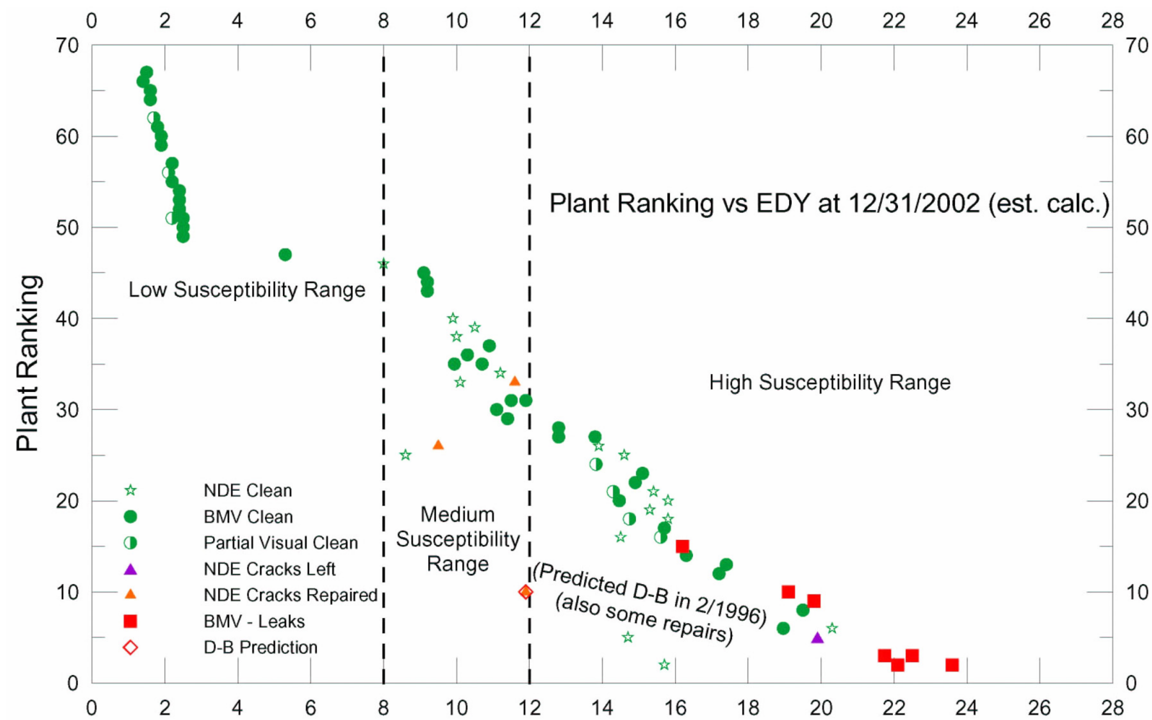



Figure 1. Ranking of domestic plants according to the EDY formula, showing results of inspections, evidence of leakage, and repairs. Many plants are shown with multiple symbols, indicating a "clean" inspection at inspection opportunity, followed by a different finding at a subsequent inspection (e.g., 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.

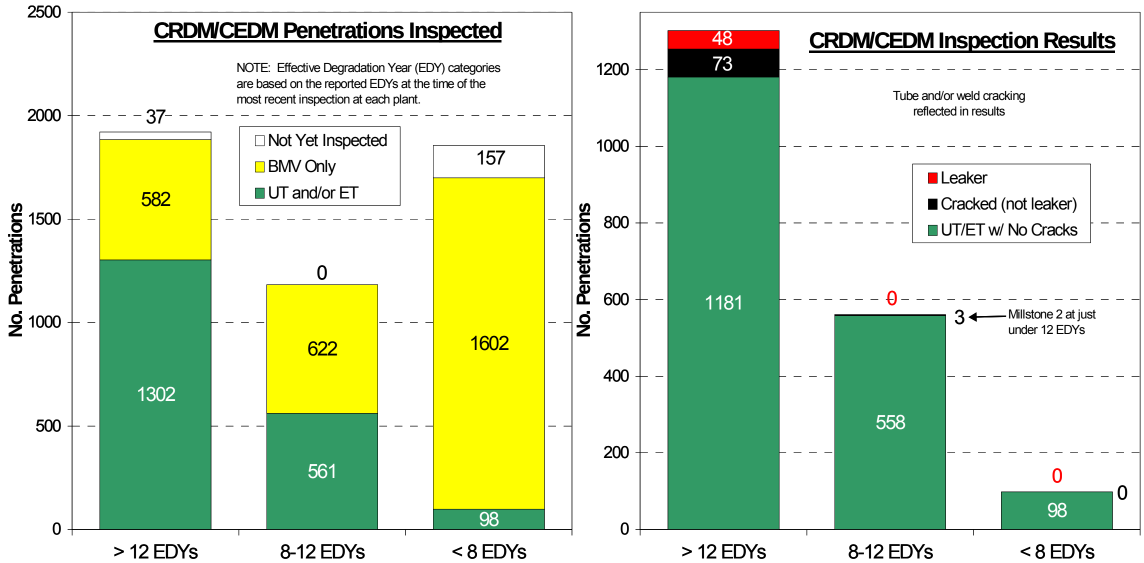



Figure 6
RPV Head Inspection Statistics – by EDY Group

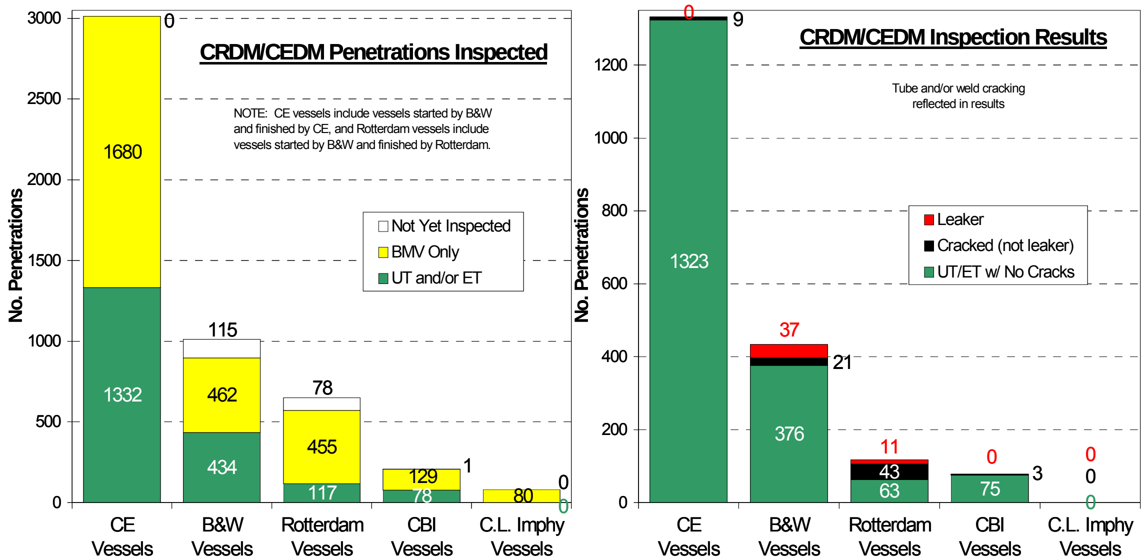



Figure 7
RPV Head Inspection Statistics – by Head Fabricator (All EDYs)

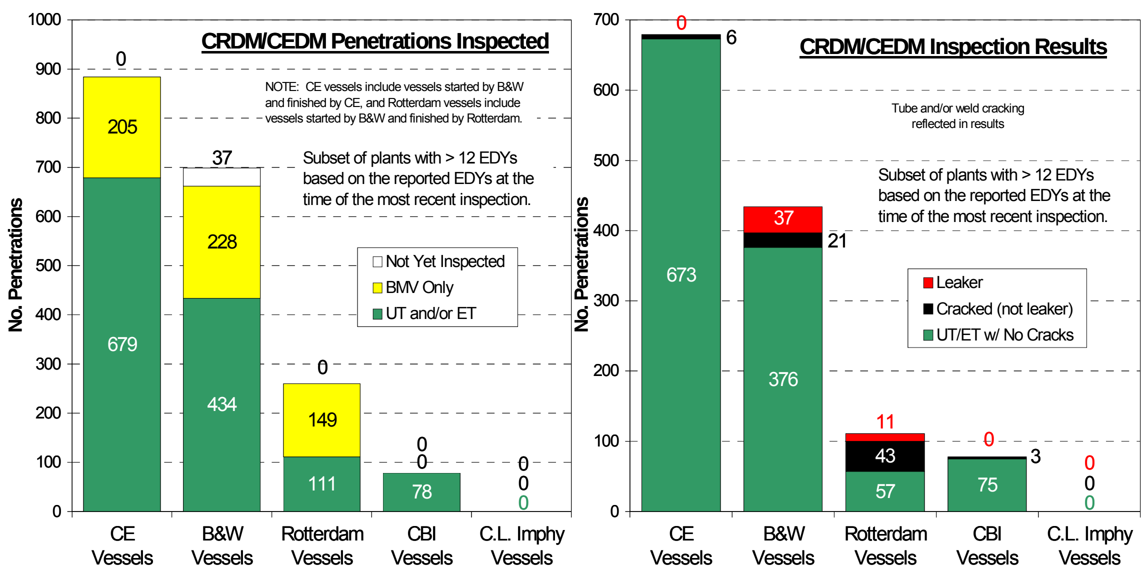



Figure 8
RPV Head Inspection Statistics – by Head Fabricator (>12 EDYs)

