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R645.1
A TECHNICAL MANUAL ON THE UNDERLYING THEORY & ASSOCIATED METHODOLOGY FOR PERFORMING OPERABILITY DETERMINATIONS THROUGH RISK CHARACTERIZATION OF DEGRADED PASSIVE COMPONENTS

Final Report
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R645.1 Final Report SPI-2018-06

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A Technical Manual on the Underlying Theory & Associated Methodology for Performing Operability Determinations Through Risk Characterization of Degraded Passive Components (R645.1: Statistical Modelling of Aging Effects in Failure Rates of Piping Components Task 7: Final Report)			
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ABBREVIATIONS & ACRONYMS

AAF	Aging Assessment Factor
AF	Aging Factor
AIChE	American Institute of Chemical Engineers
AM	Aging Management
ANS	American Nuclear Society
AOT	Allowed Outage Time
API	American Petroleum Institute
APSA	Aging PSA
ASD	Allowable Stress Design
ASME	American Society of Mechanical Engineers
ASEP	Accident Sequence Evaluation Program
ASP	Accident Sequence Precursor
ATWS	Anticipated Transient Without Scram
BS	British Standard
B&PVC	Boiler & Pressure Vessel Code
B&W	Babcock & Wilcox
CASA	Containment Accident Stochastic Analysis
CB	Chrystal-Ball (software product)
CBL	Crack-Before-Leak
CCDP	Conditional Core Damage Probability
CCPS	Center for Chemical Process Safety
CDF	Core Damage Frequency
CDP	Core Damage Probability
CFD	Computational Fluid Dynamics
CFP	Conditional Failure Probability
CGR	Crack Growth Rate
CNSC	Canadian Nuclear Safety Commission
CODAM	Corrosion and Damage Database
CODAP	Component Operational Experience, Degradation & Ageing Programme
COG	CANDU Owners Group
CRP	Conditional Rupture Probability
CSA	Canadian Standards Association
CSV	Comma Separated Value
CT	Completion Time
CUI	Corrosion Under Insulation
DDM	Data-Driven Model (of Piping Reliability)
DEGB	Double-Ended Guillotine Break
DFU	Defined Hazard & Accident Conditions
DM	Degradation Mechanism
DOE	U.S. Department of Energy
EBD	Event Based Degradation
EBS	Equivalent Break Size
ENIQ	European Network for Inspection and Qualification
ENSI	Swiss Federal Nuclear Safety Inspectorate
EPA	Environmental Protection Agency
EPRI	Electric Power Research Institute
ESD	Event Sequence Diagram
ESReDA	European Safety, Reliability and Data Association

FAC	Flow Accelerated Corrosion
FAD	Failure Assessment Diagram
FEA	Finite Element Analysis
FFS	Fitness for Service
FITNET	European Fitness-for-Service Network
FMEA	Failure Mode & Effects Analysis
FS	Factor of Safety
FSA	Fitness for Service Assessment
FSWOL	Full Structural Weld Overlay
HCF	High-Cycle Fatigue
HIC	Hydrogen-Induced Cracking
HMR	High MIC Resistance
HPI	High Pressure Injection (UK)
HSE	Health, Safety and Environment
ICCDP	Incremental Conditional CDP
IGSCC	Intergranular SCC
INES	International Nuclear Event Scale
ISI	In-service Inspection
JRC	Joint Research Centre
KNPS	Koeberg Nuclear Power Station
LAS	Low Alloy Steel
LB	Lower Bound
LBB	Leak-Before-Break
LCF	Low Cycle Fatigue
LHS	Latin Hypercube Sampling
LML	Local Metal Loss
LMR	Low MIC Resistance
LOCA	Loss-of-Coolant-Accident
LOFW	Loss of Feedwater
LOOP	Loss of Offsite Power
LOPA	Layer of Protection Analysis
LTA	Local Thin Areas
LTCC	Low Temperature Creep Cracking
MERIT	Maximizing Enhancements in Risk-Informed Technology
MIC	Microbiologically Influence Corrosion
MLE	Maximum Likelihood Estimation
MMR	Moderate MIC Resistance
MPC	Materials Properties Council
MRP	Depending on the context, “Markov Renewal Process” or “EPRI Materials Reliability Project”
MSF	Major Structural Failure
NASA	National Aeronautical and Space Administration
NCS	National Consensus Codes & Standards
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NOGA	Norwegian Oil and Gas Association
NPD	Norwegian Petroleum Directorate
NPR	NASA Procedural Requirement
NSSS	Nuclear Steam Supply System
NTWC	Non-Through-wall Crack
NURBIM	Nuclear Risk Based Inspection Methodology for Passive Components
OE	Operating Experience
OEAD	Operational Engineering Assessment Division (of the CNSC)
OPDE	OECD Pipe Failure Data Exchange Project

OPEX	Operating Experience
PBF	Pressure Boundary Failure
PDI	Plant Design Information
PE	Point Estimate
PFM	Probabilistic Fracture Mechanics
POD	Probability of Detection
POF	Depending on the context, “Physics of Failure” or “Probability of Failure”
PPOF	Probabilistic Physics of Failure Model
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PSA	Petroleum Safety Authority
PSM	Process Safety Manager
PVS	Pressure Vessels and Pressurized Systems
PWROG	PWR Owners Group
PWSCC	Primary Water SCC
RAC	Risk Assessment Code
RASP	Risk Assessment Standardization Process
RAVEN	Risk Analysis Virtual Environment
RF	Range Factor
RI-ISI	Risk-Informed ISI
RIM	Reliability & Integrity Management
RIOD	Risk-Informed Operability Determination
RISMC	Risk-Informed Safety Margins Characterization
RMTS	Risk Managed Technical Specifications
RNNP	Trends in Risk Level in the Petroleum Activity
ROY	Reactor Operating Year
RP	Recommended Practice
RSF	Remaining Strength Factor
RSL	Remaining Service Life
SDP	Significance Determination Process
SEM	Standard Error of the Mean
SGTR	Steam Generator Tube Rupture
SICC	Strain Induces Stress Corrosion Cracking
SIM	Structural Integrity Management
SINTEF	Stiftelsen for industriell og teknisk forskning
SME	Subject Matter Expert
SMP	Semi-Markov Process
SOHIC	Stress-Oriented Hydrogen-Induced Cracking
SPAR	Standardized Plant Analysis Risk (model)
SPCC	Spill Prevention, Control, and Countermeasure
SRM	Structural Reliability Model
SSC	System, Structure, Component
SW	Depending on the context, “Service Water”, “Socket Weld” or “Shop Weld”
TASCS	Thermal Stratification, Cycling and Striping
TCF	Temporal Change Factor
TM	Technical Manual
TS	Technical Specifications
UB	Upper Bound
UHS	Ultimate Heat Sink
ULS	Ultimate Limit State
USDOE	U.S. Department of Energy
VF	Vibratory Fatigue
ZOI	Zone of Influence

1. INTRODUCTION

The objective of CNSC Project No. 87055-15-0214 (“R645.1 Statistical Modelling of Aging Effects in Failure Rates of Piping Components”) is to develop a Technical Manual (TM) on the underlying theory and associated methods and techniques for performing operability determinations through risk characterization of carbon steel passive components that exhibit structural degradation such as rejectable non through-wall defects or active pressure boundary leakages. Specifically, the TM establishes a technical basis for a data-driven approach to piping reliability analysis that acknowledges the possible negative and positive impacts on pressure boundary integrity from material aging and reliability and integrity management (RIM) program implementation, respectively. The work that is documented in this report was performed during the period 2016 through 2018.

1.1 Background

The structural integrity of pressure boundary components is determined by multiple, interrelated reliability attributes and influence factors. Depending on the conjoint requirements for damage and degradation, certain combinations of material, operating environment, loading conditions together with applicable design codes and standards, some passive components are substantially more resistant to damage and degradation than others.

The field experience with safety- and non-safety related piping in commercial nuclear power reactors is extensive. Equally extensive is the experience gained from the implementation of different degradation mechanism (DM) mitigation strategies. By applying advanced piping reliability models, this body of field experience data, the associated engineering data (e.g. material properties, methods of fabrication and installation) and integrity management insights can be used to assess the projected structural integrity of a revised or new piping system design. These applications include assessments of how piping reliability parameter estimates are affected by different integrity management strategies as well as by the use of DM-resistant materials.

The overall scope of CNSC Project No. 87055-15-0214 is concerned with development of Technical Manual (TM) on the underlying theory and associated methodology for performing risk characterization of carbon steel passive components that exhibit structural degradation such as a non through-wall and through-wall defects. Specifically, the TM builds on the utilization of advanced statistical models of passive component reliability on the basis of recognized passive component reliability databases. In developing the TM, the following tasks were performed; Figure 1-1 and Table 1-1:

1. Survey of nuclear and non-nuclear risk-informed operability determination practices. Based on survey insights and results, this task formulates the salient features of a holistic approach to the assessment of the risk significance of degraded passive components such as carbon steel piping subject to cracking and/or wall thinning.
2. Development of the TM. This work builds on insights & results of applications that have been performed over a 25-year period. These applications include risk-informed operability determinations performed for a number of commercial nuclear power plants. Also included are insights from workshops that have been delivered to clients in North America as well as in Europe. The TM includes a set of review check lists especially developed for use by CNSC-OEAD subject matter experts.
3. TM Demonstrations. This aspect of the proposed project demonstrates the practical application of passive component reliability analysis methods & techniques:

a. Fitness-for-service evaluations of piping system components using advanced statistical models. Given the discovery of a degraded carbon steel piping component, this sub-task provides the step-by-step instructions for performing a risk-characterization.

b. Effect of different integrity management strategies (e.g., leak detection, leak test, non-destructive examination) on piping integrity. This sub-task addresses a practical application of Markovian reliability models to structural integrity assessments.

c. Aging management assessment of carbon steel passive components. This sub-task addresses the application of advanced statistical reliability models to the quantitative determination of aging factors.

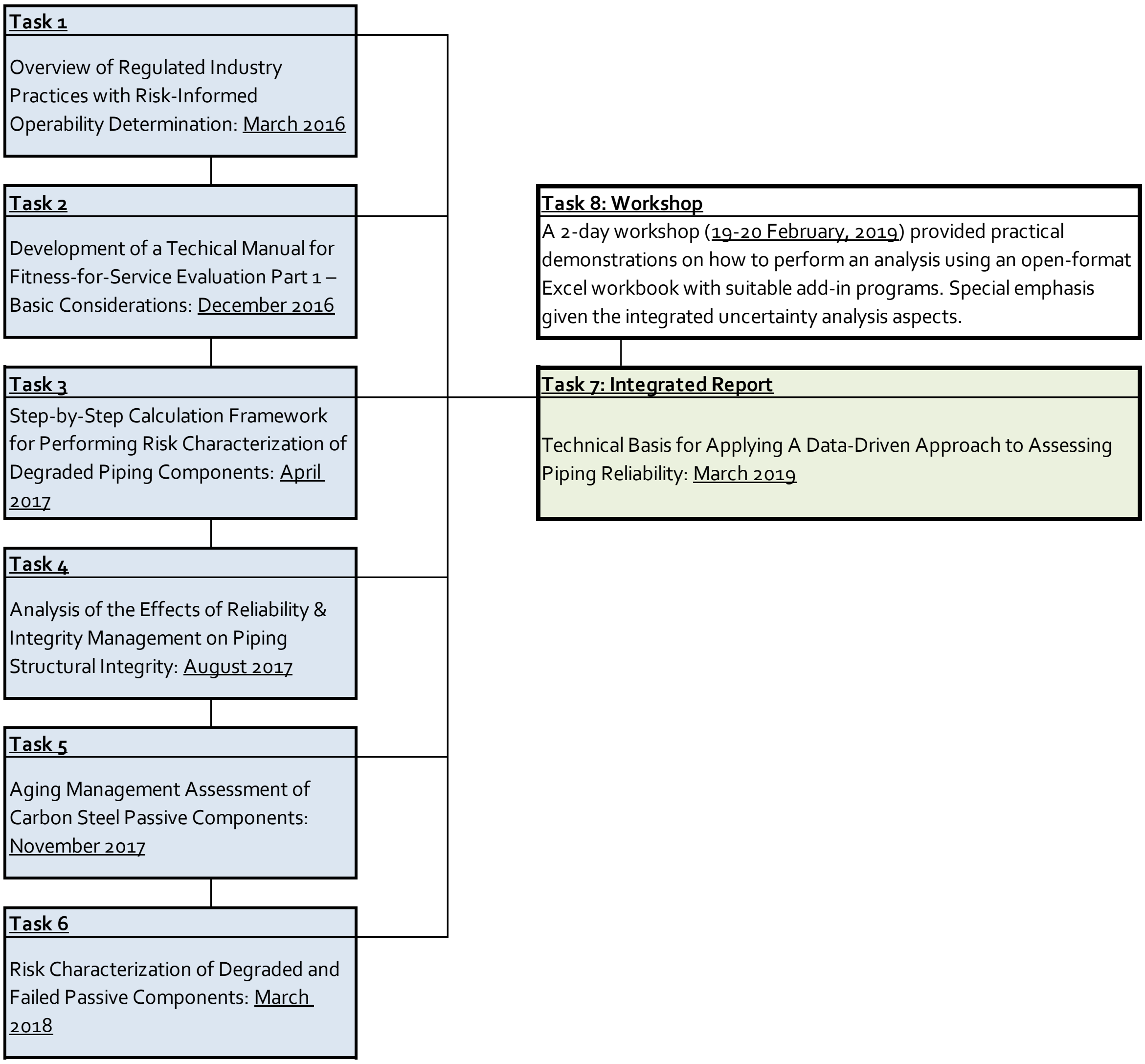

Figure 1-1: The Eight Tasks of CNSC Research Project 87055-15-0214

The TM is intended to support four types of risk informed structural integrity analyses: 1) risk-informed operability determination of degraded conditions, 2) independent assessments of licensee applications that include structural reliability analysis results, 3) structural integrity projections of influence by changes to reliability & integrity management (RIM) strategies, and 4) independent assessments of risk-informed in-service inspection (RI-ISI) program applications. Risk informed operability determination is concerned with the risk characterization of a degraded piping pressure boundary and its impact on plant risk.

Table 1-1: Overview of CNSC Research Project No. 87055-15-0214

Task ID	Definition	How the Task Was Executed
1 ☒	Overview of regulated industries practices with risk-informed operability determinations across different industries. Regulatory requirements and industry practices, as a minimum, should discuss US Nuclear Regulatory Commission, Environmental Protection Agency, US National Aeronautical and Space Administration, relevant authorities from European and Nordic countries. The contractor is encouraged to include references to other industries (oil, gas, marine, civil engineering, etc.). Date of completion: 3/24/2016; Revision 3 issued on 11/22/2017.	Performance of a survey of risk-informed operability determination approaches as practiced by the U.S. NRC through its Significance Determination Process (SDP), the Swiss Federal Nuclear Safety Inspectorate (ENSI) through the recently published ENSI-A06 (November 2015) and ENS-A06/d (November 2015), and the U.S. Department of Energy. Selected non-nuclear industry practices will be surveyed as well. Examples of non-nuclear examples include the Norwegian offshore industry through the Petroleum Safety Directorate and the Norwegian Oil & Gas industry. In the U.S. the Center for Chemical Process Safety (CCPS) of the American Institute of Chemical Engineers (AIChE) developed the guideline “Dealing with Aging Process Equipment & Infrastructure.”
2 ☒	Development of the TM. This work shall build on the existing body of insights and results of applications. This task needs to clearly acknowledge and address real-world applications involving the direct utilization of a pedigreed passive component failure (database). The TM shall include a set of review check lists especially developed for use by OEAD analysts. Date of Completion: 12/15/2016	Task 2 elaborates on a process for how to systematically perform a fitness-for-service or operability determination analysis on the basis of field experience data. Specifically, this task outlines a process for how to organize and to perform a piping reliability analysis task that utilizes a statistical modelling approach to the estimation of conditional piping reliability parameters.
3	TM Validation and Demonstrations. This aspect of the proposed project shall demonstrate the practical application of TM methods & techniques, such as:	See 3.a through 3.d
3.a ☒	Fitness-for-service evaluations of piping system components using advanced statistical models. Given the discovery of a degraded carbon steel piping component, this sub-task provides the step-by-step instructions for performing a risk-characterization; e.g. change in core damage frequency and large early release frequency. Date of Completion: 4/5/2017	Based on recent risk characterizations that have been performed to address the risk significance of degraded conditions in commercial nuclear power plants, this task outlines the details of applicable methods and techniques and how they respond to ASME/ANS RA-Sb-2013 (the “PRA Standard”) and other relevant regulatory guides.
3.b ☒	Effect of different integrity management strategies (e.g., leak detection, leak test, non-destructive examination) on piping integrity. This sub-task addresses a practical application of modern statistical reliability models to structural integrity assessments. Date of Completion: 8/18/2017	This report. Task 3.a is an extension of the CODAP 3rd Topical Report NEA/CSNI/R(2017)3: “Operating Experience Insights Into Pressure Boundary Component Reliability & Integrity Management.”
3.c ☒	Aging management assessment of carbon steel passive components. This sub-task addresses the application of advanced statistical reliability models to the determination of quantitative aging factors. Date of Completion: 11/22/2017.	Methods and techniques for quantitative assessment of “short-term” and “long-term” projected aging effects. “Short-term” aging addresses the effects of degradation mechanisms with “manifestation times” ranging from a single operating cycle up to an ASME XI inspection interval (i.e. 10 years). “Long-term” aging addresses the effects of degradation mechanisms with “manifestation times” > or >> 10 years.

Task ID	Definition	How the Task Was Executed
3.d ☒	Given the importance of passive components for operation of a plant, the TM should present details on modeling of operability determination including the factors that must be considered and how the statistical/ probabilistic fitness-for-service assessment should be performed. Date of Completion: 3/28/2018	The risk terms are defined that are used to quantitatively address the risk significance of a degraded or failed passive component. That is, terms like “change in CDF”, “change in initiating event frequency” and “conditional CDF” will be defined and exemplified. The roles of uncertainty analysis and quantitative acceptance criteria will be elaborated. Task 3.d is an extension of Tasks 1 and 3, and with focus on methodology (required inputs & expected outputs). It is anticipated that CNSC-OEAD will provide examples of probabilistic fitness-for-service assessments that the contractor will review against the TM checklists.
4 ☒	Documentation. Upon completion of Tasks 1 through 4, a Technical Manual/ Guide shall be prepared to document inputs, outputs, assumptions, methodology and results. Date of Completion: 6/30/2018. Final Report issued 3/8/2019	Each of Tasks 1 through 4 is documented as Technical Notes. Task 4 involves integrating respective Task Output into a single document. Also, feedback from participants in the February 19-20, 2019 Workshop were acknowledged in the preparation of the Final Report.

1.2 Objectives

This Technical Manual (TM) is intended as a technical basis document to enable the CNSC-OEAD subject matter experts to effectively utilize field experience data insights, either directly or indirectly, in performing operability determinations and to perform independent reviews of licensee submittals that address passive component structural integrity considerations. The work to prepare the TM was performed in six tasks and as outlined below:

- **TASK 1** (Sections 2 through 4). As a basis for the TM, the objective of Task 1 of Project 87055-15-0214 is to survey regulated industry practices with operability determination of degraded piping system components. The survey addresses operability determination as regulated and practiced by nuclear as well as non-nuclear industries. The scope of the survey is summarized in Table 1-1. Operability determinations are formalized engineering evaluations to determine whether or not a degraded passive component is fit for continued safe and reliable service over some predetermined period. An implicit facet of such an evaluation is the risk characterization of the degraded condition. That is, does the degraded condition impact a plant's risk metrics as determined by a probabilistic safety assessment (PSA)? Operating experience (OPEX) data plays an important qualitative and quantitative role in all operability determinations. Two general categories of operability determinations are performed: 1) evaluation of degraded conditions during routine power operation, and 2) evaluation of degraded conditions found during scheduled in-service inspections. In the former case the operability determination may address the viability of a temporary repair to avoid a forced outage. In the latter case, the operability determination focuses on the fitness-for-service (FFS), and whether a Code Repair or replacement is required if "codified acceptance criteria" cannot be met.
- **TASK 2** (Section 5). This task elaborates on a framework for how to systematically perform a fitness-for-service or operability determination analysis on the basis of field experience data. Specifically, this task outlines a process for how to organize and to perform a piping reliability analysis task that utilizes a statistical modelling approach to the estimation of conditional piping reliability parameters.
- **TASK 3** (Sections 6 and 7). This task elaborates on how to implement a Microsoft® Excel workbook format[1] for calculating pipe failure rates and rupture frequencies on the basis of operating experience data. Included in this task an example of how to benchmark piping reliability analysis results.
- **Task 4** (Section 8). Task 4 elaborates on the different options to model the effects of different integrity management strategies (e.g., leak detection, leak test, non-destructive examination) on piping integrity. Included in this report is an example of a Markov model application.
- **TASK 5** (Section 9). This task addresses methods for assessing the effectiveness of different aging management strategies (e.g. augmented NDE programs, improved water chemistry control, use of new materials that are resistant to environmental degradation). The analytical prospects for deriving statistics on "aging assessment factors" are explored through examples.
- **TASK 6** (Section 10). The vast majority of passive component operational events constitute precursors to more severe structural failures. In order to perform a risk characterization of these types of precursors an analyst is faced with two analytical challenges. First, a consequence analysis must be performed to accurately determine the

[1] A Microsoft Office Excel workbook is a file that contains one or more worksheets that are used to organize various kinds of related information. An Excel workbook template is provided as a separate deliverable.

potential safety impact of a degraded condition should it progress to major "event." Next the precursor information needs to be "extrapolated" by means of a conditional failure probability model. The consequence analysis consists of a qualitative element (in essence a failure mode and effects analysis, FMEA) and a quantitative element whereby a plant-specific PSA model is re-quantified by setting the failed component to probability of 1.0. Task 6 defines the risk terms that are used to quantitatively address the risk significance of a degraded or failed passive component; e.g. terms like "change in core damage frequency (CDF)", "change in initiating event (IE) frequency" and "conditional CDF." The roles of uncertainty analysis and quantitative acceptance criteria are addressed. Task 6 is an extension of Tasks 1 and 3, and with focus on methodology (required inputs & expected outputs).different aging management strategies (e.g. augmented NDE programs, improved water chemistry control, use of new materials that are resistant to environmental degradation). The results of Task 6 are documented in Section 10.

1.3 An Amplification

A fitness-for-service (FFS) evaluation implies that a deterministic or probabilistic structural reliability model (SRM) is applied to determine whether a degraded passive component can remain in service for a predetermined period. The SRMs are based on fracture mechanics theory. As articulated by A. Brückner (University of Karlsruhe, Germany), "*Most of the applications of probabilistic fracture mechanics (PFM) to nuclear components lead to very low failure probabilities which have to be determined by sophisticated numerical procedures from rather scarce and incomplete input data …."*[2] The validation of FFS results is an implicit evaluation process step, aspects of which addressed in this report.

This Technical Manual summarizes different approaches to FFS and operability determination (OD), and it elaborates on the different ways by which operating experience data can be used to validate FFS and OD results and to support the associated decision-making processes. The manual recognizes the strengths and limitations to PFM in calculating the likelihood of a pressure boundary failure, as well as the strengths and limitations of data-driven models of passive component reliability. In particular, the manual includes examples of how to use PFM and data-driven models in a synergistic analytical context. The traditional and perhaps widely held notion that it is practically impossible to validate FFS results by way of data-driven models needs re-assessing, however.[3]

[2] From Provan, J.W. (Ed.), Probabilistic Fracture Mechanics and Reliability, Martinus Nijhoff Publishers, Boston, MA, 1987, ISBN 90-247-3334-0, pp 351-386.

[3] According to Section 7 of ENIQ Recommended Practice 9 (December 2017), http://nugenia.org/eniq-reports/: "Since the objective of the SRM is to provide a realistic estimate for structural failure rates within industry, it would seem logical to argue that the historical data from the industry on such failures should be fundamental to the model validation. Unfortunately, due to the lack of adequate reliability data for the disruptive failure of components and structures there are inherent problems in using this means of validation. When comparing failure information from historical databases, several aspects and potential difficulties must be borne in mind. Generally, the historical failure data provides a point estimate determined by simply adding all the known passive component failures together and dividing by the total pipe population data, expressed for instance in weld-years. However, this data is derived from a wide variety of conditions, environments and loads, among other factors that influence failure probability. If this data is to be used to validate SRM software predictions in some way, then the SRM software must be run so as to represent the world data against which it is to be compared. This type of comparison cannot be completed unless the necessary data is available, which is not normally the case. On the other hand, qualitative trends between historical failure data and SRM software predictions can be more readily compared. In addition, large uncertainties inevitably exist with respect to rare events such as gross structural failures and failures of large pipes. More data is available on identified cracks and small leakages, which could be used for validation of the SRM software with the limitations stated above. Experience gained from application of the RI-ISI scheme can provide confidence in the overall predictions of the SRM, provided that experience aligns with SRM predictions and expected plant behavior." The author of this report disagrees with Section 7 of ENIQ RP9.

1.4 Reading Guide

This report consists of twelve sections and three appendices. Section 2 documents the nomenclature that is applicable to the technical content of the report. The term "operability determination" entails a formal decision-making process and supporting engineering analyses. The latter may involve deterministic as well as probabilistic analyses. In the context of piping reliability the term "failure" encompasses a range of structural integrity states, from a rejectable defect or flaw to a major loss of structural integrity with significant operation and safety impact.

Section 3 summarizes insights and results of a survey of different nuclear and non-nuclear technical approaches to operability determination. Section 4 is an overview of risk-informed operability determinations. A typical definition of "risk-informed" is a formalized decision-making process in which insights from industry-wide and plant-specific probabilistic safety assessments are considered with other engineering insights.

In Section 5 a framework for data-driven piping reliability analysis is presented and Section 6 provides an example for how such a framework may be implemented. Sections 7 through 10 represent amplifications of certain aspects this framework and its implementation.

Section 11 includes a summary and conclusions. A list of references is included in Section 12. Appendix A is a glossary of technical terms. Appendix B includes an abbreviated summary of carbon steel designations together with corresponding chemical compositions. Appendix C is a white paper on the development of justifications for different conditional pipe failure probability distribution parameters.

2. NOMENCLATURE

The systems, structures and components (SSC) of a nuclear power plant are subject to environmental degradation as well as to potential unusual loading conditions. Industry codes and standards govern the determination whether an SSC that has undergone some damage or degradation is fit for continued safe and reliable service, or whether it should be placed out of service to affect a permanent repair or a replacement. The terms "operability determination" and "fitness-for-service" are used to describe formalized (and in some instances codified) evaluation processes to determine whether a degraded or non-conforming condition is fit for continued service for some specified future period (e.g. operating cycle). Concepts like risk-informed decision making and risk-informed (or probabilistic) operability determination provide alternative or complementary approaches to a traditional or deterministic operability determination.

2.1 Data-Driven Models of Piping Reliability

The structural reliability of passive components may be assessed using one of, or a combination of the following modeling concepts: 1) probabilistic fracture mechanics (PFM), 2) probabilistic physics of failure (PPOF) models, 3) data-driven models (DDMs), or 4) hybrid DDM + PFM models; Figure 2-1.

Concept	Basic Principles of Analys
Data-Driven Model (DDM)	The "DDM" concept implies full use of OE data through statistical modeing and with or without explicit consideration of input from EE, PPOF and PFM. For additional information see Section 5.6 in this report.
Probabilistic Physics of Failure (PPOF)	PPOF models are developed from basic physics-based equations that describe the failure mechanisms. For example, fretting wear is the loss of material due to small-amplitude relative motion between contacting components. A probabilistic model is developed from a basic physics-based equation that describes fretting wear phenomena. The probabilistic model simulates fretting wear given certain pipe geometries and supports
Probabilistic Fracture Mechanics (PFM)	Fracture mechanics is an engineering discipline that quantifies the conditions under which a load-bearing body can fail due to the enlargement of a dominant crack contained in that body. Such enlargement can occur over an extended period, due to cyclic loading and/or adverse environmental effects. This subcritical growth of the dominant crack eventually leads to attainment of critical conditions, at which point the crack grows rapidly in an unstable manner. Monte-Carlo simulation is used to predict the probability distribution of a structural performance parameter from known distributions of input parameters used to model the problem. For additional information see for example Reference [89].
Expert Elicitation (EE)	Expert elicitation is a scientific consensus methodology. It is often used in the study of rare events. Expert elicitation allows for parametrization, an "educated guess", for the respective topic under study. Expert elicitation generally quantifies uncertainty. The "guesses" would be informed by DDM, PPOF as well as PFM. For additional details see for example NUREG-1829 [71].

Figure 2-1: The Different Piping Reliability Analysis Concepts

The DDM approach involves an explicit and optimized utilization of operating experience data. In this report the term "optimized" implies full accessibility to a comprehensive, well vetted operating experience database on piping in commercial nuclear power plants in combination with an engineering database of transcribed isometric drawing information from which pipe failure rate exposure term data can be extracted. A DDM approach has to consider three types of piping: 1) leak-before-break (LBB) piping and with no operating experience data involving major structural failures, 2) LBB piping but with operating experience data on major structural failure, and 3) break-before-leak (BBL) piping with operating experience data on the full spectrum of pressure boundary failures from minor leak to major releases of process medium involving significant spatial effects.

2.2 Definition of "Failure"[4][5]

In applying and comparing the results obtained from using different passive component reliability modeling techniques the definition of what constitutes a "failure" becomes important. It is noted that considerable ambiguity exists in the use of piping reliability terminology. In the broadest sense a passive component failure includes anything from a "rejectable flaw" that does not meet the requirements of applicable codes and standards to a major structural failure resulting in direct impact on plant operations (e.g. safety system actuation) and potentially with significant dynamic impacts on adjacent structures, systems or components (SSCs). The exact location of a flaw within a piping system, its size and orientation are inputs to any analysis. The definition of what constitutes a pipe failure is a function of a technical evaluation context and includes a consideration of:

- Rejectable defect or flaw requiring repair or replacement:
    - Weld repair to provide a leak barrier
    - Code repair that involves radiography or ultrasonic examination to verify the integrity of a weld repair
    - Full structural weld overlay
    - Replacement in-kind using same material
    - Replacement using new material
    - Replacement including re-routing/re-configuration using same material
    - Replacement including re-routing/re-configuration using new material
- Through-wall flaw, inactive leakage
- Through-wall flaw, active leakage

A structural failure mode may be divided into "ultimate" and "serviceability" types of failure modes. Ultimate failure modes are representative of a strength limit, beyond which the structural component loses capability to carry additional load. Serviceability failure modes imply structural failure without exceeding load-carrying capability which would occur prior to an ultimate failure.

In PFM a "leak" is a wall-penetrating defect which is stable, i.e. the loads are not inducing brittle fracture, plastic collapse, or unstable ductile crack growth. The rate of leakage may be limited to a "perceptible leak"; that is, visible wetness on the outside pipe wall. A wall-penetrating defect which fails due to the applied loads is considered as a "rupture." Failure probabilities are calculated for different initial flaw sizes, consequential through-wall flow rates and with consideration of the effectiveness of in-service inspection through different assumptions about the probability of detecting a pre-existing flaw.

In the DDM approach the failure probability metrics of interest are more-or-less consistent with those of PFM but with an emphasis on direct compatibility with the PSA modeling requirements. For example, initiating event frequency (number of events per reactor operating year) and as a function of well-defined consequence, for example equivalent break sizes (in

[4] Section 4.1 of the report "Failure Definition for Structural Reliability Assessment" (2000) by Hess, P.E., Ayyub, B.M. and Knight, D.E. includes a summary of a literature survey of failure definitions; https://www.researchgate.net/publication/235037126_Failure_Definition_for_Structural_Reliability_Assessment.

[5] Section 6 of "xLPR Version 2.0 Technical Basis Document, Acceptance Criteria" (2016) documents a U.S. regulatory and U.S. industry perspectives on the definition of failure of Safety Class 1 piping; https://adams.nrc.gov/wba/, Accession No. ML16271A436.

terms of through-wall mass flow rate or the equivalent diameter of pressure boundary penetration).

In a report related to probabilistic fracture mechanics computer code development the U.S. Nuclear Regulatory Commission and the Electric Power Research Institute have proposed the following metric [1]:

- "*All large break LOCAs and medium break LOCAs must previously be small break LOCAs, if even for a fraction of a second during a rupture, because the progression of degradation goes from crack, to leak, to bigger leak, and this process cannot happen in reverse. A LOCA should be defined as any leak with a rate that exceeds the leak rate of a small-break LOCA as defined in the PSA. This will vary from plant to plant, but will be approximately equal to the capacity of a single charging pump, which is typically greater than 50 gallons per minute. As a generic simplification for the 'xLPR' probabilistic fracture mechanics code, it is proposed that when the calculated leak rate at the end of a time step exceeds 50 gpm, the leak be defined as a failure, irrespective of whether unstable crack growth or rupture occurred or not. This is considered to be conservative, and can be changed for specific analyses, and should be subject to sensitivity studies, but this offers a simple starting point for the failure metric*."

2.3 Operability Determination

The term "operability determination" is used to describe a formal engineering evaluation process to assess the required SSC functions when a degraded or nonconforming condition is identified.[6] It is a holistic approach to the evaluation of the safety impact of a degraded condition and involves multiple qualitative and quantitative analysis steps; Figure 2-2 is a conceptual representation of the operability determination process.

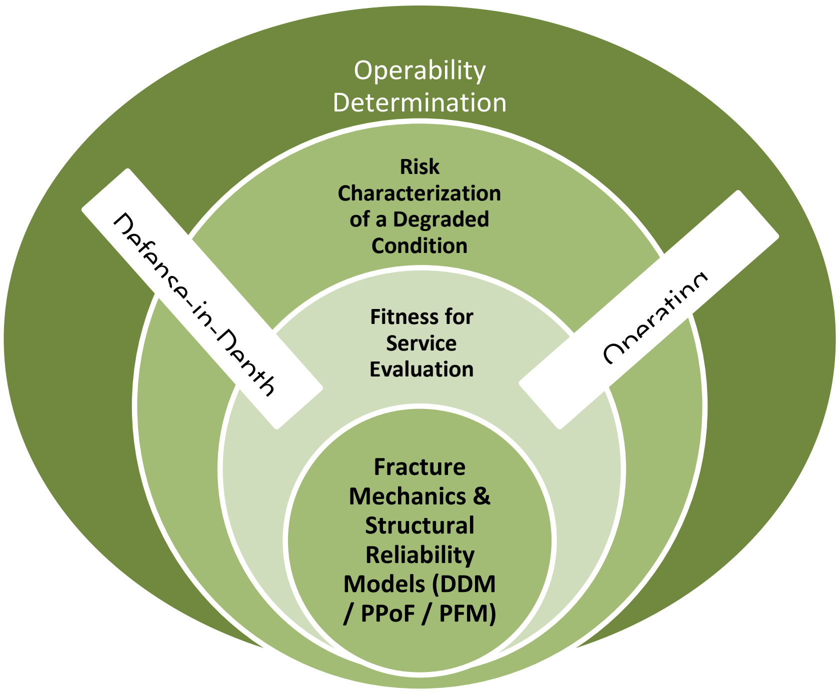

Figure 2-2: Engineering Evaluations in Support of Operability Determination

[6] For additional information see https://www.nrc.gov/reactors/operating/licensing/techspecs/operability-guidance.html. Furthermore, CSA 290.8-15 documents technical specification requirements for CANDU reactors; it is a "basis document." Functional and performance requirements are found in Sections 6.3 and 6.4 of CSA 290.8-15.

In the context of passive component reliability, conditions requiring operability and functionality evaluations are usually identified through plant walkdowns, in-service inspection and testing programs, non-destructive examinations, and augmented inspection programs. The conditions that lead to the requirement for an operability or functionality evaluation vary. Some examples of conditions identified by the in-service inspection and testing programs are pipe wall thinning from a flow-accelerated corrosion (FAC) mechanism, snubber failures and pipe or nozzle cracking from fatigue and IGSCC mechanisms, leakage exceeding Technical Specification limits for valves, and excessive pump vibrations. In some instances, a system-operating excursion subjects a system and its components to an unanalyzed condition that requires an evaluation.

Unanticipated operating events require operability evaluations to assess the margin-to-structural-failure. Examples of such events include fluid transients including water hammer; temperature and pressure excursions; and thermal stratification. Flow-induced system vibration from pump operation, cavitation, two-phase flow conditions, and acoustic pressure waves are known to lead to snubber wear, piping erosion, and fatigue failure, especially in socket-welded fittings.

When an SSC is found to be degraded or nonconforming, the operability determination should assess credible consequential failures previously considered in the design (i.e., the SSC failures that are the direct consequence of a design basis event for which the degraded or nonconforming SSC needs to function). Where a consequential failure (i.e., considering the degraded or nonconforming condition) would cause the loss of a specified safety function or functions needed for limiting or mitigating the effects of the event, the affected SSC is inoperable because it cannot perform all of its specified safety functions. Such situations are most likely discovered during design basis reconstitution studies, or when new credible failure modes are identified.

Structural integrity must be maintained in conformance with ASME Code Section XI (or equivalent code) for those parts of a system that are subject to Code requirements. The Code contains rules describing acceptable means of inspecting welds in piping, vessels, and areas of high-stress concentration. The Code also specifies acceptable flaw sizes based on the material type, location, and service of the system within which the flaw is discovered. If the flaw exceeds the generally acceptable limits, the Code also describes an alternate method by which a refined calculation may be performed to evaluate the acceptability of the flaw. At no time does the Code allow an unrepaired through-wall flaw to be returned to service. If a flaw is discovered by any means (including surveillance, maintenance activity, or ISI) in a system subject to Code requirements (whether during normal plant operation, plant transition, or shutdown operation), the flaw must be promptly evaluated using Code rules. If the flaw is through-wall or does not meet the limits established by the Code, the component and part of the system containing the flaw is inoperable. If the flaw is within the limits established by the Code, the component and part of the system is operable. An assessment must be made to determine how long the flawed component will remain operable before the flaw grows to exceed Code limits.

Probabilistic safety assessment (PSA)[7] may provide supporting information on the timeliness of a prompt operability decision and a corrective action. PSA is also useful for determining the safety significance of SSCs. The safety significance, whether determined by PSA or other analyses, is a factor in making decisions about the timeliness of operability determinations. Different types of risk monitoring software tools are used by plant operators to support decisions about SSC operability.

[7] The chemical process industries and oil and gas industries use the term quantitative risk assessment (QRA) in lieu of PSA.

The following types of activities are considered in order to define the scope of an operability determination:

1. Identify the equipment or SSC that is degraded, potentially nonconforming, or subject to the unanticipated event.
2. Establish the safety function(s) performed by the equipment.
3. Assess the possible failure or degradation mechanisms.
4. Assemble and assess relevant plant-specific and industry-wide operating experience data.
5. Define the operating events or loading conditions concurrent with the evaluation period.
6. Determine the acceptance criteria to be used to determine that the affected system would be operable through any of the following conditions:
   - Capable of performing an intended safety function(s)
     i. Pressure retaining function.
     ii. Cooling function.
     iii. Provide support to equipment given a demand.
7. Perform the necessary evaluations to determine whether the system is operable and fit for continued operation.

The methods that can be used include engineering analysis, test or partial test, analysis of operating experience data, or engineering judgment based on previous well documented experience. Any operability evaluation should consider the operational status of the SSC for the time period during which it is degraded or nonconforming. As one defines the operating events and loading conditions for the evaluation, it is important to distinguish between an SSC that was restored to its licensing basis and one that is expected to operate in the degraded or non-conforming condition. The evaluation of an SSC that was restored to its current licensing basis is initially based on all operating events that actually occurred while the SSC was in the nonconforming or degraded condition. Included in such operating events are unanticipated events that were determined to have occurred.

2.4 Fitness for Service

Fitness-for-service (FFS) assessment is a multi-disciplinary engineering approach that is used to determine if a piece of equipment is fit to continue operation for some desired future period. The equipment may contain flaws, have sustained damage (e.g., corrosion or creep damage), or have aged so that it cannot be evaluated by use of the original construction codes. As one example, the American Petroleum Institute (API) in 2000 issued the standard API 579-1 and in 2007, API and the American Society for Mechanical Engineers (ASME) jointly produced an updated document with the designation API 579-1/ASME FFS-1 [22]. As opposed to an operability determination, a FFS is discrete evaluation of a specific degraded condition with focus on material conditions and flaw growth potential. A typical FFS assessment requires collecting data from a number of sources. Some of the areas of expertise that may be part of an FFS assessment are:

- Stress Analysis. An accurate estimate of stresses acting on the component of interest is needed to assess structural integrity and remaining life.
- Metallurgy/Materials Engineering. An understanding of the performance of various materials subject to specific environments, temperatures, and stress levels is essential for ensuring safe and reliable operation.
- Nondestructive Examination (NDE). Flaws must be detected and sized before they can be assessed. The most suitable inspection technology depends on a variety of factors,

including type of the flaws or damage present and the accessibility of the region of interest.

- Corrosion. An understanding of environmental degradation mechanism(s) that led to the observed damage is a prerequisite for FFS assessments. Moreover expertise in corrosion is useful for prescribing suitable remediation measures.
- Plant Operations. Interaction with plant personnel is usually necessary to understand the operating parameters for the SSC of interest. Information such as operating temperature & pressure, process environment, and startup/shutdown procedures are key inputs to a FFS assessment.
- Fracture Mechanics. This discipline is used to analyze cracks and other planar flaws.
- Probability and Statistics. This discipline is useful for data analysis and for a probabilistic approach to FFS. An example of probabilistic FFS assessment approach has been used in a risk based pipeline integrity management program. The probabilistic assessment utilizes an Advanced Monte Carlo simulation based approach and the fracture mechanics techniques described in BS 7910 [30]. Wintle and Kenzie [31] present an overview of the basic approach and provides a demonstration of its capabilities in terms of estimating the risk of failure (or probability of failure) associated with a pipeline over time, due to the presence of a crack like flaw. The paper also discusses the sources of data and inherent assumptions used to model various input parameters required for a typical FFS analysis carried out according to BS 7910. Probabilistic FFS assessment incorporates the uncertainties in the input data. In this approach there is no absolute yes/no 'answer' as to whether or not a structure is safe for continued operation. Rather, the probabilistic analysis estimates the relative likelihood of failure, given all of the incorporated uncertainties.

Fitness-for-service assessments can range in complexity from simple screening evaluations to highly sophisticated computer simulations, including finite element analysis (FEA) and computational fluid dynamics (CFD). The necessary level of complexity varies from one situation to the next. In some cases, an advanced analysis is performed when a simple screening assessment is unable to demonstrate that the equipment in question is fit for continued service. Standardized FFS procedures typically include a range of assessment options that cover the full spectrum of complexity.

If flaws are discovered during normal operation, a fitness-for-service assessment can determine whether or not it is safe to operate the equipment until the next planned outage. If the outcome of the FFS assessment is favorable in such a case, then the operator can avoid a costly unplanned shutdown. Even during an outage, whether planned or not, it is desirable to avoid or postpone repairs, provided the FFS assessment indicates that the equipment can be safely operated until the next planned shutdown. Unscheduled retirement of components can be particularly costly, as long lead times for delivery of replacement components can result in extensive delays in production. Fitness-for-service assessments provide a rational basis to determine whether or not a damaged component can continue to operate until a replacement can be delivered.

2.4.1 ASME Code Case N-597-2

The rules of Code Case N-597-2 apply to the evaluation of Class 1, 2 and 3 piping systems subjected to internal or external wall thinning. Components affected by flow-accelerated corrosion to which this Code Case are applied must be repaired or replaced in accordance with the construction code of record and Owner's requirements prior to the value of remaining wall thickness reaching the allowable minimum wall thickness, as specified in Article 3622.1(a)(1) of this Code Case. A Class 1, 2, or 3 butt welded pipe, elbow, branch connection, or reducer piping item is acceptable for continued service without further evaluation when t_p, at all locations on the piping item meets the following requirements:

1. For straight pipe and elbows purchased to a nominal pipe specification with an allowable wall thickness under-tolerance of 12.5%, t_p shall be not less than $0.875 \times t_{nom}$ except that, for Class 1 short radius elbows, an evaluation shall be conducted to show that the requirements of NB-3642.2 are met.[8] [9]
2. For the small end of concentric and eccentric reducers, t_p shall be not less than$0.875 \times t_{nom}$ for the pipe size at the small end. For the large end, the large end transition and the conical portion, t_p shall not be less than $0.875 \times t_{nom}$ for the pipe size at the large end. For the small end transition, the required thickness shall be gradually reduced from that required at the large end to that required at the small end.
3. For tees and branch connections, t_p shall be not less than $0.875 \times t_{nom}$ for the same size pipe for regions outside the limits of reinforcement required by the Construction Code used in the evaluation. For regions within the limits of reinforcement, t_p shall be not less than the thickness required to meet the branch reinforcement requirements of the Construction Code.

- For regions of piping items designed to specific wall thickness requirements, including designed weld counter-bores and regions with integral reinforcement pads, t_p shall be not less than the minimum design thickness, including tolerances and excluding any corrosion allowances, specified in the original design analysis for the piping item.

2.4.2 ASME B31G – Non-Safety-Related Piping

ASME B31G has been commonly applied to evaluate wall thinning of below ground pipelines, using the results of in-line inspections. The criteria for corroded pipe to remain in service are based only on the ability of the pipe to maintain structural integrity under internal pressure.

2.5 Risk-Informed Decision Making

Risk-informed decision making (RIDM) [44] is different from risk-based decision making (RBDM) in that RIDM is a "fundamentally deliberative process that uses a diverse set of performance measures, along with other considerations, to inform decision making. The RIDM process acknowledges the role that human judgment plays in decisions, and that technical information cannot be the sole basis for decision making." According to NASA [45], risk management is the interaction of RIDM and continuous risk management (CRM): "The RIDM process addresses the risk-informed selection of decision alternatives to assure effective approaches to achieving objectives, and the CRM process addresses implementation of the selected alternative to assure that requirements are met."

As formulated in IAEA INSAG-25 (A Framework for an Integrated Risk Informed Decision Making Process) [49], probabilistic safety assessment (PSA) and probabilistic safety targets provide risk metrics to support decisions related to nuclear safety and to strengthen their basis. Many risk informed applications have been successfully employed for purposes of considering and comparing the safety of alternative design solutions and operating practice. Also, targets set for the probability of core damage and off-site releases have been found useful for assessing safety in an integrated manner. However, in any application the strengths and weaknesses of a PSA must be understood and taken into account. The increasing use of PSA techniques is being made in many countries because it provides valuable complementary

[8] t_p is the predicted wall thickness at the next scheduled examination.

[9] ASME states that the wall thickness of a short elbow "crotch region" shall be 20% greater than the minimum wall thickness required for the straight pipe.

insight, perspective, comprehension and balance to the deterministic safety assessment of nuclear installations. Experience has shown that an integrated decision making process, including deterministic and probabilistic analyses together with good engineering practices, consideration of operating experience and sound managerial arrangements, is effective in refining and improving safe design and safe operations of nuclear installations.

The U.S. NRC staff uses PSA models to support decisions regarding the appropriate response to a reported incident. Conditional core damage probability (CCDP) is calculated and is considered along with other factors (including uncertainty of the results) when determining the type of inspection team (an incident investigation team, an augmented inspection team, or a special inspection team) to send with a higher CCDP generally leading to a larger, more thorough inspection [50].

2.6 Significance Determination Process

The term "significance determination" (SD) is used by the U.S. Nuclear Regulatory Commission (NRC) when applying risk insights to determine the safety significance of inspection findings. The methodology for assigning risk metrics to operational event and inspection findings is formulated in SECY-99-007A [51]. The Significance Determination Process (SDP) uses change in core damage frequency (ΔCDF) and change in large early release frequency (ΔLERF) as risk metrics. This process provides an initial screening to identify those inspection findings that do not result in a significant increase in plant risk and thus need not be analyzed further (a "green" finding). Remaining inspection findings, which may have an effect on plant risk, are subjected to a more thorough risk assessment, using the next phase of the SDP. This more detailed assessment may involve NRC risk experts from the appropriate regional office and further review by the utility's plant staff. The final outcome of the review, evaluating whether the finding is green, white, yellow, or red, is used to determine further NRC actions that may be needed. The colors assigned SDP findings are defined as follows:

- Red (high safety or security significance) is quantitatively greater than 10^{-4} ΔCDF or 10^{-5} ΔLERF. Qualitatively, a Red significance indicates a decline in licensee performance that is associated with an unacceptable loss of safety margin. Sufficient safety margin still exists to prevent undue risk to public health and safety.
- Yellow (substantial safety or security significance) is quantitatively greater than 10^{-5} and less than or equal to 10^{-4} ΔCDF or greater than 10^{-6} and less than or equal to 10^{-5} ΔLERF. Qualitatively, a Yellow significance indicates a decline in licensee performance that is still acceptable with cornerstone objectives met, but with significant reduction in safety margin.
- White (low to moderate safety or security significance) is quantitatively greater than 10^{-6} and less than or equal to 10^{-5} ΔCDF or greater than 10^{-7} and less than or equal to 10^{-6} ΔLERF. Qualitatively, a White significance indicates an acceptable level of performance by the licensee, but outside the nominal risk range. Cornerstone objectives are met with minimal reduction in safety margin.
- Green (very low safety or security significance) is quantitatively less than or equal to 10^{-6} ΔCDF or 10^{-7} ΔLERF. Qualitatively, a Green significance indicates that licensee performance is acceptable and cornerstone objectives are fully met with nominal risk and deviation.

According to the NRC, the term "SDP" should be considered as an overall process description that includes all associated provisions designed to meet the Reactor Oversight Process (ROP) objectives, such as formal opportunities for licensee input (i.e., Regulatory Conferences), NRC management review for any significance characterization of greater than green (i.e., Significance and Enforcement Review Panel - SERP), and licensee appeal options (i.e. defined in IMC 0609, Attachment 2). The SDP is implemented using various cornerstone-specific SDP tools, which may be referred to by their specific names as individual "SDPs."

2.7 Risk Metrics & Risk Acceptance Criteria

In this technical report, the term "risk-informed operability determination" (RIOD) implies that the impact of a performance deficiency (PD) on operations, safety and risk is assessed relative to some unique set of plant availability/safety/risk metrics. The metrics of interest may be obtained by using a PSA model for nuclear applications, or using a quantitative risk assessment (QRA) model for non-nuclear applications. There are inconsistencies in the use of piping reliability metrics. The event sequence diagram in Figure 2-3 is a representation of the piping integrity risk "triplet." It addresses what can go wrong, how likely it is to happen, and what are the consequences of a structural failure.

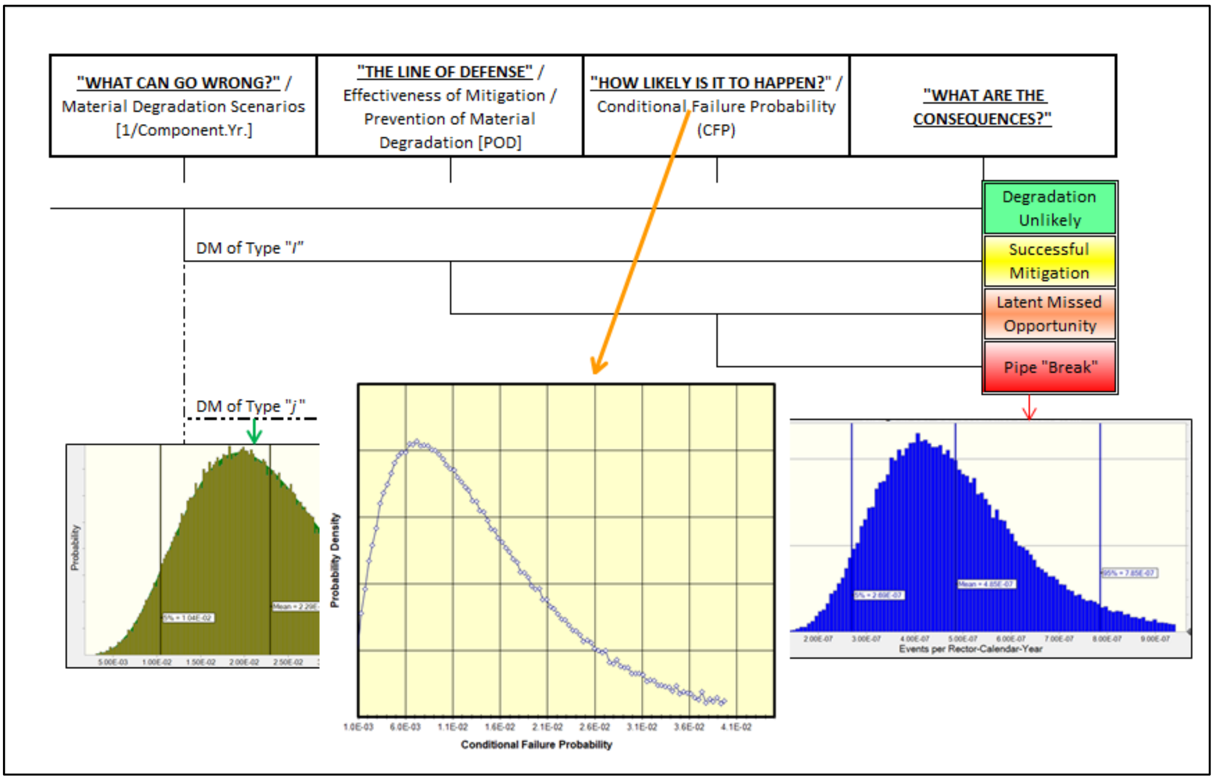

Figure 2-3: The Structural Integrity "Risk Triplet"[10]

The specific risk metric to be used is a function of the intended application [51]; e.g. consideration of potential aging effects or different inspection intervals on the structural reliability and related safety or risk significance requires different risk metrics to be used.

Two examples of nuclear industry consensus perspectives on risk metrics in the context of RIOD are found in the U.S. NRC Regulatory Guide 1.174 [53] and in EPRI TR-112657 [54]. Section 3.3 in the latter reference presents a methodology for assessing the risk impact of a pipe failure.

The instantaneous CDF varies with time as SSCs are taken in and out of service. A time dependent map of this instantaneous CDF is a representation of the risk profile as a function of time. The base CDF evaluated using a PSA model represents the average over this risk profile. The performance deficiency (PD) results in an additional increase in the instantaneous

[10] Note that each element of the "risk triplet" is represented by a probability density function.

CDF as a result of additional components being unavailable. The impact of the performance deficiency for SDP purposes is to be evaluated for the average unavailabilities of other unrelated SSCs, i.e., it is averaged over the risk profile using the following equation [55]:

$$CDF_{PD} = [T_0/(T_0 + T_1)] \times CDF_{Base} + [T_1/(T_0 + T_1)] \times CDF_1 \tag{2-1}$$

Where

CDF_{PD}	=	New CDF given that the performance deficiency has occurred
CDF_{Base}	=	Base PSA CDF
CDF_1	=	CDF conditional on the additional unavailability
T_1	=	Duration of the performance deficiency
$T_0 + T_1$	=	Nominal time over which the CDF is evaluated (1 year). If the PD is associated with an activity that only occurs at greater than 1 year intervals, $(T_1 + T_0)$ would be adjusted to the increased time between opportunities for the performance deficiency to result in the undesired consequence.

This calculation establishes a new increased average CDF that is interpreted as the CDF that would result if the performance deficiency were allowed to persist. Therefore, the change in CDF resulting from the performance deficiency is calculated by

$$\Delta CDF = CDF_{PD} – CDF_{base} = (CDF_1 – CDF_{base}) \times [T_1/(T_1 + T_0)] \tag{2-2}$$

When evaluating the change in CDF attributed to a degraded passive component, Equation (2-2) should be reformulated to explicitly address the effect on risk. As examples, a failed passive component could cause an initiating event such as “small” or “medium” loss-of-coolant-accident (LOCA) or internal flooding. Using risk-informed in-service inspection (RI-ISI) methodology, the change in CDF given a degraded piping component can be calculated as follows:

$$\Delta CDF = \Delta F_{SLOCA} \times CCDP_{SLOCA} + \Delta F_{MLOCA} \times CCDP_{MLOCA} \tag{2-3}$$

Where

ΔCDF	=	Change in CDF due to occurrence of degraded piping component. Could be a non-through-wall defect (e.g. circumferential crack) or minor leakage
ΔF_{SLOCA}	=	Change in SLOCA initiating event frequency due to occurrence of degraded condition
ΔF_{MLOCA}	=	Change in MLOCA initiating event frequency due to occurrence of leak event
$CCDP_{SLOCA}$	=	Conditional core damage probability given occurrence of a Small LOCA; this parameter is obtained from a plant-specific PSA
$CCDP_{MLOCA}$	=	Conditional core damage probability given occurrence of a Medium LOCA; this parameter is obtained from a plant-specific PSA

The change in the LOCA initiating event terms can be further defined using a DDM approach in which the frequency of a pipe rupture of a given size is expressed as the product of a pipe failure rate and a conditional probability of a rupture of a given size given pipe failure. Using this model as a basis, formulas for calculating the changes in the LOCA initiating event frequencies are as follows:

$$\Delta F_{SLOCA} = \Delta f_{CC1} \times CRP_{SLOCA} \tag{2-4}$$

$$\Delta F_{MLOCA} = \Delta f_{CC1} \times CRP_{MLOCA} \tag{2-5}$$

Where

Δf_{CC1}	=	Change in failure frequency due to a specific new degraded condition
CRP_{SLOCA}	=	Conditional rupture probability that the new degraded condition will produce a small LOCA
CRP_{MLOCA}	=	Conditional rupture probability that the new degraded condition will produce a medium LOCA

$$\Delta f_{CC1} = f_{CC1}^{new} - f_{CC1}^{old} \tag{2-6}$$

Where

$f_{CC1}^{new} =$	Plant-specific failure frequency based on the state-of-knowledge immediately after the discovery of a degraded condition
$f_{CC1}^{old} =$	Plant-specific failure frequency based on the state-of-knowledge prior to the discovery of a degraded condition

The failure frequencies on the right hand side of Equation (2-6) are obtained using a Bayes' update process where a prior distribution is derived based on relevant industry-wide operating experience. To characterize uncertainty in the failure frequency estimates, the gamma distribution may be selected. In order to provide a link to a data analysis performed, say, on the basis of CODAP and to maximize the uncertainty, the Constrained Non-Informative Distribution (CNID) method may be used [56]. The CNID method which permits an informed estimate of the mean of an uncertainty distribution but otherwise is non-informed to maximize uncertainty is comprised of the following steps.

- A gamma distribution is used to characterize uncertainty in the frequency estimates.
- The mean of the gamma distribution is set to the point estimate of the failure rate obtained from the service data, i.e. number of occurrences (n) divided by number of reactor-years (T).
- The alpha (α) and beta (β) parameters are set according to the following equations to apply the CNID method for the gamma distribution:

$$Mean = \alpha / \beta = n/T \tag{2-7}$$

$$\alpha = 0.5 \tag{2-8}$$

$$\beta = 0.5/(n/T) \tag{2-9}$$

If the gamma distribution is used as a prior distribution in a Bayes' update, there is plant specific data on the number of failures m and the plant's exposure t, the Poisson likelihood function is selected, the posterior distribution is also a gamma distribution with parameters α' and β' calculated as:

$$\alpha' = \alpha + m \tag{2-10}$$

$$\beta' = \beta + t \tag{2-11}$$

Appendix 3 of the ENSI PSA Guideline ENSI-A06/e [21] documents a procedure for probabilistic evaluation of operational events, which uses the incremental conditional core damage probability (ICCDP) as a measure of risk significance (Table 2-1). The $ICCDP_i$ of degraded or affected component "i" is estimated as:

$$ICCDP_i = (CCDF_i - CDF_{Base}) \times (\Delta t_i / 8760) \tag{2-12}$$

Where

Δt_i = Duration of component unavailability, or degraded condition
$CCDF_i$ = Conditional core damage frequency

Table 2-1: Relationship between ICCDP & INES-Scale

$ICCDP_i$	INES
$1 > \underline{ICCDP}_i \geq 1\times10^{-2}$	3
$1\times10^{-2} > \underline{ICCDP}_i \geq 1\times10^{-4}$	2
$1\times10^{-4} > \underline{ICCDP}_i \geq 1\times10^{-6}$	1
$1\times10^{-6} > \underline{ICCDP}_i \geq 1\times10^{-8}$	0

The U.S. NRC Regulatory Guide 1.174 [53] provides numerical acceptance guidelines for actions related to temporary risk, resulted from maintenance or other operational activities, random component failures, adverse weather or power grid conditions are recommended as follows:

- $10^{-4} < \Delta CDF < 5 \text{ x}10^{-4}$ - Temporary measures should be in place when restoring failed component(s).
- $5 \text{ x}10^{-4} < \Delta CDF < 10^{-3}$ - This configuration should be avoided. When non-voluntary entering, more cautions should be excised.
- $10^{-3} < \Delta CDF$ - Current industry consensus consider this as a high risk region, the reactor should be shutdown.

The offshore oil & gas industries in Norway and the United Kingdom are subject to risk regulation that includes requirements for facility-specific quantitative risk assessment (QRA). An overview of QRA as it applies to offshore installations is documented in References [57][58]. According to the paper by Mannan, Mentzer, Rocha-Valez and Minz [59]:

> "*… determining the impact of implementing new safety programs or of any strategy for that matter can only be done by having proper performance indicators that represent the risk one is trying to reduce. The importance of process-safety indicators is reflected in the many studies of the subject since the 1990s. A special issue of Safety Science was dedicated solely to safety indicators just in 2009, only 1 year before the Macondo disaster (Vol. 47, 2009)*[11]*. Perhaps one of the most extensive projects addressing safety indicators was the Risk Level Project (RNNP) in Norway. Leading and lagging indicators were used to assess annually the risk level of the Norwegian oil and gas industry.*"[12]

Advances have been made in developing PSA-centric risk metrics and software solutions that support real-time risk-informed operability determination of performance deficiencies.[13]

[11] Jon Espen Skogdalen, Ingrid B. Utne, Jan Erik Vinnem,"Developing Safety Indicators for Preventing Offshore Oil and Gas Deepwater Drilling Blowouts," Safety Science, **49**:1187-1199, 2009. The "Macondo Disaster" refers to the April 20, 2010 Deepwater Horizon Oil Spill in the Gulf of Mexico (for details, see the January 2011 report by the "National Commission on the BP Deepwater Horizon Oil Spill and Offshore Drilling" (www.oilspillcommission.gov). Blowout preventer (BOP) risk monitors have been developed to support the operability determination of degraded equipment.

[12] It was initiated in 1999-2000 timeframe in order to develop and apply a measuring tool which illustrates trends in risk level on the Norwegian continental shelf (NCS). The process plays an important role in the industry by contributing to a shared understanding of risk development by companies, unions and government agencies. For additional information, go to http://www.psa.no/about-rnnp/category911.html.

[13] See for example, "Framework of a Risk Monitor System for Nuclear Power Plant" (https://link.springer.com/chapter/10.1007/978-4-431-99779-5_58), and "Risk Monitors: The State of the Art in their Development and Use at Nuclear Power Plants" (https://www.oecd-nea.org/nsd/docs/2004/csni-r2004-20.pdf)

Many plants have implemented computer programs that enable the monitoring of plant risk as components or systems are taken out of service for repair or routine maintenance.

Risk acceptance criteria for piping may be derived from acceptance guidelines for core damage frequency. Reference [1] outlines the interdependence between pipe rupture and core damage frequency. This reference states that when the output from the 'xLPR' probabilistic fracture mechanics code corresponds to an annual Safety Class 1 total pipe rupture frequency less than 10^{-6} per year the Regulatory Guide 1.174 [53][14] criterion for acceptable increases in core damage frequency is always satisfied. Furthermore, the authors of Reference [1] recognize the inherent uncertainties in piping reliability and that the pipe rupture frequency is represented by a probability density function and that "the acceptance criterion of 10^{-6} failures per year was developed with the understanding that it is to be assessed against the 95% confidence level of mean results. When an analysis yields results that are close to the acceptance criteria it may be necessary to further address uncertainty."

2.8 The Dimensions of Piping Reliability Analysis

Summarized in Table 2-2 are the input data needed to estimate piping reliability. In most applications the parameter λ (pipe failure rate) forms one of several elements of a piping reliability model and the units of λ are per unit of time and component type. Usually the model of the λ-parameter is described by a Poisson process. The failure rate is the rate at which "failure" occurs in a specified time period.

In piping reliability analysis the term "failure" takes on many different meanings that depend on the purpose of an analysis, including the material properties, degradation susceptibilities and operating experience data. In the context of PSA "failure" usually implies a pipe break with a specific consequence; e.g. an initiating event that produces a certain through-wall mass flow rate. A PFM analysis might be performed conditional on a pre-existing defect given by crack length, orientation, aspect ratio and a "failure" may correspond to a through-wall perceptible leakage (i.e. wetness on outside diameter pipe wall). Illustrated in Figure 2-4 are the different types of piping failure manifestations.

Table 2-2: The Dimensions of Piping Reliability Analysis

Parameter	Dimension	Description
PSA Model Input Parameters		
$F(IE_x)$	[1/ROY]	(Initiating event) frequency of a pipe failure of size *x*, per reactor operating-year (ROY), subject to epistemic uncertainty & represented by a probability density function. The term ROY is the total time in operation with breakers closed to the station bus. The frequency of occurrence of an event is directly related to the likelihood of failure of a passive component that can initiate a certain consequence (or accident).
ρ_{ix}	[1/ROY]	Frequency of rupture of component type *i* with break size *x*, subject to epistemic uncertainty calculated via Monte Carlo simulation.
$P(R_x\|F_{ik})$	Probability	For leak-before-break (LBB) piping, the conditional rupture probability (CRP) of size *x* given failure of pipe component type *i* due to damage or degradation mechanism *k*, subject to epistemic uncertainty. Depending on the type of piping, this parameter may be determined on the basis of probabilistic fracture mechanics, expert elicitation or OPEX data.

[14] https://www.nrc.gov/docs/ML1731/ML17317A256.pdf

Parameter	Dimension	Description
λ_{ik}	[1/Weld.ROY] or [1/m.ROY] or [1/Elbow.ROY] etc.	Failure rate per "location-year" for pipe component type *i* due to failure mechanism *k*, subject to epistemic uncertainty. This term is a representation of the susceptibility of a given piping component to material degradation.
Piping Reliability Model Input Data (Typical)		
m_i		Number of pipe welds (or fittings, segments or inspection locations of type *i*; each type determined by pipe size, weld type, applicable damage or degradation mechanisms.
I_{ik}	Ratio (of hazard rates)	Reliability and integrity management (RIM) factor for weld type *i* and failure mechanism *k*, subject to epistemic uncertainty.
n_{ik}		Number of failures in pipe component of type *i* due to degradation mechanism *k*. The component boundary used in defining exposure terms is a function of the susceptibility to certain damage or degradation mechanisms..
τ_{ik}		Component exposure population for welds of type *i* susceptible to degradation mechanism *k*.
f_{ik}		Estimate of the fraction of the component exposure population for piping component type *i* that is susceptible to degradation mechanism *k*.
N_i		Estimate of the average number of pipe components of type *i* per reactor in the reactor operating years of exposure for the data query used to determine n_{ik}. Determined from isometric drawings reviews for a population of plants and expert knowledge of degradation mechanisms.
T_i	ROY	Total exposure in reactor operating years (ROYs) for the failure population for component type *i*. $\tau_{ik} = f_{ik} \times N_i \times T_i$

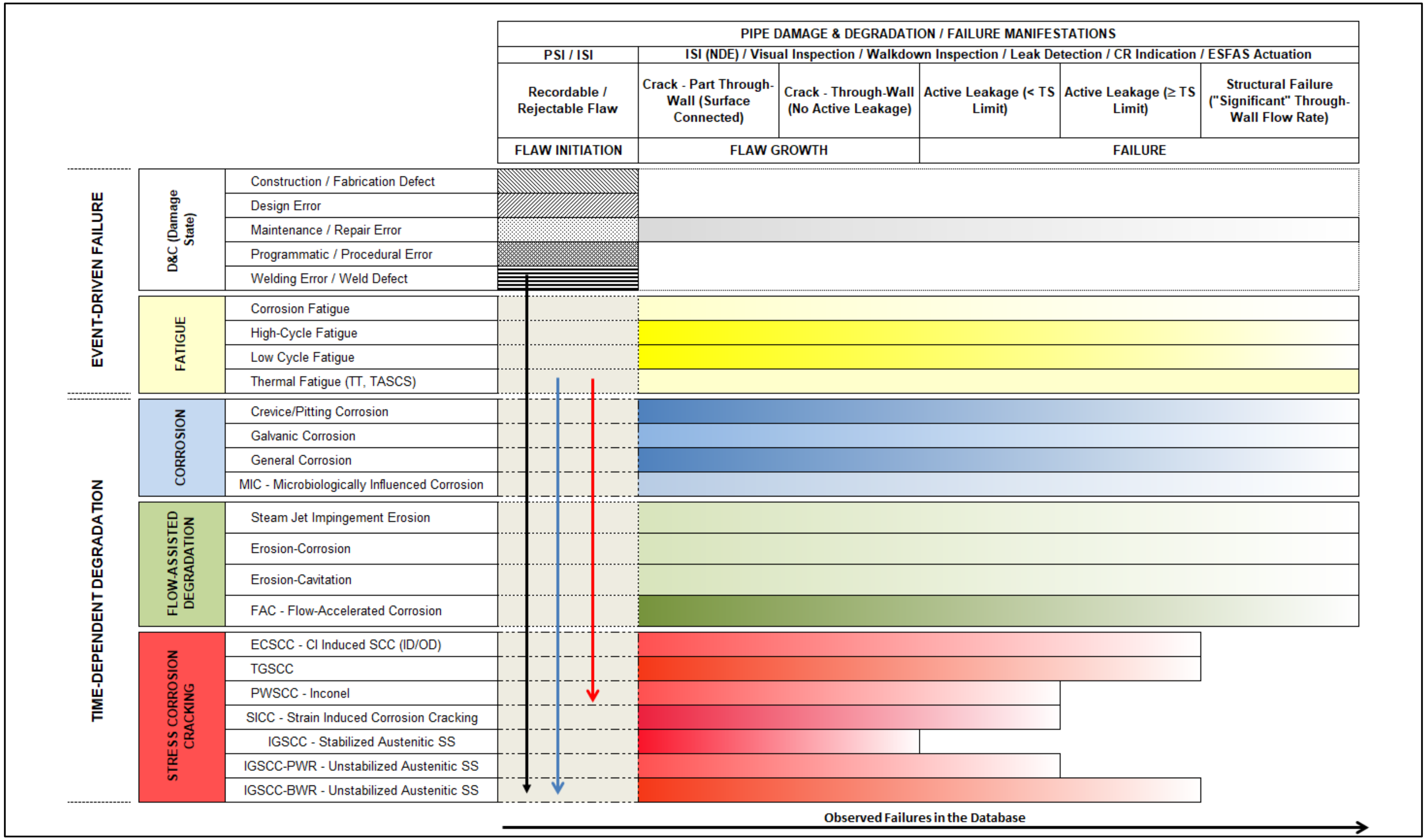

Figure 2-4: The Different Types of Pipe Failure Manifestations[15]

[15] The vertical arrows indicate the presence of potential or actual *synergistic effects*. For example, thermal fatigue may cause crack initiation, and crack propagation may occur via IGSCC. The fill-effects in the colorized horizontal bars are commensurate with the observed event populations; i.e. a strong fill corresponds ‘multiple’ events and a weak fill corresponds ‘relatively few’ major structural failures. TS = Technical Specification Requirements.

3. SURVEY OF OPERABILITY DETERMINATION APPROACHES

The results of a survey of operability determination schemes and techniques that are summarized in this section address discrete and holistic approaches to the determination of the safety significance of a degraded passive component. The term “degraded” implies a non-conforming material condition; it can be a non-through-wall flaw (crack or thinning) or a through-wall flaw that results in a direct operational impact.

3.1 Scope of the Survey

The scope of the survey is on operability determination and fitness-for-service assessment, (FFS) respectively, and its consideration of operating experience data, risk impact assessment and aging factors. Included in the survey are operability determination evaluation practices of non-nuclear industries, including:

- Oil & Gas Industry (Offshore Exploration & Drilling Operations). Survey of guidelines developed by the Norwegian Petroleum Safety Authority (PSA)[16] and the British Health and Safety Executive (HSE). In 1981 the Norwegian Petroleum Directorate (NPD) issued regulations for the safety evaluation of offshore platform conceptual design. These regulations contained a cut-off criterion of 10^{-4} per platform-year as the frequency limit for accidents that needed to be considered in order to define so called Design Accidental Events. Similarly, in 1992 the HSE issued the “Offshore Installations Safety Case Regulations,” which included requirements for “risk assessment demonstration.” The PSA and HSE safety regimes provide a technical basis for risk-informed operability determinations.
- Oil Refinery Industry. Survey of guideline on fitness-for-service developed by the American Petroleum Institute (API).
- Occupational Safety and Health. The U.S. “National Emphasis Program” policies and procedures for implementing a National Emphasis Program (NEP) to reduce or eliminate the workplace hazards associated with the catastrophic release of highly hazardous chemicals at petroleum refineries.
- U.S. Environmental Protection Agency (EPA). The survey addresses operability determination under the EPA Spill Prevention, Control, and Counter-measure (SPCC) rule.
- National Aeronautical and Space Agency (NASA). The survey addresses NASA-STD-8719.17A (NASA Requirements for Ground-Based Pressure Vessels and Pressurized Systems).

The purpose of an “operability determination” is to perform a systematic assessment of a pre-existing degraded condition to determine its potential impact on the functional capability of a system or component. Also implied are technical considerations about the resolution of degraded and nonconforming Conditions. The scope of the survey of the different operability determination concepts are summarized in Table 3-1.

[16] The NPD was created in 1972 with responsibility for managing petroleum resources. From January 1, 2004 the organizational division of the NPD with responsibility for labor and safety issues was made a separate agency, the Petroleum Safety Authority Norway (http://www.ptil.no/?lang=en_US).

Table 3-1: Scope of the Survey of Operability Determination Principles & Practices

Concept / Activity	Reference(s)	Definition
Standard Technical Specifications (STS)	--	In the US, STS have been published for each of the five domestic reactor types as a NUREG-series publication. Improved Standard Technical Specifications (STS) were developed based on the criteria in the Final Commission Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, dated July 22, 1993 (58 FR 39132). The plant-specific Technical Specifications (TS) define completion times from which it is possible to determine an appropriate time from within which an operability determination should be complete. In the STS, the surveillance frequencies and allowed outage times have been determined on a deterministic basis using engineering judgment.
Risk Management Technical Specifications	--	Based on the guidance in Regulatory Guide 1.177 and Standard Review Plan Chapter 16.1 (NUREG-0800) risk-informed amendments may be pursued to set risk informed completion times (CTs) and surveillance frequencies (SFs) using a three-tiered approach.
Operability Determination	[2][3][4][5][6][7]	A formal process, as defined in the NRC Inspection Manual. Part 9900, used to assess TS required structures, systems or components (SSCs) and their support functions for compliance with TS when a degraded or nonconforming condition is identified for a specific SSC for an associated necessary and related support function.
Operability Requirements	--	According to US regulation, the Technical Specifications are required to include limiting conditions for operation (LCO) and defines LCO as the lowest functional capability or performance levels of equipment required for safe operation of the facility. The regulation requires that when an LCO of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial actions permitted by the TS until the condition can be met. The operability determination process is used to assess TS required SSCs and their support functions for compliance with TS when a degraded or nonconforming condition is identified for a specific SSC for an associated necessary and related support function.
Short-Term Operability Evaluation	[4][6][7]	Generic Letter 90-05 provides guidance for temporary repair of moderate energy piping. Such repairs are applicable until the next scheduled outage exceeding 30 days, but no later than the next scheduled refueling outage. The guidance in the Generic Letter applies when a flaw is detected during plant operation. If a flaw is detected during a scheduled shutdown such as planned maintenance outage or refueling outage, a code repair is required before plant restart. According to Generic Letter 90-05, the integrity of a temporary non-Code repair of Class 3 piping should be assessed at least every 3 months by a suitable NDE method. This examination should involve the application of ultrasonic testing (UT) or radiographic testing (RT). Furthermore, a qualitative assessment of leakage through the temporary non-Code repair should be performed at least weekly during plant walkdown inspections to determine any degradation of structural integrity. The licensee should perform an engineering evaluation to assess the rate and extent of the degradation to determine what remedial measures are required. A temporary non-Code repair is no longer valid if the structural integrity is not assured.

Concept / Activity	Reference(s)	Definition
	ASME [10][12]	Refer to Code Case Reference Table in Section 3.1 of this report. Article IWA-4000 of Section XI of the ASME Code describes the Code repair procedures. A code repair requires the removal of the flaw and a subsequent weld repair. The repair weld is subject to post-repair NDE and a post-repair pressure test may also be required. A Code repair is practical during a scheduled shutdown. If a flaw is detected during plant operation, the plant may have to be shutdown to perform a code repair. To avoid a plant shutdown and to limit the leakage from a through-wall flaw in moderate energy piping, plant operators use temporary non-code repairs such as clamps with rubber gasketing, encapsulation of leaking pipes in cans using liquid sealants, or weld overlays. Temporary non-Code repairs are not permitted on ASME Code piping without prior relief from the NRC. For Class 2 and 3 piping, a plant operator is required to perform code repairs or request relief for temporary repairs on a case-by-case basis regardless of pipe size.
Long-Term Operability Evaluation	U.S.NRC [5][6][7][8]	Generic Letter 91-18 contains specific guidance for long-term operability by requiring that the degraded or non-conforming SSC be brought back into complete compliance with the licensing basis requirements. This task requires the repair or replacement of the SSC in accordance with the ASME Section XI requirements.
	ASME [10]	ASME Section XI, Subarticles IWB-3132.3, IWB-3142.4, IWC-2122.3, IWC-3122.3, IWE-3122.3, IWF-3112.3, IWF-3122.3 and IWL-3212 provide acceptance criteria for evaluations of nonconforming conditions.
	Norwegian Oil & Gas Association [13]	Barrier elements, such as wells, subsea templates, pipelines, risers etc. shall be qualified for the new planned service conditions and life extension. Quantitative analysis shall be used for equipment where known degradation mechanisms are dominating and where models exist to calculate degradation, remaining margins and prediction of remaining service life.
Probabilistic Operability Evaluation	NEI [14][15]	Section 11 of NUMARC 93-01 provides guidance for the development of an approach to assess and manage the risk impact expected to result from performance of maintenance activities. Assessing the risk means using a risk-informed process to evaluate the overall contribution to risk of the planned maintenance activities. Managing the risk means providing plant personnel with proper awareness of the risk, and taking actions as appropriate to control the risk. The assessment is required for maintenance activities performed during power operations or during shutdown. Management of risk involves consideration of temporary risk increases, as well as aggregate risk impacts.
Significance Determination	U.S. NRC [16]	The Significance Determination Process (SDP) uses risk insights, where appropriate, to help NRC inspectors and staff determine the safety significance of inspection findings identified within the seven cornerstones of safety at operating reactors. The SDP is a risk-informed process and the resulting safety significance of findings, combined with the results of the risk-informed performance indicator (PI) program, are used to define a licensee's level of safety performance, and to define the level of NRC engagement with the licensee.

Concept / Activity	Reference(s)	Definition
	NEI [17]	Proposes the use of different risk metrics relative to the NRC SDP to provide more realism in the determination of changes in risk profile due to a degraded condition.
Accident Precursor Analysis (Event Analysis)	U.S. NRC [17][18][19][20]	Application of a PSA model to determine to calculate conditional risk measures to determine the safety significance of an event (i.e. equipment failure).
	ENSI [21]	ENSI-A06/d provides technical guidance for how to perform risk evaluations of operating experience. The risk significance of a given event is determined by calculating conditional risk measures.
Fitness-for-Service Assessment		
American Petroleum Institute (API)	[23][24][25][26] [26]	Fitness-for-service assessment is a multi-disciplinary engineering approach that is used to determine if equipment is fit to continue operation for some desired future period. The equipment may contain flaws, have sustained damage, or have aged so that it cannot be evaluated by use of the original construction codes. API 579-1 is a consensus industry recommended practice that can be used to analyze, evaluate, and monitor equipment for continued operation. Includes a quantitative engineering evaluation to determine the structural integrity of pressure retaining equipment containing a flaw. In March 2010, API published Recommended Practice 754 which identifies leading and lagging process safety indicators that are useful for driving performance improvement. The indicators are divided into four tiers that represent a leading and lagging continuum. Tier 1 is the most lagging and Tier 4 is the most leading. Tiers 1 and 2 are suitable for nationwide public reporting and Tiers 3 and 4 are intended for internal use at individual sites.
Norwegian Petroleum Safety Authority (PSA)	[28][28]	Regulation 17551/1/08 [27] provides guidance on "Fitness for Service and Remnant Life Assessment" and makes reference to API 579-1 [22], BS 7910 [30] and DNV RP-F101 [28]. The assessments focus on wall thinning and fatigue damage.
United Kingdom Health & Safety Executive (HSE)	[30][31][31]	Contract Research Report 363/2001 [31] provides an overview of FFS assessment methodologies including API 579-1 and British Standard BS 7910. The latter uses a failure assessment diagram (FAD) methodology derived from fracture mechanics. Per HSE directive, equipment that has deteriorated to a condition assessed to be unacceptable requires immediate action before it can re-enter service. Where equipment has deteriorated but has not reached unacceptable limits, monitoring or shorter inspection intervals or other action may be required depending on the rate at which the deterioration is proceeding and the confidence with which this rate is known.
U.S. Environmental Protection Agency (EPA)	[33][34][34]	The U.S. Environmental Protection Agency's (EPAs) "Spill Prevention, Control, and Countermeasure" (SPCC) Guidance Chapter 7 ("Inspections, Evaluations, and Testing") provides guidance for FFS assessment and references API 579-1. The scope of the SPCC Guidance covers the prevention, prediction and detection of structural issues before they cause a leak, spill or discharge of oil to navigable waters or adjoining shorelines.

<table>
<tr><th>Concept / Activity</th><th>Reference(s)</th><th>Definition</th></tr>
<tr><td>CANDU Owners Group Inc. (COG)</td><td>[36][37][38][39][39]</td><td>The COG Feeder Integrity Joint Program, the 'Fitness for Service Guidelines for Feeders Affected by Wall Thinning was developed and subsequently conditionally approved by CNSC. The assessment of feeders is based on the requirements of the construction Code (Section III of the ASME Boiler and Pressure Vessel Code). Satisfying the requirements of NB-3650 for design and service loadings are sufficient for continued service and extended life if the predicted minimum wall thickness of the component is ≥ 90% of the pressure based thickness calculated as per NB-3641. Realistic stress indices can be obtained by detailed finite element analysis of the feeder bends, which can be used in meeting the requirements of NB-3650. If the predicted minimum wall thickness is < 90% of the thickness value of NB-3641, a detailed finite element analysis of the thinned feeder bend is undertaken to qualify the feeder as per the rules of NB-3221 to meet the pressure design requirements as per NB-3324. The lowest thickness value permitted by the FFSG, if the structural factors in the FFSG that were derived from Section XI of the ASME B&PV Code can be shown to be satisfied by a detailed finite element assessment, is 75% of the pressure based thickness.</td></tr>
<tr><td rowspan="2">EDF Energy R6-Procedure</td><td rowspan="2">[41][67][67]</td><td>Originally developed by the Central Electricity Generating Board (CEGB) of the UK, the R6-Procedure (“Assessment of the Integrity of Structures Containing Defects”) is used to estimate:
• Limiting load to avoid failure of a structure containing a known or postulated flaw.
• The limiting flaw size of a structure subjected to a specified loading condition
• The margins of reserve factors on the assessed condition, compared with the limiting load & limiting flaw size.
• The sensitivity of these margins and factors to the assessed conditions and assumptions.
Structural integrity is determined on the basis of a Failure Assessment Diagram (FAD). This diagram is divided in two parts by a segment called “Failure Assessment Line” (FAL).</td></tr>
<tr><td>The Bhabha Atomic Research Centre has developed a probabilistic fracture mechanics assessment procedure for the R6 method for application to the Primary Heat Transport System piping.</td></tr>
<tr><td>U.S. Occupational Safety and Health Administration (OSHA)</td><td>[42]</td><td>Directive Number CPL 03-00-010 (Petroleum Refinery Process Safety Management National Emphasis Program, 2009) makes reference to API RP 579 for fitness-for-service assessments.</td></tr>
<tr><td>National Aeronautical and Space Administration (NASA) System Safety & Risk Assessment Requirements</td><td>[44][45][46][47]</td><td>NASA-STD-8719.17A establishes tailored System Safety Requirements for pressure vessels and pressurized systems (PVS). Risks shall be identified and documented for all PVS within the scope of the standard, the risk status shall be updated during the certification/recertification process, and new risks shall be identified as appropriate throughout the life of a PVS. Tables 1 through 4 of the standard provide guidance for the "Risk Assessment Code" (RAC) determination.</td></tr>
</table>

Concept / Activity	Reference(s)	Definition
Structural Integrity Management (SIM)	[48]	SIM is a life cycle process for ensuring the continued fitness-for-service of offshore platforms, pipelines, wells, and onshore platforms. ISO 19901-3:2010 gives requirements for the design, fabrication, installation, modification and SIM for the topsides structure for an oil and gas platform. It complements ISO 19902, ISO 19903, ISO 19904-1, ISO 19905-1 and ISO 19906, which give requirements for various forms of support structure.
Remaining Service Life (RSL) Assessment of Thinned Pipe Wall	--	FFS assessment in its simplest form, a RSL assessment is performed whenever a flow-accelerated corrosion (FAC) inspection program reveals a thinned pipe wall area. A typical RSL assessment procedure includes the following steps: • Initial thickness of a component is determined by ultrasonic inspection prior to the component being placed in service or in the first ultrasonic inspection during its service life. If an examination has not previously been performed on the component, the initial thickness is determined by reviewing the initial ultrasonic data for that component. The area of maximum wall thickness within the same region as the worn area is identified. If the thickness is greater than the nominal component wall thickness, the maximum wall thickness within the relevant area is used as the initial thickness. If that thickness is less than the nominal wall thickness, the nominal wall thickness is used as the initial thickness. • The projected wear rate is calculated by dividing the wear by the time between measurements or the time between when the component was placed in service and the time of the measurement. Wear is the amount of material removed or lost from a components wall thickness since baseline or subsequent to being placed in service and time is the actual plant operating hours, although calendar hours may be used for conservatism. • The RSL is determined by subtracting the minimum acceptable wall thickness from the actual measured wall thickness, then dividing by the wear rate times a safety factor of 1.1. • If the RSL of a component is greater than or equal to the number of hours in the next operating cycle, the component may be returned to service. If the component's RSL is greater than the number of hours in the next operating cycle but is less than the number of hours in the next two operating cycles, the component should be considered for re-inspection, repair or replacement during the next scheduled outage. If the component is acceptable for continued service, it shall be re-examined before, or during the cycle during which it is projected to wear to the minimum acceptable wall thickness.
In this table, information about Technical Specifications (TS) is included for the sake of completeness. The TS define completion times (CT) and allowed outage time (AOT) for equipment taken out of service for maintenance, repair or replacement. Many plants have implemented risk-informed TS and risk monitor software that account for the risk impact of degraded or non-conforming equipment. Implicitly, the risk-informed TS applications generate conditional risk metrics for degraded or non-conforming equipment. Prepared by the Nuclear Energy Institute, the draft NEI 18-03 (Operability Determination) includes proposed guidance for operability determination in a broad sense. The topic of TS and related operability determination considerations is given no further treatment in this report, however.		

3.2 Survey Format

Illustrated in Figure 3-1 is an event sequence diagram (ESD) of the operability determination and fitness-for-service (FFS) evaluation regimes. A convention used herein is that operability determination in a broad sense addresses a degraded condition in the context of method of detection, flaw characterization and operational impact. The flaw characterization activity entails quantitative engineering assessments to demonstrate the structural integrity of an in-service passive component containing a flaw. As used in the nuclear, oil & gas, refining and petrochemical industries, the FFS activity focus on demonstration of structural integrity. Also included in the FFS activity is consideration of long-term operation and the implications of aging structures.

Using ASME Section XI terminology, an FFS evaluation is "triggered" by a discovery of a recordable or rejectable flaw. This is indicated by the shaded area in Figure 3-1. In contrast, an operability determination usually is initiated to determine compliance with Technical Specifications when a degraded or nonconforming condition is identified for a specific SSC. Examples of such conditions include situations where a non-destructive examination (NDE) fails to detect a flaw and that particular flaw grows in the through-wall direction resulting in active leakage and a forced plant shutdown. This is indicated as sequences 16 through 18 in Figure 3-1.

3.3 Nuclear Industry Guidelines

An operability determination (OD) is associated with SSCs described in the plant technical specifications (TS) and forms the basis for compliance with regulatory requirements and limiting conditions for operation (LCO). The scope of SSCs considered within the operability determination process include: SSCs required to be operable by TS and SSCs that are not explicitly required to be operable by TS, but that perform required support functions. Conversely, "functionality assessments" are performed for SSCs not described in the plant TSs. From a practical standpoint these distinctions serve as a means differentiating the evaluation processes employed to assess the fitness for service of safety related and non-safety related SSCs. In fact the fundamental basis for either "operability" or "functionality" rests in the measure of the SSC's capability to perform its intended function(s).

The U.S. NRC Standard TS[17] define "operable/operability" as follows: "A system, subsystem, train, component, or device shall be operable or have operability when it is capable of performing its specified safety functions, and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s)."

Several variations of the preceding definition may exist in the plant-specific Technical Specifications. In all cases, however, a licensee's plant-specific definition is accepted as governing how one applies the terms operable and operability. The specified functions referenced in the foregoing definition are the specified "safety" functions described in the current licensing basis for the facility. The following are some examples of specified safety functions for piping:

[17] For details, go to http://www.nrc.gov/reactors/operating/licensing/techspecs.html

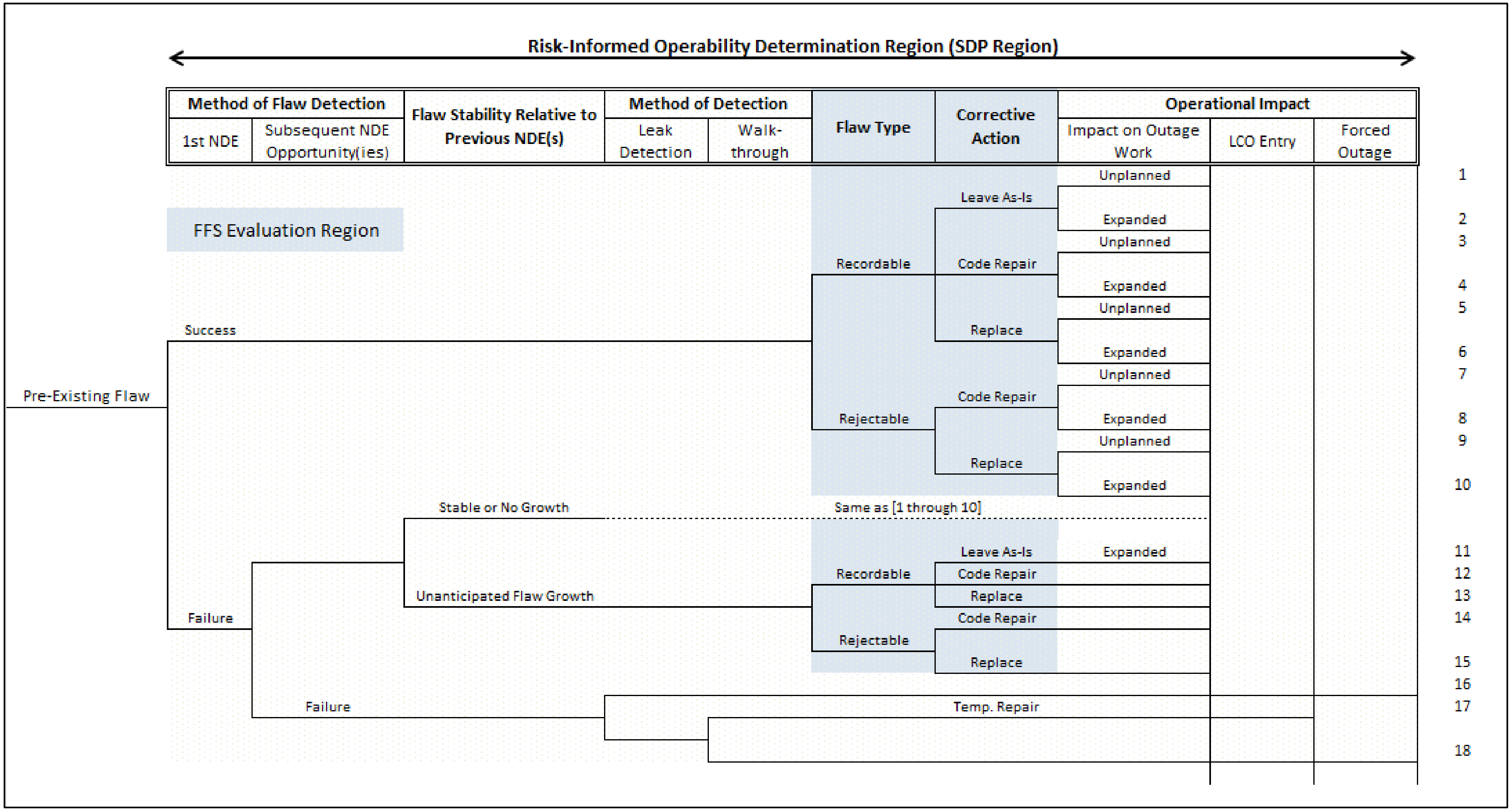

Figure 3-1: Operability & FFS Assessment Regimes

- Structural integrity where "structural failure" would interfere with other systems being able to perform their safety functions;
- Pressure integrity to the extent that leakage is limited to levels permitted by the licensing basis and through-wall flaws remain stable when subjected to faulted loads; and
- Ability to pass required flow rates.

In addition to providing the specified safety functions, a system, subsystem, train, component, or device (referred to as system in this section) is expected to perform as designed, tested, and maintained. When system capability is so degraded that it cannot perform with reasonable certainty or reliability, the system should be judged inoperable even if it is shown to provide the specified safety functions. Required action ranges included in the acceptance criteria of ASME Operation and Maintenance (O&M) Code [60] and ASME Section XI [10] are examples of degraded capabilities for SSCs. Plant Technical Specifications also contain limiting values, such as leakage rate and set point pressure, for component performance. These values constitute the technical specification-based operability verification criteria that, if they are not met, necessitate the entering of the applicable Limiting Condition for Operation (LCO) and Allowed Outage Times (AOTs).

In some cases, the 'ASME O&M Code–required' (for active mechanical components) or 'Section XI–required' action ranges for certain passive components may be more conservative than the plant technical specification limits. However, the component in question must be declared inoperable even if the existing performance meets the technical specification safety limit because of the imposed ASME operability limit. An example is a pump that is capable of delivering rated flow but exhibits vibration in excess of the reference values and falls in the required action range.

ASME Section XI specifies code-acceptable repair methods for flaws that exceed code acceptance limits in piping that is in service. A code repair is required to restore the structural integrity of flawed ASME Code piping, independent of the operational mode of the plant when the flaw is detected. Those repairs not in compliance with Section XI of the ASME Code are non-code repairs. However, the required code repair may be impractical for a flaw detected during plant operation unless the facility is shut down. In 1990 the U.S. NRC issued Generic Letter 90-05 (Guidance for Performing Temporary Non-Code Repair of ASME Code Class 1, 2 and Piping) [4]. This generic letter provides guidance that will be considered by the NRC staff in evaluating relief requests submitted by licensees for temporary non-code repairs of code Class 3 piping. Generic Letter 90-05 [4] has also been adopted by the regulatory bodies of Chinese-Taipei [61] and Spain[18].

Temporary non-code repairs are applicable until the next scheduled outage exceeding 30 days, but no later than the next scheduled refueling outage. This guideline applies when a flaw is detected during plant operation. If a flaw is detected during a scheduled shutdown, a code repair is required before plant restart. When the leakage in the class 3 moderate energy pipe components is identified, the flaw geometry must be characterized by volumetric inspection. The flaw geometry has to be considered to account for measurement uncertainties and limitations. The structural integrity of the flawed piping is assessed by fracture mechanics approach provided by Generic Letter 90-05.

Article IWA-4000 of Section XI of the ASME Code describes the code repair procedures. A code repair requires the removal of the flaw and a subsequent weld repair. The repair weld is subject to post-repair nondestructive examination and a post-repair pressure test may also be required. A code repair is practical during a scheduled shutdown. If a flaw is detected during plant operation, the plant may have to be shut down to perform a code repair. To avoid a plant

[18] https://www.oecd-nea.org/download/codap/CODAPMeetings1.html; restricted & password protected access.

shutdown and to limit the leakage from a through-wall flaw, some licensees have used temporary non-code repairs such as clamps with rubber gasketing, encapsulation of leaking pipes in cans using liquid sealants, or weld overlays. Temporary non-code repairs are not permitted on ASME Code piping without prior relief from the NRC.

For code Class 1 and 2 piping, a licensee is required to perform code repairs or request NRC to grant relief for temporary non-code repairs on a case-by-case basis regardless of pipe size. Temporary non-code repairs of code Class 1 and 2 piping must have load-bearing capability similar to that provided by engineered weld overlays or engineered mechanical clamps. Licensee requests based on repairs such as encapsulation of leaking pipes in cans using liquid sealants, clamps with rubber gasketing, or non-engineered weld overlays (patches) will not be approved by the staff.

Engineered weld overlays or engineered mechanical clamps are designed to meet the load bearing requirements of the piping, assuming that the flaw is completely through the wall for 360-degrees, that is, all around the pipe circumference, at the location of the flaw. Engineered weld overlays and engineered mechanical clamps are discussed in Generic Letter 88-01 [62].

For ASME Code Class 3 piping, a licensee is also required to perform code repairs or request NRC to grant relief for temporary non-code repairs on a case-by-case basis regardless of pipe size. Because of the rather frequent instances of small leaks in some Class 3 systems, such as service water systems, the staff is providing guidance in Enclosure 1 that will be considered by the staff in evaluating relief requests for temporary non-code repairs of code Class 3 piping. The guidance for code Class 3 piping in Enclosure 1 consists of assessing the structural integrity of the flawed piping by a flaw evaluation and assessing the overall degradation of the system by an augmented inspection. In addition, licensee evaluation should consider system interactions such as flooding, spraying water on equipment, and loss of flow. Furthermore, temporary non-code repairs should be evaluated for design loading conditions.

Temporary non-code repairs of code Class 3 piping in high energy systems, that is, the maximum operating temperature exceeds 200° F or the maximum operating pressure exceeds 275 psig, must have load-bearing capability similar to that provided by engineered weld overlays or engineered mechanical clamps. Licensee requests for high energy Class 3 piping based on repairs such as encapsulation of leaking pipes in cans using liquid sealants, clamps with rubber gasketing, or non-engineered weld overlays (patches) will not be approved by the staff. For temporary non-code repairs of code Class 3 piping in moderate energy systems, that is, other than high energy systems, the licensee may consider non-welded repairs. Furthermore, the structural integrity of the temporary non-code repair of code Class 3 piping should be assessed periodically.

For ASME Code Class 3 piping, two specific flaw evaluation approaches should be considered, namely, the "through-wall flaw" and the "wall thinning" approaches. If the flaw is found acceptable by the "through-wall flaw" approach, a temporary non-code repair may be proposed. If the flaw is found acceptable by the "wall thinning" approach, immediate repair is not required but the licensee should comply with the guideline for repair and monitoring. An augmented inspection is a part of the relief acceptance criteria. The extent of the augmented inspection is more stringent for high energy lines than for moderate energy lines because of the potential for more severe failure consequences. The Generic Letter 90-05 [4] evaluation guideline on temporary repairs consists of four parts:

1. Flaw detection during plant operation and impracticality determination. “Impractical” means that affected section of piping cannot be isolated for completing a Code repair within the Technical Specification Limiting Condition for Operation (LCO) without a plant shutdown. For example an LCO may prescribe that an affected SW train must be

restored to operable status within 72 hours or the reactor be placed in at least hot standby within the next 6 hours.

2. Root cause determination and flaw characterization. This determination should include a positive identification of root cause and an identification of the most susceptible locations in the piping system.
3. Flaw evaluation. The flawed piping should satisfy the criteria of one of two approaches, the “through-wall flaw” approach or the “wall thinning” approach for non-through wall flaws.
4. Augmented inspection. If the flaw is evaluated and found acceptable, the plant operator should perform an augmented inspection via ultrasonics or radiography to assess the overall degradation of the system. The augmented inspection, performed within 15 days of detection of the flaw, which results in a temporary non-Code repair, is part of the relief acceptance criteria of the repair. The inspection should be performed for at least the 5 most susceptible (and accessible) locations.

Flaws detected in the augmented inspection should be characterized and evaluated. If any flaw is detected having a minimum measured wall thickness, t_{meas}, less than the Code-required minimum wall thickness, t_{min}, in the augmented inspection sample, inspection of an additional sample of the same size should be performed. This process should be repeated within 15 days of each other until no flaw having $t_{meas} < t_{min}$ is detected in the additional inspection sample or until 100% of susceptible (and accessible) locations have been inspected.

