NUCLEAR ENERGY INSTITUTE

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DIRECTOR
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April 27, 2001

Dr. Brian W. Sheron
Associate Director for Project Licensing and Technical Analysis
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop O5-E7
Washington, DC 20555-0001

SUBJECT: PWR Reactor Piping and Head Penetrations

PROJECT NUMBER: 689

Dear Dr. Sheron:

At a March 23, 2001, meeting, NEI and the Materials Reliability Project (MRP) agreed to provide the NRC staff an interim safety assessment on Alloy 82/182 pipe butt weld cracking. The following enclosed reports (proprietary and non-propriety versions) and the associated affidavit complete this action.

- EPRI Report TP1001491, Part 1, *Materials Reliability Project, Interim Alloy 600 Safety Assessment for U. S. PWR Plants, Part 1: Alloy 82/182 Pipe Butt Welds,* (Proprietary), April 2001 (10 Copies)
- EPRI Report TP1001491 NP, Part 1, *Materials Reliability Project, Interim Alloy 600 Safety Assessment for U. S. PWR Plants, Part 1: Alloy 82/182 Pipe Butt Welds,* (Non-Proprietary), April 2001 (10 Copies)
- EPRI Affidavit

The report is provided for information as part of industry's effort to better understand any generic implications of the Alloy 82/182 weld metal cracking that occurred at V. C. Summer.

EPRI Report TP1001491, Part 1 contains proprietary information as discussed in the enclosed affidavit. The NRC is requested to withhold this report from public disclosure.

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1776 I STREET, NW SUITE 400 WASHINGTON, DC 20006-3708 PHONE 202.739.8000 FAX 202.785.4019

In an April 17, 2001, letter, you requested responses to several questions concerning cracking PWR reactor pressure vessel head penetrations. An interim generic assessment and responses to your questions will be provided to the NRC by May 11.

The final version of the assessment on the generic implications of the cracking that occurred at V. C. Summer and Oconee will be provided to the NRC by the end of June.

If you have questions, please contact Kurt Cozens at 202-739-8085, koc@nei.org, or me.

Sincerely,

Kurt Cozens, for

Alexander Marion

KOC/maa
Enclosures

c: Mr. Jack R. Strosnider, U. S. Nuclear Regulatory Commission
Mr. Peter C. Wen, U. S. Nuclear Regulatory Commission

POWERING PROGRESS THROUGH
SCIENCE AND TECHNOLOGY

EPRI

April 24, 2001

Document Control Clerk
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Washington, DC 20555

Subject: "PWR Material Reliability Project, Interim Alloy 600 Safety Assessments for U.S. PWR Plants, Part 1: Alloy 82/182 Pipe Butt Welds," EPRI Report TP-1001491, April 2001

Gentlemen:

This is a request under 10CFR2.790(a)(4) that the NRC withhold from public disclosure the information identified in the enclosed affidavit consisting of EPRI owned Proprietary Information identified above (the "Report"). Copies of the Report and the affidavit in support of this request are enclosed.

EPRI desires to disclose the Report in confidence to the NRC as a means of exchanging information with the NRC staff for the purpose of supporting generic regulatory improvements related to the management of the MRP Alloy 82/182 weld integrity. EPRI welcomes any discussion with the NRC regarding the Report that the NRC desires to conduct.

The Report is for the NRC's internal use and may be used only for the purposes for which it is disclosed by EPRI. The report should not be otherwise used or disclosed to any person outside the NRC without prior written permission from EPRI.

If you have any questions about the legal aspects of this request for withholding, please do not hesitate to contact me at (650) 855-2997. Questions on the contents of the Report should be directed to Mr. Al McIlree of EPRI at (650) 855-2092.

Sincerely,

Theodore U. Marston, Ph.D.
Vice President & Chief Nuclear Officer

Enclosures

c: Licensing

EPRI

AFFIDAVIT

RE: "PWR Material Reliability Project, Interim Alloy 600 Safety Assessments for U.S. PWR Plants, Part 1: Alloy 82/183 Pipe Butt Welds," EPRI Report TP-1001491, April 2001

I, THEODORE U. MARSTON, being duly sworn, depose and state as follows:

I am a Vice President at the Electric Power Research Institute ("EPRI") and I have been specifically delegated responsibility for the report listed above that is sought under this affidavit to be withheld (the "Report") and authorized to apply for their withholding on behalf of EPRI. This affidavit is submitted to the Nuclear Regulatory Commission ("NRC") pursuant to 10 CFR 2.790 (a)(4) based on the fact that the Report consists of trade secrets of EPRI and that the NRC will receive the Report from EPRI under privilege and in confidence.

The basis for withholding such Report from the public is set forth below:

(i) The Report has been held in confidence by EPRI, its owner. All those accepting copies of the Report must agree to preserve the confidentiality of the Report.

(ii) The Report is a type customarily held in confidence by EPRI and there is a rational basis therefor. The Report is a type, which EPRI considers as a trade secret(s) and is held in confidence by EPRI because to disclose it would prevent EPRI from licensing the Report at fees, which would allow EPRI to recover its investment. If consultants and/or other businesses providing services in the electric/nuclear power industry were able to publicly obtain the Report, they would be able to use it commercially for profit and avoid spending the large amount of money that EPRI was required to spend in preparation of the Report. The rational basis that EPRI has for classifying this/these Report(s) as a trade secrets is justified by the Uniform Trade Secrets Act, which California adopted in 1984 and which has been adopted by over twenty states. The Uniform Trade Secrets Act defines a "trade secret" as follows:

> "Trade secret" means information, including a formula, pattern, compilation, program, device, method, technique, or process, that:
>
> (1) Derives independent economic value, actual or potential, from not being generally known to the public or to other persons who can obtain economic value from its disclosure or use; and
>
> (2) Is the subject of efforts that are reasonable under the circumstances to maintain its secrecy.

(iii) The Report will be transmitted to the NRC in confidence.

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(iv) The Report is not available in public sources. EPRI developed the Report only after making a determination that the Report was not available from public sources. It required a large expenditure of dollars for EPRI to develop the Report. In addition, EPRI was required to use a large amount of time of EPRI employees. The money spent, plus the value of EPRI's staff time in preparing the Report, show that the Report is highly valuable to EPRI. Finally, the Report was developed only after a long period of effort of at least several months.

(v) A public disclosure of the Report would be highly likely to cause substantial harm to EPRI's competitive position and the ability of EPRI to license the Report both domestically and internationally. The Report can only be acquired and/or duplicated by others using an equivalent investment of time and effort.

I have read the foregoing and the matters stated therein are true and correct to the best of my knowledge, information and belief. I make this affidavit under penalty of perjury under the laws of the United States of America and under the laws of the State of California.

Executed at 3412 Hillview Avenue, Palo Alto, being the premises and place of business of the Electric Power Research Institute:

April 24, 2001

Theodore U. Marston

Subscribed and sworn before me this day: April 24, 2001

Sumi Yamashita

Sumi Yamashita, Notary Public

SUMI YAMASHITA
COMM. # 1255539
NOTARY PUBLIC-CALIFORNIA
SANTA CLARA COUNTY
COMM. EXP. MARCH 31, 2004

EPRI

PWR Materials Reliability Project Interim Alloy 600 Safety Assessments for US PWR Plants (MRP-44NP)

Part 1: Alloy 82/182 Pipe Butt Welds

TP-1001491-NP, Part 1

Interim Report, April 2001

PWR Materials Reliability Project Interim Alloy 600 Safety Assessments for US PWR Plants (MRP-44NP)

Part 1: Alloy 82/182 Pipe Butt Welds

TP-1001491-NP, Part 1

Interim Report, April 2001

EPRI Project Manager
A. R. McIlree

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ORGANIZATION(S) THAT PREPARED THIS DOCUMENT

Westinghouse Electric Co.
Framatome ANP
Dominion Engineering, Inc.

CITATIONS

The following members of the Alloy 600 ITG Safety Assessment Committee were actively involved in the preparation and review of this document.

Tom Alley	Duke Power Company
David Ayres	Westinghouse Electric Company LLC
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Steve Fyfitch	Framatome ANP
Steve Hunt	Dominion Engineering, Inc.
Larry Mathews	Southern Nuclear Operating Company
Al McIlree	EPRI
Gary Moffatt	South Carolina Electric & Gas
Mike Short	Southern California Edison
Vaughn Wagoner	Carolina Power & Light
Chuck Welty	EPRI

This report describes research sponsored by EPRI.

The report is a corporate document that should be cited in the literature in the following manner:

PWR Materials Reliability Project, Interim Alloy 600 Safety Assessments for US PWR Plants (MRP-44NP): Part 1: Alloy 82/182 Pipe Butt Welds, EPRI, Palo Alto, CA,: 2001. TP-1001491, Part 1.