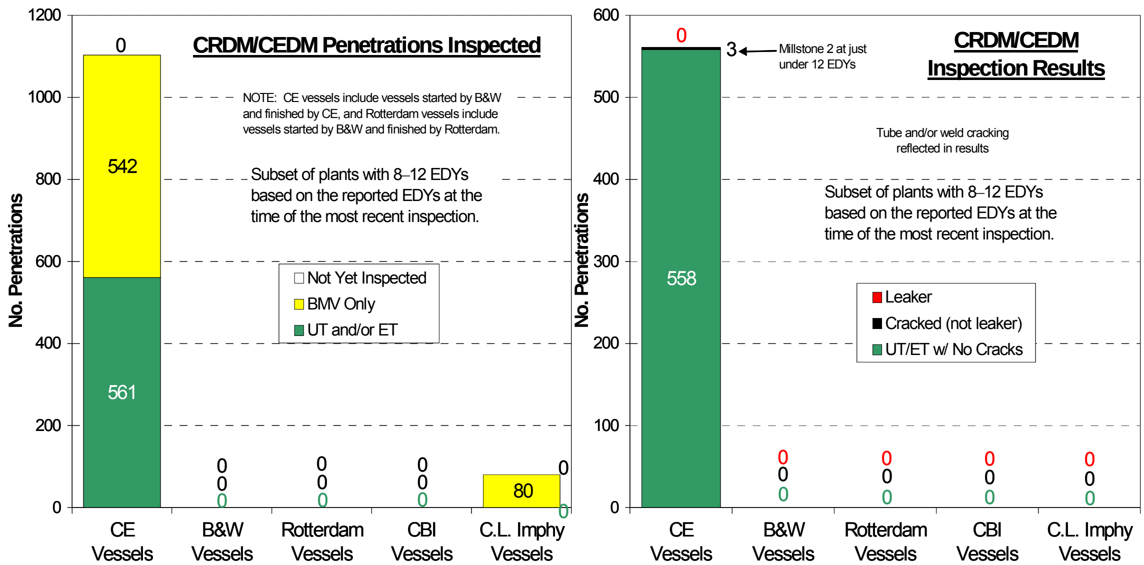



Figure 9
RPV Head Inspection Statistics – by Head Fabricator (8-12 EDYs)

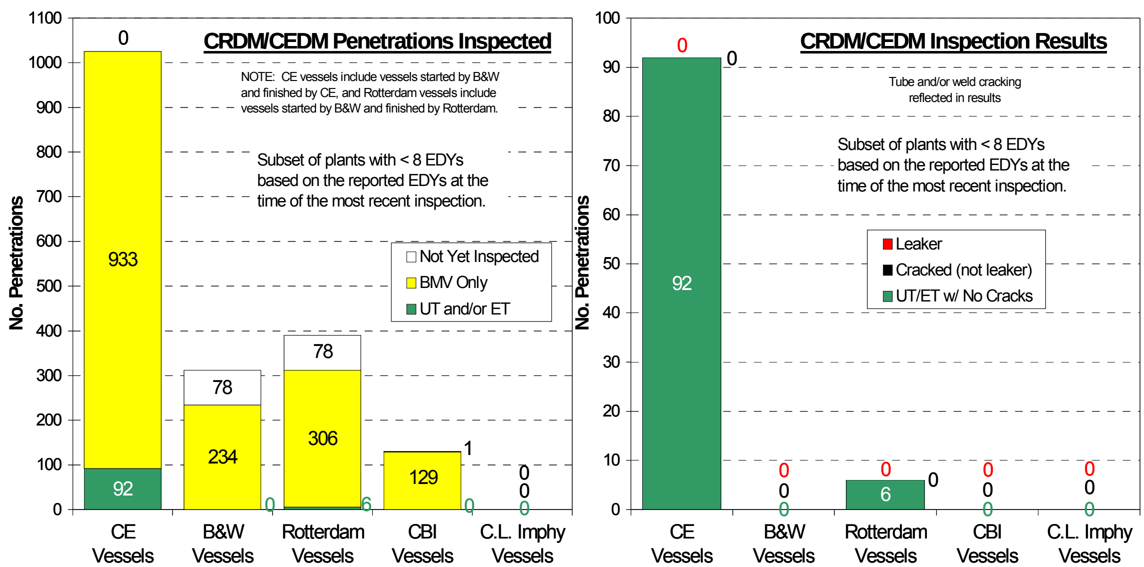



Figure 10
RPV Head Inspection Statistics – by Head Fabricator (<8 EDYs)

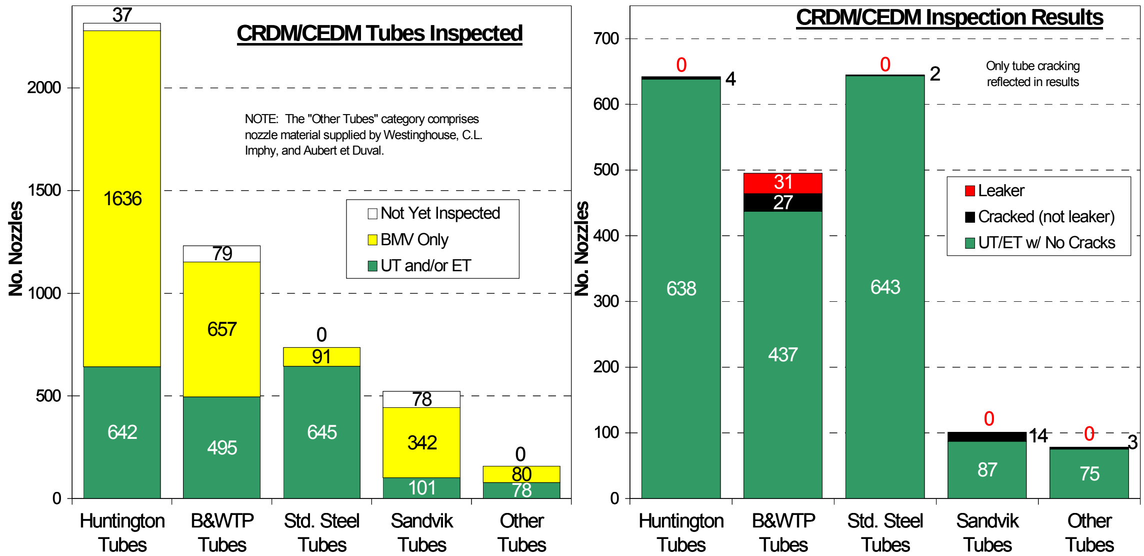



Figure 11
RPV Head Inspection Statistics – by Nozzle Material Supplier (All EDYs)

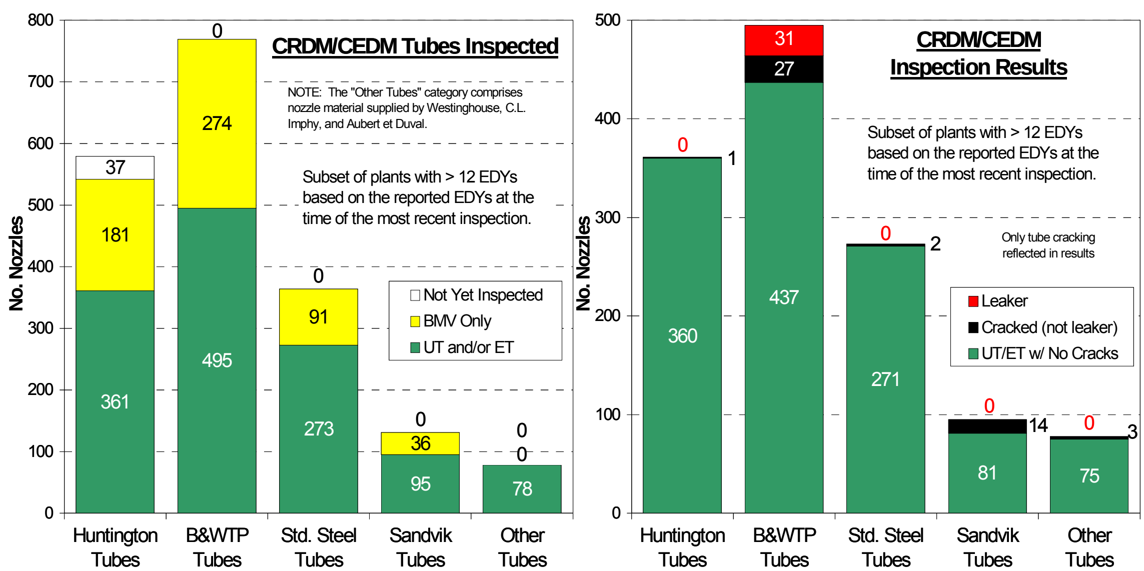



Figure 12
RPV Head Inspection Statistics – by Nozzle Material Supplier (>12 EDYs)

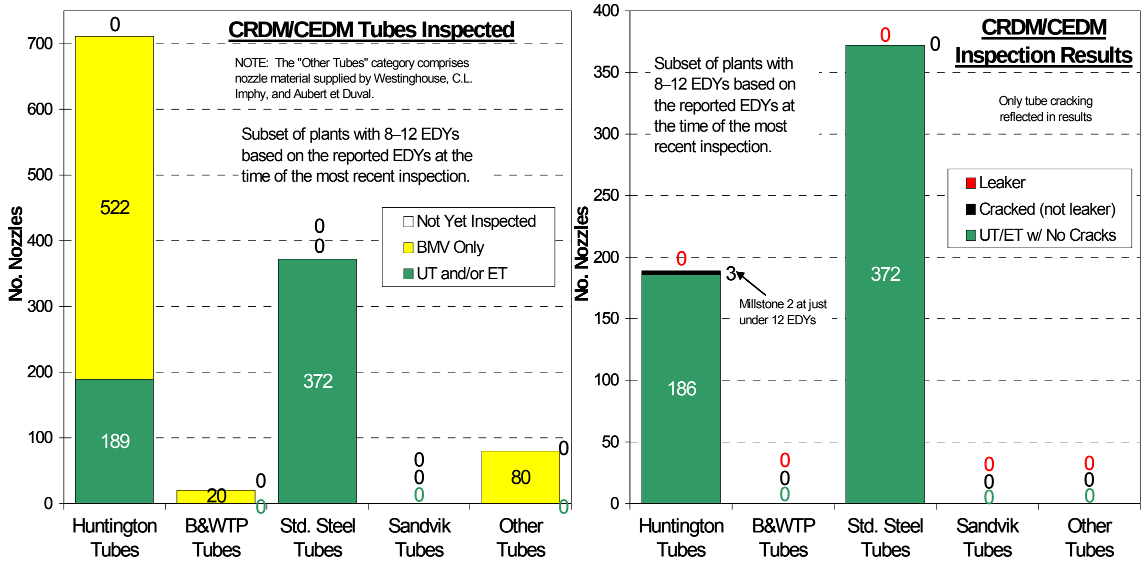



Figure 13
RPV Head Inspection Statistics – by Nozzle Material Supplier (8-12 EDYs)