In addition to the approach defined in Generic Letter 90-05, a number of alternatives are currently available and conditionally approved by the NRC for evaluating and repairing of piping wall thinning and pitting, including through-wall leaks (Table 3-2). Where Code Class 3 SW piping is included in a plant’s ISI program for flow accelerated corrosion (FAC), ASME Code Case N-561, N-562, N-597, or N-661 may be applied in lieu of Generic Letter 90-05. In addition, ASME Code Case N-523-1 may be applied in lieu of Generic Letter 90-05:[19]

- Code Case N-523-1. Mechanical Clamping Devices for Class 2 and 3 Piping.
- Code Case N-561. Alternative Requirements for Wall Thickness Restoration of Class 3 Moderate Energy Carbon Steel Piping.
- Code Case N-562. Alternative Requirements for Wall Thickness Restoration of Class 2 and High Energy Class 3 Carbon Steel Piping.
- Code Case N-597. Requirements for Analytical Evaluation of Pipe Wall Thinning.
- Code Case N-661. Alternative Requirements for Wall Thickness Restoration of Classes 2 and 3 Carbon Steel Piping for Raw Water Service.

These ASME Code Cases are not available for application at nuclear power plants without specific NRC approval. A formal FFS evaluation is required and to be available for onsite inspection by NRC Inspectors.

[19] The SW piping is susceptible to wall thinning through various flow-assisted degradation mechanisms. Hence, NDE technologies known to be effective in the monitoring FAC also would be effective in the monitoring of Service Water piping degradation.

Table 3-2: Selected ASME Code Cases for Temporary Repair of Ferritic Piping

Code Case	Title	Applicability	Comment
N-463	Evaluation Procedures and Acceptance Criteria for Flaws in Class 1 Ferritic Piping That Exceed the Acceptance Standards of IWB-3514.2, Section XI, Division 1	Provides acceptance criteria for engineered weld overlays of ferritic piping.	Approved in 1998.
N-513	Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 3 Piping	Permits through-wall flaws in moderate energy Class 3 straight piping for a limited time not exceeding the time to the next scheduled outage, if it can be demonstrated that adequate pipe integrity and leakage containment are maintained. The Code Case allows the licensee to perform a flaw growth analysis to establish the allowable time for temporary operation. Periodic examinations of no more than 90-day intervals shall be conducted to verify the analysis.	Approved in 1997.
N-513-1		Code Case N-513-1 revises the base case to expand the temporary acceptance methodology from Class 3 moderate energy piping to Class 2 and 3 moderate energy piping. Both cases provide requirements which may be followed for temporary acceptance of flaws in ASME Section III, ANSI B31.1, and ANSI B31.7 piping designated as Class 2 or 3. The allowable time for the temporary repair may not exceed the time to next scheduled refueling outage.	Approved in 2004.
N-513-2		Relative to N-513-1, this revision includes an improved flaw evaluation procedure for carbon steel piping. Approved in 2004.	Approved in 2004.
N-513-3		Does not allow evaluation of flaws located away from attaching circumferential piping welds that are in elbows, bends, reducers, expanders, and branch tees. ASME Code Case N-513-3 also does not allow evaluation of flaws located in heat exchanger external tubing or piping.	Approved in 2009
N-513-4		Relative to previous revisions, the following is a high-level summary of changes: • Revised the maximum allowed time of use from no longer than 26 months to the next scheduled refueling outage. • Added applicability to piping elbows, bent pipe, reducers, expanders, and branch tees where the flaw is located more than $(R_o t)^{1/2}$ from the centerline of the attaching circumferential piping weld. • Expanded use to external tubing or piping attached to heat exchangers. • Revised to require minimum wall thickness acceptance criteria to consider longitudinal stress in addition to hoop stress. • Daily walkdown requirement for through-wall leaks.	• R_o = Outside pipe radius • T= Evaluation wall thickness Code Case N-513-4 was approved in 2014

Code Case	Title	Applicability	Comment
N-523	Mechanical Clamping Devices for Class 2 and 3 Piping.	Permits the use of mechanical clamping devices to control leakage through the pressure boundary and maintain an acceptable level of structural integrity of piping. The clamping device limits leakage by preventing separation of piping components. This Code Case was developed to address criteria for temporary acceptance of flaws (including through-wall leaking) of moderate energy Class 3 piping where a Section XI Code repair may be impractical for a flaw detected during plant operation (i.e., a plant shutdown would be required to perform the Code repair)	
N-561	Alternative Requirements for Wall Thickness Restoration of Class 2 and High Energy Class 3 Carbon Steel Piping	The requirements of ASME Code Case N-561 allows carbon steel piping experiencing internal thinning or pitting to be restored by means of a weld-deposited carbon or low-alloy steel reinforcement (i.e., weld overlay) on the outside surface of the piping. Approved in 1996. Note that according to a NRC staff evaluation performed in 2000, it was concluded that neither the Section XI nor N-561 has sufficient rules for determining the rate or extent of the degradation of the repair or the surrounding based metal. In addition, re-inspection requirements are not specifically defined to verify the structural integrity of the component since the root cause may not be mitigated. Therefore, Code Case N-561 does not provide an acceptable level of quality and safety.	
N-561-1	Alternative Requirements for Wall Thickness Restoration of Class 2 and High Energy Class 3 Carbon Steel Piping	Provides an alternative to performing an internal weld repair or replacement of carbon steel piping components experiencing internal wall thinning or pitting in Class 2 or high energy Class 3 piping system. According to the code case, the weld metal overlay (reinforcement) is applied to the outside diameter of the piping component. the Code Case has criteria for determining the rate or extent of degradation of the repair or the surrounding base metal. Re-inspection requirements are not provided to verify structural integrity since the root cause may not be mitigated. This Code Case is invoked by licensees via Relief Requests with provisions for the operating period of the repaired component; e.g., until next scheduled refueling outage.	Approved in 1998.
N-562	Alternative Requirements for Wall Thickness Restoration of Class 3 Moderate Energy Carbon Steel Piping	Addresses an alternative for internal wall thinning of Class 3 piping systems which have experienced degradation mechanisms such as MIC that would provide an acceptable repair configuration. The primary purpose for implementing weld overlay repair is to allow for adequate time for additional examination of adjacent piping so that pipe replacement can be planned to reduce impact on system availability including Maintenance Rule applicability, availability of replacement materials and cost. This alternative repair technique involves the application of additional weld metal on the exterior of the piping system which restores the wall thickness requirement.	Approved in 1996.

Code Case	Title	Applicability	Comment
N-562-1	Alternative Requirements for Wall Thickness Restoration of Class 3 Moderate Energy Carbon Steel Piping	Repairs are limited to internal thinning or pitting caused by general localized corrosion, such as microbiological corrosion; cavitation induced pitting; erosion/corrosion and/or localized pitting corrosion; but excluded are conditions involving corrosion-assisted cracking or any other form of cracking. The repair is considered to have a maximum service life of one fuel cycle. In some applications the service life has been set to "no more than two operating cycles at which time a Section XI Code repair replaces the weld overlay."	Approved in 1996.
N-597	Requirements for Analytical Evaluation of Pipe Wall Thinning	Applicable to all Code Class piping. The Code Case was developed for the analytical evaluation of carbon and low-alloy steel piping which has experienced wall thinning due to corrosion. Limited to straight sections of piping, this Code Case provides a two-step structural integrity evaluation process; a thickness based and a stress-based evaluation.	Developed as an alternative to Code Case N-480, it was approved in 1998.
N-597-1		Differs from N-597 in that more detailed guidance is provided regarding the analytical evaluation of pipe wall thinning. Under Condition 1 to the Code Case, acceptance of the Code Case is conditioned on the use of this report. Since this is a guideline report developed by EPRI and EPRI reports by nature contain suggestions rather than requirements, the term “should” rather than “shall” is used throughout the report. Thus, implementation of the Code Case requires that the terms "should" and "shall" have the same expectation of being completed. Condition 2 requires that prior to reaching the calculated allowable minimum wall thickness, the component must be repaired or replaced in accordance with the construction code of record and owners requirements or a later approved edition of ASME Section III. Neither the ASME Code nor Code Case addresses wall thinning rates or inspection frequency..	Approved in 2001.
N-597-2	Requirements for Analytical Evaluation of Pipe Wall Thinning	Differs from 597-1 in that additional conditions for use apply.	Approved in 2003.
N-661	Alternative Requirements for Wall Thickness Restoration of Classes 2 and 3 Carbon Steel Piping for Raw Water Service	The Code Case is as an alternative under 10 CFR 50.55a(a)(3)(i) for Class 2 and 3 raw water piping system repairs resulting from degradation mechanisms such as erosion, corrosion, cavitation, or pitting as an alternative to the requirements of the ASME Section XI. These types of defects are typically identified by small leaks in the piping system or by pre-emptive non-code required examinations performed to monitor the degradation mechanisms. The alternative repair technique described in Code Case-661 involves the application of additional weld metal on the exterior of the piping system that restores the wall thickness requirement. The weld overlay will be utilized whenever the engineering evaluation determines that such a repair is suitable for the particular defect or degradation being resolved. Adjacent areas must be examined to verify that the repair will encompass the entire flawed area	If the root cause of the degradation has not been determined, a suitable re-inspection frequency cannot be established. Weld overlay repair of an area can only be performed once to ensure that ineffective repairs are not being repeatedly implemented in the same location. Performing through-wall weld repairs on surfaces that are wet or exposed to water would produce welds that include weld defects such as porosity, lack of fusion, and cracks. It is highly unlikely that a weld can be made on an open root joint with water present on the backside of the weld without having several weld defects. These

Code Case	Title	Applicability	Comment
		and that no other unacceptable degraded locations exist within a representative area dependent on the degradation mechanism present. The repair will be considered to have a maximum service life of two fuel cycles unless the re-examinations conducted during each of the two fuel cycles establish the expected life of the repair.	types of weld defects can, and many times do, lead to premature failure of a weld joint.
N-661-2		Approved in March 2007	For repairs performed on a wet surface, the overlay is only acceptable until the next refueling outage; and if the cause of the degradation has not been determined, the repair is only acceptable until the next refueling outage. Piping with wall thickness less than the diameter of the electrode shall be depressurized before welding.
N-666	Weld Overlay of Class 1, 2, and 3 Socket Welded Connection	Paragraph 1(e) states that a socket weld may not be overlaid more than one time. If the installed weld overlay is degraded in the future, the licensee needs to replace the socket weld and the weld overlay in its entirety. Paragraph 2(a) requires that the owner verify that the socket weld failure is a result of vibration fatigue. This determination shall include review of the design, operating history, including changes in the piping system, and visual inspection of the failed socket weld.	
N-786-2	Alternative Requirements for Sleeve Reinforcement of Class 2 and 3 Moderate-Energy Carbon Steel Piping	The material beneath the surface to which the reinforcing sleeve is to be applied shall be ultrasonically measured to establish the existing wall thickness and the extent and configuration of degradation to be reinforced. The adjacent area shall be examined to verify that the repair will encompass the entire unacceptable area, and that the adjacent base material is of sufficient thickness to accommodate the attachment welds at the edges of the sleeve. The cause and rate of degradation shall be determined. The extent and rate of degradation in the piping shall be evaluated to ensure that there will be no other unacceptable locations within the surrounding area that could affect the integrity of the reinforced areas for the life of the repair. Surrounding areas showing signs of degradation shall be identified and included in the Owner's plan for thickness-monitoring inspections of full structural reinforcing sleeves.	This Code Case has not been incorporated into NRC Regulatory Guide 1.147, "ln-Service Inspection Code Case Acceptability, ASME Code Section Xl Division 1." As such, ASME Code Case N-786-2 is not available for application at nuclear power plants without specific NRC approval.
N-789	Alternative Requirements for Pad Reinforcement of Class 2 and 3 Moderate-Energy Carbon Steel Piping for Raw Water Service	Temporary repair of degradation in Class 2 and 3 moderate energy raw water piping systems resulting from mechanisms such as erosion, corrosion, cavitation, or pitting, but excluding conditions involving flow accelerated corrosion (FAC), corrosion assisted cracking, or any other form of cracking. These types of defects are typically identified by small leaks in the piping system or by pre-emptive, non-code required examinations performed to monitor the degradation mechanisms.	

Code Case	Title	Applicability	Comment
N-789-2		Approved in June 2015	The alternative repair described in ASME Code Case N-789-2 involves the application of a metal reinforcing pad welded to the surface of the piping system that either restores pressure integrity or reinforces the weakened area and retain the system pressure. The reinforcing pad may be used for leak prevention only (pressure pad), or for leak prevention plus structural reinforcement of thinned areas including areas that do, or are expected to, penetrate the piping wall (structural pad). Pressure pads are designed to retain pressure and may be used only where the piping is predicted to retain full structural integrity until the next refueling outage assuming a corrosion rate of either 2 times the actual measured corrosion rate in that location, or four (4) times the estimated maximum corrosion rate for the same degradation mechanism in that system or similar system at the same plant site. Structural pads are designed for pressure plus structural reinforcement and may be used where the piping is predicted not to retain full structural integrity until the next refueling outage. In this context, "full structural integrity" means the piping maintains full capability to withstand structural (mechanical) loading for which it is designed without need for additional support or reinforcement. The appropriate repair technique will be determined based on the characterization of the degradation.

3.3.1 U.S. NRC Inspection Manual Chapter 0326

Appendix C of the U.S. NRC Inspection Manual Chapter 0326 [55] delineates the NRC position regarding operability determination (OD) of flawed piping:

- ASME Class 1 Components
  "When flaws in ASME Class 1 components do not meet ASME Code or construction code acceptance standards, the requirements of an NRC accepted ASME code case as listed in Section C.1 and C.2 of Regulatory Guide (RG) 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division, or an NRC approved alternative, then an immediate OD would not be expected to conclude a reasonable expectation of operability exists and the components should be declared inoperable. The NRC position is that satisfaction of ASME Code acceptance standards is the minimum necessary for operability of Class 1 pressure boundary components because of the importance of the safety function being performed."

- ASME Class 2 and 3 Components
  Immediate Operability Determinations: "When a defect is identified in ASME Class 2 or Class 3 components, an OD is required. In evaluating the defect during an immediate OD, the licensee must use the methodologies included in the ASME Code, construction code acceptance standard, an NRC-accepted ASME code case as listed in RG 1.147 or an NRC approved alternative. Detailed non-destructive examination (NDE) data may be necessary to determine if a component is operable during an immediate OD. If there is insufficient time to perform the required NDE testing to support the immediate OD, there must be significant operating experience with the identified degradation mechanism in the affected system in order to conclude the component is operable. If detailed NDE is necessary, the examination cannot be completed within the time frame normally expected for an immediate OD and there is no evidence (pertinent operating experience) that supports a determination of operable, the component should be declared inoperable and the appropriate LCO declared not met."

- Prompt Operability Determinations: "During the prompt OD, and in order to conclude the component is operable, the component must meet the structural integrity criteria contained in the ASME Code, construction code acceptance standards, an NRC-accepted ASME code case as listed in RG 1.147 or an NRC approved alternative. NRC issued Generic Letter (GL) 90-05, "Guidance for Performing Temporary Non-Code Repair of ASME Code Class 1, 2, and 3 Piping," to provide analysis tools, acceptance standards and allow non-code repairs of code Class 3 piping when a code repair was determined to be impractical. The scope of GL 90-05 is limited to Class 3 systems, but it does address moderate and high-energy systems. GL 90-05 preceded the ASME Code cases, which address the structural integrity of components containing flaws. However, the definition of moderate energy systems is consistent with these code cases, which came later. GL 90-05 permits licensees to consider either the "through-wall flaw" or the "wall thinning" flaw evaluation approach when assessing the structural integrity of Class 3 piping with identified flaws where no leakage is present. If the flaw is found acceptable by the "wall thinning" approach, or by the "through-wall flaw" approach, and no leakage is present, immediate repair of the flaw is not required and the component can be declared degraded but operable. However, the licensee should promptly submit a relief request and comply with the guidelines provided for flaw repair and monitoring. The relief request is to justify the impracticality of performing the required "code repair" at the time. Compensatory actions may be implemented by the licensee without NRC staff review and approval, provided the compensatory action does not involve a non-code repair to the piping system or supports and the compensatory action can be implemented in accordance with 10 CFR50.59."

"If it is identified that a flaw does not meet the criteria in ASME Code, construction code acceptance standards, an NRC-accepted ASME code case as listed in RG 1.147, or an NRC approved alternative, the component should be declared inoperable and the applicable TS action statement is to be addressed at that time. Alternatively, a relief request/alternative can be submitted and at a minimum, verbally approved by the NRC before the completion time expires."

3.3.2 U.S. NRC Risk Assessment of Operational Events

The Risk Assessment of Operational Events Handbook (also referred to as the "RASP Handbook") documents methods and guidance that NRC staff uses to achieve more consistent results when performing risk assessments of operational events and licensee performance issues. The handbook consists of two volumes; Volume 1[20] addresses internal events [63] and Volume 2[21] addresses external events [64]. Guidance is provided to ensure that the NRC-developed standardized plant analysis risk (SPAR) models used in the risk analysis of operational events represent plant design and operation to a sufficient level, and include the current operating experience data. According to the NRC, the handbook represents best practices based on feedback and experience from the analyses of over 600 precursors of events dating back to 1969 in the Accident Sequence Precursor (ASP) Program and numerous Significance Determination Process (SDP) analyses (since 2000).[22]

With respect to the evaluation of degraded passive components the Volume 1 of the handbook provides only relatively vague (as in open to interpretation and the judgment of an analyst) guidance on how to provide realistic risk characterizations, however. In analyzing degraded conditions the RASP Handbook states that "degraded failures are generally modeled by one of the following applications: 1) Adjusting the failure probability to a higher value, based on appropriate engineering analysis, to reflect increased likelihood of failure (e.g., due to aging, growth of a crack), 2) setting the basic event to its non-recovery probability (based on a recovery analysis) when it is not feasible to conduct an engineering analysis to determine the impact of the degradation on the failure probability, 3) adjusting the PRA success criteria, or 4) in some cases, refining the SPAR model to remove conservatism and thereby reducing the importance of the degradation."

In Volume 2 of the RASP Handbook, detailed guidance is given for evaluations of potential internal flooding scenarios that are attributed to degraded or failed piping. Tabulations of recommended pipe failure rates and rupture frequencies are based on published data:

- EGG-SSRE—9639, "Component External Leakage and Rupture Frequency Estimates" (1991)[23]. Prepared by Idaho National Laboratory, the report includes passive component population and exposure terms. The event population data were obtained from searches of licensee event reports. The component external leakage and rupture frequencies were calculated using Bayesian updates of a non-informative prior. "External rupture" is defined as a leakage greater than 50 gpm or a complete severance of a pipe. "External leakages' range from 50 gpm to a "lower limit" corresponding to the licensee reporting threshold value.

[20] Volume 1 addresses the following PSA technical elements: exposure time modeling, failure modeling, mission time modeling, common-cause failure modeling, recovery modeling, multi-unit considerations, initiating event analysis, human reliability analysis, loss of offsite power initiating events and support systems initiating events.

[21] Volume 2 addresses the following PSA technical elements: internal fires, internal flooding, seismic events, "other" external events and frequencies of seismically-induced loss-of-offsite-power events.

[22] An international perspective on precursor analysis is found in IAEA-TECDOC-1417: Precursor Analyses - The Use of Deterministic and PSA Based Methods in the Event Investigation Process at Nuclear Power Plants, 2004. https://www-pub.iaea.org/MTCD/Publications/PDF/te_1417_web.pdf

[23] https://www.osti.gov/scitech/biblio/5461408

- EPRI TR-102266, "Pipe Failure Study Update" (1993; EPRI proprietary). This report documents pipe failure rates and rupture frequencies for BWR and PWR piping systems. The principal source of pipe failure data were licensee event report abstracts, Nuclear Power Experience[24], the Nuclear Plant Reliability Data Systems (NPRDS)[25] and "other sources." The pipe failure rate calculations are based on analysis-of-variance (ANOVA) statistical models that account for different pipe failure attributes (e.g. material, size, plant system, degradation mechanism) and pipe failure mode definitions. This methodology is described in detail in the public domain report EPRI TR-100380[26].

3.4 Oil & Gas and Petrochemical Industry Guidelines

In 1990, a joint-industry project was organized by the Materials Properties Council (MPC)[27] to develop fitness-for-service guidelines for the refining industry. Most of the major multi-national oil companies were part of the initial sponsor group, including Exxon, Shell, BP, Amoco, Mobil, Chevron, Texaco, Pennzoil and Arco. Some of these companies had developed in-house FFS procedures, but they recognized that an industry standard was needed. One of the driving forces behind this effort was that improvements in NDE technology led to the detection of more flaws. In-service inspections were often more rigorous and sensitive than the initial (or pre-service) inspections that had been performed at the time of fabrication, so many previously undetected fabrication flaws were found in refinery equipment during turnarounds. When significant flaws are detected in pressure equipment, it is the responsibility of the owner/operator to address these flaws, even if the equipment had been operating successfully for many years before the flaws were detected.

The MPC project facilitated the development of FFS guidelines and preparation of a detailed report that outlined the procedures. A total of four versions of this report were circulated to the MPC sponsor group over the next several years. The final draft of the MPC consultants' report was turned over to the API committee on FFS. The MPC evolved into what became known as API 579, which was ultimately published in year 2000 and in revised form in 2007 [22]. API 579 has received widespread acceptance, both within and outside of the refining industry.

In the meantime, the Post-Construction Committee of the ASME Boiler and Pressure Vessel Code Main Committee formed the Task Group on Flaw Evaluation, which was charged with developing an FFS standard for pressure equipment in non-refinery applications. This task group made very little progress, however, and was disbanded shortly after the publication of API 579. Rather than duplicating efforts, ASME decided it would be beneficial to join forces with API to develop a unified FFS standard. For its part, API welcomed the alliance with ASME because of the latter's positive reputation among regulatory bodies and the public at large.

A joint API/ASME committee was formed to revise API 579 to make it applicable to broader range of industries. This broader base was reflected in the committee membership. The document was also edited to conform to ISO guidelines for standards, and recent advances in technology were adopted. The joint standard was published in the summer of 2007; API-579 /

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[24] The no longer available Nuclear Power Experience (NPE) was published by Stoller Corporation of Boulder, CO. It was available by subscription and it primarily consisted of U.S. Licensee Event Reports and other public domain information. The publication of NPE ceased in the early 1990s.

[25] This is now the INPO Consolidated Events Database (ICES).

[26] https://www.epri.com/#/pages/product/TR-100380%20%20%20%20%20%20%20%20%20/

[27] The Materials Properties Council (MPC) was established in 1966 in response to the growing need for valid data on the engineering properties of metals. MPC receives support from industry, technical organizations, codes and standards developers, and government agencies.

For additional information, go to https://forengineers.org/materials-properties-council/

ASME FFS-1 [22]. This document, which is a Standard rather than a Recommended Practice, contains numerous improvements and explicitly addresses industries outside of refining and petrochemical.

The API/ASME standard uses the remaining strength factor (RSF) concept in a number of assessment procedures. The RSF is defined by the following ratio:

$$RSF = L_{DC} / L_{UC} \tag{3-1}$$

Where L_{DC} is the limit load or burst pressure of the damaged component (i.e., the component with a flaw) and L_{UC} is the corresponding load or pressure for the undamaged component. For example, consider a pressure vessel in which corrosion has caused wall thinning over a localized region of the shell. If an RSF of 0.85 is computed for this vessel, it means that the burst pressure has been reduced to 85% of the original value as a result of the corrosion. The RSF can be computed from Level 1 or Level 2 equations, or from a finite element analysis in a Level 3 assessment. The calculated RSF is compared with an allowable value, RSF_a. If $RSF < RSF_a$, then a pressure vessel or pipe can be re-rated using the following expression:

$$MAWP_r = MAWP \times (RSF/RSF_a) \tag{3-2}$$

Where $MAWP$ is the original maximum allowable working pressure and $MAWP_r$ is the re-rated value. For atmospheric storage tanks, a similar re-rating of maximum fill height (MFH) can be performed:

$$MFH_r = MFH \times (RSF/RSF_a) \tag{3-3}$$

The API/ASME standard recommends an RSF_a value of 0.9 for most situations. There are three Parts in the standard that address corrosion assessment:

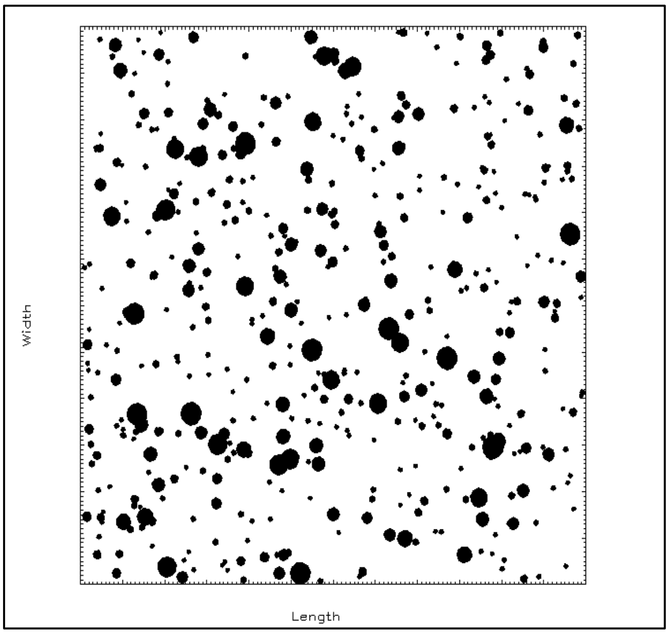
- Part 4 – Assessment of General Metal Loss.
- Part 5 – Assessment of Local Metal Loss.
- Part 6 – Assessment of Pitting Corrosion.

There is no quantitative demarcation between general and local metal loss in the API/ASME standard. The qualitative definition is that Part 4 pertains to metal loss over most or all of the components, while Part 5 is applicable to metal loss over a confined area. Either or both assessments can be applied to a given instance of wall thinning. Part 5 is usually less conservative than Part 4 because the former accounts for the finite extent of the metal loss, while the assessment in Part 4 assumes that the metal loss is over the entire component. The two assessments give similar answers when the metal loss extends over long distances. Both the Part 4 and Part 5 assessments use the RSF concept to evaluate wall thinning. Inspection data for local and general metal loss assessments typically consists of wall thickness readings in a grid pattern.

The pitting corrosion assessment entails computing an RSF that depends on the diameter, depth, and spacing of pits. In the Level 1 assessment, the RSF is estimated by visually comparing pitting charts (Figure 3-2) with the observed pitting. The Level 2 assessment requires measurement of pit dimensions and spacing and includes a series of calculations to estimate the RSF. There are two parts that pertain to brittle fracture and crack-like flaws:

- Part 3 – Assessment of Existing Equipment for Brittle Fracture.
- Part 9 – Assessment of Crack-Like Flaws.

Part 3 does not assess specific flaws and their effect on the risk of brittle fracture. Rather, this assessment procedure evaluates the material of construction relative to the temperatures at which it is subject to significant applied stress. The Part 3 assessment is based on the toughness rules and exemption curves in Section VIII of the ASME Boiler and Pressure Vessel Code.



The scale of the pitting chart is 6-by-6 inches. The surface damage from photographs and rubbings are compared to the standard pit charts (like the sample to the left), next select a chart that approximates the actual damage of the component. For additional details on the assessment procedure see https://gist.github.com/flare9x/d5af29ad7870fd49aabb8d457c58dd59

Figure 3-2: Sample "Pitting Chart" – API-579 Grade 4 Pitting

When cracks or other planar flaws are detected, Part 9 of the API/ASME fitness-for-service standard provides suitable assessment procedures. The failure assessment diagram (FAD) approach[28] is used for Level 2 crack evaluation. Engineers who apply this assessment procedure should have at least a basic understanding of fracture mechanics. A simple screening assessment is provided at Level 1, which can be applied without having a background in fracture mechanics. The FAD is a two-parameter graphical representation of the failure envelope of a cracked structure expressed in terms of the ratio of the applied stress intensity factor to the material fracture toughness (the toughness ratio, $K_r = K_{app}/K_{mat}$) and the ratio of the applied load to the plastic limit load of the structure (the load ratio, $L_r = P/PL$).

To use the FAD approach, assessment points with coordinates (L_r , K_r) are calculated based on the applicable loads, crack type and crack size(s), and material properties are compared with the failure envelope line. Assessment points that lie inside the envelope indicate non-failure, while assessment points outside the envelope indicate failure. For many fatigue crack growth analyses, the assessment points will initially be far inside the failure assessment line envelope and will gradually grow toward the envelope as the crack grows sub-critically. When the load ratio is low, the FAD predicts failure based on fracture instability; however, as the load ratio increases, the interaction of the presence of plasticity decreases the allowable stress intensity factor. If the assessment point is on, or inside, the FAD envelope, which indicates that there is remaining service life, then the pressure vessel is deemed safe, and therefore fit for service. A fatigue crack growth analysis must then be performed to determine how long the structure will remain fit for service.

Part 7 of the API/ASME standard, which is entitled "Assessment of Hydrogen Blisters and Hydrogen Damage Associated with HIC and SOHIC), has been extensively revised since

[28] See Appendix A, "Glossary of Technical Terms"

the 2000 edition of API 579 was released. A new assessment procedure for hydrogen-induced cracking (HIC) damage has been added. This methodology relies on the remaining strength factor (RSF) concept to account for loss of load-carrying capacity in HIC-damaged steel. Although the title of Part 7 mentions stress-oriented hydrogen-induced cracking (SOHIC), it does not include a Level 1 or 2 assessment procedure for SOHIC. When SOHIC is present, a Level 3 assessment using Part 9 is recommended.

Part 10, entitled "Assessment of Components Operating in the Creep Range," is a new addition to the FFS standard. Creep damage is assessed using the Omega method, which was developed in a joint-industry project in the 1990s.[29] According to the developers, the Omega model has a number of advantages over the traditional Larson-Miller approach[9]. It can be used to estimate creep rate, creep damage, and time to rupture in components operating at elevated temperatures. In a Level 3 assessment, creep deformation can be modeled with finite element analysis. The API/ASME standard lists material constants for the Omega model for a wide range of alloys.

The API/ASME fitness-for-service standard contains a number of other assessment methods, which are listed below.

- Part 8 – Assessment of Weld Misalignment and Shell Distortions.
- Part 11 – Assessment of Fire Damage.
- Part 12 – Assessment of Dents, Gouges and Dent-Gouge Combinations.
- Part 13 – Assessment of Laminations.

British Standard (BS) 7910 provides a 'Guide on methods for assessing the acceptability of flaws in metallic structures' [30]. It utilizes a failure assessment diagram (FAD) derived from fracture mechanics. The assessment process positions the flaw within acceptable or unacceptable regions of the FAD.

The flaw lying within the acceptable region of the FAD does not by itself infer an easily quantifiable margin of safety or probability of failure. Conservative input data to the fracture mechanics calculations are necessary to place reliance on the result. If key data are unavailable, (e.g. fracture toughness properties of the weld and parent material), then conservative assumptions should be made. Sensitivity studies are recommended so that the effect of each assumption can be tested.

3.4.1 API Recommended Practice 579

API's Recommended Practice 579, Fitness-For-Service (FFS) [22] is a compendium of consensus methods for assessment of the structural integrity of equipment containing identified flaws or damage. The FFS assessment procedures in this Standard are organized by flaw type and/or damage mechanism; Table 3-3. In some cases, it may be necessary to use the assessment procedures from multiple Parts if the primary type of damage is not evident. For example, the metal loss in a component may be associated with general corrosion, local corrosion and pitting. If multiple damage mechanisms are present, a damage class, e.g., corrosion/erosion, can be identified to assist in the evaluation. API 579 is a 1320 pages long document and it includes detailed implementation guidelines, for example:

- Part 2 – Fitness for Service Engineering Assessment Procedure. Annex 2A through Annex 2F includes supporting information. As an example, Annex 2D contains

[29] Mathematically, "Omega" is the rate at which the strain rate accelerates as a result of the creep strain. If the creep behavior of a material conforms to the Omega model, a plot of natural log of strain rate vs. strain in the tertiary creep regime would consist of a straight line with a slope of Omega. For additional details see Technical Paper ETAM2014-1019, Proc. ASME Symposium on Elevated Temperature Application of Materials for Fossil, Nuclear, and Petrochemical Industries, 2014.

analytical methods for stress analysis. These methods are based on ASME Boiler & Pressure Vessel Code (B&PVC) Section VIII (Rules for Construction of Pressure Vessels).

- Part 3 – Assessment of Existing Equipment for Brittle Fracture. The assessment procedures for prevention of brittle fracture for pressure vessel and piping components in Part 3 are based on the design requirements contained in the ASME B&PV Code, Section VIII.
- Part 4 – Assessment of General Metal Loss. The assessment procedures in this Part can be used to evaluate general metal loss (uniform or local) that exceeds or is predicted to exceed the corrosion allowance before the next scheduled inspection. The general metal loss may occur on the inside or outside surface of the component. Assessment procedures based on point thickness readings and thickness profiles are provided.

Additional implementation guidelines, including practical examples are included in a companion text, "Piping and Pipelines Assessment Guide" [23].

Table 3-3: Summary of Flaw & Damage Assessment Procedures in API 579-1

Flaw or Damage Mechanism	**Section in API 579-1**	**Overview**
Brittle Fracture	Part 3	Assessment procedures for evaluating the resistance to brittle fracture of existing carbon and low alloy steel pressure vessels, piping, and storage tanks. Criteria are provided to evaluate normal operating, start-up, upset, and shut-down conditions.
General Metal Loss	Part 4	Assessment procedures to evaluate general corrosion. Thickness data used for the assessment can be either point thickness readings or detailed thickness profiles. A methodology is provided to utilize the assessment procedures of Part 5 when the thickness data indicates that the metal loss can be treated as localized.
Local Metal Loss	Part 5	Assessment techniques to evaluate single and networks of Local Thin Areas and groove-like flaws in pressurized components. Detailed thickness profiles are required for the assessment. The assessment procedures can also be utilized evaluate individual pits or blisters as provided for in Part 6 and Part 7, respectively.
Pitting Corrosion	Part 6	Assessment procedures are provided to evaluate widely scattered pitting, localized pitting, pitting which occurs within a region of local metal loss, and a region of localized metal loss located within a region of widely scattered pitting. The assessment procedures can also be utilized to evaluate a network of closely spaced blisters as provided for in Part 7.
Blisters and HIC/SOHIC Damage	Part 7	Assessment procedures are provided to evaluate isolated and networks of blisters and hydrogen-induced cracking (HIC) / stress-oriented hydrogen-induced cracking (SOHIC) Damage. The assessment guidelines include provisions for blisters and HIC/SOHIC damage located at weld joints and structural discontinuities such as shell transitions, stiffening rings, and nozzles.
Weld Misalignment and Shell Distortions	Part 8	Assessment procedures are provided to evaluate stresses resulting from geometric discontinuities in shell type structures including weld misalignment and shell distortions (e.g. out-of-roundness and bulges).
Crack-Like Flaws	Part 9	Assessment procedures are provided to evaluate crack-like flaws. Solutions for stress intensity factors and reference stress (limit load) are included in Annex 9B and Annex 9C, respectively. Methods to evaluate residual stress as required by the assessment procedure are described in Annex 9D. Material properties required for the assessment are provided in Annex 9E.

Flaw or Damage Mechanism	Section in API 579-1	Overview
High Temperature Operation and Creep	Part 10	Assessment procedures are provided to determine the remaining life of a component operating in the creep regime. Material properties required for the assessment are provided in Annex 10B.
Fire Damage	Part 11	Assessment procedures are provided to evaluate equipment subject to fire damage. A methodology is provided to rank and screen components for evaluation based on the heat exposure experienced during the fire. The assessment procedures of the other Parts of API 579-1 are utilized to evaluate component damage.
Dent, Gouge, and Dent Gouge Combinations	Part 12	Assessment techniques are provided to evaluate dent, gouge, and dent gouge defect combinations.[30]
Laminations	Part 13	Assessment procedures for evaluation of laminations. The assessment guidelines include provisions for laminations located at weld joints and structural discontinuities; e.g. shell transitions, stiffening rings, and nozzles.
Fatigue	Part 14	Assessment procedures to evaluate pressurized components (including weldments) subject to cyclic loading.

3.4.2 API Recommended Practice 754

As a result of the U.S. Chemical Safety and Hazard Investigation Board's (CSB) investigation of the 2005 BP Texas City incident, the CSB issued several recommendations. One of those recommendations called for the development of an ANSI standard that creates "performance indicators for process safety in the refinery and petrochemical industries;" Figure 3-3[31]. A performance indicators program provides useful information for driving improvement and when acted upon, contributes to reducing risks of major hazards by identifying the underlying causes and taking action to prevent recurrence.

API Recommended Practice (RP) 754 [24] identifies "leading" and "lagging" process safety indicators for performance improvement. The indicators are divided into four tiers that represent a leading and lagging continuum. Tier 1 is the most "lagging" and Tier 4 is the most "leading." Tiers 1 and 2 are suitable for nationwide public reporting and Tiers 3 and 4 are intended for internal use at individual sites. This RP was developed for the refining and petrochemical industries, but may also be applicable to other industries with operating systems and processes where loss of containment has the potential to cause harm. Applicability is not limited to those facilities covered by the OSHA Process Safety Management Standard, 29 CFR 1910.119 or similar national and international regulations.

When systematically applied, the performance indicators are intended to be used to identify process safety improvements. Implicitly these performance indicators, when applied within a quantitative risk assessment (QRA) framework could be used to facilitate risk-informed operability determinations.

[30] For definitions, see https://primis.phmsa.dot.gov/comm/FactSheets/FSPipeDefects.htm

[31] For additional details see Section 13.0 (Recommendations) of the CSB Final Investigation Report https://www.csb.gov/bp-america-refinery-explosion/

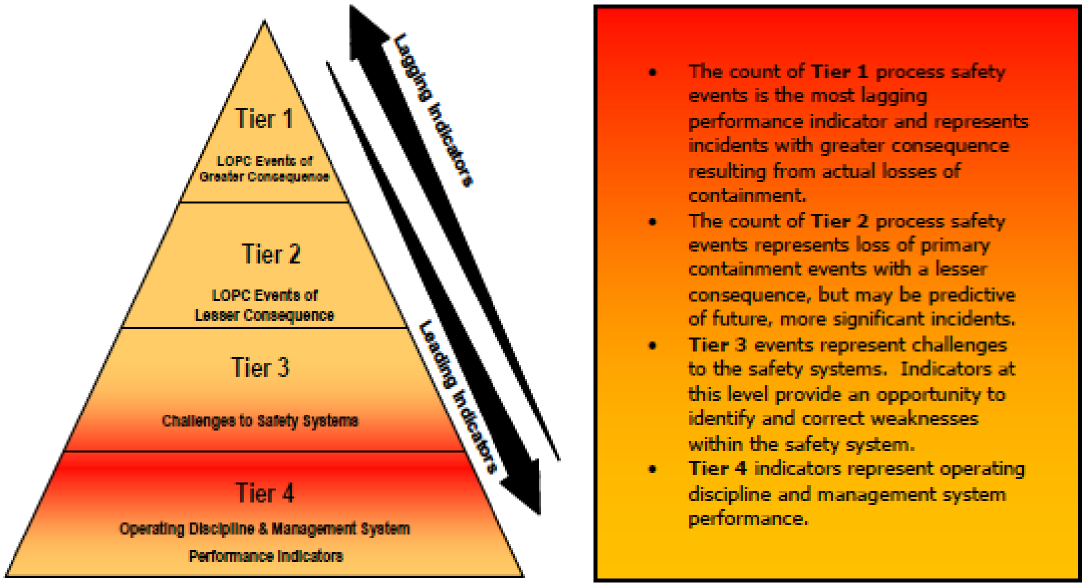


Figure 3-3: API RP-754 Performance Indicators[32]

3.4.3 European FFS Network (FITNET)

Building on the experience with API 579 [22] and BS 7910 [30], and other recognized FFS procedures, the European Fitness-for-Service Network (FITNET) was created in 2002 in order to develop an alternative FFS procedure; FITNET-FFS [65]. The objectives of FITNET-FFS are stated as:

- Review existing FFS procedures and develop an updated, unified, and verified European FFS procedure covering fracture, fatigue, creep, and corrosion in metallic load bearing components with or without welds.
- Enable European industry users to influence future editions of FFS procedure and to foster the dissemination of results of ongoing R&D, and training activities.

The FITNET-FFS flaw assessment methodologies are in the form of a step-by-step procedure set out for assessing a welded or non-welded metallic component containing a known or postulated flaw under static, dynamic, creep loading conditions or a component subject to corrosion damage. The FITNET-FFS procedure is documented in three volumes: Volume I “FITNET FFS Procedure [66],” Volume II: “Case Studies and Tutorials” [67] and Volume III “Annex” **Error! Reference source not found.**. The FFS Procedure volume provides e ngineering methodologies for assessing flaws to reach a decision about the component.

Completed in 2006, a unified flaw assessment procedure is the main output of the FITNET project. It provides a unique and comprehensive fitness-for-service document. According to the final report, “all major parts and approaches have been validated to produce a technically sound “ready-to-use” procedure for the European manufacturing and plant operating industries.” Furthermore, extensive course notes have been developed for training of early-career engineers. Training materials have been developed and used during the seminars and distributed as CD to participants of the training seminars held during the project life-time.”

[32] Reproduced from the API RP-754 Fact Sheet.

3.4.4 DNV GL RP-101[33]

DNV-RP-101 [28] describes two alternative approaches to the assessment of corrosion, and the document is divided into two parts. The first approach includes calibrated safety factors taking into account the natural spread in material properties and wall thickness and internal pressure variations. Uncertainties associated with the sizing of the defect and the specification of the material properties are specifically considered in determination of the allowable operating pressure. Probabilistically calibrated equations (with partial safety factors) for the determination of the allowable operating pressure of a corroded pipeline are given. The second approach is based on the Allowable Stress Design (ASD) format. The failure pressure (capacity) of the corrosion defect is calculated, and this failure pressure is multiplied by a single usage factor based on the original design factor. Consideration of the uncertainties associated with the sizing of the corrosion defect is left to the judgement of the user.

This recommended practice lists ultimate limit states (ULSs) that must be fulfilled in order to comply with the engineering demands for strength and stability under design loads. Table 3-4 lists target annual failure probability by safety class and ULS. Subsea oil and gas pipelines, where no frequent human activity is anticipated, will normally be classified as Safety Class Normal. Safety Class High is used for risers and the parts of the pipeline close to platforms, or in areas with frequent human activity. Safety Class Low can be considered for e.g. water injection pipelines.

Table 3-4: Target Annual Failure Probability for Ultimate Limit States

Safety Class	Target Annual Failure Probability
High	$< 10^{-5}$
Normal	$< 10^{-4}$
Low	$< 10^{-3}$

3.4.5 EPA 'SPCC Rule' & Risk Management Program

The U.S. Environmental Protection Agency's (EPA's) oil spill prevention program includes the Spill Prevention, Control, and Countermeasure (SPCC) and the Facility Response Plan (FRP) rules [32][33][34]. The SPCC rule helps facilities prevent a discharge of oil into navigable waters or adjoining shorelines. The FRP rule requires certain facilities to submit a response plan and prepare to respond to a worst case oil discharge or threat of a discharge. The scope of the SPCC Guidance Chapter 7 ("Inspections, Evaluation, and Testing") provides guidance for FFS assessment and references API RP 579-1, and with emphasis on brittle fracture assessments.

Section 112(r) of the Clean Air Act Amendments requires EPA to publish regulations and guidance for chemical accident prevention at facilities that use certain hazardous substances. These regulations and guidance are contained in the Risk Management Plan (RMP) rule. The information required from facilities under RMP addresses response to chemical emergencies. The RMP rule was built upon existing industry codes and standards; e.g. API RP 579-1. It requires companies that use certain flammable and toxic substance to develop a Risk Management Program, which entails the performance of hazard and operability analysis and QRA.

[33] DNV GL is a global quality assurance and risk management company providing classification, technical assurance, software and independent expert advisory services to the maritime, oil & gas, power and renewables industries.

3.5 CCPS Project 268

The Center for Chemical process Safety (CCPS) of the American Institute of Chemical Engineers (AIChE) is a not-for-profit, corporate membership organization that identifies and addresses process safety needs within the chemical and petroleum industries. CCPS brings together manufacturers, government agencies, consultants, academia and insurers to lead the way in improving industrial process safety. CCPS member companies, working in project subcommittees, define and develop useful, time-tested guidelines that have practical application within industry.

CCPS Project 268 "Dealing with Aging Process Equipment and Infrastructure" (2018) [43] is a text book on aging of process equipment and infrastructure. Intended for mid- to small-size facility operators, the text book is designed to help those responsible for maintaining and evaluating equipment (maintenance, equipment and asset integrity) to understand and assess the added risks of accrued damage and deterioration on the performance and safety of aging equipment. Aspects of management systems and engineering are included in the text as well as the roles and responsibilities, documentation throughout the life of the equipment. It is a non-technical text book written for the lay person. Key elements of the text book are:

- Section 2. Aging Equipment Failures, Causes and Consequences. Equipment aging is defined as "its condition, the service it is and how that is changing over time."
- Section 4: Risk-Based Decisions. To assess the likelihood of failure, reference is made to "Layer of Protection Analysis (LOPA) which is "screening approach" to quantitative risk assessment.[34]
- Section 5. Managing Process Equipment and Infrastructure Lifecycle
  - 5.3.1: Manage by Operational Integrity.
  - 5.4: Predicting Asset Service Life.
  - 5.4.1: Mean Life and Age.
  - 5.4.2: Assessing End-of-Life Failure Probability. This section is a very basic introduction to the fundamentals of the statistical theory of reliability.

3.6 NASA RAC Determination

The Office of Safety and Mission Assurance (OSMA) of the National Aeronautical and Space Administration (NAS) has overall responsibility for the Agency's Pressure Vessels and Pressurized Systems (PVS) Program and has delegated day-to-day oversight of policy and procedures to the OSMA Technical Discipline Manager for Pressure Systems. In accordance with NASA-STD-8719.17A [44][45], Process Safety Managers (PSMs) are required to determine whether both NASA owned or operated and contractor or tenant ground-based PVS pose a risk to NASA personnel, facilities, or equipment.

The purpose of NASA-STD-8719.A is to ensure the structural integrity of PVS through implementation of the minimum requirements for ground-based PVS in accordance with NASA Policy Directive (NPD) 8710.5 [68] and national consensus codes and standards (NCS). "Remaining Life Assessment Requirements" are addressed in Section 4.8.2 of NASA-STD-8719.A. "System Safety and Risk Assessment Requirements" are addressed in Section 4.9 of the same standard.

The original service life or remaining safe life of each PVS shall be documented at the time of certification or recertification through a detailed integrity assessment based on nondestructive examination (NDE) and inspection results, relevant damage mechanisms, cyclic service history, rates of degradation, and other relevant factors. According to the standard, the rate of

[34] A highly simplified, stylized approach to the quantification of initiating event frequencies.

service-related or environmentally-induced wall thinning of PVS shall be documented by means of periodic thickness inspection.

When PVS service life is limited by fatigue considerations, NCS-based fatigue or fracture life assessment shall form the basis for specified cyclic life. The fatigue life assessment methodology of ASME Section VIII, Div. 2, may be used to estimate fatigue life of Div. 1 vessels or non-Code vessels provided the allowable stress values from Div. 1 are substituted for maximum allowable stress and appropriate consideration is given to the additional requirements imposed on Div. 2 material, fabrication, and inspection. Greater factor of safety (FS) on cyclic life must be incorporated as uncertainty and unknowns increase. Similarly, the fracture assessment methodology of Div. 3 may be used to assess non-Div. 3 vessels provided additional consideration is given to uncertainties in stress intensity factors and fracture toughness for material that was not fully documented at the time of fabrication in accordance with Div. 3 requirements, which is typically the case for old PVS.

The standard requires the derivation of a numerical expression of comparative risk of a condition determined by an evaluation of both the potential severity of a consequence and the likelihood of that consequence occurring. The risk significance of a degraded or failed condition is determined on the basis of the Risk Assessment Code (RAC). The RAC is a numerical expression of comparative risk determined by an evaluation of both the potential severity of a condition and the likelihood of its occurrence causing an expected consequence. RACs are assigned a number from 1 to 7 in a risk matrix (Table 3-5). The assessed risk of in-service PVS shall be no greater than RAC 3 after mitigation unless that risk is specifically approved and accepted in accordance with paragraph 4.9.4 of the standard. The severity classification is displayed in Table 3-6. The equipment failure probability estimates of Table 3-7 shall be applied only to “certified PVS”. Without further information on a specific PVS complying with the certification requirements of this standard, the default values of Table 3-8 shall be used as the equipment failure probability in the RAC determination.

Table 3-5: RAC Risk Determination

Severity	**Likelihood**				
	A Frequent	**B Probable**	**C Occasional**	**D Remote**	**E Improbable**
I / Catastrophic	1	1	2	3	4
II / Critical	1	2	3	4	5
III / Moderate	2	3	4	5	6
IV / Negligible	3	4	5	6	7

Table 3-6: Severity Determination Table

Class	**Class Description**	**Equipment Loss [$k]**	**Downtime**	**Data Integrity**	**Environmental Effect**
I Catastrophic	A condition that may cause death or permanently disabling injury, facility destruction on the ground, or loss of crew, major systems, or vehicle during the mission.	> $1,000	> 4 months	Data Not recovered	> 5 years or >$1M to correct
II Critical	A condition that may cause severe injury or occupational illness or major property damage to facilities, systems, equipment, or flight hardware.	$1,000 to $250	4 months to 2 weeks	Repeat program	1-5 years or $250K - $1M to correct

Class	Class Description	Equipment Loss [$k]	Downtime	Data Integrity	Environmental Effect
III Moderate	A condition that may cause minor injury or occupational illness or minor property damage to facilities, systems, equipment, or flight hardware.	$250 to $25	2 weeks to 1 day	Repeat test period	< 1 yr. or $25K -$250K to correct
IV Negligible	A condition that could cause the need for minor first aid treatment though would not adversely affect personal safety or health. A condition that subjects facilities, equipment, or flight hardware to more than normal wear and tear.	$25 to $1	< 1 day	Repeat test point	Minor or < $25K to correct

Table 3-7: Probability Determination Table

Severity Level	Description	Definition
A / "Frequent"	Likely to occur immediately	$> 10^{-1}$
B / "Probable"	Probably will occur in time	$10^{-1} \geq X > 10^{-2}$
C / "Occasional"	May occur in time	$10^{-2} \geq X > 10^{-3}$
D / "Remote"	Unlikely to occur	$10^{-3} \geq X > 10^{-6}$
E / "Improbable"	Improbable to occur	$10^{-6} \geq X$

Table 3-8: Selected Catastrophic PVS Failure Rates per Year for Certified PVS

Item	PVS Component Type	Equipment Failure Probability (Median Values)[35]				
		$> 10^{-1}$	10^{-1} to $< 10^{-2}$	10^{-2} to $< 10^{-3}$	10^{-3} to $< 10^{-6}$	$\leq 10^{-6}$
Steel Piping System		Catastrophic Failures / Piping System-Yr				
10	Small bore (NPS 4" and under) piping system (no double containment, sensors/alarms, etc.) – Small system (less than 75 ft. of pipe)				10^{-3}	
11	Small bore (NPS 4" and under) piping system (no double containment, sensors/alarms, etc.) – Large system (more than 75 ft. of pipe)				$> 10^{-3}$	
12	Large bore piping system (> NPS 4', non-intergranular stress corrosion cracking (IGSCC)) – Small system (less than 75 ft. of pipe)				10^{-4} (10^{-5} if stainless steel & enhanced ISI)	
13	Large bore piping system (> NPS 4', non-IGSCC) – Large system (more than 75 ft. of pipe) with failure modes that include thermal fatigue, fluid dynamic loads, or erosion/corrosion wall thinning			> 10-3	10^{-4} if stainless steel & enhanced ISI	

For PVS whose design life is limited by fatigue or brittle fracture failure mode, and whose life has been extended through the application of NDE, in order to consider the potential for NDE

35 This table is based on failure rate data documented in S. H. Bush, "Statistics of Pressure Vessel and Piping Failures," Pressure Vessel Technology, **110**:225-233 (1988). The 99% confidence data are also provided in Spencer H. Bush, "Pressure Vessel Reliability" (J. Pressure Vessel Technology, 97(1), 54-70 (Feb 01, 1975) doi:10.1115/1.3454253. The standard states that the "… applicability of these industrial statistical data to NASA pressure vessels has not been qualified."

to miss existing crack-like flaws, the probability of failure shall be increased by a minimum of one level from Table 3-8 (i.e., an original level E (10^{-6}) becomes a level D (10^{-3} to 10^{-6}).

Severity class assessment shall include consideration of the worst credible consequence due to residual risk for all failure modes. Where Table 3-9 requires that a specific failure assessment be performed, that assessment shall consider the particular facts and condition of the PVS in question and be based on either:

1. Analysis consistent with the principles of risk management found in NPR 8000.4, Agency Risk Management Procedural Requirements [69], or
2. Informed and conservative engineering judgment that is approved and documented by the Pressure Systems Manager (PSM)[36].

Failure probabilities of PVS not included in Table 3-8 shall be specified by the PSM based on one of the following: (1) qualitative or quantitative data relevant to the PVS in question, (2) analyses performed consistent with the principles of risk management found in NPR 8000.4, or (3) "informed and conservative engineering judgment" that is documented.

3.7 An Assessment of Synergies

Fitness-for-service (FFS) assessments involve discrete operability determinations consisting of engineering analyses and deterministic or probabilistic fracture mechanics analyses. The regulatory regimes for the non-nuclear industries mostly rely on consensus codes and standards for FFS. The offshore oil & gas industries in Norway and the United Kingdom are subject to risk regulation similar to the nuclear industry. Possibilities therefore exist to analyze the safety or risk significance of degraded SSCs using QRA-centric risk metrics in combination with structural reliability information. It is noteworthy that within the Norwegian regulatory regime it is the responsibility of respective owner/operator to define acceptance criteria for major accident risk.[37]

The Norwegian Petroleum Safety Authority (PSA) has implemented the "Trends in Risk Level in the Petroleum Activity" (RNNP) process, which analyses a number of underlying indicators which are significant for assessing the chances of major accident risk.[38] A formula, where these indicators are weighted in accordance with their contribution to the overall position, yields a composite indicator for major accident risk. Viewed over time (e.g. a couple to several years), this in turn provides a picture of the trend in the probability of major accident risk.

The annual Trends in risk level report (RNNP) from the Petroleum Safety Authority (PSA) uses risk indicators to measure the status of so-called "defined hazard and accident conditions" (DFUs). But what are these DFUs? And which of them comes with a potential for causing major accidents? An owner/operator responsible for pursuing oil and gas activities acceptably must identify the occurrences it needs to guard against; known as "defined hazard and accident conditions" (DFUs).[39]

[36] PSM is the person responsible for the implementation of NPD 8710.5 at a NASA facility.

[37] For more information see https://www.ntnu.edu/documents/624876/1277591044/chapt04-rac.pdf/3eb85fbd-eadf-4f55-a9b7-d39bcad42d8a and the text "Use of Risk Acceptance Criteria in Norwegian Offshore Industry: Dilemmas and Challenges" https://www.tandfonline.com/doi/abs/10.1080/14664530490505567

[38] http://www.ptil.no/about-rnnp/category911.html

[39] http://www.ptil.no/news/defined-hazard-and-accident-conditions-dfus-article9296-878.html

The annual RNNP report uses one or more risk indicators to measure the status of most DFUs. All data acquired through various channels are processed in a statistical model. This shows how the various contributors to risk are developing, both collectively and for the individual DFU. DFUs with a potential for causing major accidents in the petroleum industry include the following:

- Leaks of flammable gas or liquids: A distinction is drawn between ignited and non-ignited leaks. A non-ignited leak, for example, could allow gas to spread over large areas so that later ignition causes an explosion and a major accident.
- Fire/explosion in other areas: An example of such incidents is a fire in the living quarters with the potential to develop into a major accident.
- Leaks from subsea production facilities with pipelines and associated equipment: Installations on the seabed can be damaged by objects dropped from above. Fishing gear may also cause substantial harm. The major accident potential of damage to subsea facilities relates primarily to pollution from possible oil spills. Any nearby surface facilities could also be threatened.

In addition to the RNNP process, the Norwegian authority has created the "**CO**rrosion and **DAM**age" (CODAM)[40] operating experience database, which captures data on degraded and failed conditions in structures, risers, and pipelines. The database content (in Norwegian) is publically available.

Nuclear industry FFS assessments are also based discrete operability determinations. With the widespread use of PSA in combination with the progress with risk-informed piping reliability analysis and the advances in probabilistic fracture mechanics possibilities exist for development of a holistic risk-informed operability determination approach. None of the reviewed operability determination and FFS procedures makes explicit reference to OPEX data, however.

[40] http://www.psa.no/aging-and-life-extention/damage-and-incidents-involving-load-bearing-structures-and-pipeline-systems-article4306-1032.html. A summary of the pipe failure events in the CODAM database can be found at http://www.ptil.no/getfile.php/1345620/PDF/Roerledningsskader%20Oktober2017.pdf and it covers the period 1975 to 2017.

4. 'RIOD' PRACTICAL CONSIDERATIONS

This section documents two examples of a risk-informed operability determination (RIOD) to assess of the risk significance of degraded conditions. The first example is an independent assessment of a degraded condition involving high-cycle fatigue damage. In contrast, the second example summarizes an evaluation based on the U.S. NRC Significance Determination Process (SDP).

4.1 An Introduction to the Use of PSA in Operability Determination

During the operation of a nuclear power plant, conditions (or events) exist that alter the risk of operating the facility. The conditions that result in a change, where "change" can be either an increase or decrease in risk, fall under three general categories:

1. Plant activities dictate that certain components will be incapable of performing their desired functions at certain times during operation. Examples of these activities that incapacitate components include preventative maintenance and testing (both scheduled activities), corrective repairs to failed components, and Technical Specification actions such as either entering a Limiting Condition of Operation (LCO) to replace a component or performing specified functional tests (these activities could be either scheduled or unscheduled).
2. Improper plant design, maintenance or in-service inspection could result in an unintended reduction in plant or component reliability, potentially over long periods of time.
3. Initiating events that occur during operation that cause challenges to plant systems and operators. Examples of these events include losses of off-site power, miscellaneous plant transients, and loss-of-coolant pipe breaks.

While it is acknowledged that operational plant risk changes over time as components are taken out of service or plant upsets are caused by initiators, measuring the risk over time can be challenging from a analytical perspective. Ideally, the process of measuring the operational risk should account for the following analytical considerations:

- A proposed risk metric should allow the analyst to calculate the risk magnitude for a particular event or over a specified period of time.
- The risk results should be consistent such that they can be summed for an operational period of time (e.g., a single 18 month fuel cycle) to obtain a cumulative risk profile over the period of interest.
- The risk metric calculation process should be tractable and represent the actual risk while still using the current modeling techniques, tools, and state of knowledge.

Since the core damage frequency (CDF) is not an appropriate measure for tracking risk over a known time period, another measure must be considered. This measure, a conditional core damage probability (CCDP), turns out to satisfy all three of the desirable attributes discussed previously. In addition, the CCDP make use of the core damage frequency, thereby providing a bridge of understanding for those analysts familiar with using the core damage frequency measure.

For the CCDP measure, the element of time is incorporated into the calculation, allowing the analyst to estimate the risk magnitude for an event at a certain point in time (e.g., at the time of an initiating event) or for a condition existing over a length of time (e.g., an improperly installed valve that remains unnoticed). Since the CCDP is dimensionless (it is a probability) and factors in time if necessary, two different events can be compared quantitatively to one another. Thus, the first desirable risk measure attribute is satisfied. This consistency in the risk measure from one event to the next allows for the integration over the time period of interest to obtain an overall risk profile. Consequently, the second desirable risk measure attribute is satisfied. Finally, since the calculation of the CCDP builds upon the core damage frequency, additional risk models and analysis tools are not required; accordingly, the third desirable attribute is satisfied.

4.2 Risk Significance of a High-Cycle Fatigue Event

In November 2013, a control room at a multi-unit U.S. PWR site (NPP1-3) received indications of potential reactor coolant system (RCS) leakage in the containment. The location of the leakage was determined to be from the vicinity of an isolation valve and a controlled power reduction was initiated. Approximately three days after the initial indication of the primary pressure boundary leakage, a containment entry determined that the leak was from a circumferential crack in the Code Class 1 safe end-to-pipe butt weld (designated as 1-RC-201-105) located between the "1B2 High Pressure Injection" (HPI) nozzle and a valve designated as 1HP-152; Figure 4-1.

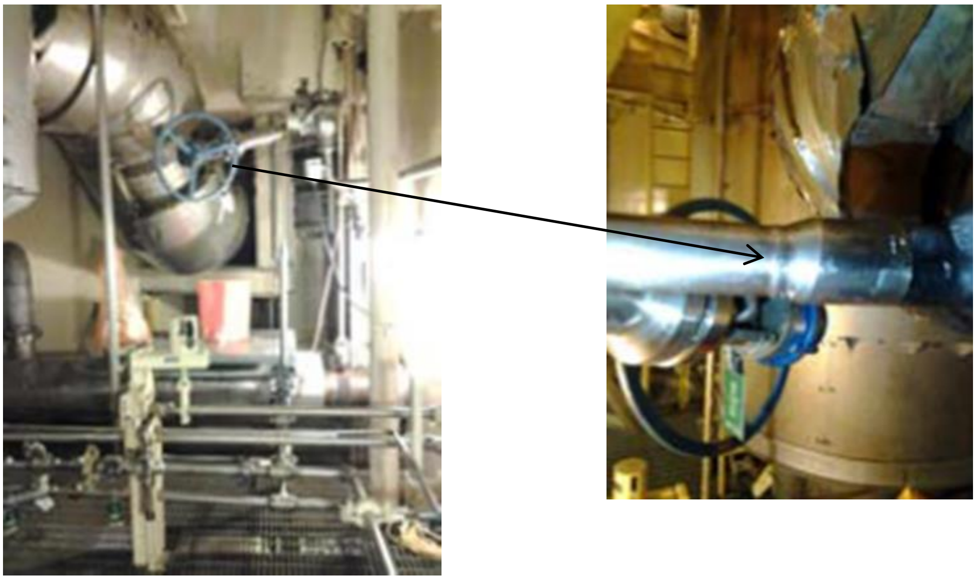


Figure 4-1: Location of Through-Wall Weld Flaw[41]

This section documents an evaluation of the risk significance of this event, which occurred at one of the units (referred to as 'NPP1'; a Babcock & Wilcox NSSS) at the multi-unit plant site. The evaluation was performed approximately 7 months after the event occurred. The inside diameter of the pipe that leaked is 2.125". In the plant-specific PSA model, pipe breaks with an effective break diameter between 0.375" and 1.4" are classified as "small LOCAs", and breaks between 1.4" and 4.5" are classified as "medium LOCAs". If the HPI line was to suffer a double ended guillotine type of break the effective break size would be about 3.0".

In this example an appropriate risk characterization of the leak event is that it can be reasonably assumed to be classified as a precursor to a small or medium LOCA depending on

[41] The photographs are reproduced from a presentation at the 2014 NRC Regulatory Information Conference RIC-2014. The presentation is available from NRC-ADAMS. https://www.nrc.gov/public-involve/conference-symposia/ric/past/2014/docs/abstracts/martinl-t5%20Nozzle%20Weld%20-hv.pdf

the size of the break being in the SLOCA or MLOCA range. This pipe is not large enough to produce a large LOCA. The risk significance of the leak event is that it alters the state of knowledge about the frequency of a small LOCA or medium LOCA initiating event.

The technical approach to characterizing the risk significance of the leak event is to express the risk significance in terms of the change in core damage frequency (CDF) associated with the occurrence of the leak event. To estimate the failure frequency of HPI lines the following key facts are extracted from an operating experience database:

- From January 1, 1970 through June 30, 2014 (the cut-off date for the analysis that is summarized herein) there had been 26 failures in ASME Class 1 HPI injection line piping in 6,342 reactor operating years of world wide experience with PWRs based on U.S. NSSS vendor designs.
- 7 of the 26 HPI line failures occurred at NPP1-3 including the November 2013 leak event at Unit 1.
- Of the 6,342 reactor operating years of service experience about 121.6 is associated with the NPP1 plant site. Up to the time of the November 2013 leak event 120.1 reactor operating years had been accumulated.
- Based on the similarity of HPI line designs, materials and operating conditions, the operating experience data from B&W, Westinghouse, and CE vendor plants was combined. Even though there are significant differences in the vendor designs for larger pipes in the hot legs and cold legs, all three vendors use essentially the same kinds of stainless steel piping for the HPI and other interfacing Class 1 piping.

Failure frequency estimates derived from these key facts are shown in Table 4-1. Plots of the plant-specific specific failure frequencies obtained by Bayes' updating the prior that excludes the NPP1 evidence are shown in Figures 4-2 and 4-3 for the state of knowledge just prior to and following the November 2013 leak event, respectively. As seen in these exhibits the HPI line failure frequencies have been significantly higher at NPP1 compared to the other PWR plants that are covered in the OPEX data query. The HPI line failure frequency has been almost 9 times higher based on the mean values in this table. Comparing the NPP1 Unit 1 estimates before and after the leak event, it is seen that the impact of the November 2013 leak event is an increase in the HPI line failure frequency of about 15% which is not significant in the context of the uncertainty in the estimates. Neither of the 26 HPI line failures nor in any of the PWR pipe failures in the OPEX database for U.S. PWR design plants, were the failures more severe than a large leak (or ≤ 0.8 kg/s). In order to produce a small LOCA as defined in the plant-specific PSA model the failure would need to have an equivalent break size of about 0.375 in. or larger (i.e. ≥ 7 kg/s). To produce a medium LOCA a break would have to exceed 1.4 in. in equivalent break size (i.e. ≥ 100 kg/s). As noted, the HPI line that leaked is not large enough to produce a large LOCA.

Table 4-1: Estimates of HPI Line Failure Frequency

HPI Failure Frequency Case	Failure Count	Exposure [Rx.Yr]	Gamma Distribution Parameters				
			α	β	Mean	5%tile	95%tile
All PWR Data Pooled	26	6,342	0.5	122	4.10E-03	1.70E-05	1.58E-02
PWR Data Excluding NPP1-3 (Bayes Prior)	19	6,220	0.5	164	3.05E-03	1.26E-05	1.17E-02
f_{HPI}^{old} = NPP1 Bayes Update Prior to the Leak Event	6	120.1	6.5	284	2.29E-02	1.04E-02	3.94E-02
f_{HPI}^{new} = NPP1 Bayes Update After the Leak Event	7	121.6	7.5	285	2.63E-02	1.28E-02	4.38E-02

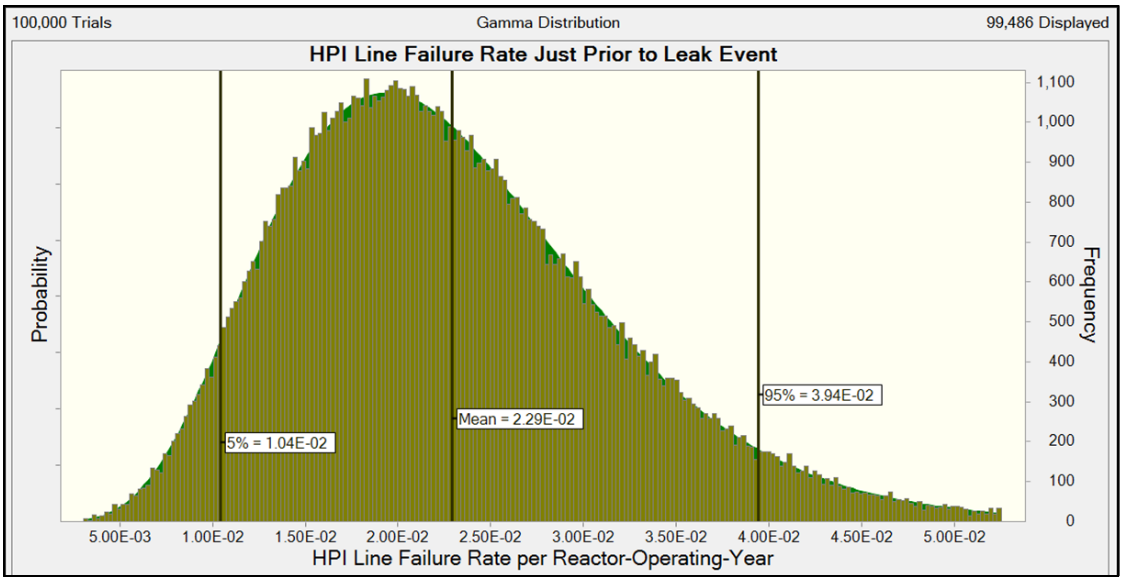


Figure 4-2: 'NPP1' HPI Line Frequency Just Prior to the Leak Event

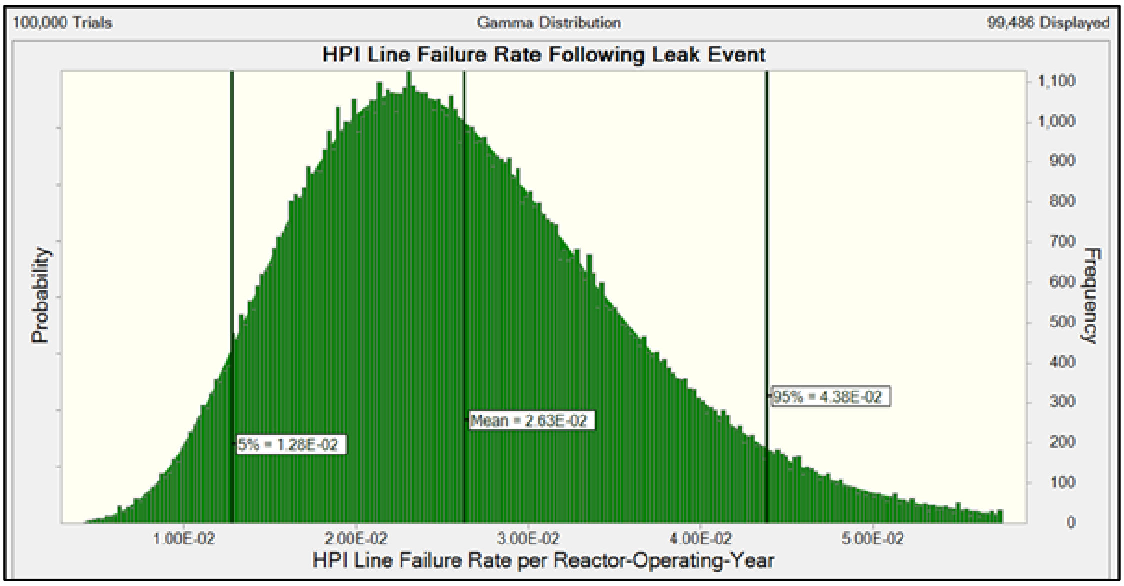


Figure 4-3: 'NPP1' HPI Line Frequency Following the Leak Event

To estimate initiating event frequencies for small and medium LOCAs it is necessary to estimate conditional rupture probabilities (CRPs) as defined by the CRP equations documented in Section 2 of the 2015 "OECD/NEA Event Database Applications Workshop" [70]. The CRP distributions that result for the HPI line are shown in Table 4-2. The distributions calculated using the CRP formulas are used as prior distributions in a Bayes' update so that the evidence collected on pipe failures and ruptures in this study can be taken into account.

The evidence in this case is 26 HPI pipe failures that involve cracks and leaks and no ruptures as large as that which would produce a small or medium LOCA. The Bayes' update was performed using RDAT-Plus™ Software assuming lognormal distributions for the prior from Table 4-2 and the binomial distribution for the likelihood function with the evidence of 0 ruptures out of 26 pipe failures in each of the 5 LOCA categories; Table 4-3.

The LOCA categories in Table 4-2 are those defined in NUREG-1829 [71]. Several additional break sizes were needed for the current analysis in order to determine LOCA frequencies at each end of the SLOCA and MLOCA categories and to identify the specific sizes of the pipe leak in question. Consistent with the methodology used in NUREG-1829 a log-log linear method was used to interpolate and extrapolate the CRP values at these special break sizes. The resulting CRP distributions that were used in this analysis are shown in Table 4-4.

Table 4-2: High Pressure Injection Line LOCA Frequencies from NUREG-1829

Component	LOCA Cat.	Break Size (Inches)	Safety Injection Line LOCA Frequencies Events per Reactor-Calendar Year				
			Mean	5%tile	50%tile	95%tile	RF
Geometric Mean of NUREG-1829 Experts [71]	1	≥ 0.5	1.27E-05	6.40E-07	5.45E-06	4.65E-05	8.5
	2	≥ 1.625	4.58E-06	1.51E-07	1.62E-06	1.74E-05	10.7
	3	≥ 3	7.21E-07	1.53E-08	2.06E-07	2.78E-06	13.5
Lydell Base Case Results Appendix D of NUREG-1829	1	≥ 0.5	1.60E-05	2.62E-07	3.93E-06	6.09E-05	15.2
	2	≥ 1.625	2.33E-06	3.30E-08	5.40E-07	9.02E-06	16.5
	3	≥ 3	9.22E-07	1.28E-08	2.14E-07	3.59E-06	16.7
Mixture Distribution of Experts and Lydell Results	1	≥ 0.5	1.39E-05	3.88E-07	4.73E-06	5.26E-05	11.6
	2	≥ 1.625	3.51E-06	5.50E-08	9.78E-07	1.37E-05	15.8
	3	≥ 3	8.11E-07	1.41E-08	2.11E-07	3.11E-06	14.9

Table 4-3: CRP Distributions for HPI Line

Component	LOCA Category	Break Size (in.)	CRP Distribution Parameters			
			Mean	5th Percentile	Median	95th Percentile
Prior Distribution for CRP	1	≥ 0.5	1.08E-02	5.77E-03	1.02E-02	1.80E-02
	2	≥ 1.625	3.00E-03	5.27E-04	2.10E-03	8.39E-03
	3	≥ 3.0	6.45E-04	1.13E-04	4.53E-04	1.81E-03
Posterior Distribution Using 0 ruptures in 26 pipe failures	1	≥ 0.5	1.05E-02	5.66E-03	9.88E-03	1.72E-02
	2	≥ 1.625	2.79E-03	5.13E-04	2.01E-03	7.64E-03
	3	≥ 3.0	6.34E-04	1.13E-04	4.48E-04	1.77E-03

Table 4-4: CRP Distributions Used in Risk Characterization of HPI Failure Event

Break Size Reference	Break Size (in.)	Mean	5%tile	50%tile	95%tile	How Obtained
SLOCA Lower Bound	0.375	1.49E-02	1.02E-02	1.46E-02	2.10E-02	Log-Log Extrapolation from LOCA Cat 1 and Cat 2
LOCA Category 1	0.5	1.05E-02	5.66E-03	9.88E-03	1.72E-02	Table 12
MLOCA Lower Bound	1.4	3.25E-03	6.98E-04	2.44E-03	8.50E-03	Log-Log Interpolation from LOCA Cat 1 and Cat 2
LOCA Category 2	1.625	2.79E-03	5.13E-04	2.01E-03	7.64E-03	Table 12
SEGB of Unit 1 HPI Pipe where leak occurred	2.125	1.46E-03	2.65E-04	1.04E-03	4.04E-03	Log-Log Interpolation from LOCA Cat 2 and Cat 3
LOCA Category 3	3.0	6.34E-04	1.13E-04	4.48E-04	1.77E-03	Table 12
DEGB of Unit 1 HPI Pipe where leak occurred	3.0052	6.32E-04	1.12E-04	4.45E-04	1.76E-03	Log-Log Interpolation from LOCA Cat 3 and Cat 4
LOCA Category 4	6.75	9.63E-05	1.03E-05	5.66E-05	3.09E-04	Table 12

Estimates of LOCA frequencies for different break sizes as well those for the SLOCA and MLOCA initiating event frequencies for the plant-specific PSA were obtained by multiplying the distributions for the HPI line failure frequency and those for the CRPs using Monte Carlo sampling and the Crystal Ball ™software. The results are shown in Table 4-5 and Table 4-6 for the state of knowledge immediately prior to and following the HPI leak event, respectively. Before-and-after comparisons of the SLOCA and MLOCA frequencies are pictorially displayed in Figures 4-4 and 4-4. In these figures "FOP" represent the plant-specific frequencies prior to the leak event, and "FO" represent the plant-specific frequencies following the leak event. In calculating the results for SLOCA and MLOCA the uncertainties in the HPI line failure frequencies before and after the leak event were assumed to be perfectly correlated, and the CRP uncertainties for each of the break sizes were also assumed to be perfectly correlated.

The results for the change in CDF due to changes in the state of knowledge before and after the occurrence of the leak are shown in Figure 4-6. The mean estimate of the change in CDF is about 5 x 10^{-7} per reactor-calendar-year and there is a high degree of confidence that it is less than 1 x 10^{-6} per reactor-calendar-year. Less than 1% of the Monte Carlo samples exceeded 1 x 10^{-6}. This finding is amplified by the very conservative treatment of uncertainty employed in these analyses (e.g. use of the constrained non-informative distribution method).

Table 4-5: Plant-Specific LOCA Frequencies Prior to Leak Event

Break Size	'NPP1-3' Bayes Updated LOCA Frequencies Prior to Leak Event							
	Mean	5%tile	50%tile	95%tile	RF1[1]	RF2[2]	RF3[3]	Mean Std Error[4]
0.375	3.42E-04	1.41E-04	3.15E-04	6.38E-04	2.2	2.0	2.1	4.98E-07
0.500	2.39E-04	8.57E-05	2.12E-04	4.87E-04	2.5	2.3	2.4	4.11E-07
1.000	1.07E-04	2.37E-05	8.21E-05	2.74E-04	3.5	3.3	3.4	2.82E-07
1.625	6.39E-05	9.24E-06	4.23E-05	1.90E-04	4.6	4.5	4.5	2.25E-07
2.000	3.87E-05	5.53E-06	2.55E-05	1.15E-04	4.6	4.5	4.6	1.37E-07
2.125	3.35E-05	4.76E-06	2.20E-05	9.96E-05	4.6	4.5	4.6	1.19E-07
3.000	1.46E-05	2.03E-06	9.50E-06	4.36E-05	4.7	4.6	4.6	5.24E-08
3.005	1.45E-05	2.02E-06	9.46E-06	4.34E-05	4.7	4.6	4.6	5.22E-08
SLOCA[5]	2.41E-04	1.06E-04	2.28E-04	4.23E-04	2.1	1.9	2.0	3.11E-07
MLOCA[6]	6.71E-05	1.11E-05	4.67E-05	1.90E-04	4.2	4.1	4.1	2.17E-07

Notes:
[1] RF1 = 50%tile/5%tile
[2] RF2 = 95%tile/50%tile
[3] RF3 = SQRT(95%tile/5%tile)
[4] Monte Carlo sampling error in the mean value estimate based on 100,000 samples
[5] SLOCA is defined as a LOCA with break size between 0.375" and 1.4"
[6] MLOCA is defined as a LOCA with break size between 1.4" and 4.5"; largest break possible in 2.125" I.D. pipe is 3.005"

Table 4-6: 'NPP1' LOCA Frequencies after Leak Event

Break Size	'NPP1' Bayes Updated LOCA Frequencies After Leak Event			
	Mean	5%tile	50%tile	95%tile
0.375	3.93E-04	1.71E-04	3.64E-04	7.14E-04
0.500	2.75E-04	1.04E-04	2.45E-04	5.47E-04
1.000	1.23E-04	2.83E-05	9.51E-05	3.11E-04
1.625	7.34E-05	1.09E-05	4.90E-05	2.16E-04
2.000	4.45E-05	6.55E-06	2.95E-05	1.31E-04
2.125	3.84E-05	5.63E-06	2.55E-05	1.14E-04
3.000	1.67E-05	2.40E-06	1.10E-05	4.97E-05
3.005	1.66E-05	2.39E-06	1.10E-05	4.95E-05
SLOCA	2.77E-04	1.30E-04	2.63E-04	4.71E-04
MLOCA	7.70E-05	1.32E-05	5.40E-05	2.17E-04

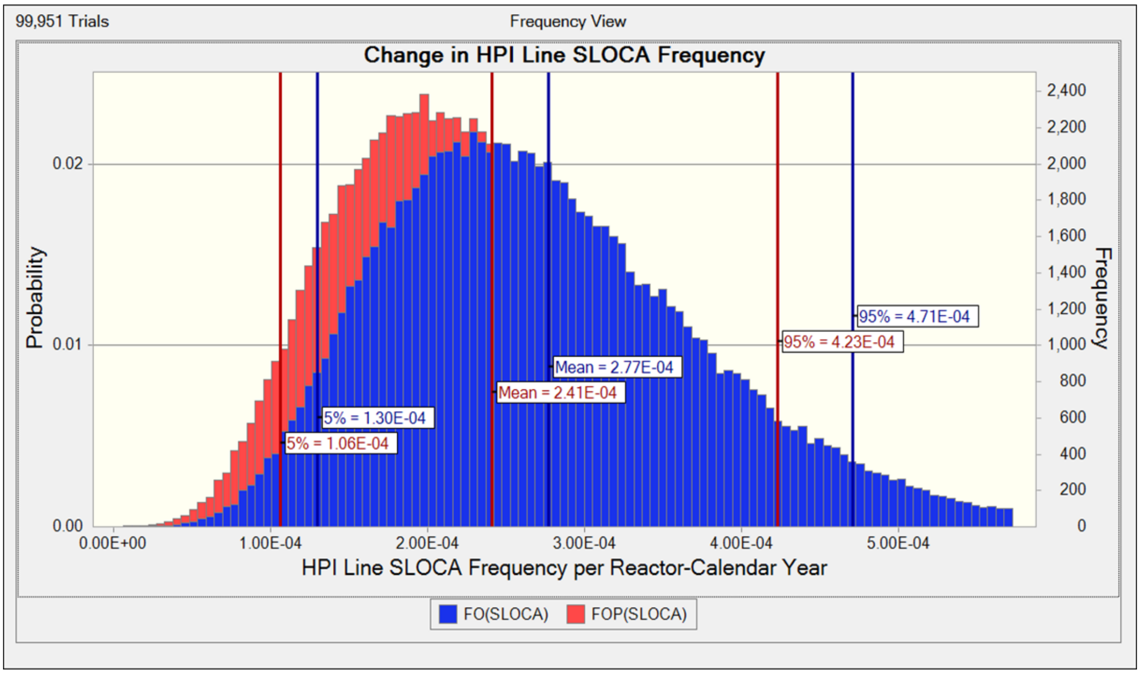


Figure 4-4: Comparison of SLOCA Frequencies Before-and-After the November 2013 Leak Event

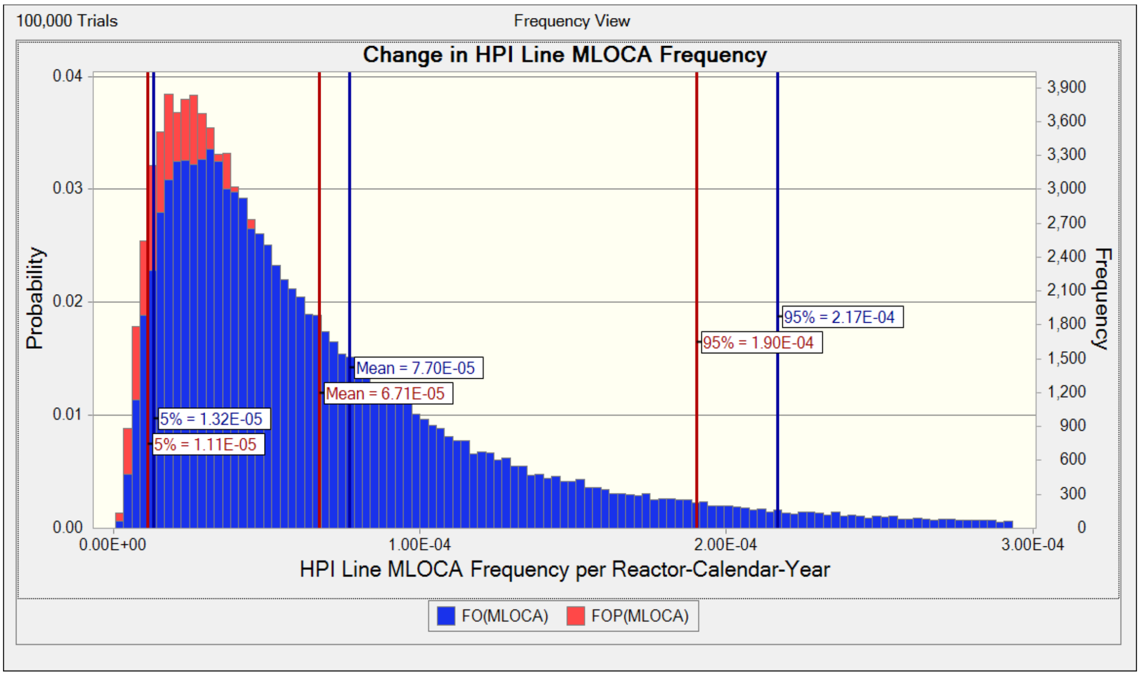


Figure 4-5: Comparison of MLOCA Frequencies Before-and-After the November 2013 Leak Event

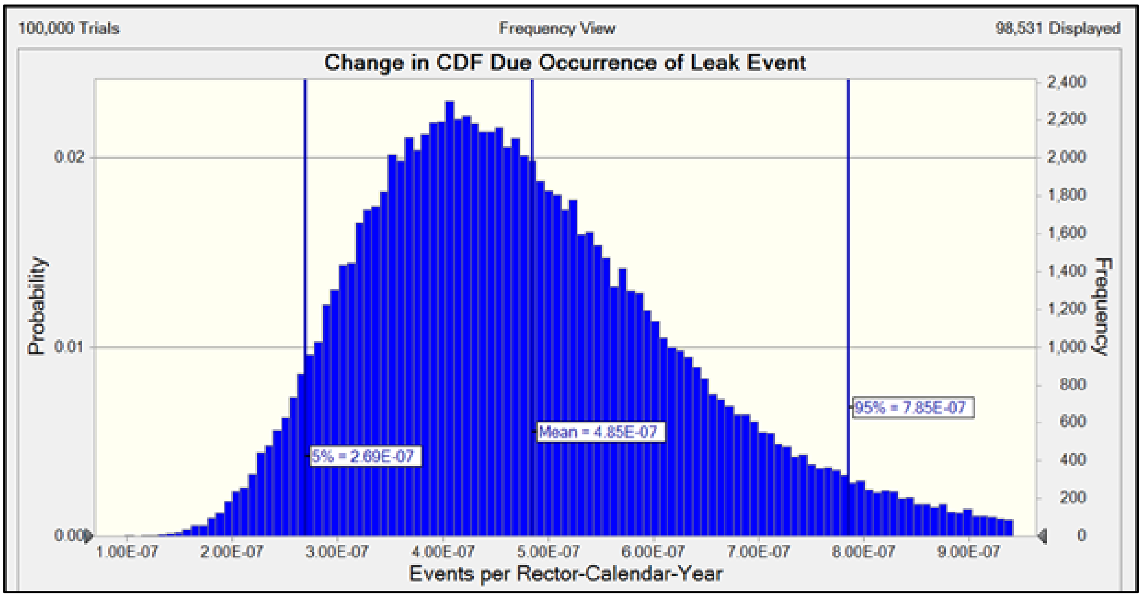


Figure 4-6: Change in CDF due to Occurrence of November 2013 Leak Event

In summary, the conditional probability that a HPI line pipe failure would result in a breach as large as a small LOCA (0.375 in.) at any of the three 'NPP1-3' units is estimated to be about 1.5 x 10^{-2} based on information derived from NUREG-1829 [71]. Similarly, the conditional probability that a re-occurrence of the event would result in a breach as large as a medium LOCA (1.4 in.) is estimated to be about 3.3 x 10^{-3}.

Using a more direct method of analyzing the service data, based on the fact that there have been a total of 268 events involving at least a small leak in PWR reactor coolant pressure boundary piping, as occurred during the November 2013 leak event, the conditional probability of a small LOCA can be estimated to be about 2 x $10^{-3.}$ If only the vibration fatigue events are considered, the conditional probability only rises to about 3 x 10^{-3}. This is regarded as a more realistic estimate than that derived from NUREG-1829 data. Hence the facts in the pipe service data show that the likelihood of a leak due to any cause including high-cycle fatigue rapidly evolving into a pipe rupture is considered to be small.

Based on estimates of the plant-specific initiating event frequencies for HPI line induced small and medium LOCA initiating event frequencies, the change in CDF due to occurrence of the leak event has been estimated to be about 5 x 10^{-7} per reactor calendar year. Uncertainty in this change has been quantified. The chance in CDF would be more than a factor of 10 lower if the CRP derived directly from the service data is used. Hence the 5 x 10^{-7} value for the change in CDF is regarded as a conservative estimate.[42]

[42] The reader of this report is referred to Section 2.7 as well as Reference [1] for additional perspectives on this conclusion.

4.3 Risk Significance of a Thermal Fatigue Event

On December 22, 2014, the North Anna Unit 1 (NAPS-1) reactor power was reduced from 100% to 30% to allow containment entry into the RCS loop rooms to investigate an increased unidentified leak rate of 0.053 gpm. During a containment walkdown, the licensee discovered steam coming from underneath the lagging on the ‘B’ RCS intermediate loop. The licensee’s investigation identified a pressure boundary leak on the ‘B’ loop drain piping between the loop connection and 1-RC-68, the ‘B’ Loop Cold Leg Drain Isolation valve. At that time, the limiting condition for operation action statement of Technical Specification (TS) 3.4.13.b, RCS 1 gpm unidentified leakage, was entered, which required placing the unit in Mode 3 within 6 hours and Mode 5 within 36 hours. On December 23, 2014, the unit was placed in Mode 3 (hot standby), and at 1629 on December 23, 2014, the unit was placed in Mode 5 (cold shutdown).

The licensee determined that the cause of the through-wall leak in the ‘B’ loop drain line elbow was thermal fatigue. Limitations in the industry generic model used to predict swirl penetration thermal fatigue in stagnant RCS branch lines allowed the non-destructive examination (NDE) test frequency of the ‘B’ Loop drain line elbow to be set non-conservatively, which resulted in the thermal fatigue cracking to go unmonitored. The inspectors reviewed the licensee’s root cause evaluation and the Electric Power Research Industry (EPRI) Material Reliability Program (MRP) 146, Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines.[43]

The NRC inspectors determined that the licensee has appropriately evaluated the 2-inch ‘B’ loop drain line elbow and implemented the NDE test frequency requirements in accordance with the licensee’s commitments on the current industry EPRI MRP 146 standard guidelines. No violations of NRC ASME Code Section XI requirements were identified. The licensee is working with EPRI and industry peers to develop a new model and/or new guidance to better predict the impact of the thermal fatigue and other sources of thermal loading. Since the cause was not reasonably within the licensee’s ability to foresee and correct, the inspectors concluded that there was no performance deficiency associated with this issue. The NRC inspectors utilized available risk-informed tools to assess the safety significance of this RCS pressure boundary leakage issue and a detailed risk assessment for this issue was performed by a regional SRA in accordance with NRC Inspection Manual Chapter 0609 Appendix A. The major analysis assumptions included: a conservative one year exposure interval, the two inch break was considered a small loss of cooling accident (SLOCA), conditional core damage probability for SLOCA from the licensee’s full scope NAPS PRA model, and conditional rupture probability from the EPRI report TR-111880 [72]. The result of the analysis was a risk increase for the condition of 1.4E-6/year, representing low to moderate safety significance.

Nonetheless, the inspectors determined a violation of TS occurred because it constituted pressure boundary leakage. North Anna TS Limiting Condition for Operation (LCO) 3.4.13.a requires, in part, that RCS leakage be limited to ‘NO PRESSURE BOUNDARY LEAKAGE’, when in Modes 1, 2, 3 and 4. The associated action statement requires, in part, that with any (RCS) pressure boundary leakage, the unit be in Hot Standby within 6 hours and in Cold Shutdown within the following 36 hours. Contrary to the above, on December 23, 2014, it was discovered that RCS pressure boundary leakage did exist while the unit was in Modes 1, 2, 3 and 4; and that the unit was not placed in Hot Standby within 6 hours and in Cold shutdown within the following 36 hours. Although a violation of the TS occurred, the violation was not attributable to an equipment failure that was avoidable by reasonable licensee quality assurance measures or management controls. Licensee corrective actions included the following:

[43] https://www.nrc.gov/docs/ML1518/ML15189A100.pdf

- Performed an extent of condition NDE of all three cold leg loop drain lines for Unit 1. Results indicated low level craze cracking and a circumferential defect in the similar elbow on ‘C’ drain loop piping. No issues were found in the “A” drain loop piping.
- Replaced the cracked elbow from the ‘B’ loop drain piping.
- Provided the failed ‘B’ loop elbow/piping for materials failure analysis. The results of the evaluation are being used to confirm the direct cause of thermal fatigue and provide more insight on the failure mechanism.
- Performed an evaluation in accordance with ASME Section XI Sub-article IWB-3640, Evaluation Procedures and Acceptance Criteria for Austenitic Piping, of the Unit 1 ‘C’ loop drain elbow. The results determined that it was acceptable to operate the unit until next refueling outage, during which, the repairs were made.
- Discontinued taking RCS cold leg chemistry samples. RCS chemistry samples are currently being taken from the RCS hot leg.
- Informed the EPRI Materials Reliability Program of this failure, and continue to work with the industry on this and similar issues.

4.4 Consideration of Uncertainty in Analyzing Structural Reliability

The sources of uncertainty that are relevant to structural reliability analysis can be classified into two categories: 1) aleatory uncertainties; being associated with physical uncertainty or randomness and, 2) epistemic uncertainties; being associated with understanding or knowledge. Aleatory uncertainty refers to the natural randomness associated with an uncertain quantity, and is often termed “Type I” uncertainty in reliability analysis. Aleatory uncertainty is quantified through the collection and analysis of data. The observed data may be fitted by theoretical distributions, and the probabilistic modelling may be interpreted in the relative frequency sense. Epistemic uncertainty reflects the lack of knowledge or information about a quantity, and is often termed “Type II” uncertainty in reliability analysis. Key aspects of uncertainty treatment in structural reliability analysis are summarized below.

For data-driven models (DDMs), the starting point for the treatment of uncertainty is the development of prior distributions for the failure rates. This is accomplished as a two-step process. The goal was to capture the uncertainty in the state of knowledge about piping system failure rates before and independent of the application of the service data which will be applied in the Bayes’ updating process. This state of knowledge corresponds to that which existed about the time the Reactor Safety Study was performed at which time there were no available estimates from nuclear power plant piping system service experience. Estimates for the overall frequency of major pipe breaks from these data sources were as high as 1×10^{-2} per reactor year and other estimates derived from probabilistic fracture mechanics methods were as low as 1×10^{-6} per reactor year. In developing the initial prior distributions this information was used and fit to the 95th and 5th percentiles of a lognormal distribution giving a median of 1×10^{-4} per plant year and a range factor of 100 to represent the overall frequency of major pipe ruptures on a per reactor year basis.

The next step to development of prior distributions was to convert the above lognormal distribution for pipe ruptures into a set of failure rates for specific systems and damage mechanisms for individual pipe components. A fixed ratio of the number of failures to ruptures was assumed based on a gross analysis of the events in the service data as. In addition, estimates were made of the system and component populations and fractions of the populations susceptible from different damage mechanisms. This information was used to develop scaling factors to convert the above lognormal distribution to the proper units. The range factor of the lognormal distributions was fixed to 100 which provides a very broad distribution and might be regarded as a “slightly informative” prior. This is viewed to be preferable to the use of a non-informative prior, or maximum entropy prior which is often used by statisticians. A very broad lognormal distribution such as this one will not overly bias the results, and it does not seem reasonable to say that we did not know anything about pipe failure rates before we looked at the service data as implied by an non-informative prior. It is also reasonable to assume that the probability distribution is unimodal, unlike the Jeffrey's non-informative prior which states that the probability of the true values of the parameters is highest at the extreme values.

Piping component population uncertainty is treated by using three estimates for the component populations and subjectively assigning probabilities to weight the best estimates and upper and lower bounds. The best estimates are derived from a sample of plant for which details on the piping component populations have been published. The upper and lower bounds are set at percentages (e.g. +/- 25%) above and below these estimates based on engineering judgment.

In early applications the uncertainty in the conditional rupture probability parameter has been treated by using a simple Beta distribution formulation. The main issue with assuming a prior Beta distribution is the estimation of its parameters. Several "constrained" approaches have been proposed. Methods to determine the parameters of the prior Beta distribution include: the method of moments, the PERT approach or the Pearson-Tukey approach. In the absence of data, non-informative priors appear to be a straightforward solution. However, there is often a good knowledge on one constraint, such as the mean probability.

The approach described in Reference [73] is to the use of a constrained non-informative prior. This approach seems to be especially relevant to situations where limited failure data are available to assess the probability that a structural failure occurs, given a degraded condition. Further details on the uncertainty treatment of the CRP are discussed in Section 10.3.

The Bayes' updating process is applied not once but three times to cover the range of estimates in the component populations (high/medium/low) and damage mechanism susceptibility fractions. That process by its standard elements accounts for the uncertainty due to the scarcity of data by the use of a Poisson likelihood function and Bayes' theorem itself performs the important task of defining the proper weights between the prior and the likelihood functions in defining the final form of the of the posterior distributions. The selection of such a large range factor for the lognormal priors provides results that have only a minor influence on the posterior distributions. The three Bayes' estimates are probabilistically combined in a Monte Carlo simulation process. The Monte Carlo simulation combines all the variables statistically to determine the overall uncertainty in the results of the analysis. This is covered in more detail in Sections 5.5 and 5.6.

The number of samples used should be large enough so that the sampling distribution obtained converges to the true distribution of the risk metric. The standard error of the mean (SEM) is a measure of this convergence:

$$SEM = \sigma/\sqrt{n} \tag{4-10}$$

In this equation, σ is the standard deviation of the sampling distribution (i.e., the square root of the variance of this distribution) and n is the number of trials. Based on Equation 4-10 it can be concluded that using a larger number of samples will produce more accurate estimates of the sampling mean.

5. SYSTEMATIC DATA-DRIVEN PIPING RELIABILITY ANALYSIS

Section 5 describes a process for how to systematically perform a risk-informed fitness-for-service or operability determination analysis on the basis of field experience data. This process is a synthesis of insights that have been from a large number of practical applications performed over a 25+ year period. The process identifies pipe failure event database infrastructure considerations and the requirements on database integrity, nomenclature (taxonomy), degradation mechanism knowledgebase, and high-level and supporting requirements for a piping reliability analysis. The term “risk-informed” implies an application that is performed using the best available and most current information concerning piping degradation mechanisms and their mitigation, and in a context of the current probabilistic safety assessment practice. “Risk-informed” also implies enabling confidence in the analysis insights and results by addressing uncertainty [74].

5.1 A Framework for Piping Reliability Analysis

In the context of probabilistic safety assessment (PSA), during the past twenty-five years efforts have been directed towards establishment of comprehensive pipe failure event databases as a foundation for exploratory research to better understand how to effectively organize a piping reliability analysis task. With the focused pipe failure database development efforts have followed good progress with the development of piping reliability analysis frameworks that utilize the full body of service experience data, fracture mechanics analysis insights, expert elicitation results that are rolled into an integrated and risk-informed approach to the estimation of piping reliability parameters with full recognition of the embedded uncertainties.

The ability of a pipe failure event database to support practical applications is closely linked to its completeness and comprehensiveness. Equally important is the knowledge and experience of an analysts in interpreting and applying a database given typical project constraints. Achievement of database "completeness" and "comprehensiveness" is driven by an in-depth understanding of application requirements. These requirements are linked to three general types of applications: 1) high-level, 2) risk-informed, and 3) advanced database applications. Extensive experience exists with PSA-oriented database applications such as:

- Evaluation of “unanalyzed conditions” that involve the potential spatial impacts of high-energy or moderate-energy line breaks.
- Internal flooding PSA; e.g., derivation of internal flooding initiating event frequencies.
- High Energy Line Break (HELB) Analysis. Consideration of HELB in PSA includes estimation of Main Steam and Feedwater line break initiating event frequency.
- Significance Determination Process (SDP) evaluations (also referred to as "accident precursor analysis") to determine the risk significance of pipe degradation or failure.

The application of a framework for performing a data-driven piping reliability analysis is demonstrated via three case studies. The analysis framework consists of seven elements; Figure 5-1:

1. Specifying the Analysis Requirements (AR). In this step the evaluation boundary and calculation cases are defined. Supporting engineering calculations are identified for the purpose of calculation case definition such as specific pipe failure modes (e.g. equivalent break size, EBS) to be considered and with reference to potential spacial impacts as characterized by a zone-of-influence (ZOI). Examples of engineering calculations include transient thermal-hydraulic analysis and simulation.