REPORT SUMMARY

Background

In October 2000, the V.C. Summer plant shut down for a normal refueling outage. During the plant walkdown to visually inspect for leakage, significant boric acid deposits were discovered in the vicinity of the reactor vessel Loop A outlet nozzle-to-pipe weld. The origin of the leak was found to be a small hole in the Alloy 82/182 weld between the low-alloy steel reactor vessel outlet nozzle and the stainless steel primary coolant pipe. A review of plant leakage records showed that the unidentified leak rate had been nearly constant at 0.3 gpm from all sources, well below the plant Technical Specification limit of 1.0 gpm.

Ultrasonic inspections from outside of the pipe were inconclusive. Ultrasonic inspections from inside the pipe revealed a single flaw near the top of the pipe. Destructive examination confirmed axial primary side initiated cracking confined to the Alloy 182 nozzle-to-pipe weld and to the Alloy 182 buttering on the inside surface of the low-alloy steel nozzle near the weld, and a short shallow circumferential crack in the buttering which arrested at the low-alloy steel nozzle.

Objective

The objective of this report is to provide interim assessments of primary water stress corrosion cracking (PWSCC) of Alloy 82/182 primary coolant system pipe butt welds in PWR plants. In addition, this assessment compiles all Alloy 82/182 pipe weld locations in the primary system. This list will be used to complete the final safety assessments later in 2001.

Approach

The report begins with a summary description of the V.C. Summer leakage incident and a general review of Alloy 82/182 butt welds in PWR plant primary coolant systems. This is followed by a description of the methodology used to assess the safety of these welds on a generic basis. The Safety Assessments include review of crack orientations and sizes, limiting flaw sizes, the ability to detect leaks before reaching a critical flaw size, and other margins provided by defense-in-depth.

Results

The Interim Safety Assessments show that there is a very low risk of pipe rupture as a result of PWSCC of Alloy 82/182 welds in primary system applications. There have been no previous reports of leakage from these types of pipe welds worldwide; the leakage at V.C. Summer was discovered by visual leak inspections well before there was a risk of failure; many of these welds are periodically inspected as part of 10-year ISI programs without any previous report of significant problems; analyses show that there is a high probability that leakage will be detected

by normal plant equipment and procedures long before failure; and significant additional margin is provided by defense-in-depth inherent in the plants design and operation.

EPRI Perspective

As a consequence of the hot leg nozzle weld leak at V.C. Summer in October 2000, the industry, acting through the EPRI Materials Reliability Project, has undertaken development of a comprehensive program to address this issue. Interim safety assessments have been completed to assure continued safe operation. The interim safety assessments will be used to guide development of comprehensive inspection and evaluation guidance, as well as potential repair and mitigation strategies, where warranted. This report provides the interim safety assessments for Alloy 82/182 pipe butt welds.

Keywords

Primary water stress corrosion cracking
PWSCC
Alloy 600
Alloy 82/182
RV nozzle
RCS piping
Butt welds

ABSTRACT

This Interim Safety Assessment summarizes industry effort to develop an integrated technical response to the issue of primary water stress corrosion cracking (PWSCC) of Alloy 82/182 butt welds in PWR plant primary coolant system applications. Emphasis in the interim report is on evaluating the most important Alloy 82/182 pipe butt weld application in each of the NSSS plant designs. More complete assessments of all such pipe butt welds will be completed later in 2001. The interim report addresses the background regarding leakage from an Alloy 82/182 hot leg nozzle to primary coolant pipe butt weld at V.C. Summer, axial cracks in Alloy 82/182 butt welds at Ringhals, a compilation of locations where Alloy 82/182 butt welds are used, the safety assessment methodology, results of the interim safety assessments on the most important locations, and inspection recommendations. Supporting appendices include assessments of Alloy 82/182 butt welds in Westinghouse, Combustion Engineering, and Babcock & Wilcox designed plants, and elastic-plastic stress analyses of typical butt welds including the effect of welding residual stresses.

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1 INTRODUCTION

Primary water stress corrosion cracking (PWSCC) has been detected in Alloy 82/182 butt welds between the reactor vessel hot leg nozzle and primary coolant pipes at three plants: V.C. Summer, Ringhals 3, and Ringhals 4. At Ringhals 3 and 4 the cracks were part-depth axial. At V.C. Summer there were several part-depth axial cracks, one through-wall axial crack that resulted in leakage that was detected by means of boric acid deposits discovered during a visual inspection, and a short circumferential crack in the Alloy 182 cladding that arrested when it reached the low-alloy steel nozzle base material.

During a regularly scheduled inservice inspection of Ringhals Unit 3 in the summer of 1999, two shallow axial surface flaws were discovered in the Alloy 182 outlet nozzle to safe end weld region. These flaws were all in a single weld, and were evaluated and allowed to remain in service.

During the regular inservice inspection of Ringhals Unit 4 in the summer of 2000, four axial surface flaws were found in one of the outlet nozzle to safe end weld regions. The deepest of these was approximately 28 mm (1.1 inch) deep, and all four were removed by taking contoured boat samples. No weld repairs were made.

The purpose of this report is to provide an Interim Safety Assessment addressing primary water stress corrosion cracking (PWSCC) of the most important Alloy 82/182 butt weld locations in PWR plant primary system applications. A Final Safety Assessment covering all Alloy 82/182 pipe butt welds will be submitted later in 2001.

2 BACKGROUND

The purpose of this section is to describe three incidents involving cracks in large diameter reactor vessel outlet nozzle to primary coolant pipe butt welds that occurred during 2000, and the industry response to these incidents.

2.1 Axial Cracks in Hot Leg Nozzle Welds at Ringhals 3 and 4

Part-depth axial cracks were discovered in Alloy 82/182 reactor vessel outlet nozzle to primary coolant pipe welds at Ringhals 4 during a refueling outage in the Fall of 2000. Similar part-depth axial cracks had been discovered at the same location in Ringhals 3 in 1999, but had been evaluated and allowed to remain in service.

2.2 Leak From Hot Leg Nozzle Weld at V.C. Summer

During the October 2000 refueling outage at V.C. Summer, over 200 pounds of boric acid crystals were discovered near the "A" reactor vessel nozzle to hot leg reactor coolant pipe weld. Subsequent examinations showed the leakage to be coming from a small hole near the centerline of the Alloy 82/182 weld near the top of the pipe [1].

Destructive examination after removing a short spool piece containing the weld showed that there was 1) a through-wall axial crack of about 2 inch maximum length covering most of the Alloy 82/182 weld width, 2) several other part-depth axial cracks, and 3) one shallow circumferential crack that initiated in the Alloy 182 cladding. These conditions are illustrated in Figure 2-1. As shown in this figure, the large axial crack arrested when it reached the low-alloy steel nozzle and the stainless steel pipe. The circumferential crack arrested when it reached the low-alloy steel nozzle material, giving it a maximum depth of 0.2 inches.

2.3 Industry Response

In early 2001, the EPRI-coordinated Materials Reliability Project (MRP) Alloy 600 Issue Task Group (ITG) developed a program to address cracking in Alloy 82/182 pipe butt welds. Three committees were formed to address this issue:

- An Assessment Committee to assess this and other Alloy 600 and Alloy 82/182 weld issues and develop overall inspection and evaluation guidance

- An Inspection Committee to assess the current inspection technology, ensure that improved technology is developed and qualified in a timely manner, and provide guidance to plants in performing the inspections
- A Repair/Mitigation Committee to assess the current repair and mitigation technology and ensure that improved technology is developed and qualified in a timely manner

2.4 Interim Safety Assessment

The Interim Safety Assessment provided in this report is the initial response to this issue. Focus in this effort was on quickly demonstrating that the most important Alloy 82/182 pipe welds in Westinghouse and Combustion Engineering designed plants retain an adequate margin of safety for the type of cracking that has been reported. Similar conclusions are expected for the Babcock and Wilcox designed plants. This interim evaluation will be followed up by a more complete evaluation of all Alloy 82/182 pipe butt welds later this year.

