SKI Report 98:30

Failure Rates in Barsebäck-1 Reactor Coolant Pressure Boundary Piping

An Application of a Piping Failure Database

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Summary

SKI Report 98:30 documents an application of a piping failure database to estimate the frequency of leak and rupture in reactor coolant pressure boundary piping. The study used Barsebäck-1 as reference plant. The study tried two different approaches to piping failure rate estimation: 1) 'PSA-style', simple estimation using Bayesian statistics, and 2) fitting of statistical distribution to failure data. A large, validated database on piping failures (like the 'SKI-PIPE' database) supports both approaches. In addition to documenting leak and rupture frequencies, the SKI report describes the use of piping failure data to estimate frequency of medium and large loss of coolant accidents (LOCAs). This application study was cosponsored by Barsebäck Kraft AB and SKI Research. Urho Pulkkinen and Kaisa Simola (Technical Research Centre of Finland; VTT Automation) performed an independent peer review of the final manuscript to this report.

Sammanfattning **[Summary in Swedish]**

Sedan 1994 har enheten för anläggninssäkerhet på SKI bedrivit ett FoU-projekt inom området rörtillförlitlighet. En viktig del av projektet har varit insamling och bearbetning av erfarenhetsdata från kärnkraftverk i Norden såväl som utomlands. Statistisk utvärdering av dessa data har möjliggjort bestämning av läckage- och brottfrekvenser för bl.a. böjar, svetsar och T-stycken ingående i rörsystem innan- och utanför reaktorinneslutningen. Projektet avslutades under 1998 med en tillämpningsstudie avseende prediktering av brottsfrekvenser i rör innanför inneslutningen i Barsebäck-1. Tillämpningsstudien under 1998 samfinansierades av Barsebäck Kraft AB och SKI. VTT Automation (Urho Pulkkinen och Kaisa Simola) utförde oberoende granskning av det slutgiltliga manuskriptet till SKI Rapport 98:30.

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Acknowledgments

The author provides special thanks to Ralph Nyman (SKI project manager) for his strong commitment to bring the 4-year program to successful completion. Special thanks are also extended to Ingemar Ingemarson (BKAB) for his foresight to commission the application study. The excellent work by Lennart Åström[1] (BKAB) in developing PSA_VER1 was key to performing the application study with Barsebäck-1 as reference plant. Mr. Harald Gmeiner (Enconet Consulting Ges.m.b.H.) assisted with the review and enhancement of PSA_VER1.

The author would like to acknowledge the invaluable contributions to the piping failure database by Walter Burchardt (Energie-Versorgung Schwaben AG), Tinko Gantchev (Bulgarian Nuclear Safety Authority), Rudolf Häussermann (Kernkraftwerk Leibstadt AG), Kalle Jänkälä (IVO International, Ltd.), Bernd Schubert (Hamburgische Electricitäts-Werke AG), Ralph-Michael Zander and Adelbert Gessler (Kernkraftwerke Gundremmingen Betriebsgesellschaft mbH), Caroline Lauron and Louise Lund (U.S. Nuclear Regulatory Commission).

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The final version of SKI Report 98:30 was peer reviewed by Urho Pulkkinen and Kajsa Simola (VTT Automation). The work to finalize the study documentation benefited greatly from their review comments.

[1] Currently with Fagerström Industrikonsult AB (Helsingborg, Sweden).

1. Introduction

This summary report presents an application of a database on piping failures to estimate frequencies of leaks and ruptures in medium- and large diameter reactor coolant pressure boundary (RCPB) piping. Together with a description of the technical approach, the report includes an overview of results and insights of a plant-specific analysis of piping reliability with Barsebäck-1 as reference plant.

1.1 Background

Of limited or unproved usefulness, several databases on piping failures exist. Factors such as accessibility (e.g., proprietary data), incompleteness or lack of validation significantly impacts their usefulness. Initiated in mid-1994, the R&D leading to the application study included the development of a validated, comprehensive database on piping failures in commercial nuclear power plants worldwide. Consistent with the data collection standard ISO 14 224 “Collection of Reliability and Maintenance Data for Equipment,” the database development program specified the following characteristics:

- Verification of the completeness of data sources through reviews of full-text event reports and in-service inspection (ISI) reports.

- Verification that data sources have the proper information and that basic information (population, material composition, diameter, wall thickness, installation data, operating period) on the piping is available.

- Well documented set of failure definitions against which the service data are collected.

- Accurate input of service data, and handling of the data using quality principles (e.g., document control).

A framework for interpreting and analyzing service data on piping evolved with the database development effort. According to that framework, the statistical parameter estimation should be performed on the basis of carefully defined piping reliability attributes and reliability influence factors. Interim results of the R&D were published as SKI Reports 95:58 (SKI, 1995a), 95:61 (SKI, 1995b) and 97:26 (SKI, 1997) and conference papers (e.g., Lydell and Nyman, 1996, 1998). Limited to ASME Class 1, 2 and 3 piping systems, the database currently (end of December, 1998) includes well over 3,000 event reports addressing significant degradations (cracks in the through-wall direction) and failures (small to major leaks and ruptures). The database also includes an additional 400 event reports addressing water hammer events leading to piping system damage (e.g., failure of hangers or supports), leaks or ruptures.

Historically, different technical approaches have been applied to estimate the frequency of pipe leaks and ruptures. These approaches have ranged from direct statistical estimation using the available service data to application of probabilistic fracture mechanics models. In the subject application, statistical estimates of leaks and ruptures were obtained through consideration of trends in event frequencies correlated to the length of service time. The statistical estimation process was intimately coupled to a framework for interpreting and analyzing service data on occurred flaws (e.g., cracks) and leaks. The reader is referred to SKI Report 97:26 for additional background information.

1.2 Conventions & Definitions

Piping failures occur because of degradation and failure mechanisms not accounted for in the original piping system design, fabrication and installation. Collecting quality data on degradations and failures enable direct statistical estimation of pipe leak and rupture frequency. Requirements for reliability data analysis differ significantly between active components and passive components, however.

Since no major RCPB piping failures in BWRs have occurred, piping reliability analysis builds on interpretations of data on incipient and degraded failures. The completeness of databases on piping failures is particularly important when estimating reliability parameters for rare events such as large leaks or ruptures. The *SKI-PIPE* database[2] contains detailed data on degradation and failure mechanisms, root cause evaluations and operating conditions for piping failure events during 1970-1998. It is a periodically updated database on failures in carbon steel and stainless steel piping in commercial nuclear reactors worldwide. This study only considered service data relevant to boiling water reactors (BWRs). It used an archived version of the database (SKI-PIPE, Revision 98:4).

All failure event records in the database are mutually exclusive events. While a single weld could contain multiple cracks, even multiple pinholes, for each such instance the database only records one weld failure representative of the most significant crack. The term ‘weld’ encompasses the weld metal and the weld heat-affected zone (HAZ).

Degradation mechanisms addressed by the application study included forms of degradation mechanisms specific to BWR operating environments. Examples of such mechanisms are intergranular stress corrosion cracking (IGSCC) in weld heat affected zones (HAZ), transgranular stress corrosion cracking (TGSCC) in cold worked pipe bends, and thermal fatigue in piping system branch points.

[2] Several recent industry reports on piping reliability, including NUREG-1661 (U.S. NRC, 1999) and EPRI TR-112657 (Dimitrijevic et al, 1999), refers to a database developed for SKI in 1995 (SKI Report 96:20; Bush et al, 1996). It should be recognized that the SKI-PIPE database is independent of the database documented in SKI Report 96:20.

As recorded in SKI-PIPE, the failures are classified as either ‘crack’, ‘pinhole leak’, ‘leak’ or ‘rupture’. An event classified as ‘crack’ implies that the crack tip did not penetrate the pipe wall. By contrast, welds containing pipe wall penetrating cracks of limited with and length but with visible water seepage or drop leakage are classified as pinhole leaks. Events involving at-power leaks discovered through normal global or visual leak detection systems are classified simply as ‘leaks’. Finally, the term ‘rupture’ implies a sudden, major piping failure having a significant effect on plant operations. The consequence could be a large release of process medium (say, 50 kg/s, depending on size and location), or complete separation of pipe-ends (e.g., double-ended guillotine break).

The term ‘failure’ implies that a corrective action was taken to refurbish a piping system. Examples of corrective actions include repair by using the weld overlay repair technique, replacement using piping component of same type and material composition as the original design, replacement using piping component of same size and schedule but of different layout or material composition, and replacement of an entire piping system using a material composition different from the original design. The term ‘failure’ also has a risk connotation. Depending on risk significance, failures are classified as incipient, degraded or complete failures.

Derived weld leak and rupture frequencies build on data interpretations and not on application of physical models of failure. The data interpretations assume weld crack *initiation* to be a function of the quality of piping fabrication and installation. Crack *propagation* is assumed to be a function of plant thermal transient history and operating conditions. Hence, the occurred weld cracks as recorded in the database represent manifestations of quality deficiencies during plant construction, and of plant thermal transient histories. Throughout the report, all pipe diameters are quoted as nominal diameters (DN) in millimeters. Also, the report uses the terms ‘small-’, ‘medium-’ and ‘large-diameter’ piping to mean piping of diameter < DN100, 100 ≤ DN ≤ 250, and > DN250, respectively.

1.3 Objectives

The application study was concerned with the estimation of RCPB piping leak and rupture frequencies in Barsebäck-1, a third design generation ABB-Atom BWR unit. These leak and rupture frequencies were input to a component-by-component model of the RCPB piping representing loss of coolant accident (LOCA) initiating events. In Barsebäck-1, the RCPB consists of the following ten piping systems; the plant-specific system IDs are given in parentheses:

1. Main steam system (System 311) up to the outside containment isolation valves.
2. Main feedwater system (System 312) from the outside containment isolation valves to the reactor pressure vessel (RPV).
3. Main reactor coolant recirculation system (System 313) in its entirety.

4. Nuclear steam pressure relief system (System 314).
5. Residual heat removal system (System 321) from the recirculation system and to the outside containment isolation valves, and from the outside containment isolation valves to the System 312 branch connections.
6. Emergency core cooling system (System 323) from the outside containment isolation valves to the RPV.
7. RPV head cooling system (System 326) in its entirety. This system connects to the recirculation pump discharge side of the main recirculation loops.
8. Auxiliary feedwater system (System 327) from the outside containment isolation valves to the System 312 branch connections.
9. Standby liquid control system (System 351) from the outside containment isolation valves to the RPV.
10. Hydraulic control rod insertion system (System 354) from the outside containment isolation valves to the respective control rod group.

The evaluation was limited to typical piping components (bends/elbows, nozzles, pipes, tees and welds). Failures of other types of passive components (e.g., pump casings, rupture discs, valve bodies) were not addressed by the study. In Barsebäck-1, the RCPB piping nominal diameter ranges from DN650 (recirculation system; System 313) to DN25 (hydraulic scram system; System 354). The impact of breaks in instrument sensing lines (DN8-DN10), sample lines and vent lines (DN20) was also accounted for.

Consideration of statistical uncertainty is an integral part of risk and reliability analysis. It is recognized that uncertainty analysis is particularly important when modeling rare events such as medium- and large-diameter pipe ruptures. The work scope did not include a comprehensive uncertainty analysis, however. Instead, the study included a qualitative evaluation of the impact by model and data uncertainties on the overall insights and results. Also, uncertainty propagation was selectively performed to illustrate the confidence intervals of derived leak and rupture frequencies.

1.4 Report Organization

This report documents the data analysis methodology together with results and insights of a plant-specific application of derived piping component leak and rupture frequencies. It is a summary report of a data analysis effort performed over a relatively long time. The report is divided into eleven chapters and seven appendices as follows:

Chapter 1 Introduction, study conventions and objectives.
Chapter 2 Service data on piping failures, description of SKI-PIPE; time-dependent failure rates; service data specific to Barsebäck-1/2; basic data analysis considerations; technical organization of application study.
Chapter 3 Models of piping reliability.

Chapter 4 Description of the Barsebäck-1 Reactor Coolant Pressure Boundary piping systems and description of the Barsebäck-1 LOCA initiating event model.

Chapter 5 Estimation of 'baseline' weld leak and rupture frequencies due to IGSCC. The impact of corrective actions on leak and rupture frequencies.

Chapter 6 Estimation of 'baseline' leak and rupture frequencies in piping susceptible to thermal fatigue.

Chapter 7 Development of a piping component reliability database for Barsebäck-1 and quantification of piping system leak and rupture frequencies.

Chapter 8 Medium and large LOCA frequencies in Barsebäck-1 PSA.

Chapter 9 Results & insights including a discussion of the sensitivity and uncertainty analysis results. Review of technical issues in piping reliability parameter estimation.

Chapter 10 Conclusions and recommendations.

Chapter 11 References.

Appendix A Abbreviations, Acronyms & Notation

Appendix B Barsebäck-1 RCPB piping component populations.

Appendix C SKI-PIPE - database content as of 12/31/98.

Appendix D Database Structures - PSA_VER2 and SKI-PIPE.

Appendix E Vibration-fatigue in small-diameter piping.

Appendix F Note on the potential for flow-assisted corrosion (FAC) in RCPB Piping.

Appendix G Note on the statistical analysis of censored data. Hazard plots for a selection of piping systems.

Numerous MS-Excel spreadsheets, spreadsheet programs and MS-Access databases were developed to facilitate the plant-specific application of service data on piping failures. These spreadsheets, spreadsheet programs and databases are not included in the report. All charts and tables displaying service data were based on queries in the archived version of SKI-PIPE. The 'model' of the Barsebäck-1 RCPB piping (PSA_VER2) is proprietary to the plant operator, BKAB (a division of Sydkraft), and therefore not included with the report.

2. Data on Piping Failures

Section 2 presents the piping failure database with data interpretation guidelines. Approximately 50% of the failures in SKI-PIPE apply to piping in BWR units. This BWR-specific service experience includes documented evidence of about 1400 piping failures in BWR plants worldwide. The database content influenced the approach to data analysis.

2.1 Summary of the BWR-Specific Failure Data

As documented in SKI-PIPE, the overall service experience with piping systems in light water reactors is summarized in Table 2-1. The data are organized according to types of degradation and failure mechanisms and pipe size. Next, Figures 2-2 and 2-3 show the BWR-specific database content.

Table 2-1: Summary of SKI-PIPE (Version 98:4) - BWR & PWR Data.

Failure Mechanism		**≤ DN50**				**> DN50**			
I.D.	**Description**	**Type of Failure**				**Type of Failure**			
		All	**Crack**	**Leak**	**Rupture**	**All**	**Crack**	**Leak**	**Rupture**
SC	Stress Corrosion Cracking	152	20	132	0	794	587	207	0
TF	Thermal Fatigue	36	7	27	2	63	31	32	0
E-C	Erosion-Cavitation	3	0	3	0	7	0	7	0
CF	Corrosion-Fatigue	9	0	9	0	11	4	7	0
E/C	Erosion / Flow Accelerated Corrosion	208	2	193	13	236	11	180	45
COR	Corrosion Attack / MIC	84	1	80	3	80	3	74	3
VF	Vibration Fatigue	670	14	592	65	96	6	85	5
D&C	Design & Construction Defects	148	2	140	6	68	5	61	2
WH	Water Hammer	71	7	47	17	89	14	31	44
HE[c]	Human Error	45	0	44	1	16	0	15	1
UNR	Unreported Cause	103	0	102	1	86	0	83	3
	All Mechanisms	1530	53	1369	108	1546	661	782	103

All failure data in Table 2-1 are for piping components external to the reactor pressure vessel (RPV). Also, the database is limited to events involving damage to piping components in the through-wall direction of base or weld metal. For welds the database is limited to failures of 'Type 1' per Figure 2-1.

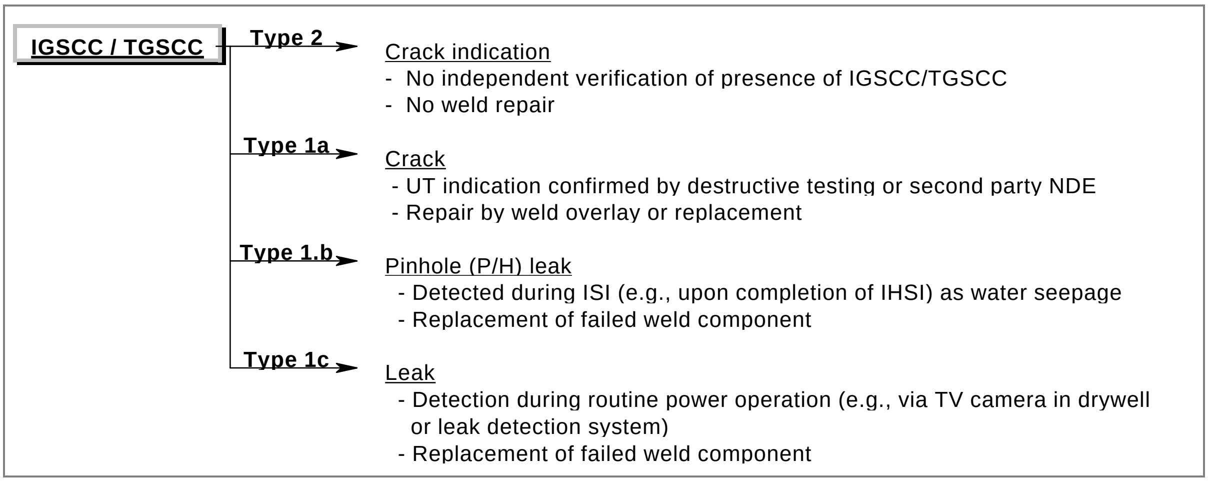

Figure 2-1: Failure Definitions Based on Manifestations of IGSCC & TGSCC.

Specifically, the definitions in Figure 2-1 apply to welds susceptible to IGSCC in BWR operating environments. Often, weld cracks originate in the transgranular mode and propagate in the intergranular mode.

In Figure 2-2 the service data on cracks, leaks and ruptures are differentiated by plant type (BWRs and PWRs) and by year of commercial operation. The piping failures are strongly time-dependent, showing a decline after the first 10 to 15 years of operation. The population of operating PWR plants is about twice that of BWRs.

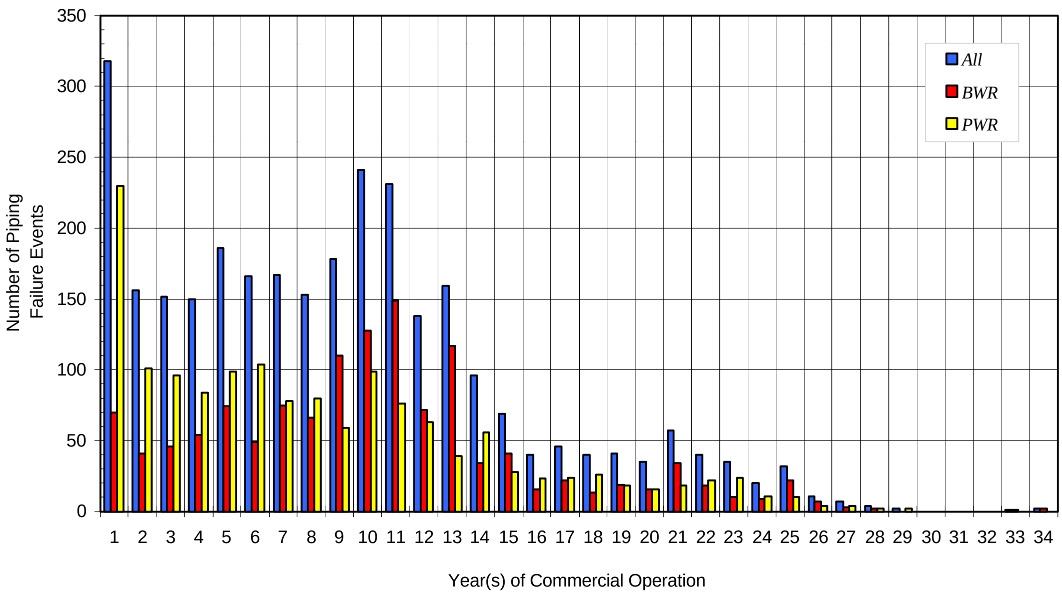

Figure 2-2: Piping Failures in BWR & PWR Plants by Year of Operation.

Figure 2-3 (page 8) shows service data specific to ASME Class 1, 2 and 3 piping systems in BWR plants. The data are organized in two groups: (1) Small-diameter piping failures by year of operation; and (2) Medium- to large diameter piping failures by year of operation. Again, this data summary emphasizes the strong time dependency of piping failures. For medium- to large-diameter piping, a sharp decline occurs after the 13^{th} year of operation. Figures 2-4 through 2-6 summarize the worldwide BWR-specific service experience with piping susceptible to flow-assisted corrosion (FAC), IGSCC and vibration-fatigue, respectively. The vibration-fatigue data almost exclusively relates to failures in small-diameter piping such as instrument lines, drain lines, sample lines and vent lines. The number of operating BWR units is not constant for each year. In Figures 2-4 through 2-6 the number of failures is scaled according to the number of operating plants for each year.

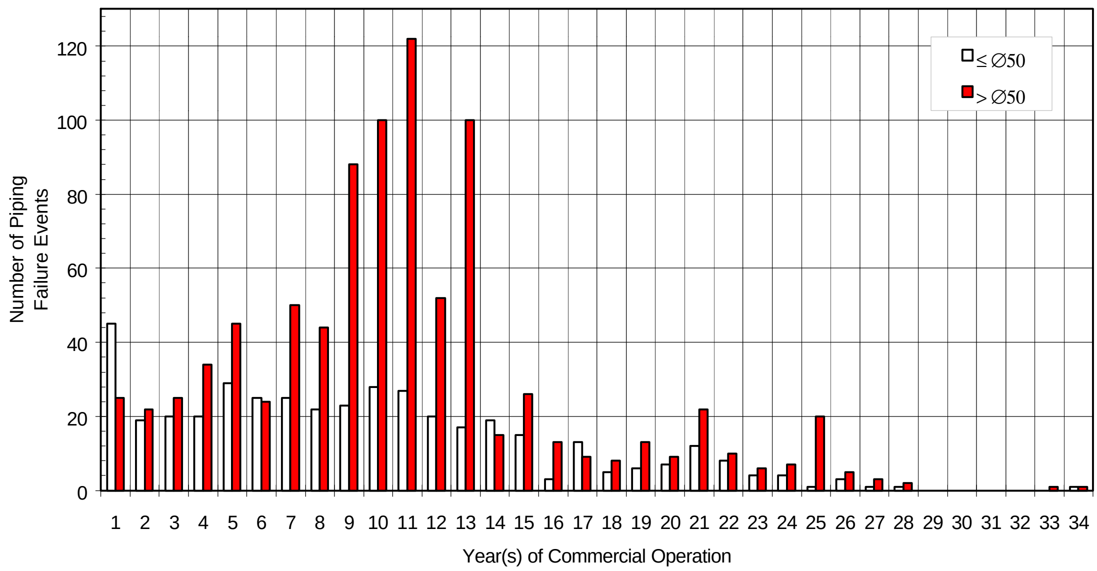

Figure 2-3: Piping Failures in BWR Plants by Pipe Size and Year of Operation.

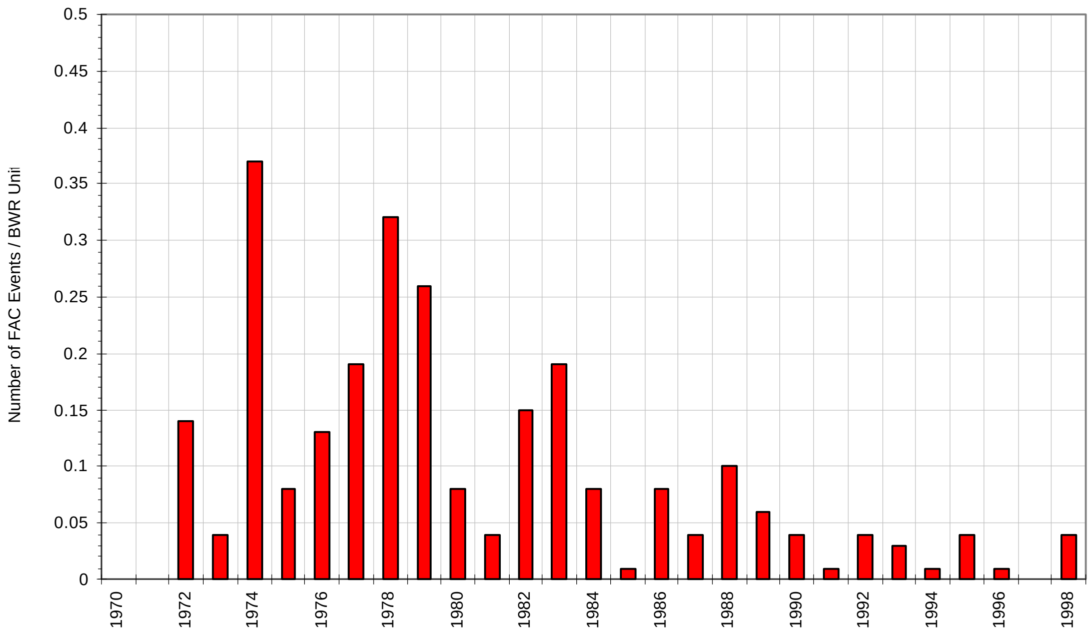

Figure 2-4: Flow-Assisted Corrosion in BWR Units Worldwide.

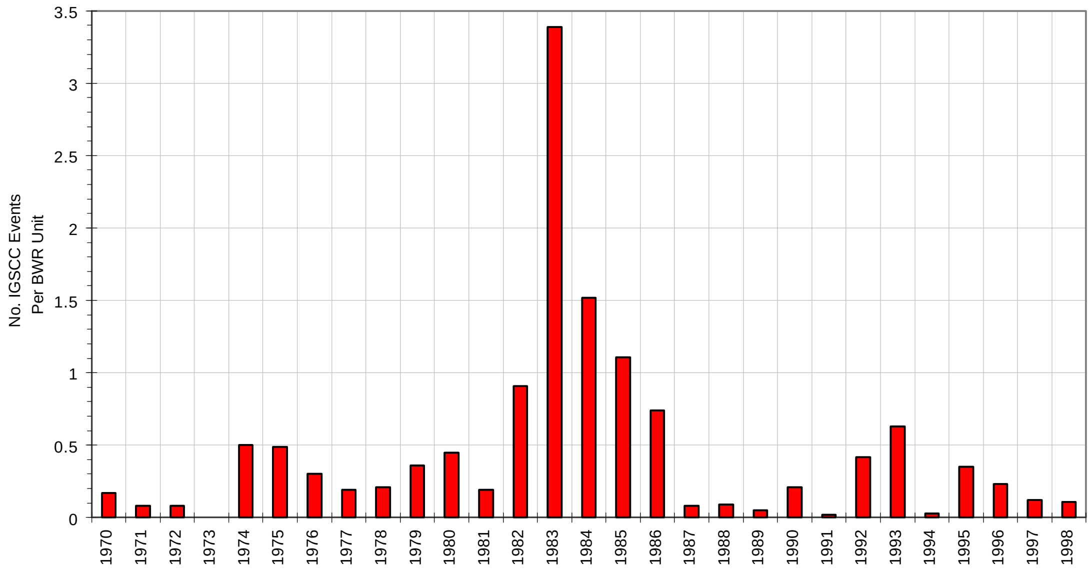

Figure 2-5: IGSCC in BWR Units Worldwide.

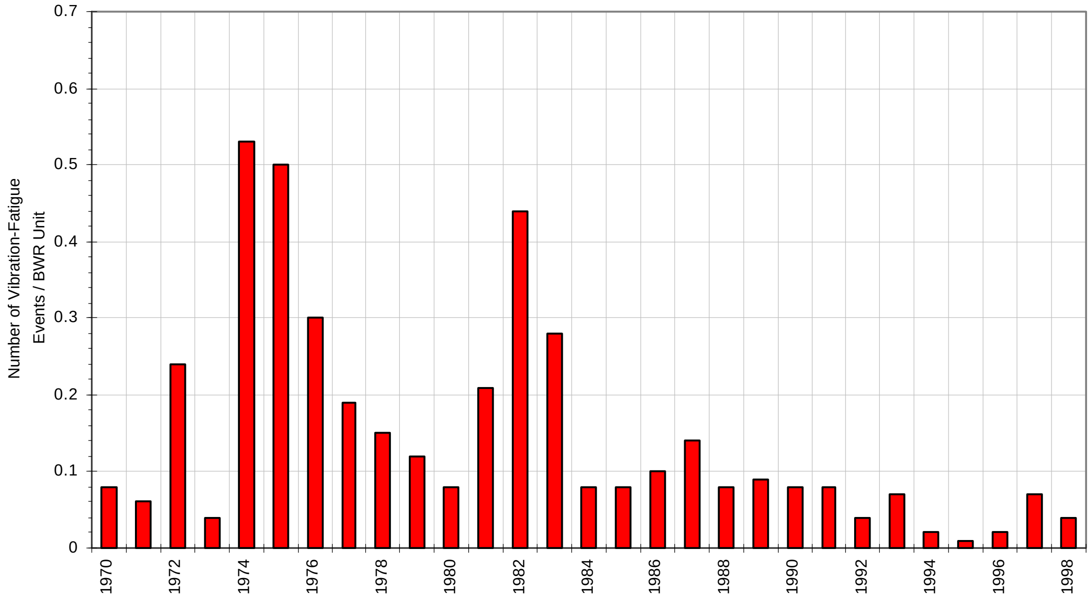

Figure 2-6: Vibration-Fatigue in BWR Units Worldwide.

An important aspect of data analysis involves grouping failure data according to reliability attributes and influence factors representative of a type of piping system for which failure parameters are to be estimated. Grouping of data could reduce the statistical significance. Therefore, the grouping should be done using sound technical justifications to ensure appropriate consideration of service.

2.2 Piping Failures in ABB-Atom BWR Units

Per Table 2-2, nine units of five design generations currently operate in Sweden. The first three generations, comprising five units, have external recirculation loops, while the BWR-4 and 5 units have internal recirculation pumps without large-diameter piping connected to the reactor pressure vessel below top-of-active fuel (TAF).

Table 2-2: ABB-Atom BWR Design Generations.

Unit	Design Generation	Main Technical Design Features
Oskarshamn-1	BWR-1	External recirculation loops. Only unit with internal feedwater riser pipes. Diversification by auxiliary condenser. Fine-motion control rods, diversified shutdown system.
Ringhals-1	BWR-2	Similar to BWR-1 but improved physical separation of the electrical systems. Diversification by steam-driven emergency core cooling and auxiliary feedwater pumps.
Barsebäck-1/2 Oskarshamn-2	BWR-3	Stronger requirements on physical separation of the safety systems. Full two-train electrical separation. Improved electrical supply reliability instead of diversification.
Forsmark-1/2	BWR-4	Full four-train electrical separation. Internal recirculation pumps; no external recirculation piping. Pipe-whip restraints.
Forsmark-3 Oskarshamn-3	BWR-5	Complete physical separation of safety systems. Internal recirculation pumps. Consideration of seismic safety.

