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Probabilities of Failure and Uncertainty Estimate Information for Passive Components – A Literature Review

Pacific Northwest National Laboratory

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Abstract

This report describes a study performed for the U.S. Nuclear Regulatory Commission by Pacific Northwest National Laboratory with subcontractor support from Sigma-Phase, Inc. The effort involved reviewing the open literature to collect estimates of failure probability and their associated uncertainties for passive reactor components. The review focused primarily on probabilistic structural mechanics evaluations of plant-specific components for domestic nuclear power plants with pressurized water reactors or boiling water reactors as well as for international plants of similar design.

A computerized search of several databases identified more than 7500 documents, of which only a small fraction were related to probabilistic structural mechanics calculations. Probabilistic treatments and characterizations of fatigue and stress corrosion cracking were found to be well represented in the literature, but only a limited number of the publications described plant-specific and component-specific probabilistic evaluations based on actual design or operating stresses. The NRC will apply these results during the development of probabilistic fracture mechanics tools to generate failure probabilities for passive reactor components for use in regulatory decision making.

Foreword

The present approach to effective materials degradation management in nuclear power plants involves selecting appropriate materials for the design of components and monitoring degradation during operations. The *Code of Federal Regulations* identifies the regulatory requirements for both component design and periodic inservice inspections to ensure that design safety margins are maintained throughout component life. Plant technical specifications also include requirements for leakage monitoring and reactor shutdown to provide defense in depth to ensure the integrity of the reactor coolant system boundary. Lastly, the U.S. Nuclear Regulatory Commission (NRC) issues generic letters, bulletins, and orders to address emergent issues.

Notwithstanding this multifaceted regulatory framework, instances of unexpected materials degradation in nuclear power plants during recent years have led to a heightened interest by the nuclear power industry and the NRC in developing a proactive approach to materials degradation management. The establishment of a proactive program requires the identification of the components and materials that are expected to experience future degradation and the associated degradation mechanisms. This report presents the results of the NRC's review of the open literature to collect estimates of failure probabilities and their associated uncertainties for passive reactor components. The review focused primarily on probabilistic structural mechanics evaluations of plant-specific components for domestic nuclear power plants with pressurized-water reactors or boiling-water reactors, as well as for international plants of similar design.

The review showed that probabilistic treatments and characterizations of fatigue and stress-corrosion cracking are well represented in the literature, but only a limited number of the publications describe plant-specific and component-specific probabilistic evaluations based on actual design or operating stresses. The NRC will apply these results during the development of probabilistic fracture mechanics tools to generate failure probabilities for passive reactor components for use in regulatory decision making.

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Executive Summary

Failure frequencies for passive pressure boundary components often have been estimated to support probabilistic risk assessments. These estimates have been based primarily on reported failures from past plant operation and have established failure frequencies on a system rather than a component level. As such, these estimates have limitations in terms of predicting future performance and in identifying priorities for managing the integrity of specific components. Being based on reported operational events, the estimates are relatively good for failure modes consisting of small leaks, which have little potential to contribute to core damage. Corresponding estimates for the frequencies of larger leaks and pipe ruptures are, however, subject to much larger uncertainties.

This report describes a study performed for the U.S. Nuclear Regulatory Commission by Pacific Northwest National Laboratory with subcontractor support from Sigma-Phase Inc. This effort collected estimates of failure probability and their associated uncertainties from the open literature for passive reactor components. The review focused primarily on probabilistic structural mechanics evaluations of plant-specific components for domestic plants with pressurized water reactors and boiling water reactors as well as for international plants of similar design.

Computerized searches of several databases identified more than 7500 documents, of which only a small fraction were related to probabilistic structural mechanics calculations. Approximately 60% of the documents were related to corrosion mechanisms (e.g., flow-accelerated corrosion, erosion-corrosion, cavitation erosion, boric acid corrosion, crevice corrosion, pitting). The remaining documents were divided approximately equally between fatigue and stress corrosion cracking mechanisms. The vast majority of the references addressed various mechanistic aspects of degradation mechanisms and applied deterministic rather than probabilistic models. The literature search did not identify many plant-specific studies completed by industry-sponsored owners' groups or by research organizations since they are reported in documents not publicly available in the open literature.

Although the probabilistic treatments and characterizations of fatigue and stress corrosion cracking are well documented in the literature, only a limited number of these publications document plant-specific and component-specific evaluations based on actual design or operating stresses. Examples from the more relevant of these studies are discussed in this report.

The literature search did not identify any applications of probabilistic structural mechanics models for local corrosion such as pitting and microbiologically influenced corrosion. However, estimates of failure probabilities have been reported in plant-specific risk-informed inservice inspection (RI-ISI) evaluations. In some applications, the failure frequency estimates were based on statistical evaluations of service failure data. Failure probability calculations have been based also on assumed estimates of corrosion rates and on qualitative evaluations made by plant RI-ISI expert panels.

The open literature search did not identify any published applications of probabilistic structural mechanics models in nuclear power plant systems subject to the various wall-thinning types of degradation mechanisms, such as flow-assisted corrosion. As with other corrosion mechanisms, failure probabilities for flow-assisted corrosion have been estimated in support of plant-specific RI-ISI evaluations. In these cases, failure frequency estimates have been based on estimated inputs for rates of material wastage and wall thinning rates. In these calculations, corrosion-specific mechanistic models were not applied, and wall thinning was assumed to have the same impact on structural integrity as a circumferential crack that penetrated the wall to the same depth.

Formal attempts to quantify uncertainties in estimated failure probabilities show that the estimated failure probabilities for specific components are subject to large uncertainties, whether the estimate is based on data from operating experience or based on an application of probabilistic structural mechanics models. The reported uncertainties are particularly large when the estimated probabilities are very small. Initial flaw size distributions were identified as an especially large source of uncertainty in calculated failure probabilities because of the unavoidable difficulty in estimating the very low probabilities for the large fabrication flaws, which (if present) have a major impact on piping integrity. Consequently there can be significant uncertainties in calculated probabilities that are related to both the probabilistic fracture mechanics models as implemented in the various computer codes and the judgments made in defining critical input parameters for plant-specific calculations.

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Abbreviations and Acronyms

ASME	American Society of Mechanical Engineers
B&W	Babcock & Wilcox
BWR	boiling water reactor
CC	component cooling
CGR	crack growth rate
Compendex	Computerized Engineering Index
CVC	chemical and volume control
DEGB	double-ended guillotine break
DET	detection
DHR	decay heat removal
EMC^2	Engineering Mechanics Corporation of Columbus
EPRI	Electric Power Research Institute
FAC	flow-accelerated corrosion
GE	General Electric
GPM	gallons per minute
GRA&I	Government Report Announcements & Index
IGSCC	intergranular stress corrosion cracking
INEEL	Idaho National Engineering and Environmental Laboratory
LAS	low alloy steel
LER	Licensee Event Report
LOCA	loss-of-coolant accident
L/min	liters per minute
METADEX	Metals Abstracts/Alloy Index
MIC	microbiologically influenced corrosion
MRP	materials reliability program
MSIP	mechanical stress improvement process
NPP	nuclear power plant
NPS	nominal pipe size
NRC	U.S. Nuclear Regulatory Commission
NTIS	National Technical Information Service

PFM	probabilistic fracture mechanics
PIRT	Phenomena Identification and Ranking Technique
PMDA	Proactive Materials Degradation Assessment
PNNL	Pacific Northwest National Laboratory
POD	probability of detection
PPM	parts per million
PRA	probabilistic risk assessment
PTS	pressurized thermal shock
PTWC	probability of through-wall crack
PWR	pressurized water reactor
PWSCC	primary water stress corrosion cracking
RI-ISI	risk-informed inservice inspection
RVH	reactor vessel head
SCC	stress corrosion cracking
SKI	Swedish Nuclear Inspectorate
SRDM	simplified strain rate damage model
SRRA	structural reliability and risk assessment
SW	service water
TGSCC	transgranular stress corrosion cracking
TWC	through-wall crack
UHS	ultimate heat sink
USNRC	U.S. Nuclear Regulatory Commission
UT	ultrasonic testing
W	Westinghouse

1 Introduction

The U.S. Nuclear Regulatory Commission (NRC) has supported the research program Proactive Materials Degradation Assessment (PMDA) (USNRC 2007). The objective of this program has been to assess the possible occurrence of materials degradation and component degradation of light water reactors from the view of long-term operation such as would be relevant to relicensed plants. A central intent of the PMDA program has been to predict future occurrences of degradation that may or may not have been observed in the field or in the laboratory. Another objective is to consider the possibility of unexpected increases in degradation rates with time.

The PMDA program has included an assessment, conducted under contract with Brookhaven National Laboratory, of past and possible future materials degradation in light water reactors. The study used the Phenomena Identification and Ranking Technique (PIRT) involving eight experts from five countries who first met to discuss the technical issues and then to perform individual assessments. The analyses focused on materials degradation modes associated with operating environments for specific components, such as stress corrosion cracking, fatigue, flow-accelerated corrosion, boric acid corrosion, thermal aging embrittlement and radiation effects for existing plants. The work encompassed passive components for which failure of the pressure boundary could lead to a release of radioactivity to the environment or would degrade the ability of safety systems to perform their intended functions. The study did not address design issues such as mechanics and thermal hydraulics, the consequences of degradation, or the failure of active components such as values and pumps.

A detailed review by the panel of experts addressed more than 2000 components in the primary, secondary, and tertiary systems of specific reactor designs. Each expert individually provided judgments to score individual components in terms of “degradation susceptibility” and the “extent of knowledge” needed to predict the potential for future failures. These inputs were compiled and were used to generate a summary that reflected the collective judgment of the experts as a whole.

This report describes a study performed for the NRC by Pacific Northwest National Laboratory (PNNL), with subcontractor support from Sigma-Phase, Inc., to collect probability of failure estimates and their associated uncertainties as reported from previous studies in the open literature for passive reactor components. Although the literature contains examples of failure probability and frequency[a] estimates from statistical treatment of service failure data, this review focused primarily on examples of probabilistic structural mechanics evaluations of plant-specific components for domestic pressurized water reactor (PWR) and boiling water reactor (BWR) plants as well as international plants of similar design.

Section 2 describes some previous studies that have generated estimates of failure frequencies for reactor components in general and piping systems in particular.

(a) In this report, component *failure*, *failure probability*, and *failure frequency* generally refer to failure of the component pressure boundary associated with through-thickness cracks and small/large leaks. In some references, *failure* has also been used when referring to the presence of part through-thickness cracks.

Section 3 provides background on past and current efforts to develop probabilistic fracture mechanics codes, both in the United States and overseas. These codes have addressed the structural integrity of both reactor pressure vessels and reactor piping systems. This section describes some of these codes with an emphasis on codes that have been developed and applied in NRC research programs.

Section 4 describes the strategy, methodology, and results of a literature search for failure probability information that covered the reactor components and degradation mechanisms within the scope of the PMDA program. Computerized searches of several databases identified more than 7500 documents, of which only a small fraction was related to probabilistic fracture mechanics calculations.

Section 5 includes summaries of pipe failure probability estimates obtained by applying probabilistic structural mechanics models for different degradation mechanisms and for different reactor components. The information was obtained from reviews of public domain literature and is organized by the degradation mechanism types, including stress corrosion cracking (e.g., intergranular stress corrosion cracking [IGSCC], transgranular stress corrosion cracking [TGSCC], and primary water stress corrosion cracking [PWSCC]), thermal and vibration fatigue, flow-assisted degradation (e.g., flow-accelerated corrosion [FAC], erosion-corrosion, and cavitation erosion), and local corrosion (e.g., crevice corrosion, pitting, and microbiologically influenced corrosion [MIC]).

Section 6 presents published results of formal attempts to quantify the levels of uncertainty in estimated failure probabilities. The uncertainties associated with the individual inputs and modeling assumptions used in probabilistic fracture mechanics (PFM) calculations for fatigue failures are presented along with the effects of these uncertainties on the calculated failure probabilities. Uncertainties assigned by participants in an expert elicitation process used to estimate loss-of-coolant accident (LOCA) frequencies also are presented.

Section 7 summarizes the results and conclusions from this literature review.

2 Summary of Past Failure Probability Studies

The NRC research on the PMDA program (USNRC 2007) has the objective of assessing possible future degradation and failures of components from the viewpoint of long-term operation. One objective has been to predict future occurrences of new types of degradation that would not be expected based on the extrapolations of prior field experience. Another objective is to anticipate unexpected and significant increases in the failure rates for previously experienced degradation mechanisms.

Failure frequencies for passive pressure boundary components have often been estimated as described in NUREG/CR-5750 (Poloski et al. 1999) for use in probabilistic risk assessments (PRAs). Such estimates have been based on reported failures (e.g., leaks, ruptures) from past plant operation and have established failure frequencies on a system rather than component level. As such, these estimates have limitations in terms of predicting future performance and in identifying priorities for managing the integrity of specific components. Being based on reported operational events, the estimates will be relatively good for failure modes consisting of small leaks that have little potential to contribute to core damage. Corresponding estimates for the frequencies of larger leaks and pipe ruptures are, therefore, subject to large uncertainties.

The development of risk-informed inservice inspection (RI-ISI) programs also has required estimates of failure frequencies. In this work, the focus has been on the component level to a greater extent than other evaluations intended to support PRAs. However, the postulated degradation mechanisms have largely been limited to the mechanisms known to be active based on past experience. As such, other mechanisms that may have long incubation times have not been considered, and time-related increases in failure frequencies for known mechanisms have not been explicitly addressed. New mechanisms and increased failure rates for known mechanisms are addressed by periodic updating of the risk-informed inservice inspection programs.

This section describes examples of some previous studies that have generated estimates of failure frequencies for reactor components in general and piping systems in particular.

2.1 Rates of Initiating Events

A study (Poloski et al. 1999) by Idaho National Engineering and Environmental Laboratory (INEEL) provided estimates of initiating events at U.S. nuclear power plants, based on operating experience as well as other engineering evaluations. Loss-of-coolant accidents including pipe breaks were a major consideration in the study. The objectives were to 1) provide revised, historical frequencies for the initiating events, 2) compare these estimates with prior estimates used in PRAs, and 3) review the plant data for trends related to specific plant types (PWR plants versus BWR plants). The data to support the INEEL work came largely from the NRC Licensee Event Reports (LERs). This review gave approximately 2000 events to support the evaluations. Because the events of interest were limited to those that resulted in reactor trips, only the more significant modes of structural failures were covered. As such, occurrences of small leaks and observations of material degradation just simply requiring the repairs would have been largely excluded from consideration. In the case of rare events such as LOCAs, the INEEL researchers supplemented their LER-based evaluations with trends from available evaluations of engineering aspects of LOCA events. In these cases, analysis of available operating data, fracture mechanics analyses, and expert elicitation processes were employed to support reasonable but conservative adjustments to previous best-estimate LOCA frequencies in WASH-1400 (USNRC 1990b) and NUREG-1150 (USNRC 1975).

The estimated frequencies for LOCA events were less than the frequencies for other important initiating events such as loss of offsite power and loss of feedwater flow. The frequency for the small pipe break LOCA was estimated to be 5×10^{-4} (per plant per year) for both PWR and BWR plants. For PWR plants, the estimated frequency for steam generator tube rupture was about a factor of 10 greater, at 7×10^{-3}. Table 2.1 lists the frequencies for small, medium, and large LOCAs. For medium and large breaks, the frequencies were based in part on the operating data for small breaks and in part on conservative estimates from available fracture mechanics evaluations for the ratios between frequencies for the different LOCA categories. The frequencies reported in Table 2.1 (Poloski et al. 1999) were based on generic definitions of small, medium and large LOCAs and break sizes assumed in NUREG-1150 (USNRC 1990b).

Table 2.1 Estimated LOCA Frequencies (per plant per year) for PWR and BWR Plants (Poloski et al. 1999)

Plant Type	Small LOCA	Medium LOCA	Large LOCA
PWR	5×10^{-4} yr^{-1}	4×10^{-5} yr^{-1}	5×10^{-6} yr^{-1}
BWR	5×10^{-4} yr^{-1}	4×10^{-5} yr^{-1}	5×10^{-6} yr^{-1}

The LOCA frequencies in Table 2.1 are of limited usefulness in meeting the needs of the PMDA program because they represent the total contribution for all of the piping of the primary coolant system ranging from small-bore to large-diameter piping. For PRA evaluations, there is no need to identify specific components, materials, and environments that contribute to the overall LOCA frequency. A simplified approach could apportion the failure frequency equally among all the welds in the system, but this approach would fail to identify the small fraction of welds that would be expected to make the dominant contributions to the system-level failure frequency.

2.2 LOCA Frequency Elicitation Process

A recent NRC study applied the expert elicitation process to consolidate service history data and insights from PFM studies with knowledge of plant design, operation, and material performance and to thereby estimate failure frequencies for pressure boundary components (Tregoning et al. 2005). The objective was similar to that of the INEEL study (Poloski et al. 1999) in that the final results were estimates of system-level LOCA frequencies. However, the study required that the experts identify the components and degradation mechanisms that were the dominant contributors to the various categories of system-level LOCA frequencies. As such, the results of the elicitation were more closely aligned with the needs of the PMDA program.

Separate piping and non-piping passive system LOCA frequency estimates were developed as a function of effective break size and operating time for both BWR and PWR plants. The elicitation focused solely on determining event frequencies that initiate by unisolable primary system side failures that can be exacerbated by material degradation with age. The expert elicitation process employed was an adaptation of the formal expert judgment process used in NUREG-1150 (USNRC 1990b).

Most panelists agreed that a complete break of a smaller pipe is more likely than an equivalent size of opening in a larger pipe. Frequency estimates were not expected to change dramatically over the next 15 years or even the next 35 years. Although aging will continue, the consensus was that mitigation procedures are in place or will be implemented in a timely manner to alleviate possible LOCA frequency increases.

The quantitative responses were analyzed separately for each panel member to develop individual BWR and PWR total LOCA frequency estimates of the mean, median, 5th, and 95th percentiles. The LOCA frequencies for the individual panelists were then aggregated to obtain group LOCA frequency estimates, along with measures of panel diversity. While there was general qualitative agreement among the panelists about important technical issues and LOCA contributing factors, the individual quantitative estimates were much more variable. Additionally, as the LOCA size increased, the panel members generally expressed greater uncertainty in their predictions, and the variability among individual panelists' estimates increased. The elicitation LOCA frequency estimates were generally much less than the previous WASH-1400 (USNRC 1975) estimates and more consistent with the NUREG/CR-5750 estimates (Poloski et al. 1999). The elicitation medium-break LOCA estimates were higher than the NUREG/CR-5750 estimates by factors of 2.5 and 10 for BWR and PWR plant types, respectively. The baseline LOCA frequency estimates for the current day and end of license extension period are provided in Table 2.2 for both BWR and PWR plants. The aggregated group estimates for the median, mean, and 5th and 95th percentiles are summarized.

Selected PFM analyses were performed as part of the elicitation process to assist the participants in estimating failure frequencies. Results from some of these calculations are described in Section 5. The experts were also provided for their consideration a collection of data and evaluations coming from the PIPExp database (Lydell et al. 2004). The experts tended to use trends from the data on the reported operating events as the primary consideration in estimating failure frequencies and, in particular, for the smaller LOCA sizes. However, trends from fracture mechanics calculations were used for insights into the relative frequencies for the larger LOCA sizes compared to the frequencies for the small LOCA sizes (e.g., through-wall crack with perceptible leak).

The pipe failure database PIPExp was used in the LOCA frequency study. This database in an extension of the OECD Piping Failure Data Exchange (OPDE) database (Nuclear Energy Agency 2006) and makes use of the Microsoft Access database software. Over the time period of its development beginning in 1998, the PIPExp database has been expanded in terms of both the absolute number of event records and the depth of the database structure. The PIPExp database includes applicable worldwide BWR- and PWR-specific service experience with Code Class 1 piping. As of December 31, 2002, the database accounted for approximately 1992 and 3621 critical reactor-years of operating experience with commercial BWR and PWR plants, respectively. The database is actively maintained and periodically updated. The effort has involved populating the database while at the same time ensuring data quality. For an event to be considered for inclusion, it undergoes screening for eligibility.

Table 2.2 Baseline LOCA frequency Estimates from Expert Elicitation (Tregoning et al. 2005)

Plant Type	LOCA Size (gpm)	Break Size (in.)	Current Day Estimate (per cal. year) (25-yr fleet average operation)				Next 15 Year Estimate (per cal. yr.) (End of original license)			
			5th	Median	Mean	95th	5th	Median	Mean	95th
BWR	>100	1/2	3.1×10^{-5}	2.3×10^{-4}	5×10^{-4}	1.6×10^{-3}	2.5×10^{-5}	2×10^{-4}	4.5×10^{-4}	1.4×10^{-3}
	>1,500	1-7/8	3.1×10^{-6}	4.1×10^{-5}	9.9×10^{-5}	3.3×10^{-4}	2.5×10^{-6}	3.5×10^{-5}	9×10^{-5}	3×10^{-4}
	>5,000	3-1/4	4.7×10^{-7}	6.8×10^{-6}	1.9×10^{-5}	6.7×10^{-5}	4.1×10^{-7}	6.8×10^{-6}	2.1×10^{-5}	7.3×10^{-5}
	>25K	7	7.3×10^{-8}	1.5×10^{-6}	4.9×10^{-6}	1.7×10^{-5}	6.4×10^{-8}	1.6×10^{-6}	6.1×10^{-6}	2.1×10^{-5}
	>100K	18	7.4×10^{-9}	1.3×10^{-7}	9.8×10^{-7}	2.6×10^{-6}	$7.7 \times {}^{10-9}$	1.4×10^{-7}	1.3×10^{-6}	3.1×10^{-6}
	>500K	41	1.1×10^{-11}	3.2×10^{-10}	4.9×10^{-9}	$9.7 \times {}^{10-9}$	1.3×10^{-11}	4.3×10^{-10}	7.2×10^{-9}	1.4×10^{-8}
PWR	>100	1/2	5.6×10^{-4}	3.1×10^{-3}	6×10^{-3}	$1.8 \times {}^{10-2}$	2.7×10^{-4}	1.9×10^{-3}	4.4×10^{-3}	1.3×10^{-2}
	>1,500	1-5/8	9.2×10^{-6}	1.3×10^{-4}	4.9×10^{-4}	1.5×10^{-3}	9.4×10^{-6}	1.5×10^{-4}	6.1×10^{-4}	1.9×10^{-3}
	>5,000	3	1.8×10^{-7}	3.6×10^{-6}	1.5×10^{-5}	5.1×10^{-5}	3.5×10^{-7}	8×10^{-6}	3.4×10^{-5}	1.1×10^{-4}
	>25K	7	1.1×10^{-8}	2.6×10^{-7}	1.5×10^{-6}	4.5×10^{-6}	2.6×10^{-8}	5.4×10^{-7}	3.4×10^{-6}	1×10^{-5}
	>100K	14	$3 \times 10^{-10}0$	1.2×10^{-8}	1.5×10^{-7}	3.1×10^{-7}	1×10^{-9}	2.9×10^{-8}	3.8×10^{-7}	7.9×10^{-7}
	>500K	31	1.4×10^{-11}	9.9×10^{-10}	2.5×10^{-8}	4.2×10^{-8}	5×10^{-11}	2.5×10^{-9}	6.5×10^{-8}	1.1×10^{-7}

1 gpm = 3.7852 lpm
1 in. = 25.4 mm

2.3 Risk-Informed Inservice Inspection Evaluations

Risk-informed inservice inspection evaluations have estimated failure (leak and rupture) frequencies. This work has been focused on a detailed component level to a greater extent than the more common system-level evaluations used for PRAs. Specific degradation mechanisms are identified for components and welds, usually on the basis of past operating experience. Future degradation mechanisms and increased failure rates for known mechanisms are addressed by periodic updating of the inspection programs.

Most U.S. plants have adopted risk-informed inservice inspection programs for their piping systems. As such, damage mechanism assessments have been performed for these plants, and failure frequencies have been estimated. In some cases, the adopted methodology has been qualitative, with failure frequencies being classified in terms of categories (e.g., high, medium, and low). In other cases, the methodology (such as described by the Westinghouse Owners Group [1997]) has involved extensive probabilistic fracture mechanics calculations. Details of these plant-specific calculations have not been published but are available to the NRC through onsite inspection. Instead, the methodology and high-level results of the evaluations are submitted to the NRC as part of the review and approval process for the proposed risk-informed inservice inspection programs.

2.4 Other Studies

Other estimates of failure frequencies have been performed on a system- or plant-wide basis either for specific plants or on a more generic basis (such as PWR or BWR plants). Various methods have been applied, including expert elicitation (Vo et al. 1991). The very early studies, such as for WASH-1400, were based on nonnuclear data from other industries (USNRC 1975).

Other studies have used databases of reported pressure boundary failure events at operating plants. A Swedish evaluation of piping failure frequencies was performed for the Barsebäck-1 plant (Lydell 1999) in coordination with the Swedish Nuclear Inspectorate (SKI) using an international piping failure database that included Class 1, Class 2, and Class 3 piping. The SKI effort resulted in guidelines for using a database to evaluate piping reliability in terms of important influence and attribute factors. The guidelines include consideration of each failure mechanism separately (e.g., IGSCC, vibration fatigue, water hammer). All the relevant mechanisms are then combined to develop total LOCA frequencies. Only medium and large LOCA frequencies were determined in the SKI effort by considering events for these modes from failure mechanisms represented within the database. Maintenance of the database has nevertheless included reports of nonleaking flaws discovered during inspections to provide insights into the types of piping locations that have a potential for more significant modes of failure.