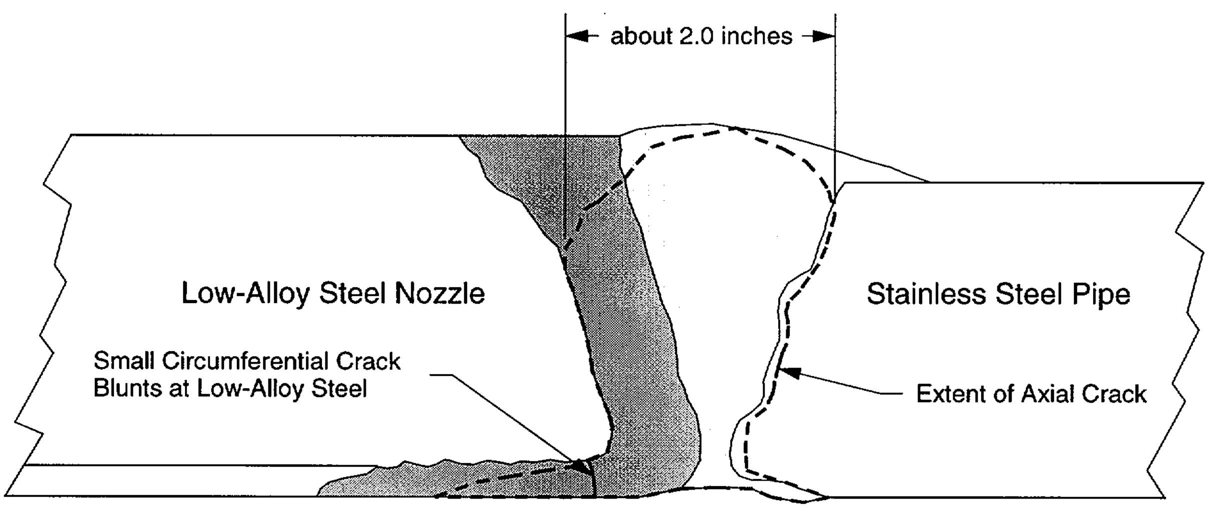

Figure 2-1
Location of Axial and Circumferential Cracks in V.C. Summer Hot Leg Nozzle to Primary Coolant Pipe Weld

3 LOCATIONS ADDRESSED IN INTERIM SAFETY ASSESSMENTS

The purpose of this section is to report the results of preliminary efforts to identify locations with Alloy 82/182 pipe butt welds, and to identify butt welds selected for the Interim Safety Assessments.

3.1 Survey of Locations with Alloy 82/182 Butt Welds

As reported in Appendices A and B, Westinghouse and Framatome ANP have completed evaluations to identify where Alloy 82/182 pipe butt welds are used in their primary coolant systems. These locations are listed in Table 3-1. More complete discussion of these locations is provided in Appendices A and B.

3.2 Important Locations Selected for Interim Safety Assessments

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Table 3-1
Important Locations Involving Alloy 82/182 Pipe Butt Welds (Preliminary)

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4 SAFETY ASSESSMENT METHODOLOGY

The purpose of this section is to briefly describe the methodology that has been implemented in the Safety Assessments in Appendices A and B. Further details will be provided in the Final Safety Assessment report to be submitted later this year.

4.1 Significant Alloy 82/182 Weld Locations

Each PWR NSSS vendor prepared a summary of all of the Alloy 82/182 butt welds in their primary system piping. The most important of these butt weld applications was then selected for initial focus. Selection criteria included factors such as size, temperature, weld design, and the presence of low-alloy and stainless steel on opposite sides of the weld.

4.2 Crack Orientation

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4.3 Tolerance for Axial and Circumferential Flaws

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4.4 Leakage Detection and Structural Margin

Calculations were performed to demonstrate that leakage will be detected prior to safety margins being exceeded. For the case of axial cracks, the calculations are of less importance since the length of the flaws is limited by the cracks arresting at the low-alloy or stainless steel pipe base material. For the case of partial-arc through-wall circumferential cracks the calculations demonstrate that leaks will be detectable by normal plant leak detection methods (1 gpm leaks) while there is still significant safety margin. As in the case for structural margin, operational experience demonstrates that there is a very low probability of 360° part-depth circumferential cracks occurring that affect structural margins, without first being detected.

4.5 Defense-in-Depth

A technical case has been made that there is significant defense-in-depth for the subject pipe butt welds. Analyses demonstrate that these cracks do not significantly increase the Core Damage Frequency. Finally, a postulated instantaneous primary pipe break is an analyzed accident per the Final Safety Analysis Report (FSAR) of every plant.

4.6 Operational Experience with Alloy 82/182 Pipe Butt Welds

There have been many inspections of primary coolant system pipe butt welds over the past 30+ years. These include non-destructive inspections from the inside of the pipes during 10-year Section XI vessel ISI inspections; visual, surface and volumetric inspections of the outside of the bimetallic welds at 10-year intervals; and visual inspections for boric acid leakage every outage as required by Generic Letter 88-05. The lack of significant findings from these inspections suggests that there are no widespread problems.

In parallel with the Interim Safety Assessments, the Inspection Committee is compiling information regarding inspections that have already been performed of the subject butt welds.

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While experience at V.C. Summer has demonstrated the potential need for improvements to the NDE technology for irregular inside surfaces such as field welds, the absence of any findings from previous inspections strongly suggests that there are no widespread problems with these joints.

4.7 Boric Acid Corrosion

Small leaks from hot pipes result in the production of dry boric acid crystals that are not corrosive to low-alloy steel materials. The absence of boric acid corrosion in this type of environment is confirmed by the fact that no significant boric acid corrosion was reported at V.C. Summer despite over 200 pounds of boric acid crystals having accumulated around the leaking weld. The V.C. Summer experience is consistent with expectations as documented in EPRI TR-104748, *Boric Acid Corrosion Guidebook* [3].

5 INTERIM SAFETY ASSESSMENT RESULTS

The Interim Safety Assessments in Appendices A and B demonstrate that plants have adequate safety margin to continue in operation. The main conclusions are as follows:

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- There has been no history of widespread problems with Alloy 82/182 pipe butt welds.
  - A number of leaks in Alloy 600 parts have been discovered by visual inspections for boric acid leakage as required by Generic Letter 88-05. In every case, the leakage was discovered by visual inspection before a significant safety risk developed. In fact, the V.C. Summer leak was discovered long before safety margins were compromised.
  - All plants in the US inspect the large diameter reactor vessel to primary coolant pipe welds during 10-year Section XI ISI programs. While experience at V.C. Summer has demonstrated the potential need for improvements to the NDE technology, the absence of findings from these inspections suggests that there are no widespread problems with these joints.
- There is no concern with boric acid corrosion as a result of the relatively low leakage rates from the PWSCC cracks and the high temperatures of the components.

6 INTERIM RECOMMENDATIONS

The MRP Inspection Committee developed short term inspection guidance for plants having spring 2001 refueling outages. This guidance was transmitted to utilities on March 1, 2001. A copy of the letter is attached.

The focus of this letter was to:

- Enhance the sensitivity of personnel performing inspections for boric acid per the requirements of Generic Letter 88-05
- Enhance the sensitivity of NDE inspection personnel to inspection capabilities, limitations and results
- Enhance the sensitivity of operations personnel to small changes in containment leak rates, and possible leak sources
- Encourage use of mockup demonstrations of NDE capabilities for any planned inspections

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REFERENCES

1. *V.C. Summer Alpha Hot Leg Evaluation and Repair*, Presented at the V.C. Summer Nuclear Station, January 18, 2001.

2. *PWSCC of Alloy 600 Materials in PWR Primary System Penetrations*, EPRI, Palo Alto, CA: 1994. TR-103696.

3. *Boric Acid Corrosion Guidebook*, EPRI, Palo Alto, CA: 1995. TR-104748.

A
INTERIM SAFETY ASSESSMENT REPORT: WESTINGHOUSE AND COMBUSTION ENGINEERING DESIGN PLANTS

The following is the Interim Safety Assessment prepared by Westinghouse for the Westinghouse and Combustion Engineering designed plants.

The Interim Safety Assessment for Westinghouse design plants and one of the Combustion Engineering design plants focuses on the large diameter and high temperature reactor vessel outlet nozzle to primary coolant pipe welds.

The Interim Safety Assessment for the remaining Combustion Engineering design plants focuses on the pressurizer surge line welds since these Combustion Engineering plants do not have Alloy 82/182 materials at the reactor vessel nozzle to primary coolant pipe joint. The pressurizer surge line welds were selected for the Interim Safety Assessment. These lines are of intermediate size but operate at the highest plant temperature and are exposed to significant load changes during plant operation.

Safety Assessments of the other Alloy 82/182 pipe butt welds in Westinghouse and Combustion Engineering designed plants will be provided in the Final Safety Assessment to be submitted later in 2001.

Citations

This report was prepared by

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, Pennsylvania 15230

Principal Investigators:
Warren H. Bamford
David Ayres
Chuck Holmes
Karl Haslinger

This report describes research sponsored by EPRI.

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INTRODUCTION

1.1 Background

In early October 2000 the V. C. Summer plant shut down for a normal refueling outage and conducted a walkdown to search for boron deposits, as is done to begin each outage. During the walkdown, significant boron deposits were discovered in the vicinity of the reactor vessel Loop A outlet nozzle to pipe weld. Insulation was removed, and leakage monitoring records were searched.