Figure 2-7 summarizes service data on piping failures in ABB-Atom units (including the two BWR-4 units in Finland; TVO-1/2). The time-dependent failure trends differ from the industry-wide data shown in Figures 2-2 and 2-3. The shape of the failure trends and its impact on parameter estimation will be addressed in more detail in Chapters 5 through 7.

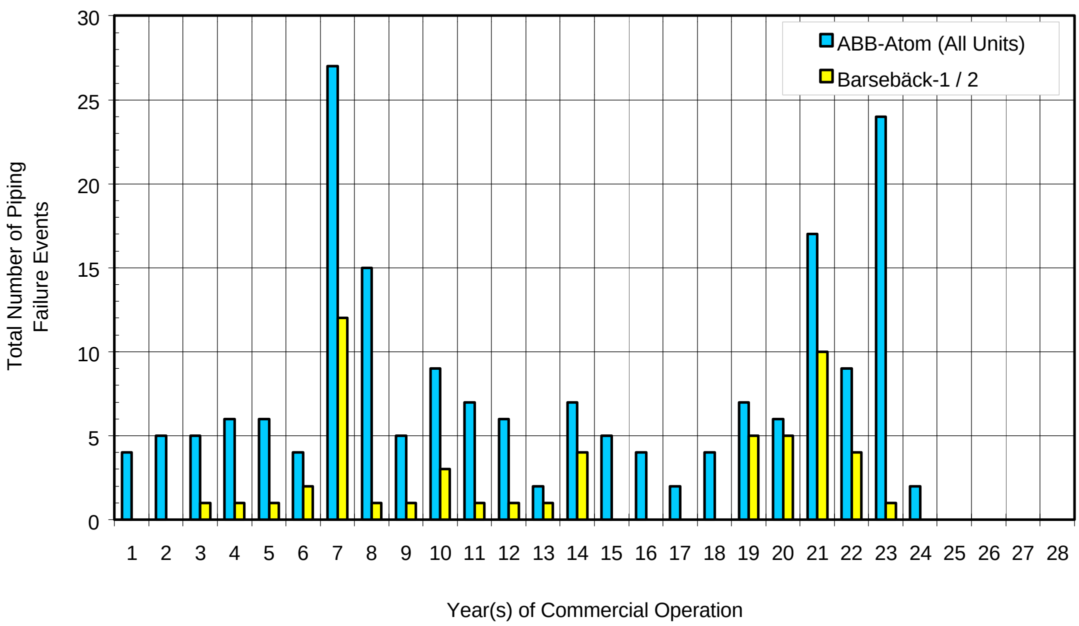

Figure 2-7: Piping Failures in ABB-Atom Plants by Year of Operation.

In-service inspection (ISI) during the cold shutdown plant state is the main method of detecting failures in medium- and large-diameter Class 1 and 2 piping. In part, the time-dependent piping failures reflect inspection practices as much as the impact of corrective actions on susceptibilities to degradation and failure mechanisms. Further, as much as the patterns and trends in Figure 2-7 reflect the ABB-Atom plant-specific piping reliability they also reflect the Nordic regulatory domains and the ISI program plans as implemented by the plant operators.

During the mid to late 1980's, the RCPB portions of the residual heat removal systems of TVO-1 and 2 were replaced with piping of nuclear grade material. Oskarshamn-1, the oldest unit, was in an extended outage during 1993-95. The outage work included modifications to the emergency core cooling system and partial replacement of RCPB piping, welds and nozzles.

The 1997 annual refueling and maintenance outage of Ringhals-1 included the replacement of the RCPB-portion of the residual heat removal system piping. The new piping material is of low carbon content austenitic stainless steel. Also, the work included the replacement of a total of 81 nozzles belonging to the six external main recirculation loops. The 1998 annual refueling and maintenance outage of Barsebäck-2 included replacing the RCPB-portion of the residual heat removal piping with nuclear grade austenitic stainless steel. Use of pre-formed piping sections reduced the weld count.

In Figure 2-8, the service data on piping systems in ABB-Atom plants are organized in two groups: (1) Small-diameter piping failures by year of operation; and (2) Medium- to large diameter piping failures by year of operation.

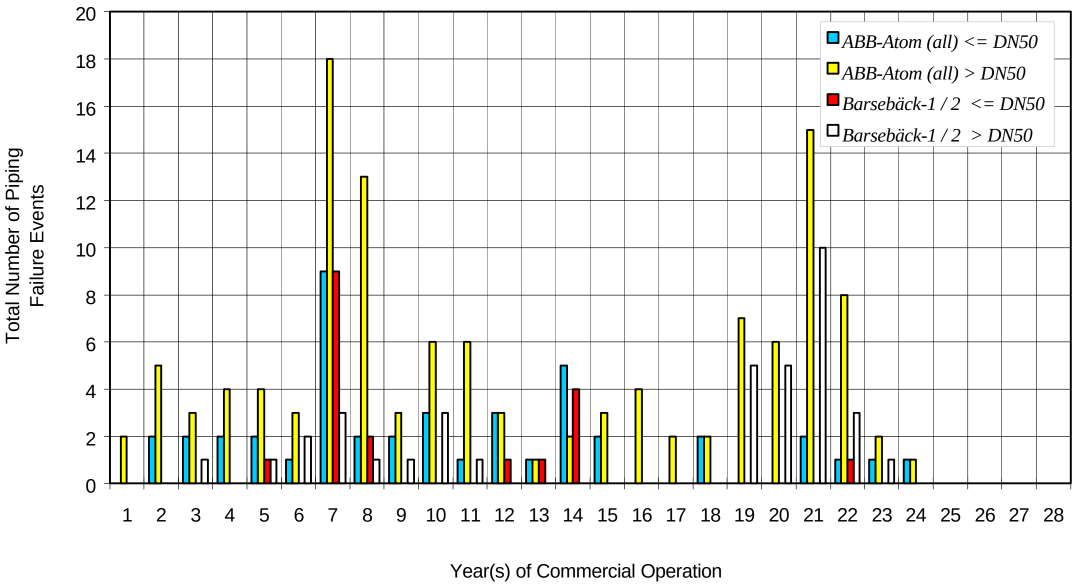

Figure 2-8: Piping Failures in ABB-Atom Plants by Pipe Size and Year of Operation.

For each failure record in SKI-PIPE detailed information on root causes, operating conditions, material, crack morphology, etc. is given by the sixty-one data fields in the database (Appendix D). Many of the data fields are filters for database queries performed to group the data according to reliability attributes and influence factors.

2.3 Data Interpretation Guidelines

Failures of piping system components are location dependent. This means that cracks, leaks or ruptures occur at the weakest points in piping systems. Examples of failure locations include bends or elbows thinned to the point of rupture due to flow-assisted corrosion, and welds cracked by stress corrosion mechanisms. An essential step in data analysis involves organizing the failure data according to reliability attributes and influence factors. Justifications for prior distributions and likelihood functions rest on data interpretations that acknowledge the *why-where-how* of occurred failure events. Evaluations of degradation and failure mechanisms determine the piping component boundary, which in turn determine the form of the failure rate estimators. The dimension of failure frequency could be failure per weld and hour, or failure per piping section and hour. A basic expression for calculating failure frequency is:

$$f_{\mathrm{FAILURE}} = (\textit{Number of Failures}) / (\textit{Time ´ Extension}) \quad (2.1)$$

where ‘Extension’ = Component boundary; e.g., number of welds as defined by a <attribute-influence> set.

In Equation (2.1), the value of the numerator is a function of the database coverage and completeness, which entails capturing all relevant failure events. Accurate event classifications and descriptions enable database queries producing accurate failure event counts. The denominator is a function of the completeness of the piping system design information. It asks for information on the component population of a specific attribute (e.g., material, diameter) known to be susceptible to an influence (e.g., IGSCC). This basic failure frequency estimator requires information on component populations and plant populations (i.e., BWR units having a system representative of a selected attribute). Figure 2-9 shows the number of BWR units covered in SKI-PIPE.

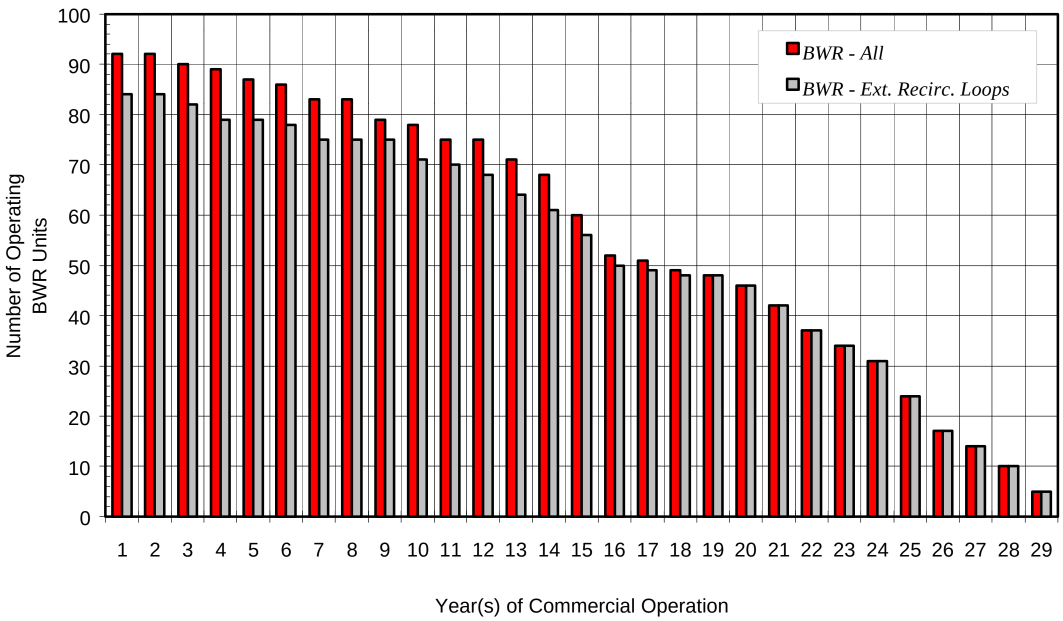

Figure 2-9: The BWR Population by Year of Commercial Operation.

Both the numerators and denominators used in calculating failure frequencies are attribute-sensitive parameters and therefore sources of statistical uncertainties. As indicated by Figure 2-9, not all BWR units have external (to the reactor pressure vessel) recirculation loops. Hence, the denominator should account for plants with external loops when calculating main recirculation weld failure frequencies.

Optimum utilization of a failure event database follows on having a well-defined analysis objective. The database contains no rupture events in medium- to large-diameter Class 1 and 2 BWR pipes. Consequently, the ways of interpreting and grouping incipient and degraded failure events influence the parameter estimates.

The database includes information on incipient and degraded failures; from shallow cracks to through-wall cracks (TWCs). For a TWC to become unstable and possibly rupture, at least 40% of the inside pipe circumference must be cracked (Figure 2-7) at a depth of 100% through-wall or 100% of the inside circumference at a depth of 70% through-wall. Of the data records on IGSCC-induced weld failures, 490 records (circa 65% of total data-base content) include crack sizing data (crack depth and/or crack length). Most of the events fall below, at or slightly above the ‘Repair Criteria’ limit line. A selection of actual data from SKI-PIPE are included in Figure 2-10. None of the events in the database were determined to lie above the ‘Collapse’ limit line.

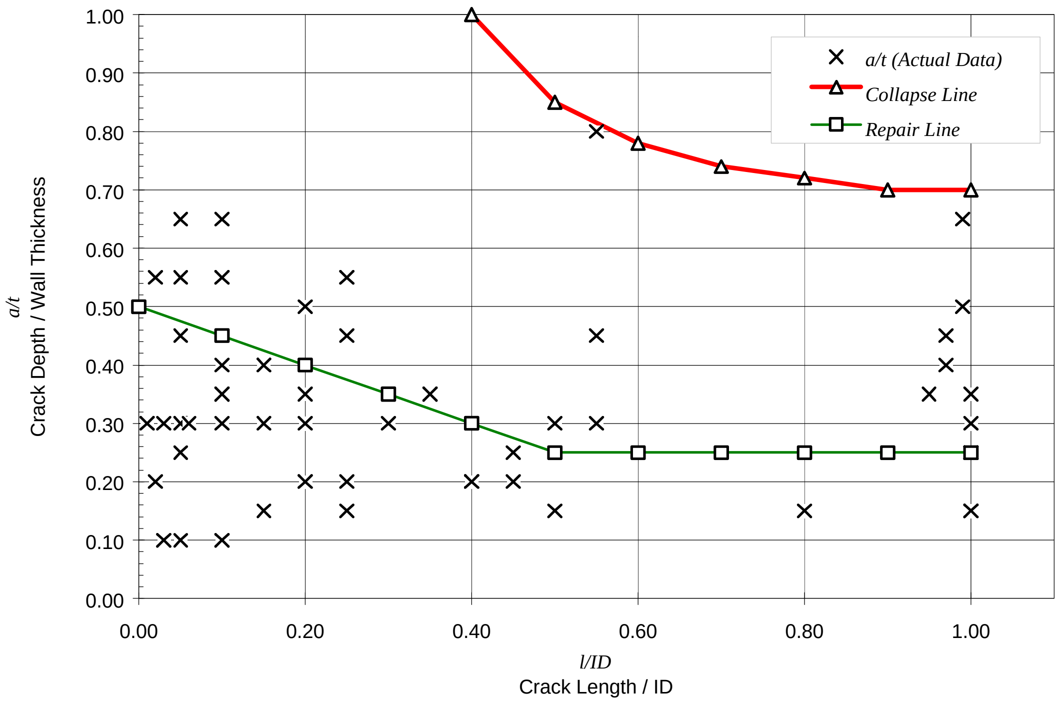

Figure 2-10: Criteria for Classifying ISI Results - Typical Interpretations.

Given the nature of the piping failure data, the statistical parameter estimation begins by defining failure mode criteria. The parameter estimation includes the following analysis steps:

1. Develop a model of piping reliability (see Chapter 3, page 21)

2. Define the failure mode (crack, P/H-leak or leak). Depending on the intended application, it could be the combination of all modes or 'leak' only in the case of rupture frequency estimation supporting the modeling of LOCA events.

3. Define prior distribution (e.g., informative versus noninformative prior, or empirical Bayes). The choice of prior should reflect the piping reliability state-of-knowledge.

4. Determination of population data (i.e., total number of welds).

In this study, the approach to calculating the conditional probability of rupture given weld failure uses the Jeffrey's noninformative prior distribution. The updating of this distribution uses the number of occurred cracks, pinhole leaks and leaks for specified sets of attributes and influence factors. Each event occurrence is assumed to be a function of the thermal transient history of respective plant. Figure 2-11 displays examples of conditional rupture probabilities for IGSCC-susceptible welds.

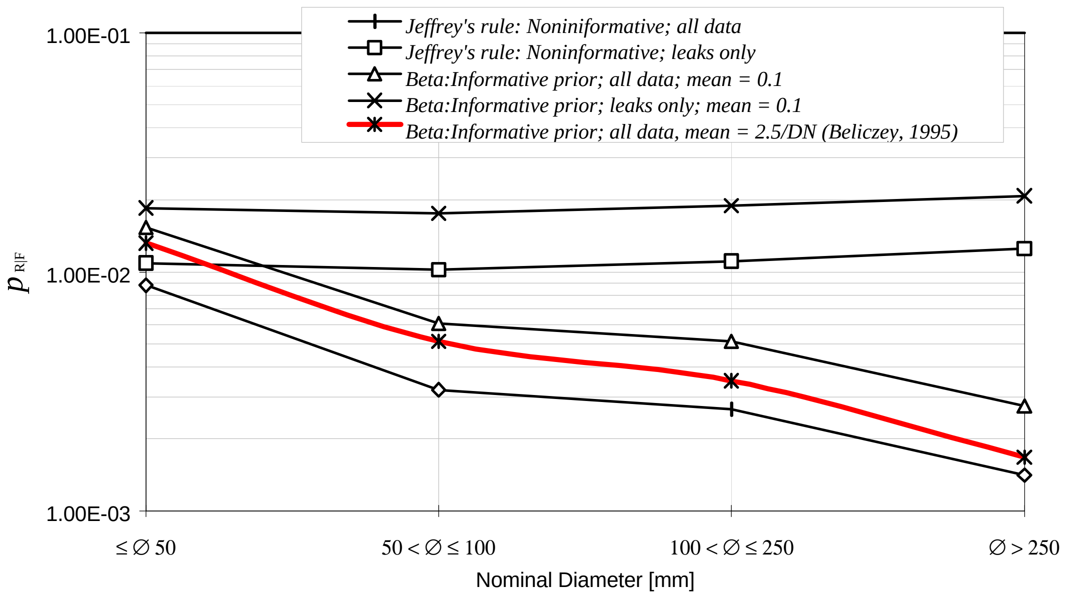

Figure 2-11: Conditional Probability of Rupture Given Weld Failure Due to IGSCC.[3]

A source of uncertainty as well as debate, the definition of prior distributions affects the numerical results. A good overview of practical aspects of Bayesian statistics is found in a recent paper by Siu and Kelly (1998). As indicated in Figure 2-11, this study addressed the topic of defining an appropriate prior through a limited sensitivity analysis using two different distributions. The choice of prior distribution is less critical than the data interpretation approach, however. As shown, the conditional rupture probabilities become insensitive to pipe size when using leak data only.

In the opinion of the author of this report, the conditional rupture probability is strongly dependent on the combination of reliability attributes and influence factors. Conversely, a derived conditional rupture probability should reflect a unique

[3] Chapter 5 documents the basis for how the chart was generated.

combination of piping diameter and material as well as unique degradation/failure mechanism. Again, this technical opinion points to the importance of having a database of sufficient depth (i.e., coverage and completeness). As an example, for IGSCC-susceptible piping *very few* at-power leaks exist in the database. Mostly, the P/H-leaks were induced by ISI-preparations (e.g., pipe surface preparations through grinding) or stress improvement treatment processes (e.g., induction heat stress improvement, IHSI). Unrealistically high conditional rupture probabilities would result if only the at-power leak events were to be used in the calculations. Chapter 5 includes more details on the estimation of conditional rupture probabilities.

As the database development process matures, more research should be directed to the methods and techniques for statistical analysis of piping failure data. In the absence of a scientifically developed basis for data interpretation and analysis, this application study used the Jeffrey's noninformative prior throughout to facilitate parameter estimation. The conditional rupture probabilities as shown in Figure 2-11 built on data interpretation. By contrast, Figure 2-12 shows results from a recent fracture mechanics evaluation (Rahman et al, 1995) of recirculation system piping in a General Electric boiling water reactor.

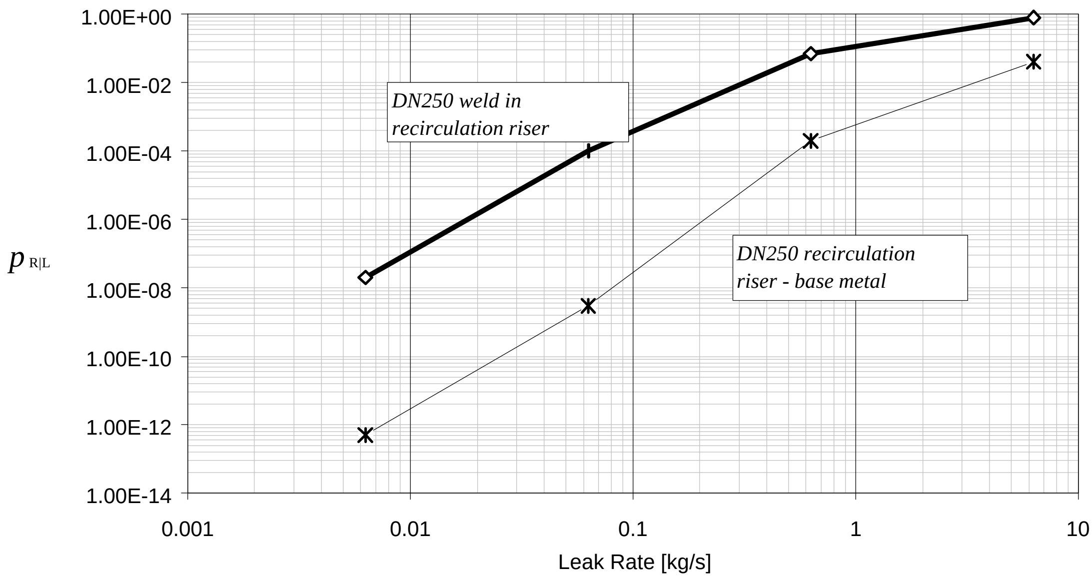

Figure 2-12: Conditional Rupture Probability According to Fracture Mechanics.

The largest IGSCC-induced leak rate recorded in SKI-PIPE is for the U.S. plant Duane Arnold. On June 17, 1978[4], a containment entry was made to identify a primary system leakage which had existed for about three days. The leak rate was determined to be approximately 0.2 kg/s coming from a through-wall crack in a DN250 safe-end weld. In February 1980, the Spanish plant Santa Maria de Garona experienced a 0.05 kg/s leakage due to IGSCC damage in recirculation system piping.

[4] The event was reported to the U.S. Regulatory Commission in LER 50-331/78-030.

2.4 Organization of the Application Study

The data parameter estimation needs (*c.f.* Equation 2.1, page 12) and data interpretation guidelines determined how the technical work was organized. First, a detailed review of isometric drawings, and fabrication and material data provided the necessary design information on the RCPB piping systems. This evaluation produced an extensive database (in MS-Access) containing reliability attribute data for each of the circa 4,000 piping components comprising the ten Barsebäck-1 RCPB piping systems. Second, reviews of the industry-wide and plant specific service data applicable to the ten systems provided the reliability influence factors used to create queries in SKI-PIPE as well as a basis for grouping of failure data. Finally, the attribute and influence data were pooled to facilitate the calculation of leak and rupture frequencies. The total work scope was accomplished in five steps as described below (and Figure 2-13).

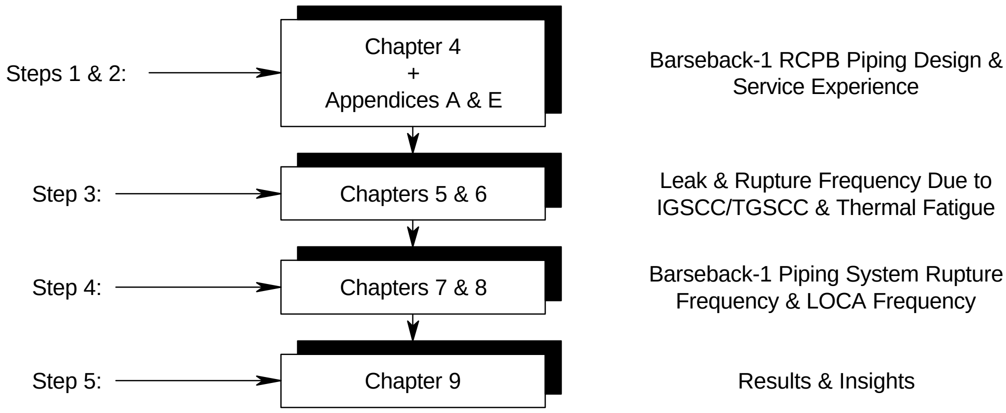

Figure 2-13: Guide to SKI Report 98:30.

Step 1 involved an independent review of the database on all RCPB piping system components. The PSA_VER1.mdb was originally prepared by BKAB personnel during 1997 (Åström, 1997). In its initial form, the database addressed both Units 1 and 2 at Barsebäck. At the conclusion of the pilot project a new database, specific to Unit 1 had been created, consisting of close to 4,000 data records on bends, pipe, welds and tees (Figure 2-14).

The enhanced database (PSA_VER2.mdb), includes information on weld locations, piping components (bends, elbows, pipes, tees), elevations (above and below top-of-active-fuel), ISI histories, material data, results of the degradation/failure mechanism evaluations, and leak and rupture frequencies. Described in Chapter 4, this database represents the model of the RCPB piping systems used to derive LOCA initiating event frequencies.

As formulated in a paper by Fleming and Gosselin (1997a) on piping failure, the service experience shows that piping failures result from degradation mechanisms and loading conditions not anticipated in the original piping design. Since the likelihood of a failure is strongly dependent upon presence of an active degradation mechanism, the data on the service experience support direct estimation of the probability of pipe rupture. This premise lead to the formulation of Step 2

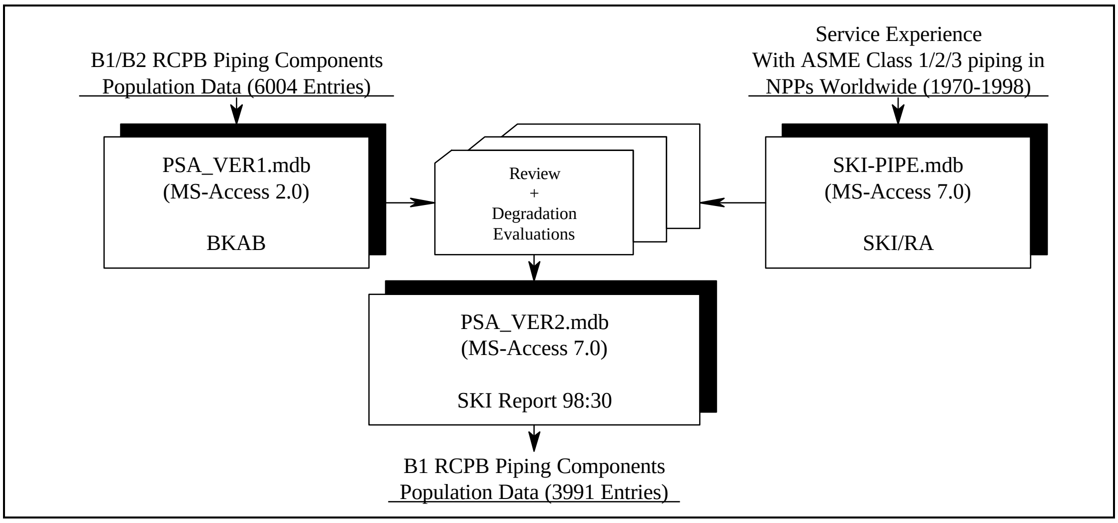

Figure 2-14: The Databases of the R&D Project.

In Step 2, the piping components in PSA_VER2.mdb were evaluated by comparing actual piping design (material, pipe size, component type) and operating conditions to service experience with piping of equivalent attributes. System descriptions included current information on material compositions, known degradation and failure mechanisms, and summaries of piping replacements due to occurred cracks and leaks. Extensive use was made of the information in SKI-PIPE.mdb, supplemented by recent topical reports on degradation and failure mechanisms; e.g., fatigue (Mehta and Gosselin, 1995), thermal fatigue (Su, 1990; Stevens, 1998), water hammer (Van Duyne, Yow and Sabin, 1992; Griffith, 1997). Table 2-3 summarizes main features of the ten RCPB piping systems subjected to degradation evaluation.

Table 2-3.1: Barsebäck-1 RCPB Piping Systems.

System	Connecting Systems	Known Degradation Mechanisms	Material	Susceptible Component[5]
311 Main steam (MS)	314, 316	IGSCC/TGSCC, corrosion, FAC, vibration-fatigue, severe overloading	Carbon steel (CS) (St 45.9/III)	Bends; Welds
312 Main feedwater	321, 327	IGSCC, thermal fatigue	Mainly stainless steel (SS)	Pipes; Tees; Welds
313 Reactor Recirculation	211, 321, 326, 352	IGSCC, weld embrittlement	SS-clad CS, venturi pipes in austenitic SS, butt rings in Inconel 182	Welds
314 MS pressure relief	311, 316	TGSCC, corrosion in lines with stagnant condensate, weld embrittlement, vibration-fatigue	Valve impulse lines in SS, relief lines leading to the suppression pool in galvanized CS	Bends; Pipes; Welds

Table 2-3.2: Barsebäck-1 RCPB Piping Systems.

[5] Susceptibility to degradation and/or failure mechanism. Based on actual occurrences in Barsebäck-1 and generic insights from the piping failure database.

System	Connecting Systems	Known Degradation Mechanisms	Material	Susceptible Component[6]
321 Residual heat removal	313, 327	IGSCC, thermal fatigue	Mainly SS	Bends; Welds
323 ECCS	213	IGSCC, weld embrittlement, vibration-fatigue	Austenitic SS	Bends; Welds
326 RPV head spray	313	IGSCC, weld embrittlement, vibration-fatigue	Austenitic SS	Bends; Welds
327 Aux. Feed	312, 321	IGSCC, vibration-fatigue of small-diameter piping	Austenitic SS	Bends; Welds
351 SLCS	213, 352	IGSCC, B/A-CC, vibration-fatigue of small-diameter piping	Austenitic SS	Bends; Pipes; Welds
354	211, 221, 316	IGSCC, vibration-fatigue	Austenitic SS	Bends; Welds

The active degradation mechanisms result from combinations of design characteristics, environment, service conditions, and operating environments. While their presence cannot be eliminated with absolute certainty, the negative effects can be minimized through the implementation of appropriate measures. Step 2 acknowledged the following attribute and influence factors:

- **Design Characteristics**. Design characteristics include material composition, pipe diameter and wall thickness, component type. The design characteristics vary between systems and can occasionally vary within a system.

- **Fabrication Practices**. These practices include material selection, weld filler material, heat treatment, forming method, bending method.

- **Operating Conditions**. The operating conditions determine the internal and external environments that impact material degradation. These include operating temperatures and pressures, fluid conditions (stagnant, laminar, turbulent flow), quality of process medium (primary water, raw water, dry steam, wet steam), chemical control, service environment (humidity, radiation, etc.).

- **Service Experience**. The operating experience with a particular piping system provide confirmation that damage mechanisms identified for a specific location are appropriate and complete.

Step 3 consisted of organizing, evaluating and analyzing the failure data according to the attributes and influence factors as identified in Step 2. Established methods of reliability life data analysis were used; see Table 2-4. The estimation of piping reliability parameters especially considered the development of hazard functions by recognizing the partial and complete failures (i.e., the penetration of pipe wall by growth-type

[6] Susceptibility to degradation and/or failure mechanism. Based on actual occurrences in Barsebäck-1 and generic insights from the piping failure database.

degradation mechanisms), renewals by repair or replacement, as well as non-failures. While some plants have experienced incipient, degraded and complete failures, many plants have operated for long periods without any failures.

Table 2-4: The Main Steps of Analyzing Service Data on Piping.