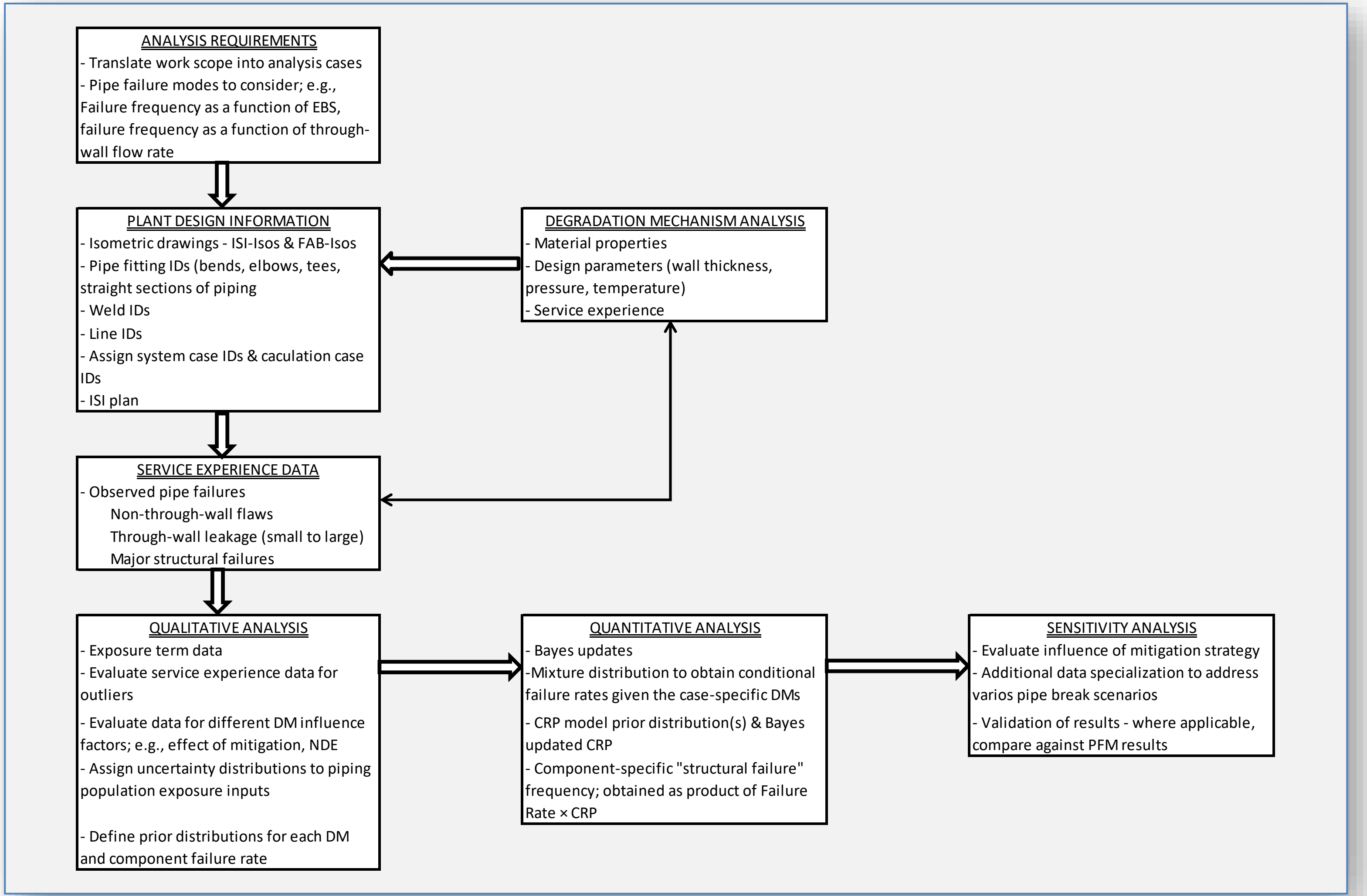


Figure 5-1: Data-Driven Piping Reliability Analysis Framework

2. Plant Design Information (PDI). Piping design information is collected in this step to correlate piping line identification (including pipe segment ID, weld ID) with the different calculation cases. The primary piping design information is contained in three types of isometric drawings: 1) fabrication isometrics, 2) in-service inspection isometrics, and 3) piping analysis/stress analysis isometrics.
3. Service Experience Data (OE). In this step pipe failure event database query definitions corresponding to the calculation cases are applied in order to obtain pipe failure event population data.
4. Qualitative Analysis (ID). This step correlates the pipe failure event population with exposure term data; i.e. the plant reactor operating years that produced the pipe failures times the piping component population at risk.
5. Quantitative Analysis (RA). In this step the piping reliability parameters are calculated for each of the calculation cases.
6. Uncertainty Analysis (UA). Consideration of uncertainties is an integral part of the quantitative analysis. Sources of uncertainty include failure data reporting issues, scarcity of data, poorly characterized component populations, and uncertainties about the physical characteristics of the failure mechanisms and root causes
7. Sensitivity Analysis (SA). This step explores the impact of different assumptions about reliability and integrity management and degradation mechanism (DM) susceptibility.

The piping reliability analysis process is elaborated using lessons learned from three recent case studies. Case Study 1 is a composite made up from three plant-specific Main Steam line piping reliability studies that were performed in the 2013-2016 time period. Case Study 2 is a composite made up from five plant-specific pressurized water reactor (PWR) loss-of-coolant-accident (LOCA) initiating event frequency calculations that were performed in the 2011-2015 time period. Finally, Case Study 3 details the assessment of buried Essential Service Water (ESW) system piping reliability.

5.2 Defining the Analysis Requirements

Case Study 1 is concerned with analysis requirements for three types of Main Steam Line Break) MSLB frequency assessments: 1) MSLB frequency for an existing Level 1 PSA, 2) support to the resolution of an “unanalyzed condition,” and 3) MSLB frequency for a design certification PSA. Hence, quite different motivating factors prompted the three analysis efforts and with profound impact on how respective study was organized.

5.2.1 AR: Case Study 1 – MSLB Frequency Assessment

Level 1 PSA studies require main steam line break (MSLB) initiating event frequencies, including the frequency of MSLB inside and outside the containment. The initiating event frequencies are estimated on a reactor-year basis.

- Development of plant-specific MSLB frequencies to replace the generic initiating event frequency data used in a Level 1 PSA of a 3-loop Westinghouse PWR (Case Study 1.1) which has been in operation since 1981. The Main Steam system (main header, turbine bypass and Auxiliary Feedwater pump steam supply) inside the Turbine Building consists of low-alloy steel (LAS) as opposed to the more commonly used carbon steel (e.g. ASTM A-106 Gr. B). The MS piping inside/outside containment is included in the scope of a risk-informed ISI program. The plant has not experienced any service induced MS pipe failures since star of commercial operation.
- Assessment of MSLB frequency for a specific location inside the Turbine Building at a PWR plant (Case Study 1.2). Based on zone-of-influence (ZOI) evaluations, this location consists of a ca. 100 ft. distance from Engineered Safeguards Switchgear structure and a

steam release with consequential elevated temperature would cause electrical equipment failure. Transient thermal-hydraulic calculations had determined that a MSLB must be ≥ 10 inch diameter to fail the electrical equipment. The reactor has been in operation since 1973 and the MS system is not equipped with Main Steam Isolation Valves (MSIV). The motivating factor for this study was the fact that existing PSA evaluations had shown that the frequency of MS breaks that would potentially benefit from MSIVs was likely significantly higher than originally estimated. These evaluations utilized high-energy line break (HELB) frequencies from the EPRI 3003000079 report [75]. The objective of the new analysis task was to develop a more realistic and plant-specific MSLB frequency.

- The third MSLB frequency study was performed in support of a design certification PSA of an advanced PWR design with a unique MS piping design for which no prior operating experience exists (Case Study 1.3). The scope of the analysis was to develop MSLB frequencies for break scenarios inside and outside the containment and by adapting the existing Main Steam piping field experience from operating commercial nuclear power plants to estimate new piping reliability parameters that account for the unique reliability attributes and influence factors of the advanced PWR MS piping design. The main steam piping material inside containment is stainless steel and outside containment low alloy steel. The MSLB frequency assessment was based on an evolving set of fabrication isometric drawings for the piping inside containment and conceptual piping layout drawings for piping outside containment.

The two MSLB initiating event frequency analyses (Case Study 1.1 and 1.3) calculated a mean value for the frequency of the initiating events and provided a probabilistic representation of the uncertainty for the parameter estimates of the initiating events. Furthermore the initiating event frequencies were estimated on a reactor-operating-year (ROY) basis. The MSLB frequency analysis to address an unanalyzed condition (Case Study 1.2) calculated a mean frequency per ROY and linear feet of MS piping.

5.2.2 AR: Case Study 2 – LOCA Frequency Assessment

In support of a series of 2011-2015 risk-informed Generic Safety Issue (GSI) 191 [76] resolution projects, plant-specific and location-specific LOCA frequencies were developed based on a statistical model of reactor coolant pressure boundary (RCPB) piping. In this approach LOCA frequencies are developed for each Class 1 weld location based on the service experience that shows that all known pipe failures in Light Water Reactor coolant systems have occurred at or near pipe welds. This approach is often referred to as the "bottom-up" method in GSI-191 evaluations to be contrasted with "top down" methods in which a fixed aggregated LOCA frequency is allocated to specific locations.

Summarized in Table 5-1 are the key RCPB design and structural integrity management features of the seven PWR units for which location-specific LOCA frequencies were derived. LOCA frequencies versus break size curves were developed for each weld within the RCPB. For each weld within the RCPB, the analysis requirements common to the seven LOCA frequency assessments included the consideration of pipe size, location within the RCPB, material and degradation mechanism susceptibility. A suite of sensitivity studies were defined to address the effect on LOCA frequency by degradation mitigation. A unique analysis requirement was the need for compatibility with the physics-based CASA Grande computer model [77]; compatibility as far as using a common naming convention for each individual RCPB break location.

5.2.3 AR: Case Study 3 – Failure Rate of Buried ESW Piping

The assessment of failure rate of buried ESW piping was done in support of the development of the 4th OECD-NEA CODAP Topical Report (NEA/CSNI/R(2018)2)[44] on operating experience insights into below ground / buried piping. No specific analysis requirements were defined other than to demonstrate how to utilize the CODAP event database together with a simple piping reliability calculation format.

5.2.4 Step 1 Check List

The practical analysis insights from the three case studies have been organized in a series of checklists to support systematic reviews of licensee submittals of fitness-for-service evaluations, and to assist the OEAD staff in validating licensee submittals by performing independent fitness-for-service evaluations. Table 5-2 represents the sub task "Analysis Requirements" (AR) checklist.

[44] https://www.oecd-nea.org/documents/2018/sin/csni-r2018-2.pdf

Table 5-1: Main RCPB Design Features of Five PWR Plants

PWR Plant ID	NSSS Design	RCPB Design Features	PWSCC Mitigation Status @ Time of Analysis
Plant A Unit 1 (2 unit site)	Westinghouse Model 414	4 closed heat transfer loops; each loop consisting 1 Hot Leg & 1 Cold Leg. The piping is made of cast austenitic stainless steel (SA-351 CF8A) Pressurizer surge line, spray and relief lines are made of austenitic stainless steel material. Three redundant safety injection trains.	• Full Structural Weld Overlay (FSWOL) using Alloy 52/152 material applied to pressurizer surge line DMWs • FSWOLs using Alloy 52/152 applied to pressurizer spray line, safety line and relief line DMWs
Plant B – Unit 1 Plant B – Unit 2	Westinghouse Model 412	4 closed heat transfer loops; each loop consisting 1 Hot Leg & 1 Cold Leg. The piping is made of cast austenitic stainless steel (SA-351 CF8A) Pressurizer surge line, spray and relief lines are made of austenitic stainless steel material. Two redundant safety injection trains.	• SG inlet/outlet nozzles are of similar design with no Alloy 600 or 82/182 material exposed to the primary water • RPV outlet nozzle-to-safe-end locations mitigated by MSIP process • FSWOL using Alloy 52/152 material applied to pressurizer surge line DMWs • FSWOL using Alloy 52/152 material applied to pressurizer safety and relief valve-to-safe-end locations • FSWOL using Alloy 52/152 material applied to pressurizer spray line DMW locations
Plant C Single unit site	Westinghouse Model 412	Same as Plant B1/B2	With one exception, same as for plant B. Exception being no PWSCC mitigation through application of MSIP.
Plant D – Unit 1 Plant D – Unit 2	Combustion Engineering; 3rd gen. NSSS design	Two closed heat transfer loops; each loop consisting of 1 Hot Leg 42” (ID) & 2 Cold Legs (30” ID). Respective loop is fabricated from stainless steel clad carbon steel, seam welded pipe sections	• Similar-metal RPV inlet/outlet and SG inlet/outlet welds. All Hot Leg/Cold Leg small-bore penetrations replaced with PWSCC-resistant material. • All DMW locations at hot leg temperature have been mitigated by the MSIP process • Pressurizer surge line @ Hot Leg • Decay heat removal @ Hot Leg • Hot Leg drains • Pressurizer relief valves
Plant E Single unit site	Combustion Engineering; 1st gen. NSSS design	Two closed heat transfer loops; each loop consisting of 1 Hot Leg 42” (ID) & 2 Cold Legs (30” ID). Respective loop is fabricated from stainless steel clad carbon steel, seam welded pipe sections	• Similar-metal RPV inlet/outlet and SG inlet/outlet welds. • MSIP applied to pressurizer surge line DMW locations • Pressurizer relief valve line DMW (Alloy 600) locations replaced with Alloy 690 material. Alloy 690 is considered resistant to PWSCC

Table 5-2: Sub Task AR Checklist

CALCULATION SPECIFICATION / INPUT		JUSTIFICATION / MOTIVATION	DEVIATION / RESOLUTION ('WORK-AROUND')
☒	Work Scope Definition	Defines overall objectives, including expected outcomes, schedule, review process, software requirements	
☒	Project Mobilization	Precise list of design information needed including a documentation of how the information will be used to support a piping reliability calculation. Project communications & meeting schedule	
☒	Definition of Analysis Case(s)	This task defines the reliability parameters to be derived including all conditional parameters including sensitivity cases to be considered. Interface(s) (if any) with other engineering and/or safety analysis activities needs to be defined.	
☒	Pipe Failure Modes to Consider	Clear definition of: • Non-conforming (i.e. not fit for continued service) • Perceptible leakage • Leakage necessitating controlled plant shutdown • Spectrum of break sizes	
☒	Identify the Parameters to be Estimated & Data Required for Estimation	Defines data source(s) and time period(s) to consider. Use of generic vs. plant-specific data. This item refers also to the piping failure database that produces event population data and exposure term.	
☒	Analysis Tools	Description of calculation format including embedded computer code(s)	
☒	Compatibility with Other Analysis Tasks or Project Tasks	Common definition of failure modes and common nomenclature	
☒	Documentation Requirements	Results presentation format, input/output data, assumptions	
☒	Technology Transfer	Training on calculation format and calculation tools	
☒	Independent Review & Comment Resolution	Identify reviewer(s), review process & schedule	

5.3 PLANT DESIGN INFORMATION

Availability of accurate and detailed plant piping design information is critical to all downstream piping reliability analysis process steps. Examples of design information include piping and instrumentation diagrams (P&IDs), isometric drawings, in-service inspection (ISI) information, and transient thermal hydraulics analysis results pertaining to zone-of-influence (ZOI) evaluations. The evaluation boundary, calculation case definition and exposure term definition rely on piping design information.

5.3.1 PDI: Case Study 1 – MSLB Frequency Assessment

Common to the three MSLB frequency studies was the detailed review of piping design information. The isometric drawing information was transcribed into a Microsoft® Excel spreadsheet with a spreadsheet tab for each initiating event with columns for pipe line number, line description, component ID, component type, material, pipe diameter and schedule number, and isometric drawing ID. Two examples of transcribed isometric drawing information are detailed in Tables 5-3 and 5-4.

Table 5-3: Sample PWR Main Steam Pipe Lengths Inside Containment

MS Segment	Pipe Diameter [NPS]	Linear ft. of Piping			
		Plant 1	Plant 2	Plant 3	Plant 4
MS Header – from Turbine Bldg. pen. to HP Turbine	34	445.8	407.6	352.0	--
	24	479.8	347.2	416.4	--
	26	--	--	--	639.6
Turbine Bypass – from Main Header to Condenser	24	261.5	169.2	[redacted]	--
	20	--	--	--	8.9
	18	74.5	[redacted]		--
	16	--	--	--	349.9
	14	--	--	--	398.4
	12	103.3	[redacted]		--
Total Pipe Length:		1364.9	MS Turbine Bypass pipe length assumed similar to B&W-1 (± 10%)		1396.8

Table 5-4: Sample PWR Main Steam Piping Component Population Data

MSLB Evaluation Boundary	3-Loop Westinghouse PWR					
	Material	Pipe Size [in]	Bend-Long Radius	Elbow	Pipe Length [ft]	Weld Count incl. Branch Connection Welds
Inside Turbine Building						
Turbine Bypass Lines to Condensers	LAS	14	1	12	398.4	23
	LAS	16	4	8	342.9	16
	LAS	20	--	--	8.9	4
MS Headers to HP Turbines	LAS	26	--	7	639.6	24
Piping Component Population:			5	27	1389.8	69
Inside Containment						
S/G-1 MS Header	CS	32	--	--	156.9	--
S/G-2 MS Header	CS	32	--	--	116.9	--
S/G-3 MS Header	CS	32	--	--	155.9	--
Total MS Header Length:			--	--	429.7	N/A
Containment Penetration Area						
S/G-1 MS Header	CS	32	--	--	17.8	--[1]
S/G-2 MS Header	CS	32	--	--	17.8	--

MSLB Evaluation Boundary	3-Loop Westinghouse PWR					
	Material	**Pipe Size [in]**	**Bend-Long Radius**	**Elbow**	**Pipe Length [ft]**	**Weld Count incl. Branch Connection Welds**
S/G-3 MS Header	CS	32	--	--	17.8	--
Total MS Header Length:			--	--	53.4	
Intermediate Building						
S/G-1 MS Header	CS	32	--	--	105.1	--
S/G-2 MS Header	CS	32	--	--	35.0	--
S/G-3 MS Header	CS	32	--	--	112.6	--
Total MS Header Length:					252.7	
24" Safety Valve Headers						
Header 1	CS	24	--	--	49.8	--
Header 2	CS	24	--	--	49.8	--
Header 3	CS	24	--	--	49.8	--
Total MS Safety Valve Header Length:					149.4	N/A
Main Steam Supply to Turbine-Driven AFW Pump						
Header 1	CS	4	--	--	89.6	
Header 2	CS	4	--	--	54.6	
Total AFW Pump Steam Supply Header Length:			--	--	144.2	--
Total MS Pipe Length:			--	--	2419.2	N/A
[1] "--" – undisclosed in this report						

5.3.2 PDI: Case Study 2 – LOCA Frequency Assessment

In pressurized water reactors (PWRs), the size and location of a reactor coolant pressure boundary (RCPB) pipe break could influence the amount and chemistry of debris formation and the timing and need for actions to initiate or terminate containment sprays and recirculation cooling. The frequency of location-specific pipe breaks and the consequential zone-of-influence by pipe insulation debris have been studied within the scope of the risk-informed resolution of Generic Safety Issues 191 [76].

Based on a detailed review of the piping system isometric drawings, the first task of the LOCA frequency assessment was to determine the piping component populations for each pipe size and system within the RCPB. The isometric drawing information was transcribed into a Microsoft® Access database (Figures 5-2 & 5-3) format consisting of 24 database categories for each line item, including:

- Pipe Line Number
- System Description; e.g. RCS Hot Leg, RCS Cold Leg, Pressurizer Surge Line
- Isometric Drawing Number
- Location Within Containment (e.g. Plant Elevation, Zone)
- ASME XI Weld Category
- Material Designation
- Nominal Pipe Size (NPS)
- DEGB Break Size (= NPS × √2)
- Degradation Mechanism per Risk-Informed ISI Plan.

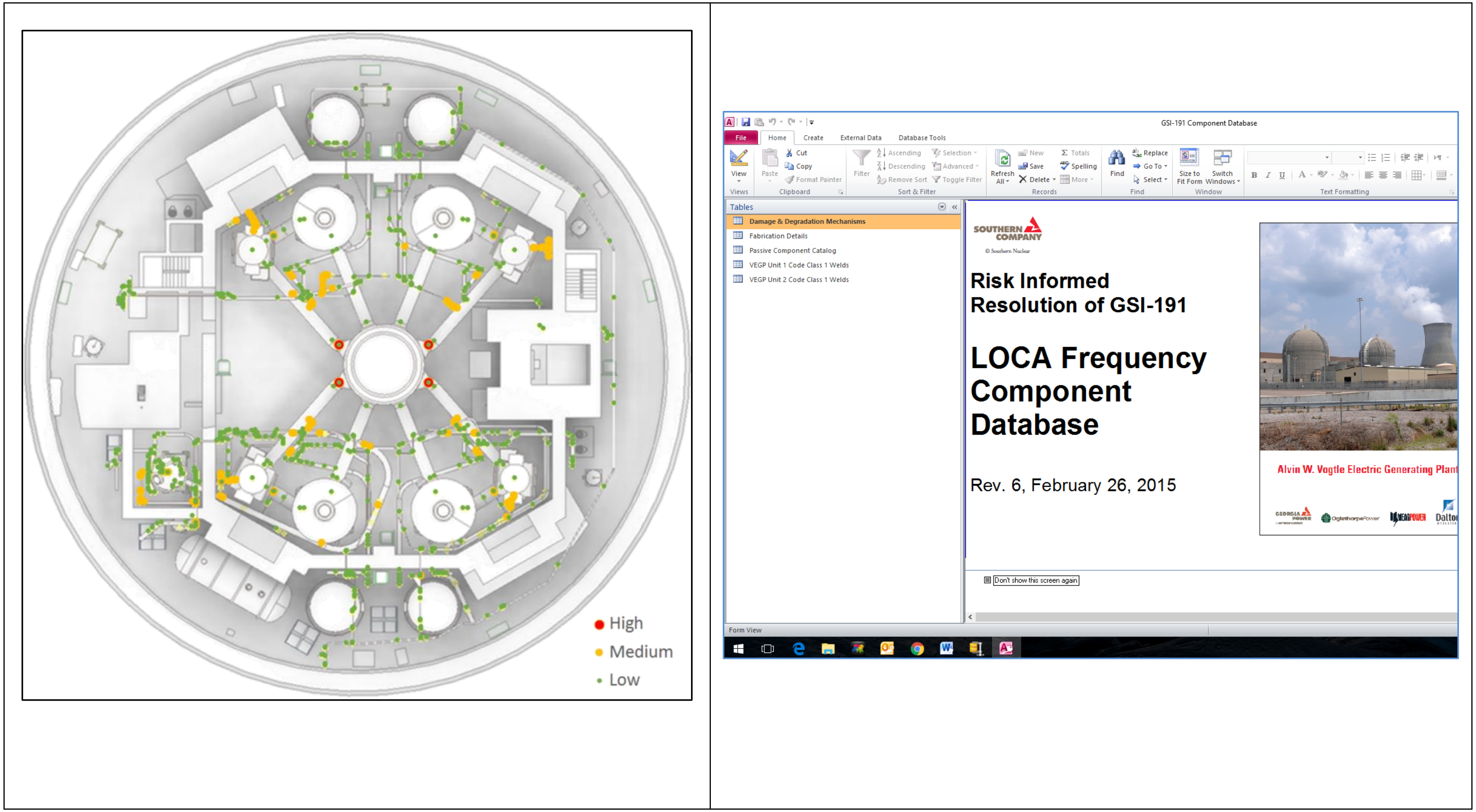
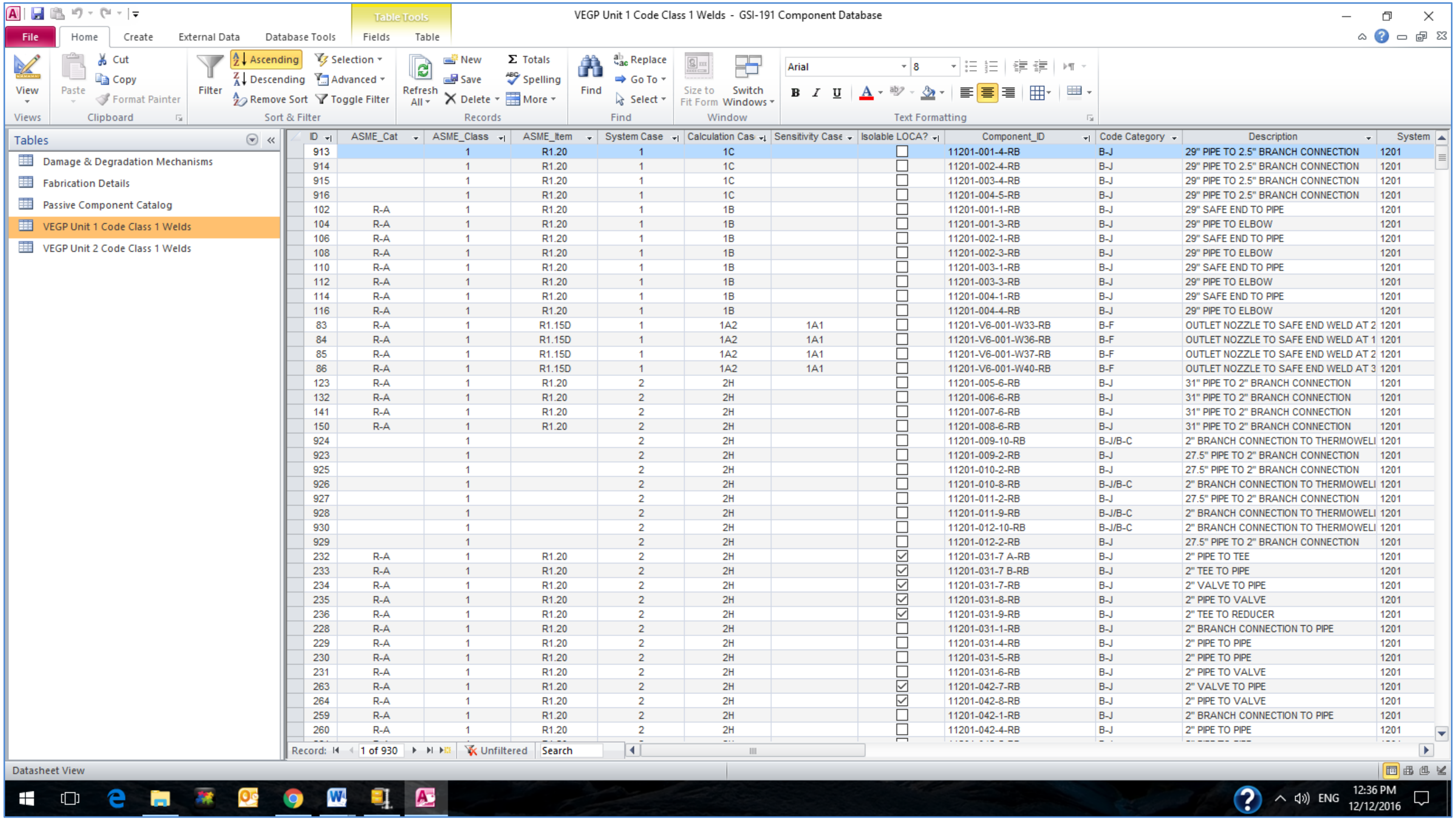


Figure 5-2: PWR Location Dependent LOCA Frequency & Associated 'LOCA Frequency Component Database'



ID	ASME_Cat	ASME_Class	ASME_Item	System Case	Calculation Cas	Sensitivity Case	Isolable LOCA?	Component_ID	Code Category	Description	System
913		1	R1.20	1	1C		☐	11201-001-4-RB	B-J	29" PIPE TO 2.5" BRANCH CONNECTION	1201
914		1	R1.20	1	1C		☐	11201-002-4-RB	B-J	29" PIPE TO 2.5" BRANCH CONNECTION	1201
915		1	R1.20	1	1C		☐	11201-003-4-RB	B-J	29" PIPE TO 2.5" BRANCH CONNECTION	1201
916		1	R1.20	1	1C		☐	11201-004-5-RB	B-J	29" PIPE TO 2.5" BRANCH CONNECTION	1201
102	R-A	1	R1.20	1	1B		☐	11201-001-1-RB	B-J	29" SAFE END TO PIPE	1201
104	R-A	1	R1.20	1	1B		☐	11201-001-3-RB	B-J	29" PIPE TO ELBOW	1201
106	R-A	1	R1.20	1	1B		☐	11201-002-1-RB	B-J	29" SAFE END TO PIPE	1201
108	R-A	1	R1.20	1	1B		☐	11201-002-3-RB	B-J	29" PIPE TO ELBOW	1201
110	R-A	1	R1.20	1	1B		☐	11201-003-1-RB	B-J	29" SAFE END TO PIPE	1201
112	R-A	1	R1.20	1	1B		☐	11201-003-3-RB	B-J	29" PIPE TO ELBOW	1201
114	R-A	1	R1.20	1	1B		☐	11201-004-1-RB	B-J	29" SAFE END TO PIPE	1201
116	R-A	1	R1.20	1	1B		☐	11201-004-4-RB	B-J	29" PIPE TO ELBOW	1201
83	R-A	1	R1.15D	1	1A2	1A1	☐	11201-V6-001-W33-RB	B-F	OUTLET NOZZLE TO SAFE END WELD AT 2	1201
84	R-A	1	R1.15D	1	1A2	1A1	☐	11201-V6-001-W36-RB	B-F	OUTLET NOZZLE TO SAFE END WELD AT 1	1201
85	R-A	1	R1.15D	1	1A2	1A1	☐	11201-V6-001-W37-RB	B-F	OUTLET NOZZLE TO SAFE END WELD AT 2	1201
86	R-A	1	R1.15D	1	1A2	1A1	☐	11201-V6-001-W40-RB	B-F	OUTLET NOZZLE TO SAFE END WELD AT 3	1201
123	R-A	1	R1.20	2	2H		☐	11201-005-6-RB	B-J	31" PIPE TO 2" BRANCH CONNECTION	1201
132	R-A	1	R1.20	2	2H		☐	11201-006-6-RB	B-J	31" PIPE TO 2" BRANCH CONNECTION	1201
141	R-A	1	R1.20	2	2H		☐	11201-007-6-RB	B-J	31" PIPE TO 2" BRANCH CONNECTION	1201
150	R-A	1	R1.20	2	2H		☐	11201-008-6-RB	B-J	31" PIPE TO 2" BRANCH CONNECTION	1201
924		1		2	2H		☐	11201-009-10-RB	B-J/B-C	2" BRANCH CONNECTION TO THERMOWELI	1201
923		1		2	2H		☐	11201-009-2-RB	B-J	27.5" PIPE TO 2" BRANCH CONNECTION	1201
925		1		2	2H		☐	11201-010-2-RB	B-J	27.5" PIPE TO 2" BRANCH CONNECTION	1201
926		1		2	2H		☐	11201-010-8-RB	B-J/B-C	2" BRANCH CONNECTION TO THERMOWELI	1201
927		1		2	2H		☐	11201-011-2-RB	B-J	27.5" PIPE TO 2" BRANCH CONNECTION	1201
928		1		2	2H		☐	11201-011-9-RB	B-J/B-C	2" BRANCH CONNECTION TO THERMOWELI	1201
930		1		2	2H		☐	11201-012-10-RB	B-J/B-C	2" BRANCH CONNECTION TO THERMOWELI	1201
929		1		2	2H		☐	11201-012-2-RB	B-J	27.5" PIPE TO 2" BRANCH CONNECTION	1201
232	R-A	1	R1.20	2	2H		☑	11201-031-7 A-RB	B-J	2" PIPE TO TEE	1201
233	R-A	1	R1.20	2	2H		☑	11201-031-7 B-RB	B-J	2" TEE TO PIPE	1201
234	R-A	1	R1.20	2	2H		☑	11201-031-7-RB	B-J	2" VALVE TO PIPE	1201
235	R-A	1	R1.20	2	2H		☑	11201-031-8-RB	B-J	2" PIPE TO VALVE	1201
236	R-A	1	R1.20	2	2H		☑	11201-031-9-RB	B-J	2" TEE TO REDUCER	1201
228	R-A	1	R1.20	2	2H		☐	11201-031-1-RB	B-J	2" BRANCH CONNECTION TO PIPE	1201
229	R-A	1	R1.20	2	2H		☐	11201-031-4-RB	B-J	2" PIPE TO PIPE	1201
230	R-A	1	R1.20	2	2H		☐	11201-031-5-RB	B-J	2" PIPE TO PIPE	1201
231	R-A	1	R1.20	2	2H		☐	11201-031-6-RB	B-J	2" PIPE TO VALVE	1201
263	R-A	1	R1.20	2	2H		☑	11201-042-7-RB	B-J	2" VALVE TO PIPE	1201
264	R-A	1	R1.20	2	2H		☑	11201-042-8-RB	B-J	2" PIPE TO VALVE	1201
259	R-A	1	R1.20	2	2H		☐	11201-042-1-RB	B-J	2" BRANCH CONNECTION TO PIPE	1201
260	R-A	1	R1.20	2	2H		☐	11201-042-4-RB	B-J	2" PIPE TO PIPE	1201

Figure 5-3: Example of Transcribed Isometric Drawing Information

5.3.3 PDI: Case Study 3 – Failure Rate of Buried ESW Piping

This analysis was performed in order to demonstrate how to utilize the CODAP event database to obtain 'order-of-magnitude' piping reliability parameters without access to any detailed buried piping design information. The assessment relied on publically available information; examples of such information is documented in NEA/CSNI/R(2018)2 (https://www.oecd-nea.org/nsd/docs/2018/csni-r2018-2.pdf).

5.3.4 Step 2 Check List

The practical analysis insights from the three case studies have been organized in a series of checklists to support systematic reviews of licensee submittals of fitness-for-service evaluations, and to assist the OEAD staff in validating licensee submittals by performing independent fitness-for-service evaluations. Table 5-5 represents the sub task "Plant Design Information" (PDI) checklist.

Table 5-5: Sub Task PDI Checklist

PIPING DESIGN INFORMATION		JUSTIFICATION / MOTIVATION	DEVIATION / RESOLUTION ('WORK-AROUND')
☒	Marked-up Piping & Instrumentation Diagram (P&ID)	For analyses that support PSA applications, a marked-up P&ID identifies the evaluation boundary (piping line IDs). For plants with a RI-ISI program, especially developed P&IDs may exist that identify pipe segment definitions per the RI-ISI analysis files.	
	<u>Isometric Drawing(s)</u>	Essential information for any piping reliability analysis: • Fabrication data • Weld IDs • Pipe fitting (bend, elbow, tee, reducer, expander) IDs • Pipe lengths The isometric drawings provide plant-specific exposure term information and enable an assessment of plant-to-plant variability in exposure term. Additional information	N/A
☒	Fabrication Isometrics		
☒	In-Service Inspection (ISI) Isometrics		
☒	Stress Analysis (or Piping Analysis) Isometrics		
☒	ISI Program Plan & Weld Database	Supplements the isometric drawing information	
☒	Isometrics Not Available?	Locate published information (e.g. EPRI TR-111880[45] or NUREG/CR-4407[46]) and / or ISI/RI-ISIProgram Plans	

[45] Fleming, K.N. et al, Piping System Failure Rates and Rupture Frequencies for Use in Risk-Informed In-Service Inspection Applications, TR-111880, Electric Power Research Institute, Palo Alto, CA, 1999. Table A-5 of the report includes "estimated number of welds and linear feet of piping per plant system and NSSS vendor. No differentiation between Code Class, pipe size or material, however. According to Table A-5, a "typical" PWR plant has 409 welds in the Reactor Coolant System. Based on isometric drawing reviews, one 4-loop PWR of Westinghouse design has a total 289 B-J (similar-metal) butt welds of pipe size 1.25" diameter or greater and 22 B-F (dissimilar metal) welds.

[46] Wright, R.E., Steverson, J.A. and Zuroff, W.F., Pipe Break Frequency Estimation for Nuclear Power Plants, NUREG/CR-4407, USNRC, Washington, DC, 1987. Tables 6 through 11 include weld count and pipe length data from 18 different nuclear power plants. The information contained in these tables shows the weld population and pipe lengths for three pipe size categories: 2", > 2" to 6", and > 6".

5.4 SERVICE EXPERIENCE DATA

Statistical models of piping reliability rely on real-life observations of pipe degradation and failure over time. The fundamental principles of statistical reliability theory and life testing apply to the determination if the performance (as in structural integrity) of piping system components under in-service conditions, with or without in-service inspection or other integrity management activity, is within specifications for a desired operating period. As well, the statistical reliability theory applies to the determination of recurring failures and the underlying times-to-failure distributions. In order to perform a meaningful analysis, a high-quality pipe failure event database is queried (or reduced) commensurate with certain calculation case definitions. This section documents practical examples of how to prepare event population data for input to piping reliability analysis.

5.4.1 Data Quality & Completeness

The term "data quality" is an attribute of the processes that have been implemented to ensure that any given database record (including all of its constituent elements, or database fields) can be traced to the source information. The term also encompasses "fitness-for-use", that is, the database records should contain sufficient technical detail to support database applications.

Most, if not all database applications are concerned with evaluations of event populations as a function of calendar time, operating time or component age at time of failure. The technical scope of the evaluations includes determination of trends and patterns and data homogeneity, and assessment of various statistical parameters of piping reliability. Therefore, an intrinsic aspect of practical database applications is the completeness and comprehensiveness of an event database. Do the results of an application correctly reflect the effectiveness of in-service inspection, aging management, and/or water chemistry programs? Does the database capture "all" relevant operational events? In summary:

- Completeness is an indication of whether or not all the data necessary to meet current and future analysis demands are available in the database. An intrinsic aspect of the case studies in this report is an assessment of data completeness through a search for additional relevant information in in-service inspection record databases and condition report databases of respective case study organization.
- Comprehensiveness is concerned with how well a pipe failure database captures the full and appropriate range of reliability attributes (e.g., material properties, dimensional data) and influence factors (e.g., operating environment, pipe stresses).

The governing documents for a pipe failure event database should be consulted for additional information regarding the processes that have been implemented to address database completeness and comprehensiveness.

Service experience with piping systems has been an important consideration in the development of RI-ISI methodologies. Implicitly, service experience considerations enter into all aspects of a plant-specific RI-ISI program development effort. Depending on the selected RI-ISI methodology, explicit pipe failure data considerations play an important role in certain aspects of the program development; degradation mechanism assessment and risk impact assessment. Once a RI-ISI program has been fully implemented, the explicit use of pipe failure data may also enter into the future updating of a program. New pipe failure data could potentially result in changes to the degradation mechanism assessment as well as the risk impact assessment. In the end, specific weld locations could be removed or added to an inspection program. Figure 5-4 illustrates the possible effect of piping reliability parameter selection on the number of weld examinations [78].

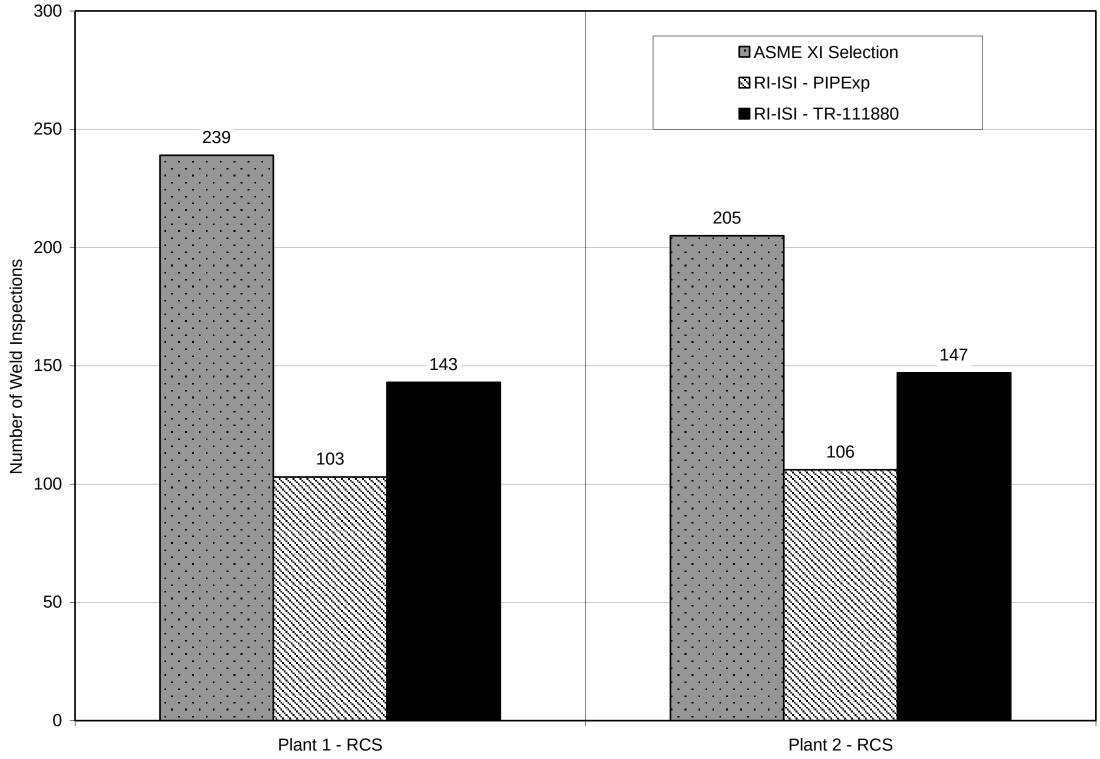


Figure 5-4: Dependence of RI-ISI Weld Selection on Pipe Failure Data

5.4.2 Plant-to-Plant OE Variability

In generating event population data by querying an event database various data screening criteria should be applied to ensure that the OE data obtained correspond to the analysis requirements. To demonstrate data screening, the industry wide experience with Service Water system piping is used as an example.

Figures 5-5 and 5-6 represent the U.S. operating experience with Service Water piping. In Figure 5-5 the U.S. Boiling Water Reactor (BWR) and PWR service experience is organized by type of ultimate heat sink and degradation mechanisms. In Figure 5-6 the U.S. PWR-specific operating experience is organized by the nominal pipe size of affected piping. In Figure 5-7 the overall operating experience with Service Water piping is organized by material type. According to Figure 5-8, there is significant plant-to-plant variability in the Service Water piping operating experience.

The most common Service Water piping materials are carbon steel (e.g. ASTM A-106 Gr. B) and 300-Series austenitic stainless steels. Some U.S. and foreign nuclear power plants have replaced the original piping with corrosion-resistant or “high-alloyed” stainless steel material such as AL6-XN or 254/654-SMO. Based on the operating experience review, it can be concluded that the following piping reliability attributes and influence factors should be considered in developing homogenous event populations:

- Type of material
- Raw water type (e.g., lake, river or sea water)
- Pipe size
- Piping integrity management practice.

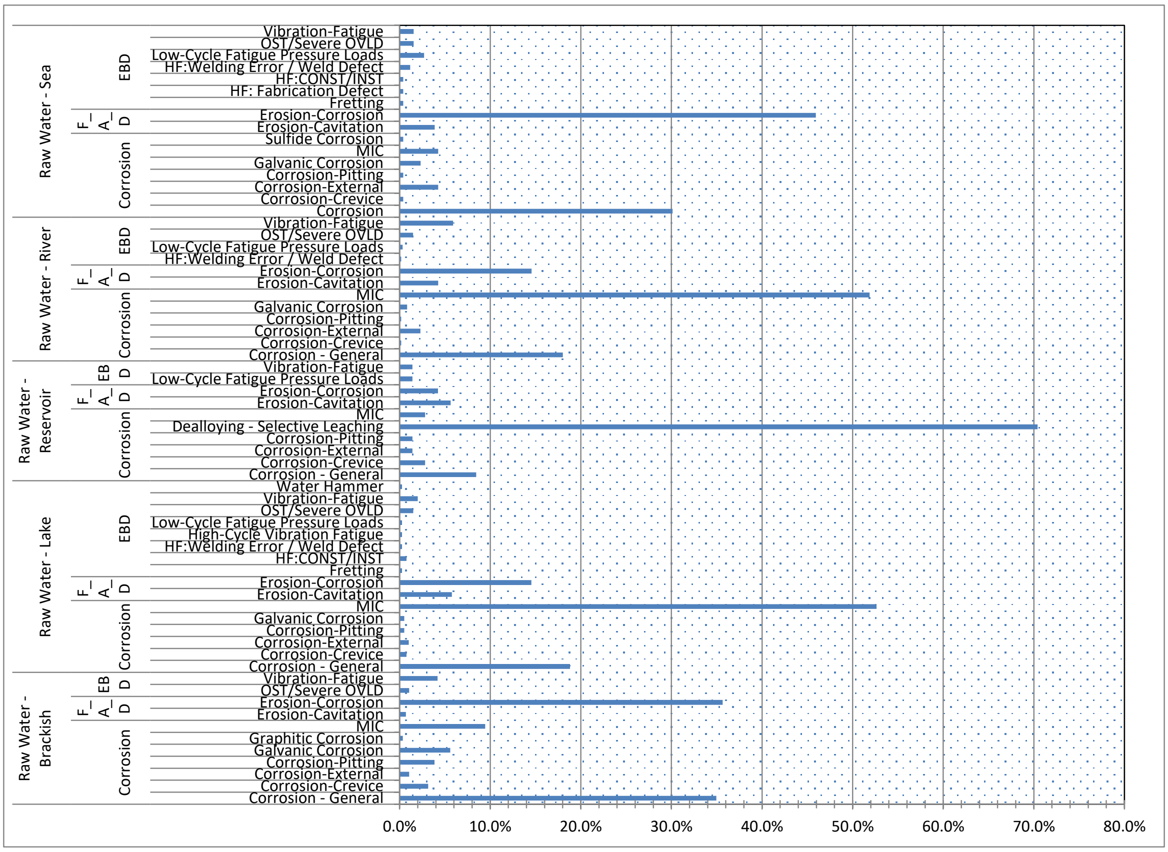


Figure 5-5: Degradation Mechanisms Acting on Service Water Piping

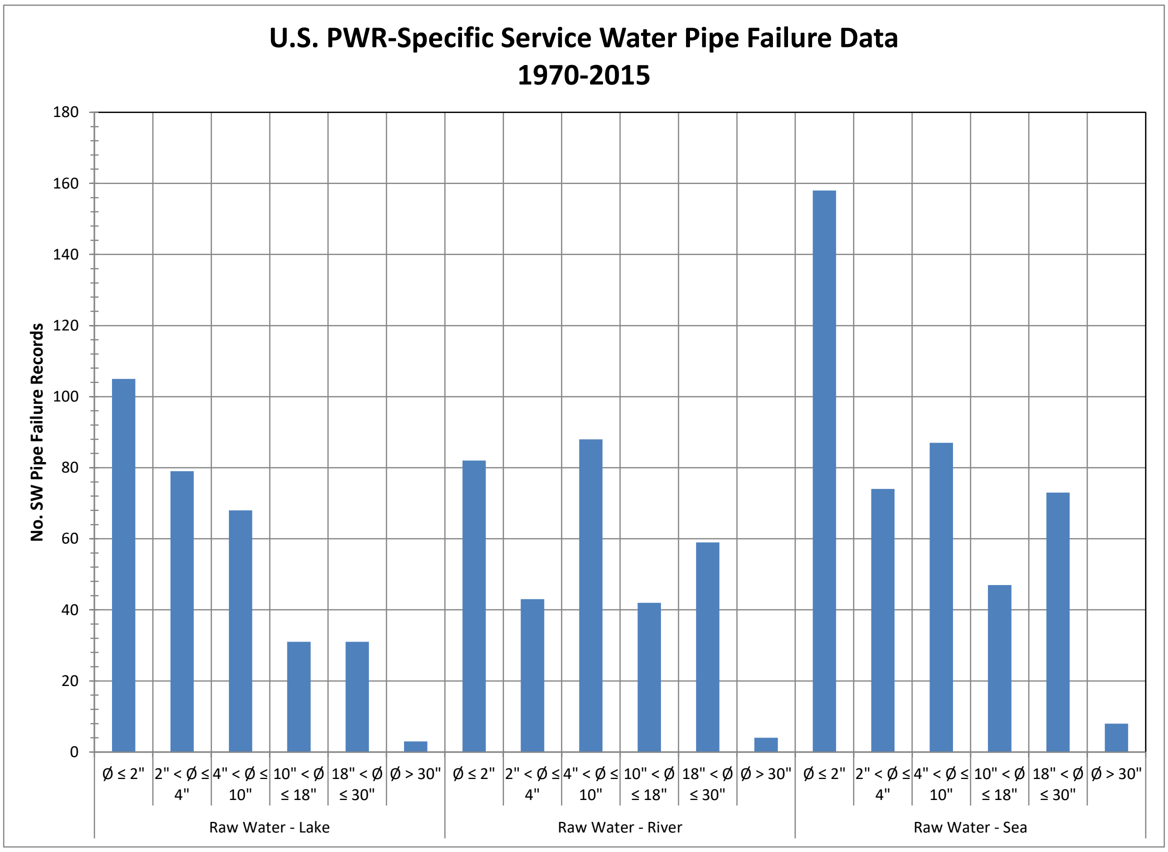


Figure 5-6: U.S. Service Water Piping Operating Experience

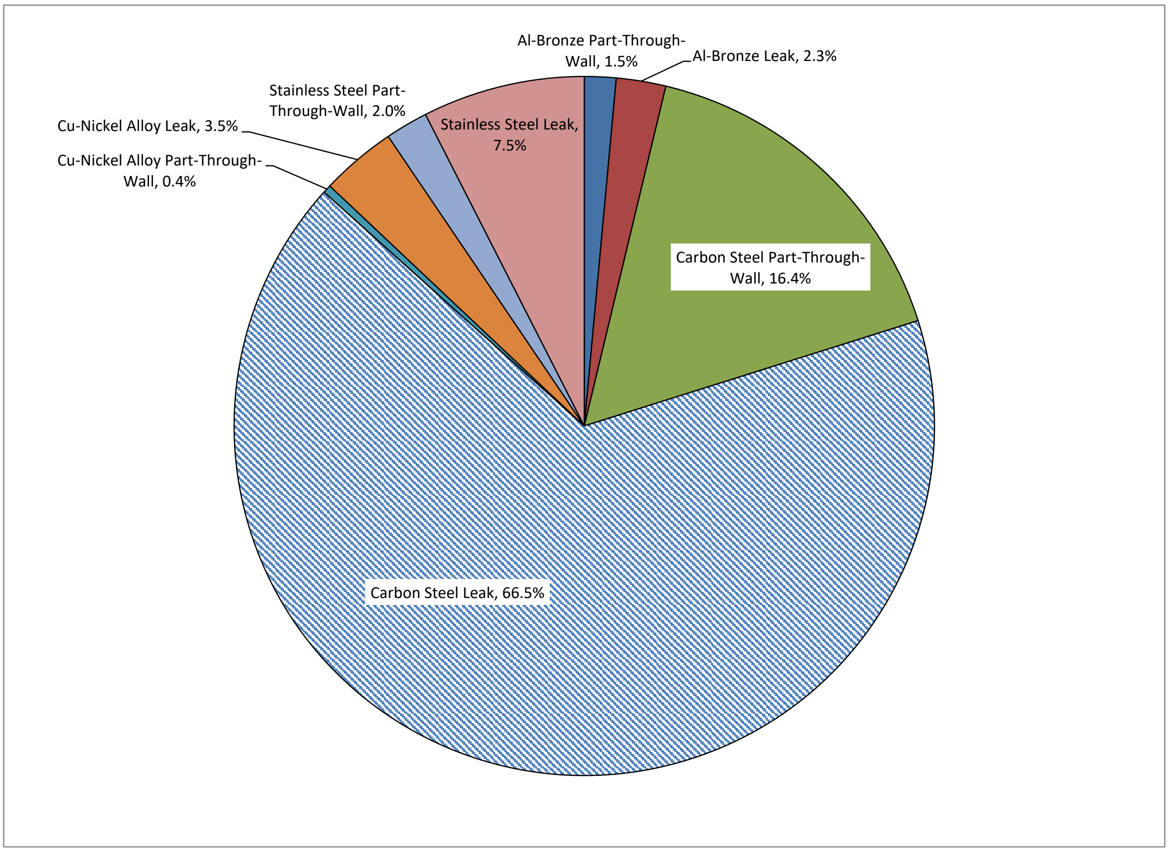


Figure 5-7: Service Water Piping Operating Experience by Material Type

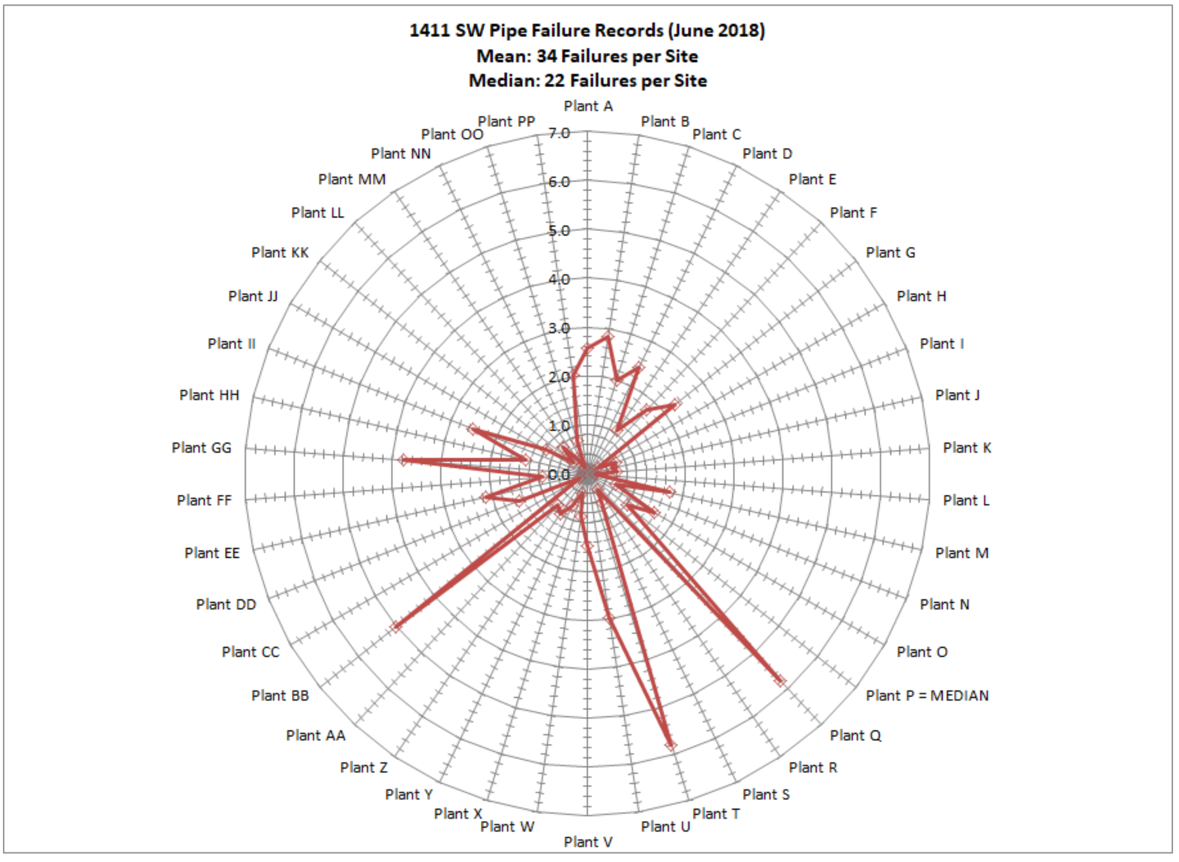


Figure 5-8: U.S. PWR Service Water Piping OE Site-to-Site Variability

5.4.3 OE: Case Study 1 – MSLB Frequency Assessment

In preparing for the assessment of the relevant operating experience data, MS piping system design information was assembled in order to define an appropriate set of database query definitions; Table 5-8. As discussed in Section 2, Case Study 1 consists of three different piping reliability assessments with a common denominator, namely, PWR MS piping. In Table 5-6 the piping design information is organized by ASME III Code Class; i.e. piping inside and outside containment (non-Code). Depending on the NSSS design generation and Architect & Engineer responsible for plant construction the MS safety relief valve header piping is located in the Intermediate Building or in open air.

The MS piping failure frequency estimates were generated using a proprietary SQL-database (PIPExp), which was originated in 1993[47]. It is a continuously maintained and updated piping operating experience database. Its technical basis, including the interpretation and classification of the field experience data, is documented in a Coding Guideline. Additional information on how PIPExp relates to other piping failure database development efforts are documented in Reference [79]. Each data record is subject to initial screening for "eligibility" followed by a formalized coding (or event classification). Each record in the database is classified using over 700 data filters/key-words and ca. 100 database fields.

[47] As of Fenruary 2019, the database contains 11,072 pipe failure records, a water hammer (WH) event database (725 records), and a piping component population database organized by system for several BWR and PWR plants.

Table 5-6: Typical PWR Main Steam Piping System Designs

PCS Layout / MS Piping Design Feature	B&W (Plant-Specific Example – Case Study 1.1))	Westinghouse 412 (WE-4 Loop, Typical)	Westinghouse (WE-3 Loop, Plant-Specific Example – Case Study 1.2) [2]
MS Inside Containment / Intermediate Building			
No. MS Headers	2 Headers per S/G combining into single header before entering into RB. Each MS header has 8 MSRVs off the 36” header pipe- these MSRVs are located in the Yard	1 Header per S/G - the 4 headers combine into 2 headers in the Steam Tunnel. Each steam line from respective S/G contains 1 PORV and 5 SRVs and 1 MSIV.	1 Header per S/G - the 3 headers combine into a single manifold in Steam Tunnel. Each steam line from respective S/G contains 1 PORV and 5 SRVs and 1 MSIV.
Header Diameter [inch]	24 & 36"	30 – 32 [1] (OD)	32
Wall Thickness			Schedule 80
Design Pressure [psig]	1093	1185	1192
Design Temperature [°F]		600	600
Operating Pressure [psig]	910	910 to 980	993
Operating Temperature [°F]		544 to 557	
Steam Quality; defined as: (Mass of Vapor) / [(Mass of Vapor) + (Vapor Mixture)]	Superheated – the OTSG design has three heat transfer regions. The saturated steam entering the superheat region is heated to a minimum of 35°F above saturation temperature.	Steam quality ≥ 99.5%	Steam quality ≥ 99.5% (0.25% moisture); centrifugal moisture separators above S/G tube bundle
MS Inside Turbine Building			
No. T/G Sets	2 - 50% T/G sets	Single	2 - 50% T/G sets
No. Main Headers	Two 34” (inside diameter) MS headers in Turbine Building. This unit has two TD-FWPs (8” steam supply lines) and two TD-EFWPs (6” steam supply lines)	From MS crosstie header (manifold) connects two 41.25” headers in close proximity to the HP Turbine. A 6-inch branch line supplies steam to a turbine-driven Main Feedwater pump.	From MS manifold, 2 headers for each of the two T/G sets. The two headers split into 4 headers that connect to HP turbine inlet valves. This unit has 1 steam-driven AF pump and 2 motor-driven AF pumps. Steam supply to the single steam driven AF pump is via 4" steam supply line.
Main Steam Header Diameter [inch]	34 (ID)	28 (NPS) & 41.25 (OD)	26 & 16 (NPS)
Turbine Bypass	Three Bypass lines to Condenser 1A, 1B &1C		Two Bypass lines to respective condenser
Turbine Bypass Diameter	8, 18, 20 & 24		20 & 14
Material	CS	CS	All MS piping in Turbine Building is Low-Alloy Steel (LAS, 15Mo3)

PCS Layout / MS Piping Design Feature	B&W (Plant-Specific Example – Case Study 1.1))	Westinghouse 412 (WE-4 Loop, Typical)	Westinghouse (WE-3 Loop, Plant-Specific Example – Case Study 1.2) [2]
Turbine Bypass Design Pressure [psig]			1192
MS Bypass Design Temperature [°F]			599
In-Service Inspection	FAC Program inspections performed on HP Turbine exhaust lines located under the turbine – these lines are not part of the MSLB evaluation boundary	See Case Study 1.1.	The piping segments corresponding to the Case Study 1.1 MSLB evaluation boundary are included in the RI-ISI program for this reactor unit
AFW Pump Steam Supply	Two (2) 6” steam supply lines		Two (2) 4” steam supply lines to the single turbine-driven AFW pump. The pump steam supply lines are branched off respective 24” SRV header
FW Pump Steam Supply	Two (2) 8” steam supply lines branched off respective MS Header 1A and 1B	Two (2) turbine-driven FW pumps. During plant startup, steam supply is from the ‘A’ and ‘C’ main steam header. During normal operation, supply is from low-pressure steam from Moisture Separator ‘A’	N/A
Notes	1. 30.25” for A & D S/G:s and 33.75” for B & C S/G:s. Difference in line size compensates for difference in S/G locations. Over 2000 linear feet of MS Header piping – from S/G to HP Turbine. 2. Piping system designed by Mannesmann-Rohrbau AG		

Summarized in Tables 5-7 and 5-8 and Figure 5-9 through Figure 5-11 is the service experience data applicable to the Main Steam piping in commercial PWR plants for the period 1970 through 2015.

Table 5-7: MS Piping Service Experience (1970-2015)

Damage / Degradation Mechanism	Pipe Diameter [inch]	Failure Mode			
		All	NTWC	Leak	MSF
Inside Containment					
Corrosion	Ø ≤ 2	5	1	4	0
	6 < Ø ≤ 10	1	0	1	0
FAC	Ø ≤ 2	6	4	2	0
	2 < Ø ≤ 4	3	1	2	0
	4 < Ø ≤ 6	4	4	0	0
	Ø > 10	3	3	0	0
D&C	Ø ≤ 2	1	1	0	0
	4 < Ø ≤ 6	1	1	0	0
	Ø > 10	14	14	0	0
HC-FAT / Fretting	Ø ≤ 2	15	3	12	0
LC-FAT	Ø ≤ 2	8	0	8	0
	2 < Ø ≤ 4	1	0	1	0
	6 < Ø ≤ 10	1	1	0	0
SH/WH	Ø > 10	1	1	0	0
	Code Class 2 Total:	64	34	30	0
Inside Turbine Building / Intermediate Building					
Corrosion	Ø ≤ 2	2	0	0	2
FAC	Ø ≤ 2	55	33	21	1
	2 < Ø ≤ 4	9	2	6	1
	4 < Ø ≤ 6	3	1	2	0
	6 < Ø ≤ 10	2	1	1	0
D&C	Ø > 10	3	3	0	0
HC-FAT / Fretting	Ø ≤ 2	35	2	31	2
	2 < Ø ≤ 4	6	0	5	1
LDIE – LAS & SS material	Ø ≤ 2	2	0	1	1
SH/WH	Ø ≤ 2	1	0	1	0
	4 < Ø ≤ 6	1	0	0	1
	Ø > 10	4	2	0	2
	Total:	123	44	68	11

Table 5-8: MS Piping Service Experience Applicable to Case Study 1.1

Damage / Degradation Mechanism	Pipe Diameter [inch]	Failure Mode			
		All	NTWC	Leak	MSF
Inside Turbine Building					
Corrosion	Ø ≤ 2	2	0	0	2
FAC	Ø ≤ 2	55	33	21	1
	2 < Ø ≤ 4	9	2	6	1
	4 < Ø ≤ 6	3	1	2	0
	6 < Ø ≤ 10	2	1	1	0
D&C (1)	Ø > 10	2	2	0	0
HC-FAT / Fretting	Ø ≤ 2	35	2	31	2
	2 < Ø ≤ 4	6	0	5	1
	Total:	114	41	66	7

(1) Plant A (foreign, September 1998) and Plant B (foreign, March 2001); during "opportunistic inspections" rejectable flaws in base metal near circumferential welds on 24-inch MS header to HP Turbine. The rejectable flaws were subjected to detailed metallographic evaluations, which attributed the flaws to fabrication defects with some evidence of flaw growth. Affected piping sections (about 15 linear feet) were subsequently replaced.

- According to the available service experience data, water/steam hammer events have been found to occur in three areas: 1) Adjacent to the main steam isolation valves, 2) piping downstream of the turbine bypass valve, and 3) in the main steam relief valve piping.
- The Main Steam piping carries dry, saturated (or super-heated) steam from the steam generators to the High Pressure Turbine. FAC is not a credible degradation mechanism in carbon steel MS header piping primarily because of the high temperature and very low-to-no moisture content. The MS locations for which FAC have been recorded are associated with moisture traps and drain lines off MS header piping and Turbine Bypass piping. These small-bore piping segments were excluded from respective MSLB case study.
- Figure 5-11 summarizes the service experience with FAC-induced MS piping failures. It is noteworthy that this operating experience is limited to small- and medium-bore piping connected to either the MS header piping or the Turbine Bypass piping. Therefore, this operating experience does not apply to the Case Study 1.1 MSLB evaluation boundary.
- Figure 5-12 summarizes the global service experience involving degraded of failed large-diameter (≥ 12-inch) piping due to weld defects (D&C) or low-cycle fatigue (LC-FAT) due to normal plant cooldown/heat-cycles and conditional on a pre-existing weld flaw. This service experience is displayed as the fraction of all large-diameter piping failures that is attributed to D&C, and the fraction of all large-diameter piping failures that is attributed to LC-FAT conditional on a pre-existing given D&C flaw.

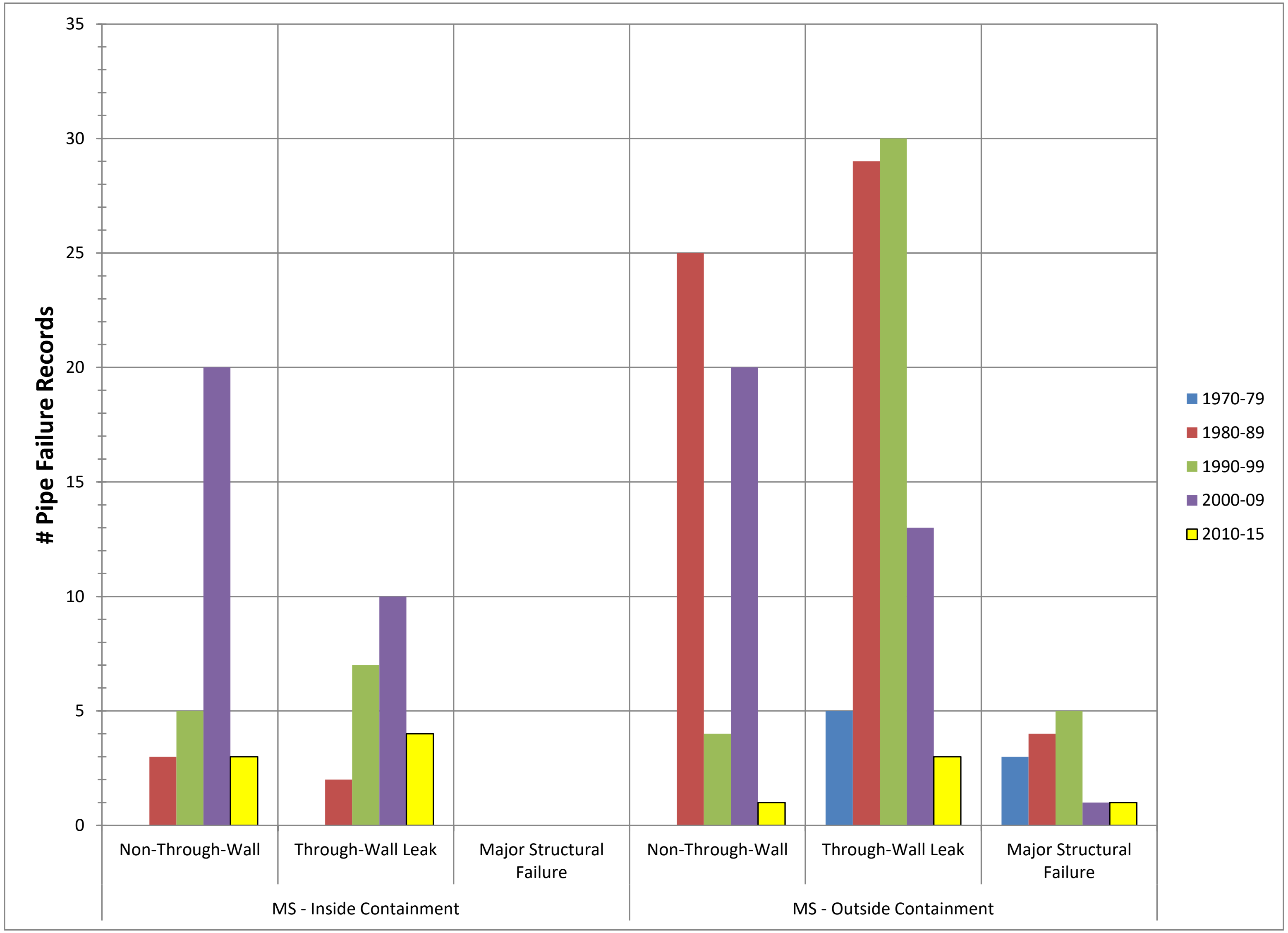


Figure 5-9: PWR Main Steam Pipe Failure Experience by Time-Period & In-Plant Location

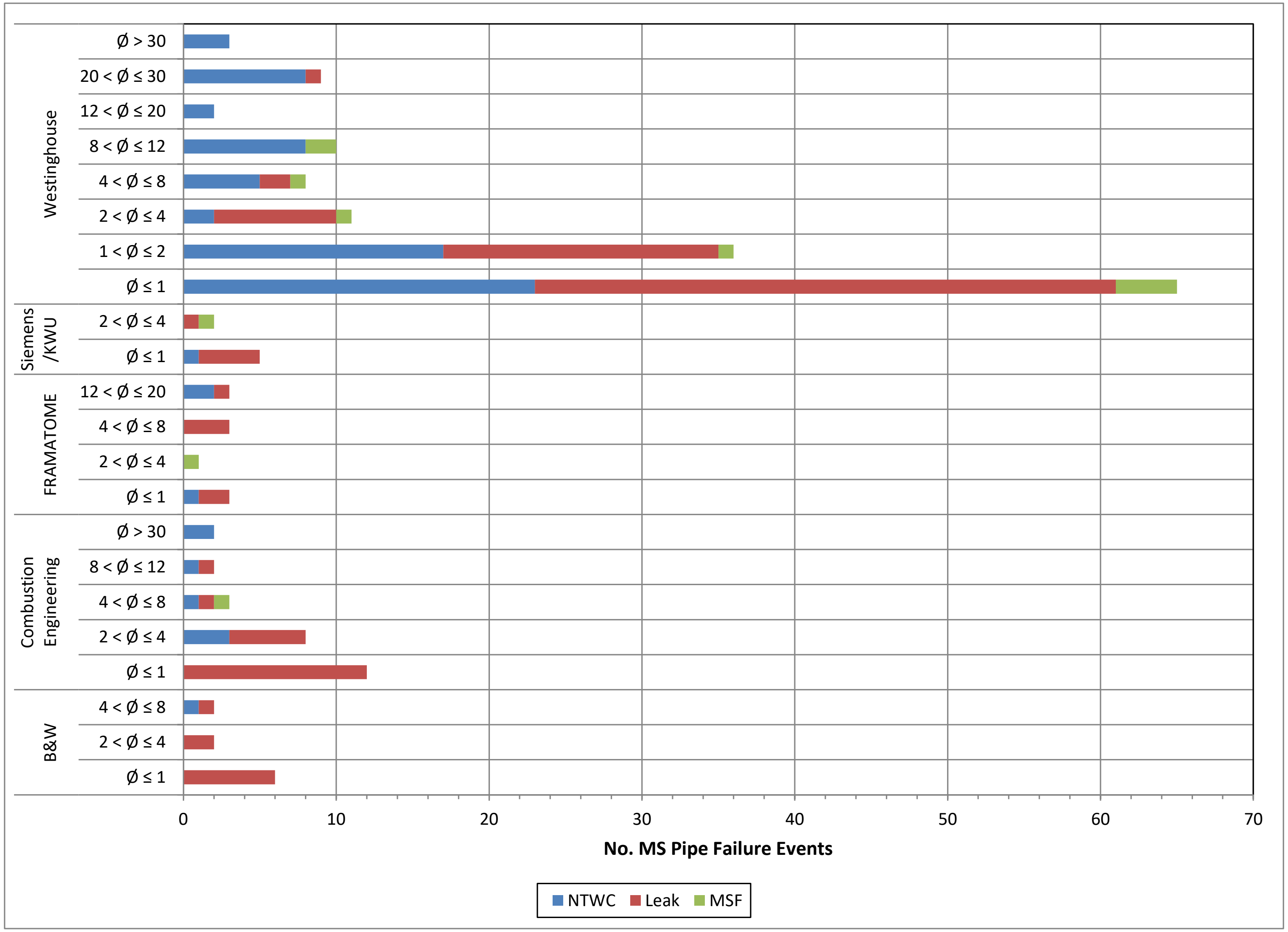


Figure 5-10: PWR Main Steam Pipe Failure Experience by NSSS Vendor, Pipe Size & Mode of Failure

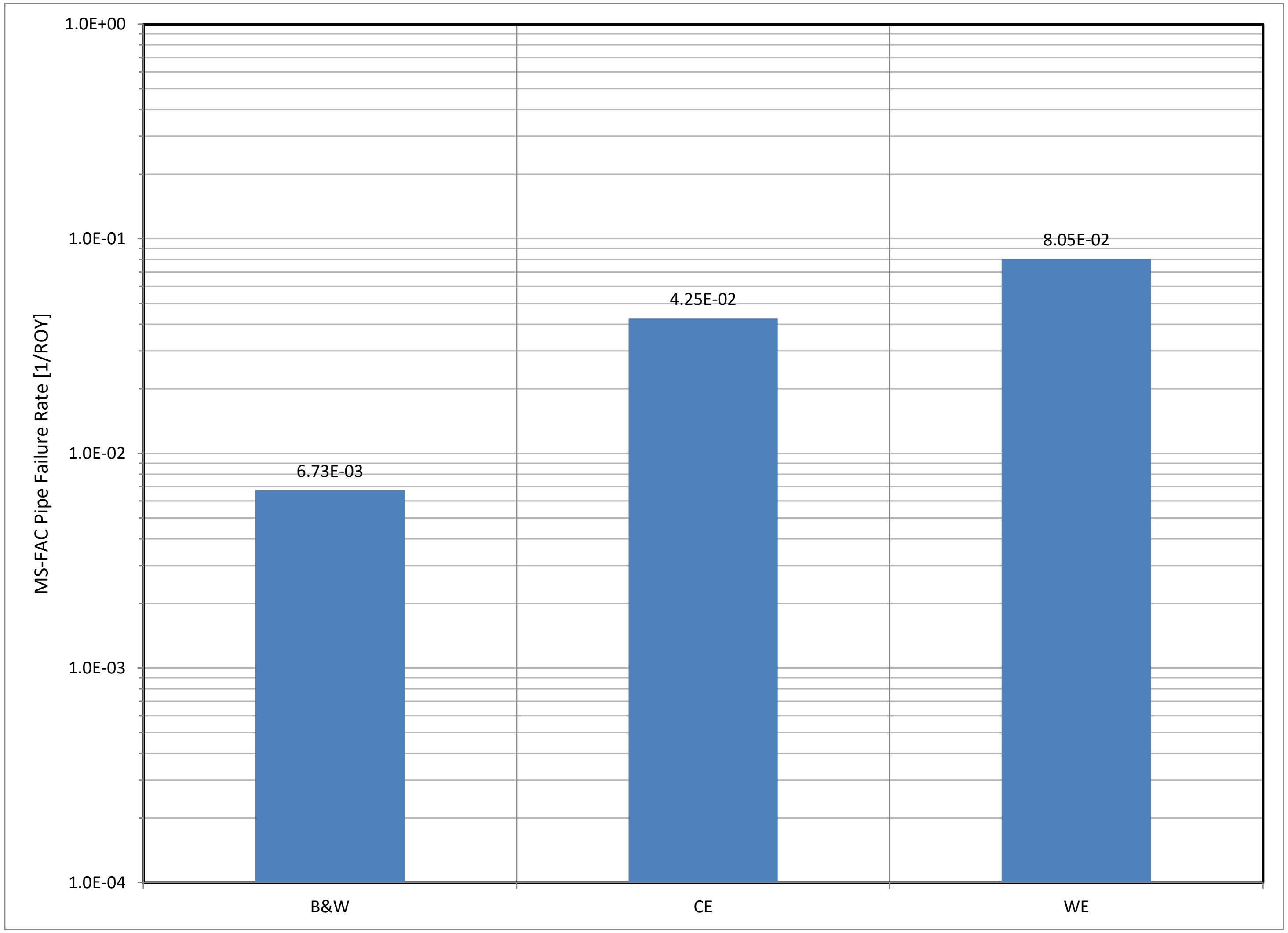


Figure 5-11: Rate of MS Piping Failure[48] due to FAC [1/System.ROY]

[48] Moisture Trap lines and drain lines off MS Header piping and Turbine Bypass piping – all < 10" diameter.

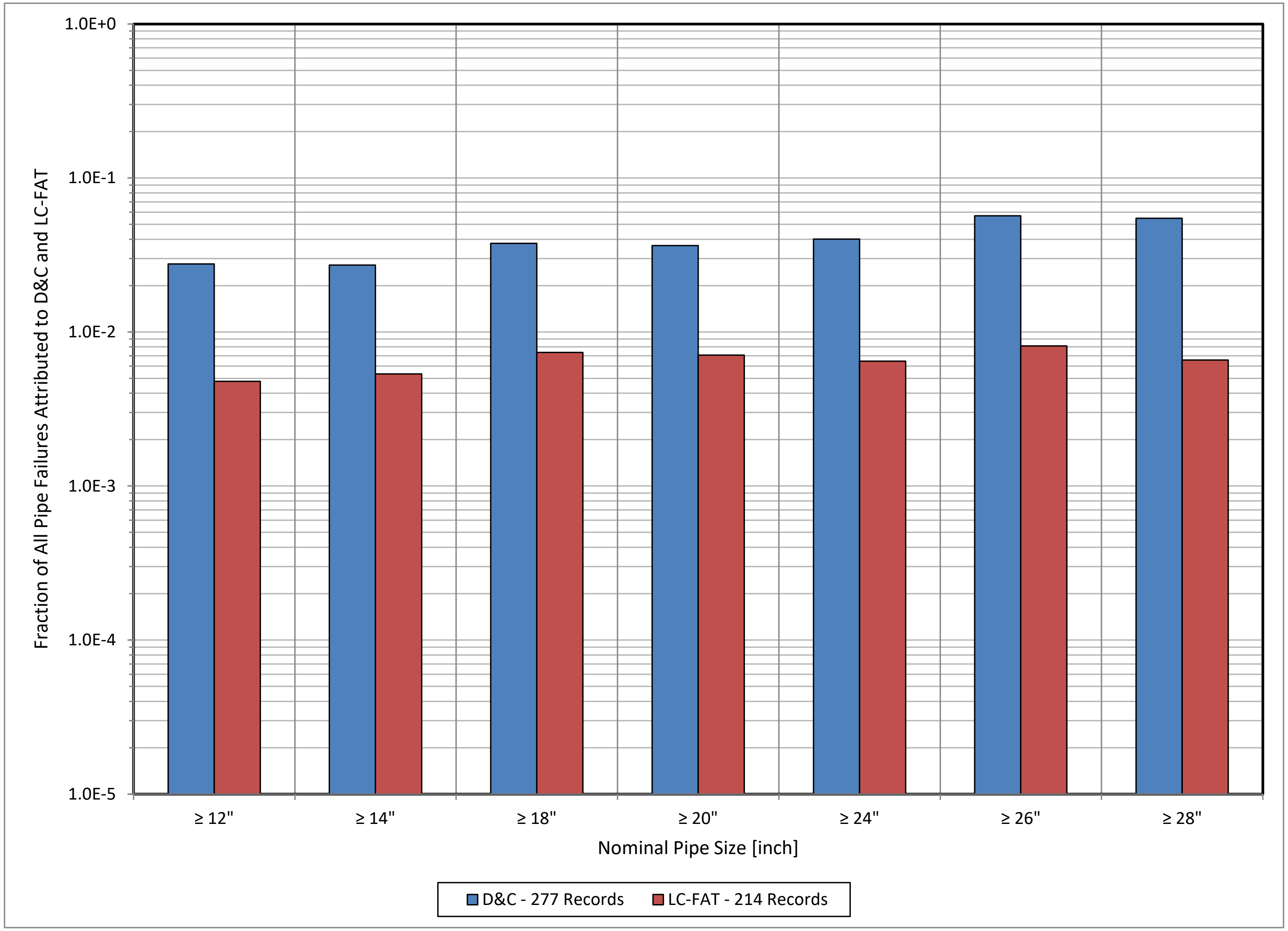


Figure 5-12: Fraction of All Pipe Failures Attributed to D&C and LC-FAT

One insight from the MSLB case studies is the importance of querying a database using good piping system design knowledge and on the basis of a succinct definition of the evaluation boundary. The database query results should not be limited to generate pipe failure event populations. Additional data processing should be performed to highlight various perceptions or assumptions about the potential conditional pipe failure propensities. Such data processing helps direct and support the "down-stream" analysis tasks.

For reference, below are descriptions of the three most significant PWR Main Steam pipe failures to date. The tree events involve Main Steam PORV/SRV branch connection failures attributed to significant overpressure/overload in excess of ultimate strength of the carbon steel piping material. Neither of these events is of relevance to the case study MSLB case study evaluation boundaries, however.

Event #1. On April 28, 1970, during hot hydraulic testing of the unfueled H.B. Robinson-2 reactor, a pipe nozzle, counter-bored and tapered from Sch. 80 to Sch. 40, between a pressurized main steam line and a safety valve failed completely; a 360-degree circumferential break. The rupture was a non-isolable break, and an uncontrolled cooldown of the plant occurred. Seven men were injured by the escaping steam. The fracture path coincided with the end of the machined taper adjacent to the weld inside the reduced section of the 6-inch pipe nozzle which connected the safety valve to the 26-inch main steam line. Plant recovery was accomplished in an orderly manner. The secondary system pressure was about 900 psi with the MSIV and bypass valves closed. The RCS temperature decreased in about minutes from 540 to 319 F. In a little over 9 minutes, the pressurizer pressure dropped from 2225 to 1860 psi, and the steam generator boiled dry. The section of pipe which failed was insulated to a point approximately 3" above the failure which minimized the temperature gradient across the failed area. A pneumatic test device was connected to the valve which loads the spring so as to balance the seating pressure on the valve disc. The operator was opening the valve regulator to balance the spring force when the failure occurred. The failure pattern, surrounding damage, and trajectories of the valve and exhaust chute indicate that the pipe first opened up on the West side, directly opposite the exhaust, emitting steam in a fan jet which sharply cut the insulation on an adjacent pipe. The pattern of the insulation cut and fracture face suggest that the jet had a fan angle of about 80-degrees wide and was directed about 45-degrees up from the horizontal. The safety valve apparently rotated as the pipe tore across its section. The exhaust chute broke off and was found under the Loop #3 steam pipe. An inspection of the exhaust elbow revealed longitudinal markings on the inner surface of the extrados. These may have been caused either by construction, by handling, or by debris being swept through with sufficient force to bare the metal.

Analyses of the failed piping showed extreme plastic strain on the fracture surface, which indicated an overload failure, but no plausible mechanism for the overloading was found. Stress analyses indicated that the branch line was undersized and that the stress calculated for a full-capacity discharge through the valve could exceed the ultimate strength of the material. Modifications were performed to all the safety valve branch lines to increase their loading capacity.

Event #2. On December 2, 1971, during hot functional testing of the unfueled Turkey Point-3 reactor, three of four safety valves were blown off the header on one of three steam loops, and the north segment of the header was split open. The main steam pipe header failed in the base metal just below and outside the nozzle-to-pipe weld. The dynamic loading resulting from actuation of the safety relief valve, combined with the condensate in the line, exerted a bending moment and torsional stress on the header at the location of the valve attachment. These overstressing forces were not considered in the design. Sixteen persons received treatment for injuries, but only two were injured seriously enough to be hospitalized overnight. The steam line header was redesigned and replaced by heavier 14 x 14 x 8 inch

Schedule 160 forged tees. Prior to the hot functional testing, the system had been hydrostatically tested at 9.3 MPa at cold conditions.

Fracture examination of the failed portion of the main steam line revealed that the failure was caused by impact loading and the origin of each valve fracture was at the weld connecting the valve to the header pipe. Pipe stress analysis indicated that opening the safety valves at design pressure would produce a reaction force exceeding the design limits of the steam line header assembly and result in the fracture of the branch connections.

It was estimated that 44 ft^3 of water could have condensed in the dead leg of steam line ‘A’ prior to the incident. The flow of this amount of water could have increased the pressure acting on the safety valves. Once opened, the reaction forces resulting from the discharge of steam and probably some water would produce a stress at the safety valve branch connection to the header pipe capable of fracturing the assembly.

Event #3. On April 16, 2005, a PWR plant was in the early stages of returning to power following completion of the 23rd refueling and maintenance outage when manual operation of the MS-PORV was performed at a low speed to reduce the pressure of the MS steam line with an MSIV (Main Steam Isolation Valve) and 5 MSSVs (Main Steam Safety valves), which remained closed. When the pressure in the MS line reached about 7.12 MPa 1033 psi) and a reactor operator switched the PORV operation mode from 'Manual' to 'Auto' to speed up the pressure reduction process. Immediately after that the PORV opened fully in a short time due to higher pressure beyond the PORV setpoint, which resulted in excessively rapid reduction of pressure in the main steam line system. Later on, the operator closed the valve manually to mitigate the unexpected transient response of system.

Due to the abrupt pressure reduction of steam generator and main steam line caused by the sudden opening of PORV, the reactor shut down by the low pressure signal of safety injection and some main steam separation valves seemed to be actuated. As the result of the rapid release of high pressure steam to the atmosphere through the PORV, the supports for the curved pipe spool of about 3.3 m long and 300 kg weight, which had been connecting the straight pipe line in the downstream of the PORV and a silencer for reducing noise generated by the steam discharging to the atmosphere, were broken away. At the same time, the pipe spool was separated and projected from the line. Finally it ejected and struck the outer wall of the RWST located about 50 m away, resulting in structural damage with the maximum permanent deformation of about 60 mm in the radial direction at the local part of wall. Upon investigation, (1) the safety injection was due to an improper mode transfer of the PORV controller, (2) reactor trip was due to the SI, (3) pipe detachment was due to the defect in pipe design in which the maximum dynamic load to that pipe was not properly reflected in the stages of design and construction. During the transient including SI and reactor trip, the key safety functions were maintained via prompt operator actions and the plant entered hot standby condition.

5.4.4 OE: Case Study 2 – LOCA Frequency Assessment

This case study used the term "pipe failure" to include any condition that leads to repair or replacement of the affected piping component. The most severe type of pipe failure found in the data query was a leak with leak flow rate less than 10 gpm. Insights from review of service experience clearly show that for failures in Code Class 1 piping systems, failures occur almost exclusively at or near welds. In preparing for the assessment of the relevant operating experience data, RCPB piping system design information was assembled in order to define an appropriate set of database query definitions; Table 5-9 through Table 5-11. Summarized in Table 5-12 is the service experience data applicable to the LOCA frequency calculations.

Table 5-9: RCPB Weld Populations for Selected PWR Plants

Description	Pipe Size [inch]	ASME XI Category	Weld Count					
			NPP1	**NPP2**	**NPP3**	**MPP4**	**NPP5**	**NPP6**
RC Hot Leg	42	B-F	N/A	N/A	N/A	N/A	N/A	N/A
		B-J	8	8	10	10	10	10
		Longitudinal	8	8	8	8	8	8
RC Hot Leg Drain - Unmitigated	2	B-F	1	1	0	0	0	0
RC Hot Leg Drain - Mitigated		B-F	1	1	0	0	0	0
RC Hot Leg Drain		B-J	9	11	2	2	2	2
RC Cold Leg - Unmitigated	30	B-F	8	8	8	8	8	8
RC Cold Leg - Mitigated	30	B-F	8	8	8	8	8	8
RC Cold Leg	30	B-J	46	44	44	48	48	48
		Longitudinal	54	56	56	32	32	32
RC Cold Leg Drain - Unmitigated	2	B-F	3	3	3	3	3	3
RC Cold Leg Drain - Mitigated		B-F	3	3	3	3	3	3
RC Cold Leg Drain		B-J	21	14	24	24	24	24
RC Hot Leg / SG Inlet	42	B-F	N/A	N/A	N/A	N/A	N/A	N/A
Pressurizer Surge Line - Unmitigated	12	B-F	1	1	1	1	1	1
Pressurizer Surge Line - Mitigated		B-F	1	1	1	1	1	1
Pressurizer Surge Line		B-J	11	11	9	11	11	11
Pressurizer Surge Line RC-HL Branch Connection		B-F	1	1	1	1	1	1
Pzr PRV/SRV & Spray Lines - Unmitigated	3, 4.5	B-F	6	6	6	6	6	6
Pzr PRV/SRV & Spray Lines - Mitigated	3, 4.5	B-F						
Pzr PRV/SRV & Spray Lines	2, 3, 4, 4.5	B-J	119	127	134	116	116	116
Class 1 Small-Bore Piping	1	B-J	Deferred					
RHR - Unmitigated	12, 14, 16	B-F	1	1	1	2	2	2
RHR - Mitigated	12, 14, 16	B-F	1	1	1	2	2	2
RHR	12, 14, 16	B-J	37	33	56	55	42	42
SIT to Cold Leg - Unmitigated	12, 14	B-F	4	4	4	4	4	4
SIT to Cold Leg - Mitigated	12, 14	B-F	4	4	4	4	4	4
SIT to Cold Leg	12, 14	B-J	50	46	68	79	79	79
Medium Bore SIR Piping	3, 6	B-J	23	18	23	52	52	52

Description	Pipe Size [inch]	ASME XI Category	Weld Count					
			NPP1	NPP2	NPP3	MPP4	NPP5	NPP6
CV (Charging & Letdown) - Unmitigated	2, 3	B-F	3	3	4	2	2	2
CV (Charging & Letdown) - Mitigated		B-F	3	3	4	2	2	2
CV (Charging & Letdown)		B-J	74	78	115	131	130	131

Table 5-10: RCPB Weld Population for a Typical 4-Loop Westinghouse PWR Plant

System	Component Case	Weld Type	Applicable Damage/ Degradation Mechanism(s)	Plant WE-1 Weld Count	Pipe Size [inch]	DEGB Size [inch]	Comment
RC Hot Leg	1A1	B-F	PWSCC, D&C	4	29	41.0	RPV Outlet
	1A2	B-F (M)	D&C	4	29	41.0	RPV Outlet
	1B	B-J	D&C	8	29	41.0	
	1C	B-J-L	--	N/A	N/A	N/A	
	1D	B-F	--	N/A	N/A	N/A	
	1E, 1F	B-J	--	N/A	N/A	N/A	
RC Hot Leg - S/G Inlet	2A1	B-F	PWSCC, D&C	4	29	41.0	This case addresses the North Anna Unit 1 March 2012 NDE Results
	2A2	B-F	D&C	4	29	41.0	
RC Cold Leg & Intermediate Leg	3A1	B-F	PWSCC, D&C	4	31	43.8	RPV Inlet
	3A2	B-F (M)	D&C	4	31	43.8	RPV Inlet
	3B	B-J	D&C	20	31	43.8	
	3C1	B-F (M)	PWSCC, D&C	4	27.5	38.9	S/G Outlet
	3C2	B-F	D&C	4	27.5	38.9	S/G Outlet
	3D	B-J	D&C	12	27.5	38.9	
	3E	B-J	D&C	8	3	4.2	Capped lines off Intermediate Legs
	3F	B-J	TASCS, D&C	8	2	2.8	RC Loop Drain
	3G	B-J	D&C	31	2	2.8	
PZR Surge Line	4A1	B-F	PWSCC, D&C	1	14	19.8	
	4A2	B-F (M)	D&C	1	14	19.8	
	4B	B-J	TASCS, D&C	4	16	22.6	
	4C	B-J	TASCS, D&C	1	14	19.8	
PZR Spray &	5A1	B-F	PWSCC, D&C	4	6	8.5	

System	Component Case	Weld Type	Applicable Damage/ Degradation Mechanism(s)	Plant WE-1 Weld Count	Pipe Size [inch]	DEGB Size [inch]	Comment
PORV/SRV Lines	5A2	B-F (M)	D&C	4	6	8.5	
	5B1	B-F	PWSCC, D&C	1	4	5.7	
	5B2	B-F (M)	D&C	1	4	5.7	
	5C	B-J	TT, D&C	9	6	8.5	
	5D	B-J	D&C	34	6	8.5	
	5E	B-J	TASCS, D&C	6	4	5.7	
	5F	B-J	D&C	52	4	5.7	
	5G	B-J	D&C	16	3	4.2	
	5H	B-J	D&C	2	2	2.8	
SDC & SIR Lines	7A	B-J	D&C	41	12	17.0	RHR Suction Lines
	7B	B-J	IGSCC, D&C	8	10	14.1	Accumulator Injection Lines
	7C	B-J	D&C	72	10	14.1	
	7D	B-J	D&C	19	8	11.3	
	7E	B-J	IGSCC, D&C	2	6	8.5	
	7F	B-J	D&C	115	6	8.5	
	7G	B-J	TT, D&C	4	3	4.2	
	7H	B-J	D&C	38	3	4.2	
	7I	B-J	D&C	64	2	2.8	
	7J	B-J	TASCS, D&C	13	1.5	2.1	
	7K	B-J	TT, D&C	3	1.5	2.1	
	7L	B-J	TT, IGSCC and D&C	4	1.5	2.1	
	7M	B-J	D&C	186	1.5	2.1	
CVC	8A	B-J	TT, D&C	27	3	4.2	
	8B	B-J	D&C	3	3	4.2	
	8C	B-J	TT, D&C	1	2	2.8	
	8D	B-J	D&C	6	2	2.8	
	8E	B-J	D&C	65	1.5	2.1	

Table 5-11: Selected BWR & PWR RCPB Small-Bore Weld Populations

ASME III Class 1 Small-Bore Piping Systems - ≤ NPS4					
Plant	**Weld Population**			**Comment**	**Information Source**
	Socket Welds	**Butt Welds**	**Total**		
Callaway	77			1" ≤ Ø < 4"	ULNRC-05981, April 15, 2013
Duane Arnold	118				NG-10-0091, March 9, 2010
Hope Creek	250	51	301		LR-N10-0415, December 15, 2010
Indian Point-2	433	195	628		NL-11-032: Indian Point 2 and 3 LRA-RAIs
Indian Point-3	333	96	429		NL-11-032: Indian Point 2 and 3 LRA-RAIs
Kewaunee	345	Unknown			Serial No. 10-665: Kewaunee LRA-RAIs
LaSalle-1	483	108	591		RS-15-193, August 6, 2015
LaSalle-2	458	94	552		RS-15-193, August 6, 2015
St. Lucie-1	493	92	585	Ø ≤ 2" (socket welds)	L-2014-265, Attachment, September 3, 2014
St. Lucie-2	440	137	577	Ø ≤ 2" (socket welds)	L-2016-063, Attachment, March 29, 2016
Prairie Island-1	50			DM-sensitive socket welds (> 1")	L-P-10-109: PINGS LRA-RAIs
Prairie Island-2	84				L-P-10-109: PINGS LRA-RAIs
River Bend	64	381	445		RBG-47817, 2/6/2018
Seabrook	150	300	450		SBK-L-11002: Seabrook LRA-RAIs
STP-1	49	182	231		NOC-AE-11002731, 9/15/2011
STP-2	59	190	249		
Waterford-3	216	371	587		WF3_LR-SER, 8/17/2018

Table 5-12: Database Query Results for RCPB Pipe Failure Events

System	Westinghouse PWRs			Combustion Engineering PWRs		
	Nominal Pipe Size (NPS)[3]	**Failures**	**Degradation Mechanism[1]**	**Nominal Pipe Size (NPS)[3]**	**Failures**	**Degradation Mechanism[1]**
RCS Hot Leg B-F at RPV Nozzle	29"	6	PWSCC	Not Applicable (2)		
RCS Hot Leg B-F at SG Nozzle	29"	19	PWSCC			
		1	D&C			
RCS Hot Leg B-J	29"	0	N/A	42"	1	D&C
RCS Cold Leg B-F at RPV and SG Nozzles	31"	4	PWSCC	30"	0	N/A
RCS Cold Leg B-J	31"	0	N/A	30"	1	BA-COR
PZR Surge B-F at PZR Nozzle	14"	3	PWSCC	12"	1	PWSCC
PZR-Surge B-J	14"	0	N/A	12"	0	N/A
Total Large Pipe Failures		33			3	
Total All Pipe Failures		209			128	
Plant Exposure (reactor-years)		4,081			593	
Point Estimate Large Pipe Failure Rate, per reactor-year		8.1E-03			5.1E-03	
Point Estimate Total Pipe Failure Rate, per reactor-year		5.1E-02			2.2E-01	

[1] PWSCC = primary water stress corrosion cracking
D&C = design and construction defects
BA-COR = external boric acid corrosion
[2] CE PWRs do not use B-F welds in large hot leg pipes
[3] For hot legs and cold legs the NPS is also the inner diameter of the pipe; for surge line pipes the ID is somewhat smaller than NPS

The database queries were limited to non-isolable ASME III Class 1 piping system pressure boundary failures. The PWR Class 1 boundary consists of the RCS Hot leg and Cold Leg, pressurizer surge, spray, auxiliary spray, relief valve, safety valve and vent lines, drain lines. It also includes branch piping to the Safety Injection System (SIS), Chemical & Volume Control System (CVCS), and Residual Heat Removal System (RHRS). All piping attached to the RCS loops or pressurizer vessel is considered Class 1 out to the second isolation valve. Class 1 SIS, CVCS and RHRS piping between the first and second valve off the RCS were not included in the scope of Case Study 2.

5.4.5 OE: Case Study 3- Failure Rate of Buried ESW Piping

In this case study a simple database query was performed to obtain the number of buries ESW failures as a function of the observed through-wall flow rate. The event database filters for observed leakage are listed in Table 5-13. The query results are summarized in Table 5-14.

Table 5-13: Database Filters for Through-Wall Leaks

Leak Class	
Class	**Definition**
1	<< 1 gpm
2	Minor < LR ≤ 1 gpm
3	1 < LR ≤ 2 gpm
4	2 < LR ≤ 5 gpm
5	5 < LR ≤ 10 gpm
6	10 < LR ≤ 50 gpm
7	50 < LR ≤ 100 gpm
8	100 < LR ≤ 500 gpm
9	500 < LR ≤ 1000 gpm
10	1000 < LR ≤ 5000 gpm
11	5000 < LR ≤ 10000 gpm
12	10000 < LR ≤ 20000 gpm
13	20000 < LR ≤ 50000 gpm
14	50000 < LR ≤ 100000 gpm
15	> 100000 gpm

Table 5-14: Buried ESW Piping Operating Experience Data

Event Classification	Equivalent Break Size Class	No. Observed Buried ESW Piping Failures
Small Leal	≥ 1 (perceptible leakage < or << 0.06 kg/s	29
Leal	≥ 6 (equal to or greater than 0.06 kg/s but less than 3 kg/s)	10
Large Leak	≥ 6 (equal to or greater than 60 kg/s but less than 300 kg/s)	4
Massive Leak / Break	≥ 15 (massive leakage greater than 6,000 kg/s)	1

5.4.6 Step 3 Check List

The practical analysis insights from the three case studies have been organized in a series of checklists to support systematic reviews of licensee submittals of fitness-for-service evaluations, and to assist the OEAD staff in validating licensee submittals by performing independent fitness-for-service evaluations. Table 5-15 represents the sub task "Operating Experience Data" (OEI) checklist.

Table 5-15: Sub Task OE Check List

SERVICE EXPERIENCE DATA		JUSTIFICATION / MOTIVATION	DEVIATION / RESOLUTION ('WORK-AROUND')
☒	Pipe failure event data collection availability / event database and database scope	The analysts needs direct access to the event database in order qualify all analysis steps	
☒	Scope of the event database	Failure event data, exposure term data	
☒	Data quality assurance program		
☒	Completeness & comprehensiveness		May be a need for further review of OPEX to determine data completeness.
☒	Database application facilities	Search / Query Definition functions	
☒	Data processing, including validation and verification		
	Through-wall leak / flow rates	Perform engineering analysis if information is not readily available; base analysis of flaw / crack / hole size and system operating pressure	
	Verify pipe size		
☒	Non-Conforming Conditions, Weld Repair Data, Condition Reports, Root Cause Analysis Reports	A review of the plant specific operating experience data is always performed to ensure the completeness of the pipe failure event database that is used to generate the event population data.	

5.5 QUALITATIVE ANALYSIS

According to the calculation scheme, the input data consists of two parts; and exposure term and an event population. For a given observation period, the former reflects the total piping component population that generated the operating experience. The latter is an integral part of a Bayesian analysis process, which starts with the definition of an appropriate distribution that is representative of the state-of-knowledge prior the application at hand.

5.5.1 Exposure Term Definition

In pipe failure rate estimation the exposure term is the product of the number of components that provide the observed pipe failures and the total time over which failure events are collected. There is variability in the population counts. In part this variability stems from differences across NSSS types (e.g., number of RC loops), and in part it stems from different design and fabrication practices (e.g., use of cold bent piping versus use of welded fittings); Table 5-16 and Table 5-17. Also, design modifications are implemented during the lifetime of a plant to accommodate steam generator replacements, enhance the access for nondestructive examination (NDE), etc. The type of exposure term used (i.e. weld vs. base metal) should be a function of degradation mechanism acting on a piping component boundary; Table 5-18.