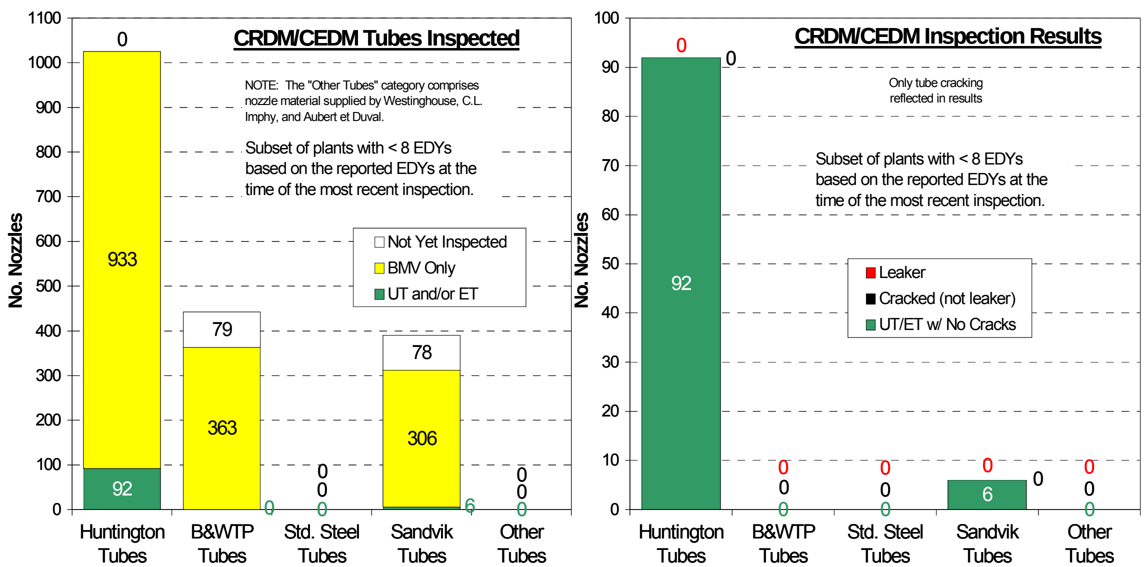



Figure 14
RPV Head Inspection Statistics – by Nozzle Material Supplier (<8 EDYs)

# 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.

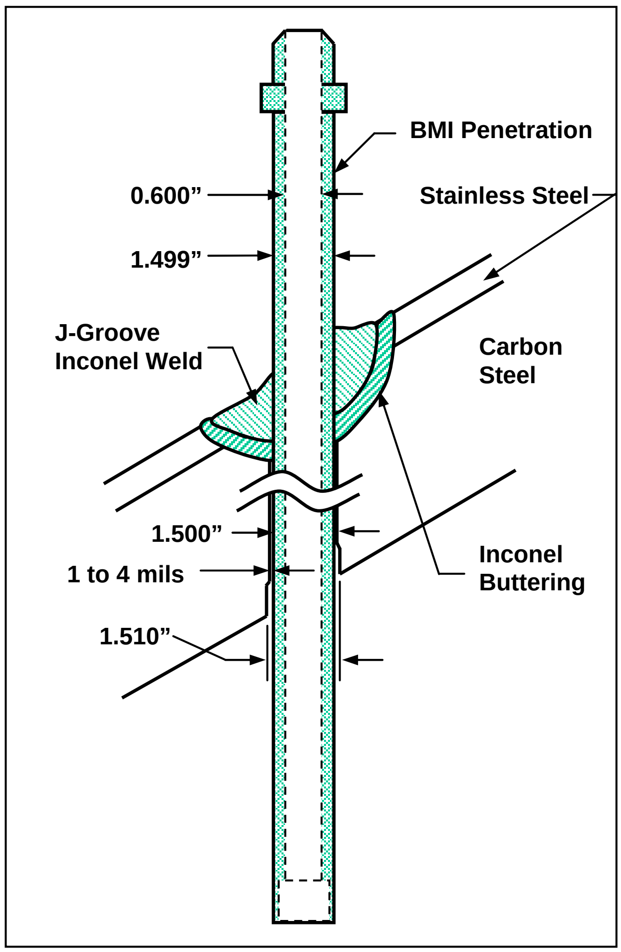


Figure 1: BMI general configuration

## Inspection History

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

**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.

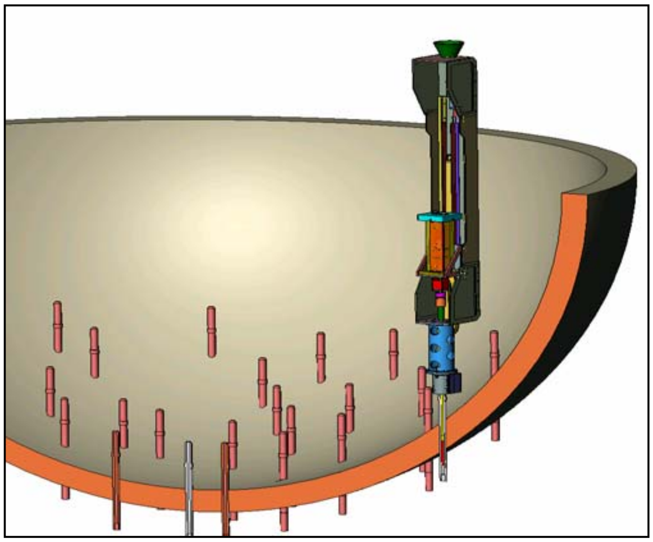


Figure 4: Representation of UT tool engaging with BMI penetrations

**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 & $0^o$ 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.

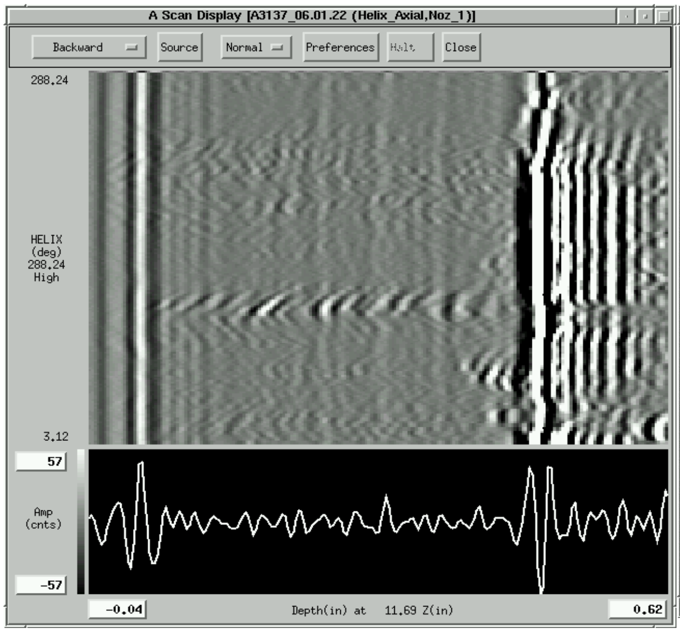



Figure 6: Penetration 1 axial probe with three axial indications and one full-penetration leak path

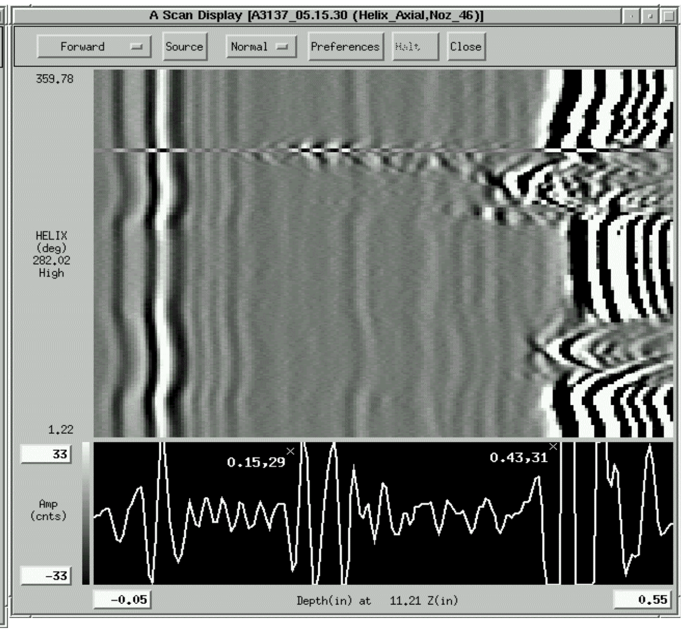



Figure 7: Penetration 46 with 2 axial indications and one leak path.

**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.

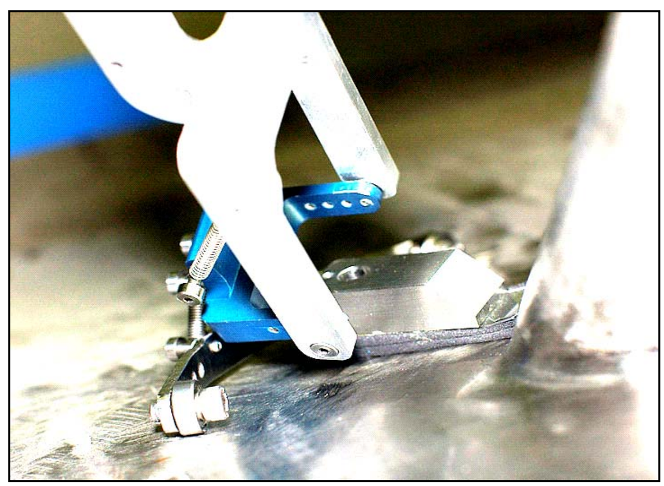


Figure 8: Compliant eddy current transducer pressed against shell and into weld fillet.

**Volumetric Phased Array Examination**

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**

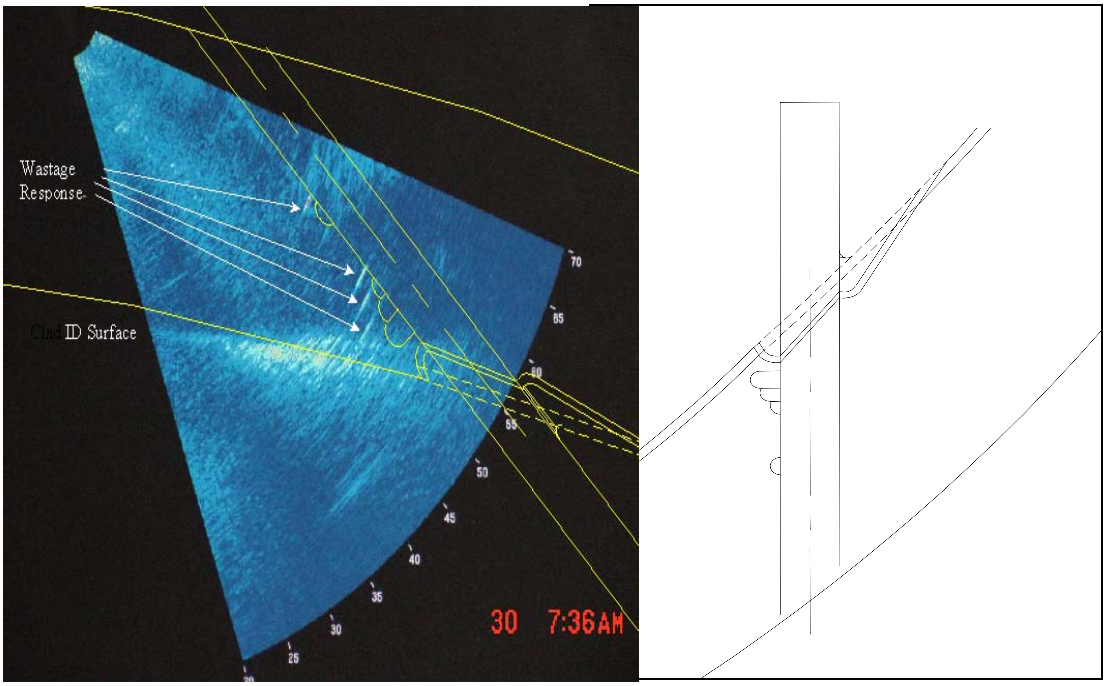



Figure 9: Phased array sector scan of shell volume calibration standard demonstrating sensitivity to wastage between surfaces.