Leakage records showed a nearly constant 0.3 gpm unidentified leakage from all sources, well below the plant Technical Specification limit of 1.0 gpm. The geometry of the V.C. Summer nozzle to pipe weld is shown in Figure 1-1. Ultrasonic tests performed on the pipe from the outside surface were inconclusive, but ultrasonic tests performed from the inside surface revealed a single axial flaw in the weld near the top of the pipe.

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Figure 1-1
Geometry of V. C. Summer Nozzle to Pipe Weld Region

Supplemental eddy current testing revealed other indications, some of which were later confirmed to be flaws.

1.2 Safety Assessment Technical Approach

Inspection findings at V.C. Summer have led to questions regarding the likelihood of similar flaws in other plants, and their impact on safe operation of those plants. This report has been prepared to provide answers to those questions.

The report begins with an identification of the Alloy 82 and 182 butt weld locations in plants designed by Westinghouse and Combustion Engineering. This is followed by a chapter which describes the expected flaw orientation if flaws were to occur. The structural and leak-before-break chapters are designed to provide confidence that large flaws are required to cause a failure, and that detectable leakage would be expected in all the geometries well before failure would occur. Finally, a complementary assessment of safety will be presented in terms of risk.

The approach used in this interim report has been to cover the most limiting locations. Detailed treatments are provided of the reactor vessel hot leg nozzle safe-end locations for all Westinghouse designs, and the surge nozzle safe-end locations for the Combustion Engineering designs. Other butt weld regions will be covered in the final report, to be issued later in 2001.

2
ALLOY 82/182 BUTT WELD LOCATIONS IN WESTINGHOUSE DESIGNED PLANTS

The reactor coolant piping and fittings in Westinghouse designed reactors are austenitic stainless steel. Smaller diameter piping, such as the pressurizer surge line, spray line, safety and relief lines, and connecting lines to other systems are also austenitic stainless steel. All of the joints and connections are welded.

The major components of the system are low-alloy steel. These include the reactor vessel, pressurizer, and steam generators. The reactor coolant pump, and loop isolating valves are austenitic stainless steel. Stainless steel safe-ends were applied to the nozzles of low-alloy steel components to simplify attachment of the austenitic pipe to the vessels. Both stainless steel and Alloy 82/182 welds were used in the nozzle-to-pipe weld regions. This section will provide safe-end and nozzle-to-pipe weld information for the reactor vessel, pressurizer, and steam generators in Westinghouse design plants.

There are 48 Westinghouse designed reactors currently in operation in the United States. This section will provide information on the nozzle safe-end geometries for these 48 units, along with one unit designed by Combustion Engineering with stainless steel main loop piping. Domestic reactors that have ceased commercial operation and domestic units that have never reached commercial operation are not included. The reactor vessel safe-end configuration for two international units has been included for comparison purposes only.

2.1 Reactor Vessel

There are five reactor vessel nozzle safe end configurations on domestic Westinghouse plants. In addition to the 48 Westinghouse designed reactor vessels, this report includes one Combustion Engineering designed reactor vessel which has Alloy 82/182 in the reactor vessel safe-end welds.

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Graphical representations of the seven reactor vessel nozzle safe end configurations discussed above are contained in Figures 2-1 through 2-7.

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Figure 2-1
Type 1 Reactor Vessel Safe End: Weld Deposited Stainless Steel

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Figure 2-2
Type 1A Reactor Vessel Safe End: Weld Deposited Stainless Steel with Alloy 182 Bands

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Figure 2-3
Type 2 Reactor Vessel Safe End: Weld Deposited NiCrFe Alloy

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Figure 2-4
Type 3A Reactor Vessel Safe End: Forged Stainless Steel Safe End with NiCrFe Butter and Single V-Weld

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Figure 2-5
Type 3B Reactor Vessel Safe End: Forged Stainless Steel Safe End with NiCrFe Single V-Weld

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Figure 2-6
Type 3C Reactor Vessel Safe End: Forged Stainless Steel Safe End with Cladding with NiCrFe Single-V Weld

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Figure 2-7
Type 3D Reactor Vessel Safe End: Forged Stainless Steel Safe End NiCrFe Buttering and Double J-Groove Weld

2.2 Steam Generators

Three classes of steam generators are covered in this report:

- Original equipment steam generators supplied by Westinghouse
- Replacement steam generators supplied by Westinghouse
- Replacement steam generators supplied by others.

The majority of the replacement steam generators supplied by "Others" were supplied by Babcock and Wilcox Canada, Ltd. B&W Canada replacement steam generators are included in this report.

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Graphical representations of the four steam generator nozzle safe end configurations discussed above are contained in Figures 2-8 through 2-11.

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Figure 2-8
Type 1 Steam Generator Safe End: Forged Stainless Steel Safe End Stainless Steel Attachment Weld

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Figure 2-9
Type 2 Steam Generator Safe End: Forged Stainless Steel Safe End Alloy 82/182 Attachment Weld

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Figure 2-10
Type 3A Steam Generator Safe End: Forged Stainless Steel Safe End Alloy 52 Butter with Alloy 152 Attachment Weld

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Figure 2-11
Type 3B Steam Generator Safe End Forged Stainless Steel Safe End Alloy 52 Attachment Weld

2.3 Pressurizers

There are four nozzles on Westinghouse supplied pressurizers which potentially contain Alloy 82/182 weld material:

- Surge nozzle
- Spray nozzle
- Safety nozzles (two on 2-loop plants, three on 3-loop and 4-loop plants)
- Relief Nozzles

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Graphical representations of the two pressurizer nozzle safe end configurations discussed above are contained in Figures 2-12 through 2-13.

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Figure 2-12
Type 1 Pressurizer Safe End: Forged Stainless Steel Safe End Stainless Steel Attachment Weld

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Figure 2-13
Type 2 Pressurizer Safe End: Forged Stainless Steel Safe End Alloy 82/182 Attachment Weld

2.4 Choice of Key Location

The evaluations completed for this interim report concentrated on the reactor vessel outlet nozzle safe-end region. This region was chosen for investigation for several reasons.

- The first reason was service experience. This was the weld found to be cracked at the V.C. Summer plant, which initiated this issue. This same weld region has also been found to contain service-induced cracks at Ringhals Units 3 and 4 in Sweden, so the occurrence at V. C. Summer cannot be considered an isolated event.
- The size of the pipe also contributed to the choice of the outlet nozzle. This is the largest butt welded pipe in the Westinghouse design plants, and its failure would therefore be of the greatest significance.
- The stress corrosion cracking mechanism which is occurring is strongly affected by temperature. This is true both for crack initiation and propagation. Higher temperatures are worse from this standpoint, so the outlet nozzle would be expected to be of greater concern than the inlet nozzle.

3
ALLOY 82/182 BUTT WELD LOCATIONS IN COMBUSTION ENGINEERING DESIGNED PLANTS

Locations of bi-metallic Alloy 82/182 weld joints in the primary system components have been identified for each Combustion Engineering designed plant. In general, these welds are limited to the primary coolant piping and larger pressurizer nozzles. Transitions from carbon or low-allow steel components to stainless steel piping are accomplished by shop-welding stainless steel safe ends to the ferritic components. Nozzles and piping components are either carbon or low-alloy steel. Safe ends are fabricated from either wrought or cast stainless steel. Welds are typically configured with nickel base alloy weld deposits (i.e. buttering) on the carbon or low-alloy steel components followed by a full penetration weld with similar material to the stainless steel safe end.

Figure 3-1 shows a typical Combustion Engineering weld configuration for the surge line nozzle. Fabrication drawing notes clearly indicate that the weld between safe end and the buttered nozzle was only to be made after final post-weld heat treatment of the ferritic component. It is also to be noted that, with exception of incore instrument (ICI) to Guide Tube welds for one plant, all Alloy 82/182 welds are shop welds.

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Figure 3-1
Typical CE Surge Line Nozzle Geometry

Table 3-1 summarizes the number of affected weld locations for all operating CE plants. The locations are grouped as follows.

Table 3-1
Affected Weld Locations in CE Design Plants

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3.1 Pressurizers

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3.2 Main Coolant Loop Piping

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3.3 Auxiliary Line Welds

With exception of one plant, all CE branch line connections have Alloy 182/82 nozzle to safe end welds. The following locations comprise these branch lines. Table 3-1 shows the number of affected welds for CE Plants A through I.

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3.4 Reactor Vessel and Control Element Drive Mechanisms (CEDMs)

Each CEDM pressure housing in all later CE plants has two Alloy 82/182 welds. Because of the relatively low temperature at these locations they are not considered to be significant.

CEDM, CRDM, and ICI nozzles attached to the vessel head by J-groove welds are covered in another Interim Safety Assessment report.

