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Strategies for Reactor Safety: Preventing Loss of Coolant Accidents

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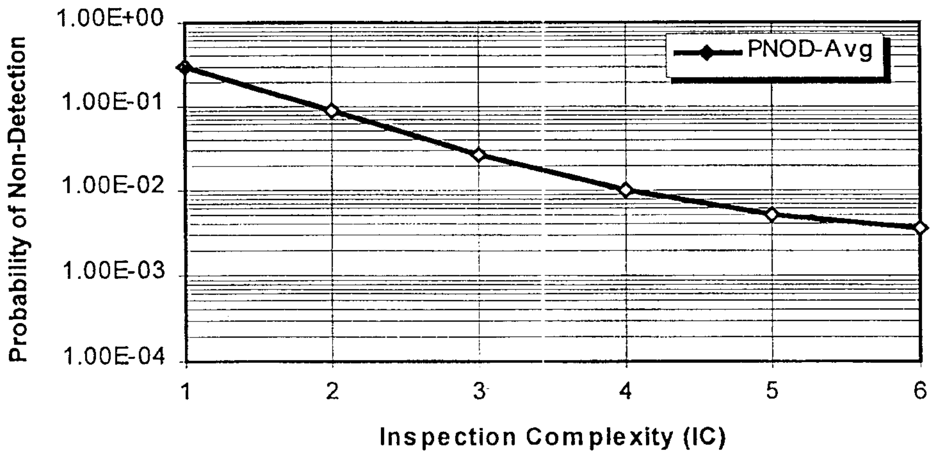

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Strategies for Reactor Safety: Preventing Loss of Coolant Accidents

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1. INTRODUCTION

The NKS/RAK-1 project forms a part of a four-year nuclear research program (1994-1997) in the Nordic countries; the NKS Programme. The NKS is a Nordic Committee for Safety Research with members from authorities, research organizations and enterprises in the nuclear field, which formulates and implements cooperative research programs with participation from the five Nordic countries. The programs are financed partly by NKS and partly by national bodies. To date, a total of five four-year research program cycles have been completed in areas such as human factors & human reliability, nuclear safety analysis, materials research, probabilistic safety assessment (PSA), severe accident management. Each research programme has addressed current nuclear safety issues, prepared topical state-of-the-art reviews, and developed strategies to promote proactive plant safety management.

This report summarizes results from one of the topic areas (LOCA frequencies) in NKS/RAK-1. The entire study is summarized in the main report (Andersson,1998).

1.1 NKS/RAK-1: Strategies for Reactor Safety

The NKS/RAK-1 project formed one element in the most recent four-year Nordic nuclear research program (1994-1997). The overall objective of the NKS/RAK-1 was to explore strategies for nuclear reactor safety; *c.f.* Table 1-1. Task 2 -- hereafter referred to as NKS/RAK-1.2 -- dealt with the loss-of-coolant-accident (LOCA) and its modeling in the context of PSA given today's state-of-knowledge about degradation mechanisms and the service experience with primary system piping.

Table 1-1: *Overview of the Tasks Within NKS/RAK-1.*

TASK	THEME	RESEARCH PROBLEM
RAK-1.1	Survey of the safety work at nuclear installations.	How can we assess the suitability and effectiveness of the safety work?
RAK-1.2 (This Report)	**LOCA frequencies**	Can we improve WASH-1400 values, and what are the LOCA risk dominant mechanisms?
RAK-1.3	Integrated sequence analysis	How should complex event sequences be analysed with new approaches integrating different disciplines?
RAK-1.4	Maintenance strategies	How can one optimize maintenance and testing?
RAK-1.5	Plant modernization	How can we reasonably meet modern safety standards?

Proposed by Nordic utility organizations, the NKS/RAK-1.2 was initiated to reevaluate the PSA practice regarding LOCA frequency estimation. In PSA, the technical approach to analysis of LOCA has mainly followed the seminal PSA pilot study, the WASH-1400, which was completed in 1975. Since that time, extensive service experience with primary system piping has accumulated. Additionally, based on insights and results from the analysis of occurred cracks, leaks and ruptures in primary and secondary side piping systems, significant advances have been made in the understanding of piping degradation mechanisms. Similarly, advances have been made in the maintenance and surveillance methods to more effectively combat aging.

Rather than adopting a formal system modeling approach or a structural reliability analysis (SRA) approach, WASH-1400 applied a combination of expert judgment and zero failure statistics. In view of today's state-of-knowledge/practice, PSA practitioners increasingly have questioned the WASH-1400 approach to LOCA frequency estimation. A current view on the estimation process at large includes a recognition of the importance of more fully exploring and exploiting the discriminatory power of PSA. That is, the estimation process should include the ability to discriminate between unique plant design and operational features. Relative to LOCA frequency estimation, a chosen approach should have the ability to discriminate between, say, primary system designs for external pump BWR versus internal pump BWR, between industrial grade and nuclear grade austenitic stainless steels, between plants with and without hydrogen water chemistry (HWC), etc. Finally, the chosen approach should have the capability to incorporate insights from detailed evaluations of the service experience.

Modern PSA applications rely on validated plant models of sufficient discrimination to support *safety improvements* and *safety optimization*. As part of plant life extension programs, older nuclear power plants are subjected to primary piping replacements. The living PSA concept encompasses the re-evaluation of LOCA frequency estimates in support of such primary piping system replacement projects. Also, PSA-based optimization of the in-service inspection (ISI) practices has emerged as a major candidate for PSA applications. Such optimization is concerned with methods that permit the examination of the impacts of changes in the ISI program on the pipe rupture risks and overall plant risk. Against this background the NKS/RAK-1.2 project was formulated to address the following technical issues:

- **Issue 1**. Incorporation of detailed probabilistic modeling of pipe ruptures in PSA. Examination of the feasibility of such modeling in view of the analytical efforts to achieve realism relative to the service experience, plant-specific piping system designs and ISI practice.

- **Issue 2**. Justification of detailed probabilistic modeling. In view of the relative contributions of LOCAs to the overall plant risk, can detailed modeling of the piping systems be justified? If the answer is yes, what would be a recommended analysis practice?

- **Issue 3**. Integration of PSA methods and material sciences. How should the methods for integrating PSA with basic sciences be pursued? In the area of rare events (such as pipe ruptures), there has been a long-standing debate on how to perform failure parameter estimation. There are opposing views regarding the fundamental modeling issues, as well as regarding the application of basic probabilistic and statistical concepts. Given the irreconcilable views on the reliability analysis of passive components, such as piping, how should PSA practitioners make the most effective use of established systems reliability analysis concepts and probabilistic fracture mechanics (PFM)?

- **Issue 4**. LOCA frequency estimation and service experience. In view of the service experience with primary piping systems worldwide, is direct parameter estimation feasible? Should direct estimation of pipe rupture frequencies implicitly be viewed as an analytical 'short-cut' alternative to PFM?

The principal result of NKS/RAK-1.2 has been the development of a user-friendly, PC-based program for estimating the probability of rupture in primary system piping, which is susceptible to intergranular stress corrosion cracking (IGSCC); *c.f.* Bergman, Brickstad and Nilsson (1997 and 1998). This program -- PIFRAP (**PI**pe **FRA**cture **P**robabilities) -- was developed by the Swedish control agency SAQ Kontroll AB with the specific objective of responding to the analytical requirements of utility engineers.

1.2 R&D on Piping Reliability by SKI

Independent of NKS/RAK-1.2, the Swedish Nuclear Power Inspectorate (SKI) in 1994 initiated a 4-year R&D project on piping reliability; *c.f.* Nyman et al (1997). The technical scope included the development of an analysis framework for the estimation of piping reliability parameters from service data. Based on systematic analysis of the service experience with piping systems in nuclear power plants worldwide, the SKI-sponsored project was prompted by requirements for an integrated, 'data-driven' analysis approach to support PSA applications. The project responded to one basic PSA-oriented question: *Does the worldwide industry experience with leaks and ruptures change the consensus perception of small-, medium- and large LOCAs?*

A consolidation of insights and results from NKS/RAK-1.2 and the parallel SKI-sponsored project was pursued through an international seminar on piping reliability; *c.f.* SKI (1997). From the Nordic point of view, the merging of insights from the two projects has broadened the analytical perspectives on piping reliability analysis as it applies to LOCA frequency estimation and PSA applications. There is no single one method of estimating piping reliability. The merging of data-driven models and SRA models holds the greatest promise in supporting the development of advanced piping reliability models.

In the PSA applications of piping reliability models, the term 'advanced' refers to piping system models, which are developed using detailed consideration of degradation and failure mechanisms of individual piping system components (e.g., welds, tees, bends/elbows). Additionally, the reliability of individual components is evaluated on basis of the potential consequences (e.g., direct and indirect impact on plant operation) of a rupture. A direct failure would be the loss of a train/system or an initiating event (LOCA). An indirect failure would be the potential spatial effects (e.g., pipe whips) acting upon equipment located in the vicinity of a pipe rupture.

1.3 Outline of This Report

This final report on the NKS/RAK-1.2 summarizes the main features of the PIFRAP PC-program and its intended implementation. Regardless of the preferred technical approach to LOCA frequency estimation, the analysis approach must include recognition of the following technical issues: a) Degradation and failure mechanisms potentially affecting piping systems within the reactor coolant pressure boundary (RCPB) and the potential consequences; b) In-service inspection practices and how they influence piping reliability; and c) The service experience with piping systems.

The report consists of six sections and one appendix. Section 2 is a Nordic perspective on LOCA and nuclear safety. It includes summaries of results from research in material sciences and current regulatory philosophies regarding piping reliability. Section 3 is a summary of the LOCA concept as applied in Nordic PSA studies. It includes a discussion on deterministic and probabilistic views on LOCA. The R&D on piping reliability by SKI is summarized in Section 4, while the PIFRAP model is summarized in Section 5. Next, Section 6 presents conclusions and recommendations. Finally, Appendix A contains a list of abbreviations and acronyms, together with a glossary of technical terms.

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2. NORDIC PERSPECTIVE ON LOCA & NUCLEAR SAFETY

Probabilistic safety assessment (PSA) is an integral activity of the Nordic nuclear safety work. Requirements for plant-specific PSA were implemented by regulatory bodies in Finland and Sweden during the early 1980's. The current Nordic approach emphasizes the practical use of PSA in support of decisions regarding aging management, plant life extension, event analysis, etc. This section summarizes Nordic perspectives on the roles of PSA and structural reliability analysis (SRA) in addressing the reliability of piping systems. Insights from the most recently completed research by the Nordic program point to the need for, and practicality of closer interactions between PSA and the material sciences in the area of piping reliability as applied to LOCA frequency estimation and risk-informed inservice inspection (RISI).

2.1 Insights from Nordic PSA Programs

Six perspectives on the evolving Nordic PSA programs are found in VTT (1973), Pershagen (1989), Hirschberg (1990), Laaksonen and Virolainen (1991), Werner et al (1995), Knochenhauer (1996). Plant-specific PSA studies of varying scope exist for the sixteen operating nuclear power plants (NPPs) in Finland (4 NPPs) and Sweden (12 NPPs). For these plants, the degree of integration between Level 1, Level 2, Level 3 and low power and shutdown PSA study phases differ, however. During the 1980s, and in response to post-TMI and post-Chernobyl safety concerns, significant R&D programs were established in Finland and Sweden to address severe accident management (SAM). Specifically, the containment performance under degraded core conditions, source term issues, and human factors issues were studied extensively as part of these SAM-programs. While the actual degree of PSA integration differs among the Nordic plant-specific PSAs, all the key technical elements for state-of-the-art, full-scope and integrated PSA exist for the sixteen operating NPPs.