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.

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.

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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**

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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.

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## 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.

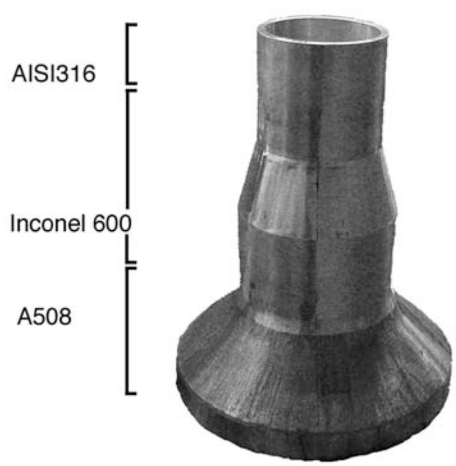


**Figure 1** Core spray nozzle mock-up with Inconel 600 safe-end.

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.

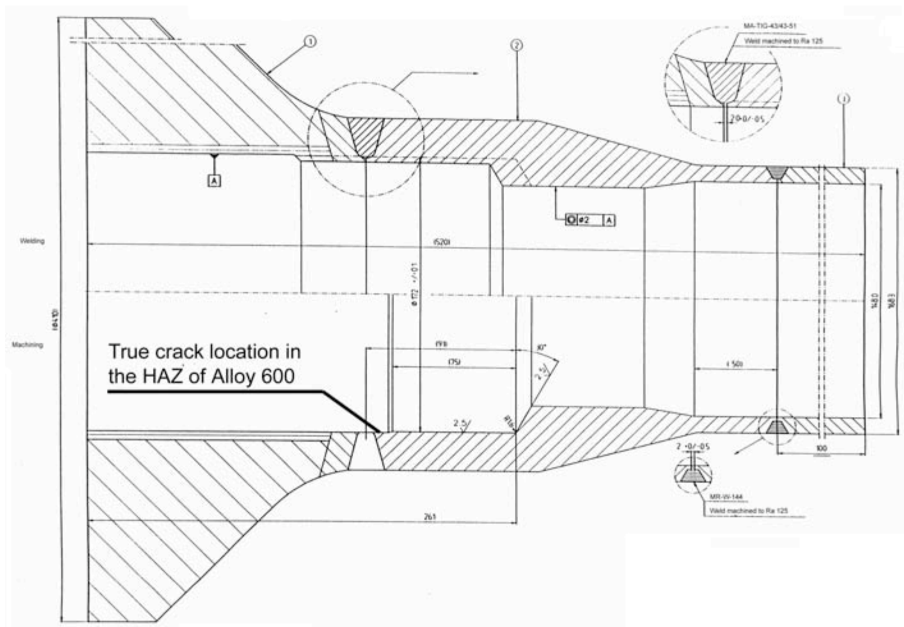


**Figure 2** Drawing of the core spray nozzle and location of the produced crack in Inconel 600 safe-end in the heat affected zone of the buttering weld.

**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.

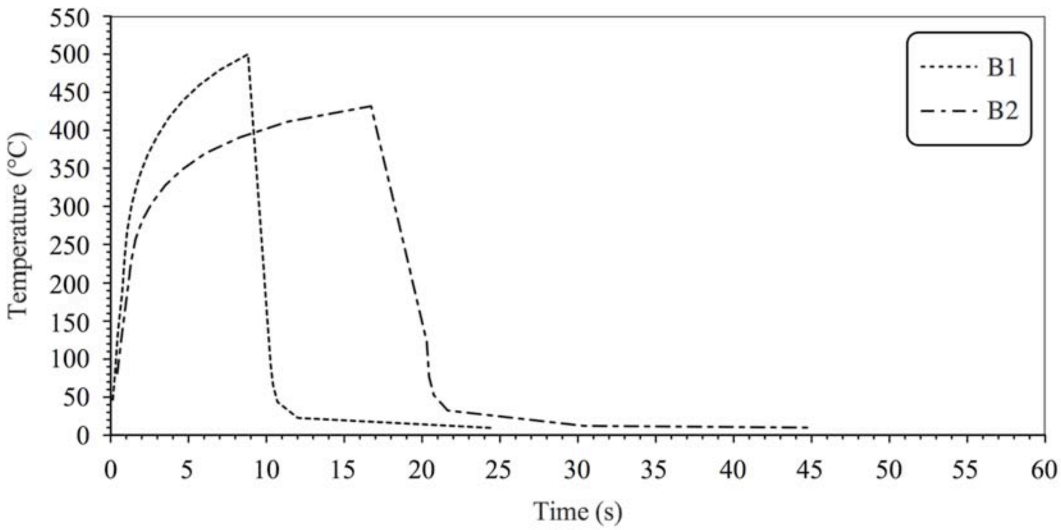


**Figure 3** Two different temperature loading cycles used in the studies.

**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.

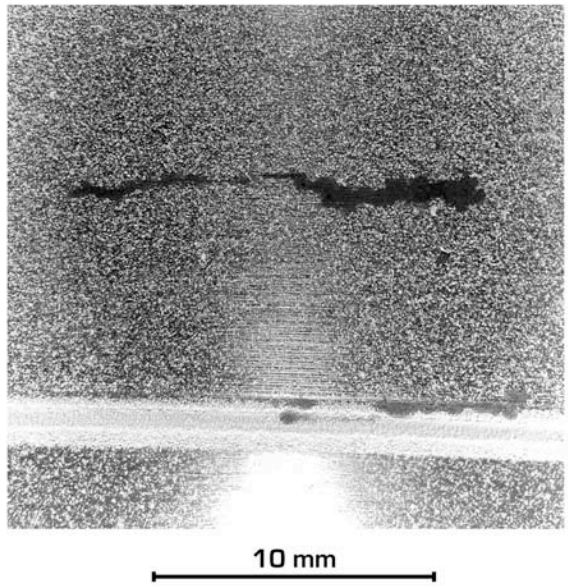


**Figure 4** Dye penetrant indication of the produced realistic crack in Inconel 600 safe-end in the heat affected zone of the buttering weld.

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).

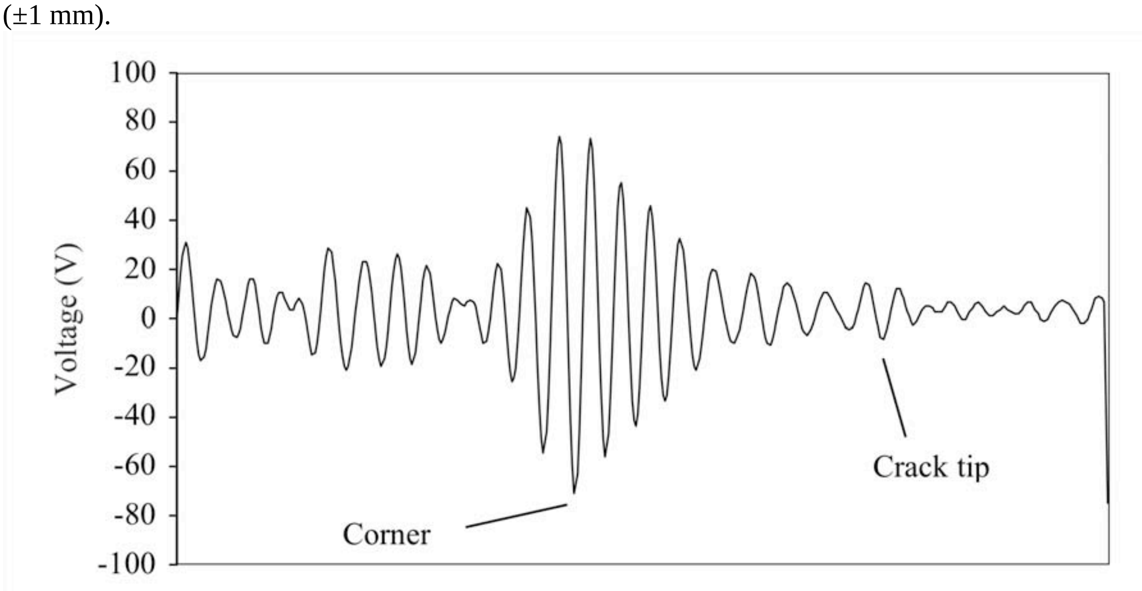


**Figure 5** A-scan obtained from the crack at room temperature (41°, 1.5 MHz, shear wave probe).

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.

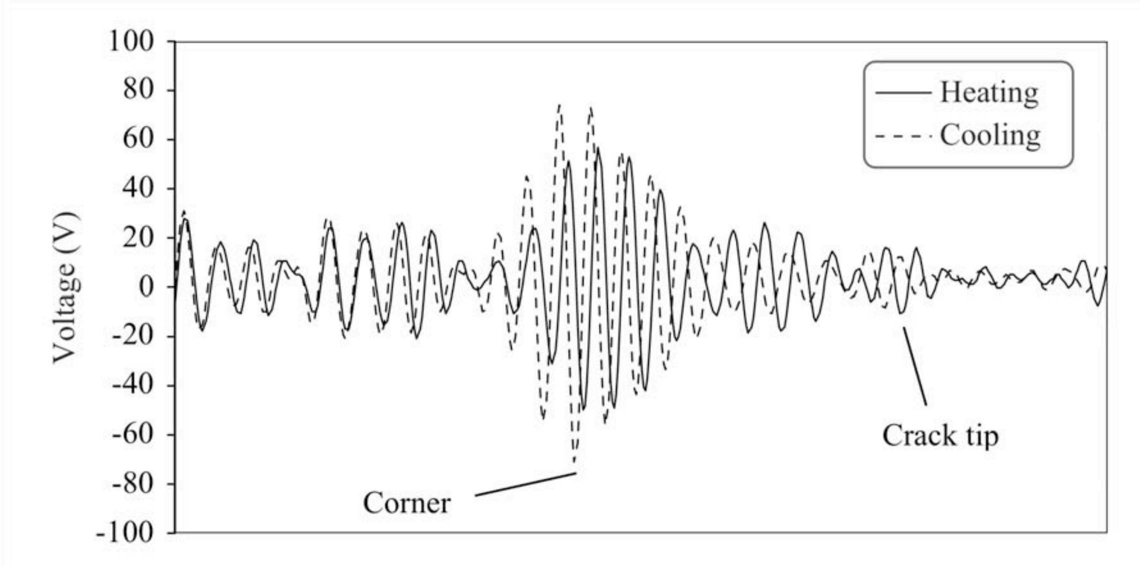


**Figure 6** A-scans from the crack in the end of cooling and in the end of heating of thermal fatigue loading cycle. Differences in ultrasonic response are related to the crack opening and closing behaviour.