The integrity of steam generator tubes has been a significant aging issue given the various degradation mechanisms that have been active at PWR plants. Methods have been developed to predict probabilities of burst for degraded tubes. These methods have been based on statistical correlations of experimental data from burst tests of degraded tubes. Correlations have related burst pressures to such parameters as crack dimensions (Keating et al. 1995) or voltage amplitudes as measured during eddy current inspections. Voltage-based correlations have been extensively used in connection with criteria for plugging degraded tubes, but the methodologies have been largely described within submittals from plant licensees rather than in the open literature. In all cases, the probabilistic evaluations for steam generator tubes have used statistical treatments of burst testing rather than probabilistic structural mechanics models as addressed by the current report.

3 Review of Existing Probabilistic Fracture Mechanics Models for Reactor Components

Since the early 1980s, the NRC has supported the development and application of probabilistic fracture mechanics codes for predicting failure probabilities of reactor pressure boundary components. These codes have addressed the structural integrity of both reactor pressure vessels and reactor piping systems. This section describes some of these codes with an emphasis on those that have been developed and applied in NRC research programs. Various available codes are discussed also in Section 5 in connection with published failure probabilities as predicted for a range of degradation mechanisms.

3.1 Reactor Pressure Vessel Codes

Early work in the 1980s supported the development of the pressurized thermal shock (PTS) rule (10 CFR 50.61) that defined acceptable limits for neutron embrittlement for materials of reactor pressure vessels. Two codes were developed during this time period: OCA-P by Oak Ridge National Laboratory (Cheverton and Ball 1984) and VISA-II by PNNL (Simonen et al. 1986). Later work beginning in the 1990s and continuing to the present has focused on the FAVOR code (Dickson 1994; Dickson et al., NRC, unpublished[(a)]). These codes as well as similar codes developed both in the United States and overseas are used to predict brittle fracture in irradiated reactor pressure vessel beltline materials. The VISA-II code was used to support the current PTS rule, and FAVOR was used to support recently proposed PTS rule changes.

Another probabilistic fracture mechanics code (VIPER) was developed to address issues specifically related to welds in boiling water reactor vessels (Tang et al. 1999). This code used a Monte Carlo approach and a fracture mechanics model similar to those used in the VISA-II and FAVOR codes. Special model features were incorporated to address weld residual stresses and crack initiation and growth by stress corrosion mechanisms.

3.2 Reactor Piping Codes

A number of codes for predicting failure probabilities of reactor piping have been developed. Each is described in the following sections.

3.2.1 PRAISE

The PRAISE code (Harris et al. 1981, 1986; Harris and Dedhia 1992) was developed as an NRC-funded effort starting in the early 1980s to predict failure probabilities for reactor piping systems. The original work was directed at fatigue failures of large-diameter reactor coolant piping due to the growth of weld fabrication flaws. The first version of PRAISE was developed for the NRC by Lawrence Livermore National Laboratory to support the seismic safety margins project and was applied in early work to address PWR piping; it was later extended to address IGSCC in BWR stainless steel piping welds.

(a) Dickson TL, PT Williams, and S Yin. Unpublished. *Fracture Analysis of Vessels – Oak Ridge, FAVOR, v04.1, Computer Code: User's Guide*. NUREG/CR-6855, ORNL/TM-2004/245, October 2004, in review by the NRC.

PRAISE has seen continued applications over the years with ongoing updates to include capabilities needed to address specific piping integrity issues, most notably the initiation of fatigue cracks (Khaleel et al. 2000).

The enhanced version of PRAISE has allowed for crack initiation at multiple sites around the circumference of girth welds and simulated the linking of adjacent cracks to form longer cracks more likely to cause large leaks and pipe ruptures. By coupling S-N crack initiation and fatigue crack growth models, simulations allow cracks to initiate and grow in a more realistic manner.

In the early 1990s with the evolution of computer technology a version of PRAISE was developed to run on personal computers (Harris and Dedhia 1992). The mid 1990s saw the development of methods for risk-informed inservice inspection, and there were many new applications of PRAISE. During this period, a commercial version of PRAISE was made available by Dr. David Harris of Engineering Mechanics Technology that simplified the input to the code with an interactive front end (Harris and Dedhia 1998). PNNL also performed numerous calculations with PRAISE that applied probabilistic fracture mechanics to examine how different inservice inspection strategies could improve component reliability (Khaleel and Simonen 1994a, 1994b; Khaleel et al. 1995; Simonen et al. 1998; Simonen and Khaleel 1998a, 1998b). These studies evaluated changes in failure probabilities when changes to inservice inspection requirements are implemented at operating nuclear power plants. Other work at PNNL for the NRC (Khaleel et al. 2000) was directed at fatigue critical components with the potential to attain calculated fatigue usage factors in excess of design limits (usage factors greater than unity). PRAISE was used also to develop the technical basis for changes to Appendix L of the ASME Code Section XI that addresses fatigue critical locations in pressure boundary components (Gosselin et al. 2005).

In summary, the PRAISE code has been extensively documented, has been successfully applied to a range of structural integrity issues, and has been available since the 1980s as a public domain computer code. However, the code has not been systematically maintained and upgraded in an ongoing manner. Upgrades have met the needs of immediate applications of the code and, as such, have served to fill very specific gaps in capabilities of PRAISE.

3.2.2 PRO-LOCA

In 2003, the NRC began the development of a new code called PRO-LOCA (Rudland et al. 2006; Rudland et al., NRC, unpublished[(a)]) that is intended to include advances in fracture mechanics models and thereby become the successor to the PRAISE code. The development of the PRO-LOCA code was motivated by an NRC need to address issues related to LOCA events. The need for an improved probabilistic fracture mechanics code became evident during an expert elicitation process funded by the NRC (Tregoning et al. 2005) to establish estimates of LOCA frequencies along with quantification of the uncertainties in the estimates. Development of the code is expected to continue. Part of this future work would include the preparation of documentation of the code with detailed user instructions needed to support the release of the code to external organizations. A report on the status of PRO-LOCA has been

(a) Rudland DL, H Xu, G Wilkowski, N Ghadiali, F Brust, and P Scott. Unpublished. *Evaluation of Loss-of-Coolant Accident (LOCA) Frequencies Using the PRO-LOCA Code*, NUREG/CR-XXXX, currently under review.

prepared by EMC^2 (Rudland et al., NRC, unpublished[a]), which describes features and the technical bases for PRO-LOCA. Like PRAISE, the PRO-LOCA code addresses the failure mechanisms associated with both preexisting cracks and service-induced cracks. Both fatigue and IGSCC are addressed. The model for fatigue is essentially the same as that in PRAISE, whereas the IGSCC model differs significantly from the model in PRAISE. PRO-LOCA can also predict failure probabilities for PWSCC. Other improved capabilities are in the areas of leak rate predictions and the prediction of critical/unstable crack sizes. Many of the enhancements are based on the results of some 20 years of NRC-supported research on the integrity of degraded piping. PRO-LOCA has also incorporated an improved basis for simulating weld residual stresses.

3.2.3 Other Piping Codes

A number of other probabilistic fracture mechanics codes have been developed by various organizations to calculate failure probabilities for piping. The SRRA code (Westinghouse Owners Group 1997; Bishop 1993, 1997) as developed by Westinghouse uses approaches similar to those of the PRAISE code. However, the fatigue model is limited to failures associated with the growth of preexisting fabrication flaws. Fatigue crack initiation has been approximated, in a conservative manner, by assuming very small preexisting cracks, but only one crack per weld. Stress corrosion cracking is similarly treated by postulating very small preexisting cracks with crack growth predicted using a crack growth law consistent with growth rates for stress corrosion cracks (SCCs). The SRRA code includes an importance sampling procedure that gives reduced computation times compared to the Monte Carlo approaches used by PRO-LOCA and PRAISE. The SRRA code also simulates uncertainties in a wide range of parameters, including operating stresses and pipe dimensions.

A European effort (Brickstad et al. 2004, NURBIM) has reviewed and tested a number of codes including PRAISE as part of an international benchmarking study. Included were a Swedish code NURBIT (Brickstad and Zang 2001), the PRODIGAL code from the United Kingdom (Bell and Chapman 2003), a code developed in Germany by GRS (Schimpfke 2003), a Swedish code ProSACC (Dillstrom 2003), and another code (STRUEL) from the United Kingdom (Mohammed 2003). A detailed discussion of the NURBIM study is beyond the scope of this report. However, it appears that none of the other benchmarked codes provides capabilities that are significantly different from or superior to the capabilities of PRAISE or PRO-LOCA. In any case, the predictions of all such codes are limited in large measure by uncertainties in establishing the input parameters needed to apply the code.

(a) Rudland DL, H Xu, G Wilkowski, N Ghadiali, F Brust, and P Scott. Unpublished. *Evaluation of Loss-of-Coolant Accident (LOCA) Frequencies Using the PRO-LOCA Code*, NUREG/CR-XXXX, currently under review.

4 Open Literature Search

The Proactive Materials Degradation Assessment (PMDA) (USNRC 2007) identified components of nuclear power plant systems and ranked their susceptibility or likelihood for damage due to various degradation mechanisms. An open literature search was performed to locate technical papers and reports with information relevant to failure frequencies for the components and degradation mechanisms identified in the NRC PMDA program. The search used computerized databases available through the Hanford Technical Library in Richland, Washington, and made use of support from information specialists from the library staff. A large list of relevant documents was generated and reviewed to identify documents with specific information on component failure frequencies with specific attention to results generated by application of probabilistic structural mechanics models.

4.1 Methodology

Research and development relevant to the structural integrity of reactor systems (reactor coolant, emergency core cooling, steam and power conversion, and support and auxiliary) of PWRs and BWRs has been ongoing since the advent of nuclear facilities, both military and civilian. Activities have included programs at nuclear laboratories and material laboratories in many countries worldwide. While some results such as new material compositions may not be freely available, numerous other results and accounts of operating experience have been published internationally in various journals and research reports. Most of these publications are electronically listed in various comprehensive bibliographic databases such as *Compendex*, *Corrosion Abstracts*, *METADEX*, and *Science Research Connection*, and those maintained by the National Technical Information Service (NTIS) and the NRC ADAMS Public Records System and Public Legacy Databases.

An initial computer-aided search was performed on every subgroup of the different systems of PWRs and BWRs. This search yielded very few results. The search criterion was revised by grouping different subgroups in PWRs and BWRs to do a broad search of the above databases. For example, different subgroups such as the hot leg piping, the crossover leg piping, and the cold leg piping were grouped into a single search item. Keywords such as nuclear power plants and major group names such as pipes along with the various degradation mechanisms were used. All the results were retrieved and downloaded into an EndNote database. EndNote is a bibliographic management software tool for publishing and managing bibliographies. This EndNote software was subsequently used for searching specific topics.

The following open literature databases were searched.

***Compendex* (computerized engineering index)** is a database that provides international coverage of literature in the engineering field, including materials science and metallurgy, bioengineering, air and water pollution, and solid waste and hazardous waste management. Citations are drawn from 5000 journals, technical reports, and conference papers and proceedings.

Corrosion Abstracts provides the world's most complete source of bibliographic information in the area of corrosion science and engineering. International sources of literature are scanned and abstracted in the areas of general corrosion, testing, corrosion characteristics, preventive measures, materials construction and performance, and equipment for many industries.

***METADEX* (Metals Abstracts/Alloy Index)** is a comprehensive source covering the worldwide literature on metals and alloys, including properties, manufacturing, applications, and development.

The National Technical Information Service (NTIS) provides a multidisciplinary database covering U.S. government-sponsored research and worldwide scientific, technical, engineering, and business-related information. The Government Report Announcements & Index (GRA&I) is completely covered by the NTIS database.

Science Research Connection provides approximately 4 million bibliographic records and over 125,000 full-text documents spanning more than six decades of DOE research.

The NRC ADAMS Public Records System and Public Legacy Databases provide technical information about nuclear facilities and NRC licensees and is a comprehensive research center for NRC documents.

Search strategies were customized to fit the databases searched. The search strategy for *Compendex, Corrosion Abstracts, METADEX,* and *NTIS* is very straightforward. Boolean operators like "and" and "or" were used to limit the searches. *Science Research Connection* follows a similar strategy, whereas NRC's ADAMS database search engine is simple and does not handle complex searching so multiple simple searches were run to retrieve the relevant important documents. Six sets of different searches covering all of the degradation mechanism, were performed, and the results were downloaded into EndNote Libraries.

4.1.1 Search Strategy for Compendex, Corrosion Abstracts, Metadex, and NTIS Search

The following six sets of keywords were used:

- "nuclear power plant*" AND (pipe OR pipes OR piping OR nozzle*) AND (erosion OR fatigue OR corrosion OR pitting OR MIC OR SCC OR PWSCC OR FAC) AND limit by date to 1980–present

- "nuclear power plant*" AND (vessel* OR steam generator* OR pressurizer*) AND (corrosion OR pitting OR fatigue OR SCC OR PWSCC) AND limit by date to 1980–present

- (BWR OR boiling water reactor*) AND core shroud* AND (304 OR "stainless steel" OR "alloy 182" OR weld*) AND (SCC OR PWSCC or IGSCC OR fatigue of corrosion) AND limit by date to 1980–present

- (BWR OR boiling water reactor*) AND jet pump* AND (weld* OR "inconel 600" OR "alloy 600" OR "alloy 182" or diffuser* OR bracket* or support* or brace* OR adapter OR riser) AND (SCC OR PWSCC or IGSCC OR fatigue of corrosion) AND limit by date to 1980–present

- (BWR OR boiling water reactor*) AND steam AND (separator OR dryer) AND (SCC OR PWSCC or IGSCC OR fatigue OR corrosion) limit by date to 1980–present

- (BWR OR boiling water reactor*) AND condensate storage tank* AND limit by date to 1980–present.

4.1.2 Search Strategy for ADAMS

This search required a specialized strategy. The Public Records System contains documents and records from November 1999 to the present, and the Public Legacy System contains records of documents published before November 1999. The advanced search option available in ADAMS database is a full-text search and proved too broad. The simple search used automatic truncation and found many irrelevant documents. As a result, multiple searches of the following keywords were done in each database. In the Public Legacy System, results were limited by date to the period from 1980 through 1999. The sets of keywords used were

- pipe erosion, pipe fatigue, pipe corrosion, pipe pitting, pipe SCC, piping erosion, piping fatigue, piping corrosion, piping pitting, and piping SCC

- nozzle erosion, nozzle fatigue, nozzle corrosion, nozzle pitting, and nozzle SCC

- vessel corrosion, vessel pitting, vessel fatigue, and vessel SCC

- steam generator corrosion, steam generator pitting, steam generator fatigue, and steam generator SCC

- pressurizer corrosion, pressurizer pitting, pressurizer fatigue, and pressurizer SCC

- core 304 SCC, core stainless steel SCC, core alloy 182 SCC, core weld SCC, core 304 fatigue, core stainless steel and fatigue, core alloy 182 fatigue, core weld fatigue, core 304 corrosion, core stainless steel corrosion, core alloy 182 corrosion, and core weld corrosion

- jet pump weld SCC, jet pump stainless steel SCC, jet pump inconel 600 SCC, jet pump alloy 600 SCC, jet pump alloy 182 SCC, jet pump diffuser SCC, jet pump bracket SCC, jet pump support SCC, jet pump brace SCC, jet pump adapter SCC, jet pump riser SCC, jet pump weld fatigue, jet pump stainless steel fatigue, jet pump inconel 600 fatigue, jet pump alloy 600 fatigue, jet pump alloy 182 fatigue, jet pump diffuser fatigue, jet pump bracket fatigue, jet pump support fatigue, jet pump brace fatigue, jet pump adapter fatigue and jet pump riser fatigue, jet pump weld corrosion, jet pump stainless steel corrosion, jet pump inconel 600 corrosion, jet pump alloy 600 corrosion, jet pump alloy 182 corrosion, jet pump diffuser corrosion, jet pump bracket corrosion, jet pump support corrosion, jet pump brace corrosion, jet pump adapter corrosion, and jet pump riser corrosion

- steam dryer fatigue, steam dryer corrosion, and steam dryer SCC, steam separator fatigue, steam separator corrosion, and steam separator SCC.

4.1.3 Search Strategy for Science Research Connection

In searching *Science Research Connection*, we used the following six sets of keywords with a date limit of 1980–2006:

- All Fields: “nuclear power plant” OR “nuclear power plants”
  Bibliographic Data: erosion OR fatigue OR corrosion OR pitting OR MIC or SCC OR PWSCC OR FAC
  Title Data: pipe or pipes or piping

- All Fields: “nuclear power plant” OR “nuclear power plants”
  Bibliographic Data: corrosion OR pitting OR fatigue OR SCC OR PWSCC
  Title Data: vessel OR vessels OR “steam generator*” OR pressurizer*

- All Fields: “BWR” OR “boiling water reactor*”
  Bibliographic Data: corrosion OR fatigue OR SCC OR PWSCC OR IGSCC
  Title Data: core shroud

- All Fields: “BWR” OR “boiling water reactor*”
  Bibliographic Data: corrosion OR fatigue OR SCC OR PWSCC OR IGSCC
  Title Data: jet pump*

- All Fields: “BWR” OR “boiling water reactor*”
  Bibliographic Data: corrosion OR fatigue OR SCC OR PWSCC OR IGSCC
  Title Data: steam AND (separator* OR dryer*)

- All Fields: “BWR” OR “boiling water reactor*”
  Title Data: condensate storage tank*.

4.2 Literature Search Results

The above open literature searches identified more than 7500 publications from 1980 to the present. Approximately 60% of the documents were related to corrosion and pitting mechanisms. The remaining documents were equally divided between fatigue and stress corrosion cracking mechanisms. The vast majority of the references addressed various mechanistic aspects of degradation mechanisms and applied deterministic rather than probabilistic models. With regard to corrosion mechanisms other than SCC (e.g., general corrosion, pitting), the literature search did not identify any probabilistic calculations that applied mechanistic models to address the types of failures observed in the field. Although the probabilistic treatments and characterizations of fatigue and stress corrosion cracking are well documented in the open literature, a relatively limited number of publications documented the results of plant-specific and component-specific probabilistic evaluations based on design or actual operating stresses. The more relevant of these are discussed in Section 5 of this report.

In many cases, the literature search did not identify numerous plant-specific studies completed by industry owners’ groups and by research organizations such as the EPRI Materials Reliability Program. These results are typically not available through the open literature. Also, plant-specific calculations, such as those performed in the support of risk-informed evaluations, were not publicly available. However, portions of these results are sometimes included in topical reports submitted for NRC approval or in selected papers in the literature. Examples can be found in the Westinghouse Owners Group topical report on RI-ISI (1999) and work reported by Bishop and McNeil (1999). Additionally, failure probability estimates have been based on statistical analyses of industry service experience. Examples include work reported by Fleming et al. (1999); Lydell (1999); Attwood et al. (1999); Fleming (2004); and Tregoning et al. (2005).

5 Failure Probabilities for Reactor Components

This section includes summaries of failure probability estimates obtained by applying probabilistic fracture mechanics models to different reactor components and for different degradation mechanisms. The information was obtained from reviews of public domain literature and is organized by degradation mechanism and plant type (BWR or PWR).

5.1 Stress Corrosion Cracking

The three variations of stress corrosion cracking of concern to reactor components are intergranular, transgranular, and primary water cracking, with each type discussed separately in this section.

5.1.1 Intergranular Stress Corrosion Cracking

In general, intergranular stress corrosion cracking (IGSCC) results from a combination of sensitized materials (caused by a depletion of chromium in regions adjacent to the grain boundaries in weld heat-effected zones), high stress (residual welding stresses), and a corrosive environment (high level of oxygen or other contaminants). IGSCC is encountered most frequently in wrought Type 304 or 316 austenitic stainless steels that become sensitized through the welding process and are subjected to BWR operating environments. To a lesser extent, IGSCC has been observed also in wrought austenitic stainless steel PWR piping having high dissolved oxygen content and stagnant flow (e.g., stagnant, oxygenated borated water systems). The susceptible areas extend into the base material a few millimeters beyond either side of the weld—the weld *heat-affected zone*. Generally materials resistant to sensitization from welding are not considered susceptible to IGSCC. The cause of and experience with IGSCC in nuclear power plants is well documented in Shao et al. (1979, 1980); Kassir et al. (1985); Miraglia (1988); Hazelton and Koo (1988); and Strosnider et al. (2000).

By 1987 and 1988, most operating BWR plant operators had implemented one or more IGSCC mitigation programs together with IGSCC inspection programs as outlined in the NRC Generic Letter 88-01 (Miraglia 1988) and NUREG-0313 Revision 2 (Hazelton and Koo 1988). Today most if not all BWR plants operate with hydrogen water chemistry, and weld residual stresses are kept at a minimum by using improved welding techniques, applying weld overlays or by induction heat stress improvement of existing welds. Although not generally used in the United States, austenitic stainless steels with additions of niobium or titanium, referred to as *stabilized stainless steels*, have become the preferred material in German BWR plants.

Summarized in Table 5.1 is the worldwide service experience involving IGSCC in the BWR operating environment. This tabulation is based on nuclear power plant (NPP) failure data reported in the proprietary PIPExp database described in Appendix D to NUREG-1829 (Tregoning et al. 2005). The weld failure data separated into two groups: stabilized and unstabilized austenitic stainless steels.

Table 5.1 Summary of IGSCC Service Experience in BWR Nuclear Power Plants

Type of Austenitic Stainless Steel	Number of Reported Weld Failures Attributed to IGSCC[a]					
	1970–1987		1988–2005		1970–2005	
	Part Through-Wall	Through-Wall	Part Through-Wall	Through-Wall	Part Through-Wall	Through-Wall
Stabilized[b]	--	--	209	12	209	12
Unstabilized[c]	623	139	195	33	818	172

(a) Through-wall leak rates range from perceptible (no active leakage during normal operation) to small active leakage (Duane Arnold weld failure on June 14, 1978, to date the most significant failure, produced a 11.4-L/min (3-gpm) leakage. Weld failure at Santa Maria de Garona, a Spanish BWR plant, on February 12, 1980, produced a 3.0-L/min (0.8-gpm) leakage.
(b) Stabilized austenitic stainless steel failures report for 15 non-U.S. operating NPPs.
(c) Unstabilized austenitic stainless steel failures report for 57 plants U.S. and non-U.S. operating NPPs.

Numerous published reports document results of PFM-based ISGCC failure probabilities for major welds in BWR ASME Code Class 1 and 2 piping. The published probabilities span a wide range from lower-bound PFM computer code truncation values of 1×10^{-11} or less per weld per calendar year to upper-bound estimates approaching 1×10^{-1} per weld per calendar year. These results reflect inherent modeling uncertainties as well as different modeling capabilities and analysis assumptions. For the most part, the more significant IGSCC probabilistic fracture mechanics studies in the open literature either employed or were benchmarked against versions of IGSCC models developed for the PRAISE computer code.

The early IGSCC probabilistic fracture mechanics evaluations (Holman and Chou 1988) were based on the semi-empirical IGSCC model developed for the PRAISE computer code as described in Harris et al. (1985, 1986a, 1986b). In the first paper the crack initiation time and crack growth rates are correlated against *damage parameters* that consolidate the separate influences of environment, applied loads, residual stresses, and material sensitization. The models were benchmarked by comparing predicted leak rates under nominal BWR applied load conditions against actual crack and leak data. To improve the agreement of the PRAISE model predictions with observed field experience, the model was tuned by adjusting residual stress magnitudes (using a multiplication factor).

The original values of residual stress adjustment factors selected by Harris et al. (1985) for large, intermediate, and small pipe sizes are provided in Table 5.2. The leak and double-ended guillotine (DEGB) probabilities for the large, intermediate, and small pipe sizes and using residual stress adjustment factors in Table 5.2 are summarized in Table 5.3. The comparison of the field observations of leak probabilities and PRAISE predictions for the large pipe range (outside diameters greater than 508 mm [>20 in.]) and various residual stress adjustment factor values is shown in Figure 5.1.

The results in Figure 5.1 show that it was necessary to substantially reduce the residual stress levels to 15-20% of the values estimated from measurements.

