Annex I
METHODOLOGY DESCRIPTIONS

I–1. DATA DRIVEN METHODOLOGY

This subsection documents an approach to implementing a DDM of piping reliability. The term 'data driven' implies that a model of piping reliability as much as possible builds on technical insights obtained from detailed evaluations of OPEX data on piping material degradation and failure. In the DDM approach the treatment of uncertainties is accomplished by using Bayesian techniques and Monte Carlo simulation.

I–1.1.Constituent elements of the data driven methodology approach

The constituent elements and input parameters of the DDM approach are summarized in Tables I–1 and I–2, respectively. The failure rate term and the CFP term are assessed using a Bayesian approach. Depending on the context of an analysis, the frequency of a pipe failure having a specific consequence is estimated on a piping component level or on a piping system level.

TABLE I–1. CONSTITUENT ELEMENTS OF THE DDM APPROACH (INPUT PARAMETERS ARE PROVIDED IN TABLE I–2)

DDM element	Description
$F(IE_x) = \sum_i m_i \times \rho_{ix}$	To derive a frequency of an initiating event caused by a pipe failure of a certain type *x* per reactor operating year (ROY). It could be a pipe failure due to a degradation mechanism acting on a well-defined location within a piping system producing a certain consequence. The frequency is subject to epistemic uncertainty
$\rho_{ix} = \sum_k \lambda_{ik} \times P(R_x \mid \mathrm{DC}_{ik}) \times I_{ik}$	To derive a frequency of failure of component type *i* with EBS *x* (size of hole in pipe [mm] or through-wall mass flow rate [kg/s]), subject to epistemic uncertainty
$\lambda_{ik} = \frac{n_{ik}}{\tau_{ik}} = \frac{n_{ik}}{f_{ik} N_i \, T_i}$	Pipe failure rate per location-year for pipe component type *i* due to degradation mechanism *k*; subject to epistemic uncertainty. The failure rate is estimated using Bayes' theorem

TABLE I–2. INPUT PARAMETERS OF THE DDM APPROACH

DDM input parameters		
Parameter	**Dimension**	**Description**
$P(R_x\|DC_{ik})$	Probability	CFP of size x given failure of pipe component type i due to damage or degradation mechanism k, subject to epistemic uncertainty. The CFP parameter is conditional on a degraded condition (DC). Depending on the type of piping (i.e. BBL or LBB), this parameter may be determined on the basis of PFM, expert elicitation or OPEX data
λ_{ik}	For example: [1/Weld.ROY] or [1/m.ROY] or [1/Elbow.ROY]	Failure rate per location year for pipe component type i due to degradation mechanism k, subject to epistemic uncertainty. This term is a representation of the susceptibility of a given piping component to material degradation. The probability distribution is developed using a one-stage or two-stage application of Bayes' theorem
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	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 ROYs 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 ROYs for the failure population for component type i. $\tau_{ik} = f_{ik} \times N_i \times T_i$

I–1.2. Bayesian techniques

The DDM produces piping reliability parameters through detailed evaluations of OPEX data via an implementation of a Bayesian treatment of uncertainty. The latter aspect involves the application of empirical Bayes' methods and standard Bayes' methods. The methodology consists of two basic elements: (1) the pipe failure rate (λ) that is calculated using a Bayesian analysis technique to update the probability distribution representing a prior state of knowledge with the evidence from a pipe failure event database, and (2) the CFP of certain magnitude (e.g. peak through-wall mass or volumetric flow rate or exceeding a flow rate threshold value). Typically, multiple calculation cases have to be addressed in order to perform a complete reliability and risk characterization of a degraded or failed pipe. These basic elements are elaborated on in this section.

The model used for relating failure rates and rupture frequencies has a simple formulation. The pipe failure modes that are 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 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 is 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 OPEX data, engineering judgement, expert elicitation and PFM. 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.

In the development of Bayesian uncertainty distributions for the above reliability parameters, prior distributions are developed for the parameters λ_{ik} and $P(R_x|F_{ik})$. These distributions are updated using evidence from the failure and exposure term data. The failure rate exposure terms are uncertain because of plant-to-plant variability in, for example, pipe length and weld populations. The ROY of service experience has a relatively small amount of uncertainty; therefore, this uncertainty is dominated by the uncertainty in the estimated pipe population; number of elbows, tees, welds or length of piping. An example of the plant-to-plant variability in socket weld populations is given in Fig. I–1. It is a summary of primary system socket weld populations in twenty (20) reactor units.

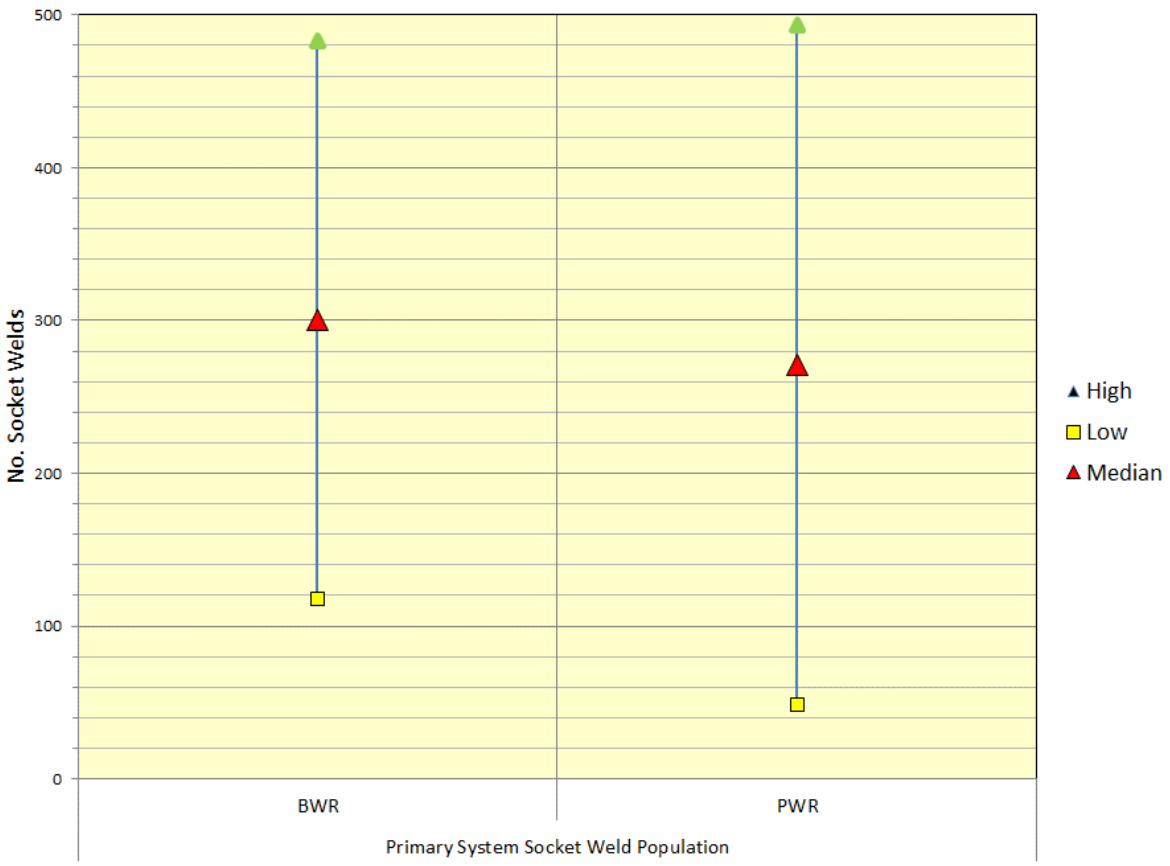


FIG. I–1. Example of plant-to-plant variability in socket weld populations.

The uncertainty in the piping component exposure term 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 are then combined using Monte Carlo sampling to obtain uncertainty distributions for the pipe failure frequencies.

I–1.3.Methodology for developing prior distribution parameters

An implementation of the Bayesian approach begins with the development of a prior distribution, either through empirical or standard Bayes' methods. In the empirical Bayes, the prior distribution is developed from the operating experience data. In the standard Bayes' applications, the prior distribution is fixed before any failures are observed. Engineering judgement and OPEX insights may be used to characterize the uncertainty in pipe failure frequency estimates.

To answer the question 'what are the conditions for developing empirical prior distributions?', two examples, the IGSCC and flow accelerated corrosion, are used. The two material degradation mechanisms are well understood and managed through specific reliability and integrity management processes. The BWR IGSCC experience is used to illustrate how the related knowledge base has evolved as shown in Fig. I–2.

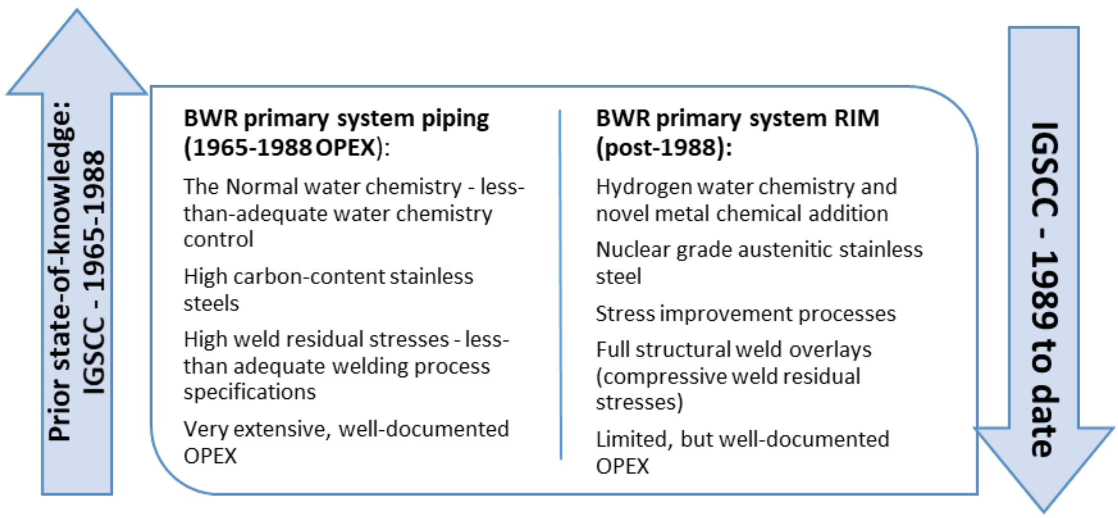


FIG. I–2. Basis for developing empirical IGSCC-centric pipe failure rates.

For the prior state of knowledge in Fig. I–2 the OPEX is extensive and well documented. The demarcation between the two domains 1965–1988 and post-1988 is logical in that it is strongly connected with the reactor regulations and industry initiatives to address and mitigate intergranular SCC. The available OPEX lends itself to the development of empirical Bayes' priors that are input to an estimation of the intergranular SCC mitigation factor of improvement.

Similar to intergranular SCC, the OPEX data on flow accelerated corrosion failures can be divided into two groups, before and after implementing effective RIM processes to monitor and mitigate pipe wall thinning. That is, well documented justifications exist for how to define the demarcation between the prior and posterior state of knowledge.

In order to provide a link to OPEX data and to maximize the uncertainty, a common approach is to use the constrained non-informative distribution method [I–1]. This method has been used, for example, in estimating initiating event frequencies for internal flooding events. The constrained non-informative distribution method: a method which permits an informed estimate of the mean of an uncertainty distribution but otherwise is non-informed to maximize, is comprised of the following steps.

— 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 to apply the constrained non-informative distribution method ($\alpha = 0.5$); the method for the gamma distribution is defined as follows:

$$\text{Mean} = \frac{\alpha}{\beta} = \frac{n}{T} \tag{I–1}$$

where:
$\alpha = 0.5$;
$\beta = 0.5/(\frac{n}{T})$.

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 exposure t, the Poisson likelihood function is selected, the posterior distribution is also a gamma distribution with parameters α' and β' calculated as follows:

$$\alpha' = \alpha + m \tag{I–2}$$

$$\beta' = \beta + t \tag{I–3}$$

The problem to define a prior distribution is now down to defining parameters n (i.e. the industry wide experience) and T (i.e. the observation period during which n was observed). The prior state of knowledge on which the analyst bases the parameter selection comes from piping OPEX data.

I–1.4.Estimation of the conditional failure probability

A statistical model of the CFP may be used to extrapolate estimates of failure frequency of minor leaks to major structural failures. The CFP can be represented as a cumulative failure probability versus an 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 an equivalent diameter in mm (i.e. an EBS).

