Materials Reliability Program: Leak-Before-Break Evaluation for PWR Alloy 82/182 Welds (MRP-140)

WARNING:
Please read the Export Control Agreement on the back cover.

Technical Report

Materials Reliability Program: Leak-Before-Break Evaluation for PWR Alloy 82/182 Welds (MRP-140)

1011808

Final Report, November 2005

EPRI Project Manager
C. King

ELECTRIC POWER RESEARCH INSTITUTE
3420 Hillview Avenue, Palo Alto, California 94304-1395 • PO Box 10412, Palo Alto, California 94303-0813 • USA
800.313.3774 • 650.855.2121 • askepri@epri.com • www.epri.com

DISCLAIMER OF WARRANTIES AND LIMITATION OF LIABILITIES

ORGANIZATION(S) THAT PREPARED THIS DOCUMENT

Structural Integrity Associates, Inc.

Westinghouse Electric Co.

AREVA

ORDERING INFORMATION

Requests for copies of this report should be directed to EPRI Orders and Conferences, 1355 Willow Way, Suite 278, Concord, CA 94520, (800) 313-3774, press 2 or internally x5379, (925) 609-9169, (925) 609-1310 (fax).

CITATIONS

This report was prepared by

Structural Integrity Associates, Inc.
3315 Almaden Expressway, Suite 24
San Jose, CA 95118

Principal Investigators
N. Cofie
A. Deardorff

Westinghouse Electric Co.
PO Box 158
Madison, PA 15663

Principal Investigators
W. Bamford
D. Bhowmick

AREVA
3315 Old Forest Rd.
Lynchburg, VA 24501

Principal Investigators
A. Nana
B. Grambau
H. Gunawardane

This report describes research sponsored by the Electric Power Research Institute (EPRI).

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

Materials Reliability Program: Leak-Before-Break Evaluation for PWR Alloy 82/182 Welds (MRP-140). EPRI, Palo Alto, CA: 2005. 1011808.

REPORT SUMMARY

General Design Criterion 4 (GDC-4) in Appendix A of Part 50 of Title 10 of the Code of Federal Regulations (10CFR50) requires postulation of sudden double-ended guillotine breaks (DEGBs) in light water reactor high-energy lines. This requirement resulted in the installation of massive protective devices such as pipe whip restraints and jet impingement shields in the vicinity of high-energy lines in some plants. These devices were extremely expensive to design and install and, in most cases, interfered with the inspection of the adjacent piping weld. Advancements in fracture mechanics technology caused the Nuclear Regulatory Commission (NRC) to amend GDC-4 in 1987 to permit use of leak-before-break (LBB) evaluation as an alternative to the DEGB postulation. Recent events, however, have caused the industry to question use of LBB at Alloy 82/182 locations.

Background

LBB evaluations demonstrate through deterministic fracture mechanics analyses that through-wall flaws in high-energy piping systems will result in leaks that can be detected by plant leak detection systems long before the flaws grow to unstable critical flaw sizes that can result in DEGB. The general guidance for performing LBB evaluations is provided in NUREG-1061, Vol. 3, and NUREG-0800, draft SRP 3.6.3.

One of the limitations imposed by NRC in NUREG-1061, Vol. 3, and draft SRP 3.6.3 is that locations on piping systems that are susceptible to stress corrosion cracking (SCC) do not qualify for LBB. Before primary water stress corrosion cracking (PWSCC) incidents occurred in Alloy 82/182 butt welds at a few domestic and foreign plants, pressurized water reactor (PWR) butt welds were generally believed to be free of SCC problems since PWRs operate in a low-oxygen environment. As such, LBB has been applied to various high-energy lines in PWRs as an alternate way of addressing the assumption of DEGB per GDC-4 of 10CFR50. Following the recent PWSCC events, the application of LBB at Alloy 82/182 locations has been questioned.

Objectives

To demonstrate that previous LBB submittals reviewed and approved by the NRC, as well as the present submittals for LBB application (currently under review by the NRC), still have adequate margins in the presence of PWSCC at Alloy 82/182 locations.

Approach

To identify where LBB had been applied to Alloy 82/182 welds, the project team completed a survey to determine the systems where LBB had been applied, or for which there is a current or pending LBB application, for the entire PWR fleet. Next, these systems were reviewed to determine if they contained Alloy 82/182 welds. This review was followed by determining through-wall critical flaw lengths and leakage flaw sizes at critical locations. The effect of PWSCC crack morphology on calculating leakage flaw sizes was evaluated. The margins between the critical flaw sizes and the leakage flaw sizes were then determined and compared to the guidelines in NUREG-1061, Vol. 3, and draft SRP 3.6.3. The team evaluated the leakage detection systems currently used at the plants. They concluded that there has been increased leak detection sensitivity by most plants such that they are trending leakage and taking action at leakage levels below the 1 gpm limit (which is normally assumed for LBB evaluation).

Results

The evaluation indicated that only the nozzle-to-safe end welds in five types of piping systems have Alloy 82/182 weldments that have current or pending LBB applications. These are the reactor main coolant loop (which includes piping connections to the reactor vessel, pumps, and steam generators), the surge line, the safety injection line, shutdown cooling, and core flood line.

The LBB evaluation for these lines considering PWSCC morphology indicated that the margins in NUREG-1061, Vol. 3, and draft SRP 3.6.3 can be met for relatively larger lines even after leak rate reduction resulting from PWSCC crack morphology is considered. For smaller diameter lines (for example, surge lines), the margin is reduced. An evaluation of leak detection capabilities indicated that most plants take actions long before the assumed Technical-Specification-required leakage of 1 gpm in one hour. This is consistent with the leakage detection capability of 1 gpm used in the LBB submittals. For PWSCC, there is ample period (greater than one year) for the leakage size flaw to grow to a critical flaw length, allowing plants to take appropriate actions long before pipe rupture.

EPRI Perspective

The results of this study have shown that there is no immediate need to apply LBB to Alloy 82/182 locations. As such, there is no need to reinstall massive protective devices, such as pipe whip restraints or jet impingement shields, at these locations. Since inspection is an important part of managing PWSCC, the absence of these devices will facilitate weld inspection.

Keywords

Alloy 82
Alloy 182
Critical flaw size
LBB
Leak-before-break
Leakage flaw size
Leak detection

CONTENTS

LIST OF FIGURES

LIST OF TABLES

1 INTRODUCTION

1.1 Background on Technical and Licensing Aspects of LBB

Historically, the assumption of a double-ended sudden pipe break for design purposes goes back to the original design of the Shippingport reactor, where the containment was designed for the sudden break in the reactor coolant system piping. In the sixties, the sudden break was used for design of emergency core cooling systems (ECCS). In the seventies, the emphasis shifted to considering the dynamic effects of the break with respect to reactor asymmetric loading, jet impingement, and pipe-whip [1-1]. This resulted in the implementation of General Design Criterion 4 (GDC-4) "Environmental and Dynamic Effects Design Bases" in Appendix A of Part 50 of Title 10 of the Code of Federal Regulations (10CFR50) which required postulation of sudden double-ended guillotine breaks (DEGB) in the high-energy lines of light water reactors. This requirement resulted in the installation of massive protective devices such as pipe whip restraints and jet impingement shields in the vicinity of high-energy lines in some plants. The cost of these protective devices was extremely high, and in many cases also made inspection of the adjacent piping welds very difficult, if not impossible. In some cases, these devices had to be removed and reinstalled during the inspections, leading to increased man-rem exposure.

The issue of asymmetric blowdown loads on PWR primary systems following a postulated DEGB in such systems was identified by the Nuclear Regulatory Commission (NRC) staff in 1975 as an Unresolved Safety Issue (USI) A-2. The resolution of this issue would have required installation of massive pipe whip restraints in some PWRs. Instead, the industry and the NRC staff resolved this issue by adoption of the leak-before-break (LBB) concept utilizing fracture mechanics techniques. Following the successful resolution of the USI A-2 issue and the work of the Pipe Break Task Group published in NUREG-1061, Vol. 3 [1-2] in 1984, the NRC accepted the concept of LBB for large diameter high energy piping as an alternate to the DEGB postulation. In 1986, NRC amended GDC-4 to permit the use of LBB analysis for primary loop piping in PWRs. While the LBB concept permitted elimination of local effects of the DEGB, the global effects of the DEGB were retained [1-3]. The containment must be designed to accommodate the effects of pressure, temperature, and flooding due to the break of up to the largest pipe in the reactor coolant system (RCS). The ECCS must be designed to accommodate the break of the largest line. Equipment inside the containment must be qualified to withstand the effects of pressure, temperature, flooding, humidity, chemical environment and radiation resulting from pipe ruptures.

The use of LBB as an alternative to the DEGB postulation required a relief request or exemption from the NRC until 1987 when a final rule was published amending GDC-4 to permit the use of LBB analyses in all qualified high-energy piping systems [1-4]. At the same time, a draft Standard Review Plan (draft SRP 3.6.3) [1-5] entitled "Leak-Before-Break Evaluation

Procedures" which provided review guidance for the implementation of the revised GDC-4 was published for comments in August 1987. This draft SRP 3.6.3 and NUREG-1061, Vol. 3 provide the guidelines for the NRC to review LBB submittals. The technical requirements in these two documents are essentially the same and hence, they are referenced interchangeably throughout this report.

The purpose of an LBB evaluation is to demonstrate through deterministic fracture mechanics analyses that through-wall flaws in high energy piping systems will result in leaks that can be detected by the plant leak detection system before the flaws grow to critical through-wall flaw sizes that can result in a DEGB. The demonstration of the LBB concept for a particular high-energy piping system permits the removal or non-installation of the massive protective devices on that system. In addition, other dynamic effects such as those due to jet impingement and reactor internals loadings need not be included as a condition of design.

General technical guidance for LBB evaluation is provided in Section 5 of NUREG-1061, Vol. 3. In particular, subsection 5.2 provides a detailed step-by-step approach for performing LBB evaluations. A summary of the key technical requirements is provided below.

- Address the limitations imposed in Section 5.1 of NUREG-1061, Vol. 3 on the use of LBB for high-energy piping. LBB is not considered applicable to systems if operating experience indicates particular susceptibility to failure from the effects of corrosion (e.g., intergranular stress corrosion cracking (IGSCC) or flow assisted corrosion (FAC)), water hammer, or low and high cycle (i.e., thermal, mechanical) fatigue.
- Determine loads and stresses. The loads to be used in LBB evaluations include normal operating loads (pressure, dead weight and thermal) for leakage determination and normal plus seismic SSE loads for critical flaw determination.
- Determine material properties to be used in the LBB evaluation. Key material properties include stress-strain curve parameters and material toughness. Specific requirements for determination of the material properties are provided in NUREG-1061, Vol. 3.
- Determine critical and leakage flaw sizes based on fracture mechanics using either elastic-plastic J-integral/tearing modulus approach or net section collapse (limit load) analyses. The critical flaw size is that flaw size at which failure or instability occurs based on the applied loading. The leakage flaw is the flaw size that will result in a particular leakage (usually 1 gpm with a margin of 10 applied per NUREG-1061, Vol. 3).
- Determine the margin between the critical flaw size and the leakage flaw size. NUREG-1061, Vol. 3 recommends a margin of two, except that the critical flaw size based on $\sqrt{2}$ on loads must also be shown to exceed the leakage flaw size. The factor of $\sqrt{2}$ can be reduced to 1.0 if faulted loads are combined by absolute summation method [1-5].
- Perform crack growth evaluation of sub-critical flaws to show that they will not grow to critical flaw size between inspections.

LBB for the piping system is demonstrated if adequate margin exists between the leakage flaw size and the critical flaw size and if there is adequate inspection interval to supplement the LBB evaluation.

1.2 Application of LBB to Alloy 82/182 Components

One of the limitations imposed by the NRC in NUREG-1061, Vol. 3 and the draft SRP 3.6.3 is that locations on piping systems that are susceptible to corrosion mechanisms such as stress corrosion cracking (SCC) do not qualify for application of LBB. However, the NRC indicated in the draft SRP 3.6.3 that non-conforming piping that has been treated by two mitigating methods may qualify for LBB if the piping contains no flaws larger than those permitted by ASME Code Section XI without repair. Alternatively, LBB is acceptable if two mitigation methods were applied within the first two years of service. No guidance was provided in these documents as to what actions to take if cracking were to be discovered after LBB was accepted by the NRC.

Before the cracking incidents in the Alloy 82/182 weld metal in CRDM penetration welds and in the butt welds at V. C. Summer and a number of other plants, PWR butt welds were generally believed to be free of SCC problems since PWRs operate in a very low dissolved oxygen environment. As such, LBB has been applied to various high-energy lines in PWRs as an alternate way of addressing the assumption of a DEGB per GDC-4 of 10CFR50. Following the event at V. C. Summer, new LBB submittals for high-energy lines for PWRs with Alloy 82/182 have been deferred until the issue is resolved.

Consideration of SCC in Alloy 82/182 material in LBB applications has three potential effects on the evaluations. First, all the LBB applications to-date have assumed that crack growth by fatigue is the only credible cracking mechanism, and hence the determination of leakage through flaws has been based on crack morphology consistent with fatigue cracks. Recent research [1-6] suggests that the assumption of SCC will have an impact on the roughness assumption since SCC introduces a tortuous path for the fluid in leakage determination. The assumption of increased crack face roughness and a flow path consisting of multiple changes of direction of the flow (turns) can increase the size of leakage flaws, or reduce the margin between critical flaw size and the flaw size needed to produce a specific leak rate. Second, crack growth due to SCC is also a credible crack growth mechanism in addition to fatigue. Studies have shown that SCC growth rate in Alloy 82/182 can be relatively high [1-7]. Third, there is some concern that PWSCC flaws may grow in the circumferential direction around the butt welds, instead of growing predominantly through-wall prior to growing in the circumferential direction, thus affecting the basic premise for LBB. The effect of these three issues will be the main focus of this report.

Three other degradation mechanisms that could potentially affect the LBB evaluation for Alloy 82/182 locations are also addressed in this report. The first is thermal aging, which leads to embrittlement and consequently loss of toughness. This phenomenon has been observed in some stainless weldments [1-8] and as such, it is prudent that it be addressed with respect to Alloy 82/182 materials. The second is the loss of toughness at low temperatures in the presence of hydrogen in Alloy 82/182 and some other nickel-based alloys [1-9, 1-10]. This mechanism is referred to as low temperature crack propagation (LTCP). The third relates to degradation of toughness at the fusion line of Alloy 82/182 welds. The limited amount of work presented in Reference 1-11 indicates that when Alloy 82/182 is welded to ferritic steels, the toughness at the fusion line may be reduced compared to the weld metal. These mechanisms could potentially affect the critical flaw size calculations.

All other aspects related to LBB evaluation remain unchanged for application in PWR reactor coolant system piping. It is recognized that existing LBB analyses for Alloy 82/182 locations may have margins that are over and beyond the minimum recommended in NUREG-1061, Vol. 3 and draft SRP 3.6.3, and as such the effect of the these issues will be investigated to determine how much the present margins in LBB evaluations for Alloy 82/182 locations are affected.

It is widely believed that since the publication of the technical methodology for LBB applications in NUREG-1061, Vol. 3 and the draft NUREG-800, draft SRP 3.6.3, there have been significant advances in technology and experience such that the criteria in these documents can be revisited and some of the conservatisms in the LBB evaluation can be addressed. In particular, work done on various aspects of the Alloy 82/182 butt weld cracking issue [1-12, 1-13, 1-14] is available, which are used in this report as a basis for developing a technical justification for application of LBB at these locations without the need for any mitigating actions. For the B&W designed plants, a separate LBB evaluation was performed to address the Alloy 82/182 locations in the cold legs of the main loop piping [1-15]. Progress in other areas of LBB evaluations is documented in several industry reports [1-16, 1-17]. In addition, plants are typically trending leakage rates at much lower levels than those assumed in original LBB submittals. Therefore, some of the margin requirements in these documents are very conservative. In addition, NDE capabilities in Alloy 82/182 butt welds locations have been significantly improved. Therefore, the restriction imposed in these documents regarding when to apply LBB especially with regard to susceptibility to SCC can be revisited in light of recent work performed by the industry in these areas.

1.3 Objective of Report and Organization

The objective of the study in this report is to demonstrate that LBB applications with Alloy 82/182 locations still have adequate margins in the presence of PWSCC at these locations. Specifically, this report includes:

- LBB locations in general and Alloy 82/182 locations with LBB applications (Section 2)
- The material properties used in the evaluations including discussion of toughness (Section 3)
- PWSCC crack shape and effect on LBB behavior (Section 4)
- Evaluation of critical flaw sizes, leakage, leakage flaw sizes, and associated margins (Section 5)
- Leak detection (Section 6)
- Crack growth evaluations (Section 7)
- The role of in-service inspection and proposed inspection guidelines for Alloy 82/182 locations (Section 8)
- Margins and Uncertainties in LBB Evaluations (Section 9)
- Finally, summary and conclusions are provided. (Section 10)

1.4 Overall Technical Approach

The technical approach employed in this report consists of determining the leakage flaw sizes and critical flaw sizes for Alloy 82/182 locations in the PWR fleet. The margin between the critical flaw size and the leakage flaw size is then determined and compared to the guidelines in NUREG-1061, Vol. 3 and draft SRP 3.6.3 and an assessment of the margin has also been performed assuming a conservative PWSCC crack morphology factor. Since the leakage detection systems of many plants are capable of measuring leakage significantly below 1 gpm, this margin is determined for different assumed leak rates to determine its sensitivity to leakage.

1.5 References

1-1. Arlotto, G. A., "Leak-Before-Break Seminar Opening Remarks," Proceedings on the Seminar on LEAK-BEFORE-BREAK: International Policies and Supporting Research," NUREG/CP-0077, U. S. Nuclear Regulatory Commission, Washington, D.C., October 28-30, 1985.

1-2. The Pipe Break Task Group, "Report of the U. S. Nuclear Regulatory Commission Piping Review Committee – Evaluation of Potential for Pipe Break," NUREG-1061, Volume 3, November 1984.

1-3. Wichman, K. and Lee, S., "Background and Status of Leak-Before-Break Applications in U. S. Nuclear Power Plants," Proceedings on the Seminar on LEAK-BEFORE-BREAK: Further Developments in Regulatory Policies and Supporting Research, NUREG/CP-0109, U. S. Nuclear Regulatory Commission, Washington, D.C., May 11-12, 1989.

1-4. Stello, Jr., V., "Final Broad Scope Rule to Modify General Design Criterion 4 of Appendix A, 10 CFR Part 50," NRC SECY-87-213, Rulemaking Issue (Affirmation), August 21, 1987.

1-5. U. S. Nuclear Regulatory Commission; Standard Revision Plan; Office of Nuclear Reactor Regulation, "Section 3.6.3, Leak-Before-Break Evaluation Procedure," NUREG-0800, August 1987. (Federal Register/Vol. 52, No. 167/August 28,1987/Notices pp. 32626-32633)

1-6. D. Rudland, R. Wolterman, G. Wilkowski and R. Tregoning, "Impact of PWSCC and Current Leak Detection on Leak-Before-Break," Proceedings of Conference on Vessel Head Penetration, Inspection, Cracking, and Repairs, Sponsored by USNRC, Marriot Washingtonian Center, Gaithersburg, MD, September 29 to October 2, 2003.

1-7. *Materials Reliability Program: Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 82, 182, and 132 Welds (MRP-115),* EPRI. Palo Alto, CA: 2004. 1006696.

1-8. D. J. Gavenda, et al, "Effects of Thermal Aging on Fracture Toughness and Charpy-Impact Strength of Stainless Steel Pipe Welds," NUREG/CR-6428, May 1996.

1-9. W.J. Mills and C.M. Brown, "Fracture Toughness of Alloy 600 and EN82H Weld in Air and Water," Bettis Atomic Power Laboratory Report No. B-T-3264.

1-10. W.J. Mills and C.M. Brown, "Low Temperature and Crack Propagation Behavior of Alloy 82H Welds, Alloy 52, and Alloy 690," Bettis Atomic Power Laboratory Report No. B-T-3549.

1-11. C. Williams, et al., "The Impact of Fracture Toughness and Weld Residual Stress of Inconel 82/182 Bimetal Welds on Leak-Before-Break Behavior," PVP-Vol. 479, Residual Stress, Fracture, and Stress Corrosion Cracking, July 25-29, 2004, San Diego, CA, PVP2004-2659.

1-12. *Materials Reliability Program: Alloy 82/182 Butt Weld Safety Assessments for U.S. PWR Plant Designs (MRP-113),* EPRI, Palo Alto, CA: 2004. 1009549.

1-13. *Materials Reliability Program: Alloy 82/182 Pipe Butt Weld Safety Assessment for U. S. PWR Plant Designs: Westinghouse and CE Design Plants (MRP-109),* EPRI, Palo Alto, CA: 2004. 1009804.