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.

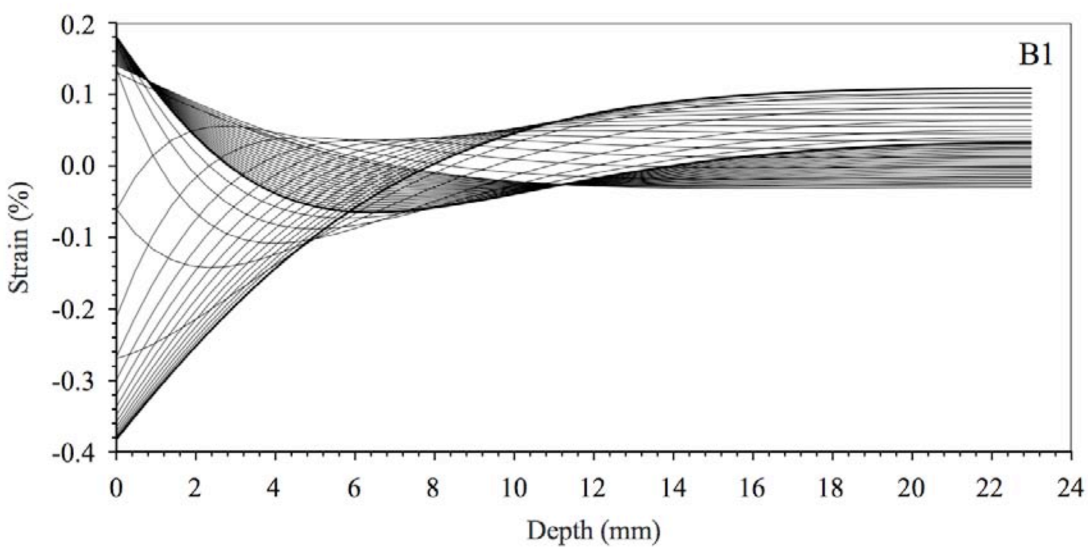


**Figure 7** Strain distribution for loading cycle B1. Nozzle ID on the left and OD on the right side of the figure.

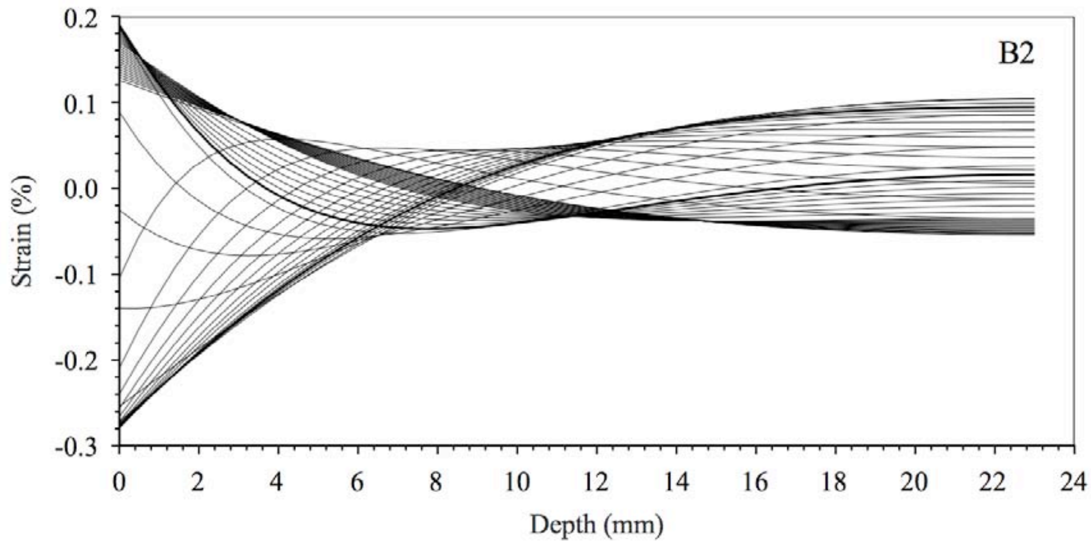


**Figure 8** Strain distribution for loading cycle B2. Nozzle ID on the left and OD on the right side of the figure.

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.

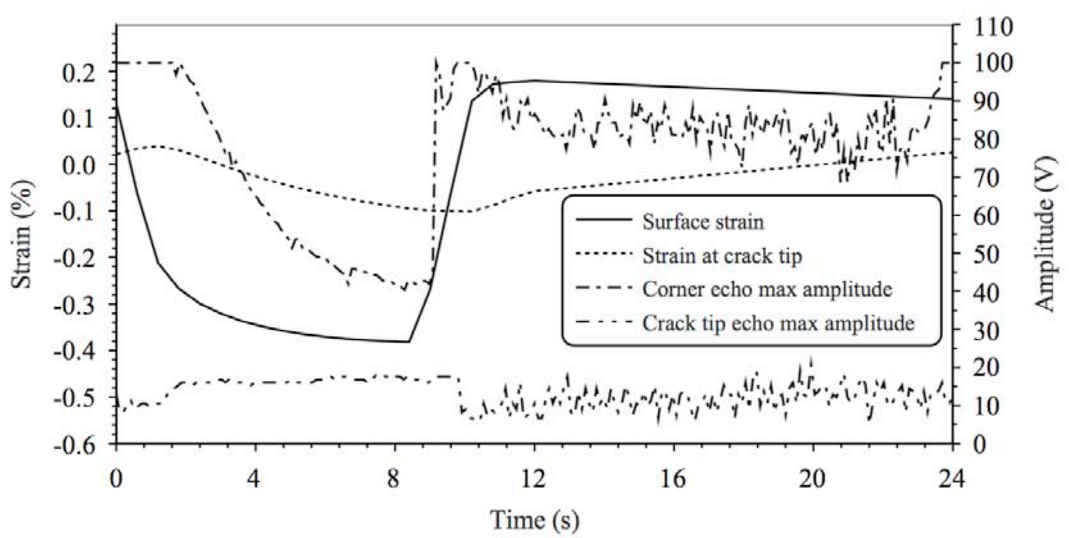


**Figure 9** Combined results of strain and ultrasonic amplitude variations from crack corner and tip caused by loading cycle B1.

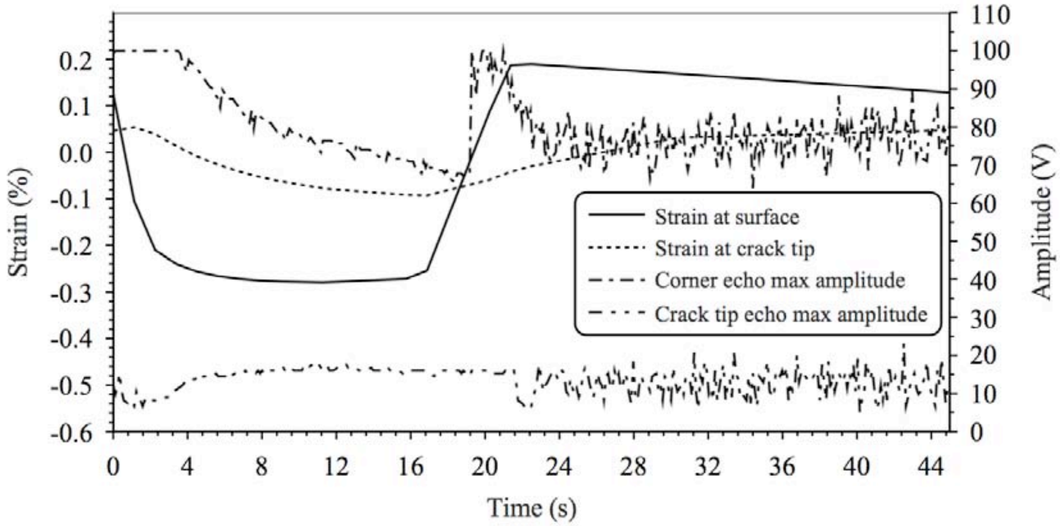


**Figure 10** Combined results of strain and ultrasonic amplitude variations from crack corner and tip caused by loading cycle B2.

**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**

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

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**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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**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 270º 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 30º 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 (≈270º) 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 30º 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.