Table 5.2 Residual Stress Adjustment Factors, f (Harris et al. 1985)

Line Size Range	Outside Diameter	Value of f
Large	>508 mm (>20 in.)	0.15
Intermediate	254–508 mm (10–20 in.)	0.30
Small	<254 mm (<10 in.)	0.20

Table 5.3 Summary of Leak and DEGB Probabilities as a Function of Time for Three Pipe Size Ranges (Harris et al. 1985)

Time (Years)	Leak Probabilities			DEGB Probabilities		
	Large	Intermediate	Small	Large	Intermediate	Small
2	$<3.0 \times 10^{-5}$	2.4×10^{-4}	7.0×10^{-4}	--	$<2.0 \times 10^{-5}$	$<1.0 \times 10^{-4}$
5	6.0×10^{-4}	8.3×10^{-4}	1.2×10^{-2}	--	4.0×10^{-5}	1.0×10^{-4}
10	3.4×10^{-4}	5.4×10^{-2}	6.7×10^{-2}	--	4.0×10^{-5}	3.0×10^{-4}
15	7.7×10^{-3}	1.2×10^{-1}	1.4×10^{-1}	$<3.0 \times 10^{-5}$	6.0×10^{-5}	4.0×10^{-4}
20	1.2×10^{-2}	1.8×10^{-1}	1.9×10^{-1}	3.0×10^{-5}	6.0×10^{-5}	5.0×10^{-4}

The original calibration of the PRAISE model in Harris et al. (1985) was redone by Khaleel et al. (1995). The original calculations predicted substantial levels of material damage from loading and unloading events (i.e., complete startup and shutdown of the plant) that used a model that applied strain-to-failure data from constant extension rate tests. A review of the damage model concluded that these predictions were extremely conservative and were inconsistent with more recent insights into stress corrosion cracking mechanisms. In the revised calculations, the loading/unloading events were decreased from once per year to once per 40 years, which essentially removed the contribution of these events to the calculated failure probabilities. Figure 5.2 compares the original PRAISE cumulative leak probabilities (Harris et al. 1985) and results obtained by Khaleel et al. (1995) for various values of residual stress adjustment factors and plant loading/unloading frequencies. In this case, the adjusted residual stress level used to limit the disagreement between predicted and observed leak probabilities was set at 75% of their original values ($f = 0.75$). The resulting predictions had a much more rational basis and were in very good agreement with operational data for time periods beyond 6 years. The less satisfactory level of agreement for time periods less than 6 years can be attributed in large measure to a lack of observed failure events for the early periods of plant operation.

NUREG/CR-4792 (Holman and Chou 1988) describes early stress corrosion cracking probabilistic fracture mechanics evaluations for BWR reactor coolant piping performed for the NRC by Lawrence Livermore National Laboratory. In these evaluations, leak and DEGB probabilities were estimated in the recirculation loop piping of a representative Mark I BWR plant. The probabilistic fracture mechanics evaluations of NUREG/CR-4792 were based on the semi-empirical IGSCC model developed for the PRAISE computer code as described in Harris et al. (1985, 1986a, 1986b).

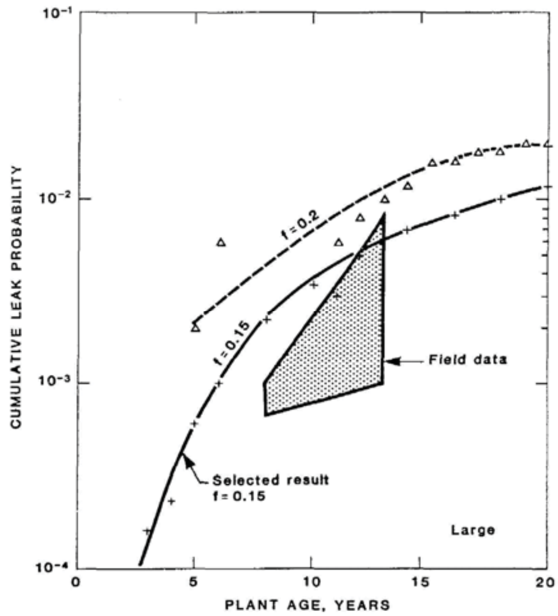


Figure 5.1 Field Observations of Leak Probabilities and PRAISE Predictions at Various Residual Stress Adjustment Factor Values (f) for 304 Stainless Steel Pipes Having Outside Diameters Greater Than 508 mm (>20 in.) (Harris et al. 1985)

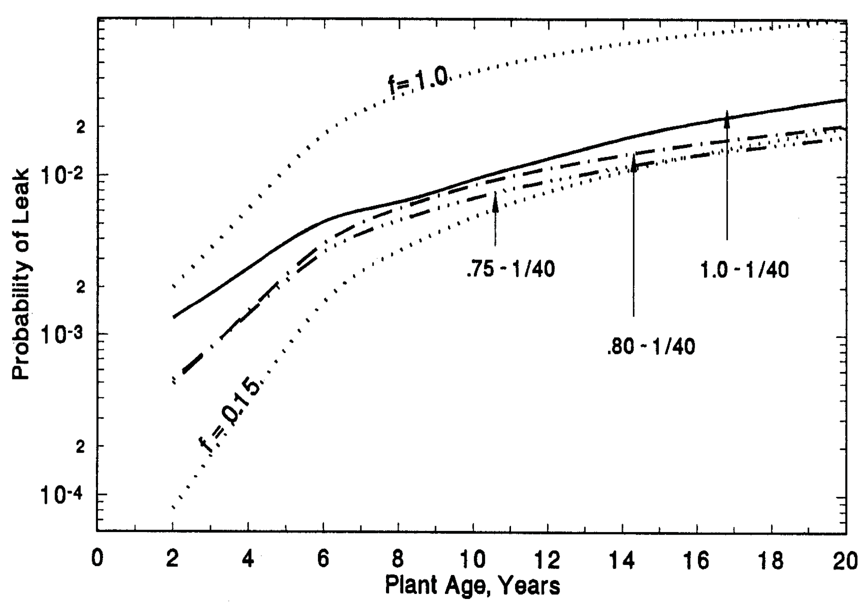


Figure 5.2 Sample Comparison of PRAISE Cumulative Leak Probabilities (Harris et al. 1985) and Results Obtained by Khaleel et al. (1995) for Various Values of Residual Stress Adjustment Factors and Plant Loading/Unloading Frequencies

The reactor recirculation system in a Mark I BWR plant has two loops, as shown in Figure 5.3. The recirculation water from the reactor vessel is delivered to the reactor coolant recirculation pump via a suction line 711 mm (28 in.) in diameter. The reactor coolant recirculation pump discharges into the reactor pressure vessel via discharge piping 711 mm (28 in.) in diameter and header piping 559 mm (22 in.) in diameter header piping, plus five 305-mm (12-in.) NPS risers. Although not shown in Figure 5.3, a bypass line 102 mm (4 in.) in diameter with a bypass valve is connected to the discharge line on either side of the reactor coolant recirculation pump discharge valve.

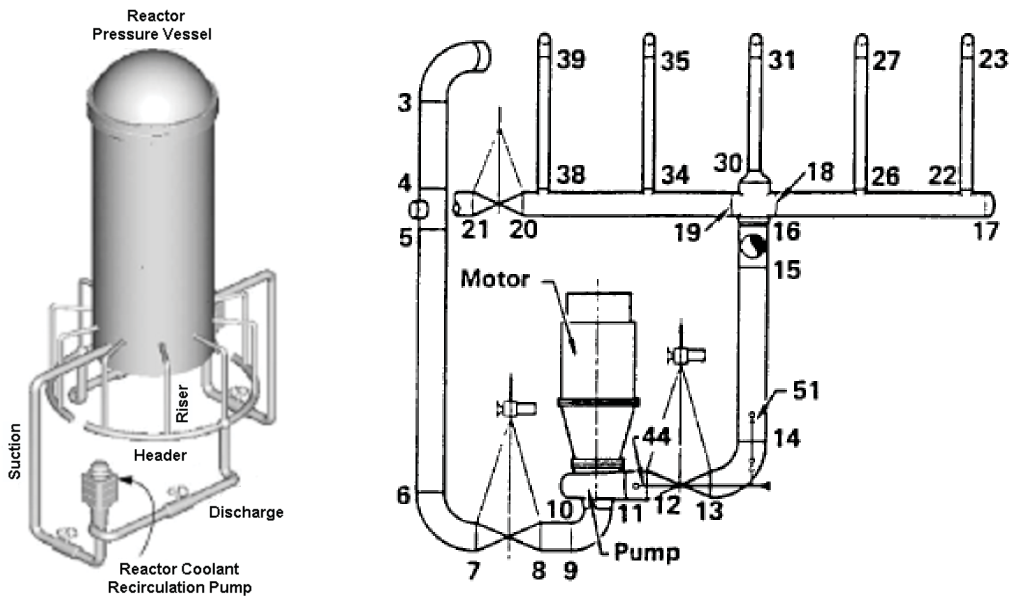


Figure 5.3 Mark I BWR Reactor Recirculation System Piping

The evaluation considered three recirculation system configurations: 1) an existing "old" configuration with 51 welds per loop and SA-240 TP304 stainless steel piping; 2) an existing "old" configuration with 51 welds per loop and SA-358 TP316 stainless steel piping; and 3) a replacement "new" configuration with SA-358 TP316 stainless steel piping and 30 welds per loop. The material types, dimensions, and number of welds for the recirculation piping in the NUREG/CR-4792 evaluations are presented in Table 5.4.

The bar charts in Figure 5.4 show the relative contribution made by each NUREG/CR-4792 weld type to the overall system probabilities of leak and DEGB. The relative contribution of different weld types is illustrated further in Figure 5.5, which shows weld-by-weld leak probabilities for the existing loop configuration. Finally, a comparison of the NUREG/CR-4792 cumulative DEGB probabilities between the old configuration with TP304 stainless steel piping and the new configuration with TP316 stainless steel piping is shown in Figure 5.6.

Table 5.4 NUREG/CR-4792 Mark I BWR Recirculation System Configurations

Material	Pipe	Suction	Discharge	Header	Riser	Bypass
SA-240	Nominal Pipe Size (in.)	28	28	22	12	4
TP304	Outside Diameter (in.)	28.169	28.519	22.003	12.706	4.5
Stainless	Wall Thickness (in.)	1.51	1.326	1.038	0.631	0.337
Steel	Number of Welds	10	6	5	20	10
SA-358	Nominal Pipe Size (in.)	28	28	22	12	NA
TP316	Outside Diameter (in.)	28.000	28.000	22.000	12.750	NA
Stainless	Wall Thickness (in.)	1.209	1.390	1.750	1.688	NA
Steel	Number of Welds	11	5	2	12	NA
25.4 mm = 1 in.						

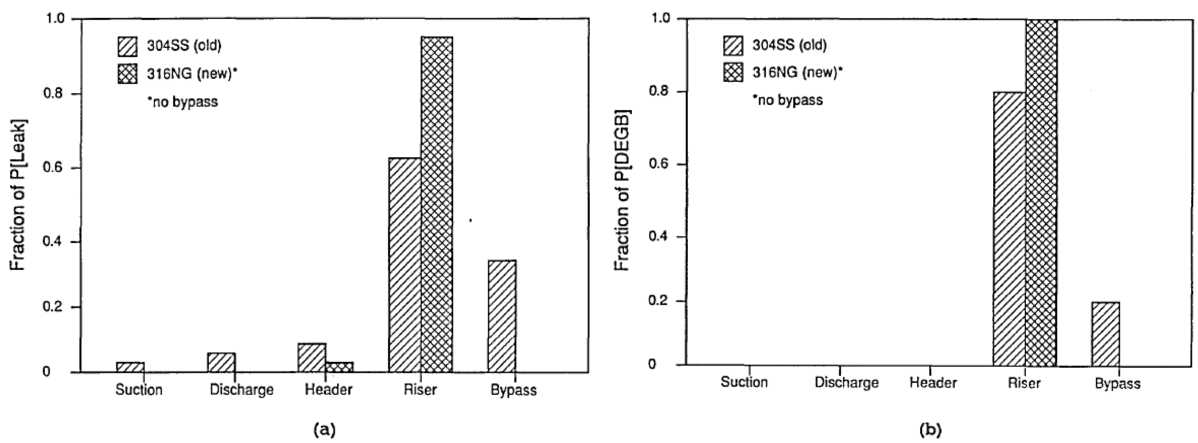


Figure 5.4 NUREG/CR-4792 Relative Contributions of Various Weld Types to BWR Recirculation System Probabilities of (a) Leak and (b) DEGB

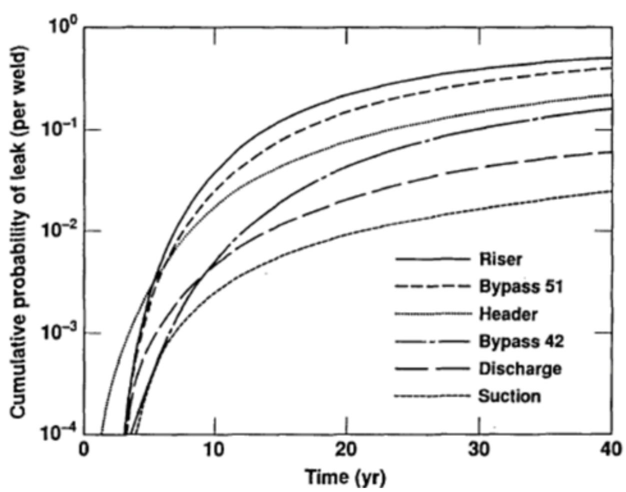


Figure 5.5 NUREG/CR-4792 Cumulative Leak Probabilities for BWR Recirculation System Weld Types in the "Old" Configuration with TP304 Stainless Steel Piping

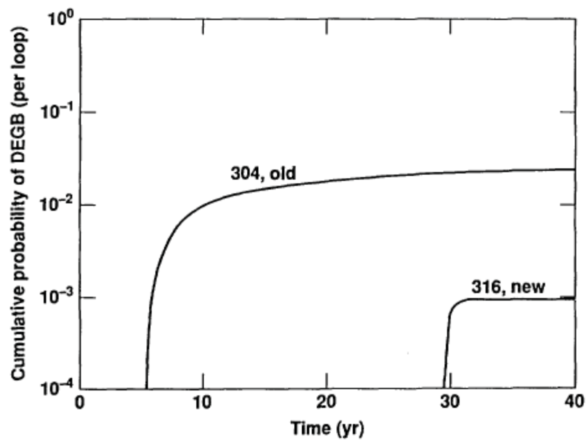


Figure 5.6 NUREG/CR-4792 Cumulative DEGB Probabilities Between the Old Configuration with TP304 Stainless Steel Piping and the New Configuration with TP316 Stainless Steel Piping

As mentioned previously, most operating BWR plants have implemented one or more of the IGSCC mitigation programs outlined the NRC Generic Letter 88-01 (Miraglia 1988) and NUREG-0313 Revision 2 (Hazelton and Koo 1988). These include minimizing the weld residual stresses through the application of improved welding techniques, use of weld overlays, or by the implementation of induction heat stress improvement or mechanical stress improvement process (MSIP) on existing welds. In addition most, if not all, operating plants control oxygen concentration below 20 ppb (parts per billion) by hydrogen addition to the reactor coolant.

In work performed by Failure Analysis Associates (1990), enhancements to the original PRAISE stress corrosion cracking models were made to account for improved piping reliability associated with the midlife implementation of IGSCC mitigation programs. These enhancements to PRAISE included mechanistic models that account for mid-life changes in: 1) residual stresses due to the implementation of thermal and mechanical stress improvement methods, 2) oxygen concentrations due to implantation of H2 water chemistry controls, coolant conductivity. Enhancements made to account for mid-life changes associated with piping replacements included changes in pipe wall thickness, dead weight stresses, thermal expansion stresses, and vibration stresses.

Figure 5.7 presents an example of the predicted effects of midlife dissolved oxygen concentrations on the cumulative leak probability of a BWR 304 stainless steel pipe. These results show that reducing the steady-state oxygen levels from 0.2 to 0.05 ppm decreases the cumulative leak probability by a factor of 2 for the following 10 operating years. Figure 5.8 presents the results for cases in which residual stresses in the weld region were changed by treatment with either induction heat stress improvement or mechanical stress improvement.

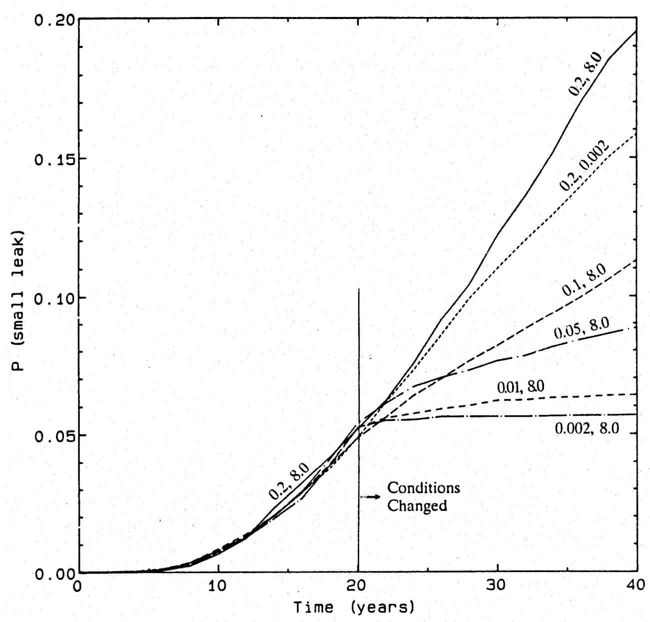


Figure 5.7 Effect of Lowering Oxygen Concentration at Midlife (20 years) in 102-mm (4-in.) Schedule 80 304 Stainless Steel BWR Pipe. The numbers by each line are the oxygen concentration (ppm) in the coolant during plant startup and steady-state operation, respectively (Failure Analysis Associates 1990).

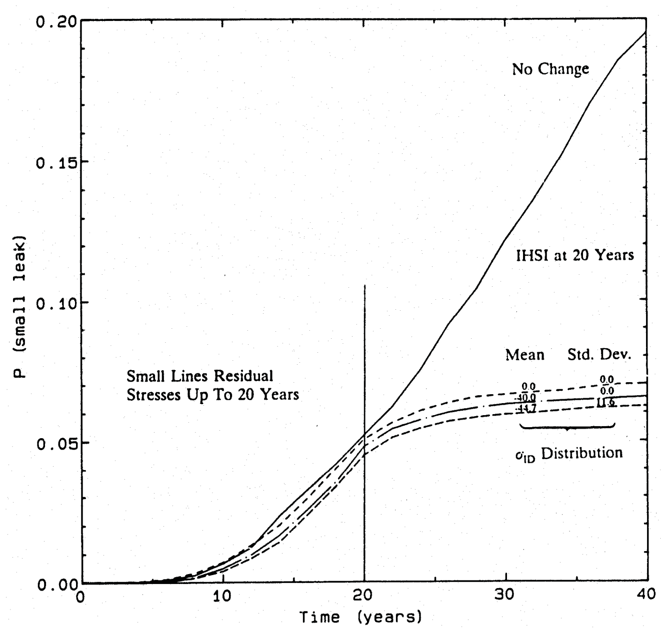


Figure 5.8 Effect of Reducing Residual Stress at Midlife (20 years) in 102-mm (4-in.) Schedule 80 Type 304 Stainless Steel BWR Pipe (Failure Analysis Associates 1990)

Many of the early studies focused on the probability of a DEGB; however, the DEGB is widely recognized as an extremely unlikely event. Therefore, the NRC is evaluating risk-informed design-basis break size requirements for operating commercial nuclear power plants. A central consideration in selecting a risk-informed break size is an understanding of conditional failure frequency as a function of break size (i.e., leak rate).

In a recent study (Tregoning et al. 2005), LOCA frequency estimates as a function of break size and operating time were developed using an expert elicitation process. This process was designed to consolidate service history data and insights from probabilistic fracture mechanics studies with knowledge of plant design, operation, and material performance. As part of this study, probabilistic fracture mechanics calculations were performed on a 304-mm (12-in.) and 711-mm (28-in.) Schedule 80 BWR recirculation piping. The recirculation piping was A-358 Class 1 Type 304 stainless steel. For the 304-mm (12-in.) pipe, analyses were performed assuming no remedial actions were taken and also assuming a weld overlay was performed at 20 years. No remedial action was the only case considered for the 711-mm (28-in.) line. Normal operating stresses were assumed to be 140.8 MPa (20.41 ksi) in the 304-mm (12-in.) line and 65.4 MPa (9.48 ksi) in the 711-mm (28-in.) line. The default residual stress distributions in NUREG/CR-4792 were assumed for the case when no mitigating actions (i.e., weld overlay) were performed. A linear approximation of the axisymmetric through-wall residual stress distribution in EPRI (1988) was assumed for the post-weld overlay. Table 5.5 summarizes the results for the 304-mm (12-in.) weld and the 711-mm (28-in.) weld in the recirculation system.

The beneficial effects of a weld overlay at 20 years are shown more clearly in Figure 5.9, which provides a plot of the cumulative probability of a leak exceeding 378.5 L/min (100 gpm) as a function of time. The slopes of the lines in Figure 5.9 are the leak frequencies; and the slope at 40 years with no overlay is approximately seven times that with an overlay at 20 years.

5.1.2 Transgranular Stress Corrosion Cracking

In addition to IGSCC, transgranular stress corrosion cracking (TGSCC) failures also have been reported for highly strained nonsensitized austenitic stainless steel piping base materials subjected to aggressive environments (e.g., oxygen/oxidizing species in conjunction with the presence of halides in PWR and BWR nuclear power plant reactor systems). TGSCC has occurred generally in regions of stagnant coolant at high elevation, where a high concentration of oxygen might have been present (Shah et al. 1998). For example, TGSCC was found in penetrations at the top of the reactor pressure vessel head. TGSCC pipe failures have also been reported for stainless steel piping subject to external chloride contamination from PVC tape applied to piping, and small- to medium-diameter cold bent piping where lubricants used in the bending machine contaminated the inside pipe wall and the bending process introduced high stresses. Summarized in Table 5.6 are the numbers of worldwide nuclear power plant service failures involving TGSCC; as reported in the proprietary PIPExp database described in Appendix D to NUREG-1829 (Tregoning et al. 2005).

Table 5.5 Cumulative Probability PRAISE Results for the 304-mm (12-in.) and 711-mm (28-in.) Recirculation Line Welds (Tregoning et al. 2005)

Leak Size (gpm)	Time (yr)	304-mm (12–in.) Schedule 80 [(a)]		711-mm (28–in.) Schedule 80 [(b)]
		No Weld Overlay	Weld Overlay at 20 Years	No Weld Overlay
>0	25	0.3674	0.2967	6.23×10^{-3}
	40	0.5986	0.3803	1.02×10^{-2}
	60	0.7435	0.4241	1.46×10^{-2}
>100	25	0.1682	0.1427	6.0×10^{-4}
	40	0.2452	0.1622	8.0×10^{-4}
	60	0.2872	0.1693	8.0×10^{-4}
>1500	25	0.1529	0.1066	6.66×10^{-5}
	40	0.2193	0.1250	6.66×10^{-5}
	60	0.2534	0.1312	1.25×10^{-4}
>5000	25	--	--	6.00×10^{-5}
	40	--	--	$6.87 \times 10^{-5}5$
	60	--	--	9.79×10^{-5}
DEGB	25	0.1529	0.0490	3.30×10^{-5}
	40	0.2193	0.0674	3.30×10^{-5}
	60	0.2535	0.0736	6.70×10^{-5}

(a) σ_{NO} = 140.8 MPa (20.41 ksi) + Seismic $\sigma \cong 0$ MPa (0 ksi)

(b) σ_{NO} = 65.4 MPa (9.48 ksi)+ Seismic σ = 7.72 MPa (1.12 ksi)

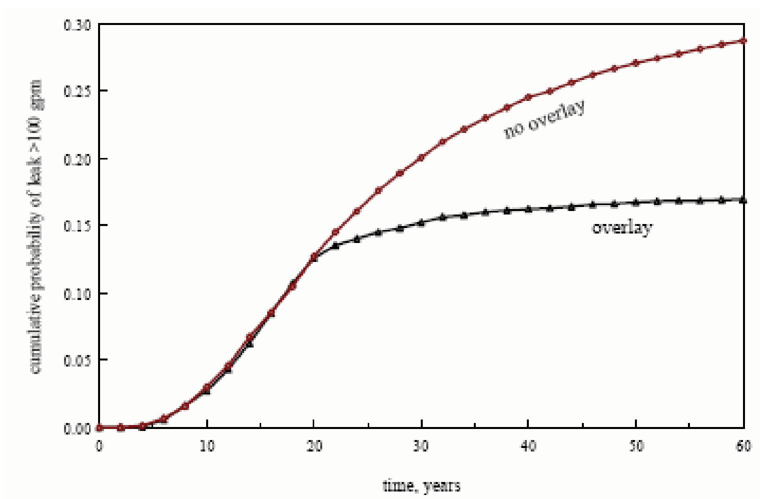


Figure 5.9 Cumulative Probability of a Leak Exceeding 378.5 lpm (100 gpm) Versus Time for a 304-mm (12-in.) Recirculation Line Weld With and Without a Weld Overlay at 20 Years (Tregoning et al. 2005)

Table 5.6 Summary of Service Experience Involving TGSCC

PlantType	Number of Reported Base Metal Failures (1970–2005)		
	All	Part Through-Wall	Through-Wall
BWR	209	149	60
PWR	123	24	99
BWR + PWR	332	173	159

- This tabulation is based on failure data as recorded in the proprietary PIPExp database; see Appendix D in NUREG-1829 (Tregoning et al. 2005).
- Maximum recorded leak rate is 6.4 lpm (1.7 gpm).