1-14. *Materials Reliability Program: Alloy 82/182 Pipe Butt Weld Safety Assessment for U. S. PWR Plant Designs: Babcock and Wilcox Design Plants (MRP-112),* EPRI, Palo Alto, CA: 2004. 1009805.

1-15. AREVA Document No. 51-5052759-00, "Safety Evaluation of Alloy 82/182 Welds in LBB Applications," February 2005.

1-16. N. Ghadalia, S. Rahman, Y. H. Choi and G. Wilkowski, "Deterministic and Probabilistic Evaluation for Uncertainty in Pipe Fracture Parameters in Leak-Before-Break and in Service Flow Evaluations," NUREG/CR-6443, BMI-2191, June 1996.

1-17. P. M. Scott, R. J. Olson and G. M. Wilkowski, "Development of Technical Basis for Leak-Before-Break Evaluation Procedures," NUREG/CR-6765, May 2002.

2
COMPILATION OF PLANT INFORMATION ON LBB

There are a total of 69 PWR units in the U.S. These plants were designed by three major vendors, Babcock & Wilcox (seven), Combustion Engineering (14), and Westinghouse (48). A summary of the Alloy 82/182 butt welds in these plants has been provided in the MRP Butt Weld Safety Assessment [2-1, 2-2, 2-3]. The number of PWR plants with current or pending LBB applications are shown in Table 2-1. All the Alloy 82/182 LBB butt weld locations are associated with the nozzle (typically SA 508 C1ass 2) to piping or safe end connections. The number of plants and pipelines or systems with Alloy 82/182 locations that have current approved or pending LBB submittals is shown in Table 2-2.

As can be seen from Table 2-1, several piping systems have been qualified for LBB. However, as shown in Table 2-2, Alloy 82/182 materials are present in five main piping systems to which LBB has been applied. These are the primary loop piping (which includes the piping connections to the reactor vessel, pumps and the steam generators), the surge line (connections to the pressurizer and the hot leg), the safety injection line, the shutdown cooling line and the core flood piping (connection to the core flood tank). The evaluations in this report will focus on these five piping systems.

Table 2-1
Plants with Current or Pending LBB Applications in PWRs

	Plant Type		
	Westinghouse	**CE**	**B&W**
Main Loop Piping	48	14	7
Surge Line Piping	23	1	0
RHR Piping	13	0	0
Accumulator Piping	19	0	0
Loop Bypass Piping	6	0	0
Main Steam Piping	2	0	0
Core Flood Piping	NA	NA	3
Shutdown Cooling	NA	1	0
Safety Injection	1	1	0

Table 2-2
Plants with Alloy 82/182 LBB Locations in PWRs

	Plant Type		
	Westinghouse	**CE**	**B&W**
Main Loop Piping	30	11	7
Surge Line Piping	16	1	0
Core Flood Piping	NA	NA	3*
Shutdown Cooling	NA	1	NA
Safety Injection	0	1	NA

* This project did not evaluate these locations because they are at containment temperature and are isolated from the RCS.

2.1 References

2-1. *Materials Reliability Program: Alloy 82/182 Butt Weld Safety Assessments for U. S. PWR Plant Designs (MRP-113),* EPRI, Palo Alto, CA: 2004. 1009549.

2-2. *Materials Reliability Program: Alloy 82/182 Pipe Butt Weld Safety Assessment for U. S PWR Plant Designs: Westinghouse and CE Design Plants (MRP-109),* EPRI, Palo Alto, CA: 2004. 1009804.

2-3. *Materials Reliability Program: Alloy 82/182 Pipe Butt Weld Safety Assessment for U. S PWR Plant Designs: Babcock and Wilcox Design Plants (MRP-112),* EPRI, Palo Alto, CA: 2004. 1009805.

3 EVALUATION OF MATERIAL PROPERTIES INCLUDING TOUGHNESS OF ALLOY 82/182 WELDS

Material properties play a key role in several aspects of LBB evaluations. In this section, material properties used in the LBB evaluations are discussed. Material properties are required for the determination of critical and leakage flaw sizes.

3.1 Material Properties for Critical and Leakage Flaw Size Determination

Two alternate methods are allowed in NUREG-1061, Vol. 3 [3-1] and NUREG 0800, draft SRP 3.6.3 [3-2] in calculating critical flaw sizes for LBB evaluation. The first is elastic-plastic analysis employing the J-integral/tearing modulus (J-T) technique. The second method involves the use of limit load (net section plastic collapse) analysis. The inherent assumption in the use of limit load analysis for determination of critical flaw sizes is that the material possesses adequate toughness such that the whole cross-section of the pipe becomes plastic prior to failure. On the other hand, in order to use J-T analysis, it is only necessary to show that the material has enough toughness to prevent brittle fracture.

Limit load analysis is used in this report to determine the critical flaw sizes at normal operating temperature because Alloy 82/182 welds have been shown to be very ductile and possess toughness comparable to forged austenitic stainless steel base metal or GTAW/GMAW weldments at the normal operating temperature. At lower temperatures (less than 300^{o}F), where the toughness is very low, elastic-plastic fracture mechanic techniques involving the J-T analysis concept is used to determine the critical flaw sizes.

3.1.1 Toughness of Alloy 82/182

Work done on Alloy 82/182 materials indicate that at temperatures corresponding to the operating conditions of PWRs, these materials have adequate toughness [3-3, 3-4] such that limit load analysis is appropriate for determination of critical flaw sizes. Figure 3-1 shows the J versus crack extension (Δa) curve of the parent base material (Alloy 600). Figures 3-2 and 3-3 show the J-Δa curve for Alloy 82 in both the longitudinal and transverse direction. As can be seen from these figures, at high temperatures corresponding to normal operating conditions, the J-Δa curves in both directions for the Alloy 82 material is comparable to the forged base metal. These curves can also be compared to a typical J-Δa curve for forged austenitic stainless steel base metal in Figure 3-4 and TIG welds in Figure 3-5 [3-5]. It can be seen that the toughness of Alloy 82/182 is very comparable to these materials. Based on these comparisons, it is clear that at high operating temperatures, Alloy 82/182 has adequate toughness such that the pipe cross-section becomes fully plastic prior to failure, allowing the use of limit load analysis for determination of critical flaw sizes.

Studies performed in Reference 3-6 on a bimetallic weld consisting of Alloy 82/182 welded to A516 Grade 70 carbon steel base at near PWR operating temperatures indicate that the toughness of the fusion line may not be as high as compared to the weld material (see Figure 3-6). However, even with this reduction in toughness at the fusion line, it is judged that the component still possesses adequate toughness to justify the use of limit load analysis for the determination of the critical flaw sizes since as noted from Figure 3-6, the weld metal has extremely high toughness to begin with and so the reduction of the toughness at the fusion line still results in a regime where limit load remains applicable. Another observation from Figure 3-6 is that at the fusion line, the material exhibits much higher ductility as evidenced by the higher crack extension slope, dJ/da, which compensates for the reduction in the J values.

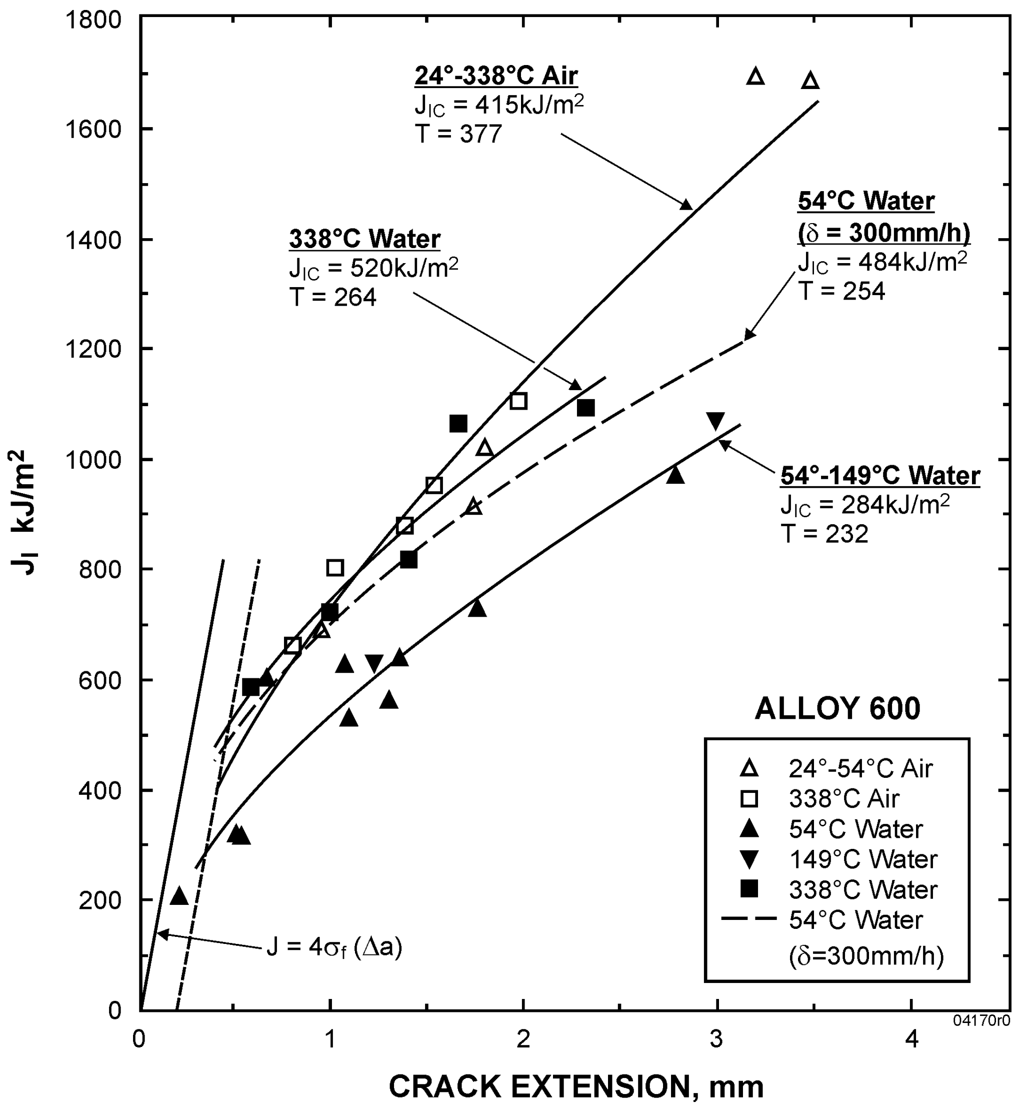


Figure 3-1
J-R Curve for Alloy 600 in Air and Water with Hydrogen Overpressure [3-3]

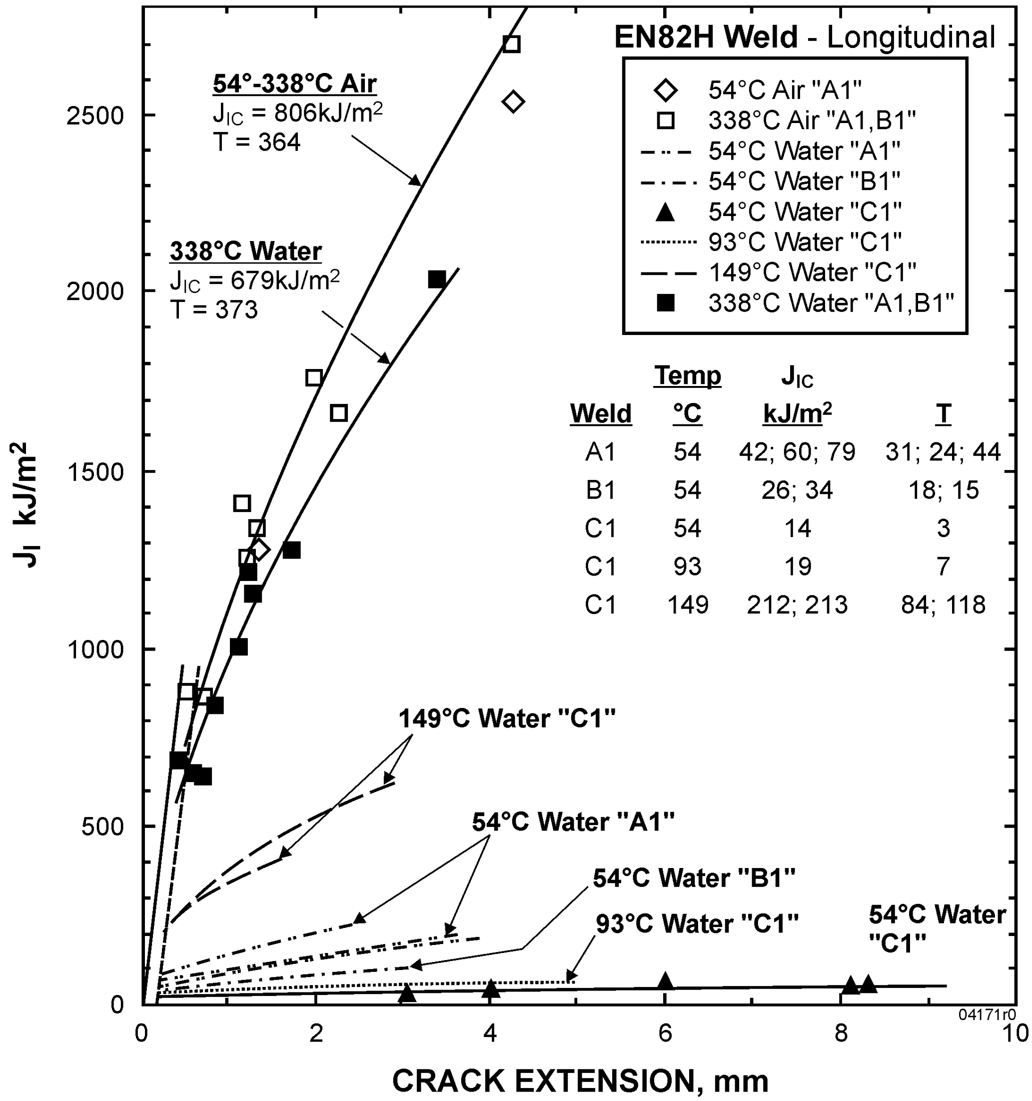


Figure 3-2
J-R Curves of EN82H Welds with Longitudinal Orientation Tested in Air and Water With Hydrogen Overpressure [3-3]

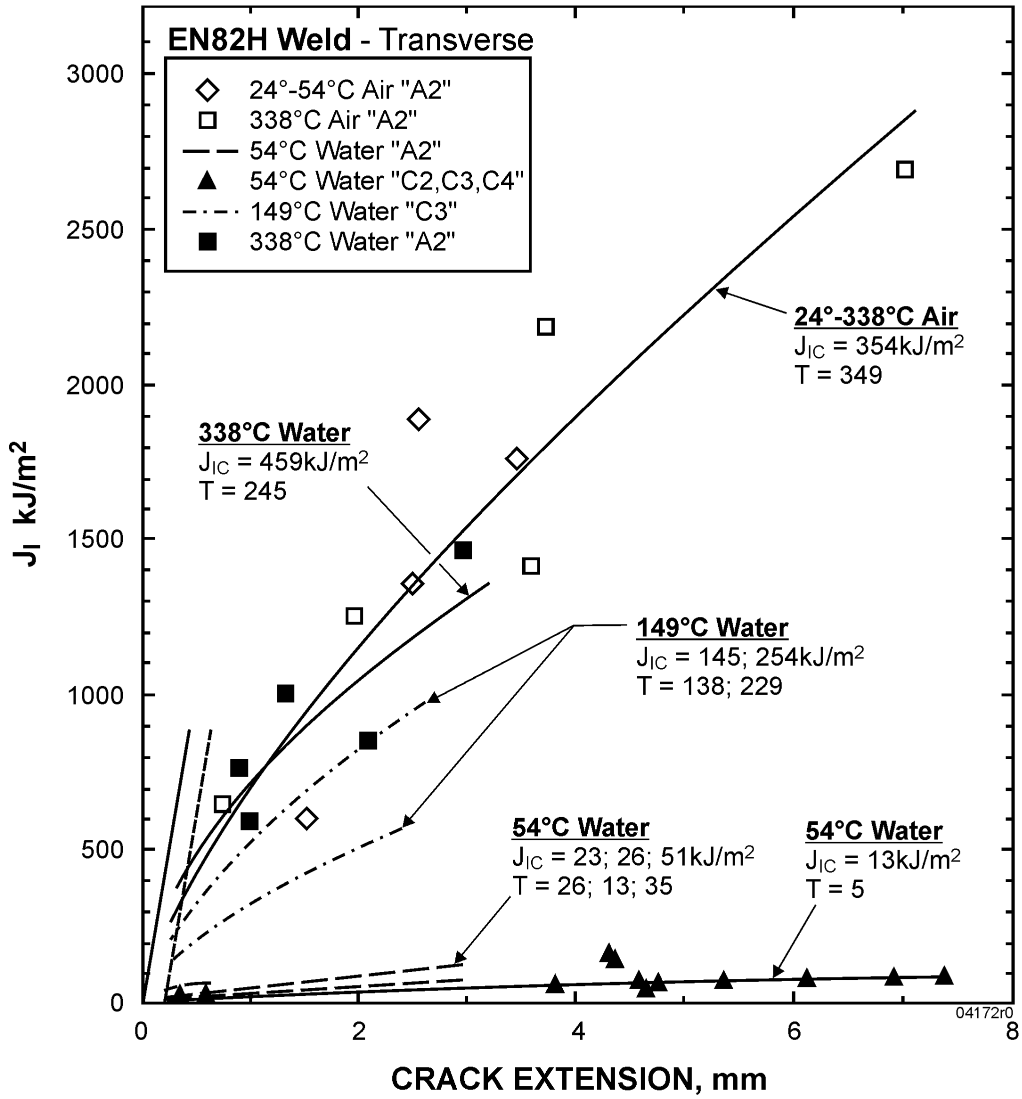


Figure 3-3
J-R Curves of EN82H Welds with Transverse Orientation Tested in Air and Water With Hydrogen Overpressure [3-3]

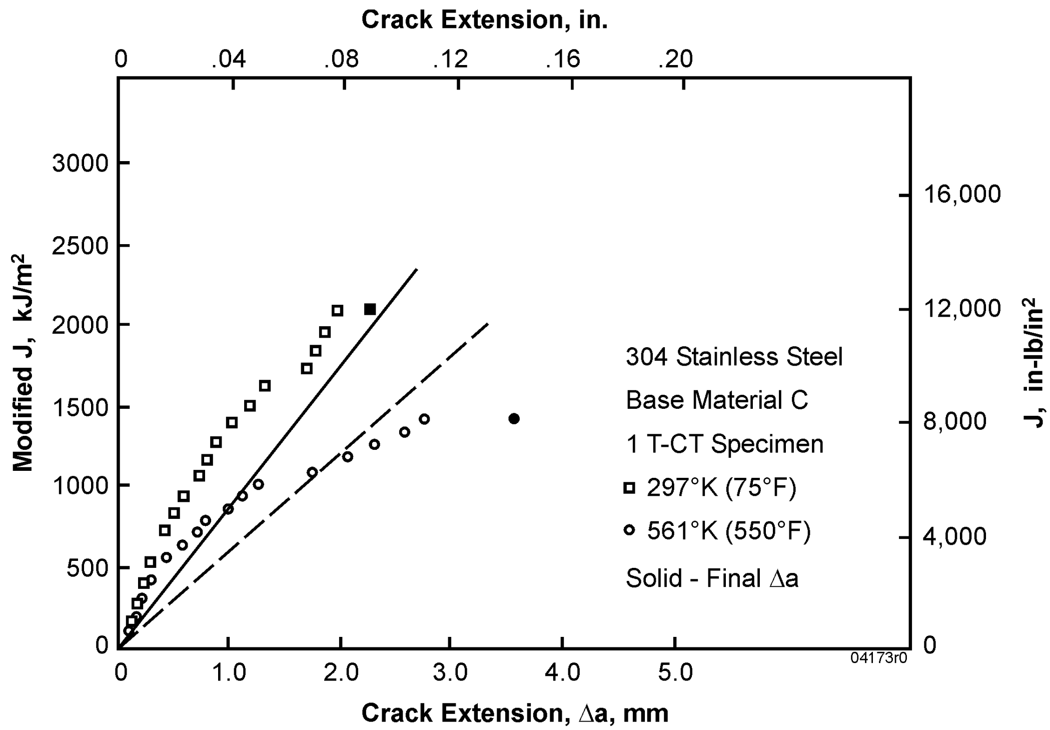


Figure 3-4
J-R Curve for Type 304 Stainless Steel Base Metal [3-5]