At the organizational level of PSA, the Nordic programs have always been based on extensive cooperation. The utility organizations have exchanged technical information, results and study insights on a continuing basis. Similarly, the regulatory bodies have participated in the PSA projects by providing support in the areas of equipment reliability database development, R&D on common cause failure analysis, and PSA software development. There is, by-and-large, a consensus-approach to PSA via the results of the Nordic nuclear safety R&D.

Plant risk is dynamic and changes with equipment and system availability, and human performance. From the outset, the concept of living PSA was embraced by the regulatory requirements in both Finland and Sweden. Some of the utility organizations perform annual updates of the PSA studies following the completion of the annual refueling outages. This way the system models reflect the current service experience and operating practices, and the as-built and as-operated plant systems.

There are at least three study generations -- i.e., original base-line study plus subsequent updates -- for each of the Nordic NPPs. Based on PSA results and insights, the Nordic NPPs compare favorably with other Western European and U.S. plants. New service experience and associated safety concerns emphasize the importance and value of PSA programs that are well integrated with plant safety management organizations. A snap-shot of some results from Finnish PSAs and Swedish PSAs are given in Tables 2-1 and 2-2, respectively.

Table 2-1: *Some Finnish PSA Results; c.f. Laaksonen and Virolainen (1991).*

TVO-1/2 PSA (Internal Recirc. Pump BWR)		**Loviisa-1/2 PSA (WWER)**	
Initiating Event Category	**Total CDF Contribution**	**Initiating Event Category**	**Total CDF Contribution**
Loss of Condenser	53%	Single + Multiple SGTR	26.4%
Loss of Offsite Power (LOSP)	19%	Loss of Offsite Power	21.1%
Small LOCA	17%	SG Collector Break	11.6%
Loss of Main Feedwater	8%	Partial Loss of Service Water	8.2%
Medium LOCA	4%	Small LOCA	6.8%
Large LOCA	0.09%	Medium LOCA	5.5%

Table 2-2: *Some Swedish PSA Results.*

Ringhals-1 PSA (External Recirc. Pump BWR) ***Source:*** **PT-80/92 (May 1992)**		**Ringhals-2 PSA (3-Loop Westinghouse PWR)** ***Source:*** **RX-PKH-R2 (May 1987)**	
Initiating Event Category	**Total CDF Contribution**	**Initiating Event Category**	**Total CDF Contribution**
Medium LOCA, bottom break	42%	Small LOCA	43%
Large LOCA, bottom break	15%	Large LOCA	21%
Loss of Offsite Power	11%	SGTR (Single Tube)	18%
Medium LOCA, top break	10%	Medium LOCA	5%
Large LOCA, top break	4%	Interfacing Systems LOCA	0.8%
Loss of Main Feedwater	3%	Loss of Offsite Power	0.5%
Small LOCA	1%		

In PSA, the analytical direction and emphasis is a function of the known or perceived importance of the various initiating event categories. Whether the PSA study objectives are retrospective or prospective, there is at least one more important PSA management consideration which determines the analytical emphasis: The model discrimination necessary to support current and future application requirements impacts the breadth and depth of the modeling efforts. In the context of PSA, the term 'model discrimination' refers to the level of detail afforded a system model and whether equipment failures and human failure events are described through decomposition or holistic models; *c.f.* Bodsberg (1993). The degree of achieved model discrimination could be a measure of completeness, and is indicative of the feasibility of PSA model exploitation and exploration in support of decisions about plant safety.

The seminal WASH-1400 concluded that NPP risk largely is driven by transient initiating events and small LOCAs, and subsequent international PSA programs emphasized these initiating event categories. Why, then, should the new PSA programs be concerned with structural reliability, and medium and large LOCAs, and would our understanding of plant risk be altered by re-qualifying the medium- and large-LOCA frequencies of WASH-1400?

Consistent with the international practice, the Nordic PSA studies have, in general, not included detailed analyses of primary piping system failures as part of the modeling of medium- and large LOCAs. Typically, the passive components have been excluded from explicit modeling using decomposition approaches. The argument for doing so has been the demonstrated high reliability of primary system piping, lack of service experience with significant piping failures, and lack of recognized models of piping reliability that are compatible with the PSA methodology. As the NPPs are getting older, a critical evaluation of the 'aged' WASH-1400 practice is needed, however.

Today's Nordic perspective on PSA and piping reliability is driven by insights from evaluations of the service experience (including inservice inspection, ISI of Class 1,2 and 3 systems and components), and the recognition of the immense potential benefit of performing PSA applications directed at optimizing the ISI programs. A re-direction of the analytical emphasis regarding LOCAs has resulted in new requirements on modeling of piping systems and data on piping failures. The absolute values of derived LOCA frequencies are less important than the way by which the modeling is performed, and how the modeling would support practical PSA applications, however.

2.2 Service Experience & Results from R&D in Material Sciences

The service experience with primary system piping in Nordic plants has paralleled the international service experience. Crack growth through stress corrosion cracking (SCC) and thermal fatigue has occurred in localized sections of reactor coolant pressure boundary (RCPB) piping.

Through-wall cracks in affected piping have resulted in minor leaks during normal operation. In all instances, there has been ample warning time to allow for mitigation by isolating affected pipe sections, providing make-up if necessary, and cooling down and depressurizing the primary system. The applicability of the leak-before-break (LBB) concept has been demonstrated through the service experience. Relative to medium- and large-diameter piping, Table 2-2 summarizes key features of the two primary RCPB piping degradation mechanisms.

Table 2-2: *Factors Affecting SCC & Thermal Fatigue; c.f. Simola and Koski (1997).*

Degradation Mechanism	**Loading Type**	**Environmental Conditions**	**Materials Affected**
Thermal fatigue	Cyclic	Temperature Pressure	Austenitic steels Ferritic steels Alloys
Stress corrosion cracking: IGSCC (TGSCC)	Steady load, residual stress	Medium (e.g., chlorides in the case of TGSCC) Temperature Oxygen pH	Austenitic steels

Austenitic steels generally are used as materials for the piping in the BWRs. In the older, external-pump units, compound materials are used in the recirculation loops; i.e., ferritic steels form the primary pressure boundary with austenitic cladding material on the inside surface. Some primary system fittings (e.g., elbows) for these older plants were manufactured from cast austenitic steels (Trolle, 1996). The type (i.e., grade) of steels used for piping and fittings is equivalent to AISI Type 304 with restrictions on carbon content (C ≤ 0.05%). For the newest, internal pump BWRs (Forsmark-3 and Oskarshamn-3), low carbon content materials was used for piping with service temperatures exceeding 100°C. To some extent, stabilized steels have been used; e.g., the titanium stabilized steels. As a practice, when piping replacement is performed, the first choice of replacement material is a modified version of the AISI Type 316 NG steel; i.e., IGSCC-resistant steel per Swedish Standard (SS 2353 with C ≤ 0.02%).

In total, during the period 1979-1997, more than 100 incidents involving IGSCC have occurred in Swedish BWR piping systems. The incidence rate at the two Finnish internal-pump BWRs has been substantially lower. A first incident of intergranular stress corrosion cracking (IGSCC) in a weld heat affected zone (HAZ) in a Swedish NPP occurred at Ringhals-1 in 1979; *c.f.* Jansson (1996). It was a through-wall crack in a connecting DN32 nozzle to a control rod drive housing. The subsequent metallographic evaluation confirmed IGSCC as the degradation mechanism. The carbon content in the affected fitting was 0.038%.

A first incident of IGSCC in base metal in a Swedish NPP occurred at Oskarshamn-1 in 1979; *c.f.* Jansson (1996). The damaged elbow was in a section of DN100 residual heat removal (RHR) system piping. In this particular case the crack propagation had been initiated through transgranular stress corrosion cracking (TGSCC) in surface scratches caused by the bending tool during fabrication. Upon initiation, the crack had propagated in the intergranular mode. Subsequent metallographic evaluations revealed extremely high cold deformation in the affected area and 8-12% martensite was measured. During the fall 1979 and winter 1980, additional through-wall cracking and leaks were detected in elbow sections of the reactor water cleanup (RWCU) system piping at 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 the equivalent piping for Barsebäck-1/2 and Ringhals-1.

Yet another notable incident of IGSCC in base metal was revealed in 1993 during a containment walk-through in preparation for the start-up of Oskarshamn-1. During the refueling and maintenance outage 1992/93, welds and heat affected zones of IGSCC-susceptible piping had been subjected to 100% volumetric testing in accordance with the Swedish regulations. Visual detection of a pinhole leak during the containment walk-through in preparation for plant start-up, rather than the ISI, revealed significant cracking in a bend of DN80 piping of the RHR/RWCU piping, however. This pinhole leak (or seepage) had developed in a 53-degree, cold bent pipe section connected to the bottom of the reactor pressure vessel. Subsequent event evaluations by the utility and the regulatory body raised renewed concerns about the effectiveness of ISI and the feasibility of 100% volumetric testing. The ISI of RCPB piping is a highly complicated undertaking influenced by human factors (e.g., accessibility, radiation exposure, proficiency of inspectors) and inspection technology.

Early indications of a generic thermal fatigue cracking problem were identified in 1978 in Oskarshamn-2 and Barsebäck-1. A significant event occurred in TVO-1 in 1979; *c.f.* Holmberg and Pyy (1993). A DN150 T-joint of the RWCU developed a fracture and about 5 m^3 (5,000 kg) of water was discharged into the Reactor Building. The pipe fracture area was approximately 2.3 cm^2 (150 mm long and maximum of 2 mm wide fracture); a small-LOCA precursor. The direct cause of the rupture was cyclic thermal stratification. The fatigue crack propagation was the result of the mixing of two process streams at different temperatures ($\Delta T \sim 50°C$). Within a period of about 12 months, extensive thermal fatigue cracking in T-joints of the RWCU system was also discovered at TVO-2 and Barsebäck-2 (maximum crack depth of 10 mm); *c.f.* Nordgren (1983). A design change, which included the use of thermal mixers to prevent the RWCU pipe walls from exposures to unacceptable temperature fluctuation, has proven effective in minimizing the particular form of thermal fatigue damage in the Nordic BWRs.

IGSCC is an example of environmental cracking - a generic term that includes various environmentally assisted cracking phenomena. It is a time-dependent phenomenon, which occurs when certain metallurgical (e.g., sensitization), mechanical (e.g., tensile stresses) and environmental conditions exist simultaneously; *c.f.* Aaltonen, Saarinen and Simola (1993). Sensitization of the stainless steel takes place when Cr-depletion areas are formed near the grain boundaries due to precipitation of Cr-rich carbides during welding and due to growth of the carbides during the plant operation. The tensile stresses in HAZ might increase locally due to the increase of the weld residual stresses as well as due to stresses from operation, fabrication and fit-up loadings. In the high-purity BWR primary water, small amounts of dissolved oxygen after start-ups and other oxidizing species, such as H_2O_2, during normal operation provide the environment that supports IGSCC.

Modifications to water chemistry, tensile stresses or metallurgy impacts the IGSCC-susceptibility. The IGSCC counter measures in the Nordic countries have included the consideration of all three conditions for IGSCC.