The literature survey did not identify any probabilistic fracture mechanics models or probabilistic evaluations to determine failure probabilities for TGSCC degradation of the kind observed in the field and reported in Table 5.6. Work reported by Lydell (1999) included a statistical assessment of service-induced TGSCC to determine the failure probability of cold bent piping. Depending on system and pipe size, the derived failure frequencies ranged from about 9.8×10^{-12} to about 1.5×10^{-7} per piping component per calendar year.

5.1.3 Primary Water Stress Corrosion Cracking

Excluding the service experience with steam generator tubes, incidents of stress corrosion cracking of Alloy 600 components including weld alloys 182/82 in domestic PWR primary systems have been observed since the early 1980s. This experience is well documented in numerous NRC bulletins, information notices, and generic letters (USNRC 1997, 2001a, 2001b, 2002, 2003a, 2003b, 2003c, 2004). An excellent history of foreign and domestic Alloy 600 cracking experience is provided in NUREG-1823 (Grimmel and Cullen 2005). Summarized in Table 5.7 are the numbers of worldwide nuclear power plant service failures involving primary water stress corrosion cracking (PWSCC), as reported in the proprietary PIPExp database described in Appendix D to NUREG-1829 (Tregoning et al. 2005).

Research, both domestically and internationally, is being performed to better characterize Alloy 600 cracking in light water reactors. The Office of Nuclear Regulatory Research programs include studies of 1) nondestructive inspection technologies, 2) material properties and crack growth rates, and 3) stress and structural integrity analysis. Additional work is being undertaken by the industry including the EPRI Materials Reliability Program, vendors, and owners' groups. With regard to structural integrity, the work reported in the literature has focused primarily on the need for qualified PWSCC crack initiation, and crack growth rate models have been reported by Garud (1997), Garud and Pathania (1999), EPRI (2002, 2004), White et al. (2003), and Cattant et al. (2005).

Garud (1997) proposed a simplified strain rate damage model (SRDM) for short PWSCC initiation times in low-temperature mil-annealed and cold-worked Alloy 600 tubing materials. In Figure 5.10, Garud (1997) compares SRDM predicted Alloy 600 initiation times with available initiation data in the literature for various test conditions in high-purity water environments.

Table 5.7 Summary of Service Experience in Piping Involving PWSCC

Pipe Size	Number of Reported Bi-Metallic Weld Failures (1970-2005)		
	All	Part Through-Wall	Through-Wall
OD ≤ 25.4 mm (≤1-in.)	62	22	40
25.4 mm (1 in.) < OD ≤ 254 mm (10-in.)	21	14	7
OD > 254 mm (10-in.)	2	1	1
All Pipe Sizes	85	37	48

- This tabulation is based on failure data as recorded in the proprietary PIPExp database; see Appendix D in NUREG-1829 (Tregoning et al. 2005).
- The data summary is limited to piping components (non-piping passive components are excluded).
- Maximum recorded through-wall leak rate is 3.1 L/min (0.8 gpm).

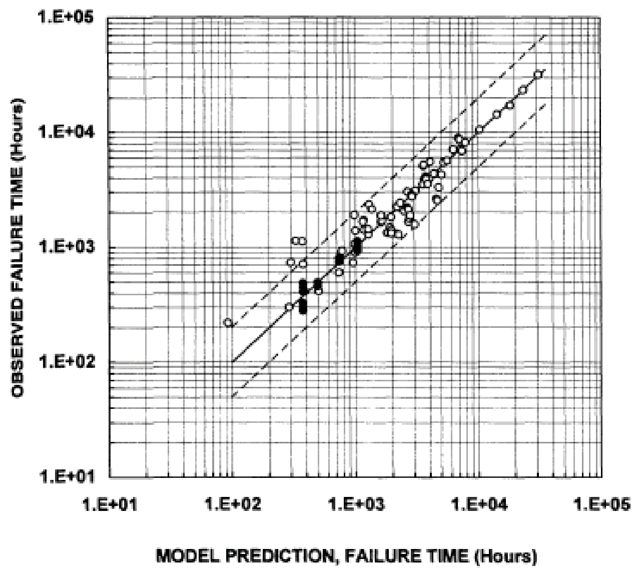


Figure 5.10 Comparisons of SRDM Estimates with All Available Data on Alloy 600 Initiation Life for Various Tests and Conditions in High-Purity Water Environments (Garud 1997)

EPRI recommended crack growth rate (CGR) curves for PWSCC of thick-wall components, such as reactor vessel head nozzles including control rod drive mechanism, control element drive mechanism, and in-core instrument nozzles, fabricated from Alloy 600 material (EPRI 2002) and Alloys 82 and182/132 (EPRI 2004), are shown in Figure 5.11 and Figure 5.12. In accordance with standard practice for evaluation of SCC, a power-law dependence on stress intensity factor, based upon the relationship originally developed by Scott (1991), was assumed. Tabulation of crack growth rates for both Alloy 600 and Alloy 182 allowed an evaluation showing that SCC growth rates in Alloy 182 weld metal were factors of 3–5 faster than rates in Alloy 600.

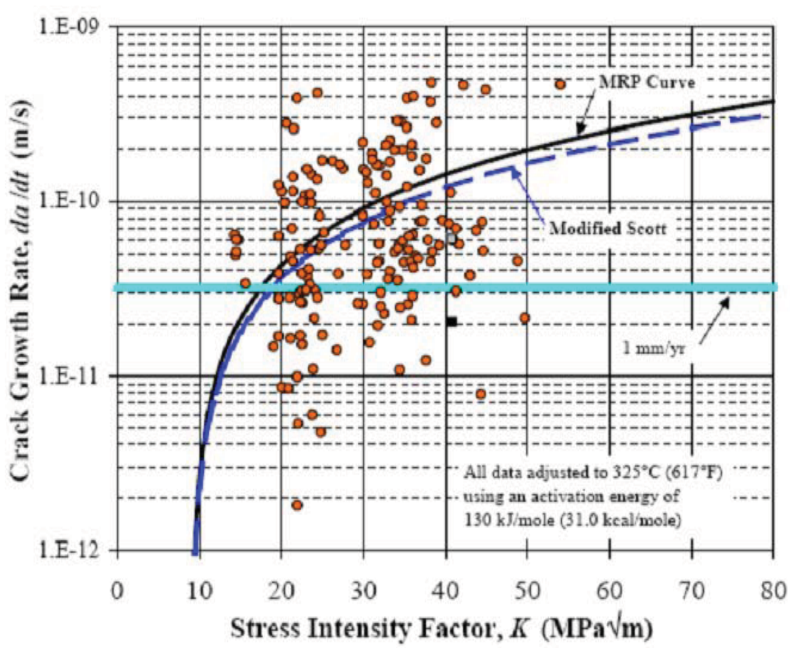


Figure 5.11 PWSCC Crack Growth Rate versus Crack Stress Intensity for Alloy 600 Laboratory Data, EPRI Material Reliability Program Crack Growth Curve, Modified Scott Curve, and CGR Data Points for Cook 2 Nozzle 75 (EPRI 2002). MPa√m = 0.91 ksi√in.

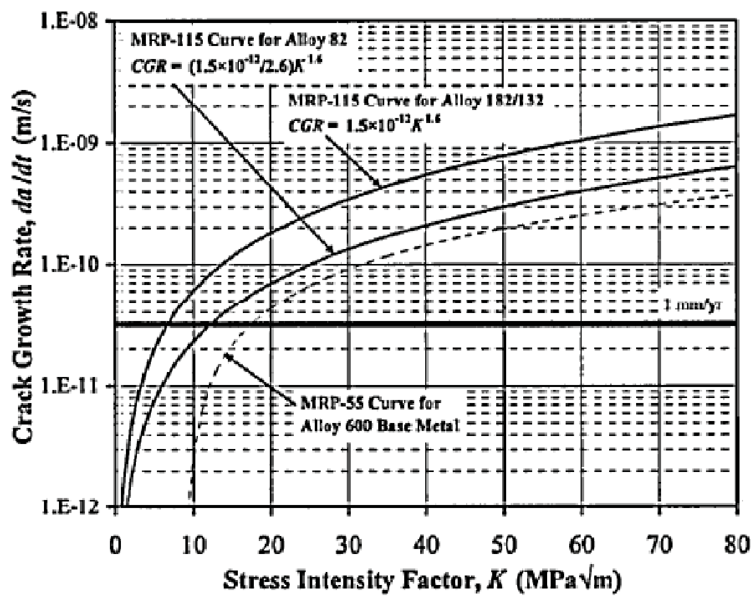


Figure 5.12 Comparisons of MRP-115 Curves for Alloys 182/132 and 82 with MRP 55 Westinghouse CGR Data for Weld Material Removed from VC Summer Reactor Hot Leg Nozzle Dissimilar Metal Weld (EPRI 2004). MPa√m = 0.91 ksi√in.

Probabilistic fracture mechanics calculations were used to estimate PWSCC failure frequencies in the 864-mm (34-in.) -diameter, 63.5-mm (2.5-in.) -thick PWR hot leg safe end-to-pressure vessel Alloy 600 weld (Tregoning et al. 2005). The hot leg material is SA-376 austenitic stainless steel. To model the initiation and growth of PWSCC cracks, in these analyses, the initiation kinetics were assumed to be the same as for Type 316NG stainless steel as currently in PRAISE (Harris et al. 1985); however, the crack growth kinetics in PRAISE were changed to be consistent with the EPRI MRP Alloy 600 crack growth rate recommendations (Garud 1997).

The cumulative probabilities for several leak rates and two modes of cracking are shown in Table 5.8. First, PWSCC for the growth of an initial fabrication defect with a depth distribution from Khaleel et al. (1999) was considered. In the second case, the initiation and growth of multiple PWSCC cracks were considered. We can see that, when a single preexisting fabrication crack is postulated, crack growth occurs at the start of plant life and all failures occurred within the first 25 years of plant operation. Consequently, the failure probability for small leaks is significantly higher than the larger leak sizes. In the second case, a significant operating time is required for PWSCC initiation and the crack growth phase is much shorter. Consequently, the small leak probability is much lower. Also, because multiple cracks tend to coalesce and grow as longer cracks, the probability of larger leaks is higher than when only a single fabrication crack is present.

As part of an ongoing NRC study, PWSCC probabilistic failure probability estimates for a PWR hot leg bi-metallic Alloy 600 weld have been performed using the PRO-LOCA code (Rudland et al., NRC, unpublished[a]). Initial results are expected to be published in the summer of 2006.

5.2 Fatigue

This section discusses two categories of fatigue degradation–thermal and vibration fatigue. Although both forms of fatigue degradation are well documented in the literature, most of the publications in the literature search were related to mechanistic aspects for initiation and crack growth modeling. The limited number of probabilistic fracture mechanics applications reported for nuclear plant components were associated mostly with thermal fatigue mechanisms.

5.2.1 Thermal Fatigue

Thermal fatigue has caused through-wall cracking and leakage in BWR and PWR plants from both low-cycle and high-cycle thermal fatigue. The service experience with thermal fatigue is extensively documented in reports by the Nuclear Energy Agency (1998) and Shah et al. (1997), as well as numerous NRC bulletins and information notices (USNRC 1979, 1988a, 1988b, 1988c, 1988d, 1989). Included in Table 5.9 is a summary of the thermal fatigue failures recorded in the proprietary PIPExp database as described in Appendix D of NUREG-1829 (Tregoning et al. 2005).

Experience (EPRI 1994, 2000; USNRC 1992) suggests that service-induced fatigue in operating plants are caused primarily by thermal stratification or hot-cold water mixing conditions, such as thermal

(a) Rudland DL, H Xu, G Wilkowski, N Ghadiali, F Brust, and P Scott. Unpublished. *Evaluation of Loss-of-Coolant Accident (LOCA) Frequencies Using the PRO-LOCA Code*, NUREG/CR-XXXX, currently under review.

Table 5.8 Cumulative PRAISE Failure Probabilities for Hot Leg Pressure Vessel-to-Safe End Alloy 600 Weld (Tregoning et al. 2005)

Leak Size	Time	PWSCC Growth of Single Fabrication Defect	PWSCC Initiation and Growth of Multiple Cracks
>0 lpm (>0 gpm)	25	0.916	0.001
	40	0.918	0.020
	60	0.919	0.068
>378.5 lpm (>100 gpm)	25	2.167×10^{-7}	1×10^{-5}
	40	2.167×10^{-7}	2.69×10^{-4}
	60	2.167×10^{-7}	1.78×10^{-3}
>5677 lpm (>1500 gpm)	25	2.78×10^{-11}	$<1 \times 10^{-4}$
	40	2.78×10^{-11}	1×10^{-4}
	60	2.78×10^{-11}	4.85×10^{-4}
>18926 lpm (>5000 gpm)	25	4.66×10^{-11}	$<1 \times 10^{-5}$
	40	4.66×10^{-11}	9×10^{-5}
	60	4.66×10^{-11}	3.77×10^{-4}
DEGB	25	2.59×10^{-13}	$<1 \times 10^{-5}$
	40	2.59×10^{-13}	9×10^{-5}
	60	2.59×10^{-13}	3.77×10^{-5}

Table 5.9 Summary of Service Experience Involving Thermal Fatigue (Tregoning et al. 2005)

Plant Type	Number of Pipe Failures Attributed to Thermal Fatigue (1970–2005)			
	All Failure Types	Part Through-Wall Crack	Small Through-Wall Leak	Large Through-Wall Leak
BWR	69	40	27	2
PWR	87	42	40	5
All	156	82	67	7

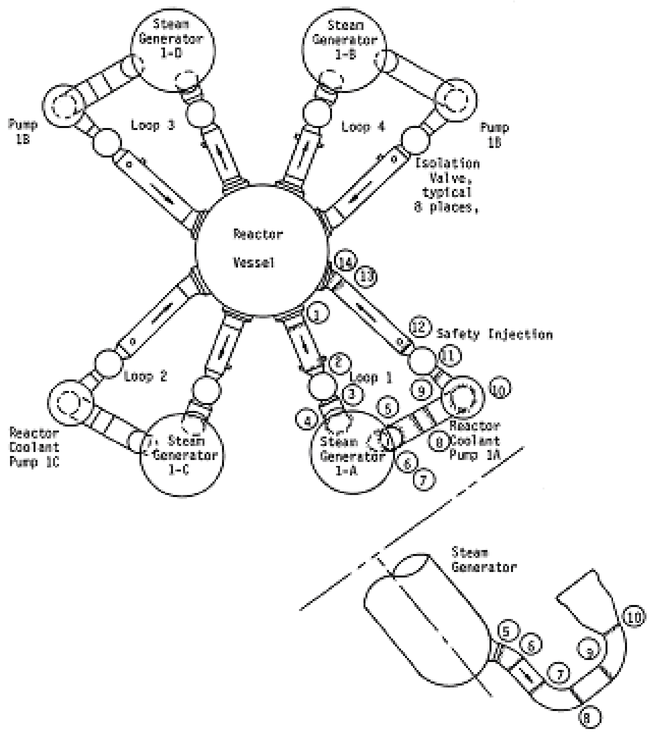
striping, and cyclic turbulent mixing not analyzed in the original plant designs. These loading conditions are known to impose many cycles, with the cyclic boundary conditions changing rapidly causing cracks to initiate and grow at multiple locations on the inside surface of the pipe. Because of the large nonlinear gradient stress profiles associated with these service conditions, cracks will tend to grow in the length direction where the highest surface cyclic stresses occur. This will cause many of these cracks to coalesce and grow as effectively longer defects. Another large body of fatigue cracking events relates to PWR piping (USNRC 1990a) and BWR feedwater and control rod drive return line nozzles (Snaider 1980; USNRC 1980). In both situations, relatively colder water flows into a nozzle of a hot vessel. For the PWRs, the cracking initiated at the weld between the nozzle/safe-end and the attached piping. In the BWRs, the cracking occurred very locally in the vessel nozzle and was due to localized mixing of the hot and cold water on the nozzle bore and blend-radius regions. Again, these loadings were unknown at the time of the original plant design.

In NUREG/CR-2189, Harris et al. (1981) documented the probabilistic fracture mechanics calculations performed with the early version of the PRAISE code to estimate the probability of a seismic-induced LOCA in the primary piping of a commercial pressurized water reactor. The primary piping system at Zion 1 was selected for these analyses. Attention was focused on the girth butt welds in the hot leg, cold leg, and cross-over piping. The primary piping was austenitic stainless steel, predominantly centrifugally cast stainless steel. Zion 1 was a Westinghouse 1100-MWe four-loop PWR design. A listing of the 11 Zion 1 primary system design thermal transients considered in these computations along with their postulated number of occurrences over the 40-year plant life is shown in Table 5.10. Appendix D of NUREG/CR-2189 included the design time-temperature profiles in the Zion 1 hot leg, cold leg, and steam lines for each of the design transients in Table 5.10. The Zion 1 primary system piping, with the 14 weld locations considered and the resulting failure rates (i.e., leaks per weld per year), is shown in Figure 5.13. Figure 5.14 provides an example of the conditional LOCA probabilities for weld joints 1 and 13 for various magnitudes of seismic events.

NUREG/CR-6674 (Khaleel et al. 2000) documents calculations performed with the pc-PRAISE code to address fatigue failures for 47 selected components for seven representative plants covering both BWR and PWR plants. The calculations also covered all four nuclear steam supply system suppliers and both older and newer vintage plants. These calculations predicted probabilities of crack initiation and probabilities of through-wall cracks as a function of time for plant operating periods up to 60 years. The calculations were made possible by the development of a new version of the pc-PRAISE code that has the capability to simulate the initiation of fatigue cracks in combination with a simulation of the subsequent growth of these fatigue cracks. Probabilities of crack initiation were calculated using fatigue-life correlations from the Argonne National Laboratory (ANL) research for NRC (Keisler et al. 1995). The number of cycles to crack initiation in the Argonne equations is a function of the material type, water/air

Table 5.10 Zion 1 Primary System Piping Design Transients and Number of Postulated Occurrences in 40-year Design Life (Harris et al. 1981)

Design Transient	Design Cycles
Heatup and Cooldown at 37.7°C/hr (100°F/hr)	200 (each)
Unit Loading and Unloading at 5% of full power/minute	18,300 (each)
Step Load Increase and Decrease at 10% of full power	2,000 (each)
Large Step Load Decrease	200
Loss of Load	80
Loss of Power	40
Loss of Flow (partial loss)	80
Reactor Trip from Full Power	400
Steam Line Break	1
Turbine Roll Test	10
Hydrostatic Test Condition	5



Weld	Leak Frequency (yr-1)
*1	2.05E-09
2	5.83E-09
3	9.03E-11
*4	1.27E-10
*5	8.20E-11
6	3.20E-09
7	3.15E-09
8	2.01E-09
9	2.55E-09
10	9.23E-11
11	6.78E-13
12	8.15E-09
13	8.60E-09
*14	1.86E-10

* Denotes terminal end

Figure 5.13 Zion 1 Primary System Piping Weld Locations and NUREG/CR-2189 Estimated Leak Frequencies (Heasler and Simonen 1990)

environment, temperature, dissolved oxygen content, sulfur content, and strain rate. The alternating stresses and anticipated number of cycles were the same as those used for the deterministic calculations of fatigue usage factors of NUREG/CR-6260 (Ware et al. 1995). Detailed inputs and results for the probabilistic fracture mechanics calculations are given in NUREG/CR-6674.

Tables 5.11 and 5.12 provide a summary of the results for all the components, with Table 5.11 accounting for the effects of reactor water environments and Table 5.12 neglecting the environmental effects. Results are given for both a 40-year and a 60-year operating period. Many of the components analyzed in NUREG/CR-6674 had cumulative probabilities of crack initiation[a] and cumulative probabilities of through-wall cracks that approach unity within the 40-year to 60-year time period. Other components, often with similar values of fatigue usage factors, show much lower failure probabilities. The maximum failure rate (through-wall cracks per year) is about 5×10^{-2}. Tables 5.11 and 5.12 also show estimated values for core-damage frequencies, which show maximum values based on the calculated failure rates of about 1.0×10^{-6} per year. These maximum values correspond to components with very high cumulative failure probabilities, and the failure rates do not change significantly from 40 years to 60 years. Failure

(a) The ANL S-N model associates fatigue life with approximately a 3-mm (0.020-in.) surface flaw depth (Gavenda et al. 1997; Chopra and Shack 2001; Chopra 2002) and is based on consideration of the 25% load drop criteria used to denote fatigue failure of smooth push-pull specimens in fatigue testing procedures.

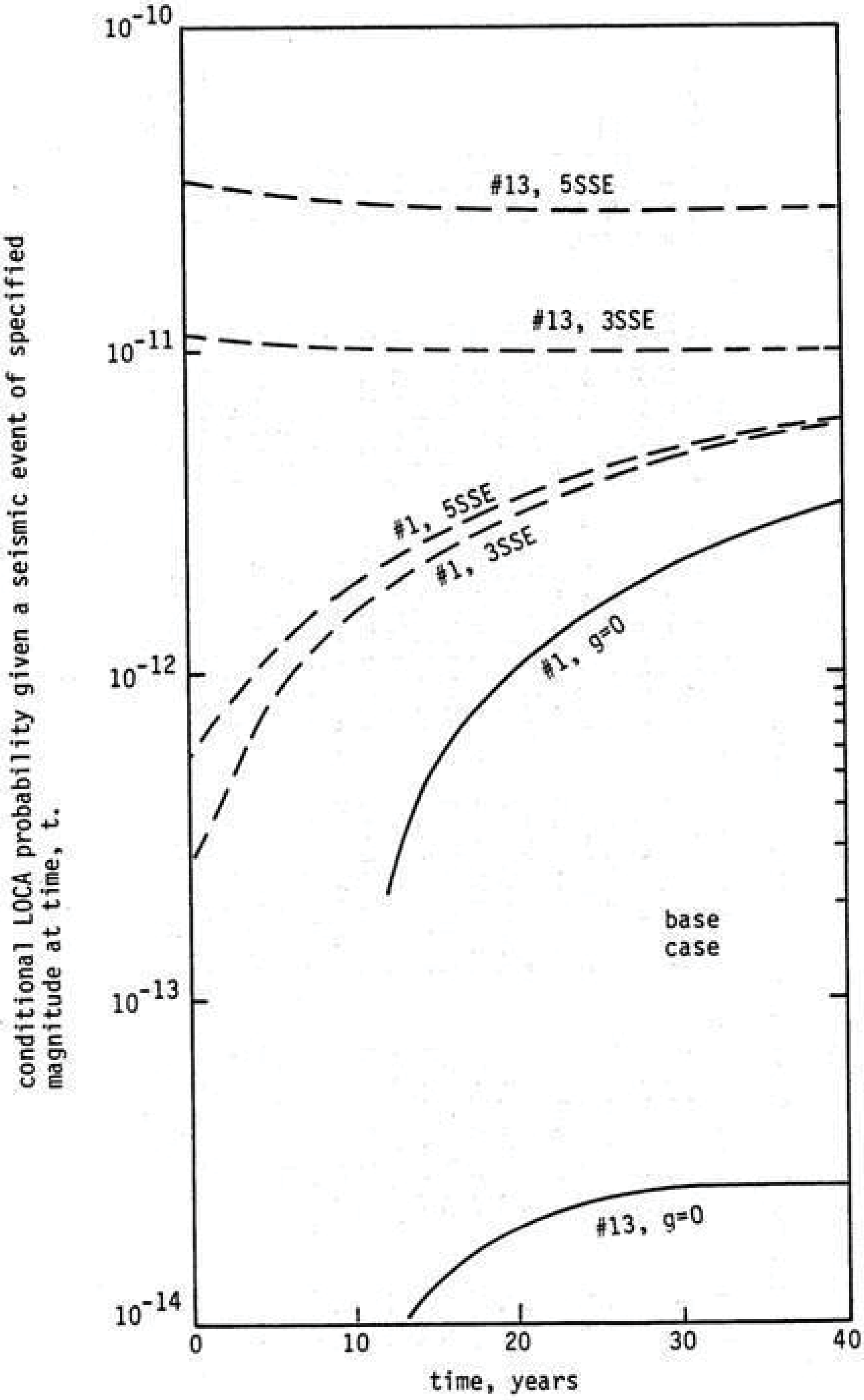


Figure 5.14 **NUREG/CR-2189 Conditional LOCA Probabilities as a Function of Time for Two Zion I Representative Weld Locations Showing the Influence of Seismic Events (Harris et al. 1981)**

Table 5.11 Summary of Results for All Seven Plants - Water Environment (Khaleel et al. 2000)