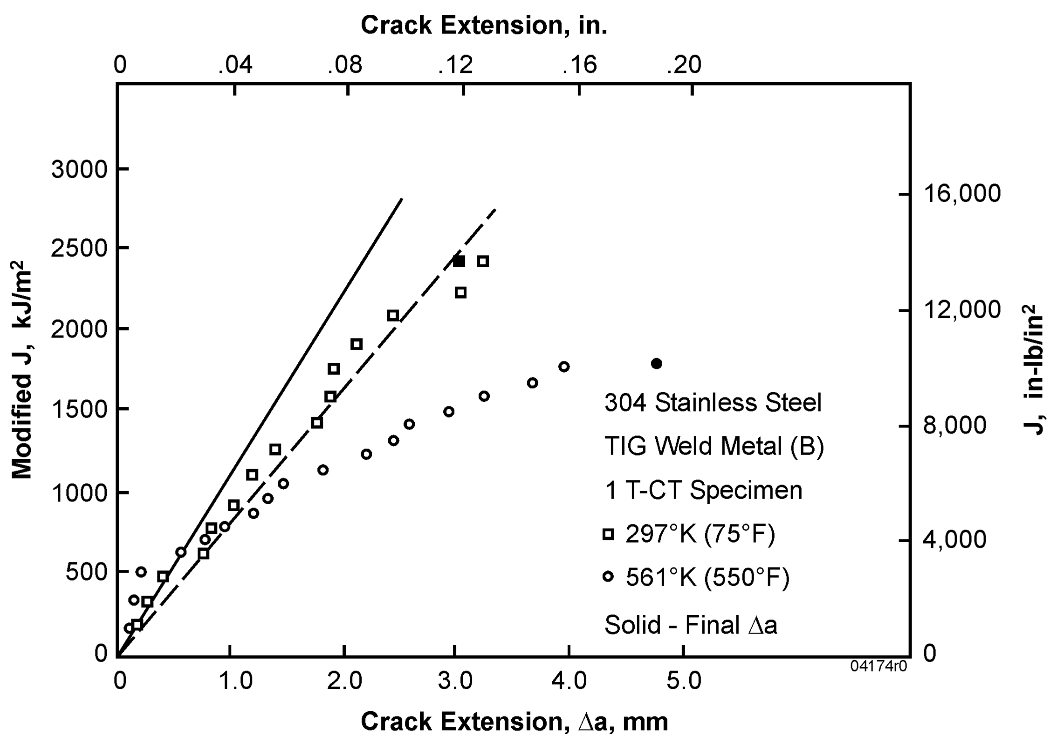


Figure 3-5
Typical J-R Curve for Stainless Steel TIG Welds [3-5]

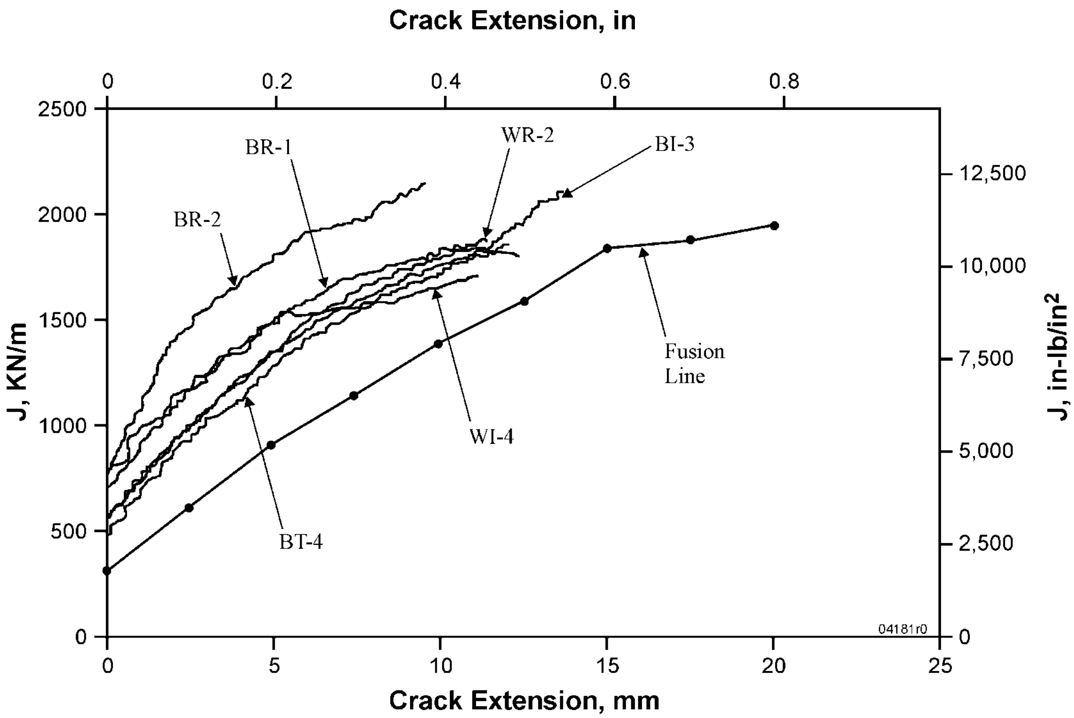


Figure 3-6
Comparison of J-R Curve Data with Fusion Line Data for Same Bimetal Weld Tested as Part of the NRC Short Crack Program [3-6]

3.1.1.1 Low Temperature Crack Propagation

As can be seen from Figures 3-2 and 3-3, at temperatures below 149°C (300°F), Alloy 82/182 begins to experience severe degradation in fracture resistance when hydrogen overpressure is present. This phenomenon is referred to as low temperature crack propagation (LTCP). LTCP occurs at temperatures below 150°C (300°F) starting from pre-existing sharp cracks, with hydrogen at grain boundaries, and under local crack tip stresses greater than the material yield strength. This degradation in fracture resistance is due to hydrogen-induced intergranular cracking. The temperature and hydrogen conditions for LTCP to occur may exist during some stages of cooldown in PWRs where high levels of hydrogen can potentially exist depending on specific plant and shutdown procedure. The combination of low temperature and hydrogen overpressure is rare, but under these conditions, the operating loads have reduced significantly and therefore even with this reduction in fracture resistance at low temperature, large critical flaw sizes may still be possible which may not invalidate the LBB evaluations performed at normal operating conditions. The effect of LTCP on critical flaw sizes is discussed in Section 5.

3.1.1.2 Thermal Aging of Alloy 82/182

Work done in Reference 3-7 using limited data indicates that Alloy 82/182 materials do not become embrittled due to long time exposure at PWR operating temperatures. This is because there are no embrittling intermetallic phases such as sigma (σ), chi (χ) or alpha prime (α') phase precipitations responsible for thermal aging embrittlement of austenitic stainless steel welds and castings. Hence, thermal aging does not have any effect on the toughness of Alloy 82/182.

3.1.2 Key Material Properties

For critical flaw size determination using limit load analysis, the only material property of interest is the flow stress, which is the average of yield and ultimate strength. Material properties consistent with the original LBB submittals were used in this evaluation. These are typically ASME Code minimum values or values slightly higher than the Code minimum but less than the actual ones. There is a wide range of ASME Code minimum properties for base material (Alloy 600) but typical and representative values are shown in Table 3-1 [3-8]. These properties are compared with typical Type 304 stainless steel base materials and it is seen that in general Alloy 600 properties are higher than the stainless steel properties. Even more importantly, typical actual values of the material properties for Alloy 82/182 materials obtained from two sources are also shown in Table 3-1 [3-3, 3-9]. As can be seen from this table, the actual properties are considerably higher than the Code minimum values for the base material used in some aspects of this evaluation. The use of actual material properties for Alloy 82/182 would increase the critical flaw sizes considerably.

Table 3-1
Material Properties of Alloy 82/182 and Alloy 600 Base Metal

Basis	Material	Material Properties (ksi)								
		70°F			129°F			640°F		
		S_y	S_u	$E \times 10^3$	S_y	S_u	$E \times 10^3$	S_y	S_u	$E \times 10^3$
ASME Code (Ref. 3-8)	Alloy 600 SB-166	35	80	31	34.33	90	30.64	27.50	80	28.5
	Alloy 600 SB-167	30	80	31	29.59	80	30.64	23.98	80	28.5
	Type 304 Stainless Steel SA-312	30	75	28.3	28.09	73.84	27.98	18.08	63.4	25.1
Actual	Alloy 600 (Ref. 3-9)	49.00	100.0	31.1	48.0	99.0	30.8	43.00	93.00	28.6
	Alloy 82 (Ref. 3-9)	57.10	96.20	31.1	53.0	97.0	NA	47.00	84.00	28.6
	Alloy 182 (Ref. 3-9)	55.10	92.40	31.1	55.0	93.0	30.8	44.00	84.00	28.6
	Alloy 82 (Ref. 3-3)	NA	NA	NA	68.73	87.15	NA	57.75	82.80	NA

The material properties for leakage flaw size determination include the modulus of elasticity, the yield stress and may include the Ramberg-Osgood stress-strain parameters. Typical material properties are shown in Table 3-1. In practice, the same consistent set of material properties are used in the leakage flaw determination as was used for determination of the critical flaw sizes.

3.2 References

3-1. The Pipe Break Task Group, "Report of the U. S. Nuclear Regulatory Commission Piping Review Committee – Evaluation of Potential for Pipe Break," NUREG-1061, Volume 3, November 1984.

3-2. U. S. Nuclear Regulatory Commission; Standard Revision Plan; Office of Nuclear Reactor Regulation, "Section 3.6.3, Leak-Before-Break Evaluation Procedure," NUREG-0800, August 1987. (Federal Register/Vol. 52, No. 167/August 28,1987/Notices pp. 32626-32633).

3-3. W. J. Mills and C. M. Brown, "Fracture Toughness of Alloy 600 and EN82H Weld in Air and Water," Bettis Atomic Power Laboratory Report No. B-T-3264.

3-4. W. J. Mills and C. M. Brown, "Low Temperature and Crack Propagation Behavior of Alloy 82H Welds, Alloy 52, and Alloy 690," Bettis Atomic Power Laboratory Report No. B-T-3549.

3-5. J. D. Landes and D. E. McCabe, "Toughness of Austenitic Stainless Steel Pipe Welds," EPRI NP-4768, Electric Power Research Institute, Palo Alto, CA, October 1986.

3-6. C. Williams, et al., "The Impact of Fracture Toughness and Weld Residual Stress of Inconel 82/182 Bimetal Welds on Leak-Before-Break Behavior," PVP-Vol. 479, Residual Stress, Fracture, and Stress Corrosion Cracking, July 25-29, 2004, San Diego, CA, PVP2004-2659.

3-7. *Materials Reliability Program: A Review of Thermal Aging Embrittlement in Pressurized Water Reactors (MRP-80),* EPRI, Palo Alto, CA: 2003. 1003523.

3-8. ASME Boiler and Pressure Vessel Code, Section II, Parts A and B, 2001 Edition.

3-9. Special Metals Corporation, "INCONEL Alloy 600," Publication No. SMC-027, September 2002.

4
PWSCC CRACK SHAPE AND EFFECTS ON LBB BEHAVIOR

One of the important premises of LBB is that when flaws develop, they will grow in the through-wall direction and leak before they grow significantly in the axial or circumferential directions. This assumption has been universally accepted as a basis for crack growth behavior if fatigue is the predominant crack growth mechanism. However, there may be some concern that flaws associated with PWSCC may not exhibit this through-wall growth behavior and therefore render the assumption of a through-wall leak before failure of the pipe questionable. In this section, some of the field experience from PWRs and BWRs is reviewed to show that flaws associated with PWSCC are expected to propagate in the through-wall direction and leak without significant growth in the axial or circumferential directions.

4.1 PWR Field Experience

Cracking of Alloy 82/182 material in PWR butt welds has been very limited. It was not until 2000 that the first incident of cracking was reported in the industry after several years of operation. Including V. C. Summer, only five such cracking events have been reported.

During the second half of 2000, cracks were discovered in Alloy 182 welds joining low alloy steel reactor vessel hot leg nozzles to stainless steel pipes at Ringhals 3 and 4, and V. C. Summer. All cases involved axial cracks in the weld metal. In the case of V. C. Summer, one of the axial cracks appears to have initiated at a weld repair location and then developed into a leak that resulted in over 100 pounds of boric acid crystals being deposited outside the pipe near the weld. At V. C. Summer, a short circumferential crack was also discovered on the inside diameter region of the Alloy 182 weld butter. This circumferential crack arrested when it reached low alloy steel base material. The root cause analyses attributed the cracking to PWSCC of the Alloy 182 weld metal [4-1].

In 2002 axial indications were discovered in the Alloy 182 butt weld between the pressurizer surge nozzle and safe end at Tihange 2. This weld had been stress relieved with the pressurizer vessel. The weld was re-inspected after six months of operation in 2003 with no evidence of crack growth. At this time, this indication has not been attributed to PWSCC and continues to be monitored by the utility.

A leak was discovered from a pressurizer safety relief line butt weld at Tsuruga 2 in 2003. This flaw was also found to be an axial flaw that arrested in the adjacent base metal. A second part through-wall axial flaw was also found in this weld.

In late 2003, a part through-wall axial flaw was detected in the hot leg nozzle-to-surge line weld at TMI Unit 1. The indication appeared to be in the Alloy 182 butter.

The key observation from the above cracking events is that PWSCC in Alloy 82/182 materials in PWRs resulted mostly in axial cracks. The orientation of the cracks is not unexpected given that the residual stresses from welding favor flaw initiation and growth in the axial direction. Since the length of axial flaws are limited to the width of the Alloy 82/182 welds, PWSCC growth will occur primarily in the through-wall direction resulting in leaks. The small circumferential flaw associated with the cracking at V. C. Summer though suggests that circumferential flaws cannot be totally ruled out and need to be evaluated with respect to LBB since they present the most limiting case.

4.2 Analytical Evaluation of PWSCC Initiation and Growth in PWR Butt Welds

Significant work has been reported in MRP-106 [4-2], MRP-114 [4-3] and MRP-113 [4-4] to evaluate PWSCC initiation and growth in Alloy 82/182 butt welds and the role played by residual stresses including the effect of weld repairs. Several weld repair sizes were simulated in these studies. Given the high hoop stresses at both repaired and unrepaired locations, the most probable outcome is an axial flaw developing and propagating through-wall to produce a visible leak prior to a circumferential part through-wall crack growing to a long length. The supporting conclusions from these studies are summarized below:

- For the as-designed case with no repairs, hoop and axial stresses on the wetted Alloy 600/82/182 material are relatively low. The highest tensile stresses for these cases are hoop stresses nearer the OD on the weld. The relatively low ID hoop and axial tensile stresses may have contributed to the small number of reports of PWSCC or leaks to date for these joints.
- Weld repairs to the inside surface of the nozzle/pipe are clearly detrimental, creating high tensile hoop stresses and tensile axial stresses of similar magnitude. These results are consistent with the axial and shallow circumferential PWSCC cracks at V. C. Summer.
- While weld repairs to the ID surfaces are clearly detrimental from the standpoint of PWSCC susceptibility, the piping containing many of these butt weld joints are believed to be too small in diameter for weld repairs to have been performed.
- If flaws initiate by PWSCC, or due to weld defects or grinding (for some repair sizes and at higher moment stress levels), significant growth may occur at locations away from the weld repair. However, given the residual stress in the weld repair, there is a very high likelihood that through-wall crack will develop if flaws were to initiate.
- The results support leak-before-break in that initiated flaws would tend to grow through-wall within the weld repair region, and, except for very high piping load cases, would grow through the wall beyond the weld repair for only short distances. The exception is the 360° weld repair case where the through-wall growth could occur anywhere. However, uniform initiation is highly unlikely, even when extensive grinding has occurred.

The effect of multiple crack initiation sites on crack growth was also investigated in MRP-113 [4-4]. It was concluded that multiple initiation sites are associated with multiple weld repairs. If multiple initiation sites were to occur in a pipe, through-wall growth of one or more of the flaws would be observed in the weld-repaired region with minimum impact on the non-repaired region. In the absence of any weld repairs, PWR butt welds have very favorable residual stress patterns that arrest PWSCC.

For the extreme case of fully circumferential inside surface repairs, the final, as-repaired residual stress state could promote PWSCC initiation and growth, possible through-wall and significantly around the circumference of the component. Even in this case, however, other factors relating to both PWSCC initiation and growth, such as complex effects of weld metallurgy and weld pass sequencing make it unlikely that circumferential cracks of uniform depth would develop. Thus, it is expected that a flaw associated with such a significant repair would still grow locally through-wall and result in leakage.

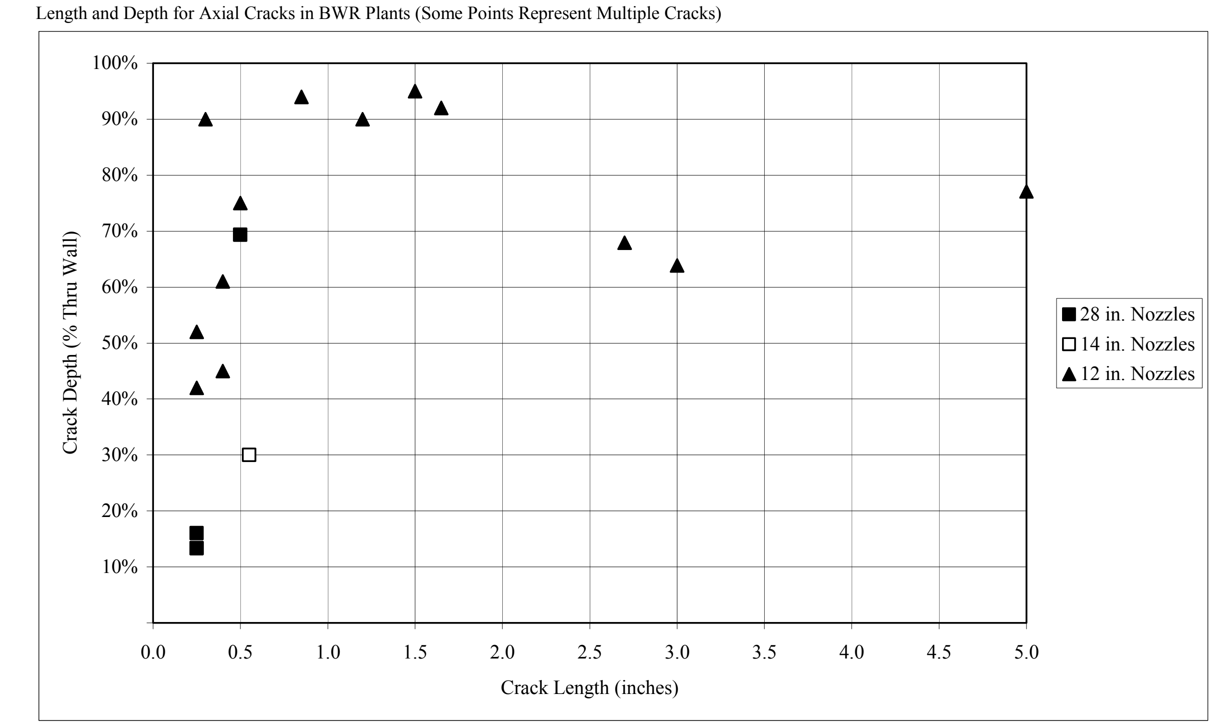
Finally, given the high hoop stresses at repaired locations, the most probable outcome is an axial flaw developing and propagating through-wall to produce a visible leak prior to a circumferential part through-wall crack growing to a long length.

4.3 BWR Field Experience

BWR plants have experienced stress corrosion cracking in early plant life including at Alloy 82/182 locations. Although the nozzle-to-safe end welds of most BWRs were fabricated from Alloy 82/182 material using welding processes similar to those used for the PWRs, the operating environments are quite different. As such, the mechanism of SCC in Alloy 82/182 locations in BWRs may be somewhat different than that in PWRs. Nevertheless, the experience from the BWRs can shed some light on the cracking behavior of these alloys.

The first incidence of cracking in Alloy 82/182 material occurred at the Pilgrim Station in the early 1980s. Since then, there have been several Alloy 82/182 cracks reported in many BWRs. GE Nuclear Energy (GE) summarized these cracking incidents in MRP-57 [4-5]. Data from the 13 BWR plants known to have experienced cracking in Alloy 182 reactor coolant pressure boundary weldments were evaluated in MRP-57.

Out of those 35 cracked Alloy 82/182 weldments, 21 were identified as having axial cracks exclusively, 11 were identified as having circumferential cracks exclusively, and 3 were identified as having both axial and circumferential cracking. The data, however, showed that the larger diameter (28-inch recirculation outlet nozzle) weldments have experienced only axial cracking consistent with what has been observed to date in PWRs. The data also shows that, overall, 78% of the cracks were confined to the Alloy 182 weld and butter, and circumferential cracking typically did not extend beyond 75° (approximately 20%) of the weld circumference. Figure 4-1 shows plots of the lengths and depths of axial cracks and the arc-lengths and depths of circumferential cracks discovered in BWR pipe butt welds. The circumferential cracking at one plant that caused a through-wall leak was only 4 inches long (~10% of the weld circumference) confirming LBB characteristics for circumferential flaws in Alloy 82/182 material.