Activity	Objective	Comment
1	Organize the service data by attribute & influence factor(s). Ensure mutually exclusive groups of data.	SKI-PIPE includes service data on a wide range of plant designs and piping systems. The service data must be organized to reflect a specific application.
2	Define failure criterion; develop/apply a physical model of failure (e.g., crack propagation law).	A failure criterion is necessary to ensure consisted data interpretations and data pooling strategies
3	Determine the total number of 'failures' for the specific combination of attribute & influence.	Unless Activity 2 & 3 are performed consistently, the parameter estimates could be overly conservative.
4	Determine the total time in service for component boundary of choice.	Very important, and time consuming activity. Requires detailed knowledge of piping system design.
5	Estimate the reliability parameters	The quality of the estimate(s) is a function of the efforts expended on Activity 1 through 4.

Any variability in estimated leak and rupture frequencies could be related to the quality of the database development effort as well as the method of estimation. Data completeness becomes particularly important when analyzing rare event data. Once the initial hurdle of establishing a database has been overcome, the parameter estimation can be streamlined and simplified. As in all reliability data work, the analyst responsible for analyzing the service data must be intimately familiar with the nature of the degradation or failure mechanisms, failure modes, and piping designs. In part, the required level of familiarity is obtained by reviewing service data and root cause analysis reports. In part, the familiarity is obtained via degradation evaluations as in Step 2.

Step 4 consisted of quantifying piping system leak and rupture frequencies, and ultimately LOCA initiating event frequencies using the component-level frequencies in Step 3. The following LOCA initiating events were quantified:

- Medium LOCA (S1); distinction between LOCA above or below top-of-active fuel (TAF).

- Large LOCA (A); distinction between LOCA above or below TAF.

- Medium LOCA (S1); above or below TAF and with requirement for back-flush operations due to dynamic effects resulting in stripped pipe insulation material clogging the containment sump ECCS strainers.

- Large LOCA (A); above or below TAF and with requirement for back-flush operations due to dynamic effects resulting in stripped pipe insulation material.

- Large LOCA (A); above or below TAF and with dynamic effects from severance of adjacent small- and medium-size piping due pipe whip. The potential consequences of severed small- and medium-size piping could be loss of reactor vessel level indication system, loss of containment spray function, or loss of or degraded core spray function.

In the final Step 5 the interpretation of the results included performing sensitivity and selected uncertainty analysis. Due to the nature of the failure data, reliability parameter estimation involves a series of assumptions. Each of these assumptions could have a large or small impact on results. The impact of changing the assumptions in Step 3 were addressed by re-quantifying the LOCA model. Finally, the statistical uncertainty in predicted results was qualitatively addressed by identifying the sources of data and modeling uncertainty.

3. Models of Piping Reliability

This chapter presents ‘data-influenced’ models of piping reliability that acknowledge different degradation and failure mechanisms. These models reflect the data collection process and the classification of failure events. Except implicitly, the models do not account for crack growth or other complex physical phenomena. Their application assumes access to a large body of observational data on piping failures. An analytical challenge involves the consistent interpretation of service data per the guidelines in Chapter 2.

3.1 Basic Models & Classification of Pipe Failures

A framework for analyzing service data and calculating piping leak and rupture frequencies was presented in SKI Report 97:26. According to the analysis framework, evaluations of service data should be done on the basis of reliability attributes and influence factors. Depending on specific attributes of piping (size, material, material composition, method of fabrication), a piping system could be more or less susceptible to degradation or failure mechanisms (i.e., different influences). Using service data alone for estimating leak and rupture frequencies, three simple models of piping reliability are:

(1) **Active Degradation Mechanism**
Examples include flow accelerated corrosion (FAC), stress corrosion cracking (e.g., IGSCC, TGSCC), thermal fatigue.
Freq.{Rupture} = Freq.{Failure} x Prob.{Rupture½Failure}

(2) **No Active Degradation Mechanism**
Mainly affecting small-diameter piping, a typical failure mechanism includes vibration-fatigue.
Freq. {Rupture} is developed directly from rupture data.

(3) **Piping Which Is Susceptible to Water Hammer**
Freq.{Rupture} = Freq.{Water Hammer} x Prob.{Rupture½Water Hammer}

Within any given piping system, individual piping sections could be susceptible to degradation mechanisms *or* failure mechanisms *or* water hammer. The known or potential degradation or failure susceptibilities must be determined prior to organizing and interpreting a database content. Such determination requires reviewing full text event reports and root cause evaluation reports. In other words, the piping failure database must include enough information to support good data pooling strategies.

It is seldom straightforward to identify underlying causes of an event. For example, pipe ruptures due to water hammer could occur due to combinations of aggressive flow accelerated corrosion, lack of in-service inspection, poor system operating procedures and/or inadequate piping system design. Before pursuing parameter estimation, a systematic degradation mechanism evaluation determines the types of degradation and/or failure mechanisms to consider when organizing and interpreting service data. The symbolic Equation (3.1) gives the frequency of pipe rupture for a general piping system, consisting of piping components of different size and material and susceptible to different degradation and failure mechanisms:

$$F_R\{System\} = \sum\lambda_j(D)n_jP_j\{R|D\} + \sum\lambda_k(R) + \sum\lambda_l(WH)n_lP_l\{R|WH\} \qquad 3.1)$$

$F_R\{System\}$	= Frequency of pipe rupture;
$\lambda_j(D)$	= Failure rate per component (e.g., weld, foot of piping, bend, tee) for all degradation mechanisms in category 'j';
$\lambda_k(R)$	= Rupture rate per component for all failure mechanisms in category 'k' piping;
n_j	= Number of components in category 'j';
$\lambda_l(WH)$	= Frequency of water hammer in category 'l' piping;
n_l	= Number of components in category 'l';
$P_j\{R\|F\}$	= Probability of rupture given failure of component in category 'j' piping;
$P_l\{R\|WH\}$	= Probability of rupture given water hammer in category 'l' piping.
D	= Number of piping degradations; e.g., crack in through-wall direction, leak or rupture;
R	= Number of ruptures;
WH	= Number of water hammer events.

According to Equation (3.1), determining piping system rupture frequency is reduced to estimating piping component leak and rupture frequencies from observational data. The coverage and completeness of the piping failure database and the accuracy of the degradation mechanism evaluation determine the quality of the rupture frequency estimate. Before application, some specialization of Equation (3.1) may be required.

As an example, for IGSCC-susceptible piping the likelihood of weld failure depends on the location the weld in a piping system. As derived from piping failure data, this weld location dependency is shown in Figure 3-1. For a given reliability attribute (e.g., pipe size and material), and if IGSCC is determined to be a predominant degradation mechanism, the frequency of piping system failure should be determined by the frequency of weld failure per symbolic Equation (3.2):

$$F_{D\text{-}System} = \sum_{i=1}^{m} f\,(\boldsymbol{Weld}_i) + \sum_{j=1}^{n} f\,(\boldsymbol{Weld}_j) +,..,+ \sum_{k=1}^{r} f(\,\boldsymbol{Weld}_k) \qquad (3.2)$$

That is, the failure frequency is a function of the contributions from weld failures of type 'i', 'j',…. , 'k'. Index 'D' in $F_{D\text{-}System}$ stands for degradation by IGSCC in the example. This study defines 'weld type' as a characteristic according to the location of a weld in piping; e.g., elbow-to-pipe weld, pipe-to-pipe weld. Figure 3-1 represents the full range of medium- and large diameter stainless steel piping systems. An intrinsic assumption behind symbolic Equation (3.2) is that there is a direct relationship between the achieved quality of welding and the location of a weld in the piping. Some locations (due to factors such as piping geometry, accessibility) could be more amenable to adverse combinations of weld sensitization and tensile stresses.

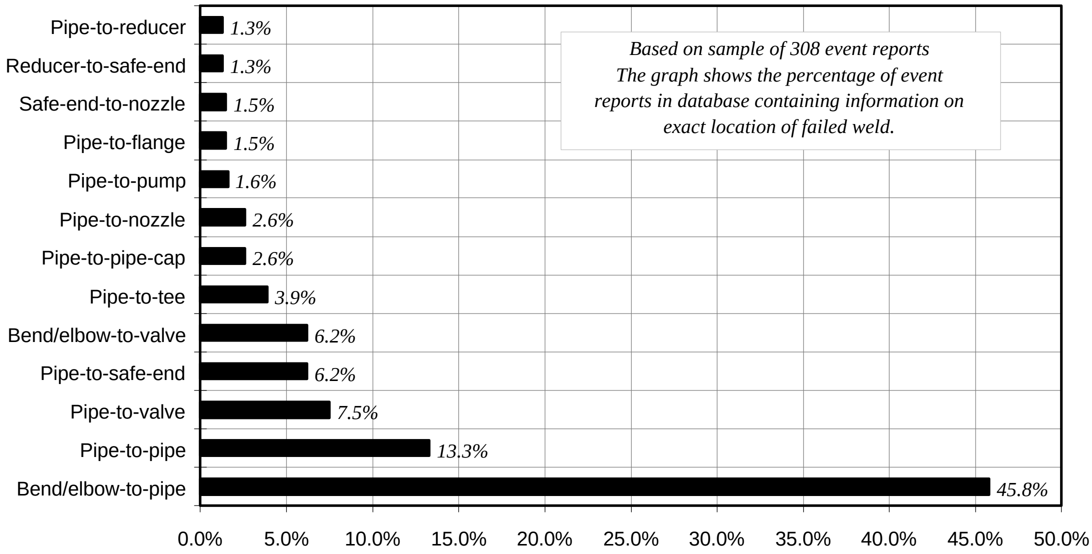

Figure 3-1: Location-Dependency of Weld Failure in IGSCC-Susceptible Piping.

Summarized in Figures 3-2 through 3-5 is information on the 'location-dependency' of weld failures in different systems. Not only do these data summaries reflect the susceptibilities of different weld locations, but also the actual counts of different welds in different systems. Additionally, these data summaries reflect the full range of plant types in the piping failure database.

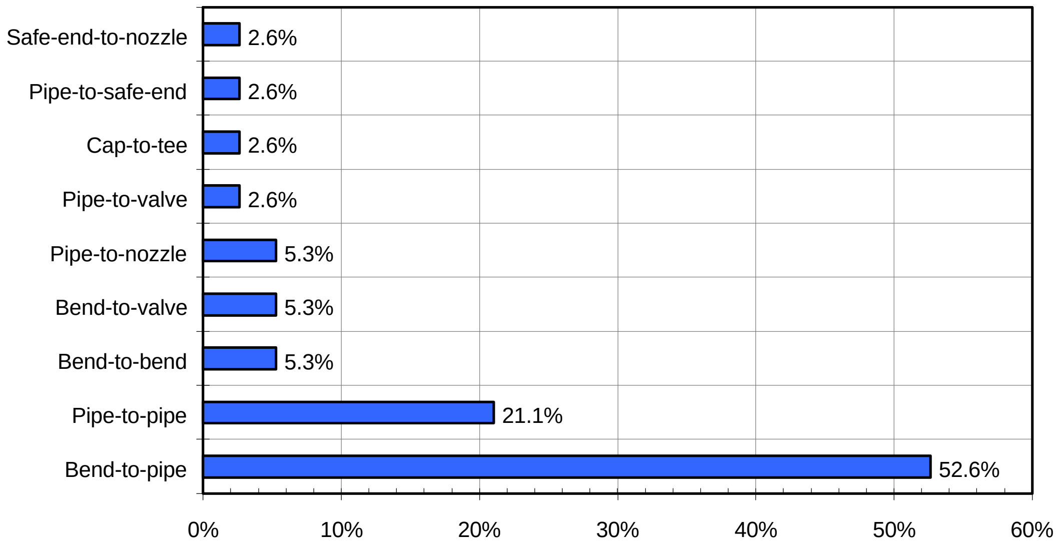

Figure 3-2. Location Dependency of Weld Failure in Medium-Diameter Safety Injection Piping.

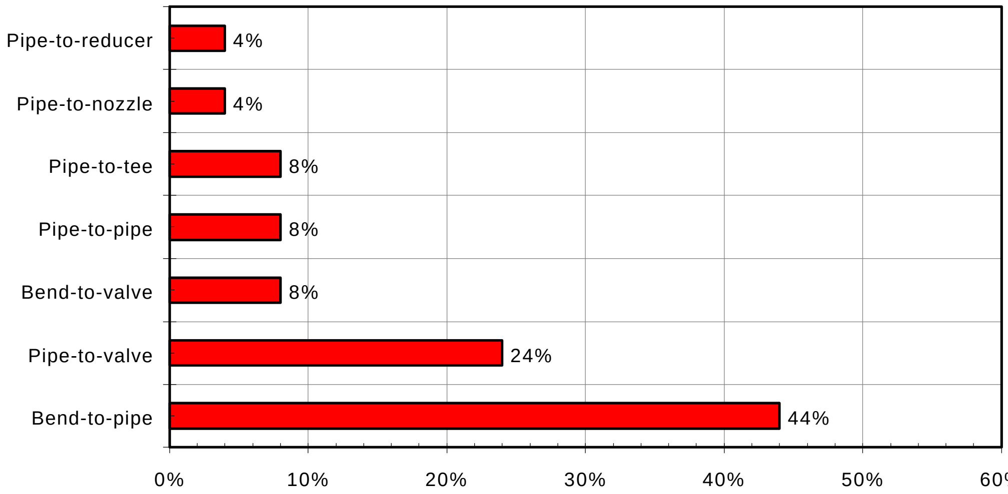

Figure 3-3: Location-Dependency of Weld Failure in Medium-Diameter RHR Piping.

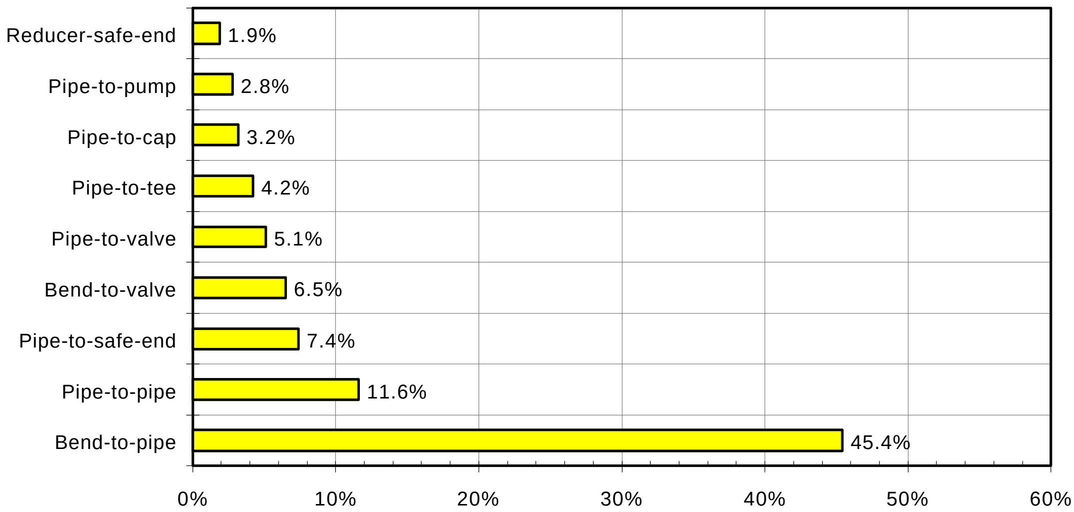

Figure 3-4: Location-Dependency of Weld Failure in Large-Diameter RCS Piping.

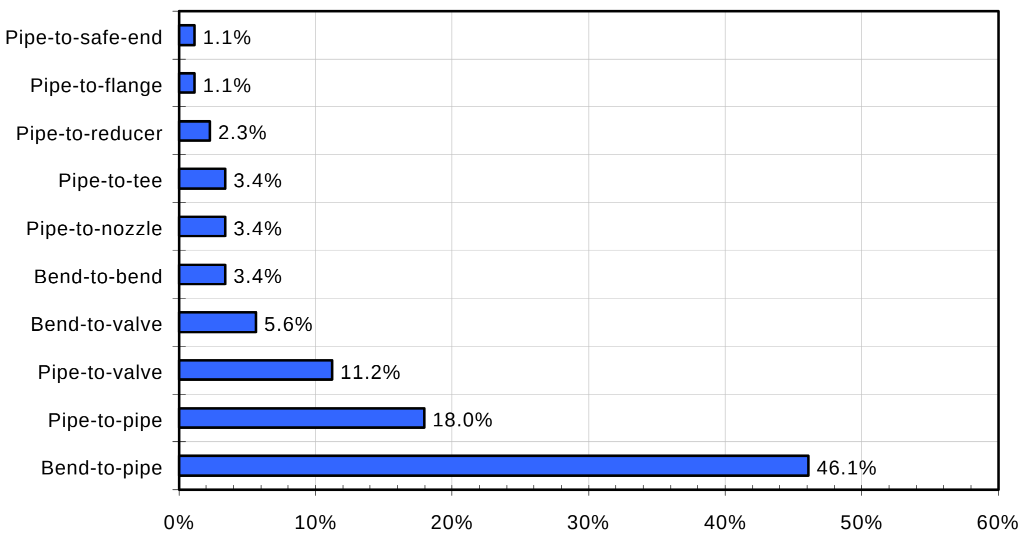

Figure 3-5: Location-Dependency of Weld Failures in SIR Piping.

Another important example of specialization of Equation (3.1) entails consideration of TGSCC in the base metal of bends in cold worked medium-diameter stainless steel piping. For a given reliability attribute, and if TGSCC is determined to be a predominant degradation mechanism, the frequency of piping system failure should be determined by the frequency of bend failure per symbolic Equation (3.3):

$$f_{\text{D-System}} = \sum_{i=1}^{m} f\,(Bend_i) \qquad (3.3)$$

That is, the failure frequency is a function of the contributions from through-wall cracking in base metal of bends. Theoretically, failure propensity of bends in cold worked piping could be a function of bending angle; i.e., 30-degree bend versus, say, 90-degree bend. This study did not explore the piping failure database to determine such correlations, however.

3.2 Time-dependence of Crack Growth

Considering historical service data on austenitic stainless steel piping, IGSCC has been a major degradation mechanism in BWRs. Few circumferential cracks have actually penetrated the pipe wall, however. Mostly, the cracks have been shallow and attributed to improper welding procedures during initial plant construction. Even in the case of significant cracking, the margin to pipe wall penetration is substantial.

An ‘a/t-ratio’ (crack depth to wall thickness) of 60% or more implies a requirement for prompt corrective action. Current service experience indicates that below 60% crack depth, design weld overlay repairs allow continued operation indefinitely. According to an analysis by the New York Power Authority (1990), without corrective action a crack depth of 60% would be reached after approximately 30,000 hours of plant operation. For a through-wall crack to become unstable it must extend more than 40% of the pipe circumference. Figure 3-6 is an example of estimated crack growth applicable to FitzPatrick Nuclear Power Plant.

Kassir (1985) and Andresen (1998) summarize experimental and actual crack growth rates in BWR construction materials. Aaltonen, Saarinen and Simola (1993) demonstrate possible relationships between the TVO-1 transient history and crack propagation in RHR piping welds. Actual crack growth rates can be determined directly from the extensive crack morphology data in the piping failure database. To facilitate statistical parameter estimation using the hazard plotting technique (Nelson, 1972), the data in SKI-PIPE were extrapolated to estimate a fictitious time when a given crack would penetrate the pipe wall. A linear crack growth rate of 5.0E-9 cm/s was used to calculate the time of pipe wall penetration.

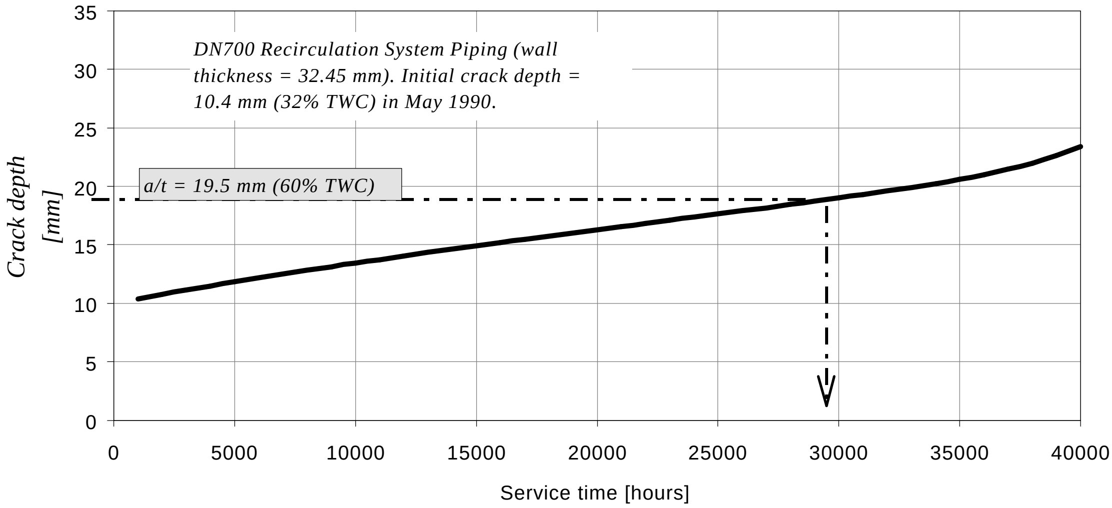

Figure 3-6: Crack Growth in DN700 Reactor Recirculation System Piping.[7]

As an example, assume ISI detected a crack of 80% TWC (a/t = 0.8) in thick-walled, large diameter recirculation piping (35 mm wall thickness) on 1/1/90. The pipe wall penetration could occur after approximately 4.4 years given no corrective action. Conversely, an at-power leak could feasibly have occurred around mid-1994 had the ISI failed to detect this particular flaw. This simplified crack growth extrapolation was used in this study to evaluate the sensitivity of derived piping failure parameters to the completeness of the database. Appendix G includes a selection of hazard plots based on crack growth extrapolation.

3.3 Time-dependence of Thermal Fatigue Damage

In BWRs, crack growth induced by thermal fatigue could be a function of operating cycles involving the injection of hot water from the Reactor Water Cleanup System into cold main feedwater. The operating practices vary from plant-to-plant. Some plants operate with a continuous cleanup flow while others operate with intermittent flow. It is not possible to derive plant operating profiles from service data collections, however. Plant service time was selected as the basis for the parameter estimation in this project, and without adjustments for scheduled or unscheduled outages

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[7] James A. FitzPatrick Nuclear Power Plant, 1990. *Summary of Intergranular Stress Corrosion Cracking Inspection During 1990 Refueling Outage*, JPN-90-041, New York Power Authority, White Plains (NY).

4. Brief Description of Barsebäck-1 RCPB Piping

The Sydkraft Group, through its wholly-owned subsidiary Barsebäck Kraft AB (BKAB) operates the two 615 MWe units at Barsebäck. The contract for Barsebäck-1 was placed with ABB-Atom in 1969 and commercial operation began in July 1975. The two units at Barsebäck and Oskarshamn-2 are nearly identical BWR-3 units (see Table 2-2, page 8). Chapter 4 gives a brief summary of key RCPB piping design features and operating experience.

4.1 RCPB Piping Design Features

In Barsebäck-1, the major RCPB piping consists of four external recirculation loops (Figure 4-1), four main steam lines and two main feedwater lines (System 312). Inside the containment, each of the two incoming feedwater lines split into two risers that connect to the RPV. Except for the recirculation loops and main steam lines, the piping systems are made of austenitic stainless steel material (average carbon content of about 0.04%, or slightly above).

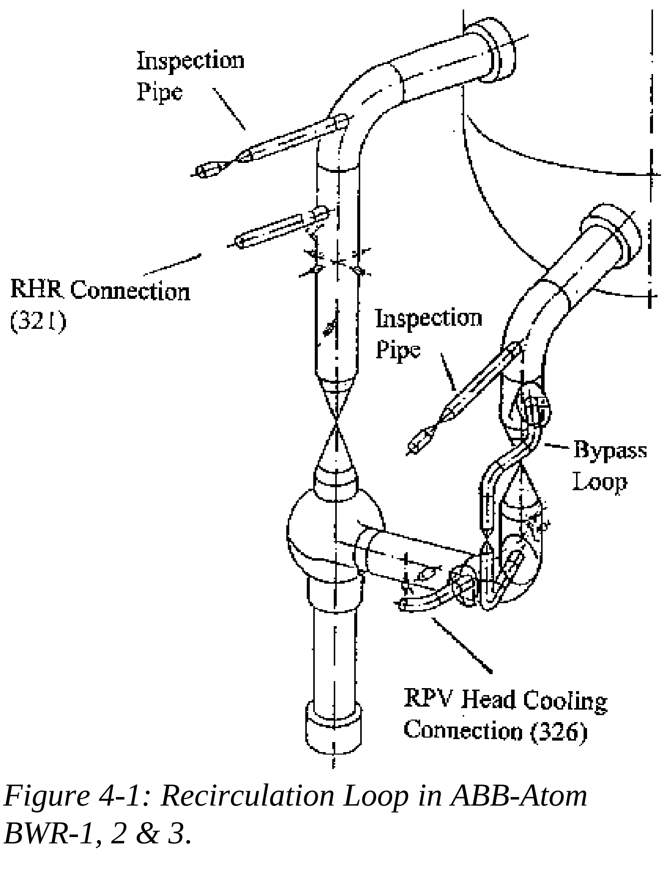


Figure 4-1: Recirculation Loop in ABB-Atom BWR-1, 2 & 3.

The main recirculation loops are made of duplex material (carbon steel with a stainless steel coating). The venturi sections of the recirculation loops consists of austenitic stainless steel piping; vertical pipe sections with nozzles for venturis. Butt-welding rings are Inconel 182 fittings. Welding material between main loops and venturi pipes is Inconel 182. Pump casings and valves in the recirculation loops are cast austenitic stainless steel units. The medium- and large-diameter sections of the main steam lines are made of carbon steel, while the small-diameter sections are made of austenitic stainless steel.

In connection with repairs and scheduled replacements, sections of the original RCPB piping have been replaced with piping of low or extra-low carbon content ('nuclear grade') stainless steels. Main contractor for the original RCPB piping design, including fabrication, was Mannesmann-Rohrleitungsbau AG (Düsseldorf, Germany).

4.2 Operating Experience

Currently at the mid-point of its expected technical service life, the Unit 1 has operated with a capacity factor exceeding 80% for most of its life. Exceptions were calendar years 1979 and 1992. In 1979 the unit experienced a major turbine-generator failure and ensuing turbine building fire (Sydkraft, 1985). In 1992 the unit was shutdown for modifications of the emergency core and containment spray systems following the July 28, 1992 incident at Unit 2 when a rupture disc at the outlet of a safety relief valve inadvertently opened (OECD-NEA, 1996). That incident revealed a weakness in the defense-in-depth, which under another set of circumstances could have led to the emergency core cooling system failing to provide water to the core.

Like other BWRs worldwide, the units at Barsebäck have been affected by stress corrosion cracking (IGSCC) damage in the stainless steel piping (mainly the ECCS and RHR piping). Unit 1 has operated with intermittent hydrogen water chemistry (HWC) since April 1991. For about four months in early 1992 the unit operated without any hydrogen injection. After that period the unit has operated with HWC for 70-90% of the time. The most recent IGSCC occurrence was discovered during the 1997 annual refueling outage when inspecting the RPV Head Spray piping. During the 1996 annual refueling outage, IGSCC was detected in ten different locations of the residual heat removal piping.

Early indications of a thermal fatigue cracking problem generic to ABB-Atom plants were identified during 1978 in Barsebäck-1 and Oskarshamn-2, respectively. Both units experienced thermal fatigue in base metal of piping close to a RWCU/RHR branch connection. In 1979, a significant thermal fatigue incident occurred in TVO-1 (BWR-4 per Table 2-2, page 8) when a DN150 tee in the RWCU System[8] fractured (Holmberg and Pyy, 1993). That event resulted in the release of about 5,000 kg of reactor water in the reactor building. The pipe fracture area was approximately 2.3 cm^2 (maximum 150 mm long and 2 mm wide fracture).

The 1980 ISI-program for Barsebäck-2 was implemented in response to IGSCC-concerns. Liquid penetrant and gamma radiography examinations were carried out externally but gave no positive indications of possible defects in the stainless steel base material close to the Main Feedwater and Residual Heat Removal branch connections. Sections of the branch connections were removed. This revealed visible cracks on the inside of the pipe sections even before cleaning. Using liquid penetrant, and after cleaning, reveled more extensive cracking similar in appearance to a spider's web. That discovery led to an investigation of each similar branch in Barsebäck-1, Oskarshamn-2,

[8] The RWCU System is not part of the RCPB in the ABB-Atom designed BWR units.

Ringhals-1 and Forsmark-1 (which at the time was in the final stages of startup). Except Forsmark-1, all inspected units exhibited thermal fatigue cracks.

During a shiftly walk-down in Oskarshamn-1 in 1979, plant personnel detected a small leak in a DN100 bend in the residual heat removal piping system. The leak was determined to be caused by the combination of TGSCC and IGSCC in the cold worked pipe section. The metallographic evaluations revealed extremely high cold deformation in the affected area, and 8-12% martensite was measured. During the fall/winter 1979-80, additional through-wall cracks and leaks were detected in bends in the RWCU system piping in Oskarshamn-1. Later it was revealed that the particular pipe bending machine used in fabricating the RHR and RWCU piping for Oskarshamn-1 also had been used to fabricate corresponding piping at Barsebäck-1/2 and Ringhals-1. Specifically for Barsebäck-1, the piping failure database does not include any records involving TGSCC/IGSCC in base metal, however.

4.3 RCPB Piping Design Database

Development of piping leak and rupture frequencies representative of a specific plant design generation requires access to accurate and complete information on reliability attributes and piping component populations. For Barsebäck-1, all RCPB piping design information was contained in the PSA_VER1 (Åström, 1997).

Prepared by plant personnel, this database stored essential data for each of the approximately 4,000 RCPB piping components (type, diameter, material, location, inspection data). The database work entailed detailed reviews of the entire isometric drawing package for the ten systems, supplemented by system walk-downs, reviews of inspection records and fabrication data (material compositions for each charge/heat number). An additional aspect of the database work consisted of identifying locations inside containment susceptible to dynamic effects from double-ended guillotine breaks (DEGBs). An example of a dynamic effect could be the break of a small- or medium-diameter pipe caused by pipe whip effects by a large-diameter DEGB. The PSA_VER1 represented the complete LOCA initiating event model for the plant.

In <u>Step 1</u> of the application study, PSA_VER1 was subjected to an independent review which entailed all isometric drawings and of how the information off these drawings had been transferred to the database. The review focused on the accuracy and completeness of the original database. Equally important, the independent review accomplished piping system design familiarization by the data analyst. During the review, the database content was augmented as follows:

- For each system, pipes were added to the database. Originally, the database included bends, tees, welds. In the revised database all pipes were given a unique ID with appropriate reference to the weld components joining a pipe to other components in the piping system.