An insight from the assessment of the OPEX is that the slope of the CFP curve has a relatively strong decreasing probability with an increasing failure magnitude; Fig. I–3 shows the CFP estimated on the basis of a statistical analysis of all relevant operating experience data. In this case, below ground raw water piping susceptible to corrosion mechanisms and severe loading conditions through external impact [I–2], and safety class 1 small diameter piping subjected to high cycle fatigue.

Figure I–4 shows a larger set of plots of the number of pipe failures of a certain safety class versus the consequence of failure. The y axis indicates the number of observed pipe failures. With information about the total pipe failure populations it becomes a relatively trivial task in theory to convert the y axis to indicate a CFP on the basis of empirical data. This figure accounts for WCR and advanced WCR operating experience. It highlights some of the complexities of estimating the CFP solely on OPEX. As indicated, for safety class 1 piping the most significant pipe failures to date have produced through-wall flow rates of no more than about 10 kg/s. Some form of extrapolation is needed to address more significant failure consequences.

With no OPEX data available to support direct statistical estimation of the CFP, an assessment can draw insights from PFM. A practical approach to calculating conditional pipe failure probabilities is to use a Bayesian approach in which the prior CFP uncertainty distribution is expressed by a beta distribution [I–3]. The beta distribution takes on values between 0 and 1 and is defined by two parameters A and B.

The mean of the beta distribution is expressed as $\text{CFP}_{\text{Mean}} = A/(A+B)$. 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 correspond to that expert's estimate of the mean failure probability.

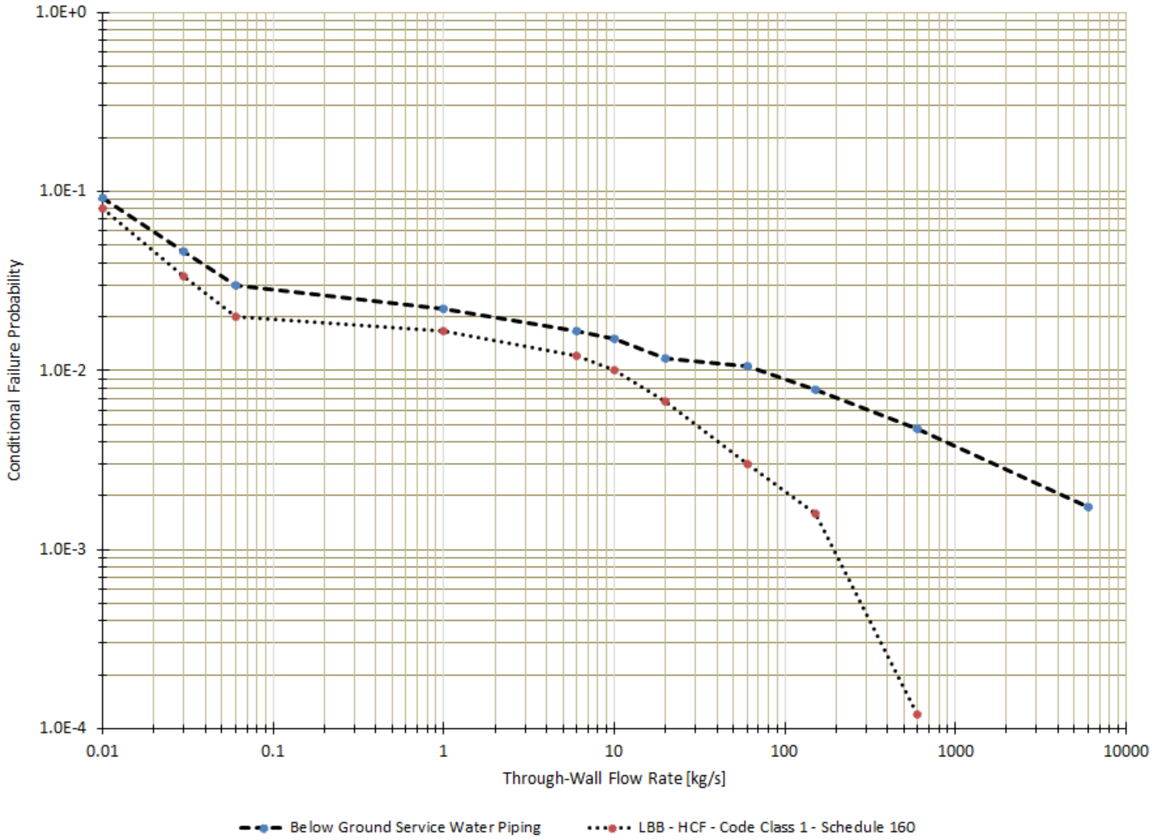


FIG. I–3. Two classes of CFP curves.

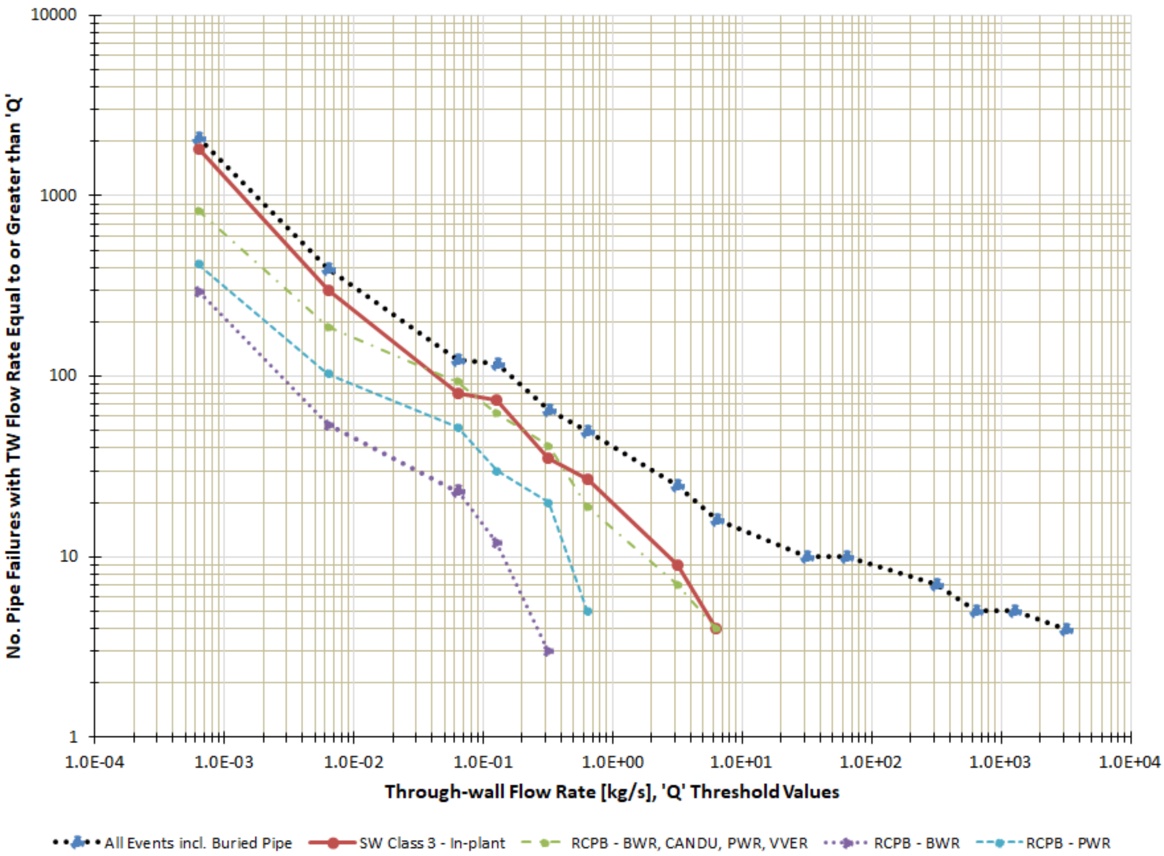


FIG. I–4. OPEX on piping pressure boundary failures involving active leakage.

To illustrate a meaning of this, for example, in order to represent an expert estimate of 10^{-2}, the two parameters can be selected as $A = 1$ and $B = 99$ as per $\mathrm{CFP}_{\mathrm{Mean}} = A/(A+B)$. These abstract parameters can be associated with the number of failures and the number of successes in examining OPEX data to obtain an estimate of the conditional pipe failure probability. The following explains how the beta distribution can be used to estimate the CFP. If a statistical basis is sought for the estimation of the CFP, the questions to be asked are: 'exactly how are data extrapolations to be expressed?' and 'how are the uncertainties in such extrapolations stated?'. The beta distribution has some convenient and useful properties for use in Bayes' updating. A prior distribution, representing an analyst's understanding of piping performance, can be represented by selecting an appropriate set of initial A and B values, denoted as A_{Prior} and B_{Prior}. These prior parameters are to 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. 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 $A_{\mathrm{Posterior}} = A_{\mathrm{Prior}} + N$ and $B_{\mathrm{Posterior}} = B_{\mathrm{Prior}} + M$. Here, 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. Since many different parameter combinations will produce the same mean value, selecting well justified A and B parameters is not a trivial task. Where very little evidence is available about the parameters, non-informative priors may be tested. For leak before break piping it can be concluded that parameter A has to be a small number. A non-informative A_{Prior} assumption would be to assign a value of approximately ½ (assigned value is always less than 1).

The two parameter beta distribution has been well accepted to model the conditional probability of failure because of its versatility and ease of use in a Bayesian framework. However, assessing the parameters A and B is not trivial since an infinite number of combinations of A and B can produce the correct value of p. Insights from OPEX are an entry point for developing engineering-oriented bounding values for the CFP. That is, our Bayesian prior state of knowledge of the CFP for different combinations of systems, operating environments and loading conditions.

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). Extensive work has been performed to determine the best approximations of the mean and standard deviation of a beta distribution, based on three percentiles. As an example, the experts are asked to provide the 5th, 50th, 95th percentiles ($C_{.05}$, $C_{0.50}$, $C_{0.95}$, respectively). The information provided by the expert may be based on PFM results. The estimated mean μ and variance σ^2 are obtained from the following equations [I–4]:

$$\mu = 0.185\,(C_{0.95} + C_{0.05}) + 0.63\,C_{0.50} \tag{I–4}$$

$$\sigma^2 = \left[\frac{C_{0.95} - C_{0.05}}{3.29 - \frac{0.1(C_{0.95} + C_{0.05} - 2C_{0.50})^2}{\sigma_0^2}}\right]^2 \tag{I–5}$$

$$\sigma_0^2 = \left(\frac{C_{0.95} - C_{0.05}}{3.25}\right)^2 \tag{I–6}$$

where the values of 0.181 and 0.63 are the upper/lower and mid-point probabilities, respectively, as assigned by an expert, while the coefficients 3.29 and 3.25 were established using a procedure as defined in [I–5]. Then, the parameters A and B can be obtained as follows:

$$\begin{cases} A = \left[\frac{(1-\mu)\,\mu}{\sigma^2} - 1\right]\mu \\ B = \left[\frac{(1-\mu)\,\mu}{\sigma^2} - 1\right]\,(1-\mu) \end{cases} \tag{I–7}$$

Whenever the current state of knowledge establishes a basis for assessing the 5th, 50th, 95th percentiles of the CFP, the expert can use the above approach to define a defensible set of prior distribution parameters. The results of two Bayesian updates of CFP parameters are illustrated in Fig. I–5. One assessment used NUREG-1829 expert elicitation results [I–6], and another used PFM to develop the prior *A* and *B* parameters of the beta distribution.