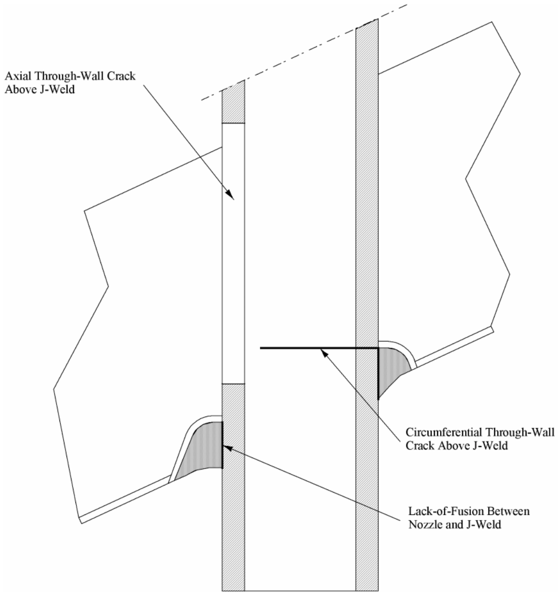
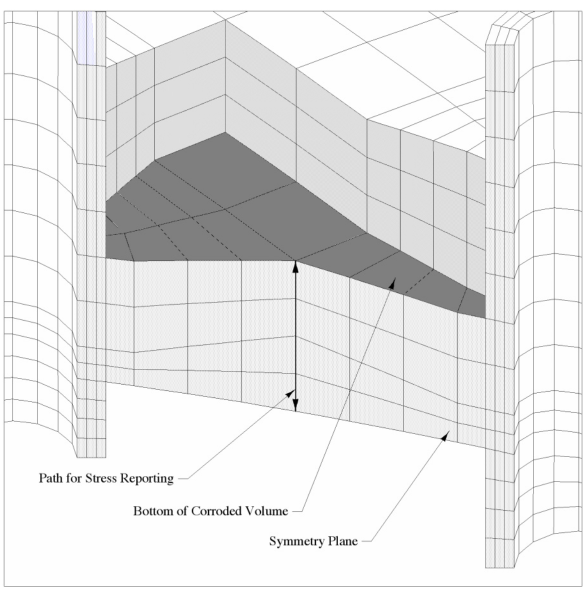


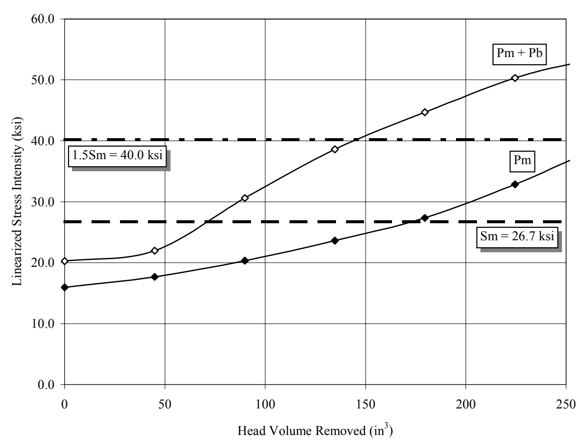

Figure 1
Tolerance to Axial and Circumferential Cracks

### 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° |

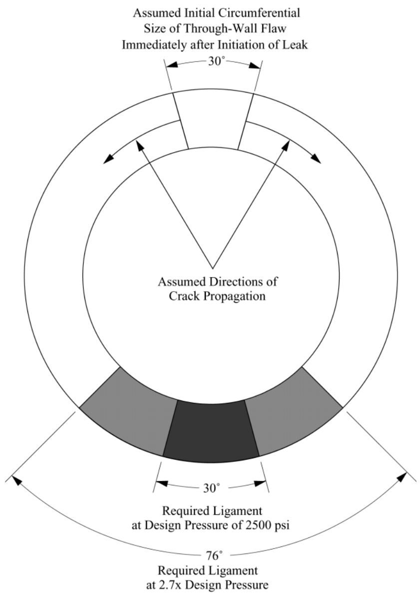



**Finite Element Model of Representative Head**

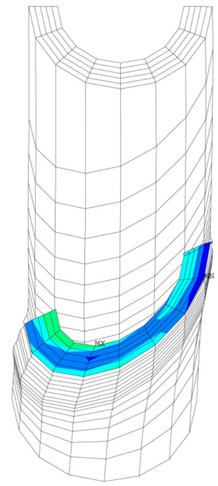



**Primary Membrane and Membrane Plus Bending Stress as Function of Wastage Volume**

Figure 2
Tolerance to Boric Acid Wastage

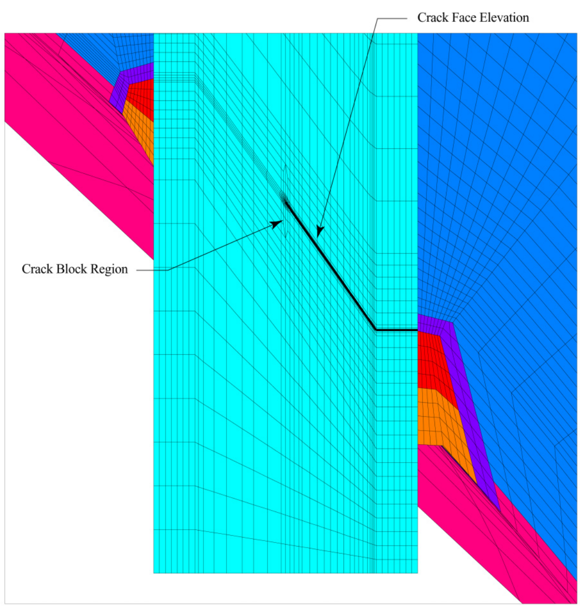



***Typical Critical Flaw Size of 330°***

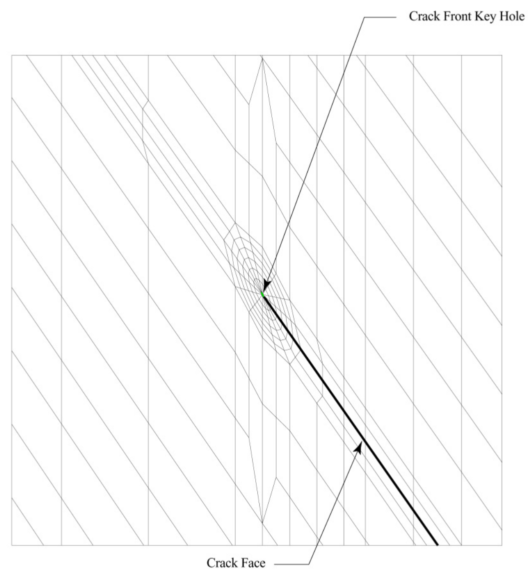


***Example of Operating Stress Perpendicular to Circ Crack Plane***

Figure 3
Crack Growth Around Nozzle Circumference




**180° Downhill-Centered Crack**




**Crack Mesh Detail**

Figure 4
Example Fracture Mechanics Model for Nozzle Circumferential Cracks

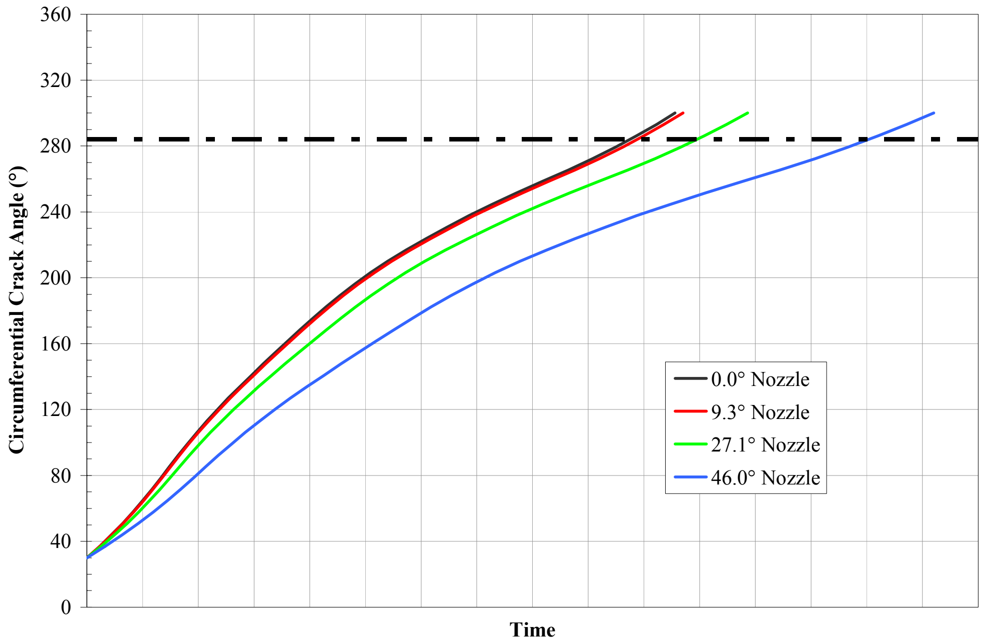
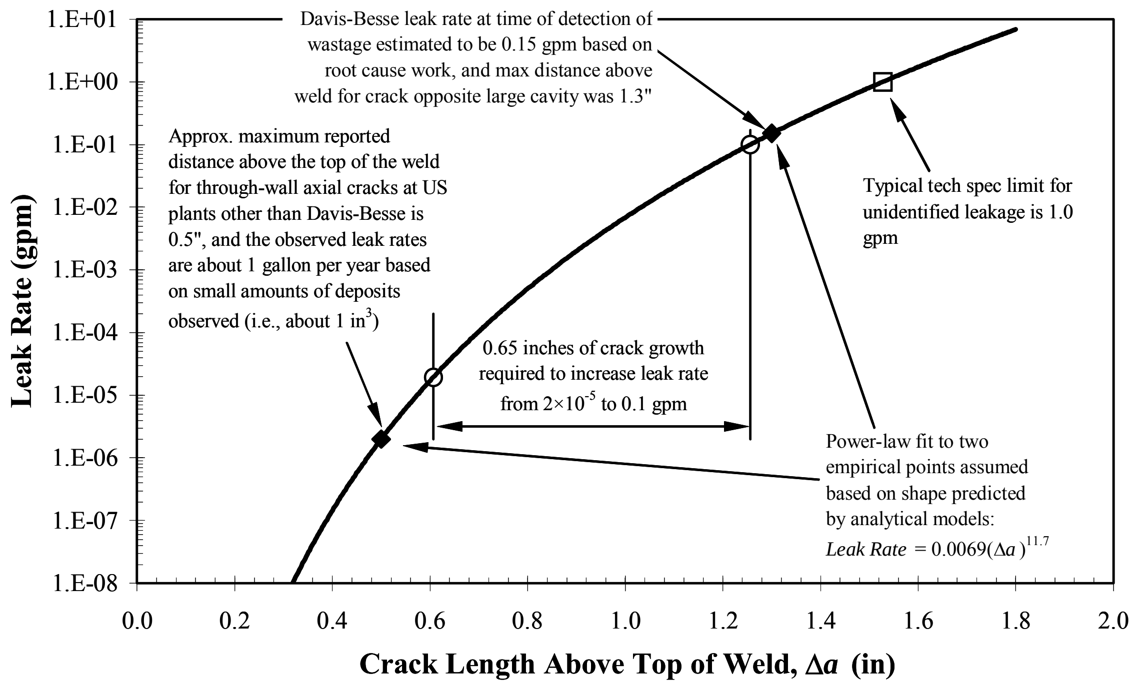