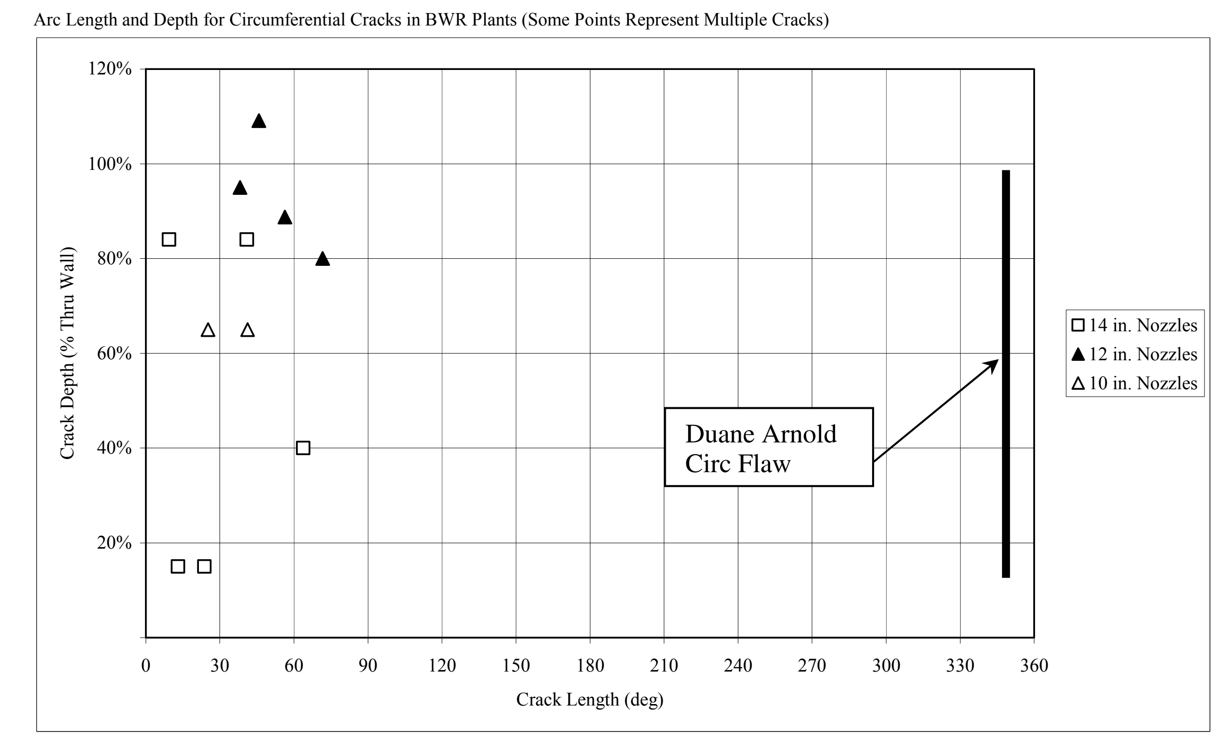


Figure 4-1
Lengths and Depths of Axial and Circumferential Cracks in BWR Butt Welds [4-4]

4.4 Duane Arnold Energy Center (DAEC) Alloy 600 Safe-End Cracking

The case of the 360° part-depth crack at DAEC (a BWR) due to IGSCC has received significant attention. An extensive discussion of the cracking at of the DAEC safe-ends including causal factors and applicability to Alloy 82/182 PWSCC cracking in PWR butt welds has been presented in MRP-114 [4-3]. The following provides a summary of that event, and shows that it is very unique and not typical of PWSCC in PWRs.

4.4.1 Description of Event

A slow increase in unidentified leakage was identified at DAEC on May 1, 1978 [4-6]. By June 14, 1978 the unidentified drywell leakage had increased from approximately 1 gpm to 3 gpm. At 00:55 am on June 17, 1978, an automatic reactor scram occurred due to problems in the reactor protection system relays during weekly control valve testing. Plant operators took the action to reduce reactor pressure, de-inert the containment, enter the containment and investigate the leakage. A survey of the drywell identified a leak in the vicinity of the N2A recirculation inlet nozzle. A cross section through the crack is shown in Figure 4-2 [4-7]. The safe-ends on all eight recirculation inlet nozzles were visually, ultrasonically, and radiographically examined. A through-wall circumferential crack was observed on the N2A nozzle that had an extent of approximately 90 degrees. Cracks were later identified on all of the remaining recirculation inlet nozzle safe ends.

4.4.2 Causal Factors for DAEC Safe-End IGSCC

The cracked DAEC safe end was destructively examined by Southwest Research Institute (SWRI) [4-6] and independently verified by GE. Results from metallographic examination indicated that the mode of cracking was IGSCC and existed 360 degrees around the safe end as sketched in Figure 4-2 based on a figure from Reference 4-7. The results of the examination yielded several causal factors for the IGSCC at DAEC:

1. The design of the safe-end was deficient for the intended application in that an electrochemical crevice was created where the thermal sleeve was attached to the safe-end with a partial penetration weld, as shown in Figures 4-2 and 4-3. The presence of a crevice results in premature IGSCC initiation in high purity BWR-type environments [4-8]. When detrimental impurities such as sulfate and/or chloride are present, the detrimental effects of crevices are intensified. A discussion of the effect of crevices on IGSCC propensities are provided in the DAEC report and summarized in Figure 4-4 [4-6]. Small specimen test results for creviced and uncreviced Alloy 600 are shown in Figure 4-5 [4-9].

2. As noted above, sulfate accelerates IGSCC initiation and propagation [4-8, 4-9, 4-10]. During startup at the plant, approximately 800 pounds of resin was inadvertently released into the DAEC reactor coolant in 1975 [4-11]. Although the system was cleaned up as best as could be achieved at the time, some traces of residual resin would be expected to remain. This would degrade into dissolved sulfates upon irradiation. Sulfur was clearly identified in the crack tip at the cracked safe-end.

3. The thermal sleeve was welded with a fillet weld. This left a crevice at the intersection of the thermal sleeve, the Alloy 600 safe end and the attachment weld that existed 360 degrees around the weld location. High residual tensile stresses resulted from the shrinkage of the attachment weld as it cooled [4-6].

4. Estimates were made of the rate of cracking in the report based on the stress rule index [4-6]. It was reported that the creviced safe end stress rule index (SRI) was 2.24 compared to 0.73 for the later design. This SRI was the highest in the BWR fleet. The result was confirmed by Battelle Columbus Laboratories (BCL) by a time history analysis that considered weld input parameters and utilized a temperature dependent material stress-strain curve [4-6]. The evaluation indicated that above yield tensile stresses existed over the entire safe end side of the crevice. The location of the cracking was consistent with the stress analysis.

5. The susceptible region formed adjacent to the weld in the safe end was caused by re-solution of the carbide phase and subsequent grain boundary precipitation. This was confirmed by the metallographic results from samples etched by phosphoric acid [4-6].

6. In addition, several weld repairs and associated grinding were performed on the weld. This would have contributed to both the premature crack initiation and propagation of cracking by further reducing the corrosion resistance of Alloy 600 due to additional heating from the weld repairs and the increase in the yield strength. The cold working of the Alloy 600 from the grinding would also reduce the creviced material's IGSCC resistance.

4.4.3 Applicability of DAEC Alloy 600 Safe End Cracking to Alloy 82/182 PWSCC Cracking

Alloy 82/182 butt welds in PWRs are generally between carbon steel/low-alloy steel nozzles and stainless steel piping. At DAEC, a thermal sleeve was attached to the ID of the safe-end by a partial penetration weld. An electrochemical crevice was formed between the pipe ID and the closely fitting OD of the thermal sleeve at DAEC. The residual stresses from the attachment were highly tensile at the tip of the crevice. For PWR piping, a thermal sleeve may exist, but there is always a relatively large annular gap that does not create the same electrochemical crevice that existed at DAEC.

IGSCC tests of uncreviced Alloy 600 in a BWR environment have shown that high tensile stresses are needed for crack initiation. In fact, the occurrence of IGSCC in uncreviced Alloy 600 component in BWRs is very rare. The only uncreviced Alloy 600 IGSCC identified in an operating BWR occurred on a pressurized tube test specimen that had a 90% through-wall crack after seven years of exposure at an applied stress ratio of 1.4. (See the solid point pinning the uncreviced Alloy 600 IGSCC curve in Figure 4-5.)

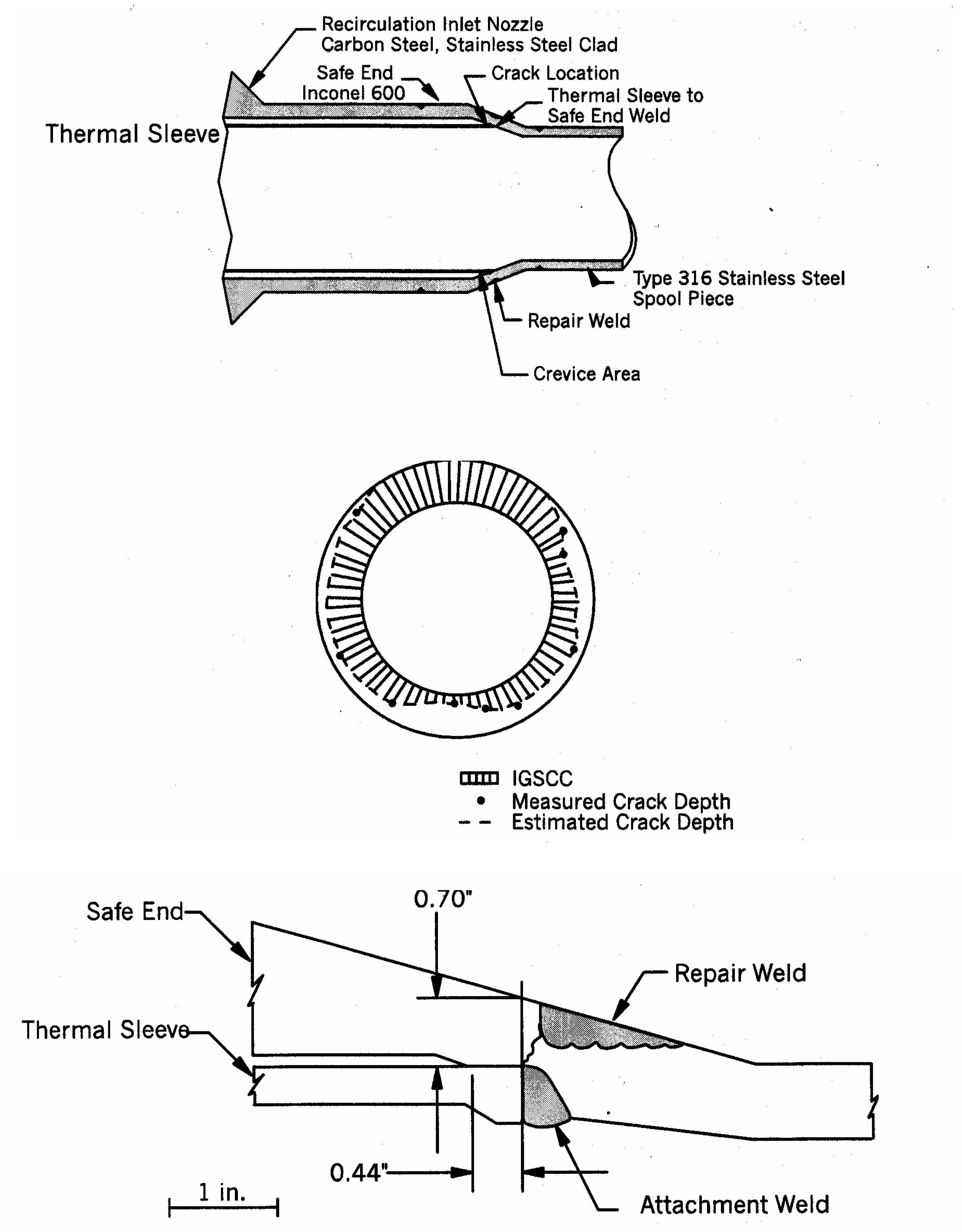


Figure 4-2
Cross Section Through 360 Degree Part Depth Crack at DAEC [4-5]

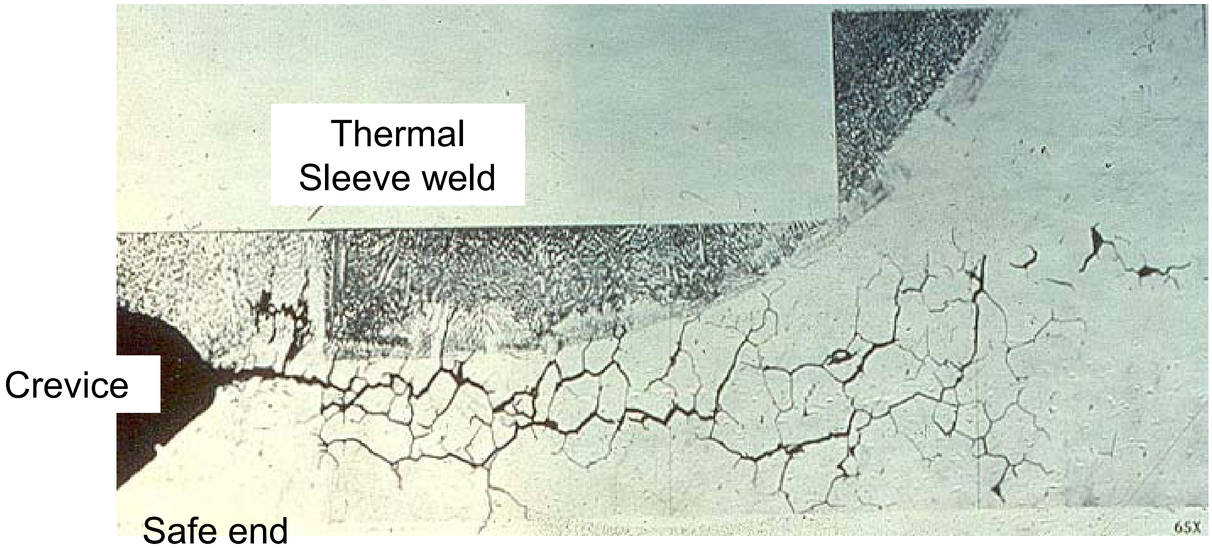


Figure 4-3
DAEC Alloy 600 Recirculation Inlet Safe End Crevice with IGSCC

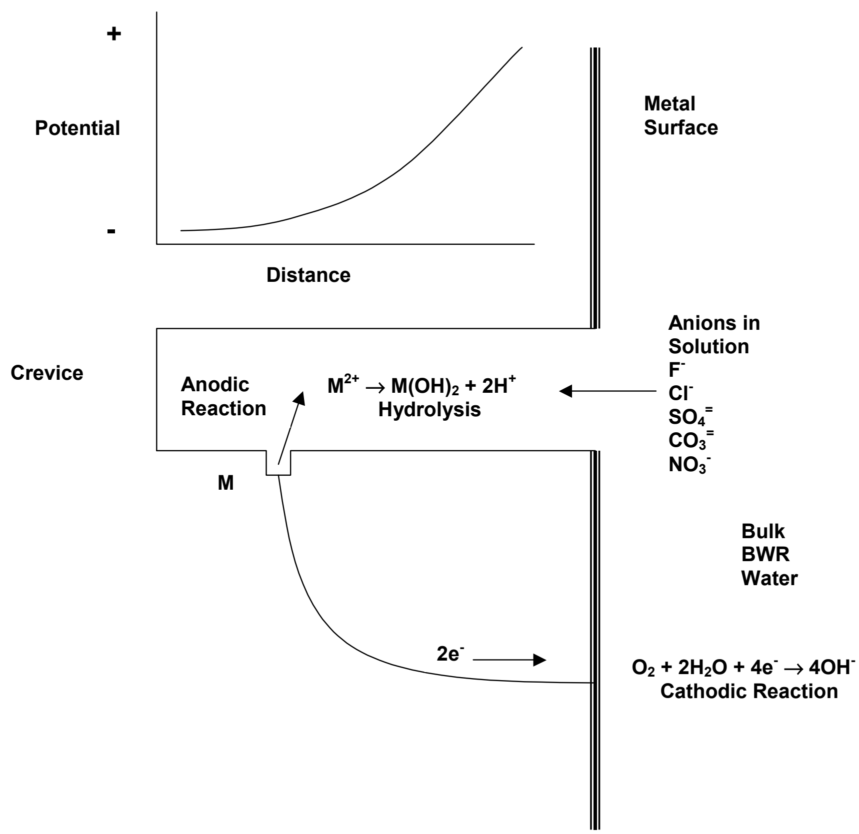


Figure 4-4
Crevice Effects in Oxygenated Environments

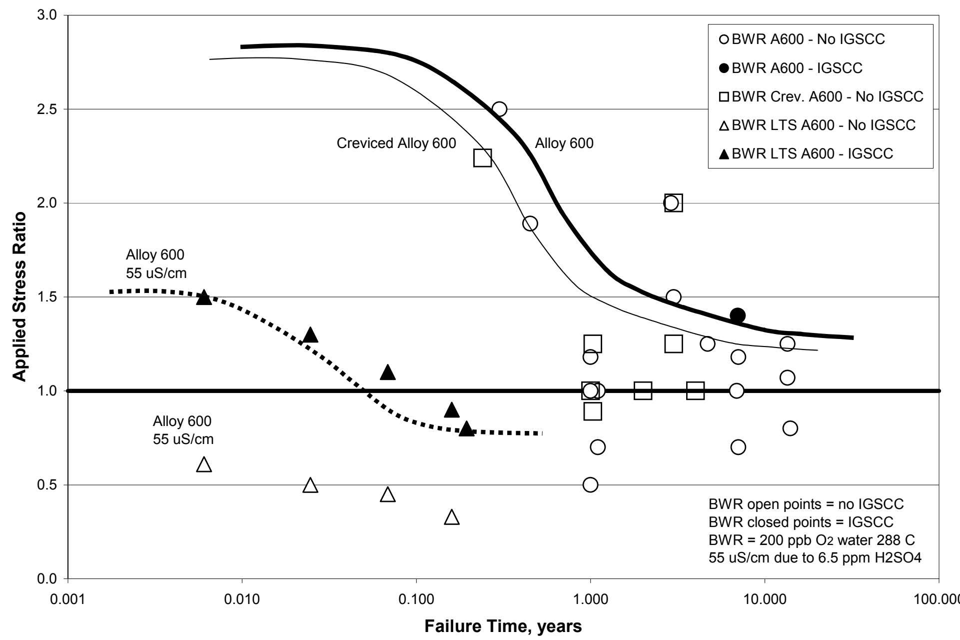


Figure 4-5
Effect of Crevices on IGSCC of Alloy 600 in BWR Environments

The following is an item-by-item assessment of the DAEC causal factors as applied to the PWR butt-welded locations.

1. The severe 360° crevice at DAEC and the associated high stresses at the crevice tip increased the likelihood of a 360° crack. Even if the PWR butt-weld location were repaired along the full circumference on the ID, it is not as likely that a full 360° flaw would develop since an electrochemical crevice is unlikely to be established in the deaerated PWR primary water environment [4-12]. Although the possibility of multiple initiation points cannot be eliminated, the same fully circumferential condition thus does not exist in the PWR butt weld case.

2. Sulfates are the most detrimental anion in the BWR environment [4-9] and were responsible for premature IGSCC of the Alloy 600 safe ends at DAEC. Although the presence of sulfates can increase crack propagation in deaerated, neutral environments, Figure 4-6 [4-13], these and other anions, such as chloride and fluoride, do not have a detrimental effect to the same degree on PWSCC of Alloy 600 in the buffered primary water environment [4-14]. Therefore, if proper cleaning controls are maintained prior to welding and EPRI PWR water chemistry guidelines are followed during start-up, operation and shutdown, it is much less likely that sufficient impurities would be present to facilitate cracking in any local crevice.

3. If a significant number of repairs and excessive grinding were performed on the ID of a butt weld in a PWR, high stresses and abnormal stress distributions could be developed. In addition, significant grinding on the ID surface would cold work the materials and decrease time for crack initiation. However, it is not likely that cracks would grow in a uniform manner similar to the DAEC safe-end since the residual stresses associated with repairs would not be expected to be uniform around the circumference.