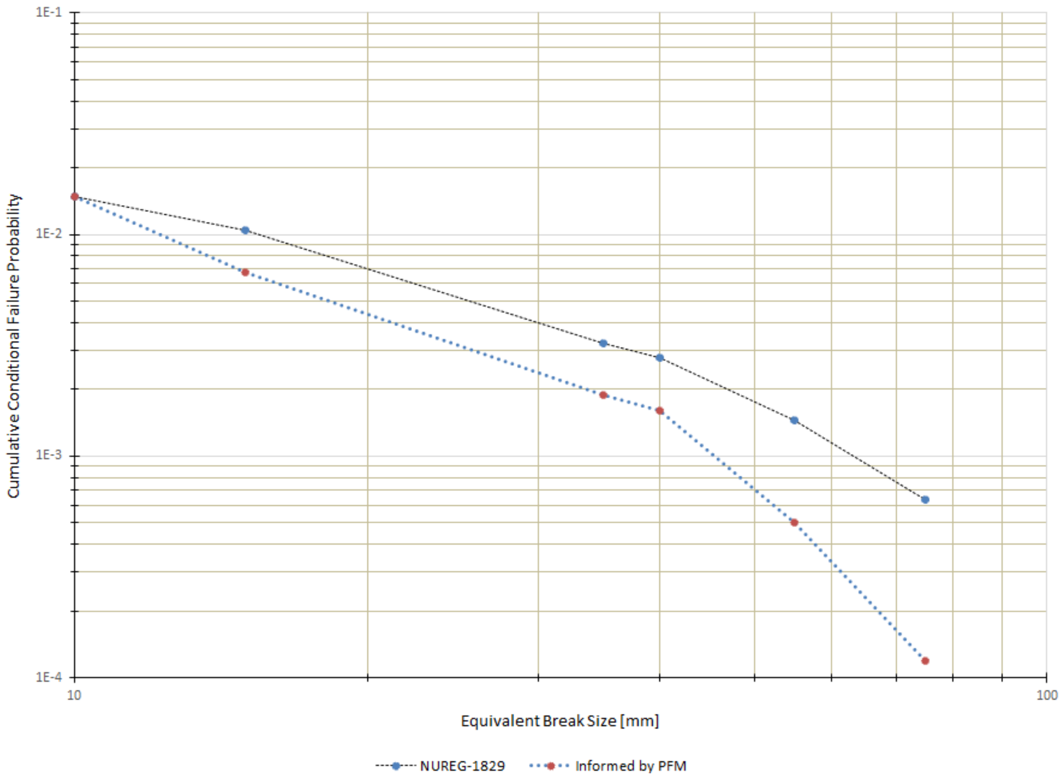


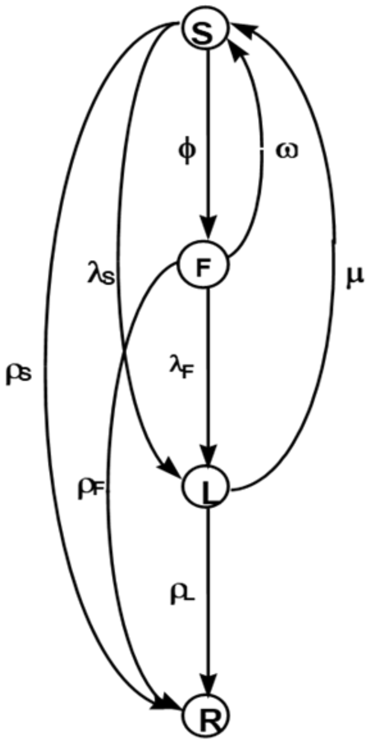
FIG. I–5. CFP models based on expert elicitation and PFM.

I–1.5.Markov model extension

An implicit consequence of relying on OPEX data is that the derived reliability parameters account for different types of RIM processes; for example, leak detection technologies, ISI, non-destructive examination [I–7]. The influence of RIM on pipe failure rates is accounted for by a structural integrity management factor, I_{ik}, in the equation for ρ_{ix} (Table I–3). A Markov model is used to quantify I_{ik}.

The Markov model address the interactions between piping material degradation, ISI, flaw detection and repair. The Markov modelling starts with a representation of the integrity of a piping component in a set of discrete and mutually exclusive states. At any instant in time, the piping component is permitted to change state in accordance with competing processes appropriate for that NPP state. A Markov model state refers to flaws, leaks or ruptures. The processes that can create a state change are degradation mechanisms acting on the pipe and the processes for 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.

A general four-state Markov model of piping reliability is shown in Fig. I–6. All failure processes of this model can be evaluated using OPEX data. A piping component can be in four mutually exclusive states: *S* (success or free of flaws), *F* (flawed or cracked), *L* (leaking, non-active leakage, or active leakage with leak rate within technical specification limit) or *R* (significant structural failure). The repair rates μ and ω are estimated using the simple models in Table I–3.



PIPING INTEGRITY 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 or major structural failure

State Transitions:	
ϕ	Occurrence of non-through wall flaw
λ_S	Occurrence of a leak (L) given no flaw – from success state to leak
λ_F	Occurrence of leak given a pipe flaw (‘*F*’)
ρ_S	Occurrence of rupture given no flaw
ρ_F	Occurrence of rupture given a flaw
ρ_L	Occurrence of rupture given a leak
μ	Detect through in-service inspection and repair leak
ω	Inspect and repair a non-through wall flaw

FIG. I–6. Four-state Markov model of piping reliability [I–7].

Markov model parameters are provided in Table I–3. It can be seen that the hazard rate is a function of time and the parameters of the Markov model. For the four-state Markov model the time dependent frequency of pipe rupture, *h{t}*, is expressed as a function of six parameters: occurrence rate for detectable flaws (ϕ), 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), repair rate for detectable flaws (ω) and repair rate for leaks.

TABLE I–3. MARKOV MODEL PARAMETERS

Model/parameter	Description
$\omega = \frac{P_I P_{FD}}{(T_{FI}+T_R)}$	Rate of inspection and repair non-through-wall defect
$\mu = \frac{P_{LD}}{(T_{LI}+T_R)}$	Rate of detection and repair of a leak
P_I	Probability that a piping element with a flaw will be inspected per inspection interval
P_{FD}	Probability that a flaw will be detected given that this segment is inspected. Also referred to as the POD
P_{LD}	Probability that a through-wall flaw will be detected during a walk-down inspection
T_{FI}	Mean time between inspections for flaws (inspection interval)
T_R	Mean time to repair when detected. If the detection is made with reactor @ power, the repair time should be the minimum of the actual repair time and the time associated with any technical specification limiting condition for operation if the leak rate exceeds technical specification requirements
T_{LI}	Mean time between inspections for leaks
$I_{ix} = \frac{h_{ix}(t)}{h_i(t)}$	Integrity management factor for component *i* for RIM strategy *x* at piping system *t*. As shown in this equation, this factor can be expressed as a product of separate integrity management and piping system factors. The RIM strategy includes a specification of how often leak tests and inspections and non-destructive examination are performed on the piping system component and how effective they are
$h_{ix}(t)$	Hazard rate for component *i* at piping system *t* and RIM strategy *x*. This is determined from the solution of the ordinary differential equations that describe the Markov model
$h_i(t)$	Hazard rate for component *i* at piping system of *t* and RIM strategy corresponding to an 'average' component in the service data. For example, for ASME class 1 components, the average component is determined by 25% of the components being subjected to non-destructive examination exams every 10 years, the remaining 75% not being subjected to non-destructive examination, and 100% of the components being subjected to a system leak test once every refuelling outage

Opportunities for leak detection depend on the system in which the leak occurs as well as the specific location and size of the leak. For example, in the primary coolant system, leaks of a significant magnitude (e.g. $>6.3 \times 10^{-2}$ kg/s) would prompt a containment high radiation alarm in the main control room. In these cases, the time to inspection and repair is limited by the technical specifications on primary coolant system leakage and the time to cool down the plant and begin the process of repair. Other, smaller leaks may not cause an alarm but would be subject to possible detection during periodic (e.g. every 12 h) walk-down by plant personnel or other opportunities 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 codes and standards rules for different classes of piping.

For most leaks, the detection possibilities are not normally limited to some predetermined population of welds that are inspected. Therefore, there is a difference between the leak repair term μ in comparison to the flaw repair term ω. Leak testing provides an opportunity to inspect all locations system wide, and therefore for a given leak of significant magnitude anywhere in a piping system, the probability of leak detection tends to be high. For uninspected locations the flaw repair rate term ω is zero. 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. Therefore, the Markov model provides the capability to take into account the leak before break principle.

An example of Markov model input parameters is given in Table I–4. This example addresses the impact of RIM on pipe failure induced LOCA and the analysis was performed in support of the NUREG-1829 expert elicitation [I–6].

TABLE I–4. EXAMPLE OF MARKOV MODEL INPUT DATA

Parameter	Assumed or estimated value	Basis
ω	2.1×10^{-2}/year {= (0.25) × (0.90)/(10 + (200/8760))}	The weld is assumed to have a 25% chance of being inspected for flaws every 10 years with a 90% detection probability. In the example detected flaws will be repaired within 200 h
μ	7.92×10^{-1}/ year {= (0.90) × (0.90)/(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_F	$N x\ 10^{-x}$/year	If an element is already leaking, the conditional frequency of rupture is assumed to be determined by the frequency of a severe pressure pulse from a water hammer
ϕ	Variable	The occurrence rate of a flaw is estimated from OPEX data
P_{FI}	1 or 0	Probability per inspection interval that the pipe element will be included in the ISI programme
P_{FD}	Variable	Probability per inspection interval that an existing flaw will be detected; POD
P_{LD}	Variable 0 — no leak detection to 0.9 with leak detection	Probability per detection interval that an existing leak will be detected
T_{FI}	10 years (per ASME XI)	Flaw inspection interval, mean time between ISIs
T_{LD}	Variable 1.5 — Once per refuelling outage/1.92×10^{-2} for weekly/9.13×10^{-4} for daily	Leak detection interval, mean time between leak detections
T_R	Variable	Mean time to repair given detection of a flaw or leak

I–1.6.Calculation procedure

Details on how to implement the DDM calculation procedure are found in [I–8, I–9, I–10]. The methodology 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. A posterior weighting procedure is used to synthesize the results of these three Bayes' updates into a single composite uncertainty distribution for the pipe failure rate. The posterior weighting procedure is implemented in Microsoft Excel with a suitable add-in program to facilitate uncertainty propagation using a Monte Carlo simulation routine. The tasks involved to calculate probabilistic failure metrics are illustrated in Fig. I–7.

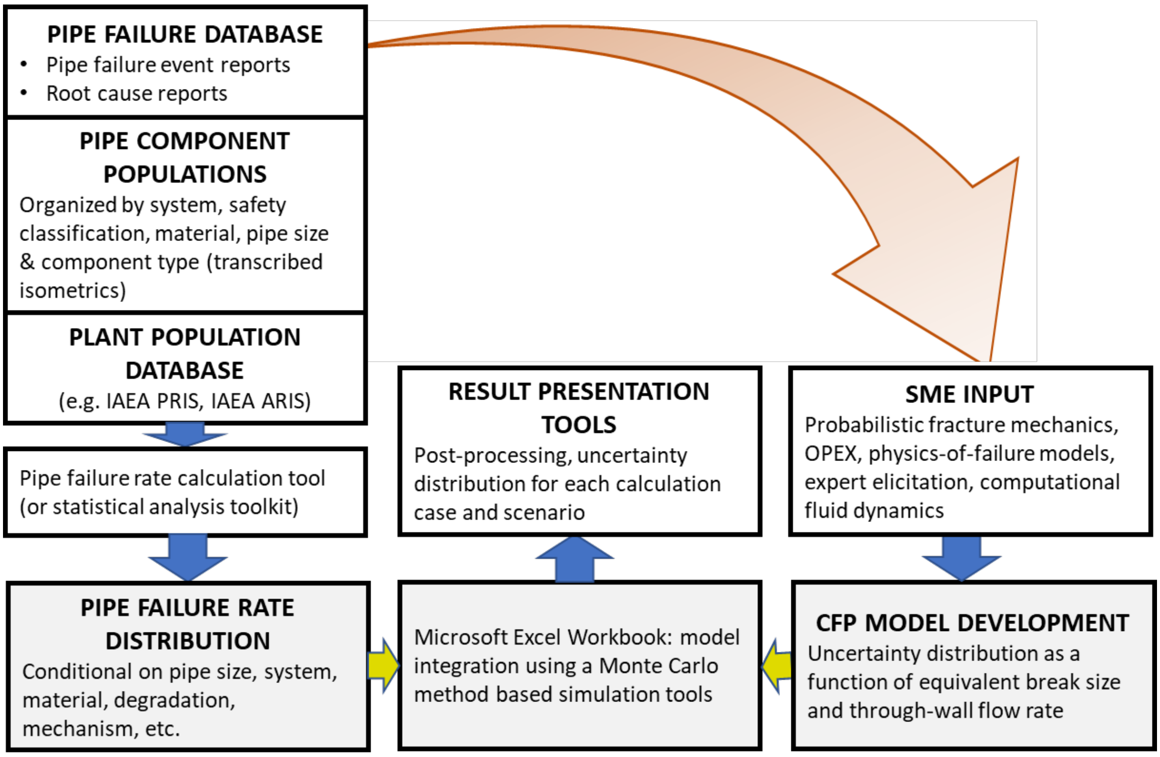


FIG. I–7. Implementation of the piping reliability calculation format.

An implementation in Microsoft Excel workbook format can be done as follows. Each analysis step is implemented on a worksheet (or tab):

(1) Definition of the evaluation boundary(-ies);
(2) Definition of the pipe failure modes together with results of the supporting engineering calculations that correlate the consequence of a passive component failure with the size(-s) of a through-wall flaw;
(3) Definition of exposure terms and results of OPEX review to obtain failure populations;
(4) Pipe failure rate calculation based on Bayesian statistics. This step starts with the definition of appropriate prior state of knowledge probability density functions;
(5) CFP calculations. This step may include input from OPEX, PFM and/or expert elicitation;
(6) Integrated quantitative analysis;
(7) Results presentation which involves extracting and organizing the output in accordance with end user requirements.