Figure 5
Example Results for Growth of Circumferential Through-Wall Cracks



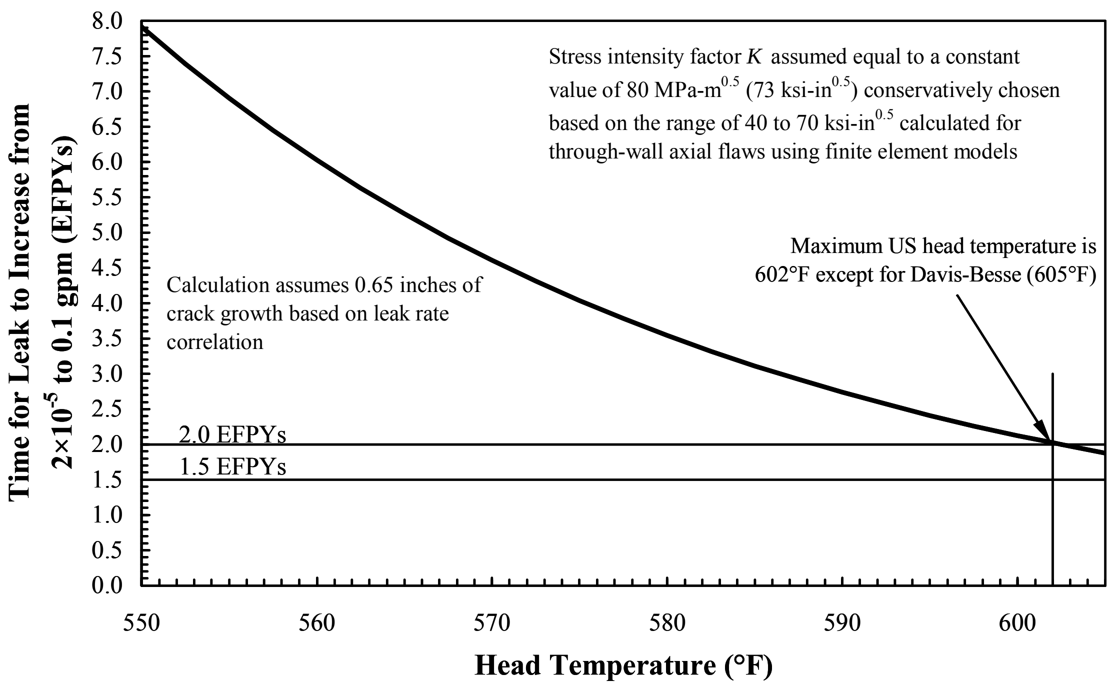




Figure 6
Deterministic Model for Boric Acid Wastage

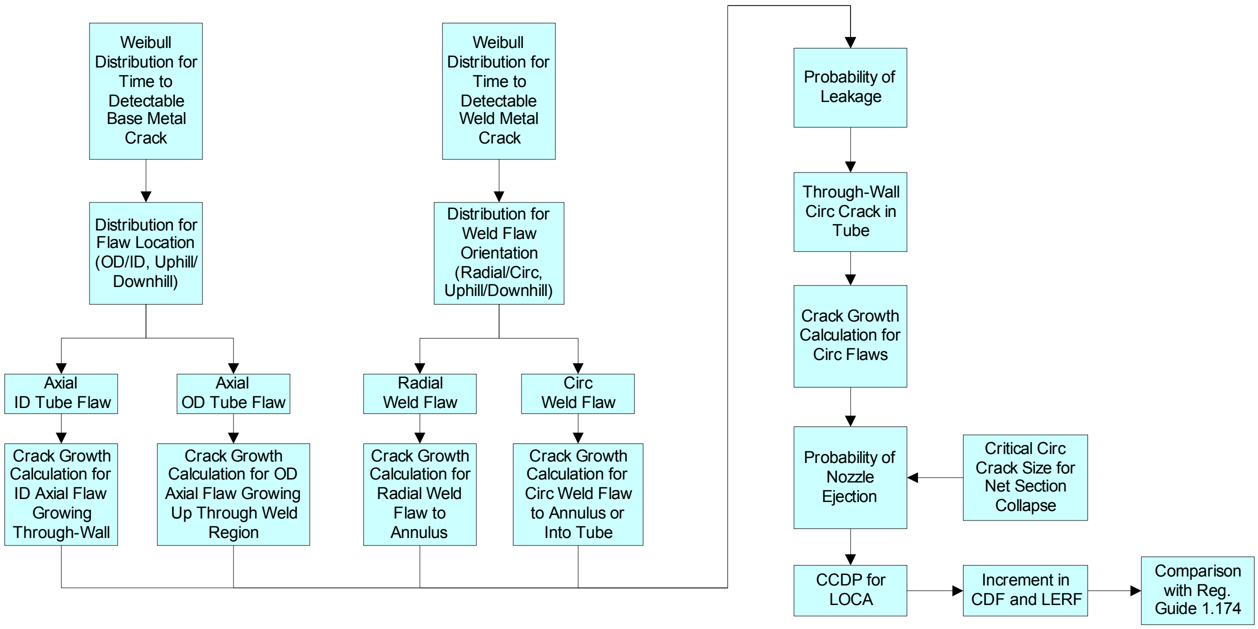



Figure 7
Simplified Flowchart of Monte Carlo Risk Assessment Model

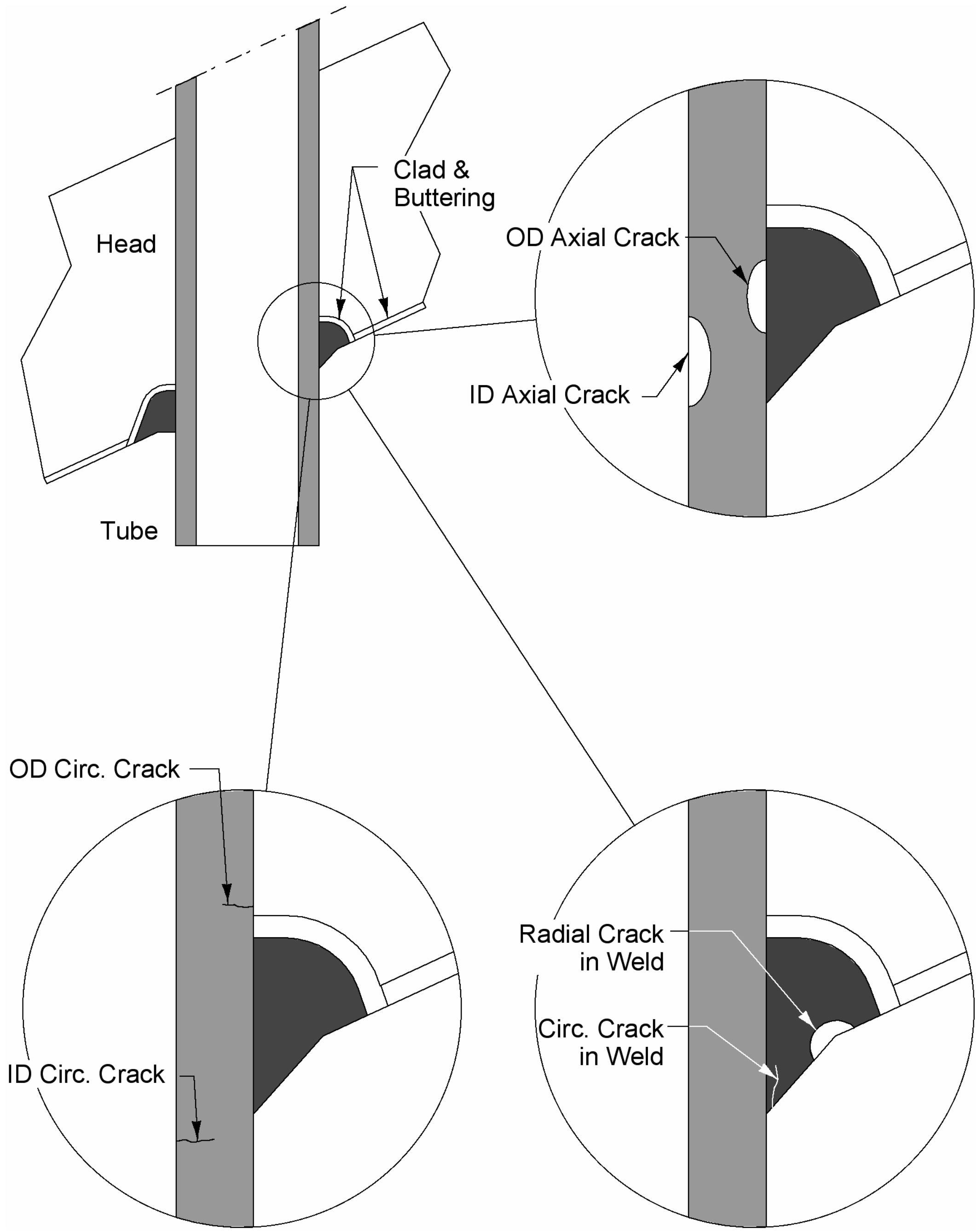



Figure 8
Modeled Flaw Geometries

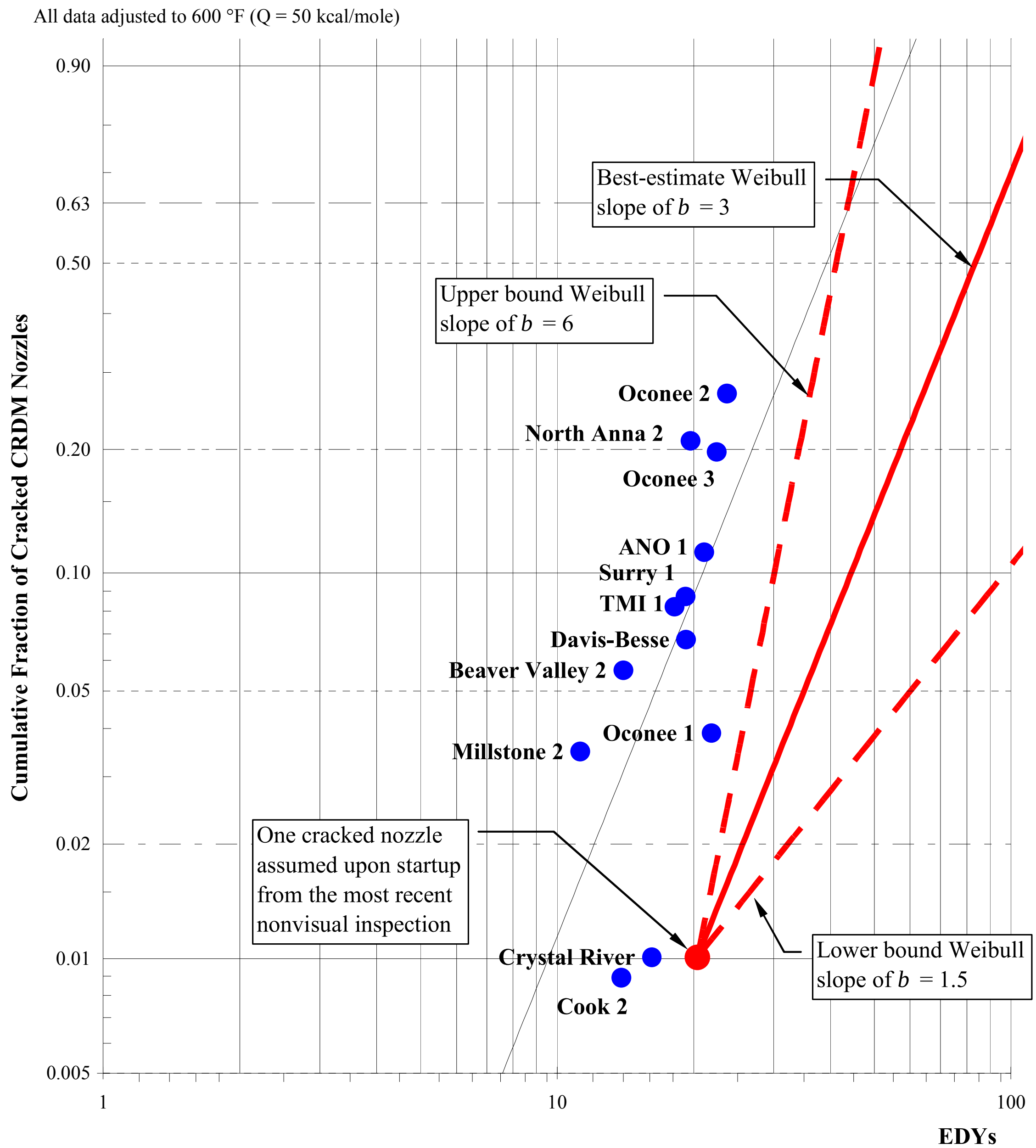



Figure 9
Weibull Statistical Model for Crack Initiation