One remedy in addressing water chemistry is hydrogen water chemistry (HWC). Hydrogen injected in the feedwater reacts with oxygen in the radiation field in the downcomers of the reactor pressure vessel. HWC reduces the oxidation power of the primary water and decreases the corrosion potential for stainless steel below the critical potential for IGSCC. The oxygen concentration is decreased from about 200 parts per billion (ppb) in normal water chemistry (NWC) to less than 1-2 ppb in HWC. Finding the optimum HWC-strategy is a complex undertaking. The effectiveness of HWC is affected by several factors such as the concentration of chlorides and sulfates in reactor water, the reliability of condensate purification, etc. In Sweden, HWC strategies were implemented selectively during 1988. There is a significant variation in the IGSCC incidence frequency among the Nordic BWR plants during 1988-1997, a variation that can be attributed to differences in HWC strategy as well as other plant operational conditions. Potential negative side-effects of HWC include increased erosion-corrosion propensity in secondary side piping, inadvertent turbine trips caused by increased radiation levels in steam piping following plant startup.

Different stress remedies include: heat sink welding (HSW), last pass HSW, induction heat stress improvement (IHSI), weld overlay repair (WOR), and mechanical stress improvement (MSIP). The IHSI and WOR are commonly used where cracking has not yet occurred, or where the crack depth is still shallow in the through-wall direction. When IGSCC is too deep, the crack tip will be situated within the tensile stress zone. In such case, IHSI or MSIP would greatly enhance the crack growth and would therefore be unsuitable. For piping components with such deep cracks in the through-wall direction the application of WOR is required. The WOR method was authorized for use in Sweden in 1987. The MSIP method has been tried on one weld in Oskarshamn during the 1989 annual refueling outage.

2.3 Service Experience & Requirements for PSA Applications

The service experience with piping systems has raised questions about the traditional approach to the estimation of LOCA frequencies. Should this approach be modified to more adequately reflect the available service experience (including compensatory measures to mitigate or eliminate specific degradation mechanisms), and would the modified approach be amenable to applications involving the definition of inspection locations for safety significant piping in NPPs? What analytical techniques should be applied to the quantification of an advanced LOCA model? The specific technical issues include:

- **Detailed modeling of LOCA in the PSAs**. An old PSA convention has been to model LOCA events as single basic events. The service experience and new PSA application needs point to the need for a modified PSA convention by developing detailed, systems-oriented models of the primary system piping. Such models should recognize the applicable service experience by accounting for occurred piping degradations and failures, *and* the effectiveness of ISI. An advanced LOCA modeling concept should discriminate between the plant-specific piping system design (e.g., method of fabrication, metallurgy, geometry, type and number of fittings) and the observed degradation mechanisms. Enhanced model discrimination would result from a section-by-section (or component-by-component) representation of piping systems.

- **Effectiveness of ISI**. The recent 1995 event at Oskarshamn-1 addressed the complex nature of performing volumetric testing of IGSCC-susceptible piping inside the containment. Factors such as the access to specific inspection locations, and the radiation exposure from the inspection locations influence the effectiveness of ISI. Furthermore, the reliability of the inspection methods influence the likelihood of detecting cracks within the inspection locations.

- **Plant-specific PSA and RISI**. Within the international nuclear safety R&D community, the topic of RISI is under extensive exploration; *c.f.* ASME (1991 and EPRI (1996). The philosophy of RISI encompasses the application of detailed LOCA models to identify inspection locations based on their relative risk contribution. Rather than, say, 100% volumetric testing of IGSCC-susceptible piping, the RISI approach could reduce the examination scope while maintaining, or possibly improving plant safety.

Each of the above technical issues have been addressed to some degree by earlier Nordic programs. During the 1981-1984 research program, the issue of pipe rupture frequency estimation using service experience was studied by Risø National Laboratory; *c.f.* Lauridsen (1982). With funding from the Swedish Nuclear Power Inspectorate, a simple model for the estimation of the pipe rupture probability due IGSCC was developed during 1987-1989; *c.f.* Nilsson, Brickstad and Skånberg (1989). This particular model was an application of probabilistic fracture mechanics, and it included the consideration of the effects on ISI on pipe rupture probability.

Initial steps to develop an advanced LOCA model building on the piping component-by-component concept were taken by the Swedish utility OKG AB in 1993. This model development was motivated by PSA application requirements that included the consideration of RISI to optimize the selection of locations for ISI. The NKS/RAK-1.2 programme represents a consolidation of the Nordic work on the integrity of RCPB piping in the context of PSA. Sections 3 through 5 of this report summarize views on LOCA modeling and applications of advanced LOCA models.

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3. LOCA CLASSIFICATION IN TRADITIONAL SAFETY ANALYSIS & PSA

The definition of LOCA categories for PSA is based on a combination of qualitative and analytical concepts. Mostly, the definition is largely historical, resulting from the design basis accident (DBA) concept of the deterministic safety analysis philosophy, however. In this philosophy, a large LOCA is a postulated large pipe break in the main coolant system and of such magnitude that the capacity of the make-up systems is insufficient to replace the lost coolant.

In this section we review the different approaches to the definition of LOCA categories by deterministic safety analysis and PSA, respectively. The presentation addresses the one PSA-oriented question: Does the worldwide industry experience with leaks and ruptures in piping systems change the consensus perception of small-, medium- and large LOCAs? An underlying question is this: To support decisions about plant safety, how should today's PSA studies and the future PSA applications model the potential pipe rupture events that could contribute to plant risk?

3.1 Overview of Basic Considerations

Adopted from published work by SKI (Tomic et al, 1996), this section summarizes basic PSA-oriented considerations in defining LOCA categories. The possible LOCA situations vary from primary coolant losses of a few kg/hour (i.e., at or below the leak detection threshold) to major losses, e.g. more than or much more than, say, 5 kg/s. At the lower end of this spectrum of coolant losses, a small primary coolant leak would be indicative of a trend which, if left unattended, could result in a significant plant disturbance. At the upper end, a very large coolant loss would result in the actuation of the engineered safeguard features. For practical reasons, in PSA the potential LOCA events are grouped into a manageable set of categories.

In the categorization according to design principles, plant response arguments determine how a LOCA is classified. In essence, the LOCAs are categorized according to the number of trains of emergency core cooling needed to accomplish the safety function, the approximate timing of critical evolutions in an accident scenario, the ability of control room operators to initiate manual back-up actions in case of failed auto-initiation of a safety function, etc.

Typically, a small LOCA is an event where the high pressure safety injection system (or equivalent system) is required to maintain coolant inventory, but the heat removal through the cracked or ruptured pipe would not be sufficient to remove decay heat. Therefore, the secondary side cooling is still needed, placing additional requirements on both safety and support systems. Typically, such an event is terminated before the need for recirculation of the primary water which has accumulated in the containment sump. In some PSA studies, the 'small-small' LOCA is explicitly analyzed by considering scenarios where a regular (i.e., non-safety grade) make-up system could cope with the loss of primary coolant and maintain the inventory. The plant is shut down and cooled using secondary side cooling.

A medium LOCA is typically an event where the high pressure safety injection system(s) is needed early in the event sequence. But to ensure long-term heat removal, low pressure safety systems are needed later on. The energy released via a cracked/ruptured pipe could be sufficient to remove the heat from the core, thus minimizing the need for the secondary side cooling. Ultimately, there would be a requirement for long-term recirculation cooling, however. A large LOCA is an event where the loss of primary water inventory is so rapid that the active and passive low pressure systems need to be initiated immediately to prevent the overheating of the core. If successful, injection of water from the low pressure systems would be sufficient to remove the decay heat and to prevent overheating of the core. In addition, there is a requirement for recirculation cooling in the short-term.

In PSA, the indicated LOCA events represent the categories as defined in their most basic forms. Designers, operators and PSA practitioners alike do recognize that within the three basic categories there may be distinct sub-categories. These are not only dependent on the break sizes, but also on specific locations, detectability, means of leak isolation, consequential effect(s), etc. In summary, the following approaches to LOCA categorization have been applied in PSA:

- **Categorization on the basis of pipe diameter**. This approach assumes that the predominant failure mode is a rupture with at least a single-side unrestricted flow. According to available service experience the frequency of rupture is much less than the frequency of leaks or through-wall cracks. Therefore, categorization on the basis of diameter leads to conservative LOCA evaluations.

- **Categorization on the basis of flow area**. The flow area is defined as the area through which the reactor coolant is lost. An advantage of the approach is its gradation of pipe failure. That is, a large opening in a small-diameter pipe could have the same effect on operations as a small opening in a large-diameter pipe.

- **Categorization on the basis of leak rate**. This is the most logical approach since the loss of inventory is the actual parameter of interest in LOCA evaluations. Not only does a leak rate calculation take into account the flow area, it also considers the pressure differential, flow restrictions, etc.

Although the leak rate definition is the most appropriate basis for LOCA categorization, it has not been widely used in PSA. An argument against using leak rate calculations has been the complexity of such analyses. New analytical tools (*c.f.* Grebner, 1995) allow for more realistic LOCA evaluations, however.

3.2 LOCA and Traditional Safety Analysis

Traditional (or deterministic) safety analysis is the study of selected LOCA events (and transients) using calculation models which provide the time history of essential plant variables after the initiating event. The purpose is to verify the safety design, to show that the licensing requirements are fulfilled, and to make realistic safety assessments for actual or anticipated events. Essential variables include fuel cladding temperature, control rod surface heat flux, reactor pressure, as well as temperature and pressure in the reactor containment. Thermal hydraulic calculation models are set up based on mass, energy and momentum balance.

A LOCA in a BWR is initiated by a leak or rupture in the primary system. A distinction is made between leaks/ruptures above and below the upper edge of the core ('top breaks' and 'bottom breaks', respectively) as well as between large and small breaks. In external pump reactors, a break in a DN650 main recirculation line connected to the bottom of the reactor vessel constitutes a large LOCA design basis accident. The initial break flow is estimated at 20,000 kg/s. In BWRs with internal recirculation pumps an assumed break in a main steam line inside the reactor containment is representative of a large LOCA. Typically, the deterministic analysis of LOCA in PWRs differentiate between large LOCAs, which are characterized by a break flow area corresponding to a rupture in piping of at least DN250, medium LOCA (DN80 - DN250) and small LOCA (DN10 - DN 80).

3.3 LOCA in PSA

PSA is a complement to traditional safety analysis. Farmer (1979) has defined the basic philosophy behind PSA in approximately the following way: The PSA approach requires the use of historic data (e.g., service experience) *and* predictive techniques to arrive at a spectrum of risks versus consequences. It emphasizes the evaluation of the spectrum of possible incident scenarios using probabilistic techniques by including uncertainties. A leading motivation for supplementing deterministic analyses with probabilistic analyses is the need for realism. Rather than focusing on incredulous events (e.g., the double-ended guillotine pipe rupture of a main coolant line), PSA includes the range of events, from the credulous to the incredulous by developing in-depth '*what-if*' assessments.

Against the background of the Farmer philosophy, how well/poorly have the completed plant-specific PSA studies addressed the highly complex group of LOCA initiating events? Rather than developing detailed models of the initiating event possibilities, the majority of studies have followed the convention established by WASH-1400. Hence, the analyses have emphasized the responses by engineered safety features and control room operators *given* a LOCA of certain size and location. Instead of high-discrimination, probabilistic models of the initiator, that are directly compatible with the overall PSA architecture, the categorization of LOCA mostly follows some qualitative grouping criteria. Such criteria consider the systems that can be used to provide cooling following a leak of a given size and a given location.