PLANT	COMPONENT	MAT	DET USEAGE@40	DET USEAGE@60	CUMULATIVE PI@40 YR	CUMULATIVE PI@60 YR	CUMULATIVE PTWC@40 YR	CUMULATIVE PTWC@60 YR	TWC/YEAR @40 YR	TWC/YEAR @60 YR	CDF @40 YR	CDF @60 YR
CE-NEW	RPV LOWER HEAD/SHELL	LAS	1.40E-02	2.10E-02	7.89E-06	4.82E-05	6.71E-15	1.44E-12	1.13E-15	1.90E-13	1.13E-16	1.91E-14
CE-NEW	RPV INLET NOZZLE	LAS	4.75E-01	7.12E-01	1.40E-02	4.44E-02	5.90E-05	9.01E-04	7.50E-06	7.59E-05	2.03E-11	2.05E-10
CE-NEW	RPV OUTLET NOZZLE	LAS	4.72E-01	7.08E-01	4.22E-01	6.89E-01	1.74E-03	2.90E-02	3.58E-04	2.57E-03	9.65E-10	6.93E-09
CE-NEW	SURGE LINE ELBOW	304/316	2.60E+00	3.90E+00	9.95E-01	9.99E-01	9.81E-01	9.98E-01	7.60E-02	9.38E-02	2.17E-06	2.67E-06
CE-NEW	CHARGING NOZZLE NOZZLE	LAS	1.04E-01	1.56E-01	9.56E-04	3.84E-03	2.61E-06	5.50E-05	3.46E-07	5.06E-06	2.77E-12	4.05E-11
CE-NEW	CHARGING NOZZLE SAFE END	304/316	5.02E-01	7.53E-01	1.06E-02	6.75E-02	9.00E-05	1.03E-03	1.75E-05	1.15E-04	1.40E-09	9.21E-09
CE-NEW	SAFETY INJECTION NOZZLE NOZZLE	LAS	4.57E-01	6.85E-01	1.01E-03	4.81E-03	1.00E-06	1.90E-05	3.75E-07	1.50E-06	1.88E-12	7.50E-12
CE-NEW	SAFETY INJECTION NOZZLE SAFE E	304/316	2.86E-01	4.29E-01	8.68E-03	3.16E-02	2.61E-06	5.50E-05	3.46E-07	5.06E-06	1.73E-11	2.53E-10
CE-NEW	SHUTDOWN COOLING LINE ELBOW	304/316	4.87E-01	7.30E-01	1.13E-02	5.75E-02	2.00E-05	4.53E-04	7.00E-06	4.40E-05	1.89E-10	1.19E-09
CE-OLD	RPV LOWER HEAD/SHELL	LAS	1.30E-02	1.95E-02	2.68E-06	1.93E-05	6.36E-16	1.85E-13	1.07E-16	1.85E-13	1.08E-17	1.86E-14
CE-OLD	RPV INLET NOZZLE	LAS	1.72E-01	2.58E-01	1.88E-03	7.89E-03	4.11E-07	1.33E-05	5.87E-08	1.33E-05	1.58E-13	3.59E-11
CE-OLD	RPV OUTLET NOZZLE	LAS	5.53E-01	8.29E-01	5.91E-01	8.46E-01	7.05E-02	3.53E-01	8.98E-03	2.27E-02	2.42E-08	6.13E-08
CE-OLD	SURGE LINE ELBOW	304/316	6.61E-01	9.92E-01	9.39E-01	9.87E-01	6.27E-01	8.85E-01	4.36E-02	5.48E-02	1.24E-06	1.56E-06
CE-OLD	CHARGING NOZZLE SAFE END	304/316	5.62E-01	8.43E-01	1.18E-02	5.31E-02	4.10E-05	5.98E-04	8.75E-06	5.05E-05	7.00E-10	4.04E-09
CE-OLD	SAFETY INJECTION NOZZLE SAFE E	304/316	3.17E-01	4.75E-01	7.56E-03	3.59E-02	1.40E-05	2.00E-04	2.25E-06	1.85E-05	1.13E-10	9.25E-10
CE-OLD	SHUTDOWN COOLING LINE ELBOW	304/316	8.40E-02	1.26E-01	3.94E-02	1.19E-01	2.10E-04	2.36E-03	4.38E-05	1.98E-04	1.18E-09	5.35E-09
B&W	RPV NEAR SUPPORT SKIRT	LAS	2.23E-01	3.35E-01	8.25E-03	2.50E-02	7.85E-06	1.52E-04	1.04E-06	1.36E-05	1.04E-07	1.36E-06
B&W	RPV OUTLET NOZZLE	LAS	4.69E-01	7.04E-01	7.74E-01	8.99E-01	1.83E-01	5.44E-01	1.94E-02	3.35E-02	5.25E-08	9.03E-08
B&W	MAKEUP/HPI NOZZLE SAFE END	304/316	1.05E+00	1.58E+00	1.30E-01	4.79E-01	2.10E-03	3.09E-02	5.88E-04	2.22E-03	2.94E-08	1.11E-07
B&W	DECAY HEAT REMOVAL/REDUCING T	304/316	5.30E-01	7.95E-01	5.72E-02	2.08E-01	3.00E-03	2.54E-02	4.26E-04	1.79E-03	1.15E-08	4.82E-08
W-NEW	RPV LOWER HEAD/SHELL	LAS	1.80E-02	2.70E-02	3.21E-05	1.71E-04	7.52E-13	9.64E-11	1.23E-13	1.21E-11	1.24E-14	1.50E-12
W-NEW	RPV INLET NOZZLE	LAS	2.90E-01	4.35E-01	2.49E-03	1.05E-02	9.17E-07	2.84E-05	1.30E-07	2.83E-06	3.51E-13	7.64E-12
W-NEW	RPV OUTLET NOZZLE	LAS	6.58E-01	9.87E-01	8.62E-01	9.49E-01	3.65E-01	7.42E-01	3.17E-02	4.50E-02	8.57E-08	1.22E-07
W-NEW	CHARGING NOZZLE NOZZLE	316NG	3.37E+00	5.06E+00	9.51E-01	9.83E-01	8.72E-01	9.63E-01	5.38E-02	5.66E-02	4.31E-07	4.53E-07
W-NEW	SAFETY INJEC NOZZLE NOZZLE	316NG	1.46E+00	2.19E+00	4.34E-03	3.69E-02	5.00E-04	1.09E-02	5.33E-05	1.30E-03	2.67E-10	6.50E-09
W-NEW	RESIDUAL HEAT INLET TRAN	304/316	2.73E+00	4.10E+00	9.58E-01	9.99E-01	7.80E-01	9.80E-01	6.25E-02	1.18E-01	1.69E-06	3.17E-06
W-OLD	RPV LOWER HEAD SHELL	LAS	8.91E-01	1.34E+00	1.11E-01	1.28E-01	7.20E-07	1.11E-05	8.38E-08	9.08E-07	8.44E-09	9.15E-08
W-OLD	RPV INLET NOZZLE INNER SURFACE	LAS	3.02E-01	4.53E-01	3.91E-01	6.44E-01	4.38E-03	5.04E-02	7.53E-04	3.96E-03	2.03E-09	1.07E-08
W-OLD	RPV INLET NOZZLE OUTER SURFACE	LAS	4.96E-01	7.44E-01	6.81E-02	1.11E-01	4.48E-04	3.32E-03	4.75E-05	2.18E-04	1.28E-10	5.89E-10
W-OLD	RPV OUTLET NOZZLE INNER SURF	LAS	4.99E-01	7.48E-01	4.90E-01	7.53E-01	9.33E-03	9.60E-02	1.56E-03	7.54E-03	4.21E-09	2.04E-08
W-OLD	RPV OUTLET NOZZLE OUTER SURF	LAS	3.47E-01	5.20E-01	1.63E-01	2.38E-01	7.77E-03	3.60E-02	6.99E-04	1.83E-03	1.89E-09	4.94E-09
W-OLD	CHARGING NOZZLE NOZZLE	304/316	3.19E-01	4.79E-01	4.67E-04	3.75E-03	3.00E-07	5.20E-06	7.50E-08	6.00E-07	6.00E-13	4.80E-12
W-OLD	SAFETY INJECTION NOZZLE NOZZLE	304/316	3.27E-01	4.90E-01	1.88E-03	1.31E-02	4.00E-06	8.80E-05	8.75E-07	1.05E-05	4.38E-12	5.25E-11
W-OLD	RESIDUAL HEAT REMOVAL TEE	304/316	2.05E-01	3.08E-01	1.34E-02	5.16E-02	1.15E-04	1.14E-03	1.63E-05	9.26E-05	8.13E-10	4.63E-09
GE-NEW	RPV NEAR CRDM PENETRATION	LAS	6.28E-01	9.42E-01	7.89E-05	3.49E-04	7.88E-12	6.82E-10	1.25E-12	8.26E-11	1.25E-13	8.26E-12
GE-NEW	FEEDWATER NOZZLE SAFE END	LAS	1.88E+00	2.82E+00	1.04E-01	2.53E-01	1.31E-03	1.47E-02	2.38E-04	1.23E-03	3.57E-11	1.84E-10
GE-NEW	RECIRC SYS - TEE SUCTION PIPE	304/316	8.30E-01	1.25E+00	4.23E-02	1.39E-01	4.80E-04	4.67E-03	7.13E-05	3.66E-04	1.07E-11	5.49E-11
GE-NEW	CORE SPRAY LINE SAFE END EXT	LAS	4.36E-01	6.54E-01	3.83E-04	1.27E-03	1.45E-07	3.25E-06	1.97E-08	3.04E-07	7.09E-15	1.09E-13
GE-NEW	RHR LINE STRAIGHT PIPE	LAS	1.13E+01	1.69E+01	4.73E-01	6.71E-01	4.10E-01	6.21E-01	1.35E-02	2.25E-02	2.54E-11	2.03E-10
GE-NEW	FEEDWATER LINE ELBOW	LAS	3.69E+00	5.53E+00	1.59E-01	3.65E-01	1.01E-03	1.46E-02	1.69E-04	1.35E-03	3.04E-09	5.06E-09
GE-OLD	RPV LOWER HEAD TO SHELL	LAS	7.90E-02	1.19E-01	2.71E-10	2.76E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
GE-OLD	RPV FEEDWATER NOZZLE BORE	LAS	3.17E+00	4.75E+00	7.27E-02	2.42E-01	1.00E-05	8.80E-04	2.50E-06	9.76E-05	3.75E-14	1.46E-12
GE-OLD	RECIRC SYSTEM RHR RETURN LINE	304/316	3.90E+00	5.85E+00	9.43E-01	9.99E-01	7.12E-01	9.85E-01	7.20E-02	1.23E-01	1.08E-08	1.84E-08
GE-OLD	CORE SPRAY SYSTEM NOZZLE	LAS	5.20E-01	7.80E-01	1.41E-04	7.89E-04	1.91E-08	8.84E-07	2.85E-09	9.51E-08	6.41E-17	2.14E-15
GE-OLD	CORE SPRAY SYSTEM SAFE END	304/316	1.77E+00	2.65E+00	3.33E-01	7.64E-01	1.46E-02	1.10E-01	2.08E-03	8.04E-03	4.68E-10	1.81E-09
GE-OLD	RESIDUAL HEAT TAPERED	304/316	4.78E-01	7.17E-01	1.47E-03	7.89E-03	9.21E-05	1.02E-03	1.07E-05	7.82E-05	1.61E-12	1.17E-11
GE-OLD	FEEDWATER LINE - RCIC TEE	LAS	6.98E+00	1.05E+01	3.86E-01	7.82E-01	2.99E-03	5.92E-02	6.96E-04	5.54E-03	1.04E-10	8.30E-10

Table 5.12 Summary of Results for All Seven Plants - Air Environment (Khaleel et al. 2000)

PLANT	COMPONENT	MAT	ASME DESIGN USEAGE@40	ASME DESIGN USEAGE@60	CUMULATIVE PI@40 YR	CUMULATIVE PI@60 YR	CUMULATIVE PTWC@40 YR	CUMULATIVE PTWC@60 YR	TWC/YEAR @40 YR	TWC/YEAR @60 YR	CDF @40 YR	CDF @60 YR
CE-NEW	RPV LOWER HEAD/SHELL	LAS	7.00E-03	1.05E-02	1.30E-06	1.00E-05	8.40E-23	3.96E-19	1.64E-23	6.37E-20	1.65E-24	6.39E-21
CE-NEW	RPV INLET NOZZLE	LAS	1.82E-01	2.73E-01	4.24E-03	1.64E-02	1.75E-07	7.64E-06	2.57E-08	8.18E-07	6.94E-14	2.21E-12
CE-NEW	RPV OUTLET NOZZLE	LAS	3.34E-01	5.01E-01	2.21E-01	4.55E-01	1.00E-07	7.70E-06	2.50E-08	1.02E-06	6.75E-14	2.75E-12
CE-NEW	SURGE LINE ELBOW	304/316	9.81E-01	1.47E+00	6.48E-01	8.26E-01	4.52E-01	7.11E-01	2.69E-02	5.16E-02	7.68E-07	1.47E-06
CE-NEW	CHARGING NOZZLE NOZZLE	LAS	5.00E-02	7.50E-02	2.27E-04	1.16E-03	6.48E-11	5.96E-09	1.04E-11	7.22E-10	8.32E-17	5.78E-15
CE-NEW	CHARGING NOZZLE SAFE END	304/316	7.78E-01	1.17E+00	3.83E-04	3.16E-03	1.62E-04	1.76E-03	1.90E-05	1.33E-04	1.52E-09	1.06E-08
CE-NEW	SAFETY INJECTION NOZZLE NOZZLE	LAS	8.98E-01	1.35E+00	2.15E-03	8.69E-03	1.22E-06	3.18E-05	1.66E-07	3.09E-06	8.30E-13	1.55E-11
CE-NEW	SAFETY INJECTION NOZZLE SAFE E	304/316	3.60E-01	5.40E-01	2.12E-05	1.64E-04	1.04E-06	1.59E-05	1.29E-07	1.37E-06	6.45E-12	6.85E-11
CE-NEW	SHUTDOWN COOLING LINE ELBOW	304/316	8.94E-01	1.34E+00	1.56E-04	1.00E-03	1.83E-05	2.01E-04	2.10E-06	1.55E-05	5.67E-11	4.19E-10
CE-OLD	RPV LOWER HEAD/SHELL	LAS	8.00E-03	1.20E-02	4.03E-07	3.47E-06	4.85E-24	3.80E-20	9.51E-25	5.95E-21	9.56E-26	5.98E-22
CE-OLD	RPV INLET NOZZLE	LAS	7.30E-02	1.10E-01	4.44E-04	2.27E-03	7.99E-11	7.84E-09	1.26E-11	9.69E-10	3.40E-17	2.62E-15
CE-OLD	RPV OUTLET NOZZLE	LAS	2.84E-01	4.26E-01	7.89E-02	1.28E-01	6.72E-04	4.79E-03	7.07E-05	3.09E-04	1.91E-10	8.34E-10
CE-OLD	SURGE LINE ELBOW	304/316	7.05E-01	1.06E+00	2.62E-01	4.91E-01	1.74E-02	9.10E-02	2.02E-03	5.46E-03	5.76E-08	1.56E-07
CE-OLD	CHARGING NOZZLE SAFE END	304/316	2.66E-01	3.99E-01	5.62E-04	3.16E-03	9.92E-05	8.92E-04	1.09E-05	6.45E-05	8.72E-10	5.16E-09
CE-OLD	SAFETY INJECTION NOZZLE SAFE E	304/316	8.80E-02	1.32E-01	2.39E-04	1.34E-03	2.03E-05	2.18E-04	2.36E-06	1.65E-05	1.18E-10	8.25E-10
CE-OLD	SHUTDOWN COOLING LINE ELBOW	304/316	-	-	8.69E-04	3.16E-03	5.13E-05	3.78E-04	5.28E-06	2.53E-05	1.43E-10	6.83E-10
B&W	RPV NEAR SUPPORT SKIRT	LAS	3.60E-01	5.40E-01	2.61E-03	9.56E-03	4.07E-09	2.64E-07	6.23E-10	3.01E-08	6.24E-11	3.01E-09
B&W	RPV OUTLET NOZZLE	LAS	1.43E-01	2.15E-01	7.89E-02	1.05E-01	2.92E-03	1.27E-02	2.49E-04	6.50E-04	6.72E-10	1.76E-09
B&W	MAKEUP/HPI NOZZLE SAFE END	304/316	7.40E-01	1.11E+00	3.03E-03	1.98E-02	1.91E-03	1.48E-02	2.07E-04	1.02E-03	1.04E-08	5.10E-08
B&W	DECAY HEAT REMOVAL/REDUCING T	304/316	-	-	1.05E-03	4.89E-03	1.70E-04	1.34E-03	1.80E-05	9.13E-05	4.86E-10	2.47E-09
W-NEW	RPV LOWER HEAD/SHELL	LAS	1.00E-02	1.50E-02	6.19E-06	4.10E-05	1.04E-19	8.98E-17	1.79E-20	1.39E-17	1.80E-21	1.40E-18
W-NEW	RPV INLET NOZZLE	LAS	1.10E-01	1.65E-01	5.62E-04	3.03E-03	7.03E-10	6.54E-08	1.11E-10	7.93E-09	3.00E-16	2.14E-14
W-NEW	RPV OUTLET NOZZLE	LAS	3.98E-01	5.97E-01	7.26E-01	8.76E-01	3.00E-04	2.20E-03	2.50E-05	2.00E-04	6.75E-11	5.40E-10
W-NEW	CHARGING NOZZLE NOZZLE	316NG	8.29E-01	1.24E+00	6.18E-01	8.11E-01	3.17E-01	6.06E-01	2.18E-02	2.75E-02	1.74E-07	2.20E-07
W-NEW	SAFETY INJEC NOZZLE NOZZLE	316NG	9.66E-01	1.45E+00	2.89E-04	2.61E-03	1.93E-04	2.06E-03	2.25E-05	1.50E-04	1.13E-10	7.51E-10
W-NEW	RESIDUAL HEAT INLET TRAN	304/316	8.96E-01	1.34E+00	1.66E-01	3.79E-01	2.90E-02	1.32E-01	3.09E-03	8.11E-03	8.34E-08	2.19E-07
W-OLD	RPV LOWER HEAD SHELL	LAS	2.90E-01	4.35E-01	9.56E-02	1.11E-01	1.06E-09	3.34E-08	1.37E-10	3.27E-09	1.38E-11	3.29E-10
W-OLD	RPV INLET NOZZLE INNER SURFACE	LAS	1.35E-01	2.03E-01	2.06E-01	4.21E-01	2.00E-07	1.03E-05	1.00E-07	1.15E-06	2.70E-13	3.11E-12
W-OLD	RPV INLET NOZZLE OUTER SURFACE	LAS	2.08E-01	3.12E-01	6.82E-02	1.11E-01	4.48E-04	3.33E-03	4.75E-05	2.19E-04	1.28E-10	5.91E-10
W-OLD	RPV OUTLET NOZZLE INNER SURF	LAS	1.93E-01	2.90E-01	2.67E-01	5.22E-01	2.00E-05	6.00E-05	1.25E-06	2.50E-06	3.38E-12	6.75E-12
W-OLD	RPV OUTLET NOZZLE OUTER SURF	LAS	4.31E-01	6.47E-01	1.63E-01	2.39E-01	7.77E-03	3.60E-02	7.00E-04	1.83E-03	1.89E-09	4.94E-09
W-OLD	CHARGING NOZZLE NOZZLE	304/316	4.60E-01	6.90E-01	1.53E-05	1.48E-04	8.90E-07	1.68E-05	1.15E-07	1.52E-06	9.20E-13	1.22E-11
W-OLD	SAFETY INJECTION NOZZLE NOZZLE	304/316	1.99E+00	2.99E+00	4.10E-05	3.67E-04	4.77E-06	7.46E-05	5.95E-07	6.40E-06	2.98E-12	3.20E-11
W-OLD	RESIDUAL HEAT REMOVAL TEE	304/316	2.20E-02	3.30E-02	4.89E-04	2.27E-03	3.01E-05	2.86E-04	3.38E-06	2.07E-05	1.69E-10	1.04E-09
GE-NEW	RPV NEAR CRDM PENETRATION	LAS	4.07E-01	6.11E-01	1.94E-05	1.00E-04	3.59E-18	1.55E-15	6.03E-19	2.18E-16	6.03E-20	2.18E-17
GE-NEW	FEEDWATER NOZZLE SAFE END	LAS	3.01E-01	4.52E-01	4.31E-02	1.19E-01	2.00E-06	2.00E-05	1.25E-07	1.75E-06	1.88E-14	2.63E-13
GE-NEW	RECIRC SYS - TEE SUCTION PIPE	304/316	2.98E-01	4.47E-01	2.61E-03	1.00E-02	6.50E-04	3.59E-03	5.96E-05	2.16E-04	8.94E-12	3.24E-11
GE-NEW	CORE SPRAY LINE SAFE END EXT	LAS	5.00E-02	7.50E-02	1.22E-04	4.65E-04	6.45E-12	5.21E-10	1.02E-12	6.26E-11	3.67E-19	2.25E-17
GE-NEW	RHR LINE STRAIGHT PIPE	LAS	4.07E-01	6.11E-01	3.71E-01	4.99E-01	2.08E-01	3.02E-01	7.21E-03	6.09E-03	1.35E-11	1.14E-11
GE-NEW	FEEDWATER LINE ELBOW	LAS	4.35E-01	6.53E-01	6.77E-02	1.82E-01	1.00E-05	6.00E-05	1.25E-06	5.00E-06	2.25E-11	9.00E-11
GE-OLD	RPV LOWER HEAD TO SHELL	LAS	3.20E-02	4.80E-02	0.00E+00	4.72E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
GE-OLD	RPV FEEDWATER NOZZLE BORE	LAS	7.58E-01	1.14E+00	3.83E-02	9.13E-02	4.53E-05	6.76E-04	5.79E-06	5.58E-05	8.69E-14	8.37E-13
GE-OLD	RECIRC SYSTEM RHR RETURN LINE	304/316	3.97E-01	5.96E-01	1.11E-01	4.18E-01	1.44E-02	1.33E-01	2.38E-03	1.04E-02	3.57E-10	1.56E-09
GE-OLD	CORE SPRAY SYSTEM NOZZLE	LAS	2.30E-02	3.45E-02	2.50E-05	1.79E-04	4.44E-14	7.67E-12	7.63E-15	1.01E-12	1.72E-22	2.27E-20
GE-OLD	CORE SPRAY SYSTEM SAFE END	304/316	1.82E-01	2.73E-01	6.11E-03	4.30E-02	3.40E-06	1.08E-04	7.12E-07	1.19E-05	1.60E-13	2.68E-12
GE-OLD	RESIDUAL HEAT TAPERED	304/316	1.56E+00	2.34E+00	6.19E-06	7.19E-05	2.62E-07	6.36E-06	3.56E-08	6.10E-07	5.34E-15	9.15E-14
GE-OLD	FEEDWATER LINE - RCIC TEE	LAS	5.84E-01	8.76E-01	1.37E-01	4.35E-01	4.30E-06	2.06E-04	1.33E-06	2.29E-05	2.00E-13	3.44E-12

rates for other components having much lower probabilities of failure are seen to increase by as much as an order of magnitude from 40 years to 60 years, but these components make relatively small overall contributions to core damage frequencies.

The pc-PRAISE code was applied in the calculations, with crack initiation depending only on the peak local stresses. Crack initiation was modeled as occurring at multiple sites with the pc-PRAISE model using a 5-cm (2-in.) -long initiation site. The Monte Carlo calculations with the pc-PRAISE code were run to a maximum of 10^6 simulations. Because some components had very low failure probabilities, this number of simulations was sometimes inadequate to establish probabilities of through-wall cracks. Rather than reporting a probability as less than 10^{-6}, additional calculations were performed with a Latin Hypercube approach developed by PNNL on a previous NRC research project (Simonen and Khaleel 1998b). The italicized values in Table 5.11 were derived from these supplementary calculations. The importance sampling procedure of the Latin Hypercube code permitted calculations of through-wall crack probabilities as small as 10^{-15}.

The pc-PRAISE calculations were based on a number of conservative modeling assumptions and input parameters, balanced by other non-conservative assumptions and inputs. In the balance, the calculations are believed to provide realistic predictions of through-wall crack frequencies for the assumed cyclic stress pairs. The inputs for stress cycles as taken from NUREG/CR-6260 (Ware et al. 1995) were believed in most cases to conservatively bound the stresses experienced during actual plant operation. These stresses were taken from design stress reports that assumed bounding conditions for thermal stress transients and other loads. In addition, the method used to derive load pairs from the transient descriptions assumed worst-case sequencing of loads. The method used in the current calculations to estimate through-wall stress distributions (uniform tension versus through-wall gradient) were intended to overestimate the fraction of the stress assigned to the uniform tension category. Inputs for strain rates, oxygen, and sulfur were all assigned as bounding values that are unlikely to be present simultaneously for any given component.

In summary, the calculations of NUREG/CR-6674 indicated that the components with the very high probabilities of failure can have through-wall crack frequencies that approach about 5×10^{-2} per year. In contrast, other components with much lower failure probabilities can have their failure frequencies increase by factors of about 10 from 40 years to 60 years. Calculations were also performed to address the effects of reactor water environments (versus air) and to compare these effects to the effects of extended plant operation from 40 years to 60 years. The environmental effects were predicted to increase through-wall crack probabilities by as much as two orders of magnitude. The calculated failure frequencies of NUREG/CR-6674 were sufficiently low to make relatively small contributions of fatigue failures to core-damage frequencies for the seven representative plants addressed by the study. However, it was also observed that the calculated failure frequencies were higher than the corresponding frequencies based on reported events at operating plants. As discussed above, the differences can be explained in terms of conservative inputs for cyclic stresses and assumptions regarding plant operating environments.