Table 5-16: Weld Population Data for a Typical 3-Loop Westinghouse PWR

System	Code Class	Weld Count by Pipe Size [inch]											
		2	**3**	**4**	**6**	**8**	**10**	**12**	**14**	**16**	**24**	**28**	**30**
AF	2	--	--	--	8	--	--	--	--	--	--	--	--
CS	2	--	--	--	--	30	132	85	--	--	--	--	--
CV	1	60	68	--	--	--	--	--	--	--	--	--	--
	2	40	86	26	38	115	--	--	--	--	--	--	--
FW	2	--	--	--	--	--	--	--	--	74	--	--	--
MS	2	--	--	--	15	21	--	--	15	--	--	--	93
RC	1	29	6	--	--	--	--	--	--	--	--	25	25
Pressurizer	1	--	--	77	--	4	--	--	14	--	--	--	--
RH	1	4	--	--	--	--	--	24	--	--	--	--	--
	2	--	--	--	--	14	165	66	30	--	--	--	--
SI	1	34	--	--	59	--	--	27	--	--	--	--	--
	2	244	154	10	159	21	98	46	50	--	12	--	--

Table 5-17: Weld Population Data for a Typical 4-Loop Westinghouse PWR

System	Code Class	Weld Count by Pipe Size [inch]											
		2	**3**	**4**	**6**	**8**	**10**	**12**	**14**	**16**	**24**	**28**	**30**
AF	2	--	--	19	--	--	--	--	--	--	--	--	--
CS	2	--	--	--	87	4	181	--	48	52	--	--	--
CV	1	238	22	--	--	--	--	--	--	--	--	--	--
	2	7	6	71	46	70	--	--	--	--	--	--	--
FW	2	--	--	--	233	--	--	--	--	140	--	--	--
MS	2	--	--	--	54	47	--	11	--	--	--	73	177
RC	1	377	160	38	23	56	24	17	--	--	--	43	33
Pressurizer	1	14	18	22	62	--	--	--	13	--	--	--	--
RH	1	--	--	--	--	--	--	66	--	--	--	--	--
	2	--	--	--	--	286	--	78	19	87	--	--	--
SI	1	127	48	--	94	42	40						
	2	2	7	19	75	224	119	66	26	5	126	--	--

The variability in weld population data is treated by using three estimates for the component populations with subjectively assigned probabilities to weight the best estimates and upper and lower bounds. The upper and lower bounds are set at percentages above and below these estimates based on engineering judgment; Figure 5-13.

Table 5-18: Technical Basis for Exposure Term Definition

Damage / Degradation Mechanism: Class	Damage / Degradation Mechanism: Type	Location of Flaw Initiation	Crack Morphology	Controlling Parameters (Environment / Metallurgy) [b]	Piping Reliability Model (Exposure Term Definition)
SCC	ECSCC [a]	Base metal	Transgranular	Chloride contamination, surface temperature below 100C	Linear feet of SS piping
	IDSCC	Weld metal	Interdenditric	Alloy 182 weld metal – cracking occurs along weld fusion line	Number of susceptible welds
	IGSCC	Weld heat-affected zone (HAZ)	Intergranular	Corrosion potential of process medium, chromium content of material, sensitization, pressure, temperature	Number of susceptible welds
	PWSCC	Weld metal, safe-ends	Intergranular	Stress, surface finish, temperature. Susceptible material includes Alloy 600 (base metal) and Alloy 82/182 (weld metal).	Number of bi-metallic (dissimilar metal) weld locations
	SICC	Weld metal	Transgranular	Strain rate, dynamic loading	Number of susceptible welds
	TGSCC	Base metal (bend, elbow)	Transgranular	Chloride or fluoride contaminants, formation of strain-induced cold worked areas, lack of or insufficient heat treatment	Number of bends or elbows (cold formed piping) or linear feet of piping
Flow-Assisted Degradation	Erosion	Base metal (bend, elbow, pipe)	N/A	Destruction of metals by the abrasive action of moving fluids, usually accelerated by the presence of solid particles or matter in suspension	For raw water piping, linear ft. of piping may be an appropriate component boundary. The definition of component boundary also should reflect a specific analysis objective
	Erosion-Cavitation	Base metal, weld metal	N/A	Downstream of directional change or in the presence of an eddy.	Determined through systematic DM evaluation; highly system-dependent
	Erosion Corrosion	Base metal (bend, elbow, pipe)	N/A	Corrosion occurs simultaneously with erosion	See "erosion"
	FAC	FAC-sensitive fittings (bend, elbow, expander, reducer, tee)	N/A	Temperature, flow velocity, fluid pH, oxygen, alloying element (e.g. chromium content)	Component boundary is defined by FAC program (i.e. susceptible components)
	LDIE	Elbow, area immediately downstream an orifice	N/A	Caused by impact of high velocity droplets or liquid jets. Normally, LDI occurs when a two-phase stream experiences a high-pressure drop	Highly localized mechanism, component boundary definition requires detailed DM evaluation coupled with review of field experience data

Damage / Degradation Mechanism		Location of Flaw Initiation	Crack Morphology	Controlling Parameters (Environment / Metallurgy) [b]	Piping Reliability Model (Exposure Term Definition)
Fatigue	High-cycle	Branch connections, welds (butt welds and socket welds)	Sharp crack turning paths	Presence of stress riser due to geometrical discontinuity, inadequate or failed pipe support	Depending on system, counts of branch connections and welds
	Low-cycle	Branch connections, welds	Sharp crack turning paths	Presence of stress riser, inadequate or failed pipe support	Depending on system, counts of branch connections and welds
	Thermal	Base metal at or near mixing tees or branch connections	"Torturous" crack path, in some cases transgranular	Intermittent cold water injection, low flow, little fluid mixing, stress risers, very frequent cycling, unstable turbulence penetration into stagnant line, bypass leakage in valves with large ΔT	Nozzles, branch connections, safe ends, welds, HAZ, and base metal regions of high stress concentration
Corrosion	General	Base metal	N/A	pH of water, calcium compounds	Linear feet of piping
	MIC	Base metal of CS and weld heat affected zone of SS	N/A	Presence of microbes in process medium	Depending on material, linear feet of piping or weld count
	Pitting	Base metal or crevices	N/A	Presence of chlorides in process medium and corrosion passivation layer	Depending on material (e.g. SS, nickel-base) and system, linear feet of piping or counts of susceptible locations

Note:
(a) Sub-categories include chloride-assisted SCC and "corrosion-under-insulation" (CUI)

PIPExp Database

Reactor Operating Years	
PWR MS Class 2	ROYs
2-Loop	1761
3-Loop	2750
4-Loop	2162
Total	6673

	> 10"			≤ 10"		
Plant	3-Loop	2-Loop	4-Loop	3-Loop	2-Loop	4-Loop
R3	96	–	–		–	–
K1	117	–	–	45	–	–
K2	121	–	–	45	–	–
VC	108	–	–	36	–	–
REG	–	64	–	–	21	–
K	–	72	–	–	26	–
Br1	–	–	139	–	–	88
Br2	–	–	130	–	–	90
By1	–	–	119	–	–	115
By2	–	–	99	–	–	112
Average	110.5	68	121.75	42	23.5	101.25
Min	96	64	99	36	21	88
Max	121	72	139	45	26	115

= ((G26+H26+I26)/3)*(2*D7+3*D8+4*D9)

	Welds per Loop					
	> 10"			≤ 10"		
Average	36.8	34	30.4	14	11.75	25.3
Min	32.0	32	24.8	12	10.5	22.0
Max	40.3	36	34.8	15	13	28.8

Weld-Years		> 10"	≤ 10"
	Low	604091.7	302896.7
	Medium	689316.8	347565.4
	High	756107.2	386278.3

Figure 5-13: Process for Exposure Term Definition

5.5.2 Prior Distributions for Bayesian Updating

The starting point for the treatment of uncertainty is the development of prior distributions for the pipe failure rates. This is accomplished by using a two-step process. The objective of this process is to capture the uncertainty in the state of knowledge about piping system failure rates before and independent of the application of the service data which is applied in the Bayes' updating process.

There are two types of priors: informative and non-informative. A non-informative prior is defined as one that provides little information relative to the analysis case. Informative prior distributions, on the other hand, summarize the evidence about the parameters concerned from many sources and often have a considerable impact on the results. Some priors are chosen to be "non-informative," that is, diffuse enough that they correspond to very little prior information. In many situations, it is not at all obvious which prior is the most appropriate and it is suggested that the sensitivity of the results to the choice of the prior be examined; c.f. Section 5.7.1.

The simplest prior distribution is discrete prior distribution. The examples of these distributions are Poisson distribution, Binomial distribution, discrete uniform distribution and Bernoulli's distribution. The posterior can be calculated easily by using a spread sheet. The next simplest prior is called conjugate; this prior combines neatly with the likelihood to give a posterior that can be evaluated by simple formulas. Finally, the most general priors are considered; the posterior distribution in such a case can only be found by numerical integration or by random sampling.

As noted by Box and Tiao [80], the prior distribution should accurately reflect prior knowledge or belief about the unknown parameter. However, quantifying belief is not easy. The prior belief should be internally consistent and should be accurately quantified.

In the past it was common practice to develop prior distribution parameters on the basis of the state of knowledge that existed at a time when there were no publically available, "pedigreed" estimates from the nuclear power plant piping system service experience. In the PSA context, when the Reactor Safety Study was performed (1972-1975)[49] there were no available estimates from nuclear power plant piping system service experience. Estimates for the overall frequency of major pipe breaks from the then available data sources were as high as $1x10^{-2}$ per reactor year and other estimates derived from probabilistic fracture mechanics methods were as low as $1x10^{-6}$ per reactor year. In developing the initial prior distributions this information could be used and fit to the 95th and 5th percentiles of a lognormal distribution giving a median of $1x10^{-4}$ per plant year and a range factor (RF)[50] of 100 to represent the overall frequency of major pipe ruptures on a per reactor year basis.

While a common practice in PSA applications, the choice of lognormal distribution is somewhat arbitrary. It should be noted that the form of the chosen distribution is not as important as the first two moments (mean and standard deviation). If one were to replace the lognormal distribution with another distribution that has the "right" properties, the final answers (i.e., posterior distribution parameters) would not be affected, however.

The next step to development of prior distributions is to convert the above lognormal distribution for pipe ruptures into a set of failure rates for specific systems and damage mechanisms for individual pipe components. A fixed ratio of the number of failures to ruptures is assumed based on a gross analysis of the events in the service data. In addition,

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[49] For a historical perspective, see NUREG/KM-0010 (https://www.nrc.gov/docs/ML1622/ML16225A002.pdf)

[50] RF = (95^{th} percentile/5^{th} percentile)$^{0.5}$

estimates are made of the system and component populations and fractions of the populations susceptible from different damage mechanisms. This information is used to develop scaling factors to convert the above lognormal distribution to the proper units.

The range factor of the lognormal distributions is fixed to 100 which provides a very broad distribution and might be regarded as a “slightly informative” prior. This is viewed to be preferable to the use of a non-informative prior, or maximum entropy prior which is often used by statisticians. A very broad lognormal distribution such as this one will not bias the results too much. It is also reasonable to assume that the probability distribution is unimodal, unlike the Jefferey’s non-informative prior which states that the probability of the true values of the parameters are highest at the extreme values.

An alternative to the non-informative prior is the empirical prior distribution, which is appropriate when sufficient pipe failure data is available. Two such examples include IGSCC in BWR plants and corrosion in raw water piping (e.g. Essential Service Water). For IGSCC, a good empirical prior distribution should account for the service experience “prior” to the implementation of mitigation programs such as weld overlay repairs and improved water chemistry. Standard statistical analysis techniques (such as ‘total-time-on-test’, ‘goodness-of-fit’, ‘hazard plotting’) readily apply to these types of pipe failure data. Table 5-19 compares prior distributions used in the Braidwood and Byron RI-ISI program development and the updated prior distributions developed for the Koeberg NPS RI-ISI project. New prior distributions have been developed for Corrosion, PWSCC, Thermal Fatigue, and Vibration Fatigue.

Numerous published sources exist of generic pipe failure rates and rupture frequencies. Is it feasible to update a generic pipe failure rate distribution using plant-specific pipe failure data? Due to the large uncertainties and relatively low failure rates associated with piping systems, performance of plant specific Bayes’ updates are not typically done. The reason for this is that there is normally insufficient plant specific evidence to justify this procedure. It has always been assumed that there would be only very small changes in pipe failure rate estimates if this type of Bayes’ update were to be performed. In order to perform a technically sound Bayes’ update of pipe failure rates the following questions arise:

- Is the plant specific data for failures and pipe exposure being collected and analyzed in a manner that is consistent with the treatment of generic data in the generic estimates provided in published reports?
- Is there significant plant-to-plant or site-to-site variability in the failure rate data that is reflected in the generic distributions?
- There is a question whether plant-specific data should be removed from the generic data to avoid over-counting the same evidence in two places. This is a generic issue in Bayes’ updating with plant specific data but it is usually ignored under the assumption that the contribution to the generic distributions from any specific plant is small. This might not be true in the pipe failure rate case especially if the plant in question has an unusually high incidence of failure relative to the rest of the industry.
- If the operating experience data points to some evidence of aging (e.g. a progressive trend upwards in the calculated average failure rate as new evidence is applied) additional work is needed to establish a good definition of the term “aging” and then to establish an appropriate statistical model. Typically only averaged failure rates are calculated over progressively longer periods. Subdividing a time period into smaller intervals might be a better approach to addressing temporal changes in calculated pipe failure rates.

Table 5-19: Examples of Prior Failure Rate Distributions for Pipe Failure Rate Estimation

Degradation or Damage Mechanism	Unit	RI-ISI Project "I" (2000)			RI-ISI Project "II" (2005)			Note
		Distr.	Mean	RF	Distr.	Mean	RF	
TF - Thermal Fatigue (global)	TWC × $(Weld.Yr)^{-1}$	LN	1.34E-05	100		--	--	
TF - BWR carbon steel (∅ > DN300)	TWC × $(Weld.Yr)^{-1}$		--		LN	2.48E-05	10	1, 2
TF – PWR Code Class 1 SI (100 <∅ ≤ 250)	TWC × $(Weld.Yr)^{-1}$		--		LN	6.64E-04	10	3
SCC - Stress Corrosion Cracking (global)	TWC × $(Weld.Yr)^{-1}$	LN	4.27E-05	100		--	--	
SCC - IGSCC in BWR operating environment	TWC × $(Weld.Yr)^{-1}$		--	--	LN	2.78E-04	10	2, 4
SCC - PWSCC in PWR operating environment	TWC × $(Weld.Yr)^{-1}$		--	--	LN	3.22E-06	100	5
SCC - TGSCC	TWC × $(Weld.Yr)^{-1}$		--	--				
Erosion Cavitation	TWC × $(Weld.Yr)^{-1}$	LN	1.01E-04	100				
Design & Construction Defects	TWC × $(Weld.Yr)^{-1}$	LN	1.24E-06	100				
BWR Main Steam piping (DN500 ≤ ∅ ≤ DN650)	TWC × $(Weld.Yr)^{-1}$					2.75E-06		6
Flow Accelerated Corrosion	TWC × $(m.Yr)^{-1}$	LN	1.71E-06	100				7
Corrosion Attack	TWC × $(m.Yr)^{-1}$	LN	3.38E-06	100				
Corrosion Attack - Carbon steel (DN50 < ∅ ≤ DN100)	TWC × $(m.Yr)^{-1}$		--	--	LN	2.07E-04	5	8
Corrosion Attack - Stainless steel (DN50 < ∅ ≤ DN100)	TWC × $(Weld.Yr)^{-1}$		--	--	LN	5.06E-05	5	9
High-Cycle Fatigue	TWC × $(System.Yr)^{-1}$	LN	6.30E-04	100				
RI-ISI scope systems	TWC × $(Weld.Yr)^{-1}$					1.00E-04	100	10
Steam/Water Hammer	Rupture × $(System.Yr)^{-1}$	LN	4.20E-05	100				
Human Error (Small-bore Piping)	Severance × $(System.Yr)^{-1}$	LN	4.20E-05	100				
Overstress or Overpressure	Rupture × $(System.Yr)^{-1}$	LN	4.20E-05	100				

Notes:

1. Applies to BWR main feedwater piping.
2. The weld failure rate is unadjusted for different weld locations, including weld residual stresses, influence by in-service inspection, or mitigation program.
3. The prior distribution applies to the most TF-susceptible inspection locations. It acknowledges service experience through calendar year 1996. The last, significant TF-induced SI piping failure occurred in Dampierre-2 on 21-Dec-1996 (0.63 gpm leak rate). TF mitigation programs have been implemented at most LWR plants since 1996; augmented inspections, modified operating procedures, design changes).
4. This distribution applies to NPS28 Reactor Recirculation piping without IGSCC mitigation.
5. This distribution applies to RC Cold Leg and Hot Leg piping; it accounts for the pre-2000, worldwide service experience (0 failures in 155,106 weld-years).
6. For Code Class 1 Main Steam piping subjected to ASME XI ISI requirements, there is no recorded information on service-induced flaws (non-TWC or TWC).
7. There are on the order of 80 Class 1 welds in a typical BWR. Zero (0) failures in 2247 critical reactor years gives a P.E. of about 2.75E-06 per weld-year.
8. Unadjusted for impact of ISI (volumetric examination). Susceptibility to FAC different for high-energy piping (e.g., Code Class 4 FW piping) and low-energy SW piping.
9. Applies to Code Class 3 low-energy Service Water piping. The given failure rate includes contributions from corrosion attack (pitting, MIC) and flow-accelerated corrosion and is unadjusted for influence by chemical treatment of raw water and volumetric examination. Applies to Code Class 3 low energy Service Water piping susceptible to micro-biologically influenced corrosion (MIC) [25].
10. Applies to PWR systems ≤ 3-inch diameter. Derived from service data.

While additional research is needed, it can be concluded that traditional Bayesian updating of pipe failure rates is not generally applicable. The analysis case definition needs to specifically account for the evaluation boundary and a thorough qualitative evaluation of all sources of failure data to obtain a reasonable level of confidence in the data completeness. Any underlying temporal shifts in the pipe failure data need to be explored further.

5.5.3 Step 4 Check List

The practical analysis insights from the three case studies have been organized in a series of checklists to support systematic reviews of licensee submittals of fitness-for-service evaluations, and to assist the OEAD staff in validating licensee submittals by performing independent fitness-for-service evaluations. Table 5-20 represents the sub task “Qualitative Data Analysis” (QAD) checklist.

Table 5-20: Sub Task QAD Check List

INPUT DATA PREPARATION		JUSTIFICATION / MOTIVATION	DEVIATION / RESOLUTION ('WORK-AROUND')
☒	Identify degradation mechanism(s) acting on piping component boundary(ies)	This is obtained via the operating experience data review and degradation mechanism analysis	
☒	Assemble piping population information per the "Plant Design Information" step		
☒	Assemble relevant piping population data for a representative set of plants		
☒	Determine the plant-to-plant variability in piping population data	For non-RCPB piping, expect this variability to be "considerable"	Obtain a reasonable set of isometric drawing packages to obtain a justifiable plant-to-plant variability ratio.
☒	Assign a discrete probability distribution (DPD) to the combined plant population data to characterize the plant-to-plant variability		
☒	Assemble the reactor operating years that produced the event population data		

5.6 QUANTITATIVE ANALYSIS

Pipe failure frequency is calculated using Bayesian analysis to update the probability distribution representing our prior state of knowledge with the evidence from a pipe failure event database such as CODAP. This section presents an example of how the CODAP event database can be applied in support of buried piping reliability analysis. The example is concerned with an assessment buried Essential Service Water (ESW) piping failure rates and break frequencies on the basis of the corresponding ESW buried piping operating experience as recorded in CODAP.

5.6.1 Quantification Scheme

The technical approach to estimating the frequency of an ESW pipe break on the basis of operating experience data is expressed by Equations (5-1) and (5-2); Equation 5-2 represents the structural integrity "risk triplet". The magnitude (i.e. size of a pressure boundary breach) is expressed by an equivalent break size (EBS) "*x*" and corresponding peak through-wall flow rate. The parameter *x* is treated as a discrete variable representing different equivalent break-size ranges.

$$F(IE_x) = \sum_i m_i \rho_{ix} \tag{5-1}$$

$$\rho_{ix} = \sum_k \lambda_{ik} P(R_x | F_{ik}) I_{ik} \tag{5-2}$$

Where:

$F(IE_x) =$ Frequency of pipe break of size *x*, per reactor operating-year, subject to epistemic (or state-of-knowledge) uncertainty calculated via Monte Carlo simulation.

$m_i =$ Number of pipe welds (or fittings, segments or inspection locations of type *i*; each type determined by pipe size, weld type, applicable damage or degradation mechanisms, and inspection status (leak test and non-destructive examination). While not explicitly addressed in the given example, for the buried ESW piping the parameter m_i corresponds to the total length of piping being analyzed.

$\rho_{ix} =$ Frequency of rupture of component type *i* with break size *x*, subject to epistemic uncertainty calculated via Monte Carlo simulation.

$\lambda_{ik} =$ Failure rate per "location-year" for pipe component type *i* due to failure mechanism *k*, subject to epistemic uncertainty, Equation 5-3 below. In this analysis the failure rate is calculated on the basis of per linear meter and reactor operating year.

$P(R_x | F_{ik}) =$ For leak-before-break (LBB) piping, the conditional rupture probability (CRP) of size *x* given failure of pipe component type *i* due to damage or degradation mechanism *k*, subject to epistemic uncertainty. This parameter may be determined on the basis of probabilistic fracture mechanics, expert elicitation or service experience insights. However, a CRP model is not developed for the ESW case study since there is sufficient operating experience data available for the full spectrum of pipe break sizes.

$I_{ik} =$ Integrity (RIM) management factor for weld type *i* and failure mechanism *k*, subject to epistemic uncertainty. This parameter is not explicitly addressed in this example, however.

For a point estimate of the failure rate of piping component type *i* and degradation mechanism *k*:

$$\lambda_{ik} = \frac{n_{ik}}{\tau_{ik}} = \frac{n_{ik}}{f_{ik} N_i T_i} \qquad (5\text{-}3)$$

Where:

n_{ik} = Number of failures in pipe component of type *i* due to degradation mechanism *k*. The component boundary used in defining exposure terms is a function of the susceptibility to certain damage or degradation mechanisms..

τ_{ik} = Component exposure population for welds of type *i* susceptible to degradation mechanism *k.*

f_{ik} = Estimate of the fraction of the component exposure population for piping component type *i* that is susceptible to degradation mechanism *k*, estimated from results of a formal degradation mechanism evaluation. In this example it is assumed that each section of ESW piping is equally susceptible to degradation through internal or external corrosion.

N_i = Estimate of the average number of pipe components of type *i* per reactor in the reactor operating years of exposure for the data query used to determine n_{ik}. Determined from isometric drawings reviews for a population of plants and expert knowledge of degradation mechanisms. In this analysis N = linear meter of ESW piping on a per plant basis. The plant-to-plant variability is accounted for using three estimates for the component populations and subjectively assigning probabilities to weight the best estimates and upper and lower bounds. The best estimates are derived from a sample of plant for which details on the linear meter of ESW piping is available. The upper and lower bounds were set at percentages above and below these estimates based on engineering judgment (Figures 5-14 & 5-15).

T_i = Total exposure in reactor-years for the data collection for component type *i*. CODAP event database provides the number of reactor operating years that produced the operating experience data. In this example, the ESW failure population resulted from 3042 reactor operating years (ROYs).

For a Bayes' estimate, a prior distribution for the failure rate is updated using n_{ik} and τ_{ik} with a Poisson likelihood function. The formulation of Equation (2) enables the quantification of conditional failure rates, given the known susceptibility to the given damage or degradation mechanism. When the parameter f_{ik} is applied, the units of the failure rate are failures per piping component susceptible to the degradation mechanism of concern.

The above calculation format has been implemented in a Microsoft® Excel spreadsheet with two add-in programs for Bayesian reliability analysis and Monte Carlo simulation, respectively. A first step in this data processing involves querying the event database by applying data filters that address the conjoint requirements for pipe degradation and failure. These data filters are integral part of a database structure. Specifically, these data filters relate to unique piping reliability attributes and influence factors with respect to piping system design characteristics, design and construction practice, in-service inspection (ISI) and operating environment. A qualitative analysis of service experience data is concerned with establishing the unique sets of calculation cases that are needed to accomplish the overall analysis objectives and the corresponding event populations and exposure terms.

Update Type	Description	Units	Prior Mean	Prior Range Factor	Evidence Failures	Evidence Exposure	Posterior Mean	Posterior 5th	Posterior 50th	Posterior 95th	Posterior Range Factor	Calc Date
TIME	Case 1: ESW BP Low	pipe m.yr	1.03E-07	100	1	1,670,058	2.86E-07	7.21E-09	1.51E-07	1.03E-06	11.93	11/9/2016 0:00
TIME	Case 1: ESW BP Medium	pipe m.yr	1.03E-07	100	1	3,249,617	1.66E-07	5.12E-09	9.29E-08	5.74E-07	10.59	11/9/2016 0:00
TIME	Case 1: ESW BP High	pipe m.yr	1.03E-07	100	1	4,648,176	1.23E-07	4.21E-09	7.10E-08	4.19E-07	9.98	11/9/2016 0:00
TIME	Case 2: ESW BP Low	pipe m.yr	1.03E-07	100	4	1,670,058	1.89E-06	5.61E-07	1.70E-06	3.86E-06	2.62	11/9/2016 0:00
TIME	Case 2: ESW BP Medium	pipe m.yr	1.03E-07	100	4	3,249,617	9.94E-07	3.01E-07	8.96E-07	2.02E-06	2.59	11/9/2016 0:00
TIME	Case 2: ESW BP High	pipe m.yr	1.03E-07	100	4	4,648,176	7.04E-07	2.16E-07	6.36E-07	1.43E-06	2.57	11/9/2016 0:00
TIME	Case 3: ESW BP Low	pipe m.yr	1.03E-07	100	10	1,670,058	5.39E-06	2.83E-06	5.19E-06	8.62E-06	1.75	11/9/2016 0:00
TIME	Case 3: ESW BP Medium	pipe m.yr	1.03E-07	100	10	3,249,617	2.80E-06	1.47E-06	2.70E-06	4.46E-06	1.74	11/9/2016 0:00
TIME	Case 3: ESW BP High	pipe m.yr	1.03E-07	100	10	4,648,176	1.96E-06	1.04E-06	1.89E-06	3.13E-06	1.74	11/9/2016 0:00
TIME	Case 4: ESW BP Low	pipe m.yr	1.03E-06	100	29	1,670,058	1.69E-05	1.20E-05	1.67E-05	2.24E-05	1.37	11/9/2016 0:00
TIME	Case 4: ESW BP Medium	pipe m.yr	1.03E-06	100	29	3,249,617	8.69E-06	6.19E-06	8.59E-06	1.15E-05	1.36	11/9/2016 0:00
TIME	Case 4: ESW BP High	pipe m.yr	1.03E-06	100	29	4,648,176	6.08E-06	4.34E-06	6.01E-06	8.08E-06	1.36	11/9/2016 0:00

Figure 5-14: Implementation of Calculation Scheme in Microsoft® Excel (Part 1)

CB Input						Probability/Susceptible Pipe-Feet Years												Weighting Dist	
Median	95th	Pipe length	Prob	Suscept	Prob	LL	LM	LH	ML	MM	MH	HL	HM	HH	Low	Mean	High	VALUE	PROB
1.45E-07	9.86E-07	549	0.25	1.00	0.25	0.0625	0.125	0.0625	0.125	0.25	0.125	0.0625	0.125	0.0625	1,670,058	3,249,617	4,648,176	1	0.1
8.97E-08	5.55E-07	1,098	0.50	1.00	0.5	1670058	1670058	1670058	3340116	3340116	3340116	4648176	4648176	4648176				2	0.8
6.88E-08	4.06E-07	1,528	0.25	1.00	0.25													3	0.1
1.66E-06	3.79E-06	549	0.25	1.00	0.25	0.0625	0.125	0.0625	0.125	0.25	0.125	0.0625	0.125	0.0625	1,670,058	3,249,617	4,648,176	1	0.1
8.80E-07	1.99E-06	1,098	0.50	1.00	0.5	1670058	1670058	1670058	3340116	3340116	3340116	4648176	4648176	4648176				2	0.8
6.24E-07	1.40E-06	1,528	0.25	1.00	0.25													3	0.1
5.14E-06	8.53E-06	549	0.25	1.00	0.25	0.0625	0.125	0.0625	0.125	0.25	0.125	0.0625	0.125	0.0625	1,670,058	3,249,617	4,648,176	1	0.1
2.67E-06	4.42E-06	1,098	0.50	1.00	0.5	1670058	1670058	1670058	3340116	3340116	3340116	4648176	4648176	4648176				2	0.8
1.87E-06	3.10E-06	1,528	0.25	1.00	0.25													3	0.1
1.66E-05	2.23E-05	549	0.25	1.00	0.25	0.0625	0.125	0.0625	0.125	0.25	0.125	0.0625	0.125	0.0625	1,670,058	3,249,617	4,648,176	1	0.1
8.55E-06	1.15E-05	1,098	0.50	1.00	0.5	1670058	1670058	1670058	3340116	3340116	3340116	4648176	4648176	4648176				2	0.8
5.99E-06	8.04E-06	1,528	0.25	1.00	0.25													3	0.1

Figure 5-15: Implementation of Calculation Scheme in Microsoft® Excel (Part 2)[51]

[51] Figure 5-15 is the continuation of Figure 5-14; the single Excel spreadsheet has been split in two parts for the purpose of the presentation in this report.

The piping component failure rate estimates can be specialized further by accounting for a "location-dependency" factor, which accounts for local environmental factors (e.g. flow conditions or pipe stresses). Obtained directly from operating experience data, the location-dependent weld failure rate correction factor is shown in Figure 5-16.

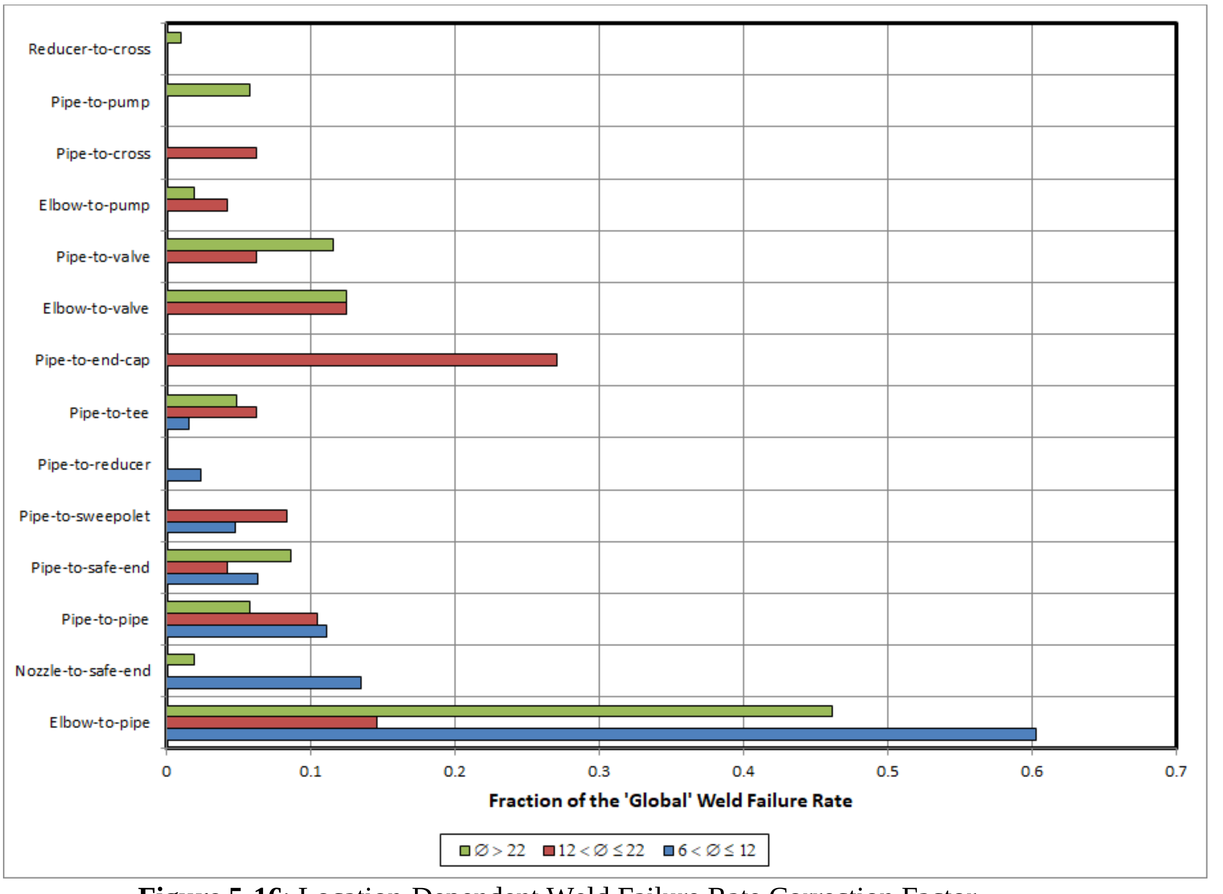


Figure 5-16: Location-Dependent Weld Failure Rate Correction Factor

For a weld of type "i" and size "j" the failure rate is expressed as follows:

$$\lambda_{ij} = F_{ij}/(W_{ij} \times T) \quad (5\text{-}4)$$

$$S_{ij} = F_{ij} / F_j \quad (5\text{-}5)$$

$$A_{ij} = W_j/W_{ij} \quad (5\text{-}6)$$

$$\lambda_{ij} = (F_j \times S_{ij}) \times 1/(W_{ij} \times T) \quad (5\text{-}7)$$

$$\lambda_{ij} = (F_j \times S_{ij}) \times A_{ij} /(W_j \times T) \quad (5\text{-}8)$$

$$\lambda_{ij} = S_{ij} \times A_{ij} \times \lambda_j \quad (5\text{-}9)$$

Where:

λ_{ij}	=	Failure rate of an IGSCC-susceptible weld of type "i", size "j"
λ_j	=	Failure rate of an IGSCC susceptible weld of size 'j'
F_j	=	Number of size "j" weld failures
F_{ij}	=	Number of type "i" and size "j" weld failures
W_j	=	Size "j" weld count
W_{ij}	=	Type "i" and size "j" weld count
Susceptibility (S_{ij})	=	The service experience shows the failure susceptibility to be correlated with the location of a weld relative to pipe fittings and other in-line components (terminal ends, flanges, pump casings, valve bodies). For a given pipe size and system, the susceptibility is expressed as the

		fraction of welds of type "ij" that failed due to a certain degradation mechanism). This fraction is established from the operating experience data.
Attribute (A_{ij})	=	In the above expressions the attribute (A) is defined as the ratio of the total number of welds of size "j" to the number of welds of type "i". In expression (5-6) A_{ij} is a correction factor and accounts for the fact that piping system design & layout constraints impose limits on the number of welds of a certain type. For example, in a given system there could be more elbow-to-pipe welds than, say, pipe-to-tee welds.

Combining a global (or averaged) failure rate with the location-dependency information in Figure 5-16 and Table 5-21 provides the apriori failure rates.

Table 5-21: Selected Stainless Steel Weld Susceptibility & Attribute Factors

NPS [inch]	Weld Location	Location-Dependency Parameters	
		Susceptibility (S_{ij})	Attribute (A_{ij})
12	Elbow-to-pipe	6.03E-01	2.8
	Nozzle-to-safe-end	1.35E-01	5.0
	Pipe-to-reducer	2.38E-02	25.0
22	Pipe-to-end-cap	2.71E-01	4.0
	Pipe-to-sweepolet	8.33E-02	2.0
	Pipe-to-cross	6.25E-02	4.0
28	Elbow-to-pipe	4.62E-02	5.6
	Pipe-to-pipe	5.77E-02	3.1
	Cross-to-reducer	9.60E-03	28.8

5.6.2 Results Presentation

The quantitative results may be presented in tabular or graphical form. The latter may be displayed as cumulative pipe failure rate versus break size or through-wall flow rate. For the ESW buried piping case, Illustrated in Figure 5-17 are the results of the ESW piping reliability analysis as a function of through-wall flow rate threshold values:

- Equivalent Break Size Class (EBSC) >1 corresponds to a very small, perceptible leakage. This calculation case captures all events of EBSC > 1, from very small to very large through-wall flow rates.
- EBSC > 6 corresponds to a relatively small leak; greater than 0.06 kg/s. Over time, this would be a leak of sufficient magnitude to cause soil erosion adjacent to leak site with the potential of propagating to a large leak.
- EBSC > 10 corresponds to a significant mass flow rate greater than 60 kg/s.
- EBSC > 15, finally, corresponds to a very significant buried ESW pipe failure with a through-wall mass flow rate greater than 6,000 kg/s.

Included in Figure 5-17 for comparison is the calculated ESW pipe failure rate for accessible, in-plant ESW piping based on U.S. operating experience [75].[52] The difference in reliability of inaccessible (buried) versus the accessible ESW piping is attributed to differences in damage/degradation susceptibility and aging management program.

[52] From Table 3-23 of Reference [75].

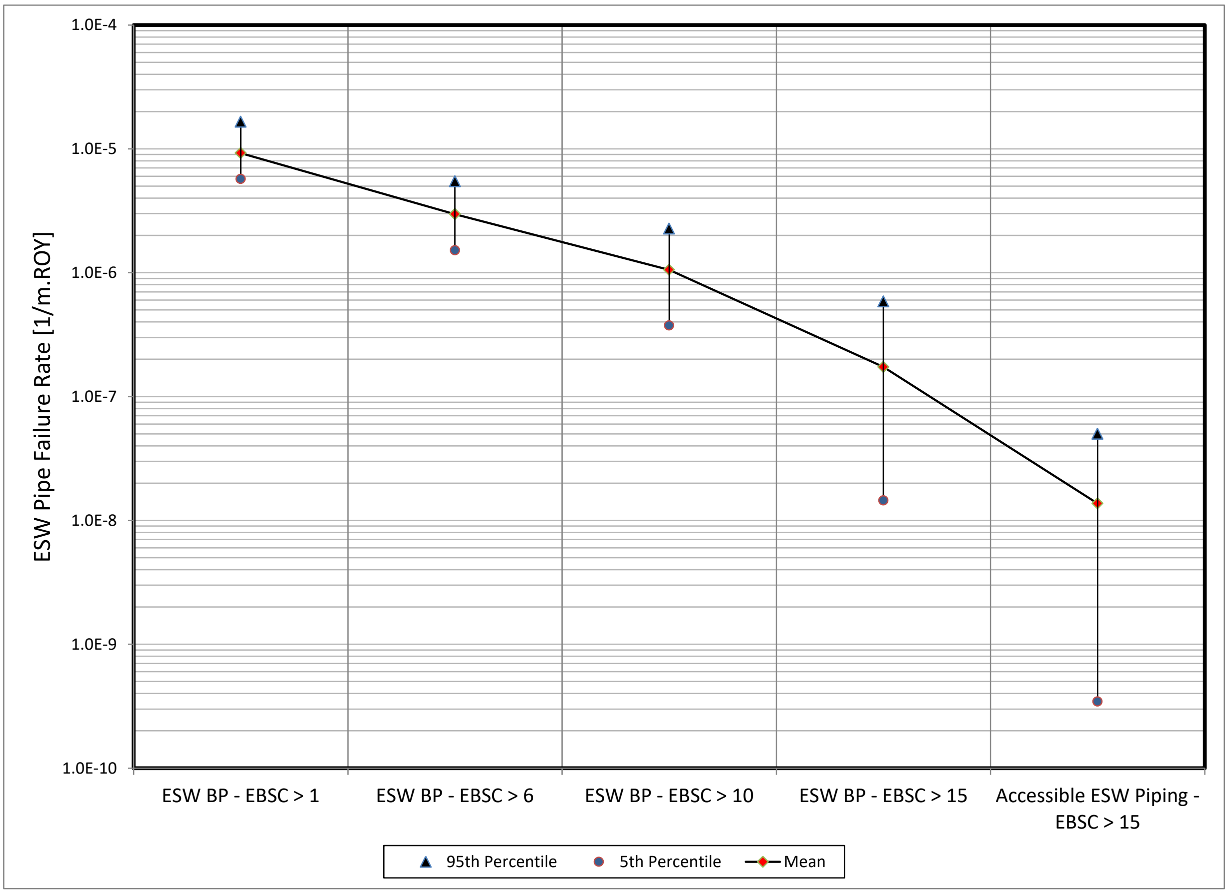


Figure 5-17: Comparison of Calculated Mean Cumulative ESW Large-Diameter (≥ DN600) Pipe Failure Rates as a Function of Break Size

5.6.3 Conditional Rupture Probability

The scope of a pipe break frequency calculation determines the type of CRP model to be used. A selected calculation strategy oftentimes involves the consideration of a suite of different CRP models in order to establish a plant-specific aggregate that acknowledges that a given piping systems may be susceptible to environmental degradation in addition to effects of stress risers. As an example, for a feedwater piping system that consists of cold-formed stainless steel piping section and low alloy steel piping systems, the following proposed CRP models that are summarized in Table 5-22 would be required. Section 10.3 includes additional details on how to obtain CRP distribution parameters.

Table 5-22: Proposed CRP Distribution Parameters

TASCS - FW Nozzle Weld Area						
Class	**EBS**	**Mean – TASCS**	**5%-tile**	**Median**	**95%-tile**	**RF**
1	0.5	1.70E-02	5.77E-03	1.02E-02	1.80E-02	1.8
2	1.5	2.88E-03	5.27E-04	2.10E-03	8.39E-03	4
3	3	6.40E-04	1.13E-04	4.53E-04	1.81E-03	4
4	6	9.67E-05	1.03E-05	5.67E-05	3.11E-04	5.5
5	14	2.27E-05	2.43E-06	1.33E-05	7.30E-05	5.5
FAC - Single-Phase Flow						
Class	**EBS**	**Mean - FAC - Single Phase Flow**	**5%-tile**	**Median**	**95%-tile**	**RF**
1	0.5	1.27E-01	1.57E-02	7.87E-02	3.93E-01	5.0
2	1.5	3.05E-02	3.78E-03	1.89E-02	9.45E-02	5.0
3	3	1.53E-02	5.74E-04	5.74E-03	5.74E-02	10.0
4	6	5.09E-03	2.00E-05	6.00E-04	1.80E-02	30.0
5	14	2.54E-03	9.99E-06	3.00E-04	8.99E-03	30.0
FAC - Two-Phase Flow						
Class	**EBS**	**Mean - FAC - Two-Phase Flow**	**5%-tile**	**Median**	**95%-tile**	**RF**
1	0.5	2.33E-01	2.89E-02	1.44E-01	7.22E-01	5.0
2	1.5	5.27E-02	6.53E-03	3.27E-02	1.63E-01	5.0
3	3	3.01E-02	1.13E-03	1.13E-02	1.13E-01	10.0
4	6	1.18E-02	4.64E-05	1.39E-03	4.18E-02	30.0
5	14	4.31E-03	1.69E-05	5.08E-04	1.53E-02	30.0
LDIE - Low Alloy Steel						
Class	**EBS**	**Mean - LDIE**	**5%-tile**	**Median**	**95%-tile**	**RF**
1	0.5	1.43E-03	1.77E-04	8.86E-04	4.43E-03	5.0
2	1.5	3.28E-04	1.23E-05	1.23E-04	1.23E-03	10.0
3	3	1.64E-04	6.16E-06	6.16E-05	6.16E-04	10.0
4	6	5.74E-05	2.16E-06	2.16E-05	2.16E-04	10.0
5	14	2.49E-05	4.28E-07	6.42E-06	9.63E-05	15.0
LC-FAT - No Active DM						
Class	**EBS**	**Mean - LC-FAT**	**5%-tile**	**Median**	**95%-tile**	**RF**
1	0.5	4.47E-04	5.54E-05	2.77E-04	1.39E-03	5.0
2	1.5	1.01E-04	3.80E-06	3.80E-05	3.80E-04	10.0
3	3	4.20E-05	1.58E-06	1.58E-05	1.58E-04	10.0
4	6	2.58E-05	9.68E-07	9.68E-06	9.68E-05	10.0
5	14	5.94E-06	1.02E-07	1.53E-06	2.30E-05	15.0

5.6.4 Water Hammer Analysis

Certain piping systems are vulnerable to water hammer events that can fail a piping pressure boundary. This section outlines an approach to water hammer impact analysis. According to the PIPExp database, a total of 691 significant water hammer events were identified for the time period 1970-2015 covering a plant exposure of 4924 reactor operating years. This includes 3863 reactor operating years in the U.S. fleet and 1061 reactor operating years with selected French, German, Swedish, and Swiss nuclear power plant experience. The term "significant" is used to note that the 691 events were severe enough to be reported given the knowledge that many water hammer events are experienced but never reported and hence to be available to be collected into the pipe database. Of the 691 water hammer events only 65 resulted in a major pressure boundary failure (PBF-M) including 61 at U.S. plants, and of these a single event involved a rupture of a valve body. In this analysis major pressure boundary failure is defined as a breach of the pressure boundary with through wall leakage of at least 100gpm. Analyses of the water hammer events into time periods, systems, causes, and pressure boundary impacts are executed in Figures 5-18 through 5-20. The nomenclature used in this data evaluation is summarized as follows:

- HF; procedural error, operator error contributing to erroneous valve operation
- N-PBF; no visible Pressure Boundary damage.
- D-PB; degraded Pressure Boundary, including detectable through-wall leakage on piping less than or much less than 100 gpm.
- M-PBF; major Pressure Boundary failure with through wall leakage greater than 100 gpm.
- Valve Failure-C; spurious valve closure caused by valve control(s)
- Valve Failure-M; spurious valve closure caused by internal mechanical failure

Summarized in Table 5-22 are selected results of a statistical analysis of the water hammer event data. Illustrated in Figure 5-21 is the conditional probability of a pressure boundary failure given a pressure pulse caused by water hammer. Illustrated in Figure 5-22 is a conceptual water hammer event sequence diagram for the quantification of the frequency of a pressure boundary failure.

5.6.5 Step 5 Check List

Practical analysis insights have been organized in a checklist to support systematic reviews of licensee submittals of fitness-for-service evaluations, and to assist the OEAD staff in validating licensee submittals by performing independent fitness-for-service evaluations. Table 5-24 represents the sub task “Quantitative Analysis” checklist.

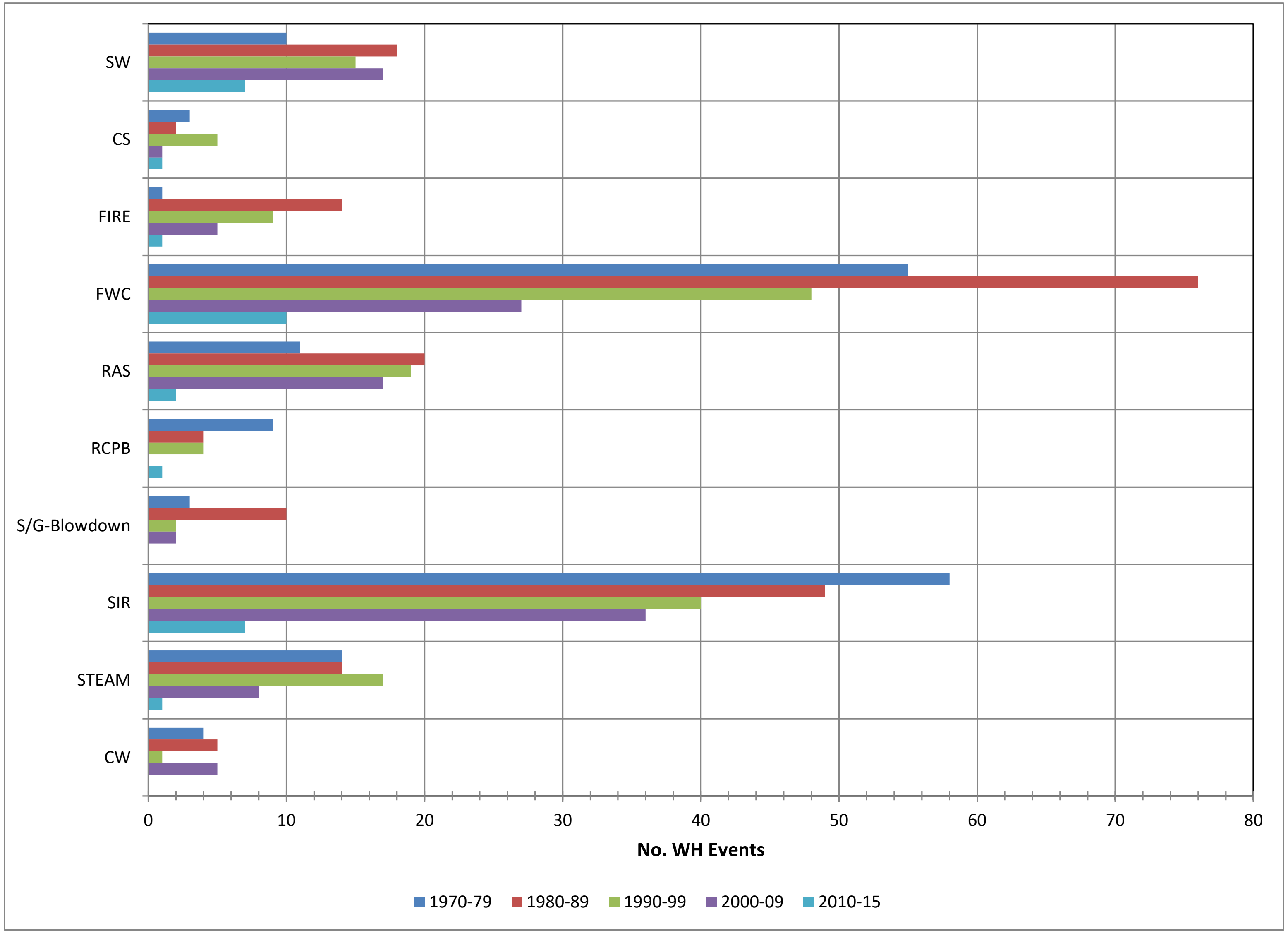


Figure 5-18: Water Hammer Events by Plant System & Time Period

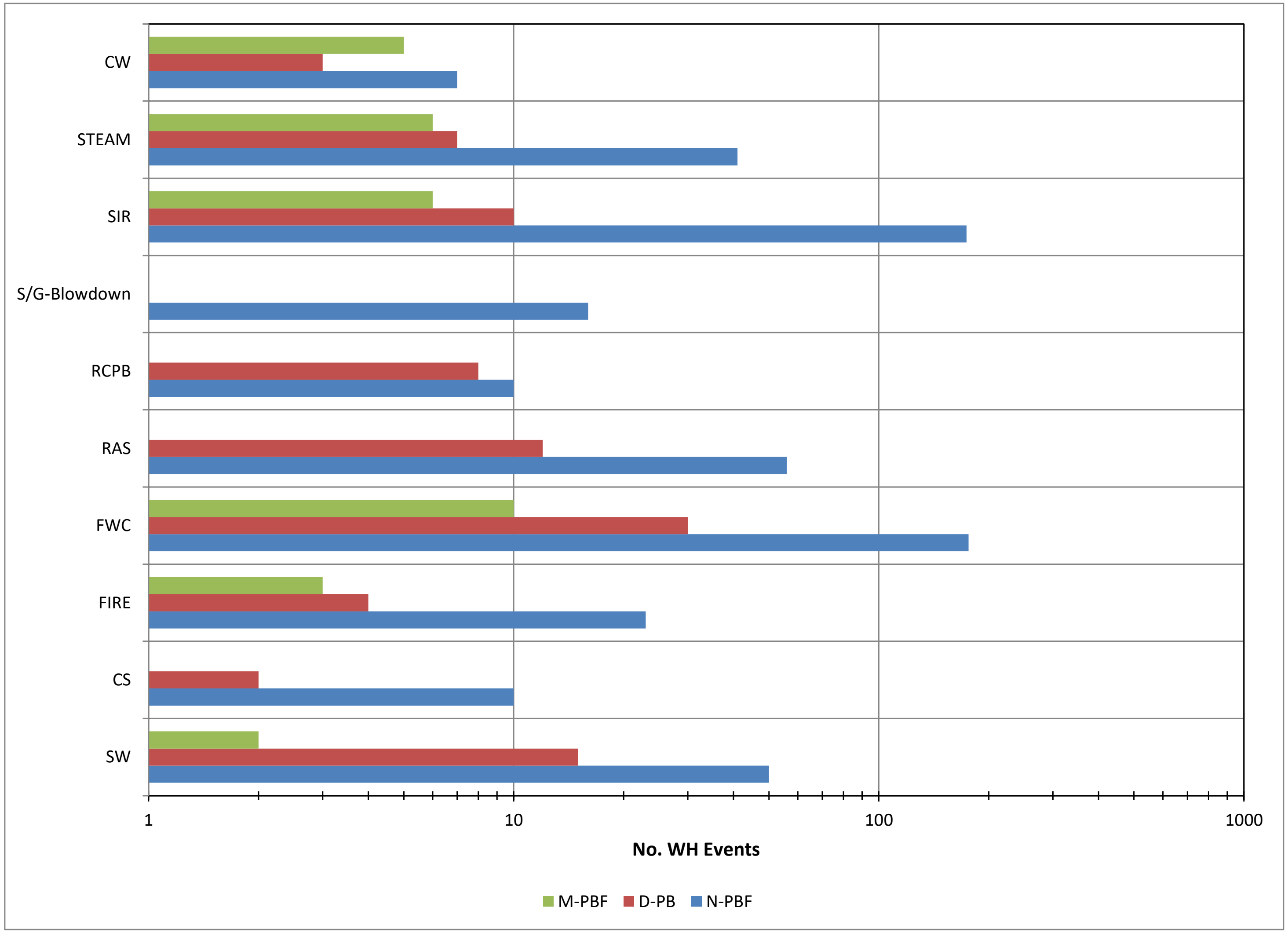


Figure 5-19: Water Hammer Events by Plant System & Impact on Pressure Boundary Integrity[53]

[53] 'M-PBF' = Major Pressure Boundary Failure; 'D-PB' = Degraded Pressure Boundary; 'N-PBF' = No Pressure Boundary Failure

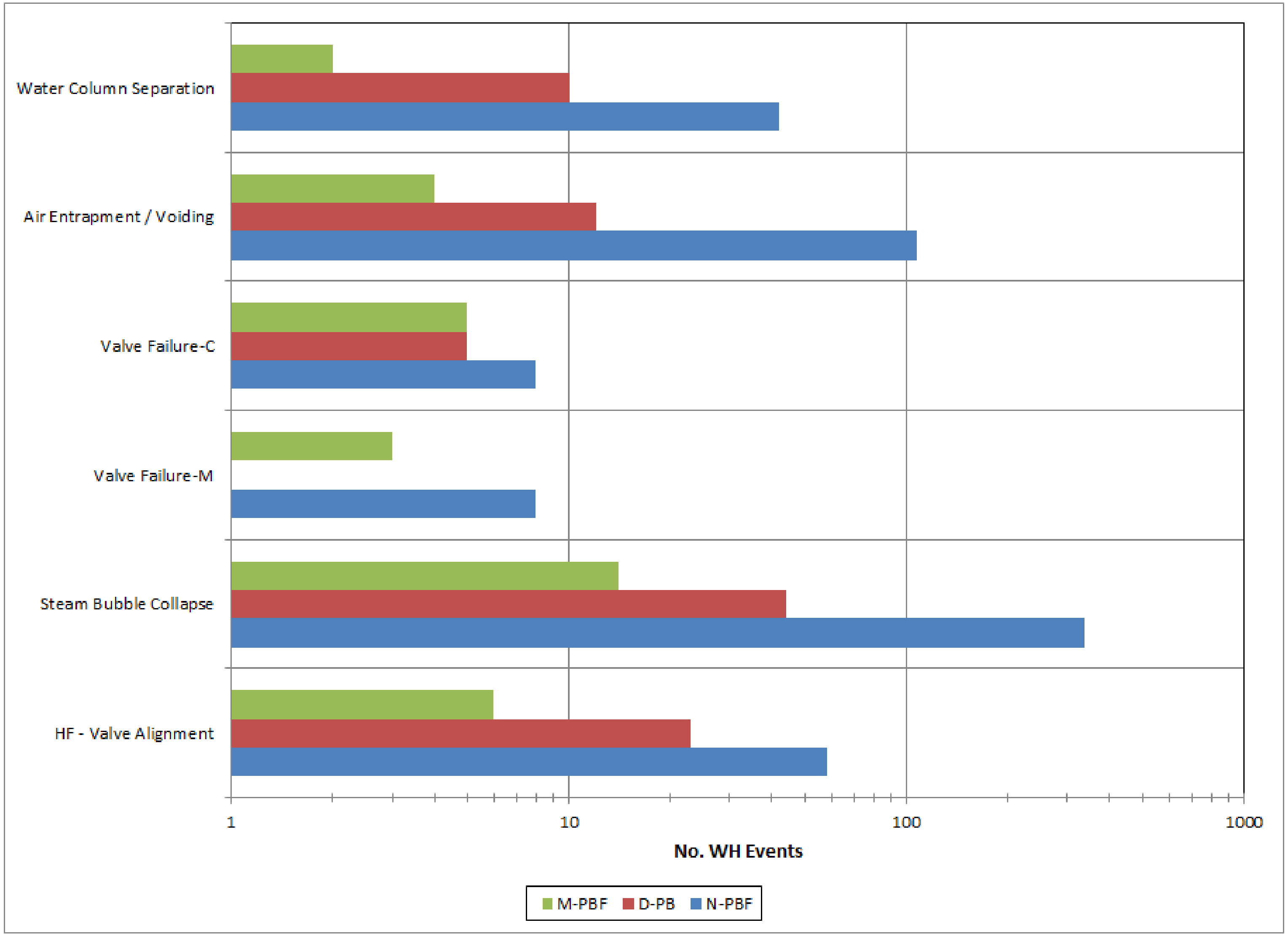


Figure 5-20: Water Hammer Events by Cause & Impact

Table 5-23: Statistical Analysis of Water Hammer Event Data

Cause of WH	Major PBF by WH - All System Groups		IE Frequency [1/ROY]					
			Prior		Posterior			
	Impacted Component	# Events	Mean	RF	Mean	5th	50th	95th
HF: Valve Alignment/Operation	Elbow	2	Not evaluated					
	EXJ_M (Metal Expansion Joint)	3						
	Flange	1						
	Nozzle	1						
	Pipe	12						
	Reducer	1						
	Tube_H/X	1						
	Total Event Count:	21						
LP Discharge - Hot Water Entering Low Pressure Line	Pump Casing - S/G Blowdown	1	Not evaluated					
	Total Event Count:	1						
Steam Bubble Collapse	Elbow	1	Not evaluated					
	End Cap	1						
	EXJ-M	5						
	Pipe	8						
	Rupture Disc	2						
	Valve Body	1						
	Weld	4						
	Total Event Count:	22						
Valve_C - note, a typical commercial nuclear power plant has 12 CW EXJ_R Joints - 4/inlet side & 8/discharge side	CW EXJ_R (Rubber Expansion Joint) ≤ 150,000 gpm	1	2.20E-03	100	5.11E-05	2.60E-07	1.38E-05	2.30E-04
	Pipe - ≤ 70 gpm - CW	2	Not evaluated					
	Pump Casing	1						
	Reducer - 2" RWCU line	1						
	Vacuum Breaker - CW Blowdown	1						
	Total Event Count:	6						
Valve_M	EXJ_R - ≤ 2000 gpm - CW	1	1.03E-03	100	4.36E-05	1.45E-07	9.23E-06	2.05E-04
	Pipe - ≤ 40,000 gpm - CW	1	8.18E-04	100	3.42E-05	1.15E-07	7.30E-06	1.16E-04
	Pump Casing - ≤ 200,000 gpm - CW	1	1.63E-03	100	4.39E-05	2.03E-07	1.12E-05	1.99E-04
	Weld - ≤ 15.8 gpm - FW	1	Not evaluated					
	Total Event Count:	4						
Air Entrapment/Voiding	Rupture Disc	3	Not evaluated					
	Weld	2						
	Total Event Count:	5						
Water Column Separation - SW Piping System	EXJ_R	1	Not evaluated					
	Pipe	1						
	Total Event Count:	2						
	Total No. M-PBF Events:	61						

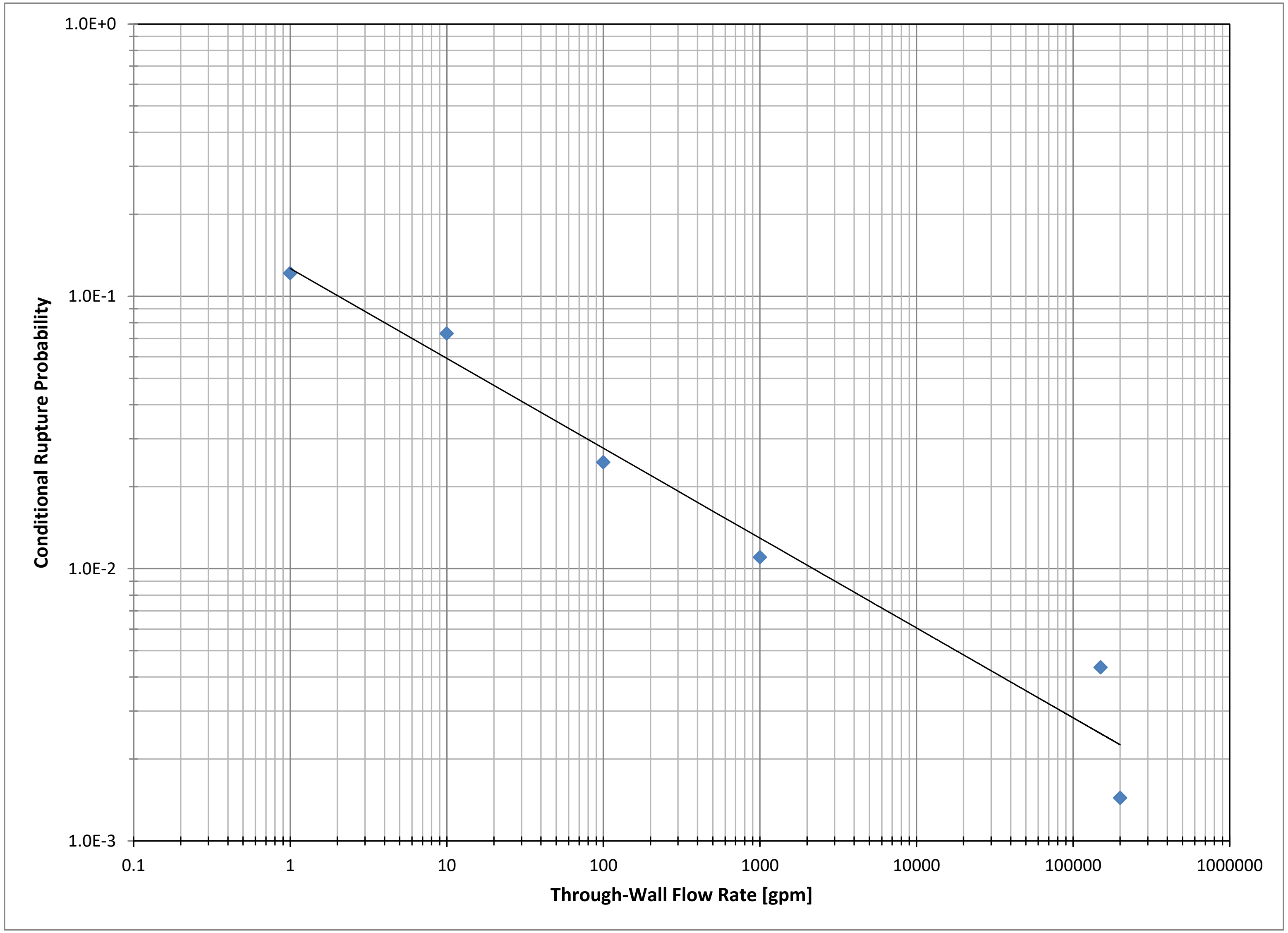


Figure 5-21: Conditional Probability of Pressure Boundary Failure Given a Water Hammer Induced Pressure Pulse

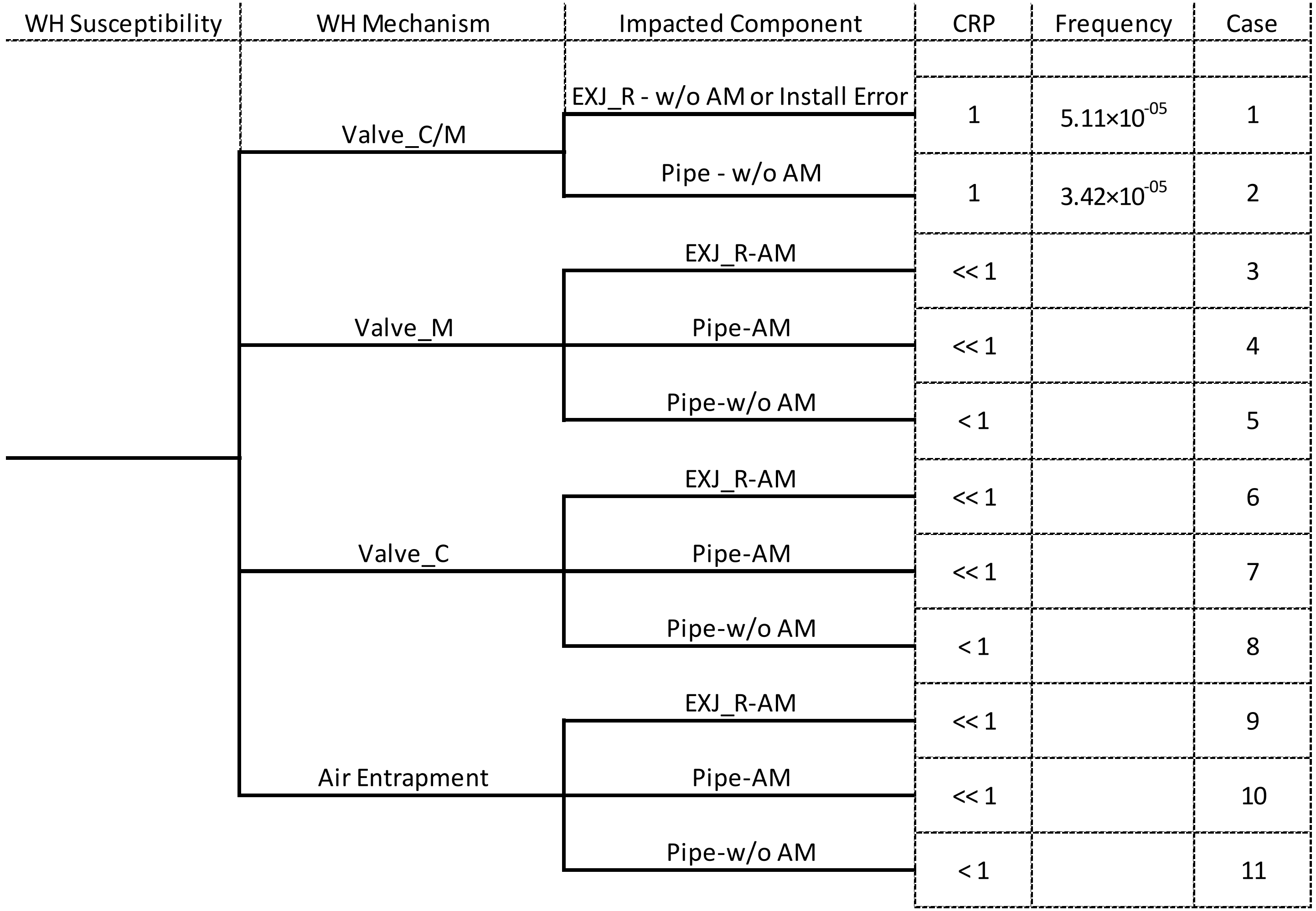


Figure 5-22: Water Hammer Event Sequence Diagram

Table 5-24: Sub Task QUANT Check List

PREPARE FOR QUANTIFICATION		JUSTIFICATION / MOTIVATION	DEVIATION / RESOLUTION ('WORK-AROUND')
☒	Define the technical approach to the quantification scheme	Justification may exist for using an alternative quantification scheme as part of a verification / validation of a licensee submitted analysis.	When it is deemed necessary, develop & submit for peer review a White Paper on the selected CFP model; refer to Appendix C for an example
☒	Define the types of calculations to be performed		
☒	Develop / obtain quantification tool(s) / program(s) & define the user(s)	In-house developed tool & use of licensed / proprietary tool. Determine the extent of training needed.	
☒	Results presentation format(s)	Determine what the acceptance criteria might be as well as asking the question: Do the piping reliability metrics used and results presentation format properly align with expectations?	
☒	Define the uncertainty analysis methodology	Review the industry practice and make adjustments as needed	
☒	Method of results validation	Review the industry practice and make adjustments as needed	

5.7 SENSITIVITY ANALYSIS

Certain follow-up (or sensitivity) studies may have to be performed once a base case set of piping reliability parameters have been obtained. In this section, two examples are provided to illustrate how to address the impact of certain assumption on base case results.

5.7.1 Impact of Choice of Prior Distribution Parameters on Results

Using a Bayesian approach, tabulated pipe failure rates in a referenceable information source are treated as "generic" or "industry-wide" data that represent informed prior distributions. A Bayesian approach has been extensively used to analyze risk significant internal flooding accident scenarios. A first step in the Bayesian process is to collect plant-specific failure and exposure term data, followed by a careful selection of the relevant prior distribution parameters.

A demonstration of the process uses the operating experience with Safety Class 3 Service Water piping of NPS30 is used. Restricted to the brackish-/sea-water environment, illustrated in Figure 5-23 is SW pipe operating experience for plant "X" versus the overall SW operating experience less the event population from plant "X". This comparison of the operating experience data raises the following question with respect to how to best perform a Bayesian update:

- Given the relative weight of the evidence provided by the prior and plant-specific data[54] what would be an appropriate strategy for performing a Bayesian update? Should a selected prior be taken straight from referenceable reliability parameter source or should a new, informed prior distribution be developed?

Summarized in Table 5-24 is an example of a strategy for developing plant-specific pipe failure rates. In a first step, a generic pipe failure rate for SW piping of diameter > 24" is calculated by using the applicable event population and exposure term data and with the plant-specific data subtracted. Next the new results are updated using the plant-specific input data for the same time period covered by the referenceable data source. This yields a plant-centric prior distribution, which is used to perform an update with new data.

[54] The plant-specific-specific exposure term is 84 ROY times 396 linear feet of NPS30 SW piping (safety-related and non-safety-related piping).

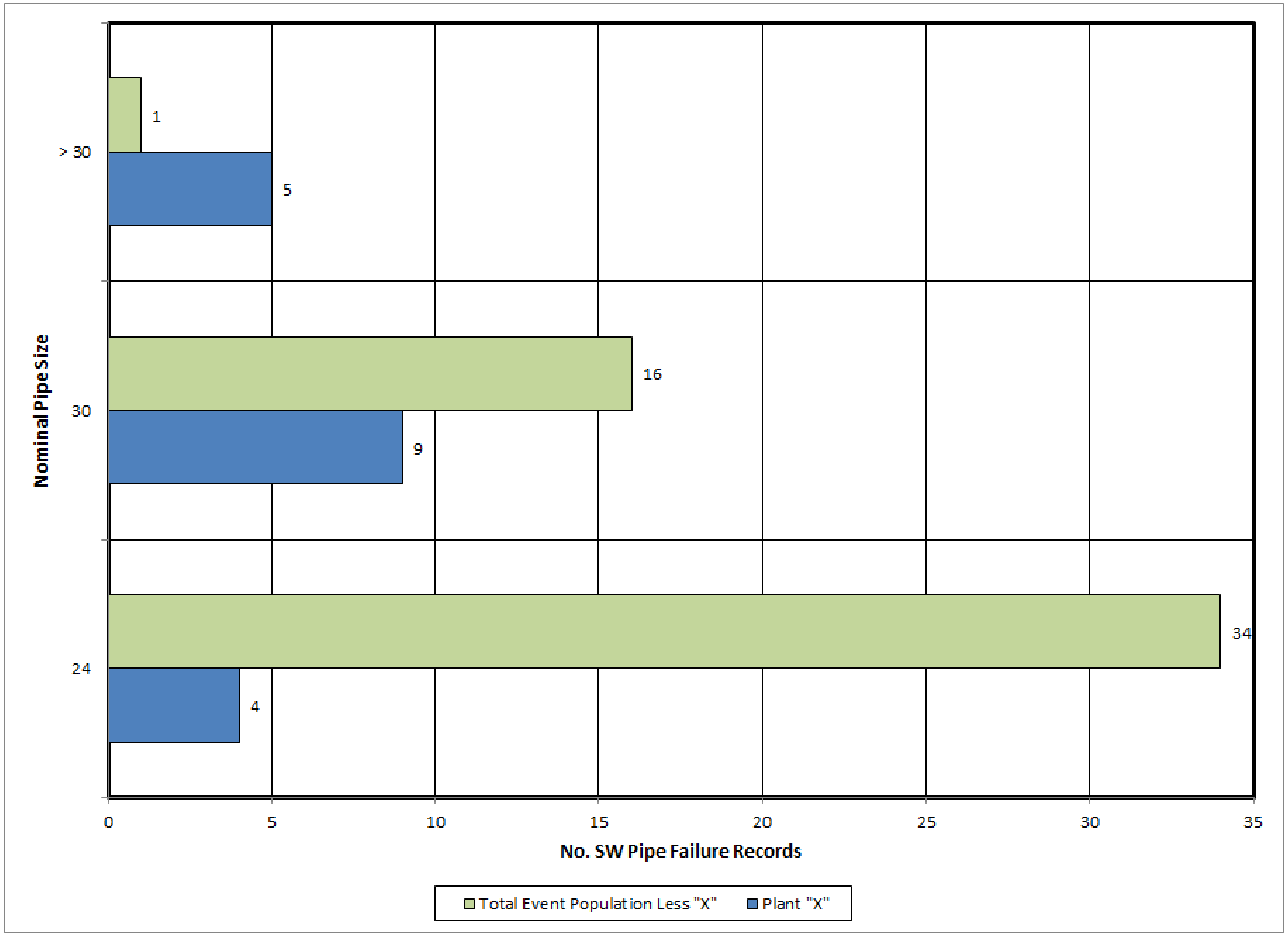


Figure 5-23: SW Pipe Failure Population – Brackish/Sea Water Environment

Table 5-25: Development of a Plant-Specific Failure Rate for In-Plant 30" Service Water Piping

Case	Approach	Plant-Specific Service Experience		Prior Distribution		Plant-Specific Posterior			
		# SW Pipe Failures	Exposure Term	Mean	RF	Mean	5th	50th	95th
1	Use the "semi-informed prior distribution" as reported in TR-1013141 (Pipe Rupture Frequencies for Internal Flooding PRA, R1, 2006) and update with the plant-specific experience (both units); pre 03/31/2009 data	8	33232 ROY-ft	2.00E-05	100	2.17E-04	1.01E-04	2.07E-04	3.63E-04
		This is the total NPS30 plant-specific experience thru July 2015							
2	Develop a site-specific prior by re-quantifying the failure rate in the selected referenceable data source with plant-specific data removed. That is, event population less "X" and same for the exposure term. A two-step process is applied:	2	4431	1.86E-05	1.24	1.92E-05	1.54E-05	1.91E-05	2.36E-05
	Step 1: Extract the plant-specific contribution to failure data & exposure term and calculate a new plant-neutral FR distribution.	This is the plant-specific experience post 03/31/2009							
	Step 2: Use results from Step 1 and perform Bayesian update with plant-specific data (thru 03/31/2009). This yields a plant-centric prior distribution to be used when calculating a new FR that accounts for new experience (03/2009 to 07/2015)								

5.7.2 Impact of DM Susceptibility Assumption on Results

This section outlines an approach to a bounding-type calculation of the projected failure rates and rupture frequencies of high-performance stainless steel (HP-SS) piping. The adopted technical approach builds on three different hypotheses about the corrosion-resistance of HP-SS piping. In view of the scarce service experience data on HP-SS piping, data screening rules are applied to the existing service experience data in order to obtain event population data sets that correspond to the three hypotheses; Table 5-25. Next, these data sets are input to the calculation framework used in Reference [75] **Error! Reference source not found.**to o btain a quantitative estimate of the factor of improvement in service water piping reliability that is obtainable by replacing the original piping material with HP-SS material.

Table 5-26: Probability Weights for Screening Hypotheses

MIC Resistance		
Hypothesis	**Lake / River Water Environment**	**Brackish / Sea Water Environment**
HMR – High MIC Resistance	0.65	1.0
	The realized level of MIC-resistance is assumed closely related to the quality of the welding, welder qualification and weld filler material selection. While unlikely, continued MIC susceptibility cannot be ruled out	Inverse relationship between salinity & bacterial growth – as salinity concentration goes up the MIC susceptibility diminishes, approaching zero. MMR & LMR hypotheses do not apply
MMR – Moderate MIC Resistance	0.25	N/A
	This hypothesis assumes a less-than-adequate welding procedure specification & implementation	
LMR – Low MIC Resistance	0.1	N/A
	This hypothesis assumes a poor welding procedure specification & implementation	

The formulation of different hypotheses about the corrosion resistance of HP-SS is based on results of corrosion experiments and the SW piping operating experience data, from which the following observations are made:

- In the brackish/sea water environment, the 300-Series austenitic stainless steel material has performed well with no recorded instance of erosion-corrosion damage, relatively few occurrences of crevice corrosion and pitting, and a few instances of MIC damage. The latter observation reflects the inverse relationship between salinity and bacterial growth; as salinity concentration goes up the bacterial growth is reduced or eliminated. Replacing the traditional SW piping materials with HP-SS would be expected to result in a significant reduction of degraded material conditions such as through-wall flaws.
- MIC is the dominant degradation mechanism in lake and river water environments with relatively high concentration of micro-organisms. It is noteworthy that the 300-Series austenitic stainless steel material has proven vulnerable to MIC. The rate by which MIC develops is a function of the surface treatment of the weld and weld heat affected zones, and the degree of sensitization (lowered chromium content). In other words, the quality of

the welding process influences the MIC susceptibility. With one important caveat, replacing the traditional SW piping materials with HP-SS would be expected to result in a significant reduction of degraded material conditions. With a less-than-adequate (LTA) welding process a MIC-vulnerability cannot be excluded, however.

Based on the above, the following hypotheses are defined regarding the projected, long-term performance of HP-SS material in the three different raw water environments:

1. High MIC-resistance (HMR Hypothesis). The influence of corrosion (including crevice corrosion, external corrosion, and pitting) and erosion-corrosion is screened out as the material properties of HP-SS would preclude them in the judgment of the authors. A conservative degradation propensity reduction factor of 90% is applied to MIC and galvanic corrosion in order to obtain an HMR-population of projected failure events given all plants that produced this experience operate with HP-SS material instead of CS or SS material. This is described as conservative because much higher reductions would be realistically expected.
2. Moderate MIC-resistance (MMR hypothesis). This hypothesis includes an assumption of poor welding procedure during fabrication and installation of the service water piping. The influence of corrosion (including crevice corrosion, external corrosion, and pitting) and erosion-corrosion is screened out. A very conservative degradation propensity reduction factor of 50% is applied to MIC and galvanic corrosion in order to obtain a MMR-population of projected failure events given that all plants that produced this experience operate with HP-SS material instead of CS or SS material.
3. Low MIC-resistance (LMR hypothesis). This hypothesis includes an assumption of very poor welding procedure during fabrication and installation of the service water piping. The influence of corrosion (including crevice corrosion, external corrosion, and pitting) and erosion-corrosion is screened out. An extremely conservative degradation propensity reduction factor of only 10% is applied MIC and galvanic corrosion in order to obtain a LMR-population of projected failure events given that all plants that produced this experience operate with HP-SS material instead of CS or SS material.

In applying the percentages in the screening process to the existing SW operating experience (Table 5-26) the resulting failure counts were converted to integers. The likelihood of galvanic corrosion of bi-metallic welds is not excluded from the bounding-type calculation. The underlying assumption being that not all non-piping passive components in the service water system are replaced with HP-SS material. By assigning probability weights to the three hypotheses, a pipe failure rate mixture distribution is developed that combine the impact of the different hypotheses. The projected event populations corresponding to each of the three hypotheses are summarized in Table 5-27 for a lake water environment.

Table 5-27: Operating Experience with Service Water Piping in U.S. PWR Plants

Pipe Size	Process Medium	Material	Total No. Failure Records	Corrosion - Crevice/Pitting / General/ External	Galvanic Corrosion	MIC	Erosion-Cavitation	Erosion-Corrosion	D&C	LC-FAT	VF	WH
ø ≤ 2"	Raw Water - Lake	Carbon Steel	55	15		18		9	2	1	8	2
		Stainless Steel	42		1	40					1	
2" < ø ≤ 4"		Carbon Steel	57	12		36		8	1			
		Stainless Steel	21			21						
4" < ø ≤ 10"		Carbon Steel	49	15		16	5	10	1			2
		Stainless Steel	16			16						
ø > 10"		Carbon Steel	60	22		13	10	15				
			300	64	1	160	15	42	4	1	9	4

Table 5-28: Projected Event Populations – Lake Water Environment

HMR Hypothesis											
Pipe Size	**Material**	**Total No. Failure Records**	**Corrosion - Crevice/Pitting/ General/External**	**Galvanic Corrosion**	**MIC**	**Erosion-Cavitation**	**Erosion-Corrosion**	**D&C**	**LC-FAT**	**Vibration-Fatigue**	**Water Hammer**
ø ≤ 2"	Carbon Steel	15	Screened Out		2		Screened Out	2	1	8	2
	Stainless Steel	6		1	4					1	
2" < ø ≤ 4"	Carbon Steel	5			4			1			
	Stainless Steel	2			2						
4" < ø ≤ 10"	Carbon Steel	10			2	5		1			2
	Stainless Steel	2			2						
ø > 10"	Carbon Steel	11			1	10					
		51	0	1	17	15	0	4	1	9	4
MMR Hypothesis											
Pipe Size	**Material**	**Total No. Failure Records**	**Corrosion - Crevice/Pitting/ General/External**	**Galvanic Corrosion**	**MIC**	**Erosion-Cavitation**	**Erosion-Corrosion**	**D&C**	**LC-FAT**	**Vibration-Fatigue**	**Water Hammer**
ø ≤ 2"	Carbon Steel	22	Screened Out		9		Screened Out	2	1	8	2
	Stainless Steel	22		1	20					1	
2" < ø ≤ 4"	Carbon Steel	19			18			1			
	Stainless Steel	11			11						
4" < ø ≤ 10"	Carbon Steel	16			8	5		1			2
	Stainless Steel	8			8						
ø > 10"	Carbon Steel	17			7	10					
		115	0	1	81	15	0	4	1	9	4

LMR Hypothesis											
Pipe Size	**Material**	**Total No. Failure Records**	**Corrosion - Crevice/Pitting/ General/External**	**Galvanic Corrosion**	**MIC**	**Erosion-Cavitation**	**Erosion-Corrosion**	**D&C**	**LC-FAT**	**Vibration-Fatigue**	**Water Hammer**
ø ≤ 2"	Carbon Steel	29	Screened Out		16		Screened Out	2	1	8	2
	Stainless Steel	38		1	36					1	
2" < ø ≤ 4"	Carbon Steel	33			32			1			
	Stainless Steel	19			19						
4" < ø ≤ 10"	Carbon Steel	22			14	5		1			2
	Stainless Steel	14			14						
ø > 10"	Carbon Steel	22			12	10					
		177	0	1	143	15	0	4	1	9	4

6. A DATA-DRIVEN MODEL OF PIPING RELIABILITY

A statistical model of piping reliability consists of two elements: 1) the pipe failure rate (λ) can be calculated using a Bayesian analysis technique to update the probability distribution representing our prior state of knowledge with the evidence from a pipe failure event database such as CODAP, and 2) the conditional probability that a failed condition produces a consequence of certain magnitude (e.g. peak through-wall flow rate or exceeding a flow rate threshold value). Typically multiple calculation cases have to be addressed in order to perform a comprehensive risk characterization of a failed pipe. These basic elements are elaborated in this section.

6.1 Technical Approach

The model used for relating failure rates and rupture frequencies uses the following simple model, which is widely used in piping reliability assessment. The pipe failure modes being considered cover all failures requiring repair or replacement, including wall thinning, cracks, leaks, and ruptures of various sizes up to and including complete severance of the pipe. The piping reliability is expressed as the product of a failure rate and a conditional failure probability (CFP). The reason for this approach is that in many cases there are insufficient data available to estimate rupture frequencies directly from the service data. There are sufficient data from which to estimate failure rates but not major structural failures. This approach also facilitates the use of different sources of information for the different parameters of interest. The conditional failure probabilities are estimated from a combination of limited data, engineering judgment, expert elicitation and structural reliability models. Finally, this approach makes it possible to divide the service data on failure rates into different cells to isolate different factors that are expected to influence failure rates such as system, pipe size, service conditions, and applicable degradation mechanisms.

Based on reviews of commercial nuclear power plant piping system service experience, all known pipe failures to date have resulted from the following damage and degradation mechanisms:

- Corrosion-fatigue (CF)
- Corrosion attack (COR), including microbiologically influenced corrosion (MIC), pitting, crevice corrosion
- Design and construction flaws and defects (D&C)
- Erosion corrosion (E/C)
- Erosion-cavitation (E-C)
- Flow-accelerated corrosion (FAC)
- Severe overloading caused by external impact or hydraulic transient (e.g. steam hammer or water hammer) (OVL)
- Stress corrosion cracking (SCC)
- Thermal fatigue (TF)
- Vibration-fatigue (VF)

These failure mechanisms include damage or degradation mechanisms that led to failure under normal service conditions (for example, thermal fatigue) and severe loading conditions such as water hammer that do not necessarily involve a degraded condition. Note that all failure modes that result in pipe repair are included in the failure rate and that all failures thus defined are regarded as precursors to rupture. The events counted as major structural failures are based on a specific failure mode definition, which is application specific.

The technical approach to estimating the frequency of a pipe break on the basis of operating experience data is expressed by Equations (6-1) and (6-2). The magnitude (i.e. size of a pressure boundary breach) is expressed by an equivalent break size (EBS) "*x*" and corresponding peak through-wall flow rate. The parameter *x* is treated as a discrete variable representing different equivalent break-size ranges.

$$F(IE_x) = \sum_i m_i \rho_{ix} \tag{6-1}$$

$$\rho_{ix} = \sum_k \lambda_{ik} P(R_x|F_{ik}) I_{ik} \tag{6-2}$$

Where:

$F(IE_x)$ =	Frequency of pipe break of size *x*, per reactor operating-year, subject to epistemic (or state-of-knowledge) uncertainty calculated via Monte Carlo simulation.
m_i =	Number of pipe welds (or fittings, segments or inspection locations of type *i*; each type determined by pipe size, weld type, applicable damage or degradation mechanisms, and inspection status (leak test and non-destructive examination).
ρ_{ix} =	Frequency of rupture of component type *i* with break size *x*, subject to epistemic uncertainty calculated via Monte Carlo simulation.
λ_{ik} =	Failure rate per "location-year" for pipe component type *i* due to failure mechanism *k*, subject to epistemic uncertainty, Equation 6-3 below. In this analysis the failure rate is calculated on the basis of per linear meter and reactor operating year.
$P(R_x\|F_{ik})$ =	For leak-before-break (LBB) piping, the conditional rupture probability (CRP) of size *x* given failure of pipe component type *i* due to damage or degradation mechanism *k*, subject to epistemic uncertainty. This parameter may be determined on the basis of probabilistic fracture mechanics, expert elicitation or service experience insights.
I_{ik} =	Integrity (RIM) management factor for weld type *i* and failure mechanism *k*, subject to epistemic uncertainty. This parameter is not explicitly addressed in this example, however.

For a point estimate of the failure rate of piping component type *i* and degradation mechanism *k*:

$$\lambda_{ik} = \frac{n_{ik}}{\tau_{ik}} = \frac{n_{ik}}{f_{ik} N_i T_i} \tag{6-3}$$

Where:

n_{ik} =	Number of failures in pipe component of type *i* due to degradation mechanism *k*. The component boundary used in defining exposure terms is a function of the susceptibility to certain damage or degradation mechanisms.
τ_{ik} =	Component exposure population for welds of type *i* susceptible to degradation mechanism *k*.
f_{ik} =	Estimate of the fraction of the component exposure population for piping component type *i* that is susceptible to degradation mechanism *k*, estimated from results of a formal degradation mechanism evaluation.
N_i =	Estimate of the average number of pipe components of type *i* per reactor in the reactor operating years of exposure for the data query used to determine n_{ik}. Determined from isometric drawings reviews for a population of plants and expert knowledge of degradation mechanisms.

T_i = Total exposure in reactor-years for the data collection for component type *i*. CODAP event database provides the number of reactor operating years that produced the operating experience data.

In the development of Bayesian uncertainty distributions for the above reliability parameters [82][83], prior distributions are developed for the parameters λ_{ik} and $P(R_x|F_{ik})$. As in standard Bayesian updating, these distributions are updated using evidence from the failure and exposure data. The failure rate exposure terms, which are identified in the denominator on the right-hand side of Equation 6-3, are uncertain because the estimates of pipe length per plant are based on a small subset of plants and must be extrapolated to represent the average across the entire industry represented in the data set for the failure counts in the numerator of this equation. The reactor years of service experience has a relatively small amount of uncertainty; therefore, this uncertainty is dominated by the uncertainty in the estimated pipe lengths per reactor. In this process, the uncertainty is treated by adopting three hypotheses about the values of the exposure terms, which requires three Bayesian updates for each failure rate. The resulting posterior distributions for each parameter on the right-hand side of Equation (6-2) are then combined using Monte Carlo sampling to obtain uncertainty distributions for the pipe rupture frequencies.

6.2 Microsoft® Excel Workbook Example

Section 5 describes how to organize a piping reliability analysis task. Key considerations involve the definition of the evaluation boundary (i.e. system or system(s) to consider, pipe size(s) and specific failure locations) and the calculation cases (i.e. combinations of pipe size, damage/degradation mechanism and consequence). Special statistical evaluations to develop appropriate a priori distribution parameters should be an integral part of a piping reliability analysis on the basis of operating experience data. Details on this topic can be found in NUREG/CR-6823 [84]. Furthermore, engineering calculations may have to be performed to establish relationships between the size of a through-wall defect and the resulting leak/flow rate and zone of influence (ZOI). Guidance for pipe failure consequence analysis is found in industry guidelines such as EPRI 1019194 (Section 6 and Appendix C) [85] and ASME/ANS PRA Standard RA-Sb-2013 (Part 3) [86]. Thermal-hydraulic computer codes such as GOTHIC [55] may also be used to support detailed pipe break consequence analyses. A Bayes' methodology may be to estimate failure rates and rupture frequencies to support the risk characterization of degraded or failed piping components.

6.3 Calculation Procedure

The Bayes' methodology is illustrated in Figure 6-1. This methodology explicitly accounts for the plant-to-plant variability in piping system design and layout, including weld populations, degradation susceptibility, pipe lengths, and the population of pipe fittings (e.g., bends, elbows and tees).

The approach taken to address the uncertainty in the piping component population is to apply a Bayes' posterior weighting procedure. A set of three estimates is obtained for the susceptible component population exposure, one for the best estimate, one for an upper bound estimate and one for a lower bound estimate. For each of these three estimates the number of pipe failures and the exposure population estimate is used to perform a Bayes' update of a generic prior distribution. Then a posterior weighting procedure is applied to synthesize the results of these three Bayes' updates into a single composite uncertainty distribution for the failure rate. The posterior weighting procedure is implemented in an Excel spreadsheet format

[55] For details, see http://www.numerical.com/gothic.php

with the Oracle Crystal Ball (or equivalent software[56]). While there is no "single-format-fits-all" worksheet format, described here are the basic elements. In summary, the pipe failure rate estimation involves the following steps:

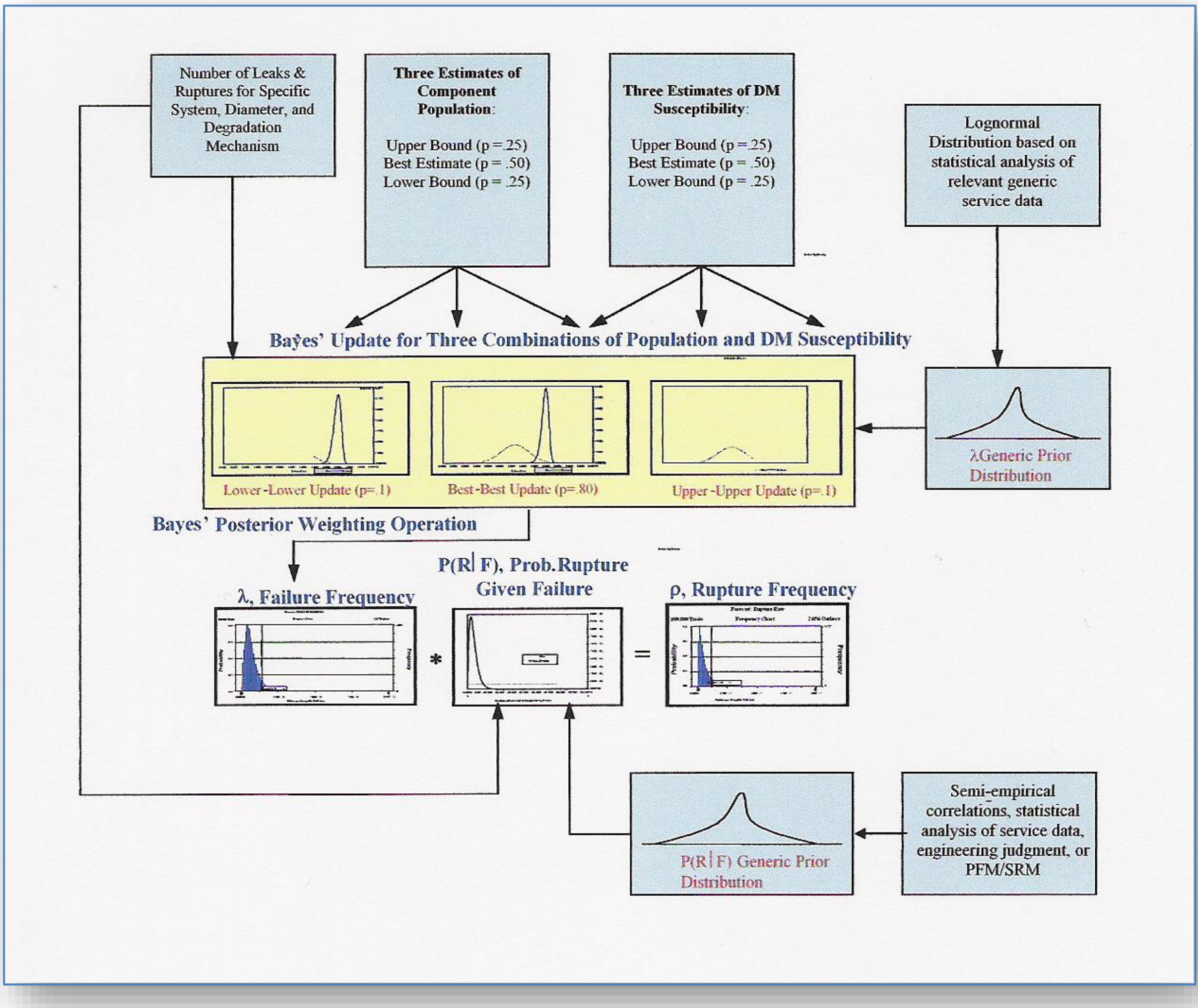


Figure 6-1: Calculation Procedure[57]

- For each calculation case, identify the applicable prior failure rate distribution to be applied together with the event population to be input to the Bayesian update.
- Develop the exposure term that produced the event population identified in Step 1. Illustrated in Figure 6-2 is an example of how to capture the plant-to-plant piping component population variability. In the given example, weld population data was obtained from a sample of ten (10) commercial PWR plants of different NSSS design vintage.
- The standalone software "R-DAT Plus" (or equivalent software) is used to calculate pipe failure rates for each unique combination of pipe size, damage/degradation mechanism and exposure term (low-medium-high).[58]

[56] For example the @Risk software product (http://www.palisade.com/products.asp); Chrystal Ball and @Risk computer software output files are compatible.

[57] Adapted from Reliability Engineering & System Safety, 86:227-246 (2004) [87].

PIPExp Database

Reactor Operating Years	
PWR MS Class 2	ROYs
2-Loop	1761
3-Loop	2750
4-Loop	2162
Total	6673

	> 10"			≤ 10"		
Plant	3-Loop	2-Loop	4-Loop	3-Loop	2-Loop	4-Loop
R3	96	–	–		–	–
K1	117	–	–	45	–	–
K2	121	–	–	45	–	–
VC	108	–	–	36	–	–
REG	–	64	–	–	21	–
K	–	72	–	–	26	–
Br1	–	–	139	–	–	88
Br2	–	–	130	–	–	90
By1	–	–	119	–	–	115
By2	–	–	99	–	–	112
Average	110.5	68	121.75	42	23.5	101.25
Min	96	64	99	36	21	88
Max	121	72	139	45	26	115

	Welds per Loop					
	> 10"			≤ 10"		
Average	36.8	34	30.4	14	11.75	25.3
Min	32.0	32	24.8	12	10.5	22.0
Max	40.3	36	34.8	15	13	28.8

		> 10"	≤ 10"
Weld-Years	Low	604091.7	302896.7
	Medium	689316.8	347565.4
	High	756107.2	386278.3

Figure 6-2: Assessment of Exposure Term Variability

- In 'R-DAT Plus', define a project with "subsystems." Each subsystem representing a Calculation Case serve to facilitate the failure rate and CFP calculations; Figure 6-3. One set of failure rate parameters per exposure term assumption (low-medium-high). A CFP posterior distribution is calculated for each of a predefined set of pipe failure consequence categories; e.g. in terms of equivalent break size (EBS). An example of R-DAT input data is given in Table 6-1. For each case, select an appropriate a priori distribution; examples of a priori distributions are found in Tables 6-2 and 6-3.
- Export the 'R-DAT Plus' results (Figures 6-4 and 6-5) to the Excel workbook (Figure 6-6) to facilitate the posterior weighting procedure component-specific "rupture mode" frequency calculations. Use 'Export' which creates a *.txt or *csv file for input to an Excel spreadsheet (see below).

[58] Other software products exist for Bayesian statistics; e.g. JASP, winBUGS. Also, Appendix J of the Center for Chemical Process Safety text "Guidelines for Chemical Process Quantitative Risk Analysis" (ISBN 0-8169-0402-2) presents a procedure for combining generic and plant-specific reliability data.