I–1.7. Practical insights

The DDM responds to the needs of PSA and risk-informed operability determinations. It is not a substitute for structural reliability models that build on fracture mechanics theories. It allows for calibrating the reliability metrics used as inputs to PFM and I-PPoF. As it is stated in [I–11]:

"Regardless of a chosen technical approach to piping reliability analysis, independent peer review processes invariably raise questions about the achieved level of realism and statistical uncertainty of quantitative results. How well do the results compare with the applicable service experience data? Has the plant-to-plant variability in piping system layout and degradation mitigation practice been properly accounted for? A particularly challenging peer review question is the one posed when no relevant service experience data is available. How should an analysis best be performed in view of zero pipe failures? Also frequently asked is whether or not a certain type of technical approach has been formally endorsed by a regulatory agency? An assessment of the consistency of calculated pipe failure rates and rupture frequencies with operating experience improves confidence in the calculated values. There are strengths and weakness associated with each of the technical approaches to pipe failure probability calculation."

The insights from practical implementations of the DDM can be summarized as follows:

— *Advantages*

- A Bayesian analysis scheme is used to estimate the parameters of the DDM. This approach has two distinct advantages over classical estimation techniques: (1) when zero failures have been recorded the Bayesian estimate is not zero, and (2) uncertainty (posterior) distributions are obtained in a form which can be readily used to propagate these uncertainties, using Monte Carlo simulation, through the piping reliability equation.
- DDM implementations are not limited to a specific form of material degradation. The methodology applies to all combinations of materials and operating environments.
- The methodology allows for reliability parameter updates as new OPEX becomes available.
- It includes a methodology for relating prior distribution parameters to specific combinations of materials and material degradation mechanisms.

— *Technical challenges*

- The Bayesian estimates are based on the use of appropriate prior distributions. It is not a trivial task to define what an appropriate distribution is. In-depth knowledge of piping material degradation and the associated OPEX is required. The steps to validate a chosen prior distribution are to be well documented. The model validation needs to address the sensitivity of the posterior distribution to assumptions of what constitutes 'an appropriate' prior distribution, especially when no or limited OPEX data is available.
- A successful DDM implementation relies on having access to an extensive pipe failure database for extracting and screening of data. However, development of such a database is resource intensive. The model validation includes steps to verify data quality control; for example, what is the process for populating the database and how is the accuracy of pipe failure classifications verified?
- The work to correlate a certain population of failure with the population of plants and susceptible locations within piping systems that produced the failures can be time-consuming. It requires in-depth knowledge of the plant-to-plant variability in piping designs (i.e. material selections, method of fabrication, welding technology, piping codes and standards).
- The plant-to-plant variability in piping system design (e.g. length of piping, number of welds) can be extensive. It is not sufficient to simply apply engineering judgement in the development of pipe failure rate exposure terms. Access to a sufficiently complete piping system design database is required. If not readily available, it is important to obtain the information needed to support the definition of exposure terms and to characterize the uncertainties.

- To be aware of Section 7 of ENIQ Recommended Practice 9 (December 2017) [I–12]:

  "Since the objective of the structural reliability models 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 kept 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 behaviour."

  An analyst may opt to agree or disagree with this opinion.

I–2. PROBABILISTIC FRACTURE MECHANICS METHODOLOGY

This subsection explains the basic principles of PFM. Fracture mechanics is an engineering discipline that quantifies the conditions under which a load-bearing structure can fail due to the enlargement of a pre-existing dominant crack. The crack can be embedded within the pressure boundary component or be surface connected, for example, to an inside pipe wall. The key ingredients in deterministic fracture mechanics analysis are the initial crack size, crack driving force solution (i.e. stress intensity factors for linear elastic fracture mechanics analysis problems), applied stresses, and material properties describing the subcritical crack growth characteristics and conditions for final crack instability. PFM is fracture mechanics that considers some or all of the inputs to be random variables (e.g. the initial crack size). The PFM discipline has evolved over a period of more than four decades. The principal application of PFM in the nuclear industry continues to be in the area of primary system piping integrity and reactor pressure vessel integrity.

In a fracture mechanical analysis, the behaviour of cracked structures under certain loads and damage mechanisms are assessed, and the growth of cracks due to ageing as well as the instability of cracks (which is associated with a type of failure) are computed. This deterministic study can be used in a safety analysis of a design or the assessment of indications during inspections. PFM is the extension of this concept to treat input parameters as distributed random variables, such that the failure can be expressed as a probability. Hence, PFM has its origin in the structure mechanics discipline, with the probabilistic reliability aspect added.

The computation of failure probabilities or failure rates can for itself be a challenging task, which involves also numerical techniques of parameter sampling. The complexity of solving a PFM task (computing the failure probability) gave rise to several computational benchmark studies, which compared the results obtained with different computing codes. An important insight from the many benchmark case studies is the importance to specify input parameters which are meaningful for a selected evaluation boundary.

I–2.1.Basic considerations

I–2.1.1. Introduction

The computational approach of PFM can be seen as a part of the field of structural reliability. The investigation of pipe failure rates during the operation of a NPP and its prediction is a type of time variant structural reliability analysis [I–13]. The underlying approach is the computational simulation of a structure and the assessment of its failure by the means of mechanics and material models, with a special emphasis on structures with defects. The behaviour of a structure is characterized by the input data modelling the geometry, the material, the loads, damage mechanisms and so on. Within these input quantities, a finite set of real variables is of importance, which correspond to the parameters associated with (explicit) uncertainties in the PFM approach, which are called basic variables, $x_1, \dots, x_n$, collected in a vector $x = (x_1, \dots, x_n)$. A function $g(x)$ describes the structural state, with a criterion for failure corresponding to: $g(x) \leq 0$. The set $\{x : g(x) = 0\}$ is called the limit state. The probability of failure is computed by assuming a probability density function $\rho(x)$ for the vector of basic variables. The total failure probability is then given by the integral over the area that represents a failure:

$$p = \int_{g(x) \leq 0} \rho(x) \mathrm{d}x \qquad \text{(I–8)}$$

Thus, the computational task for solving a PFM problem is in practice to find the solution of a multidimensional integral. As temporal processes of change are essential for long term operations and ageing effects of nuclear facilities, the scope has to be extended to a time variant reliability problem. In the time variant approach, the limit state becomes a function of the operation time. The failure frequency in a specific time interval in such an approach is the integral restricted to the specific failure region. A graphical representation of the variable space, the distribution function and the limit state is shown in Fig. I–8.

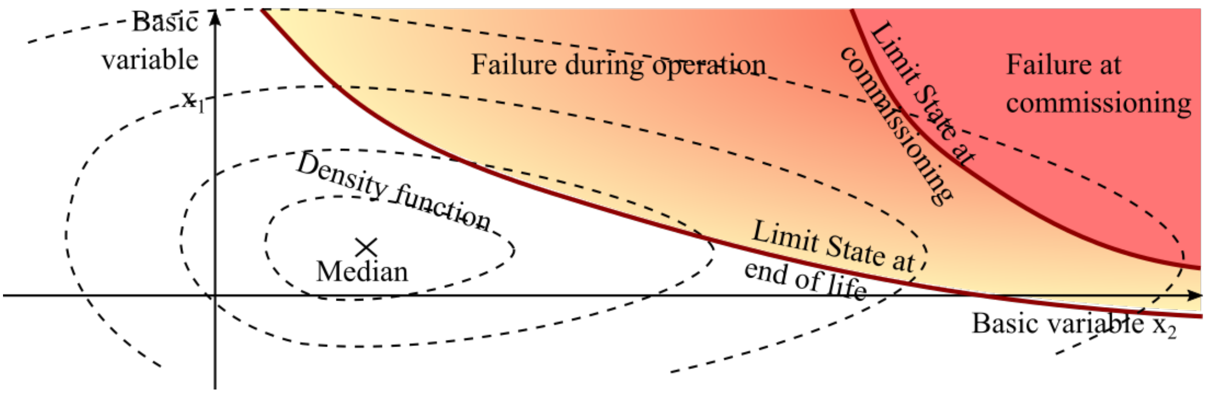


FIG. I–8. Illustration of the general approach to probabilistic structural reliability.

This introduction to the concept of probabilistic structural reliability highlights the relevant aspects of the computation of failure rates of structures: Influence parameters and their statistical distributions, the description of ageing and degradation, of failure, and of the numerical computation of the probability integral. These topics are reviewed in the following sections. As the computational part turns out to be complex, special software tools have been developed and compared in international benchmark case studies. A review of these benchmark activities is given at the end of the section.

I–2.1.2. Ageing mechanisms and failure

The understanding of the mechanics or physics of structural degradation mainly enters via the description of ageing and failure. Flaws are the most relevant defect for the structural integrity, both from a theoretical perspective (stress concentration on crack tips), and from operational experience (cracks as relevant defects). Moreover, the failure assessment of a component is a key part of fracture mechanics.

Fatigue crack growth and SCC are the main considered ageing mechanisms. Fatigue crack growth is driven by cyclic loading, while SCC is driven by a constant load acting on a crack and depends on the material as well as the surrounding medium. The crack elongation per cycle (or per time) as a function of the load can be well measured on test specimens in laboratory experiments and transferred by fracture mechanical parameters.

The failure of a simulated cracked component is associated with the instability of a crack. The result might either be a leak (wall penetrating crack) or a large rupture (if a leak of the given size is unstable itself). Similarly, it is possible to simulate an intact component and consider crack formation as failure due to accumulation of fatigue damage or due to corrosion driven crack initiation frequencies. These different failures are illustrated in Fig. I–9.

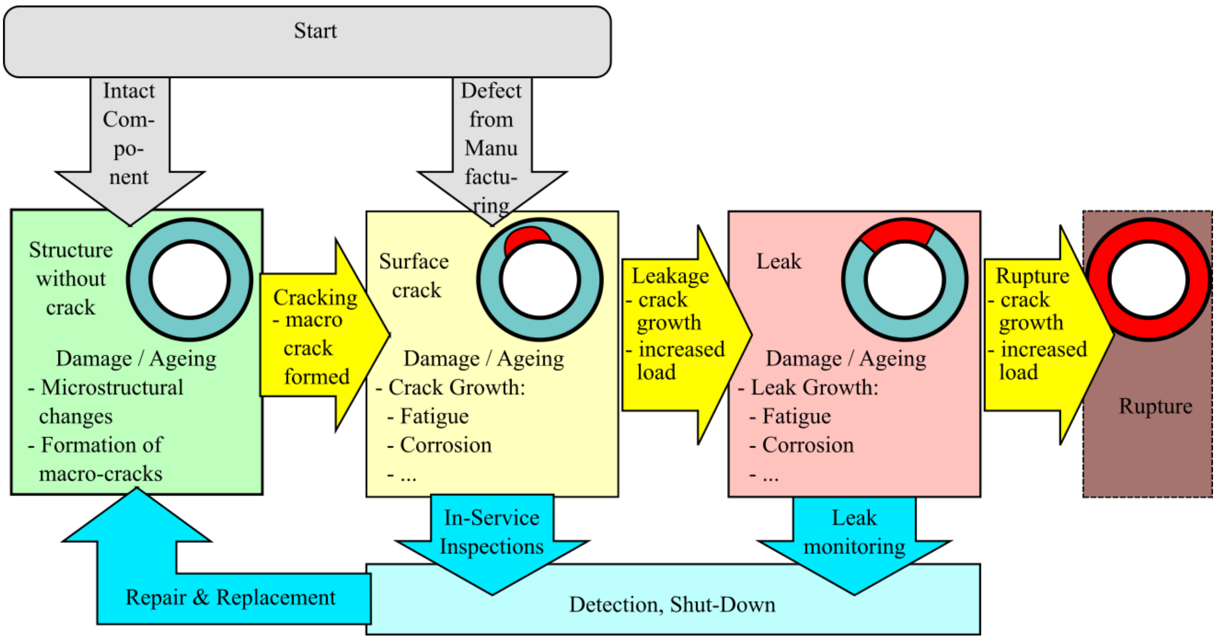


Fig. I–9. Illustration of the different modes of structural degradation and failure.

Thus, these different failures represent the transition between different degradation states of a pipe. Each of them is associated with a separate limit state function, and for each one, a probabilistic simulation will deduce a transition probability (failure rate).