In a more recent study (Tregoning et al. 2005), probabilistic fracture mechanics models in pc-PRAISE were applied to estimate LOCA frequencies as a function of break size and operating time for a PWR stainless steel pressurizer surge line, a stainless steel high-pressure injection nozzle safe-end weld, and a

BWR carbon steel feedwater line elbow. In all cases, the dominant failure mode was assumed to be thermal fatigue. Examples of the cumulative probability results for each line are summarized in Table 5.13.

Table 5.13 NUREG-1829 Appendix F Cumulative Probabilities for PWR Surge Line Elbow, PWR HPI Nozzle Safe End (with and without thermal sleeve) and BWR Feedwater Line (Tregoning et al. 2005)

Leak Size (gpm)	Time	PWR Surge Line Elbow	PWR HPI Nozzle Safe End (thermal sleeve)	PWR HPI Nozzle Safe End (no thermal sleeve)	BWR Feedwater Line
>0 L/min (>0 gpm)	5	--	--	5.67×10^{-5}	--
	25	0.216	1.01×10^{-5}	3.69×10^{-3}	$<1.00 \times 10^{-8}$
	40	0.593	6.80×10^{-4}	1.26×10^{-2}	5.69×10^{-6}
	60	0.889	1.04×10^{-2}	2.98×10^{-2}	2.57×10^{-4}
>378.5 L/min (>100 gpm)	25	1.13×10^{-4}	4.50×10^{-8}	6.49×10^{-4}	--
	40	4.00×10^{-4}	4.90×10^{-7}	2.68×10^{-3}	1.03×10^{-11}
	60	8.02×10^{-4}	1.79×10^{-5}	5.31×10^{-3}	6.44×10^{-7}
>5677 L/min (>1500 gpm)	25	3.41×10^{-8}	2.00×10^{-8}	--	--
	40	3.56×10^{-7}	2.10×10^{-7}	--	5.29×10^{-21}
	60	1.52×10^{-6}	4.56×10^{-6}	--	7.84×10^{-11}
>18926 L/min (>5000 gpm)	25	2.42×10^{-9}	--		--
	40	3.73×10^{-8}	--		3.88×10^{-24}
	60	2.04×10^{-7}	--		3.74×10^{-12}
DEGB	25	9.86×10^{-13}	--	6.49×10^{-4}	--
	40	4.86×10^{-11}	--	2.68×10^{-3}	2.44×10^{-35}
	60	5.43×10^{-10}	--	5.31×10^{-3}	7.14×10^{-17}

5.2.2 Vibration Fatigue

Vibration fatigue failures are normally a result of poor piping design or installation and welding practices. A relatively large share of vibration fatigue failures initiates at the fillets of socket and support attachment welds due to a high stress concentration at the juncture of the weld and base metal. Small-bore pipe socket welded vent and drain connections less than 25.4 mm (<1.0 in.) in the immediate proximity of vibration sources tend to be most susceptible to this failure mechanism (Olson 1985; EPRI 1994; Riccardella et al. 1997; Shah et al. 1998).

Unlike the previously discussed mechanisms, vibration fatigue may not always lend itself to periodic inservice examinations (i.e., volumetric, surface) as a means of managing this degradation mechanism. This is especially true if the inspections occur at the normal inservice inspection frequency of once every 10 years. The nature of this mechanism is such that, generally, almost the entire fatigue life of the component is expended during the initiation phase. Once a crack initiates, failure quickly follows. Therefore, the absence of any detectable crack may not ensure reliable component performance. In addition, for many of these components, the plant conditions when vibration levels are unacceptable may be very difficult to predict since they may be limited to short time periods associated with unique plant or system configurations. This would explain why we continue to observe cases where failures occur late in the plant's operating life. Therefore, the fact that a vibration fatigue failure has not occurred within the first few years of plant operation does not preclude future failures.

A review of domestic fatigue failures in U.S. nuclear power plants (EPRI 1994) indicated that although a few well-known thermal fatigue failures had affected numerous plants, by far the majority of all fatigue failures (approximately 80%) were a result of high-cycle vibration fatigue. Simonen and Gosselin (2001) report that vibration fatigue accounts for approximately 30% of all reported piping failures in nuclear power plant systems. EPRI (1994) and Shah et al. (1998) report that industry service experience (reported failure events, industry surveys, and discussions with plant maintenance personnel) suggest that, on average, small-bore vibration failures occur as often as 0.2–0.3 times per plant year.

Figure 5.15 provides a perspective on the service experience with socket weld failures in safety-related piping as reported in the proprietary PIPExp database described in Appendix D of NUREG-1829 (Tregoning et al. 2005). For the 30-year period 1976–2006, this chart displays the annual number of socket weld failures as a percentage of the total number of pipe or weld failures in safety-related systems. It includes U.S. and foreign experience. Only piping of nominal diameter 102 mm (4 in.) or less are accounted for in this service experience summary. The chart does not convey information about general failure trends. There is significant plant-to-plant variability with respect to vibration fatigue susceptibilities. Also noteworthy is the observation that some national design practices and standards no longer allow the use of socket welds. Table 5.14 is a summary of vibration fatigue service experience involving failures of butt welds and socket welds.

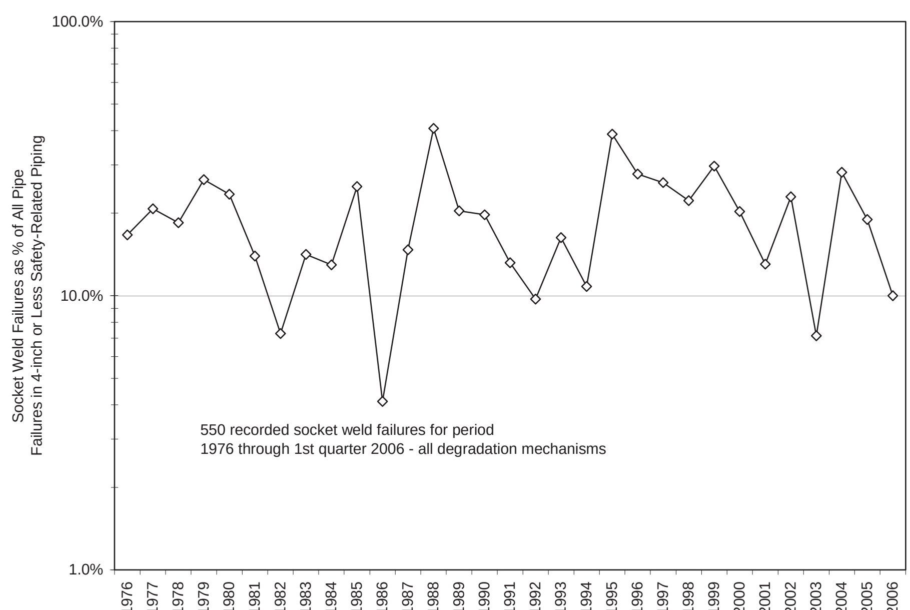


Figure 5.15 Socket Weld Failure as Percentage of All Failures of Safety Related Piping of Nominal Pipe Size ≤102-mm (≤4-in.) (Tregoning et al. 2005)

Table 5.14 Service Experience Involving Butt Weld and Socket Weld Failure Due to Vibration Fatigue

Weld Type	Pipe Size	Recorded Weld Failures Due to Vibration-Fatigue			
		All	Part Through-Wall	Leak	Severance
Butt Weld/ Attachment Weld	≤25.4 mm (≤ 1 in.)	622	50	508	64
	>25.4 mm (> 1 in.)	205	23	172	10
Socket Weld	≤25.4 mm (≤ 1 in.)	290	28	253	9
	>25.4 mm (> 1 in.)	116	48	68	0
This tabulation is based on failure data as recorded in the proprietary PIPExp database; see Appendix D in NUREG-1829 (Tregoning et al. 2005).					

The current study could not identify any probabilistic fracture mechanics models that can address the types of vibration fatigue failures as observed in the field. The field failures have largely been associated with socket welds and from vibration stresses. Major uncertainties relate to when and where the vibrations occur and the stress amplitudes when the vibrations do occur, which are factors not addressed by any of the probabilistic fracture models discussed in this report. There are some limited-scope treatments of vibration stresses that have been part of the other probabilistic models designed to specifically address other failure mechanisms such as fatigue and intergranular stress corrosion cracking (e.g., PRAISE and SRRA computer codes). Pipe failure probabilities reflective of high-cycle vibration fatigue can quite readily be obtained through application of service experience data in a statistical analysis framework.

5.3 Flow-Assisted Degradation

This section addresses three forms of flow-assisted pipe degradation: 1) flow-accelerated corrosion[a] (FAC) of high-energy carbon steel and low alloy piping; 2) erosion-corrosion of moderate energy carbon steel piping; and 3) cavitation erosion of high- or moderate-energy carbon steel and stainless steel piping. All three degradation mechanisms cause localized or global internal pipe wall thinning. Depending on the specific system and location, cavitation erosion is also known to cause flow-induced vibratory-fatigue induced damage.

5.3.1 Flow-Accelerated Corrosion[b]

Flow-accelerated corrosion (FAC) is defined as a chemical process whereby the normally protective oxide layer on carbon or low-alloy steel dissolves into a stream of flowing water or water-steam mixture. FAC corrosion rate controlling conditions are primarily electrochemical. FAC occurs in high-energy piping systems and can occur in single- and two-phase flow regions. The cause of FAC is a specific set of water chemistry conditions (for example, pH, level of dissolved oxygen), and absent of any mechanical

(a) The term “flow-assisted corrosion” also has been used to describe this process.
(b) In the United States, flow-accelerated corrosion (FAC) is commonly but incorrectly known as “erosion-corrosion.” Unlike FAC, the accelerated corrosion rates in the erosion-corrosion process are dominated by mechanical factors such as the impact of water droplets on the surface in two-phase flow steam systems, cavitation effects, or entrained particles.

contribution to the dissolution of the normally protective iron oxide (magnetite) layer on the inside pipe wall (as in pipe degradation by erosion-corrosion). The cause and effect of FAC is well understood, and the industry has implemented FAC inspection programs, as well as piping replacements using FAC-resistant materials such as stainless steel, carbon steel clad on the inside diameter with stainless steel, or chrome-molybdenum alloy steel.

Comprehensive reviews of FAC-induced carbon steel degradation are documented in Brown (1984); Cragnolino et al. (1988); Fultz et al. (1988); and Shah et al. (1997). Included in Table 5.15 is a summary of the pipe failure experience attributed to FAC. It shows the pre-1987 and post-1987 service experience as an indication of the effectiveness of FAC mitigation programs implemented by industry in the aftermath of lessons learned from FAC-induced pipe failures at Trojan in 1985 and Surry Unit 2 in 1986. It is noted that BWR piping is less susceptible to FAC than corresponding systems in the PWR operating environment. This difference is attributed to the inherently different water chemistries. The service experience that is summarized in Table 5.15 covers the following plant systems:

- condensate piping from the condensate booster pump and to the low pressure feedwater heater(s)
- extraction steam piping
- main feedwater piping from low-pressure heater(s) to outboard containment isolation valves; for PWR plants, also feedwater piping from inboard containment isolation valves to the steam generators
- main steam piping from outboard containment isolation valves to the high-pressure turbine steam admission valve and turbine cross-over/cross-under piping
- feedwater heater drain and vent piping
- moisture separator reheater piping.

Table 5.15 Summary of Service Experience Involving Flow-Accelerated Corrosion

Plant Type	Number of Reported Pipe Failures Attributed to FAC					
	1970–1987		1988–2005		1970–2005	
	Part Through-Wall	Through-Wall	Part Through-Wall	Through-Wall	Part Through-Wall	Through-Wall
BWR	85	94	61	100	146	194
PWR	818	89	195	150	972	239

- This tabulation is based on failure data as recorded in the proprietary PIPExp database; see Appendix D in NUREG-1829 (Tregoning et al. 2005).
- "Through-wall" includes small leaks to structural failure (for example, Surry-2 in December 1986 and Mihama-3 in August 2004).

The current study did not identify any published applications of probabilistic structural mechanics models with a mechanistic treatment of material degradation in the open literature that address FAC failures as observed in the field. However, nonmechanistic treatments for wall-thinning rates have been used for probabilistic structural mechanics calculations. Plant-specific calculations have been performed to estimate leak and break probabilities for piping subject to user-specified rates of FAC degradation. These assessments were performed as part of RI-ISI evaluations. These calculations are normally not publicly available; however, some results have been reported in topical reports submitted for NRC approval or selected papers in the literature. Examples can be found in work reported by Westinghouse Owners Group (1999) and Bishop and McNeil (1999).

Table 5.16 shows small leak probability results for a 305-mm (12-in.) carbon steel feedwater system (FW-12) line at Surry Unit 1. The dominant failure mechanism for this line was pipe material wastage due to FAC. The analysis was performed using the Westinghouse SRRA computer code (Westinghouse Owners Group 1997; Bishop 1993, 1997). In the SRRA code, the reduction in wall thickness by FAC was conservatively assumed to have the same impact on structural integrity as a circumferential crack that penetrated the wall to the same depth. Calculations were based on user-provided estimated inputs for material wastage rates. The results of this evaluation were included in an RI-ISI topical report (Westinghouse Owners Group 1997) submitted to the NRC for its review and approval.

Other perspectives on the application of probabilistic models to estimate failure probability in wall-thinned piping are found in Lee et al. (2006) and Vinod et al. (2003). The latter applies the models to FAC-susceptible piping to determine Markov model transition rates for use in RI-ISI program development (Fleming 2004).

5.3.2 Erosion-Corrosion

Erosion is the deterioration of pipe material because of the relative movement between a corrosive fluid and the metal surface. Generally the movement is quite rapid, and wear effect and abrasion are involved. The abrasive action of moving fluids is usually accelerated by the presence of solid particles or matter in suspension. When corrosion occurs simultaneously, the term erosion-corrosion is used. Erosion-corrosion is characterized in appearance by grooves, gullies, waves, rounded holes, and valleys, and usually exhibits a directional appearance. Most metals and alloys are susceptible. In nuclear power plants, the observed failures have largely been associated with pipe-wall degradation and is primarily of concern for raw water piping systems such as service water. Summarized in Table 5.17 is relevant service experience for carbon steel service water piping in U.S. BWR and PWR plants. It is organized by type of ultimate heat sink.

The open literature search did not identify any published applications of probabilistic structural mechanics models with a mechanistic treatment of material wastage rates that address the erosion-corrosion failures as observed in the field. However, failure probability estimates for erosion-corrosion have been made based on structural reliability and risk assessment (SRRA) wall-thinning models (Westinghouse Owners Group 1997, 1999; Bishop and McNeil 1999) for RI-ISI applications. Bishop and McNeil (1999) point out that leak probabilities were found to be extremely sensitive to the potential for erosion-corrosion wastage.

Table 5.16 Surry Unit 1 305-mm (12-in.) Carbon Steel Feedwater System Piping Segment Small Leak Failure Probability by the Westinghouse SRRA Wastage Model Output (Westinghouse Owners Group 1999)

PROBABILITIES OF FAILURE MODE: SMALL OR LARGE LEAK OR BREAK BY WASTAGE

NUMBER FAILED = 400 NUMBER OF TRIALS = 1111

END OF YEAR	FAILURE PROBABILITY WITHOUT FOR PERIOD	CUM. TOTAL	AND WITH IN-SERVICE INSPECTIONS FOR PERIOD	CUM. TOTAL
2.0	9.00090D-04	9.00090D-04	9.00090D-04	9.00090D-04
3.0	9.00090D-04	1.80018D-03	9.00090D-04	1.80018D-03
4.0	9.00090D-04	2.70027D-03	9.00090D-04	2.70027D-03
6.0	2.70027D-03	5.40054D-03	1.35015D-05	2.71377D-03
7.0	4.50045D-03	9.90099D-03	2.25053D-05	2.73628D-03
8.0	3.60036D-03	1.35014D-02	1.80347D-05	2.75431D-03
9.0	6.30063D-03	1.98020D-02	3.20940D-05	2.78641D-03
10.0	5.40054D-03	2.52025D-02	2.95670D-05	2.81597D-03
11.0	7.20072D-03	3.24032D-02	4.80018D-05	2.86397D-03
12.0	6.30063D-03	3.87039D-02	5.77420D-05	2.92172D-03
13.0	4.50045D-03	4.32043D-02	8.24102D-05	3.00413D-03
14.0	3.60036D-03	4.68047D-02	1.11338D-04	3.11546D-03
15.0	1.08011D-02	5.76058D-02	7.06374D-04	3.82184D-03
16.0	1.08011D-02	6.84068D-02	5.97774D-06	3.82782D-03
17.0	9.90099D-03	7.83078D-02	8.39504D-06	3.83621D-03
18.0	1.53015D-02	9.36094D-02	1.86417D-05	3.85485D-03
19.0	6.30063D-03	9.99100D-02	9.40229D-06	3.86425D-03
20.0	6.30063D-03	1.06211D-01	1.25075D-05	3.87676D-03
21.0	1.08011D-02	1.17012D-01	2.56855D-05	3.90245D-03
22.0	1.53015D-02	1.32313D-01	4.27736D-05	3.94522D-03
23.0	1.08011D-02	1.43114D-01	3.39219D-05	3.97914D-03
24.0	1.08011D-02	1.53915D-01	3.48654D-05	4.01401D-03
25.0	1.53015D-02	1.69217D-01	5.80230D-05	4.07203D-03
26.0	1.53015D-02	1.84518D-01	2.93011D-07	4.07232D-03
27.0	1.17012D-02	1.96220D-01	2.59212D-07	4.07258D-03
28.0	1.71017D-02	2.13321D-01	3.85467D-07	4.07297D-03
29.0	8.10081D-03	2.21422D-01	1.98333D-07	4.07317D-03
30.0	1.17012D-02	2.33123D-01	3.10748D-07	4.07348D-03
31.0	9.90099D-03	2.43024D-01	2.85966D-07	4.07376D-03
32.0	1.44014D-02	2.57426D-01	4.77358D-07	4.07424D-03
33.0	1.17012D-02	2.69127D-01	4.43106D-07	4.07469D-03
34.0	1.53015D-02	2.84428D-01	6.72403D-07	4.07536D-03
35.0	1.80018D-02	3.02430D-01	9.64182D-07	4.07632D-03
36.0	1.26013D-02	3.15032D-01	4.13114D-09	4.07633D-03
37.0	7.20072D-03	3.22232D-01	2.97484D-09	4.07633D-03
38.0	1.26013D-02	3.34833D-01	6.19516D-09	4.07633D-03
39.0	1.53015D-02	3.50135D-01	9.67723D-09	4.07634D-03
40.0	9.90099D-03	3.60036D-01	7.49518D-09	4.07635D-03

DEVIATION ON CUMULATIVE TOTALS = 1.44075D-02 1.91244D-03

Table 5.17 Summary of Service Experience Involving Erosion-Corrosion of Service Water Piping

Ultimate Heat Sink	Number of Pipe Failure Records 1972–2005		
	All	Part Through-Wall	Through-Wall
Brackish Water	49	2	47
Lake/Pond Water	39	6	33
River Water	71	26	45
Sea Water	97	3	94
All	256	37	219
This tabulation is based on failure data as recorded in the proprietary PIPExp database; see Appendix D in NUREG-1829 (Tregoning et al. 2005).			

5.3.3 Cavitation-Erosion

Mechanical degradation of a solid material caused by cavitation is called cavitation-erosion. Cavitation erosion can be formed when cavity implosions are violent enough and they take place near the solid material. Cavitation-erosion can be identified from a specific rough mark on surfaces of component flow paths.

Cavitation results from the formation and collapse (implosion) of vapor cavities in a liquid. As the liquid flows into a region where the pressure is reduced to the vapor pressure, it boils, and vapor pockets (bubbles) are formed. The bubbles are carried with the liquid until a region of higher pressure is reached, where they collapse. When the vapor bubbles are near (or in contact with) a solid boundary when they collapse, pitting and fatigue cracking have been observed on the surface. Cavitation can occur in a variety of nuclear plant system components such as pumps, turbines, elbows, valves, and flow orifices. With regard to piping pressure boundary degradation, the last two locations are considered more significant.

The observed failures have largely been associated with piping base materials downstream of flow control valves and orifices of such systems as chemical and volume control, component cooling, decay heat removal, and service water. A summary of relevant service experience is given in Table 5.18.

Table 5.18 Summary of Service Experience Involving Cavitation-Erosion

Material	Number of Pipe / Weld Failures 1974–2005		
	All	Part Through-Wall	Through-Wall
Carbon Steel	43	12	31
Stainless Steel	37	11	26
All	80	23	57
This tabulation is based on failure data as recorded in the proprietary PIPExp database; see Appendix D in NUREG-1829 (Tregoning et al. 2005).			

The current study could not identify any applications of probabilistic models with mechanistic treatments of wastage rates that can address the cavitation-erosion failures as observed in the field.

5.4 Local Corrosion Mechanisms

Local corrosion damage generally occurs over relatively small areas; however, it can occur at high penetration rates. Examples of these types of degradation mechanisms in nuclear systems include crevice corrosion, pitting (oxygen or chloride), and microbiologically induced corrosion (MIC); the most prevalent of these mechanisms is MIC.

MIC is normal electrochemical corrosion in which the microorganisms either chemically or physically change the conditions at the inner pipe wall to be favorable to corrosion. The microorganisms have the capability of changing the local environment, which then can be totally different from the bulk environment. MIC is usually the result of a complex interaction between different microorganisms, and a pure culture alone is seldom responsible for the corrosion. MIC appears as localized corrosion. MIC is of concern in raw water piping systems, such as fire protection and service water, and where low-flow or stagnant-flow conditions occur. MIC could be a problem in closed-loop piping systems in the presence of contaminants. Carbon steel and, under certain conditions, stainless steels are susceptible to MIC. Where stainless steel material is used, MIC can attack welds and weld heat-affected zones. The susceptibility of raw water piping systems is also a function of water quality. As an example, systems using brackish water or seawater are not conducive to MIC attack. Summarized in Table 5.19 is relevant service experience for two raw water piping systems, namely fire protection water system and service water system.

Table 5.19 Summary of Service Experience Involving MIC

System	Number of Pipe Failures		
	All	Part Through-Wall	Through-Wall
Fire Protection	30	--	30
Service Water	212	43	169
All	242	43	199
This tabulation is based on failure data as recorded in the proprietary PIPExp database; see Appendix D in NUREG-1829 (Tregoning et al. 2005).			

The current study could not identify any applications of probabilistic models with mechanistic treatments of wastage rates that can address the types of local corrosion failures as observed in the field.

6 Uncertainty in Estimated Failure Probabilities

The estimated failure probability for any specific component is subject to large uncertainties, whether the estimate is based on data from operating experience or based on an application of a probabilistic structural mechanics model. When only operating experience is used, uncertainties are particularly large for low-probability events such as ruptures or large breaks and/or failure predictions for specific piping locations. From Section 5, it is clear that there have been significant differences between independent estimates of failure frequencies that are intended to address the same component or a category of components. This section presents results from the literature that describe formal attempts to quantify levels of uncertainty in estimated failure probabilities. In one case (Khaleel and Simonen 1999), the approach estimated the uncertainties associated with individual inputs and modeling assumptions used for a probabilistic fracture mechanics calculation for a fatigue failure and then established the effects of these uncertainties on the calculated failure probabilities. The other case (Tregoning et al. 2005) reflects uncertainties assigned by participants in an expert elicitation process used to estimate LOCA frequencies.

6.1 Uncertainties in Fatigue Calculations with PRAISE

A methodology was developed and applied for quantifying the level of uncertainty associated with calculated fatigue failure probabilities for piping components (Khaleel and Simonen 1999). This study showed large uncertainties in calculated failure probabilities, with the uncertainties being particularly large when the calculated probabilities were very small, such as for pipe ruptures. The objective was to quantify the uncertainties associated with individual inputs to the PFM calculations and the uncertainties in the PFM model itself and then to determine the associated uncertainties in the resulting calculated piping failure probabilities.

All calculations presented used the pc-PRAISE computer code (Harris and Dedhia 1992) to address both leak and break probabilities for piping components. A two-step process was used to quantify the uncertainties in calculated failure probabilities. The first step was a sensitivity study that identified those uncertainties having the greatest effect on the results from pc-PRAISE. The second step was a quantitative uncertainty analysis that addressed the most critical parameters as identified by the sensitivity calculations.

While the selected methodology is generally applicable to PFM calculations for piping, the calculations addressed specific examples as follows:

- mechanical and thermal fatigue of stainless steel piping due to the presence of fabrication defects (the initiation of fatigue cracks is not addressed)

- a single 152.4-mm (6-in.) pipe with an inner radius of 70 mm (2.75 in.) and a wall thickness of 14.3 mm (0.562 in.)

- A wide range of cyclic stresses is addressed using the Q-factor approach of Khaleel and Simonen (1994a, b). The Q-factor depends both on the number of stress cycles (N_f) and on the corresponding magnitude of the cyclic stress (σ). It is defined by the equation $Q = A \bullet N_f \bullet \sigma^4$ where A is a constant or normalizing parameter. The Q-factor has the same functional form as the Paris law for fatigue crack growth and thereby reflects the predicted amount of fatigue crack growth. Q factors of 1, 100, and 10,000 were addressed. The lowest value corresponds to typical locations in piping systems for which fatigue failures are not expected, whereas the highest value corresponds to locations exposed to thermal fatigue by which the pipe experiences unexpected and excessive levels of cyclic stress.