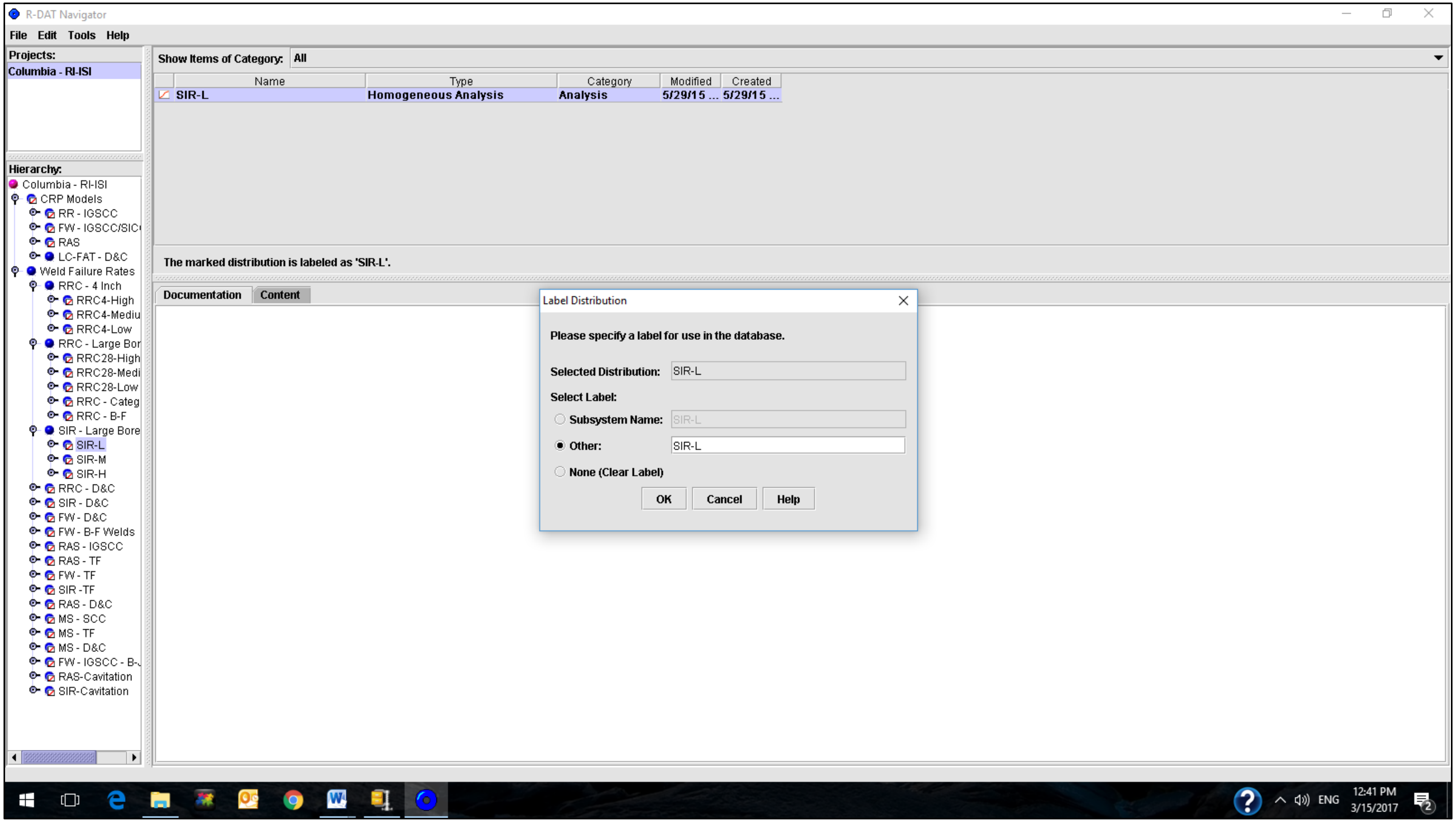
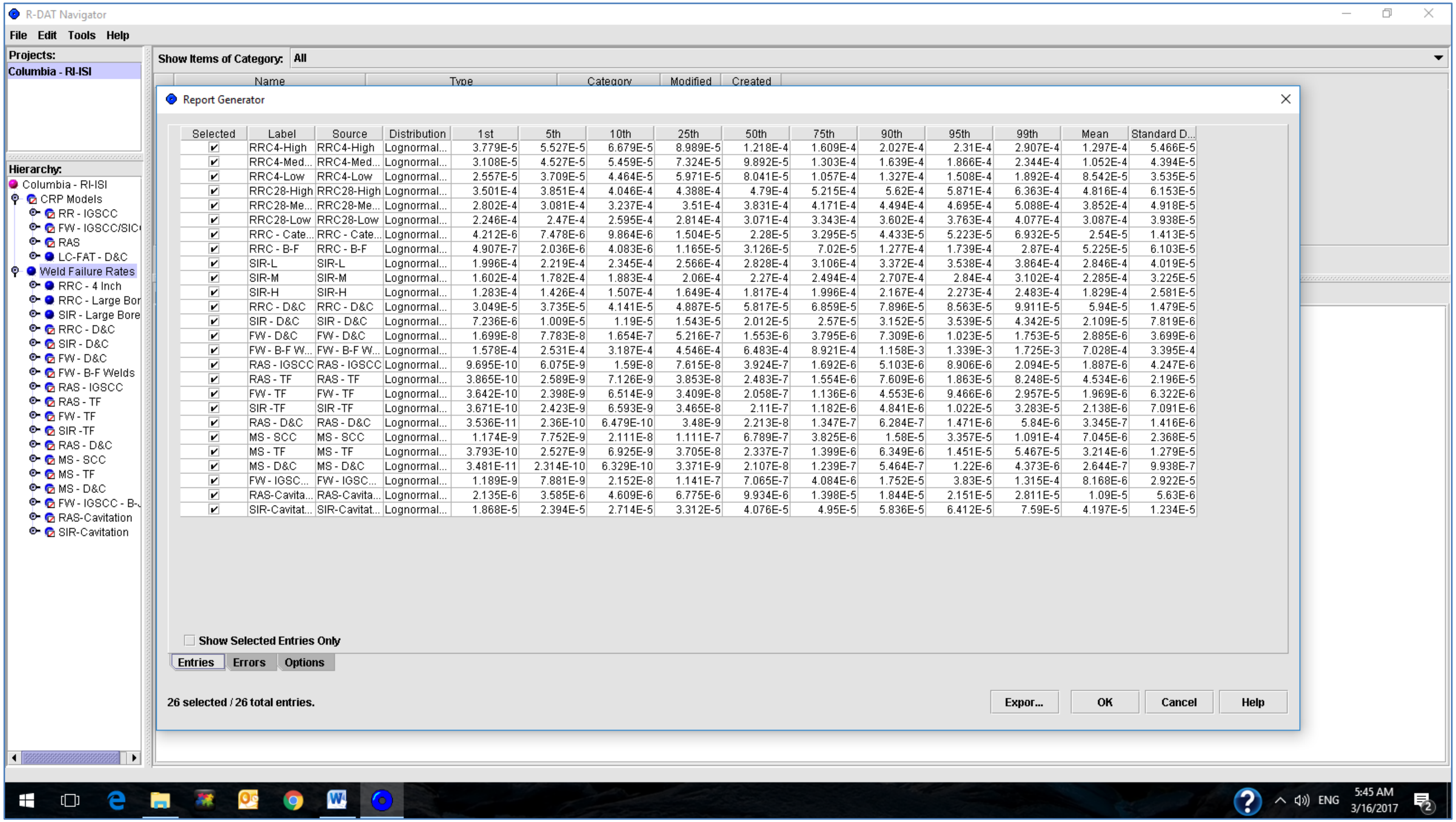


Figure 6-3: 'R-DAT Plus' Navigator



Selected	Label	Source	Distribution	1st	5th	10th	25th	50th	75th	90th	95th	99th	Mean	Standard D...
✔	RRC4-High	RRC4-High	Lognormal...	3.779E-5	5.527E-5	6.679E-5	8.989E-5	1.218E-4	1.609E-4	2.027E-4	2.31E-4	2.907E-4	1.297E-4	5.466E-5
✔	RRC4-Med...	RRC4-Med...	Lognormal...	3.108E-5	4.527E-5	5.459E-5	7.324E-5	9.892E-5	1.303E-4	1.639E-4	1.866E-4	2.344E-4	1.052E-4	4.394E-5
✔	RRC4-Low	RRC4-Low	Lognormal...	2.557E-5	3.709E-5	4.464E-5	5.971E-5	8.041E-5	1.057E-4	1.327E-4	1.508E-4	1.892E-4	8.542E-5	3.535E-5
✔	RRC28-High	RRC28-High	Lognormal...	3.501E-4	3.851E-4	4.046E-4	4.388E-4	4.79E-4	5.215E-4	5.62E-4	5.871E-4	6.363E-4	4.816E-4	6.153E-5
✔	RRC28-Me...	RRC28-Me...	Lognormal...	2.802E-4	3.081E-4	3.237E-4	3.51E-4	3.831E-4	4.171E-4	4.494E-4	4.695E-4	5.088E-4	3.852E-4	4.918E-5
✔	RRC28-Low	RRC28-Low	Lognormal...	2.246E-4	2.47E-4	2.595E-4	2.814E-4	3.071E-4	3.343E-4	3.602E-4	3.763E-4	4.077E-4	3.087E-4	3.938E-5
✔	RRC - Cate...	RRC - Cate...	Lognormal...	4.212E-6	7.478E-6	9.864E-6	1.504E-5	2.28E-5	3.295E-5	4.433E-5	5.223E-5	6.932E-5	2.54E-5	1.413E-5
✔	RRC - B-F	RRC - B-F	Lognormal...	4.907E-7	2.036E-6	4.083E-6	1.165E-5	3.126E-5	7.02E-5	1.277E-4	1.739E-4	2.87E-4	5.225E-5	6.103E-5
✔	SIR-L	SIR-L	Lognormal...	1.996E-4	2.219E-4	2.345E-4	2.566E-4	2.828E-4	3.106E-4	3.372E-4	3.538E-4	3.864E-4	2.846E-4	4.019E-5
✔	SIR-M	SIR-M	Lognormal...	1.602E-4	1.782E-4	1.883E-4	2.06E-4	2.27E-4	2.494E-4	2.707E-4	2.84E-4	3.102E-4	2.285E-4	3.225E-5
✔	SIR-H	SIR-H	Lognormal...	1.283E-4	1.426E-4	1.507E-4	1.649E-4	1.817E-4	1.996E-4	2.167E-4	2.273E-4	2.483E-4	1.829E-4	2.581E-5
✔	RRC - D&C	RRC - D&C	Lognormal...	3.049E-5	3.735E-5	4.141E-5	4.887E-5	5.817E-5	6.859E-5	7.896E-5	8.563E-5	9.911E-5	5.94E-5	1.479E-5
✔	SIR - D&C	SIR - D&C	Lognormal...	7.236E-6	1.009E-5	1.19E-5	1.543E-5	2.012E-5	2.57E-5	3.152E-5	3.539E-5	4.342E-5	2.109E-5	7.819E-6
✔	FW - D&C	FW - D&C	Lognormal...	1.699E-8	7.783E-8	1.654E-7	5.216E-7	1.553E-6	3.795E-6	7.309E-6	1.023E-5	1.753E-5	2.885E-6	3.699E-6
✔	FW - B-F W...	FW - B-F W...	Lognormal...	1.578E-4	2.531E-4	3.187E-4	4.546E-4	6.483E-4	8.921E-4	1.158E-3	1.339E-3	1.725E-3	7.028E-4	3.395E-4
✔	RAS - IGSCC	RAS - IGSCC	Lognormal...	9.695E-10	6.075E-9	1.59E-8	7.615E-8	3.924E-7	1.692E-6	5.103E-6	8.906E-6	2.094E-5	1.887E-6	4.247E-6
✔	RAS - TF	RAS - TF	Lognormal...	3.865E-10	2.589E-9	7.126E-9	3.853E-8	2.483E-7	1.554E-6	7.609E-6	1.863E-5	8.248E-5	4.534E-6	2.196E-5
✔	FW - TF	FW - TF	Lognormal...	3.642E-10	2.398E-9	6.514E-9	3.409E-8	2.058E-7	1.136E-6	4.553E-6	9.466E-6	2.957E-5	1.969E-6	6.322E-6
✔	SIR -TF	SIR -TF	Lognormal...	3.671E-10	2.423E-9	6.593E-9	3.465E-8	2.11E-7	1.182E-6	4.841E-6	1.022E-5	3.283E-5	2.138E-6	7.091E-6
✔	RAS - D&C	RAS - D&C	Lognormal...	3.536E-11	2.36E-10	6.479E-10	3.48E-9	2.213E-8	1.347E-7	6.284E-7	1.471E-6	5.84E-6	3.345E-7	1.416E-6
✔	MS - SCC	MS - SCC	Lognormal...	1.174E-9	7.752E-9	2.111E-8	1.111E-7	6.789E-7	3.825E-6	1.58E-5	3.357E-5	1.091E-4	7.045E-6	2.368E-5
✔	MS - TF	MS - TF	Lognormal...	3.793E-10	2.527E-9	6.925E-9	3.705E-8	2.337E-7	1.399E-6	6.349E-6	1.451E-5	5.467E-5	3.214E-6	1.279E-5
✔	MS - D&C	MS - D&C	Lognormal...	3.481E-11	2.314E-10	6.329E-10	3.371E-9	2.107E-8	1.239E-7	5.464E-7	1.22E-6	4.373E-6	2.644E-7	9.938E-7
✔	FW - IGSC...	FW - IGSC...	Lognormal...	1.189E-9	7.881E-9	2.152E-8	1.141E-7	7.065E-7	4.084E-6	1.752E-5	3.83E-5	1.315E-4	8.168E-6	2.922E-5
✔	RAS-Cavita...	RAS-Cavita...	Lognormal...	2.135E-6	3.585E-6	4.609E-6	6.775E-6	9.934E-6	1.398E-5	1.844E-5	2.151E-5	2.811E-5	1.09E-5	5.63E-6
✔	SIR-Cavitat...	SIR-Cavitat...	Lognormal...	1.868E-5	2.394E-5	2.714E-5	3.312E-5	4.076E-5	4.95E-5	5.836E-5	6.412E-5	7.59E-5	4.197E-5	1.234E-5

Figure 6-4: 'R-DAT' Sample Output File

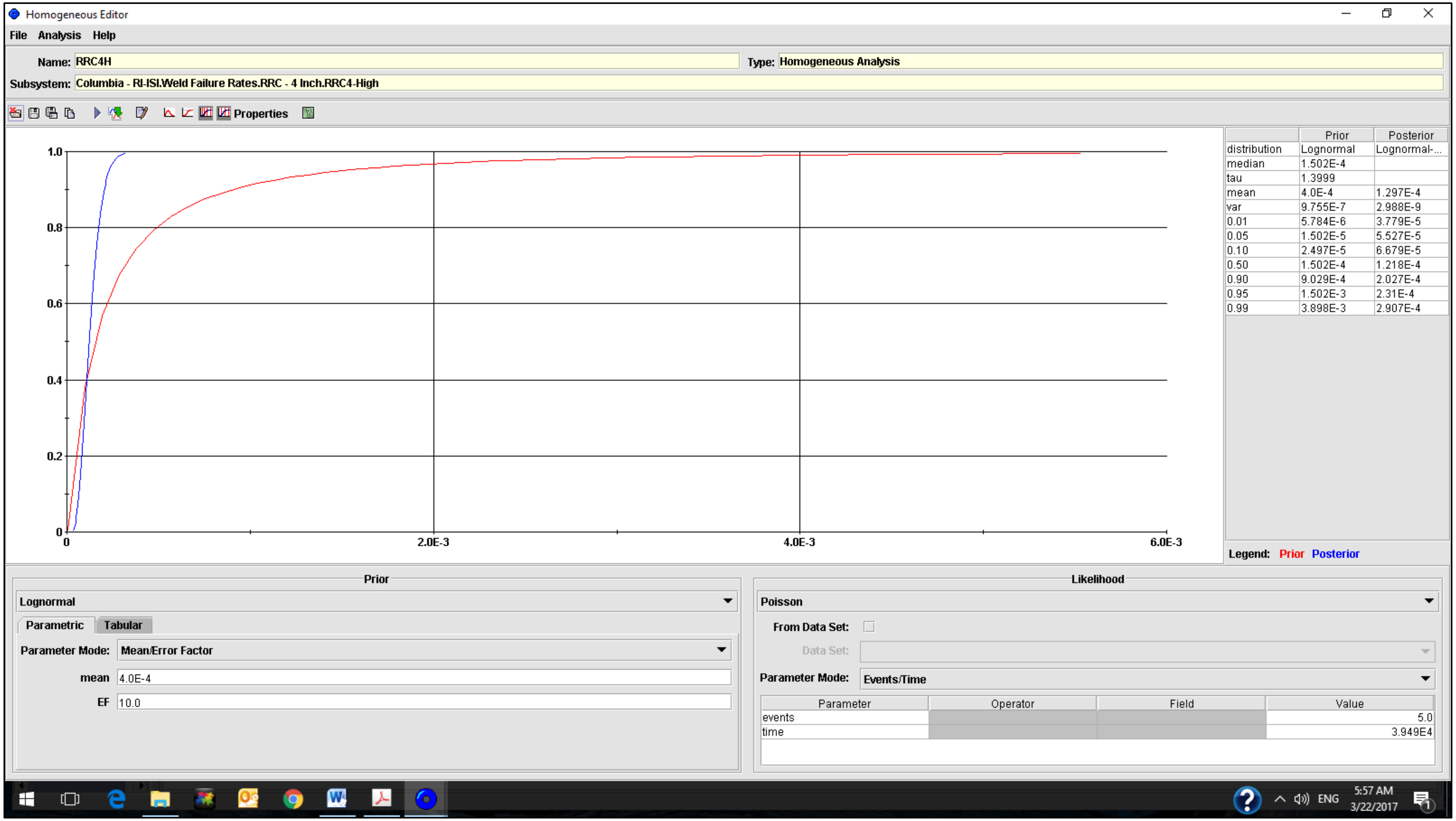
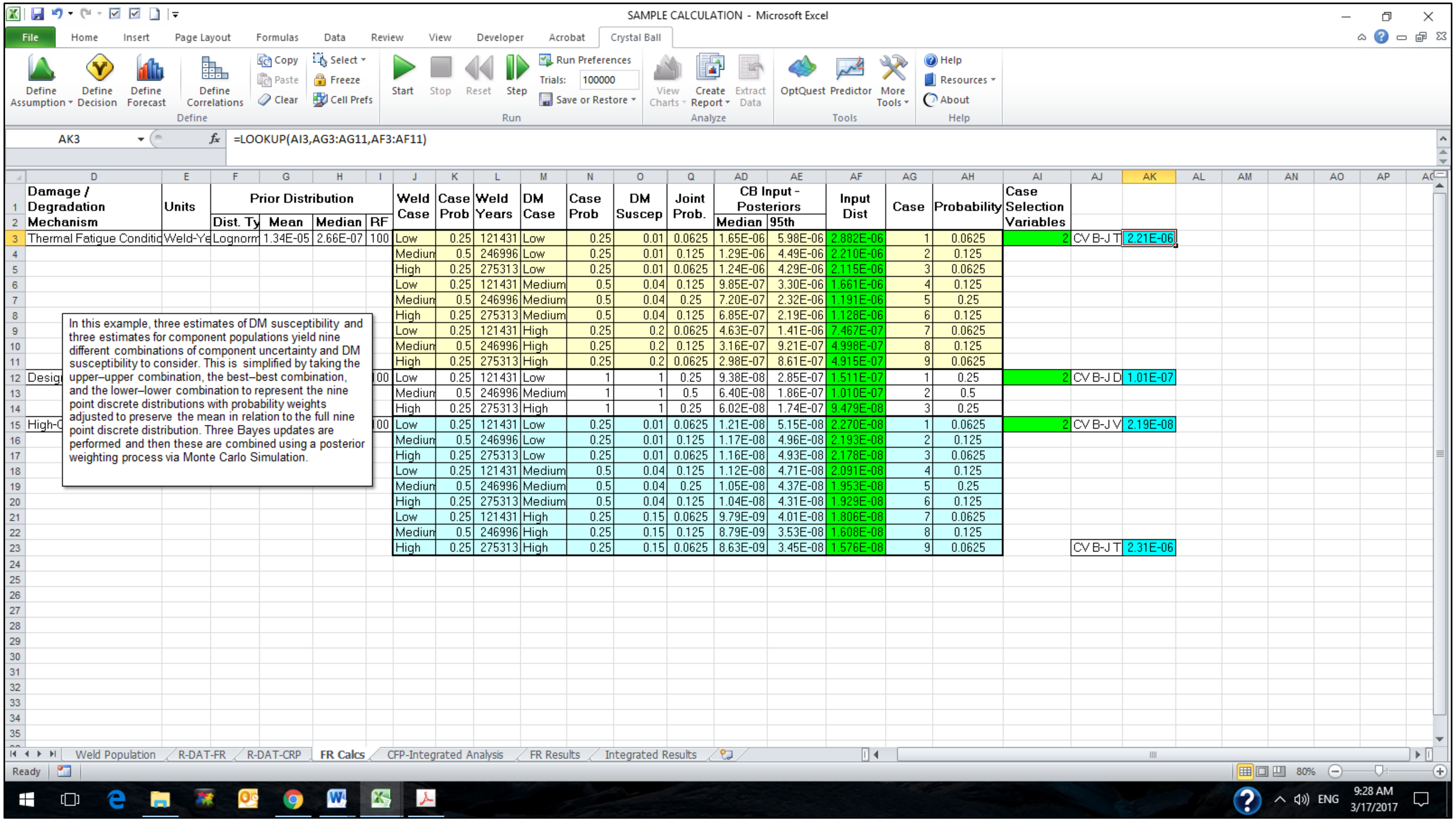


Figure 6-5: 'R-DAT' Prior & Posterior Distributions



	D	E	F	G	H	I	J	K	L	M	N	O	Q	AD	AE	AF	AG	AH	AI	AJ	AK
1	Damage / Degradation	Units	Prior Distribution				Weld Case	Case Prob	Weld Years	DM Case	Case Prob	DM Suscep	Joint Prob.	CB Input - Posteriors		Input Dist	Case	Probability	Case Selection Variables		
2	Mechanism		Dist. Ty	Mean	Median	RF								Median	95th						
3	Thermal Fatigue Conditic	Weld-Ye	Lognorm	1.34E-05	2.66E-07	100	Low	0.25	121431	Low	0.25	0.01	0.0625	1.65E-06	5.98E-06	2.882E-06	1	0.0625	2	CV B-J T	2.21E-06
4							Medium	0.5	246996	Low	0.25	0.01	0.125	1.29E-06	4.49E-06	2.210E-06	2	0.125			
5							High	0.25	275313	Low	0.25	0.01	0.0625	1.24E-06	4.29E-06	2.115E-06	3	0.0625			
6							Low	0.25	121431	Medium	0.5	0.04	0.125	9.85E-07	3.30E-06	1.661E-06	4	0.125			
7							Medium	0.5	246996	Medium	0.5	0.04	0.25	7.20E-07	2.32E-06	1.191E-06	5	0.25			
8							High	0.25	275313	Medium	0.5	0.04	0.125	6.85E-07	2.19E-06	1.128E-06	6	0.125			
9							Low	0.25	121431	High	0.25	0.2	0.0625	4.63E-07	1.41E-06	7.467E-07	7	0.0625			
10							Medium	0.5	246996	High	0.25	0.2	0.125	3.16E-07	9.21E-07	4.998E-07	8	0.125			
11							High	0.25	275313	High	0.25	0.2	0.0625	2.98E-07	8.61E-07	4.915E-07	9	0.0625			
12	Desig					00	Low	0.25	121431	Low	1	1	0.25	9.38E-08	2.85E-07	1.511E-07	1	0.25	2	CV B-J D	1.01E-07
13							Medium	0.5	246996	Medium	1	1	0.5	6.40E-08	1.86E-07	1.010E-07	2	0.5			
14							High	0.25	275313	High	1	1	0.25	6.02E-08	1.74E-07	9.479E-08	3	0.25			
15	High-C					00	Low	0.25	121431	Low	0.25	0.01	0.0625	1.21E-08	5.15E-08	2.270E-08	1	0.0625	2	CV B-J V	2.19E-08
16							Medium	0.5	246996	Low	0.25	0.01	0.125	1.17E-08	4.96E-08	2.193E-08	2	0.125			
17							High	0.25	275313	Low	0.25	0.01	0.0625	1.16E-08	4.93E-08	2.178E-08	3	0.0625			
18							Low	0.25	121431	Medium	0.5	0.04	0.125	1.12E-08	4.71E-08	2.091E-08	4	0.125			
19							Medium	0.5	246996	Medium	0.5	0.04	0.25	1.05E-08	4.37E-08	1.953E-08	5	0.25			
20							High	0.25	275313	Medium	0.5	0.04	0.125	1.04E-08	4.31E-08	1.929E-08	6	0.125			
21							Low	0.25	121431	High	0.25	0.15	0.0625	9.79E-09	4.01E-08	1.806E-08	7	0.0625			
22							Medium	0.5	246996	High	0.25	0.15	0.125	8.79E-09	3.53E-08	1.608E-08	8	0.125			
23							High	0.25	275313	High	0.25	0.15	0.0625	8.63E-09	3.45E-08	1.576E-08	9	0.0625		CV B-J T	2.31E-06

Figure 6-6: Posterior Weighting Procedure

Table 6-1: Sample R-DAT Input Data

Calculation Case (R-DAT ID)	Comment	Pipe Failure Count (1)	Exposure Term [Weld-Years]		
			Low	Medium	High
FW - D&C		1	137920	172400	215500
FW - B-F - IGSCC	BWR-5/6 field experience	5	4733	5916	7395
FW - TF		0	19216	24020	30025
MS - D&C		0	9232	11540	14425
MS - IGSCC		0	4733	5916	7395
MS - TF		0	6460	8075	10094
RAS - D&C		0	52064	65080	81350
RAS - IGSCC		0	52064	65080	81350
RAS - TF		0	2603	3254	4068
RRC4 - IGSCC	Base case - 2 < Ø ≤ 4" - post-1988 field experience	5	39757	49696	62120
RRC - Large Bore - IGSCC	Base case - > 4" - post-1988 field experience - AISI 304/316/316NG	61	126381	157976	197470
RRC - Category A - IGSCC	BWR-5/6 field experience - Japanese BWR-5/6 IGSCC data screened out	3	97811	122264	152830
RRC - B-F - IGSCC	BWR-5/6 dissimilar metal welds	1	11836	14795	18494
RRC - D&C	BWR - RRC - all plants	17	215680	269600	337000
SIR - D&C		8	271120	338900	423625
SIR - IGSCC	Post-1988 field experience	50	271120	338900	423625
SIR - TF		0	16184	20230	25288
(1)	From CODAP database				

Table 6-2: Prior Distribution Parameters Example #2

DM	Prior Failure Rate Distribution			Applicability	Technical Basis
	Mean	RF	Type		
FAC - Single-Phase Flow	1.22E-04	100	LN	CS - FAC-susceptible locations - can be, elbow, reducer, tee, locations immediately downstream flow-control valves, and other flow constriction.	The prior distribution accounts for post-1989 service experience and assumes that a managed FAC inspection program is in place (e.g., CHECHWORKS, COMSY, BRT-CICERO). The exposure term is derived from FAC program plans that identify all inspection locations that are monitored for FAC wear. For a commercial PWR, the monitored pipe locations are on the order of 320 locations.
LDIE liquid droplet impingement erosion	2.44E-05	100	LN	LAS or SS bends and elbows	According to R&D by French utility EDF, FAC resistance is highly chromium dependent. Based on experimental data (c.f. Trevin, S. in Nuclear Corrosion Science and Engineering, Woodhead Publishing, 2012), for LAS with 0.3% chromium content the relative FAC rate is about 0.2 compared with carbon steel (e.g. ASTM A-106 Grade B).
TASCS	2.27E-03	100	LN	FW piping inside containment - restricted to pipe weld-to-SG nozzle, and the pipe section next to the first elbow. For NUSCALE there are two potentially TASCS-susceptible locations.	The prior accounts for the pre-2000 service experience.
LC-FAT	1.80E-06	100	LN	All welds for which no active DM susceptibility is identified	Engineering judgment - review of available service experience data on FW large-bore piping.
IGSCC/TGSCC	5.10E-05	100	LN	Base metal failure; cold-worked piping	Empirical prior based on review of Code Class 2 field experience from BWR & PWR plants. Total of 163 failure records in database (January 1970 through July 2014)
SH - MS System	1.50E-04	100	LN	SH susceptible locations	The unit is "Failures/Reactor-Yr." - the given mean value represents the frequency of a "severe steam hammer of sufficient magnitude to challenge a piping pressure boundary." There has been a single SH-event causing a complete circumferential through-wall crack of a MS line branch connection. It occurred in 1970 during hot functional testing at H.B. Robinson-1.

DM	Prior Failure Rate Distribution			Applicability	Technical Basis
WH - FW System	9.00E-04	100	LN	WH susceptible locations	The unit is "Failures/Reactor-Yr." - the given mean value represents the frequency of a "severe water hammer of sufficient magnitude to challenge a piping pressure boundary. The basis for given value is service experience involving FAC susceptible carbon steel piping.
VF	3.70E-04	100	LN	Small-bore branch connections	Derived from service experience data - applies to branch connections ≤ NPS3. Total of 361 failures in small-bore Non-Code piping are attributed to VF. Assume 100 susceptible locations per plant. The service experience represents 9750 reactor-years of operation.
The Range Factor, RF = (95^{th} Percentile/5^{th} Percentile)$^{0.5}$					

- Post processing of results. The calculation procedure provides results in the form of pipe failure rates by pipe size, component type and degradation mechanism. The pipe rupture frequencies are calculated in terms of cumulative rupture frequencies versus different equivalent break sizes (EBS); for example, 0.5”, 1.5”, 3”, 6” and 14”.

- Open Excel and create three linked tabs to form a workbook: 1) R-DAT-FR, 2) FR Calcs., and 3) CFP. On the ‘R-DAT-FR’ sheet, import (or paste) the ‘R-DAT’ output-file. On the ‘FR Calcs.’ sheet organize the pipe failure rate calculations as shown in Figures 6 and 7. Note that the calculation procedure explicitly addresses the uncertainties in the failure population data as well as in the exposure term data. The Oracle Crystal Ball (CB) add-in software facilitates the posterior weighting process and through Monte Carlo simulation generates a single composite distribution for the pipe failure rate. A Monte Carlo merge technique is used to develop a distribution that has a mean value equal to a weighted average of the three (low – medium – high), while maintaining the full range of values representing the three input distributions (refer to Figures 6-4 and 6-5). In Figure 6-7, the three distributions are assumed to be lognormal, where the mean value and the ratio of the 95%-tile and the 5%-tile are preserved. The Crystal Ball software includes a statistical analysis package with a complete suite of statistical distribution functions. The program also allows for user-defined distributions; i.e. discrete probability distributions.

- On the ‘CFP’ tab, organize the conditional failure probability calculation by consequence category; Figure 6-8. A selected calculation strategy may involve a suite of different consequence categories; Table 6-3. Different CFP analysis strategies may involve the use of Beta-distribution (for details see Section 10) or a mixture distribution derived from an expert elicitation process [71].

Table 6-3: An Example of CFP Distribution Parameters

Category	EBS	Mean	5%-tile	Median	95%-tile	RF
Thermal Fatigue						
1	0.5	1.70E-02	5.77E-03	1.02E-02	1.80E-02	1.8
2	1.5	2.88E-03	5.27E-04	2.10E-03	8.39E-03	4
3	3	6.40E-04	1.13E-04	4.53E-04	1.81E-03	4
4	6	9.67E-05	1.03E-05	5.67E-05	3.11E-04	5.5
5	14	2.27E-05	2.43E-06	1.33E-05	7.30E-05	5.5
FAC - Single-Phase Flow						
1	0.5	1.27E-01	1.57E-02	7.87E-02	3.93E-01	5.0
2	1.5	3.05E-02	3.78E-03	1.89E-02	9.45E-02	5.0
3	3	1.53E-02	5.74E-04	5.74E-03	5.74E-02	10.0
4	6	5.09E-03	2.00E-05	6.00E-04	1.80E-02	30.0
5	14	2.54E-03	9.99E-06	3.00E-04	8.99E-03	30.0

6.4 An Alternative Bayesian Calculation Procedure

In Bayesian statistics, if the posterior distributions $p(\theta|x)$ are in the same family as the prior probability distribution $p(\theta)$, the prior and posterior are then called conjugate distributions, and the prior is called a conjugate prior for the likelihood function. In piping reliability analysis, if the failure data are independent and identically distributed Poisson (λ), then a Gamma (α,β) prior on λ is a conjugate prior. Choosing a Gamma prior over the mean will ensure that the posterior distribution is also a gamma distribution. Using the conjugate characteristics makes it possible to perform Bayesian updates in Microsoft® Excel as follows:

- Select a Gamma prior over the mean; e.g. 1E-06 (mean) and $\alpha = 0.5$ and $\beta = 50000$, which corresponds to slightly non-informative prior.

- Define the new evidence (failure population) from an operating experience database; i.e., number of failures and the corresponding exposure term (e.g. linear feet of piping and the reactor operating years that produced the pipe failures).
- Calculate the posterior α and β parameters:

  $\alpha_{Posterior} = \alpha_{Prior}$ + New Evidence (#Failures)
  $\beta_{Posterior} = \beta_{Prior}$ + New Evidence (#ft.ROYs)
- In Oracle Crystal Ball, define the new "forecast" using the above described Monte Carlo Merge technique.

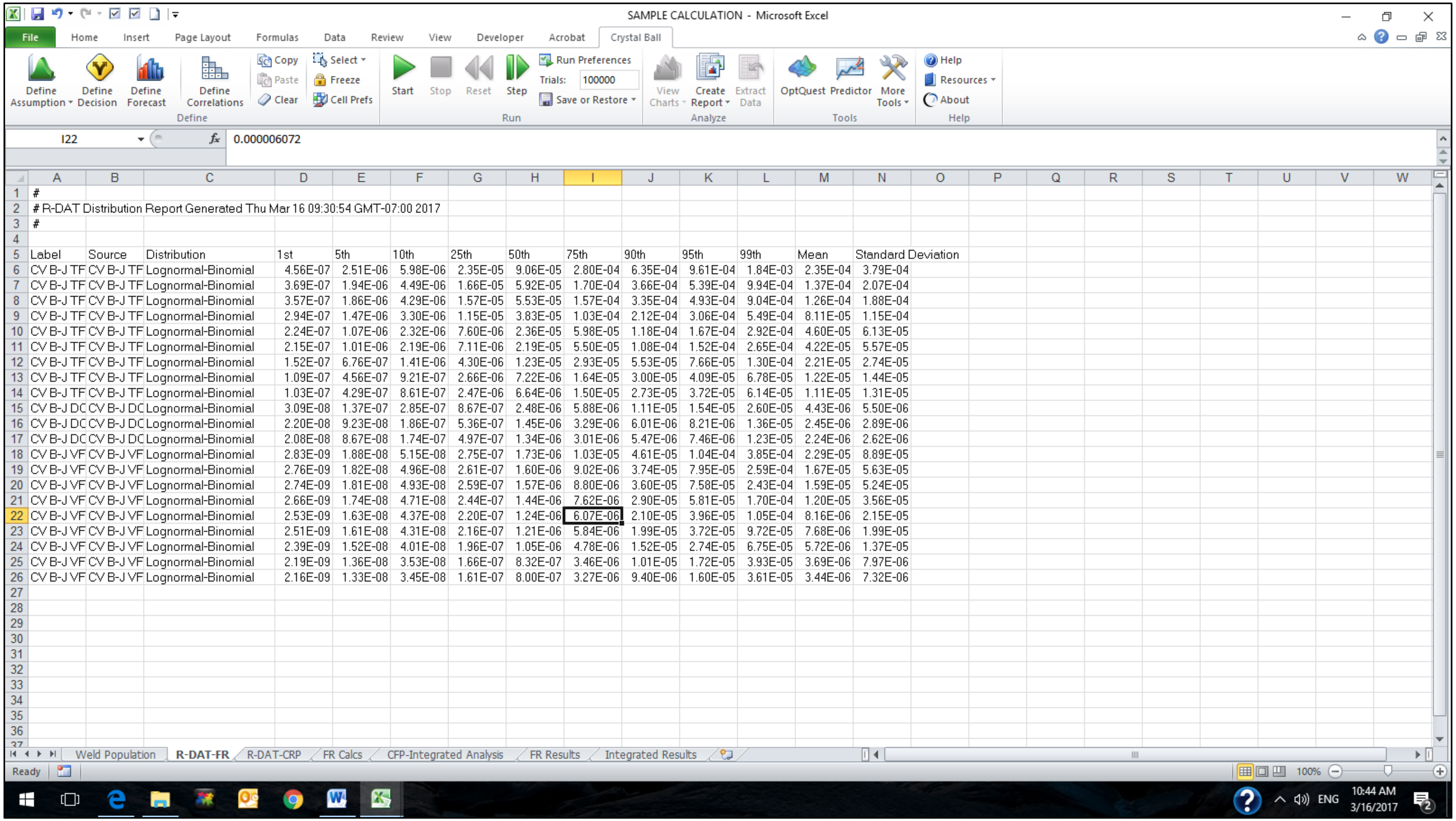
This alternative Bayesian calculation procedure eliminates the need for using a software product such as 'R-DAT Plus.'

6.5 Integrated Analysis

The integrated analysis consists of combining CB-assumptions about failure rates and conditional failure probabilities (CFP). The pipe break frequency calculation is performed by multiplying the FR-assumptions with the CFP-assumptions. The products are represented as CB 'forecasts' (Figure 6-10).

6.6 Implementation Guidance

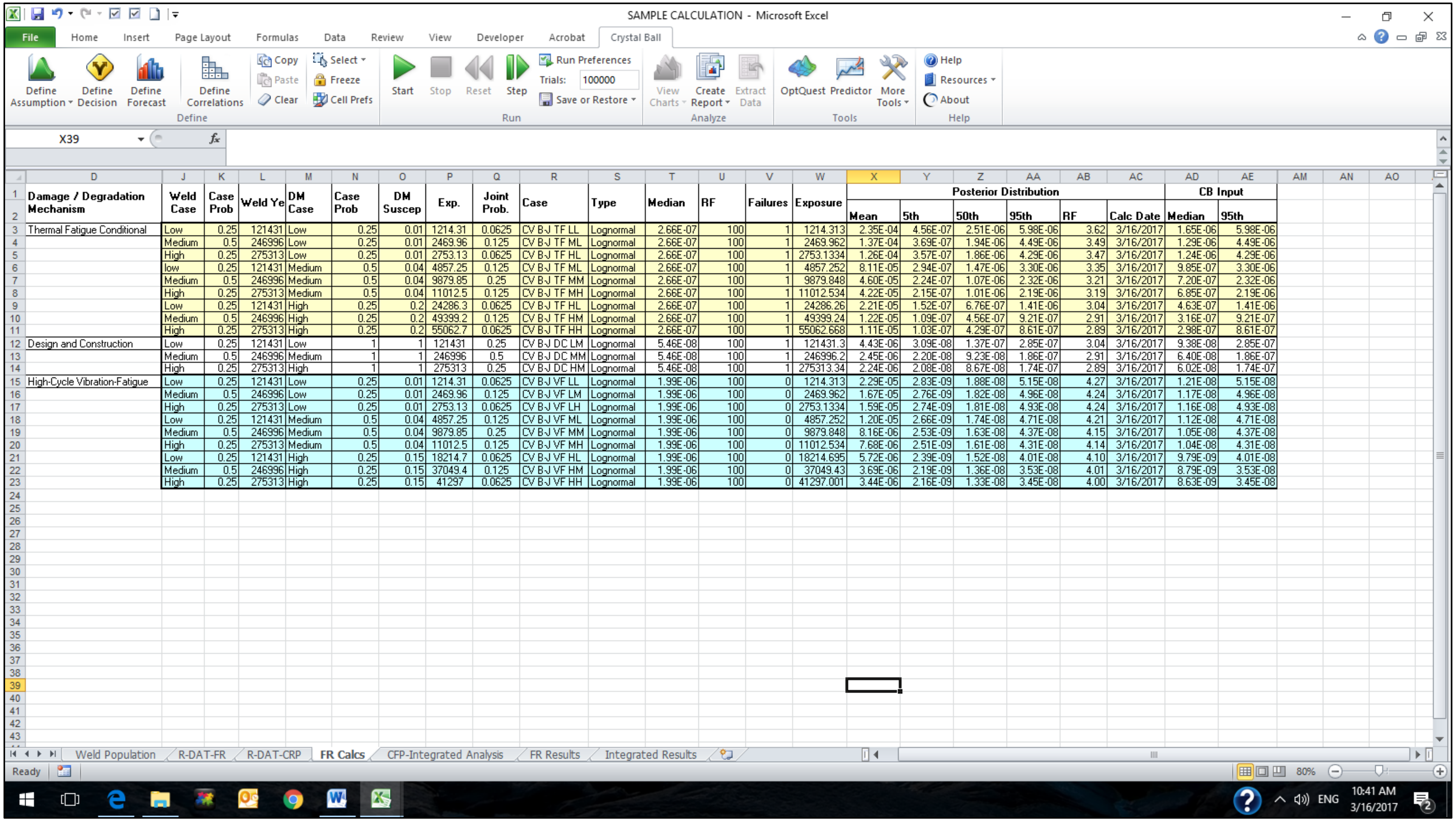
The implementation of a Microsoft® Excel workbook format is done in steps: 1) definition of the evaluation boundary, 2) definition of the structural failure modes to be addressed including the performance of engineering calculations that correlate the consequence of a passive component failure with the size(-es) of a through-wall flaw, 3) development of a conditional rupture probability model, 4) operating experience review to obtain failure populations, 5) piping system design reviews to develop exposure terms that reflect total piping component population that produced a given event population; and 6) designing the workbook. The flowchart in Figure 6-11 is a representation of the tasks involved. In it, the boxes with 'blue-grey' fill represent required preparatory activities that are to be performed before developing the workbook with its different tabs.



R-DAT Distribution Report Generated Thu Mar 16 09:30:54 GMT-07:00 2017

Label	Source	Distribution	1st	5th	10th	25th	50th	75th	90th	95th	99th	Mean	Standard Deviation
CV B-J TF	CV B-J TF	Lognormal-Binomial	4.56E-07	2.51E-06	5.98E-06	2.35E-05	9.06E-05	2.80E-04	6.35E-04	9.61E-04	1.84E-03	2.35E-04	3.79E-04
CV B-J TF	CV B-J TF	Lognormal-Binomial	3.69E-07	1.94E-06	4.49E-06	1.66E-05	5.92E-05	1.70E-04	3.66E-04	5.39E-04	9.94E-04	1.37E-04	2.07E-04
CV B-J TF	CV B-J TF	Lognormal-Binomial	3.57E-07	1.86E-06	4.29E-06	1.57E-05	5.53E-05	1.57E-04	3.35E-04	4.93E-04	9.04E-04	1.26E-04	1.88E-04
CV B-J TF	CV B-J TF	Lognormal-Binomial	2.94E-07	1.47E-06	3.30E-06	1.15E-05	3.83E-05	1.03E-04	2.12E-04	3.06E-04	5.49E-04	8.11E-05	1.15E-04
CV B-J TF	CV B-J TF	Lognormal-Binomial	2.24E-07	1.07E-06	2.32E-06	7.60E-06	2.36E-05	5.98E-05	1.18E-04	1.67E-04	2.92E-04	4.60E-05	6.13E-05
CV B-J TF	CV B-J TF	Lognormal-Binomial	2.15E-07	1.01E-06	2.19E-06	7.11E-06	2.19E-05	5.50E-05	1.08E-04	1.52E-04	2.65E-04	4.22E-05	5.57E-05
CV B-J TF	CV B-J TF	Lognormal-Binomial	1.52E-07	6.76E-07	1.41E-06	4.30E-06	1.23E-05	2.93E-05	5.53E-05	7.66E-05	1.30E-04	2.21E-05	2.74E-05
CV B-J TF	CV B-J TF	Lognormal-Binomial	1.09E-07	4.56E-07	9.21E-07	2.66E-06	7.22E-06	1.64E-05	3.00E-05	4.09E-05	6.78E-05	1.22E-05	1.44E-05
CV B-J TF	CV B-J TF	Lognormal-Binomial	1.03E-07	4.29E-07	8.61E-07	2.47E-06	6.64E-06	1.50E-05	2.73E-05	3.72E-05	6.14E-05	1.11E-05	1.31E-05
CV B-J DC	CV B-J DC	Lognormal-Binomial	3.09E-08	1.37E-07	2.85E-07	8.67E-07	2.48E-06	5.88E-06	1.11E-05	1.54E-05	2.60E-05	4.43E-06	5.50E-06
CV B-J DC	CV B-J DC	Lognormal-Binomial	2.20E-08	9.23E-08	1.86E-07	5.36E-07	1.45E-06	3.29E-06	6.01E-06	8.21E-06	1.36E-05	2.45E-06	2.89E-06
CV B-J DC	CV B-J DC	Lognormal-Binomial	2.08E-08	8.67E-08	1.74E-07	4.97E-07	1.34E-06	3.01E-06	5.47E-06	7.46E-06	1.23E-05	2.24E-06	2.62E-06
CV B-J VF	CV B-J VF	Lognormal-Binomial	2.83E-09	1.88E-08	5.15E-08	2.75E-07	1.73E-06	1.03E-05	4.61E-05	1.04E-04	3.85E-04	2.29E-05	8.89E-05
CV B-J VF	CV B-J VF	Lognormal-Binomial	2.76E-09	1.82E-08	4.96E-08	2.61E-07	1.60E-06	9.02E-06	3.74E-05	7.95E-05	2.59E-04	1.67E-05	5.63E-05
CV B-J VF	CV B-J VF	Lognormal-Binomial	2.74E-09	1.81E-08	4.93E-08	2.59E-07	1.57E-06	8.80E-06	3.60E-05	7.58E-05	2.43E-04	1.59E-05	5.24E-05
CV B-J VF	CV B-J VF	Lognormal-Binomial	2.66E-09	1.74E-08	4.71E-08	2.44E-07	1.44E-06	7.62E-06	2.90E-05	5.81E-05	1.70E-04	1.20E-05	3.56E-05
CV B-J VF	CV B-J VF	Lognormal-Binomial	2.53E-09	1.63E-08	4.37E-08	2.20E-07	1.24E-06	6.07E-06	2.10E-05	3.96E-05	1.05E-04	8.16E-06	2.15E-05
CV B-J VF	CV B-J VF	Lognormal-Binomial	2.51E-09	1.61E-08	4.31E-08	2.16E-07	1.21E-06	5.84E-06	1.99E-05	3.72E-05	9.72E-05	7.68E-06	1.99E-05
CV B-J VF	CV B-J VF	Lognormal-Binomial	2.39E-09	1.52E-08	4.01E-08	1.96E-07	1.05E-06	4.78E-06	1.52E-05	2.74E-05	6.75E-05	5.72E-06	1.37E-05
CV B-J VF	CV B-J VF	Lognormal-Binomial	2.19E-09	1.36E-08	3.53E-08	1.66E-07	8.32E-07	3.46E-06	1.01E-05	1.72E-05	3.93E-05	3.69E-06	7.97E-06
CV B-J VF	CV B-J VF	Lognormal-Binomial	2.16E-09	1.33E-08	3.45E-08	1.61E-07	8.00E-07	3.27E-06	9.40E-06	1.60E-05	3.61E-05	3.44E-06	7.32E-06

Figure 6-7: Data Input File – Pipe Failure Rate Distributions



Damage / Degradation Mechanism	Weld Case	Case Prob	Weld Ye	DM Case	Case Prob	DM Suscep	Exp.	Joint Prob.	Case	Type	Median	RF	Failures	Exposure	Posterior Distribution Mean	5th	50th	95th	RF	Calc Date	CB Input Median	95th
Thermal Fatigue Conditional	Low	0.25	121431	Low	0.25	0.01	1214.31	0.0625	CV B-J TF LL	Lognormal	2.66E-07	100	1	1214.313	2.35E-04	4.56E-07	2.51E-06	5.98E-06	3.62	3/16/2017	1.65E-06	5.98E-06
	Medium	0.5	246996	Low	0.25	0.01	2469.96	0.125	CV B-J TF ML	Lognormal	2.66E-07	100	1	2469.962	1.37E-04	3.69E-07	1.94E-06	4.49E-06	3.49	3/16/2017	1.29E-06	4.49E-06
	High	0.25	275313	Low	0.25	0.01	2753.13	0.0625	CV B-J TF HL	Lognormal	2.66E-07	100	1	2753.1334	1.26E-04	3.57E-07	1.86E-06	4.29E-06	3.47	3/16/2017	1.24E-06	4.29E-06
	low	0.25	121431	Medium	0.5	0.04	4857.25	0.125	CV B-J TF ML	Lognormal	2.66E-07	100	1	4857.252	8.11E-05	2.94E-07	1.47E-06	3.30E-06	3.35	3/16/2017	9.85E-07	3.30E-06
	Medium	0.5	246996	Medium	0.5	0.04	9879.85	0.25	CV B-J TF MM	Lognormal	2.66E-07	100	1	9879.848	4.60E-05	2.24E-07	1.07E-06	2.32E-06	3.21	3/16/2017	7.20E-07	2.32E-06
	High	0.25	275313	Medium	0.5	0.04	11012.5	0.125	CV B-J TF MH	Lognormal	2.66E-07	100	1	11012.534	4.22E-05	2.15E-07	1.01E-06	2.19E-06	3.19	3/16/2017	6.85E-07	2.19E-06
	Low	0.25	121431	High	0.25	0.2	24286.3	0.0625	CV B-J TF HL	Lognormal	2.66E-07	100	1	24286.26	2.21E-05	1.52E-07	6.76E-07	1.41E-06	3.04	3/16/2017	4.63E-07	1.41E-06
	Medium	0.5	246996	High	0.25	0.2	49399.2	0.125	CV B-J TF HM	Lognormal	2.66E-07	100	1	49399.24	1.22E-05	1.09E-07	4.56E-07	9.21E-07	2.91	3/16/2017	3.16E-07	9.21E-07
	High	0.25	275313	High	0.25	0.2	55062.7	0.0625	CV B-J TF HH	Lognormal	2.66E-07	100	1	55062.668	1.11E-05	1.03E-07	4.29E-07	8.61E-07	2.89	3/16/2017	2.98E-07	8.61E-07
Design and Construction	Low	0.25	121431	Low	1	1	121431	0.25	CV B-J DC LM	Lognormal	5.46E-08	100	1	121431.3	4.43E-06	3.09E-08	1.37E-07	2.85E-07	3.04	3/16/2017	9.38E-08	2.85E-07
	Medium	0.5	246996	Medium	1	1	246996	0.5	CV B-J DC MM	Lognormal	5.46E-08	100	1	246996.2	2.45E-06	2.20E-08	9.23E-08	1.86E-07	2.91	3/16/2017	6.40E-08	1.86E-07
	High	0.25	275313	High	1	1	275313	0.25	CV B-J DC HM	Lognormal	5.46E-08	100	1	275313.34	2.24E-06	2.08E-08	8.67E-08	1.74E-07	2.89	3/16/2017	6.02E-08	1.74E-07
High-Cycle Vibration-Fatigue	Low	0.25	121431	Low	0.25	0.01	1214.31	0.0625	CV B-J VF LL	Lognormal	1.99E-06	100	0	1214.313	2.29E-05	2.83E-09	1.88E-08	5.15E-08	4.27	3/16/2017	1.21E-08	5.15E-08
	Medium	0.5	246996	Low	0.25	0.01	2469.96	0.125	CV B-J VF LM	Lognormal	1.99E-06	100	0	2469.962	1.67E-05	2.76E-09	1.82E-08	4.96E-08	4.24	3/16/2017	1.17E-08	4.96E-08
	High	0.25	275313	Low	0.25	0.01	2753.13	0.0625	CV B-J VF LH	Lognormal	1.99E-06	100	0	2753.1334	1.59E-05	2.74E-09	1.81E-08	4.93E-08	4.24	3/16/2017	1.16E-08	4.93E-08
	Low	0.25	121431	Medium	0.5	0.04	4857.25	0.125	CV B-J VF ML	Lognormal	1.99E-06	100	0	4857.252	1.20E-05	2.66E-09	1.74E-08	4.71E-08	4.21	3/16/2017	1.12E-08	4.71E-08
	Medium	0.5	246996	Medium	0.5	0.04	9879.85	0.25	CV B-J VF MM	Lognormal	1.99E-06	100	0	9879.848	8.16E-06	2.53E-09	1.63E-08	4.37E-08	4.15	3/16/2017	1.05E-08	4.37E-08
	High	0.25	275313	Medium	0.5	0.04	11012.5	0.125	CV B-J VF MH	Lognormal	1.99E-06	100	0	11012.534	7.68E-06	2.51E-09	1.61E-08	4.31E-08	4.14	3/16/2017	1.04E-08	4.31E-08
	Low	0.25	121431	High	0.25	0.15	18214.7	0.0625	CV B-J VF HL	Lognormal	1.99E-06	100	0	18214.695	5.72E-06	2.39E-09	1.52E-08	4.01E-08	4.10	3/16/2017	9.79E-09	4.01E-08
	Medium	0.5	246996	High	0.25	0.15	37049.4	0.125	CV B-J VF HM	Lognormal	1.99E-06	100	0	37049.43	3.69E-06	2.19E-09	1.36E-08	3.53E-08	4.01	3/16/2017	8.79E-09	3.53E-08
	High	0.25	275313	High	0.25	0.15	41297	0.0625	CV B-J VF HH	Lognormal	1.99E-06	100	0	41297.001	3.44E-06	2.16E-09	1.33E-08	3.45E-08	4.00	3/16/2017	8.63E-09	3.45E-08

Figure 6-8: Composite Failure Rate Calculation – Uncertainty in Failure Event Population, Exposure Term & DM Susceptibility

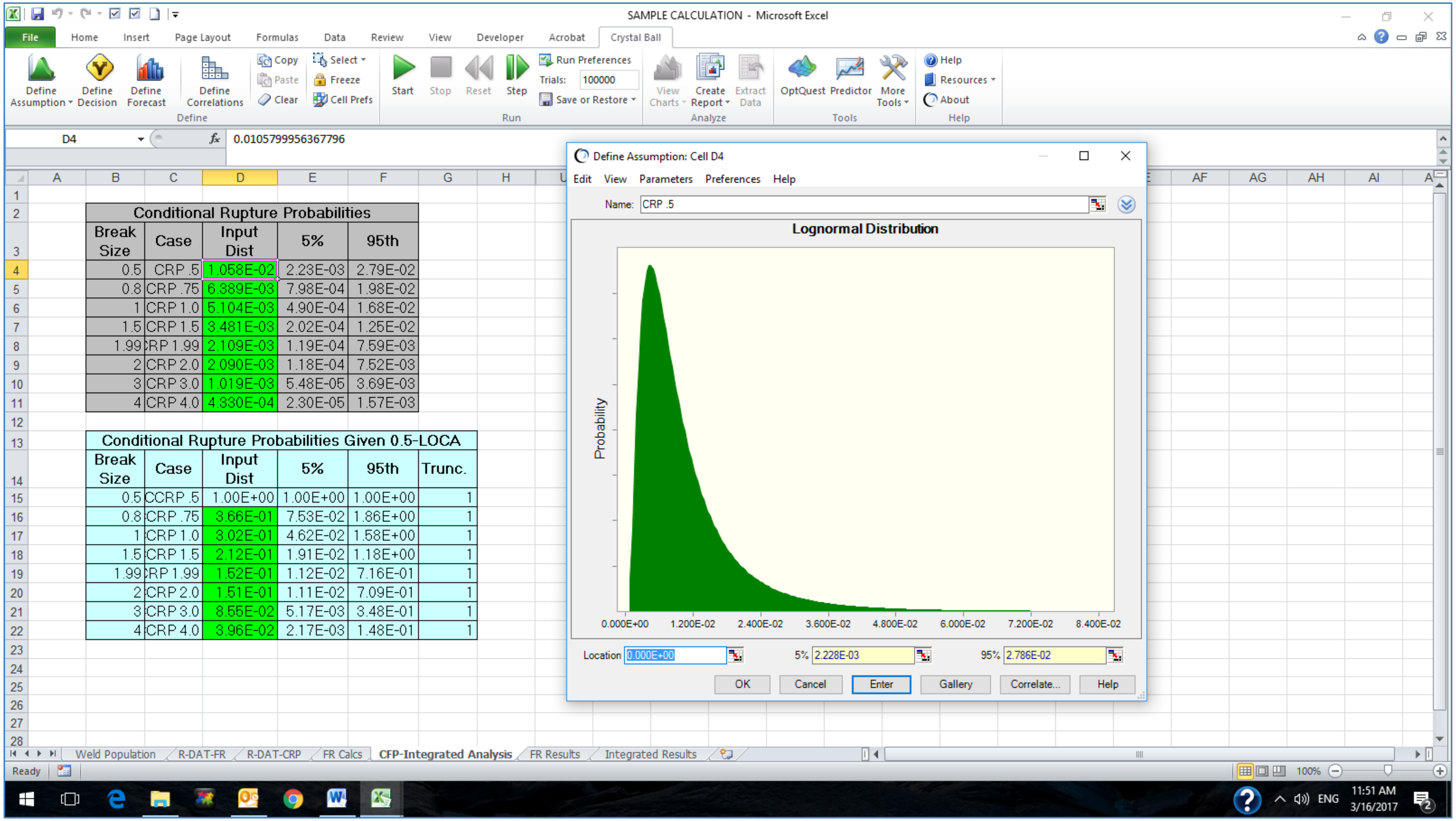


Figure 6-9: Pipe Failure Consequence Calculation[59]

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59 As used in this CB application, green cells represent CB-input parameters (or 'Assumptions').

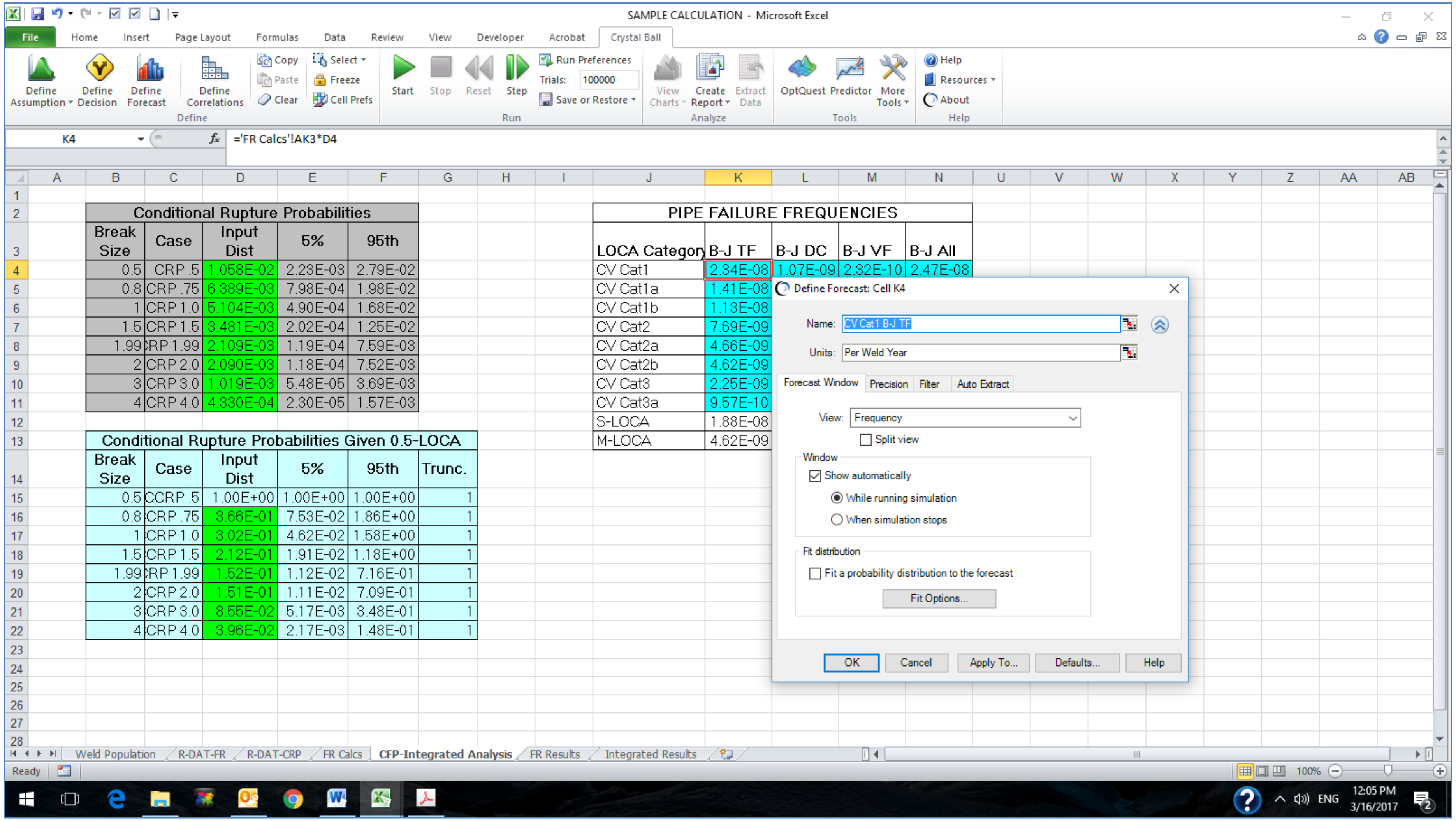


Figure 6-10: Integrated CFP Analysis[60]

[60] In this example, a total of 8 × 4 pipe failure rate distributions are defined. The blue cells represent CB forecasts. The cell values represent the sampling results at the end of the simulation.

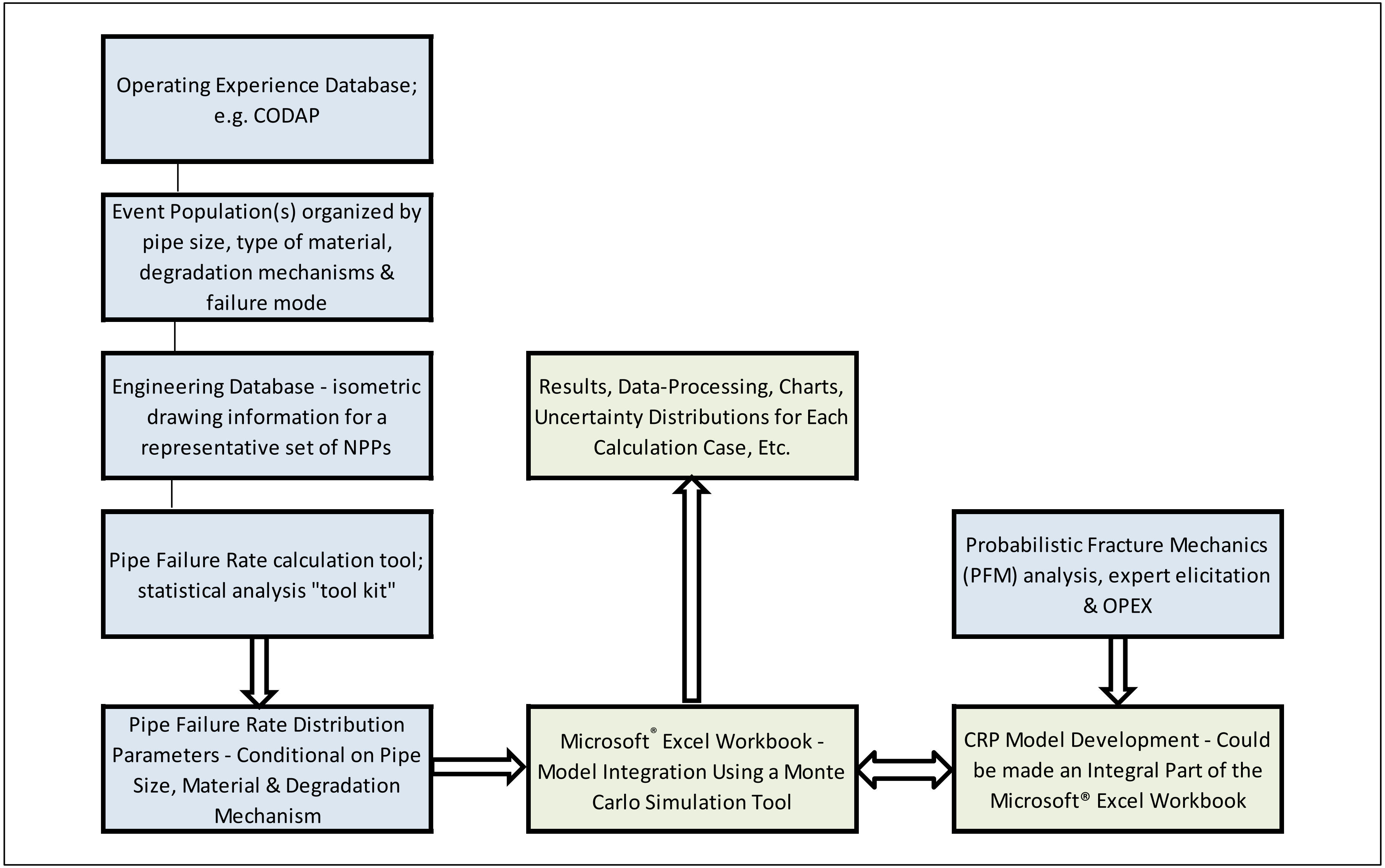


Figure 6-11: Implementation of the Piping Reliability Calculation Format

7. BENCHMARKING OF PIPING RELIABILITY ANALYSIS RESULTS[61]

Invariably, the piping reliability calculations produce small numbers; e.g. frequency of "failure" being < 10^{-6} per piping component and reactor operating year. Some form of validation against different modeling approaches may be warranted to determine the level of realism achieved. This section describes three types of validation strategies: 1) for a given problem statement, comparing results obtained using different probabilistic fracture mechanics (PFM) models, 2) same as "1" but with different probabilistic physics-of-failure (PPOF) models, and 3) comparing PFM or PPOF results with operating experience data through an application of a data-driven model (DDM).

7.1 The 'Joint Properties' of the Different Modeling Concepts

In order to establish a meaningful basis for comparing results obtained using different piping reliability models it becomes necessary to start by defining the specific "figures of merit" of interest. In the context of PSA and risk-informed applications piping reliability typically is considered as part of the initiating event frequency modeling element. The PSA parameter of interest is usually expressed in the form of a cumulative pipe break frequency versus a set of consequence threshold values expressed by an equivalent break size (EBS) with dimension of mass flow rate [kg/s] or diameter of the hole in the pipe wall [mm]. Therefore, the pipe "failure mode" of concern involves a spectrum from small through-wall leaks up to major structural failure (e.g. a double-ended guillotine break, DEGB). Embedded in an analysis are multiple sets of input parameters that account for the effects of different degradation mechanism on structural integrity, the effectiveness of in-service inspection and leak detection, the effectiveness of degradation mitigation techniques, and the chemical and mechanical properties of the material.

The "semantics" of respective modeling technique oftentimes becomes in a disabler for meaningful results comparisons. As an example in PFM the term "leak" corresponds to a wall-penetrating defect which is stable, i.e. the loads are not inducing brittle fracture, plastic collapse, or instable ductile crack growth. And the term "rupture" corresponds to a wall-penetrating defect which fails due to the applied loads. In a risk-informed application it becomes essential to characterize the modes of structural failure in more detail; e.g. geometry and orientation of the pipe outside diameter wall defect and the resulting through-wall leak rate. A follow-on issue relates the conditional failure probability given a defect of certain geometry.

Two public domain documents summarize the current status of PFM code development and the role of PFM in risk-informed applications [89][90]. The former reference points to the intrinsic values of applying advanced PFM computational tools to problems for which no operating experience data exists; e.g. no through-wall defects experienced during the existing in-service component lifetimes. The latter reference addresses a regulator's perspective on past difficulties experienced in reviewing licensee submittals of PFM calculation results. One difficulty arises when a choice of model input parameters does not reflect observations from operational experience.

7.2 Background

As reported in Reference [91], the Pacific Northwest National Laboratory (PNNL) has performed benchmark PFM calculations for selected components using the PRAISE [92] and PRO-LOCA [93] computer codes. The calculations addressed primary water stress corrosion

[61] This section is an abstract of the Pacific Northwest National Laboratory Report No. PNNL-16625 [91].

cracking (PWSCC), intergranular stress corrosion cracking (IGSCC) and fatigue for materials and operating conditions that were known to have failed components. The first objective was to benchmark the calculations against field experience. A second objective was a review of uncertainties in the treatments of the data from observed failures and in the structural mechanics models. Comparisons of predicted and observed failure frequencies showed that PFM codes correctly predicted relatively high failure probabilities for components that had experienced field failures. However, the predicted frequencies tended to be significantly greater than those estimated from plant operating experience. A review of the PFM models and inputs to the models showed that uncertainties in the calculations were sufficiently large to explain the differences between the predicted and observed failure frequencies.

PRAISE was originally developed for the US Nuclear Regulatory Commission (NRC) and has been applied to a range of risk-informed applications, most notably for risk-informed in-service inspection (RI-ISI). PRO-LOCA was also developed for NRC with the intent to incorporate the best elements of other PFM codes, advances in the fracture mechanics and data on fracture behavior of reactor materials. The scope of both codes is the prediction of piping failure probabilities for various degradation mechanisms including failures from preexisting welding flaws, fatigue crack initiation, IGSCC), and (PWSCC). Both codes simulate the progress of degradation from the initiation of small cracks, growth of cracks to small through-wall leaking flaws, and the occurrence of large leaks and piping ruptures.

Calculations were performed to allow the two computer codes to be benchmarked against each other, and more importantly benchmarked against field experience. An objective was to determine the extent to which uncertainties and modeling assumptions may impact calculated failure probabilities. The term “failure” implies any degraded state requiring remedial action. Remedial actions include repairs and replacements with or without more resistant material. The comparisons with field experience were intended to establish if the codes were capable of predicting relatively high failure probabilities for materials and operating conditions that have resulted in field failures. Sensitivity calculations have also been performed to address uncertainties associated with residual stresses, applied stresses and temperatures.

7.3 Benchmark Cases

Benchmark cases were selected to address components for which numerical results from PFM calculations were available and for which data from reported field failures was sufficient to estimate failure frequencies. Calculated failure probabilities came from both the PRO-LOCA and PRAISE codes. The cases were selected to cover a range of degradation mechanisms (fatigue, IGSCC, and PWSCC). The selected components were as follows:

- Case 1. PWR Hot Leg Bi-Metallic Weld. The bi-metallic hot leg weld at the joint between the reactor coolant piping and the reactor pressure vessel nozzle.
- Case 2. PWR Pressurizer Surge Nozzle Bi-Metallic Weld. The bi-metallic weld at the joint between the surge line and the pressurizer.
- Case 3. PWR Pressurizer Spray Line Nozzle Bi-Metallic Weld. The bi-metallic weld at the joint between the spray line and the pressurizer.
- Case 4. BWR Reactor Recirculation 12-Inch (305- mm) Weld. Circumferential welds in stainless steel piping in the BWR recirculation systems for time periods prior to 1988 and before mitigation measures (augmented inspections, water chemistry improvements, etc.) were implemented at BWR plants.
- Case 5. PWR Thermal Fatigue - Thermal fatigue applicable to the nozzle cracking event that occurred at the Oconee-2 plant [94]. This calculation addressed a single unique event, and the comparison focused on predicted versus the observed times to fail the nozzle component.

7.4 Application of Database on Field Experience

In general there are five basic approaches for estimating piping reliability:

1. Structural reliability modeling (SRM) based on probabilistic fracture mechanics,
2. Analytical modeling using Markov theory and statistical analysis of service data,
3. Direct statistical estimation using service data,
4. Expert judgment/expert elicitation,
5. Any combination of "1" through "4."

The discussion here addresses statistical estimation using service data (Method 3). The term "failure" implies any degraded state requiring remedial action. Precise definitions of failure are important to make distinctions between different through-wall flaw sizes that have different effects on plant operation and safety. In NUREG-1829 [71] the structural failure modes listed in Table 7-1 were used. For the benchmarking calculations an additional "failure" mode of crack initiation was defined which included cracks of less than through-wall depth.

Table 7-1: Definitions of Structural Failure for PWR LOCA

Mode of Structural Failure	Equivalent Pipe Break Diameter (EBD) [mm]	Peak Through-wall Flow Rate (FR) [kg/s]
Perceptible Leak	> 0	FR > 0
Large Leak	15 < EBD ≤ 50	0.5 < FR ≤ 5
Small Breach	50 < EBD ≤ 100	5 < FR ≤ 20
Breach	100 < EBD ≤ 250	20 < FR ≤ 100
Large Breach	250 < EBD ≤ 500	100 < FR ≤ 400
Major Breach	EBD > 500	FR > 400 (6,300 gpm)

Table 7-2 summarizes the input data and results of evaluations for the selected components. The number of welds found to have through-wall cracks is seen to be very small ranging from zero to seven reported events per component category. Because the number of events has been small, there are large statistical uncertainties in the estimates of frequencies of through-wall cracks. Uncertainties are particularly large for the pressurizer surge nozzle and pressurizer spray nozzle, because there have only two cases of repairs (cracks with less than through-wall depths) and no cases of through-wall cracks.

Table 7-2: Summary of Input Data and Results from Estimation of Failure Frequencies from Operating Experience Data

Component		Weld-Years	Number of Cracked/Repaired Welds	Number of TWCs	λ_{TWC} (PE) [1/Weld.Year]
PWR RCS Hot Leg Bi-metallic Weld (RPV Nozzle-to-Safe-end)		All: 10,784 2-Loop: 2,510 3-Loop: 3,570 4-Loop: 4,704	2 Note 1	1 Note 1	9.3E-05 Note 2
PWR Pressurizer Spray Line Nozzle Bi-Metallic Weld	Case 1 Note 3	1 × 3621 = 3,621	1	0	1.4E-04
	Case 2 Note 4	5 × 3621 = 18,105	5	1	5.5E-05
	Case 3 Note 5	1 × 3621 = 3,621	1	Assume 1	2.8E-04
PWR Pressurizer	Case 1 Note 6	2 × 3621 = 7,242	2	0	6.9E-05

<table>
<tr><th colspan="2">Component</th><th>Weld-Years</th><th>Number of Cracked/Repaired Welds</th><th>Number of TWCs</th><th>λ_{TWC} (PE) [1/Weld.Year]</th></tr>
<tr><td rowspan="2">Surge Line Nozzle Bi-Metallic Weld</td><td>Case 2 Note 7</td><td>7 × 3,621 = 25,347</td><td>7</td><td>1</td><td>3.9E-05</td></tr>
<tr><td>Case 3 Note 8</td><td>2 × 3621 = 7,242</td><td>2</td><td>Assume 1</td><td>1.4E-04</td></tr>
<tr><td colspan="2">BWR Reactor Recirculation 12-inch (305-mm) Weld (pre-1988) U.S. BWR/3 & BWR/4</td><td>25,137</td><td>120</td><td>7</td><td>2.8E-04 Note 9</td></tr>
<tr><td colspan="2">BWR Reactor Recirculation 28-inch (711-mm) Weld (pre-1988) U.S. BWR/3 & BWR/4</td><td>19,551</td><td>72</td><td>5</td><td>2.6E-04 Note 9</td></tr>
<tr><td colspan="6">Notes:
1. Service experience through December 2005
2. This is a composite failure rate under assumption of equal susceptibility to PWSCC in 2-loop, 3-loop and 4-loop PWR plants with bi-metallic welds
3. Case 1 accounts for existing service experience with bi-metallic pressurizer spray line bi-metallic welds
4. Case 2 assumes equal PWSCC susceptibility for bi-metallic pressurizer spray line weld and relief line welds – 5 welds per plant
5. Same as Case 1 except that the flaw found at Millstone-3 is assumed to be near or at through-wall
6. Case 1 accounts for existing service experience with bi-metallic surge line welds (hot leg side and pressurizer side)
7. Case 2 assumes equal PWSCC susceptibility for bi-metallic surge line welds, pressurizer spray line weld and pressurizer relief line welds (7 welds per plant)
8. Same as Case 1 except that 1-of-2 flaws in the service experience is near or at through-wall.
9. Average failure rate across all welds in a typical Reactor Recirculation System.</td></tr>
</table>

When there are no reported "failures" for a particular component of interest, there are methods that can be applied to estimate (or bound) failure frequencies. The two methods can be summarized as follows:

1. A bounding frequency is calculated based on the assumption that one failure (as in a through-wall leak) occurs. The key input is then the number of relevant weld-years of operation for which no failures have been reported. The results can be viewed as an upper bound to the failure frequency. The referenced benchmark assumed a single weld with a through-wall crack in the pressurizer spray line nozzle weld. In this case, there however have been reported cracks (less than through-wall) that have required repairs. Under an assumption of no PWSCC mitigation[62], it is reasonable to assume that a through-wall crack could occur in the future.
2. The population of components is expanded to include a larger number of components that have similar materials, designs, and operating conditions as the particular component of interest. A successful outcome of this approach depends on appropriate judgments regarding components to be included in the larger population. By considering more components, the relevant data is more likely to show actual failure events and will cover a much larger number of weld-years of operation. By expanding the population, estimated failure frequencies can either increase (the number of events increases significantly) or can decrease (a much larger number weld years of operation with no significant increase in failure events).

[62] Methods of PWSCC mitigation include the application of a full structural weld overlay (FSWOL) or replacing Alloy 600/82/182 materials with Alloy 690/52/152 materials.

Based on these considerations the following results were obtained for the five components of interest.

- Case 1. PWR Hot Leg Bi-Metallic Weld: The data showed one event with a through-wall crack (i.e. the V.C. Summer event of 2000)[63]. Consideration of cracked welds with less than though-wall crack depths added another event. The calculated failure frequency is listed in Table 7-2 as 1.5E-04 per weld-year based on the number of relevant welds per plant and the number of reactor years of operation up to the year 2000.
- Case 2. PWR Pressurizer Surge Nozzle Bi-Metallic Weld: At the time of the analysis, the data showed no events with through-wall cracks. Consideration of cracked welds with less than though-wall crack depths showed two events. Two calculated failure frequencies are listed in Table 7-2 as 1.2E-06 and 1.6E-05 per weld-year based on the number of relevant welds per plant (one) and the number of reactor years of operation up to the year 2005. In this evaluation, two assumptions were made regarding the relevant population. One assumption limited the population to only the surge line nozzle weld. With the other assumption bi-metallic welds in PWR plants (PWR bi-metallic welds in piping of various diameters but not the hot leg weld) were included. The order of magnitude difference in the two estimated failure frequencies comes for the different number of weld-years of operations between the two assumptions regarding the population of relevant welds and the failure history (one through-wall flaw).
- Case 3. PWR Pressurizer Spray Line Nozzle Bi-Metallic Weld: The data showed no events with through-wall cracks. Consideration of welds with less than through-wall crack depths showed two events. Two calculated failure frequencies are listed in Table 7-2 as 1.5E-06 and 2.1E-05 per weld-year based on the number of relevant welds per plant (one) and the number of reactor years of operation up to the year 2005. Two alternative assumptions were made regarding the relevant population. In one case the population was limited only to the spray line nozzle weld. In the other case bi-metallic welds in PWR plants (other bi-metallic welds in piping of various diameters but not the hot leg weld) were included. The order of magnitude difference in the two estimated failure frequencies comes for the different number of weld-year of operations between the two assumptions regarding the population of relevant welds and the failure history (one through-wall flaw).
- Case 4. BWR Reactor Recirculation 12-Inch (305-mm) Weld: This example considered circumferential welds for the time period pre-1988 before mitigation measures (augmented inspections, water chemistry improvements, etc.) that were implemented at BWR plants. The PIPExp database showed 7 events with through-wall cracks. Consideration of cracked welds with less than though-wall crack depths, added 120 events. The calculated failure frequency (through-wall cracks) is calculated as 2.8E-04 per weld-year based on the number of relevant welds per plant and the number of reactor years of operation up to the year 1988. In this case, because of the number of reported failure events and the already broad scope of the selected population, there was no reason to consider a wider population of welds to provide a more robust basis for estimating a failure frequency.
- Case 5. PWR Thermal Fatigue: This evaluation addressed a unique event in the Oconee Nuclear Station. The Oconee event involved a leaking crack that occurred at a small diameter nozzle after the unexpected loss of a thermal sleeve exposed the inner surface of the nozzle to cyclic thermal stresses. The crack was found to extend over a large fraction of the pipe circumference as indicated by Figure 7-1. An evaluation of the event indicated that the fatigue cracking initiated in a relatively short time period (one year or less).

[63] LER 50-395/2000-008-01 https://lersearch.inl.gov/PDFView.ashx?DOC::3952000008R01.PDF

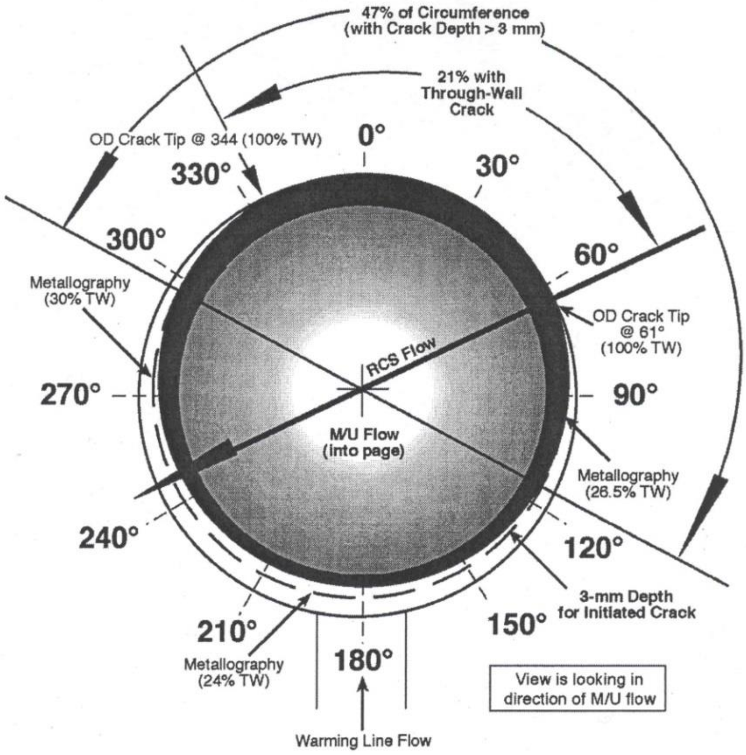


Figure 7-1: Configuration of Thermal Fatigue Crack in Failed Oconee-2 Nozzle

7.5 PFM Predictions

Details of the PFM calculations including component designs, materials, temperature/ environmental conditions, and the sources and levels of stress imposed during plant operation are described in References [95][96][97]. Table 7-3 lists specific input parameters for the five cases that were addressed.

Table 7-3: Input Parameters for PFM Calculations

Case	Temp. [°C]	ID [mm]	Wall Thickness [mm]	#Circ. Subunits of 2-inch length	Pre-Existing Flaw Size	
					Crack Depth [mm]	Crack Length [mm]
Case 1 – PWR Hot Leg Bi-metallic Weld PWSCC	315	737	63.5	44	3.0	10.0
Case 2 – PWR Surge Nozzle Bi-metallic Weld PWSCC	345	282	35.7	44	3.0	10.0
Case 3 – PWR Spray Nozzle Bi-metallic Weld PWSCC	345	87	13.5	11	3.0	10.0
Case 4 -BWR Stress Corrosion Cracking	288	324	17.2	44	3.0	10.0
PWR Thermal Fatigue	311	74	7.62	5	3.0	Sampled from Length Distribution

Figures 7-2 and 7-3 show the predicted probabilities of crack initiation and through-wall cracks as a function of time for all five cases. It is seen that all but one of the failure probabilities approach 100% before the nominal end of plant operating life of 40 years. The following provides brief summaries of the individual PFM calculations along with comparisons of calculated versus observed probabilities.

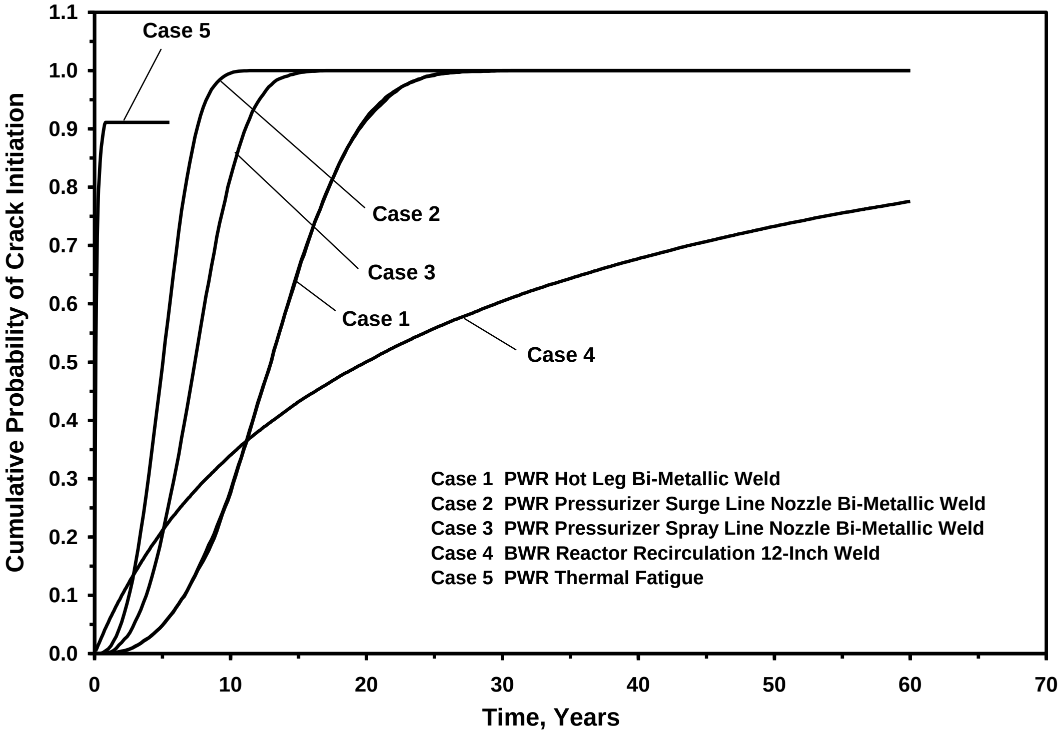


Figure 7-2: Calculated Probabilities for Crack Initiation.

- Case 1. PWR Hot Leg Bimetallic Weld PWSCC: At T = 16 years the PFM calculations gave a cumulative probability of through-wall cracking of about 0.05 (Figure 7-3). Service failure data showed that through-wall cracking was observed at the V.C. Summer and Ringhals Units 3 and 4 PWR plants in Year 2000. The failure frequency calculated based on the number of report failures, the number hot leg-to-vessel welds and the number of plant years of operation was 9.1E-05 failures per weld per year after about 20 years of operation. Using a plant availability of 80%, 20 years of plant operation would correspond to about the 16 years in the PFM calculations. On this basis the operating data gave a cumulative probability of about 20x9.1E-05 = 1.82E-03 per weld. The PFM model therefore over predicted the probability of through-wall PWSCC cracks in the hot-leg weld by a factor of about 30. Reasons for the relatively high predicted failure probabilities are discussed in Section 7.6.
- Case 2. PWR Surge Nozzle Weld: Figures 7-2 and 7-3 show a 50% probability of crack initiation by about 3 years and a 50% probability of a through-wall crack by about 6 years. There were two reported events for the surge nozzle location in the mode of cracked or repaired welds, but no failures that involved through-wall cracks. A failure frequency was calculated with consideration of the welds in the relevant population of plants and the corresponding number of plant years of operation. The resulting frequency of through-wall cracks was estimated to be 1.2E-06 failures per weld per year. With a plant availability of 80%, plant operation for 6.0 years would correspond to 4.8 years for the PFM calculations. At 4.8 years the probabilistic fracture mechanics calculations predict a cumulative probability of through-wall cracking about 0.50. In contrast, the operating data gives a cumulative probability of 6x1.2E-06 = 7.2E-06. The PFM calculations are seen to over predict the probability of through-wall PWSCC cracks in the

surge nozzle by about four orders of magnitude. Possible reasons for the large difference are discussed in Section 7.6.

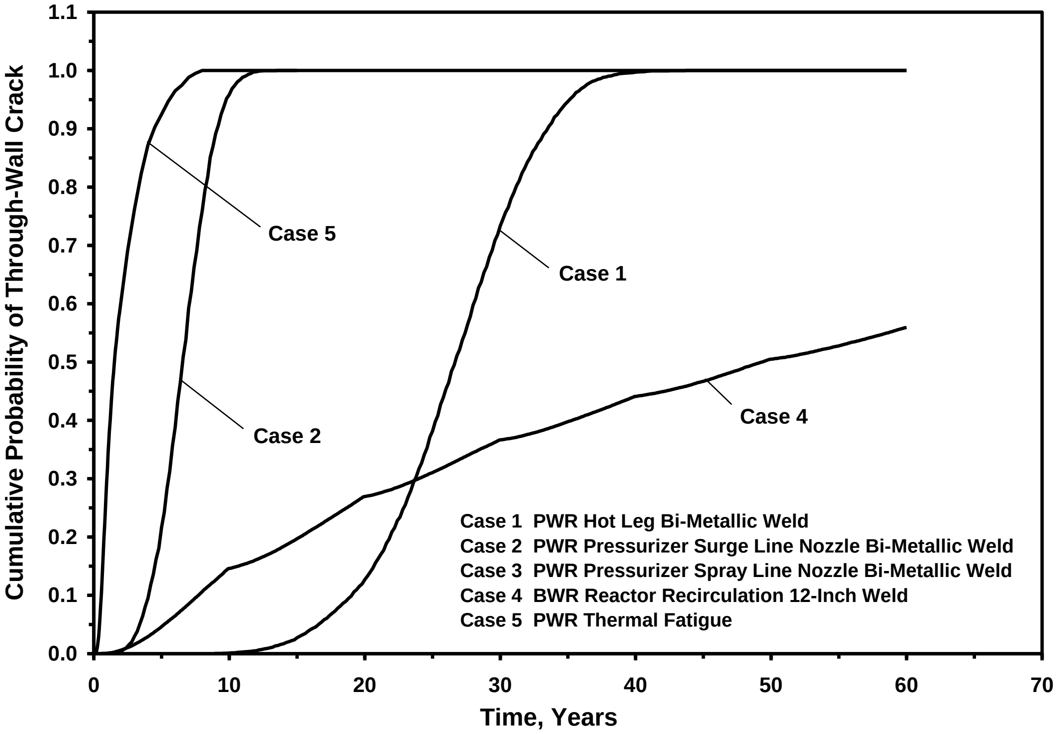


Figure 7-3: Calculated Probabilities for Through-Wall Cracking

- Case 3. PWR Spray Nozzle Weld. Figures 7-2 and 7-3 show a 50% probability of a through-wall cracking at about 5 years. Failure data were evaluated to estimate a probability of though-wall cracking based on operating experience to give frequencies ranging from 1.6E-06 to 7.3E-05 failures per weld per year. With a plant availability of 80%, 5 years of plant operation would correspond to about 4 years for the probabilistic fracture mechanics calculations. At 4 years, operating data then gives a cumulative probability of through-wall crack from 6.4E-06 to 2.9E-04 compared to the much higher 50% probability of crack initiation as predicted by the PFM calculations. The PFM calculations are seen to significantly over predict the probability of cracking by about four orders of magnitude. Possible reasons for the large difference are discussed in Section 7.6.
- Case 4. BWR, IGSCC. Figures 7-2 and 7-3 show calculated failure probabilities as a function of time for both crack initiation and through-wall cracks which give probabilities at 15 years of about 20% for crack initiation and about 40% for through-wall cracking. Failure data from field experience covering the time period prior to about 1988 were used to estimate a probability of though-wall cracking for 12 inch piping from operating data. Table 7-2 gives a frequency of through-wall cracks of 2.8E-04 from which a probability of through-wall cracking after 15 years of operation is calculated to be 4.2E-03. The calculated probability is a factor of about 50 greater than this observed probability. Sensitivity calculations for IGSCC of BWR piping performed using the PRAISE code show that uncertainties in modeling IGSCC (e.g. levels of welding residual stresses) can explain relatively large differences between calculated and observed failure probabilities.
- Case 5. PWR Thermal Fatigue. Probabilistic calculations were performed with PRAISE for several values of cyclic stress ranging from 30 ksi to 75 ksi (207 to 517 MPa). The results shown in Figures 7-2 and 7-3 corresponding to a stress range of 60 ksi cycled at a rate of one cycle per minute. For the Case 5 calculations the time scale of Figures 7-2 and

7-3 should be interpreted as the time span over which the cyclic thermal stress condition is active and not the total operating life of the component. Probabilities of crack initiation and of through-wall cracking were predicted to approach or exceed 100% within one year of operation at the assumed level of cyclic stress. The calculations for Case 5 therefore demonstrated that predictions of failure probabilities are consistent with the field experience as reported for the Oconee plant. The calculations also demonstrated that the multiple cracking feature of the PFM model can predict long circumferential flaws also consistent with field failure data.

7.6 Reconciliation of Predicted vs. Observed Failure Probabilities

Whereas it has been demonstrated that computer codes such as PRO-LOCA and PRAISE can predict essentially the same failure probabilities, both codes have predicted significantly higher failure probabilities than probabilities derived from field failure data. This section discusses uncertainties in the calculations that can explain the differences between the calculated and observed failure probabilities.

<u>Welding Residual Stresses</u>. Inputs for residual stresses have been based on finite element calculations that simulated steps of the welding processes. While some experimental stress measurements have also been made on piping welds, measurements are lacking for residual stresses in bimetallic welds. The finite element calculations assumed a sequence of weld passes along with heat inputs, and material properties for the range of temperatures as a weld cools. For particular welds the welding parameters could differ from those assumed in the finite element calculations. Welds can also be subject to repairs involving local grind outs and repair welding. Some calculations have explicitly accounted for such repairs and the calculated failure probabilities show significant effects of such repairs. The results are generally from axi-symmetric analyses, which have been shown to overestimate the residual stress. In addition, modeling assumptions, such as lumping weld passes, can also overestimate weld residual stresses.

The stress inputs have not accounted for changes in residual stresses that could occur both during construction and from plant operation. For example, safe-end bimetallic welds may experience additional residual stress changes from welding of the other side of the safe end. Hydro testing and imposed service stresses can cause yielding and redistribution of the welding residual stresses. If residual stresses are already at or near yield, the operational stresses could cause decreases in peak stresses.

<u>Crack Initiation Predictions</u>. The PRO-LOCA calculations for PWSCC have predicted the initiation of cracking in piping welds by application of cracking experience for control rod drive mechanism (CRDM) nozzles. Factors were applied to crack initiation times to account for effects of temperature and stress levels in piping welds compared to the CRDM component. This approach could be subject to significant uncertainties.

<u>Crack Growth Rates</u>. There are indications that PWSCC cracks may not grow as fast for components in the field as predicted from laboratory data. Various explanations have been offered to explain the differences. Crack growth has been observed to become retarded after a period of laboratory testing [96]. Inhomogeneity in metallurgy along the crack path [97] can cause changes in the direction of the crack growth, crack branching, changes in cracking mode (intergranular versus transgranular), creation of remaining ligaments in the crack wake, and other ill-defined resistance for crack growth. Other explanations involve effects of corrosion products within the crack on the local chemistry at a crack tip. Observations of changes in the aspect ratios of growing cracks have indicated faster growth rates at surface locations as compared to the growth rates for crack tips located deep within the wall thickness.

Another issue is the appropriate values for threshold stress intensity factors needed to sustain crack growth. Laboratory testing methods have also been an issue. For example, testing procedures often impose light fatigue cycling to maintain SCC crack growth in the presence of retardation effects.[64] Such procedures can give more reproducible crack growth rates that are useful for quantifying alloy susceptibility for cracking, but may not be representative of the conditions that control crack growth under field conditions. An additional challenge for evaluating the growth of service cracks is defining the actual stress intensity driving crack growth in structures having complex variations in metallurgical characteristics and local stresses caused by such factors as welding or surface cold-working. Apart from the calculation of stress intensity factors, service cracks are non-ideal cracks in terms of crack fronts, remaining ligaments and mode of cracking. In summary it has been difficult even for laboratory tests to generate predictable and reproducible data on PWSCC crack growth rates. Therefore one should also expect differences for growth rates from laboratory measurements compared to growth rates for field conditions.

<u>Circumferential Stress Variations</u>. PFM calculations neglected variations of stress around a pipe circumference, whereas some stress categories (dead weight and thermal expansion bending moments) will have large circumferential stress variations. The calculations conservatively have assumed peak values of stress around the full circumference. Circumferential stress variations reduce predicted probabilities of crack initiation and also reduce crack growth rates at lower stress locations.

<u>Temperatures</u>. Predicted times to crack initiation and growth rates are sensitive to temperature. Probabilistic calculations have been based on nominal or design temperatures, whereas actual operating temperatures could differ from these often conservative values.

<u>Temperature Factor</u>. Temperature has a significant effect on calculated failure probabilities due to stress corrosion cracking. These effects are predicted using an Arrhenius equation that requires an input for activation energy. For PWSCC this activation energy was taken to be 50 kcal/mole for crack initiation and 31 kcal/mole for crack growth.

<u>Reporting of Field Data</u>. There are also uncertainties in failure probabilities as estimated from field events. For the analyzed cases, the number of failure events has been small (in many cases no failures for many components of interest) giving arise to large statistical uncertainties. Upper and lower bounds for failure frequencies can nevertheless be estimated based on the number of components and reactor operating years covered by the database. The numbers of events are larger for cracks of less than through-wall depth. However, the number of events can be under reported because of NDE limitations and unreported cracks.

<u>Multiple Cracking Model</u>. The crack propagation models used in PRO-LOCA and PRAISE, respectively, account for the initiation, growth and linking of multiple cracks around the circumference of a weld. Dimensions of the circumferential subunits (i.e. potential locations for initiated cracks) are assigned to be generally consistent with observed cracks in failed components or are based on the sizes of laboratory test specimens. The depths and lengths of the initiated cracks are based on judgments regarding the initiated crack sizes that can be detected for the test procedure. Another important modeling consideration is the criterion to link growing cracks in adjacent circumferential subunits, with the ASME Section XI flaw proximity rule being used to determine when two adjacent flaws should be linked. Alternative rules could delay crack linking and thereby give lower probabilities for through-wall cracks.

<u>Correlation of Crack Initiation Times & Growth Rates</u>. Sampled parameters for crack initiation and crack growth are assumed to have no statistical correlation. In addition crack

[64] Under certain loading conditions and when applied repeatedly crack growth has been observed in laboratory settings to retard (slow).

initiation in one subunit does not imply an increased probability of initiation for other subunits of a weld. The PRO-LOCA code has considered strategies to simulate correlations between subunits, but such approaches were not included in the calculations. Such correlations would increase the probability of linking cracks in adjacent subunits along with an increase in the probabilities of through-wall cracks. The computer code does however use a common sampled parameter to predict the growth of all cracks in a given weld.

Insights from NUREG/CR-6674 Calculations [95]: Fatigue calculations with PRAISE have predicted many leaks for PWR and BWR piping systems, whereas no such leaks have been reported. A review of possible reasons for this inconsistency noted that cyclic stress inputs were based on conservative values used for the original design fatigue calculations. Cyclic stresses were believed to be unrealistically high because 1) actual thermal transients during operation are often less severe than assumed for design, and 2) load pairs were conservatively estimated assuming worst case sequences of transients. This approach will give some very high cyclic stresses not experienced during service. The calculations of NUREG/CR-6674 also used conservative inputs for cyclic strain rates and for environmental parameters such as oxygen content. Although each of the individual inputs was consistent with possible operating conditions, it is unlikely that the conservative inputs would all be present at the same time for any given component.

7.7 The NURBIM Project

Under the Fifth Framework of EURATOM, the European Commission in 2001 established the (Nuclear Risk Based Inspection Methodology for Passive Components (NURBIM) project.[65] Work Package number 4 was concerned with the review and benchmarking of structural reliability models and associated software [98]. Benchmark calculations using six different PFM computer codes were performed for different stress corrosion cracking (SCC) cases and fatigue cases and as defined by different sets of piping dimensions and loading conditions. The final report of the benchmark study is critical of the value of operating experience data in validating the calculated failure probabilities. With respect to the use of PFM in support of risk-informed in-service inspection (RI-ISI), the benchmark study provides the following requirements and recommendations relative to the underlying structural reliability models (SRMs):

Requirements

1. The SRM theory and technical basis should be published and independently reviewed.
2. The SRM and the associated software should address the relevant degradation mechanisms under consideration.
3. The SRM and the associated software should be able to evaluate failure probabilities both for leak events and ruptures.
4. A sensitivity study using the SRM and the associated software should be presented, addressing the relevant damage mechanism under consideration. In the sensitivity study failure probabilities, for events varying from small leaks to ruptures, should be evaluated for variations of input parameters and shown to be consistent with expectations and the given SRM theory assumptions.
5. Sample calculations of the SRM and the associated software should be presented where the assigned input parameters should be described and sources of the data assignments should be given. The probability distributions and internally assigned (hardwired)

[65] The paper by Duan, X., Wang, M. and Kozluck, M.J., "Benchmarking PRAISE-CANDU with Nuclear Risk Based Inspection Methodology Project Fatigue Cases," J. Pressure Vessel Technology, **137** (Oct. 2015) documents the results of a benchmarking between PRAISE-CANDU Version 1.0 and the NURBIM fatigue cases. doi: 10.1115/1.4028202.

parameters (if any) in the SRM software should be documented and the reasons stated. Also the limitations of the SRM software should be clearly identified.

Recommendations

1. The SRM software should be benchmarked against at least one other publicly available SRM software (commercial or non-commercial) for the relevant damage mechanism under consideration. The report of this benchmark study should be published and independently reviewed.
2. The SRM software should be benchmarked against operating experience using actual plant failure frequencies. For damage mechanisms where no ruptures have occurred, leak frequencies may be used for the comparison.
3. Hardwired formulations should be avoided as far as possible except to avoid the risk of a misuse of the software. Formulations of input data (type of probability distribution and its random properties) should, as far as possible, be decided by the user. This also involves the possibility to use a deterministic input (zero scatter) for the variables.
4. The SRM software should enable the user to extract control variables from the results. This is important for the user to be able to check the solutions and understand why a certain result is achieved. Examples of control variables (for a cracking damage mechanism) are:
   - Initial crack sizes, crack shapes during the sub-critical growth and critical crack sizes at leak and rupture.
   - Time to leak and rupture.
   - Stress intensity factors and J-integrals.
   - Crack opening areas and leak flow rates for through-wall cracks.
5. The influence of inspections should be included in the SRM and the associated software in order to quantify risk reductions from repeated inspections.
6. When rupture probabilities are evaluated, it is important to model LBB events. In this context an adequate model of crack opening areas, leak flow rates and leak flow rate detection is important.
7. The used software should be clearly identified. It is desired that new information or better modelling assumptions should be continuously incorporated into the SRM and the associated software so that the generated results may reflect the best current knowledge.

7.8 NUREG-1829 Limited Scope Benchmark

As part of the NUREG-1829 Expert Elicitation project [71], a limited scope benchmarking exercise was performed to compare predicted weld failure rates with the reported service experience. The benchmarking was limited to NPS12 BWR reactor recirculation welds susceptible to IGSCC and in view of the ample operating experience the failure mode considered was “perceptible leakage.”

Probabilistic fracture mechanics (PFM) calculations using the WinPRAISE computer code generated predictions about the weld failure rate for different assumptions about the normal operating stresses (σ_{NO}).[66] A data-driven approach using Bayesian reliability methodology was used to derive weld failure rates from service experience data. Figure 7-4 shows the results of the benchmarking exercise. Table 7-4 includes a description of the different analysis cases of the benchmarking exercise.