Extracted from the Surry IPE (Virginia Power, 1991), the definition of LOCA classes inside and outside containment was done as follows:

- **LOCA Inside Containment**. All components that could cause a LOCA inside containment were identified to determine the range of possible break sizes. The identification also included individual pipe segments, so that potentially a range of methods of assessing the frequencies of LOCAs, such as those based on counting the number of pipe sections, could be utilized; *c.f.* Table 3-1. Ultimately, LOCAs within the capacity of the injection systems were grouped into three categories; large, medium and small. The lower bound for a small LOCA was a break in a DN10 pipe. A Surry plant-specific analysis showed that a breach in the primary system boundary equivalent to a pipe size of DN10 is large enough to eventually lead to the generation of a Safety Injection signal.

- **LOCA Outside Containment**. Interfacing system LOCA (the V-sequence) can be the result of the failure of the closed valves representing the interface between high and low pressure systems (pressure interface valves). Or, interfacing LOCA can result from the failure of high pressure piping outside containment and subsequent failure of isolation valves to close. Table 3-2 summarizes V-sequence locations considered in the Surry IPE.

Table 3-1: *LOCA Components and Range of Equivalent Break Sizes - PWR.*

Component	Size Range (Nominal Diameter)
RCS & Connected Piping	DN10 - DN775
RCS Loop Stop Valves	DN675 - DN725
Steam Generator (S/G) Tubes (treated as a separate initiating event)	DN20
S/G Manways (RCS Side)	DN400
Pressurizer (Pzr) Main & Auxiliary Sprays	DN50 - DN100
Pzr Relief & Safety Valves Piping	DN50 - DN150
Reactor Coolant Pump Seals	0 - DN25

Table 3-2: *Some V-Sequence Locations Considered in Surry-IPE.*

Location	Size Range (Nominal Diameter)
Hot Leg Safety Injection (Loop A/B/C)	DN75 - DN250
Cold Leg Safety Injection (Loop A/B/C)	DN150
Pressurizer Spray Drain Line (Loop C)	DN38
Auxiliary Spray Line	
Charging Line (Loop B)	
Excess Letdown Line (Loop C)	
Loop Fill Line (Loop A/B/C)	

Relative to small LOCA scenarios, reviews of available service experience indicate that the PSA studies have provided realistic assessments; *c.f.* Shah et al (1997). In fact, the service experience has always been the basis for small LOCA frequency calculations (Gentillion et al, 1994). For medium and large LOCAs, the frequency estimation has to be based on analytical models.

In the spirit of the basic PSA philosophy, the analysis of medium and large LOCAs always should include the detailed evaluation of applicable service experience with piping. Such an elaborate undertaking must be supplemented by SRA models to ensure correct interpretations of the implications of incipient and degraded piping failures.

3.4 References

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4. SERVICE DATA & LOCA FREQUENCY ESTIMATION

In PSA, the approach to the estimation of loss-of-coolant-accident (LOCA) and inter-system LOCA (ISLOCA) frequency has mainly followed the seminal PSA pilot study, the WASH-1400. Historically, the LOCA was an important consideration in reactor safety analysis and licensing. WASH-1400 demonstrated the large LOCA caused by a double-ended break of the largest main coolant pipe to be less important than transients, transient-induced small LOCAs, and small LOCAs, however. Subsequent PSA studies have often validated the selection of LOCA frequencies solely by referencing the WASH-1400.

Increasingly, PSA applications are performed to support evaluations of modified primary system piping designs, aging management, and definition of enhanced strategies for in-service inspection (ISI). With the new requirements for PSA applications have followed a need for an improved piping reliability analysis methodology. A methodology which incorporates service experience, methods and techniques of PSA, and results from structural reliability analysis (SRA). The estimation of LOCA frequencies based on service experience is discussed in the following; *c.f.* Nyman et al (1997).

4.1 The Need for Change

The maturity of PSA is reflected in current code cases, regulatory guidance documents (e.g., NRC 1997a, 1997b), and guidance documents for PSA applications (e.g., EPRI 1995). Over the years, PSA practitioners have devoted considerable efforts to quality PSA, peer review, validation and verification, etc. Some technical aspects in today's PSA technology remain in their infancy, however. As examples, aging risk assessment and analysis of the reliability of structures such as vessels and piping are not supported by models, data or analysis procedures that are commensurate with the expectations on the accomplishments by PSA.

From the PSA perspective, piping reliability analysis remains largely influenced by the WASH-1400 study. A study, which reflected the late 1960's and early 1970's state-of-knowledge about nuclear safety, systems reliability, data, etc. The reliability of primary system piping remains a controversial issue.

In January 1975, during a special inspection, cracks were found in the high-pressure piping of the emergency core cooling system (ECCS) of the Dresden-2 BWR plant in the U.S. The reactor had been shut down for the inspection. The observed cracks penetrated the full wall thickness of the piping, as evidenced by primary water leakage that was found when the pipe insulation was removed. The cause of the cracking was later determined to be intergranular stress corrosion cracking (IGSCC). Events like this raised questions about the applicability of the approach to LOCA frequency estimation by WASH-1400. Did that study consider all potential degradation mechanisms in sufficient detail? The stage was set for the development of a methodology for interpretation and analysis of service experience to be input to PSA models.

Based on the currently available service experience, the WASH-1400 assumption about the high reliability of piping cannot be disproved. Those early piping reliability estimates were based on approximately 150 reactor-years of service experience with piping systems forming the reactor coolant pressure boundary (RCPB). Now, twenty-five years later and with over 8,500 reactor-years of service experience, we have not yet witnessed a large- or medium-size LOCA as a result of a double-ended-guillotine-break (DEGB) of primary system piping. Hence, it could be argued that the importance of pipe leaks or ruptures relative to plant risk remains negligible. Especially when viewed against the many other, current safety concerns; e.g., human factors and human reliability, organizational factors, software reliability. Problem is, our views on PSA and PSA applications have evolved in a major way since the mid-1970's. Invariably, PSA is seen as a key safety management tool -- a tool that should enable plant engineers and regulators to turn up the microscope on the PSA models, the data and the results to evaluate the significance of very subtle (e.g., precursor events) to major equipment degradations.

During the past twenty-five years, numerous R&D projects have been pursued to improve the technical basis for estimating piping reliability from service experience. While recognizing the role of probabilistic fracture mechanics (PFM), these projects have largely been performed to generate alternative, more cost-effective analyses of piping reliability. A review of these past R&D projects reveals that: a) The proposed, data-driven methodologies have been limited to direct estimation from counts of pipe rupture events; and b) The scope of the surveys of service experience has been limited.

New methodologies for piping reliability need to be broadened by including considerations of root cause analysis, PFM, and deeper analysis of the available service experience. Old concepts, based on variations of WASH-1400, are not fit for today's PSA applications. As has been stated by the American Society of Mechanical engineers (Balkey et al 1992), "*... the task of estimating piping reliability is complex, uncertain and costly ...*" Therefore, it is essential that the new PSA application programs are supported by sufficiently developed databases and methods of piping reliability.

4.2 SKI's R&D on Piping Reliability

During 1994-1997, the Swedish Nuclear Power Inspectorate (SKI) funded a project to develop a PSA-oriented database on the service experience with piping systems in nuclear power plants worldwide. The database covers the period 1970 to the present. In developing the database, the scope of the work emphasized failures in light water reactors (LWR). Currently the database includes well over 2,500 failure records; *c.f.* Table 4-1. Failures of piping within the RCPB, balance-of-plant (BOP) and support systems were considered. The work also included the development of a framework for interpreting and analyzing the service experience with the objective of deriving statistical failure parameter estimates.

Table 4-1: *The Content of SKI's Database on Piping Failures.*

PLANT TYPE[a]	NUMBER OF PLANTS SURVEYED	DATABASE COVERAGE [Reactor-Years]	FAILURE MODE		
			Crack[b]	Leak	Rupture[c]
BWR	72	1,479.7	199	663	66
LWGR	19	333.6	4	43	14
PHWR (CANDU)	20	367	11	75	14
PWR+WWER	167	2,827.4	65	1254	123
TOTALS:	277	5,007.7	279	2035	217

Notes: (a) LWGR = Light Water-cooled and Graphite-moderated Reactor (or RBMK). WWER = Water-cooled & Water-moderated Energetic Reactor (Russian type PWR).
(b) Significant events only with generic implications. Cracking in the through-wall direction extending > 20% of pipe wall thickness. Temporary repair by weld overlay method or replacement of affected pipe section.
(c) Catastrophic loss of structural integrity and/or leak rate > 5 kg/s, no advance warning to control room operators.

4.3 Philosophy of the Service Data Collection Effort by SKI

PSA studies consider pipe ruptures, which result in primary system leak rates large enough to actuate the emergency core cooling system (ECCS). The philosophy that was adopted in developing the database on the service experience considered: a) The PSA requirements for parameter estimates on piping reliability; and b) A model of piping reliability based on interpretations of the service experience, and evaluations of the impact of different degradation mechanisms on mechanical integrity.

In PSA, the modeling of piping failures is mainly determined by functional considerations. These considerations include the makeup capability of engineered safety systems such as the ECCS given a leak or rupture of RCPB piping. That is, pipe failures that could challenge the make-up capability as well as the overall accident management through control room operator actions would be considered as candidate events for explicit modeling in the event tree and fault tree structures. Also included are dynamic considerations whereby the effects of a major structural failure on adjacent piping systems are evaluated. An example of dynamic effects would be the potentially uncontrolled motion of a ruptured pipe (i.e., pipe whip effects), which could sever vital instrument lines as well as support system piping.

A LOCA is an event caused by a pipe break or leakage in the primary system. For BWRs it is practical to distinguish between breaks occurring above and below the top of the active core ('top breaks' and 'bottom breaks') as well as between large, medium and small breaks. In older, external pump reactors, a break in the main recirculation line connected to the bottom of the reactor vessel constitutes a design basis accident (DBA). In modern, internal pump reactors, large bottom breaks cannot occur since the external recirculation loops have been eliminated. For these BWR types, an assumed guillotine break in a main steam line inside the reactor containment is representative of a large LOCA. While the definition of LOCA types are highly plant-specific, some illustrative examples of LOCA types for BWR and PWR plants are included in Tables 4-2 and 4-3, respectively. The needs for parameter estimates on pipe failures are based on the definitions of LOCA types.

Table 4-2: *Example LOCA Types for an 'External Pump' BWR.*

TYPE OF PRESSURE BOUNDARY	LOCA SIZE RANGE
1. DN500 main steam (MS) line piping, DN250 or DN300 main feedwater (MFW) piping; MS- and MFW-piping inside containment.	Large top break
2. DN650 to DN200 main recirculation piping, DN250 residual heat removal (RHR) piping, or DN250 ECCS piping.	Large bottom break
3. DN50 to DN80 main steamline piping, DN200 RHR piping, DN150 reactor pressure vessel head sprinkler piping, or DN250 auxiliary feedwater system piping.	Medium top break
4. DN100 to DN150 main recirculation piping, DN50 to DN150 RHR piping, DN100 ECCS piping, or DN50 standby liquid control system (SLCS)	Medium bottom break

Table 4-3: *Example LOCA Types for PWR.*

TYPE OF PRESSURE BOUNDARY	LOCA SIZE RANGE [cm^2]
1. Reactor pressure vessel (RPV) failure	
- Within safeguards capability	0 - 'large'
- Beyond safeguards capability	'large'
2. Reactor coolant system (RCS) loop piping (i.e., hot leg from RPV to steam generator (SG), cross-over pipe connecting the SG and the coolant pump, and cold leg between the coolant pump and the RPV).	0-1800
3. Steam generator man-way	
- Leak	0-500
- Complete failure	500
4. Control rod mechanism housing (leak)	0-180
5. Accumulator line (leak)	0-300
6. Pressurizer surge line (leak 6 rupture)	0-180
7. Relief valve piping	0-70
8. Safety injection system piping	0-70
9. Miscellaneous DN50 and DN100 piping	0-37
10. Miscellaneous DN20 instrument lines	0-1
11. Incore instrument lines	0-9

Pipe failures are reported as flaws/cracks, leaks and ruptures, corresponding to incipient, degraded and complete failure, respectively; *c.f.* Figure 4-1. In SKI's data collection on pipe failures, a 'rupture' was interpreted as a catastrophic loss of mechanical integrity resulting in a large leak rate (> 5 kg/s).