- The location of each weld was identified by referencing its position in the piping system per the example below (and Figure 4-3).

	Database Fields				
	Bend ID[9]	**Weld ID**	**Weld 'Type'**[10]	**Pipe ID**	**Weld ID**
PSA_VER1.mdb (original)	5522	5523	N/A	N/A	5524
PSA_VER2.mdb (after review)	5522	5523	Bend/pipe	5523-S	5524

- All connecting instrument lines, drain lines, and sample lines were recorded in the revised database.

- Results from the degradation evaluation (Step 2) were entered into the data-base by adding a field for the degradation/failure mechanism potentially impacting respective component. Also, derived leak and rupture frequencies were added.

The modified database, PSA_VER2 (Figure 4-2), provided population data for different sets of reliability attributes and influence factors (Appendix B). In facilitating parameter estimation, queries in PSA_VER2 defined the combinations of attributes and influences for which queries were made in SKI-PIPE.

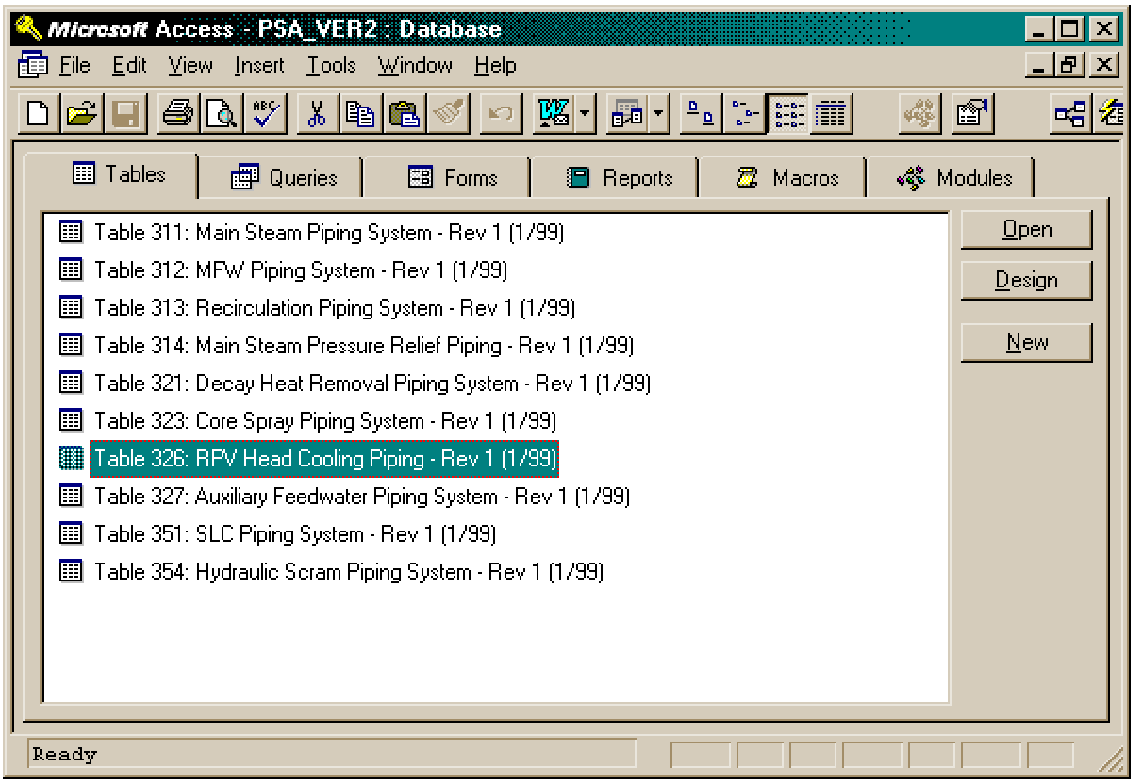
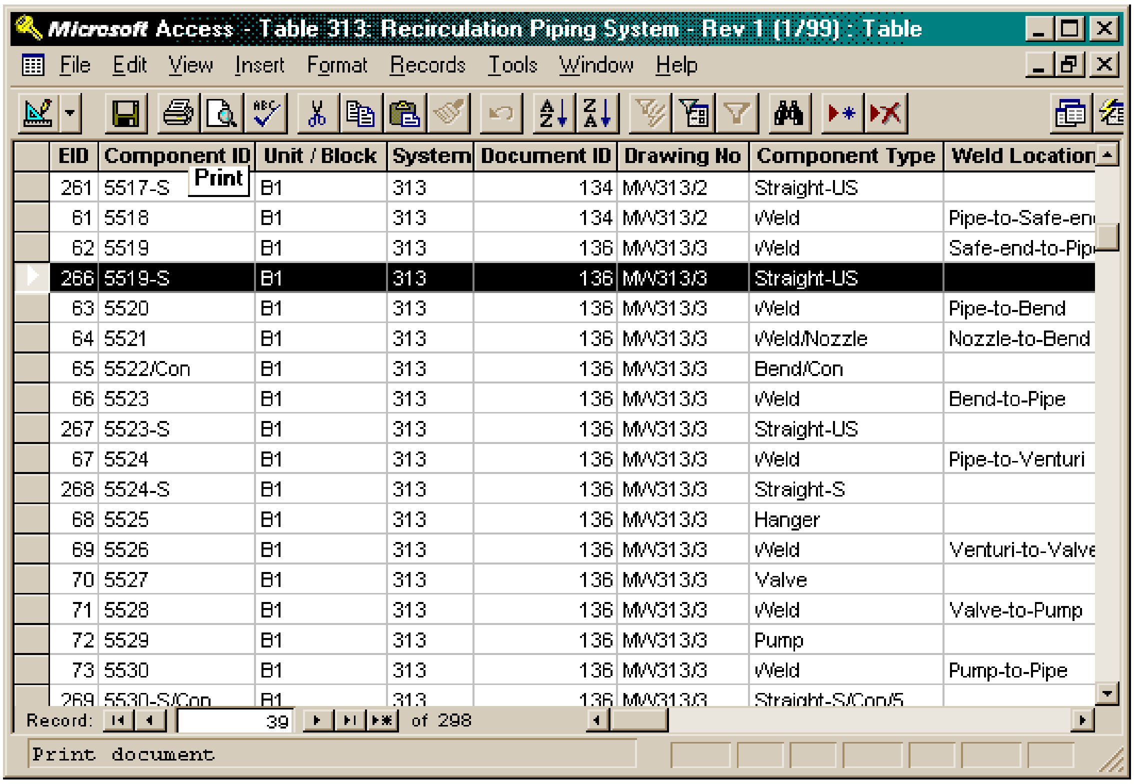


Figure 4-2: Overview of the PSA_VER2 Access Database.

[9] Each piping component has a unique alpha-numeric component ID. An archived, marked-up set of isometric drawings is kept on-site. Any changes to the isometric drawings will be included in PSA_VER2.mdb.

[10] This data field relates to SKI-PIPE; see Chapter 3 and Figures 3-1 through 3-4 for details.

As shown in Figure 4-2, each RCPB piping system is represented by its own Access-table; for a list of database field definitions, see Appendix D. An example of a table in the database is given in Figure 4-3 (see also Table on top of page 27). It shows a small portion of the design information for the Reactor Recirculation System piping as captured in the database. The second column from the left includes the unique piping component IDs. As an example, line 4 (EID #266) includes data on a pipe (component ID 5519-S). By scrolling to the right on the display screen, information about size, fabrication, material, dynamic effects, etc. becomes visible to the viewer/database-user.



EID	Component ID	Unit / Block	System	Document ID	Drawing No	Component Type	Weld Location
261	5517-S	B1	313	134	MW313/2	Straight-US	
61	5518	B1	313	134	MW313/2	Weld	Pipe-to-Safe-en
62	5519	B1	313	136	MW313/3	Weld	Safe-end-to-Pip
266	5519-S	B1	313	136	MW313/3	Straight-US	
63	5520	B1	313	136	MW313/3	Weld	Pipe-to-Bend
64	5521	B1	313	136	MW313/3	Weld/Nozzle	Nozzle-to-Bend
65	5522/Con	B1	313	136	MW313/3	Bend/Con	
66	5523	B1	313	136	MW313/3	Weld	Bend-to-Pipe
267	5523-S	B1	313	136	MW313/3	Straight-US	
67	5524	B1	313	136	MW313/3	Weld	Pipe-to-Venturi
268	5524-S	B1	313	136	MW313/3	Straight-S	
68	5525	B1	313	136	MW313/3	Hanger	
69	5526	B1	313	136	MW313/3	Weld	Venturi-to-Valve
70	5527	B1	313	136	MW313/3	Valve	
71	5528	B1	313	136	MW313/3	Weld	Valve-to-Pump
72	5529	B1	313	136	MW313/3	Pump	
73	5530	B1	313	136	MW313/3	Weld	Pump-to-Pipe
269	5530-S/Con	B1	313	136	MW313/3	Straight-S/Con/5	

Figure 4-3: Example of Database on Reliability Attributes & Component Populations.

The original piping installation includes piping components supplied by German and Swedish vendors. For each component the database records the appropriate material designation, charge/heat-number, and for stainless steels the carbon content.

PSA_VER2 is part of the overall PSA documentation for Barsebäck-1. Therefore, the database is subject to the specific QA/QC requirements defined for the Barsebäck-1 PSA project.

5. Weld Leak & Rupture Frequency Due to IGSCC

Chapter 5 develops the baseline weld leak and rupture frequencies applicable to medium-and large-diameter piping susceptible to IGSCC in the BWR operating environment. The chapter covers applicable service data, groupings of service data, and data interpretations.

5.1 Reviewing the Service Data

Figures 5-1 through 5-3 summarize the data in SKI-PIPE on weld failures due to IGSCC. Figures 5-1 shows the overall, BWR-specific data. It differentiates the data according to mode of failure at the time of discovery (i.e., crack, pinhole leak and at-power leak). Figures 5-2 and 5-3 show the number of weld failures as a function of years of commercial operation and pipe diameter in two system groups: 1) Reactor Recirculation System (RCS) including bypass loops and RPV head cooling lines (ABB-Atom units); and 2) Emergency Core Cooling, Reactor Water Cleanup and Residual Heat Removal Systems (SIR).

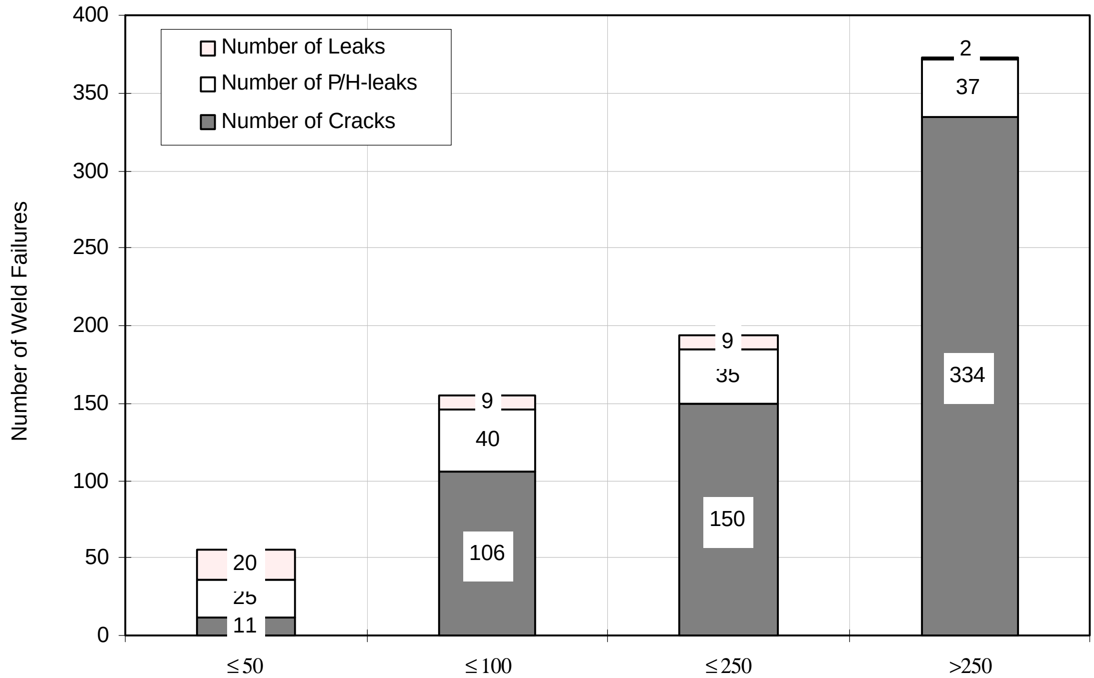


Figure 5-1: Weld Failures Due to IGSCC in BWR Units Worldwide (1970-1998).

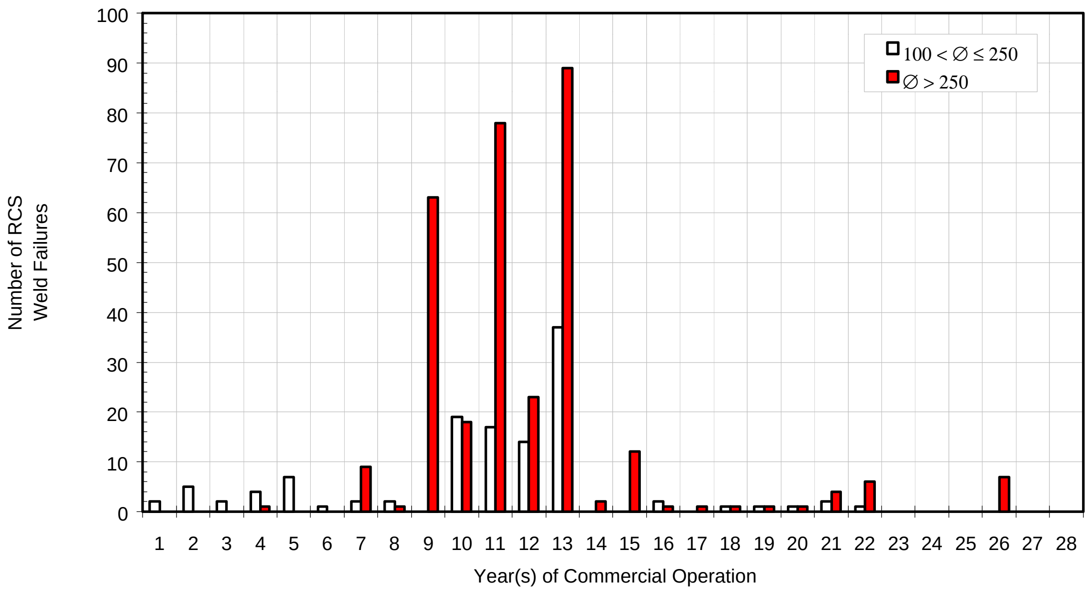


Figure 5-2: Weld Failures in RCS Piping in BWR Units Worldwide (1970-1998).

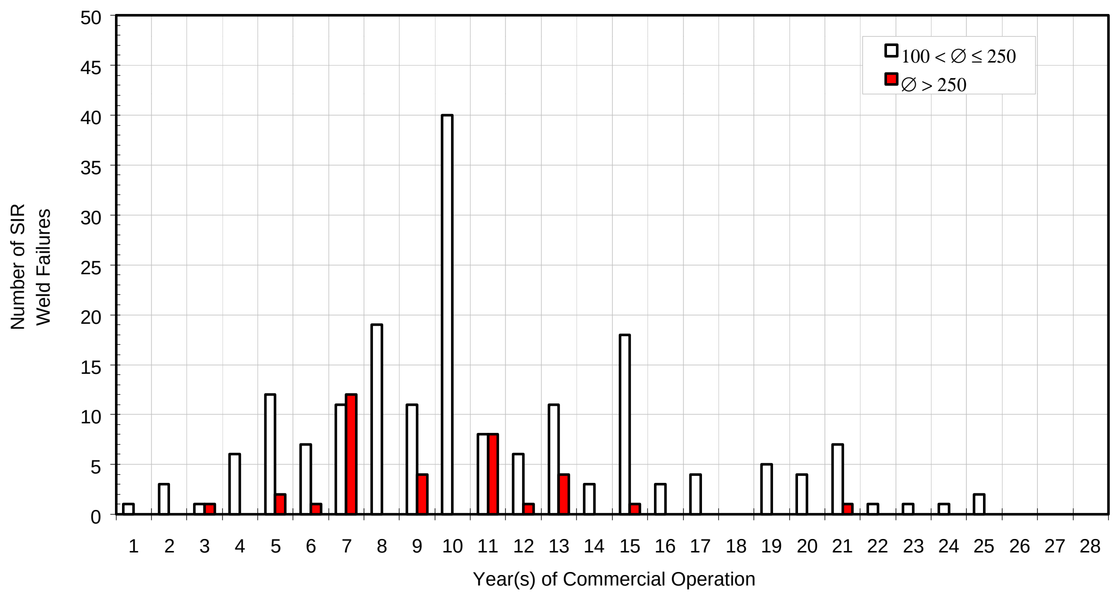


Figure 5-3: Weld Failures in SIR Piping in BWR Units Worldwide.

For welds in RCS-piping, Figure 5-2 shows a very sharp decline in the number of IGSCC occurrences. The large number of events during the 9th through 13th year of commercial operation mainly reflects industry response in the United States to the U.S. Nuclear Regulatory Commission's Inspection and Enforcement Bulletins 82-03[11] (dated October 14, 1982) and 83-02[12] (dated March 4, 1983). These Bulletins were issued in response to extensive cracking found at Browns Ferry-2, Monticello, Nine Mile Point-1

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[11] IEB 82-03: *Stress corrosion Cracking in Thick-Wall, Large Diameter, Stainless Steel Recirculation System Piping at BWR Plants*, U.S. Nuclear Regulatory Commission, Washington (DC).
[12] IEB 83-02: *Stress Corrosion Cracking in Large-Diameter Stainless Steel Recirculation System Piping at BWR Plants*, U.S. Nuclear Regulatory Commission, Washington (DC).

and other BWR units and involved weld heat affected zones in recirculation distribution headers and jet pump risers. Further, the Bulletins requested that augmented in-service inspections be performed in all plants. At approximately the same time, extensive RCS weld cracking was reported at Mühleberg (Switzerland), Santa Maria de Garona (Spain) and several German BWR units. In January 1999, Commonwealth Edison in the USA reported IGSCC flaws in the large-diameter Reactor Recirculation System at Quad Cites-1 (ComEd, 1999). Those recent data are included in Figure 5-2.

For system group 'SIR', most of the failure data apply to medium-diameter piping (up to DN250). According to Figure 5-3, up to the 10th to 15th year of commercial operation, the IGSCC occurrence rate in SIR-piping is lower than in RCS-piping. The failure trend for the two system groups is similar beyond the 15th year of operation. The low IGSCC occurrence rate is attributed to corrective actions such as HWC, more robust piping material, effectiveness of weld overlay repairs in combination with stress improvement treatments, etc.

According to above weld failure trends, the service data were divided according to events within the first 14 years of operation versus events beyond the first 14 years of operation. The former group of event data is denoted as representing a 'literal prior'. After the first 14 years of operation at least one of several options for mitigating IGSCC had been implemented at all plants susceptible plants; e.g., HWC, weld stress remedies, new material.

The division of the data according to events within and after the first 14 years of operation, respectively, is significant for the following reasons: Most of the data on IGSCC are for U.S. plants. As documented in NUREG-1061 (U.S. NRC, 1984), the regulatory and industry initiatives to mitigate IGSCC had largely been implemented by the 14th year of operation. That is, events to the left of year 14 in Figures 5-2 and 5-3 occurred as the result of insufficient attention to water chemistry and residual stresses on inner surfaces of piping. Beyond year 14, BWR piping reliability has benefited from improved water chemistry and weld stress improvement[13]. Using these arguments, the 'literal prior' represents the service experience before implementing improved water chemistry and weld overlay techniques.

Tables 5-1 through 5-8 summarize IGSCC event data for medium- and large-diameter RCPB piping; grouped as RCS and SIR piping. Also, the data are differentiated according to US versus worldwide service data. The tables compare the worldwide failure data with the U.S. data. As discussed on page 9, the comparison highlights different regulatory domains. The comparison also reflects differences in corrective action strategies.

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[13] Techniques have been developed to ensure that the internal surfaces of welds exposed to reactor water are in compression. In the U.S., the external weld overlay technique is widely used.

Table 5-1: Weld Failures in RCS Piping (100 £ DN £ 250) - Worldwide BWR Data.

Time Period	Total Number of Events	Ratio Leak: Crack[14]	Number of Cracks	Number of P/H-Leaks	Number of Leaks
Within the First 14 Years of Operation	112	0.15	97	15	0
Beyond the First 14 Years of Operation	8	0.14	7	1	0

Table 5-2: Weld Failures in RCS Piping (100 £ DN £ 250) - US BWR Data.

Time Period	Total Number of Events	Ratio Leak: Crack	Number of Cracks	Number of P/H-Leaks	Number of Leaks
Within the First 14 Years of Operation	28	1	14	14	0
Beyond the First 14 Years of Operation	0	--	0	0	0

Table 5-3: Weld Failures in SIR Piping (100 £ DN £ 250) - Worldwide BWR Data.

Time Period	Total Number of Events	Ratio Leak: Crack	Number of Cracks	Number of P/H-Leaks	Number of Leaks
Within the First 14 Years of Operation	137	0.63	84	39	14
Beyond the First 14 Years of Operation	44	0.16	38	6	0

Table 5-4: Weld Failures in SIR Piping (100 £ DN £ 250) - US BWR Data.

Time Period	Total Number of Events	Ratio Leak: Crack	Number of Cracks	Number of P/H-Leaks	Number of Leaks
Within the First 14 Years of Operation	110	0.71	64	34	12
Beyond the First 14 Years of Operation	24	0.17	20	4	0

Table 5-5: Weld Failures in RCS Piping (> DN250) - Worldwide BWR Data.

Time Period	Total Number of Events	Ratio Leak: Crack	Number of Cracks	Number of P/H-Leaks	Number of Leaks
Within the First 14 Years of Operation	284	0.12	252	30	2
Beyond the First 14 Years of Operation	34	0.06	32	2	0

[14] In calculating the 'leak:crack' ratio, the leak-term encompasses pinhole leaks and at-power leaks.

Table 5-6: Weld Failures in RCS Piping (> DN250) - US BWR Data.

Time Period	Total Number of Events	Ratio Leak: Crack	Number of Cracks	Number of P/H-Leaks	Number of Leaks
Within the First 14 Years of Operation	258	0.14	227	30	1
Beyond the First 14 Years of Operation	24	0.09	22	2	0

Table 5-7: Weld Failures in SIR Piping (> DN250) - Worldwide BWR Data.

Time Period	Total Number of Events	Ratio Leak: Crack	Number of Cracks	Number of P/H-Leaks	Number of Leaks
Within the First 14 Years of Operation	33	0.14	29	4	0
Beyond the First 14 Years of Operation	2	1	1	1	0

Table 5-8: Weld Failures in SIR Piping (> DN250) - US BWR Data.

Time Period	Total Number of Events	Ratio Leak: Crack	Number of Cracks	Number of P/H-Leaks	Number of Leaks
Within the First 14 Years of Operation	25	0.19	21	4	0
Beyond the First 14 Years of Operation	2	1	1	1	0

5.2 Parameter Estimation

Calculating pipe rupture frequency entails determining the frequency of leaks and the conditional rupture probability for specific sets of attributes and influence factors. For the leak frequency, Bayesian statistics using gamma priors and noninformative priors gave the results as given by Tables 5-9 and 5-10.

The choice of prior distributions remains an area of technical debate and a source of statistical uncertainty. For details on different strategies for selecting distributions, see the texts by Kapur and Lamberson (1977) and Martz and Waller (1982). In Tables 5-9 and 5-10, the given frequency values are based on:

- A gamma prior corresponding to 0.5 was applied to weld crack data for medium-diameter- and large-diameter RCS and medium-diameter SIR piping. The posterior mean was calculated from $(\delta + r)/(\rho + T)$ where $\delta/\rho = 0.5$ represent parameters determining the shape of the gamma distribution[15].

[15] δ = shape parameter and ρ = scale parameter. That is, in this example the shape and scale parameter are chosen in such a way that $\delta/\rho = 0.5$ or $\delta = 1$ and $\rho = 2$.

- A gamma prior corresponding to 5E-02 was applied to weld crack data for large-diameter SIR piping. The posterior mean was calculated from $(\delta + r)/(\rho + T)$ where δ/ρ = 5.0E-02.
- A gamma prior corresponding to 4.0E-3 was applied to weld leak data for large-diameter RCS piping (δ = 0.5 and ρ = 126)[16].
- A gamma prior corresponding to 2.7E-2 was applied to weld leak data for medium-diameter SIR piping (δ = 2 and ρ = 74).
- For medium-diameter RCS piping and large-diameter SIR piping the statistical evidence is 0 'at-power leaks' during the entire observation interval. The respective leak frequency is derived from a noninformative prior, and according to Jeffrey's rule the posterior mean is (2r + 1)/2T (Martz and Waller, 1982).

Table 5-9: RCS Weld Crack & Leak Frequencies Due to IGSCC - Worldwide Data.

System Group	**Statistical Evidence (Tables 5-1 through 5-8)**	**δ / ρ**	**Posterior Data**	
			Crack Frequency[17]	**Leak Frequency**
RCS (100 ≤ DN ≤ 250) (Table 5-1)	Literal prior: 112 cracks in 504 years Literal posterior: 8 cracks in 1078 years	1 / 2	8.33E-3	
	Literal prior: 0 leaks in 504 years Literal posterior: 0 leaks in 1078 years	--		3.16E-4
RCS (> DN250) (Table 5-5)	Literal prior: 284 failures in 504 years Literal posterior: 34 failures in 1078 years	1 / 2	3.24E-2	
	Literal prior: 2 leaks in 504 years Literal posterior: 0 leaks in 1078 years	0.5/126		4.15E-4

Table 5-10: SIR Weld Crack & Leak Frequencies Due to IGSCC - Worldwide Data.

System Group	**Statistical Evidence (Tables 5-1 through 5-8)**	**δ / ρ**	**Posterior Data**	
			Crack Frequency	**Leak Frequency**
SIR (100 ≤ DN ≤ 250) (Table 5-3)	Literal prior: 137 failures in 522 years[18] Literal posterior: 44 failures in 1105 years	1 / 2	4.07E-2	
	Literal prior: 14 leaks in 522 years Literal posterior: 0 leaks in 1105 years	2 / 74		1.70E-3
SIR (> DN250) (Table 5-7)	Literal prior: 33 failures in 522 years Literal posterior: 2 failures in 1105 years	1 / 20	2.67E-3	
	Literal prior: 0 leaks in 522 years 'Literal' posterior: 0 leaks in 1105 years	--		3.07E-4

Without statistical evidence on ruptured medium- to large-diameter Class 1 and Class 2 piping, the conditional probability of rupture given failure built on a noninformative prior (Jeffrey's rule[19]). Its relative merit was illustrated through comparisons with results from Bayesian updates using different prior distributions.

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[16] The prior distribution parameters were selected to give more weight to the fact that no at-power leaks have occurred. Furthermore, the parameters were selected to reflect a belief that NDE methods become more effective as the pipe size (diameter and wall thickness) decreases.

[17] All frequencies have dimension (1/System-Group-Year). For BWR units with external recirculation loops, the total service experience covers circa 1582 reactor-years.

[18] The frequency of leak in SIR piping is based on the total BWR service experience (i.e., BWRs with and without external recirculation loops), which is approximately 1627 reactor-years as of 12/31/98.

[19] The mean value of the conditional rupture probability is calculated from (2r + 1)/(2n + 2), where r = number of ruptures and n equals the total number of demands (cracks + leaks + ruptures).

Figure 5-3 includes mean values associated with the updated noninformative prior. The four discrete data points in Figure 5-3 are joined by a line for display only. Each estimate has its own uncertainty distribution. According to Beliczey (1995), as a rule-of-thumb, in small diameter piping (< DN25) the $p_{R|F} \propto$ 1.0E-1 and in larger diameter piping $p_{R|F} \propto$ 2.5/DN. Table 5-11 summarizes the suggested weld rupture frequencies.

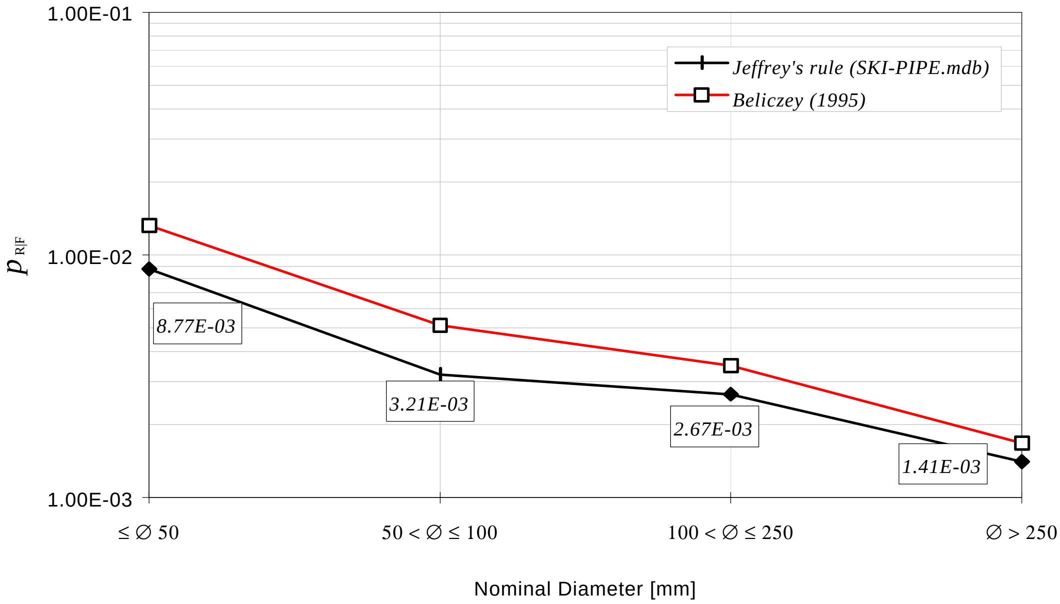

Figure 5-3: Conditional Probability of Rupture Given Weld Failure (Mean Values).