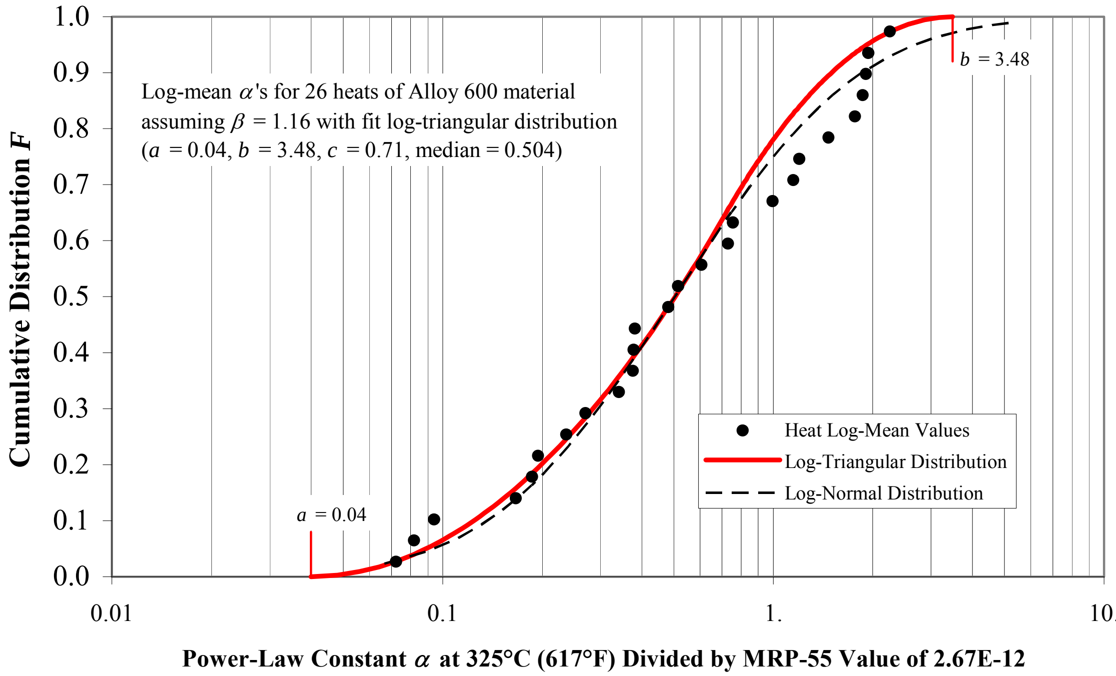



Figure 10
Crack Growth Rate for Alloy 600 Base Metal (MRP-55)

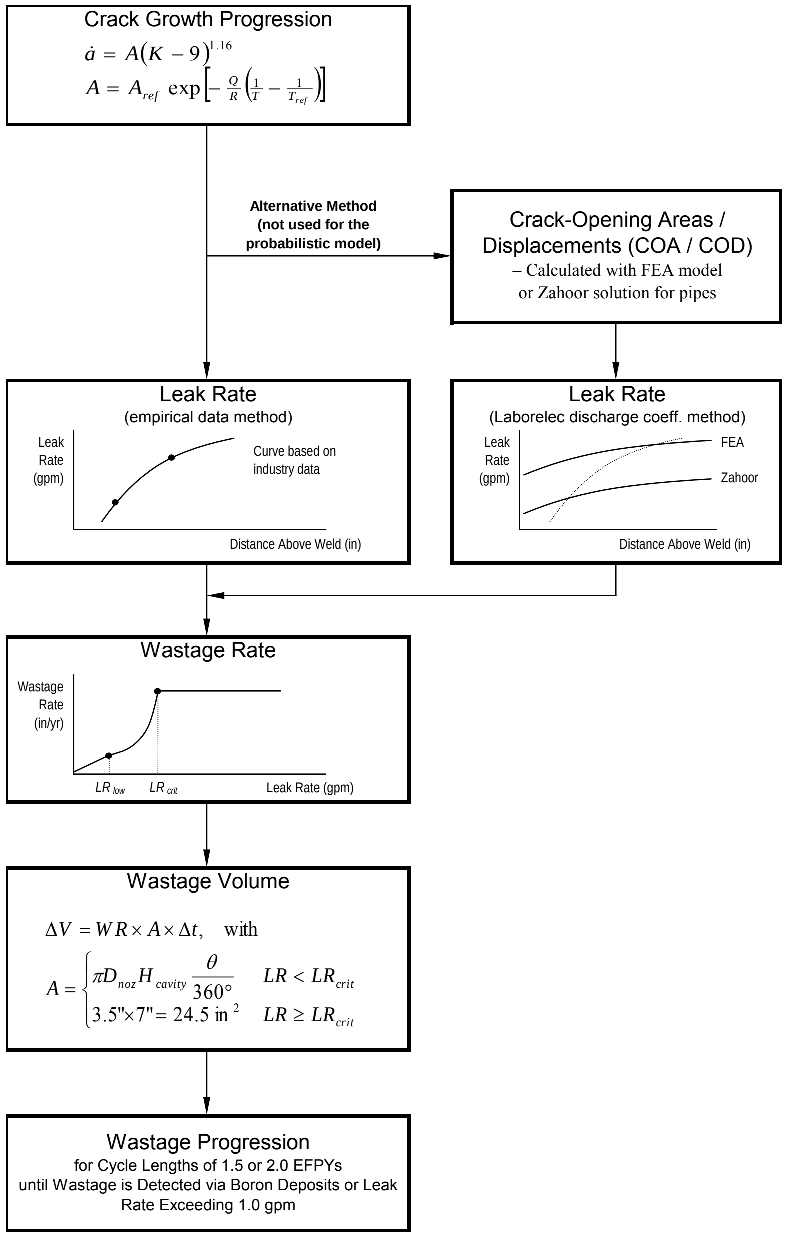



Figure 11
Probabilistic Model for Boric Acid Wastage (MRP-75)

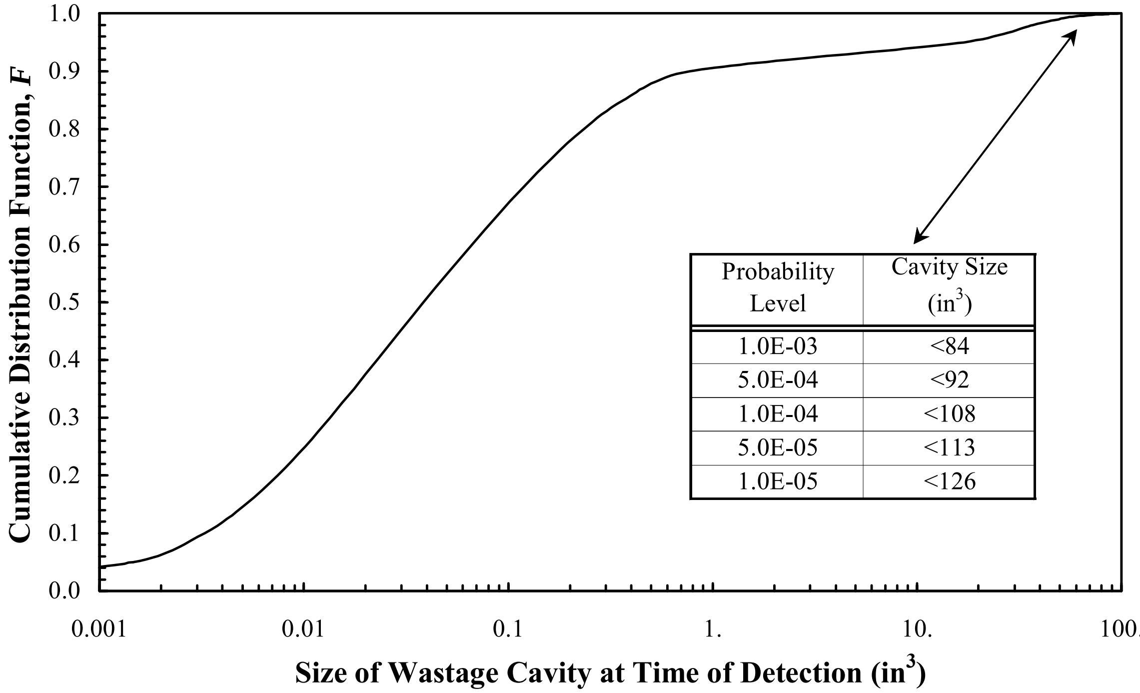



Figure 12
Typical Results of Probabilistic Wastage Analysis for a Single Leaking Nozzle (Assuming Bare Metal Visual Inspections are Performed During Each Refueling Outage)

**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**