For the assessment of failure of cracks in structures, several techniques are available. Beside the finite element analysis, several codes propose assessment schemes of fracture mechanical parameters, and plastic collapse limit loads are an additional option. In recent examples, it was shown that for the failure frequency, the choice of the failure criterion is even of secondary importance, due to the high crack growth rate of large cracks and the large safety margins of nuclear components [I–14].

I–2.1.3. Detection and repair

Detection and repair actions which may reduce failure rates are included in many probabilistic fracture mechanical codes. The three approaches discussed in the following paragraphs comprise ISIs, leak monitoring (and leak before break), and pressure tests (and hydrostatic testing). Their effect is indicated by the blue arrows in Fig. I–9.

The most established approach is the consideration of ISIs, which are assumed to be done in specific inspection intervals. These considerations are interesting for the effectiveness of ultrasonic testing in plants and can be a basis for risk-informed maintenance approaches. This was a key aspect in the benchmark studies [I–15].

ISI, for example, performed with ultrasonic sensors, is only one monitoring instrument of the plant operator to identify defects and to prevent pipe failures. Another important instrument is leak monitoring, which has the task of identifying leaks before a larger failure of the pipe (rupture) can occur. The leak before break concept is the approach behind this idea. It mainly applies to large diameter piping with high reliability, where a large rupture would question the controllability of a situation. Approaches and discussions of a probabilistic treatment of leak before break for piping in NPPs can be found in [I–16] to [I–19].

A further possibility to consider preventive actions and repair measures in a probabilistic fracture mechanical evaluation is the simulation of hydrostatic testing or pressure tests. In these tests (performed before regular operation, e.g. after an outage) the interior pressure of the system is increased to a specific value — the operating pressure (in the case of a hydro test) or an elevated value even above the design pressure (pressure test). An approach to connecting PFM with the results of pressure tests has been proposed [I–20].

I–2.1.4 Reliability and sampling

Typically, pipes in NPPs are designed and operated in accordance with applicable national codes and standards, in order to attain a high reliability. Thus, the failure frequencies of the pipes are low, and the domain of the integral Eq. (I–8) is only a small part of the basic variable space. As the computation, in general, has to be done numerically and each evaluation of $g(x)$ involves a simulation of one pipe specimen, it is important to compute the probability integral in an efficient way, with as few evaluations as possible.

Examples of reviews and comparisons of sampling strategies are given in [I–21]. Basic approaches are the Monte Carlo simulation, which has shortcomings for very small failure probabilities, and stratification techniques, which are limited to a small number of basic variables. Tailored solutions for the structural reliability problem are the first and second order reliability methods, which are used to approximate the limit state functions. More general approaches extend the insights of the first order reliability method and second order reliability method with importance sampling, leading to design point based importance sampling, spherical (or radial) sampling and the Vegas algorithm.

I–2.1.5 Software tools

Implementing PFM requires highly specialized software tools. The codes pc-PRAISE and WinPRAISE [I–22, I–23] represent early software solutions for fatigue and SCC in pipes. This work has influenced the later development of this field. Examples of later developments are the computer codes PRO-LOCA [I–24], PROST [I–19] and PINTIN [I–25].

A large number of PFM computer code benchmarks have been performed. For example, the NURBIM project [I–26], the computer codes NURBIT, WinPRAISE, PROST, PRODIGAL, ProSACC and STRUREL were compared. Within the framework of the Nordic Nuclear Safety Research programme, the codes VTTBESIT, NURBIT, the JRC code, and ProSACC were applied to SCC test cases [I–27].

I–2.2 Calibration of probabilistic input parameters with operating experience data

In probabilistic analysis the term 'calibration' implies an analytical process that is used to adjust a set of parameters associated with a computational code so that the model agreement is maximized with respect to a set of experimental data or OPEX data [I–28, I–29]. This process consists of identifying the controlling input parameters, and to compare the results against, for example, the estimated frequency of a crack of certain dimensions.

I–2.2.1 Definition of PFM input data

The following paragraphs summarize the required input data of a PFM analysis and how they are specified based on the evaluation boundary specification. In this discussion, an evaluation boundary is assumed where operational experience is available — the application to advanced systems without OPEX is discussed thereafter. Some input quantities are available from plant specific piping stress analyses or from the design rules as given in codes and standards. For other input parameters, laboratory tests are performed to characterize the underlying effects in detail. Also, some input parameters are based on a more or less accurate estimate (i.e. engineering judgement). On the other hand, some input quantities have a larger effect on the results than others as illustrated in Table I–5.

TABLE I–5. REQUIRED INPUTS OF A PFM CASE WITH ESTIMATED SENSITIVITIES

Input complex	Examples	Knowledge base	Typical sensitivity
Material characteristics	Yield stress, fracture toughness	— Nominal values from codes and standards — Realistic values from laboratory tests — Distributions from literature PFM cases	Low
Damage initiation	Initiation frequency, initial crack size	Poor knowledge	High
Damage progress	Crack growth rates, SN curves	Crack growth rates are measured in laboratory tests	Medium
Loads	Operational and exceptional loads	— Nominal pressure and temperature are given — Deadweight bending, thermal fluctuations and accident conditions are — Uncertain — Location-dependent	Medium
ISI, leak detection	Probability of detection curve, sizing error, leak detection threshold	— Several reports are dedicated to realistic probability of detection curves — Leak detection requirements have default values or can be plant specific	Medium

The statement about typical sensitivity has to be considered carefully, because it depends on the degradation mechanism that is being considered. The expected sensitivity has to be seen with respect to a case which includes a longer plant lifetime and ageing mechanisms, with plant-typical loadings.

One example is the low sensitivity of material properties; bulk material properties have, in the end, a small effect on failure frequencies, which is confirmed in the OPEX and PFM. While most PFM cases are single location analyses, the treatment of a whole plant part with PFM requires the consideration of uncertainties due to different crack locations. A prominent example is the input complex loads, since bending loads, thermal fluctuation spectra and accident loads are strongly location dependent.

Crack growth data sets primarily come from laboratory tests. The process of ongoing damage and crack growth, as it is the key element of the PFM simulation, is usually unobserved in operating power plants: The final failure is reported, or snapshot-like crack sizes from indications during ISIs. Required input parameters even originate more often from laboratory tests. However, additional available observables (in the physical sense that the quantity can be measured and used for a comparison) come from the operational experience, which give leak events, rupture events, and the associated damage mechanisms, all with the dependency on the piping class. This situation is shown in Fig. I–10 where the thickness of the lines from PFM input to PFM indicates the sensitivity to the input quantities. Thus, a path from estimates to thick lines indicates a major problem in the construction of PFM cases tailored to a specific plant situation. Such a path indicates that the PFM input parameters are not well defined by the prior knowledge. However, more information originates from the operational experience which, in a usual PFM case, is only considered with respect to the anticipated damage mechanism.

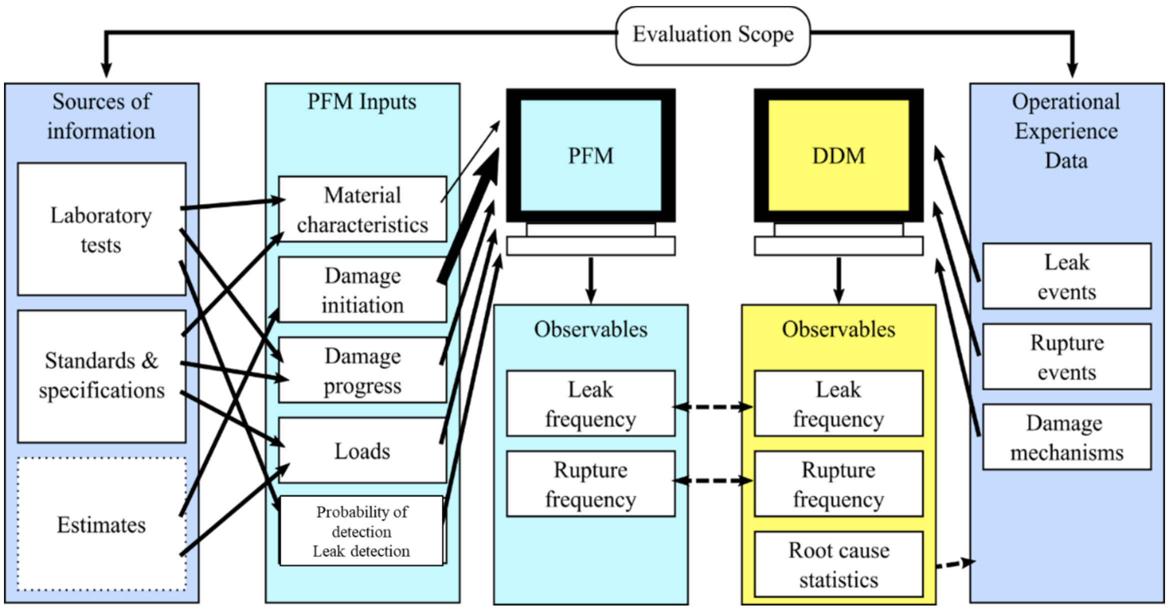


FIG. I–10. PFM input construction and comparison of results with observables.

I–2.2.2 Calibrating a PFM case study

The starting point is to set up a PFM case corresponding to a system accessible for the operational experience based failure rate computation with data driven methods. The key idea in this setup of a PFM case is the variation of uncertain input quantities of the PFM case in order to match the key observables of operational experience:

— DDM computed leak frequency;
— DDM computed rupture frequency;

— Statistics of present damage mechanisms; and
— Other data from operational experience, such as flaws found during ISIs and their size.

More precisely, the uncertainties in the DDM computation (e.g. low event number, uncertain weld count/leak relevant locations) also have an effect on the comparison, which may lead to statistical hypothesis testing. Typically, the uncertainties of the PFM input quantities can be classified as epistemic uncertainties, whereas the consideration of aleatory uncertainties is the native task of PFM as illustrated in Fig. I–11.

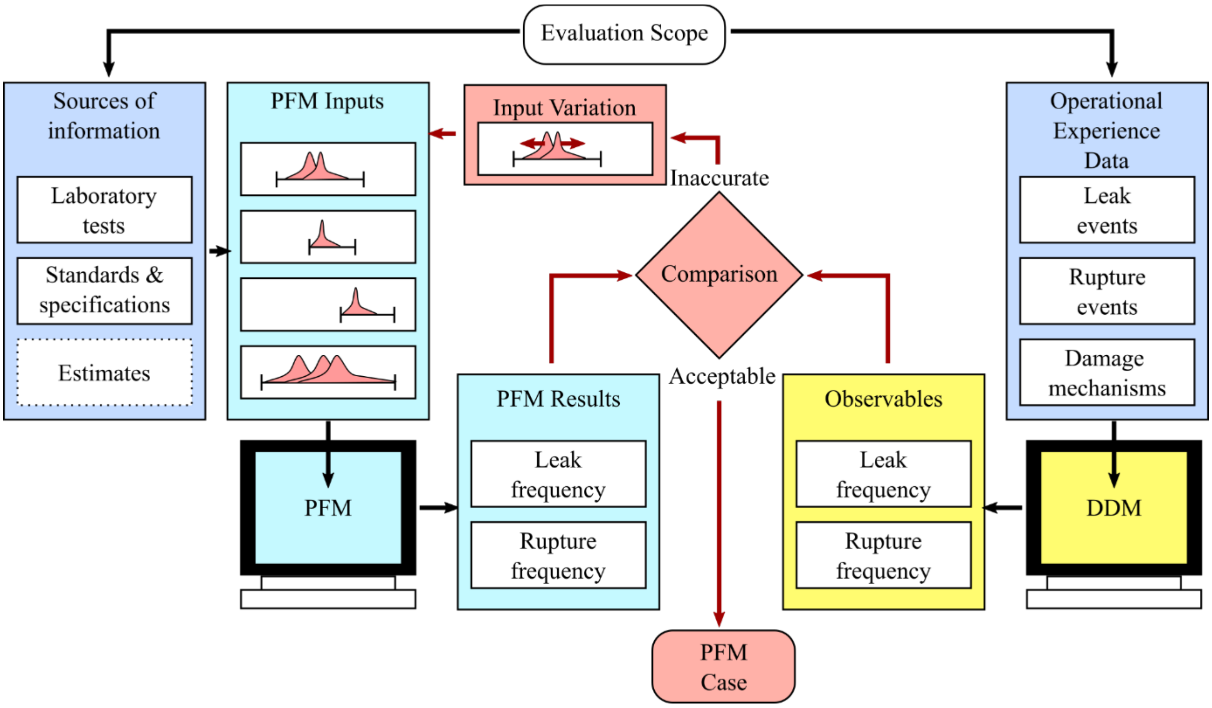


FIG. I–11. Proposed approach (red), connecting the (cyan) PFM approach and the (yellow) DDM approach for constructing a PFM case for a given evaluation boundary.