3.5 Choice of Key Location

In order to rank the CE plant Alloy 82/182 weld locations with respect to criticality to plant safety, the following criteria were applied.

- Operating Temperature: pressurizer temperatures are higher than hot leg temperatures and hot leg temperatures are higher than cold leg temperatures
- Nozzle Size and Location: larger diameter pipes tend to be more important to plant safety than smaller diameter pipes
- Service Conditions: large normal operating loads, thermal stratification loads, high fatigue usage factors, and high seismic loads, all contribute to higher service conditions.

Considering these factors, the reactor coolant system surge line nozzle to safe end weld was chosen for this safety assessment as the most critical location in Combustion Engineering design plants. This will be followed in order of significance by the pressurizer spray, surge, relief and safety valve nozzles, then by the only affected portions in the main coolant loop piping, namely the reactor coolant pump suction and discharge safe end welds, and finally by the shut down cooling outlet nozzles. It is expected that all other Alloy 82/182 butt-weld types will be enveloped by these evaluations. The results of the prioritization are indicated in Table 3-2.

Table 3-2
Prioritization of Alloy 182/82 Welds in CE Plants

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4
MOST PROBABLE FLAW ORIENTATIONS

The orientation of potential flaws in Alloy 82/182 welds is of great significance from the standpoint of structural integrity, leakage rate, and safety. The service experience thus far has been that, with one exception at V.C. Summer, all flaws have been oriented axially. This section will detail this experience and discuss some engineering reasons why that behavior is expected to continue.

4.1 Service Experience

All the significant flaws found in Alloy 82/182 weld regions to date have been oriented axially. At V. C. Summer, six axial flaws were discovered in the loop A hot-leg weld that was removed and replaced.

At Ringhals 4 in Sweden, four flaws werediscovered and removed. All of these flaws were axially oriented. At Ringhals 3, two axial flaws remain in service.

Only two circumferential indications have been found to date, both at V. C. Summer. One was found to be an artifact, and the second was confirmed to be a shallow flaw with depth limited to the cladding, about 0.20 inches.

Efforts are underway to further characterize the service experience for Alloy 82/182 welds in nuclear plants, and these results will be provided in the final safety assessment report.

4.2 Stress and Crack Driving Force

A number of outlet nozzle safe-end regions have been evaluated, and in all cases the hoop stress exceeded the axial stress. This would lead to the conclusion that axial flaws would be more likely than circumferential flaws. It is obvious that the pressure stresses in the hoop direction will be double those in the axial direction, but piping loads and residual stress also need to be considered.

Perhaps the best way to compare the probability of axial flaws vs. circumferential flaws is to compare the crack driving force, or stress intensity factor, for the two orientations. Figure 4-1 shows a comparison for a flaw with length equal to six times its depth. This calculation was performed using the stress intensity factor expression of Raju and Newman [1], and the loadings included thermal, deadweight and pressure, as well as residual stresses. The residual stresses were taken from the technical basis document for the ASME Section XI pipe flaw evaluation procedures [2], and are presented here as Figure 4-2. For shallow flaws the driving force for axial and circumferential flaws is roughly equal. For flaws with depths greater than 15-20 percent of the wall thickness, the hoop stresses take over, and for all flaws greater than 50 percent of the wall thickness, the ratio of driving force from hoop stresses to that from axial stresses is greater than two to one.

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Figure 4-1
Comparison of Crack Driving Force from Hoop vs. Axial Stresses, Aspect Ratio 6:1

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Figure 4-2
Recommended Axial and Circumferential Residual Stress Distributions for Austenitic Pipe Welds [2]

Figure 4-3 shows similar results for a flaw with length twice its depth, and the hoop stress driving force is even more pronounced compared to the axial driving force. For flaws greater than 35 percent of the wall thickness, the driving force for an axial flaw is more than double that for a circumferential flaw.

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Figure 4-3
Comparison of Crack Driving Force from Hoop vs. Axial Stresses, Aspect Ratio 2:1

5 FRACTURE EVALUATION

The purpose of this section is to calculate critical flaw sizes in both the axial and circumferential direction.

5.1 Methodology for Westinghouse Design Plants

The following calculations consider all the appropriate loadings, including dead weight, thermal expansion and pressure. For critical flaw size calculations, the seismic loads were also included. For the leak rate calculations, the normal loads were used.

The loadings for both the governing normal/upset condition and the governing emergency/faulted condition were updated to include all design changes to the system. Such changes include, where appropriate, the following:

- Steam generator replacement and uprating
- Steam generator snubber elimination
- Steam generator center of gravity and weight revisions

The forces and moments for each condition were obtained from calculations previously performed by Westinghouse, or by others who have been involved with system changes as described above. The stress values were calculated using the following equations:

$$\sigma_m = \frac{F_x}{A}$$

$$\sigma_b = \frac{1}{Z}[M_x^2 + M_y^2 + M_z^2]^{0.5}$$

where

F_x = axial force component (membrane)
M_y, M_z = moment components (bending)
A = cross-section area
Z = section modulus

The section properties A and Z at the weld location were determined based on the actual pipe dimensions. The following load combinations were considered.

For circumferential flaws:

- Thermal normal – 100 percent power
- Dead weight
- Steady state pressure
- Safe shutdown earthquake (SSE)

For axial flaws: Steady state pressure

It should be noted that other piping loadings have no impact on axial flaws.

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Rapid, non-ductile failure is possible for ferritic materials at low temperatures, but is not applicable to austenitic steels. In these materials, the higher ductility leads to two possible modes of failure, plastic collapse or unstable ductile tearing. The second mechanism can occur when the applied J integral exceeds the J_{Ic} fracture toughness, and some stable tearing occurs prior to failure. If this mode of failure is dominant, the load carrying capacity is less than that predicted by the plastic collapse mechanism.

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5.2 Results for Westinghouse Fleet

The critical flaw sizes were determined for three flaw types:

Axial through-wall flaw - Critical length
Circumferential through-wall flaw - Critical length
Continuous part-depth circumferential flaw - Critical depth

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Figure 5-1
Plant A Reactor Vessel Outlet Nozzle Longitudinal Through-Wall Crack Length vs Limit Moment

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Figure 5-2
Plant A Reactor Vessel Outlet Nozzle Circumferential Through-Wall Crack Length vs Limit Moment

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Figure 5-3
Plant A Reactor Vessel Outlet Nozzle Continuous Circumferential Part-Through Flaw Depth/Wall Thickness vs Limit Moment

The results of the fracture assessment for all four plants are shown in Table 5-1, for both stainless steel and Alloy 182 materials. The results show very large critical flaw sizes for both materials, for both normal/upset and emergency/faulted conditions. Both materials have such high fracture toughness that failure is governed by the plastic limit load, which is calculated using the material yield and ultimate tensile strength. Slightly larger critical flaw sizes resulted for the Alloy 182, because the material tensile properties are slightly higher.

Table 5-1
Critical Flaw Size Results

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5.3 Methodology for CE Design Plants

The limit load approach discussed in Section 5.1 was also used for the evaluation of the bounding weld location for the CE plants. The loads necessary for determining the critical crack size in the surge line welds include pressure, dead-weight and thermal loads plus the most limiting thermal stratification transient or the SSE event. This approach leads to a conservative load condition because the limiting thermal stratification typically does not coincide with the maximum pressure. In order to obtain a bounding crack size, the loadings for all CE designed plants were reviewed. The greatest bending moment at the hot leg/ surge line nozzle resulted from a conservatively selected set of elastically computed nozzle loads for one plant. These conservative loads were used for the assessment. A more rigorous elastic plastic analysis of that surge line would achieve a significant load reduction. The bending moment in combination with the pressure is the dominant loading on a circumferential crack. The torsion and mechanical axial force do not significantly affect the limit load results.

Circumferential cracks were assumed to be located in the weld material. Axial cracks, longer than the width of the weld, were assumed to be in the safe end / pipe material.

5.4 Results for CE Fleet

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Figure 5-4
Limit Moment vs Circumferential Crack Length (ID): Surge Line Weld

The limit moment as a function of crack depth for a uniform 360 degree crack, extending from the inside of the pipe for the surge line weld geometry of Figure 3-1, is shown in Figure 5-5. The maximum normal operation + SSE moment, and the maximum normal operation + maximum thermal stratification transient moment, are also shown in Figure 5-5, so that the critical crack depth for these loadings can be determined. For this bounding case the critical crack depth is also governed by the stratified flow condition, and is 55% through-wall.

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Figure 5-5
Limit Moment vs Depth of 360 Degree Circ Crack: Surge Line Weld

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Figure 5-6
Limit Pressure vs Axial Crack Length in Pipe/Safe-End

6
LEAK-BEFORE-BREAK (LBB) ASSESSMENT

The purpose of this section is to assess leak-before-break (LBB) on the Alloy 82 and 182 welds and to show that safety is maintained.