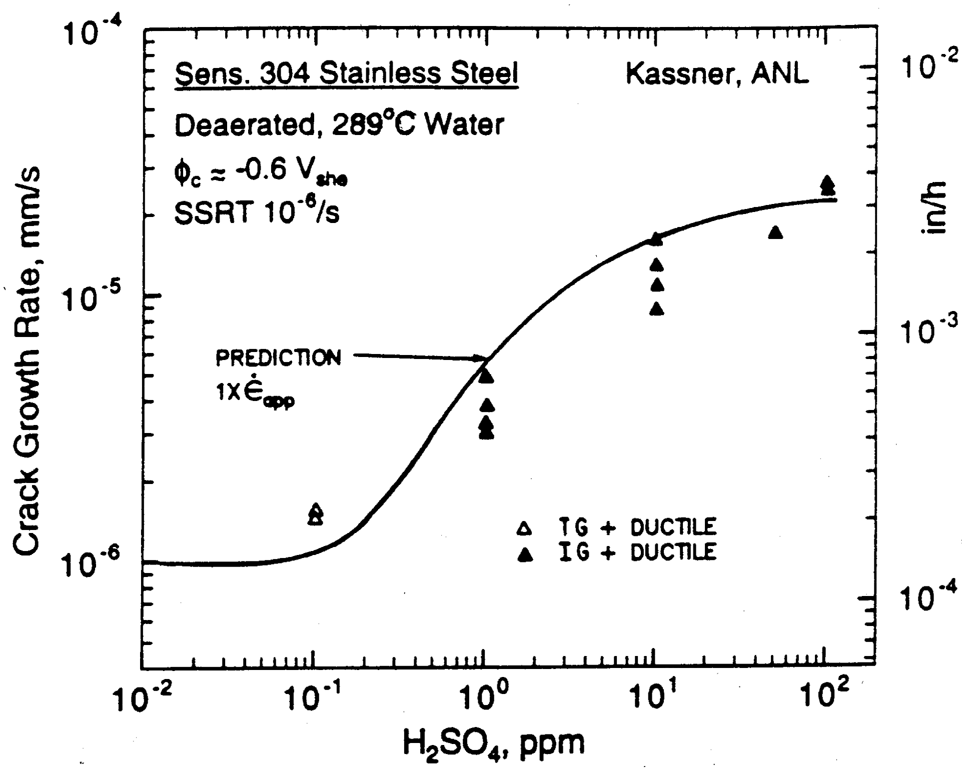


Figure 4-6
Effect of Sulfate on Crack Propagation Rates in Deaerated Water

In summary, it is concluded that cracking similar to the Duane Arnold safe-ends is not expected to occur in PWR Alloy 82/182 butt welds. Crack initiation and growth at Duane Arnold were attributed to the presence of a fully circumferential crevice that exhibited behavior like a crack initiation site. It also led to development of an acidic environment because of the oxygen in the normal BWR water chemistry. These conditions, when combined with high residual and applied stresses as a result of the geometry and nearby repaired welds, led to a condition conducive to cracking around the entire circumference of the weld root. The water chemistry conditions that contributed to cracking at Duane Arnold do not exist for the case of Alloy 82/182 butt welds in PWR plants. In addition, the butt welds in PWRs do not have the crevice geometry that existed at Duane Arnold.

4.5 Concluding Remarks

This section has shown that PWSCC flaws associated with Alloy 82/182 locations are more likely to grow through-wall first, especially in the axial direction as shown by experience, before propagating in the length direction, which will support LBB. Hence, the existence of PWSCC does not invalidate this basic premise required for LBB evaluation. Experience from the BWR industry has been used to support this finding. Although the current proposed corrosion mechanism for PWSCC, i.e., internal oxidation [4-15], is different than the identified corrosion mechanism for IGSCC in BWRs, i.e., slip-oxidation [4-16], both mechanisms involve an intergranular crack growth path where the more active grain boundary region, regardless of reason, preferentially corrodes rather than the bulk material matrix. This suggests that the results of the analysis for the crack geometry for IGSCC should apply to PWSCC and, therefore, the conclusions for the IGSCC experience can be justified for PWSCC. It is also concluded that a Duane Arnold type safe end cracking scenario is not applicable to PWR butt welds.

4.6 References

4-1. G. Moffat, et al., "Development of the Technical Basis for Plant Startup for the V. C. Summer Nuclear Plant," Proceedings of ASME 2001 Pressure Vessels and Piping Conference, ASME International, Atlanta, GA, 2001.

4-2. *Materials Reliability Program: Welding Residual and Operating Stresses in PWR Plant Alloy 182 Butt Welds (MRP-106)*, EPRI, Palo Alto, CA: 2004. 1009378.

4-3. *Materials Reliability Program: Evaluation of the Effect of Weld Repairs on Dissimilar Metal Butt Welds (MRP-114),* EPRI, Palo Alto, CA: 2004. 1009559.

4-4. *Materials Reliability Program: Alloy 82/182 Butt Weld Safety Assessments for U.S. PWR Plant Designs (MRP-113),* EPRI, Palo Alto, CA: 2004. 1009549.

4-5. *Materials Reliability Program: GE Experience Report on Cracking in Alloy 182 (MRP-57): BWR Alloy 182 Stress Corrosion Cracking*, EPRI, Palo Alto, CA: 2001. 1006603.

4-6. L. Liu, "Recirculation Inlet Safe End Repair Program," Duane Arnold Energy Center, December 8, 1978.

4-7. "Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants," NUREG-0531, February 1979.

4-8. B. M. Gordon, "Corrosion and Corrosion Control in BWRs," GE NEDO-24819A, Class 1, December 1984.

4-9. P. Andresen, "The Effects of Sulfate Impurities in 288 °C Water on the IGSCC of Inconel 600 in Constant Load and SSRT Experiments," paper #177 presented at Corrosion 84, New Orleans, LA, April 2-6, 1984, NACE, Houston, TX.

4-10. "BWR Vessel and Internals Project, BWR Water Chemistry Guideline – 2000 Revision (BWRVIP-79)," EPRI TR-103515-R2, Palo Alto, CA, February 2000.

4-11. J. W. Bagg, et al., "Chemical and Radiochemical Analysis Results from the Duane Arnold Energy Center Start-up Test Program," NEDE-13405, March 21, 1975.

4-12. P. L. Andresen and L. M. Young, "Characterization of the Role of Electrochemistry, Convection and Crack Chemistry in Stress Corrosion Cracking," paper presented at the 7th International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, August 7-10, 1995, Breckenridge, CO, NACE, Houston, TX, 1995, p. 579.

4-13. P. L. Andresen, "SCC Growth Rate Behavior in BWR Water of Increasing Purity, paper presented at the 8th International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors,'' August 10-14, 1997, Amelia Island, FL, ANS, La Grange Park, IL, 1997, p. 675.

4-14. "PWR Primary Water Chemistry Guidelines," EPRI TR-105714-R4, Palo Alto, CA, March 1999.

4-15. J. Panter, et al., "Surface Layers on Alloys 600 and 690 in PWR Primary Water: Possible Influence on Stress Corrosion Cracking Initiation," paper 02519 presented at Corrosion 2002, April 7-12, 2002, Denver, CO.

4-16. F. P. Ford, "The Crack Tip System and it's Relevance to the Prediction of Cracking in Aqueous Environments," Proceedings of First International Conference on Environmentally Assisted Cracking of Metals, Kohler Wisconsin October 2-7 1988, pp.139-165. Eds. R. Gangloff and B. Ives, NACE, Houston, TX.

5 DETERMINATION OF CRITICAL FLAW SIZES (STABILITY) AND LEAKAGE FLAW SIZES

5.1 Determination of Critical Flaw Size at Normal Operating Conditions

The critical flaw length for a through-wall flaw is that length at which under a given set of applied stresses, the flaw would become marginally unstable. Per NUREG-1061, Vol. 3 [5-1] and NUREG-0800, draft SRP 3.6.3 [5-2], the critical flaw length can be determined by elastic plastic-fracture mechanics using the J-integral/tearing modulus approach or by net-section plastic collapse (limit load). The load combination used in determining the critical flaw length is normally that due to normal full-power plant operation (pressure, dead weight and thermal expansion) plus the design basis safe shutdown earthquake (SSE). For pressurizer surge lines, stratified conditions that occur during limited periods of heatup/cooldown conditions without the earthquake loads may be controlling. These loads, as used in this evaluation for typical PWR plants, are presented in Table 5-1 [5-3, 5-4, 5-5]. For some cases, the locations used in References 5-3, 5-4, and 5-5 were not licensed for LBB, so loadings for other similar LBB locations were used.

In this evaluation, limit load analysis was employed in the determination of the critical flaw size. The use of limit load analysis assumes that the material has very good toughness. As discussed in Section 3 of this report, Alloy 82/182 weldments at normal operating conditions of PWRs have toughness very similar to that of Type 304 stainless steel base material and TIG welds. ASME Section XI flaw evaluation methodology for these stainless steels assumes limit load analysis, which therefore justifies the use of limit load analysis for Alloy 82/182 weldments at operating temperature.

The ASME Section XI limit load source equations [5-6, 5-7, 5-8] were used to determine the critical flaw sizes for circumferential and axial flaws. For through-wall flaws, the flaw depth to thickness ratio of unity was used in these equations. The Alloy 182 butter on the nozzle of most of these locations was deposited using the shielded metal arc welding (SMAW) process which would require the use of the so called Z factors in ASME Section XI in the limit load analysis. However, because of the very good toughness of these alloys at operating temperature, the use of the Z factor will result in conservative determination of the critical flaw sizes. The results of the critical flaw size determination are summarized in Table 5-2.

Table 5-1
Summary of Loads Used in the Evaluation [5-3, 5-4, 5-5][(1)]

Plant Type	Weld Locations	Plant	Geometry		Pressure (ksi)	Loads				
						Critical Flaw Size Determination			Leakage Flaw Size Determination	
			OD (in)	ID (in)		Axial Forces (kips)	Torsion (in-k)	Bending Moment (in-k)	Axial Forces (kips)	Bending Moment (in-k)
Westinghouse	Reactor Vessel Outlet Nozzle	C	34.115	29	2.25	1060	2002	44675	8.8	25057
	Reactor Vessel Inlet Nozzle	C	32.552	27.7	2.25	364	7371	12863	11.6	2728
	Pressurizer Surge Nozzle	F	15	11.88	2.25	4.5	908	3795	-3.3	2644
CE	Steam Generator Outlet Nozzle	D	28.812	24.00	2.1	5.8	1550	5287	-14.1	3251
	Steam Generator Inlet Nozzle	D	38.25	32	2.1	79	1370	19020	-174	13252
	Reactor Coolant Pump Suction Nozzle	J	36.125	29.75	2.25	0	39875*	39875	55	17107.6
	Reactor Coolant Pump Discharge Nozzle	J	36.125	29.75	2.25	0	47703*	47703	152.86	24657.6
	Surge Line (Hot Leg)	O	12.75	10.125	2.25	-0.66	-	2296	3.57	308.2
	Surge Line (Pressurizer)	O	13	10.125	2.25	-1.28	-	2201	1.73	1577
	Shutdown Cooling	O	12.75	10.125	2.25	-	-	735	-	441
	Safety Injection	O	12.75	10.125	2.25	-	-	1844	-	1372
B&W	Reactor Coolant Pump Suction/Discharge Nozzle	-	33.5	28.0	2.20	304	-	29496	256.7	21694

Notes: 1) Some loads revised from original submittals to report loads only for LBB locations

* The torque and bending loads originated in the form of a single resultant value. Given that the torque and bending moment components were not readily available; the value listed is conservatively assumed to act as both a torque and a bending moment.

Table 5-2
Summary of Critical Flaw Sizes and Reported Leakage Flaw Sizes [5-3, 5-4, 5-5][(1)]

Plant Type	Weld Locations	Plant	Geometry		Critical Flaw Size (in)		Reported Leakage Flaw Size[(2)] (in)			
							Axial		Circumferential	
			OD (in)	ID (in)	Axial Through-Wall	Circumferential Through-Wall	1.0 gpm	10 gpm	1.0 gpm	10 gpm
Westinghouse	Reactor Vessel Outlet Nozzle	C	34.115	29	27.4	25.3 (24%)	2.4	5.9	1.99	4.91
	Reactor Vessel Inlet Nozzle	C	32.552	27.7	25.9	37.2 (36%)	2.3	5.6	3.56	8.3
	Pressurizer Surge Nozzle	F	15	11.88	20.5	15.4 (33%)	2.7	5.9	2.0	4.7
CE	Steam Generator Outlet Nozzle	D	28.812	24.00	30.0	39.3 (43)	2.5	6.2	3.58	8.37
	Steam Generator Inlet Nozzle	D	38.25	32	38.5	49.1(41)	2.8	6.8	3.39	8.33
	Reactor Coolant Pump Suction Nozzle	J	36.125	29.75	38.2	36.5 (32%)	2.31	5.52	2.65	6.46
	Reactor Coolant Pump Discharge Nozzle	J	36.125	29.75	38.2	33.2 (29%)	2.31	5.52	2.24	5.52
	Surge Line (Hot Leg)	0	12.75	10.125	24.1	17.6	-	-	4.1	8.1
	Surge Line (Pressurizer)	0	13	10.125	27.3	18.6	-	-	2.7	6.0
	Shutdown Cooling	0	12.75	10.125	NA	20.3	-	-	3.7	7.5
	Safety Injection	0	12.75	10.125	NA	17.0	-	-	2.3	5.2
B&W	Reactor Coolant Pump Suction/ Discharge Nozzle	-	33.5	28.0	31.6	32.3	3.7	9.1	3.1	7.9

Notes: 1) Some flaw sizes revised from original submittals to report loads only for LBB locations

2) Westinghouse and CE leakage flaw sizes are based on fatigue morphology; B&W leakage flaw sizes are based on IGSCC morphology.

5.2 Leakage Flaw Size Determination

The leakage flaw size is that flaw size that will result in a particular leakage. In LBB evaluation, a leakage of 10 gpm has usually been assumed. This corresponds to a plant leak detection capability of 1 gpm and a margin of 10 on leakage as recommended in NUREG-1061, Vol. 3 and draft SRP 3.6.3. As will be discussed in Section 6, PWR plants are capable of detecting leakage below 1 gpm in one hour or less. Also, since the publication of NUREG-1061, Vol. 3 and draft SRP 3.6.3, PWR plant operators are more sensitive regarding leak detection and have implemented various actions to deal with leakage much smaller than 1 gpm, so lower margins on leakage or lower detection limits could be justified for some plants.

A critical aspect of LBB evaluations is the computation of leak rate to determine the leakage flaw size. The first step in the leakage computation is the determination of the crack opening area based on fracture mechanics principles. The loading combination considered for leakage per NUREG-1061, Vol. 3 and draft SRP 3.6.3 is that due to normal full-power operating loads (pressure, dead weight and thermal). The crack opening area is typically determined by assuming an elliptical shape where the crack length on the ID of the pipe is equal to the crack length on the OD of the pipe. Consistent with fracture mechanics principles for ductile materials, a plastic zone correction factor may be included in the computation of the crack opening area.

In this evaluation, the leakage flaw sizes have been determined for a leakage of 10 gpm corresponding to the assumption in the current LBB submittals and also 1 gpm to demonstrate the margins that exist under a more realistic assumption regarding plant leak detection capability.

5.2.1 Evaluation of Leakage Morphology

The leakage flaw size may be a strong function of the crack face morphology assumed in the leakage calculation. In current LBB submittals, fatigue cracks were assumed. However, with the discovery of PWSCC in Alloy 82/182 locations, the effects of PWSCC morphology may be important in the analysis to determine the leakage flaw sizes. For leakage determination with the assumption of fatigue cracks, no local turning losses are assumed in the computation. However, for the cases involving SCC, the dendritic and/or intergrannular nature of the crack surface would indicate additional local roughness and flow path turning losses in the computation of leakage. The roughness and the number of turns assumed in the computation may have a significant impact on the resulting leakage rate and leakage flaw size.

In the early 1980s, there was considerable work completed to develop and qualify methods to determine leakage rates from cracks. EPRI and Battelle Columbus worked together to conduct experiments, data correlations and analytical model development that would accurately predict leakage for both fatigue and IGSCC cracks. In the Battelle Phase II program, extensive testing, involving Alloy 82 experiments was conducted using specimens that had been cracked by IGSCC. This work has been summarized [5-11] and the PICEP model was developed. In this work, it was shown that IGSCC leakage could be predicted quite accurately if 24 45-degree turns per inch (equivalent to about 12 90-degree turns per inch) with an associated surface roughness of 200μ inches were included in the flow modeling. Both Westinghouse and AREVA have qualified their leak rate prediction models using this published data.

Over the past several years, research, funded by the NRC has been completed by Battelle Columbus and Engineering Mechanics Corporation of Columbus (EMC^2) to assess the technology used in determining leakage through cracked piping [5-9, 5-10]. One of the key issues that has been studied is the crack morphology (roughness, local flow path turns, total leakage path length) differences between fatigue cracking and SCC. For fatigue cracks, the flow path is relatively smooth and straight, whereas for SCC, the flow path consists of a relatively tortuous path. A procedure has been proposed in NUREG/CR-6004 [5-8] that defines the surface roughness, effective flow path length and number of flow path turns as a function of the ratio of the crack opening displacement (COD) to the global roughness (μ_G). For very tight cracks, there is a relatively longer flow path with many local turns, but the roughness is relatively low. For cracks with a much larger opening, the roughness is better represented by the global roughness, but the number of turns and effective flow path length is reduced. This study showed that the ratio between PWSCC leakage flaw size and air-fatigue flaw size was 1.69 [5-10]. It also reported that the ratio between IGSCC flaw size and air-fatigue flaw size was 1.89 using the proposed model. This model is based on determination of PWSCC crack morphology parameters from limited service cracks. It has not been benchmarked to the extensive Battelle Phase II experiments or field service data such as DAEC. Also, the model has not been confirmed by the proposed detailed fluid mechanics analysis [5-10].

In References 5-3, 5-4 and 5-5, calculations were conducted to determine critical and leakage size flaws (1 gpm and 10 gpm). Since some of the previously reported results were for locations that did not have LBB applied, additional evaluations were conducted for the Alloy 82/182 LBB locations to determine critical and leakage flaw sizes. For evaluation of surge lines, the critical flaw sizes were based on the limiting cases of either normal operation + SSE or startup/shutdown operation with maximum allowed pressurizer-to-hot leg stratification.

For the Westinghouse and CE plants, the leakage calculations were based on fatigue cracks. For the B&W plants, the leakage calculations were conducted based on IGSCC cracks (with number of turns as suggested in EPRI studies [5-11]), except that for this study, additional calculations were performed to determine leakage for fatigue cracks as well. It was determined that the ratio between the IGSCC and fatigue crack leakage size for the one B&W location was a factor of 1.4. The leakage flaw size factor is considerably less than the factor of 1.89 shown by EMC^2 [5-10]. In fact, the EMC^2 presentation showed that the factor would be less (1.69) for PWSCC as compared to IGSCC.

5.3 Margins Between Critical and Leakage Flaw Size

Table 5-3 presents the margins between the critical flaw sizes and the leakage flaw sizes for Alloy 82/182 locations based on fatigue (or IGSCC) morphologies. The margins were determined for leakage rates of 10 gpm and 1 gpm to determine the sensitivity of the margins to leak detection capability. As can be seen from this table, the recommended margin of two between the critical flaw size and the leakage flaw size is met for all the piping configurations when fatigue or IGSCC morphologies are considered, even with the assumption of 10 gpm leakage. The margins increase significantly when a leak rate detection capability of 1 gpm is assumed in the evaluation.

For the large main-loop piping as shown in Table 5-3, the minimum margin for a 10 gpm LBB location would be reduced from 4.5 to 2.7 if the very conservative factor of 1.69 proposed by EMC^2 were used to increase the leakage flaw size. For a 1 gpm leakage rate, the margin is reduced from 10.4 to 6.2. Thus, there is a significant margin available to account for potential effects due to PWSCC morphology for main loop piping.

Table 5-3
Summary of Reported Leakage Flaw Sizes and Margins for Circumferential Flaws [5-3, 5-4, 5-5][(1)]

Plant Type	Weld Locations	Plant	Geometry		Reported Leakage Flaw Size (in)[(1)]		Margins[(2)]	
			OD (in)	ID (in)	1.0 gpm	10 gpm	1.0 gpm	10 gpm
Westinghouse	Reactor Vessel Outlet Nozzle	C	34.115	29	1.99	4.91	12.7	5.2
	Reactor Vessel Inlet Nozzle	C	32.552	27.7	3.56	8.3	10.4	4.5
	Pressurizer Surge Nozzle	F	15	11.88	2.0	4.7	7.7	3.3
CE	Steam Generator Outlet Nozzle	D	28.812	24.00	3.58	8.37	11.0	4.7
	Steam Generator Inlet Nozzle	D	38.25	32	3.39	8.33	14.5	5.9
	Reactor Coolant Pump Suction Nozzle	J	36.125	29.75	2.65	6.46	13.8	5.7
	Reactor Coolant Pump Discharge Nozzle	J	36.125	29.75	2.24	5.52	14.8	6.0
	Surge Line (Hot Leg)	0	12.75	10.125	4.0	8.0	4.3	2.2
	Surge Line (Pressurizer)	0	13	10.125	2.7	6.0	6.9	3.1
	Shutdown Cooling	0	12.75	10.125	3.7	7.5	5.5	2.7
	Safety Injection	0	12.75	10.125	2.3	5.24	7.4	3.3
B&W	Reactor Coolant Pump Suction/ Discharge Nozzle	-	33.5	28	3.1	7.9	10.4	4.0

Notes: 1) Westinghouse and CE leakage flaw sizes are based on fatigue morphology; B&W leakage flaw sizes are based on IGSCC morphology.
2) Margin is defined as critical flaw size from Table 5-2 divided by leakage flaw size.