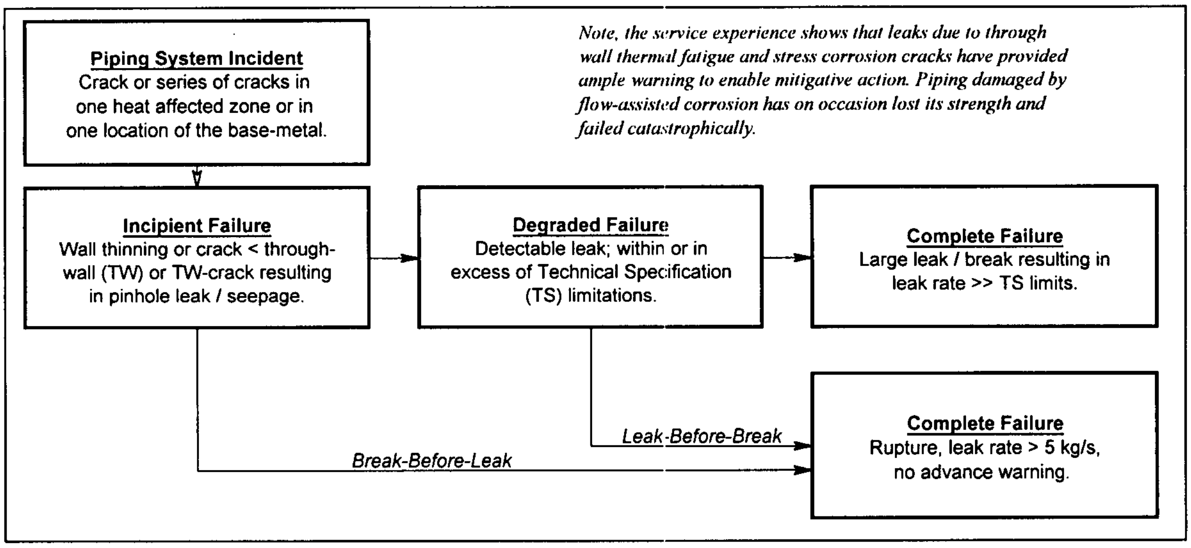

Figure 4-1: *Pipe Failure Mode Definitions Used in Developing SKI's Service Data Collection.*

The evaluation of service experience should be done against a model of failure. For piping it is convenient, and also necessary to distinguish between two types of conditional factors of failure: a) Reliability *attributes*; and b) Reliability *influence* factors. An attribute represents the unique design features of a piping system. Hence, an attribute cannot be modified without modifying the basic design. Examples of attributes include diameter/wall-thick, material, high pressure piping, moderate pressure piping. An influence factor relates to the operating environment (and the related degradation susceptibilities), and the inspection practices. A practical way of defining primary influence factors is to ask the following question: 'In view of an occurred failure, what is the best (e.g., most cost-effective) remedial action to prevent recurrence?' This is the root cause analysis perspective on reliability influence factors.

Based on the attribute and influence concepts, the service experience is organized according to exposure fields and event fields in the database. Each record fits one unique exposure field, and each failure is the realization of one and only one degradation mechanism and one and only one failure mode. The pipe rupture frequency, f_R, associated with a particular attribute may be estimated from:

$$f_R = f_F \cdot p_{R|F} \qquad (4.1)$$

where $f_F = (2F + 1)/2T$ (4.2)

$p_{R|F} = (2R + 1)/(2F + 2)$ (4.3)

Index R= rupture;

Index F= failure, which could be a through-wall flaw/crack, leak or rupture (see below);
T = exposure time in reactor-years (i.e., the in-service time).

In Equation (4.1) the parameter estimation problem is separated into two steps. First, the occurrence rate of a failure, f_F, resulting in a plant shutdown for repair or replacement is estimated from the service experience. Next, the conditional rupture probability given a failure, $p_{R|F}$, is estimated. Equation (4.1) is useful for degradation mechanisms that progress from leakage to rupture if the leak is not detected and repaired. The estimates derived through Equations (4.2) and (4.3) are the mean values of aposteriori Γ- and β-distributions, respectively, using non-informative priors; *c.f.* Martz and Waller (1982). Some global failure frequency and rupture frequency estimates are given in Table 4-4.

Table 4-4: *Some Examples of Failure*[1] *& Rupture Frequency Estimates Based on Service Data.*

Degradation Mechanism	**Number of Failures**	**Number of Ruptures**	**Mean Failure Frequency [1/Reactor-Year]**	**Mean Conditional Rupture Probability**	**Mean Rupture Frequency [1/Reactor-Year]**
Boric Acid Corrosion (BAC)	21	0	$4.3 \cdot 10^{-1}$	$2.3 \cdot 10^{-2}$	$9.9 \cdot 10^{-5}$
Corrosion (COR)	146	5	$3.0 \cdot 10^{-2}$	$3.7 \cdot 10^{-2}$	$1.1 \cdot 10^{-3}$
Erosion-corrosion (E/C)	433	54	$1.4 \cdot 10^{-1}$	$1.3 \cdot 10^{-1}$	$1.1 \cdot 10^{-2}$
Vibration-fatigue (VF)	710	65	$1.4 \cdot 10^{-}$	$9.7 \cdot 10^{-2}$	$1.4 \cdot 10^{-2}$
Thermal fatigue (TF)	90	2	$1.8 \cdot 10^{-2}$	$2.8 \cdot 10^{-2}$	$5.0 \cdot 10^{-4}$
Stress corrosion cracking (SCC) PWR environment	129	0	$4.6 \cdot 10^{-2}$	$3.8 \cdot 10^{-3}$	$1.8 \cdot 10^{-4}$
Intergranular SCC (IGSCC) BWR environment	304	0	$2.1 \cdot 10^{-1}$	$1.6 \cdot 10^{-3}$	$3.4 \cdot 10^{-4}$
Transgranular SCC (TGSCC)	38	0	$7.4 \cdot 10^{-3}$	$1.4 \cdot 10^{-2}$	$1.0 \cdot 10^{-4}$

These tabulated values represent the industry-wide experience for piping subjected to respective degradation mechanism. Next the attribute of concern must be defined more precisely, and the dimension of exposure must also be determined. For PSA applications, and as indicated by Tables 4-2 and 4-3, an attribute could be *[diameter -*

[1] Includes incipient, degraded and complete failures per definitions in Figure 4-1.

type-of-system - process-medium]. This leads to the necessity of organizing the service experience according to exposure fields by defining appropriate reliability attributes.

Does it matter in what way the service data are disaggregated? Figure 4-2 represents a comparison of conditional rupture probabilities for a selection of attributes. This comparison demonstrates the importance of defining strategies for disaggregation of service data. Equally important is the qualification of the service data. That is, the relevance of a particular service data aggregation to a specific application must be validated relative to application requirements.

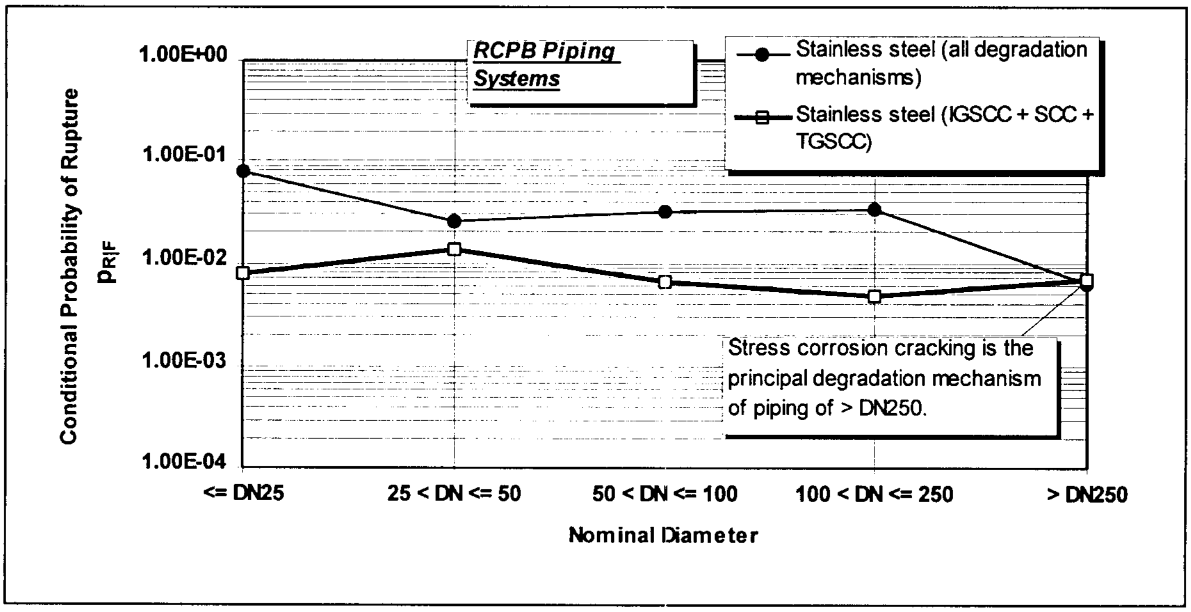

Figure 4-2: *Examples of Conditional Rupture Probabilities for Different Attributes.*

Before inputting the parameter estimates in the PSA models, the proper failure parameter dimension must be applied. For piping system components the dimension of exposure is [*time* · *extension*]. Hence, the parameters given in Table 4-4 are incomplete estimates. The extension cannot be universally defined, however. It is a function of the applicable reliability attributes and influence factors. For austenitic steels susceptible to IGSCC, the flaws/cracks or leaks develop in welds or weld-heat-affected zones (HAZ). Therefore,, the extension would be the number of welds/HAZ in the piping system(s) under consideration. The extent of erosion/corrosion (or flow-assisted corrosion) damage in ferritic steels is strongly influenced by flow velocity and geometry. Hence, for piping susceptible to erosion/corrosion (or flow-assisted corrosion) the extension would be given by the number of elbows, tees, reducers and straights.

Assuming that the average number of welds in IGSCC-susceptible piping is about 2000 per plant, the mean rupture frequency then becomes:

$$f_R = (f_F / 2000) \cdot p_{R|F} = (1.7 \cdot 10^{-1} / 2000) \cdot 2.0 \cdot 10^{-3} = 1.7 \cdot 10^{-7} \text{ / Weld.Plant-year}$$

The above parameter estimate is provided for illustrative purposes. It does not distinguish between IGSCC-susceptible piping of different diameter. Nor does it distinguish between different grades of austenitic steel piping. Accurate piping component population counts should be extracted from design information (e.g., isometric drawings).