6.1 Westinghouse Design Plants

Parametric leak rate calculations were performed using the same methodology used by Westinghouse for all LBB applications which have been reviewed and approved by the NRC. Loads were compiled for all Westinghouse designs with Alloy 182 welds, and one CE design. Primary loop piping at the reactor vessel outlet nozzle location was selected for this evaluation. Three Westinghouse plants with the highest piping faulted loads were selected for evaluation along with one CE design plant. Pipe outside diameters and the pipe wall thicknesses of 33 to 38 inches and 2.4 to 3.1 inches, respectively, at the reactor vessel outlet nozzle to pipe weld region were evaluated. Geometries include both three and four loop Westinghouse design plants. In addition, a plant with high seismic loads was also evaluated.

Loads from the plant piping stress analysis of record were used. Loads included the effects of power upratings, steam generator snubber reductions and steam generator replacements as applicable. The leak rate in gpm was calculated for various circumferential through-wall flaws (inches) using the deadweight, thermal normal 100% power, and steady state normal operating pressure. Loads are combined by the algebraic sum method.

The leak rate in gpm was also calculated for various through-wall axial flaws (inches) using the steady state normal operating pressure.

Steps involved in the leak rate predictions were to calculate the crack opening area and then to determine the leak rate using two-phase flow formulation taking into account surface roughness. Using the results of the leak rate calculations, plots were generated for the leak rate in gpm vs. flaw sizes. Plots are shown in Figures 6-1 to 6-4 for plants A, B, C, and D.

The reactor coolant system pressure boundary leak detection capability of the plants is 1 gpm. By comparing the critical flaw sizes shown in Section 5 with the 2-3 inch flaw sizes predicted to produce a 1 gpm leak rate, it can be demonstrated that there is sufficient margin to show that the intent of LBB is satisfied for all the plants evaluated.

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Figure 6-1
Leak Rate vs Flaw Size for Plant A

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Figure 6-2
Leak Rate vs Flaw Size for Plant B

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Figure 6-3
Leak Rate vs Flaw Size for Plant C

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Figure 6-4
Leak Rate vs Flaw Size for Plant D

6.2 CE Design Plants

For the CE design plants, the leakage was computed using the same methodology as for the Westinghouse designs, as discussed in Section 6.1. Inputs to the analysis include the pipe geometry, material properties, a range of crack sizes of interest and several parameters defining the resistance to flow through the crack.

For consideration of "leak before break" (LBB), leakage at 100% power "steady state" is addressed. The 100% power steady state loading for all CE design plants was reviewed and the average value of the bending moment was selected as a typical value. Leakage is computed as a function of flaw length for a through-wall circumferential crack in the weld material (Figure 6-5).

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**Figure 6-5
Leak Rate vs Circumferential Length of Through-Wall Flaws**

Leakage was computed as a function of crack length for a through-wall axial flaw in the weld material as shown in Figure 6-6

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Both axial and circumferential flaws that are sufficient to leak at a detectable rate of 1 gpm are much smaller than the critical cracks determined in paragraph 5.4, indicating that LBB for these configurations will be maintained.

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Figure 6-6
Leak Rate vs Axial Length of Through-Wall Flaws Subject to Pressure Only

7 RISK EVALUATION

As indicated in Section 4, significant PWSCC induced flaws in the reactor pressure vessel outlet nozzle welds at V. C. Summer and Ringhals have all been axially oriented and limited to the width of the weld, even for a variety of weld repair situations. While the probability of this type of flaw is relatively high, the consequence of the very small leak, which was less than 0.1 gpm, on core damage is negligible. This is because the observed leak rate is well below the plant make-up capability and is three orders of magnitude less than that for a small-break LOCA (~100 gpm per NUREG/CR-4550 [5]). The evaluation summarized in this section shows that the risk of core damage due to larger PWSCC related leaks is also expected to be insignificant.

7.1 Likelihood of a LOCA

The first part of the risk evaluation is to estimate the probability of a 100 to 5000 gpm leak rate at the reactor pressure vessel outlet nozzle weld. This leak rate range was selected because it could have a significant risk consequence in terms of potential core damage. This leak rate range is also assumed for a small-break to large-break LOCA in NUREG/CR-4550 [5]. With PWSCC as the degradation mechanism, there are two situations that must be evaluated. The low probabilities for both these evaluations are supported by the large margins in the fracture evaluations of Section 5 and the leak-before-break evaluations of Section 6.

In the first situation, the probability that an axial flaw initiates by PWSCC, and that it grows through wall and long enough to produce a large (100-5000 gpm) leak rate, is expected to be very very low (~10^{-8} in 40 years).

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In the second situation, it is postulated that a fabrication-induced flaw exists in the weld at start of life and that it grows with time to produce a large (100-5000 gpm) leak rate by end of license. The probability of this occurring is higher but still is expected to be very low ($\leq 10^{-4}$ in 40 years).

The failure probability estimates are based upon probabilistic models that have been used for piping risk-informed in-service inspection (RI-ISI) programs (see ASME Code Case N-577 [6] and Rev. 1-NP-A of WCAP-14572 [7]). The results are typical for Westinghouse designed plants, for example, Surry Unit 1 [7] and Turkey Point Unit 3 [8]. The results are for the weld between the reactor vessel outlet nozzle safe end and hot-leg piping, which is in close proximity to the Alloy 82/182 nozzle weld of concern for PWSCC

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7.2 Consequences of Large Leaks

The next part of the risk evaluation is to estimate the consequences of the large leak, should it occur. Using the probabilistic risk analysis (PRA) results that have been used at 36 Westinghouse designed plants, the average conditional core damage probability is 4×10^{-3} for a small-break LOCA (100 gpm), 7×10^{-3} for a medium-break LOCA (1500 gpm) and 1.1×10^{-2} for a large-break LOCA (5000 gpm). The conditional core damage probabilities for each type LOCA are needed because they will be individually combined with the estimated probabilities of 100, 1500 and 5000 gpm leaks at the hot-leg piping nozzle weld at 40 years.

7.3 Risk Estimate

To obtain an estimate of the risk, the higher leak probabilities at 40 years for the fabrication-induced circumferential flaw are divided by 40 years to give expected large leak frequencies of 100, 1500 and 5000 gpm leaks. Combining these frequencies with the corresponding conditional core damage probabilities gives the expected risk increase, expressed as core damage frequency, of less than 4×10^{-8}/year.

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Although the risk of core damage due to PWSCC related large leaks in the RPV outlet nozzle weld is expected to remain insignificant, there are a number of potential actions available to reduce uncertainty and manage the PWSCC degradation of the Alloy 82/182 welds.

8
SUMMARY AND CONCLUSIONS

The work presented in this report has dealt with the structural integrity of Alloy 82/182 butt welds in operating nuclear plants of Westinghouse and Combustion Engineering designs.

There are a number of locations where this weld metal is employed, and attention was focused on piping butt welds as a result of recent inspection findings at V.C. Summers and Ringhals 3 and 4. A compilation has been made of the various butt weld geometries, and this information, along with loading information, was used to identify a single key location for the interim safety assessment for the Westinghouse and CE designs. For the Westinghouse designs, this was the outlet nozzle-to-pipe weld region, and for the CE designs the surge nozzle-to-pipe weld region. These two locations were the focal points for this interim report; all the butt weld locations will be assessed for the final report.

For each of the key locations, the piping loads were compiled for all operating plants, and the limiting cases were evaluated to determine the size flaw that could lead to piping failure, and the leak rate as a function of through-wall flaw size. The results showed that there is a substantial margin between the size flaw which would lead to detectable leak (one gallon per minute) and the size flaw which could lead to failure.

A risk analysis was also carried out for the outlet nozzle-to-pipe weld region, and it was determined that the change in core damage frequency due to the potential for stress corrosion cracking was very low (estimated at 4×10^{-8} per year). This value is a factor of 25 below the threshold for significance (10^{-6}/year) identified by the NRC in Regulatory Guide 1.174. Thus the results of the risk evaluation complement the deterministic fracture evaluation in the conclusion that the safety of the operating plants will be maintained, even with this recently discovered cracking mechanism.

9 REFERENCES

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3. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1995 edition through 96 Addends.

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7. WCAP-14572, Rev. 1-NP-A, "Westinghouse Owners Group - Application of Risk-Based Methods to Piping Inservice Inspection - Topical Report," Westinghouse Energy Systems, February 1999.

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B
INTERIM SAFETY ASSESSMENT REPORT: BABCOCK & WILCOX DESIGN PLANTS

The following is the Interim Safety Assessment prepared by Framatome ANP for Babcock & Wilcox designed plants. This Interim Safety Assessment addresses the reactor vessel inlet and outlet nozzle to primary coolant pipe welds. Safety Assessments of the other Alloy 82/182 pipe butt welds will be provided in the Final Safety Assessment to be submitted later in 2001.