While probabilities of both leaks (through-wall cracks) and pipe breaks were calculated, the main interest was in the uncertainties for leak probabilities. Uncertainties in leak probabilities were of particular interest to the development of a regulatory guide on RI-ISI (Parry 1997).

The methodology and the resulting estimated uncertainties in the calculated probabilities were found to be similar to those for PFM calculations for reactor pressure vessels subject to pressurized thermal shock (Bozarth et al. 1985). While additional uncertainty evaluations were recommended to address other pipe sizes and failure mechanisms in addition to fatigue, the uncertainties as characterized by the results are believed to be representative of results expected for piping components in general.

Inputs to pc-PRAISE consist of both probabilistic and deterministic parameters. Flaw depth, flaw-aspect ratio, crack-growth rate, and flow and ultimate stresses are probabilistic, while the pipe radius and wall thickness, internal pressure, dead-weight load, thermal stresses, and the number and level of cyclic stresses are deterministic. For preliminary sensitivity analyses, probabilities were first calculated using best-estimate values for all input parameters. These analyses were followed by calculations in which the best estimate inputs changed one by one. The numerical change for each input corresponded to the estimated uncertainty associated with that input parameter. Calculations indicated that leak probabilities were most sensitive to changes in the flaw depth, flaw-aspect ratio, crack-growth rate, and level and number of cyclic stresses. Changes in some of these parameters resulted in leak probabilities higher/lower than the baseline by as much as 2.5 orders of magnitude. The flow stress of the pipe material was nevertheless included as one of the more important input parameters due to large uncertainties in the simplified fracture-mechanics model used to predict pipe breaks (Wilkowski et al. 1994). In this regard, the scatter in flow stress was used to represent the scatter in measured failure loads from degraded pipes as compared to predictions based on limit load theory.

The objective of the uncertainty analysis was to estimate the mean, median, and standard deviation of the calculated leak and break probabilities. We estimated these statistical parameters using Monte Carlo simulations on the variables described above. Many of the eight most critical parameters were related to probabilistic inputs, whereas others (thermal stress, cyclic-stress level, and number of stress cycles) are deterministic inputs to pc-PRAISE. The uncertainty analyses required that both probabilistic and deterministic inputs be randomized.

Calculations were performed for different sets of inputs obtained by sampling from triangular distributions that described the estimated uncertainties in the pc-PRAISE input parameters. Each Monte Carlo trial was a pc-PRAISE run that gave values of the failure probabilities. There were 100 trials for each of the three Q values, which provided a sample of 100 failure probabilities from which means, medians, and standard deviations for the calculated probabilities were established. There were 100 pc-PRAISE runs for each of the three Q values. This gave a total of 300 sets of piping-failure probabilities (leaks and breaks) as a function time, with the time span going from the start of plant operation to the end of life at 40 years. The focus was limited to cumulative failure probabilities at

40 years. The results gave conditional probabilities, corresponding to the assumption that one pre-existing crack exists in the weld.

Figure 6.1 shows an example histogram (leak probabilities for Q = 100) along with the best-estimate probabilities. From the 100 calculated values of leak and break probabilities, it was possible to calculate mean and median values for comparison with the corresponding values from the best-estimate calculation from pc-PRAISE. The uncertainty calculations generally agreed (within an order of magnitude) with the best-estimate calculations. Figure 6.2 summarizes the results from all of the uncertainty analyses. Whereas there are relatively small differences between the best-estimate and median values of leak probabilities, Figure 6.2 shows more significant differences for mean values. The mean values for leak probabilities were about a factor of ten greater than the best estimate. The mean values for break probabilities were several orders of magnitude greater than the best estimate. It is noted that

- Median values of probabilities correlated relatively well with the best estimates.

- Differences between best-estimate and mean values for leak probabilities were 1 to 3 orders of magnitude, and 4 to 6 orders of magnitude for break probabilities.

- The greater uncertainties for pipe-break probabilities compared to leak probabilities were largely due to the sensitivity of calculations to inputs for flaw depths and aspect ratios. There are little data to support the estimates for the low occurrence rates for very deep and long flaws. Estimates of probabilities for these rare defects are based on extrapolations having high levels of uncertainty.

- The uncertainties in calculated leak probabilities become relatively small for leak probabilities greater than 10^{-4}. Given a 40-year operating life for a reactor piping system, this 10^{-4} cumulative probability corresponds to a failure frequency of 2.5×10^{-6} leaks per weld per year.

Figure 6.3 shows that the scatter in leak probabilities and break probabilities follows similar trends. It is noted that

1. The uncertainties in calculated failure probabilities are highest when the probabilities themselves are very low (e.g., 10^{-8} per weld per 40-year life) and are much smaller when the failure probabilities become larger (e.g., 10^{-2}).

2. Large uncertainties for components with low failure probabilities may have a relatively small impact on the conclusions of risk-based assessments, because risk-based decisions are dominated by components with the higher values of failure probabilities.

3. Failure probabilities for components with higher calculated failure probabilities can be more readily checked for consistency with plant operating experience; the ability to make such comparisons helps to minimize the uncertainties in these calculated probabilities.

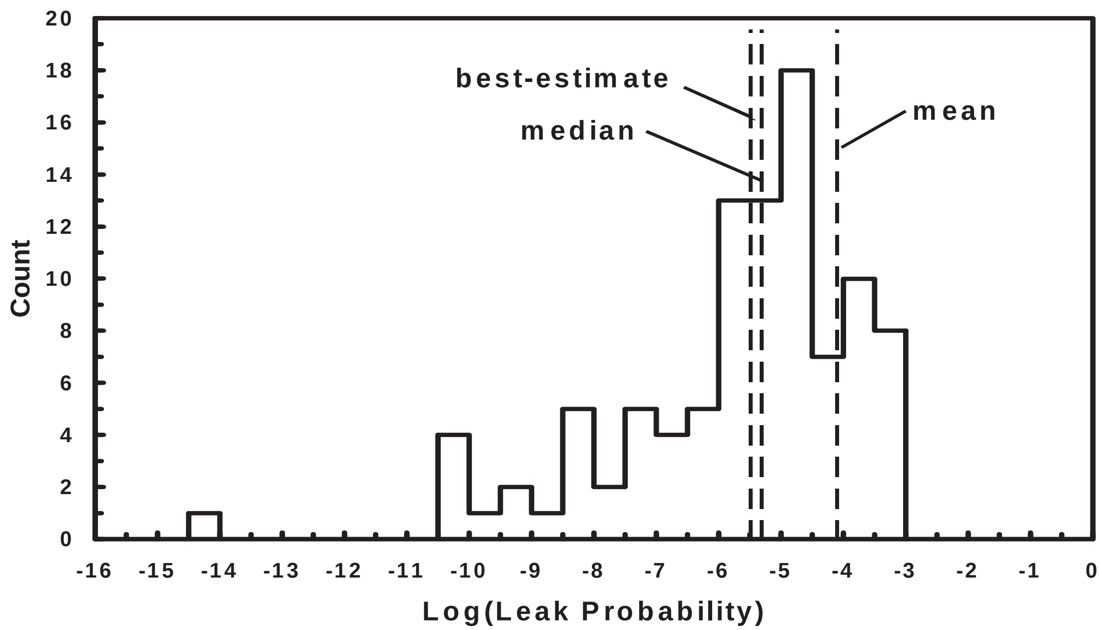


Figure 6.1 Histogram for Probability of Leak for Q = 100

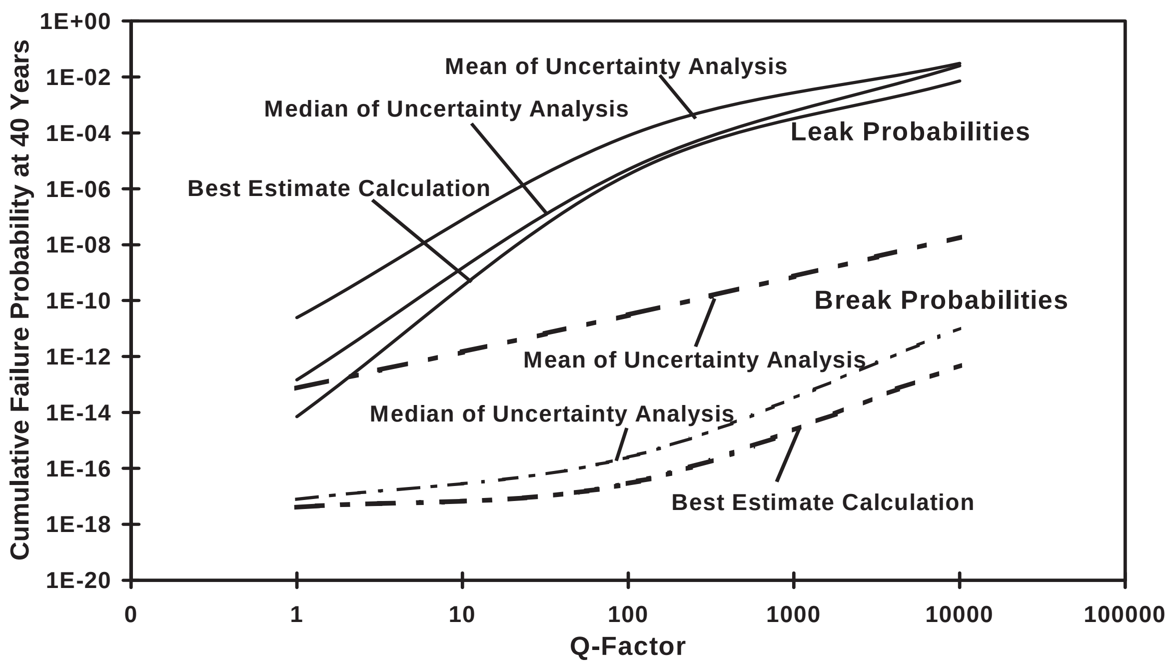


Figure 6.2 Comparisons of Probabilities from Uncertainty Analyses with Best-Estimate Calculations

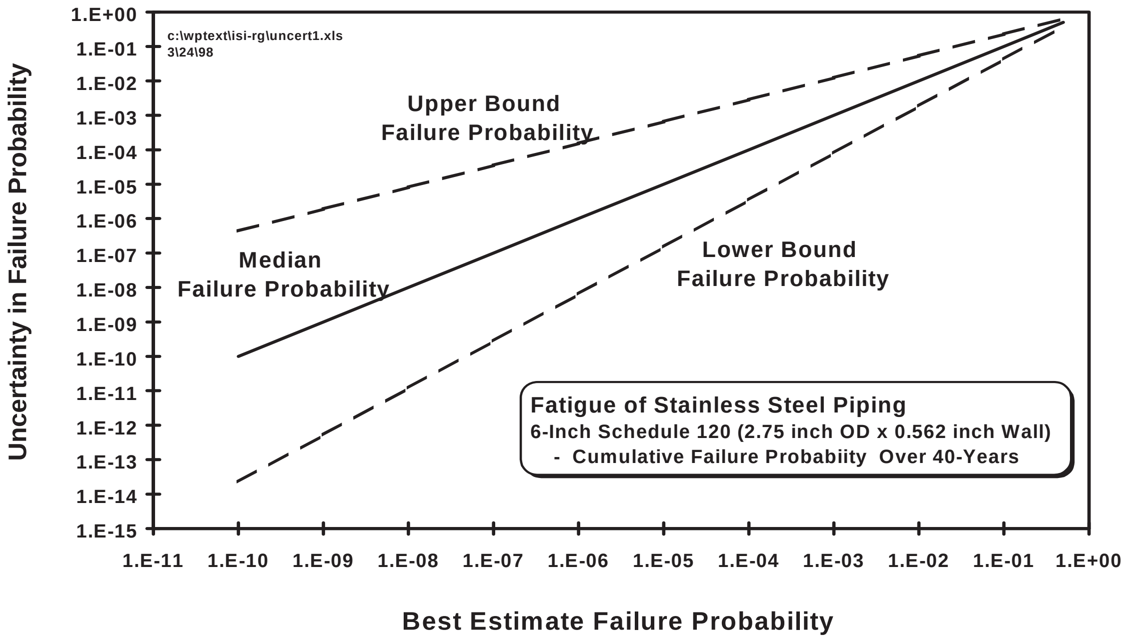


Figure 6.3 Maximum Failure (Leak and Break) Probabilities from Sample of 100 Cases Evaluated by Uncertainty Analyses (25.4 mm = 1 in.)

In summary, the results clearly show large uncertainties in calculated failure probabilities, with the uncertainties being particularly large when the best-estimate probabilities are relatively small. The median values of leak probabilities from the uncertainty calculations generally agree (within an order of magnitude) with the best-estimate calculations. This correlation supports the viability of using best-estimate calculations for risk-informed decision making. Initial flaw-size distributions were the greatest source of uncertainty in calculated failure probabilities because of the unavoidable difficulty in estimating the very low probabilities for the large fabrication flaws, which (if present) have a major impact on piping integrity.

6.2 Uncertainties in LOCA Frequencies from Expert Elicitation

Uncertainties were assigned by participants in an expert elicitation process used to estimate LOCA frequencies (Tregoning et al. 2005). Best-estimate values and the uncertainties for the failure frequencies from the experts served as inputs to a procedure that combined the individual inputs to establish overall trends that reflected the judgments of the group as a whole. These trends are shown by Figures 6.4 and 6.5 for BWR and PWR plants, respectively.

The objective of the study was to develop separate frequency estimates for BWR and PWR piping and non-piping passive system LOCAs. The sole focus was on event frequencies that initiate in unisolable primary system failures that can be exacerbated by age-related material degradation. While the central LOCA frequencies to reflect the consensus of the experts were the objective, quantifying the uncertainties and variability among the individual panel members was also an important objective.

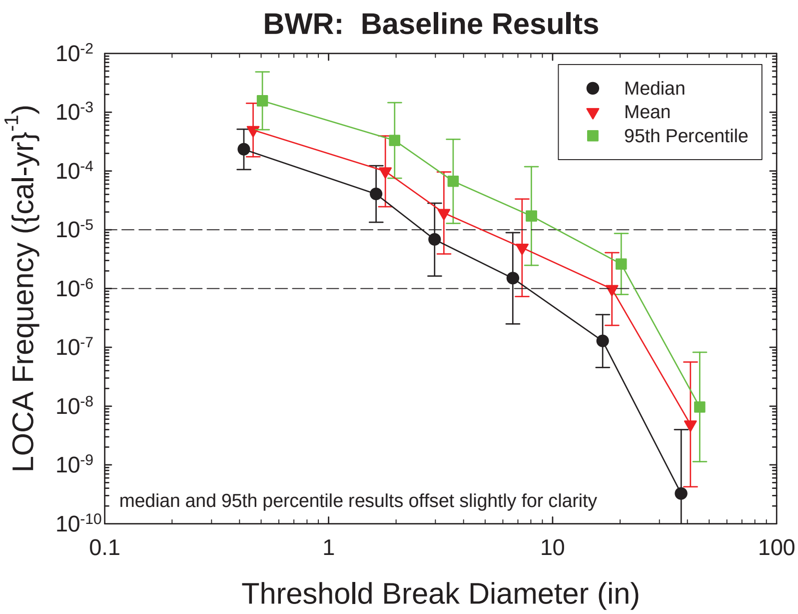


Figure 6.4 Total BWR LOCA Frequencies (means, medians, and 95th percentile values) as a Function of the Threshold Break Diameter at 25 Years of Plant Operations

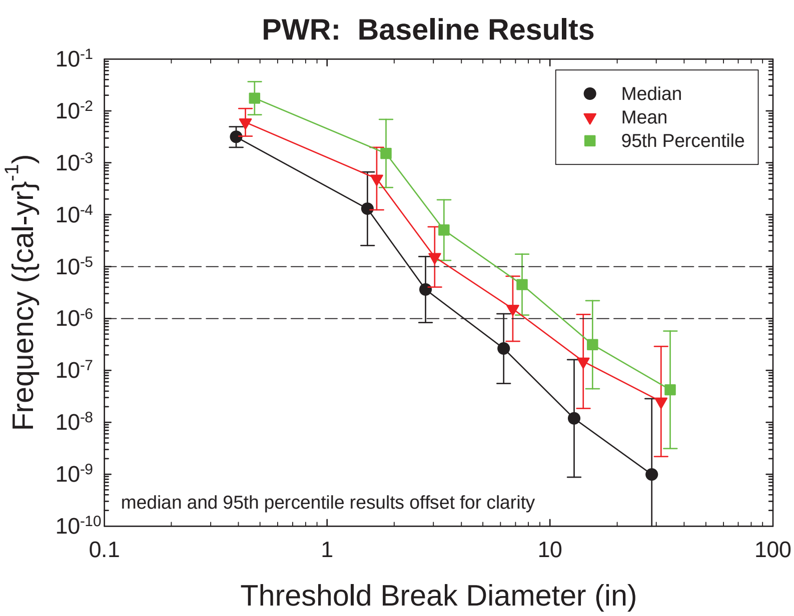


Figure 6.5 Total PWR LOCA Frequencies (means, medians, and 95th percentile values) as a Function of the Threshold Break Diameter at 25 Years of Plant Operations

The elicitation developed generic, or average, values for the commercial fleet of nuclear power plants. Thus, the uncertainty bounds do not represent LOCA frequency estimates for individual plants that may differ from the generic values. This study partitioned LOCA sizes into three smaller categories, which were consistent with historical definitions of small break, medium break, and large break flow rates. Additionally, three larger LOCA categories were defined within the classical large break LOCA regime to examine trends with increasing break size, up to and including, a DEGB of the largest piping system in the plant.

The expert elicitation process employed in this study is an adaptation of the formal expert judgment processes used in NUREG-1150 (USNRC 1990b). To begin, the project staff first identified the issues to be evaluated before a panel of 12 experts was selected. At its initial meeting, the panel discussed the issues and, using the staff formulation as a starting point, developed a final formulation for the elicitation structure. This structure included the decomposition of the complex technical issues that impact LOCA frequencies into fundamental elements. A subset of the panel was created to develop quantitative estimates of LOCA frequencies associated for selected base-case conditions. At this initial meeting, the panel was also trained in subjective elicitation of numerical values through exercises and discussion of potential biases.

After this initial meeting, the staff prepared a draft elicitation questionnaire and iterated with the panel to refine the questionnaire. A second meeting was held with the entire panel to review the base-case results, review the elicitation questions, and finalize the formulation of remaining technical issues. After the meeting, the individual panel members performed analyses and computations to develop answers to the elicitation questionnaire.

The base-case evaluation required the panelists to assess the accuracy and uncertainty in the base-case analyses, and to also choose a particular base-case approach for anchoring their elicitation responses. The piping and non-piping LOCA frequency questions required each panelist to first identify important LOCA contributing factors (i.e., piping systems, materials, degradation mechanisms) and select appropriate base-case conditions for comparison. The panelists would then provide relative ratios between their important contributing factors and the base-case conditions based on their knowledge of passive system component failure. Each relative comparison required mid-value, upper-bound, and lower-bound values. The mid value is defined such that, in the panelist's judgment, there is a 50% chance that the unknown true answer lays above the mid value. The upper and lower bounds are defined such that there is a 5% chance that the true answer lays above the upper bound or below the lower bound, respectively. Each panelist was also required to provide their qualitative rationale supporting their quantitative values.

A facilitation team met separately with each panel member in day-long individual elicitation sessions. At these sessions, each panel member provided answers to the elicitation questionnaire along with their supporting technical rationales. After this meeting, the panel members provided updated responses, and the project staff compiled the panel's responses to develop preliminary estimates of the LOCA frequencies. These preliminary estimates were presented to the panel at a wrap-up meeting. Panel members were invited to modify any of their responses based on group discussions. Final estimates of the LOCA frequencies were then calculated and provided to the panel members for final review and quality assurance.

The quantitative responses were analyzed separately for each panel member to develop individual LOCA frequency estimates. A unified analysis format was developed to ensure consistency and commonalty in processing the panelists' inputs. The panelists' mid-value and upper bound and lower bound were assumed to represent the median, 95th, and 5th percentiles, respectively, of their subjective uncertainty distributions for each elicitation response. The final output for each panelist was BWR- and PWR-specific total LOCA frequency estimates of the mean, median, 5th and 95th percentiles.

While there was general qualitative agreement among the panelists about important technical issues and LOCA contributing factors, the individual quantitative estimates were much more variable. The LOCA frequencies for the individual panelists were aggregated to obtain group LOCA frequency estimates, along with measures of panel diversity. The methodology (Tregoning et al. 2005) was consistent with the elicitation objectives and structure and ensures that the LOCA frequency parameter estimates were not dominated by outlier estimates.

The median, mean, and 95th percentile baseline estimates are graphically presented in Figures 6.4 and 6.5. The 95% confidence interval calculated for these parameters are also illustrated. A measure of the individual uncertainty is illustrated by the difference between the median and 5th or 95th percentile estimates. Differences of group opinion are reflected in the confidence bounds. As the LOCA size increased, the panel members generally expressed greater uncertainty in their predictions, and the variability among individual panelists' estimates increased. This is to be expected because of the greater extrapolation required from available service data.

In summary, the plots of Figures 6.4 and 6.5 reflect both the expert-to-expert variations in the estimated LOCA frequencies and the uncertainties expressed by the individual experts. Small breaks (e.g., through-wall cracks) are seen to have much greater estimated frequencies (e.g., 10^{-2} per plant per year) than the corresponding frequencies (e.g., 10^{-7} per plant per year) for larger breaks (pipe ruptures). In terms of the logarithmic scale used to plot the frequencies, the uncertainties are much greater for low values of failure frequencies (about one order of magnitude) than for the higher values of failure frequencies (about three orders of magnitude).

7 Conclusions

A search of the open literature was performed, which identified more than 7500 publications from 1980 to present that have addressed the degradation of reactor pressure boundary components. Approximately 60% of the documents were related to corrosion mechanisms (e.g., FAC, erosion-corrosion, cavitation erosion, boric acid corrosion, crevice corrosion, pitting). The remaining documents are divided equally between fatigue and stress corrosion cracking mechanisms. The vast majority of the references address various mechanistic aspects of degradation mechanisms and apply deterministic rather than probabilistic models.

The literature search did not generally identify the numerous plant-specific studies completed by industry owners' groups and by research organizations such as the EPRI Materials Reliability Program. These results are typically not publicly available. In addition, many plant-specific probabilistic fracture mechanics calculations have been performed in support of RI-ISI evaluations. Although these calculations are normally not publicly available, portions of these results are sometimes included in topical reports submitted for NRC approval or in selected papers in the open literature.

Although the probabilistic treatments and characterizations of fatigue and stress corrosion cracking are well documented in the literature, only a limited number of these publications document plant-specific and component-specific probabilistic evaluations based on actual design and operating stresses. Examples from the more relevant of these studies were discussed in Section 5 of this report.

The literature search did not identify any applications of probabilistic structural mechanics models for local corrosion. Failure probability estimates have, however, been reported in plant-specific RI-ISI evaluations. In some applications, failure frequency estimates were based on statistical evaluations of service failure data. Failure probability estimates have also been based on assumed estimates of corrosion rates and qualitative evaluations made by plant expert panels.

The literature search did not identify any applications of probabilistic structural mechanics models for flow-assisted degradation mechanisms (FAC, erosion-corrosion, and cavitation erosion) known to be active in nuclear power plant systems. As with other corrosion mechanisms, failure probability estimates have been performed in support of plant-specific RI-ISI evaluations. In these cases, failure frequency estimates have been based on estimated inputs for material wastage/wall thinning rates. In these models, wall thinning was simplistically assumed to have the same impact on structural integrity as a circumferential crack that penetrated the wall to the same depth.

Formal attempts to quantify uncertainties in estimated failure probabilities show that the estimated failure probabilities for particular components are subject to large uncertainties, whether the estimate is based on data from operating experience or based on application of probabilistic structural mechanics models. The reported uncertainties are particularly large when the best-estimate probabilities are relatively small. Initial flaw size distributions were identified as particularly large sources of uncertainty in calculated failure probabilities because of the unavoidable difficulty in estimating the very low probabilities for the large fabrication flaws, which (if present) have a major impact on piping integrity.

There can also be differences in calculated probabilities related both to the probabilistic fracture mechanics models as implemented in the various computer codes and the judgments made in defining input parameters for plant-specific calculations. Ongoing efforts are needed for improved data on material degradation rates and damage models that describe the performance of components under field conditions. In many cases, such as for damage mechanisms associated with wall thinning, there are no recognized probabilistic structural mechanics codes to predict failure probabilities.

8 References

10 CFR 50.61. "Fracture toughness requirements for protection against pressurized thermal shock events." *Code of Federal Regulations*, Part 50.61, Title 10, 1996.

Attwood CL, VN Shah, and WJ Galyean. 1999. "Analysis of Pressurized Water Reactor Primary Coolant Leak Events Caused by Thermal Fatigue." In *Proc. ESREL '99 – The Tenth European Conference on Safety and Reliability*. AA Balkema, Rotterdam (The Netherlands).