4.4 Validity of Parameter Estimates Based on Service Data

Complete failures of piping systems in nuclear power plants are rare events. This observation raises an important question regarding the applicability of service experience to failure parameter estimation. The analytical problems associated with rare events in the context of PSA have been debated for a very long time; *c.f.* OECD-NEA (1978). There are opposing views regarding the appropriate analytical approach to LOCA frequency estimation; *c.f.* Table 4-5. Such is the situation in other technical aspects of PSA involving rare events such like the evaluation of human error susceptibilities during the responses to plant disturbances, or common cause failures in modern plants with 4-train safety and support systems.

Table 4-5: *Examples of Opposing Views Regarding the State-of-Practice in PSA.*

Technical PSA Element	Some Opposing Views
LOCA Frequency Estimation (i)	<u>PSA Methodology</u>: Substantial uncertainties in the models used to represent the physical aspects of safety assessment; uncertainty with respect to impact of aging on equipment; uncertainty with respect to defects introduced during manufacturing, design, construction/installation. The PSA methodology includes techniques and methods to explicitly account for these uncertainties. The models should reflect the available service experience. Fundamentally, the LOCA frequency estimation is no different than, say, the estimation of human error probability, or estimation of common cause failure probability. <u>Material Sciences</u>: There have been no ruptures in medium- or large-diameter piping. Therefore, it is impossible to estimate rupture frequencies based on zero occurrences. The only approach to LOCA frequency estimation is through probabilistic fracture mechanics (PFM).
LOCA Frequency Estimation (ii)	<u>PSA Methodology</u>: 'Extended LOCA definitions' required to achieve realism. That is, rupture locations in other than BWR recirculation piping or PWR cold leg/hot leg/crossover piping must be considered. This means that degradation mechanisms other than IGSCC/SCC need to be considered. The service experience indicates that degradations due to thermal fatigue and thermal stratification can result in ruptures. <u>Material Sciences</u>. Sometime focuses on the DBA-definition of LOCA; i.e., events that would be within the ECCS makeup capability are not considered. Primary degradation mechanism limited to IGSCC in the BWR primary piping system environment.

Indeed, there are limitations and uncertainties associated with the collection of service experience on piping system. First of all, knowing only the number of failures in an exposure cell without knowing the exposure, does not provide enough information to enable failure rate estimation. Collecting exposure data from the industry-wide service experience is a formidable undertaking. Short of retrieving the exposure information from piping and instrumentation diagrams (P&IDs) and isometric drawings on a plant-by-plant basis, it might be possible to let experts estimate this exposure; e.g., for a given attribute, what is the count of welds, elbows, tees, straight sections?

The coverage and completeness of a database on the service experience impact the validity of parameter estimates. Our ability to develop reasonable (i.e., complete) databases is a direct function of the reporting practices for piping failures. There are extensive regional differences in the reporting practices. Piping failures within the RCPB resulting in detectable leaks tend to be reported, especially where the plant Technical Specifications include explicit Action Statements. Degradations (e.g., flaws/cracks) revealed during maintenance or refueling outages may only be reported in special inspection or outage reports. The scope of the work in SKI's R&D-project was restricted to licensee event reporting systems. Nevertheless, the depth and breadth of the database exceeds other known national or international databases on piping failures. Despite the many limitations and uncertainties, parameter estimation based on service experience is feasible.

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5. DEVELOPMENT OF A PROBABILISTIC PIPE RUPTURE MODEL

One of the main problems related to the modeling of structural reliability is the difficulty to obtain empirical data. The Class 1, 2 and 3 piping systems[1] of NPPs are highly reliable and major losses of structural integrity are rare events. There are two ways to estimate the probability of pipe rupture: a) Direct estimation using service experience as discussed in the previous section of this report; and b) Development of a structural reliability model based on fracture mechanics.

Advanced structural reliability models build on probabilistic fracture mechanics (PFM) analyses. The development of a PFM model based largely on service experience from Swedish BWRs has been implemented in the PC-based program PIFRAP. This program estimates the probability of rupture in primary system piping, which is subjected to IGSCC. The special considerations of this new analysis tool are presented in the following.

5.1 Constituent Elements of PFM

PFM models can take into account specific initial conditions of pipe damage and thermal and mechanical conditions that a piping system has experienced or is expected to encounter. The postulated degradation mechanism leading to failure (either leak or rupture) is the growth of cracks at welded pipe joints. As a result of imperfect welding or heat treatment, cracks can exist before a NPP begins service. Cracks can also initiate during plant operation due to the corrosive environment. If allowed to grow unchecked, such cracks could penetrate the pipe wall, causing leaks or (potentially) ruptures. It is therefore important to understand not only how cracks grow, but also the be able to detect and monitor existing cracks during plant operation.

In PFM, the size, shape, location and orientation of a crack are given as probability distributions, as well as material and stress parameters. The variables involved in PFM studies can be grouped as follows; *c.f.* Nilsson (1993) and Simola and Koski (1997):

– Material properties such as fracture toughness, fatigue crack growth data, stress corrosion cracking susceptibility;

[1] Per the U.S. Regulatory Commission's Regulatory Guide 1.26 (1976), Class 1 includes the reactor coolant system, Class 2 encompasses piping connected to the reactor coolant system, and Class 3 encompasses systems important to function (i.e., systems important to safety that are designed to provide cooling water and auxiliary feedwater for the frontline systems.

- Loads, transients, fatigue loading, residual stresses;
- Fabrication defects, inservice-induced defects;
- Inservice inspection, including reliability of inspection methods, inservice inspection plans.

The analytical processes implemented by PFM models estimate the conditional probabilities of a leak and rupture at individual weld joints, given that a crack exists at that joint, and the plant experiences various loading conditions at any time. It is important to emphasize that the PFM models are not equivalent to PSA. Instead, PFM incorporates deterministic (either empirical or analytic) models into a probabilistic framework that consolidates the results of deterministic crack growth calculations, along with the effects of other factors (e.g., leak detection, reliability of ISI, water chemistry). Therefore, a PFM model represents a parametric model of piping reliability. The results could provide input for that part of PSA event tree or fault tree using the probability of pipe rupture.

5.2 Requirements on PFM Models

Deterministic safety analysis of NPPs include the consideration of the large LOCA design basis accident, defined as a double-ended pipe break (DEPB) of the largest main coolant pipe. Piping, structures and equipment have to be designed to withstand loads resulting from DEPB's. Such designs require extensive pipe supports and pipe whip restraints. In the late 1970's, the U.S. NRC initiated projects to estimate the probability of DEPB's. Influences of preservice and inservice inspections, and the effects of various operational and accidental loads on the DEPB probability were also studied as part of these investigations.

The basic analysis tool used in most of these studies was a PFM analysis computer program called PRAISE (Piping Reliability Assessment Including Seismic Events) or derivatives of that program. PRAISE was initially developed to estimate the probability of a simultaneous occurrence of an earthquake and a large LOCA in a PWR; *c.f.* Harris, Lim and Dedhia (1981).

A later version of PRAISE (Harris et al, 1986) was applied by VTT with input data corresponding to the operating conditions of a Finnish BWR; *c.f.* Simola (1992). The objective of this trial application was to compare the analytical results obtained using PRAISE with the service experience involving IGSCC in Finnish BWRs (TVO-1/2). Additionally, the sensitivity of PRAISE to the variation of some parameters important to IGSCC were studied. These parameters were the start-up and steady state oxygen concentrations, conductivity, operating temperature, the heat-up transient time, and the number of heat-up transients per year. The effect of residual stress modeling was also studied. Some of the insights from the parametric studies are summarized below.

The probability of leakage was found to have a linear dependence on the number of transients. Hence, in the model, the cracking is totally governed by the number and duration of start-up conditions. This is due to the calculation of crack initiation time distribution. If the cumulative time spent in the heat-up state is short, the probability of crack initiation within the observation period of ten years is very small. After a crack has initiated, and the crack growth is calculated by fracture mechanics, the steady-state periods become more significant. The effect of the heat-up duration is slightly smaller than the effect of the transient frequency.

Remedial measures to alleviate IGSCC have concentrated on improving the water coolant purity and injecting hydrogen in the feedwater (HWC). Experience has shown (Scott, 1996) that where the water conductivity can be controlled to $\leq$ 0.3 mS/cm the risk of crack initiation diminishes considerably. With HWC, the conductivity normally stays below 0.1 mS/cm. The application by VTT of the PRAISE code showed the conductivity to have its strongest influence at small values, and the probability of leakage to have a logarithmic dependence on the conductivity.

The dissolved oxygen content of the reactor water, and thus the electrochemical potential, is one of the most critical variables in controlling crack initiation. According to the parametric studies, an increase in the start-up oxygen concentration appeared to have more importance at low concentration levels, and its effect seemed to saturate at higher concentration levels. An increase of the oxygen concentration from 0.1 to 0.4 ppm increased the leakage probability by about 10%.

The leakage probabilities obtained by the PRAISE code, with input parameters based on the Finnish service experience, were much higher that expected. According to the simulation results, several leakages should already have occurred, when in fact the service experience so far remains at zero leak events. Furthermore, the crack shapes obtained in the simulations differed in a substantial way from the actual shape of the cracks that had been subjected to metallographic evaluations. The crack depths of detected cracks were less than 8 mm but the length of several cracks were greater than 25 mm, a few of them exceeding 100 mm. In the simulation, the crack lengths which did not result in leakage were all less than 25 mm. In PRAISE, the crack shapes are affected by the number of the possible initiation sites because of the crack link-up model. The fast crack growth in the through-wall direction in the simulation is in disagreement with the service experience.

The modeling of IGSCC includes several uncertainties, which are considered by the PRAISE code. Although the amount of experimental data and actual service experience is considerable, the large scatter in the data leads to wide confidence bounds. More knowledge about the relevance of the various data sources for the analysis of a specific structure is needed before the PFM models can be routinely applied to PSA. The information obtained from the service experience is extremely valuable in order to improve the validity of the PFM models.

As summarized in Gosselin and Fleming (1997), the PFM models as implemented in the PRAISE code, and similar computer programs, are computationally intensive. They require expert users both in terms of developing the input parameters and performing the calculations. The results of the analyses are often driven by external assumptions that are subject to significant uncertainties. For example, the analyst must make assumptions regarding crack size distributions in the material, stress history (cyclic stresses, mean stresses, number of stress cycles, etc.), crack detection probability. In the absence of a comprehensive database to draw upon or industry consensus regarding acceptable values, these assumptions are very qualitative and will be subject to much uncertainty. Consequently, different analyses of the same problem done by different people can produce differences of several orders of magnitude in pipe rupture frequency. Published results of PFM evaluations include estimates of pipe rupture that are too small to validate and have yet to be fully reconciled against service experience.

An apparent strength of the PFM models lies in their ability to support parametric studies and sensitivity analyses. The models address the impact by, say, water chemistry, inspection strategy, or inspection reliability on the probability of pipe rupture. Routine applications in the context of PSA requires access to user-friendly, transparent computer implementations with sufficient parameter databases, however.

The level of details and the probabilistic crack growth models varies among the different PFM models. The simplest models consider the crack growth process of a single existing flaw. The advanced models include the crack growth process, crack initiation, inservice inspections, leak detection, etc. Also, several welds can be considered simultaneously. The most common approach is to base a model on some physical laws or empirically identified relationships between the crack growth and material properties, stresses and other environmental factors. Since some of the factors affecting the rupture probability include uncertainties that dominate the results, the value of overly complex models is questionable. Uncertainties of the analysis results are related to the amount and quality of data. In order to improve the quality of the input data and the validity of the results, the systematic collection of service experience and the application of data reduction procedures especially developed for piping failures are particularly important.