Citations

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This report describes research sponsored by EPRI.

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1
PURPOSE

The purpose of this report is to assess the safety of reactor vessel inlet and outlet nozzle to primary coolant system pipe welds in B&W-design plants as it relates to primary water stress corrosion cracking (PWSCC). This assessment applies to the B&W designed plants listed in Table 1-1.

A secondary purpose is to compile information on all Alloy 82/182 pipe weld locations in B&WOG plants. This list will be used to prepare the final safety assessment covering all Alloy 82/182 pipe weld locations later this year.

Table 1-1
List of B&W Design Plants Evaluated in Safety Assessment

Plant*	Owner
Davis-Besse (D-B)	First Energy Nuclear Operating Company
Oconee Nuclear Station Units 1, 2, and 3 (ONS-1, -2, and -3)	Duke Energy Corporation
Arkansas Nuclear One Unit 1 (ANO-1)	Entergy Operations, Incorporated
Crystal River Unit 3 (CR-3)	Florida Power Corporation
Three Mile Island Unit 1 (TMI-1)	Exelon Corporation

* This group will subsequently be identified as the "B&WOG plants".

2 BACKGROUND

Leakage was identified at the reactor vessel outlet nozzle-to-pipe weld in the "A" hot leg loop of the V.C. Summer Nuclear Station in October 2000 [1,2]. As shown in Figure 2-1, this weld is in a 29-inch inside diameter pipe and is located approximately 36 inches from the reactor vessel wall. The pipe wall and weld thickness are 2.33 inches minimum.

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Figure 2-1
V.C. Summer "A" Hot Leg Nozzle to Pipe Weld

An axial flaw, approximately 2.7 inches long and located approximately 7° clockwise from the top of the pipe (as viewed from inside the reactor vessel), was identified on the inside surface of the pipe at V.C. Summer. Further examinations also identified a short circumferential flaw, approximately 1.5 inches long, intersecting the axial flaw as shown in Figure 2-2. This document provides a safety assessment to address the potential for a similar concern at the B&WOG plants.

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Figure 2-2
Location of Inside Surface Cracking at V.C. Summer "A" Hot Leg Nozzle

3
B&W REACTOR VESSEL NOZZLE TO PRIMARY COOLANT PIPE WELD DESIGN

Fabrication drawings for the B&W-design reactor vessel inlet and outlet nozzles and the attached hot and cold leg piping were reviewed to identify the materials utilized and the weld joint configuration (see Figure 3-1). Quality Assurance Data packages for reactor coolant piping and microfilm rolls containing shop and site records were reviewed.

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Figure 3-1
Typical Reactor Vessel Nozzle-to-Pipe Weld Configuration in B&WOG Plants

The hot and cold leg primary coolant pipes in B&W-design plants were manufactured from carbon steel material clad internally with austenitic stainless steel. Clad carbon steel was used instead of wrought stainless steel to take advantage of the lower coefficient of thermal expansion of carbon steel, thereby minimizing the forces and moments on the reactor vessel and once-through steam generator nozzles [3].

Four clad low-alloy steel inlet nozzles connect the reactor vessel to the reactor coolant system cold leg piping. Two clad low-alloy steel outlet nozzles connect the reactor vessel to the reactor coolant system hot leg piping. Both the inlet and outlet nozzles were buttered with carbon steel at their terminal ends to facilitate field attachment of the piping [4]. Buttering of the inlet and outlet nozzles was performed before final post-weld heat treatment (PWHT).

The carbon steel field welds used to join the carbon steel pipe and nozzle materials were back clad with austenitic stainless steel typically using the shielded metal arc welding (SMAW) process. The welds were subjected to final PWHT at 1125° F $\pm$ 25°F for 1 hour per inch of weld thickness [3].

Table 3-1 is a comparison of the materials used at V.C. Summer and the B&WOG plants [3,5].

Table 3-1
Comparison of Reactor Vessel Nozzle-to-Pipe Weld Materials in V.C. Summer and B&WOG Plants

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In summary, the B&WOG plants do not have Alloy 82/182 field welds at the subject location and are not susceptible to the type of PWSCC experienced at V.C. Summer.

4
LOCATIONS OF ALLOY 82/182 BUTT WELDS IN B&WOG PLANTS

In 1996, the B&WOG performed a record search to determine the locations where Alloy 82 and Alloy 182 weld materials were utilized [6]. The records search included review of available B&WOG plant fabrication and construction data from the B&W Mt. Vernon Works and B&W Barberton facilities. Only the original construction and fabrication records were reviewed; no changes, modifications, or repairs following initial plant startup were included. Information obtained from the record search consists of the type of weld, weld metal heat number, and the component information. In addition, material test reports for the Alloy 82 and Alloy 182 weld consumables were obtained, when available.

As noted in Section 2, the materials used at V.C. Summer for the reactor vessel nozzle-to-pipe design include a low alloy steel nozzle with Alloy 182 buttering welded to a stainless steel pipe with an Alloy 82/182 full penetration field weld. At B&WOG plants there are several full penetration welds that utilize a similar combination of materials. These locations are listed in the following tables.

In addition to the Alloy 82/182 weld locations listed in the following tables, each of the B&WOG plants has numerous other full penetration, partial penetration, and fillet-type weld locations that utilize Alloy 82/182 weld metal. An example is the RCS drain nozzle, which in some plants is a carbon steel nozzle clad with stainless steel connected to an Alloy 600 safe end with an Alloy 182 full penetration weld. Evaluations are underway to assess the safety significance of all the Alloy 82/182 weld locations at the B&WOG plants relative to the V.C. Summer incident. The results of these evaluations will be included in the Final Safety Assessment to be completed later this year.

Table 4-1
Davis-Besse: Locations of Alloy 82/182 Full Penetration Welds Joining Carbon (or Low Alloy) Steel to Stainless Steel

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Table 4-2
Oconee Unit 1: Locations of Alloy 82/182 Full Penetration Welds Joining Carbon (or Low Alloy) Steel to Stainless Steel

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Table 4-3
Oconee Unit 2: Locations of Alloy 82/182 Full Penetration Welds Joining Carbon (or Low Alloy) Steel to Stainless Steel

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Table 4-4
Oconee Unit 3: Locations of Alloy 82/182 Full Penetration Welds Joining Carbon (or Low Alloy) Steel to Stainless Steel

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Table 4-7
Three Mile Island Unit 1: Locations of Alloy 82/182 Full Penetration Welds Joining Carbon (or Low Alloy) Steel to Stainless Steel

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5
SUMMARY AND CONCLUSIONS

As a result of the previously described activities performed by the B&WOG, the following conclusions have been reached regarding degradation of reactor vessel nozzle-to-pipe attachment welds at B&WOG plants:

- Through-wall leakage of primary coolant will not occur at the reactor vessel nozzle-to-pipe locations in B&WOG plants. This is because B&WOG plants have different materials from those used at V.C. Summer and these materials are not susceptible to PWSCC. B&WOG plants also involve different fabrication and field installation procedures, including post weld heat treatment of both the shop and field welds.
- The B&WOG plants comply with 10CFR50.55a and meet the intent of General Design Criterion 14 of Appendix A of 10CFR50.
- Augmented inspections of reactor vessel nozzle-to-pipe welds for PWSCC degradation are not necessary from a safety perspective. However, inspections of these weld locations are performed in accordance with ASME Code Section XI ISI requirements.
- The B&WOG will continue to evaluate and share B&WOG plant inspection data on Alloy 82/182 nozzles and welds and participate in agreed upon joint Owners Group activities with the U.S. nuclear industry on this issue.
- The B&WOG will continue to monitor this issue.

Additional evaluations are underway to assess the safety significance of other Alloy 82/182 weld locations at B&WOG plants. The results of these analyses will be provided in the Final Safety Assessment to be completed later this year.

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C
INTERIM SAFETY ASSESSMENT REPORT: WELDING RESIDUAL AND OPERATING STRESSES

The following is an interim report covering finite element stress analysis of single-V and double-V butt welds similar to those which developed cracks at V.C. Summer and Ringhals 4. The purpose of this analysis work is to demonstrate that the hoop stresses dominate axial stresses in these welds such that most cracking should be axial.

More complete results will be provided in the Final Safety Assessments to be submitted later in 2001.

Citations

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This report describes research sponsored by EPRI.

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1
PURPOSE

The purpose of this document is to describe elastic-plastic finite element analyses performed to determine the welding residual and operating condition stresses in idealized single-V and double-V weld joints similar to those at V.C. Summer and Ringhals 4, including the effect of reported weld repairs at V.C. Summer.