Thus, after an initial guess, the PFM result can be compared with the DDM frequencies. In order to align the two results, the input quantities are varied depending if they have sufficiently large epistemic uncertainties to allow for an alternative modelling, which affects the result significantly (sensitivity).

This parameter variation or calibration has not a guaranteed existence or uniqueness of a solution. If the desired failure frequencies are not within the plausible range identified by the sources of information, one is to expect that these sources are not accurate or complete enough, and a careful data review is required to reveal possible alternative modelling decisions. If one (more multiple) solution(s) is/are found, a PFM case is constructed which shares all statistics with the real plant data.

I–2.2.3 Parameter calibration strategies

Introduction: Calibration is the key step for the approach, which is driven by the comparison of leak and rupture frequency with values obtained by applying a DDM. This section is dedicated to proposals of parameter variations. The degradation mechanisms and the loads both have significant effects to the computed frequencies (i.e. the epistemic uncertainties are significant). These two properties impel this first calibration proposal to concentrate on damage mechanisms and loadings. As the different damage types are usually sensitive to different load types, it makes sense to discuss them together. Finally, PFM simulations are not principally limited to but in practice often only consider one single damage mechanism. If the failure frequencies are small, the different mechanisms can be thought to act independently, and each mechanism can be treated with its own PFM case. As an introduction, an illustrative survey of damage relevant for leaks from a database evaluation is shown in Fig. I–12.

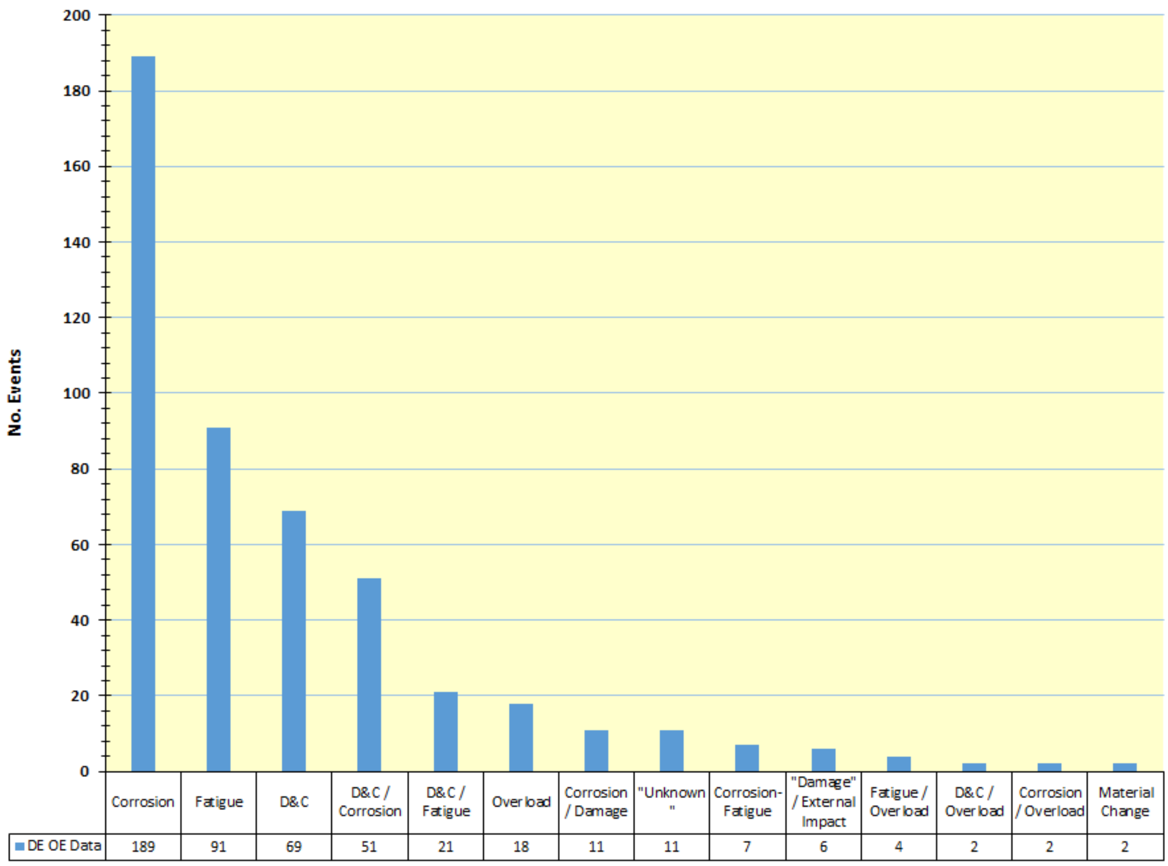


FIG. I–12. German OPEX in the time period 1972–2013 (extracted from German KOMPASS database; details provided in ANNEX III).

Thus, two relevant ageing mechanisms for leaks are corrosion and fatigue, and manufacturing defects are also a relevant source of pressure boundary failures. These root causes are selected for a further discussion in the following subsections as examples; other mechanisms require a similar approach.

Practical calibration strategy for fatigue: Fatigue damage, caused by variation of subcritical loadings, can be classified to multiple types. The typical damages seen in the operational experience are:

— Vibrational fatigue due to mechanical vibrations;

— Thermal fatigue due to thermal fluctuations at stratification layers or mixing branches; and
— Thermomechanical fatigue due to operational cycles.

For fatigue damage, fatigue life (S-N curves) and crack growth relations (da/dN) are usually well investigated — even for the usual case of environmental assisted fatigue. The operational cycles are prescribed by changes in the operational state in the component, ranging from rare (startup and shutdown transients) to frequent (operational, intermittent spraying) changes in loads. An important source for uncertainties is the load spectrum for (unintended) mechanical vibrations, as well as for (inevitable) thermal fluctuations. More precisely, the values of 'low amplitudes' and 'load cycles' are the quantities which can be varied within the uncertainties. These uncertainties represent ignorance about the sources, but also about the location within the plant. Thus, for the high cycle thermal and vibration fatigue, the load spectrum would be the quantity for variation in the calibration approach.

Practical calibration strategy for corrosion: Corrosive cracking is governed by chemical reactions in the confined zone inside a crack, where the local concentration of ions is as important as the (steady) mechanical stresses in the component. The susceptibility of the component material for a corrosion attack is of importance. For leaks identified as break precursors, SCC is most relevant for the analysis.

In contrast to the fatigue damage mechanism, the local load condition which influences the cracking process is typically less complex. A typical steady and uncertain load in welds which influences the cracking is the distribution of residual stresses. Other uncertainties arise from the crack growth rate, which is observed to scatter significantly. The initiation of the crack and the initial size, which represent the artificial separation in micro- and macro-cracks necessary for the assessment method, are further examples for uncertainties. Thus, for corrosive damage mechanisms, the following parameters are candidates for the calibration:
— Weld residual stresses;
— Crack growth rates;
— Initial flaw size;
— Initiation rates/crack initiation model.

The modelling of initiation and crack growth has probably often the most direct influence on the flaw size.

Practical calibration for initial manufacturing flaws: The distribution of undesired defect size from component manufacturing usually has a strong influence on the leak and break probabilities. There are many different distributions proposed for initial flaw size in PFM applications. In terms of the calibration approach, the initial crack length distribution and the initial crack depth distribution are relevant candidates for using epistemic uncertainties in the calibration procedure. However, probabilistic results can be dominated by very deep initial cracks, which can be unrealistic in direct comparison with operational experience. Moreover, the matching of this operational experience will probably match the manufacturing techniques from the mid-twentieth century, and with less relevance for more modern facilities.

I–2.3 Cross-validation and application to advanced WCRs

The output of the calibration approach itself is not yielding new information, the aim is to reproduce exactly the previous knowledge about current plant technologies with existing OPEX. However, it can be a starting point for a transfer to the desired evaluation scope, by modifying it. This modification is to be done in multiple steps, based on the evaluation scope and the available operational experience which does not contain the evaluation scope itself as shown in Fig. I–12:

(1) Selection of a reference evaluation scope, which resembles the application evaluation scope, but has operational experience data. A PFM input data set is generated by the previously described calibration approach.
(2) Selection of one or several validation evaluation scopes, which are slight modifications of the reference scope, and have also related data in the operational experience. The calibrated input data set of the reference scope is modified to match the validation scopes, and the computed failure frequencies are compared to the results of the operational experience data. If they are not in agreement which each other, an improvement of the calibrated reference case is necessary.
(3) Modification of the reference case to match the actual evaluation scope, and computation of the failure frequencies.
(4) Sensitivity study to investigate the different possible input data sets and their effect to the failure frequency.

In Fig. I–13, the pure application case is represented by the bottom line, from the application evaluation scope to the resulting frequencies. However, the previous steps are necessary to construct the input data for the PFM computation. Typical validation cases can be constructed by, for example, varying the piping size, where the same database can be reused.

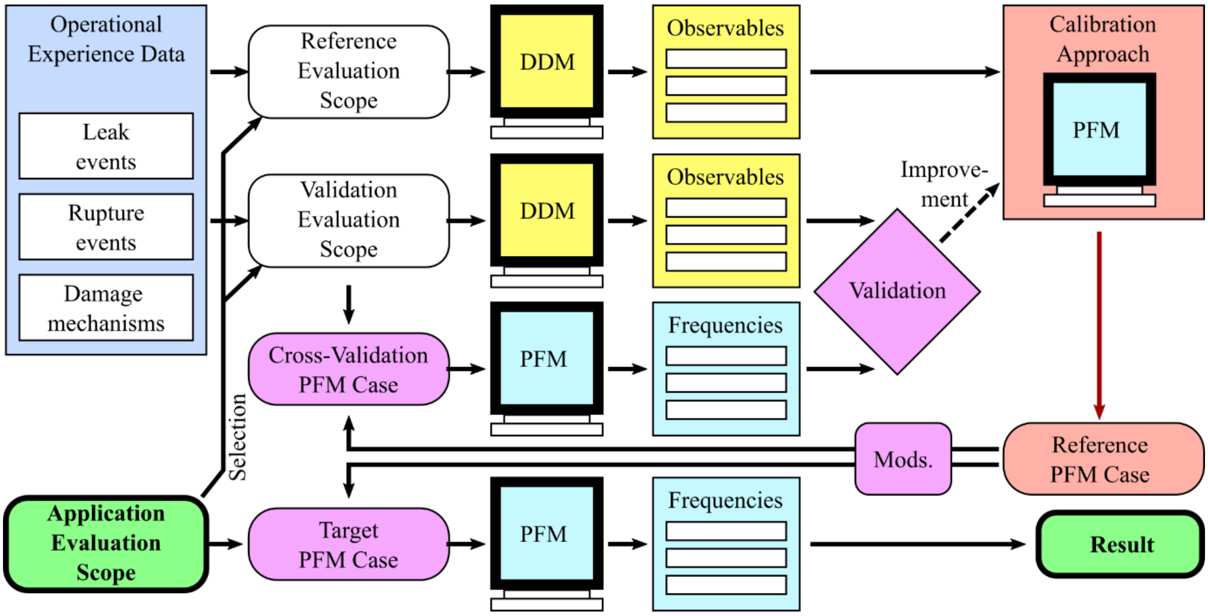


FIG. I–13. Schematics of calibration, validation and application process.

The approach also considers the comparison of different possible model assumptions consistent with the operational experience and knowledge about the system and the associated assessment of the predicted failure rates. This leads in turn to a probability distribution for the failure rates, which allows one to estimate the range of failure rate expected for the application evaluation scope. One can expect that, in the case of little information about the system, this range of possible failure rates will be very wide, and the prediction very uncertain. The validation offers the opportunity to narrow the range of parameters as far as possible, and to quantify effectively the uncertainties of the calibration cross-validation approach.

I–3. IMPLEMENTING THE INTEGRATED PHYSICS-OF-FAILURE METHODOLOGY

In the I-PPoF methodology, an integrated model of piping reliability is built with an explicit consideration of the physics-of-failure of material degradation from initiation to propagation, and RIM processes to detect and repair a pipe flaw and leak before developing into a significant structural mode of failure [I–30]. The PoF module (in Fig. I–14(b)) simulates the physical degradation and failure mechanisms and accounts for the underlying controlling parameters of a given material degradation and failure mechanism. The RIM processes are modelled by the maintenance work process (MWP) module (in Fig. I–14(c)) to estimate the maintenance performance metrics, for example, timing and probability of successful repair and replacement of a detected flaw or leak. The renewal process module (in Fig. I–14(a)) computes probabilistic failure metrics of a piping component (e.g. annual and cumulative probabilities of flaw, leak and rupture) using outputs from the PoF and MWP modules.