5.3 The PIFRAP Model

Based on work initiated over ten years ago, a PFM procedure with accompanying PC-based computer software was developed within the scope of the NKS/RAK-1.2 program (Bergman, Brickstad and Nilsson, 1997). Objective of this development work was to provide a user-friendly analysis procedure intended for utility engineers working on PSA applications. The program estimates the rupture probability for a specific pipe section with prescribed local loading. In its present form, the program is specifically developed for the evaluation of IGSCC-susceptible piping in BWRs. Since the rupture probability is strongly dependent on the actual loading conditions, pipe systems are analyzed on a weld-by-weld basis. The following model assumptions are made:

(1) The stresses and the IGSCC crack growth law are assumed to be deterministic, and only depend on the initial crack length for a given geometry, stresses and material data. As part of PIFRAP, a procedure for calculating the crack

growth for a circumferential crack has been implemented in a PC-program denoted LBBPIPE (Bergman and Brickstad, 1995). LBBPIPE, based on fracture mechanics theory, gives the time-dependent crack shape (depth and length) for surface cracks and leaking cracks. The program also gives the time to leakage, the time to rupture, and the leak rate. No correlation is made between crack growth and plant transient history, however; *c.f.* (Aaltonen, Saarinen and Simola, 1993).

(2) The initial crack depth is assumed fixed to 1.0 mm. The growth of this crack is calculated using the LBBPIPE program.

(3) The probability that a crack with the assumed depth is initiated during the time interval (t_i, $t_i + dt$) is given by $f_i(t_i)dt$ (the crack initiation frequency). In the present work, and based on Swedish service experience, the crack initiation frequency is assumed constant and set to $1.25 \cdot 10^{-4}$/year for welds in straight sections. No differentiation made between crack initiation in small-, medium- and large-diameter piping.

(4) The initial length of the crack is random with the probability density function $f_{al}(l_0)$. This distribution is also assumed to be independent of the loading situation.

(5) The probability of not detecting a crack during inservice inspection is $p_{nd}(a,l)$. In the present implementation, the actual function used is independent of the crack length.

(6) The probability of not detecting a leak rate for a given leak rate detection limit d is p_{ld}.

(7) Leaking cracks or cracks detected by inservice inspection do not contribute to the rupture probability. This implies that the methods for sizing detected cracks and the leak detection systems must be reliable. A crack that penetrates the pipe wall will continue to grow circumferentially and eventually result in pipe rupture if it is not detected and repaired.

The PIFRAP is the program manage for performing PFM evaluations using the algorithms of LBBPIPE. The preparation of input parameters would require structural evaluations of a piping system (and its welds) to establish the loading conditions (e.g., internal pressure, weld residual stress, thermal stress and vibration stress amplitude). Table 5-1 summarizes the input data requirements and some typical parameter values. The program does not perform uncertainty propagation, but the sensitivity to different assumptions about the input parameters is quite easily established. The program result is given as the annual probability of rupture per weld in IGSCC-susceptible piping and should be interpreted as best estimate values. Preliminary trial applications, under assumptions of 40 year expected service life, 20 years of total inservice time and inservice inspection every 6 years, yielded a rupture (DEGB) frequency of:

$f_R = 2.4 \cdot 10^{-7}$ / Weld.Plant-year

The above parameter best estimate is provided for illustrative purposes. It does not distinguish between IGSCC-susceptible piping of different diameter. Nor does it distinguish between different grades of austenitic steel piping.

Table 5-1: *PIPFRAP Input Parameters With Examples.*

Input Data	Example Parameter Values
***Deterministic Input Data*:** Pipe geometry (radius of pipe, wall thickness) Loading conditions (internal pressure, weld residual stress, thermal stress and vibration stress amplitude) Material data (yield strength, ultimate tensile strength, initiation fracture toughness, elastic modulus, Poisson's ratio, vibration fatigue threshold value) Leak detection limit Inspection interval Crack morphological data for leak rate calculations	DN114 / 10 mm 7 MPa / ± 198 MPa / 18.3 - 16.6 - 48.9 MPa / 0 150 MPa / 450 MPa / 385 kJ/m^2 / 18,000.00 MPa / 0.3 / 4 MPa.m$^{1/2}$ 0.3 kg/s 6 years (for details, *c.f.* Nilsson, Brickstad and Skånberg, 1989)
***Probabilistic Input Data*:** Occurrence rate of IGSCC Initial crack length (a random variable with probability density function $f_{al}(l_0)$) Probability of not detecting a crack of depth a during ISI ($p_{nd}(a)$) Probability of no leak detection for a given leak rate detection limit d (p_{ld})	$1.45 \cdot 10^{-4}$ / reactor-year 12.1% of inner surface (mean value) (for details, *c.f.* Section 5.4 below) (for details, *c.f.* Brickstad, Bergman and Nilsson, 1997)

5.4 Effectiveness & Reliability of NDE Techniques

In Finland and Sweden, the inservice inspections (ISI) based on non-destructive examinations (NDE), such as ultrasonic inspections are largely performed in accordance with the *Rules for Inservice Inspection of Nuclear Power Plant Components* as specified by Section XI of the ASM Boiler and Pressure Vessel Code (BPVC). An adaptation of the Section XI by the Swedish regulatory body is documented in Hedner (1994) and SKI (1994). ISI's do not reveal all flaws in the inspected structures, however. Also, the accuracy of the flaw size determination depends on several factors; *c.f.* Table 5-2.

Table 5-2: *Some Influences on the Evidence of Ultrasonic Tests (Bauer, 1994).*

Test Pieces	Test Condition	Environmental Influences	Human Influences
Varying grain structures influence indications	Difficulty of selecting appropriate sensor for sufficient indication	Temperature of test pieces influence ultrasonic reflection. Human factors: accessibility, radiation	Short time reflection can be overlooked.

The reliability of ISI of LWR systems (primary piping and reactor pressure vessel) has been studied extensively since the 1970's. As an example, in Europe, the 'Programme for the Inspection of Steel Components' (PISC) was started in 1974. There have been a total of three PISC's to date (Doctor, Lemaitre and Crutzen, 1995). The U.S. Nuclear Regulatory Commission in 1978 initated the program 'Integration of NDE Reliability and Fracture Mechanics' (Becker et al, 1981). In order to meet the objectives of measuring the effectiveness and reliability of European and U.S. ISI procedures, respectively, cracked samples were manufactured according to a designed test matrix. Next a round robin was conducted under simulated field conditions with teams that were representative of the then current field practice. Finally, the inspection data were analyzed, correlated with destructive assay, and the results reported along with recommendations.

Based on the empirical data from PISC and similar programs, correlations have been developed for the relationship between the probability of detection (POD) and the crack size in different grade forged/wrought stainless steels and cast stainless steel piping. As summarized in Simola and Pulkkinen (1997), the data from reliability experiments on NDE are typically of three types: a) POD or so called 'hit-miss' data, where we have information on whether or not a flaw has been detected; b) 'Signal response versus size' data, e.g., flaw size determined from experiments versus actual size; and c) 'Mean sizing error and standard deviation data.'

Within the NKS/RAK-1.2 project, models of NDE reliability data analysis (Berens, 1989) were applied to flaw detection and sizing data from PISC-III. The inspection results of three assemblies in the PISC-III study were considered. These wrought stainless steel assemblies contained only IGSCC flaws. Based on results by the Finnish team participating in PISC-III, the probability of detection was estimated using two statistical models of inspection reliability (Figure 5-1).

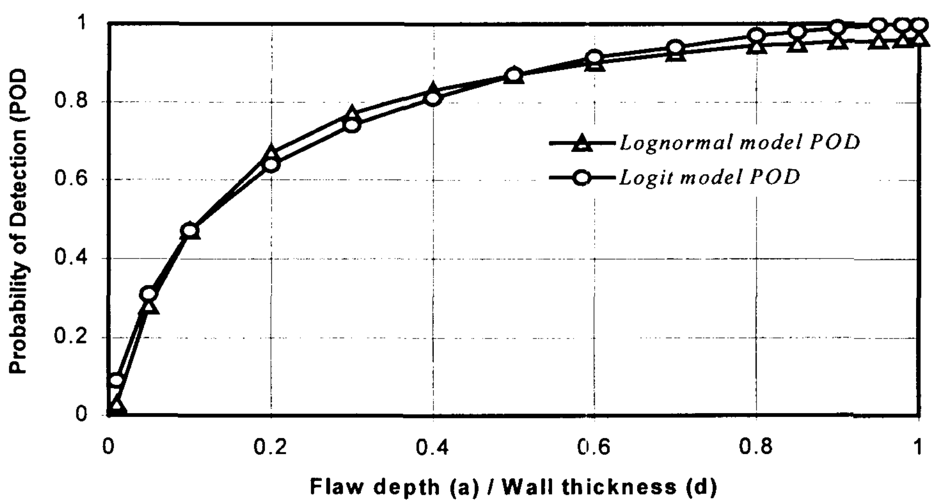

Figure 5-1: *Probability of Flaw Detection as a Function of Flaw Size (Simola & Pulkkinen, 1997).*

A limitation of the statistical models, such as the ones applied to the PISC-III data, is the requirement for extensive calibration measurements to estimate the reliability of inspection methods. Then there is always the question about the realism of round robin tests. Based on published, empirical correlations for the probability of detecting a flaw versus the size the following correlation would enable the direct consideration of human factors:

$$p_{\text{NON-DETECT}} = p_{\text{BASE}} + exp(-k \cdot IC) \qquad (5.1)$$

where p_{BASE} = Base-line probability of not detecting a flaw in a pipe segment under nominal (ideal) conditions. Interpreted as a human error probability (HEP);
k = 1.24; an assigned constant or anchor-value, which, by using p_{BASE} = 3.0E-3, gives $p_{\text{NON-DETECT}}$ = 1.0E-2;
IC = Inspection Complexity; representing the difficulty of finding a defect (Table 5-3) due to environmental conditions, accessibility.

Table 5-3: *An Example of the Definition of Inspection Complexity (IC) - Average Test Crew Performance.*

INSPECTION COMPLEXITY (IC)	DESCRIPTION OF COMPLEXITY	PNOD [$p_{\text{NON-DETECT}}$]
1	Large diameter pipe; incomplete inspection history; no known pipe replacements. Access may be difficult / radiation exposure potentially high. Ultrasonic testing possible.	2.9E-1
2	Large diameter pipe, full inspection history available; pipe replacements known to have been performed. Steps to prevent recurrence taken through material selection, water chemistry. Ultrasonic testing possible.	8.6E-2
3	Medium diameter pipe; incomplete inspection history; no known pipe replacements. Ultrasonic testing (UT) examinations possible. X-ray surveys challenging to perform due to the layout.	2.7E-2
4	Medium diameter pipe; full inspection history available; pipe replacements known to have been performed. X-ray surveys feasible for portion of pipe-run.	9.9E-3
5	Small diameter pipe; full inspection history available; history of previous pipe replacements; inspections ≤ 2 years. Visual testing and penetrant testing the preferred methods. Full-length X-ray surveys feasible.	5.0E-3
6	Small diameter pipe; full inspection history available; history of numerous pipe replacements in the past; inspections ≤ 2 years; potential for external impacts; vibrations possible during normal unit operation. Full-length X-ray surveys can be performed.	3.58E-3

The above correlation (Eq. 5-1) is intended as an alternative to the estimation of the probability of non-detection in the absence of empirical data such as the PISC-III data. A value of $3.0 \cdot 10^{-3}$ for p_{BASE} is used to represent the nominal inspections conditions. It is derived from the Human Reliability Handbook (NUREG/CR-1278) in which it is taken to represent nominal (or ideal) maintenance/inspections conditions. Figure 5-2 shows the probability of non-detection versus the inspection complexity (IC).