Analyses are also included for the case of an idealized bimetallic pipe joint with low-alloy steel on one side and stainless steel on the other side. This elastic analysis case is included for information purposes to demonstrate the role of differential thermal expansion between low-alloy and stainless steel materials on stresses in bimetallic joints.

2
GEOMETRIES AND MATERIALS ANALYZED

Three geometries were analyzed.

2.1 Idealized Bimetallic Pipe Weld

An idealized bimetallic pipe weld with low-alloy steel material on one side and stainless steel material on the other was analyzed to demonstrate the effects of differential thermal expansion on stresses near the weld. The inside and outside pipe diameters and the materials are the same as for the single-V pipe welds.

- SA 508 Class 2 low-alloy steel
- Type 304 stainless steel pipe

2.2 Single-V Hot Leg Nozzle to Pipe Weld

The second finite element model, shown in Figure 2-1, represents a single-V weld similar to that at V.C. Summer. The pipe inside diameter and wall thickness are 29.00" and 2.33", respectively. The cladding is 0.193" thick.

Materials assumed were:

- SA 508 Class 2 low-alloy steel
- Type 304 stainless steel pipe
- Alloy 82/182 weld and buttering
- Type 309 stainless steel cladding

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2.3 Double-V Hot Leg Nozzle to Pipe Weld

The third model, shown in Figure 2-2, represents a double-V weld similar to that at Ringhals 4. In order to compare analysis results for the two designs on an equal basis, the pipe inside and outside diameters were assumed to be the same as for the single-V geometry.

The materials for the double-V nozzle are assumed to be the same as for the idealized bimetallic pipe and single-V nozzle.

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Figure 2-1a - Dimensions

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Figure 2-1b - Overall Model Geometry and Materials

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Figure 2-1c – Weld Area Geometry

Figure 2-1
Nozzle and Weld Geometry – Single-V Groove Weld

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Figure 2-2a Dimensions

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Figure 2-2b Overall Model Geometry and Materials

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Figure 2-2c Weld Area Geometry

Figure 2-2
Nozzle and Weld Geometry – Double-V Groove Weld

3
ANALYSIS METHOD

The following is a brief description of the analysis method used for the three models.

3.1 Finite Element Program

Finite element analyses were performed using ANSYS Revision 5.6.

3.2 Geometric Models

Nozzles were analyzed using axisymmetric models. The model of the idealized bimetallic pipe joint consists of lengths of carbon steel and stainless steel pipe joined at the centerline of the model (no weld metal is included). The single-V and double-V nozzle-to-pipe weld models consist of a short length of the carbon steel nozzle, stainless steel cladding on the inside of the nozzle, Alloy 82/182 buttering, stainless steel pipe, and Alloy 82/182 welds.

All elements are four-node quadrilateral elements.

Alloy 82/182 weld passes are simulated by rings of weld metal that are deposited sequentially in layers two elements thick across the weld surface.

3.3 Methods

Analyses were performed using typical room temperature material yield strengths

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All materials were modeled as elastic-perfectly plastic.

Analyses did not include the effects of any cold working due to machining or grinding of the material surfaces before or after welding. These cold worked layers are relatively thin and, while they can have a significant effect on the time to crack initiation, they have no effect on crack growth.

The nozzle ends of the models (i.e., the left edge of the model as shown in Figures 2-1b and 2-2b) were fixed in the axial direction. The lines of nodes at which the pipe was terminated (i.e., the right edge of the models as shown in Figures 2-1b and 2-2b) were coupled in the axial direction (constrained to have the same axial displacement) to simulate continuation of the pipe beyond the model boundary.

3.4 Loading Steps

The models were loaded in a series of steps as follows:

3.4.1 Welding

The welding process was simulated by combined thermal and structural analyses. A transient thermal analysis was used to generate nodal temperature distributions throughout the welding process. These nodal temperatures were then used as inputs to the structural analysis which calculated resultant thermally-induced residual stresses as the welds cooled and gained strength. The sequence of thermal analysis followed by structural analysis was duplicated for each simulated weld pass.

3.4.2 Weld Repairs

Weld repairs were simulated by deactivating elements associated with previously welded material and reapplying new weld metal in its place. Deactivation of elements essentially results in elimination of the conductive capacity or stiffness of the deactivated element in heat transfer and structural analyses, respectively.

3.4.3 Hydrostatic Testing

Components were hydrostatically tested to approximately 3,125 psi after installation. This step was included in the analysis since applied hydrostatic pressure further yields any material stressed to near yield by welding and, therefore, results in a reduction of the peak residual tensile stresses after the hydrostatic test pressure is released. In this manner, the hydrostatic testing represents a form of "mechanical stress improvement" in areas of high stress. Aside from applying pressure to all wetted inside surfaces, an axial tensile stress was applied to the end of the pipe equal to the longitudinal pressure stress in the pipe wall.

3.4.4 Operating Conditions Superimposed on Welding Residual Stresses

Operating conditions were simulated by pressurizing the inside of the model to 2,250 psi and heating all of the material uniformly to an assumed operating temperature of 615°F. The constant operating temperature produces thermal stresses due to the difference in coefficient of thermal expansion between the carbon steel nozzle, the Alloy 82/182 weld and buttering, and the stainless steel pipe. The pressure and thermal conditions were added to the model which had already been subjected to welding (and weld repairs) and hydrostatic testing.

4
ANALYSIS RESULTS

The following is a brief summary of the analysis results.

4.1 Results for Idealized Bimetallic Pipe Weld

The idealized pipe weld was analyzed only for the case of operating temperature and pressure to demonstrate the stresses created by the bimetallic material combination. Analyses for this case did not include welding residual stresses.

Hoop (Sz) and axial (Sy) stresses near the idealized bimetallic joint are shown in Figure 4-1. The stress distribution is consistent with the higher coefficient of thermal expansion of the stainless steel pipe relative to that of the low-alloy steel pipe (Note: the interface between the nozzle and pipe material is located at the center of each figure; the lower edge of each image is the pressurized inside surface of the pipe.)

4.2 Results for Single-V Nozzle-to-Pipe Weld

Due to some uncertainty in the actual weld repairs performed at V.C. Summer, the single-V pipe weld was analyzed for three cases:

- The joint welded from the inside to the outside without repairs
- The joint weld repaired on the inside first followed by the outside
- The joint weld repaired on the outside first followed by the inside

Figures 4-2 and 4-3, show the hoop and axial stresses respectively, for the three cases. These results show that:

- Hoop stresses (which tend to initiate and drive axial cracks) are generally higher than axial stresses (which tend to initiate and drive circumferential cracks) for all three welding sequences. This suggests that most cracking of these welds should be axial as was experienced at V.C. Summer, Ringhals 3 and 4.
- Hoop stresses on the inside surface are highest for the case where weld repairs are made last on the inside surface. For the as-designed case, or for cases where the inside surface of the weld is repaired first, additional weld passes added to the outside of the nozzle tend to apply a radially compressive load on the earlier welds, thereby reducing the operating condition hoop stress on the inside of the weld.

- There is a small location under the buttering for the case of the V.C. Summer nozzle where the calculated axial stress exceeds the hoop stress (see Figure 4-3.c). This is the location where a small circumferential crack was discovered at V.C. Summer.

Key stresses for the case of the single-V weld are summarized in Table 4-1.

4.3 Results for Double-V Nozzle-to-Pipe Weld

The double-V pipe weld was analyzed for two cases:

- The inside weld completed first followed by the outside weld as is expected to be standard practice
- The outside weld completed first followed by the inside weld

Figures 4-4 and 4-5, show the hoop and axial stresses respectively, for the two cases. These results show the same patterns as for the single-V weld. Hoop stresses tend to exceed axial stresses by a significant margin and hoop stresses are highest on the inside surface for the case where the inside of the weld is completed last.

Key stresses for the case of the double-V weld are summarized in Table 4-1.

Table 4-1
Summary of Hot Leg Nozzle to Pipe Weld Finite Element Analysis Results

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Figure 4-1
Idealized Bimetallic Weld – Operating Pressure and Temperature Only

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Figure 4-2
Single V-Weld – Hoop Stress (Weld Residual + Hydro Test + Operating Conditions)

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Figure 4-3
Single V-Weld – Axial Stress (Weld Residual + Hydro Test + Operating Conditions)

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Figure 4-4
Double V-Weld – Hoop Stress (Weld Residual + Hydro Test + Operating Conditions)

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Figure 4-5
Double V-Weld – Axial Stress (Weld Residual + Hydro Test + Operating Conditions)

5 DISCUSSION OF ANALYSIS RESULTS

A number of conclusions can be drawn from the analyses of the single-V and double-V weld cases performed in this study.

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