For the small diameter LBB locations as shown in Table 5-3, this same factor (1.69) would bring the margin down to 1.3 for 10 gpm leakage rate and to 2.5 for a 1 gpm leakage rate. However, as discussed above, this factor of 1.69 is much higher than has been justified by the earlier leak rate testing and model calibrations using actual IGSCC specimens.

The limiting location was for a plant surge line. The evaluation to predict surge line loads is known to be conservative in that the models predict maximum moments since the stratification loading is based on the most severe mid-pipe stratification level for the entire length of the surge line. Realistically, the stratification moments would be less.

This evaluation shows that for the 10gpm leakage case, all the lines with the exception of the relatively smaller surge, safety injection and shutdown cooling lines exhibited a margin of much greater than two. If a leak rate of 1 gpm is considered, all the lines far exceeded the factor of two margin. Considering the conservatisms in the evaluation and the fact that many plants are capable of detecting leakages well below the traditional 1 gpm limit (1 gpm with a margin of 10) as discussed in Section 6, it is judged that these locations are also acceptable for LBB in the presence of PWSCC.

Note that two main coolant loop piping locations (Steam Generator inlet and Steam Generator outlet) for a plant were not evaluated. These would be expected to have margins similar to the other main loop piping locations that were considered.

5.4 Consideration of Low Temperature Crack Propagation on Critical Flaw Sizes

As noted in Section 3, the fracture resistance of Alloy 82/182 welds decreases significantly at low temperatures (<300°F) in hydrogenated water. This phenomenon is called low temperature crack propagation (LTCP). The hydrogen and water conditions for LTCP may exist during cooldown, as when the temperature is below 300°F while hydrogen concentration is maintained at a significant level. Under these conditions however, the operating loads are also significantly low. In this section, the critical flaw sizes at LTCP conditions are calculated to determine if they are bounding. Because of the degradation in fracture toughness, limit load analysis is judged not to be applicable for the determination of the critical flaw sizes. As such, elastic-plastic fracture mechanics evaluation involving the J-T analysis approach is employed to determine the critical flaw sizes under these conditions. Locations for the hot leg and for a surge line for a Westinghouse plant are evaluated.

5.4.1 Determination of Loads

Stresses for normal operation are used, based on previous butt weld safety assessment work [5-3]. The normal operating stresses are modified for low temperature operating conditions. Low temperature operating moments for both locations are determined for 129ºF (54ºC) and 300ºF (149ºC), the temperatures where J-Δa curves are available. The following assumptions are made in determining low temperature operating modes.

- The loads/stresses due to dead weight are not provided separately. Since dead weight stresses are generally fairly low, it will be assumed that these are 5 percent of the normal operating dead weight (DW) + thermal expansion (TE) + thermal stratification (TS – applies only to surge line location). These loads/stresses are held constant regardless of temperature changes.
- The contributions of thermal expansion and thermal stratification (for the surge line only) are not provided separately. It is assumed that the thermal loads are the remaining 95% of the DW + TE + TS normal operating loading. For the purposes of determining the low temperature loadings, it will be assumed that the loads/stresses may be linearly interpolated between 70ºF and an assumed operating hot leg temperature of 605ºF. This is considered conservative as discussed below.
- For determining pressurizer pressure, it is assumed that the difference between the hot leg (RCS) and the pressurizer is 320ºF. This is the normal maximum allowed and observed surge line temperature differential. This assumption maximizes the pressure loads for the LTCP evaluation. Thus, the pressurizer pressure will be estimated by determining the pressurizer temperature at 320ºF above the two LTCP conditions (129ºF and 300ºF) or 449ºF and 620ºF, respectively. This yields low temperature operating pressures of 390 psig and 1770 psig, respectively.
- The special consideration for the surge line is that there is no stratification. In order for the pressurizer surge nozzle to remain at the specified cold condition, there must be an in-surge from the hot leg to the pressurizer so that the metal will be at the cold temperature. However, by ratio the total of TE + TS loads, it is conservatively assumed that there is still some stratification.

For the above analysis, it is estimated that the pressurizer surge line at the pressurizer nozzle would have to be in a continuous in-surge condition to achieve the lower of the cold conditions postulated above at the location. For the in-surge condition, the entire surge line is at the hot leg low temperature conditions. Thus, there would be no stratification for low temperature conditions at the lower of the two evaluation temperatures. This is certainly the case for the surge nozzle.

For out-surge conditions at the pressurizer surge nozzle, it would be possible to have stratified conditions for out-surge:

- For the 129ºF out-surge case, no pressure would exist, and the density difference between the hot (129ºF) water and 70ºF (ambient conditions) in the hot leg would not result in stratification.
- For the 300ºF case, there could be up to 230ºF (300ºF - 70ºF) stratification plus thermal expansion due to the average surge line temperature of 185ºF. However, the pressure would only be 53 psig due to the lower pressurizer temperature. It is judged that this case would not be controlling due to much higher fracture toughness at the higher temperature.

For a location at the hot leg, it might be possible to postulate that the surge line was stratified such that the full stratification moment would be applied simultaneously with the high pressure conditions due to the out-surge of hot water into the top of the surge line that is assumed not to heat the hot leg nozzle end. In this case, the TE stress would have to be based on $T_{hot\ leg}$ + 320/2, with 320ºF stratification (TS) loading.

Based on the above assumptions, the final stresses used in the critical flaw size determination under LTCP conditions are shown in Tables 5-4 and 5-5 for the reactor vessel outlet nozzle and the surge line nozzle. It should be noted that seismic SSE loads are not included in the load combination since the probability of an SSE during a cooldown transient is very small.

Table 5-4
Stresses Used for LTCP Evaluation of Westinghouse Reactor Vessel Outlet Nozzle

Parameter	129°F	300°F
OD (in)	34.115	34.115
ID (in)	29	29
Area (in^2)	253.6	253.6
Section Modulus (in^3)	1862.6	1862.6
Moment Stresses (ksi)		
Pressure	0.674	0.674
Thermal	1.413	5.508
Total	2.087	6.182
Axial Stresses (ksi)		
Pressure	1.050	4.611
Thermal	-0.003	-0.014
Total	1.046	4.597

Table 5-5
Stresses Used for LTCP Evaluation of Westinghouse Pressurizer Surge Nozzle

Parameter	129°F	300°F
OD (in)	15	15
ID (in)	11.88	11.88
Area (in^2)	66.4	66.4
Section Modulus (in^3)	202.3	202.3
Moment Stresses (ksi)		
Pressure	0.634	0.634
Thermal	1.328	5.179
Total	1.962	5.813
Axial Stresses (ksi)		
Pressure	0.669	2.939
Thermal	-0.005	-0.019
Total	0.664	2.919

5.4.2 Material properties

For EPFM J-T analysis, two material properties are required. The first is the J-R material resistance curve. The lower bound curves for Alloy 82 presented in Figures 3-2 and 3-3 [5-12] for the two temperatures of interest (129°F and 300°F) were used for the evaluation. The lower bound curves were curve fit to a power law function as shown in Figure 5-1.

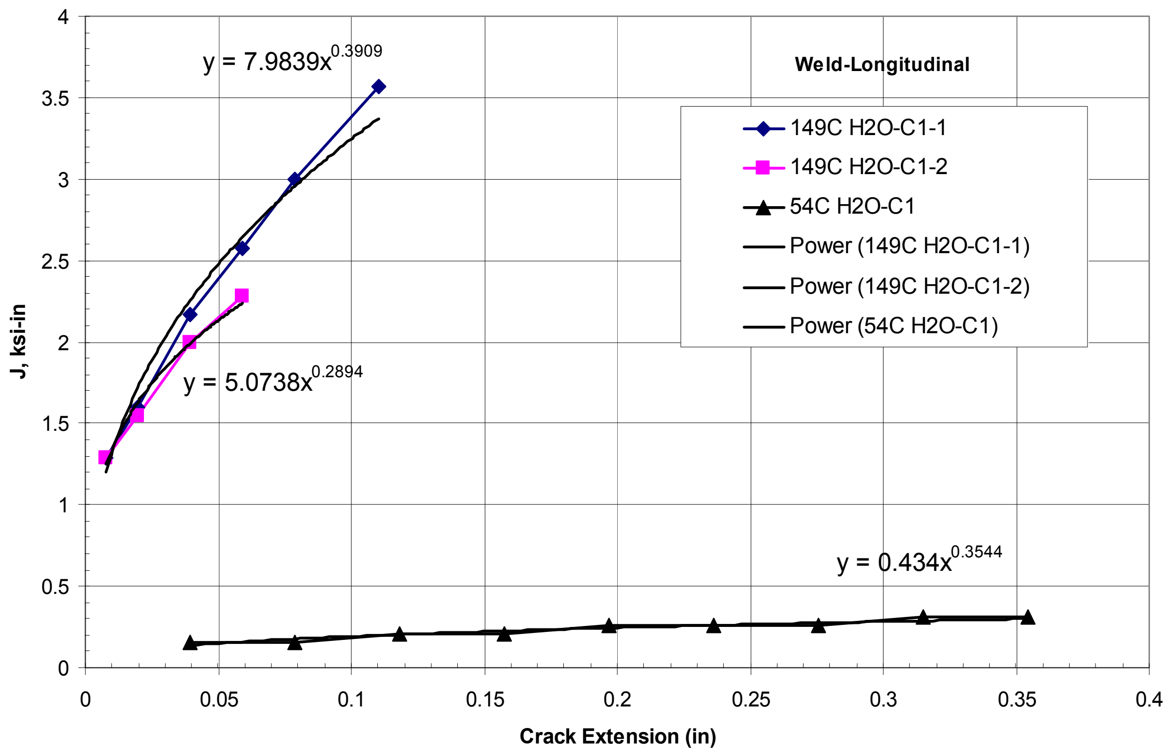


Figure 5-1
Curve Fit of Alloy 182 J-R Curves

The second material property required is the Ramberg-Osgood stress-strain curve parameters from the stress strain relationship:

$$\frac{\varepsilon}{\varepsilon_0} = \frac{\sigma}{\sigma_0} + \alpha\left(\frac{\sigma}{\sigma_0}\right)^n \tag{5-1}$$

where: ε, σ are the true strain and true stress,

ε_o, σ_o are the yield strain and yield stress, and

α, n are the Ramberg-Osgood parameters.

In general these parameters are derived if the stress-strain curve of the material is available. In the absence of the actual stress-strain curves for Alloy 82/182, the values of α and n are obtained from the relationship provided in Reference 5-14 as:

$$n = \frac{1}{\ln(1 + e_u)} \tag{5-2}$$

$$\alpha = \left[\frac{\ln(1+e_u)}{\ln\left(1+\frac{S_y}{E}\right)} - \frac{S_u(1+e_u)}{S_y\left(1+\frac{S_y}{E}\right)} \right] \left[\frac{S_u(1+e_u)}{S_y\left(1+\frac{S_y}{E}\right)} \right]^{-n} \tag{5-3}$$

where, e_u is the ultimate elongation, and E is the elastic modulus.

Actual material properties from References 5-12 and 5-13 were used in Equations 5-2 and 5-3 to calculate the Ramberg-Osgood parameters. A summary of the material properties used in the evaluation is provided in Table 5-6. To determine the sensitivity of the results to the material properties, both actual values presented in References 5-12 and 5-13 and Code minimum properties [5-15] are used in the evaluation.

Table 5-6
Material Properties Used for LTCP Evaluation

Parameter	Actual Properties			ASME Code Minimum Properties	
	54°C (129°F)	338°C (640°F)	149°C[1] (300°F)	129°F	300°F
S_y (ksi)	68.73	57.57	65.00	29.42	26.6
S_u (ksi)	87.15	82.8	85.69	75.0	75.0
% Elongation	18	32	22.7	18[2]	22.7[2]
E (ksi)	30,833[3]	28,600[3]	30,086[3]	30,555	29,900
n	8.18	4.18	6.84	8.18	6.14
α	2.87	9.49	5.23	0.022	0.11

Notes
1) Interpolated
2) No Code values. Used values from Reference 5-13
3) Values from Reference 5-14

5.4.3 EPFM J-T Evaluation and Results

The EPFM J-T evaluation for the LTCP conditions was performed using three fracture mechanics models presented in References 5-16 and 5-17 to determine the instability flaw sizes. These models are:

1. Through-wall flaw under remote tension
2. Through-wall flaw under remote bending
3. Part through-wall flaw under remote tension

The results of the evaluation are summarized in Tables 5-7. It can be seen that the critical through-wall flaw sizes are relatively large and compared to those reported in Table 5-2 at operating conditions, these critical flaw sizes at LTCP conditions are not bounding. This is because at LTCP conditions, the loads are relatively smaller and therefore compensates for the reduction in fracture resistance associated with LTCP. Hence, in spite of the reduction in fracture resistance due to LTCP, the critical flaw size at normal operating conditions is still bounding in the LBB evaluations.

Table 5-7
Critical Flaw Sizes Under LTCP Conditions

Material Properties	Nozzle	54°C			149°C		
		Half Critical Through-wall Flaw Length (in.)		Part Through-Wall Flaw Depth (in.)	Half Critical Through-wall Flaw Length (in.)		Part Through-Wall Flaw Depth (in.)
		Tension	Bending		Tension	Bending	
Actual	Outlet	> 21.8	> 25	> 2.0.	> 22.	> 25	> 2.0
	Surge	> 11	> 11	> 1.2.	> 11	> 11	> 1.2
Code Minimum	Outlet	> 25	> 25	> 2.0	> 21	> 25	> 2.0
	Surge	> 11.	> 11.	> 1.2	> 11	> 11	> 1.2.

5.5 References

5-1. The Pipe Break Task Group, "Report of the U. S. Nuclear Regulatory Commission Piping Review Committee – Evaluation of Potential for Pipe Break," NUREG-1061, Volume 3, November 1984.

5-2. U. S. Nuclear Regulatory Commission; Standard Revision Plan; Office of Nuclear Reactor Regulation, "Section 3.6.3, Leak-Before-Break Evaluation Procedure," NUREG-0800, August 1987. (Federal Register/Vol. 52, No. 167/August 28,1987/Notices pp. 32626-32633)

5-3. *Materials Reliability Program: Alloy 82/182 Pipe Butt Weld Safety Assessment for U.S PWR Plant Designs: Westinghouse and CE Design Plants (MRP-109),* EPRI, Palo Alto, CA: 2004. 1009804.

5-4. _Materials Reliability Program: Alloy 82/182 Pipe Butt Weld Safety Assessment for U.S PWR Plant Designs: Babcock and Wilcox Design Plants (MRP-112),_ EPRI, Palo Alto, CA: 2004. 1009805.

5-5. AREVA Document No. 51-5052759-00, "Safety Evaluation of Alloy 82/182 Welds in LBB Applications," February 2005.

5-6. ASME Boiler and Pressure Vessel Code, Section XI, Appendix C, 1998 Edition.

5-7. EPRI Report NP-4690-SR, "Evaluation of Flaws in Austenitic Steel Piping," July 1989.

5-8. ASME Section XI Task Group for Piping Flaw Evaluation, ASME Code, "Evaluation of Flaws in Austenitic Steel Piping," Journal of Pressure Vessel Technology, Vol. 108, August, 1986, pp. 352-366.

5-9. Rahman, S., Ghadiali, N., Paul, D., and Wilkowski, G, "Probabilistic Pipe Fracture Evaluations for Leak-Rate Detection Applications," NUREG/CR-6004, April 1995.

5-10. D. Ruland, R. Wolterman, G. Wilkowski, R. Tregoning, "Impact of PWSCC and Current Leak Detection on Leak-Before-Break," Proceedings of Conference on Vessel Head Penetration, Inspection, Cracking, and Repairs, Sponsored by USNRC, Marriot Washingtonian Center, Gaithersburg, MD, September 29 to October 2, 2003.

5-11. D. Abdollahian and B. Chexal, "Calculation of Leak Rates Through Cracks in Pipes and Tubes," EPRI NP-3395, Electric Power Research Institute, Palo Alto, CA, December 1983.

5-12. W. J. Mills and C. M. Brown, "Fracture Toughness of Alloy 600 and EN82H Weld in Air and Water," Bettis Atomic Power Laboratory Report No. B-T-3264.

5-13. Special Metals Corporation, "INCONEL Alloy 600," Publication No. SMC-027, September 2002.

5-14. T. L. Gerber, A. Y. Kuo, J. F. Copeland and D. Abdollahian, "Evaluation of High-Energy Pipe Rupture Experiments," EPRI NP-5531, January 1988.

5-15. ASME Boiler and Pressure Vessel Code, Section II, Part B, 2001 Edition.

5-16. Kumar V., et al., "Engineering Approach for Elastic-Plastic Fracture Analysis," EPRI NP-1931, Electric Power Research Institute, Palo Alto, CA, July 1981.

5-17. Kumar V., et al., Advances in Elastic-Plastic Fracture Analysis," EPRI-3607, Electric Power Research Institute, Palo Alto, CA, August 1984.

6
LEAK DETECTION

The application of LBB is predicated on a plant having adequate leak detection capabilities to provide an early warning indication to plant personnel. For the purpose of LBB application, NUREG-1061, Vol. 3 [6-1] refers to the leak detection requirements in Regulatory Guide 1.45 [6-2]. This Reg. Guide requires that at least three different detection methods be employed in the reactor. Sump flow monitoring and airborne particulate radioactivity monitoring are specifically recommended. A third method could be based on monitoring of condensate flow from air coolers or monitoring airborne gaseous activity. Many plants that have implemented LBB have met or exceeded these minimum requirements. The plant staff sensitivity to leak detection has also increased over the years such that they can address leakage at a level which is a fraction of that required by their technical specifications (generally 1 gpm).

A recent survey by Westinghouse [6-3] for plants indicated that the leak rate accuracy and/or sensitivity for most plants varied from 0.01 to 0.1 gpm. Sensitivity in this case is defined as the smallest leak rate that can be measured and/or smallest change in leak rate over a specified period of time. This is at least an order of magnitude smaller than the 1 gpm leakage typically assumed in LBB evaluation. The survey also indicated that several plants use on-line monitoring systems to track and determine leak rate. In addition, there is an increased sensitivity to unidentified leakage by plant personnel such that plants respond to leak rates before the Technical Specifications limit on unidentified leakage (generally 1 gpm in one hour) is reached. Typical responses to the survey are summarized in Table 6-1. Plants are typically trending the unidentified leak rate at much lower levels. It should be noted that the plant designations in this table are different from those in the tables in Section 5.

From these survey results, it is clear that plant personnel have become sensitive to leakage and take actions at detectable leakages significantly below 1 gpm leakage, which has traditionally been used in LBB evaluations. Although this can only be applied on a plant specific basis, it is believed that lower leakage rates can be justified by several plants in the LBB evaluation. As indicated in the previous section of this report, the margins between the critical flaw size and the leakage flaw size increases as the leakage rate used in the LBB evaluation is reduced.

6.1 References

6-1. The Pipe Break Task Group, “Report of the U. S. Nuclear Regulatory Commission Piping Review Committee – Evaluation of Potential for Pipe Break,” NUREG-1061, Volume 3, November 1984.

6-2. USNRC Regulatory Guide 1.45, “Reactor Coolant Pressure Boundary Leakage Detection Systems,” May 1973.

6-3. Proceedings of Leak Rate Monitoring Workshop, Sponsored by the Westinghouse Owner's Group (WOG Operations) Subcommittee, Foxwood Resort, Mashantucket, CT, June 21-23, 2004.