Table 5-11: Baseline Rupture Frequencies Due to IGSCC in Welds

System Group	Leak Frequency[20]	Conditional Probability of Rupture	Rupture Frequency[21] [1/Reactor-Year]
RCS (> DN250)	4.15E-4	1.41E-3	5.87E-7
RCS (100 ≤ DN ≤ 250)	3.16E-4	2.67E-3	8.44E-7
SIR (> DN250)	3.07E-4	1.41E-3	4.33E-7
SIR (100 ≤ DN ≤ 250)	1.70E-3	2.67E-3	4.54E-6

Up to this point, all frequencies are per plant-year and system-group. The given leak and rupture frequencies represent the total contributions from leaking and ruptured welds, respectively (see Chapter 3, pp 20-21). Conversion to per-weld frequencies would require access to weld population data for all plants in the database. That is, for each plant in the database, information equivalent to PSA_VER2 (Appendix B) would be required. Collecting such information is time consuming, and was beyond the defined work scope. Instead, the study utilized the information about location dependency of IGSCC-induced weld failures (Chapter 3, pp 20-21). As an example, on average 45.8% of the total rupture frequency of large-diameter RCS piping comes from a rupture in a weld between bend/elbow-to-pipe. Using the information in Figure 3-1 (page 20), Table 5-12 summarizes rupture frequencies for a selection of weld locations.

[20] Frequency of leak per RCS-Year (SIR-Year); mean values.
[21] Rupture frequency per RCS-Year (SIR-Year); mean values.

Table 5-12: Examples of Weld Rupture Frequencies for Different Weld Types.

System Group	Leak Frequency	Conditional Probability of Rupture	Rupture Frequency [1/Reactor-Year]
RCS (> DN250)			
Bend/elbow-to-pipe weld	1.90E-4	1.41E-3	1.41E-3
Pipe-to-pipe weld	5.52E-5	1.41E-3	1.41E-3
Pipe-to-valve weld	3.11E-5	1.41E-3	1.41E-3
RCS (100 ≤ DN ≤ 250)			
Bend/elbow-to-pipe weld	1.45E-4	2.67E-3	3.86E-7
Pipe-to-pipe weld	4.20E-5	2.67E-3	1.12E-7
Pipe-to-valve weld	2.37E-5	2.67E-3	6.33E-8
SIR (> 10 inch)			
Bend/elbow-to-pipe weld	1.41E-4	1.41E-3	1.98E-7
Pipe-to-pipe weld	5.53E-5	1.41E-3	7.80E-8
Pipe-to-valve weld	3.44E-5	1.41E-3	4.85E-8
SIR (4-10 inch)			
Bend/elbow-to-pipe weld	7.84E-4	2.67E-3	2.09E-6
Pipe-to-pipe weld	3.06E-4	2.67E-3	8.17E-7
Pipe-to-valve weld	1.90E-4	2.67E-3	5.08E-7

All values in Table 5-12 represent base-line values for BWR units that have implemented at least one of following corrective action strategies: 1) Stress improvement of welds using IHSI; 2) hydrogen water chemistry; or 3) IHSI and hydrogen water chemistry. As explained under ‘Study Conventions’ in Chapter 1, the term rupture implies a fractured pipe with a hole size big enough to cause a release of primary water > 5 kg/s (> 80 gpm). A leakage of this magnitude would result in the actuation of a primary system make-up system.

5.3 Parameter Estimation - A Different Perspective

The simple Bayesian (or ‘PSA-style’) approach to piping reliability analysis adequately accounts for trends and uncertainties in failure data. An alternate approach to estimating weld leak and rupture frequency would be to fit a statistical distribution to the failure data. However, it is important to recognize that modeling and data analysis go hand-in-hand. Changing the approach to data analysis could enable applications of more detailed reliability models. As an example, a parametric data analysis method would support applications of a parametric model of piping reliability such as proposed in the work by Fleming and Mikschl (1998).

An application of PSA-style piping reliability analysis highlights questions about validity. Would derived leak and rupture frequencies be affected by different data groupings or interpretations? A large database on piping failures enables different sensitivity studies. Also, such a database responds to questions about completeness. Parallel to the simple Bayesian approach, this study included extensive parametric data analyses to fully explore the database.

As a test for trends in the failure data, this study applied the hazard plotting technique (Nelson, 1972). In reliability analysis, the hazard function h(t) is the instantaneous failure rate. For a 2-parameter Weibull distribution, the hazard function is:

$$h(t) = (b / q)\times(t/q)^{\beta-1} \qquad (5.1)$$

$$Mean = q \times G(1 + 1/b) \qquad (5.2)$$

β= shape parameter

θ = characteristic life

Γ(•) = Gamma function

Queries in SKI-PIPE for specific attribute-influence sets generated summaries of failure times. Next, using the procedure in Appendix G, an Excel spreadsheet program generated the distribution parameters from hazard plots. Under the assumption of pre-existing cracks, Table 5-13 summarizes distribution parameters for piping susceptible to IGSCC and Figure 5-4 shows the corresponding hazard functions.

Table 5-13: Distribution Parameters for IGSCC-Susceptible Piping.

System Group	Pipe Size	Shape Parameter β	Scale parameter - θ [Years]
RCS	DN > 250	1.29	41.5
RCS	100 ≤ DN ≤ 250	1.14	66.4
SIR	150 ≤ DN ≤ 250	0.91	55.7
RWCU	150 ≤ DN ≤ 250	0.97	66.4
SIR	80 ≤ DN ≤ 100	0.89	38.7
CRD	DN50	1.75	40.7

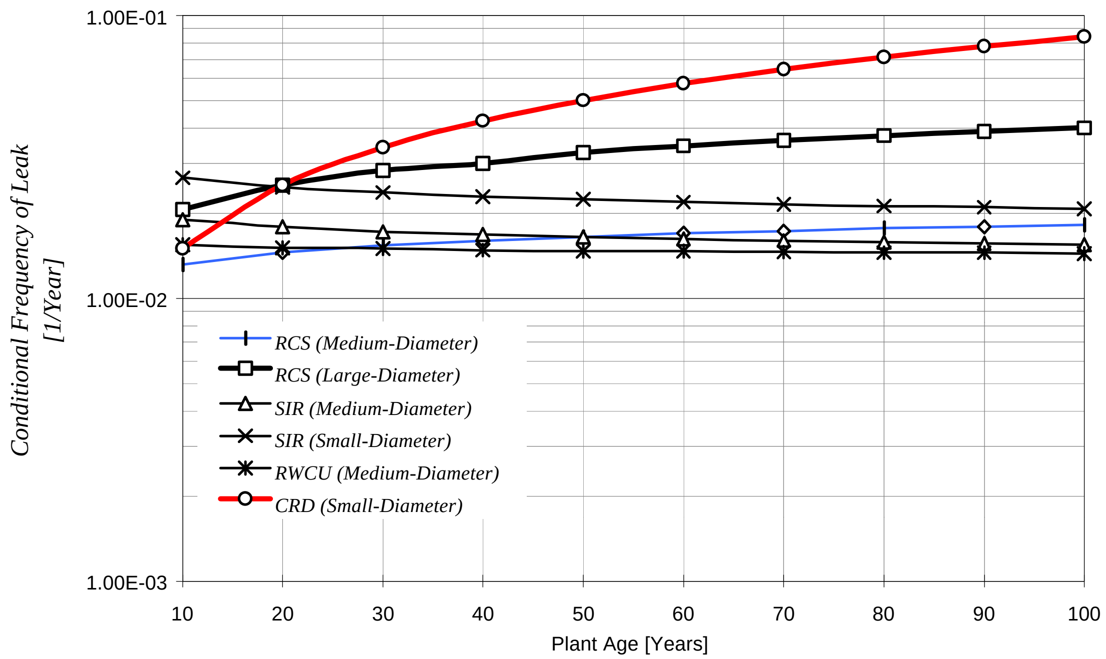

Figure 5-4: Hazard Functions for IGSCC-Susceptible Piping.

A shape factor near 1.0 is indicative of a constant failure rate. The shapes of respective hazard function in Figure 5-4 indicate where corrective actions have been effective. Also, the shapes of respective hazard function shape reflect data interpretation and data completeness issues.

Derived failure parameters that are based on relatively few data points have a high degree of variability. Using a spreadsheet program for parameter estimation makes it convenient to analyze the impact on the parameters by different data interpretation strategies. Table 5-14 is an interpretation of data on known P/H-leaks and at-power leaks in large-diameter RCS piping. Figure 5-5 shows the corresponding hazard plot.

Table 5-14: Through-wall Pipe Cracks in Large-Diameter Recirculation Piping.

Plant	**Event Date**	**SKI-PIPE EID**	**DN**	**Comment**
Santa Maria de Garona	2/13/80	2419 (IAEA/NEA-IRS #22.00)	550	The second known 'near-miss' DEGB; through-wall crack and leak from Inconel-600 safe-end.
Nine Mile Point-1	3/23/82	437 (LER 82-009)	700	The third known 'near-miss' DEGB; through-wall crack and leak from Inconel-600 safe-end. Similar event occurred at Duane Arnold in June 1978 (DN250 safe-end).
Hatch-1	11/6/82	2850 (LER 82-089)	550	During WOR, small leakage was observed on end-cap-to-manifold weld. Unit was in cold shutdown for refueling.
Dresden-2	2/25/83	1144 (LER 83-015)	750	Small leak was detected on a DN750 recirculation pump suction pipe weld.
Browns Ferry-2	2/21/85	1397	300/700	Leaking weld in junction between DN700 recirculation loop 'A' main header and DN300 jet pump riser.
Duane Arnold	3/10/85	2855 (LER 85-010)	250/700	Leaking weld between junction of main recirculation header and jet pump riser.
Brunswick-2	1/9/86	1723 (LER 86-002)	750	Visual and ultrasonic testing revealed TWC indications within the HAZ of 5 welds in the recirculation piping.

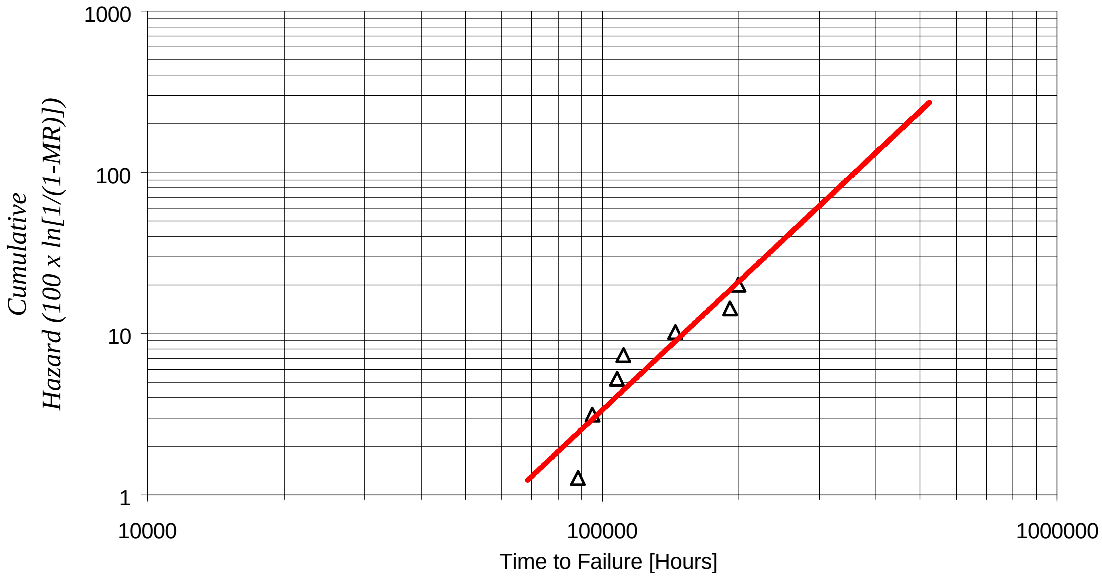

Figure 5-5: Hazard Plot of RCS Weld Failure Data in Table 5-14.

The same data analysis approach could be applied to crack data (e.g., welds in specific system and location). Combining a hazard function for cracked weld with a corresponding hazard function for leaking weld *given* crack would then produce a time-dependent estimate of the overall weld leak frequency. Preservation of analytical consistency depends on having a well-structured approach to data analysis. An effort to estimate piping reliability data parameters must correspond with the selected modeling approach. Development of parametric data suggests a parametric model of piping reliability; e.g., Markov model (Fleming and Mikschl, 1998). By contrast, in PSA-style analyses a simple Bayesian approach is adequate.

6. Rupture Frequency Due to Thermal Fatigue

Limited to thermal fatigue and with support of the database on piping failures, Chapter 6 develops the baseline weld leak and rupture frequency applicable to medium- and large-diameter RCPB piping in BWR operating environments. The chapter covers applicable service data, groupings of service data, and data interpretations.

6.1 Reviewing the Service Data

Figure 6-1 shows the number of piping failures caused by thermal fatigue as a function of years of commercial operation and pipe diameter. Affected system involve the main feedwater, reactor water cleanup and residual heat removal systems. Unlike IGSCC/TGSCC, the thermal fatigue failures mostly have affected the ABB-Atom plants in Finland and Sweden.

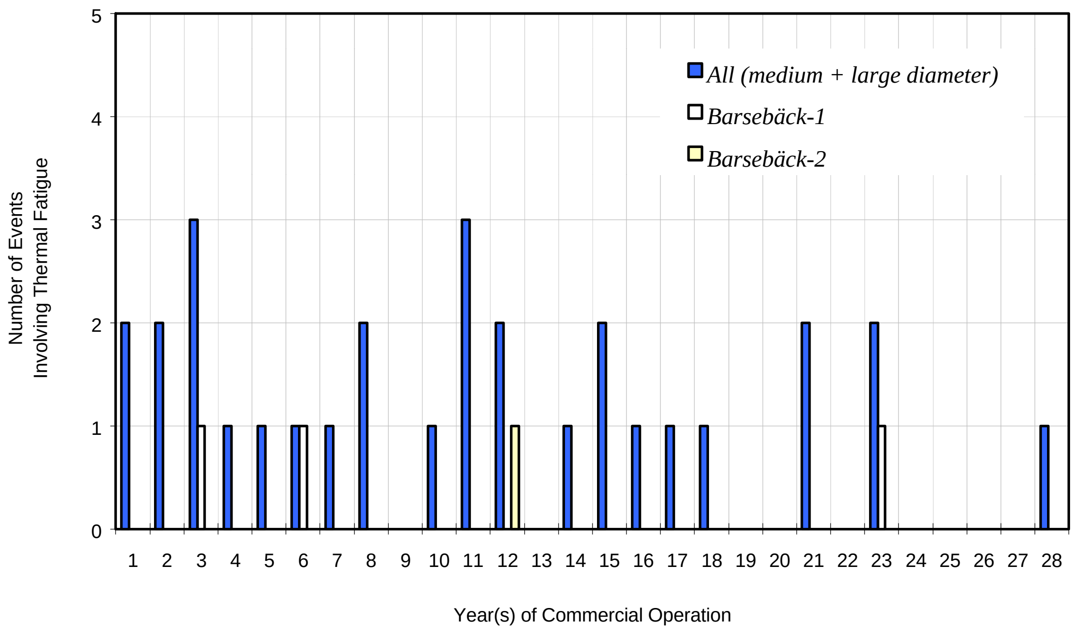

Figure 6-1: Thermal Fatigue in BWRs Worldwide.

Table 6-1 summarizes selected thermal fatigue events. The Finnish plant TVO-1 experienced a major event in 1979 (Holmberg and Pyy, 1994). A branch connection in the Reactor Water Cleanup System fractured after a short period of operation. The root cause analysis attributed the event to design error and human factors deficiencies. Approximately 5,000 kg of primary water was released through a 2.3 cm² hole onto the Reactor Building floor. Table 6-2 summarizes the thermal fatigue data for different pipe sizes. Table 6-3 summarizes the Nordic experience by system and pipe size.

Table 6-1: Service Experience Involving Thermal Fatigue in BWR Plants.

Plant	Date	SKI-PIPE Event ID	DN	Comment
Oskarshamn-2	5/26/78	341 (O2-RO-08/78)	150	Leak from branch connection between 321 and 331.
Barsebäck-1	6/20/78	2904	125	Significant cracking of a section of piping in 331.
TVO-I[22]	8/29/79	1619 (NEA-IRS #0014)	150	Fracture of tee in 331 resulting in substantial leak outside containment.
Barsebäck-1	8/1/80	375 (NEA-IRS #0010)	300	Cracking of tee (312/321); crack depth exceeded 26% TWC.
Oskarshamn-2	8/1/80	2429	250	Cracking of tee (312/321); crack depth approximately 25% TWC.
Ringhals-1	8/1/80	2428	250	Cracking of tee (312/321); crack depth approximately 20% TWC.
Dodewaard	1/1/82	28 (NEA-IRS #305.00)	450	Cracking of nozzles; up to 46% TWC. The plant has been de-commissioned.
Forsmark-2	5/31/83	2555 (F2-RO-009/83)	400	Cracking of tee connecting 321 and 331; max. of 78% TWC.
Forsmark-1	7/21/83	356 (F1-RO-13/83)	150	Cracking of branch connection between 321/331; crack depths unknown.
Forsmark-1	7/20/83	357 (RO-F1-17/83)	400	Crack in tee joint (312/321).
TVO-II	8/10/83	87 (INPO-SER 5-85)	150	Cracking of branch connection between Systems 321 and 331.
WNP-2	8/22/84	3057 (AEOD/S902)	150	Feedwater (FW) piping deflection and small leak from RWCU/FW branch connection.
Leibstadt	10/1/84	2411 (AEOD/S902)	25	During startup testing, leaks were observed at flanges of venturi flow meter in FW-line.
Forsmark-1	7/28/90	359 (F1-RO-20/90)	150	Through-wall crack leading to leak in mixer point between valve 331-V20 and heat exchanger 331-E5.
Forsmark-2	7/17/91	365 (F2-RO-11/91)	150	Cracking of branch connection in System 331; crack depths unknown.
Ringhals-1	8/1/92	350 (RO-R1-19/92)	300	Cracking of nozzles in 312; up to about 50% TWC
Oskarshamn-1[23]	6/18/93	335 (O1-RO-07/93)	150	Cracking of branch connection between 321 & 331 (just before System 312 distributor); max. crack depth of about 25% TWC.
Forsmark-1	11/16/94	361 (F1-RO-32/94)	150	Small leak from mixer point in 331.
Barsebäck-1	9/1/97	2918	200	Cracking of tee in 331; crack depths unknown.
KKI Isar-1	9/19/97	3046	150	Cracking of branch connection between 321/331; crack depths unknown.

[22] System operating procedure had been incorrectly translated from Swedish to Finnish causing the mis-positioning of a flow control valve.

[23] The metallurgical evaluation indicated no crack growth since 1981; cracks first identified around 1972/73. The initial crack growth was attributed to lack of operating experience among plant personnel.

Table 6-2: Thermal Fatigue Failures by Pipe Size.

Plant Type	Pipe Size	Total Number of Events	Crack	P/H-Leak + Leak	Rupture
ABB-Atom[24]	100 ≤ DN ≤ 250	22	12	10	0
ABB-Atom	DN > 250	8	6	2	0
BWR	100 ≤ DN ≤ 250	24	13	11	0
BWR	DN > 250	9	7	2	0
PWR	100 ≤ DN ≤ 250	18	4	14	0
PWR	DN > 250	8	0	8	0
	Sum[25]:	59	24	35	0

Table 6-3: Thermal Fatigue in Nordic Plants by System and Pipe Size.

System	Pipe Size	Total Number of Events	Crack	P/H-Leak + Leak	Rupture
AFWS	100 ≤ DN ≤ 250	22	1	0	0
ECCS			0	2	0
FWS			3	0	0
RHRS			2	1	0
RWCU			6	5	0
Other			0	2	0
FWS	DN > 250	8	4	1	0
MS			0	1	0
RHRS			1	0	0
RWCU			1	0	0
	Sum:	30	18	12	0

6.2 Thermal Fatigue Issue Summary

Thermal fatigue occurs where a piping component is exposed to thermal fluctuations caused by the mixing of process media at different temperatures. Assuming sufficient (or excessive) thermal cycling, the thermal fatigue cracks develop in the walls of austenitic stainless steel piping. Ferritic materials are considered immune to thermal fatigue. The underlying physical phenomena of thermal fatigue is termed thermal stratification. It is a condition in which two streams of fluid of different temperatures flow in separate layers without appreciable mixing. When the flow rate is low, turbulence is low and the potential for mixing is minimal. The lighter hot fluid stays above the heavier cold fluid.

The potential for stratification increases as the temperature difference between the hot and cold fluids increases. Increasing fluid temperature difference increases the effect of density variation and the buoyancy force. Fluid velocity has a negative effect on the potential for thermal stratification. As the fluid velocity increases, the potential for turbulent flow and mixing increases. The 'Richardson number' (R_i) is a measure of these effects (Su, 1990); this number is used as a screening criterion for addressing the

[24] Total of 217 reactor-years of operating experience with ABB-Atom plants as of 12/98.
[25] Less data in the rows for ABB-Atom plants.

potential for thermal stratification. The thermal stratification phenomena are classified as:

- Global thermal stratification;
- Cyclic thermal stratification;
- Thermal striping.

Consequences of global thermal stratification include: a) macroscopic movements of the piping and resulting in hanger damage; b) generation of stress in the piping, which might not have been considered in the design of the piping system; and c) low cycle fatigue. Consequences of cyclic thermal stratification and thermal striping are pipe cracks from high cycle fatigue.

Most of the service experience involving thermal fatigue damage due to global thermal stratification is concerned with the PWR operational environment. As an example, the main degradation mechanisms affecting the PWR feedwater lines are thermal fatigue and flow-accelerated corrosion (FAC); *c.f.* Shah, Ware and Porter (1997). By contrast, cyclic thermal stratification is the dominant cause of thermal fatigue damage in the BWR operating environment.

6.3 Parameter Estimation

Compared to IGSCC, the thermal fatigue data are scarce. Additionally, there is no clear trend in thermal fatigue occurrences (Figure 6-1). Operational practices (e.g., intermittent versus continuous RWCU flow) strongly influence the susceptibility to thermal fatigue. The database content could be indicative of: 1) Incompleteness; 2) Thermal fatigue being less significant than IGSCC; or 3) Inadequacy of ISI methods or ISI programs in detecting thermal fatigue degradations. On the other hand, as summarized on page 40, the data could be indicative of a thermal fatigue phenomena specific to ABB-Atom plants. Figure 6-2 shows hazard functions for data in Table 6-2.

Considering data for ABB-Atom plants and BWRs worldwide, the time-averaged failure frequencies are 3.4E-2/Year and 2.6E-2/Year, respectively. The hazard function representing BWRs worldwide does not include failures occurring within the first 4 years of commercial operation. Although the time-averaged values are quite close, the shapes of respective function differ significantly. The analysis of data on thermal fatigue raises the following questions:

- Mixing data for different BWR plant designs. Should the analysis be limited to data on the ABB-Atom experience? In view of design differences, could the pooling of ABB-Atom with GE service data be justified?
- Choice of prior distribution. Ideally, the prior distribution should reflect what is known about thermal fatigue. In view of the limited data, how should the prior best be defined?

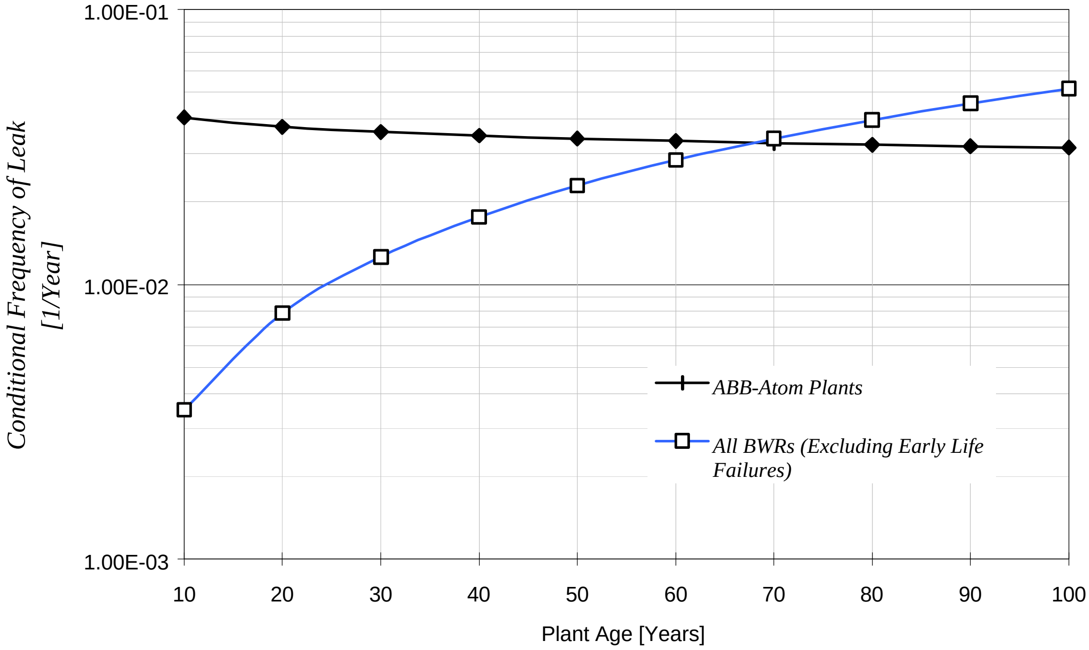

Figure 6-2: Hazard Functions for Piping Susceptible to Thermal Fatigue.

This study used non-informative priors (Jeffrey's rule) to estimate the frequency of leak and rupture. Table 6-4 summarizes the proposed leak and rupture frequencies for branch connections in Auxiliary Feedwater, Main Feedwater and Residual Heat Removal systems (system group 'F+R'). It uses the data in Table 6-3 while acknowledging that the Reactor Water Cleanup System (System 331) is not part of the RCPB in ABB-Atom plants.

Table 6-4: Baseline Leak and Rupture Frequency Due to Thermal Fatigue.

System Group	Pipe Size	Evidence	Leak Frequency [1/Plant-Year]	Rupture Frequency [1/Plant-Year]
F+R ABB-Atom plants	$100 \leq DN \leq 250$	• 1 leak in 217 reactor-years • 0 ruptures	6.91E-3	-- 1.11E-4[26]
F+R BWR plants worldwide	$100 \leq DN \leq 250$	• 1 leak in 1582 reactor-years • 0 ruptures	9.48E-4	7.90E-6
F+R ABB-Atom plants	DN > 250	• 1 leak in 217 reactor years • 0 ruptures	6.91E-3	-- 1.11E-4
F+R BWR plants worldwide	DN > 250	• 1 leak in 1582 reactor years • 0 ruptures	9.48E-4	-- 7.90E-6

[26] The conditional rupture probability uses pooled data for medium- and large-diameter piping; 0 ruptures, 30 failures (59 failures worldwide) per Table 6-3. Using a noninformative prior, according to Jeffrey's rule, the posterior mean is (2R + 1)/(2F +2) = 1.6E-2. Using an informative prior; e.g., beta with mean = 0.1 gives posterior mean of 2.5E-2.

7. Barsebäck-1 Piping Reliability Database

Chapter 7 summarizes the conversion of baseline rupture frequencies to plant-specific rupture frequencies input to PSA_VER2, the model of the RCPB piping reliability (Chapter 4 and Appendix B). In PSA_VER2 each of the ten RCPB piping systems is represented by a table of piping system components (bends, pipes, tees, welds) and associated reliability attributes and influence factors.

7.1 Base Values versus Adjusted Values

For BWRs, Chapters 5 and 6 developed base values for the frequency of leak and rupture in piping susceptible to IGSCC, TGSCC and thermal fatigue. These frequencies are 'base values' in that they were developed at the system level, not accounting for plant-specific piping designs such as layouts, or material compositions. The conversion of a system-level base frequency to a component-level, plant-specific frequency is not trivial. Adjusting a baseline frequency involves:

- Apportioning of baseline frequency for system group to the systems making up the group. As an example, SIR includes service data on ECCS, RHRS and RWCU Systems; Table 7-1.
- Collecting piping component population data for the BWR units represented in the piping failure database. Ideally, the population data should be of the same quality as for the reference plant in this study. Since such data are not readily available, an approximation of the population data should be established.
- Accounting for the effects on piping reliability by different material compositions (e.g., stainless steels of extra-low carbon content stainless steel versus high carbon content).
- Accounting for the effects on piping reliability by different water chemistry (e.g., hydrogen water chemistry versus normal water chemistry).

Table 7-1: Barsebäck-1 Plant System Names and the System Groups.

System Group	Rupture Frequency [1/Reactor-Year]	Plant Name (ID)	Rupture Frequency [1/System-Year]-	Comment
RCS (> DN250)	5.87E-7	313	5.87E-7	
RCS (100≤ DN ≤ 250)	8.44E-7	313 (bypass); 326	8.44E-7	
SIR (> DN250)	4.33E-7	RHR - 321 ECC - 323	1.44E-7 1.44E-7	Equal contribution from 3 SIR.
SIR (100≤ DN ≤ 250)	3.66E-6	RHR - 321 ECC - 323	5.89E-7 1.07E-6	16.1% contr.[27] 29.3% contr.

[27] Based on SKI-PIPE.mdb.

Each step to adjust an estimated frequency from the system-to the component-level is a source of uncertainty, however. As an example, for a given system, pipe size group and design vintage the weld population could differ by a factor 2 among plants. As an added complexity, the piping failure database includes several cases where an original piping design was modified to reduce the weld count, thus minimizing the IGSCC susceptibility. Using pre-formed piping sections, eliminating elbows, etc., could reduce an original weld count by at least a factor of 2, and in some cases by as much as a factor of 10. It would be a formidable undertaking to develop detailed population data.

One solution to this aspect of failure frequency estimation would be to apply DPD-formalism[28] to characterize the uncertainty in population data. This was done in the work by Fleming et al (1998). In the present application with Barsebäck-1 as reference plant, the problem of not having a complete set of population data was addressed using data on the location dependency of weld failures (Chapter 3, page 20 and Chapter 5, page 34).

The location dependency data are yet another source of uncertainty, however. Do these data accurately portray the IGSCC-sensitivity of different weld locations? Do the data accurately account for the number of different locations in the piping systems covered by the database? Expanding the SKI-PIPE database content on IGSCC-induced weld failures yielded an additional 450 records; an expansion from initially 350 events to 800 events. Figure 7-1 compares the location dependency data before and after the database scope expansion.