[66] D.O. Harris, “Progress in Benchmarking SCC for 12 Inch Recirculation Line, July 1, 2003. NUREG-1829 work file.

Table 7-4: Benchmarking PFM with Operating Experience Data

Case	Case Definition
Data-Driven Methodology	OPEX for NPS12 Reactor Recirculation pipe-to-reducer weld with weld overlay. T = 25 years. Total of 303 failure records involving non-through-wall and through-wall flaws.
WinPRAISE Case 1	NPS12 reactor recirculation system weld with normal operating stress, σ_{NO} = 10 ksi[67]
WinPRAISE Case 1	NPS12 reactor recirculation system weld; σ_{NO} = 12 ksi
WinPRAISE Case 1	NPS12 reactor recirculation system weld; σ_{NO} = 15 ksi
WinPRAISE Case 1	NPS12 reactor recirculation system weld; σ_{NO} = 20 ksi

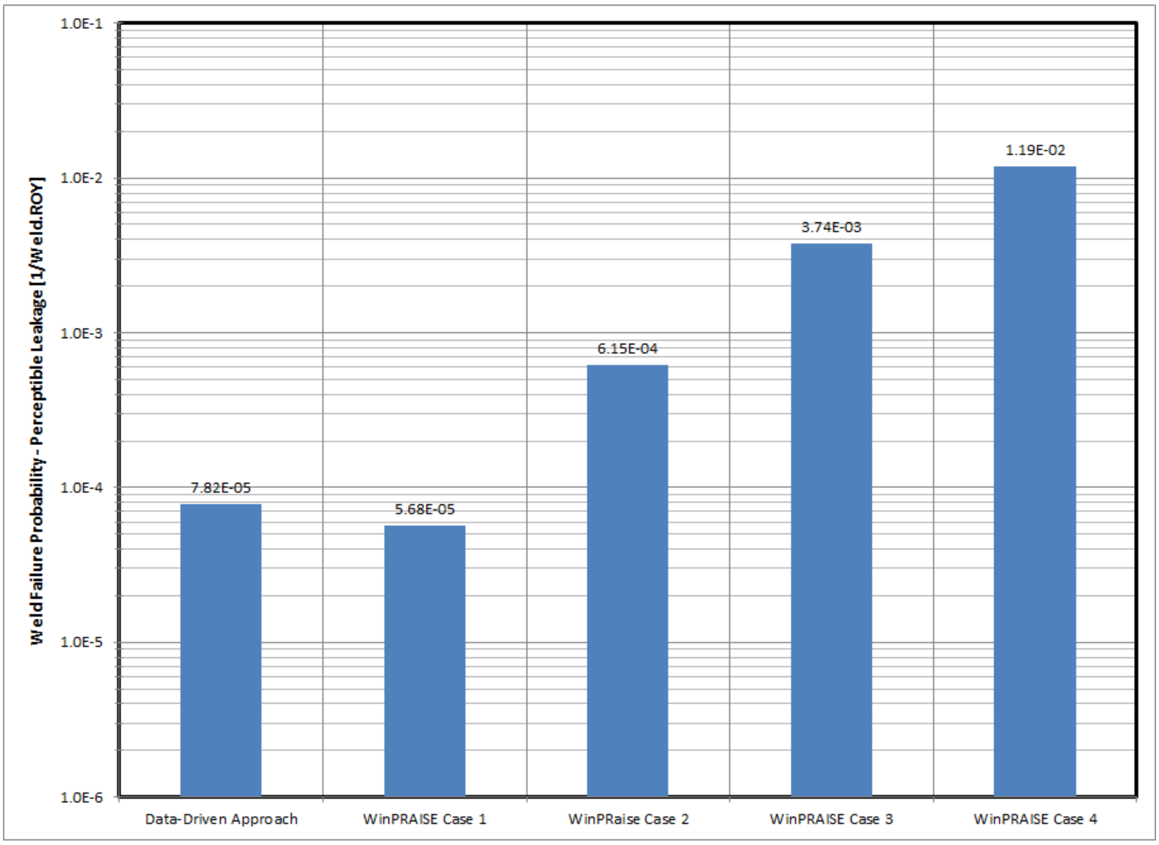


Figure 7-4: Benchmark Results – WinPRAISE & Data-Driven Approach

[67] 1 ksi = 6.9 MPa

8. ENHANCED PIPING RELIABILITY MODELS

Different Markov (MRP) and semi-Markov (SMP) model formulations have been proposed to address the influence of RIM on the structural integrity of piping. The two general forms of modeling are also referred to as “statistical flow-chart modeling.” Four- and five-state Markov models have been proposed for use in the context of risk-informed in-service inspection (R-ISI). Comparisons of Markov and semi-Markov model results have been conducted. This section summarizes some salient features of the two formulations and summarizes their analytical strengths and limitations.

8.1 Background

The reliability characteristics of piping systems are influenced by material properties and operating environment, and by various surveillance programs including leak detection systems, system leak and pressure tests, and in-service inspections involving visual and volumetric examinations using non-destructive examination (NDE) techniques. An objective of in-service inspection is to detect flaws before propagating through-wall. In piping reliability analysis the influences by leak detection and in-service inspection are accounted for by the “integrity management factor” I_{ik} in Equation 8-1 below.

$$\rho_{ix} = \sum_{k} \lambda_{ik} P(R_x | F_{ik}) I_{ik} \qquad (8\text{-}1)$$

Where:

- ρ_{ix} = Frequency of pipe failure of component type *i* with break size *x*, subject to epistemic uncertainty calculated via Monte Carlo simulation
- λ_{ik} = Failure rate per "location-year" for pipe component type *i* due to failure mechanism *k*, subject to epistemic uncertainty determined by Bayes method
- $P(R_x | F_{ik})$ = Conditional rupture probability (CRP) of size *x* given failure of pipe component type *i* due to damage or degradation mechanism *k*, subject to epistemic uncertainty
- I_{ik} = Integrity management factor for weld type *i* and degradation mechanism *k*, subject to epistemic uncertainty which may be determined through data analysis and Monte Carlo simulation or by application of a Markov model.

In the above equation the integrity management factor accounts for the probability of a certain RIM program to successfully identify degradation before the degradation has eroded the safety margin of the pipe to unacceptable levels. The integrity management factor can be determined on the basis of RIM qualification data, expert judgment or field experience data. A different approach is based on a Markov model formulation of piping reliability. This model of piping reliability enables a quantitative assessment of the level of risk reduction that is achievable with in-service inspection. RIM also involves various activities to mitigate or prevent material degradation such as use of material that is more resistant to degradation, full structural weld over lay technique to reverse mechanical stress gradients, and enhanced water chemistry control measures.

Whether a statistical estimation approach, probabilistic fracture mechanics approach or an extended statistical estimation approach using Markov modelling is used, an intrinsic aspect of piping reliability is to explicitly account for certain RIM attributes. These attributes include, for example, NDE assumptions of different probabilities of detection (PODs)

[99][100][101] and inspection intervals. The quality of the input data used to support piping reliability analysis applications remains a key to realistic and robust analysis results and insights.

Embedded in the operating experience (OE) data collections on pipe failures are effects of in-service inspection, leak detection (remote and local), routine walkdown inspections, and other integrity management strategies. An example of OE data analysis insights is shown in Figure 1. Using an appropriate reliability model it is feasible to “isolate” the effect of a RIM strategy on structural reliability and to project how it affects structural integrity in the long-term. In Figure 1 ASME Code Class 1 OE data is organized by pipe size and mode of failure; non-through-wall flaws versus through-wall flaws. In general, the though-wall flaws reflect instances where NDE failed to detect a flaw prior to it propagating through-wall.

In the analysis of the Reactor Coolant Pressure Boundary (RCPB, ASME Code Class 1) piping OE data (Figure 8-1) some general conclusions about the effectiveness of RIM can be drawn on the basis of the observed crack-before-leak (CBL) ratios. Of the large-bore (> NPS10) failure events, over 92% are part-through-wall flaws discovered by NDE. Less than 8% are minor leak events as the result of axial or circumferential flaws that were left to propagate in the through-wall direction. As a somewhat simplistic observation it can be concluded that the probability of detecting a rejectable flaw is on the order of 92% on the basis of OE data. In contrast, the small-bore piping (≤ NPS2) is excluded from the ASME Section XI in-service inspection requirements. Also, this class of piping tends to by susceptible to cyclic fatigue that tends to cause through-wall leaks over relatively short in-service periods (less than or much less than an 18- to 24-month period).

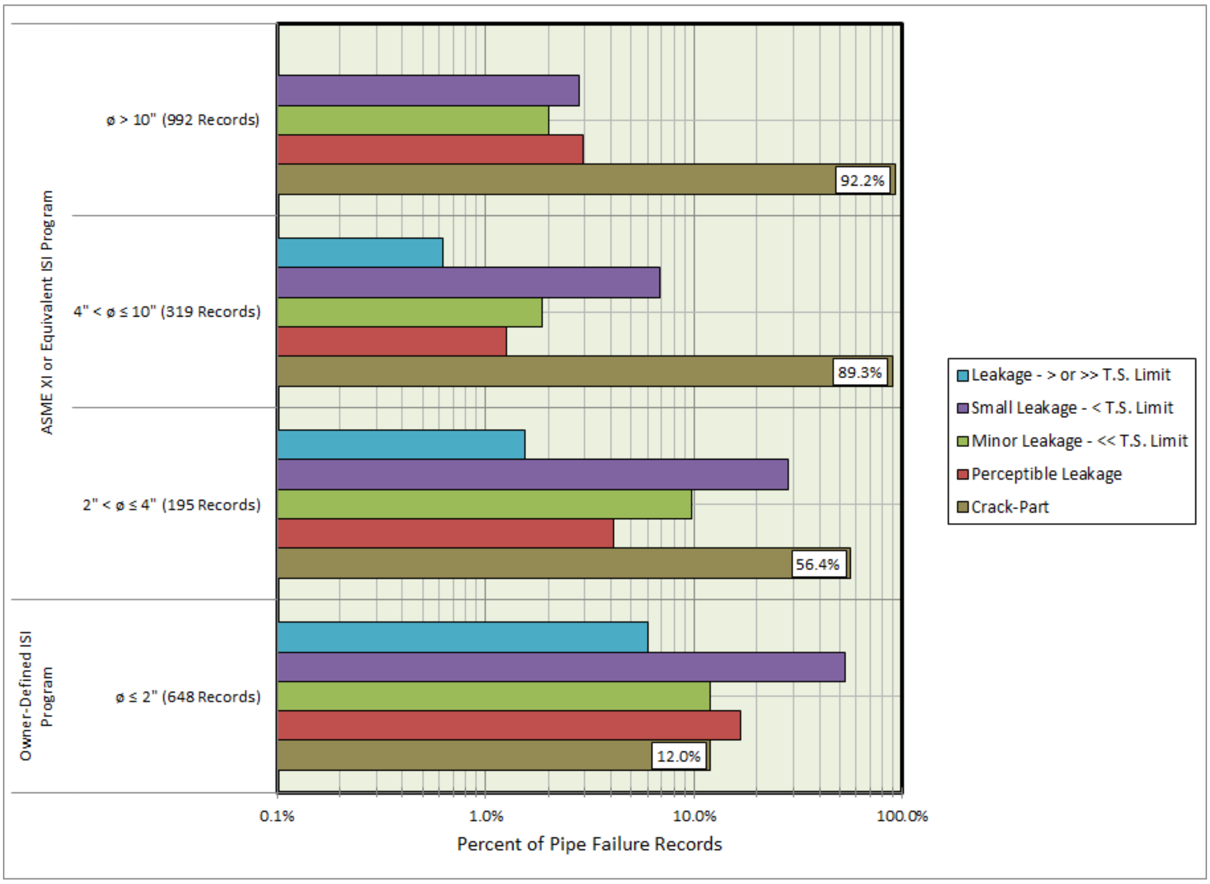


Figure 8-1: Reactor Coolant Pressure Boundary Piping Failure Data Summary

Similar to the above example, having access to a well-qualified OE database enables an evaluation of the impact of different structural materials on piping reliability. Combining the available experimental and simulation data on new materials is an essential aspect of performing certain “before-and-after” assessments, however. As an example, the results of an analysis to determine the impact of primary water stress corrosion cracking (PWSCC) mitigation on the PWR Reactor Coolant System hot leg break frequency is shown in Figure 8-2.

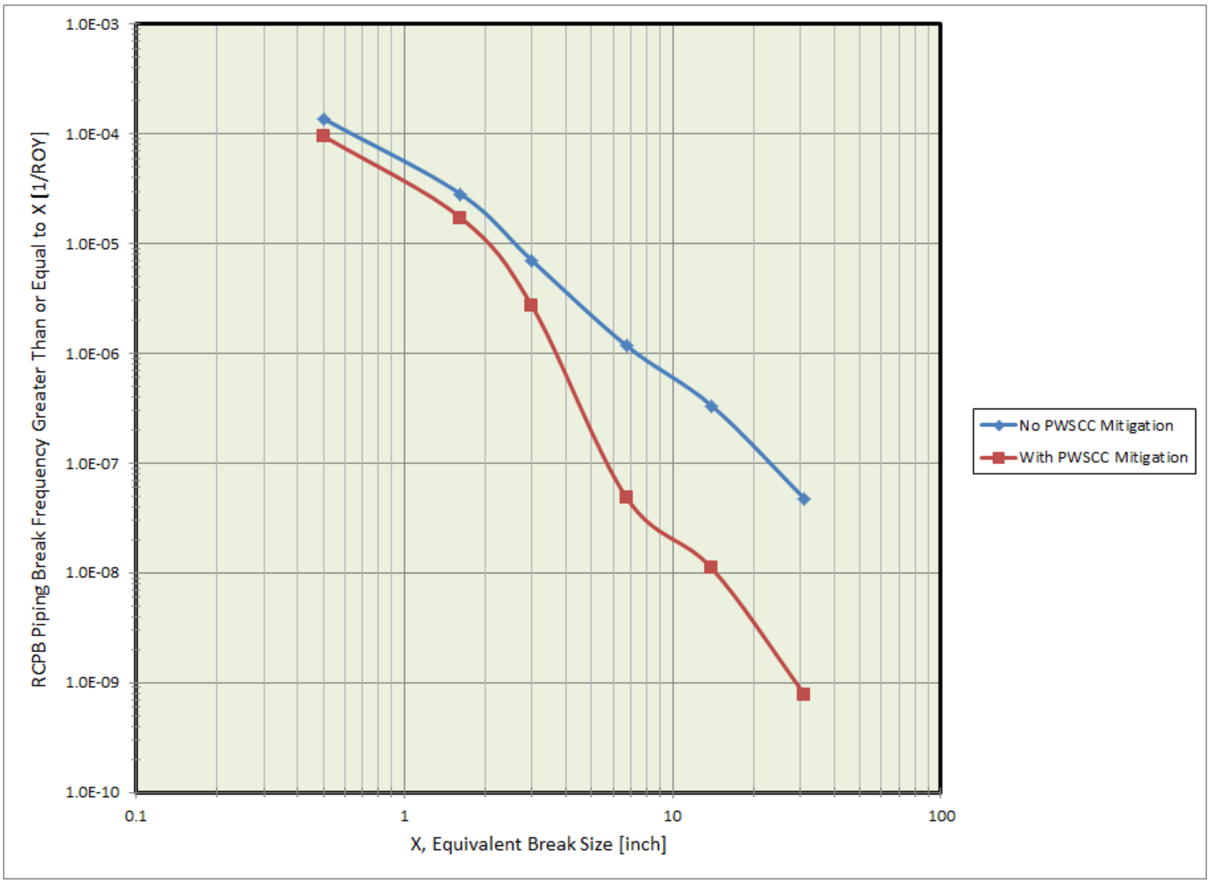


Figure 8-2: Effect of PWSCC on Loss-of-Coolant-Accident Frequency[68]

The two examples are provided to instill the basic premise of using a comprehensive knowledgebase as the foundation for any piping reliability analysis task. Enhanced modelling techniques come into play when addressing the change in structural integrity by different RIM strategies.

Reliability and integrity management (RIM) involves those aspects of a plant design process that are applied to provide an appropriate level of reliability of systems, structures and components (SSCs) and a continuing assurance over the life of the plant that such reliability is maintained. These include design features important to reliability performance such as design margins, selection of materials, testing and monitoring, provisions for maintenance, mitigation of degradation processes, repair and replacement, leak monitoring, pressure and leak testing, and in-service inspection (ISI). In the context of the ASME Boiler and Pressure Vessel Code Section XI, “Rules for In-Service Inspection of Nuclear Power Plant Components” (ASME XI), RIM is an extension of ISI and is performance-based for non-destructive examination (NDE) and on-line monitoring of structural integrity using advanced technologies such as acoustic monitoring or guided ultrasonic waves. In this report the term

[68] In the given example “PWSCC mitigation” means that Ni-base material (e.g. ALLOY 600) known to be susceptible to stress corrosion cracking is replaced with ALLOY 690 material. The analysis was performed by the author of this report for the Wolf Creek Nuclear Operating Company in 2014.

“RIM” is used broadly to describe all processes that are used to control and monitor the structural integrity of reactor components.

8.2 Markov Models for Time-Dependent Piping Reliability

The objective of Markov modeling is to analytically address the interactions between degradation mechanisms and the in-service inspection, flaw detection, and degradation mitigation strategies (e.g. full structural weld overlay) that can reduce the probability that failure occurs or the failure progresses to rupture [102][103][104][105]. Markov modeling starts with a representation of a piping component in a set of discrete and mutually exclusive states. At any instant of time, the piping component is permitted to change state in accordance with whatever competing processes are appropriate for that plant state. A Markov model state refers to the existence of flaws, leaks, or ruptures. The processes that can create a state change are degradation mechanisms acting on the pipe and the process of inspecting or detecting flaws and leaks, and repair of damage before progressing to a complete structural failure. The degradation mechanisms that act on a piping component are represented by failure rates obtained directly for OE data.

Shown in Figure 8-3 is a general four-state Markov model of piping reliability. All failure processes of this model can be evaluated using operating experience data. According to the Markov model diagram a piping component can be in four mutually exclusive states: S (= Success), F (= Flawed or Cracked), L (= Leaking, non-active leakage, or active leakage with leak rate within Technical Specification Limit) or R (= Leaking, with leak rate in well excess of Technical Specification Limit.

The time-dependent probability that a piping component is in each state S, C, F, or L is described by a differential equation. Under the assumption that all the state transition rates are constant the Markov model equations will consist of a set of coupled linear differential equations with constant coefficients. The reliability term needed to represent pipe rupture is the hazard rate $h\{t\}$, which is time-dependent. The hazard rate is defined as:

$$h\{t\} = (1/(1- R\{t\})) \times dR\{t\}/dt \qquad (8\text{-}2)$$

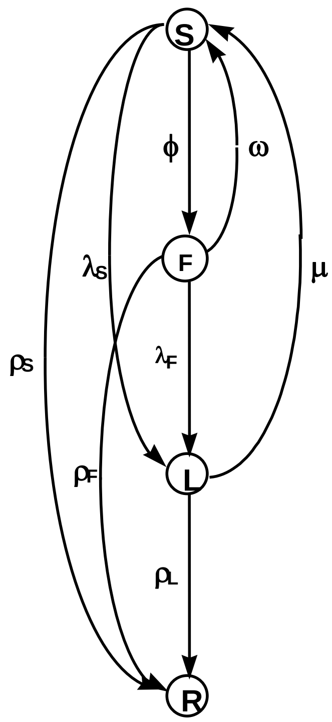
Where:

$$1 - R\{t\} = S\{t\} + F\{t\} + L\{t\} \qquad (8\text{-}3)$$

The hazard rate is a function of time and the parameters of the Markov model; $h\{t\}$ is the time-dependent frequency of pipe rupture. For the 4-state Markov model $h\{t\}$ is expressed as a function of the six parameters: An occurrence rate for detectable flaws (ϕ), a failure rate for leaks given the existence of a flaw (λ_F), two rupture frequencies including one from the initial state of a flaw (ρ_F) (break-before-leak, BBL) and one from the initial state of a leak (ρ_L) (leak-before-break, LBB), a repair rate for detectable flaws (ω), and a repair rate for leaks (μ).The latter two parameters dealing with repair are further developed by the following simple models.

The hazard rate is a function of time and the parameters of the Markov model; $h\{t\}$ is the time-dependent frequency of pipe rupture. For the 4-state Markov model $h\{t\}$ is expressed as a function of the six parameters: An occurrence rate for detectable flaws (ϕ), a failure rate for leaks given the existence of a flaw (λ_F), two rupture frequencies including one from the initial state of a flaw (ρ_F) (break-before-leak) and one from the initial state of a leak (ρ_L) (leak-before-break), a repair rate for detectable flaws (ω), and a repair rate for leaks (μ). The latter two parameters dealing with repair are further developed by the following simple models.

$$\omega = \frac{P_{FI} P_{FD}}{(T_{FI} + T_R)} \qquad (8\text{-}4)$$



Piping Reliability States:

S = Success (or undamaged state);

F = Crack (non-through wall flaw);

L = Leaking through-wall flaw (leak rate is within Technical Specification limit);

R = Rupture; major structural failure

State Transitions:	
ϕ	Occurrence of non-through wall flaw
λ_C	Occurrence of small leak given a flaw ('C')
λ_F	Occurrence of large leak given a through-wall flaw ('F')
ρ_S	Occurrence of rupture given no flaw
ρ_C	Occurrence of large leak given a non-through wall flaw
ρ_F	Occurrence of rupture given a flaw
ρ_L	Occurrence of rupture given a leak
μ	Detect and repair a through-wall flaw
ω	Inspect and repair a non-through wall flaw

Figure 8-3: Four-State Markov Model of Piping Reliability

Where:

P_{FI} = Probability that a piping element with a flaw will be inspected per inspection interval. This parameter has a value of 0 if it is not in the inspection program and 1 if it is in the inspection program. For the inspected elements, a value of 1 is used for any ISI inspection case and 0 for the case of no ISI. The element may be selected for inspection directly by being included in the sections sampled for ISI inspection, or indirectly by having a rule such that if degradation is detected anywhere in the system, the search will be expanded to include examination of that element.

P_{FD} = Probability that a flaw will be detected given this element is inspected. This is the reliability of the inspection program and is equivalent to probability of detection (POD). This probability is conditioned on the occurrence of one or more detectable flaws in the segment according to the assumptions of the model.

T_{FI} = Mean time between inspections for flaws, (inspection interval).

T_R = Mean time to repair once detected. Depending on the location of the weld to be repaired, the actual weld repair could take on the order of several days to much more than a week. Accounting for time to prepare for repair, NDE, root cause evaluation, etc., the total outage time attributed to the repair of a Class 1 weld is on the order of 1 month or more. However, since this term is always combined with T_{FI}, and T_{FI} could be 10 years, in practice the results are insensitive to assumptions regarding T_R.

Similarly, estimates of the repair rate for leaks can be estimated according to:

$$\mu = \frac{P_{LD}}{(T_{LI} + T_R)} \tag{8-5}$$

Where:

P_{LD} = Probability that the leak in the inspection location will be detected per leak inspection or detection period

T_{LI} = Mean time between inspections for leaks. For RCPB piping the time interval between leaks can be essentially instantaneous if the leak is picked up by radiation alarms, to as long as the time period between leak tests performed on the system.

T_R = Mean time to repair. This time should be the minimum of the actual repair time and the time associated with cooldown to enable repair and any waiting time for replacement piping.

Opportunities for leak detection are highly dependent on the system in which the leak occurs as well as the specific location and size of the leak. For example, in the reactor coolant system (RCS) of a Light Water Reactor, leaks of a significant magnitude would create an immediate high containment radiation alarm in the control room. In these cases, the time to inspection and repair is limited by technical specifications on RCS leakage and the time to cool down the plant and begin the process of repair. Other leaks may not cause an alarm but would be subject to possible detection during periodic (e.g. every 8 hours) walk-down by plant personnel or other opportunity for leak detection. There are some leaks that may only be detected upon periodic leak testing which may occur less often as required to meet ASME rules for different classes of pipe per ASME Section XI and other requirements for leak testing.

An important observation about the leak repair term μ in comparison to the flaw repair term ω is that for most leaks the detection possibilities are not normally limited to some predetermined population of welds that are inspected. However leak testing provides an opportunity to inspect all locations system wide. Hence, given a leak of significant magnitude anywhere in the system, the probability of leak detection tends to be high. For locations that are not inspected the flaw repair rate term ω is zero. Also, the time between successive inspections for leaks tends to be much shorter than for volumetric examination of welds with virtually instantaneous detection in cases when the leak would trigger an alarm in the control room. Hence, the Markov model provides the capability to take into account for the LBB principle.

The 4-state model in Figure 8-3 can be represented by the following system state transition matrix (or the transition matrix of the Markov chain):

$$T = \begin{bmatrix} -\varphi & \phi & 0 & 0 \\ \omega & -(\omega + \lambda_F + \rho_F) & \lambda_F & \rho_F \\ \mu & 0 & -(\mu + \rho_L) & \rho_L \\ 0 & 0 & 0 & 0 \end{bmatrix} \quad (8\text{-}6)$$

Let S(t), F(t), L(t), and R(t) represent the time-dependent probabilities of being in the states S, F, L, and R, respectively. These state probabilities can be obtained by solving the system of differential Equations (8-7) though (8-10) with the initial condition $S(0) = 1$ and $F(0) = L(0) = R(0) = 0$ and subject to the condition $S(t)+F(t)+L(t)+R(t) = 1$. The initial condition corresponds to a flaw-less pipe.

$$dS(t)/dt = \omega F(t) + \mu L(t) - \phi S(t) \quad (8\text{-}7)$$

$$dF(t)/dt = \phi S(t) - (\omega + \lambda_F + \rho_F)F(t) \quad (8\text{-}8)$$

$$dL(t)/dt = \lambda_F F(t) - (\mu + \rho_L)L(t) \quad (8\text{-}9)$$

$$dR(t)/dt) = \rho_F F(t) + \rho_L(L(t) \quad (8\text{-}10)$$

When the above equations are solved, the time dependent probabilities of the piping component occupying each state can be determined. Under the assumption that all the transition rates are constant the Markov model equations consist of a set of coupled linear differential equations with constant coefficients. These equations can be solved analytically or numerically.

The time-dependent hazard rate starts at t = 0 (beginning of plant life) and eventually reaches an asymptotic value. A limitation of the Markov model is that the transition rates are assumed to be constant. In other words, the model does not account for any temporal changes in the transition rates that are attributed to aging or degradation mitigation processes. A strength of the Markov model is that it is relatively easy to implement (e.g. in an Excel spreadsheet format) and it allows for parametric studies that address the influence of different leak detection and NDE strategies on structural integrity. An example of the effect of different RIM strategies on the calculated LOCA frequency is shown in Figure 8-4.

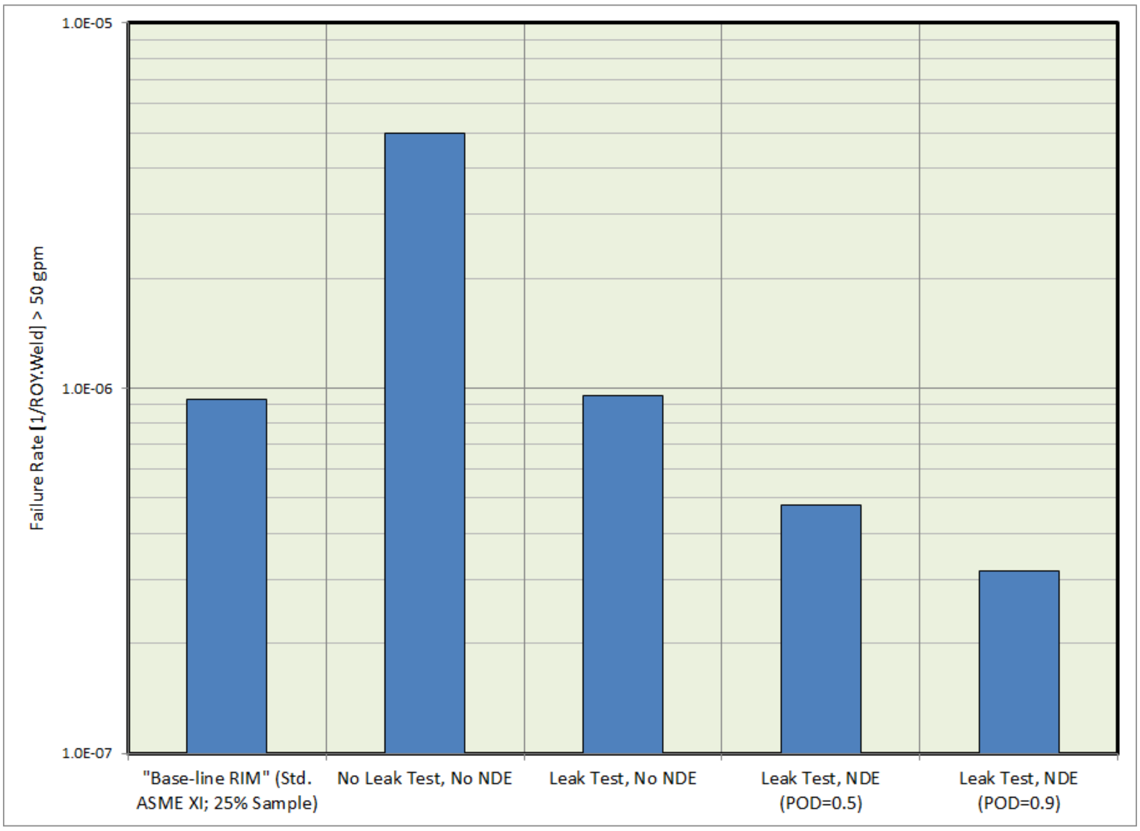


Figure 8-4: An Example of Markov Model Application Results

8.3 Semi-Markov Models of Piping Reliability

As shown in the previous section, the Markov model formulation assumes constant transition rates. Asymptotically, this model shows a constant hazard rate and thus does not account for a potentially increasing or decreasing hazard rate. A semi-Markov process (SMP) model [106], on the other hand, assumes user-defined distributions for time spent during state transitions and hence allows for a physics-of-failure oriented piping reliability analysis. Typically, a statistical distribution (lifetime distribution such as a Weibull or lognormal distribution) is used to represent crack initiation and the growth can be represented by a Paris-Erdogan crack growth law. Motivations for a SMP model formulation are usually based on an assumption of inadequate OE data but may also be based on a desire for more in-depth assessment of material aging phenomena.

Stochastic differential equations (SDEs) may be used resolve the SMP transitions. A SDE is a differential equation in which one or more of the terms are a stochastic process, resulting in a solution which is also a stochastic process. Numerical solution of stochastic differential equations is an evolving field, and especially 'stochastic partial differential equations' is a young field relatively speaking. Almost all algorithms that are used for the solution of ordinary differential equations will work very poorly for SDEs, having poor numerical convergence. Reference [107] is a standard textbook on SDE, its theory and application.

8.3.1 SMP Application Insights

Semi-Markov Process (SMP) models have been applied to different piping reliability analysis problems. The paper by Di Maio et al [108] summarizes the results of a SMP application to involving thermal fatigue of a mixing tee in a Pressurized Water Reactor (PWR) primary system operating environment. The Monte Carlo simulation method is used to model the time-dependent transition rates. According to Di Maio et al the integration of physical models of fatigue (e.g. initiation and propagation) allows for a more realistic degradation process modeling.

Included in the paper by Di Maio et al are results from a benchmark exercise where a Markov Renewal Process (MRP) and a SMP modeling approach is applied to the same thermal fatigue problem. Both approaches indicate that pipe rupture due to thermal fatigue is a non-credible event for the first 10 to 15 years of plant operation. In the long-term (e.g. beyond 15 years of operation) it is claimed that the probability of pipe rupture given thermal fatigue is an order of magnitude greater using a SMP as opposed to a MRP modeling approach.

It should be noted that thermal fatigues is an "event-based" mechanism that is strongly dependent on factors such as mode of operation, operating procedures, method of piping fabrication, leak tightness of isolation valves, mass flow rates in "run" pipes and branch pipes connected to a mixing tee. Furthermore, ample operating experience data on thermal fatigue failures is available indicating that significant structural failures can occur within a short period of plant operation; e.g. less than 1 year. Figure 8-5 summarizes the operating experience with thermal fatigue.

Aldemir et al [109] address the application of a physics-based model to piping degradation caused by flow accelerated corrosion (FAC). No analysis results are provided, however. Extensive OE data on FAC is available and quite realistic modeling of this degradation mechanism is feasible for different types of secondary side piping systems in single- or two-phase flow conditions.

Collins et al [110][111] have applied as SMP model to the problem of primary water stress corrosion cracking (PWSCC) of Alloy82/182 in a PWR environment. A Weibull distribution is selected for the characterization of crack initiation and the MRP-135 model [112] is selected for crack propagation. The results indicate that when using a Weibull scale parameter of 4 years, a maximum hazard rate of about 1×10^{-5} per year occurs at about 10 years and then appears to decrease towards an asymptote of about 1×10^{-7} per year.

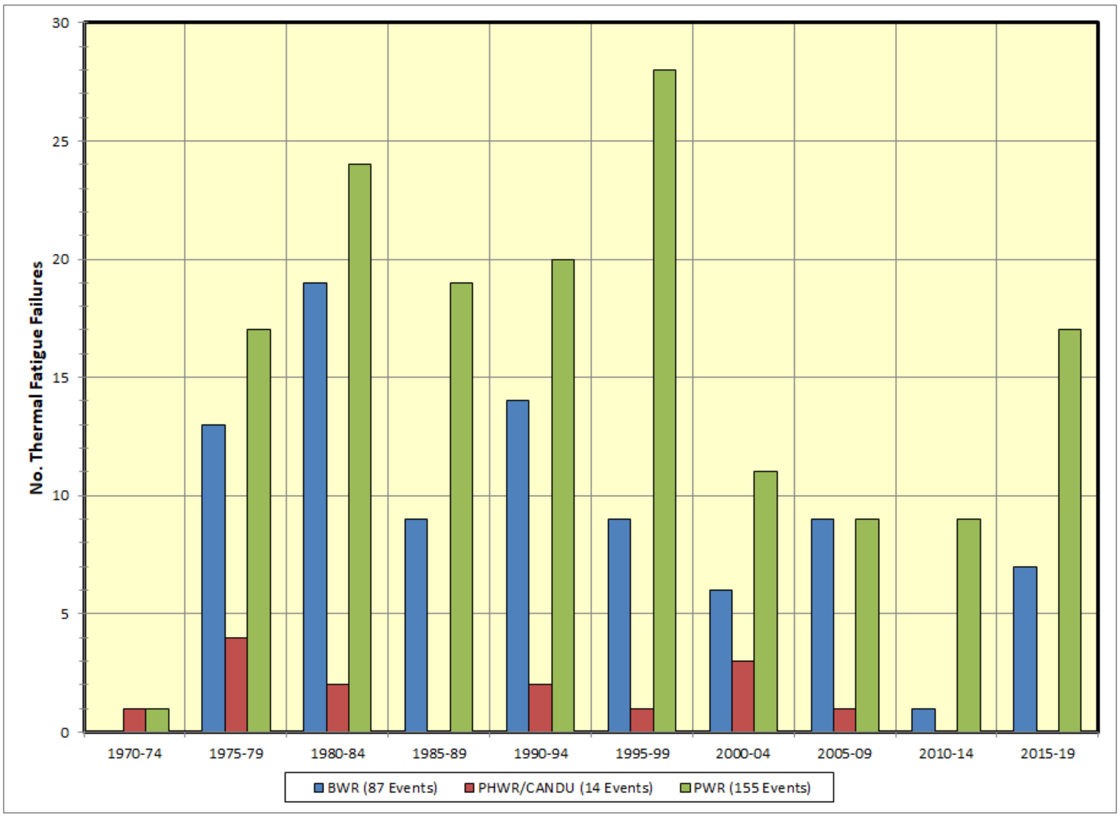


Figure 8-5: Thermal Fatigue OE Data

Veeramany [113][114] explores different assumptions about the pipe flaw initiation time on the basis of the coefficient of variation[69]. The Weibull distribution is used to model flaw initiation and the distributions for the time spent between various states are represented in a matrix form called the "kernel matrix" of the process. The kernel matrix and the initial state occupied by the process completely define the stochastic behavior of the semi-Markov process. Given these as the input, the statistical time behavior of the process is described by a system of linear integral equations.

8.4 Markov Model Application Insights

A demonstration of a MRP model of loss-of-coolant-accident (LOCA) due to IGSCC is presented in this section. The original analysis was performed in support of the U.S. Nuclear Regulatory Commission's "Expert Panel LOCA Frequencies" [71]. The analysis was done to demonstrate the impact of different RIM strategies on LOCA frequency as well as to provide insights into LOCA frequencies beyond 40 years of nuclear power plant operation. An updated perspective on the of MRP models is also presented in this section.

8.4.1 Basic Markov Model Formulation

The subject demonstration involves the assessment of Boiling Water Reactor (BWR) Reactor Recirculation pipe break induced LOCA due to intergranular stress corrosion cracking (IGSCC) in similar- and dissimilar metal weldments. A 4-state Markov model is used and the assessment is limited to piping fabricated using AISI Type 304 stainless steel and Alloy 82/182 nickel-base materials. A summary of the root parameters of the Markov model is given in Table 8-2. And the results in terms of the hazard rates are given in Figure 8-6.

[69] The dimensionless coefficient of variation provides a relative measure of data dispersion compared to the mean.

Table 8-1: Four-State Markov Model Root Input Parameters

Parameter	Assumed or Estimated Value	Basis
ω	2.1×10^{-2}/year {=(.25) × (.90)/(10+(200/8760))}	Element assumed to have a 25% chance of being inspected for flaws every 10 years with a 90% detection probability. In the given example detected flaws will be repaired in 200 hours
μ	7.92×10^{-1}/ year {=(.90) × (.90)/(1+(200/8760))}	Element is assumed to have a 90% chance of being inspected for leaks once a year with a 90% leak detection probability
ρ_C	Table 13, 14 and 15 of NUREG-1829, Appendix D	The basis is developed in Sections 4 and 5; NUREG-1829, Appendix D.
λ_C	Table 13 and 14 of NUREG-1829, Appendix D	The basis is developed in Sections 4 and 5; NUREG-1829, Appendix D.
ρ_F	2.0×10^{-2}/year	If the element is already leaking, the conditional frequency of ruptures is assumed to be determined by the frequency of severe overloading events; the given value is equal to the frequency of severe water hammer (estimated from OE data).
ϕ	Variable (for IGSCC $\phi = 7.58 \times (\lambda_C + \rho_C)$)	The occurrence rate of a flaw is estimated from service data. As an example, IGSCC in the BWR operating environment will create ca. 7.58 flaws for every through-wall leak that is observed (estimated from OE data; year 2003 state-of-knowledge)).
P_{FI}	1 or 0	Probability per inspection interval that the pipe element will be included in the inspection program.
P_{FD}	Variable (see text above for details)	Probability per inspection interval that an existing flaw will be detected; POD. A chosen estimate is based on NDE reliability performance demonstration results and difficulty and accessibility of inspection for particular weld.
P_{LD}	Variable (0 – no leak detection to 0.9 for leak detection using current methods/technology)	Probability per detection interval that an existing leak will be detected. Estimate based on system, presence and type of leak detection system, and locations and accessibility.
T_{FI}	10 years (per ASME XI)	Flaw inspection interval, mean time between in-service inspections.
T_{LD}	Variable (1.5 – Once per refueling outage / 1.92E-2 – weekly / 9.13E-4 - shiftly)	Leak detection interval, mean time between leak detections. Estimate based on method of leak detection; ranges from immediate/ continuous to frequency of routine inspections for leaks (incl. hydrostatic pressure testing).
T_R	Variable	Mean time to repair the affected piping element given detection of a critical flaw or leak. Estimate of time to tag out, isolate, prepare, repair, leak test and tag into service.

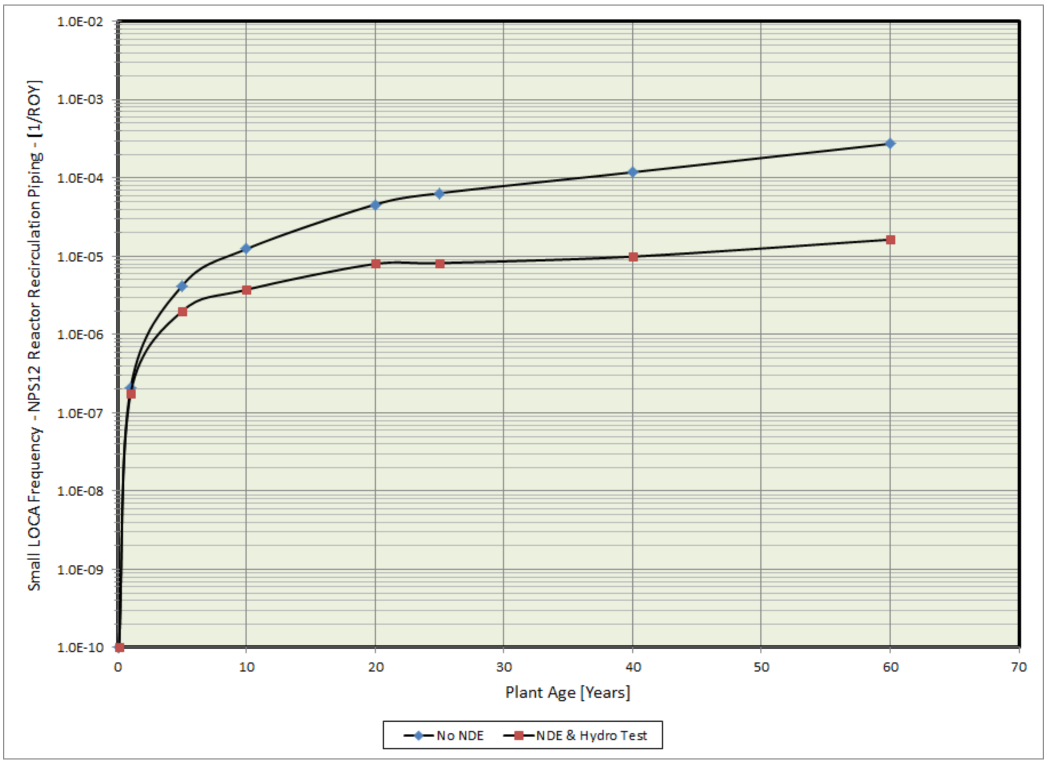


Figure 8-6: Hazard Rates for BWR Reactor Recirculation Pipe Break

8.4.2 Modified Markov Model Formulation

There are several limitations of a MRP model approach. The results that are presented in Figure 8-6 are counterintuitive. Temporal changes (Figure 8-7) are attributed to new aging management processes, including the use of degradation resistant material, etc. One way of overcoming the assumption of a constant flaw rate is to perform a step-by-step analysis to account for temporal changes in the observation of flawed welds.

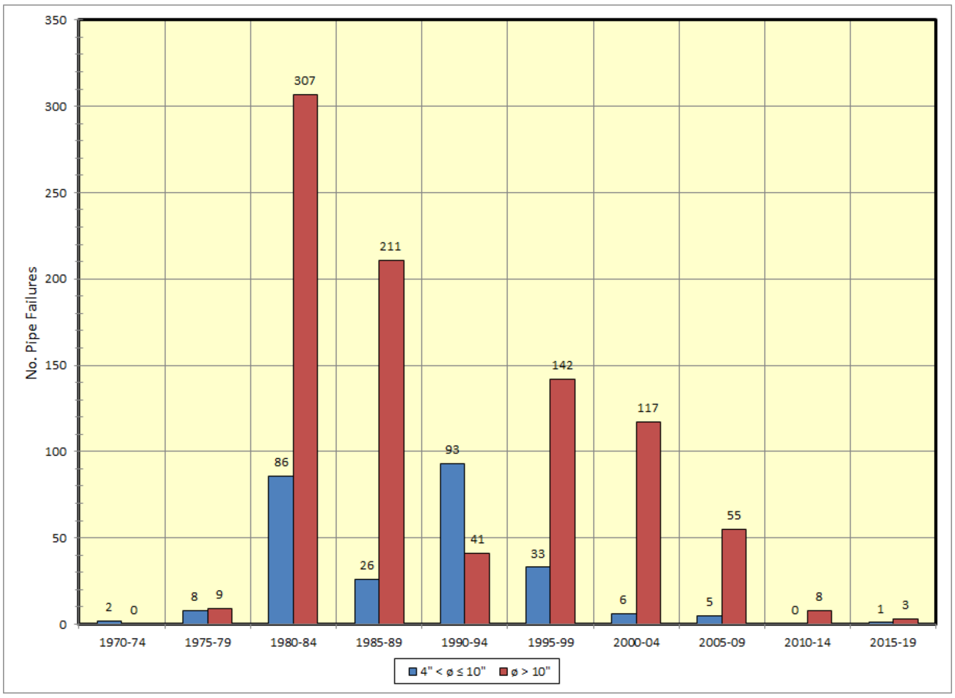


Figure 8-7: Temporal Changes in the IGSCC Occurrence Rate

The example in Section 8.4.1 was re-calculated by using a new flaw rate for IGSCC to reflect post-1990 changes in inspection and mitigation practices. The results of the re-analysis are summarized in Figure 8-8.

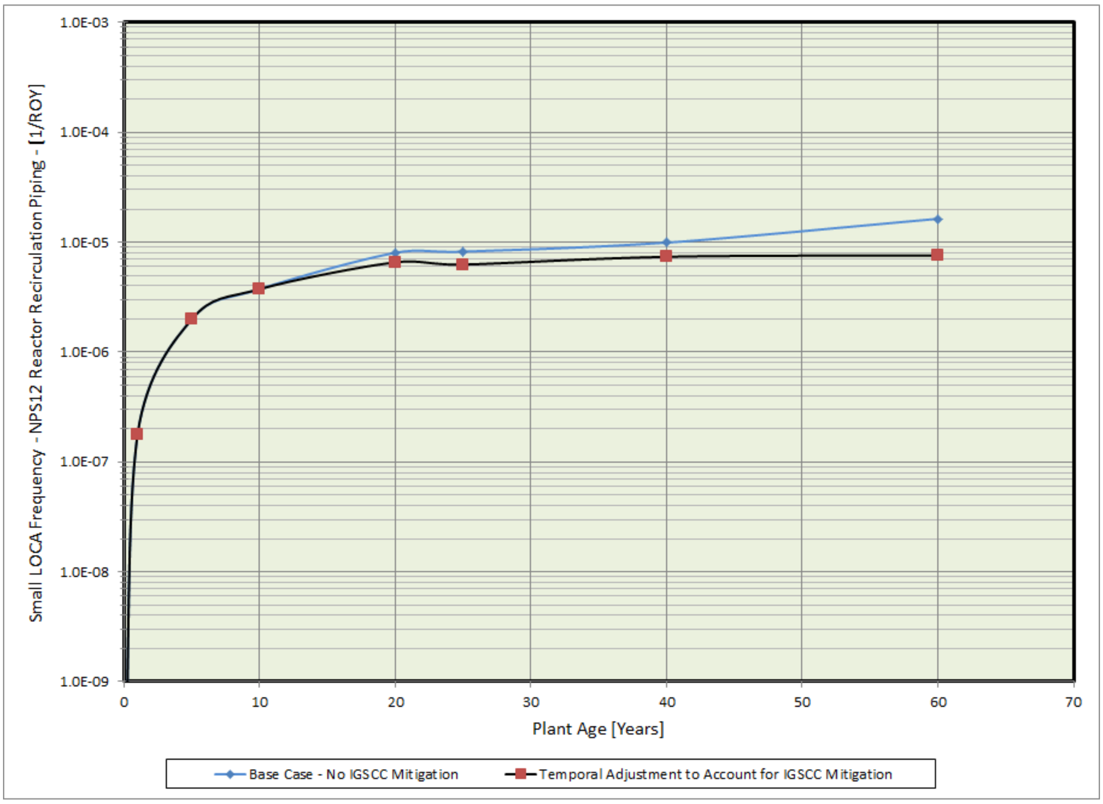


Figure 8-8: Hazard Rates for Pipe Break with & without IGSCC Mitigation

9. ASSESSEMENT OF AGING EFFECTS

Documented in this section is an overview of methods to assess the effectiveness of different aging management strategies (e.g. augmented NDE programs, improved water chemistry control, use of new materials that are resistant to environmental degradation). The prospects for deriving statistics on "aging management factors" are explored through examples.

9.1 Background

Nuclear power plant materials aging research has been ongoing for about five decades; References [115] through [133]. Reviews and evaluations of laboratory data and field experience data have been central to this research. As one example, in the United States the Nuclear Regulatory Commission (NRC) has sponsored the Nuclear Plant Aging Research (NPAR) program to collect information about aging phenomena.[70] Mainly, this program collected a large body of qualitative information on materials aging phenomena and their potential effects on plant safety. The NPAR-collected information has supported the formulation of the License Renewal Rule (10 CFR Part 54, 1995)[71], and it has been utilized in subsequent NRC-sponsored research on quantitative aging management assessments [121][122][123].

Report NUREG/CR-5378 (1992) [121] documents a methodology for identifying and quantifying age-dependent component failure rates. Central to this approach is a detailed data analysis of component operating experience data, including an evaluation of failure trends. The chosen example is an Auxiliary Feedwater System and pipe failure caused by flow-accelerated corrosion (FAC). Report NUREG/CR-6157 (1994) [123] surveys the NRC sponsored work on aging of SSCs and how it relates to plant risk. A preliminary framework is developed for how to integrate aging of SSCs into probabilistic safety assessment (PSA) and it includes the identification of necessary data for such integration.

Building on a "physics-of-failure" (PoF) concept, report NUREG/CR-5632 (2001) [123] documents a technical approach that uses a so called "compound plug-in" calculation in which an aging effect such as FAC is first modeled using a load-capacity probability calculation. To obtain the probability that a pipe segment subject to FAC will rupture, the extent of wall thinning is used to determine the remaining pressure capacity. This pressure capacity would then be "balanced" against the pressure loading that the pipe segment would see over the course of operation. This type of "load-versus-capacity" evaluation is prevalent in the reliability literature and falls under various names such as load-capacity or stress-strength analysis.[72] Next, the FAC piping failure event is inserted in a fault tree that models a loss of feedwater initiating event. The 'compound plug-in' module computes the feedwater piping failure probability due to FAC and as a function of operational time.

The United States Department of Energy (USDOE), the Idaho National Laboratory (INL), the Pacific Northwest National Laboratory (PNNL) and the Electric Power Research Institute (EPRI) have entered into a collaboration to conduct coordinated research to identify and address the technological challenges and opportunities that likely would affect the safe operation of existing nuclear power plants over the projected long-term time horizons, up to

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[70] http://www.nrc.gov/reactors/operating/licensing/renewal/guidance.html#npar

[71] https://www.nrc.gov/reading-rm/doc-collections/cfr/part054/

[72] See for example Chapter 8 ("Reliability Physics Models and Statistical Parameter Estimation") in Probabilistic Reliability: An Engineering Approach by M.L. Shooman (1968, McGraw Hill Book Company, ISBN 07-057015-9).

80 years. The Light Water Reactor Sustainability (LWRS) Program[73] addresses materials aging research and aging management assessment research is done under the Risk-Informed Safety Margins Characterization (RISMC) Pathway. The research is performed to provide methods and techniques for assessing the effect of material degradation on plant operation and the ability plant safety barriers to respond to plant transients; Figure 9-1.

Initiated within the framework of the European Commission's Joint Research Centre (JRC) Sixth Framework Program (FP-6) Institutional Project No. 3131 "Analysis and Management of Nuclear Accidents", the "Network on the Use of PSA for Evaluation of Aging Effects to the Safety of Energy Facilities" (EC-JRC-IE Aging PSA Network)[74] was created in 2004. Several Network Project and Meetings have been organized [124][125]. Reference [126] is a summary of case studies that have been performed within the framework of the Aging PSA Network to demonstrate statistical approaches to identify component aging factors. Since 2015 the Aging PSA Network is no longer active, however.

Relevant research in the area of aging and life extension is pursued by the offshore oil and gas industries. As an example, in Norway the Petroleum Safety Authority in collaboration with the Norwegian research institute SINTEF supports the development of methods and techniques for the assessment of safety margins of structures [127]. Furthermore, the no longer active "Project Group on Aging" of the European Safety, Reliability and Data Association (ESReDA) has organized seminars and workshops on aging management topics [128].

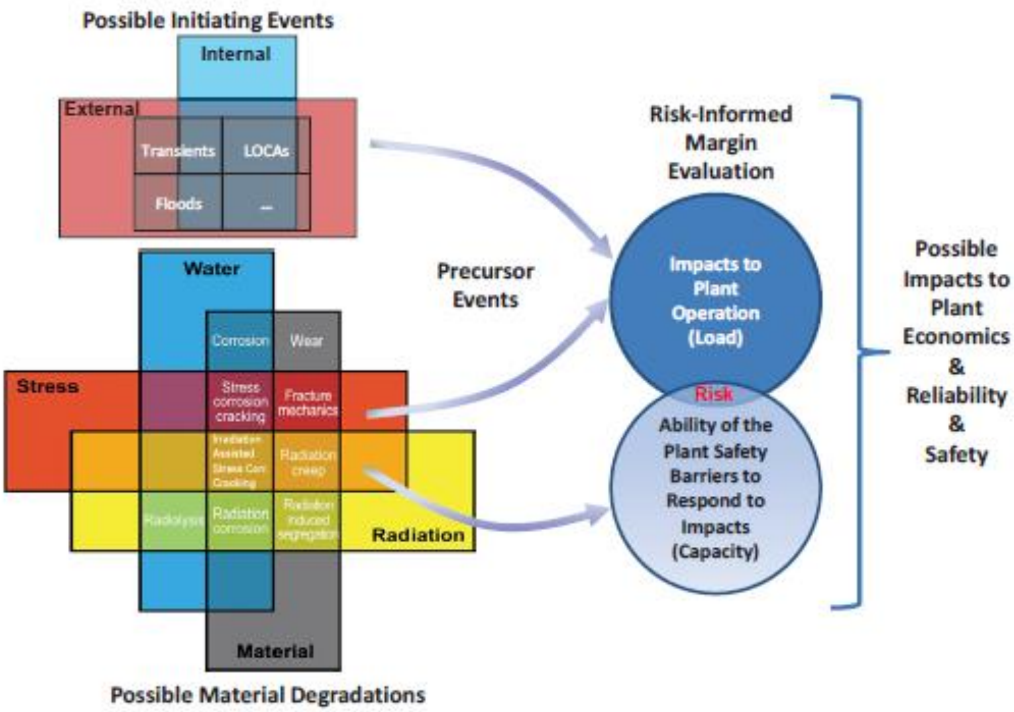

Figure 9-1: Representation of the Interaction of Material Degradation Mechanisms that May Impact Plant Operations & Safety Barriers if Left Unmitigated[75]

[73] https://lwrs.inl.gov/SitePages/Home.aspx
[74] Institute for Energy, European Commission's Joint Research Centre, Petten, The Netherlands.
[75] Reproduced from INL/EXT-11-22977 (R4): Risk-Informed Safety Margins Characterization (RISMC) Pathway Technical Program Plan, September 2016.
https://lwrs.inl.gov/RiskInformed%20Safety%20Margin%20Characterization/Risk-informed_Safety_Margins_Charterization_(RISMC)_Pathway_Technical_Program_Plan.pdf

In analyzing the operating experience data for potential aging trends different methods can be explored, from simple visual examination of data plots [129][130] to formal statistical tests (to obtain parametric models of aging) [131][132] and use of "physics of degradation" models [133][134][135][136][137]. The goal of aging management assessments include the derivation of "aging factors" that reflect current operating experience and state of knowledge of material aging. The quality of statistical approaches hinges on access to field experience data that is sufficiently current and complete.

Illustrated in Figure 9-2 is an example of how field experience data evolves over time. The field experience datasets in Figure 9-2 were systematically collected using a continuos and systematic data collection and analysis processwhich is based on a uniform data validation scheme. Any field experience data collection effort provides a retrospective assessment of (or historical perspective on) material performance over time. Comparing data collection Period 1 and Period 2 indicates a) an increase in number of failures over time, and b) the latent effects inherent in any operating experience data project. While Period 2 points to some potentially significant trends in the field experience. It also highlights the effects of a continuos data collection process whereby pre-2004 field experience data is added to the Period 2 data collection, etc. The quality of a statistical aging assessment factor (AAF) analysis is therefore higly correlated with the completeness of a field experience database. A major analytical challenge in quantitative aging management assesments is to predict future material performance, especially when a potential aging effect evolves very slowly.

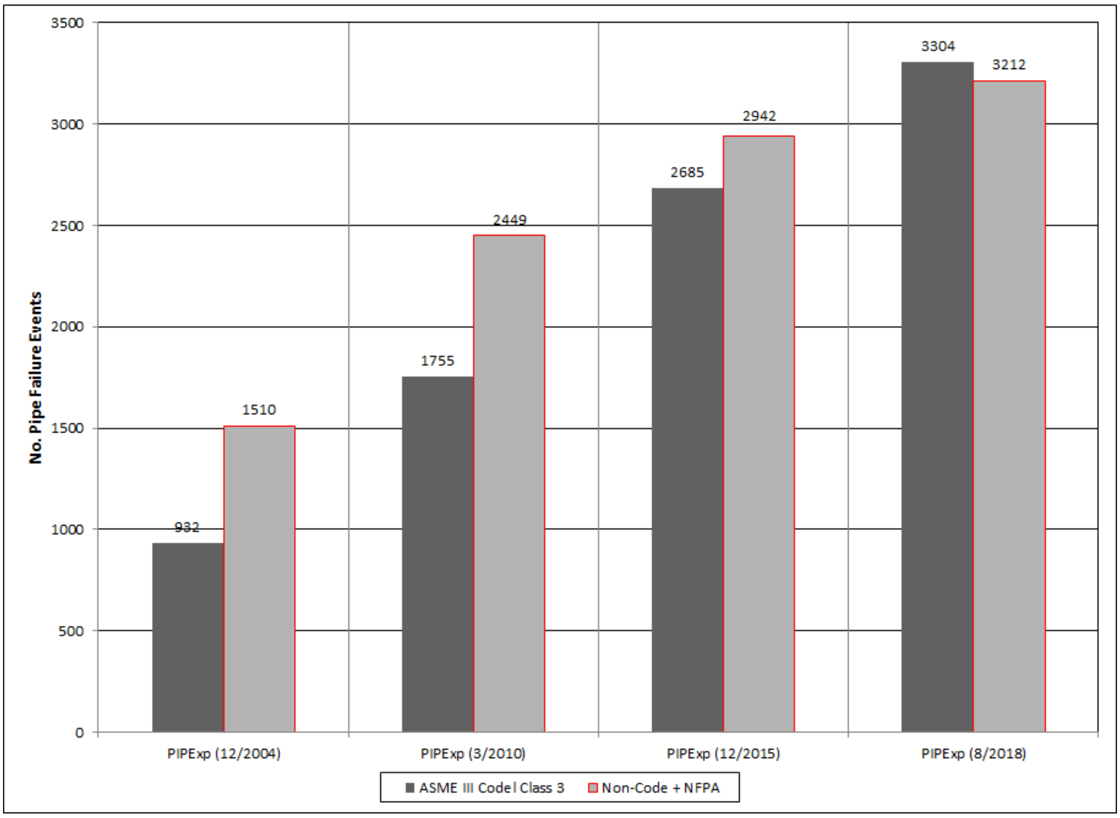

Figure 9-2: Evolution of Carbon Steel Piping Field Experience Data

9.2 Aging Management Nomenclature

Multiple technical considerations enter into the assessment of aging effects on risk metrics. It involves the following factors:

- Degradation of Non-Replaceable Items. The design of certain systems-structures-and-components (SSCs) is based on a pre-determined service-life. This is so not only because of cost of replacement, but also because of radiation protection considerations. These SSCs require close surveillance to ensure that their aging characteristics are as expected. Examples include the reactor pressure vessel, designed for a service life of at least 40 years. The main mode of vessel aging is irradiation which affects the mechanical properties of steel. Plant owners therefore take steps to predict any changes to the vessel's mechanical properties and demonstrate that despite any identified changes, the vessel is able to withstand all normal and abnormal operating conditions.
- Degradation of Replaceable Items. SSC aging is the result of phenomena such as flow-assisted wear and corrosion. Therefore, the design and construction of SSCs must consider choice of material and maintainability. Carefully implemented surveillance and maintenance programs allow for repair and replacement as warranted by operating conditions and test results.

Aging effects include loss of material, cracking, loss of fracture toughness, loss of preload, loss of heat exchanger performance, loss of adhesion, and change in material properties. Aging effect definitions of relevance for this report include:

- Change in Material Properties. Any change in a material which is detrimental to that material's ability to meet its design requirements. Mechanisms that may result in a change in material properties include galvanic corrosion, thermal degradation, strain aging and irradiation.
- Cracking. Service-induced cracking of materials includes both crack initiation and crack propagation within base metals and weld metals. Aging mechanisms that may result in crack initiation and propagation include fatigue, intergranular attack and stress corrosion cracking.
- Loss of Fracture Toughness. Changes in the material properties of a metal such that design requirements are potentially compromised. Aging mechanisms that contributed to loss of fracture toughness include irradiation embrittlement and thermal embrittlement.
- Loss of Material. A reduction in the material content of a component or structure may occur evenly over the entire component surface or be confined to localized areas. Aging mechanisms that may result in loss of material include: corrosion of below ground piping, corrosion of piping encased in concrete, crevice corrosion, erosion-corrosion, flow-accelerated corrosion, galvanic corrosion, general corrosion, selective leaching, microbiologically influenced corrosion, pitting, thermal degradation, and wear.

Some aging effects are readily observable (e.g. through leak detection, visual examination or non-destructive examination, NDE) and others ("Change in Material Properties" and "Loss of Fracture Toughness") are observable via destructive examination followed by mechanical tests to measure change in fracture toughness[76]. The focus of this report is on the "readily observable" aging effects.

[76] http://www.calce.umd.edu/TSFA/Hardness_ad_.htm (University of Maryland)

9.3 Carbon Steel Degradation Mechanisms

Carbon and low-alloy steels make up a major portion of the materials of construction for pressure retaining components in water cooled reactors (WCRs). The choice of these materials was based on their low cost (in comparison to the higher-alloyed materials such as stainless steels and nickel-base alloys), their good fracture resistance (in unirradiated condition), and their ease of fabrication. Environmentally assisted degradation has been observed in these materials by degradation modes that were not considered at the early WCR design stages. This section surveys selected carbon steel degradation mechanisms and the current state-of-knowledge of their significance relative to WCR operability and safety. The information in this section builds on information that has been assembled and amended from multiple, information sources; References [138] through [155].[77]

9.3.1 Manifestations, Monitoring & Assessment of Material Degradation

Aging management begins with plant design. Many design criteria explicitly or implicitly address aging. The long-lived SSCs in a nuclear plant, for example, were originally designed with sufficient margins to meet minimum lifetime requirements. Nuclear power plant piping systems are designed with industry codes based on assumed service conditions, with some allowance for pipe wall thinning from erosion and corrosion. In addition, fatigue analyses used to establish design criteria for piping, pumps, and valves estimate the number of on/off cycles a power plant experiences during its life, as well as the resulting temperature variations and thermal stresses from those cycles.

To account for a variety of engineering uncertainties at the time of plant design, original SSC designs were generally based on what were then thought to be conservative assumptions of operating and material conditions. Decades of operating experience and research have determined that some of the original design assumptions were in fact not conservative, while others were. As this experience suggests, aging degradation rates for SSCs are in some cases quite different than originally anticipated. Also, some of the early operating experience is no longer valid because of piping system design changes and/or implementation of effective aging management programs.

Improvements in analytical techniques and material property examination techniques have allowed the review of original plant design bases for more accurate assessments of material degradation. More accurate predictive methods may allow for less conservatism in assessing the adequacy of SSC performance and predicting their remaining useful life.

In evaluating potential ageing effects a differentiation is made between short- and long-term effects. Short-term "aging effects" (e.g. equipment wear-out) tend to be highly predictable and, hence, pose a less challenging analysis problem than the long-term aging effects for which there is limited service experience data available to support statistical analysis for trends. An aging effect can be defined as:

- Age-dependent change in a passive system, structure, or component (SSC) performance caused by an active degradation mechanism or by synergistic effects of multiple degradation mechanisms. Examples of changes in performance include:
- Change in structural integrity of a piping or non-piping passive component. This change may be characterized by the estimated aging factor (AF), which can be calculated as the ratio of a projected hazard rate to the present-day hazard rate. This could be the hazard rate at end of current operating license or at some time increment from the current state-of-knowledge

[77] https://antinternational.com/ is another source of current information on material degradation issues.

- Change in success criteria or functionality. Such a change can occur due to degraded heat transfer capability of a heat exchanger due to fouling or plugging of heat exchanger tubes. Similarly, a worn pump impeller would affect the shape of a pump curve and hence the flow capacity.
- Change in physical or chemical properties resulting from one or more active degradation mechanisms.

The prospects for developing phenomenological ageing models hinge on a well-defined characterization of what constitutes an aging effect as opposed to readily identifiable and correctable, well understood temporal changes in equipment performance and human performance (e.g. via non-destructive examination). Access to high quality data that reflect several decades of plant operation is an important element of the analysis of potential ageing effects.

The physical degradation of metallic passive reactor components involves a complex interaction of material properties (e.g. chemical compositions, fracture toughness), operating environment (e.g. local flow conditions, pressure, temperature, water chemistry, and loading conditions. The effects of a certain degradation mechanism can be mitigated or eliminated through the applications of proactive aging management, including in-service inspection, stress improvement, chemical treatment of process medium. As an example, the CODAP database structure is a reflection of the physics of material degradation, and the database attempts to capture the subtleties of the many factors that contribute to material degradation and failure. Therefore, by utilizing the tools and techniques for querying the event records that are included, for example, in CODAP a basis exists for in-depth evaluation of temporal changes in the failure data, including positive and negative trends in passive component performance.

9.3.2 Strain Aging of Carbon & Low-Alloy Steels

The cracking of steam, feedwater and condensate piping systems due to strain-induced stress corrosion cracking (SICC) has been extensively analyzed for German BWRs where these components have been fabricated with relatively fine-grained, higher-strength steels (WB 35, WB36; see Appendix B) that allow the use of thinner walled piping without stress relief treatment of the welds. The features that aggravated the cracking susceptibility in these incidents were:

- Dynamic straining associated with, for instance, reactor start-up or thermal stratification during low feedwater flow or hot standby conditions. Such operations lead to a wide range of applied strain rates, and would be expected to increase the crack propagation rate.
- High local stress at or above the high temperature yield stress, thereby giving a lack of plastic constraint at the incipient crack tip, and consequently an anomalous increase in crack propagation rate due to the effective increase in crack tip strain rate. Such high local stresses were attributed in the failure analyses to weld defects (e.g. misalignment of weld edges, presence of root notches, etc.), piping fit-up stresses and, in some cases inadequate pipe support at elbows. The combination of this high stress adjacent to the weld and the high applied strain rate led to a distribution of multiple cracks around the circumference of the pipe that was no longer confined by the asymmetric azimuthal distribution of weld residual stresses. These cracks propagated on separate planes and did not interlink, thereby potentially alleviating concerns about leak before break (LBB) safety analyses that would be raised for a fully circumferential crack propagating evenly through the pipe wall.

- Oxidizing conditions, in conjunction with intermediate temperatures and potential anionic impurities. The affected piping generally operates in the temperature region 220°C - 250°C where, as discussed earlier, the cracking susceptibility is at a maximum. Moreover, cracking was often observed in stagnant steam lines where the dissolved oxygen concentration may be in excess of 100 ppb, that is well in excess of the 30 ppb quoted to be the “threshold” value above which strain-induced cracking is to be expected in these steels at 250°C. This conjunction of environmental factors was further aggravated by the fact that during reactor shut-down stagnant water was sometimes left exposed to air in horizontal portions of piping; pitting and general corrosion occurred under these low temperature conditions, and these pits were observed to act as crack initiators during subsequent power operation conditions.

SICC was identified in several German Boiling Water Reactor (BWR) plants in the 1970s. This led to the introduction of the basic safety concept in Germany built on the use of steels with high fracture toughness and moderate strength. An extensive replacement action was performed in five operating BWRs involving the piping inside the drywell. Between 1980 and 1985 all high strength steels were replaced there by steels of high fracture toughness with carefully controlled chemical composition. No further cracking has been observed since that time.

In contrast, secondary-side piping systems made of high-strength ferritic steels were not replaced at that time. Instead a program was implemented to investigate the critical boundary conditions for SICC. Based on the results of this programme, pipe sections in the turbine building sensitive to SICC were identified. These pipe sections became the subject of augmented NDE. Several operational and systems engineering provisions were implemented to avoid the critical boundary conditions, such as reduction of the oxygen content in the water phase during start-up and avoidance of corrosion during the shut-down period. In most cases, partial replacement or local repair was performed. However, some incidents of minor safety significance occurred due to SICC outside the containment in the 1990s, which indicated that the issue had not been fully resolved.

9.3.3 Microbiologically Influenced Corrosion

Microbiologically influenced corrosion (MIC) refers to corrosion which results from the presence and activities of microorganisms. MIC can result from microbial processes that produce corrosive environments such as organic acids or lower valence sulfur. The modes of corrosion, which can result from microbiologically produced local environments, include general corrosion (GC), pitting (PIT), crevice corrosion, dealloying, galvanic corrosion, intergranular corrosion (IGC), stress corrosion cracking (SCC), and corrosion fatigue (CF).

MIC is relatively common in LWR systems such as fire water, service water and low temperature cooling water systems (Figure 9-3) and components, and typically occurs in Light Water Reactor (LWR) plants in two general locations. One is on external surfaces where there is moisture and other materials, such as organic debris buildup; and slimes; secretions; etc. which contain nutrients suitable for bacterial or fungal growth. The second occurs on internal surfaces in low temperature components; primarily those where water is flowing slowly or is periodically flushed; both situations provide a good supply of nutrients for microbiological activity and growth. This is particularly true for systems where deposits can build up, and where these deposits could accumulate bacterial or fungal populations by exposure to water that has been air-exposed. MIC manifestations tend to be in the form of localized corrosion and pinhole leaks.

Stagnant systems with no replenishment of nutrients (e.g. not exposed to air) are not favorable for significant MIC activity. Vertical “dead legs”, with water flowing by the end of the “dead leg,” are areas particularly at risk for bacterial growth. Periodic flushing, by introducing nutrients and bacteria, is one of the major factors in promoting MIC in piping and tanks. Usually this flushing is carried out for testing purposes and consideration should be given to reducing the frequency of such testing to minimize the risk of MIC. Chemicals that are common in LWR waters can affect the growth of microorganisms, for instance hydrazine and boric acid. MIC is often associated with fouling, the fouling being a combination of bacterial colonies and associated corrosion products, and the MIC damage found under the deposits. These deposits can be significant, resulting in blockage of piping and much reduced water flows MIC and MIC-related fouling have been found in a wide range of systems, from fire protection and service water systems to ECC storage systems and spent fuel pools.

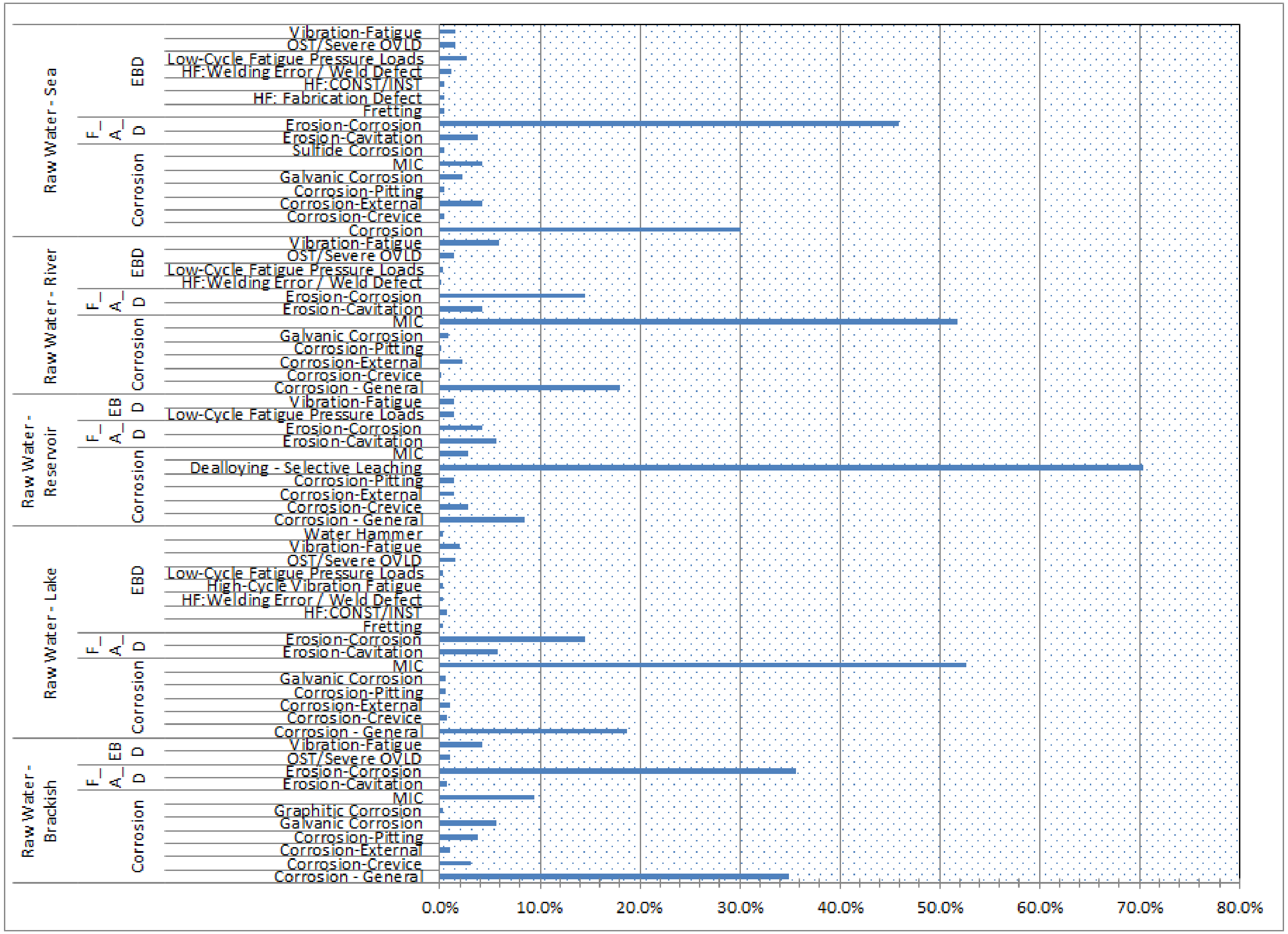

Figure 9-3: Degradation Mechanisms in Raw Water Piping Systems[78]

[78] Based on an evaluation of 2374 Service Water pipe failure events (October 2018).

Most MIC mitigation efforts require that physical cleaning, chemical cleaning, or both be used as a first step in the process. The application of a biocide is usually a second step following cleaning, or in some cases a supplementary factor in conjunction with the chemical/physical cleaning. During mitigation, the environmental and operational conditions of the system are very atypical to those during actual operation.

MIC has historically been a significant degradation mechanism in LWRs in terms of cost to manage it, and is also an issue in CANDU Service Water Systems and Fire Protection Systems (FPS). According to Tapping (Chapter 17 of Reference [138]), the number of FPS pipe failures within Canadian nuclear power plants “has been relatively low, but some have involved severe degradation and rupture.”

The most severe example of degradation was in a common ring header fed from four separate High-Pressure Service Water Systems at Pickering NGS. Numerous secondary headers lead off the ring header to feed risers supplying fire hose cabinets arranged vertically on the elevations above and below the secondary headers. Each secondary header forms a dead-leg[79] attached to a continuously flowing system. After nearly 30 years of operation a number of weeping leaks[80] were seen, appearing primarily in the risers at the same elevation as the ring header, as well as a few leaks in piping on the elevation below. External inspection revealed that most of these leaks were located at either welded or threaded pipe joints, or on the seam welds. A number of leaking sections were replaced and the removed pipe was sent for analysis. For the ring header it was determined that the station conditions were ideal for iron oxidizing bacterial growth. With a constant flow of oxygenated water, there was a steady supply of oxygen to diffuse into the headers and risers.

9.3.4 Stress Corrosion Cracking of Carbon Steel

Stress corrosion cracking (SCC) failures in carbon steel components are well-known phenomena, which have been observed in fossil fuel plants, chemical plants, and paper plants. Such failures are frequently associated with hydroxide, nitrate, chloride, and carbonate environments. Historically, SCC of carbon steel components in the nuclear industry has been of far less concern than materials such as austenitic stainless steels and nickel-base alloys [142][143][144][145]. Typically, SCC failures of carbon steel in the nuclear industry have been found in components such as feed water tanks, deaerators, and primary piping in heavy water reactors. SCC failures in carbon steel component cooling water systems have also been observed in the nuclear industry. Such failures have been correlated to high residual stresses (i.e., association with welds) as well as environmental conditions such as aerated water with certain corrosion inhibitors, biofouling, and copper depositions.

SCC of CANDU feeder piping was first observed on a CANDU-6 (Point Lepreau NGS) [146] and occurred at outlet tight-radius bends. The original belief was that the axially oriented feeder cracks were initiated from the inside but subsequent examinations determined that the cracks initiated on the outside surface as well. From all the data collected during destructive post-examination of removed feeders, two failure mechanisms were postulated: Intergranular Stress Corrosion Cracking (IGSCC) assisted by hydrogen for inside initiated cracks, and Low Temperature Creep Cracking (LTCC) assisted by hydrogen for outside initiated cracks. Both mechanisms would not be active without the contribution of stresses, and in the case of feeder bends, residual stresses from bending was the necessary condition to propagate an incipient

[79] An area of a piping system that rarely see flow, yet are still exposed to process, even if not explicitly cut off. Dead-legs are prone to contamination and corrosion. They can take the form of blanked branches, lines with normally closed block valves, lines with one end blanked, support legs, stagnant control valve bypass piping, spare pump piping, level bridles, relief valve inlet and outlet header piping, pump trim bypass lines, high-point vents, sample points, drains, bleeders, and instrument connections.
[80] See for example event reports P-2010-03448, P-2012-11551, P-2012-15599 and P-2013-05920.

crack. Not all CANDUs are at risk and only those stations with none stress-relieved bends are susceptible to feeder bend cracking.

9.3.5 Flow-Accelerated Corrosion

Flow accelerated corrosion (FAC, also termed flow-assisted corrosion, and sometimes wrongly erosion-corrosion) leads to wall thinning (metal loss) of steel piping exposed to flowing water or wet steam; Figure 9-4. The wall thinning is the result of the dissolution of the normally protective oxide layer formed on the surfaces of carbon and low alloy steel piping. The rate of metal loss depends on a complex interplay of several parameters including water chemistry, material composition, and hydrodynamics. Carbon steel piping components that carry wet steam are especially susceptible to FAC.

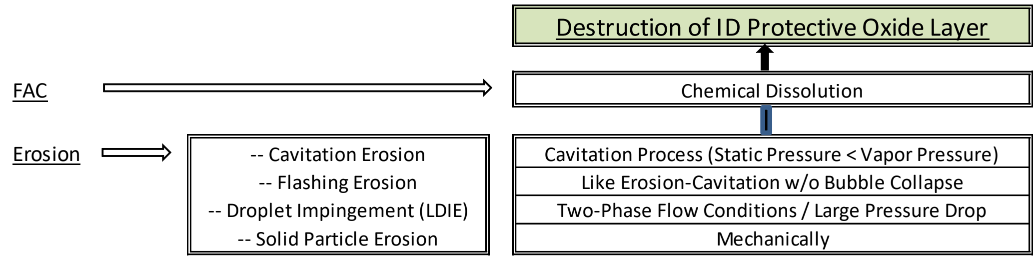

Figure 9-4: Examples of Flow-Assisted Pipe Wall Thinning Mechanisms[81]

The dominant controlling variables that promote FAC are temperature, fluid velocity, fluid pH, the water amine, oxygen content, steam quality, void fraction of the fluid, piping geometry, and the pipe material composition. It is important that FAC degradation is diagnosed correctly so that the appropriate mitigation methods can be implemented. Historically the terminology was ambiguous since erosion-corrosion was used for both the chemical mechanism now known as FAC and the mechanisms in which the oxide is broken down mechanically by the impingement of particles, solids or gaseous bubbles. There are also differences in the surface morphology of FAC and erosion-corrosion. Single phase FAC has a scalloped or orange-peel appearance and two phase damage often has a characteristic pattern known as tiger striping. These surface features are absent in surfaces damaged by erosion mechanisms. Another difference is that FAC is often more widespread than the localised erosion damage. It should also be noted that most of the codes developed to predict wall thinning do not distinguish between FAC and erosion-corrosion.

FAC mainly affects the secondary circuit of pressurized heavy water reactor (PHWR) plants [146] and pressurized water reactor (PWR) plants, but also BWR feedwater piping is susceptible to single phase FAC induced damage. In all plant types, several main steam line sub-systems, including the high-pressure turbine exhaust piping, the turbine crossover piping, the extraction steam lines, and certain straight portions of the steam lines are susceptible to two-phase FAC. The moisture content in the main steam leaving the reactor pressure vessel is about 0.1% and increases as the steam reaches the main turbines. The high moisture content in the steam extraction and exhaust lines and turbine crossover lines makes these lines particularly susceptible to FAC. The main steam line pipes are not susceptible to FAC unless moisture is present. The feeder tubes of CANDU-600 plants have experienced FAC damage during the pre-refurbishment phase of operation [147][148]; Point Lepreau NGS and Wolsong Units 1 & 2, respectively). Feeders at other CANDU Units have also experienced FAC damage The FAC controlling parameters include [149][150]:

- Effect of Temperature. An important variable affecting the FAC resistance of carbon and low alloy steels is temperature. Most of the reported cases of FAC damage under single-

[81] Reproduced from NEA/CSN/R(2014)6 https://www.oecd-nea.org/nsd/docs/2014/csni-r2014-6.pdf.

phase conditions have occurred within the temperature range of 80 to 230°C, whereas the range is displaced to higher temperatures (140 to 260°C) under two-phase flow. The exact location of the maximum wear rate changes with pH, oxygen content, and other environmental variables. Experience has shown that the wear rate is highest at around 150ºC and increases with fluid velocity. Furthermore, FAC can occur in low temperature single phase systems under unusual and severe operating conditions.

- Effect of Flow Velocity. Flow rate of the liquid has been found to have a linear effect on the FAC wear rate. As higher velocities are experienced, higher wear rates are expected. Since the enhanced mass transfer associated with turbulent flows is the fundamental process in the accelerated dissolution of the pipe wall protective oxide layer, the effect of flow is best described in terms of the mass transfer coefficient, which is a function of flow velocity and geometry. Local flow velocities can differ by a factor of 2 to 3 from the bulk flow velocity.

- Effect of Fluid pH. FAC wear rates are strongly dependent on pH. In general, increasing the pH value reduces the wear. The FAC wear rate of carbon steels increases rapidly in the pH range of 7 to 9, and drops sharply above pH 9.2. As the fluid becomes more acidic, more pipe wall losses are expected. The pH value can be affected by the choice of control agents (e.g., morpholine or ammonia) and by impurities in the water. In two-phase flows the critical parameter is the pH of the liquid phase. This can be significantly affected by the partitioning of the control agent between the steam and liquid phase. There is no adjustment of pH performed in BWR plants.

- Effect of Oxygen. FAC rates are inversely affected by the amount of dissolved oxygen (DO) in the feedwater, and too low an oxygen level is harmful to carbon steel piping. The FAC rate decreases rapidly when the water contains more than 20 ppb oxygen, but the precise oxygen level required to prevent FAC depends on other factors such as pH and the presence of contaminants.[82]

- Effect of Alloy Additions. The FAC rate is highest in carbon steel piping with very low levels of alloying elements. The presence of chromium, copper and molybdenum, even at low percentage levels, reduces the FAC rate considerably. The relative corrosion rate of steels is reduced by 80 % at chromium content as low as 0.2%. The FAC rate is decreased by a factor of 4 with the steel type 2-1/4% Cr and 1% Mo (2-1/4 Cr- 1 Mo steel). Austenitic stainless steels are virtually immune to FAC.

- The Entrance Effect. About 30 years ago new FAC wear effect was documented. This effect has been called the “leading edge effect” or the “entrance effect.” This effect occurs when flow passes from a FAC-resistant material to a non-resistant material, which causes a local increase in the corrosion rate. This effect is normally manifested by a groove up- or downstream of the attachment weld between the corroding and the resistant

The effect of piping and piping component geometry is also a contributing factor to the occurrence of FAC. The general layout of the piping such as the positioning of elbows, Tees and inner surface geometry such as reduction of the internal diameter, surface finish of weld roots, flow changes in valve bodies, orifices, pressure reducers, areas where flow, pressure

[82] In BWRs, hydrogen water chemistry (HWC) can be applied with the main intention to suppress intergranular stress corrosion cracking (IGSCC) susceptibility and crack growth rate. The FAC rate has been measured in a laboratory test to be higher for a time period of 8 months after starting HWC. After this time the FAC rate appears to be similar to that in a reference normal water chemistry (NWC) environment. General Electric guidelines consider an oxygen level of 20 to 50 ppb desirable for hydrogen for hydrogen water chemistry. Some plants must add oxygen in their feedwater when using HWC, while others do not. The effects of higher hydrogen levels under NWC conditions are plant specific and must be taken into account as for HWC conditions. The use of noble metals to reduce the quantities of hydrogen required to establish HWC conditions has to date not had a more pronounced effect on FAC than the application of HWC itself.

and temperature are measured, and regions where the inner surface finish or geometry change over short distances, are all contributing factors to the occurrence of FAC.

9.4 Assessment of Aging Impacts on Carbon Steel Material

This section is concerned with the different methods for estimating age-dependent pipe failure rates and aging trends and their positive and negative impacts on pipe failure rates. Two different approaches to parameter estimation are presented by way of practical examples.

9.4.1 'Holistic' Aging Factor Assessment

To paraphrase Section 2.1 of NUREG/CR-5378 [121], when systematically collecting and analyzing operating experience data it is of interest to know if "*aging" is present, and, if so, how to identify and characterize it so that the insights can be input to a decision making process; e.g. operability determination, evaluations of different material selections, viability of long-term operation. A fundamental problem facing an analyst is that there is no clear dividing line between aging and no-aging. As an example, without enormous amounts of data, extremely slow aging cannot be distinguished from no-aging, and a decision-maker probably does not wish to make the distinction between the two. It is, therefore, more informative to replace the yes-or-no question, "Is there aging?" by a quantitative question, "How much aging is there?"*

The term "aging" is used in a high-level, holistic manner to analytically evaluate certain potential time-dependent effects on material performance. Objectives of aging factor analysis are to account for:

- Aging plant fleet. As shown in Figures 9-5 through 9-9, numerous nuclear plants have entered an extended period of operation (> 40 years). How effective is an existing aging management program (Figures 9-10 and 9-11)?
- Renewal processes. Piping systems are replaced-in-kind or upgraded, for example, by replacing original material with material 'supposedly' resistant to degradation. Also, there is plant-to-plant variability piping system design, which impacts the material degradation propensity; see below and Figure 9-12).
- New operating experience data. The operating experience with metallic passive components is continuously being updated. Is the data collection process sufficiently complete to support quantitative aging factor assessment?
- Enhancements in reliability and integrity management. Plant life extension initiatives together with applications of non-destructive examination (NDE) techniques and inspection qualification processes continue to evolve. Embedded in the operating experience data are effects of NDE and changes in the reporting of pipe failures.

Referring to Figure 9-12 and as addressed in report NSAC-148 [156], the PWR Service Water (SW) system designs are highly plant-specific. NSAC-148 documents a review of six plants, and it also makes reference to a study performed in support of the U.S. NRC Accident Sequence Evaluation Program (ASEP) which determined "… that there were essentially no common piping configurations among the more than 50 plants analyzed by the program" [157]. In Figure 9-12 the respective site-specific SW pipe failure population is normalized against a "median plant". The plant sites with normalized SW pipe failure event populations < 1.0 have piping systems that support relatively few to a single load (e.g. component cooling water system heat exchanger via large-diameter supply and discharge SW lines; "once-through" system). The plant sites with SW piping system supporting a large number of essential cooling loads tend to have normalized SW pipe failure event populations > or >> 1.0. This observation needs to be taken into account in a pipe failure rate estimation effort.

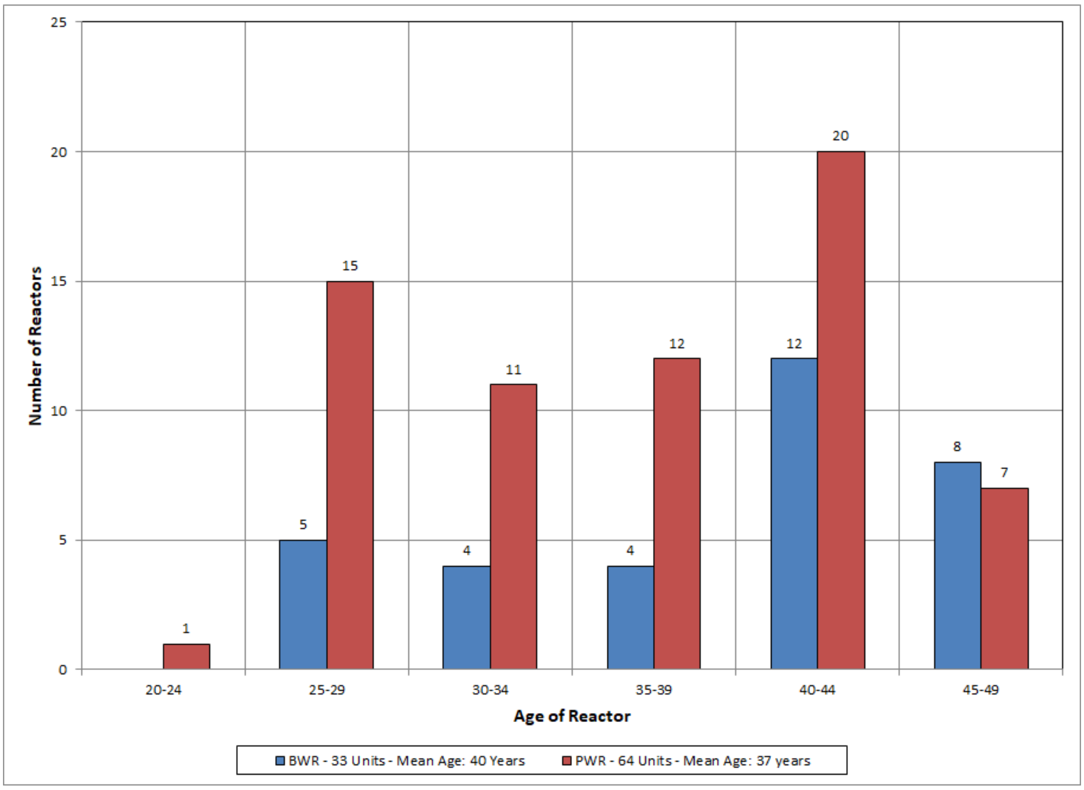

Figure 9-5: Age of the U.S. Nuclear Plant Fleet (October 2017)[83]

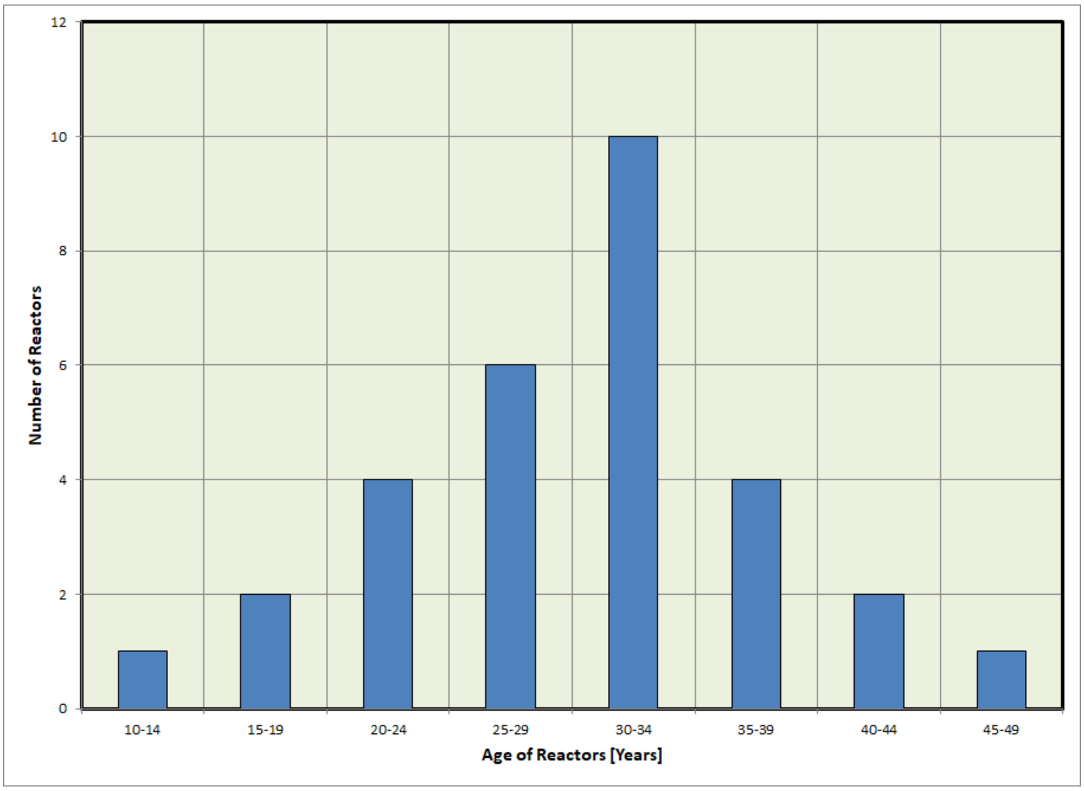

Figure 9-6: Age of the Worldwide CANDU Plant Fleet (October 2017)

[83] Excludes Watts Bar Unit 2 which entered commercial operation in October 2016.