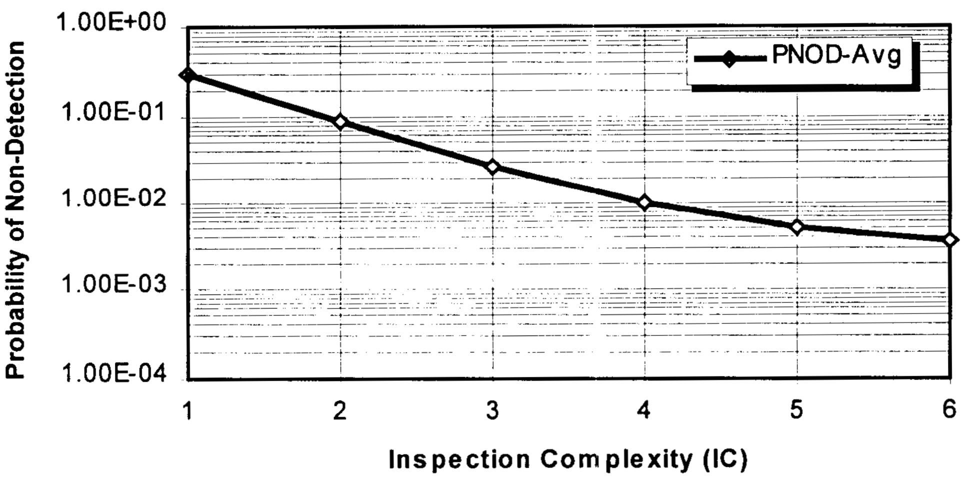

Figure 5-2: *Probability of Non-Detection of Defect - Screening Correlation.*

5.5 References

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6. CONCLUSIONS AND RECOMMENDATIONS

This report on NKS/RAK-1.2 has discussed two approaches to the estimation of pipe rupture frequency. One approach builds on interpretations of service experience through identification of reliability attributes and reliability influence factors, and parameter estimation, which uses non-informative priors in a Bayesian framework. The other approach builds on probabilistic fracture mechanics, and includes considerations of imperfect leak detection and inservice inspection. As implemented in a PC-based computer code, the procedure for performing the probabilistic fracture mechanics evaluations considers IGSCC-susceptible piping in BWRs.

Both approaches have unique analytical strengths and limitations, and they complement rather than replace each other. The approach which builds on interpretations of service experience does not go beyond the estimation of the frequency of pipe failure modes contained in the service data collections. As such, it is not a physical model of failure. In probabilistic fracture mechanics, the crack growth initiation and propagation phenomena are modeled for the purpose of estimating the probability of double-ended guillotine pipe breaks.

While outside the scope of the NKS/RAK-programme, pilot applications of the two approaches will be performed during 1998. The Barsebäck-1 PSA Update Project utilizes service data to develop plant-specific LOCA frequencies, and the Oskarshamn-1 PSA Update Project utilizes the PFM approach to generate LOCA frequencies. It is recommended that the insights and results be summarized and compared upon completion of the two pilot projects.

APPENDIX A

A.1 Abbreviations & Acronyms

AISI	American Iron and Steel Institute
ASME	American Society of Mechanical Engineers
BOP	Balance of Plant
DEGB	Double-Ended Guillotine Break
DN	Nominal Diameter (in mm)
ECCS	Emergency Core Cooling System
EPRI	Electric Power Research Institute
HAZ	Heat-Affected Zone
HSW	Heat Sink Welding
HWC	Hydrogen Water Chemistry
IGSCC	Intergranular stress corrosion cracking
IHSI	Induction Heating Stress Improvement
ISI	Inservice Inspection
LBB	Leak-Before-Break
LOCA	Loss of Coolant Accident
LWGR	Light Water Cooled and Graphite Moderated Reactor
MSIP	Mechanical Stress Improvement Process
NDE	Non-Destructive Examination
NKS	Nordic Ad Hoc Committee on Nuclear Safety Research
NPP	Nuclear Power Plant
NWC	Neutral/normal Water Chemistry
PFM	Probabilistic Fracture Mechanics
PIFRAP	Pipe Fracture Probabilities Computer Program
PRAISE	Piping Reliability Analysis Including Seismic Events
PSA	Probabilistic Safety Assessment
RCPB	Reactor Coolant Pressure Boundary
RHRS	Residual Heat Removal System
RISI	Risk-informed Inservice Inspection
RWCU	Reactor Water Cleanup System
SCC	Stress Corrosion Cracking
SLAP	SKI's LOCA Affected Piping Database
SS	Swedish Standard
TGSCC	Transgranular Stress Corrosion Cracking
TWC	Through-Wall Crack
TWD	Through-Wall Defect
UT	Ultrasonic Testing
WOR	Weld Overlay Repair

A.2 Glossary

Aging: Degradation of a component resulting in the loss of function or reduced performance caused by some time-dependent agent or mechanism. The agent or mechanism can be cyclic (e.g., caused by repeated demand) or continuously acting (e.g., caused by the operational environment). The change in the component failure probability resulting from the degradation will be monotonically increasing with the time of exposure to the agent or mechanism unless the component is refurbished, repaired, or replaced. In reliability statistics, aging is represented by that part of the 'bathtub curve' where the failure rate changes from being approximately constant to increasing.

Balance of Plant: The turbine-generator portion of a nuclear power plant with the associated piping and controls.

Break-Before-Leak: Used to describe the ratio of ruptures to total number of events involving ruptures and leaks. Various, experience-based correlations exist for determining this ratio.

Complete Failure: A failure that causes termination of one or more fundamental functions. If the failure is sudden and terminal it is also referred to as 'catastrophic'. The complete failure requires immediate corrective action to return the item to satisfactory condition. The effect of the complete failure on the unit can be a reduction in the feed rate or unit shutdown.

Degraded Failure: A failure that is gradual or partial. If left unattended (no immediate corrective action) it can lead to a complete failure.

Erosion/Corrosion (E/C): A form of materials degradation that affects carbon-steel piping systems carrying water (single-phase) or wet steam (two-phase) in both BWRs and PWRs. E/C-damage due to single-phase flow conditions usually manifest as uniform wall thinning similar to that caused by general corrosion. E/C-damage due to two-phase flow is less uniform and often has the appearance of "tiger-striping". Piping systems susceptible to E/C-damage include feedwater, condensate, extraction steam, turbine exhaust, feedwater heater, heater and moisture separator reheater vents and drains. There has been no documented evidence of E/C in dry steam lines.

Incipient Failure: An imperfection in the state or condition of equipment such that a degraded or complete failure can be expected to result if corrective action is not taken in time.

Induction Heating Stress Improvement: Heat treatment process which is preventing stress corrosion cracking by reducing tensile residual stresses.

Intergranular Stress Corrosion Cracking (IGSCC): A condition of brittle cracking along grain boundaries of austenitic stainless steel caused by a combination of high stresses and a corrosive environment. Primarily a problem in BWR environments.

IGSCC has also been discovered (mid-1970's) in the PWR environment, especially in piping containing stagnant boric acid solutions.

Leak-Before-Break (LBB): Most nuclear high-energy piping is made of high-toughness material, which is resistant to unstable crack growth. This type of piping would leak a detectable amount well in advance of any crack growth that could result in a sudden catastrophic break.

Noncritical Piping Failure: A local degradation of the pressure boundary that is limited to localized cracking with or without minor leakage. Such a crack would not reach critical size and lead to disruptive piping failure.

Pipe Rupture: Loss of pressure integrity of a pipe run in the form of a circumferential break, longitudinal break or through-wall crack.

[*Reference: ANSI/ANS-58.2-1980*]

Pipe Section (as defined by WASH-1400): A segment of piping between major discontinuities such as valves, pumps, reducers, etc. WASH-1400 indicated that, on average, a pipe section consists of 12 feet (3.6 m) of piping.

Pipe Section: A segment of piping between welds as indicated on isometric drawings. A pipe section can be either an elbow (e.g., 90E or 180E), a straight or a tee.

Piping Component: The passive components in a pipe run whose failure result in leakage or rupture. Includes pipe section, valves, flanges, fittings (elbow, tee, cross, reducer).

Piping Failure Attribute: Factor(s) that is believed to have a significant impact on pipe reliability; e.g., combination of metallurgy and application, type of pipe section, exposure time, load cycles; *c.f.* reliability attribute.

Probabilistic Fracture Mechanics: A procedure for determining pipe failure (leak or break) probabilities, especially large-diameter piping in the RCS. The procedure incorporates deterministic (either empirical or analytic) models into a probabilistic framework that allows the results of deterministic growth calculations for literally thousands of individual cracks to be consolidated, along with the effects of other factors such as NDE intervals and earthquake occurrence rates, into a single convenient result. The PFM models only apply for anticipated degradation mechanisms; e.g., IGSCC with long time between crack initiation and leak.

Reactor Coolant Pressure Boundary: All pressure containing components of light water reactor nuclear power plants, such as pressure vessels, piping, pumps, and valves that are either:

(1) Part of the reactor coolant system (RCS); or
(1) Connected to the RCS up to and including any or all of the following:
    (b) the outermost primary containment isolation valve in system piping that penetrates the primary containment;

(c) the second of two valves normally closed during normal reactor operation in system piping that does not penetrate primary containment; or
(d) the RCS safety and relief valves.

For a direct cycle BWR, the RCS extends to and includes the outermost primary containment isolation valve in the main steam and feedwater piping.
[Reference: ANSI/ANS-58.14-1993]

Reliability Attribute: (See 'piping failure attribute'). The inherent piping reliability established through application of recognized (e.g., nominated) piping system design principles and engineering standards. The inherent reliability cannot be changed without making design modifications.

Reliability Influence Factor: The achieved reliability through controlled/manageable environmental impacts (i.e., influences) or NDE, ISI, etc.

Round Robin: In the context of piping reliability and inspection, the purpose of round robin is to define reliability and effectiveness of inservice inspection procedures. Cracked pipe samples are manufactured, and then sent to expert teams who under simulated field conditions determine crack size and location. Test results are then analyzed, and correlated with the destructive assay. Next, results are reported along with recommendations.

Sensitization: Precipitation of carbides during welding. When austenitic stainless steels are heated in the range of about 425 C - 870 C, carbon in excess of about 0.02% will come out of solution and diffuse to the grain boundaries where it will combine with adjacent chromium to form chromium carbide ($Cr_{23}C_6$). These grain boundaries are then preferentially attacked by corrosive media.

Stabilization: To minimize the formation of carbides in austenitic stainless steels, niobium (Nb) or titanium (Ti) is added to the grain boundary area so that Nb- or Ti-carbides are formed. Purpose of stabilization is to minimize the susceptibility to sensitization.

Transgranular Stress Corrosion Cracking (TGSCC): A form of environment-assisted cracking (just as IGSCC); complex interaction of metallurgy, process medium and stresses. The resistance against corrosion that stainless steel has is depending on a passive oxide film that has low electron movement. Chlorides and sulphides travel into the film to create oxide chlorides/sulphides that result in high electron movement. Outside and inside diameter TGSCC have been observed.

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