To simulate the interactions between physical degradation and failure mechanisms and RIM processes, a combination of different types of multi-state Markov models have been used, for example, discrete-time and continuous-time Markov models. Connecting the PoF, MWP and renewal process modules is done by using the interface module (in Fig. I–14(d)).

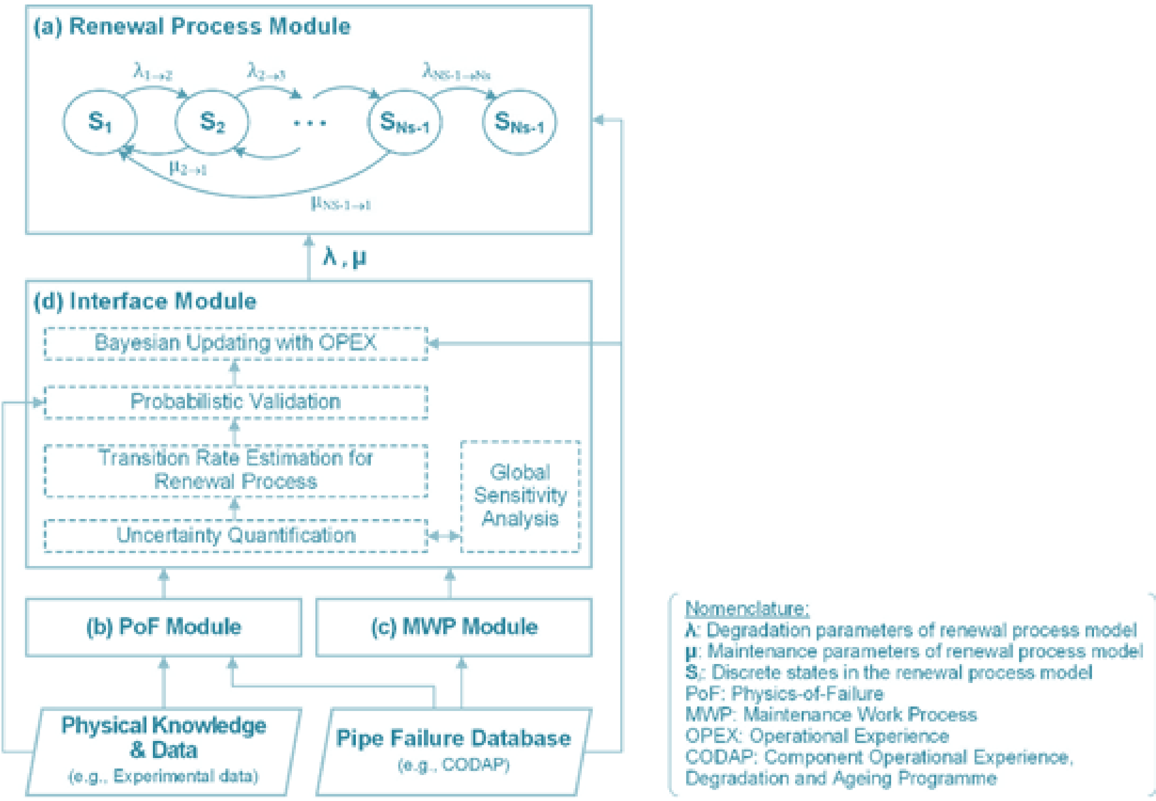


FIG. I–14. Integrated probabilistic physics-of-failure methodological framework.

The interface module performs uncertainty analysis considering two types of uncertainties: the aleatory uncertainties induced by the natural and inherent variability are quantified by uncertainty quantification, while the epistemic uncertainties originated from the incomplete or lack of knowledge about the phenomena and process are analysed by the probabilistic validation. If relevant OPEX data is available, the simulation based estimations of the probabilistic failure metrics are updated as needed using Bayesian techniques. An implementation of the I-PPoF methodology consists of the following activities:

— Developing a renewal process model of piping reliability;

— Estimating the pipe material degradation parameters of the renewal process model;
— Estimating the RIM parameters of the renewal process model;
— Calculating the probabilistic pipe failure metrics;
— Performing probabilistic validation;
— Conducting Bayesian updating of the simulation based estimations with OPEX; and
— Conducting a global sensitivity analysis to rank the input parameters.

I–3.1 Developing a renewal process model of piping reliability (step 1)

In renewal process models, the occurrence and timings of multiple types of events (e.g. physical degradation with various damage states and RIM activities) can be treated within a coherent stochastic modelling framework. In the I-PPoF applications to piping components, a multiphase Markov process model [I–31] is utilized, considering two phases: operation and inspection. In the operation phase, the component is subjected to physical degradation and leak monitoring continuously. Meanwhile, the inspection phase is entered when periodic flaw or leak inspection takes place at discrete times t_q ($q = 1, 2, \ldots, N_m$) during the lifetime of the piping component, where N_m represents the total number of inspection events. In this context, the time variable t represents the operating time (rather than the real clock time); hence, whenever the plant is shut down or the piping is isolated, the corresponding period should not be counted toward t. It is assumed that, in the multiphase Markov process model for piping components, the possible damage states are represented by discrete states: new, flaw, leak and rupture. The piping component is assumed to be free from degradation during inspection and repair, considering that (i) ISI and the subsequent repair of piping are usually during plant outages; and (ii) the inspection and repair/replacement time is typically much shorter than the operating time.

Figures I–15 and I–16 show an example of a multiphase Markov process model developed for a piping component. For the initial condition of the multiphase Markov model, it is assumed that the component begins in an as-new state; thus, the state distribution at $t = 0$ is provided as [1, 0, 0, 0]. During the operation phase, physical degradation takes place as represented by a continuous time Markov model shown in Fig. I–15.

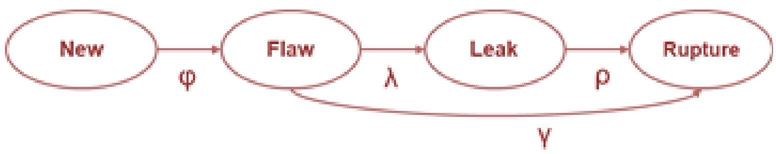


Fig. I–15. Multiphase Markov model for operation phase without RIM: φ and λ are degradation transitions, ρ and γ are degradation transition rates for the transition paths to rupture.

The impact of inspection and the resultant repair/replacement is incorporated into the inspection phase of the multiphase Markov process model shown in Fig. I–16. At $t = t_q$, when each RIM activity is applied, the repair or replacement process due to inspection (non-destructive examination or visual inspection) is evaluated.

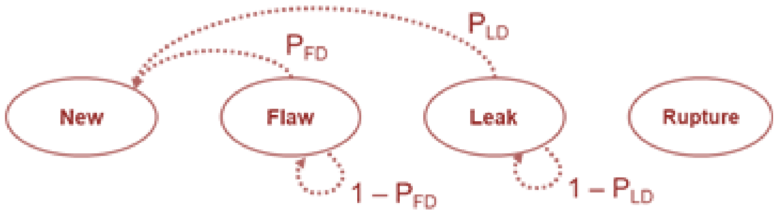


FIG. I–16. Multiphase Markov model for inspection phase with RIM: P_{FD} and P_{LD} represent the RIM transition probabilities.

The specific piping segment being analysed is assumed to be subject to an ISI programme, so there is no consideration of the possibility that the component is not inspected during each inspection cycle. Additionally, perfect repair is assumed (i.e. given that a flaw or leak is detected in the inspection, the component is restored to 'as-new' with certainty). If the flaw or leak is not detected by the inspection, it is assumed that the component remains in its state at $t = t_q$.

I–3.2 Estimating the pipe material degradation parameters of the renewal process model (step 2)

The material degradation transition rates in the renewal process model are estimated based on a physics-of-failure analysis by conducting two sub-steps: (2.1) development of the PoF models, and (2.2) uncertainty quantification and transitions rates estimation with consideration of aleatory uncertainties.

Development of physics-of-failure models: The physics-of-failure module predicts the progression of material degradation using physics based models, which explicitly incorporate (a) physical knowledge associated with the dominant failure mechanisms, and (b) the underlying physical parameters that can affect material degradation. In support of the CRP benchmark, physics-of-failure modules were developed for two cases: (i) thermal fatigue due to thermal stratification, and (ii) primary water SCC in a dissimilar metal weld consisting of nickel base alloys 52/152. The physics-of-failure module incorporates the physical degradation mechanism and associated physical contributing factors, for instance,

- The pipe's outside diameter and wall thickness.
- Young's modulus (i.e. the relationship between stress and strain in a material in the linear elasticity regime of a uniaxial deformation).
- Poisson ratio (i.e. the expansion or contraction of a material in directions perpendicular to the direction of loading).
- Yield stress (i.e. the value of stress at a yield point). The yield point is the point on a stress–strain curve that indicates the limit of elastic behaviour and the beginning of plastic behaviour.

As outputs of the physics-of-failure module, the physical key performance measures *Y*, representing the degree of progression of material degradation or component failure processes (e.g. a crack size) are predicted. Assuming there is one dominant physical degradation mechanism, the physics-of-failure module can be represented by a functional form:

$$\frac{dY}{dt} = g(t; \mathbf{x}, \boldsymbol{\eta}) \tag{I–9}$$

with the initial condition:

$$Y_0 \sim f_{Y_0}(y_0) \tag{I–10}$$

and where $\mathbf{x}$ represents the input parameters of the physics-of-failure models, representing physical factors that can initiate or accelerate the degradation mechanisms, such as system geometry, material properties and operational conditions, while $\boldsymbol{\eta}$ denotes the coefficients of correlation based physical models; and f_{Y_0}: a distribution function for the initial value of Y, denoted by Y_0. By numerically integrating Eq. (I–9) with respect to time over the component lifetime considering the initiation condition in Eq. (I–10), the time profile of the physical key performance measure $Y(t;\mathbf{x},\boldsymbol{\eta})$ is predicted.

As an example, Fig. I–17 illustrates the PoF module developed for thermal fatigue due to cycling and stratification [I–32]. As shown in this figure, this module consists of: (i) a mechanistic model for thermal-stress analysis developed in the finite element analysis software ABAQUS, (ii) a crack initiation model based on an *S-N* curve executed in the MATLAB code, and (iii) a crack propagation model composed of a stress intensity factor model assuming a semi-elliptical axial crack subjected to hoop stress on the infinite plate (mode I behaviour) and a crack growth rate model using Paris' law, executed in the MATLAB code. The PoF module was applied for two different piping configurations; one before recognition of the susceptibility to thermal fatigue, and one after implanting a design change to mitigate future failures.

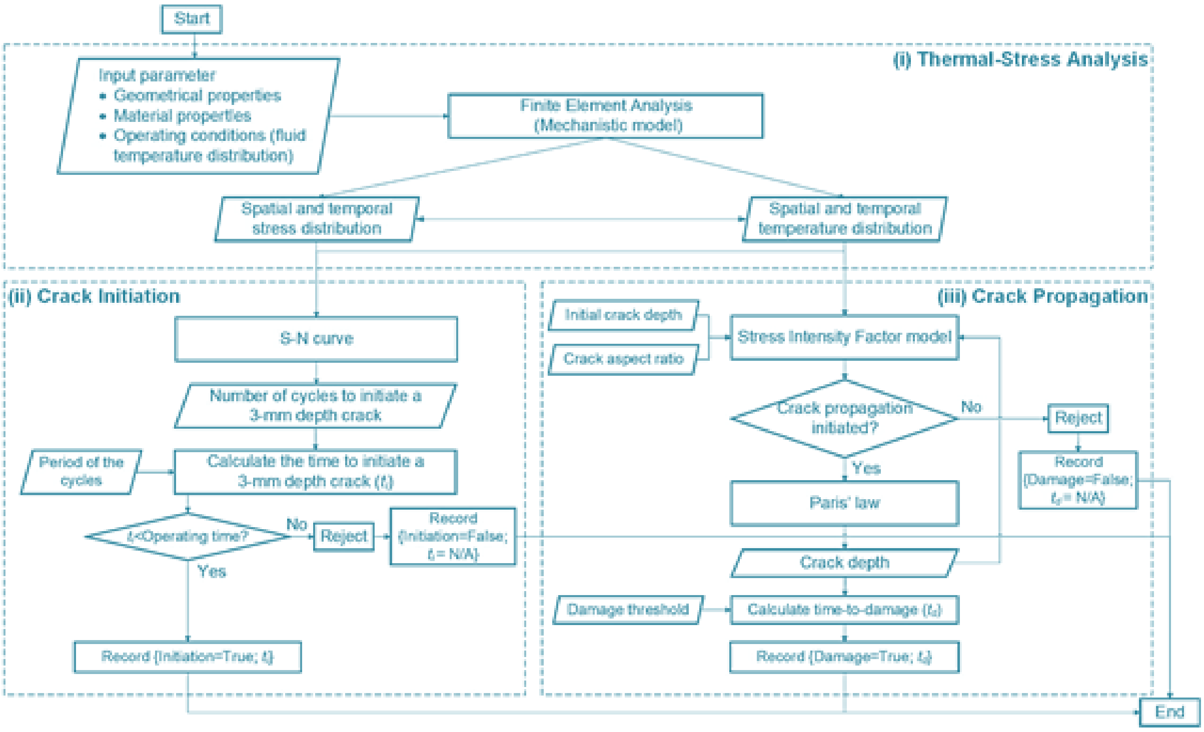


FIG. I–17. Flow chart for the physics-of-failure module for thermal fatigue.