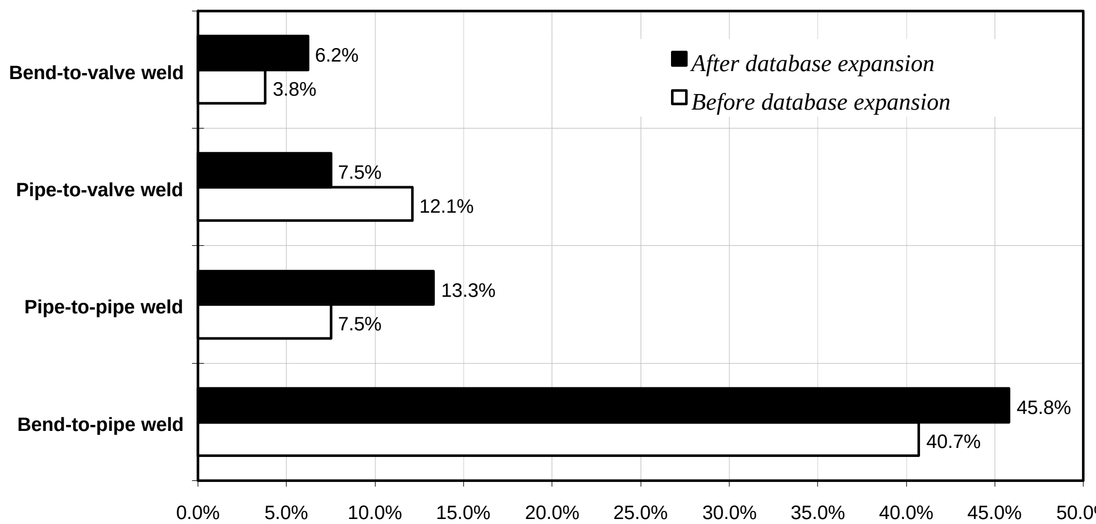

Figure 7-1: Information on IGSCC-Sensitivity of Welds.

Figure 7-1 conveys two insights about weld failures due to IGSCC. First, a weld between a bend and pipe is more prone to cracking than other locations in a piping system. Second, the insights about location dependency changes with the completeness of the database.

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[28] DPD = discrete-probability distribution

Within the given work scope limitations, the study on location dependency was not taken beyond comparing the before-and-after insights. The location dependency data after database expansion were taken to accurately portray IGSCC-susceptibility of different weld locations. The baseline rupture frequency was apportioned across the different locations, without accounting for uncertainty in the dependency information.

Different material compositions and primary water chemistry strategies significantly impact weld reliability. Many studies have evaluated the effects on IGSCC-susceptibility by different carbon content stainless steels. Lowering the carbon content improves the IGSCC-resistance. Extra-low carbon content ('nuclear grade', NG) stainless steels[29] are considered immune to IGSCC. The piping failure database includes data on plants where piping systems have been partially or completely replaced using NG-steels. There are no reports on IGSCC in NG-steels in the database. According to studies by the Electric Power Research Institute, the use of NG-steels (ELC) should increase the time to failure by a factor of 20 or more (Danko, 1983). This study applied a factor of 10 improvement in failure times by low carbon (LC) content stainless steels

Hydrogen water chemistry (HWC) has proven effective in reducing or eliminating IGSCC. The SKI-PIPE database identifies BWR units with and without HWC. According to the database, approximately 50% of the BWR units operate with HWC. The water chemistry strategy differs among these plants, however. Some units have been operating with 100% HWC for many years, while others have operated with intermittent HWC. Plant-specific factors such as the operability of demineralizers and the integrity of condenser tubes affect the HWC-effectiveness.

The piping failure database summaries in Chapter 5 reflect industry programs to minimize IGSCC. Implicitly, the baseline rupture frequencies account for improved water chemistry, weld overlay repairs, different materials, etc. The application study did not pursue formal statistical analysis of the correlation between IGSCC and different influence factors.

7.2 Barsebäck-1 Piping Reliability Database

Converting the IGSCC baseline frequencies to Barsebäck-1 specific frequencies accounted for influences by material composition and weld location. In addition to analyzing data on weld failures, the data analysis also considered cracks and leaks in base metal of cold worked piping. According to SKI-PIPE, for medium-diameter cold worked stainless steel piping, approximately 1-of-6 reported events involved damage in base metal of residual heat removal piping. Table 7-1 shows plant-specific (i.e., adjusted) component rupture frequencies applicable to base and weld metal of the Residual Heat Removal System (System 321). The study developed similar data tabulations for all Barsebäck-1 RCPB piping systems.

[29] For example, stainless steels of type SS 2353EL (Sweden) or AISI-316NG (USA); ≤ 0.02% carbon.

Table 7-2: Excerpt from Barsebäck-1 Piping Component Reliability Database.

Component	$F_{R\text{-TOTAL}}$ (Base-line)	Influence by Location - Weld	Influence by Material	Population	F_R - Mean [1/Comp.-Year]
Weld Weld - DN200	5.89E-07[30]	Bend-to-pipe	--	12	7.00E-09[31]
			LC	8	7.00E-10
SYSTEM 321		Pipe-to-penetr.		--	--
(RHR System)			LC	2	8.71E-10
		Pipe-to-pipe	--	1	1.25E-08
		Pipe-to-reducer	--	3	5.89E-09
			LC	1	5.89E-10
		Pipe-to-valve	--	2	1.77E-08
			LC	2	1.77E-09
		Tee-to-pipe	--	2	9.42E-09
Weld Weld - DN250	5.89E-07	Bend-to-pipe	--	2	7.00E-09
			LC	5	7.00E-10
SYSTEM 321			ELC/NG	1	3.50E-10
(RHR System)		Bend-to-valve	--	2	1.18E-08
		Pipe-to-penetr.	--	--	--
			LC	1	1.25E-09
		Pipe-to-valve	--	2	1.77E-08
		Tee-to-pipe	--	2	9.42E-09
			LC	1	9.42E-10

7.3 Weld 'Location-Dependency' - Limitations & Pitfalls

For a given system, the proposed 'location-dependency' concept (Chapter 3) reflects populations of different welds *and* different susceptibilities to IGSCC. Uncritical use of the concept could lead to unwarranted reduction in estimated rupture frequency, however. Data specialization as demonstrated in this chapter should always include a check for reasonableness; see Table 7-3. Unless a piping system has been subjected to repair/replacements and/or modification, the baseline and adjusted rupture frequencies should be the same. In the given example, the data specialization was model-driven.

Table 7-3: Comparison of Baseline and Specialized Pipe Rupture Frequency.

Baseline Rupture Frequency in RHR Piping [1/Reactor-Year]	Barsebäck-1 Rupture Frequency in RHR Piping [1/Reactor-Year]	Comments
5.89E-7	4.59E-07	The data specialization resulted in a rupture frequency reduction by about 22%. The reduction is attributed to influence by low- and extra-low carbon content stainless steels.

[30] From Table 7-1, page 45.
[31] Calculated from: 5.89E-07 × 0.44, where 0.44 is taken from Figure 3-3. Total of 37 bend-to-pipe welds in medium-diameter RHR-piping (Table B-5, page 68); (5.89E-07 × 0.44)/37 = 7.00E-09.

8. LOCA Frequencies in Barsebäck-1 PSA

The application study resulted in a set of new LOCA initiating event frequencies for input to the Barsebäck-1 PSA. As such, the estimation of new LOCA frequencies was subservient to the overall R&D program objective, however. The ultimate objective was to demonstrate an application of a piping failure database. Chapter 8 presents results, insights and implications relative to future PSA applications.

8.1 LOCA Categories in Barsebäck-1 PSA

The definition of LOCA categories (Figure 8-1) built on an integrated evaluation of location-dependent pipe ruptures and dynamic effects of pipe whips and steam/water jets on piping and pipe insulation adjacent to the rupture. This approach to defining LOCA considered structural failures in any of the circa 4,000 RCPB-piping components. An evaluation of pipe break consequences lead to identification of dynamic effects impacting plant and operator response to LOCA.

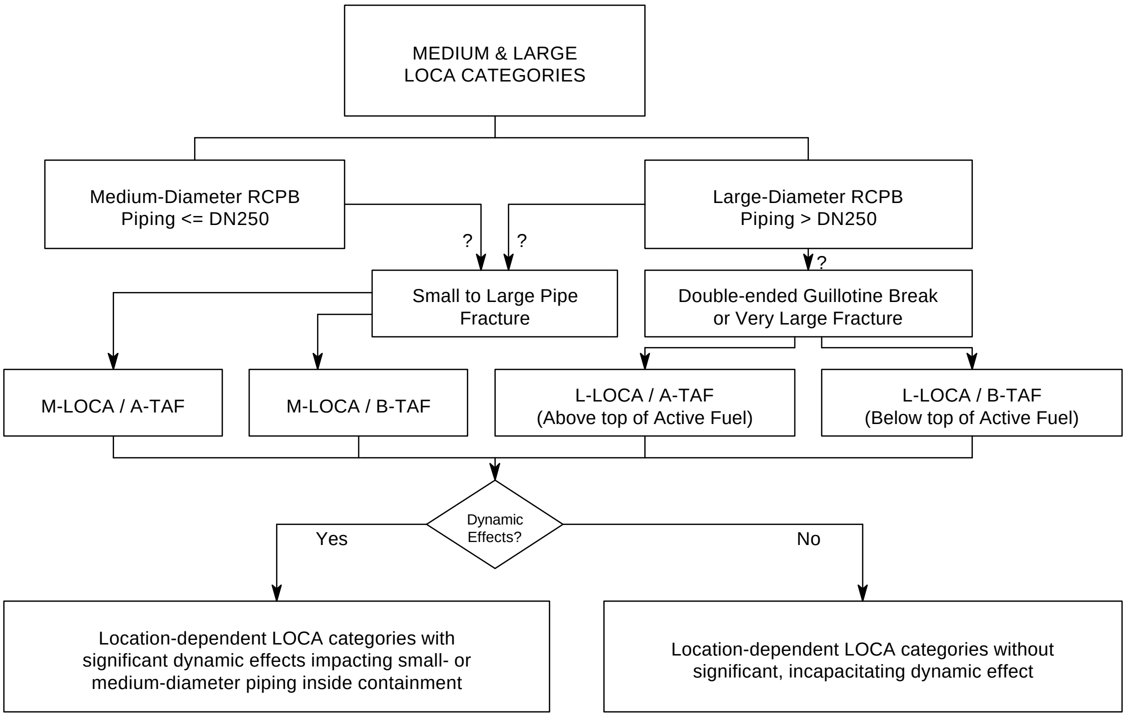

Figure 8-1: Medium & Large LOCA Categories in Barsebäck-1 PSA.

As a simplifying assumption, a large- or medium LOCA was classified on the basis of pipe diameter at the location of failure. A rupture in large-diameter piping causes the equivalent of a large LOCA. A large leak in a large-diameter piping causes the equivalent of a medium- to small-LOCA.

8.2 New LOCA Frequencies

Table 8-1 summarizes the new LOCA frequencies. Table 8-2 is a comparison with the current version of Barsebäck-1 PSA (December 1998) using old data (based on WASH-1400), and NUREG/CR-5750 (Poloski et al, 1999), respectively.

Table 8-1: LOCA Frequencies in Barsebäck-1 PSA - Proposed Values.

Category	**Definition / Boundary Condition(s)**	**Frequency - Mean [1/year]**
AT.1	DEGB of System 323, Loop 1 (DN200-250)	1.25E-07
AT.1.1	DEGB of System 323, Loop 1 (DN250). Consequential rupture in System 322.	1.49E-08
AT.2	DEGB of System 323, Loop 2 (DN250)	1.64E-07
AT.3	DEGB of System 312 (DN200-350), or System 321 (DN200), or System 327 (DN250).	8.48E-06
AT.3.1	DEGB in the portion of System 312 (DN250) that could cause consequential rupture in System 322.	1.12E-07
AT.4	DEGB in System 311/314 (DN125-500)	3.74E-07
AT.4.1	DEGB of System 311 (DN500). Consequential break in System 313/321.	5.18E-08
AT.4.2	DEGB of System 311 (DN500). Consequential break in System 312/321.	8.61E-09
AT.4.3	DEGB of System 311 (DN500). Consequential pipe break in System 723 (CCW).	1.39E-09
AT.4.4	DEGB of System 311 (DN500). Consequential breaks in RPV level indication system and System 723.	1.27E-08
AT.4.5	DEGB of System 311 (DN500). Consequential break in RPV level indication system.	1.82E-08
	Large LOCA (A-TAF); total:	9.36E-06
S1T.1	Rupture in System 323 Loop 1 (DN100), or fracture in piping corresponding to AT.1	2.77E-07
S1T.2	Rupture in System 323 Loop 2 (DN100), or fracture in piping corresponding to AT.2	3.88E-07
S1T.3	DEGB of System 351, or fracture in piping corresponding to AT.3	1.86E-06
S1T.4	DEGB of System 311/314 (DN50-80), System 326, or fracture in piping corresponding to AT.4 and AT.4.1-5	2.36E-07
	Medium LOCA (A-TAF); total:	2.76E-06
AB.1	DEGB of System 313 (DN200-600) piping or DEGB of System 321 (DN250) connecting to System 313.	1.83E-06
	Large LOCA (B-TAF); total:	1.83E-06
S1B.1	Rupture of System 313 (DN100-150) or System 321 (DN150) or System 326 connecting to System 313 or fracture in piping corresponding to AB.1	3.99E-07
S1B.1.1	DEGB of System 326 piping and consequential break in RPV level indication piping/tubing.	1.96E-07
S1B.2	DEGB of System 354 piping (Sum of 17 ‘scram groups’)	4.14E-05
	Medium LOCA (B-TAF); total:	4.20E-05

Table 8-2: Comparison of LOCA Frequencies (Mean Values).

Category	This Project [1/year]	Barsebäck-1 PSA (BKAB, 1998) [32] [1/year]	NUREG/CR-5750 (Poloski et al, 1999) [1/year]
Large LOCA (A-TAF); total:	9.36E-06	1.39E-04	--
Medium LOCA (A-TAF); total:	2.76E-06	4.56E-04	--
Large LOCA (B-TAF); total:	1.83E-06	1.08E-04	--
Medium LOCA (B-TAF); total:	4.20E-05	7.63E-04	--
TOTAL LARGE LOCA:	1.1E-05	2.5E-04	3.0E-05
TOTAL MEDIUM LOCA:	4.5E-05	1.2E-03	4.0E-05

8.3 Study Insights

The above results relate to specific degradation and failure mechanisms and vulnerabilities specific to RCPB piping systems in Barsebäck-1. As examples, Figure 8-1 through 8-3 display contributions to large and medium LOCA from different pipe break locations above top-of-active-fuel (TAF).

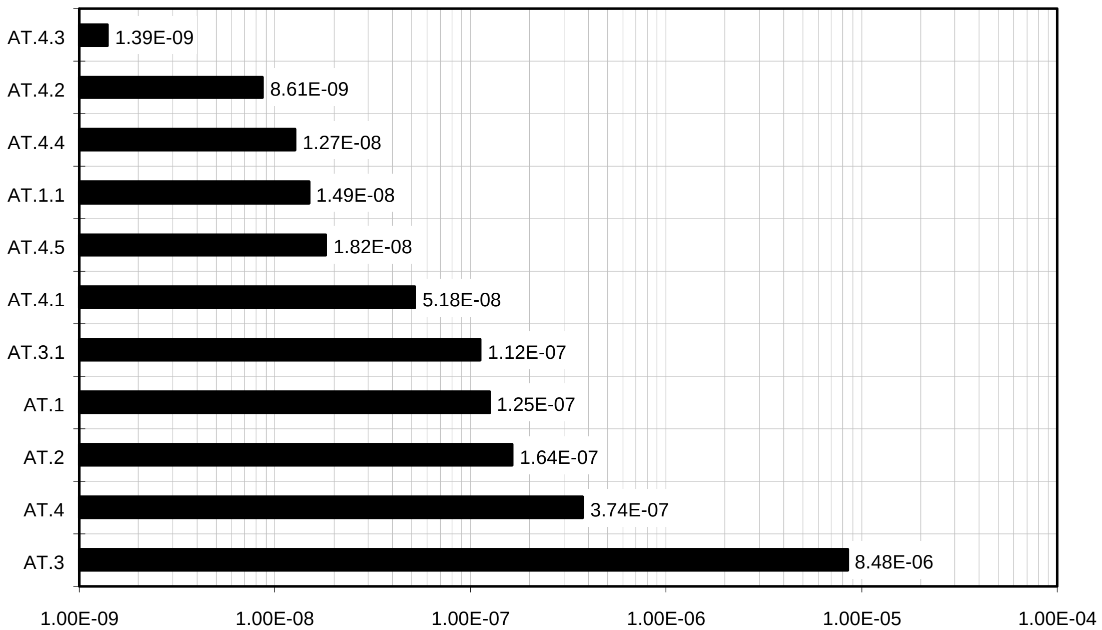

Figure 8-1: Contributors to Large LOCA - Pipe Break Above TAF.

[32] Based on PSA_VER1 (see Chapter 4) and data from WASH-1400.

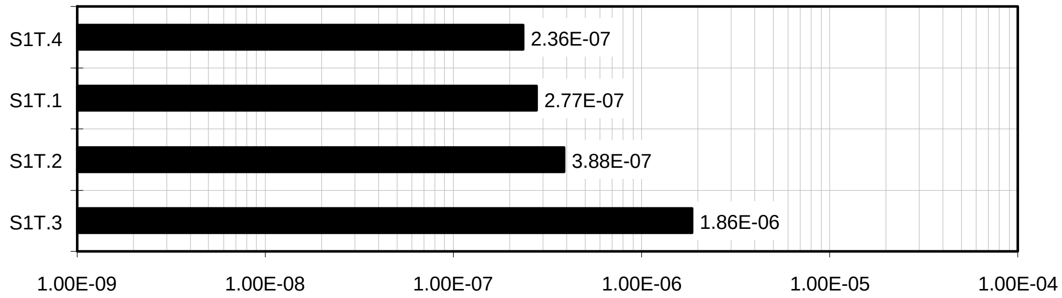

Figure 8-2: Contributors to Medium LOCA - Pipe Break Above TAF.

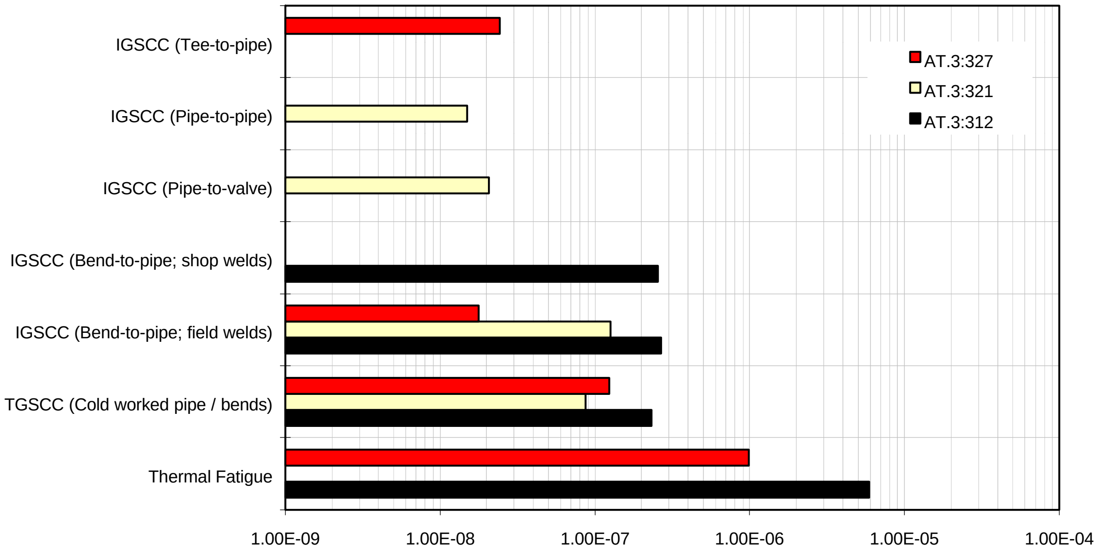

Figure 8-3: Contributors to Large LOCA by Degradation Mechanism.

Above figures represent a small excerpt of study insights. Each LOCA category frequency relates to specific piping components in PSA_VER2 (the LOCA model), which in turn relates to specific events as recorded in SKI-PIPE. The strength of this technical approach to modeling piping systems and analyzing piping failure data is its ability to perform integrated evaluations of degradation occurrences and their risk impact. The entire analysis process as described in preceding chapters is amenable to computer automation. That is, an integrated workstation, and using the Microsoft Access and Excel computer software programs, could be developed in a relatively short time.

9. Review of Technical Issues

Chapter 9 discusses five technical issues in piping reliability analysis: 1) Sensitivity and uncertainty analysis; 2) Justifications of detailed probabilistic modeling; 3) Integration of PSA methods and material sciences; 4) Direct estimation of pipe rupture frequency versus probabilistic fracture mechanics; and 5) Location-dependency of weld failures due to IGSCC.

9.1 Issues in Sensitivity & Uncertainty Analysis

Much of the failure data on RCPB piping reflects results of ISI programs and detection of flaws in base metal and weld metal. Therefore, the coverage and completeness of the piping failure database are directly proportional to the effectiveness (i.e., reliability) of ISI. An inherent source of data uncertainty stems from the evolving inspection methods. In recognition of human factors (Bauer, 1994; Enkvist, Edland and Svenson, 1999) and technical limitations (Doctor, Lemaitre and Crutzen, 1995), the reliability of ISI using non-destructive examination methods[33] has been the subject of significant R&D. All along, power plant organizations have made efforts to ensure high ISI reliability by relying on independent verifications of results. It is difficult to determine by how much the piping failure database under-estimates the absolute number of piping degradations due to the evolving ISI programs, however.

The given pipe rupture and LOCA frequencies are predicted values based on interpretation and statistical analysis of piping failure data. The rupture frequencies are sensitive to the completeness of service data and the intrinsic assumptions of the data analysis approach.

The results are specific to Barsebäck-1. Over the years, piping replacements have been made using low- and extra-low carbon content stainless steels to mitigate IGSCC. The reliability parameter estimation explicitly accounted for different grades of stainless steel in the RCPB-piping. Because of the unique piping design, the new LOCA frequencies are not directly applicable to other BWR units. Validation of LOCA frequencies for application to any other PSA project by referencing SKI Report 98:30 would be inadmissible. The estimation of pipe rupture frequencies is non-trivial, and the statistical analysis involved several steps:

- <u>Degradation evaluation</u> to determine the predominant degradation mechanisms. A qualitative evaluation of susceptibility to degradation mechanisms was done on the basis of service experience specific to Barsebäck-1 and industry-wide service experience as documented in SKI-PIPE.

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[33] For example, automated or manual ultrasonic testing, X-ray techniques, visual testing, eddy current tetsting.

- Organizing service data according to specific attributes and reliability influence factors. A critical step involved collecting data of sufficient quality, and evaluating the data against specific definitions of failure.

- Verification of data quality. The quality of failure data and parameter estimates was ensured by validation for completeness and accuracy. Full text event reports were reviewed so that 'illegal data' were removed from pooled data prior to parameter estimation.

- The parameter estimation involved PSA-style Bayesian updates of prior failure distributions as well as formal statistical analysis by fitting statistical distributions to the data. In the Bayesian analysis, the definition of prior distributions was influenced by the way the service data were organized and evaluated.

Determining accurate piping component exposure data proved difficult, tedious and uncertain. Except for the subject plant and selected systems in other plants, a detailed count of RCPB piping components in the range of plants of different design vintage in the SKI-PIPE was not pursued. For IGSCC-susceptible piping the database supported determination of the rate of cracking by weld location. This then was used to determine reliability parameters on the basis of expected failure location. Next, component counts specific to Barsebäck-1 were used to estimate the reliability parameters on a component basis.

An omission in the qualitative failure data evaluations would impact the predicted reliability parameters. Repeated database queries and data verification minimized the opportunity for omissions. Following the initial data analysis during April - September 1998, the piping failure database was expanded to determine the robustness of the original analysis insights. The impact on estimated rupture frequencies would have been considerable had the database expansion revealed new significant leaks or ruptures. As an example, for any given set of attribute-influence, the impact of a single rupture event would raise an original mean rupture frequency by at least a factor of 3.

A formal uncertainty analysis was outside the work scope. The application study was a 'proof-of-principles study.' The sources of statistical uncertainty are many, however. Each step to specialize service data impacts the uncertainty bounds on estimated parameters. Qualitative as well as quantitative information on the service experience supported the grouping of data according to attribute-influence sets. In a formal uncertainty analysis, each step to specialize should be evaluated. As outlined in Chapter 10, additional R&D should be pursued to more fully explore the entire failure database.

9.2 Justifications for Detailed Modeling

Pursuit of detailed modeling solely to demonstrate low or very low LOCA frequencies is an insufficient justification. Instead, the justifications behind the detailed modeling of RCPB piping extend far beyond the prediction of plant-specific LOCA frequencies. Two technical reasons should determine the need for detailed modeling.

First, an added realism to the event tree and fault tree models of the plant response to LOCAs of different size and location, and with different dynamic effects on important plant equipment enable pro-active risk management or monitoring. As an example, the Barsebäck-1 PSA considers dynamic effects of a pipe whip (following a DEGB) involving the stripping of insulation material from adjacent piping.

Second, using reliability models and parameter estimation approach by this project supports PSA applications such as risk-informed ISI, and PSA-based event analysis. Successful risk-informed ISI projects would require validated pipe rupture frequencies, where validation is accomplished through evaluations of applicable service experience data. This application study with Barsebäck-1 as reference plant is an alternative to the technical options to risk-informed ISI described in NUREG-1661 (Guttmann et al, 1999).

9.3 PSA Methods & Material Sciences

The application study was a demonstration of how to use of a piping failure data to derive pipe rupture frequencies. The author of this report does recognize the strengths and weaknesses of probabilistic fracture mechanics (PFM). An important analytical strength of PFM evaluations is the correlation of pipe leaks and ruptures to specific leak rates and size of pipe fracture.

In its present form, the data analysis approach to frequency estimation does not characterize pipe ruptures by leak rate. A rupture was interpreted to be an event causing a release of process medium well beyond the plant technical specification leak rate limits. A significant advancement would be to couple data analysis to a physical model (such as a PFM model), and thereby relating frequency of failure to pipe fracture size.

9.4 Direct Estimation versus PFM

Direct estimation using a piping failure database is technically viable, and a cost-effective approach to piping reliability analysis. The direct estimation is not a short-cut approach, however. The level of effort involved in collecting and analyzing data is considerable. Yet, on a system-wide basis, the direct estimation could provide substantial savings in engineering effort versus the PFM-approach. Also, direct estimation is compatible with the tools and techniques of PSA, and it makes a closer connection with ISI-findings and operational data.

9.5 Location Dependency of Weld Failures

The ‘location dependency’ concept was introduced in Chapter 3. It was formulated to address two problems in piping *component* reliability analysis: 1) Lack of detailed piping component population data for the full range of BWR plants covered by the piping failure database; and 2) The possibility that some weld locations could be more prone to IGSCC than others because due to the ease or difficulty of welding. Also, the study needed a simplified way of generating piping component reliability data to fully support the Barsebäck-1 RCPB piping reliability model. The concept is a source of uncertainty, however.

10. Conclusions and Recommendations

The application study on RCPB piping reliability was performed to demonstrate basic piping failure data analysis principles. It was a 'proof-of-principle' study, and should not be seen as an all-inclusive, ultimate piping reliability analysis. As outlined below, routine applications of the technical approach would require additional methodological enhancements.

10.1 Conclusions

In Barsebäck-1, important contributions to the LOCA frequency come from pipe breaks that are due to thermal fatigue. Other significant LOCA frequency contributions are due to TGSCC/IGSCC in base metal of bends in cold worked medium-diameter piping. Compared with the seminal WASH-1400, the new medium- and large LOCA frequencies are lower by about an order of magnitude.

The study tried two different approaches to piping failure rate estimation: 1) 'PSA-style', simple estimation using Bayesian statistics, and 2) Fitting of statistical distribution to failure data. A large, validated database on piping failures (like SKI-PIPE) supports both approaches. The ability to perform failure rate estimation is limited by the completeness of the piping failure database. There is significant statistical uncertainty in failure rates for RCPB piping susceptible to thermal fatigue and flow-assisted corrosion in RCPB piping systems.

10.2 Recommendations on Future R&D Directions

Initiated in mid-1994, the R&D supported by SKI's Department of Plant Safety Assessment[34] resulted in a major piping failure database. With financial and engineering support from BKAB, the 4-year program came to closure through the Barsebäck-1 RCPB piping reliability application study.

Without this application, the full potential of the piping failure database could not have been demonstrated. Also, the application study identified areas in need of further R&D. Listed below are recommendations for future R&D and applications:

[34] SKI/RA is the organizational acronym.

SKI

- SKI should continue its support of the SKI-PIPE database. Ideally, database management should eventually be shared by the international nuclear safety community to expand the content, and to enable authorized access by PSA analysts, structural engineers and ISI personnel. An authorized user would come from an organization actively supporting the database with information (e.g., root cause analysis reports, ISI summary reports) on occurred failures and degradations.

  In the short term, SKI should ensure that new events be added to the database as they occur. Also, full validation of the quality and completeness of all event records for BWRs as well as PWRs should be prioritized. In the longer term, SKI should promote the database in ongoing national, Nordic and international programs in risk-informed ISI.

- SKI should actively facilitate new R&D in the area of statistical analysis of piping failure data, including uncertainty analysis. Such research should address the development of appropriate prior distributions for leaks and ruptures in piping characterized by typical light water reactor attribute-influence sets. Furthermore, some R&D should be performed on data pooling guidelines. Specific questions to be addressed include: a) Guidelines for pooling of data from different plant designs; b) The concept of location dependent piping failures should be explored in further detail, including establishment of a database on piping component population data. SKI-PIPE tracks failure data on field welds versus shop welds. Within the scope of the application study, that particular subset of failure data was not explored.

- The application study utilized several qualitative and semi-quantitative failure data insights to facilitate parameter estimation. These 'database insights' should be converted into probabilistic quantities; i.e., concepts like the 'location dependency' of weld failures should be characterized by appropriate statistical distributions.

- As discussed in Chapter 8, the total approach to pipe rupture frequency estimation and LOCA frequency estimation is amenable to computer automation. A relatively limited effort would be required to develop an integrated computer workstation concept building on commercially available spreadsheet and database programs.

- Chapters 3, 5, 6 and 7 presented some baseline and adjusted leak and rupture frequencies. To prevent misapplications of published data, SKI Report 98:30 does not include the full Barsebäck-1 piping reliability database. It is desirable that high quality, generic reliability data on piping become available to PSA analysts. Therefore, SKI should consider R&D to develop requirements for generic data; e.g., recommended prior and posterior distribution types for different piping, sets of raw data for a range of typical attribute-influence sets.

BKAB

- In BKAB's living PSA program, familiarity with the LOCA model and data analysis approach should be established through model decomposition and sensitivity studies. Chapter 8 presented a very small subset of results. It is recommended that a detailed summary of results and insights be prepared by PSA engineers for presentation to operations, maintenance, ISI personnel.