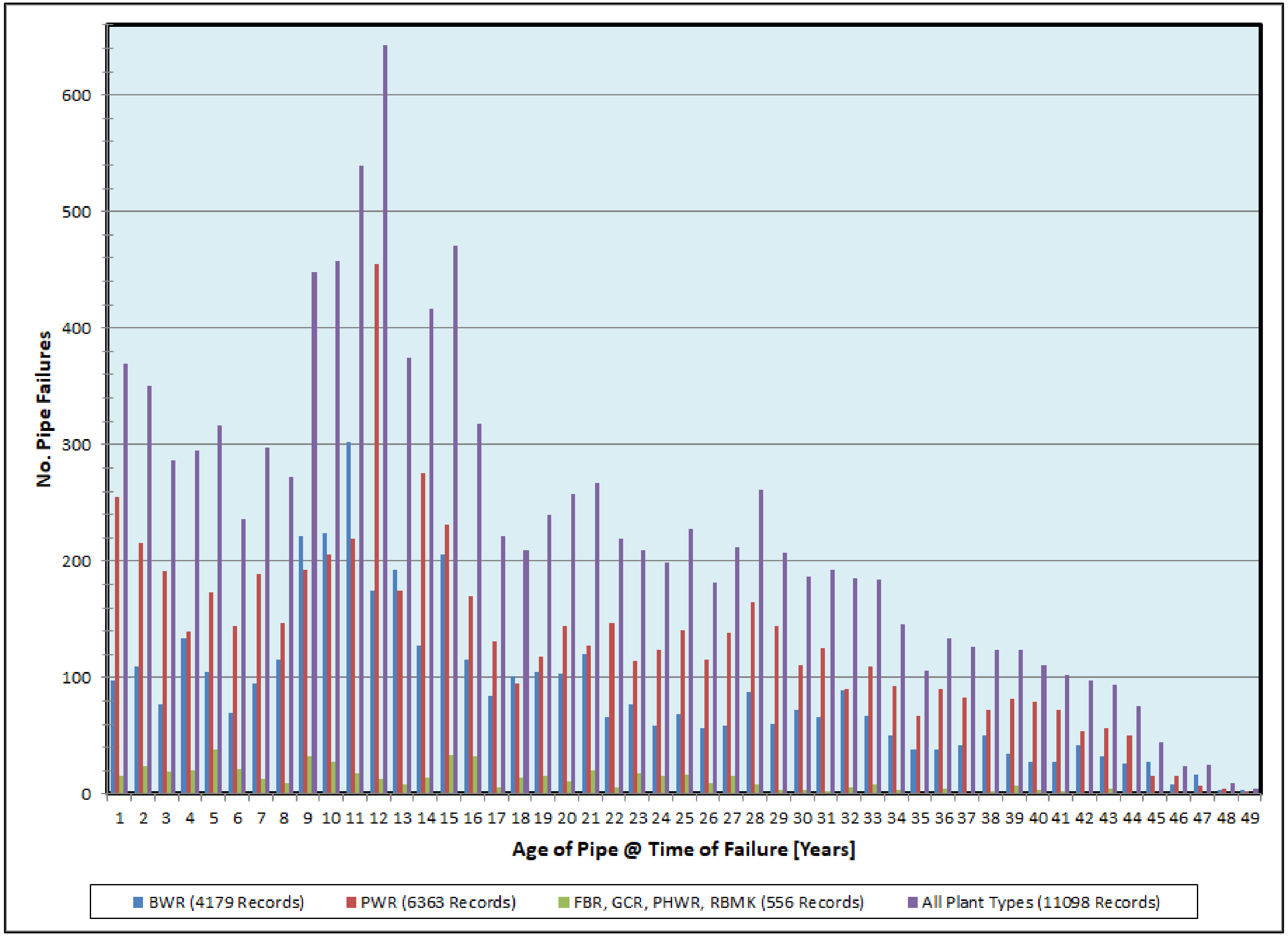

Figure 9-7: Number of Pipe Failure Events vs. Age of Failed Component (as of March 2019)

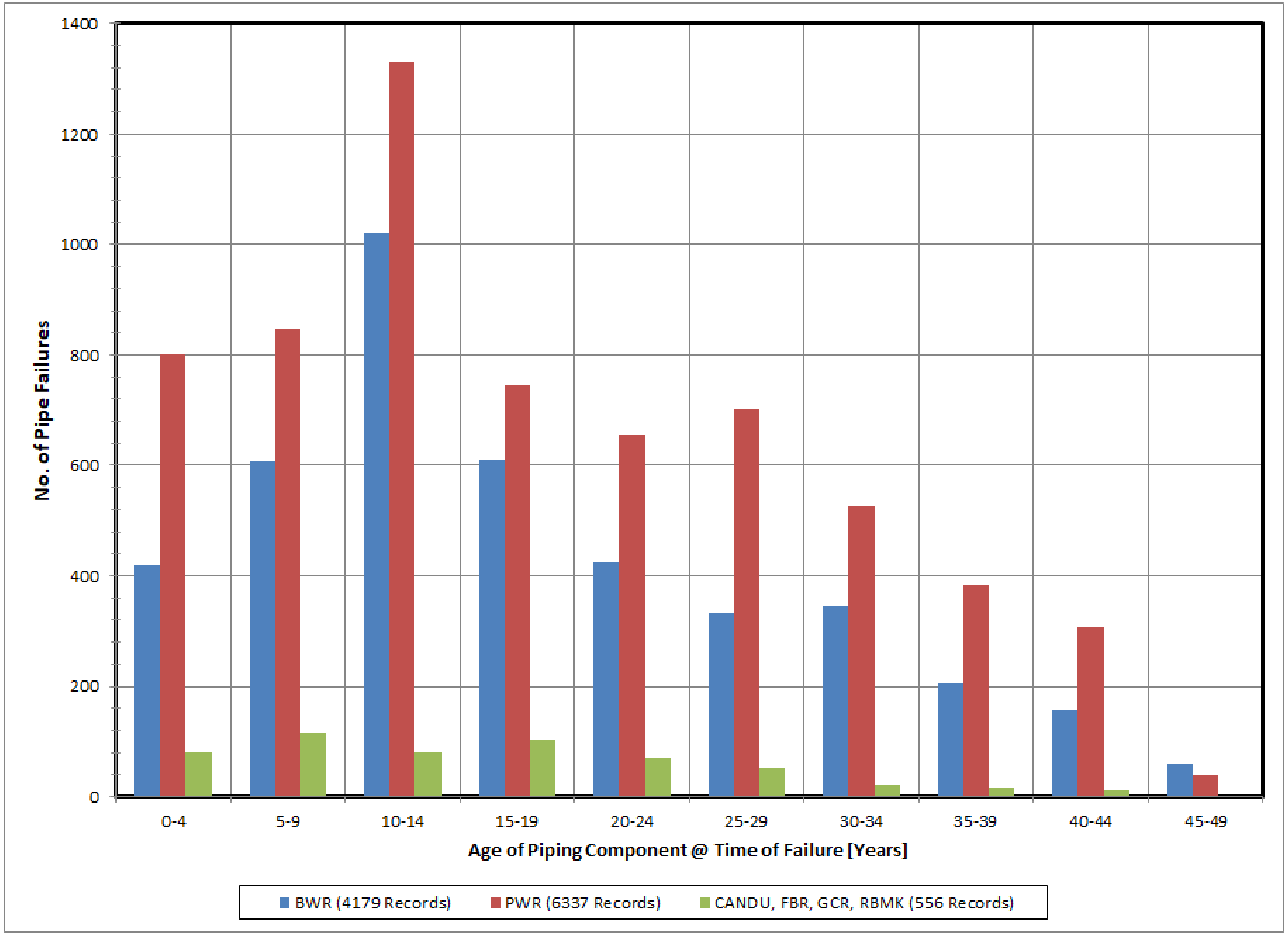

Figure 9-8: Number of Pipe Failure Events vs. Age of Failed Component (as of January 2019)

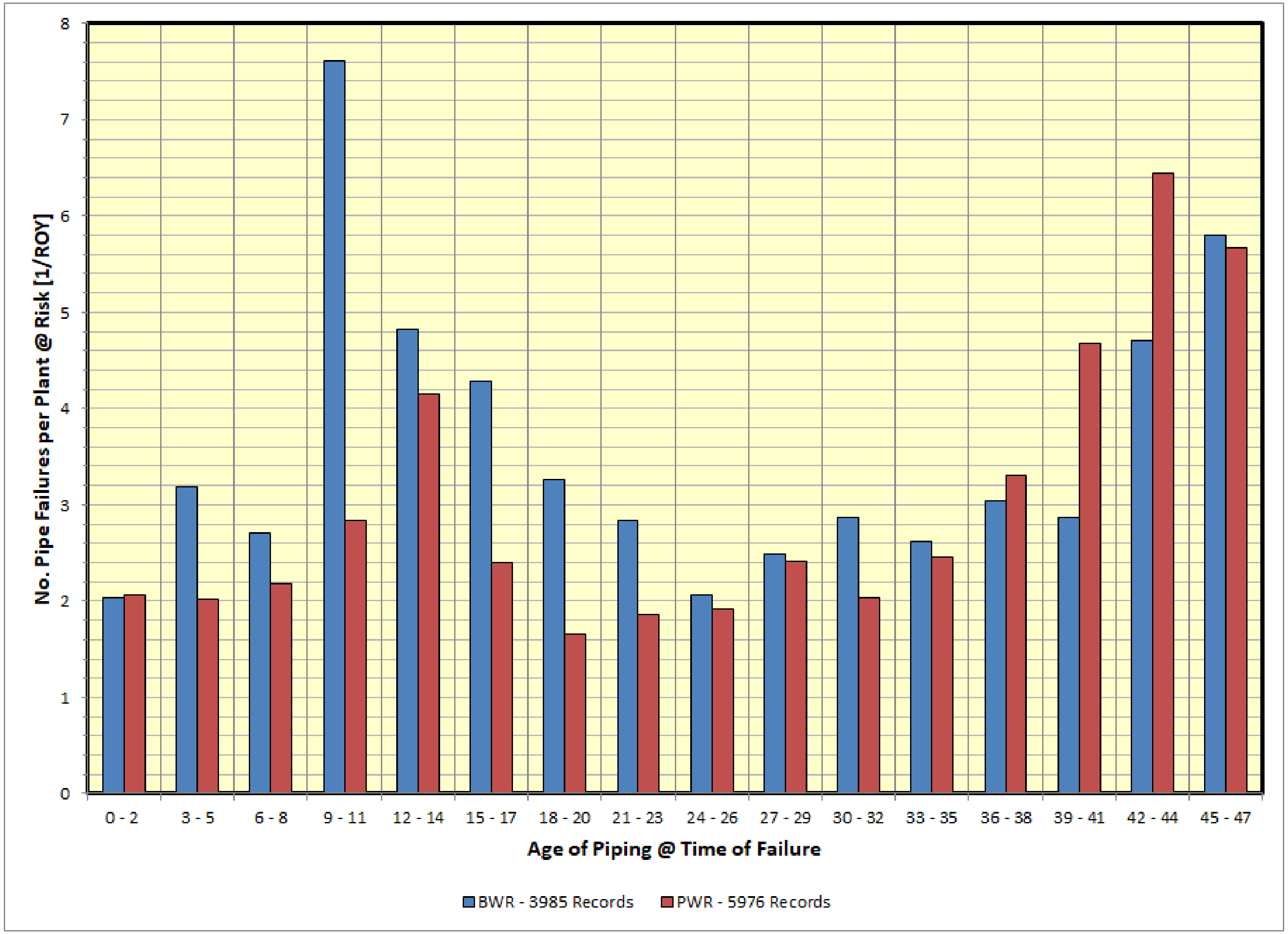

Figure 9-9: Number of Pipe Failures per Plant @ Risk

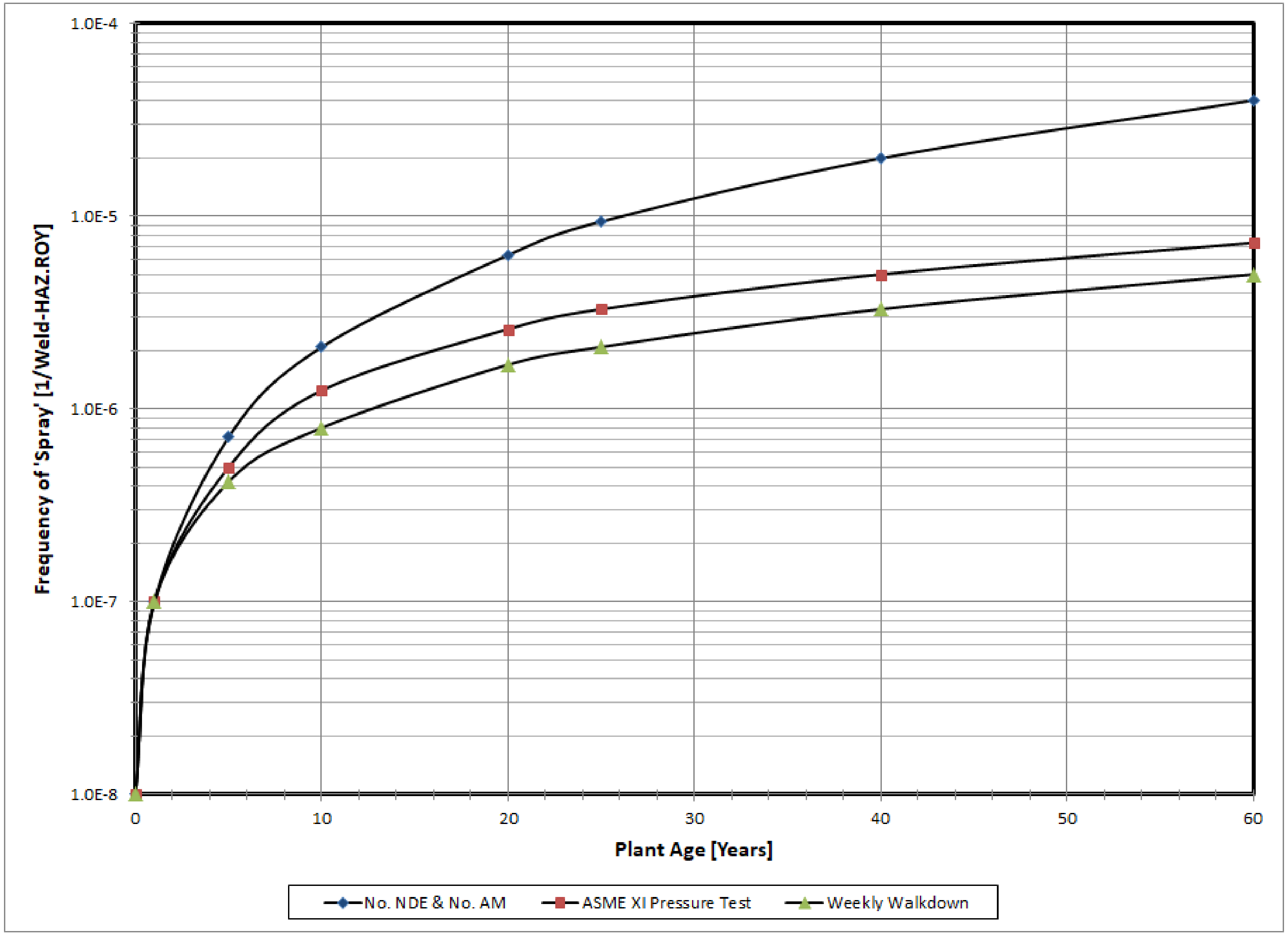

Figure 9-10: Impact of Aging Management on Service Water Piping Reliability[84]

[84] In this example the term "spray" corresponds to a through-wall flow rate of up to 6 kg/s. The time-dependent results are for DN150 SW piping subjected to MIC.

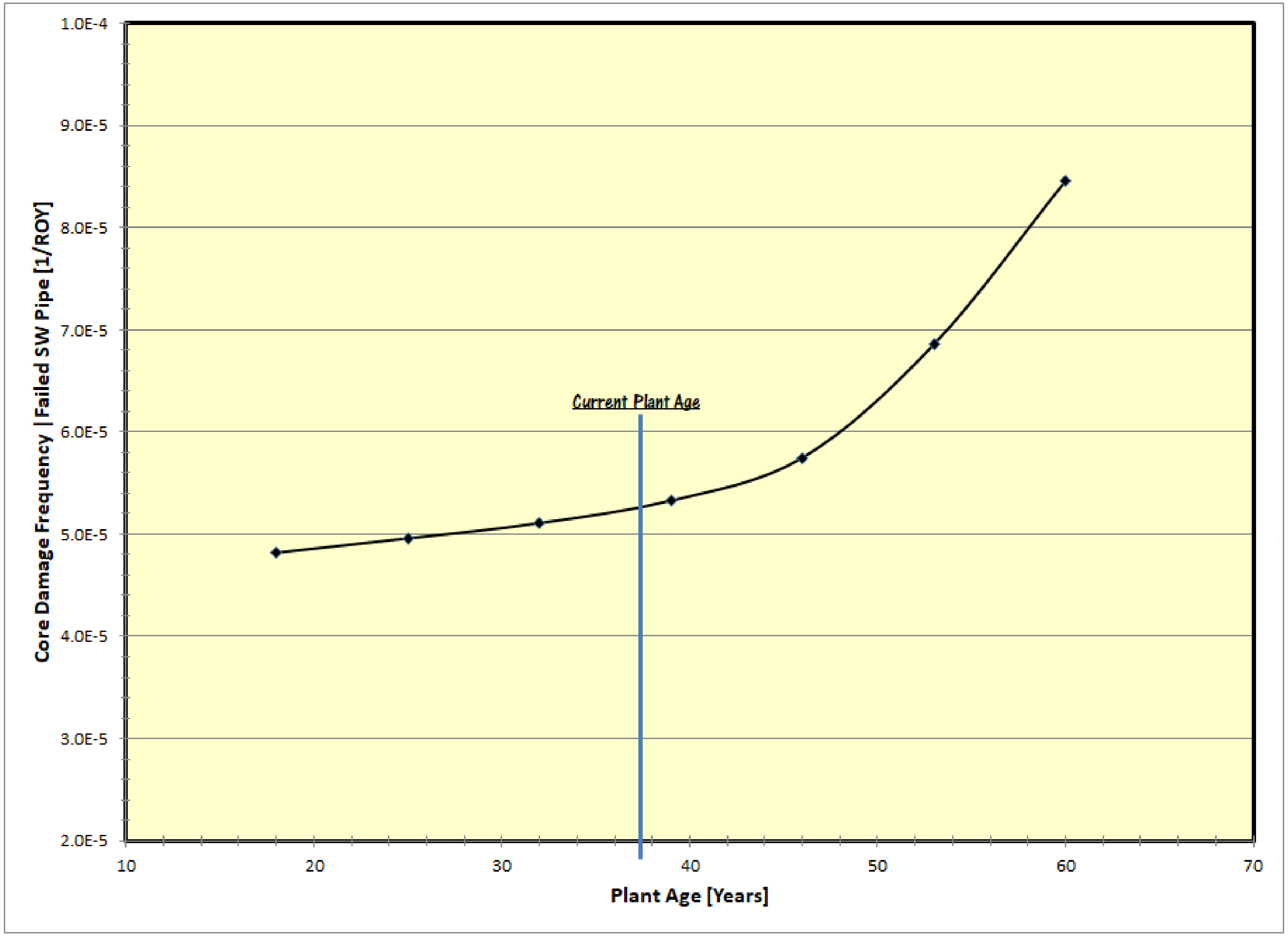

Figure 9-11: Impact of Aged Service Water Carbon Steel Piping on Core Damage Frequency

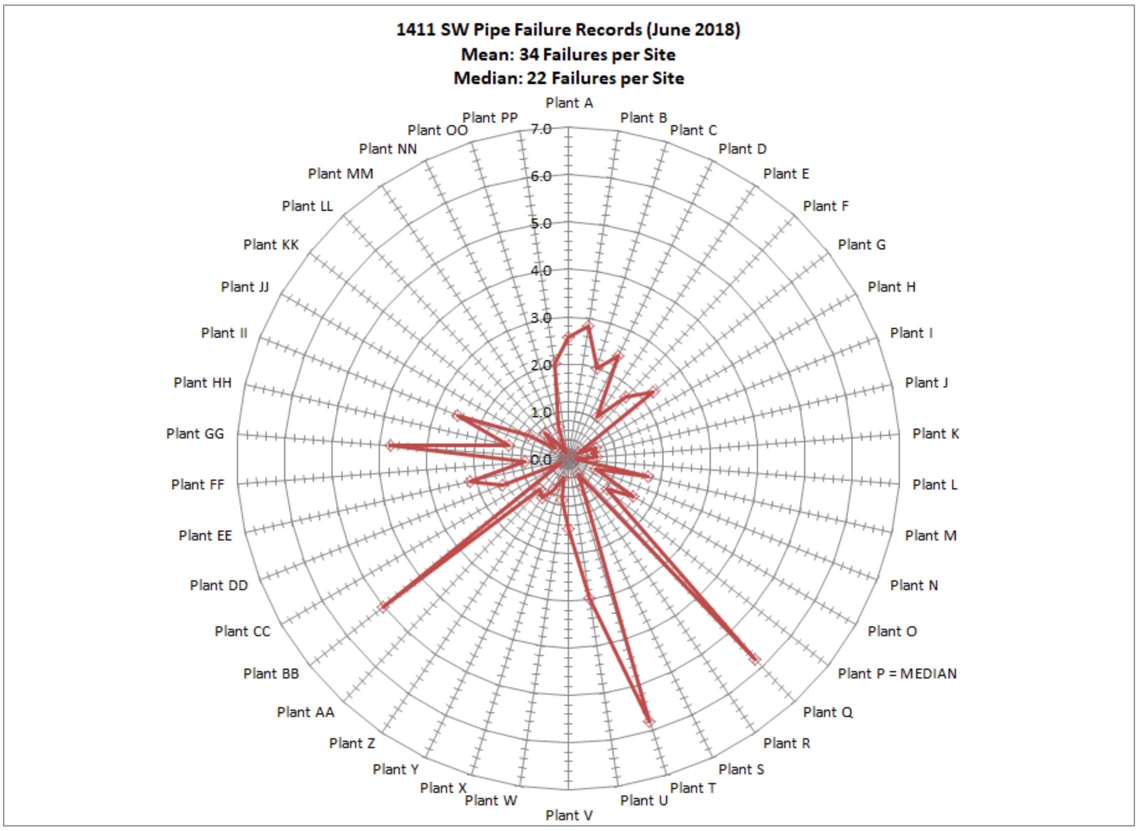

Figure 9-12: Normalized PWR Site-to-Site Variability in SW Pipe Failure Event Population[85]

9.4.2 Cohort Effects & Temporal Change

Age-dependent pipe failure rate estimation can be performed according to different analysis strategies to obtain piping reliability parameters as a function of the age of an affected piping component at the time of failure observation, or as a function of the temporal changes in the piping operating experience. The term "temporal change" means that descriptive failure statistics (mean, upper/lower bound, etc.) change over time. The temporal trends may be due to aging, such as change in the physical properties of piping material (e.g., thinning or cracking), but may also be due to changing reporting routines and the data collection processes.

In the analysis of piping reliability the term "cohort effect" is sometimes used to describe variations in observed structural integrity factors (e.g. onset of crack initiation and subsequent crack growth) as a function of operating time, plant age, plant design generation, and degradation mitigation implementation strategies (e.g. full structural weld overlay, peening, induction heating stress improvement). The commercial nuclear power plant designs have evolved and those reactor units designed in the 1960s and commissioned in the early 1970s exhibit quite different service experience histories than reactors designed and commissioned at later stages. As one example, cohort effects relating to intergranular stress corrosion cracking (IGSCC) of boiling water reactor (BWR) primary system unirradiated stainless steel piping are particularly strong. Those BWR units commissioned in the 1960s and 1970s exhibited a very high IGSCC incidence rate, which was attributed to a lack of recognition of the relationships between the operating environments (e.g. water chemistry), material characteristics (e.g. carbon content) and stress conditions. The worldwide service experience with IGSCC in Code Class 1 piping is summarized in Figures 9-13 and 9-14 (1126 IGSCC

[85] OE data from 41 PWR plant sites in the United States.

failure records). The service experience is organized by "IGSCC Class" where each class corresponds to a uniquely defined event population:

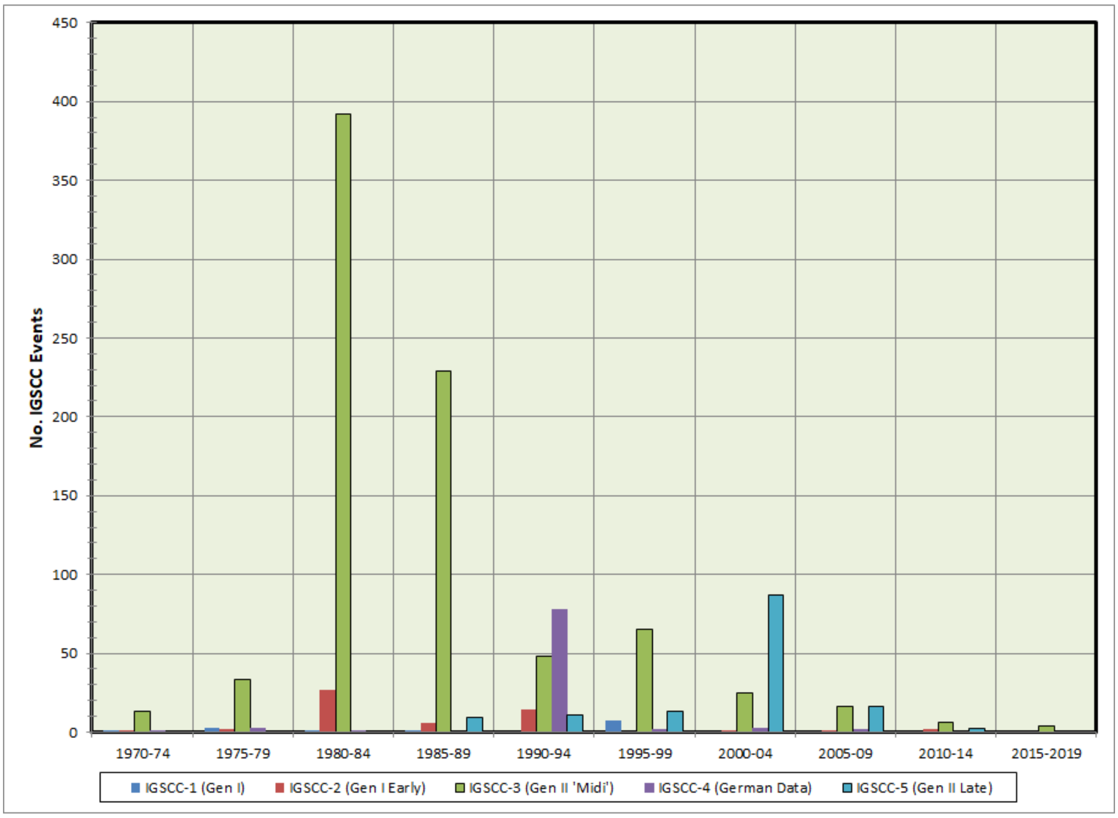

Figure 9-13: IGSCC Operating Experience Data

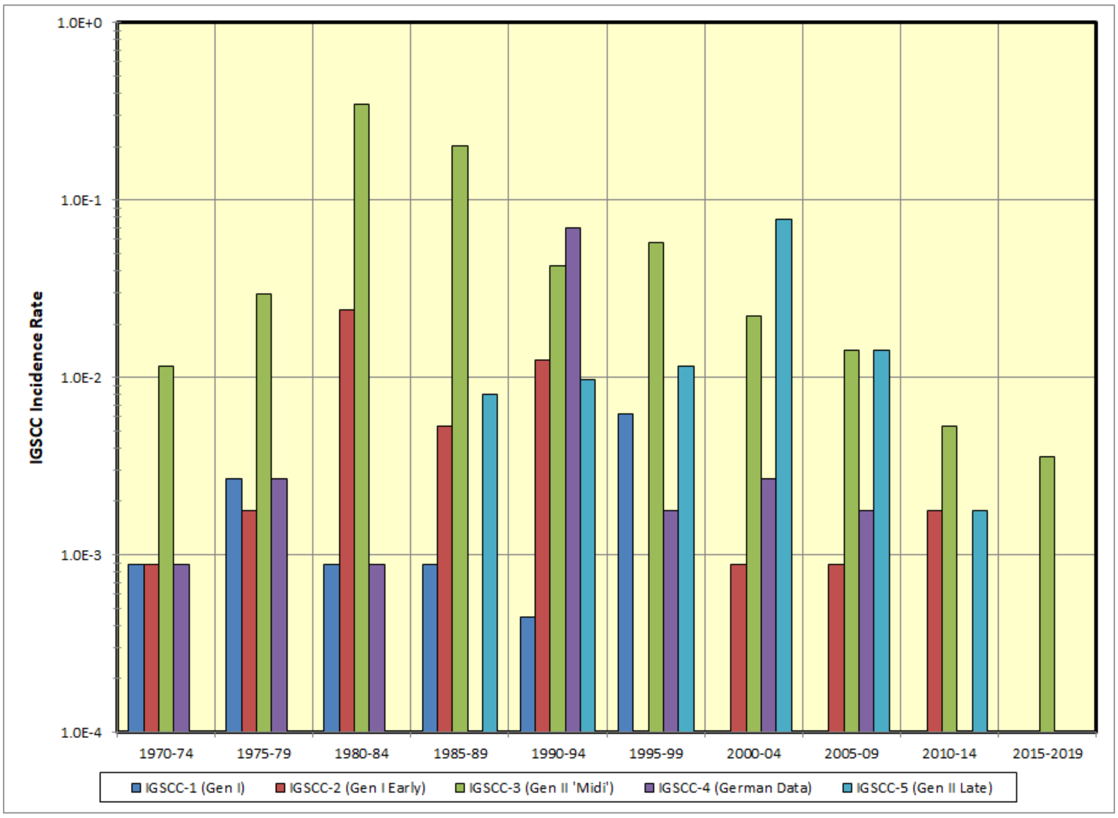

Figure 9-14: IGSCC Incidence Rates

- IGSCC-1 (Gen I) corresponds to the first BWR design generation (essentially demonstration plants). These units were brought online in the 1960s and except for one unit, they have all been decommissioned.
- IGSCC-2 (Gen II 'Early') corresponds to the second BWR design generation; plants commercialized in early 1970s. Several units have entered into an extended period of operation, beyond 40 years)), and a few of units are undergoing decommissioning.
- IGSCC-3 (Gen II 'Midi') corresponds to the third BWR design generation. Multiple IGSCC-mitigation projects were implemented in the latter part of the 1980s.
- IGSCC-4 corresponds to two specific BWR NSSS design generations (SWR69 and SWR72, respectively) that utilized stabilized austenitic stainless steel materials. This class is limited to German plants for which IGSCC-mitigation consisted of a re-design of the primary piping systems. The steps taken have virtually eliminated the IGSCC susceptibility. Except for KRB-II-B (Gundremmingen 'B'; to be permanently shut down by 31-December-2017) and KRB-II-C (Gundremmingen 'C'), all German BWR plants have been permanently closed.
- IGSCC-5 (Gen II 'Late') corresponds to BWR designs for which IGSCC-mitigation consideration was an integral part of the original piping design (e.g. use of Nuclear Grade austenitic stainless steel) and plant operating practices (e.g. enhanced primary water chemistry control).

Not only do Figures 9-13 and 9-14 (same data set as in Figure 9-13 but converted to occurrence rate) portray five fundamentally different sets of service experience data, they also portray how the IGSCC incidence rate has changed over time. In general, there is sufficient understanding of the parametric dependencies (e.g. between weld residual stresses, material chemical properties, cold work and water purity) to justify the five IGSCC categories as indicated in Figures 9-13 and 9-14. Correlating the five event populations with corresponding exposure terms (plant population that generated the failures, operating time and weld populations) provides a sound basis for establishing an 'apriori' failure rate distributions for each of the IGSCC categories. Oversimplifying a statistical analysis, by ignoring cohort effects, results in incorrect conclusions about the structural integrity of a piping system. Similar qualitative assessments apply to other degradation mechanisms, for example, flow-accelerated corrosion (FAC). A model of the pipe failure rate as a function of aging and cohort effects is expressed by a power law relationship [158][159]:

$$\lambda(t) = \Theta t^{\alpha} \times e^{\gamma\tau} \qquad (9\text{-}1)$$

Where the first term on the right-hand side represents the time-dependent reliability and the second term on the right-hand term corresponds to the cohort effects. The parameters in Equation (1) are defined as:

Θ = Scale parameter
t = Plant age
α = Shape parameter
γ = Cohort effects coefficient
τ = Cohort age

The three unknown parameters: Θ, α and γ, can be determined using the maximum likelihood estimation (MLE) method [84]. The determination of presence of any cohort effects depends on the approach to subdividing the OE data, for example, by pipe size and other piping system design factors.

As an alternative approach to the examination of cohort effects on pipe failures, the following analysis is performed for large-bore PWR Service Water System piping. In this example, failure rate estimation is performed for four (somewhat) arbitrarily selected time periods[86]:

Period 1: 01/01/1970 to 12/31/2004
Period 2: 01/01/2005 to 03/31/2009
Period 3: 04/01/2009 to 12/31/2015
Period 4: 01/01/1970 to 12/31/2015

The point estimate failure rate in the second time period (P2) increased a factor of about 5 over that for the first period (P1), and the increase in the third period (P3) is a factor of almost 7 times higher than that for the first period. If one simply re-computes the average failure over all three time periods, the revised failure rate is about 2.4 times higher than that for the first period; P4 versus P1. The evolution of the carbon steel piping operating experience is summarized in Figure 9-15. There is clearly an increasing trend that suggests that piping is experiencing the effects of aging. The first time period spans the period when the Maintenance Rule and other aging management initiatives for many U.S. plants lead to improvements to performance and changes to reporting requirements. Note that in this example, the calculation case represents a safety related system and clearly within the scope of the Maintenance Rule[87] and, hence, the time-dependent material performance is closely monitored.

An uncertainty analysis was performed for the four time periods exhibited in Table 9-1 and the results of this analysis are shown in Table 9-2 and Figure 9-16. The uncertainty about the point estimates is seen to be much larger for the second and third time periods, which is a consequence of the fact that these time periods have less evidence to support the estimates. The conclusion from this analysis is that the practice of averaging data analysis over the entire time periods for the successive updates yields optimistic prediction on two fronts: The increasing failure trends over the three time periods are muted, and the uncertainty about the failure rate estimates is under predicted.

Table 9-1: Service Water Piping Operating Experience Data for Four Time Periods[88]

Data Collection Period		Failures	Reactor Operating Years [ROY]	Pipe length per plant [ft.]	Exposure ROY-ft.	Point Estimate Failure Rate
Start	End					
01/01/70	12/31/04	40	523	6307	3.30E+06	1.21E-05
01/01/05	03/31/09	30	76.6	6307	4.83E+05	6.21E-05
04/01/09	12/31/15	61	121.5	6307	7.66E+05	7.96E-05
01/01/70	12/31/15	131	721	6307	4.55E+06	2.88E-05

Table 9-2: Uncertainty Distribution Parameters for Pipe Failure Rates for Four Time Periods

Period	Time Frame	Point Estimate	Mean	5%tile	50%tile	95%tile	RF
1	1970-2004	1.21E-05	1.28E-05	7.79E-06	1.18E-05	2.35E-05	1.7
2	2005-2009	6.21E-05	6.48E-05	3.86E-05	5.98E-05	1.19E-04	1.8
3	2009-2015	7.96E-05	8.39E-05	5.18E-05	7.81E-05	1.56E-04	1.7
4	1970-2015	2.88E-05	3.03E-05	2.28E-05	2.87E-05	4.29E-05	1.4

[86] The origin of the analysis of the PWR Service Water piping is documented in an unpublished report "A Risk-Informed Perspective on the Reliability of Raw Water Piping Systems in Commercial Nuclear Power Plants" (March 2003). An updated version of the report was published as EPRI 1013141 "Pipe Rupture Frequencies for Internal Flooding PRAs" (2006). It has since been updated in 2010 and 2018.
[87] U.S. NRC, 10 CFR 50.65,"Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."
[88] The analysis is restricted to PWR Service Water Systems that use sea water as heat transfer medium and NPS > 10".

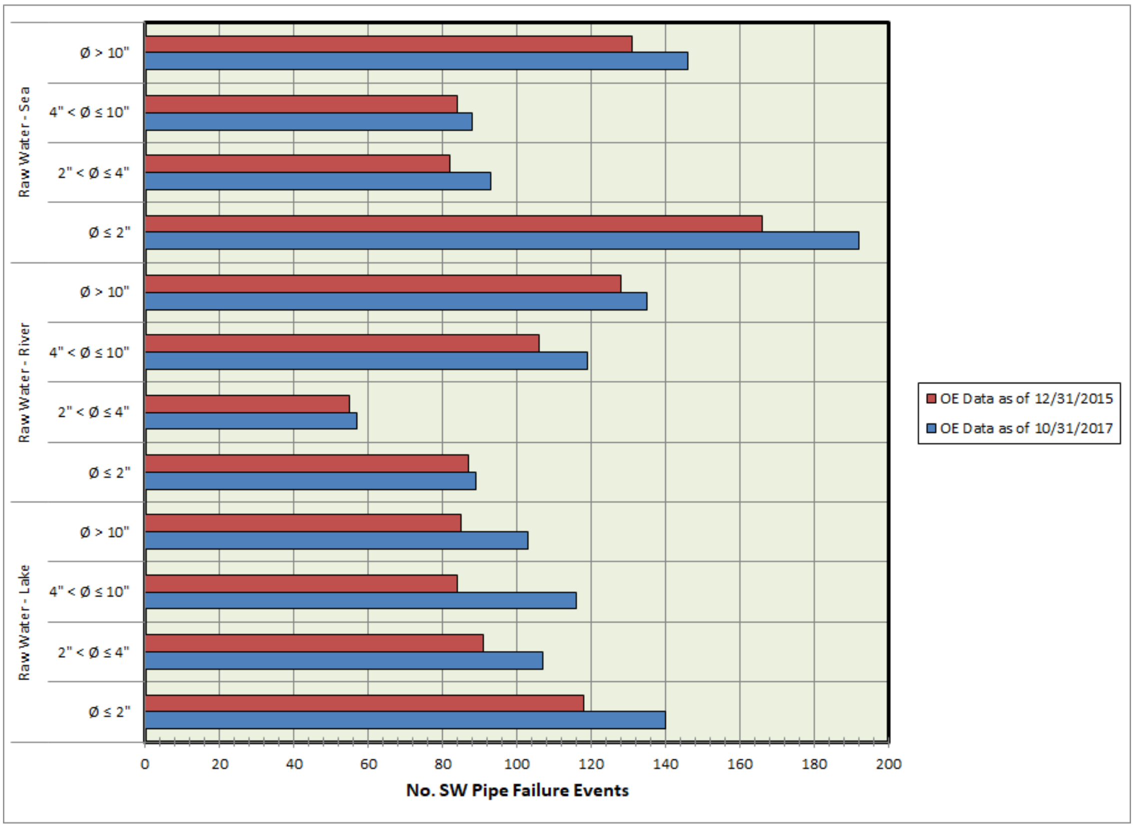

Figure 9-15: Evolution of the U.S. SW Piping Operating Experience Data

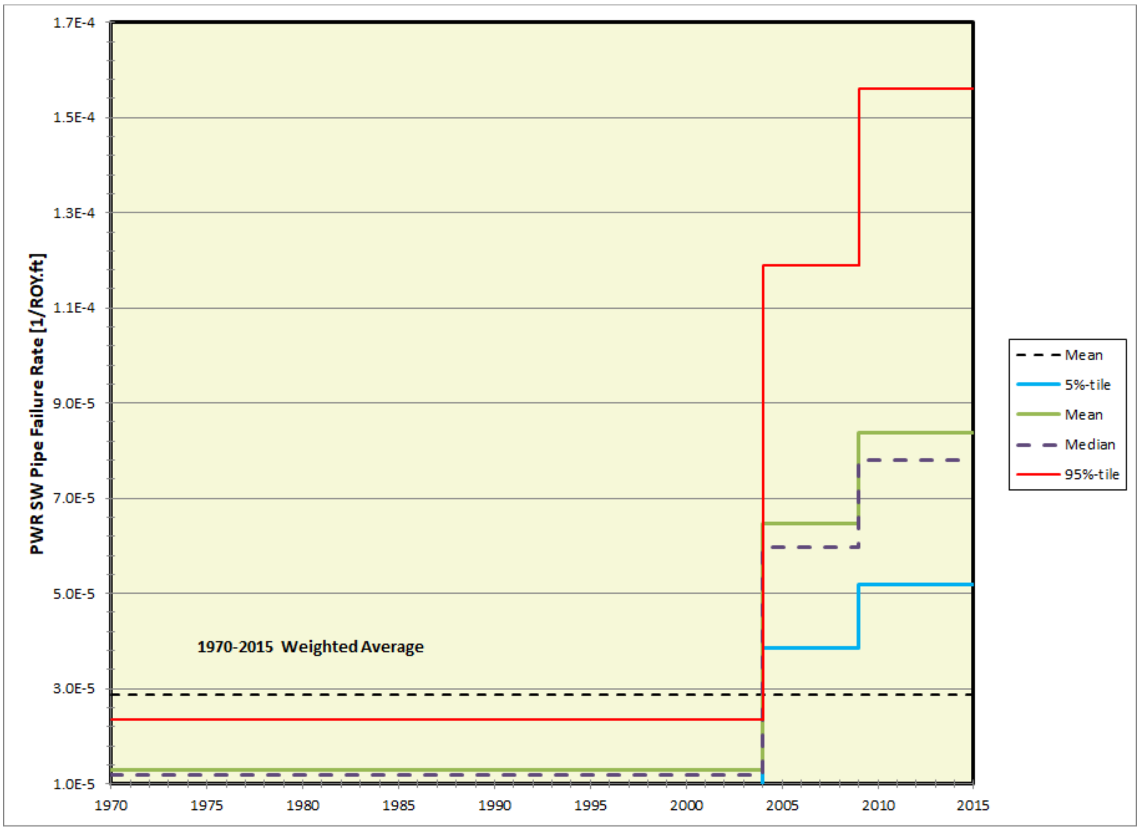

Figure 9-16: Uncertainty Distributions for SW Piping in Different Time Periods

The conclusion from this analysis is that the practice of averaging data analysis over the entire time periods for successive pipe failure rate updates yields optimistic prediction on two fronts: 1) the increasing failure trends over the three time periods are muted, and 2) the uncertainty about the failure rate estimates is under predicted.

9.4.3 "Streamlined" Aging Factor Assessment

In its most simplistic form age-dependent pipe failure rates may be calculated for pre-defined age-bins to obtain failure rates for plants-at-risk; e.g. 3-year bins, 5-year bins. Next, each calculated failure rate is assumed to represent the median value of each bin. With reference to the median age of a given plant population, the aging factor (AF) is calculated as follows:

$$AF = \lambda_{AGE='A'} / \lambda_{AGE=Median} \tag{9-2}$$

The estimated carbon steel Service Water piping aging factors for the U.S. PWR fleet are summarized in Table 9-3. At the end 2016 the median U.S. PWR plant age was 35 years.

Table 9-3: Estimated Aging Factors for Carbon Steel Service Water Piping

Age of Plant	Pipe Size			
	ø ≤ 2"	2" < ø ≤ 4"	4" < ø ≤ 10"	ø > 10"
44	1.08E+00	1.23E+00	1.26E+00	1.34E+00
43	1.07E+00	1.20E+00	1.23E+00	1.30E+00
42	1.06E+00	1.17E+00	1.20E+00	1.26E+00
41	1.05E+00	1.15E+00	1.17E+00	1.22E+00
40	1.04E+00	1.12E+00	1.14E+00	1.18E+00
39	1.03E+00	1.10E+00	1.11E+00	1.14E+00
38	1.03E+00	1.07E+00	1.08E+00	1.10E+00
37	1.02E+00	1.05E+00	1.05E+00	1.07E+00
36	1.01E+00	1.02E+00	1.03E+00	1.03E+00
35 - Median	1.00E+00	1.00E+00	1.00E+00	1.00E+00
34	9.92E-01	9.78E-01	9.75E-01	9.68E-01
33	9.83E-01	9.56E-01	9.50E-01	9.37E-01
32	9.75E-01	9.34E-01	9.26E-01	9.06E-01
31	9.67E-01	9.13E-01	9.03E-01	8.77E-01
30	9.59E-01	8.93E-01	8.80E-01	8.49E-01

According to Table 9-3, the extent of aging on the carbon steel performance is quite modest. Do these results accurately reflect the existing body of operating experience data? Results of a "simplistic" aging factor assessment are more a function of the completeness of the database and the fact that invariably there is a considerable latent effect of the data collection process. At any given time, failure data for plants that have entered into an extended period of operation does not become available until long after an initial data collection effort has been completed, and, hence, a calculated aging factor for "aging plants" tends to be underestimated. In addition, in numerous instances piping renewals occur; replacements are made in-kind or by design change. It is not uncommon to observe multiple failures occurring in exactly the same locations. Hence, when this is the case the piping component 'time-clock' is reset in the OE database whenever a replacement occurs.

9.4.4 Screening of Data for Temporal Trends

In order to better interpret and organize the operating experience data it is proposed that an assessment of age-dependent pipe failure rates is preceded by a screening analysis to determine presence of negative or positive aging trends. Screening may be performed given that the pipe failure rate has been monitored over some significant period of time and on the basis of a systematically applied failure rate estimation process. If so, a 'temporal change factor' (TCF) is calculated as:

$$\mathrm{TCF} = \lambda_{(Pi)} / \lambda_{Ref} \tag{9-3}$$

Where

$\lambda_{(Pi)}$ = Failure rate for Period "i"
λ_{Ref} = Failure rate for a "reference" period (or base case)

Assuming there is sufficient OE data available, when TCF > 1.0 pipe failure rates and associated uncertainty distributions may be derived for each of the time periods under consideration. The temporal change factor accounts for numerous influences, including the potential change in material properties (e.g. loss of material), changes in data collection process, changes in reporting processes/requirements, etc. It is recognized that some significant plant-to-plant variability exist in the piping operating experience and instructions for how to interpret and potentially apply the temporal change factor concept must be developed for each aging factor study. Results of an aging factor screening analysis are summarized in Table 9-4. With TCF > 2, the raw water cooling water systems (CW, FP and SW) all show signs of some form of aging.

Included in Table 9-4 are results of a temporal change factor analysis performed for selected FAC-susceptible carbon steam-cycle systems. These systems have TCF < 1, which can be interpreted as a result of an effective aging management program (Table 9-5) through improvements in secondary-side water chemistry and use of FAC-resistant materials.

Table 9-4: Selected Aging Factor Screening Analysis Results

Piping System	Nominal Pipe Size (NPS)	Temporal Change Factor			Pipe Failure Rate – Point Estimate [1/(ROY.ft)]			
		P2	**P3**	**P4**	**P1 (1970-2004)**	**P2 (2004-2009)**	**P3 (2010-2017)**	**P4 (1970-2017)**
Raw Water Piping Systems								
In-Plant Fire Protection	Ø ≤ 4"	0.73	2.02	1.15	1.03E-05	7.51E-06	2.08E-05	1.19E-05
	4'"< Ø ≤ 6"	3.35	4.75	1.94	5.63E-06	1.89E-05	2.67E-05	1.09E-05
	Ø > 6"	2.18	2.18	1.34	2.54E-05	5.53E-05	5.53E-05	3.41E-05
BWR Service Water UHS: Lake Water	Ø ≤ 2"	1.84	2.88	1.43	3.15E-05	5.82E-05	9.09E-05	4.50E-05
	2 < Ø ≤ 4"	5.16	4.03	2.00	5.08E-05	2.63E-04	2.05E-04	1.01E-04
	4 < Ø ≤ 10"	7.53	4.03	2.26	1.86E-05	1.40E-04	7.52E-05	4.21E-05
	Ø > 10"	6.45	8.64	2.95	4.67E-06	3.01E-05	4.03E-05	1.38E-05
BWR Service Water UHS: River Water	Ø ≤ 2"	2.11	4.75	1.77	7.80E-05	1.65E-04	3.70E-04	1.38E-04
	2 < Ø ≤ 4"	3.32	5.50	2.03	1.76E-04	5.84E-04	9.69E-04	3.57E-04
	4 < Ø ≤ 10"	4.54	6.81	2.39	4.64E-05	2.11E-04	3.16E-04	1.11E-04
	Ø > 10"	11.25	9.44	3.57	1.21E-05	1.36E-04	1.14E-04	4.30E-05
PWR Service Water UHS: Sea Water	Ø ≤ 2"	1.14	3.01	1.35	3.22E-04	3.68E-04	9.68E-04	4.35E-04
	2 < Ø ≤ 4"	1.46	3.19	1.42	3.42E-04	4.98E-04	1.09E-03	4.84E-04
	4 < Ø ≤ 10"	2.82	2.74	1.49	1.02E-04	2.87E-04	2.80E-04	1.52E-04
	Ø > 10"	5.09	6.59	2.38	1.36E-05	6.92E-05	8.95E-05	3.23E-05
Steam Cycle Piping System								
Extraction Steam (BWR)	2" < Ø ≤ 10"	1.34	0.36	0.93	3.81E-05	5.09E-05	1.37E-05	3.54E-05
	Ø > 10"	0.94	0.30	0.88	4.63E-05	4.36E-05	1.37E-05	4.07E-05
Extraction Steam (PWR)	2" < Ø ≤ 10"	0.61	0.01	0.81	1.19E-04	7.27E-05	1.14E-05	9.59E-05
	Ø > 10"	0.60	0.89	0.80	7.73E-05	4.63E-05	6.86E-06	6.20E-05

Table 9-5: Aging Factor Analysis Proposed 'Screening Rules'

TCF	Period(s)	Interpretation	Impact on Pipe Failure Rates & Rupture Frequency Calculations
< 1	P2, P3, P4	Effective FAC aging management. No significant trend change anticipated beyond 2017. FAC-free piping performance is achievable.	Applies to FAC-susceptible steam cycle piping systems. Extensive operating experience data available. Most FAC-susceptible piping systems have been replaced with material that is resistant to flow-assisted wall thinning. Simple update; average across chosen time period.
> 1 but < 2	All	No significant adverse trend in the recorded operating experience data.	Insufficient data to support aging factor assessments. Alternatively, existing aging management programs sufficiently effective to prevent adverse trends. Simple update; average across chosen time period
> 2	P2 & P3 or All	Indicative of aging of raw water piping systems; e.g. Circulating Water, Fire Protection and Service Water systems[89]	Results of formal aging assessment factor analysis could be factored into failure rate calculations for input to PSA to address, for example, internal flooding or loss of ultimate heat sink scenarios.

[89] For an international perspective of aging raw water systems, refer to https://www-news.iaea.org/ErfView.aspx?mId=24a2e076-631a-4ba6-bbb8-5f3af7053934 ("Risk of Loss of Heat Sink in Case of an Earthquake - 20 Reactors Concerned").

9.4.5 Accounting for Temporal Trends in Aging Factor Assessment

A simplified technical approach to aging factor assessment utilizes the calculated pipe failure rates for the periods P1 through P4 (1970-2017). Each of the periods is representative of the accumulated operating experience against an average U.S. BWR and PWR fleet age. The estimated pipe failure rates for each calculation case are used as 'anchor values' when plotted against the average plant age for periods P1 through P3. In the example below (Figure 9-17) and an additional "calibration value" is calculated for the period (P0) 1970 through 1992 corresponding to an average plant age on the order of 15 years. While somewhat arbitrary, the year 1992 as end point is selected so that the operating experience data incorporates the results of approximately two years of U.S. NRC Generic Letter 90-05 relief requests[90]. In Microsoft® Excel a best-fit curve and equation are added to the four calibration points, enabling the calculation of age dependent pipe failure rates; Equation (9-4).[91] For respective calculation case the aging factor (AF) is determined from Equation (9-5):

$$h(t) = \eta\times\beta(EXP(-\eta))\times AGE(EXP(\eta-1) \qquad (9\text{-}4)$$

$$AF = \lambda_{Age\ 'i'} / \lambda_{Avg.\ (1970\text{-}2017)} \qquad (9\text{-}5)$$

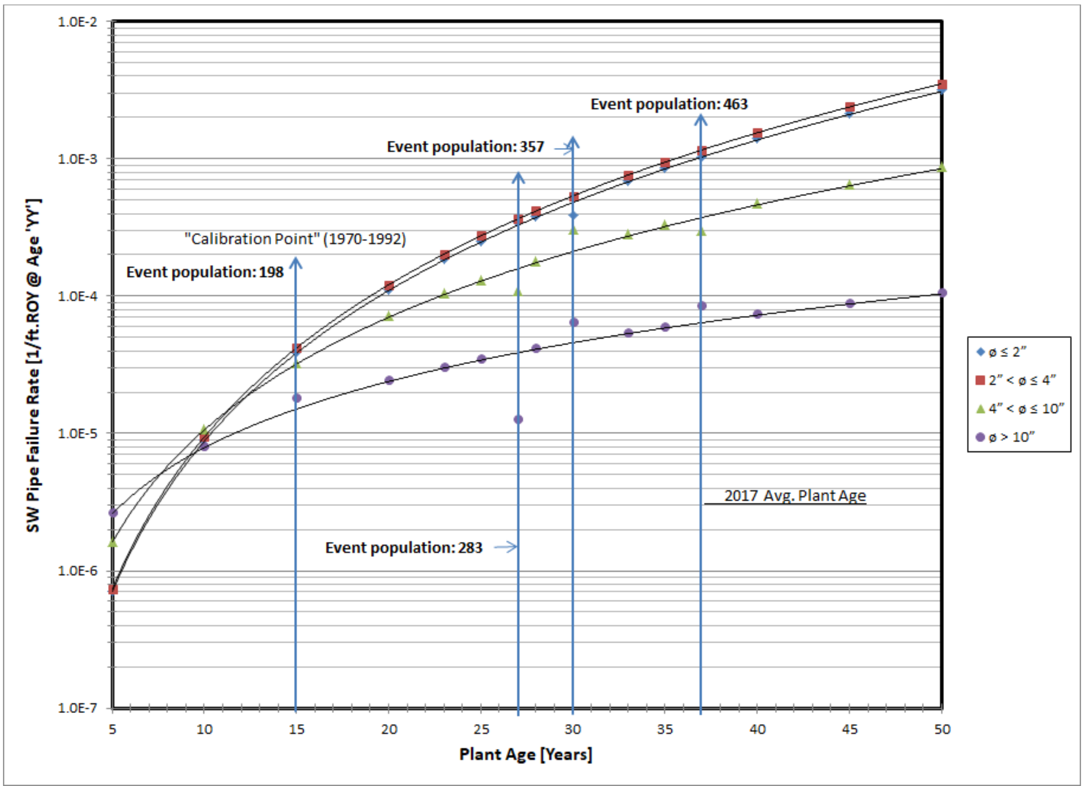

Figure 9-17: Age-Dependent Pipe Failure Rates for Carbon Steel Service Water Piping[92]

[90] Effective June 15, 1990, Generic Letter 90-05 provides guidance for preparing relief request for temporary non-Code code repair of moderate-energy ASME Code Class 2 and 3 piping. The safety-related SW piping is mainly Code Class 3. In the context of this analysis, the significance of Generic Letter 90-05 is tied to the extent and quality of the reporting of degraded SW piping. Refer to the 1st deliverable ("Overview of Regulated Industry Practices with Risk-Informed Operability Determination") for additional details.

[91] This equation is written in terms of a scale parameter (β) and a shape parameter (α). AGE is the age of the piping components.

[92] For details, see Paper #367 in Proceedings of PSAM14, Probabilistic Safety Assessment and Management, Beverly Hills, CA, September 16-21, 2018. Full details are documented in EPRI 3002012997 (October 2018) https://www.epri.com/#/pages/product/000000003002012997/?lang=en-US

Summarized in Table 9-6 [160] are the estimated aging factors for three carbon steel Service Water pipe failure rate calculation cases. It is important to note that these aging factors assume that no corrosion degradation mitigation processes are implemented beyond regular walkdown inspections, on-demand Code repairs to restore structural integrity, and ASME Section XI leak tests.

Table 9-6: Calculated Aging Factors Based on the Simplified Approach

Plant Age [Years]	Aging Factor - PWR 'Lake Water'				Aging Factor - PWR 'River Water'			
	ø ≤ 2”	2” < ø ≤ 4”	4” < ø ≤ 10”	ø > 10”	ø ≤ 2”	2” < ø ≤ 4”	4” < ø ≤ 10”	ø > 10”
20	0.8	0.8	0.5	0.3	0.8	0.3	0.3	0.4
25	1.0	1.0	1.0	0.7	1.0	0.9	0.8	1.0
30	1.2	1.2	1.7	1.5	1.1	2.3	1.8	2.0
35	1.3	1.4	2.6	2.9	1.2	4.7	3.7	3.0
40	1.5	1.6	3.9	5.1	1.2	8.8	7.0	5.1
45	1.6	1.8	5.5	9.5	1.3	15.4	12.2	8.1
50	1.8	2.1	7.6	14.7	1.4	25.5	20.1	12.2

9.4.6 Physics-Based Models of Material Aging

Material degradation is defined by the conjoint requirements that have to be met for a given mode of failure. There are very specific sets of tensile stress (applied, residual, fit-up), environment (for example, temperature, corrosion potential) and material (for example, microstructure, yield stress, fracture toughness) that will lead to degradation susceptibility of a certain magnitude. In physics-based models of material degradation the controlling parameters, including the influences of in-service inspection and degradation mitigation, are characterized by a statistical distribution. References [123][152][153] document physics-based models of carbon steel piping degradation and examples of FAC parameters modeled are included in Table 9-7.

Table 9-7: Example of FAC Parameters & Their Uncertainty Information[93]

Parameter	Symbol	Value	Standard Deviation	Distribution
Operational Time [hr]	t	e.g. 100,000c	N/A	Point Estimate (PE)
Piping Material (Cr + Mo in %)	h	0.08	0.0053	Normal
Piping Geometry (Elbow-after-Tee)	k_c	0.75	N/A	PE
Fluid Velocity [m/s]	w	5.1816	0.0345	Normal
Dissolved Oxygen Concentration [ppb]	g	1.2	0.08	Normal
Water Chemistry [pH]	pH	8.9	0.089	Normal
Water Temperature [°K]	T	463	0.772	Normal
Steel Pipe Density [kg/m3]	ρ_{st}	8500	N/A	PE
Pipe Failure Stress [ksi]	σ_f	61	N/A	PE
Pipe Hoop Strain [%]	ε_f	6	N/A	PE
Pipe Radius [cm]	r	22.9	N/A	PE
Pipe Initial Thickness [cm]	$W_{Pipe}(t=0)$	1.27	N/A	PE
Steam Quality	x_{st}	0 (Conditioned Feedwater)	N/A	None

[93] Reproduced from page 93 of Reference [123].

Physics-based models addressing various stress corrosion cracking mechanisms tend to characterize the crack initiation phase by a Weibull distribution and the models for the crack growth rate (CGR) phase build on laboratory data and field experience data; e.g. the Paris-Erdogan crack growth law [154] or the EPRI/MRP-115 CGR-curves [155]. Next, semi-Markov model formulations are employed to allow for an integrated analysis of the controlling parameters of a certain degradation mechanism; from crack initiation to structural failure (e.g. pipe break).

To paraphrase Reference [161], there are significant uncertainties associated with crack initiation and propagation for some degradation mechanisms. It is therefore not possible to predict when such a degradation mechanism will lead to failure of a pipe during any given time interval. However, there is some probability that it will occur in that interval and the uncertainties associated with that probability may be significant. Similarly, there is some likelihood that in-service inspection will successfully identify a degraded state prior to a through-wall flaw so that the needed repair or replacement can be performed. Thus, as time progresses there are a large number of alternative scenarios that are possible. One technical approach to solving a complex problem is to integrate a passive component multi-physics model into PSA using a simulation tool such as RAVEN[94] [162][163]. Yigitoglu [136] illustrates how to incorporate aging model results into a dynamic PSA by coupling a physics-based semi-Markov model with the RAVEN/RELAP-7 simulation environment. RELAP-7 is used to model the effects of a location-specific pipe break inside containment and RAVEN facilitates parametric and probabilistic analyses based on the response of the plant to response to the outcome of the semi-Markov model (or ‘pipe break model’). The degradation mechanism modeled is primary water stress corrosion cracking (PWSCC) in Ni-base Alloys 182 and 600; Figure 9-18.

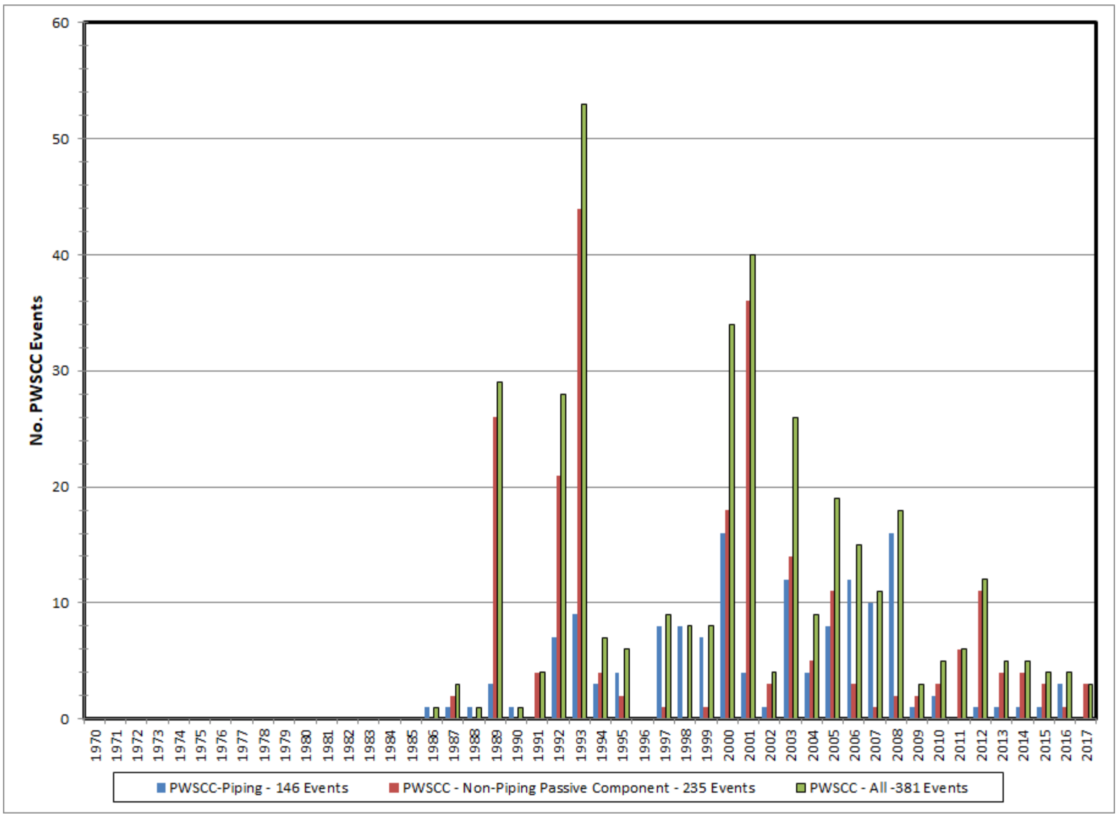

Figure 9-18: The Operating Experience with PWSCC[95]

[94] For additional details go to https://raven.inl.gov/

[95] This figure clearly shows the impact of mitigation on PWSCC susceptibility through the replacement of ALLOY600/82/182 materials with ALLOY69/52/152 materials.

In using the RAVEN simulation tool, the "outcome" can be in the form of a certain degraded state that evolves over time; from a small break size to a very large break size. It is envisaged that RAVEN could work in a computational fluid dynamics (CFD) simulation environment to model spatial effects of a pipe break in plant locations other than the reactor containment.

An alternative simulation tool for investigating the potential effects of passive component degradation is the CASA (Containment Accident Stochastic Analysis) Grande software tool developed by the Los Alamos National Laboratory. It is applied to address the U.S. NRC Generic Safety Issue (GSI) 191. GSI-191 involves assessments of high-energy pipe break scenarios that create pipe insulation debris that finds its way to a plant's emergency core cooling (ECC) recirculation sump strainers [164][165]. Debris passing through the ECC strainers can also accumulate in fuel channels, adversely impacting reactor cooling.

The software tool automates the evaluation of a single postulated accident so that thousands of possible scenarios can be assessed. CASA Grande enables generation of a spectrum of possible outcomes that range from successful performance of the plant's safety systems to various "failure" states defined by regulatory levels of concern. CASA Grande statistically samples probability distributions defined for each factor, propagating uncertainty on the input into an assessment of uncertainty on the measures of failure. Non-uniform Latin Hypercube Sampling (LHS) is used to sample and propagate uncertainty through a basic event scenario that includes debris generation, debris transport, and debris accumulation. Inclusion of plant-state timing in the uncertainty sampling is a novel adaptation of LHS that generates randomized event sequences that are not easily handled by traditional PSA methods. The code has been extensively applied in GSI-191 risk-informed resolution processes.

9.4.7 Aging Management Assessment Factor Analysis Check-List

Aging management assessment entails 1) a determination of the effectiveness of a certain degradation mitigation technique or process, and 2) a determination of aging factors given and adverse trend in material performance. Operating experience (OE) data is essential to the two aspects of aging management assessment. When there is insufficient OE data to support straightforward trend analysis, advanced statistical estimation techniques may be employed to address the uncertainties in the state-of-knowledge about material performance over time. The essential steps in aging management assessment are as follows:

1. Evaluate the current state-of-knowledge about the degradation mechanism of interest.
2. Evaluate the relevant degradation mitigation techniques; extent of application, extent of maturity, potential detrimental effects (e.g. effect of hydrogen water chemistry or noble metal chemical addition on FAC resistance).
3. In-depth OE review, including an assessment of the completeness of the OE data.
4. Perform a temporal change factor analysis to determine the presence of negative trends in material performance.
5. Perform quantitative aging factor assessment.
6. Perform a quantitative assessment of aging on the initiating event frequencies (e.g. LOCA, high-energy-line break, moderate-energy line break, loss-of-heat-sink) of a PSA.
7. Perform a quantitative assessment of the impact(s) of an aging mechanism on plant safety barriers.

10. RISK CHARACTERIZATION OF DEGRADED COMPONENTS

This section addresses methodologies for performing a risk characterization of a degraded passive component. The "risk characterization" may be performed in several different contexts. It may be a fracture mechanics-oriented approach to determine for how long a degraded condition can be allowed to exist before a plant shutdown is needed to perform repair or replacement, including an evaluation of the change in plant risk metrics. Another context is that of using probabilistic safety assessment (PSA) methods to determine the risk significance of a certain event; e.g. determine the change in core damage frequency (CDF) and large early release frequency (LERF). Regardless of the technical context of a risk characterization task, the quantitative assessment of structural integrity reliability parameters is needed.

10.1 Historical Perspectives

Invariably, the risk characterization of a "failed" passive component encompasses all of the analytical challenges that are associated with so called "precursor events"; a concept that was recognized 40-plus years ago. The Risk Assessment Review Group[96] was organized by the U.S. Nuclear Regulatory Commission (NRC) on July 1, 1977, with four elements to its charter [166]:

1. Clarify the achievements and limitations of WASH-1400, the "Rasmussen Report."
2. Assess the peer comments thereon, and responses to those comments.
3. Study the present state of such risk assessment methodology.
4. Recommend to the NRC how (and whether) such methodology can be used in the regulatory and licensing process.

Among its recommendations, the Group introduced the notion of defining "precursor events" and subjecting them to probabilistic safety analysis (PSA) formalism in order to determine their risk significance and to establish a technical basis for proactive risk management.[97] The definition of a precursor may vary from highly specific criteria such as exceeding a specific quantitative threshold to broad definitions that encompass a wide range of events and circumstances.

Identification of precursors requires the review of operational events for instances in which plant functions that provide protection (defense-in-depth) against core damage have been challenged or compromised. Most operational events can be directly or indirectly associated with four initiators: trip [which includes loss of main feedwater (LOFW) within its sequences], loss of offsite power (LOOP), small-break LOCA, and steam generator tube rupture (SGTR). These four initiators are primarily associated with loss of core cooling. Licensee event reports (LERs) and other event documentation are examined to determine the impact that operational events have on potential core damage sequences associated with these initiators.

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[96] Also referred to as the "Lewis Committee" after the chairman of the Risk Assessment Review Group, Professor Harold W. Lewis (1923-2011), Department of Physics, University of California at Santa Barbara.

[97] An example of a precursor to the Three Mile Island Accident in 1979 was the August 20, 1974 event at Beznau Unit 1. A trip of one of the two turbines followed by failure of the steam dump system to operate resulted in a reactor trip and the opening of the pressurizer relief valves. One of these valves subsequently failed to close and the extended blowdown of the pressurizer resulted in the rupture of the pressurizer relief tank rupture disk. A low pressurizer level actuated the safety injection system and once level was restored the operators manually stopped safety injection to commence a normal reactor cooldown. A formalized evaluation of this "precursor event" could possibly have prevented the TMI accident.

In response to recommendations of the Risk Assessment Review Group, the NRC initiated the Accident Sequence Precursor (ASP) Program in 1979. The ASP Program is concerned with the identification and documentation of operational events that have involved portions of core damage sequences and with the estimation of associated frequencies and probabilities. Examples of non-US nuclear industry ASP perspectives are documented in References [167][168]. Examples of non-nuclear industry ASP perspectives are documented in References [169][170][171]. The term risk impact assessment is sometimes used in lieu of ASP analysis; Figure 10-1.

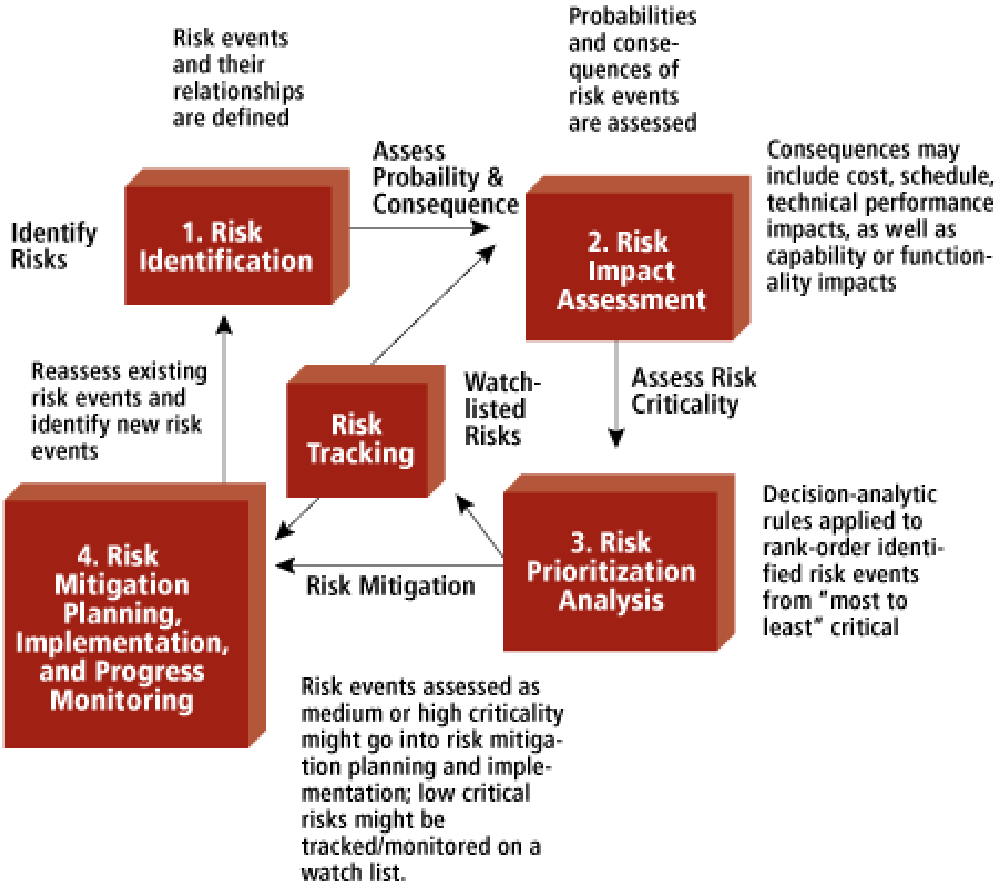

Figure 10-1: Risk Impact Assessment & Prioritization Process[98]

There are many reasons for pursuing ASP-type evaluations. In the U.S. it is an integral aspect of the Regulatory Oversight Process. As another example, an ASP-type evaluation may be performed in order to determine whether a certain event is reportable. An overriding objective of an ASP evaluation is to enable sound risk reduction management and to avoid recurring events [171].[99] From a NEA CODAP perspective, the database contains failure information from a very broad range on NSSS-designs. Using risk models could help to translate events into comparable information at a functional level [172].

[98] From Garvey, P.R., Analytical Methods for Risk Management: A Systems Engineering Perspective, Wiley Online Library, 2010. https://doi.org/10.1002/0471667196.ess7135

[99] An interesting quote from Reference [173] is: "…. What I would really like to have in the Phillips Building [*the NRC Headquarters at the time*] would be a risk simulator panel in which, when you walk into your room each morning, you've got all the accident scenarios listed along one column and you see what the major contributors to risk are in each plant that day and you pick those out and you go to work on those …."

10.2 METHODS & TECHNIQUES

This section outlines the analysis steps of a risk characterization process. It is process that begins by performing a qualitative assessment of the potential consequences of a precursor event. The qualitative assessment provides a "connection" to a plant-specific PSA model with the specifications for how to extract needed quantitative PSA information that is input to the formal risk characterization to obtain risk insights to be used in a subsequent risk management effort.

10.2.1 Problem Statement

The passive component precursor events of concern involve pipe wall thinning of a certain extent, part through-wall cracking and through-wall flaws of a certain magnitude. Depending of the outcomes from augmented non-destructive examination performed in the immediate aftermath of the initial discovery of a precursor event, the risk characterization process may also involve the consideration of common cause events. The majority of the information contained in the CODAP event database (and other similar information sources) represent precursor events. Subjecting the CODAP database content to a risk characterization process would provide a sound basis for making logical comparisons of certain event information from widely differing NSSS and BOP designs. Risk information of relevance to a degraded condition and depending on the specific circumstances may be obtained from a Level 1 PSA, Shutdown PSA, Seismic PSA [175], Fire PSA or internal flooding PSA.

10.2.2 Relevant PSA Relationships

The types of passive component failures under consideration include major structural failures and different types of precursor events. A major structural failure such as a double-ended guillotine break invariably involves spatial (or secondary effects) such as spray impact on adjacent equipment or flooding of plant areas. In a PSA context, the treatment of these events is summarized as follows:

- <u>Catastrophic Failures</u>.
  - Catastrophic failures require some kind of repair or replacement action on the component in order to restore the component to operability.
  - Catastrophic failures are generally modeled by setting the basic event to TRUE (basic event probability of 1.0) and setting its non-recovery probability, if applicable, to TRUE.
- <u>Degraded Failures</u>.
  - Degraded failures can prevent a system or train from meeting the success criteria modeled in the PSA model.
  - Degraded structures may fail from a more severe external event or fail at a condition outside its rated specifications.
  - Degraded failures are generally modeled by one of the following applications:
    - Adjusting the failure probability to a higher value, based on appropriate engineering analysis, to reflect increased likelihood of failure (e.g., due to aging, crack growth).
    - Setting the basic event to its non-recovery probability (based on a recovery analysis) when it is not feasible to conduct an engineering analysis to determine the impact of the degradation on the failure probability.
    - Adjusting the PSA success criteria.

- Incipient Failures.
  - Incipient failures have no significant degradation in performance but there are indications of a developing fault. An incipient failure that does not conform to its safety analysis basis may be classified as inoperable. The term "inoperable" has regulatory significance. It does not necessarily imply a state of physical failure. A component can be inoperable and still able to perform its PSA mitigation function over its assumed mission time.
  - Although an incipient failure will typically lead to a corrective action, the corrective action may or may not make the component unavailable to perform its function.

The figure of merit for ASP analyses is the conditional core damage probability (CCDP)[100] for initiating events and the increase in core damage probability (ΔCDP)[101] or importance for conditions. The importance is the measure of the incremental increase between the CCDP for the period in which the condition existed and the nominal CDP[102] for the same period. According to the NRC guidelines [176], two thresholds are noted in the ASP Program in the order of increasing severity.

- An initiating event with a CCDP or a condition with a ΔCDP greater than or equal to 1×10^{-6} is called a precursor.
- An initiating event with a CCDP or a condition with a ΔCDP greater than or equal to 1×10^{-3} is called a "significant precursor."

For the "initiating event impact group," pressure boundary failure (PBF) results in an initiating event. An initiating event could occur as a result of a loss of fluid (e.g., LOCA, potential LOCA due to isolation valve failure, etc.), a loss of a system (e.g., reactor auxiliary system or support system) or due to an indirect effect (e.g., spraying of an electrical bus, flooding of the rooms, etc.).

For the "loss of mitigating ability impact group," a PBF during a demand on the system is evaluated. This evaluation identifies those pipe failures that can result in a loss or degradation of a system/train, or possibly, multiple systems/trains. Reference [176] provides guidance on the form of PSA calculations to assign the consequence category to pipe failure that affects the plant mitigating ability, but does not cause an initiating event. This guidance is designed to determine the CCDP range as defined in Table 10-1. Each PSA model computation generates a CCDP from the equation given below:

CCDP (given impact from PBF) =
*[CDF (given impact from PBF) – CDF (Base)] * [Exposure Time]* (10-1)

As described in the EPRI "Risk-Informed In-Service Inspection Evaluation Procedure Reference" [177], this calculation accounts for the frequency of challenge, number of back-up trains unaffected, and exposure time. The exposure time, as described in Section 3.3.3.2.3 of Reference [177], falls into one of two categories. For pipe failures that would be discovered immediately, the exposure time is just the Allowed Outage Time (AOT) for the system/train.

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[100] The CCDP represents the probability that a core would have gone to a damaged state given that (i.e., "conditioned on") a specific initiating event occurred AND the actual plant equipment and operator responses are accounted for. This "event" use of CCDP represents the remaining probabilistic "margin" (related to defense-in-depth) to core damage at a precise moment in time, that of the event itself.

[101] ΔCDP = CCDP – CDP; the CDP accounts for testing and maintenance during the time that a degraded condition exists in the plant.

[102] The ASP estimates the CCDP given a degraded condition for the time this degradation existed. The nominal CDP, which accounts for normal maintenance, during this time, is subtracted from the CCDP to obtain the change in CDP due to the degraded condition alone (without consideration of any specific maintenance configuration that might have existed). This numerical result is then normalized by dividing it by 1 year to arrive at a delta CDF in units of "per year."

For pipe failures that could go undetected, the exposure time is taken as the test interval, when the test causes a pressure/flow demand on the pipe segment. For pipe segments that do not experience a test demand, an exposure time of one year is used.

The "combination impact group" includes pipe failures which cause both an initiating event and affect the mitigating ability, in addition to the expected and modeled effects of the initiator. For example, when a loss of an injection leg occurs with a LOCA, this is considered an expected LOCA effect on the mitigating ability, and typically analyzed as a simple initiating event.

10.2.3 Consequence Categorization of Passive Component Failure

This section gives an overview of how consequences of passive component failures are treated in RI-ISI. As one example, the consequence evaluation in PWROG methodology [178] consists of two parts, definition of consequences associated with piping failures and calculation of the CCDF/CCDP and CLERF/CLERP[103] for the defined consequences using the plant PSA model.

In the definition of consequences associated with piping failures, both direct consequences and indirect consequences are considered. The direct consequences are the effects on the plant if the fluid medium (water or steam) in the pipe not reaching its intended destination. The indirect effects are the effects on the plant from the released medium (water or steam) following a piping failure which can give flooding, spray, jet impingement, pipe whip, high temperature and high humidity. To ensure that the direct consequences are properly identified, two questions are asked.

- What are the consequences, if the piping failure occurs while the plant is operating normally (i.e., does the failure cause an initiating event)?
- What are the consequences, if the piping failure occurs as a secondary failure in a transient/accident?

Therefore, the direct and indirect consequences to be considered include:

- Failures causing initiating events such as LOCA or reactor trip (initiator)
- Mitigating system failures such as disabling a single train or system or multiple trains or systems (hidden failures or fails as a consequence of the initiating event).
- Failures that cause both the initiating event and failure of the mitigating system.

When defining direct mitigating system failures, two primary criteria are used: flow diversion and loss of inventory:

- “Flow diversion” occurs when a certain percentage of flow is diverted from its intended flow path. Typically the flow diversion criterion used in a PSA model is also used for the PWROG RI-ISI methodology [178]. Typically this is a 1/3 pipe diameter. If a branch line fails that is 1/3 the diameter (or greater) of the main line, then the function of the main line is assumed to fail.

- “Loss of inventory” occurs in a closed loop system. A piping failure may not cause sufficient diversion of flow to fail a train to perform its main function for a period of the sequence but over time, the water level in the closed loop system can lower to the point where there is pump cavitation or loss of net positive suction head and thus a loss of all flow in the system. Based on this, the effect of small bore piping failure can after long

[103] CLERF = Conditional Large Early Release Frequency.

time in a sequence can have the same effect as larger bore piping and give severe consequences.

In determining the consequences, especially in calculating the loss of inventory, a 24 hour mission time is used to be consistent with the PSA model. The actual mission time for a given function may vary based on the initiating event. Likewise the exact size piping failure is a range from a small leak to a full break with potentially different consequences. Thus to keep the process manageable, the potential direct consequences are typically based on the largest break size that could occur on the pipe and a mission time of 24 hours is used. Exceptions to basing the consequence on the largest break size include piping failures that can result in different size LOCAs, jet impingement or spray.

To determine the potential indirect effects existing documentation are reviewed and a plant walk through is conducted. The plant walk through are performed to evaluate potential flooding, spray, pipe whips and jet impingement. Typically a walkdown inside the containment is not necessary due to existing analyses and documentation.

Effects of piping failures are evaluated both without operator actions and with operator actions. Without operator action consequences assume that the operators take no action to isolate or mitigate the specific piping failure. With operator action consequences assume that the operators are perfect in taking the appropriate actions to isolate the leak (no human error probabilities are used). By evaluating consequences both without and with operator action consequences, the range of potential effects of operator actions is bounded. The probability of a success for the operator action is not specified. The following restrictions are specified to be able to credit operator action:

- There is an alarm or clear indication in the control room, to which the operator will respond.
- The operator response would be expected.
- The method for identifying the expected location of the piping failure is available such that a train or system can be recovered. Included in this is that there is sufficient time for the operators to identify, diagnose, and take the corrective actions.
- The isolation equipment is not affect by the failure.
- The action can be performed within the control room. As part of the PWROG methodology's expert panel process, the expert panel can credit operator actions that are taken outside the control room.

Core damage frequencies (CDF) and large early release frequencies (LERF) are evaluated with PSA, based on the prejudged effect of pipe failure in the different systems/segments over the specified mission time (24 hour) With a few exceptions (LOCAs, main steam piping failure etc.), piping failures are not normally included in the PSA model. Thus, a surrogate component, or a group of components, is (are) defined such that its (their) failure(s) would simulate the postulated consequences of the system/segment's failure. For failures causing initiating events such as LOCA or reactor trip (initiator) the PSA model generates a CCDP and a CLERP. For mitigating system failures such as disabling a single train or system or multiple trains or systems the PSA-model generates a CCDF and a CLERF. These results along with the failure probabilities and test intervals are used as inputs to the risk evaluation for the quantitative risk ranking and expert panel categorization of the segments.

The PWROG methodology addresses defence-in-depth at levels 1, 2 and, 3 (Table 10-1) and also to take into account the support from non-safety functions and diversified safety systems. In evaluating conditional large early release probability (CLERP) also defence in depth level 4 is addressed.

Table 10-1: Levels of Defense in Depth According to IAEA-INSAG-10 [179]

Levels of Defense-in-Depth	Objective	Essential Means to Fulfil the Objective
Level 1	Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation
Level 2	Control of abnormal operation and detection of failures	Control, limiting and protection systems and other surveillance features (work within the technical specification limits)
Level 3	Control of accidents within the design basis	Engineered safety features (safety systems) and accident procedures
Level 4	Control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents	Complementary measures and systems to avoid further release of radioactivity and accident management
Level 5	Mitigation of radiological consequences of significant releases of radioactive materials	Off-site emergency responses

The purpose of the consequence evaluation phase in the EPRI RI-ISI methodology [177] is to evaluate pipe failures in terms of their impact on Core Damage Frequency (CDF) and Large Early Release Frequency (LERF). The consequence evaluation focuses on the impact of a pipe section failure (loss of pressure boundary integrity) on plant operation. This impact can be direct, indirect or a combination of both:

- Direct Impacts. A failure results in a diversion of flow and a loss of the train and/or system or an initiating event (such as a LOCA).
- Indirect Impacts. A failure results in a flood, spray, or pipe whip, spatially affecting neighboring structures, systems and components or results in depletion of a tank and loss of the systems supplied by the tank.

Spatial effects are an example of indirect effects caused by pressure boundary failures. These include the effects of flood, spray, and pipe whip on equipment located in the vicinity of the break. Spatial consequences of the break are determined based on the location of the analyzed break and the relative position of important equipment. The presence of important equipment in a specific location can be identified through existing analyses (e.g., internal flood analysis or fire analysis) and should be confirmed by a walkdown.

The possibility of isolating a break is also identified and accounted for as part of the consequence analysis. A break could be isolated by a protective check valve, a closed isolation valve, or it could be automatically isolated by an isolation valve that closes on a given signal. If not automatically isolated, a break can be isolated by an operator action, given successful diagnosis. Depending upon the scenario and other factors, operator action can be taken within or outside the control room. The likelihood of success of these actions depends on the availability of isolation equipment, a means of detecting the break, the amount of time available to prevent specific consequences (e.g., flooding of the room or draining of the tank), and human performance. If isolation is possible, the consequence assessment should be conducted for both cases: successful and unsuccessful isolation.

For each run of piping under evaluation, a spectrum of break sizes is evaluated. The break size ranges from a small leak to a rupture. Larger leaks and breaks have the potential to disable system or trains and to cause initiating events, flooding, or diversions of water sources. Typically, small breaks (minor leakage) would not render a train inoperable. They may, however, depending on the energy level of the system, spray onto adjacent equipment and cause equipment malfunction.

Plant evaluations have shown that the large break scenarios (worst-case breaks) result in the most limiting consequences. However, the methodology was specifically developed to require that a spectrum of break sizes be evaluated so that, if smaller breaks can cause a measurable or the dominant consequence, they are identified and input into the risk ranking process.

The goal of the consequence evaluation is to establish a process that consistently ranks consequences caused by a pipe failure, based on its risk impact or safety significance. For example, in order to rank piping failures consistently, one needs to address the question of whether a pipe break that results in a loss of coolant accident (LOCA) is more safety significant than a pipe break that leads to a loss of feedwater? Similarly, is a pipe break that disables one train of high pressure injection more safety significant than a pipe break, which disables an auxiliary feedwater train? In order to answer these questions consistently, consequences are categorized into different importance categories. The consequences are ranked into those categories based on a combination of plant-specific PSA insights and results, and methodology lookup tables. The methodology lookup tables were developed, in order to standardize and streamline the consequence ranking process.

Four consequence importance categories have been defined based upon PSA evaluation. They are: high, medium, low, and none. The high category represents events with a significant impact on plant safety, while the low category represents events with a minor impact on plant safety. The “none” category defines those locations that have no impact on plant safety and are typified by “abandoned in place” piping. These categories are defined by a range of Conditional Core Damage Probability (CCDP) or Conditional Large Early Release Probability (CLERP), associated with the impact of specific Pressure Boundary Failure (PBF). The ranges used to numerically define each category are shown in Table 10-2 and Figure 10-2. The CCDP and CLERP ranges are determined based on the estimates of the total risk associated with the piping failure. Risk is measured by Core Damage Frequency (CDF) or Large Early Release Frequency (LERF) as:

$$\text{CDF [for a PBF]} = \text{[PBF frequency]} \times \text{[CCDP]} \quad (10\text{-}2)$$

$$\text{LERF [for a PBF]} = \text{[PBF frequency]} \times \text{[LERF]} \quad (10\text{-}3)$$

Table 10-2: EPRI RI-ISI Consequence Categories

CCDP	CLERP	EPRI classification
CCDP > 10^{-4}	CLERP > 10^{-5}	High
10^{-6} < CCDP ≤ 10^{-4}	10^{-7} <CLERP≤ 10^{-5}	Medium
CCDP ≤ 10^{-6}	CLERP ≤ 10^{-7}	Low

Based on the above expression, and using a conservative estimate of the total PBF frequency for the plant (estimated in the order of 10^{-2} per year), CCDP and CLERP ranges are selected to guarantee that all pipe locations ranked in the low consequence category do not have a potential CDF impact higher than 10^{-8} per year or a potential LERF impact higher than 10^{-9} per year; Figure 10-2. The boundaries between the high and medium consequence categories, at CCDP and CLERP values of 10^{-4} and 10^{-5} respectively, are set to correspond with the definitions of small CDF and LERF values of 10^{-6} and 10^{-7} per year.

The assumption that 10^{-6} and 10^{-7} represent suitably small CDF and LERF values is consistent with the decision criteria for acceptable changes in CDF and LERF found in RG 1.174 [180]. The medium category is selected to cover the area between high and low categories, and to address uncertainties in the CCDP and CLERP estimates.

For each inspection location, an inspection for cause program is implemented to ensure that appropriate examination methods, procedures, acceptance criteria, and evaluation standards are applied to address the degradation mechanisms of concern.

Potential for Pipe Failure	Consequence Category (CCDP)			
	"None" CCDP < 10^{-8}	Low 10^{-8} < CCDP ≤ 10^{-6}	Medium 10^{-6} < CCDP ≤ 10^{-4}	High CCDP > 10^{-4}
High FAC, High-Cycle-Fatigue Water Hammer	**LOW** Category 7	**MEDIUM** Category 5	**HIGH** Category 3	**HIGH** Category 1
Medium Corrosion (MIC, Pitting), Erosion-Cavitation	**LOW** Category 7	**LOW** Category 6	**MEDIUM** Category 5	**HIGH** Category 2
Low No Damage / Degradation Mechanism	**LOW** Category 7	**LOW** Category 7	**LOW** Category 6	**MEDIUM** Category 4

Figure 10-2: EPRI RI-ISI Matrix for Pipe Failure Risk Characterization

In the alternative RI-ISI methodology as articulated in ASME Code Case N-716 [181], the change-in-risk evaluation shall be performed prior to the initial implementation and in the following manner:

a. Bounding Failure Frequency. The failure frequencies of 2E-06 per weld-year for welds in the high failure potential category, 2E-07 per weld-year for welds in the medium failure potential category, and 1E-08 per weld-year in the low failure potential category may be used as bounding failure frequencies.

b. Conditional Risk Estimates. The estimated conditional core damage probability (CCDP) and conditional large early release probability (CLERP) may be used if available. Bounding values of the highest estimated CCDP and CLERP may be used if specific estimates are not available.

c. The following general equations shall be used to estimate the change-in-risk. One estimate shall be made for the change in core damage frequency (CDF) and one for large early release frequency (LERF). The equations only illustrate the change in CDF. The change in LERF due to application of the process shall be estimated by substituting the CLERP for CCDP in the equations.

$$\Delta R_{\mathrm{CDF}} = \Sigma_{\mathrm{j}}\, (I_{rj} - I_{ej}) \times PF_j \times \mathrm{CCDP}_j \qquad (10\text{-}4)$$

Where

Σj = summation of locations selected for examination.
ΔR_{CDF} = change in CDF due to replacing the prior deterministic ISI program with the ISI program developed in accordance with the Code Case.
I_{rj} = factor of reduction in pipe rupture frequency at location j associated with the ISI program developed by the Code Case.
I_{ej} = factor of reduction in pipe rupture frequency at location j associated with the prior deterministic ISI program.
PF_j = piping failure frequency at location j without examination.
CCDP_j = conditional core damage probability at location j.

According to Code Case N-716, it is acceptable to use bounding estimates for pipe failure frequency, conditional core damage probability, and conditional large early release probability, to simplify the calculations. Any increase in CDF and LERF for each system shall be less than 1E-07 per year and 1E-08 per year, respectively, and the total increase in CDF and LERF should be less than 1E-06 per year and 1E-07 per year respectively. If necessary, additional examinations shall be selected to meet this acceptance criterion.

10.2.4 Risk Categorization According to SDP

The main purpose of the SDP is to determine the safety significance of inspection findings. The SDP is part of the Reactor Oversight Process (ROP). The SDP uses a three-phased approach to determine the significance of inspection findings in the initiating events, mitigating systems, and barrier integrity [182][183].

The SDP process is designed to estimate the increase in annualized CDF risk due to identified deficiencies in licensee performance that led to unavailability of equipment or safety functions, or to the increase in initiating event frequencies. This increase is measured from the normal annualized CDF that results from routine plant operation. The additional risk contributions caused by deficient licensee performance (as characterized by the SDP) are assumed to be additive to this normal annualized CDF which already includes the risk contribution due to the probabilities of equipment failures expected occasionally for industrial facilities of this size and complexity.

Another contribution to normal annualized CDF is caused by planned preventive maintenance and testing activities which cause the CDF at any particular moment in time to fluctuate dependent upon the changes in plant equipment status. The additional annualized CDF risk due to deficient licensee performance must be dependent only upon the performance issue itself and not the particular plant configurations during which the issue occurred. Therefore, if a degraded equipment or function is identified to exist simultaneously with equipment outages for preventive maintenance or testing, the SDP inputs cannot include the contribution of the maintenance or testing, since this is already included in the normal annualized CDF against which the change is being measured. This non-consideration of routine maintenance and testing is a departure from the historical enforcement practice of including the consideration of any additional equipment unavailability that made the loss of function due to a deficiency more severe, even if the added unavailability were due to routine maintenance.

If equipment outages due to maintenance were included in this delta CDF estimation, the result would potentially render results of higher significance. This would result in assessments of the risk impact of licensee performance that inappropriately would depend as much on the licensee's appropriate conduct of on-line maintenance as on the licensee's deficient performance. The objective of using the SDP is to characterize the significance of inspection findings in a manner that is comparable to performance indicators for use in the NRC Action Matrix. The reactor safety cornerstone performance indicator thresholds were developed based on the increase to annualized CDF represented by the value of the indicators. Thus, in comparing and "adding" the effects of PIs and inspection findings within the Action Matrix, it is necessary to use the same risk metric.

A basis document for establishing risk guidelines is Regulatory Guide (RG) 1.174 [180]. The metrics that have been adopted in RG 1.174 for the characterization of risk are CDF and LERF. In RG 1.174, acceptance guidelines were established for assessing changes to the licensing basis of a plant; Figure 10-3. Acceptance is predicated on increases in CDF and LERF implied by the change to the licensing basis being small.

The instantaneous CDF varies with time as SSCs are taken in and out of service. A time dependent map of this instantaneous CDF is a representation of the risk profile as a function of time. The base CDF evaluated using the PSA model represents the average over this risk profile. The performance deficiency results in an additional increase in the instantaneous CDF as a result of additional components being unavailable. The impact of the performance deficiency for SDP purposes is to be evaluated for the average unavailabilities of other unrelated SSCs, i.e., it is averaged over the risk profile. This is achieved using the following equation:

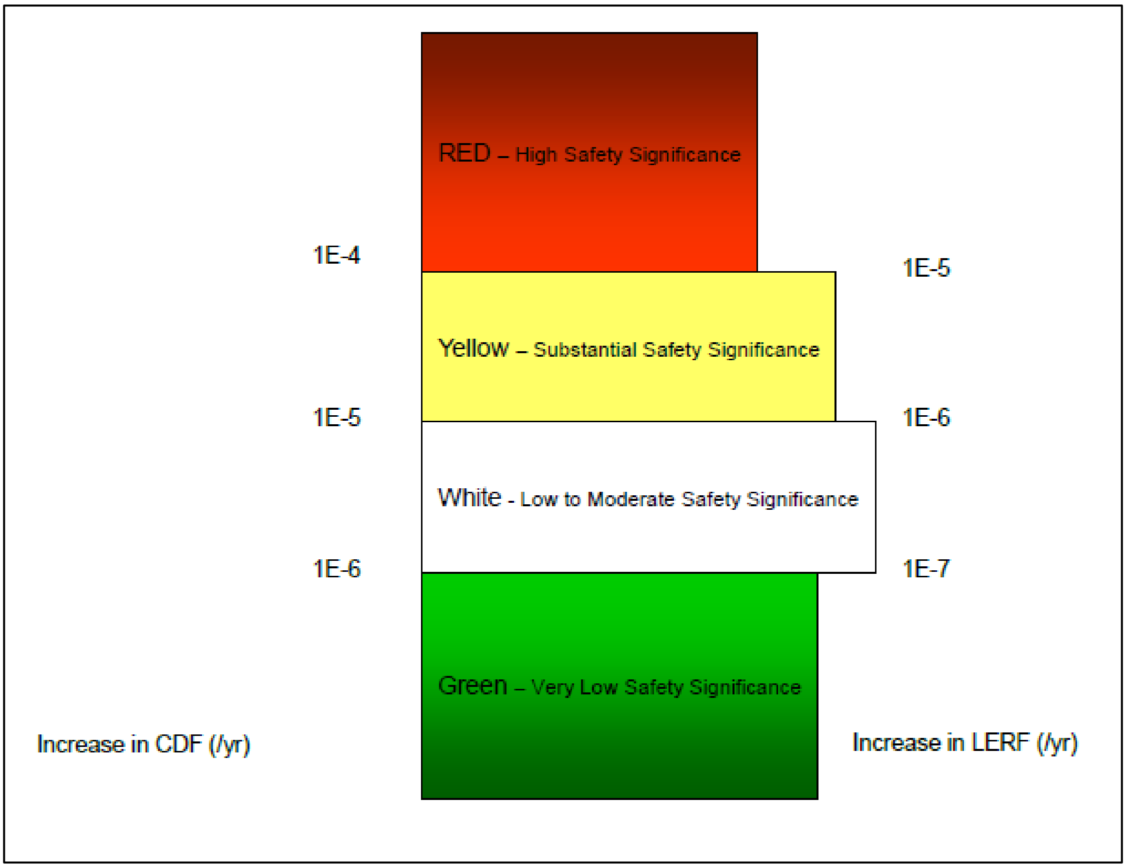

Figure 10-3: Quantitative Significance of SDP Finding

$$CDF_{PD} = [T_0/(T_0 + T_1)] \times CDF_{base} + [T_1/(T_0 + T_1)] \times CDF_1 \qquad (10\text{-}5)$$

Where CDF_{PD} is the new CDF given that the performance deficiency has occurred, CDF_{base} is the base PSA CDF, CDF_1 is the CDF conditional on the additional unavailability, T_1 is the duration of the performance deficiency, and $T_0 + T_1$ is the nominal time over which the CDF is evaluated (one year). This calculation establishes a new increased average CDF that is interpreted as the CDF that would result if the performance deficiency were allowed to persist. Therefore, the change in CDF resulting from the performance deficiency is calculated by:

$$\Delta CDF = CDF_{PD} - CDF_{base} = (CDF_1 - CDF_{base}) \times [T_1/(T_1 + T_0)] \qquad (10\text{-}6)$$

From a mathematical point of view, this formulation is equivalent to, and has been interpreted as, calculating an ICDP, i.e., the incremental core damage probability caused by the performance deficiency. However, in the above formulation, the duration factor is in fact a fraction and dimensionless, and not time, and therefore, it is a direct calculation of ΔCDF.

10.3 Conditional Failure Probability

In the risk characterization of precursor events such as degraded passive components the assessment of conditional failure probabilities is an essential step. The likelihood of a pipe flaw propagating to a significant structural failure is expressed by the conditional failure probability (CFP). A typical presentation format for the CFP term in the piping reliability equation is as a cumulative failure probability versus an equivalent break size (EBS) expressed in terms of through-wall mass flow rate in kg/s or the size of the hole in the pressure boundary converted to and equivalent diameter in mm. The slope of the CFP curve has a relatively strong decreasing probability versus an increasing failure magnitude; Figure

10-4. In the given example, the CFP versus EBS relationship for the BBL piping is estimated directly from service experience data and by utilizing a Bayesian model of piping reliability [184]. For the LBB piping the CFP versus EBS relationship is estimated using a methodology described Section 10.3.3.

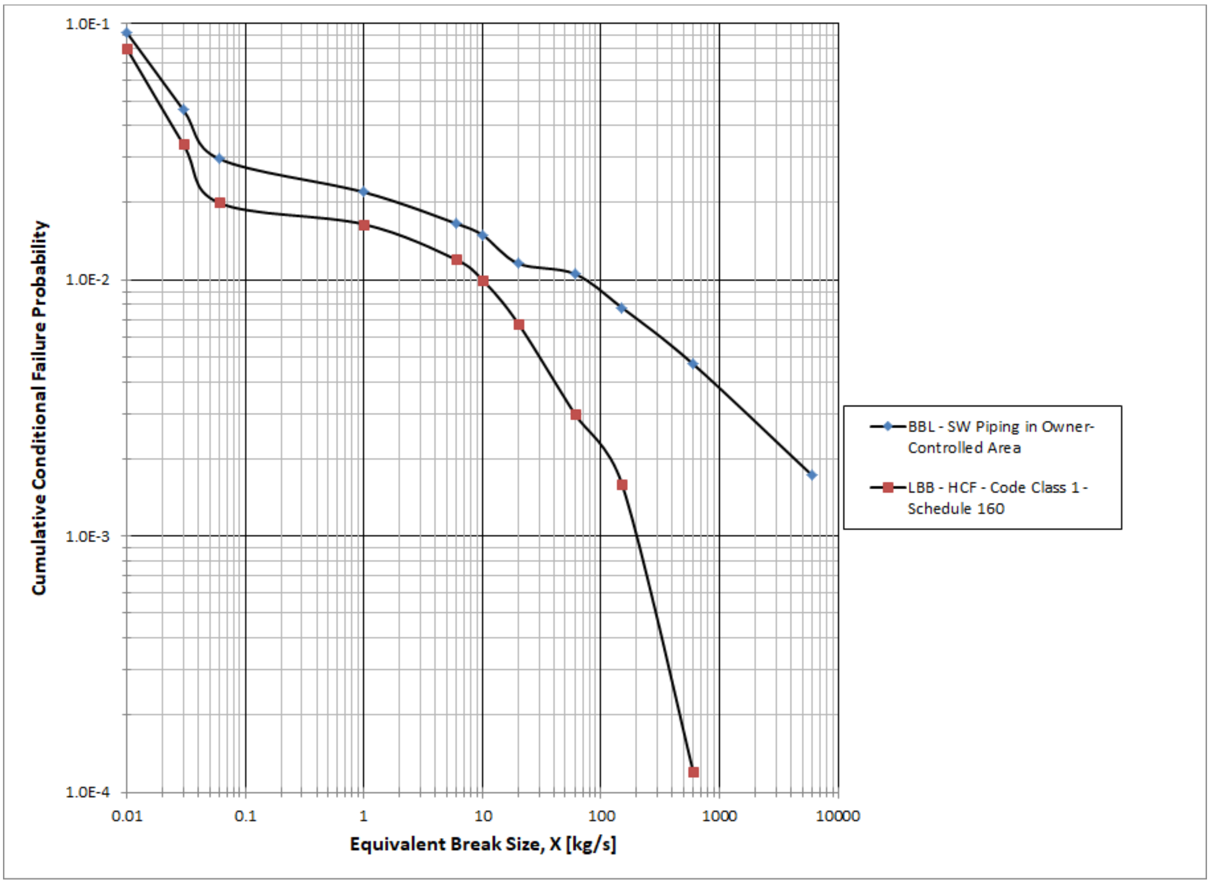

Figure 10-4: Example Cumulative Conditional Failure Probability Curve

With no service data available to support a direct statistical estimation of the conditional probability the assessment can be based on probabilistic fracture mechanics (PFM), expert judgment, or a combination of service data insights, expert judgment and PFM. Different PFM algorithms have been developed, but with focus on fatigue growth through vibration fatigue, thermal fatigue) and stress corrosion cracking in Reactor Coolant Pressure Boundary (RCPB) piping. There remain issues of dispute with respect to reconciliation of results obtained through statistical estimation and extrapolation versus the physical models of PFM, however.

Results from studies to benchmark PFM calculations against field experience have shown PFM computer codes to over-predict pipe failure rates by more than an order magnitude relative to statistical estimates of field experience data. In general, the results obtained with PFM computer codes are quite sensitive to assumptions about weld residual stresses, crack growth rates, and correlations of crack initiation times and growth rates.

10.3.1 Basic Two-Parameter Beta Distribution Approach

From a statistical perspective, a practical approach to calculating conditional pipe failure probabilities is to use a Bayesian approach in which the prior conditional failure probability uncertainty distribution is expressed by a Beta Distribution. The Beta Distribution takes on values between 0 and 1 and is defined by the two parameters "A" and "B" (or "α" and "β" in some texts). It is often used to express the uncertainty in on-demand failure rates. The mean of the Beta Distribution is given by:

$$\text{Mean} = A/(A + B) \quad (10\text{-}7)$$

If A = B = 1, the Beta Distribution takes on a flat distribution between 0 and 1. If A = B = ½, the distribution is referred to as Jeffrey's non-informative prior and is a U-shaped distribution with peaks at 0 and 1. Expert opinion can be incorporated by selecting A and B to match up with an expert estimate of the mean failure probability. For example, to represent an expert estimate of 1.0×10^{-2}, A = 1 and B = 99 can be selected. These abstract parameters can be associated with the number of "failures" and the number of "successes" in examining service data to estimate the conditional pipe failure probability.

Figure 10-5 is a summary of operating experience with piping of all types. It includes plots of the fraction of pipe failure of a certain safety class versus the consequence of failure (expressed as the peak leak/flow rate threshold value, and irrespective of pipe size) to all failures in that safety class. Also included in Figure 10-5 are plots for a group categorized as "Class 1 & 2 Small-Diameter Piping". For comparison, the chart includes a correlation for the conditional pipe failure probability derived through an expert elicitation process (this is denoted as "NUREG-1829" [71]), and a semi-empirical correlation, which is referred to as the "Beliczey-Schulz correlation" [185].

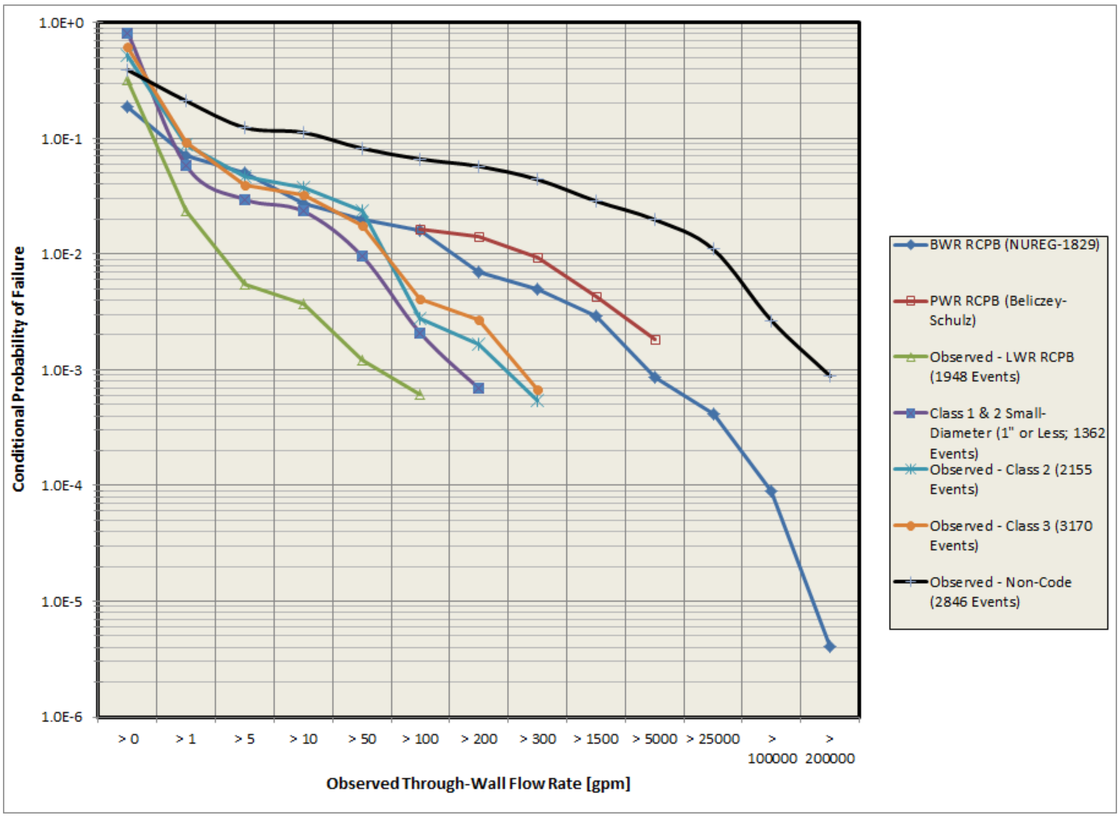

Figure 10-5: Likelihood of Pipe Failure According to Empirical Data & Theoretical Studies

According to Figure 10-5, the conditional pipe failure probability is bounded upwards by a curve identified as "Observed-Non-Code." This category includes high-energy Turbine Building carbon steel piping that is susceptible to flow-accelerated corrosion (FAC).