Table 6-1
Summary of Sample Survey Results

Plant[1]	Primary Detection Method	Sensitivity	Capability and if any Operating Experience
A	Continuous Online Detection Program	For very tight RCS, unidentified leak rate varied both + and – around 0 gpm Redundant inputs are averaged, uses an average of the reliable values over 10 hour leak rate calculation	The administrative requirements are in place: Leak rates greater than 0.25 gpm identified require investigation as to the source and initiation of a condition report. Leak rates greater than 0.15 gpm unidentified require investigation as to the source and initiation of a condition report. An increase of 0.05 gpm from the previous leak rate or adverse trends over time requires investigation into the cause.
B	Continuous online method using Integrated Plant Computer (IPC).	PZR level, PZR pressure, VCT level, least squares fit, 2 hour usual sample, 0.05 gpm accuracy, 0.01 sensitivity	IF RCS unidentified leakage increases >0.1 gpm in a 30 day period, leak investigation team formed. If RCS unidentified leakage increases to >0.25 gpm, CNTMT Rad monitor particulate filter removed from service and sampled for iron. Inventory balance good down to 0.05 gpm

Note: (1) The plant designations in this table are different from those in the tables in Section 5

Table 6-1
Summary of Sample Survey Results (continued)

Plant[1]	Primary Detection Method	Sensitivity	Capability and if any Operating Experience
C	Action triggers based upon unidentified leakage, cumulative volume, step change, rate of change, slow increasing leak rate, and adverse trends of indirect leakage indicators	One time test performed to check sensitivity at RCS NOP/NDT conditions.	Improved management involvement Improvements in instrumentation uncertainty associated with leak rate calculation Improvements to RCS inventory balance Three action levels set much lower than the Tech. Spec. limit of 1 gpm. Investigation into sources of leakage initiated when there is a sustained step change of 0.1 gpm.
D	Online method combined with linear regression.	Linear regression mass balance has a lower sensitivity of about 0.06 gpm; The standard deviation is also about 0.06 gpm.	Online monitoring allows for better tracking/trending. Continuous atmosphere monitor can be utilized for early detection of low RCS leakage.
E	Online monitoring with linear regression.	Normal daily fluctuation of approx. 0.05 gpm.	Linear regression method has greatly improved the stability of leak rate results. If unidentified leakage increases by > 0.2 gpm then Operations initiates procedures to track down the source of the leak.
F	Online monitoring with Integrated Computer System (ICS).	Not Available	Capable of identifying leakage of 0.12 gpm.
G	Inventory balance: a) Total leakage measurement b) Quantifiable leakage measurement.	0.09 to 0.13 gpm sensitivity.	

Note: (1) The plant designations in this table are different from those in the tables in Section 5

Table 6-1
Summary of Sample Survey Results (continued)

Plant[1]	Primary Detection Method	Sensitivity	Capability and if any Operating Experience
H	RCS leakage surveillance every 72 hours	Heightened sensitivity to leaks by maintaining the rate of unidentified leakage very low.	Investigation commences if the identified leak rate exceeds 0.3 gpm and or the unidentified leak rate exceeds 0.1 gpm.
I	On-Line Inventory Balance	Installed instrumentation: Level, Temperature and Pressure Transmitters Sensitivity: 0.01 gpm Detection Time: The measurement is performed over a two hour time frame.	The computer program uses a 10-minute average of inputs, which has greatly improved the accuracy of the data. The administrative limits were based on historical data from Plant data and data obtained from other plant.
J	RCS Inventory Balance performed manually every 72 hours. Online monitoring of containment sump inputs and containment particulate and gaseous radiation level provide alarm features to trigger RCS inventory balance outside normal schedule.	varies from +/- 0.04 to 0.07 gpm. A value greater than 0.1 gpm above historic trends is needed. Long-term changes on the order of 0.03 gpm are detectable after a sufficient number of sample data points are collected. The inventory calculations are conducted once every 72 hours unless RCS leakage monitoring systems (sump and/or containment radiation) are unavailable or there is a change in measured leakage.	The Least-Squares method attempts to improve accuracy by performing linear regression of calculated RCS mass values over time. Recent examples include elevated containment gaseous radiation levels leading to a physical search for leaks resulting in identification of gas leaks on the Pressurizer Relief Tank system and a 0.12 gpm change in the RCS inventory calculation resulting in the identification of a leaking filter housing vent valve.

Note: (1) The plant designations in this table are different from those in the tables in Section 5

Table 6-1
Summary of Sample Survey Results (continued)

Plant[1]	Primary Detection Method	Sensitivity	Capability and if any Operating Experience
K	on-line program assessable on the plant operator aid computer (OAC).	Program sensitivity / accuracy has not (i.e., cannot) been determination using process instrumentation uncertainty. A typical program run of 2 hours provide reasonable program results. The daily results are repeatable with +/- 0.02 gpm.	The established administrative limit for "unidentified" leakage is 0.1 gpm. This challenges program day-to-day repeatability (considered to be +/- 0.02 gpm with a 2-hour run of leak rate calculation). Unidentified Leakage (Tiered Action Plan) - >0.1 gpm - evaluate any system alignment changes that may have caused an increase in system leak rate >0.15 gpm - initiate a formal leak hunt activity (procedure driven) to locate and correct system leakage source(s), which includes accessible areas inside the reactor containment >0.3 gpm - schedule a power decrease (15% reactor power) to expand leak hunt activity within the reactor containment >0.6 gpm - schedule a reactor shutdown (Mode 3) to perform reactor containment walkdown >1 gpm - Tech Spec Limit (Mandatory Shutdown) Identified Leakage - >0.35 gpm - evaluate source(s) and initiate corrective action > 10 gpm - Tech Spec Limit (Mandatory Shutdown)
L	Manual inventory balance.	Sensitivity is between 0.05 and 0.10 gpm and is highly dependent on stability of the plant systems.	If leak rate rises above 0.2 gpm, we will perform another leak rate calculation and begin looking for possible sources.

Note: (1) The plant designations in this table are different from those in the tables in Section 5

Table 6-1
Summary of Sample Survey Results (continued)

Plant[1]	Primary Detection Method	Sensitivity	Capability and if any Operating Experience
M	A program on the plant process computer, which takes VCT level, pressurizer level, RCS temp, Boric Acid and Primary Water integrator readings.	Reported leakage is 0.3 – 0.5 gpm, currently getting leakages in the 0.010 gpm range based on some refinements to the current procedure.	In the process of completing a mass balance calc of all the RCSPB points we have a temperature for in the PPC. This includes the vapor and liquid spaces in the pressurizer. Preliminary results give a consistent normal leakage of 0.020 to 0.050 gpm, which we feel is pretty accurate.
N	Manual inventory balance and radiation monitoring. This includes trending the on-line inventory balance for the past 30 days.	Sensitivity is based on minimum process change of inputs by the Plant Computer. This would be about .0122% sensitivity to input point changes.	method provides reliable data that would easily indicate an adverse trend of approximately 0.05 gpm increase. Plant stability is required for accurate data. The test takes over two hours to complete, but the on-line data is normally available for information use only. Boric acid corrosion control program provides for leak identification, evaluation and repair. <u>IF</u> any abnormal leakage is detected, <u>THEN</u> perform an inspection and evaluation to identify and document the leakage path(s), any corrective actions, and the affects of the leakage <u>IF</u> unidentified leakage is determined to be ≥ 0.25 gpm, <u>THEN</u> perform an additional STP-9.0 to confirm the result. If the result is confirmed to be ≥ 0.25 gpm, then request chemistry to collect an R-67 sample for Iron analysis.

Note: (1) The plant designations in this table are different from those in the tables in Section 5

Table 6-1
Summary of Sample Survey Results (continued)

Plant[1]	Primary Detection Method	Sensitivity	Capability and if any Operating Experience
O	Two methods of on-line inventory balance. One method is a continuous monitoring program the other is a “snapshot” of data where the operator controls the starting and stopping of the program. Manual method is also available if the plant computer is unavailable. Any of the three methods can be utilized to satisfy the Technical Specification requirement.	The program sensitivity is 0.001gpm and accuracy is approximately 0.03gpm with the fluctuations in T_{HOT} causing the largest swings in the calculation. The continuous on-line method provides an operator immediate feedback on RCS leak rate, if one of the other methods is used the typical timeframe is two hours. Instrument uncertainty is not factored into the If an operator must perform a manual calculation using board indications versus plant computer data, the results can vary 0.5gpm.	Multi-channel inputs are averaged and linear regression is applied to the continuous RCS leak rate program. The Reactor Coolant System volumes are divided into three sections (T_{HOT}, T_{COLD} & T_{AVE}), density compensation is used to determine the change in lb-mass, then converted to gallons at 68°, 1 atmos. Accuracy of program during testing and validation was within 0.5%, primarily due to differences in steam and liquid properties and rounding. Instrument uncertainty is not accounted for. Total leakage greater than 0.104gpm determine the existence of primary-to-secondary leakage. Total leakage greater than or equal to 0.25gpm, attempt to identify the source, initiate a condition report. This will initiate the organization getting involved to identify the source. If RCS leakage then increases by 0.1gpm, another condition report is generated to heighten awareness. 0.104gpm based upon EPRI guidelines for identifying primary-to-secondary leakage 0.25gpm selected by plant management.

Note: (1) The plant designations in this table are different from those in the tables in Section 5

Table 6-1
Summary of Sample Survey Results (continued)

Plant[1]	Primary Detection Method	Sensitivity	Capability and if any Operating Experience
P	Inventory balance	The real answer for detection time is approximately four hours. This is the minimum expected time in the procedure. We use our permanent plant instrumentation. Accuracy is industry standard for these applications.	Procedures require more extensive evaluation prior to technical specification limits for RCS leakage. The current leak rates are discussed each morning at our operations focus meeting. Identified leak rates of 0.1 gpm foster collaboration with Engineering to evaluate and resolve (Historical sources have been demineralizer drains/vents and charging pump seal leakage.). Unidentified leak rates above 0.1 gpm would prompt other actions to localize and resolve. Do not have a formal troubleshooting plan. In the event we would see an elevated leak rate, we would develop an Operational Decision Making Issue (ODMI) action plan with trigger levels. This would be captured in condition reporting system.
Q	On line inventory balance	Accuracy is +/- .05 but results are given to .001.	Linear regression is used on one minute averaged input. The model provides an updated result each minute so the operator can see if the result is settling on a good result before the calculation is terminated. Action plan is in place. the plan has progressive action up to shutdown for leakage from 0.1 gpm to the tech spec limit. Containment entry and inspection is made for any leak of 0.15 gpm.

Note: (1) The plant designations in this table are different from those in the tables in Section 5

Table 6-1
Summary of Sample Survey Results (continued)

Plant[(1)]	Primary Detection Method	Sensitivity	Capability and if any Operating Experience
R	Plant process computer inventory balance.	Accuracy of about 0.2 gpm. A minimum duration of 4 hours is required.	"Reactor Coolant Leak" procedure provides a structured method for diagnosing and responding to RCS leaks. "Containment Entry" procedure provides a structured method for performing RCS leakage investigations of accessible areas inside containment during power operation. Unidentified leak rate is typically very small and operators are sensitive to changes especially when there are also other indications of a potential RCS leak such as CTMT rad monitors or CTMT sump level change. These instruments are best used for trending purposes and to trigger a check of other indications of RCS leakage. Experience has shown this to be true. CTMT particulate rad monitors have shown to be a good early warning indicator of even small RCS leaks (between 0.05 and 0.1 gpm). Walk downs performed during plant shutdowns have confirmed only very small leaks.

Note: (1) The plant designations in this table are different from those in the tables in Section 5

Table 6-1
Summary of Sample Survey Results (continued)

Plant[1]	Primary Detection Method	Sensitivity	Capability and if any Operating Experience
S	Plant Process Computer inventory balance. Radiation monitors Sump pump run times	Present mean is 0.05 gpm with one standard deviation of 0.06 gpm.	The inventory balance provides a good indication of RCS leakage. However, for accuracy to the second decimal place, a month of reading is required. T. S. limits are 10 gpm identified and 1 gpm unidentified. Maintenance Rule limits are 7.5 gpm and .75 gpm. Procedural triggers are a sudden increase of 0.2 gpm or an absolute of 0.6 gpm. The system engineer initiates an investigation if the average leak rate goes above 0.1 gpm.
T	Computer based mass balance, manually calculated mass balance, Radiation monitors are provided to detect increases, Operations walkdown if conditions do not permit the computer or manual leak rate calculation.	Uncertainty in computer-based measurement is considered to be ~ 0.1 gpm.	Linear regression is not used. Both computer and manual-based calculation have an acceptance criterion that identified RCS leakage be less than 9.0 gpm (T.S. limit is 10 gpm). If unidentified leakage increases by > 0.2 gpm then Operations initiates procedures to track down the source of the leak. An Operations procedure is used to identify the source of the leakage.

Note: (1) The plant designations in this table are different from those in the tables in Section 5

Table 6-1
Summary of Sample Survey Results (continued)

Plant[1]	Primary Detection Method	Sensitivity	Capability and if any Operating Experience
U	Currently, performs a manual hand calculation over a 3 hour period to determine PCS leakage. During transient periods a one hour leak rate is utilized.	Monitor the 3 hour leak rate with a 0.1 gpm visible band and the 1 hour (usually where the variable time interval is set)at a .15 gpm band.	Use containment sump fill rate as a check value to PCS leak rates. We have a rate determination for the fill of the sump. We also use containment air monitor as a rough means for determining the changes in the leak rate. Use linear regression for the snapshot leak rate calculation. Monitor the leak rate on shift using the plant process computer for trending purposes. The hand calculation is performed on a every other day basis. Off normal procedure provides direction to shutdown the plant at half of the TS limit. Also required to do additional monitoring and additional leak rate calculations if unidentified leakage is greater than 0.1 gpm.
V	Manual RCS inventory balance	Accuracy/sensitivity has never been calculated, but repeatability is assessed at approx 0.15 gpm. Monitoring of containment conditions (moisture, pressure, activity, RBFC standpipe levels etc) under the Boric Acid Corrosion Prevention Program is very sensitive and has detected leakage rates of a few drips per minute.	The method uses plant indications that are averaged within the plant computer system. Linear regression is not used. Density compensation is used, but only when a significant temperature change has occurred in the RCS, VCT, RCDT, etc. Considering moving to an on line computer based inventory balance.

Note: (1) The plant designations in this table are different from those in the tables in Section 5

Table 6-1
Summary of Sample Survey Results (concluded)

Plant[1]	Primary Detection Method	Sensitivity	Capability and if any Operating Experience
W	On-line inventory balance when plant process computer is available, or manual calculation otherwise.	Least Squares or Point-to-Point method (user choice) If using the Least Squares method, Data scattering Is approximately +/- 0.02 gpm and Sensitivity is about 0.005 gpm. The Point-to-Point method yields inconsistent results, depending on quality of the data. In general this method is less accurate.	Use Trend Charts to validate computer calculation. Currently the program is run on demand. This has some limitation since optimum data stability is not guaranteed. This results in some data scattering. We investigate when the leakage rate approaches 0.20 GPM. At 0.60 gpm, we will schedule a shutdown. We need to determine if the unknown leak is a RCS boundary leakage. If it is a RCS pressure boundary leak, the leak can be found even if it is as small as 0.10 gpm.

Note: (1) The plant designations in this table are different from those in the tables in Section 5

7
CRACK GROWTH EVALUATIONS

One of the important aspects of LBB is that when a leak occurs, there is adequate time for the plant to take appropriate action before the critical through-wall flaw length is reached. Reg. Guide 1.45 requirements for detection in one hour are not necessary for assurance of LBB. This section summarizes results of crack growth evaluations performed to determine the time interval from the leakage flaw size to the critical flaw size.

In most of the LBB submittals to date, PWSCC has not been considered in the evaluation since PWR butt welds were considered immune to this mechanism. Hence, crack growth was calculated mainly due to fatigue. There are a few cases where both fatigue and SCC have been considered. The evaluation typically involves the assumption of a flaw whose size is equal to that of the acceptance standards of ASME Code, Section XI, IWB-3500. All the design transients in the system's Design Specification are considered to show that crack growth is acceptable especially when combined with the plant in-service inspection (ISI) program.

As PWSCC has been observed in Alloy 82/182 butt welds, PWSCC growth has to be considered in addition to fatigue crack growth. A considerable amount of work has been performed by the MRP Butt Weld Working Group on crack growth of surface connected cracks in the through-wall direction [7-1, 7-2, 7-3], which provide insight to the contribution to crack growth by PWSCC. In addition, the growth of through-wall cracks from the leakage flaw size to the critical flaw size performed in these reports is also used to determine if there is adequate time for a plant to take remedial action.

Both PWSCC and fatigue mechanisms were considered in the crack growth evaluations presented in References 7-1, 7-2, 7-3 and 7-4. Crack growth for PWSCC was calculated in References 7-1, 7-2 and 7-3 using the MRP Alloy 82/182 crack growth rate model from MRP-21 [7-5], while later work in Reference 7-4 used the crack growth rate model from MRP-115 [7-6]. The stresses used in the evaluation included the normal sustained operating stresses (pressure, deadweight, and thermal) in addition to weld residual stresses. The fatigue crack growth analysis was performed using the model of Chopra, et al. [7-7] for Alloy 82/182. All relevant design transients were considered in the evaluation. Details of the PWSCC and fatigue crack growth evaluations are provided in References 7-1 and 7-2.

The results of the crack growth evaluations are shown in Table 7-1 for PWSCC at the various Alloy 82/182 locations for Westinghouse and CE plants. Two types of evaluations are reported. First, the time to grow from a part through-wall circumferential flaw to a through-wall flaw with two different aspect ratios was calculated. Second, the time for through-wall flaws with 1 gpm and 10 gpm leakage flaw lengths to grow to the critical flaw sizes determined in Section 5 was calculated. The results of these calculations are presented in Table 7-1. The results show that for the reactor vessel nozzles, it takes at least 5.2 years for these initial flaws to reach critical flaw size. For the surge line, it takes at least 1.1 years. These relatively long time periods indicate that there is sufficient time for a plant to take action when a leak is detected before it reaches the critical flaw size. In fact, plants are required by their Technical Specification to take action within 24 hours following identification of a 1 gpm leakage.

Table 7-2 shows the results of fatigue crack growth analysis for the outlet nozzle safe end for Westinghouse Plant C. As can be seen from this table, fatigue crack growth is not a concern even for very deep flaws.

For the B&W plants, the evaluation was performed by combining fatigue and SCC for circumferential through-wall flaws with an aspect ratio of 6:1. The results are presented in Table 7-3. As can be seen from this table, there is an adequate time period for plants to take action when leakage is detected before crack growth results in a length equal to the critical flaw size.

7.1 References

7-1. *Materials Reliability Program: Alloy 82/182 Pipe Butt Weld Safety Assessment for U. S. PWR Plant Designs: Westinghouse and CE Design Plants (MRP-109),* EPRI, Palo Alto, CA: 2004. 1009804.

7-2. *Materials Reliability Program: Alloy 82/182 Pipe Butt Weld Safety Assessment for U. S. PWR Plant Designs: Babcock and Wilcox Design Plants (MRP-112),* EPRI, Palo Alto, CA: 2004. 1009805.

7-3. *Materials Reliability Program: Alloy 82/182 Butt Weld Safety Assessment for U. S. PWR Plant Designs (MRP-113),* EPRI, Palo Alto, CA: 2004. 1009549.

7-4. AREVA Document No. 51-5052759-00, "Safety Evaluation of Alloy 82/182 Welds in LBB Applications," February 2005.

7-5. *Materials Reliability Program: Crack Growth of Alloy 182 Weld Metal in PWR Environments (MRP-21),* EPRI. Palo Alto, CA: 2004. 1000037.

7-6. *Materials Reliability Program: Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 82, 182, and 132 Welds (MRP-115),* EPRI. Palo Alto, CA: 2004. 1006696.

7-7. O. K. Chopra, W. K. Soppet, and W. J. Shack, "Effects of Alloy Chemistry, Cold Work, and Water Chemistry on Corrosion Fatigue and Stress Corrosion Cracking of Nickel Alloys and Welds," NUREG/CR-67221, April 2001.

Table 7-1
SCC Growth Results for Westinghouse and CE Plants [7-1]

Plant Type	Weld Location	Limiting Plant	Period from Assumed SCC Initiation to Through-Wall (years)		Period from a 1 or 10 gpm Through-Wall Crack to Critical Flaw Length (years)		Critical Circumferential Through-Wall Flaw Length (in.)
			Aspect Ratio 6:1	Aspect Ratio 2:1	1 gpm	10 GPM	
Westinghouse	Reactor Vessel Outlet Nozzle	C	2.9	9.8	11.9	5.2	25.3
	Reactor Vessel Inlet Nozzle	C	>40	>40	>40	>40	37.2
	Pressurizer Surge Nozzle	F	1.4	3.9	2.6	1.1	15.4
CE	Reactor Coolant Pump Suction Nozzle	J	27.0	>40	>40	>40	36.5
	Reactor Coolant Pump Discharge Nozzle	J	19.7	>40	>40	38.5	33.2
	Surge Line (Hot Leg)	M	8.8	13.8	14.2	2.7	10.3
	Surge Line (Pressurizer)	N	3.9	6.5	5.3	1.6	14.2

Notes:

1. Aspect ratio defined as: Flaw length: Flaw depth.
2. Through-wall defined as producing either 1 gpm or 10 GPM leak.
3. Critical condition occurs prior to 10 GPM. Refer to time from 1 gpm to critical condition.