- Training of PSA engineers in statistical analysis of piping failure data. A training program should be developed around SKI Report 98:30.

- The project developed a set of 'look-up' tables[35] summarizing the piping reliability data applicable to the ten RCPB piping systems in Barsebäck-1. In edited form, these tables should be included in the PSA engineers' work books and be updated as new service data become available.

- Limited to estimation of pipe rupture frequency and LOCA frequency, the work scope produced PSA_VER2 and a piping component reliability database. These databases represent major elements towards risk-informed ISI for Barsebäck-1. A near-term application should involve a formal risk-informed ISI program development.

[35] Not included in SKI Report 98:30.

11. References

Aaltonen, P., K. Saarinen and K. Simola, 1993. "The Correlation of IGSCC Propagation with the Power Plant Transient History," *Int. J. Pres. Ves. & Piping*, **55**:149-162.

Advisory Committee on Reactor Safeguards (ACRS), 1998. *Risk-based Inservice Inspection & Loss of Shutdown Cooling at River Bend*, Meeting Transcript of the 451st ACRS, Washington (DC).

Andresen, P.L., 1998. *IGSCC Crack Propagation Rate Measurement in BWR Environments. Executive Summary of a Round Robin Study*, SKI Report 98:27, Swedish Nuclear Power Inspectorate, Stockholm (Sweden).

Bauer, C.-O., 1994. "Reliability, Uncertainties, Their Origin and Consequences in Presenting Test Results," *Int. J. Pres. Ves. & Piping*, **59**:313-322.

Beliczey, S. (Ed.), 1995. *Ermittlung der Häufigkeiten von Lecks und Brüchen in druckführenden Systemen für probabilistische Sicherheitsanalysen*, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Köln (Germany).

BKAB, 1998. *Barsebäck-1 PSA. Section 1.2,* T-9812-31 (in Swedish), Barsebäck (Sweden).

Bush, S.H. et al, 1996. *Piping Failures in United States Nuclear Power Plants: 1961-1995*, SKI Report 96:20, Swedish Nuclear Power Inspectorate, Stockholm (Sweden).

Commonwealth Edison, 1999. *Results of the Root Cause Analysis of Flaws in Induction Heat Stress Improvement Treated Recirculation System Piping*, SVP-98-376 (with Attachment), Cordova (IL).

Crowder, M.J. et al, 1994. *Statistical Analysis of Reliability Data*, Chapman & Hall, London (UK), ISBN 0-412-59480-3.

Danko, J.C., 1983. "Boiling Water Reactor Pipe Cracking: The Problem and Solution," *Proc. Materials in Nuclear Energy*, American Society for Metals, Metals Park (OH).

Dimitrijevic, V. et al, 1999. *Revised Risk-Informed Inservice Inspection Evaluation Procedure*, TR-112657, Electric Power Research Institute, Palo Alto (CA).

Doctor, S.R., P. Lemaitre and S. Crutzen, 1995, "Austenitic Steel Piping Testing Exercises in PISC," *Nuclear Engineering and Design*, **157**:231-244.

Enkvist, J., A. Edland and O. Svenson, 1999. *Human Factors Aspects of Non-Destructive Testing in the Nuclear Power Context. A Review of Research in the Field*, SKI Report 99:8, Swedish Nuclear Power Inspectorate, Stockholm (Sweden).

Erve, M. et al, 1997. "Planned and Implemented Remedial Measures to Prevent IGSCC and to Ensure Safe Plant Operation with BWR Piping Systems Made of Stabilized Austenitic Steels," *Nuclear Engineering and Design*, **171**:125-136.

Fleming, K.N. and S.R. Gosselin, 1997a. "Evaluation of Pipe Failure Potential via Degradation Mechanism Assessment," *Proc. ICONE 5: 5th International Conference on Nuclear Engineering*, May 26-30, Nice (France).

Fleming, K.N. and T.J. Mikschl, 1998. "Application of Markovian Technique to Modeling Influences of Inspection on Pipe Rupture Frequencies," *Proc. PSAM-IV*, Springer Verlag, New York (NY), pp 463-470.

Fleming, K.N. et al, 1998. *Piping System Reliability and Failure Rate Estimation Models for Use in Risk-Informed In-Service Inspection Applications*, TR-110161 (Proprietary Licensed Material), Electric Power Research Institute, Palo Alto (CA).

Griffith, P., 1997. *Screening Reactor Steam/Water Piping Systems for Water Hammer*, NUREG/CR-6519, U.S. Nuclear Regulatory Research, Washington (DC).

Kapur, K.C. and L.R. Lamberson, 1977. *Reliability in Engineering Design*, John Wiley & Sons, Inc., New York (NY), ISBN 0-471-51191-9.

Kassir, M. et al, 1985. *An Evaluation of Stress Corrosion Crack Growth in BWR Piping Systems*, BNL-NUREG-51874 (NUREG/CR-4221), Brookhaven National Laboratory, Upton (NY)

Lydell, B.O.Y. and R. Nyman, 1996. "Incorporation of Piping System Failures in PSA. Some Basic Analysis Considerations," *Proc. PSA'96*, American Nuclear Society, La Grange Park (IL).

Lydell, B.O.Y. and R. Nyman, 1998. "Systematic Evaluation of Service Data on Piping: A Framework for Effective Aging Management," *Proc. of ICONE-6: 6th International Conference on Nuclear Engineering*, Paper #6303, American Society of Mechanical Engineers, New York (NY).

Martz, H.F. and R.A. Waller, 1982. *Bayesian Reliability Analysis*, John Wiley & Sons, Inc., New York (NY), ISBN 0-89464-395-9.

Mehta, H.S. and S.R. Gosselin, 1995. *An Environmental Factor Approach to Account for Reactor Water Effects in Light Water Reactor Pressure Vessel and Piping Fatigue Evaluations*, EPRI TR-105759, Electric Power Research Institute, Palo Alto (CA).

Moosemiller, M. and B. Weber, 1998. *Guidelines for Improving Plant Reliability through Data Collection and Analysis*, Center for Chemical Process Safety, American Institute of Chemical Engineers, New York (NY), ISBN 0-8169-0751-X.

Nelson, W., 1972. “Theory and Application of Hazard Plotting for Censored Failure Data,” *Technometrics*, **14**:945-966.

New York Power Authority, 1990. *Summary of Intergranular Stress Corrosion Cracking Inspection During the 1990 Refueling Outage*, JPN-90-041, James A. FitzPatrick Nuclear Power Plant, White Plains (NY).

Nyman, R. et al, 1997. *Reliability of Piping System Components. Framework for Estimating Failure Parameters from Service Data*, SKI Report 97:26, Swedish Nuclear Power Inspectorate, Stockholm (Sweden).

Organization for Economic Cooperation and Development, Nuclear Energy Agency (OECD-NEA), 1996. *Knowledge Base for Emergency Core Cooling System Recirculation Reliability*, NEA/CSNI/R (95)11.

Poloski, J.P. et al, 1999 *Rates of Initiating Events at U.S. Nuclear Power Plants 1987 through 1995*, NUREG/CR-5750, U.S. Nuclear Regulatory Commission, Washington (DC).

Rahman, S. et al, 1995. *Probabilistic Pipe Fracture Evaluations for Leak-Rate-Detection Applications*, BMI-2174 (NUREG/CR-6004), Battelle, Columbus (OH).

Sandstedt, J. (1998). *Barsebäck-1 PSA Level 1. Main Report. Section 6.5: Pipe Breaks and Consequential Events*, (in Swedish) BKAB.

Shah, V.N., A.G. Ware and A.M. 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 (DC).

Siu, N.O. and D.L. Kelly, 1998. “Bayesian Parameter Estimation in Probabilistic Risk Assessment,” *Reliability Engineering and System Safety*, **62**:89-116.

Stevens, G.L., 1998. *Evaluation of Environmental Thermal Fatigue on Selected Components in a Boiling Water Reactor Plant*, EPRI TR-110356, Electric Power Research Institute, Palo Alto (CA).

Su, N.T., 1990. *Review of Thermal Stratification Operating Experience*, AEOD/S902, U.S. Nuclear Regulatory Commission, Washington (DC).

Sydsvenska Värmekraft AB (Sydkraft), 1985. *Säkerhetsgranskning BARSEBÄCK 1984*, Malmö (Sweden), pp 2-25--26.

U.S. Nuclear Regulatory Commission, 1984. *Investigation and Evaluation of Stress Corrosion Cracking in Piping of Boiling-Water Reactor Plants*, NUREG-1061, Washington (DC).

U.S. Nuclear Regulatory Commission, 1999. *Technical Elements of Risk-Informed Inservice Inspection Programs for Piping*, NUREG-1661 (Draft), Washington (DC).

Van Duyne, D.A., W. Yow, J.W. Sabin, 1992. *Water Hammer Prevention, Mitigation, and Accommodation*, EPRI NP-6766, Electric Power Research Institute, Palo Alto (CA).

Åström, L., 1997. *PSA B1/B2: Survey of Pipe Break Locations Inside the Containment*, T-9706-180 (in Swedish), BKAB.

A: Abbreviations, Acronyms & Notation

AISI	Americal Iron and Steel Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for the Testing of Materials
B/A-CC	Boric acid induced stress corrosion
BFS	Bundesamt für Strahlenschutz (Germany)
BKAB	Barsebäck Kraft Aktiebolag
BOP	Balance of Plant
BRS	Bibliographical Retrieval System of the U.S. NRC
BWR	Boiling water reactor
CACD	Code allowable crack depth
DBA	Design Basis Accident
D&C	Design & construction
DEGB	Double-ended guillotine break
DN	Nominal diameter [mm]
ELC	Austenitic stainless steel of extra-low carbon content ($C < 0.03\%$)
EPRI	Electric Power Research Institute
FAC	Flow Assisted Corrosion
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit
HAZ	Heat affected zone
HSK	Swiss Federal Nuclear Safety Inspectorate
HSW	Heat sink welding
HWC	Hydrogen water chemistry
IC	Inside containment
IGSCC	Intergranular stress corrosion cracking
IHSI	Induction heating stress improvement
INEEL	Idaho National Engineering and Environmental Laboratory
ISI	In-service inspection
LC	Austenitic stainless steel of low carbon content ($0.03\% \leq C \leq 0.04\%$)
LER	Licensee Event Report
LOCA	Loss of coolant accident
LWR	Light water reactor
MIC	Microbiologically-induced corrosion
MSIP	Mechanical stress improvement process
NEA	Nuclear Energy Agency of the OECD(Organization for Economic Cooperation & Development)
NDE	Non-destructive examination
NRC	Nuclear Regulatory Commission
PDR	Public Document Room of the NRC
PFM	Probabilistic fracture mechanics
PSA	Probabilistic Safety Assessment
PWR	Pressurized water reactor
PWSCC	Primary Water Stress Corrosion Cracking
RCPB	Reactor coolant pressure boundary

RHR	Residual heat removal
RPV	Reactor Pressure Vessel
RWCU	Reactor water cleanup
SCC	Stress Corrosion Cracking
SHT	Solution heat treatment
SKI	Statens Kärnkraftinspektion
SLCS	Standby liquid control system
SPIN	Standardized Plant Identification Number
TAF	Top of active fuel
TGSCC	Transgranular stress corrosion cracking
TTF	Time to failure
TWC	Through-wall crack
T&M	Test and maintenance
UT	Ultrasonic testing
VGB	Technisches Vereinigung der Grosskraftwerkbetreiber (Germany)
WH	Water hammer
WOR	Weld overlay repair

Notation

C-d	Censored data. The process of including non-failed items in the parameter estimation. This becomes important when considering rare events.
F	Number of failure events. In the context of calculating the conditional probability of pipe rupture, 'F' has a specific meaning (see below). In equation (6.2), 'F' also has the meaning of frequency.
F_R	Frequency of pipe rupture.
m	Mean time to failure (= λ^{-1}, where λ is the constant failure rate).
MR	Median rank; used for skewed distributions, it is a non-parametric estimate of the cumulative distribution based on ordered failures.
r	Number of rupture events.
$p_{R\|F}$	Conditional probability of rupture given a failure, where 'failure' implies a through-wall crack extending 30% of the pipe circumference (i.e., unstable crack propagation).
T	Exposure time
α	Significance level.
β	Shape factor.
δ	Location parameter (or minimum life).
$\Gamma(n)$	Gamma function
χ^2	Chi-square distribution.
μ	Mean
θ	Characteristic life

Swedish Standardized Plant System Identification Numbers (SPINs)

200-Series, A Selection

211	Reactor pressure vessel (RPV)
213	Core spray nozzle assemblies
221	Control rod drive assemblies

300-Series, A Selection

311	Main steam system
312	Main feedwater system
313	Recirculation system (Reactor coolant system, PWR)
314	Main steam pressure relief system
316	Containment pressure suppression system
321	Residual heat removal system
322	Containment spray system
323	Emergency core cooling system
326	RPV head cooling system
327	Auxiliary feedwater system
331	Reactor water cleanup system
351	Boron injection system
352	Reactor controlled drain system
354	Hydraulic scram system

B: Barsebäck-1 RCPB Piping Component Counts

Table B-1: Main Steam Piping System Components - System 311.

Component Type	Location of Weld	Nominal Diameter [DN]	No. of Components (ELC-SS)	No. of Components (HC-SS)	No. of Components (CS)
Bend	N/A	50	0	0	68
		80	0	0	26
		500	0	0	23
Pipe	N/A	50	0	1	81
		80	0	26	9
		500	4	25	0
Tee	N/A	50	0	0	12
		80	0	0	1
		500	0	0	10
Weld	Bend-to-bend	50	0	0	2
	Bend-to-pipe		0	0	111
	Bend-to-valve		0	0	9
	Bend-to-tee		0	1	5
	Nozzle-to-pipe		0	0	1
	Safe-end-to-bend		0	0	5
	Pipe-to-pipe		0	0	1
	Pipe-to-valve		0	0	24
	Safe-end-to-pipe		0	0	3
	Safe-end-to-tee		0	0	3
	Tee-to-pipe		0	0	23
	Tee-to-valve		0	0	2
Weld	Bend-to-pipe	80	0	22	30
	Bend-to-valve		0	0	1
	Bend-to-tee		0	3	0
	Pipe-to-pipe		0	1	3
	Pipe-to-valve		0	0	1
	Safe-end-to-pipe		0	4	0
	Tee-to-pipe		0	0	1
	Tee-to-valve		0	0	1
Weld	Pipe-to-valve	150	0	2	0
	Tee-to-valve		0	0	2
Weld	Tee-to-valve	175	0	0	10
Weld	Bend-to-pipe	500	21	17	2
	Bend-to-tee		3	4	0
	Bend-to-valve		1	1	0
	Nozzle-to-pipe		0	0	1
	Pipe-to-pipe		0	1	0
	Pipe-to-valve		5	2	0
	Safe-end-to-bend		0	1	1
	Safe-end-to-pipe		1	1	0
	Tee-to-pipe		2	6	0
	Tee-to-valve		2	0	1

Table B-2: Main Feedwater Piping System Components - System 312.

Component Type	Location of Weld	Nominal Diameter [DN]	No. of Components (ELC-SS)	No. of Components (LC-SS)	No. of Components (HC-SS)
Bend	N/A	250 350	0 1	0 0	25 0
Pipe	N/A	250 350	0 1	0 0	25 2
Tee	N/A	200 350	0 0	0 0	2 4
Weld	Tee-to-tee	200	1	0	0
Weld	Bend-to-bend Bend-to-pipe Bend-to-tee Bend-to-valve Nozzle-to-pipe Pipe-to-valve Tee-to-pipe	250	0 0 0 0 0 0 0	0 14 3 2 0 3 0	4 26 0 1 4 3 1
Weld	Bend-to-pipe Bend-to-valve Pipe-to-pipe Pipe-to-valve Safe-end-to-safe-end Safe-end-to-tee Tee-to-pipe	350	0 1 0 1 0 0 2	0 0 0 1 0 2 2	1 0 1 2 1 0 0

Table B-3.1: Recirculation Piping System Components - System 313.

Comp. Type	Location of Weld	Nominal Diam. [DN]	No. of Components (ELC-SS)	No. of Components (LC-SS)	No. of Components (HC-SS)	No. of Components (CS/SS)
Bend	N/A	100 200 600	2 0 0	0 0 0	7 8 0	0 0 12
Pipe	N/A	100 200 600 650	3 0 0 0	0 0 0 1	12 8 6 3	0 0 14 0
Weld	Bend-to-pipe Nozzle-to-pipe Pipe-to-pipe Pipe-to-tee Pipe-to-valve	100	4 0 2 0 0	0 0 0 0 0	12 2 1 2 7	0 0 0 0 0
Weld	Bend-to-nozzle Buttring-to-nozzle Buttring-to-valve	150	0 0 0	0 0 0	10 16 16	0 0 0
Weld	Bend-to-pipe Bend-to-valve Nozzle-to-pipe Pipe-to-valve	200	0 0 0 0	3 1 0 0	9 2 7 1	0 0 0 0

Table B-3.2: Recirculation Piping System Components - System 313.

Comp. Type	Location of Weld	Nominal Diam. [DN]	No. of Components (ELC-SS)	No. of Components (LC-SS)	No. of Components (HC-SS)	No. of Components (CS/SS)
Weld	Nozzle-to-pipe	250	0	0	1	0
Weld	Bend-to-pipe	600	0	0	0	20
	Bend-to-valve		0	0	0	4
	Nozzle-to-pipe		0	0	0	2
	Pipe-to-pump		0	0	4	0
	Pipe-to-valve		0	0	1	3
	Pipe-to-venturi		0	0	4	0
	Safe-end-to-pipe		0	0	8	0
	Valve-to-pump		0	4	0	0
	Valve-to-venturi		0	4	0	0

Table B-4.1: Main Steam Pressure Relief Piping System Components - System 314.

Component Type	Location of Weld	Nominal Diameter [DN]	No. of Components (ELC-SS)	No. of Components (LC-SS)	No. of Components (CS)
Bend	N/A	125	0	0	72
		150	0	0	39
		250	0	0	17
Pipe	N/A	50	0	0	2
		125	0	0	129
		150	0	0	57
		250	0	0	31
Reducer	N/A	125 x 150	0	0	14
		150 x 250	0	0	5
Tee	N/A	125	0	0	2
		150	0	0	12
		250	0	0	14
Weld	Nozzle-to-bend	50	0	0	3
	Nozzle-to-pipe		0	0	1
Weld	Bend-to-pipe	125	0	0	124
	Pipe-to-pipe		0	0	49
	Pipe-to-reducer		0	0	2
	Tee-to-pipe		0	0	14
Weld	Bend-to-bend	150	0	0	2
	Bend-to-pipe		0	0	66
	Bend-to-reducer		0	0	4
	Bend-to-tee		0	0	3
	Bend-to-valve		0	0	1
	Pipe-to-pipe		0	0	4
	Pipe-to-reducer		0	0	4
	Pipe-to-valve		0	0	1
	Safe-end-to-bend		0	0	1
	Safe-end-to-pipe		0	0	20
	Tee-to-pipe		0	0	16
	Tee-to-reducer		0	0	2

Table B-4.2: Main Steam Pressure Relief Piping System Components - System 314.

Component Type	Location of Weld	Nominal Diameter [DN]	No. of Components (ELC-SS)	No. of Components (LC-SS)	No. of Components (CS)
Weld	Bend-to-pipe	250	0	0	27
	Bend-to-tee		0	0	4
	Bend-to-valve		0	0	2
	Pipe-to-pipe		0	0	10
	Pipe-to-valve		0	0	3
	Tee-to-pipe		0	0	17
	Tee-to-reducer		0	0	5

Table B-5: Residual Heat Removal Piping System Components - System 321.

Component Type	Location of Weld	Nominal Diameter [DN]	No. of Components (ELC-SS)	No. of Components (LC-SS)	No. of Components (HC-SS)
Bend	N/A	150	0	0	5
		200	0	0	10
		250	0	0	4
Pipe	N/A	150	0	0	5
		200	0	0	15
		250	1	0	6
Tee	N/A	250	0	0	1
Weld	Bend-to-pipe	150	0	4	4
	Bend-to-penetration		0	1	0
	Bend-to-tee		1	0	0
	Pipe-to-valve		0	0	2
Weld	Bend-to-pipe	200	0	8	12
	Bend-to-valve		0	2	0
	Pipe-to-penetration		0	0	1
	Pipe-to-pipe		0	1	3
	Pipe-to-reducer		0	2	2
	Pipe-to-valve		0	0	2
	Tee-to-pipe		0	0	2
Weld	Bend-to-pipe	250	1	5	2
	Bend-to-valve		0	0	2
	Pipe-to-penetration		0	1	0
	Pipe-to-valve		0	0	2
	Tee-to-pipe		0	1	2

Table B-6: Core Spray Piping System Components - System 323.

Component Type	Location of Weld	Nominal Diameter [DN]	No. of Components (ELC-SS)	No. of Components (LC-SS)	No. of Components (HC-SS)
Bend	N/A	80	0	0	2
		100	0	0	38
		200	0	0	1
		250	0	0	9
Pipe	N/A	80	0	0	4
		100	0	0	32
		250	1	1	9
Tee	N/A	250	0	0	1
Weld	Bend-to-pipe	80	0	0	4
	Pipe-to-pipe		0	0	2
	Pipe-to-valve		0	0	2
Weld	Bend-to-bend	100	3	0	8
	Bend-to-pipe		34	0	17
	Bend-to-tee		2	0	0
	Pipe-to-pipe		0	0	7
	Tee-to-pipe		0	5	1
Weld	Bend-to-pipe	200	0	0	1
	Bend-to-tee		0	0	1
	Tee-to-valve		0	0	1
Weld	Bend-to-pipe	250	0	11	4
	Bend-to-tee		0	1	0
	Bend-to-valve		0	0	1
	Pipe-to-valve		0	0	6
	Tee-to-pipe		0	1	0

Table B-7: RPV Head Cooling Piping System Components - System 326.

Component Type	Location of Weld	Nominal Diameter [DN]	No. of Components (ELC-SS)	No. of Components (LC-SS)	No. of Components (HC-SS)
Bend	N/A	150	0	0	15
Pipe	N/A	150	0	0	21
Tee	N/A	150	0	0	1
Weld	Bend-to-pipe	150	0	14	15
	Pipe-to-flange		1	1	0
	Pipe-to-pipe		0	0	1
	Pipe-to-valve		0	0	4
	Pipe-to-venturi		0	0	2
	Safe-end-to-bend		1	0	0
	Safe-end-to-nozzle		0	0	1
	Tee-to-pipe		1	1	3

Table B-8: Auxiliary Feedwater Piping System Components - System 327.

Component Type	Location of Weld	Nominal Diameter [DN]	No. of Components (ELC-SS)	No. of Components (LC-SS)	No. of Components (HC-SS)
Bend	N/A	125 150	3 2	0 0	3 2
Pipe	N/A	125 150	0 0	6 0	1 6
Reducer	N/A	150 x 200	0	0	1
Tee	N/A	150	0	0	1
Weld	Bend-to-pipe Pipe-to-pipe Pipe-to-valve	125	0 0 0	6 2 1	0 1 0
Weld	Bend-to-pipe Bend-to-tee Nozzle-to-pipe Pipe-to-pipe Pipe-to-reducer Pipe-to-valve Tee-to-pipe	150	0 0 0 0 0 0 0	0 0 0 0 0 0 0	2 1 2 2 1 3 1
Weld	Reducer-to-valve	200	0	1	0

Table B-9: Standby Liquid Control Piping System Components - System 351.

Component Type	Location of Weld	Nominal Diameter [DN]	No. of Components (ELC-SS)	No. of Components (LC-SS)	No. of Components (HC-SS)
Bend	N/A	50	11	0	11
Pipe	N/A	50	0	0	18
Weld	Bend-to-bend Bend-to-pipe Nozzle-to-bend Pipe-to-penetration Pipe-to-valve	50	3 11 1 0 0	0 0 0 0 1	5 16 1 2 7

Table B-10: Hydraulic Scram Piping System Components - System 354.

Component Type	Location of Weld	Nominal Diameter [DN]	No. of Components (ELC-SS)	No. of Components (LC-SS)	No. of Components (HC-SS)
Bend	N/A	15	0	0	50
		25	0	0	235
		65	0	0	51
Pipe	N/A	15	0	0	7
		65	0	0	111
Reducer	N/A	65 x 80	1	0	29
Tee	N/A	15	0	0	7
Weld	Bend-to-pipe	15	0	0	1
	Bend-to-tee		0	0	6
	Bend-to-valve		0	0	23
	Tee-to-pipe		0	0	13
Weld	Nozzle-to-pipe	25	5	0	5
	Pipe-to-valve		110	0	98
Weld	Bend-to-bend	65	0	0	3
	Bend-to-pipe		0	0	52
	Bend-to-valve		0	0	1
	Nozzle-to-pipe		0	0	98
	Pipe-to-pipe		0	0	2
	Pipe-to-valve		0	0	1
	Reducer-to-bend		1	0	30
Weld	Reducer-to-valve	80	0	12	18

C: SKI-PIPE - The Database Content[59]

Covering the period 1970 to the present, SKI-PIPE (MS-Access 7.0) is a periodically updated database on piping failures in commercial nuclear power plants worldwide. Figure C-1 shows an overview of the database content as of December 1998. The 'Master Database' includes 3080 piping failure reports. Four subsets of the Master Database are: 1) Piping failures in Soviet designed reactors (about 180 failure reports); 2) Weld failures due to IGSCC in BWRs (about 800 failure reports); 3) Piping failures specific to Barsebäck-1/2 (55 failure reports); and 4) Database supporting aging evaluations.

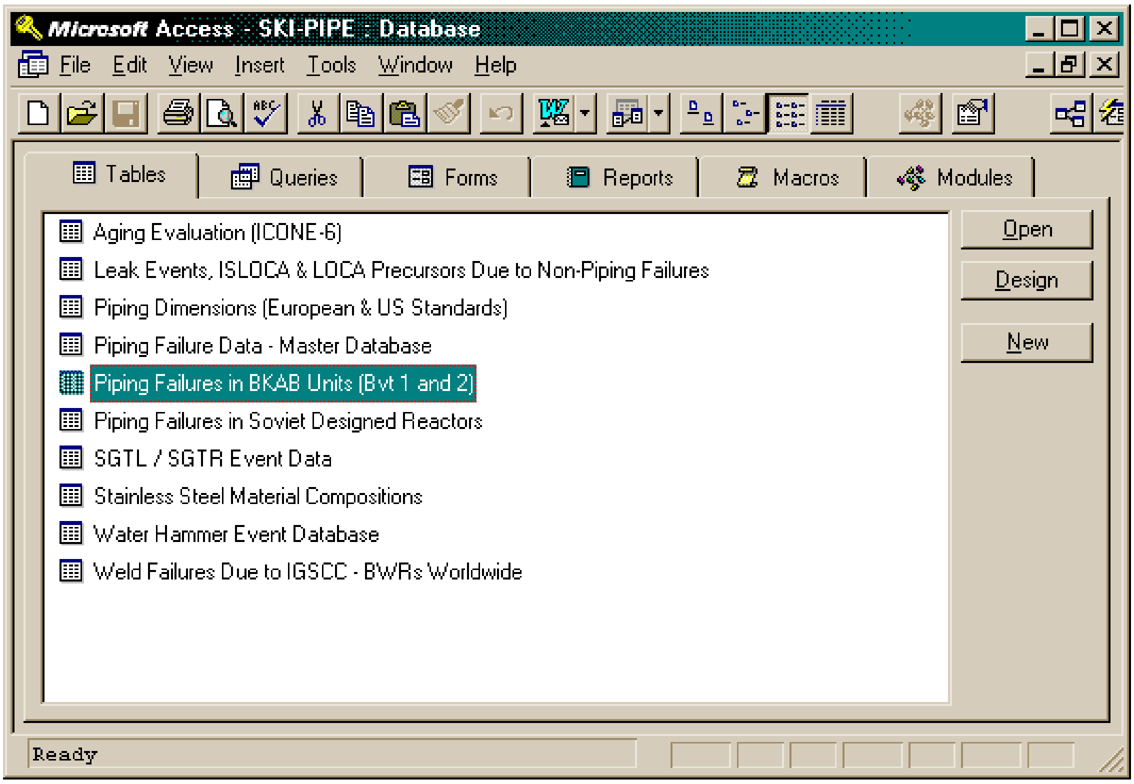

Figure C-1: Overview of SKI-PIPE Database - Status as of 12/98.

The database includes public domain and proprietary data as indicated in Figure C-2. Examples of proprietary data include failure reports supplied by utilities to the database development project. These were reports excluded from normal licensee reporting to safety authorities. Figure C-3 shows the coverage of BWR units. Except for Japanese BWR units, the database covers almost all, currently operating commercial BWR units. Figure C-4 shows piping failures by degradation mechanisms in Swedish BWR units. Most of the database records are for U.S. plants, supplemented with data from European (France, Germany, Switzerland) and Nordic plants (Table C-1). Finally, Table C-2 shows the database coverage of IGSCC events in BWRs worldwide.

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[59] Status as of December 31, 1998.

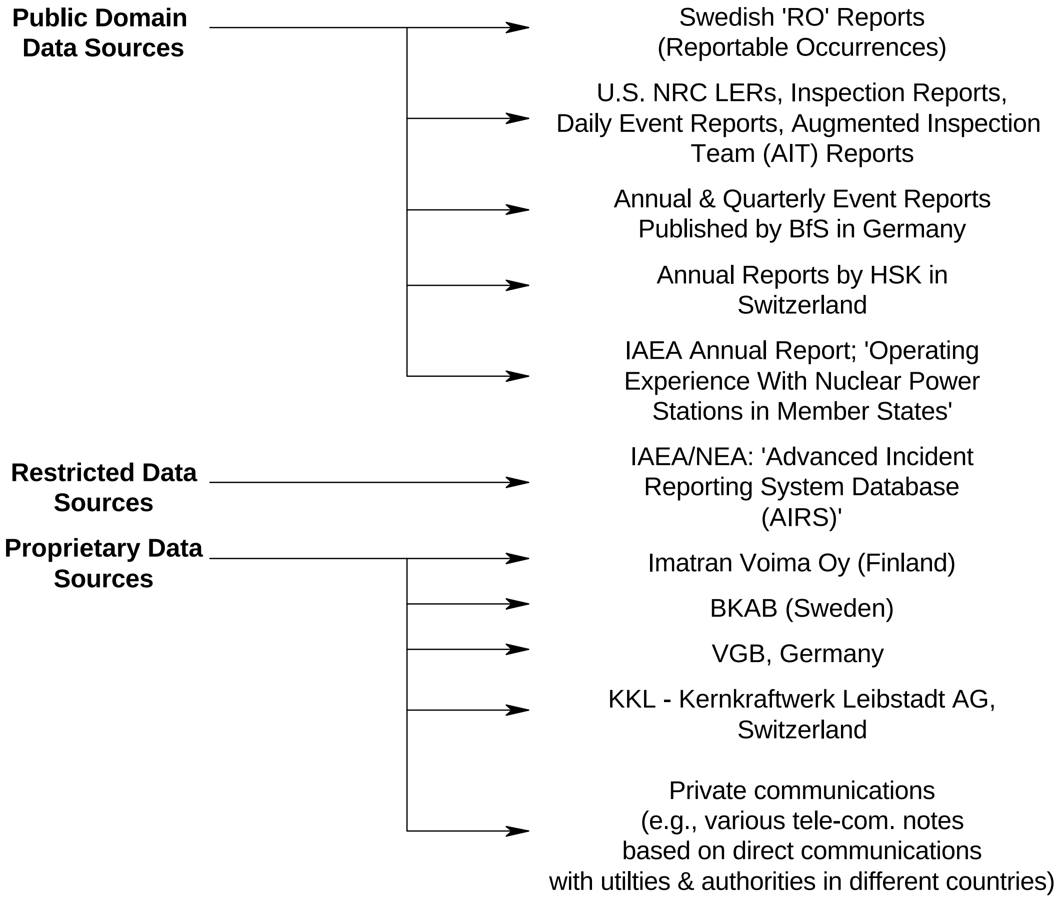

Figure C-2: Examples of Data Sources in SKI-PIPE.