Bell CD and OJV Chapman. 2003. *Description of PRODIGAL*. NURBIM Report D4/Appendix F, Rolls-Royce plc., Derby, United Kingdom.

Bishop BA. 1993. *Benchmarking of Probabilistic Fracture Mechanics Analyses of Reactor Vessels Subjected to Pressurized Thermal Shock Loading.* EPRI Research Project 2975-5, Electric Power Research Institute, Palo Alto, California.

Bishop BA. 1997. "An Updated Structural Reliability Model for Piping Risk Informed ISI." In *Fatigue and Fracture - 1997, Volume 2*, PVP-Vol. 346, pp. 245–252. American Society of Mechanical Engineers, New York.

Bishop BA and A McNeil. 1999. "Use of Structural Reliability Models for Piping Risk Informed ISI at 40 and 60 Years." In *Pressure Vessel and Piping Codes and Standards*, PVP-Vol. 383, pp. 61–67. American Society of Mechanical Engineers, New York.

Bozarth D, CL Mason, and JW Minarick. 1985. "Sensitivity and Uncertainty Analyses of Estimated Through-Wall Crack Frequencies for H.B. Robinson Unit 2." Chapter 7 in *Pressurized Thermal Shock Evaluation of the H.B Robinson Unit 2 Nuclear Power Plant*, NUREG/CR-4183, Volume 1, U.S. Nuclear Regulatory Commission, Washington, D.C.

Brickstad B and W Zang. 2001. *NURBIT Nuclear RBI Analysis Tool, A Software for Risk Management of Nuclear Components*. Technical Report No 10334900-1, DNV, Stockholm, Sweden.

Brickstad B, OJV Chapman, T Schimpfke, H Schulz, and A Muhammed. 2004. *Review and Benchmarking of SRM and Associated Software*. NURBIM Final Report D4, Contract FIKS-CT-2001-00172, DNV, Stockholm, Sweden.

Brown EJ. 1984. *Erosion in Nuclear Power Plants*. AEOD/E4 16, Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, Washington, D.C.

Cattant F, F Vaillant, JM Boursier, and S De-Vito. 2005. "Contribution of Hot and Cold Laboratory Investigations to the Resolution of EDF Alloys 600/182/82 PWSCC Issues." In *Materials Reliability Program: Proceedings of the 2005 International PWSCC of Alloy 600 Conference and Exhibit Show* (MRP-154), TR-1012089, Electric Power Research Institute, Palo Alto, California.

Cheverton RD and DG Ball. 1984. *OCA-P, A Deterministic and Probabilistic Fracture-Mechanics Code for Applications to Pressure Vessels*. NUREG/CR-3618, U.S. Nuclear Regulatory Commission, Washington, D.C.

Chopra OK. 2002. "Mechanism of Fatigue Crack Initiation in Austenitic Stainless Steel in LWR Environments." In *Pressure Vessel and Piping Codes and Standards*, PVP-Vol. 439, pp. 133–142, RD Rana (ed.). American Society of Mechanical Engineers, New York.

Chopra OK and WJ Shack. 2001. *Environmental Effects on Fatigue Crack Initiation in Piping and Pressure Vessel Steels*. NUREG/CR-6717, U.S. Nuclear Regulatory Commission, Washington, D.C.

Cragnolino G, C Czajkowsi, and WJ Shack. 1988. *Review of Erosion-Corrosion in Single-Phase Flows*. NUREG/CR-5156, U.S. Nuclear Regulatory Commission, Washington, D.C.

Dickson TL. 1994. *FAVOR: A Fracture Mechanics Analysis Code for Nuclear Reactor Pressure Vessel, Release 9401*. ORNL/NRC/LTR/94/1, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Dillstrom P. 2003. *A Short Description of ProSACC*. NURBIM Report D4/Appendix G, DNV, Stockholm, Sweden.

EPRI. 1988. *Assessment of Remedies for Degraded Piping*. NP-5881-LD, Electric Power Research Institute, Palo Alto, California.

EPRI. 1994. *EPRI Fatigue Management Handbook*. TR-104534, Electric Power Research Institute, Palo Alto, California.

EPRI. 2000. *Operating Experience Regarding Thermal Fatigue of Unisolable Piping Connected to PWR Reactor Coolant System (MRP-25)*. TR-1001006, Electric Power Research Institute, Palo Alto, California.

EPRI. 2002. *Crack Growth Rates for Evaluating Primary Water Stress-Corrosion Cracking (PWSCC) of Thick-Wall Alloy 600 Materials (MRP-55) Revision 1*. TR-1006695, Electric Power Research Institute, Palo Alto, California. NRC ADAMS Accession No. ML02301050.

EPRI. 2004. *Materials Reliability Program: Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 82, 182, and 132 Welds (MRP-115NP)*. TR-1006696, Electric Power Research Institute, Palo Alto, California. NRC ADAMS Accession No. ML051450555.

Failure Analysis Associates. 1990. *PRAISE Enhancements to Include General Strain Hardening Exponents and Mid-Life Residual Stress and Water Chemistry Changes*. FaAA-SF-R-90-06-06, UCRL-CR-105339. Prepared for Lawrence Livermore National Laboratory, Livermore, California.

Fleming KN. 2004. "Markov Models for Evaluating Risk-Informed In-service Inspection Strategies for Nuclear Power Plant Piping Systems." *Reliability Engineering and System Safety* 83:27–45.

Fleming KN, SR Gosselin, and J Mitman. 1999. "Application of Markov Models and Service Data to Evaluate the Influence of Inspection on Pipe Rupture Frequencies." In *Pressure Vessel and Piping Codes and Standards*, PVP-Vol. 383, pp. 69–75. American Society of Mechanical Engineers, New York.

Fultz KO, MA Kruslicky, WD McDowell, Jr., KR Bartlett, WD Campbell, and RL Coleman. 1988. *Nuclear Regulation: Action Needed to Ensure That Utilities Monitor and Repair Pipe Damage*. RCED-88-73, U.S. General Accounting Office, Washington, D.C.

Garud YS. 1997. *A Simplified Model for Assessment of SCC Initiation Time in Alloy 600.* TR-109137, Electric Power Research Institute, Palo Alto, California.

Garud YS and RS Pathania. 1999. "A Simplified Model for SCC Initiation Susceptibility in Alloy 600 with the Influence of Cold Work Layer and Strength Characteristics." In *Ninth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems—Water Reactors*, pp. 261–268. PP Ford, SM Bruemmer, and GS Was (eds). The Minerals, Metals & Materials Society (TMS), Warrendale, Pennsylvania.

Gavenda DJ, PR Luebbers, and OK Chopra. 1997. "Crack Initiation and Crack Growth Behavior of Carbon and Low-Alloy Steels." In *Fatigue and Fracture* – Volume 1, PVP-Vol. 350, pp. 243–255. S Rahman, KK Yoon, S Bhandari, R Warke, and JM Bloom (eds). American Society of Mechanical Engineers, New York.

Gosselin SR, FA Simonen, RG Carter, JM Davis, and GL Stevens. 2005. "Enhanced ASME Section XI Appendix L Flaw Tolerance Procedure." In *Proceedings of the PVP2005*, 2005 ASME Pressure Vessels and Piping Conference. July 17–21, 2005, Denver, Colorado. PVP2005-71100, American Society of Mechanical Engineers, New York.

Grimmel B and WH Cullen Jr. 2005. *U.S. Plant Experience with Alloy 600 Cracking and Boric Acid Corrosion of Light-Water Reactor Pressure Vessel Materials*. NUREG-1823, U.S. Nuclear Regulatory Commission, Washington, D.C.

Harris DO and DD Dedhia. 1992. *Theoretical and User's Manual for pc-PRAISE, A Probabilistic Fracture Mechanics Computer Code for Piping Reliability Analysis*. NUREG/CR-5864, U.S. Nuclear Regulatory Commission, Washington, D.C.

Harris DO and DD Dedhia. 1998. *WinPRAISE 98 PRAISE Code in Windows*. Engineering Mechanics Technology, Inc., San Jose, California.

Harris DO, DD Dedhia, and ED Eason. 1986a. "Probabilistic Analysis of Initiation and Growth of Stress Corrosion Cracks in BWR Piping." ASME Paper 86-PVP-11. American Society of Mechanical Engineers, New York.

Harris DO, DD Dedhia, ED Eason, and SD Patterson. 1985. *Probabilistic Treatment of Stress Corrosion Cracking in Sensitized 304 Stainless Steel Weldments in BWR Piping*. Failure Analysis Associates Report to Lawrence Livermore National Laboratory, Livermore, California.

Harris DO, DD Dedhia, ED Eason, and SD Patterson. 1986b. *Probability of Failure in BWR Reactor Coolant Piping: Probabilistic Treatment of Stress Corrosion Cracking in 304 and 316NG BWR Piping Weldments*. NUREG/CR-4792, Vol. 3, U.S. Nuclear Regulatory Commission, Washington, D.C.

Harris DO, EY Lim, and DD Dedhia. 1981. *Probability of Pipe Fracture in the Primary Coolant Loop of a PWR Plant Volume 5: Probabilistic Fracture Mechanics Analysis - Load Combination Program Project 1 Final Report*. NUREG/CR-2189, Vol. 5, U.S. Nuclear Regulatory Commission, Washington, D.C.

Hazelton WS and WH Koo. 1988. *Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping*. NUREG-0313, Rev. 2, U.S. Nuclear Regulatory Commission, Washington, D.C.

Heasler PG and FA Simonen. 1990. "A Probabilistic Evaluation of Alternative Weld Inspection Sampling Plans." In *Uncertainty Modeling and Analysis – Proceedings of the First International Symposium on Uncertainty Modeling and Analysis (ISUMA '90)*, BM Ayyub (ed), pp. 432–436. IEEE Computer Society Press, Los Alamitos, California.

Holman GS and CK Chou. 1988. *Probability of Failure in BWR Reactor Coolant Piping*. NUREG/CR-4792, Vol. 1, U.S. Nuclear Regulatory Commission, Washington, D.C.

Kassir M, S Sharma, M Reich, and M-T Chang. 1985. *An Evaluation of Stress Corrosion Crack Growth in BWR Piping Systems*. NUREG/CR-4221, U.S. Nuclear Regulatory Commission, Washington, D.C.

Keating RF, P Hernalsteen, and JA Begley. 1995. *Burst Pressure Correlation for Steam Generator Tubes with Throughwall Axial Cracks*. TR-105505, Electric Power Research Institute, Palo Alto, California.

Keisler J, OK Chopra, and WJ Shack. 1995. *Fatigue Strain–Life Behavior of Carbon and Low–Alloy Steels, Austenitic Stainless Steels, and Alloy 600 in LWR Environments*. NUREG/CR-6335, U.S. Nuclear Regulatory Commission, Washington, D.C.

Khaleel MA and FA Simonen. 1994a. "A Parametric Approach to Predicting the Effects of Fatigue on Piping Reliability." In *Service Experience and Reliability Improvement: Nuclear, Fossil, and Petrochemical Plants*, ASME PVP-Vol. 288, pp. 117–125. American Society of Mechanical Engineers, New York.

Khaleel MA and FA Simonen. 1994b. "The Effects of Initial Flaw Sizes and Inservice Inspection on Piping Reliability." In *Service Experience and Reliability Improvement: Nuclear, Fossil, and Petrochemical Plants*, PVP-Vol. 288, pp. 95–07. American Society of Mechanical Engineers, New York.

Khaleel MA and FA Simonen. 1999. "Uncertainty Analyses for Calculated Failure Probabilities or Piping Welds." In *Pressure Vessel and Piping Codes and Standards - 1999*, PVP-Vol. 383, pp. 77–90. American Society of Mechanical Engineers, New York.

Khaleel MA, OJV Chapman, DO Harris, and FA Simonen. 1999. "Flaw Size Distribution and Flaw Existence Frequencies in Nuclear Piping." In *Probabilistic and Environmental Aspects of Fracture and Fatigue*, ASME PVP-Vol. 386, pp. 127–144. American Society of Mechanical Engineers, New York.

Khaleel MA, FA Simonen, DO Harris, and D Dedhia. 1995. "The Impact of Inspection on Intergranular Stress Corrosion Cracking for Stainless Steel Piping." In *Risk and Safety Assessments: Where Is the Balance?* PVP-Vol. 296, pp. 411–422. American Society of Mechanical Engineers, New York.

Khaleel MA, FA Simonen, HK Han, DO Harris, and D Dedhia. 2000. *Fatigue Analysis of Components for 60-Year Plant Life*. NUREG/CR-6674, U.S. Nuclear Regulatory Commission, Washington, D.C.

Lee SM, YS Chang, JB Choi, and YJ Kim. 2006. "Failure Probability Assessment of Wall-Thinned Nuclear Pipes Using Probabilistic Fracture Mechanics." *Nuclear Engineering and Design* 236(4):350–358.

Lydell BOY. 1999. *Failure Rates in Barsebäck-1 Reactor Coolant Pressure Boundary Piping: An Application of a Piping Failure Database*. SKI Report 98:30, Swedish Nuclear Power Inspectorate, Stockholm, Sweden.

Lydell BOY, E Mathet, and K Gott. 2004. "Piping Service Life Experience in Commercial Nuclear Power Plants: Progress with the OECD Pipe Failure Data Exchange Project." In *Proceedings of the 2004 ASME Pressure Vessels and Piping Conference*, PVP2004-2961. American Society of Mechanical Engineers, New York.

Miraglia F J Jr. 1988. "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping." Generic Letter 88-01, U.S. Nuclear Regulatory Commission, Washington, D.C.

Mohammed AA. 2003. *A Short Description of STRUEL*. NURBIM Report D4/Appendix H. The Welding Institute, Cambridge, United Kingdom.

Nuclear Energy Agency. 1998. "Experience with Thermal Fatigue in LWR Piping Caused by Mixing and Stratification." NEA/CSNI/R(98)8, Issy-les-Moulineaux, France.

Nuclear Energy Agency. 2006. "NEA Joint Projects – OECD/NEA OPDE Project." Available at http://www.nea.fr/html/jointproj/opde.html (September 2006).

Olson DE. 1985. "Piping Vibration Experience in Power Plants." In *Pressure Vessel and Piping Technology 1985: A Decade of Progress*, pp. 689–705. Pressure Vessel and Piping Division, American Society of Mechanical Engineers, New York.

Parry GW. 1997. "Quality of PRA for Use in Regulatory Decision Making." In *Risk Informed Decision Making*, PVP-Vol. 358, pp. 81–85, American Society of Mechanical Engineers, New York.

Poloski JP, DG Marksberry, CL Atwood, and WJ Galyean. 1999. *Rates of Initiating Events at U.S. Nuclear Power Plants: 1987-1995*. NUREG/CR-5750, U.S. Nuclear Regulatory Commission, Washington, D.C.

Riccardella PC, DC Rosaio, and SR Gosselin. 1997. "Fracture Mechanics Analysis of Socket Welds under High Cycle Vibrational Loading." PVP-Vol. 353, pp. 23–29. American Society of Mechanical Engineers, New York.

Rudland DL, H Xu, G Wilkowski, P Scott, N Ghadiali, and F Brust. 2006. "Development of a New Generation Computer Code (PRO-LOCA) for the Prediction of Break Probabilities for Commercial Nuclear Power Plants Loss-of-Coolant Accidents." In *2006 Proceedings of the ASME Pressure Vessels and Piping Conference*, July 23–27, 2006, Vancouver, British Columbia, Canada. American Society of Mechanical Engineers, New York.

Schimpfke T. 2003. *A Short Description of the Piping Reliability Code PROST*. NURBIM Report D4/Appendix C, Gesellschaft fur Anlagen-und Reaktorsicherheit (GRS), Berlin, Germany.

Scott PM. 1991. "An Analysis of Primary Water Stress Corrosion Cracking in PWR Steam Generators." Presented at NEA/CSNI Specialist Meeting on Operating Experience with Steam Generators, September 16–20, 1991, Brussels, Belgium.

Shah VN, AG Ware, and AM Porter. 1997. *Review of Industry Efforts to Manage Pressurized Water Reactor Feedwater Nozzle, Piping, and Feedring Cracking and Wall Thinning*. NUREG/CR-6456, U.S. Nuclear Regulatory Commission, Washington, D.C.

Shah VN, MB Sattison, CL Atwood, AG Ware, GM Grant, and RS Hartley. 1998. "*Assessment of Pressurized Water Reactor Primary System Leaks*. NUREG/CR-6582, U.S. Nuclear Regulatory Commission, Washington, D.C.

Pipe Crack Study Group. 1979. *Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants*. NUREG-0531, U.S. Nuclear Regulatory Commission, Washington, D.C.

Pipe Crack Study Group. 1980. *Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors*. NUREG-0691, U.S. Nuclear Regulatory Commission, Washington, D.C.

Simonen FA, KI Johnson AM Liebetrau, DW Engel, and EP Simonen. 1986. *VISA-II - A Computer Code for Predicting the Probability of Reactor Pressure Vessel Failure*. NUREG/CR-4486, U.S. Nuclear Regulatory Commission, Washington, D.C.

Simonen FA, DO Harris, and DD Dedhia. 1998. "Effect of Leak Detection on Piping Failure Probabilities." In *Fatigue Fracture and Residual Stress*, PVP-Vol. 373, pp. 105–113. American Society of Mechanical Engineers, New York.

Simonen FA and MA Khaleel. 1998a. "A Probabilistic Fracture Mechanics Model for Fatigue Crack Initiation in Piping." In *Fatigue, Fracture and Residual Stress*, PVP-Vol. 373, pp. 27–34. American Society of Mechanical Engineers, New York.

Simonen FA and MA Khaleel. 1998b. "Effects of Flaw Sizing Errors on the Reliability of Vessels and Piping." *ASME Journal of Pressure Vessel Technology* 120:365–373.

Simonen FA and MA Khaleel. 1998c. “On the Contribution of Buried Flaws to Piping Failure Probabilities.” In *Fatigue Fracture and Residual Stress*, PVP-Vol. 373, pp. 69–75. American Society of Mechanical Engineers, New York.

Simonen FA and SR Gosselin. 2001. “Life Prediction and Monitoring of Nuclear Power Plant Components for Service-Related Degradation,” *Transactions of the ASME*, J. PVT, Vol. 123, 1:58–64.

Snaider R. 1980. *BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking: Resolution of Generic Technical Activity A-10*. NUREG-0619-Rev-1, U.S. Nuclear Regulatory Commission, Washington, D.C.

Strosnider JR Jr, WH Koo, JA Davis, and MC Modes. 2000. *Pipe Cracking in U.S. BWRs: A Regulatory History*. NUREG-1719, U.S. Nuclear Regulatory Commission, Washington, D.C.

Tang SS, PC Riccardella, ML Herrera, D Covill, R Carter, and L Steinert. 1999. “The Effect of Adjusted Reference Temperature on the Probability of Failure in Boiling Water Reactor Vessel Welds.” In *Fracture, Design Analysis of Pressure Vessels, Heat Exchangers, Piping Components, and Fitness for Service – 1999*, PVP-Vol. 388, pp. 121–126. American Society of Mechanical Engineers, New York.

Tregoning R, L Abramson and P Scott. 2005. *Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process (NUREG-1829) - Draft Report for Comment*. Draft NUREG-1829 (June 30, 2005). U.S. Nuclear Regulatory Commission, Washington, D.C. www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1829/index.html

USNRC. 1975. *Reactor Safety Study: An Assessment of Accident Risks I U.S. Commercial Nuclear Power Plants*. NUREG-1150, U.S. Nuclear Regulatory Commission, Washington, D.C.

USNRC. 1979. “Cracking in Feedwater System Piping.” Bulletin 79-13, Rev. 2, U.S. Nuclear Regulatory Commission, Washington, D.C., October 16, 1979.

USNRC. 1980. *BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking*. NUREG-0619, U.S. Nuclear Regulatory Commission, Washington, D.C.

USNRC. 1988a. “Safety Injection pipe Failure.” Information Notice 88-01, U.S. Nuclear Regulatory Commission, Washington, D.C., January 27, 1988.

USNRC. 1988b. “Thermal Stresses in Piping Connected to Reactor Coolant Systems.” Bulletin 88-08, U.S. Nuclear Regulatory Commission, Washington, D.C., June 22, 1988.

USNRC. 1988c. “Thermal Stresses in Piping Connected to Reactor Coolant Systems.” Bulletin 88-08, Supplement 1, U.S. Nuclear Regulatory Commission, Washington, D.C., June 24, 1988.

USNRC. 1988d. “Thermal Stresses in Piping Connected to Reactor Coolant Systems.” Bulletin 88-08, Supplement 2, U.S. Nuclear Regulatory Commission, Washington, D.C., August 4, 1988.

USNRC. 1989. "Thermal Stresses in Piping Connected to Reactor Coolant Systems." Bulletin 88-08, Supplement 3, U.S. Nuclear Regulatory Commission, Washington, D.C., April 11, 1989.

USNRC. 1990a. *Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors*. NUREG-0691, U.S. Nuclear Regulatory Commission, Washington, D.C.

USNRC. 1990b. "Severe Accident Risks: An Assessment of Five U.S. Nuclear Power Plants." WASH-1400, U.S. Nuclear Regulatory Commission, Washington, D.C.

USNRC. 1992. "Cracking of Valves in the Condensate Return Lines in a BWR Emergency Condenser System." Information Notice 92-50, U.S. Nuclear Regulatory Commission, Washington, D.C., July 2, 1992.

USNRC 1997. "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations." Generic Letter 97-01, U.S. Nuclear Regulatory Commission, Washington, D.C., April 1, 1997.

USNRC. 1998. "Cavitation Erosion of Letdown Line Orifices Resulting in Fatigue Cracking of Pipe Welds." Information Notice 98-45, U.S. Nuclear Regulatory Commission, Washington, D.C., December 15, 1998.

USNRC. 2001a. "Through-Wall Circumferential Cracking of Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzles at Oconee Nuclear Station, Unit 3." Information Notice 2001-05, U.S. Nuclear Regulatory Commission, Washington, D.C., April 30, 2001.

USNRC. 2001b. "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles." Bulletin 2001-01, U.S. Nuclear Regulatory Commission, Washington, D.C., August 3, 2001.

USNRC. 2002. "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs." Bulletin 2002-02, U.S. Nuclear Regulatory Commission, Washington, D.C., August 9, 2002.

USNRC. 2003a. "Leakage Found on Bottom-Mounted Instrumentation Nozzles." Information Notice 2003-11, U.S. Nuclear Regulatory Commission, Washington, D.C., August 13, 2003.

USNRC. 2003b. "Leakage Found on Bottom-Mounted Instrumentation Nozzles." Information Notice 2003-11, Supplement 1, U.S. Nuclear Regulatory Commission, Washington, D.C., August 13, 2003.

USNRC. 2003c. "Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity." Bulletin 2003-02, U.S. Nuclear Regulatory Commission, Washington, D.C., August 21, 2003.

USNRC. 2004. "Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized-Water Reactors." Bulletin 2004-01, U.S. Nuclear Regulatory Commission, Washington, D.C., May 18, 2004.

USNRC. 2007. *Expert Panel on Proactive Materials Degradation Assessment*. NUREG/CR-6923. U.S. Nuclear Regulatory Commission, Washington, D.C.

Vinod G, SK Bidhar, HS Kushwaha, AK Verma, and A Srividya. 2003. "A Comprehensive Framework for Evaluation of Piping Reliability Due to Erosion-Corrosion for Risk-Informed Inservice Inspection." *Reliability Engineering and System Safety* 82:187–193.

Vo TV, PG Heasler, SR Doctor, FA Simonen, and BF Gore. 1991. "Estimates of Rupture Probabilities for Nuclear Power Plant Components: Expert Judgment Elicitation." *Nuclear Technology* **96**:259–271.

Wachter O and G Brümmer. 1997. "Experiences with Austenitic Stainless Steels in Boiling Water Reactors." *Nuclear Engineering and Design* 168:35–52.

Ware AG, DK Morton, and ME Nitzel. 1995. *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components*. NUREG/CR-6260, U.S. Nuclear Regulatory Commission, Washington, D.C.

Westinghouse Owners Group. 1997. *Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report*. WCAP-14572, Rev. 1, Westinghouse Electric Co., Pittsburgh, Pennsylvania.

Westinghouse Owners Group. 1999. *Westinghouse Owners Group Application of Risk Informed Methods to Piping Inservice Inspection Topical Report*. WCAP-14572, Revision 1-NP-A, Westinghouse Electric Co., Pittsburgh, Pennsylvania.

White GA, J Hickling, and LK Mathews. 2003. "Crack Growth Rates for Evaluating PWSCC of Thick-Wall Alloy 600 Material." In *Proceedings of the Eleventh International Conference on Environmental Degradation of Materials in Nuclear Systems – Water Reactors*, pp. 166-179. August 10–14, 2003, Stevenson, Washington. American Nuclear Society, LaGrange Park, Illinois.

Wilkowski G, G Kramer, P Veith, R Francini, and P Scott. 1994. "The Effect of Dynamic and Cyclic Loading During Ductile Tearing on Circumferentially Cracked Pipe: Analytical Results." In *Fatigue, Flaw Evaluation and Leak Before Break Assessments*, PVP–Vol. 280, pp. 221–239. American Society of Mechanical Engineers, New York.