Table 7-2
Example Fatigue Crack Growth Results for the Reactor Vessel Outlet Nozzle Safe End Weld Region for Westinghouse Plant C (Circumferential Flaw length: Flaw depth = 6:1, Wall thickness = 2.35 in.) [7-1]

Initial Crack Depth (inch)	Crack Depth After			
	10 years	20 years	30 years	40 years
0.6	0.61	0.62	0.63	0.64
0.8	0.82	0.84	0.86	0.89
1.0	1.04	1.08	1.12	1.17
1.2	1.25	1.32	1.39	1.47
1.4	1.48	1.57	1.67	1.78
1.6	1.70	1.80	1.90	1.99
1.8	1.90	1.98	2.07	2.14
2.0	1.08	2.14	2.20	2.25
2.2	2.25	2.29	2.32	2.35

Table 7-3
Combined Crack Growth Results B&W Plants – Circumferential Through-Wall Flaws

Plant	Weld Location	Period from Assumed SCC Initiation to Through-wall (years) Aspect Ratio 6:1	Period from a 1 or 10 gpm Through-Wall Crack to Critical Flaw Length (years)		Critical Flaw Length of Circumferential Through-Wall Flaw (in.)
			1 gpm	10 gpm	
-	Reactor Coolant Pump Suction/Discharge Nozzle	17	35.1	22.6	32.3

8 ROLE OF IN-SERVICE INSPECTIONS IN LBB EVALUATION

In-service inspection (ISI) is a key element of the LBB process. A successful LBB implementation can only be assured if ISI and/or leak detection provide early indication of flaws or leaks in the system. Alloy 82/182 weld locations have been part of plants' ISI programs. Although recent risk-informed in-service inspection (RI-ISI) programs using NRC accepted Code Cases N-560, N-577 and N-578 have eliminated some of the Alloy 82/182 weld locations from the inspection programs due to low risk and consequence of failure effects, the inspection of these welds as required by the Alloy 82/182 butt weld Inspection and Evaluation (I&E) guidelines [8-1] further complements the overall LBB approach.

8.1 Inspection and Evaluation Guidelines for Alloy 82/182 Locations

With the consideration of PWSCC as a crack growth mechanism, the MRP is in the process of developing I&E guidelines for PWSCC susceptible butt weld locations [8-1]. These guidelines among other things provide the frequency for inspection of the Alloy 82/182 welds based on the category of the weldment. The philosophy used in the development of these guidelines is very similar to that provided by the NRC in NUREG-0313, Rev. 2 [8-2] for the BWRs for effective management of IGSCC in the susceptible stainless steel piping welds. The I&E guidelines proposed by the MRP assures that in-service inspection plays an integral role in LBB evaluation and provides further assurance that the whole LBB process has a defense in depth philosophy.

8.2 Volumetric Examination of Alloy 82/182 Locations

Alloy 82/182 locations in PWRs are dissimilar metal (DM) welds, classified as ASME Section XI category B-F and B-J piping welds. As required by ASME Section XI, they are inspected by ultrasonic examinations every 10 years. These dissimilar metal welds pose an inspection challenge due to the microstructure of the weld combined with access constraints and weld geometry features.

The need for improving ultrasonic examination technology for austenitic piping, including DM weldments, multiple material types, and microstructures in the scan path, became evident during the early 1980s when extensive stress corrosion cracking was discovered in BWR stainless steel piping systems [8-3]. During this period, several international round robin exercises were completed [8-4] that showed large scatter in the performance among inspection teams. This experience created an impetus to improve ultrasonic examination technology. Also at this time, formal requirements for demonstrating the performance of inspection procedures and personnel came into effect, but only for BWR piping inspections. The BWR piping examination [8-5] experience spurred improvements of ultrasonic testing (UT) instrumentation, procedures, and personnel training and performance was formally assessed and documented. Since no instances of similar cracking had been reported in PWR units, there was no corresponding effort to demonstrate performance for PWR piping inspection at that time [8-6]. However, the UT technology improvements that came from the BWR experience contributed to improving the technology applied to PWR units, although there were no regulatory requirements at the time to demonstrate capability for PWR applications [8-7].

General performance demonstration requirements first appeared as Appendix VIII to the 1989 Addenda of Section XI of the ASME Boiler and Pressure Vessel Code [8-4]. Appendix VIII requires demonstration of the capability to detect, discriminate, and size defects by examination of realistic mockups containing intentional defects with well-known size and location. Essential variables used in the performance demonstrations are recorded and become part of the qualification record. Supplements in Appendix VIII address specific components such as piping welds, vessel welds, vessel nozzles, bolting, etc. Supplement 10 of Appendix VIII addresses UT of dissimilar metal welds, and was incorporated into 10 CFR50.55a requiring implementation by November 22, 2002. All dissimilar metal weld examinations after that date have been required to be performed with Appendix VIII qualified procedures and personnel. Thus, incorporation of Supplement 10 into the rule introduced formal performance demonstration requirements for the PWR and BWR piping DM weld inspections.

Discovery of a leak from the V.C. Summer hot leg weld in 2000, and the associated UT and eddy current testing (ECT) experience, showed that the geometry of the weld can dramatically affect the reliability of UT for examinations conducted from the inside surface of the pipe. Other experience, including Supplement 10 qualification results, confirmed the importance of knowing the weld configuration to enable adequate preparation for the examination. For examinations performed from the outside surface, the weld and nozzle geometry, and the roughness or waviness of the surface, have a particularly strong influence on the examination effectiveness.

The industry responded to these events with further improvements of UT technology coupled with intense efforts to qualify procedures and personnel to Supplement 10 for PWR applications. The qualification to Supplement 10 was modified to include challenging weld configurations such as were encountered at V. C. Summer to ensure that procedures and tooling address the range of inside surface contours. These experiences have identified the most effective techniques and practices and these practices have been incorporated into production examination procedures [8-8]. In many situations, procedures and equipment in place prior to Supplement 10 implementation had to be modified to improve performance to meet the new requirements. Another practical outcome of implementation of Appendix VIII, in addition to documentation of performance relative to standards, is formal documentation of procedure limitations. That is, the qualification record specifically documents the range of conditions, such as surface roughness or waviness, for which the procedure is qualified. This enables the licensee to identify where the procedures would not be effective and allows assessment and application of alternatives to address the limitations. This kind of formal documentation was not available prior to implementation of Appendix VIII. The most significant limitations pertain to surface conditions and weld configurations that preclude effective scanning. Licensees can assess the applicability of qualified procedures only if the site-specific surface conditions and as-built weld configurations are known.

In summary, while volumetric inspections prior to about 2002 may not have had the same detection capability or pedigree as inspections performed subsequent to the implementation of Appendix VIII, Supplement 10, they have provided some assurance, in combination with the results of visual and surface examinations, that PWSCC is not widespread in dissimilar metal welds. Implementation of Supplement 10 to Appendix VIII has resulted in development and application of improved procedures for UT detection and characterization of PWSCC in pipe butt welds. Structural integrity assessments can be made with confidence for those situations in which a qualified UT procedure can be applied. The improvement in UT capabilities, combined with the inspection intervals provided in the I&E guidelines for these weldments provides the assurance that the overall inspection program for Alloy 82/182 locations plays an integral role in the application of LBB to these locations.

8.3 References

8-1. *Materials Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guidelines (MRP-139) (DRAFT),* EPRI, Palo Alto, CA: 2005.

8-2. Hazelton, W. S, and Koo, W. H., "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping", NUREG-0313, Revision 2, Nuclear Regulatory Commission, January 1988.

8-3. USNRC, "Stress Corrosion Cracking in Thick-Wall Large-Diameter, Stainless Steel, Recirculation System Piping at BWR Plants," IE Bulletin 82-03, October 14, 1982.

8-4. L. Becker, "Performance Demonstration-25 Year of Progress," Proceedings of the Third International Conference on NDE in Relation to Structural Integrity for Nuclear and Pressurized Components, 14-16 November 2001. Seville, Spain.

8-5. Letter R. DeYoung, USNRC to G. Neils, BWROG, "Coordination Plan for NRC/EPRI/BWROG Training and Qualification Activities of NDE Personnel," August 24, 1984.

8-6. USNRC, "Ultrasonic Inspection Techniques for Dissimilar Metal Welds," Information Notice 90-3, May 1, 1990.

8-7. "Examination of Dissimilar Metal Welds in BWR and PWR Piping Systems," EPRI, Palo Alto, CA.: 1993. TR-102148.

8-8. "Dissimilar Metal Weld Examination- Guidance and Technical Basis for Qualification," EPRI, Palo Alto, CA: 2003. 1008007.

9
MARGINS AND UNCERTAINTIES IN LBB EVALUATION

There are two areas in both NUREG-1061, Vol. 3 [9-1] and NUREG-0800, draft SRP 3.6.3 [9-2] where margins have been specified to account for safety and uncertainties in the LBB evaluation. First, there is a recommended margin of two between the critical flaw length and the leakage flaw size. In addition, there is a recommended margin of ten between the detectable leakage at the plant and the leakage assumed in calculating the leakage flaw size. These margins have been used in this evaluation. In addition to these margins, several conservative assumptions have been used to perform the evaluation. For instance, as noted in Section 3, the material properties used in the limit load analysis to determine the critical flaw sizes are typically lower than the actual values. The use of actual properties would have resulted in much larger critical flaw sizes, thus providing much larger margins than reported in this evaluation. Also, the guidance provided in NUREG-1061, Vol. 3 and draft SRP 3.6.3 regarding load combinations is such that secondary loads (e.g., thermal expansion loads and stratification related loads) are included as primary loads. Since stratification loads can exceed primary loads by a factor of three or more, this is very conservative and not consistent with ASME Sections III and XI philosophy for design and flaw evaluation of piping systems where lower safety factors are used for secondary loads. The development of a crack in a piping system will result in a “kink” angle, which will relieve a large fraction of the secondary loads and limit crack extension forces. Hence, application of LBB to systems such as the surge line where there are significant secondary thermal stratification loads is conservative.

Further, the explicit margins in NUREG-1061, Vol. 3 and draft SRP 3.6.3 are conservative when applied to modern day LBB evaluations. Since the publication of these documents, there have been advances in fracture mechanics evaluation and materials characterization that have reduced some of the conservatism inherent in the fracture mechanics and leak rate determinations. In addition, the leak rate margin is higher than necessary since plants routinely shut down when detecting leakage of 1 gpm or less, and inclusion of the conservative roughness/turns approach to address PWSCC inherently includes some of the margins in leakage rate that were originally required. In addition, plants are more sensitive to leak detection than they were at the time these documents were published. Thus, the combined margins are overly conservative for modern day LBB evaluations. The NRC in NUREG-1061, Vol. 3 on page ES-2 provided an avenue for addressing the margin on leakage by stating, “Licensees and applicants have the option of requesting a decrease in leakage margin provided that their leakage detection systems are sufficiently reliable, redundant, diverse and sensitive.” In addition, margins on loads and margins in general used in LBB evaluations are discussed in Section 5.10 of NUREG-1061, Vol. 3. It is stated in that section, “Thus specific margins recommended in the previous paragraphs could be modified provided that equivalent conservatisms are included elsewhere in the LBB approach. It is the Task Group’s opinion that the NRC staff should have the flexibility to use engineering judgments on a case-by-case basis.”

Recognizing the conservatisms in the LBB evaluations, the NRC staff has accepted margins of slightly less than two on the critical-to-leakage flaw size in some previous LBB submittals [9-3]. It is stated on page 11 of Reference 9-3, "However, in previous LBB evaluations, the staff has concluded that margins of slightly less than two on critical-to-leakage flaw size are acceptable provided that a full margin of 10 is maintained on the leakage uncertainty. It is the staff's position that relaxation from the guidance written in 1984 on this point is acceptable based on the work which has been completed in the areas of piping fracture (e.g. the International Piping Integrity Research Group (IPIRG) work) and the evaluation of minimum material properties to more appropriately bound the behavior of primary system piping materials."

In spite of the conservatism discussed above, the margins discussed in Section 5.3 indicate that LBB will be assured at these Alloy 82/182 locations and as such, there is no technical basis that indicates the current licensing basis for plants with existing LBB applications should be modified.

9.1 References

9-1. The Pipe Break Task Group, "Report of the U. S. Nuclear Regulatory Commission Piping Review Committee – Evaluation of Potential for Pipe Break," NUREG-1061, Volume 3, November 1984.

9-2. U. S. Nuclear Regulatory Commission; Standard Revision Plan; Office of Nuclear Reactor Regulation, "Section 3.6.3, Leak-Before-Break Evaluation Procedure," NUREG-0800, August 1987. (Federal Register/Vol. 52, No. 167/August 28,1987/Notices pp. 32626-32633).

9-3. Letter for Ronald B. Eaton (USNRC) to R. P. Necci (Northeast Nuclear Energy Company), "Staff Review of the Submittal by Northeast Nuclear Energy Company to Apply Leak-Before-Break Status to the Pressurizer Surge Line, Millstone Nuclear Power Station, Unit 2 (TAC No, MA4126)," with Enclosure: Safety Evaluation, dated May 4, 1999.

10
SUMMARY AND CONCLUSIONS

Based on the evaluations presented in the preceding sections of this report, the following summarizes the application of LBB to Alloy 82/182 locations in PWRs.

- A comprehensive evaluation was performed to identify all the Alloy 82/182 locations in PWRs and those locations for which LBB application has been submitted or in the process of being submitted. The evaluation indicated that the piping systems that have Alloy 82/182 locations for which LBB has been applied include the main reactor coolant loop nozzle-to piping welds, the surge line connections to the pressurizer, the surge line, shutdown cooling and safety injection A comprehensive evaluation was performed to identify all the Alloy 82/182 locations in connections to the main reactor coolant loop piping. The RCS main loop piping for all three vendors is in excess of 30 inches outside diameter and the surge line piping is between 12 inches and 15 inches nominal diameter. No piping smaller than 12 inches, which contain Alloy 82/182, have been qualified for LBB. The importance of this observation is that generally LBB is difficult to qualify for pipe sizes below 12 inch nominal OD which provided some optimism that even in the presence of Alloy 82/182 welds at these locations LBB will still be justified.
- The only change from existing LBB evaluations that needs to be addressed for Alloy 82/182 locations is consideration of PWSCC in these alloys. In this respect, three issues need to be revisited:
  - Will the presence of PWSCC assure that cracks will grow in the through-wall direction before growing in the circumferential direction such that crack profiles consistent with the DAEC safe end IGSCC pattern in the early 1980s will not result?
  - Will the morphology associated with PWSCC have a significant impact on the leakage rate calculation (or leakage flaw sizes) so as to affect the LBB margins and conclusions?
  - Will the crack growth associated with PWSCC in Alloy 82/182 materials affect leak detection before the critical flaw size is reached?
- An evaluation was performed to determine the growth direction of flaws associated with PWSCC. It was determined that based on both experimental studies and field behavior that flaws resulting from PWSCC will most likely grow in the through-wall direction and result in leak before growing in the length direction thus assuring LBB.
- Even though the PWSCC morphology has an effect on the leakage rate calculation by increasing the leakage flaw size, adequate margins are still maintained between the critical flaw size and the leakage flaw size for all piping.
- The time to grow a flaw from a leakage size flaw to a critical flaw size is on the order of years. This shows that there is adequate time to detect leaks without concern for imminent pipe rupture.

Other observations and conclusions from the work reported in this report are as follows:

- Critical flaw sizes and leakage flaw sizes were determined for the various Alloy 82/182 locations. The critical flaw sizes were determined using limit load (net section plastic collapse) since it was established in this report that Alloy 82/182 materials have very high toughness consistent with that of forged stainless steel base metal and TIG welds at the operating temperature of PWRs. In spite of this observation, thermal loads were conservatively included in the load combination for determining the critical flaw sizes. The through-wall critical flaw sizes were found to be relatively large (in excess of 16 inches in the axial direction and greater than 20% of circumference in the circumferential direction).
- For LBB evaluations, circumferential flaws are more critical than axial flaws since, for axial flaws, PWSCC is limited to the width of the weld, which is very small in comparison to the critical flaw sizes calculated for the axial flaws.
- Leakage flaw sizes for butt weld circumferential flaws were determined using fatigue and IGSCC morphologies. An assessment of margins was also performed using conservative PWSCC morphologies. There is an increase in the leakage flaw size when IGSCC or PWSCC morphology is considered relative to fatigue.
- The margin between the critical flaw size and the leakage flaw size is greater than two for the piping considered in this evaluation if fatigue or IGSCC morphologies are assumed and assuming a leak rate of 10 gpm (detectable leakage of 1 gpm and a margin of 10 on leakage). When PWSCC morphology factor of 1.69 is considered relative to fatigue, all main loop piping locations meet the critical-to-leakage flaw size margin of two; for smaller lines, the minimum margin was 1.3 (for one plant). When a leak rate of 1 gpm is assumed in the evaluation, all lines far exceed the margin of two when PWSCC morphology is considered.
- Plant staff has become more sensitive with regard to leak detection and several plants take action long before the Technical Specification limit of 1 gpm is reached. Several plants can detect leakage far lower than 1 gpm. Over longer periods, there is more assurance that low leakage rates could be detected. If this lower leak detection limit is used, all the Alloy 82/182 locations considered in this report will have a margin of at least two between the critical flaw size and the leakage flaw size.
- Due to the increased sensitivity to leak detection at most plants, it is believed that the margin of 10 on leakage in the existing LBB submittal is conservative and the reduction of this margin will further justify LBB for these Alloy 82/182 locations in the presence of PWSCC.
- Inspection guidelines have been developed to ensure that cracks will be detected long before they reach critical sizes adding further conservatism to application of LBB to these Alloy 82/182 locations.
- Other potential degradation mechanisms that can invalidate application of LBB to Alloy 82/182 locations were evaluated and it was concluded that only low temperature crack propagation and thermal aging and reduction of toughness at the fusion line could have any impact on the LBB evaluations. The effect of these mechanisms on the critical flaw size was evaluated and was found to have no adverse effects on the LBB evaluations.
- There are several conservatisms inherent in the LBB evaluations such as the use of lower bound material properties and the treatment of secondary load such as thermal expansion and stratification related loads as primary loads.

- The MRP is developing inspection guidelines for Alloy 82/182 locations in PWRs that will ensure that inspections support continued application of the LBB approach for these locations.

Based on the above observations and the conservatisms inherent in the analysis, it is concluded that there is no concern for LBB applied to Alloy 82/182 locations in PWRs. The main loop piping system has critical-to-leakage flaw size margins of at least two when PWSCC morphology is considered and therefore qualifies for LBB. Even though some of the smaller lines have critical-to-leakage flaw size margins slightly less than two, they also qualify for LBB in the presence of PWSCC in light of the conservatisms inherent in the evaluation, which has been recognized by the NRC staff in previous LBB submittals.

WARNING: This Document contains information classified under U.S. Export Control regulations as restricted from export outside the United States. You are under an obligation to ensure that you have a legal right to obtain access to this information and to ensure that you obtain an export license prior to any re-export of this information. Special restrictions apply to access by anyone that is not a United States citizen or a permanent United States resident. For further information regarding your obligations, please see the information contained below in the section titled "Export Control Restrictions."

Export Control Restrictions

Access to and use of EPRI Intellectual Property is granted with the specific understanding and requirement that responsibility for ensuring full compliance with all applicable U.S. and foreign export laws and regulations is being undertaken by you and your company. This includes an obligation to ensure that any individual receiving access hereunder who is not a U.S. citizen or permanent U.S. resident is permitted access under applicable U.S. and foreign export laws and regulations. In the event you are uncertain whether you or your company may lawfully obtain access to this EPRI Intellectual Property, you acknowledge that it is your obligation to consult with your company's legal counsel to determine whether this access is lawful. Although EPRI may make available on a case-by-case basis an informal assessment of the applicable U.S. export classification for specific EPRI Intellectual Property, you and your company acknowledge that this assessment is solely for informational purposes and not for reliance purposes. You and your company acknowledge that it is still the obligation of you and your company to make your own assessment of the applicable U.S. export classification and ensure compliance accordingly. You and your company understand and acknowledge your obligations to make a prompt report to EPRI and the appropriate authorities regarding any access to or use of EPRI Intellectual Property hereunder that may be in violation of applicable U.S. or foreign export laws or regulations.

Printed on recycled paper in the United States of America

The Electric Power Research Institute (EPRI)

The Electric Power Research Institute (EPRI), with major locations in Palo Alto, California, and Charlotte, North Carolina, was established in 1973 as an independent, nonprofit center for public interest energy and environmental research. EPRI brings together members, participants, the Institute's scientists and engineers, and other leading experts to work collaboratively on solutions to the challenges of electric power. These solutions span nearly every area of electricity generation, delivery, and use, including health, safety, and environment. EPRI's members represent over 90% of the electricity generated in the United States. International participation represents nearly 15% of EPRI's total research, development, and demonstration program.

Together...Shaping the Future of Electricity

Program:
Nuclear Power

1011808

ELECTRIC POWER RESEARCH INSTITUTE
3420 Hillview Avenue, Palo Alto, California 94304-1395 • PO Box 10412, Palo Alto, California 94303-0813 USA
800.313.3774 • 650.855.2121 • askepri@epri.com • www.epri.com