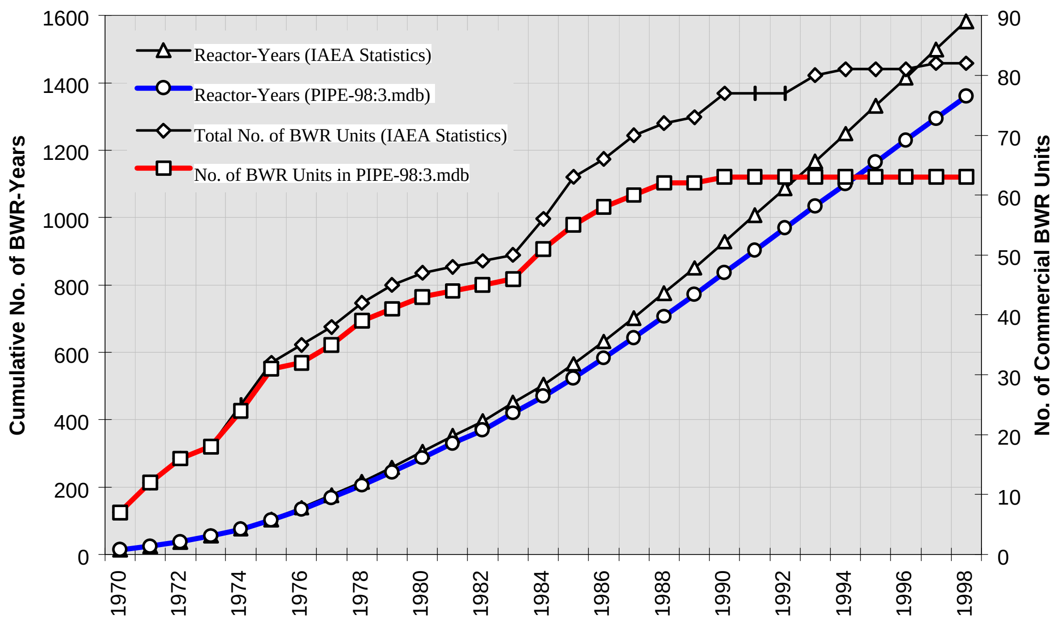

Figure C-3: The Coverage of SKI-PIPE - Currently Operating BWR Units

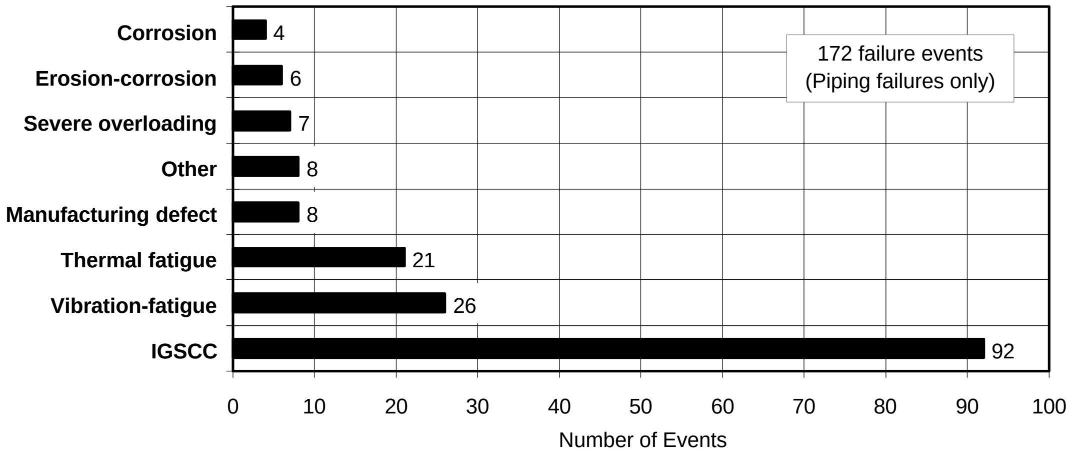

Figure C-4: Piping Failures in Swedish BWRs 1972-1998.

Table C-1: Database Coverage by Country.

Country	All Events	Cracks [No. of Events]	Leaks [No. of Events]	Ruptures [No. of Events]
Argentina	6	5	1	0
Belgium	5	1	3	1
Brazil	1	0	1	0
Bulgaria	8	0	7	1
Canada	84	6	68	10
Czech Republic	9	0	9	0
Finland	25	8	15	2
France	46	12	27	7
Germany (5%)	160	98	57	5
Hungary	9	0	4	5
India	6	0	4	2
Japan	22	0	21	1
Korea	5	0	3	2
Lithuania	9	0	7	2
Netherlands	3	1	1	1
Pakistan	5	0	3	2
Russia	85	5	64	16
Slovak Republic	12	0	11	1
Slowenia	3	0	2	1
Spain	9	4	3	2
South Africa	2	0	1	1
Sweden (6%)	192	99	75	18
Switzerland	46	36	8	2
Ukraine	26	2	22	2
United Kingdom[60]	1	0	1	0
USA (75%[36])	2324	468	1712	144
Column Summary:	3103	745	2130	228

[60] Sizewell ‘B’; no data on piping failures in gas-cooled reactors included in database.
[36] The percentage of U.S. event reports to the total number of reports.

Table C-2: Summary of IGSCC and Weld Overlay Repairs in BWRs.

Plant Name	NUREG-1061 (1984)		SKI-PIPE (1999)		Remark
	# Cracks	**# WORs**	**# Cracks**	**# WORs**	
Barsebäck-1	--	--	20	0	
Barsebäck-2	--	--	13	3	New RHR in 1998.
Forsmark-1	--	--	2	0	
Forsmark-2	--	--	6	0	
Forsmark-3	--	--	3	0	
Gundremmingen-B	--	--	1	0	
KKB Brunsbüttel	--	--	36	0	
KKI Isar-1	--	--	25	0	
KKK Krümmel	--	--	2	0	
KKP1 Phillipsburg-1	--	--	28	0	
Muehleberg	--	--	34	7	New piping in 1985.
Oskarshamn-1	--	--	15	0	New piping 1993-95.
Oskarshamn-2	--	--	7	0	
Ringhals-1	--	--	25	1	
Santa Maria de Garona	--	--	5	4	
TVO-1	--	--	4	4	
TVO-2	--	--	1	1	
Big Rock Point	--	--	4	0	
Browns Ferry-1	47	42	35	0	
Browns Ferry-2	2	0	7	1	
Browns Ferry-3	0	0	4	0	
Brunswick-1	3	3	15	8	
Brunswick-2	16	8	41	36	
Cooper	20	13	20	15	
Dresden-1	0	0	4	0	
Dresden-2	10	7	53	15	
Dresden-3	64	61	72	63	IHSI in 12/83
Duane Arnold	0	0	20	11	IHSI in 3/85
FitzPatrick	1	0	23	14	
Grand Gulf-1	--	--	1	0	
Hatch-1	7	6	42	23	
Hatch-2	39	27	2	1	
Hope Creek-1	--	--	1	1	
Millstone-1	0	0	31	21	
Monticello	6	6	11	1	
Nine Mile Point-1	53	0	11	0	
Nine Mile Point-2	--	--	1	0	
Oyster Creek	0	0	4	0	
Peach Bottom-2	26	21	4	0	
Peach Bottom-3	15	15	22	20	
Pilgrim	--	--	9	0	'NG' piping in 1984
Quad Cities-1	0	0	33	19	
Quad Cities-2	22	9	58	18	IHSI in 83/84
Vermont Yankee	34	22	19	9	
Totals:	365	240	227+547[37]	20+276	

[37] Foreign + U.S. data.

D: Database Structures - PSA_VER2 & SKI-PIPE

The Barsebäck-1 RCPB piping design information was collected in PSA_VER2 (MS-Access 7.0). Originally developed by BKAB personnel, this database was independently reviewed for completeness and accuracy as part of the work scope of the R&D project. The database structure is summarized in Table D-1. The new data fields, which were added to support the reliability parameter estimation and LOCA frequency calculation are indicated by asterisks.

SKI-PIPE (MS-Access 7.0) is SKI's database on piping degradations and failures in commercial nuclear power plants worldwide. Queries in this database were instrumental in processing the raw data for statistical analysis. The database structure is summarized in Table D-2.

Table D-1.1: PSA_VER2 - Description of Data Fields.

Field Name	Type	Description
Component ID	Number	Unique component ID; each piping component on the isometric drawings were given a unique identity. All isometric drawings were marked-up accordingly.
System	Number	Unit 1 SPIN; i.e., 311, 312, 313, etc.
Drawing No.	Text	Isometric drawing number.
Component Type	Text	Type of component; e.g., bend, pipe, weld, tee. This data field distinguishes between field-fabricated (F) and shop-fabricated (S) welds.
* Weld Location	Text	Defines where in a system a particular weld is located; e.g., BP = bend-to-pipe weld, PP = pipe-to-pipe weld, PV = pipe-to-valve weld, etc.
Object	Text	Weld number (as given by isometric drawing), bend angle (e.g., 45°, 90°)
* LOCA Class	Text	LOCA category as defined by PSA; A = large LOCA, S1 = medium LOCA, T = break location above TAF, B = break location below TAF, etc.
DN	Text	Nominal diameter
* f-DEGB	Number	Predicted break frequency.
Temperature	Text	Operating temperature of process medium; < 100 °C, 100-150°C, or > 150 °C
CHARGE_A	Text	Material composition identity as provided by manufacturer. For base metal in weld joint towards the lower isometric drawing number.
CHARGE_B	Text	Material composition identity as provided by manufacturer. For base metal in weld joint towards the higher isometric drawing number.
Material_A	Text	Type of base material (in 'A-side') per national standard; e.g., SS 2343-24 (austenitic stainless steel).
Material_B	Text	Type of base material (in 'B-side') per national standard; e.g., SS 2343-24 (austenitic stainless steel).
Carbon_A	Text	For stainless steels, carbon content of base material (in 'A-side'); < 0.03, 0.03-0.04, > 0.04% C.
Carbon_B	Text	For stainless steels, carbon content of base material (in 'B-side'); < 0.03, 0.03-0.04, > 0.04% C.

Table D-1.2: PSA_VER2 - Description of Data Fields.

Field Name	Type	Description
fGB	Number	Frequency of DEGB based on WASH-1400 data and engineering judgment as documented in BKAB Report T 9710-54 (in Swedish)
fUB	Number	Frequency of fracture (large leak < DEGB) based on WASH-1400 data and engineering judgment as documented in BKAB Report T 9710-54 (in Swedish)
Back-flush	Text	Qualitative evaluation of the likelihood of the need of back-flush operations to ensure ECCS operability given dynamic effects of steam/water jets on pipe insulation; 'low', 'medium' or 'high'.
ISI-Year	Date	Year of the most recent ISI of weld.
ISI-Frequency	Number	Frequency in years of ISI.
ISI-Next	Date	Year of the next planned ISI.

Table D-2.1: SKI-PIPE - Description of Data Fields.

Field Name	Type	Description
MER	Yes/No	Multiple Events Report; some reports include information on more than one crack/leak in one system. Used to identify events where a discovery resulted in an investigation to identify further piping degradations due to a common cause.
EDT	Date	Event date; date of discovery.
PTY	Text	Plant type; e.g., BWR, PWR, WWER.
NSSS-VENDOR	Text	Reactor vendor; e.g., ABB-Atom, KWU/Siemens, Westinghouse
FNM	Text	Plant name
CONSTRUCTOR	Text	Name of company responsible for the original piping system design. The default name is the architect engineering firm.
COD	Date	Date of commercial operation as default. If known, date of initial criticality. For U.S. data, based on NUREG-0020.
POS	Text	Plant operational state (at the time of discovery).
DSA	Text	Reference(s)
ETY	Text	Event type; 'Crack', 'Leak', 'Severance', 'Rupture'
ECA	Text	Type of Corrective Action; e.g., 'Repair'; 'Replacement'; 'Weld Overlay Repair (WOR).
ISS	Yes/No	Safety system actuation
IRT	Yes/No	Automatic reactor trip
TTR	Number	Repair time
NARRATIVE	Memo	Event narrative
LQT	Number	Quantity of process medium released [kg]
DOL	Text	Duration of release
LRT	Number	Leak rate [kg/s]
FLO	Text	Location of crack/leak/rupture; description of where in the piping system a degradation or failure occurred by referring to isometric component ID.
MSA	Text	Name of the affected plant system
OSA	Text	Name of other systems affected by the degradation or failure. Secondary effects of piping failure
S-TYPE	Text	Category of system affected by the degradation or failure; e.g., BOP, FIRE, RCPB, SUPPORT)
ISO	Yes/No	Isolateable?
DET	Text	Method of detection; e.g., ISI, ST = surveillance testing, WT = walk-through, etc.
CRS	Text	Crack morphology; size/geometry of crack or fracture.

Table D-2.2: SKI-PIPE - Description of Data Fields.

Field Name	Type	Description
CRACK-DEPTH	Number	Crack depth in percent of wall thickness.
CRACK-LENGTH	Number	Circumferential crack length in percent of inside diameter.
WELD-LOCATION	Text	Location of affected weld; e.g., BP = bend-to-pipe weld, PP = pipe-to-pipe weld, etc.
FIELD-WELD	Yes/No	Check if ‘Yes’
SHOP-WELD	Yes/No	Check if ‘Yes’
WOR	Yes/No	Weld overlay repair; check if ‘Yes’
REPLACEMENT	Yes/No	Check if ‘Yes’
REPL-DATE	Date/Time	Date of component replacement
CTA	Text	Type of piping component; e.g., bend/elbow, pipe, weld, tee.
YOO	Number	Year of commercial operation when failure occurred.
AGE	Number	Age of component socket [hours]
CLASS	Number	Based on diameter; events grouped in six diameter classes; 1= $\leq$ DN15, 15 < DN $\leq$ 25, 25 < DN $\leq$ 50, 50 < DN $\leq$ 100, 100 < DN $\leq$ 250, > DN250
THOMAS	Number	Ratio of diameter and pipe wall thickness
CSI	Number	Nominal diameter [DN]
WTK	Number	Wall thickness [mm]
MTR	Text	Material; e.g., carbon steel, stainless steel, etc.
MTR-DES	Text	Material designation according to national standard; e.g., AISI 304, SS2343, etc.
PMD	Text	Process medium
HWC	Yes/No	For BWRs; hydrogen water chemistry; check if ‘Yes’
HWC-START	Date/Time	Date when HWC was introduced.
IHSI	Yes/No	Induction heat stress improvement; check if ‘Yes’
IHSI-DATE	Date/Time	Date when IHSI was performed
STG	Yes/No	Normally stagnant process medium?
OPA	Number	Operating temperature [°C]
OPB	Number	Operating pressure [MPa]
OPC	Text	Process medium chemistry (for primary system); e.g., NWC = neutral water chemistry, HWC = hydrogen water chemistry
SYS	Yes/No	Systematic failure?
RFL	Text	Description of the extent and nature of a systematic failure
REST	Yes/No	Failure due to deficient system restoration?; e.g., no venting prior to fill procedure, etc.
CEA	Text	Apparent cause of failure; e.g., IGSCC, PWSCC, TGSCC, etc.
RC1	Text	Root cause (i)
RC2	Text	Root cause (ii)
CEC	Memo	Description of events and causal factors.
CMT	Memo	Any other information of relevance to the understanding of the underlying causal factors. Also, information on the type and extent of repair/replacement.
ISI	Yes/No	Deficient ISI; e.g., ISI not performed, or ISI failed to detect a flaw
ISI-CMT	Memo	Comments on ISI history

E: Vibration-Fatigue in Small-Diameter Piping (BWR Operating Environment)

In developing PSA_VER2, information was collected on all instrument sensing lines, RCPB drain and vent lines, and sample lines. Based on the information in Table E-1 and Table E-2, the frequency of rupture in small-diameter piping was estimated to be in the range of 7.4E-6/reactor-year (15 < DN ≤ DN25) to 1.0E-5/reactor-year (≤ DN15).

Table E-1: Weld Counts in Small-Diameter RCPB Piping in Barsebäck-1.

DN	E	311	312	313	314	321	323	326	327	351	354
≤ 15	100	5	3	16	0	5	0	8	0	4	59
15 < DN ≤ 25	230	0	0	12	0	0	0	0	0	0	218

Table E-2: Service Data on Small-Diameter RCPB Piping per SKI-PIPE.

Diameter [DN]	Row Summary	Crack [#]	Leak [#]	Rupture [#]
≤ 15	12	0	11	1
15 < DN ≤ 25	63	1	60	2

Bayesian update of Jeffrey's noninformative prior, and pipe reliability model '2' gives:

≤ **DN15**: $f_{R|V\text{-}F}$ = (2R+1)/2T = 3/[2(1479.7 x 100)] = 1.01E-05/r.y.

15 < DN ≤ 25: $f_{R|V\text{-}F}$ = (2R+1)/2T = 5/[2(1479.7 x 230)] = 7.35E-06/r.y.

These estimates were derived under the assumption that the weld counts for Barsebäck-1 are representative of the overall BWR plant population.

F: Erosion-Corrosion in RCPB Piping (BWR Operating Environment)

According to SKI-PIPE, there have been no reported degradations due erosion-corrosion in carbon steel piping within the RCPB of the worldwide LWR population. A judgmental conditional rupture probability ($p_{R|E\_C}$) of 2.0E-3 was assigned the carbon steel piping in Systems 311 (Main Steam) and 314 (Main Steam Pressure Relief). Furthermore, it is assumed that major steam piping replacements within the RCPB would not occur within the first 40 years of plant operation. Table F-1 summarizes the component counts (bends and welds) of the respective system in Barsebäck-1.

Table F-1: Piping Component Counts in Carbon Steel Steam Systems in Barsebäck-1.

Component	Row Summary	DN50	DN80	DN125	DN150	DN175	DN250	DN500
311 - Bend	117	68	26	0	0	0	0	23
314 - Bend	128	0	0	72	39	0	17	0
311 - Weld[61]	243	189	37	0	2	10	0	5
314 - Weld	583	4	0	387	124	0	68	0
	1071	261	63	459	165	10	85	28

Bayesian update of Jeffrey's noninformative prior, and pipe reliability model '1' gives:

$$f_{R|E-C} = [(2R+1)/2T] \quad p_{R|E-C} = [1/[2(40 \text{ x } 63)]] \quad 2.0\text{E-}3 = 4.0\text{E-}7/\text{r.y.}$$

$$f_{R|E-C} = 4.0\text{E-}7 / 1071 = 3.7\text{E-}10/\text{component.r.y.}$$

Virtually all known major degradations due to erosion-corrosion have occurred in secondary side wet-steam and water systems. The physical phenomena in erosion-corrosion processes are reasonably well understood[62], and the discussion above is a believed to be a reasonable approximation. The steam systems within the RCPB should be subjected to further evaluations to search for vulnerabilities to severe overloading caused by improperly placed pipe whip restraints in the main steam system and water hammer effects in pressure relief system.

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[61] It is assumed that vulnerability to erosion-corrosion damage exists in the pipe section immediately downstream a weld in carbon steel.

[62] Cragnolino, G., C. Czajkowski and W.J. Shack, 1988. Review of Erosion-Corrosion in Single-Phase Flows, NUREG/CR-5156, U.S. Nuclear Regulatory Commission.

G: Note on Statistical Analysis of Censored Data

In practical reliability work, it is often necessary to estimate the failure rate from a sample data set of time-to-failure. Some of these data points may be incomplete; e.g., when an observation ends before all the items have failed. A common method of dealing with censored data is to arrange all the data in time order and assign an order number to each failure for the particular mode being studies. In general, the order numbers of the failures following the first censoring will no longer be integers, but will take fractional values to allow for the censored item. When analyzing the censored data in this study, the derivation of median ranks (MRs) was done as follows[63]:

1. List order number (i) of failed items;

2. List increasing ordered sequence of life values (t_i) of failed items;

3. Against each failed item, list the number of items which have survived to a time between that of the previous failure and this failure (or between $t = 0$ and the first failure);

4. For each failed item, calculate the *mean order number* $i_{t.i}$ using the formula

$$i_{t.i} = i_{t.i-1} + N_{t.i}$$

where

$$N_{t.i} = [(n + 1) - i_{t.i-1}]/[1 + (\text{n - number of preceding items})]$$

in which n is sample size.

5. Calculate median rank (MR) for each failed item, using the formula[64]

$$MR_{t.i} = (i_{t.i} - 0.3) / (n + 0.4)$$

In this project the hazard plotting technique was used extensively to explore the piping failure data. Shown below is a set of hazard plots for different sets of failure data.

[63] From O'Connor, P.D.T., 1991. Practical Reliability Engineering. Third Edition, John Wiley & Sons (New York), ISBN 0-471-92696-5.
[64] For details, see Kapur, K.C. and L.R. Lamberson, 1977. Reliability in Engineering Design, John Wiley & Sons (New York), ISBN 0-471-51191-9, pp 297-311.

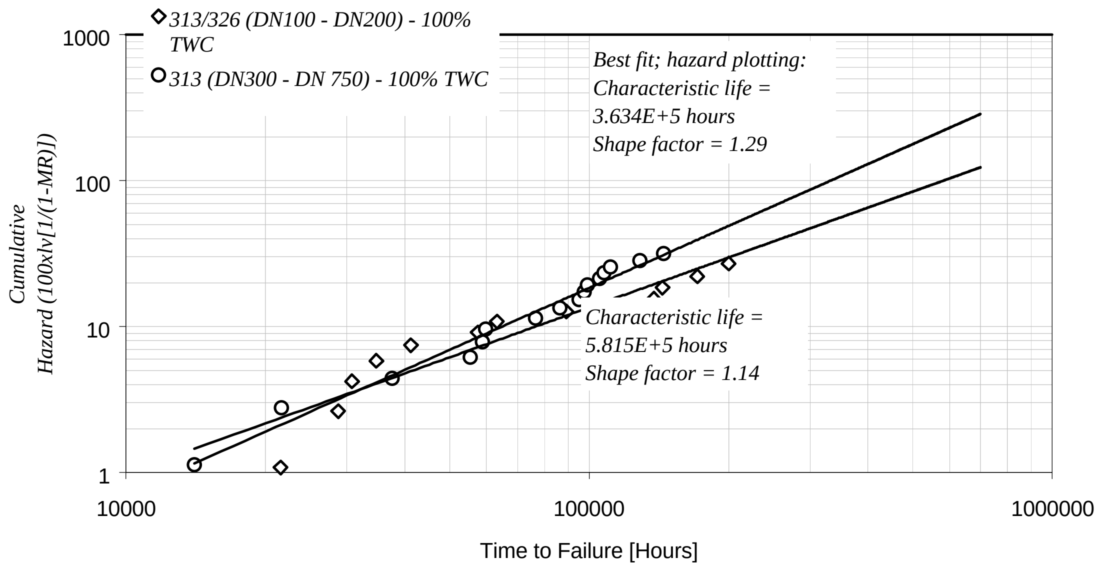

Figure G-1: Hazard Plots of Least-Square Fits to Recirculation Piping Failure Data.[38]

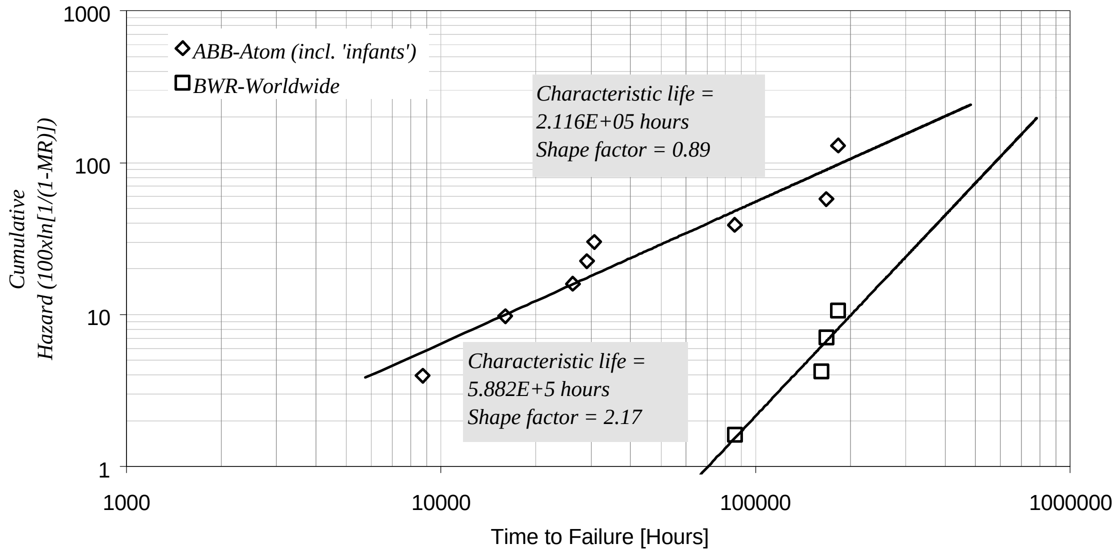

Figure G-2: Hazard Plot of Least-Square Fits to Thermal Fatigue Data.

[38] Using crack extrapolation and failure of ISI to detect flaws lead to 12 at-power leaks in large-diameter recirculation piping; 2 actual leaks plus 10 fictitious at-power leaks.

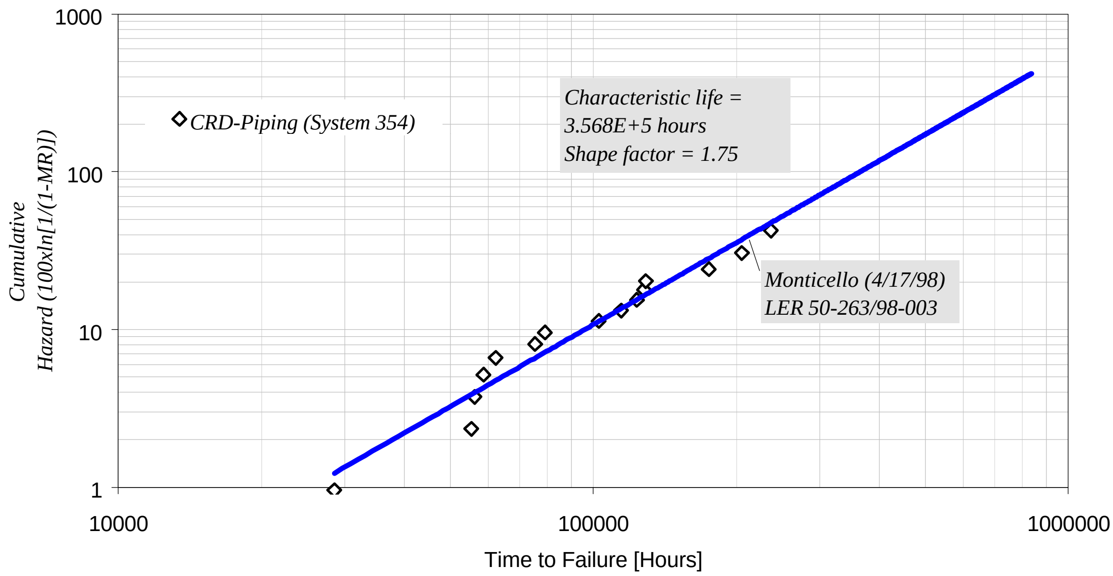

Figure G-3: Hazard Plot of Least-Square Fit to CRD Piping Failure Data.

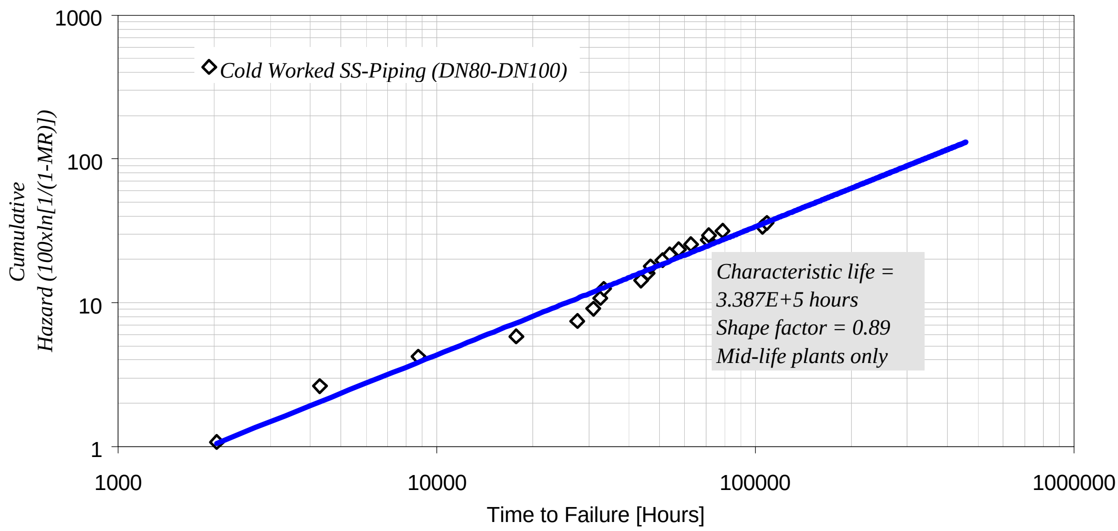

Figure G-4: Hazard Plot of Least-Square Fit to DN80-100 Pipe Failure Data.