Examples of such piping systems include Condensate, Feedwater, and Extraction Steam. The process medium in this class of piping is super-heated water, steam, or wet steam. In high-energy piping susceptible to FAC a major structural failure may occur without a precursor such as a minor leakage through a pinhole size flaw in the pipe wall. This is referred to as break-before-leak (BBL) as opposed to leak-before-break (LBB). There have been numerous, well documented major structural failures involving FAC-susceptible; or "BBL piping," for example:

- Surry-2 (PWR, December 1986), the peak through-wall mass flow rate was estimated to be about 5,050 kg/s.
- Loviisa-1 (PWR, May 1990), peak through-wall mass flow rate was estimated to be about 665 kg/s.
- Mihama-3 (PWR, August 2004), peak through-wall mass flow rate was estimated to be about 475 kg/s.

The empirical data used to construct the chart in Figure 10-5 represents more than 11,000 recorded pipe failure events, representing approximately 14,300 reactor-years of commercial nuclear power plant operation. The chart covers a wide variety of piping systems, from RCPB piping, Safety Injection & Recirculation piping, Auxiliary Cooling piping, to Balance of Plant piping. Based on the available operating experience data and expert opinions, Figure 10-5 represents our current state-of-knowledge with respect to the probability of pipe failure. Some additional insights from service data reviews:

- For Code Class 1 (or Reactor Coolant Pressure Boundary) piping the most significant failures to date with respect to the observed peak through-wall flow rates have involved small-diameter piping (25 mm or smaller). Of all failures involving through-wall flaws, about 14% have involved socket weld failures in DN20 or DN25 stainless steel lines. So far the largest observed through-wall flow rate is about 8 kg/s.

- Failure of large-diameter, thick-walled Class 1 piping is unlikely. A primary reason for this is the presence of mid-wall compressive residual stresses that tend to retard formation of deep cracks.

- To date there have been six Class 1 pipe failures involving piping of size > DN50 and > 6.3×10^{-2} kg/s peak flow rate but less than 1.9×10^{-1} kg/s peak flow rate.

- The plots of empirical data are based on estimated peak flow rates. In Class 1 piping and connecting Class 2 piping, the cracks that develop in the through-wall direction tend to be very tight and producing only minor detectable leakages, if any, while at full operating pressure.

The five data points in Figure 10-5 that represent the "Beliczey-Schulz correlation" [185] correspond respectively to failed DN15, DN20, DN25, DN50 and DN100 Class piping in a PWR operating environment. A break in a DN100 pipe at full operating pressure (15 MPa) would generate a liquid peak through-wall flow rate of about 545 kg/s. According to Equation (5) below, the conditional failure probability of a structural failure of this magnitude would be approximately 1.8×10^{-3}. Equation (10-8) reflects a German industry perspective on the conditional pipe failure probability, which is based on service experience as well as fracture mechanics evaluations and experimental data as of the mid-1980s. According to this research the conditional pipe failure probability $P_{ik}\{SF_x|F\}$ is:

$$P_{ik}\{SF_x|F\} = (9.6 \times DN/2.5 + 0.4 \times DN^2/25)^{-1} \quad (10\text{-}8)$$

Where

$P_{ik}\{SF_x|F\}$ = Conditional probability of structural failure (SF) of magnitude "x" for piping component "i" (e.g., weld of certain configuration) and damage and/or degradation mechanism "k".
DN = nominal pipe diameter [mm]

The “aggregate state-of-knowledge correlation” from NUREG-1829 [71] is a result of an expert elicitation process. It applies to BWR primary system piping and is derived from Figure 7.6 in NUREG-1829 using a total pipe failure rate (including all Class 1 system piping in a typical BWR) of 3.0×10^{-2} per reactor-year, based on available service experience data. Different conclusions can be drawn from the information embedded in the chart in Figure 10-5.

For non-Code piping there is no need to perform any data extrapolations since there is ample empirical data to support the statistical estimation of conditional pipe failure probabilities across the full range of pipe sizes and peak through-wall flow rates. For all other piping some form of data extrapolation is required, unless PFM modeling is pursued. The validity of the results of PFM modeling is a function of the assumptions behind input data and the extent by which verification and validation of calculation algorithms have been performed.

Since the empirical correlations are derived from historical data, any future, significant pipe failure would alter the characteristics of these correlations. For medium- and large-diameter safety-related piping (Code Class 1, 2 or 3), the effect of a single significant failure would cause an empirical correlation to approach the “aggregate state-of-knowledge correlation” of NUREG-1829.

Assuming a statistical basis is sought for the estimation of conditional pipe failure probabilities, exactly how are data extrapolations to be formulated and how are the uncertainties in such extrapolations expressed? The Beta Distribution has some convenient and useful properties for use in Bayes’ updating. A prior distribution, representing a analyst’s understanding of piping performance, can be assigned by selecting an appropriate set of initial values for parameters A and B, denoted as A_{Prior} and B_{Prior}. Then, when looking at the relevant service experience, if there are “N” failures and “M” successes, the Bayes’ updated, or posterior distribution is also a Beta Distribution with the following parameters:

$$A_{Posterior} = A_{Prior} + N \tag{10-9}$$

$$B_{Posterior} = B_{Prior} + M \tag{10-10}$$

The above explains how the Beta Distribution can be used to estimate conditional pipe failure probabilities. The prior parameters should be selected in such a way that they are representative of engineering estimates (e.g., PFM results) “prior” to the collection of evidence in the form of pipe failure data. Equations (10-9) and (10-10) are used to calculate the parameters of the Bayes’ posterior distribution after applying the results of a database query to determine N and M. “N” corresponds to the number of structural failures of some well-defined magnitude and in some specialized combination of pipe size and material and “M” corresponds to the total number of failures that do not result in a structural failure in the corresponding pipe size/material combination. This model assumes that all instances of a degraded condition are precursors to a structural failure.

Selecting well justified “A” and “B” parameters is not a trivial task. Many different parameter combinations will produce the same mean value. As implied, insights and results from PFM studies could be utilized in defining application-specific parameters. In general, for situations where a lot of prior information is available the parameter selections should be determined by utilizing the available evidence to the fullest extent possible. Where very little evidence is available about the parameters non-informative priors may be selected.

Using the information in Figure 10-4 there is sufficient evidence across the full range of pipe failure consequences for non-Code piping to select a robust set of A and B parameters; with A known based on field experience a corresponding B parameter can be calculated in a straightforward manner. By contrast, for Class 1 piping there is no evidence of pipe failures involving significant consequences (e.g., peak through-wall flow rates much greater than 6.3 kg/s (or 100 gpm). For such a case one can say that the A parameter has to be a small number. A reasonable non-informative A_{Prior} should be approximately ½. The empirical correlations as displayed in Figure 2 combined with the underlying information should be a guide for how to best select the most appropriate set of A and B parameters.

According to this figure, a Class 1 pipe failure producing a through-wall peak flow rate greater than 380 kg/s corresponds to mean conditional failure probability of about 5.0×10^{-4} according to NUREG-1829, which would be our prior mean value given A = ½ and B = 999.5. Note that these parameters account for a relatively broad composite of different plant systems. Risk-informed applications typically require highly specialized pipe failure probabilities, for example to address a specific system and pipe size. Next, the posterior uncertainty distribution would be determined by applying query results from a pipe failure database.

Figure 10-6 includes results of an application of this approach to the estimation of the conditional failure probability for two classes of piping systems in PWR plants: 1) small-diameter Code Class 1 and 2 stainless steel piping susceptible to high-cycle fatigue (HCF), and 2) non-Code carbon steel medium-size Main Feedwater piping susceptible to flow-accelerated corrosion (FAC).

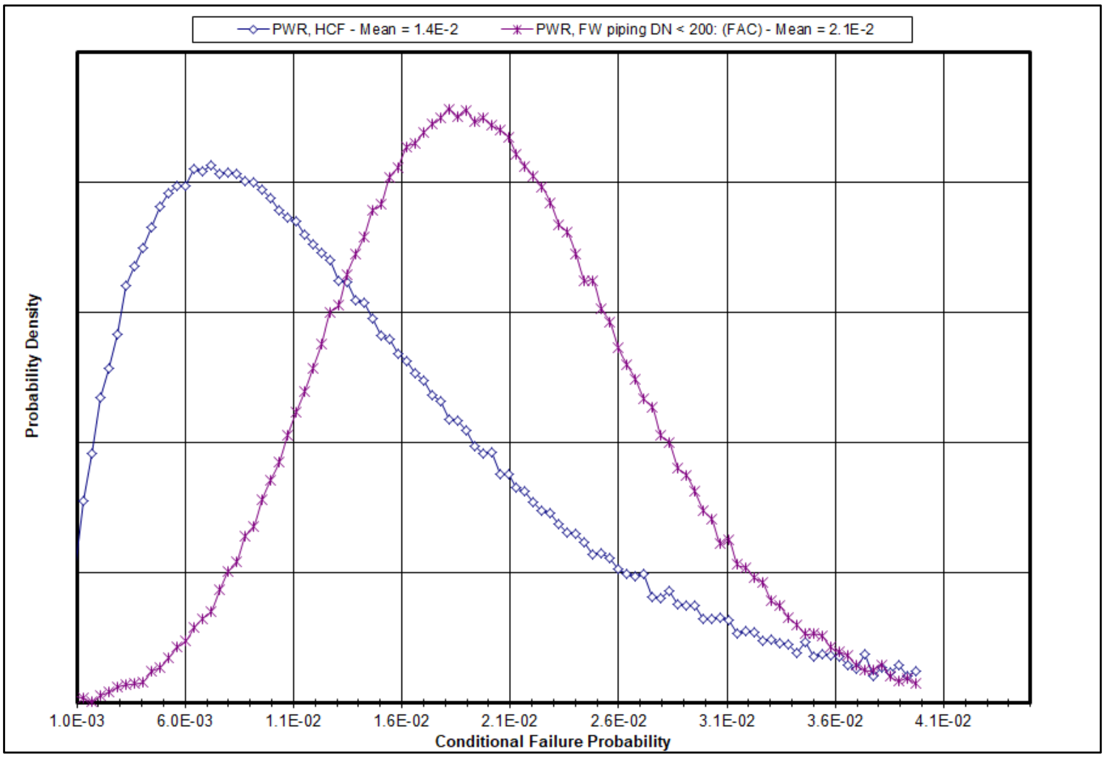

Figure 10-6: Conditional Pipe Failure Probability Given Flow-Accelerated Corrosion & High-Cycle Fatigue

To demonstrate how to select prior and posterior Beta distribution parameter two raw water piping systems are selected. Structural failure of Service Water piping system is considered by most internal flood risk assessments. It is a piping system with infinite supply of cooling water from a power plant's ultimate heat sink (UHS). Three types of structural failure as defined by consequence of a pipe failure may be accounted for by such assessments: 1) spray, 2) significant liquid release but within the capacity of a plant building floor drain system, and 3) major liquid release in excess of the capacity of a floor drain system. The conditional failure probability is determined from service experience insights and engineering judgment, with the uncertainty treated using the Beta Distribution. The "A" parameter corresponds to the consequence of pipe failure, and the "B" parameter corresponds to the failure experience. Table 10-3 summarizes a set of proposed prior and posterior Beta Distribution parameters for Service Water piping.

Table 10-3: Proposed Parameters of the Beta Distribution for the Conditional Probability of Service Water Pipe Failure

Pipe Failure Consequence	Prior Beta Parameters			Posterior Beta Parameters		
	Constraint	A	B	A_{Post}	B_{Post}	Mean
Spray	5.0E-1	1	1	1+34	1021	3.2E-2
Significant Liquid Release	5.0E-2	1	19	1+4	1069	4.7E-3
Major Liquid Release	1.0E-3	½	500	½+0	1120	4.5E-4

It is noted that it is the "weight" of the operating experience data that ultimately determines the shape of the Beta Distribution. In this example, the justifications for the parameters listed in Table 10-3 are as follows:

- Spray Event. The prior Beta parameters are defined so that the mean value corresponds to 0.5 (i.e., 50% chance that a through-wall flaw is large enough to generate a water spray rather than a drop leakage); i.e., $A_{Prior} = B_{Prior} = 1$. According to the available service experience data, there have been 34 events with recorded leak rates in excess of 6.3×10^{-2} kg/s (1 gpm) but no more than 6.3 kg/s (100 gpm). Furthermore, there have been a total of 1054 SW pipe failures. Therefore, the posterior Beta parameters become $A_{Post} = A_{Prior} + 34$, and $B_{Post} = B_{Prior} + (1054\text{-}34) = 1021$. This corresponds to a predicted mean of 3.21×10^{-2} (*Mean* = 35/(35 + 1054).
- Significant Liquid Release. The prior Beta parameters are defined so that the mean value corresponds to 5.0E-02; again, this is based on the corresponding empirical correlation in Figure 10-5. There have been 4 events with recorded spill rates in excess of 6.3 kg/s (100 gpm), which is the evidence for determining the posterior Beta parameters. All pipe size classes can potentially generate a spill rate in excess of 6.3 kg/s (100 gpm) given a sufficiently large through-wall flaw. This corresponds to a predicted mean of 4.65×10^{-3}.
- Major Liquid Release. The prior Beta parameters are defined so that the mean value corresponds to 1.0E-03; this is based on engineering judgment. There have been 0 events with recorded spill rates in excess of 127 kg/s (2,000 gpm), which is the evidence for determining the posterior Beta parameters. All pipes of 80 mm diameter or greater can potentially generate a spill rate in excess of 127 kg/s given a sufficiently large through-wall flaw. The total service experience includes 620 records for Service Water piping of diameter $\varnothing \geq 80$ mm. This corresponds to a predicted mean value of 4.46×10^{-4} for the conditional failure probability.

10.3.2 Enhanced Beta Distribution Approach

The two-parameter Beta distribution has been well-accepted to model the conditional probability of failures, because of 1) the fact that a beta distribution is used in the case of failure on demands, 2) its versatility and 3) its ease of use in a Bayesian framework. However, assessing the parameters A and B of the beta distribution is not a trivial task. Two "fuzzy" (as in engineering judgment) approaches have been applied in PSA projects. The first one consists in assessing the conditional probability of failure p by using the relationship p = A/(A+B). One can then assess the parameters A and B to match the correct probability of failure. However, this approach suggests that an infinite number of combinations of A and B can lead to one value of p.

The second approach is a well-accepted approach, in which A represents the consequence of a pipe failure (rupture) and B represents the failure experience (that results in repair). Data do not allow to assess A as very often, A = 0 (according to the applicable service experience there have been zero events of specified consequence such as major structural failure). However, B can be determined in a robust manner as data is available to quantify B. Therefore, an approach consists in setting A = 1 and determining B based on available data. Although this approach has been well accepted, there is no technical justifications on why A = 1, other than being a conservative value; i.e., the "true" value lies between 0 and 1. To cope with the fuzziness of these approaches, this report presents enhanced methods for selecting realistic (as in defensible) parameters A and B. These methods include:

- Method of moments,
- PERT approach,
- Pearson-Tukey approach,
- Method based on skewness,
- Method based on the degree of confidence.

These methods often rely on estimations of experts but often, they require knowledge from the experts that they sometimes do not have. The use of a constrained non-informative prior is then a solution to study the reliability of piping systems for which little information is available but there is information on the mean probability of failure.

Three-point approximations are widely used when assessing parameters of distributions because there is usually a good sense of what the median, and two bounds are (for instance 5th, 50th, 95th percentiles, or 2.5th, 50th, 98.5th percentiles). Extensive work has been performed to determine the best approximations of the mean and standard deviation of a beta distribution, based on three percentiles. Keefer and Bodily [186] claimed that the approach by Pearson and Tukey [187] is the best between all three-point estimate approximations. Subject matter experts (SMEs) are asked to provide the 5th, 50th, 95th percentiles, noted $C_{.05}$, $C_{0.50}$, $C_{0.95}$. The information provided by an expert may be based on results obtained by advanced structural reliability models; e.g. PFM models. According to the approach proposed by Pearson and Tukey, the estimated mean μ and variance σ^2 can be obtained through the following equations:

$$\mu = 0.185\ (C_{0.95} + C_{0.05}) + 0.63\ C_{0.50} \qquad (10\text{-}11)$$

$$\sigma_0^2 = \left(\frac{C_{0.95} - C_{0.05}}{3.25}\right)^2 \qquad (10\text{-}12)$$

$$\sigma^2 = \left[\frac{C_{0.95} - C_{0.05}}{3.29 - \frac{0.1(C_{0.95} + C_{0.05} - 2C_{0.50})^2}{\sigma_0^2}}\right]^2 \qquad (10\text{-}13)$$

Then, parameters *A* and *B* can be obtained through the formulas:

$$\begin{cases} A = \left[\frac{(1-\mu)\,\mu}{\sigma^2} - 1\right]\mu \\ B = \left[\frac{(1-\mu)\,\mu}{\sigma^2} - 1\right]\ (1-\mu) \end{cases} \qquad (10\text{-}14)$$

It is assumed that a Beta distribution describes well the conditional probability of failure of piping systems. Under this assumption, statistical values of percentiles (Pearson-Tukey approach) can be assessed by experts and can be used to estimate the values of the parameters A and B of the Beta distribution. These values represent the prior distribution parameters. Instead of assuming a Beta distribution, a non-informative prior can be used. It is especially appropriate to use a non-informative prior when little information is available (for example in the case of ruptures). However, experts often have a good estimate of the mean of the probability of failure. The prior can later be updated with experience data that are modeled with a binomial likelihood distribution. Whenever the current state-of-knowledge establishes a foundation for assessing the 5th, 50th, 95th percentiles of the conditional probability of failure, the SME can use the Pearson-Tukey approach to define a defensible set of prior distribution parameters. Figure 10-7illustrates the Pearson-Tukey approach.

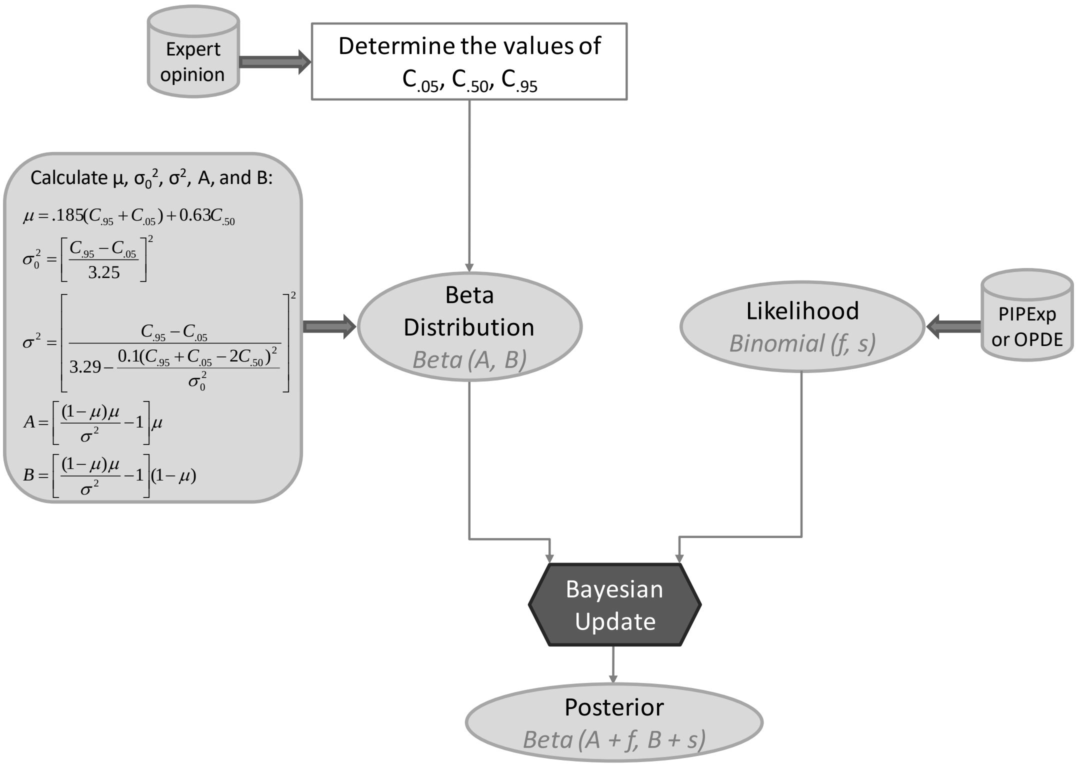

Figure 10-7: Analysis Process for the Pearson-Tukey Approach

To demonstrate the application of the Tukey-Pearson approach, the pipe rupture model of EPRI 1021086 [188] is selected. The EPRI model uses a Beta distribution to estimate conditional failure probabilities and with parameter A fixed at 1 to allow a closed form solutions to derive corresponding B parameters. Table 10-4 is a comparison between the EPRI approach and Tukey-Pearson approach as applied to Service Water (SW) piping reliability analysis. The results differ mainly in the uncertainty characterization.

Table 10-4: Comparison of Distribution Parameters

Analysis Case		**Prior Distribution Parameters**			**Posterior Distribution Parameters**			
EBS 1)	**Evidence**	**Model**	**A**	**B**	**Mean**	**5th**	**50th**	**95th'**
70 mm	0 failures at EBS, 1216 total no. failures	EPRI	1	999	4.51E-4	Assumes LN-distribution with Range Factor 2) = 10		
		Tukey-Pearson 3)	0.424	423.3	2.58E-4	3.88E-7	9.99E-5	1.05E-3

1. Range Factor = $(95^{th} / 5^{th})^{0.5}$
2. The SME input is LN mean = 1×10^{-3} and with RF = 10

10.3.3 Development of CFP Model Parameters Using PFM Results

Two different CFP models were used as input to a reliability analysis of a High Pressure Safety Injection nozzle susceptible to thermal and mechanical high-cycle fatigue. The first CRP model was based on the NUREG-1829 Expert Elicitation Study [71], and the second CRP model was based on a probabilistic fracture mechanics (PFM) analysis that was performed in support of the 2014 Duke Energy response to the U.S.NRC Significance Determination Process (SDP) evaluation (refer to Section 4.2 of this report). In the PFM analysis a set of different calculations were performed using the "beyond-PRAISE Version 1.0" fracture mechanics code.[104] These calculations were performed to simulate different HPI nozzle flaw characteristics (e.g. stable-through-wall versus through-wall with variable crack length) and cyclic stresses that promote crack growth. These results were interpreted as lower bound, median and upper bound CFPs. A two-parameter Beta distribution was used to characterize the CFP and the parameters of the distribution were obtained by using the "enhanced Beta distribution approach."

Summarized in Figure 10-8 are the two CFP models, the base case corresponding to the NUREG-1829 Expert Elicitation and the alternative model in which the A and B parameters of the Beta distributions were obtained using PMF results. As indicated, based on the PFM results for the HPI nozzle, the mean CFP = 1.14E-04 versus 6.32E-04 at an equivalent break size of 3-inch (equal to a DEGB of the nozzle.

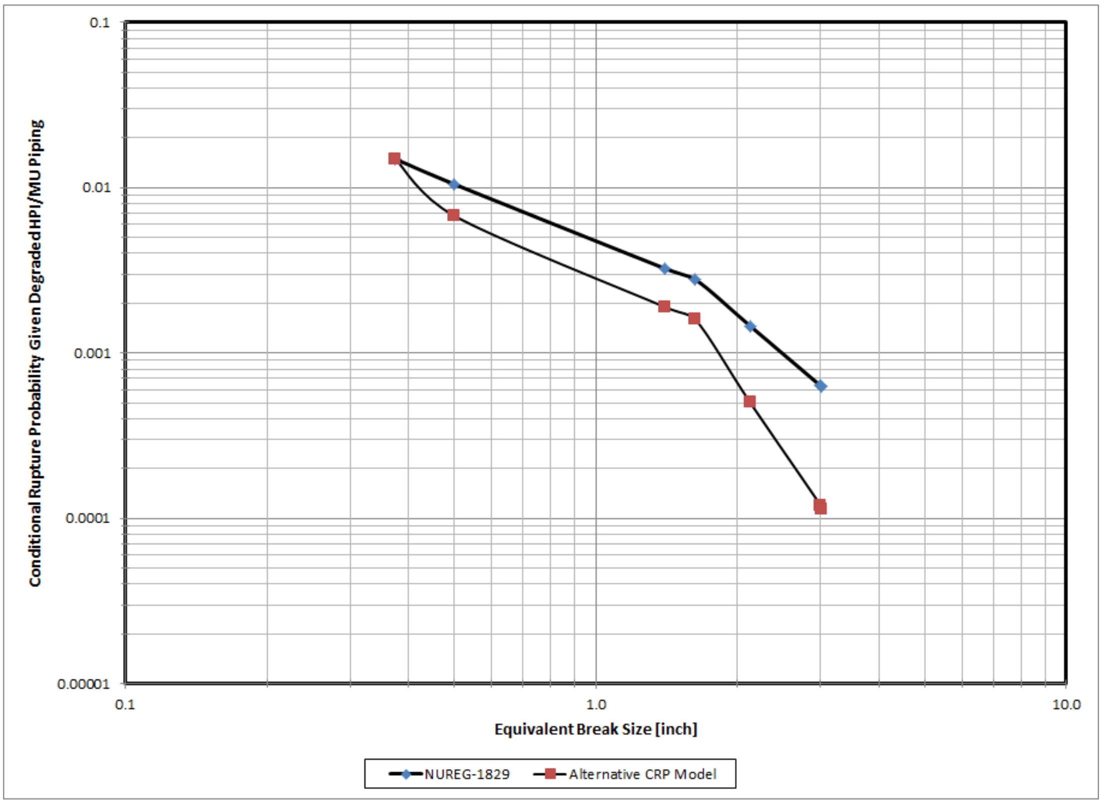

Figure 10-8: CFP Models Based on Expert Elicitation and PFM

10.3.4 Common Cause Initiating Events

The SDP process addresses the potential for common cause events in the risk characterization of events. However, no guidance exists for events involving passive component degradation or failure. The traditional common cause failure (CCF) methodologies have been developed

104 https://www.structint.com/competency/software/beyond-praise

for active components. Extending a risk characterization of passive component failure to also account for common cause events is a complex undertaking requiring modeling of specific event sequences coupled with some unique input data requirements. This is elaborated using the symbolic relationship in Figure 10-9.

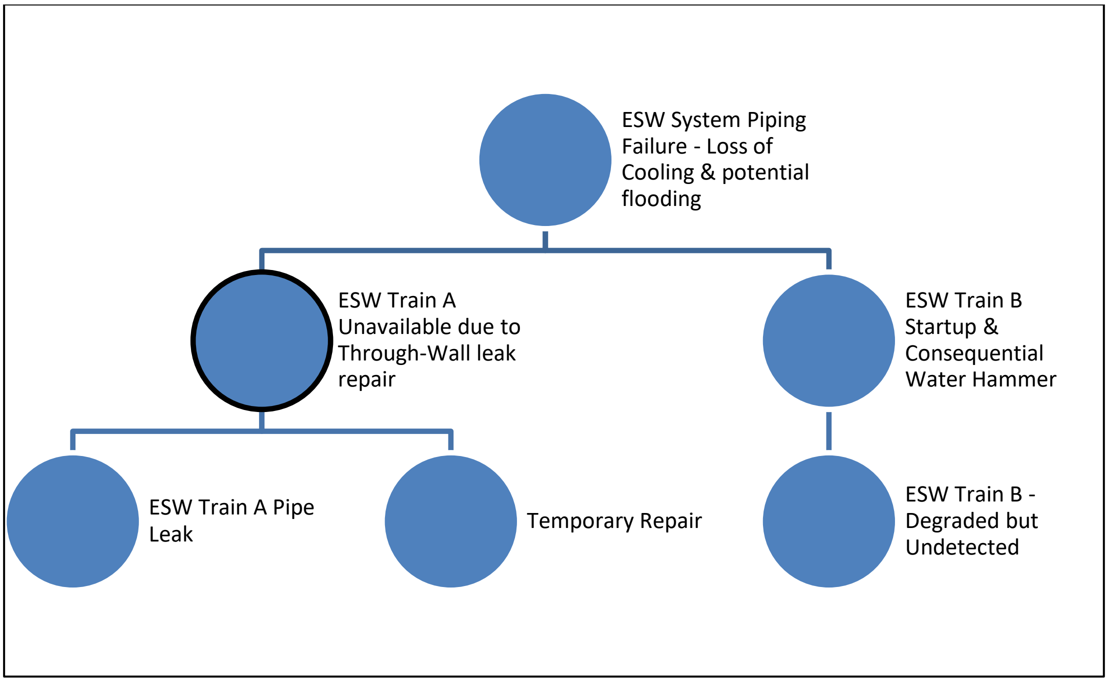

Figure 10-9: Loss of Essential Service Water Cooling due to Material Degradation

Figure 10-9 symbolizes a situation in which a pinhole leak is discovered on Train A Essential Service Water pump discharge elbow. It is assumed that the material degradation is attributed to a combination of microbiologically influenced corrosion and turbulent flow. Following the discovery of the pinhole leak Train A is declared inoperable and Train B is placed in operation. Next, the following scenario is postulated. Because of the ESW piping layout the system is prone to water hammer transients every time the standby pump is placed in operation. Furthermore, Train B is susceptible to the same localized material degradation as Train B. Prior to the discovery of the pinhole leak no non-destructive examinations had been performed on either pump train.

To correctly perform a risk characterization of this hypothetical situation the analyst would have evaluate the conditional failure probability of a failure of Train B conditional on an hydraulic transient to fail Train B piping while Train A is unavailable due to repair.

11. CONCLUSIONS & RECOMMENDATIONS

The field experience with safety- and non-safety related piping in commercial nuclear power reactors is quite extensive. Equally extensive is the experience gained from the implementation of different degradation mechanism (DM) mitigation strategies. By applying advanced piping reliability models, this body of field experience data, engineering data and integrity management insights can be used to assess the projected structural integrity of revised or new piping system designs.

Against the background of an overview of "Regulated Industry Practices with Risk-Informed Operability Determination," this report documents a technical basis for performing risk characterization of degraded passive components. Different methodologies are available to address the oftentimes complex relationships between material degradation models, material properties and reliability and integrity management techniques versus pressure boundary integrity. However, there is no single, all-encompassing methodology for solving plant-specific reliability problems.

This Technical Manual summarizes the different approaches to FFS and operability determination (OD), and it elaborates on the different ways by which operating experience data can be used to validate FFS and OD results and to support the associated decision-making processes. The manual recognizes the strengths and limitations of the different analytical approaches, including that of PFM in calculating the likelihood of a pressure boundary failure, as well as the strengths and limitations of data-driven models of passive component reliability. This manual includes examples of how to use PFM and data-driven models in a synergistic context. The traditional and perhaps widely held notion that it is practically impossible to validate FFS results by way of data-driven models needs re-assessing, and as stated by Cronvall [189]:

> "*Databases are one possible solution to the problems related to the input data needed in the risk and ageing analyses. The amount of NPP piping component degradation data in the international databases has increased and quality improved, respectively, which consequently allows improving the accuracy in the statistical piping degradation estimates. An example of such a database is CODAP which is an advanced good quality piping failure database containing data from 12 countries using nuclear energy. On the other hand, the inclusion of plant specific features and characteristics in the degradation analyses necessitates the availability of plant specific databases. The development and better availability of component degradation/failure databases will hopefully provide means for how the problems related to small failure probabilities, i.e. to very rare events, could be alleviated or even overcom*e."

Under the auspices of the International Atomic Energy Agency (IAEA)[105], efforts are underway to develop a "good practices" framework for estimating the probabilistic failure metrics for piping components in nuclear power plants. This framework will evolve from the insights and results of a series of benchmark problems that have been solved using different algorithms, computational tools and data. When these results become available it is recommended that this Technical Manual be updated as needed.

[105]IAEA Coordinated Research Project (CRP), titled "Methodology for Assessing Pipe Failure Rates in Advanced Water-Cooled Reactors" (2018-2022). The main objective of this CRP is to develop a new "best practices" methodology among IAEA member states for predicting pipe failure rates in advanced WCR designs through the analysis of data, best practices, and methodologies used in the existing reactors. Further goals of the CRP include the completion of a set of benchmark exercises, with documentation of lessons learned, as well as the creation of educational and training materials based on this newly developed methodology for assessing pipe failure rates.

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APPENDIX A

GLOSSARY OF TECHNICAL TERMS

Austenitic Alloy Steel. Also high-alloy steels with the main alloying elements being chromium (Cr) and nickel (Ni). Some high-alloy steels include niobium (Nb) to improve welding properties, or titanium (Ti) to prevent intergranular corrosion and weld decay.

Code Repair. As an example, the definitions of and requirement for a "Code Repair" are defined in ASME Section XI, Article IWA-4000, "Repair/Replacement Activities."

Component Boundary. Defines the physical boundary of a component required for system operation. A component boundary definition should be consistent with the parameter database supporting PRA model quantification. For piping components, the component boundary is established through degradation mechanism evaluations (see below).

Corrosion Fatigue. The behavior of materials under cyclic loading conditions is commonly considered as consisting of two broad categories of material properties. One category relates to cyclic life for the formation of a fatigue crack in a smooth test specimen, the so-called S-N fatigue properties. The second relates to the growth of a pre-existing crack. Laboratory test have shown that LWR coolant water can have a detrimental effect on both S-N fatigue properties and fatigue crack growth.

Corrosion-Under-Insulation (CUI). A severe form of localized external pipe wall corrosion that occurs in carbon and low alloy steel piping that has been insulated. This form of corrosion occurs when water is absorbed by or collected in the insulation. The piping begins to corrode as it is exposed to water and oxygen

Damage Mechanism. Excessive internal or external loading conditions that cause physical damage to a component pressure boundary. Pressure shocks from a water hammer might damage pipe hangers and snubbers, or distort a piping section.

DN (Nominal Diameter). The nominal diameter is used to define a pipe by the outside diameter neglecting the tolerance hand. It is based on the German industry standards and used in some other European countries.

Defect. A flaw of such size, shape, orientation, location or properties as to be unacceptable for continued service (i.e. exceeds the acceptance criteria of the American Society of Mechanical Engineers (ASME) Section XI Code, the applicable construction code or an NRC approved ASME Code Case).

Degradation Mechanism. Phenomena or processes that attack (wear, erode, crack, etc.) the pressure-retaining material over time and might result in a reduction of component pressure boundary integrity. It should be noted that damage mechanisms and degradation mechanisms could interact to cause major, catastrophic passive component failures.

Degradation Mechanism Evaluation. The identification of degradation mechanisms in a pipe segment by comparing actual piping design and operating conditions to a well-defined set of material and environmental attributes. The evaluation considers plant-specific service experience involving cracking and leakage.

Dent. A dent in a pipeline is a permanent plastic deformation of the circular cross section of the pipe. A dent is a gross distortion of the pipe cross-section. Dent depth is defined as the maximum reduction in the diameter of the pipe compared to the original diameter (i.e. the nominal diameter less the minimum diameter).

Discontinuity. A lack of continuity or cohesion; an interruption in the normal physical structure of material or a product.

Erosion Cavitation (E-C). This phenomenon occurs downstream of a directional change or in the presence of an eddy. Evidence can be seen by round pits in the base metal and is often misdiagnosed as FAC (see below). Like erosion, E-C involves fluids accelerating over the surface of a material; however, unlike erosion, the actual fluid is not doing the damage. Rather, cavitation results from small bubbles in a liquid striking a surface. Such bubbles form when the pressure of a fluid drops below the vapor pressure, the pressure at which a liquid becomes a gas. When these bubbles strike the surface, they collapse, or implode. Although a single bubble imploding does not carry much force, over time, the small damage caused by each bubble accumulates. The repeated impact of these implosions results in the formation of pits. Also, like erosion, the presence of chemical corrosion enhances the damage and rate of material removal. E-C has been observed in PWR decay heat removal and charging systems.

Erosion/Corrosion (E/C): "Erosion" is the destruction of metals by the abrasive action of moving fluids, usually accelerated by the presence of solid particles or matter in suspension. When corrosion occurs simultaneously, the term erosion-corrosion is used. In the CODAP database the term "erosion/corrosion" applies only to moderate energy carbon steel piping (e.g., raw water piping).

Failure Assessment Diagram (FAD) Method. To determine if a crack may cause a structural failure, the failure assessment diagram (FAD) method uses two ratios: brittle fracture and plastic collapse. The FAD method is described in the engineering best practice code API 579/ASME FFS-1 (API 2007). "Failure" is assumed if the assessment point falls on or outside the FAD curve while safe conditions are assumed if the assessment point falls inside the FAD-curve

Ferritic Alloy Steel. Also low-alloy steels, which have a carbon content less than 0.2% and contain a total of < 12% alloying elements (e.g., Cr, MN, Mo, Ni).

FITNET. FITNET was a 4-year (2002 to 2006) European thematic network with the objective of developing and extending the use of fitness-for-service procedures throughout Europe. It was part-funded by the "Competitive and Sustainable Growth Program" and formed part of the European Union's Framework 5 research program. Additional funding came from the participants in the form of in-kind contributions, including information from related EU- and European Coal and Steel Community (ECSC) funded projects.

Flaw. An imperfection or unintentional discontinuity that is detectable by nondestructive examination (NDE).

Flaw Aspect Ratio. Ratio of length of deepest crack to depth of deepest crack.

Flow Accelerated (or Assisted) Corrosion (FAC). EPRI defines FAC as "a process whereby the normally protective oxide layer on carbon or low-alloy steel dissolves into a stream of flowing water or water-steam mixture." It can occur in single phase and in two phase regions. According to EPRI, the cause of FAC is a specific set of water chemistry conditions (e.g., pH, level of dissolved oxygen), and absent a mechanical contribution to the dissolution of the normally protective iron oxide (magnetite) layer on the inside pipe wall.

General Corrosion. An approximately uniform wastage of a surface of a component, through chemical or electrochemical action, free of deep pits or cracks.

Gouge. A gouge is created when metal is removed from the pipe wall through mechanical means. This usually happens when the teeth of a back hoe scrape across the pipe. The sharp edges of the gouge act as stress concentrators and pose a threat to the integrity of the pipe.

Hanger. An item that carries the weight of components or piping from above with the supporting members being mainly in tension.

Hydrogen-Induced Cracking (HIC). Stepwise internal cracks that connect adjacent hydrogen blisters on different planes in the metal, or to the metal surface. An externally applied stress is not needed for the formation of HIC. In steels, the development of internal cracks (sometimes referred to as blister cracks) tends to link with other cracks by a transgranular plastic shear mechanism because of internal pressure resulting from the accumulation of hydrogen. The link-up of these cracks on different planes in steels has been referred to as stepwise cracking to characterize the nature of the crack appearance. HIC is commonly found in steels with: (a) high impurity levels that have a high density of large planar inclusions, and/or (b) regions of anomalous microstructure produced by segregation of impurity and alloying elements in the steel.

High Energy Piping. A piping system for which the maximum operating temperature exceeds 200 ºF (94.33 ºC) or the maximum operating pressure exceeds 275 psig (1.896 MPa).

INES. The International Nuclear Event Scale (INES) is a tool for promptly and consistently communicating to the public the safety significance of events associated with sources of ionizing radiation. The INES scale applies to any event associated with the use, storage and transport of radioactive material and radiation sources, whether or not the event occurs at a facility; this includes events involving the loss or theft of radioactive sources or packages and the discovery of orphan sources, such as sources being discovered in scrap metal. Events are rated at seven levels: Levels 1–3 are “incidents” and Levels 4–7 “accidents”.

Inservice Inspection (ISI). An inspection performed after pre-service inspections and test runs are satisfactorily completed and the system or component has been certified or accepted for normal service operation. The objective of such inspections is to detect degradation that might have occurred during plant operation.

Intergranular Stress Corrosion Cracking (IGSCC). IGSCC is associated in particular with a sensitized material (e.g., sensitized austenitic stainless steels are susceptible to IGSCC in an oxidizing environment). Sensitization of unstabilized austenitic stainless steels is characterized by a precipitation of a network of chromium carbides with depletion of chromium at the grain boundaries, making these boundaries vulnerable to corrosive attack.

Irradiation Assisted Stress Corrosion Cracking (IASCC). IASCC refers to intergranular cracking of materials exposed to ionizing radiation. As with SCC, IASCC requires stress, aggressive environment and a susceptible material. However, in the case of IASCC, a normally non-susceptible material is rendered susceptible by exposure to neutron irradiation. IASCC is a plausible ageing mechanism, in particular for PWR internal components (e.g., baffle bolts).

Lamination. Laminations are planes within a steel plate across which there is no metallic bond. They are typically a result of nonmetallic inclusions and gas pockets formed in the ingot when it has been cast and as it solidifies. These most often occur in the top of the ingot in the "pipe end" formed as the ingot solidifies. They may include oxide coating of the bubbles, slag inclusions, refractory inclusions from erosion of the furnace, its spout, the ladle and its spout or orifice, and the passing through of slag on the surface of the molten steel in the ladle.

Larson-Miller Parameter (LMP). The LMP is a means of predicting the lifetime of material vs. time and temperature using a correlative approach based on the Arrhenius rate equation. The value of the parameter is usually expressed as $LMP=T(C+\log t)$ where C is a material specific constant often approximated as 20, t is the time in hours and T is the temperature in Kelvin. According to the LMP, at a given stress level the log time to stress rupture plus a constant of the order of 20 multiplied by the temperature in kelvins or degrees Rankine remains constant for a given material.

Microsoft® Office Excel Workbook. An Excel file that contains one or more worksheets that organizes various kinds of related information.

Moderate Energy Piping. A piping system for which the maximum operating temperature is less than 200 ºF (94.33 ºC) or the maximum operating pressure is less than 275 psig (1.896 MPa).

Nondestructive Examination. An examination by the visual, surface, or volumetric method.

Operable and Operability. According to the Standard Technical Specifications (NUREG-1430 through NUREG-1434), a SSC shall be operable or have operability when it is capable of performing its specified function(s), and when all attendant instrumentation, controls, electrical power, cooling and seal water, lubrication and other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

Pattern Recognition. Pattern recognition is applied to the interpretation of event reports with scarce (or unclear) details on failure location and root cause. In the context of event data analysis, 'pattern recognition' is a structured process of determining the cause of degradation using known failure patterns for similar piping systems. Data analysis and classification builds on the retrieval of data on similar events, and performing a comparative analysis to determine the nature of apparent similarities between industry data and the specific event.

Pipe Schedule Designation. The schedule number (SN) is defined as SN = 1000 × (P/SE), where P is operating pressure in lb/in^2 and SE is allowable stress range multiplied by joint efficiency in lb/in^2. Most U.S. pipe failure reports include pipe schedule information.

*Pressure Boundary Failur*e. Piping component failure involving leakage or rupture that results in reduction or loss of the component pressure-retaining capability. Also, a through-wall flaw, which results from degradation or damage.

Primary Water Stress Corrosion Cracking (PWSCC). PWSCC is a form of IGSCC and is defined as intergranular cracking in primary water within specification limits (i.e., no need for additional aggressive species – for example, IGSCC of Alloy 600 in primary water).

Repair. The process of restoring a nonconforming item by welding, brazing, or metal removal such that existing design requirements are met.

Restricted. In the context of distribution of the OPDE database or OPDE database record, the term "restricted" means that once a transmittal is made to OPDE National Coordinators, it is their responsibility to decide on its further distribution for official use within their country.

Screening of Data. Against a clear component boundary definition, the screening is performed to ensure the relevancy of database entries. It is a check for reasonableness so that any suspicious data are either rejected or subjected to further evaluation once additional information has become available.

Snubber. A mechanical snubber is a mechanical device designed to protect components from excess shock or sway caused by seismic disturbances or other transient forces. During normal operating conditions, the snubber allows for movement in tension and compression. When an impulse event occurs, the snubber becomes activated and acts as a restraint device. The device becomes rigid, absorbs the dynamic energy, and transfers it to the supporting structure.

Strain Induced Corrosion Cracking (SICC). SICC is used to refer to those corrosion situations in which the presence of localized dynamic straining is essential for crack formation (i.e., initiation and propagation) to occur, but in which cyclic loading is either absent or restricted to a very low number of infrequent events. SICC has been observed in particular in pressurized components in German nuclear power plants made of higher-strength carbon steel and low-alloy steel.

Stress Corrosion Cracking (SCC). SCC is a localized non-ductile failure which occurs only under the combination of three factors: 1) tensile stress, 2) aggressive environment, and 3) susceptible material. The SCC failure mode can be intergranular (IGSCC), or transgranular (TGSCC). In a nuclear power plant operating environment, primary water SCC (PWSCC), and irradiation assisted SCC (IASCC) are also defined.

*Stress-Oriented Hydrogen-Induced Cracking (SOHIC)***.** Arrays of cracks that are aligned nearly perpendicular to the applied stress, which is formed by the link-up of small HIC cracks in steel. Tensile stress (residual or applied) is required to produce SOHIC. SOHIC is commonly observed in the base metal adjacent to the heat-affected zone (HAZ) of a weld, oriented in the through-thickness direction. SOHIC may also be produced in susceptible steels at other high stress points such as from the tip of mechanical cracks and defects, or from the interaction between HIC on different planes in the steel.

Sweepolet (Weldolet). Tradename for a contoured, integrally reinforced, butt-welded branch connection.

Thermal Stratification. Hot water can flow above cold water in horizontal runs of piping when the flow (hot water into a cold pipe or cold water into a hot pipe) does not have enough velocity to flush the fluid in the pipe. The temperature profiles in the pipe where the top of the pipe is hotter than the bottom causes the pipe to bow along with the normal expansion at the average temperature.

Transgranular Stress Corrosion Cracking (TGSCC). TGSCC is caused by aggressive chemical species especially if coupled with oxygen and combined with high stresses.

Unified Numbering System (UNS). A worldwide universal system of material identification. In this system a letter is followed by a five-digit number which, taken together, uniquely defines each particular material composition.

Water Hammer. If the velocity of water or other liquid flowing in a pipe is suddenly reduced, a pressure wave results, which travels up and down the pipe system at the speed of sound in the liquid. Water hammer occurs in systems that are subject to rapid changes in fluid flow rate, including systems with rapidly actuated valves, fast-starting pumps, and check valves.

APPENDIX B

SELECTED CARBON STEEL DESIGNATIONS

Table B-1: Selected Carbon Steel Designations[106]

National Standard				Chemical Composition [Weight %] - Max				
ASTM	**CZ**	**SS**	**EN Grade**[107]	**C**	**Mn**	**P**	**Cr**	**Mo**
105				0.35	1.05	0.040	--	--
106 Gr. A		1233-06	P235GH-TR1/2 (St 35.8)	0.17			--	--
106 Gr. B		1435-05	P265GH-TR1/2 (St 45.8)	0.30	1.06	0.048	0.30	0.12
106 Gr. C				0.35	1.20	0.050	0.40	0.15
53 Gr. A		1233-05	P235TR1/2 (St. 34.2)				--	--
53 Gr. B		1434-05	P265TR1/2 (St. 37.2)	0.30	1.20	0.05	--	--
	12022.1			0.20	0.60	0.04	0.25	--
334 WP22				0.15	0.60	0.04	2.60	1.13
335 P12			13 CrMo 44	0.15	0.61	0.045	1.25	0.65
335 P22			10 CrMo 9 10	0.15	0.60	0.030	2.60	1.13
B179 170.1			17 MnMoV 6 4 (WB 35)	0.21	1.80	0.035	--	0.55
			15 NiCuMoNb 5 S 1 (WB 36)	0.17		0.016	--	0.40

- ASTM A 105 mainly used for forged fittings (elbows, flanges)
- ASTM A 106 is for high-temperature service (e.g., feedwater and steam piping)
- ASTM A 333 is for low temperature service
- ASTM A 335 is for high-temperature service (more resistant to FAC than A 106 steel)

[106] Reproduced from the CODAP 5th Topical Report (draft report, 18-October-2017)
[107] Designation per DIN Standard in parenthesis.

APPENDIX C

A WHITE PAPER[108] ON THE CONDITIONAL PIPE FAILURE PROBABILITY CONCEPT

[108] "A White Paper serves an important educational function and uses plain language to explain complex issues in a straightforward manner. Remember that White Papers should be objective, provide adequate and appropriate detail, and be written in a clear, concise, and logical way." *Anon.*

Appendix C is adapted from a white paper originally prepared in 2011 by K. Fleming & B. Lydell in support of the South Texas Project Risk-Informed Resolution of GSI-191 and presented to the Office of Nuclear Regulation of the U.S. Nuclear Regulatory Commission. For additional background, see Nuclear Engineering and Design, **305**:433-450 (2016) as well as https://adams.nrc.gov/wba/ (Accession No. ML111890408, ML111890376 and ML111890380).

C.1 INTRODUCTION

Central to any piping reliability analysis is the technical basis for the model used to express the conditional failure probability (CFP). The main body of this report addresses certain aspects of a practical approach to calculating CFPs using Bayesian theory in which the prior CFP uncertainty distribution is expressed by a Beta distribution. It is one of many technical approaches, however. The purpose of this white paper is to present additional perspectives on how to justify a selection of CFP prior distribution parameters.

C.2 EXPERT ELICITATION

The expert elicitation that was performed and documented in NUREG-1829[109] provides estimates of the frequencies for loss-of-coolant-accidents (LOCAs) based on a set of LOCA categories selected to span the break sizes and leak rates that are normally modeled in BWR and PWR probabilistic safety assessments (PSAs). The expert elicitation that is documented in NUREG-1829 included a request for estimates of loss-of-coolant-accident (LOCA) frequencies for specific piping system locations.

The expert elicitation process consolidated operating experience and insights from probabilistic fracture mechanics studies with knowledge of plant design, operation, and material performance. The elicitation required each member of an expert panel to qualitatively and quantitatively assess important LOCA contributing factors and quantify their uncertainty. The quantitative responses were combined to develop BWR and PWR total "unconditional"[110] LOCA frequency estimates for each contributing panelist. The distributions for six LOCA size categories (Table C-1) and three time periods evaluated are represented by four parameters (mean, median, 5th and 95th percentiles). Finally, the individual estimates were aggregated to obtain group estimates; Table C-2.

Table C-1: The NUREG-1829 LOCA Categories

LOCA Category	Representative Light Water Reactor PSA Model LOCA Categories	Effective Break Size [Inch]	Through-Wall Flow Rate [gpm]
1	Small LOCA	≥ 0.5	≥ 100
2	Medium LOCA	≥ 1.5	≥ 1,500
3	Large LOCA	≥ 3	≥ 5,000
4		≥ 6.75	≥ 25,000
5		≥ 14	≥ 100,000
6		≥ 31.5	≥ 500,000

Table C-2: PWR LOCA Frequencies According to NUREG-1829

LOCA Size [gpm]	Eff. Break Size [inch]	Current Day (2008) Estimate [1/ROY] (25 Yr. Fleet Average Operation)			
		5%	Median	Mean	95%
>100	½	7.28E-04	3.7E-03	6.22E-03	1.99E-02
>1,500	1 5/8	6.95E-06	9.9E-05	2.32E-04	8.51E-04
>5,000	3	1.59E-07	4.9E-06	1.60E-05	6.20E-05
>25K	7	1.05E-08	6.3E-07	2.27E-06	8.85E-06
>100K	14	5.72E-10	7.5E-09	3.91E-08	1.50E-07
>500K	31	4.18E-11	1.4E-09	2.32E-08	6.98E-08

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109 Tregoning, R., Abramson, L. and Scott, P., Estimating Loss-of-Coolant-Accident (LOCA) Frequencies Through the Elicitation Process, NUREG-1829, U.S. Nuclear Regulatory Commission, Washington, DC, 2008.

110 In this paragraph "unconditional" means that the frequencies as presented Table C-2 account for contributions from multiple systems, multiple piping locations, different degradation mechanisms, full range of loading conditions, and all relevant operating experience.

C.3 USE OF NUREG-1829 DATA TO OBTAIN CFP PRIOR DISTRIBUTION PARAMETERS

The approach to using information in NUREG-1829 to develop estimates of the conditional failure probability is based on the basic DDM representation of piping reliability given by:

$$F(LOCA_j) = \sum_{l} m_l \lambda_l P(R_j|F) \qquad \text{(C-1)}$$

Where:

$F(LOCA_j) =$	Unconditional frequency of LOCA Category *j* due to pipe failures in selected component, per reactor calendar-year
$m_l =$	Number of pipe welds of type *i* in selected component having the same failure rate
$\lambda_l =$	Failure rate per weld-year for pipe weld type *i* within the selected component
$P(R_j\|F) =$	Conditional failure probability in LOCA Category *j* given failure in selected component

Using Equation (C-1), the LOCA frequencies in Table C-2 together with the pipe failure rate λ_i estimated from operating experience it is in principle straightforward to obtain a corresponding CFP value; this would be a "reverse-engineering approach" to solving the problem of obtaining CFP parameters that are "anchored" to a public domain peer reviewed reference. In reality, the solution to the problem is somewhat more complex.

Each term in the DDM representation is subject to epistemic uncertainty. Therefore, this model (Equation C-1) and the Base Case Analysis[111] of the failure rates (Appendix D of NUREG-1829) are used to derive epistemic uncertainties for the CFPs in each LOCA category. This produces a set of target LOCA frequency distribution parameters that have been selected to incorporate the epistemic uncertainties developed in NUREG-1829.

This approach makes use of there being a technical basis for the failure rate estimates from service data and a well-reviewed and extensively applied Bayes' uncertainty analysis method. These estimates were part of the information that was available to each NUREG-1829 expert panel member to anchor her/his inputs. Since there have been no Category 1, 2, 3, 4, 5, or 6 LOCAs, the expert elicitation results constitute an extrapolation from the existing service data. Therefore, this technical approach simply assumes that the variability in the expert elicitation inputs for LOCA frequency represents the epistemic uncertainty in the LOCA frequency for each component. This epistemic uncertainty is then assumed to result from the combination of the epistemic uncertainty in the failure rate and the epistemic uncertainty in the conditional probability of each LOCA category.

C.4 USE OF DATA FROM NUREG-1829 EXPERT ELICITATION

The expert elicitation that was performed for NUREG-1829 included a request for estimates of LOCA frequencies for specific piping component.[112] Nine experts provided input; one set of numbers provided by the experts was LOCA frequencies by LOCA category in terms of a mid-value (MV), an upper bound (UB), and a lower bound (LB), with the understanding that

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[111] Base case frequencies were developed by a subset of the expert panel. These frequencies were then provided to the other panelists as possible anchoring frequencies for use during their elicitations. Alternatively, panelists were free to develop a different approach as the basis for their elicitation responses. Base case frequencies were developed for five piping systems, two BWR systems and three PWR systems.

[112] The expert panel input data sets are available from https://adams.nrc.gov/wba/ (Accession No. ML080560005).

those would be interpreted as medians, 95%-tiles, and 5%-tiles of a lognormal uncertainty distribution. For symmetric inputs (i.e., when UB/MV = MV/LB), which were provided in most cases, these distributions were assumed to be lognormal distributions. For asymmetric inputs provided by the experts, a specific split lognormal distribution[113] was assumed.

The first set of LOCA frequencies was for the existing U.S. fleet of plants, which involves a mixture of plant ages and an average plant age of about 25 years at the time the elicitation was performed (2003 to 2005). The experts provided multipliers for normalizing these LOCA frequencies to plant ages of 25 years, 40 years, and 60 years prior to the occurrence of a LOCA. These multipliers enabled the experts to express whether LOCA frequencies could be affected by aging effects and whether such effects might be mitigated. Only the 40-year values are used in this white paper. The expert elicitation inputs for the Reactor Coolant System Hot Leg piping (Figure C-1) are provided in Table C-3.

C.5 DEVELOPMENT OF 40-YEAR LOCA FREQUENCY DISTRIBUTIONS

To make the NUREG-1829 estimates representative of plants in long-term operation, the “LOCA Frequencies for System” distributions are multiplied by the 40-year multiplier distributions, to obtain the 40-year LOCA frequency distributions. This is straightforward because the product of two lognormal distributions is also a lognormal distribution. When the two input distributions are lognormal, the parameters of the lognormal distribution for the 40-year LOCA frequencies can be directly computed using the following formulas.

$$median_{40YLF} = median_{Base} * median_{40YM} \tag{C-2}$$

$$RF_{40YLF} = e^{1.645\sigma_{40YLF}} \tag{C-3}$$

Where:

$$\sigma_{40YLF} = \sqrt{\left(\frac{\ln(RF_{Base})}{1.645}\right)^2 + \left(\frac{\ln(RF_{40YM})}{1.645}\right)^2} \tag{C-4}$$

$median_{40YLF}$ =	Median of the lognormal distribution for the 40-year LOCA frequency, evaluated for each combination of expert and LOCA Category
$median_{Base}$ =	Median of the lognormal distribution for the base LOCA frequency (“LOCA Frequency for System” provided by each expert for each LOCA Category)
$median_{40YM}$ =	Median of the lognormal distribution for the 40-year multiplier provided by each expert for each LOCA Category
RF_{40YLF} =	Range factor of the lognormal distribution for the 40-year LOCA frequency; RF = SQRT(95%-tile/5%-tile)
σ_{40YLF} =	Logarithmic standard deviation for the lognormal distribution for the 40-year LOCA frequency, evaluated for each combination of expert and

[113] A split lognormal distribution is a combination of two halves of separate lognormal distributions. Because a lognormal distribution can be determined by its median and one other percentile, two lognormal distributions can be determined from the response to each elicitation question; one by the MV and LB and another by the MV and UB. For values less than the median, the split lognormal is the lower half (i.e., the part less than its median of the lognormal distribution determined by the MV and LB). For values greater than the median, the split lognormal is the upper half (i.e., the part greater than its median) of the lognormal distribution determined by the MV and UB. If the UB and LB are symmetric about the MV, then the split lognormal is identical to the lognormal distribution determined by the MV, UB, and LB.

LOCA Category

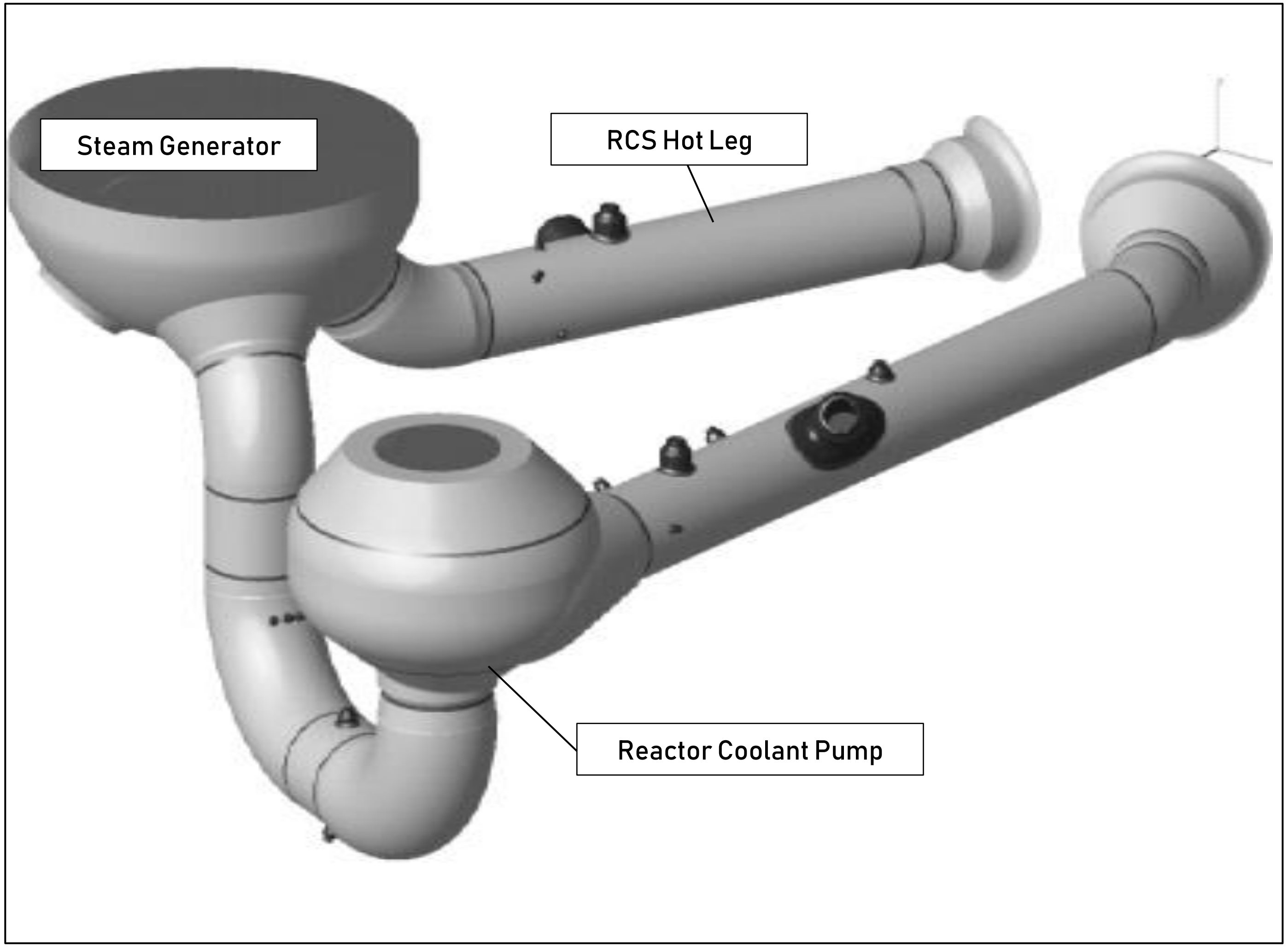

Figure C-1: Typical PWR Reactor Coolant System Hot Leg & Cold Leg[114]

[114] The Cold Leg and Hot Leg branch connections are addressed by separate evaluation boundaries. In this white paper, the RCS Hot Leg piping locations of interest are the dissimilar metal weld locations between the reactor pressure vessel nozzle and Hot Leg piping and between the Hot Leg piping and steam generator inlet nozzle.

Table C-3: Selected NUREG-1829 Expert Distributions for PWR RCS Hot Leg LOCA Frequencies

Expert ID	LOCA Category	LOCA Frequency for System[1] (Per Reactor-Calendar Year)				40-Yr Multiplier[1]				40-Yr LOCA Frequency[1] (Per Reactor-Calendar Year)	
		LB	Mid	UB	RF95=UB/Mid[2]	LB	Mid	UB	RF95=UB/Mid[2]	Mid[3]	RF95[4]
A	1 (> 100)	5.33E-08	1.60E-07	4.80E-07	3.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.60E-07	3.00E+00
	2 (> 1,500)	5.33E-08	1.60E-07	4.80E-07	3.00E+00	1.50E-02	3.00E-01	5.85E-01	1.95E+00	4.80E-08	3.62E+00
	3 (> 5,000)	5.33E-08	1.60E-07	4.80E-07	3.00E+00	5.00E-03	1.00E-01	1.95E-01	1.95E+00	1.60E-08	3.62E+00
	4 (> 25,000)	5.33E-08	1.60E-07	4.80E-07	3.00E+00	1.50E-03	3.00E-02	5.85E-02	1.95E+00	4.80E-09	3.62E+00
	5 (> 100,000)	5.33E-08	1.60E-07	4.80E-07	3.00E+00	5.00E-04	1.00E-02	1.95E-02	1.95E+00	1.60E-09	3.62E+00
	6 (> 500,000)	5.33E-08	1.60E-07	4.80E-07	3.00E+00	1.50E-04	3.00E-03	5.85E-03	1.95E+00	4.80E-10	3.62E+00
B	1 (> 100)	3.00E-07	3.00E-07	3.00E-07	1.00E+00	1.00E-01	1.00E+00	1.00E+01	1.00E+01	3.00E-07	1.00E+01
	2 (> 1,500)	1.20E-07	1.20E-07	1.20E-07	1.00E+00	1.00E-01	1.00E+00	1.00E+01	1.00E+01	1.20E-07	1.00E+01
	3 (> 5,000)	4.80E-08	4.80E-08	4.80E-08	1.00E+00	1.00E-01	1.00E+00	1.00E+01	1.00E+01	4.80E-08	1.00E+01
	4 (> 25,000)	1.92E-08	1.92E-08	1.92E-08	1.00E+00	1.00E-01	1.00E+00	1.00E+01	1.00E+01	1.92E-08	1.00E+01
	5 (> 100,000)	7.68E-09	7.68E-09	7.68E-09	1.00E+00	1.00E-01	1.00E+00	1.00E+01	1.00E+01	7.68E-09	1.00E+01
	6 (> 500,000)	3.07E-09	3.07E-09	3.07E-09	1.00E+00	1.00E-01	1.00E+00	1.00E+01	1.00E+01	3.07E-09	1.00E+01
C	1 (> 100)	6.00E-07	6.00E-07	6.00E-07	1.00E+00	3.00E-02	1.00E+00	3.00E+01	3.00E+01	6.00E-07	3.00E+01
	2 (> 1,500)	5.00E-08	5.00E-08	5.00E-08	1.00E+00	3.00E-02	1.00E+00	3.00E+01	3.00E+01	5.00E-08	3.00E+01
	3 (> 5,000)	2.00E-08	2.00E-08	2.00E-08	1.00E+00	3.00E-02	1.00E+00	3.00E+01	3.00E+01	2.00E-08	3.00E+01
	4 (> 25,000)	3.00E-09	3.00E-09	3.00E-09	1.00E+00	5.00E-02	1.67E+00	1.67E+02	1.00E+02	5.01E-09	1.00E+02
	5 (> 100,000)	1.00E-09	1.00E-09	1.00E-09	1.00E+00	6.00E-02	2.00E+00	2.00E+03	1.00E+03	2.00E-09	1.00E+03
	6 (> 500,000)	2.00E-10	2.00E-10	2.00E-10	1.00E+00	6.00E-02	2.00E+00	2.00E+03	1.00E+03	4.00E-10	1.00E+03
E	1 (> 100)	3.07E-07	9.22E-07	2.77E-06	3.00E+00	3.33E-04	2.83E-02	3.33E-01	1.18E+01	2.61E-08	1.49E+01
	2 (> 1,500)	3.07E-07	9.22E-07	2.77E-06	3.00E+00	3.33E-04	2.83E-02	3.33E-01	1.18E+01	2.61E-08	1.49E+01
	3 (> 5,000)	3.07E-07	9.22E-07	2.77E-06	3.00E+00	3.33E-04	2.83E-02	3.33E-01	1.18E+01	2.61E-08	1.49E+01
	4 (> 25,000)	3.67E-09	1.10E-08	3.30E-08	3.00E+00	1.00E-03	1.00E-01	1.50E+00	1.50E+01	1.10E-09	1.86E+01
	5 (> 100,000)	1.27E-09	3.80E-09	1.14E-08	3.00E+00	1.00E-04	5.00E-02	1.00E+00	2.00E+01	1.90E-10	2.43E+01
	6 (> 500,000)	4.33E-10	1.30E-09	3.90E-09	3.00E+00	1.00E-04	3.00E-02	3.00E+00	1.00E+02	3.90E-11	1.14E+02

Expert ID	LOCA Category	LOCA Frequency for System[1] (Per Reactor-Calendar Year)				40-Yr Multiplier[1]				40-Yr LOCA Frequency[1] (Per Reactor-Calendar Year)	
		LB	Mid	UB	RF95=UB/Mid[2]	LB	Mid	UB	RF95=UB/Mid[2]	Mid[3]	RF95[4]
G	1 (> 100)	5.13E-08	1.54E-07	4.62E-07	3.00E+00	1.00E-01	1.14E+00	1.00E+01	8.77E+00	1.76E-07	1.14E+01
	2 (> 1,500)	7.50E-09	2.25E-08	6.75E-08	3.00E+00	1.00E-01	1.14E+00	1.00E+01	8.77E+00	2.57E-08	1.14E+01
	3 (> 5,000)	2.78E-09	8.33E-09	2.50E-08	3.00E+00	1.00E-01	1.14E+00	1.00E+01	8.77E+00	9.50E-09	1.14E+01
	4 (> 25,000)	9.50E-10	2.85E-09	8.55E-09	3.00E+00	1.00E-01	1.14E+00	1.00E+01	8.77E+00	3.25E-09	1.14E+01
	5 (> 100,000)	1.71E-10	8.53E-10	4.27E-09	5.01E+00	1.00E-01	1.14E+00	1.00E+01	8.77E+00	9.72E-10	1.49E+01
	6 (> 500,000)	1.58E-11	1.58E-10	1.58E-09	1.00E+01	1.00E-01	1.14E+00	1.00E+01	8.77E+00	1.80E-10	2.37E+01
H	1 (> 100)	1.48E-07	4.45E-07	1.34E-06	3.01E+00	2.50E+00	2.50E+01	2.50E+02	1.00E+01	1.11E-05	1.28E+01
	2 (> 1,500)	2.03E-08	6.10E-08	1.83E-07	3.00E+00	1.00E+00	1.00E+01	1.00E+02	1.00E+01	6.10E-07	1.28E+01
	3 (> 5,000)	7.33E-09	2.20E-08	6.60E-08	3.00E+00	5.00E-01	5.00E+00	5.00E+01	1.00E+01	1.10E-07	1.28E+01
	4 (> 25,000)	2.60E-09	7.80E-09	2.34E-08	3.00E+00	5.00E-01	5.00E+00	5.00E+01	1.00E+01	3.90E-08	1.28E+01
	5 (> 100,000)	8.83E-10	2.65E-09	7.95E-09	3.00E+00	5.00E-01	5.00E+00	5.00E+01	1.00E+01	1.33E-08	1.28E+01
	6 (> 500,000)	2.93E-10	8.80E-10	2.64E-09	3.00E+00	5.00E-01	5.00E+00	5.00E+01	1.00E+01	4.40E-09	1.28E+01
I	1 (> 100)	4.00E-11	2.00E-09	1.00E-07	5.00E+01	5.00E-01	5.00E-01	5.00E-01	1.00E+00	1.00E-09	5.00E+01
	2 (> 1,500)	4.00E-11	2.00E-09	1.00E-07	5.00E+01	5.00E-01	5.00E-01	5.00E-01	1.00E+00	1.00E-09	5.00E+01
	3 (> 5,000)	4.00E-11	2.00E-09	1.00E-07	5.00E+01	5.00E-01	5.00E-01	5.00E-01	1.00E+00	1.00E-09	5.00E+01
	4 (> 25,000)	4.00E-11	2.00E-09	1.00E-07	5.00E+01	5.00E-01	5.00E-01	5.00E-01	1.00E+00	1.00E-09	5.00E+01
	5 (> 100,000)	4.00E-11	2.00E-09	1.00E-07	5.00E+01	5.00E-01	5.00E-01	5.00E-01	1.00E+00	1.00E-09	5.00E+01
	6 (> 500,000)	4.00E-11	2.00E-09	1.00E-07	5.00E+01	5.00E-01	5.00E-01	5.00E-01	1.00E+00	1.00E-09	5.00E+01

Notes:
[1] Data shaded in yellow are taken from NUREG-1829 expert questionnaires. Data shaded in blue were calculated per Notes [2] through [4].
[2] RF = Range Factor of a lognormal distribution defined by the Mid value as the median and by the UB value as the 95%tile.
[3] Median of a lognormal distribution for the 40-year LOCA frequency created by the product of two lognormal distributions: the medians of the lognormal distributions for LOCA frequency for system and the 40-year multiplier (see Equation [C-1]).
[4] Range Factor of the 40-year LOCA frequency lognormal distribution (see Equation [C-2]).

$RF_{Base} =$ Range factor of the lognormal distribution for the base LOCA frequency provided by each expert for each LOCA Category

$RF_{40YM} =$ Range factor of the lognormal distribution for the 40-year multiplier provided by each expert for each LOCA Category

In this analysis, the inputs provided by the experts were fit to lognormal distributions by preserving the medians and the 95%-tiles of the input distributions, while ignoring the asymmetries on the left side of the distributions. An alternative procedure was also tested, in which the median and the range factor defined as the square root of the ratio of the 95%-tile to the 5%-tile were preserved in the input distributions, which were again assumed to be lognormal. In an independent review of an early draft of this analysis procedure it was recommended that the former procedure be used. The adopted approach retains the simplicity of using lognormal distributions in lieu of the more complicated split lognormal distributions as used in NUREG-1829[115], retains the identification of the best estimates with the medians of the distributions, and by preserving the 50th and higher percentiles is more effective in preserving the means of the underlying input distributions.

In Table C-3, the first procedure is applied as indicated in the blue-shaded cells. The RF95 values were calculated based on UB/MV for the base LOCA frequencies and the 40-year multipliers, and the RF95 values for the 40-year LOCA frequencies were calculated from Equation (C-4).

As a result of the above procedure, there is a single lognormal distribution defined for each LOCA category frequency at 40 years of operation. This distribution is applicable to each component provided by each of the 9 experts who provided component level inputs in NUREG-1829. In some cases, experts provided fixed values for one parameter (base LOCA frequency or multiplier) and a distribution for the other, in which case the distribution for the 40-year LOCA frequency was found simply by scaling the provided distribution parameters with the supplied fixed values.

C.6 EXPERT COMPOSITE DISTRIBUTIONS FROM NUREG-1829

In this step, the nine expert distributions for 40-year LOCA frequencies are combined into a single composite distribution. NUREG-1829 discussed two approaches for developing expert composite distributions: 1) the Mixture Distribution Method and, 2) the Geometric Mean Method. NUREG-1829 adopted the latter approach, whereas this study evaluated both approaches, briefly described below.

The Mixture Distribution Method

A single mixture distribution was developed for each combination of component and LOCA category by combining the 40-year LOCA frequency distributions provided by each expert. A single mixture distribution was developed by sampling a discrete distribution on each Monte Carlo trial to determine which expert's lognormal distribution for the 40-year LOCA frequency to be sampled for that trial. The discrete distribution has a value for each expert, with each value's being assigned the same probability in order to give all experts equal weight. In the several cases where experts did not provide inputs for each LOCA category, the mixture distribution was developed only for those experts providing inputs for that category. In all cases, a minimum of seven experts provided input, and the vast majority of cases had nine. This method is discussed in NUREG-1829 but was rejected in favor of the Geometric Mean method.

[115] For details, See Section 5.3 in NUREG-1829.

The Geometric Mean Method

When this method was used in NUREG-1829, it was oriented toward the calculation of the total LOCA frequency rather than the LOCA frequency for multiple locations. Another contrast was the use in NUREG-1829 of split lognormals, whereas this study used lognormal fitting based on preserving medians and 95%tiles. A a single lognormal distribution for each component and each LOCA category was defined by taking the geometric mean of the medians of the experts' lognormal distributions as the composite distribution median, and the geometric means of the range factors of the experts' lognormal distributions for the 40-year LOCA frequencies as the composite distribution range factor. The input lognormal distributions provided by the experts were fit to lognormal distribution by matching the 50th and 95th percentiles.

A summary of the derived composite distribution parameters is provided in Table C-4. A comparison of the resulting composite distributions using both methods is provided in Figure C-2. As seen in this figure, the composite distributions generated by the Mixture Distribution method produce much broader ranges of uncertainty than those obtained by the Geometric Mean method. The upper and lower bounds of the mixture distributions are heavily influenced by the experts' extreme high-side and low-side inputs, respectively, whereas the distribution percentiles from the Geometric Mean method more fairly represent the experts' inputs.

Table C-4: Composite Distributions Based on the Geometric Mean Method

LOCA Category	Break Size	Mean	5%tile	50%tile	95%tile
1	≥ 0.5	4.08E-07	9.32E-09	1.21E-07	1.57E-06
2	≥ 1.5	1.28E-07	2.25E-09	3.34E-08	4.95E-07
3	≥ 3	6.51E-08	1.01E-09	1.59E-08	2.52E-07
4	≥ 6.75	2.59E-08	2.49E-10	4.96E-09	9.88E-08
5	≥ 14	1.50E-08	6.70E-11	1.90E-09	5.37E-08
6	≥ 31.5	3.16E-09	4.84E-12	2.18E-10	9.78E-09

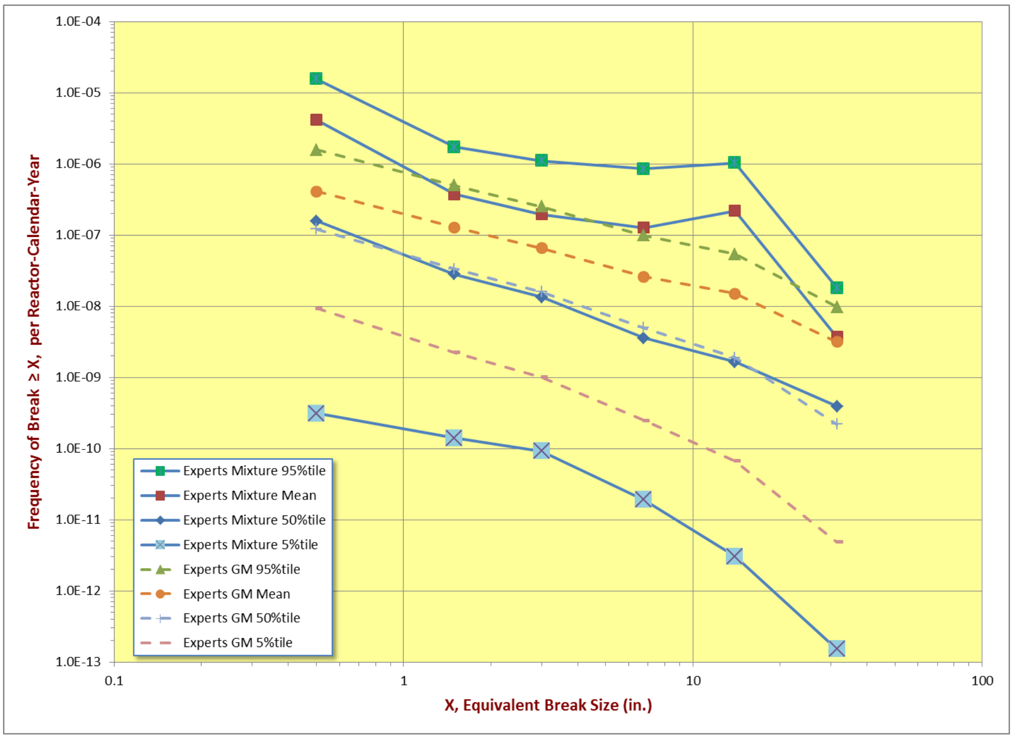

Figure C-2: Comparison of Mixture &Geometric Mean Composite Distributions

C.7 BENCHMARK OF "LYDELL's BASE CASE ANALYSIS"

This step establishes inputs to the selection of target LOCA frequencies from the Lydell Base Case Analysis (Appendix D of NUREG-1829). A secondary purpose is to establish the corresponding failure rate and CFP distributions that produce the Base Case results. The failure rate distribution parameters are used to convert the target LOCA frequency distributions to CFP distributions.

Using the same Microsoft Excel™ and Oracle Crystal Ball™ files that Lydell used to develop his Base Case results, the simplified model of Equation (C-1) was applied to the same failure rate estimates that Lydell derived, assuming a lognormal distribution for the conditional LOCA category probability for each component. This resulted in lognormal parameters that essentially reproduce Lydell's NUREG-1829 Appendix D results, as shown in Figure C-3 for the RCS Hot Leg, respectively. The CFP distribution parameters were obtained by first developing the LOCA frequencies and then calculating the CFP distribution parameters using formulas for calculating the parameters for the product of two lognormal distributions; consistent with Equations (C-2) and (C-3). As illustrated in Figure C-3, the Base Case results from Appendix D in NUREG-1829 and the results obtained using the equivalent lognormal distributions indicate excellent agreement. The underlying lognormal distribution parameters for the conditional LOCA probabilities in Table C-5 are seen as reasonable, i.e. they are neither very large nor very small. The conditional probability of a given break size is indicated to be inversely proportional to pipe size which is in agreement with previous estimates of LOCA frequencies. The uncertainty distribution parameters for the LOCA frequencies from this reconstruction of the Lydell Base Case results are shown in Table C-6.

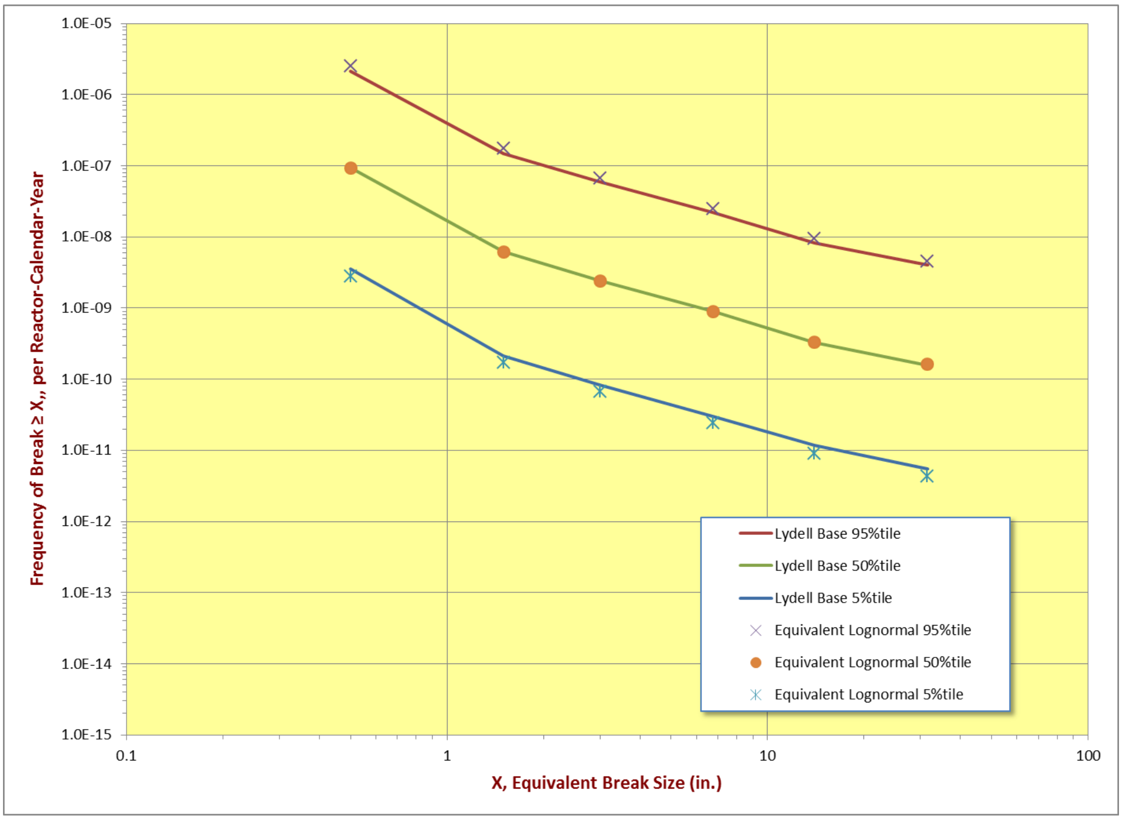

Figure C-5: Benchmarking of Lognormal Distributions to Lydell Base Case Results for the RCS Hot Leg

Table C-5: Lognormal Distributions for Failure Rates and CFPs Matching Lydell's Base Case Results

Component	LOCA Category	Break Size (in.)	Mean	5%-tile	Median	95%-tile
RCS Hot Leg	Failure Rate		3.46E-04	1.01E-05	1.15E-04	1.32E-03
	1	≥ .5	1.67E-03	9.49E-05	7.55E-04	6.01E-03
	2	≥ 1.5	1.18E-04	5.38E-06	4.85E-05	4.37E-04
	3	≥ 3	4.73E-05	2.13E-06	1.93E-05	1.75E-04
	4	≥ 6.75	1.76E-05	7.71E-07	7.09E-06	6.52E-05
	5	≥ 14	6.59E-06	2.97E-07	2.69E-06	2.43E-05
	6	≥ 31.5	3.23E-06	1.38E-07	1.28E-06	1.20E-05

Table C-6: LOCA Frequency Distributions from Benchmarking of Lydell Base Case Results

Component	LOCA Cat.	Break Size (in.)	Lydell Base Case Distribution Parameters Events per Reactor-Calendar Year				
			Mean	5%tile	50%tile	95%tile	RF
RCS Hot Leg	1	≥ 0.5	6.65E-07	3.55E-09	9.39E-08	2.14E-06	24.6
	2	≥ 1.5	4.87E-08	2.10E-10	6.15E-09	1.49E-07	26.6
	3	≥ 3	1.83E-08	8.33E-11	2.42E-09	5.95E-08	26.7
	4	≥ 6.75	6.99E-09	3.03E-11	8.93E-10	2.21E-08	27.0
	5	≥ 14	2.55E-09	1.16E-11	3.29E-10	8.29E-09	26.7
	6	≥ 31.5	1.26E-09	5.44E-12	1.58E-10	4.04E-09	27.3

C.8 SELECT TARGET LOCA FREQUENCIES FROM NUREG-1829 DATA

In selecting the 'target LOCA' frequencies, four options are considered:

1. Option 1: Use only the Lydell Base Case results
2. Option 2: Use only the Experts' Mixture Distribution results
3. Option 3: Use only the Experts' Geometric Mean results
4. Option 4: Use a hybrid of the Experts' Geometric Mean and Lydell Base Case results

Option 1 would not be making full use of the expert elicitation results of NUREG-1829. Option 2 would be making use of the expert elicitation but would produce unreasonably large spreads between the upper and lower percentiles, which would overemphasize the most extreme expert inputs. Option 3 would be preferred over Options 1 and 2 in that it would better represent the diverse inputs of the expert panel and would include the input of Lydell. However, the option selected, Option 4, is a hybrid of Options 1 and 3 and is comprised of a mixture distribution of the LOCA frequencies produced by those options.

Option 4 places equal weight on the Lydell Base Case results and the Expert Geometric Mean results. This option's mixture distribution was developed by Monte Carlo simulation, which involved a binary variable to select either Lydell Base Case results or Expert Geometric Mean results, after which a random sample was obtained from that selected distribution. Option 4 is preferred over Option 3 as it exhibits a larger degree of epistemic uncertainty while providing mean values that are very close to those of Option 3.

These 'target LOCA' frequencies are used in the next step to derive CRPs for LOCAs in each of the LOCA break size categories given a pipe failure; Table C-6. Figure C-4 compares the resulting target LOCA frequencies and those for Option 3, for the RCS Hot Leg. The net effect is to increase the uncertainty with slight reductions in the mean and 95%-tile and larger reductions for the 5%-tiles compared to Option 3 for the RCS Hot Leg.

Table C-6: Selected Mixture Distribution of Geometric Mean and Lydell Base Case for the 'Target LOCA' Frequencies

Component	LOCA Cat.	Break Size (in.)	Target LOCA Frequency Distribution Parameters				
			Mean	5%tile	50%tile	95%tile	RF
RCS Hot Leg	1	≥ 0.5	5.07E-07	5.39E-09	1.05E-07	1.83E-06	18.4
	2	≥ 1.5	8.22E-08	4.29E-10	1.49E-08	3.30E-07	27.7
	3	≥ 3	4.10E-08	1.68E-10	6.47E-09	1.60E-07	30.9
	4	≥ 6.75	1.57E-08	5.65E-11	2.09E-09	6.07E-08	32.8
	5	≥ 14	8.69E-09	2.09E-11	7.64E-10	2.93E-08	37.4
	6	≥ 31.5	2.11E-09	5.01E-12	1.79E-10	6.63E-09	36.4

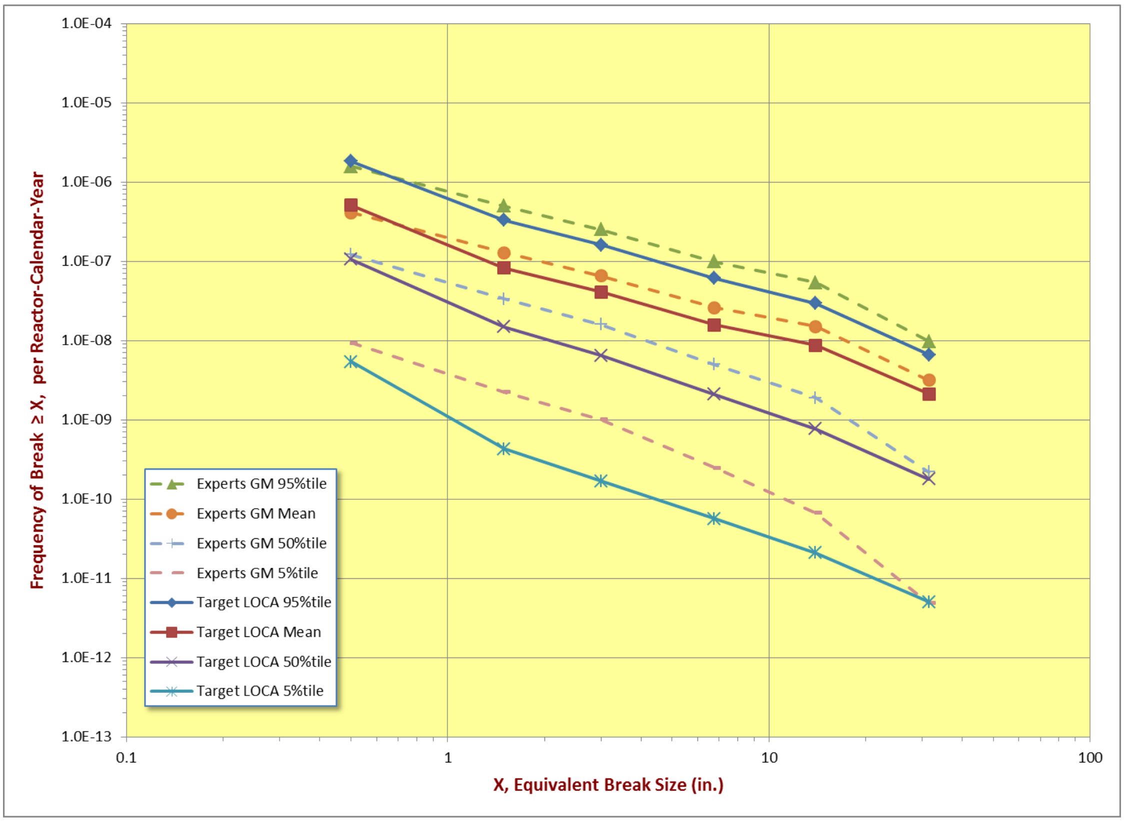

Figure C-6: Comparison of Experts' Geometric Mean and Target LOCA Model for the RCS Hot Leg

C.9 DEVELOP CONDITIONAL FAILURE PROBABILITIES FROM 'TARGET LOCA' FREQUENCIES

This step uses the 'target LOCA' frequencies in Table C-6 and information from the Lydell Base Case results (NUREG-1829, Appendix D) on the underlying failure rates for each component, to derive a CFP model that when linked with the Lydell Base Case failure rate model, will reproduce these target LOCA frequencies. The results for the Base Case failure rates for the piping components associated with the target LOCA frequencies are shown in Table C-7. In order to derive the model for conditional probability of rupture, the Base Case pipe failure rates were fit to lognormal distributions by matching the 5th and 95th percentiles and the range factor calculated from these percentiles.

Since the use of lognormal distributions enables the LOCA frequency to be expressed as the product of a lognormally distributed failure rate and a lognormally distributed CRP, the parameters of the CRP distributions may be calculated directly. Using the same methodology as used in Equations (C-2, (C-3) and (C-4), the following relations are established.

$$median_{CRP_k} = \frac{median_{TLF_k}}{median_{FR}} \tag{C-5}$$

$$RF_{CRP_k} = e^{1.645\sigma_{CRP_k}} \tag{C-6}$$

Where

$$\sigma_{CRP_k} = \sqrt{\left(\frac{\ln(RF_{TLF_k})}{1.645}\right)^2 - \left(\frac{\ln(RF_{FR})}{1.645}\right)^2} \tag{C-7}$$

$median_{CRP_k} =$	Median of the lognormal distribution for the conditional probability of pipe rupture in LOCA Category *k* given pipe failure
$median_{TLF_k} =$	Median of the lognormal distribution for the target LOCA frequency for LOCA Category *k*
$median_{FR} =$	Median of the lognormal distribution for the pipe failure rate
$RF_{CRP_k} =$	Range factor of the lognormal distribution for the conditional probability of pipe rupture in LOCA Category *k* given pipe failure, equal to SQRT(95%-tile/5%-tile) of the lognormal distribution
$\sigma_{CRP_k} =$	Logarithmic standard deviation for the lognormal distribution for the conditional probability of pipe rupture in LOCA Category *k* given pipe failure
$RF_{TLF_k} =$	Range factor of the lognormal distribution for the target LOCA frequency for LOCA Category *k*
$RF_{FR} =$	Range factor of the lognormal distribution for the pipe failure rate

The medians and range factors of the CRP distributions were computed from the medians and range factors of the target LOCA frequency distributions using the above formulas. Using the properties of the lognormal distribution, the remaining parameters of the distributions may be directly calculated; Table C-8.

Table C-7: Parameters of Target LOCA Frequencies

Component	**LOCA Category**	**Break Size (in.)**	**Cumulative LOCA Frequency[1], per Reactor-Calendar-Year**			
			Mean	**5%tile**	**50%tile**	**95%tile**
RCS Hot Leg	1	≥ 0.5	4.45E-07	3.55E-09	7.72E-08	1.68E-06
	2	≥ 1.5	1.95E-07	2.10E-10	1.09E-08	5.68E-07
	3	≥ 3	1.05E-07	8.33E-11	4.89E-09	2.87E-07
	4	≥ 6.75	3.75E-08	3.03E-11	1.77E-09	1.03E-07
	5	≥ 14	2.02E-08	1.16E-11	7.75E-10	5.17E-08
	6	≥ 31.5	2.41E-09	5.44E-12	2.08E-10	7.94E-09
[1] Frequency of LOCA with break size greater than or equal to the indicated value.						

Table C-8: Derived RCS Hot Leg Prior CFP Distribution Parameters[116]

Through-Wall Flow Rate [gpm]	Break Size (in.)	Conditional Failure Probability Distribution Parameters				
		Median	**Mean**	**5th Percentile**	**95th Percentile**	**Range Factor**
>100	≥ 0.5	1.46E-03	1.84E-04	9.10E-04	4.50E-03	4.9
>1,500	≥ 1.5	3.31E-04	1.35E-05	1.29E-04	1.23E-03	9.6
>5,000	≥ 3	1.65E-04	5.01E-06	5.61E-05	6.28E-04	11.2
>25,000	≥ 6.75	5.74E-05	1.49E-06	1.81E-05	2.20E-04	12.2
>100,000	≥ 14	2.49E-05	4.54E-07	6.62E-06	9.65E-05	14.6
>500,000	≥ 31.5	5.84E-06	1.06E-07	1.55E-06	2.26E-05	14.6

C.10 SOME OBSERVATIONS

This white paper documents an approach that details the justifications for apriori CFP distribution parameters. Reflected in the proposed parameters is the state-of-knowledge as of 2008 regarding primary stress corrosion cracking (PWSCC) and its effect on structural integrity. To date only very minor through-wall defects that are attributed PWSCC have been observed. For reference, summarized in Figure C-7 is the full history of significant RCPB leak events – all of the significant piping leak events have been caused by failure mechanisms other than PWSCC.

[116] These distribution parameters apply to piping locations that are susceptible to primary water stress corrosion cracking (PWSCC).

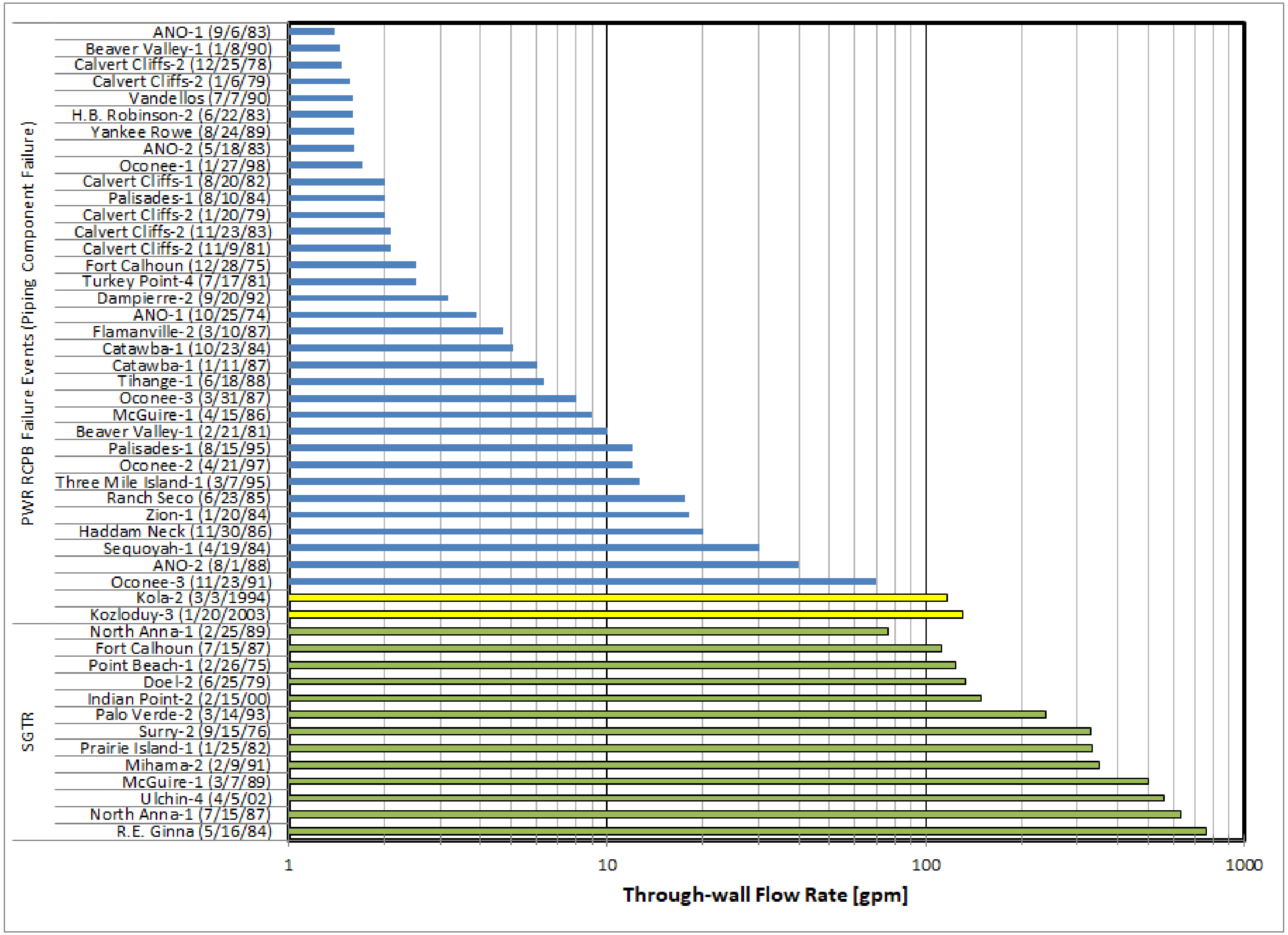

Table C-7: Observed PWR RCPB Through-Wall Flow Rates

BIBLIOGRAPHIC DATA SHEET

Report Title & Subtitle:

A Technical Manual on the Underlying Theory & Associated Methodology for Performing Operability Determinations Through Risk Characterization of Degraded Passive Components

(R645.1: Statistical Modelling of Aging Effects in Failure Rates of Piping Components
Task 7: Final Report)

Author(s)	Type of Report:
B.O.Y. Lydell	Technical
Report Date:	**CNSC Reference No.:**
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Performing Organization(s):	**Sponsoring Organization:**
Sigma-Phase Inc. 16917 S. Orchid Flower Trail Vail, AZ 85641-2701 USA	Operational Engineering Assessment Division Canadian Nuclear Safety Commission 280 Slater Street Ottawa, Ontario, K1P 5S9 CANADA

Abstract:

The objective of CNSC Project No. 87055-15-0214 ("R645.1: Statistical Modelling of Aging Effects in Failure Rates of Piping Components") is to develop a Technical Manual (TM) on the underlying theory and associated methods and techniques for performing operability determinations through risk characterization of carbon steel passive components that exhibit structural degradation such as rejectable non through-wall defects or active pressure boundary leakages. Specifically, the TM establishes a technical basis for a data-driven approach to piping reliability analysis that acknowledges the potential negative and positive impacts on pressure boundary integrity from material aging and reliability and integrity management (RIM) program implementation, respectively. The work that is documented in this technical report was performed during the period 2016 through 2018.