The mechanistic model for stress and temperature analysis was developed using ABAQUS with a coupled thermal stress analysis and ABAQUS/standard procedure [I–33]. The finite element analysis calculates the spatial and temporal stress distribution induced by the fluid temperature fluctuation given initial and boundary conditions such as spatial and temporal distribution of the fluid temperature around the interface of the thermal stratification layer. The spatial and temporal von Mises stress distribution is compared with the yield stress of piping material to determine the zones of plastic deformation, where there is a higher probability of crack initiation and propagation. A location in the zones of plastic deformation that has the highest von Mises stress is identified. The maximum hoop stress range (ΔS) at this location can be calculated by considering the stress variation within one period of the cycle from the results of finite element analysis. The temperature and stress outputs from finite element analysis are used as input to the crack initiation and propagation models.

In the crack initiation model, an *S-N* curve is used to estimate the time to crack initiation as a function of the temperature and stress range predicted by finite element analysis. If the estimated time to crack initiation is within the component lifetime, the occurrence of crack initiation (Initiation = True) and the time to crack initiation are recorded; otherwise, Initiation = False is recorded.

In the crack propagation model, the stress intensity factor is calculated using the stress and temperature distributions predicted by finite element analysis as input. The calculated stress intensity factor is used as input to the crack growth rate model based on Paris' law. The time profile of the crack depth is computed by repeatedly updating the stress intensity factor and Paris' law calculation over time, using the predefined initial crack depth assumed in the crack initiation model as an initial condition. In the PoF module, multiple damage/failure states, corresponding to the discrete states considered in the MP Markov process model, are evaluated. Each damage/failure state (e.g. Flaw, Leak) is characterized by comparing the crack depth with the predefined threshold value (e.g. 15% of the thickness for Flaw state, and 100% wall thickness for Leak state). The time when the physical key performance measure reaches the threshold value for each damage/failure state is recorded.

Uncertainty quantification and degradation transition rate estimation (Sub-step 2.2): The aleatory uncertainties associated with the PoF input variables are propagated by conducting uncertainty quantification in the interface module (Fig. I–14(d)), which makes the PoF module probabilistic, generating the probabilistic PoF analysis. The uncertainty quantification for the PoF module is performed by running the Monte Carlo simulation, where the input parameters **x** are randomly sampled from their associated probability distributions, and the PoF module is evaluated repeatedly with each set of randomly sampled **x**. As a result, random samples of *Y* as a function of time are generated:

$$Y^{(j)}(t) = Y\left(t; \mathbf{x}^{(j)}, \boldsymbol{\eta}\right); \; j = 1,2,\ldots,N_S, 0 < t < t_l, \qquad \text{(I–11)}$$

where $\mathbf{x}^{(j)}$ is the *j*th set of random samples of **x**; N_S is the sample size; and t_l is the component lifetime. Then, using the uncertainty quantification outputs, the degradation transition rates are estimated. In the MP Markov process model, the sojourn times are identically and independently distributed in exponential distributions with constant rates. The point estimate of the transition rate between states $s_i \rightarrow s_{i'}$ is computed as follows:

$$\hat{\lambda}_{i \rightarrow i'} = n(s_i \rightarrow s_{i'})/T_{s_i} \qquad \text{(I–12)}$$

where T_{s_i} denotes the summation of the duration for which the Monte Carlo samples stayed in state s_i, and $n(s_i \rightarrow s_{i'})$ is the number of transitions from s_i to $s_{i'}$ observed in the PoF runs.

To find an adequate sample size leading to converged Monte Carlo estimations, a convergence study is conducted. The degree of convergence of the Monte Carlo simulation is measured by the relative half-width of the confidence intervals normalized by the point estimates of the degradation transition rates:

$$\frac{\delta}{\hat{M}_J} \leq \frac{r}{(r+1)}, \; J = N_{S,0}, N_{S,0} + \Delta_N, N_{S,0} + 2\Delta_N, \ldots \qquad \text{(I–13)}$$

where $\hat{M}_J$ is the point estimate of the transition rate *M* based on the Monte Carlo simulation with sample size of J, δ is the relative half-width of the 95% confidence intervals of the point estimate of each transition rate, and *r* is the desired relative error [I–34]. The 95% confidence intervals are developed using the bootstrap method [I–35, I–36]. Solving Eq. (I–13) is repeated by increasing the sample size with a predefined increment Δ_N, starting from the initial sample size $N_{S,0}$ until the desired level of relative error is achieved.

I–3.3 Estimating the RIM parameters of the renewal process model (step 3)

In step 3, the RIM related transition rates in the renewal process module, $\boldsymbol{\mu}$, are estimated by the MWP module (in Fig. I–14(c)). The MWP module explicitly models the RIM processes (e.g. ISI, leak monitoring, repair/replacement), with consideration of multiple layers of underlying performance contributing factors, such as training, procedures and availability of tools and resources. Maintenance scenarios are developed using human reliability analysis decision trees [I–37], including maintenance phases (i.e. ISI programme coverage, flaw and leak detection by non-destructive examination techniques, leak indication from leak monitoring systems, repair/replacement) as the pivotal events. The probabilities of the pivotal events are then quantified either by a DDM or an existing human reliability analysis method. For instance, for the flaw detection by non-destructive examination, a DDM is used for the probability estimation, where an analytical relationship between $\boldsymbol{\mu}$ and the maintenance characteristics estimated from empirical data (i.e. an inspection interval (T_{I}) and the probability of flaw detection (P_{FD}), is used). Meanwhile, for the leak detection by the leak monitoring systems, the top event probabilities are quantified using an existing human reliability analysis method [I–37].

As an example, illustrated in Fig. I–18 is an RIM decision tree developed for the analysis of leak detection. This event tree analyses the RIM scenarios where the leak monitoring systems detect leakage that exceeds the limiting condition for operation (LCO) limits for primary system operational leakage. As an example, according to the standard technical specifications for PWR reactors, there are two primary ways that the response to leakage is initiated. In the first path, unidentified leakage in the form of an imbalance in the RCS water inventory exceeds 6.3×10^{-2} kg/s and is not brought below 6.3×10^{-2} kg/s within 4 h. If this LCO is not satisfied, the plant is required to move to lower pressure conditions by entering Mode 3 within 6 h and Mode 5 within 36 h (i.e. an unplanned shutdown occurs). This shutdown prompts an inspection, and the RIM activities are initiated.

In the second path to leak action, the unidentified RCS operational leakage does not exceed 6.3×10^{-2} kg/s, but upon inspection, the leakage is identified as a primary pressure boundary leakage. In this inspection, the source of leakage is to be located and evaluated to meet the ASME standards. Additionally, an evaluation of the structural integrity of the pressure boundary is performed. The LCO for reactor coolant pressure boundary leakage requires zero leakage at all times during operation. In either leak action path, the repair plans for the leaking component of size x will initiate if the LCO action statement cannot be satisfied.

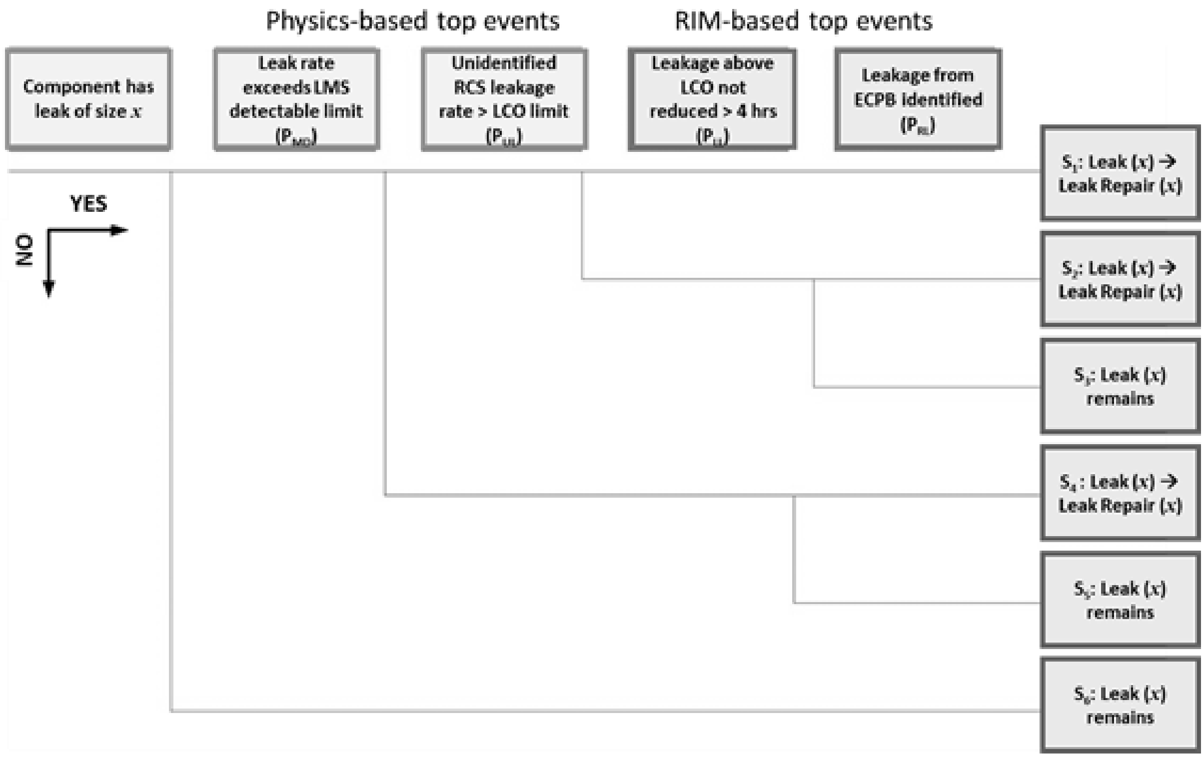


FIG. I–18 Decision tree for the physics-of-failure module.

The RIM decision tree in Fig. I–18 begins with the physics based initiating event 'component has a leak of size *x*', followed by two physics based top events and two RIM based top events. In the first top event, it is to be determined whether the leak rate is detectable by the leak monitoring systems or to be defined as undetectable leakage. If the leakage is detectable, then the next top event specifies whether the leakage exceeds the unidentified leakage limit (6.3×10^{-2} kg/s). If the leakage rate exceeds 6.3×10^{-2} kg/s, then the operators would have 4 h to reduce the leakage amount below this threshold or an unplanned shutdown will occur. However, if the leakage is identified as a primary pressure boundary leakage, then the component will need to undergo leak repair, regardless of the success or failure of previous top events. As a result of the success or failure of each top event, the component will either undergo leak repair following an unplanned shutdown (scenarios S_1, S_2, S_3, S_4) or the leak will remain (scenarios S_5 and S_6). The quantification of the RIM event tree in Fig. I–17 requires the following input parameters:

— *Probability of leak rate exceeding detectability limit (P_{MD}):* This value is dependent on the calculated leak rate from the PoF model and its comparison to the minimum detectable leak rate of 6.3×10^{-2} kg/s. This probability is already addressed as part of the degradation transition rates in the renewal process module; thus, it is not considered in the MWP module.
— *Probability of unidentified leakage rate exceeding LCO limit (P_{UL}):* This value is dependent on the calculated leak rate from the PoF model and its comparison to the maximum unidentified leakage rate for RCS leakage of 6.3×10^{-2} kg/s. This probability is already addressed as part of the degradation transition rates in the renewal process module; thus, it is not considered in the MWP module.
— *Probability of unidentified leakage rate not being reduced below LCO limit for >4 h (P_{LL}):* This value is dependent on the feasibility of isolating the leaking component to reduce leakage to below the LCO limit within 4 h. By definition, reactor coolant pressure boundary leakage is leakage from a non-isolable crack in the RCS piping, so for our case study we assume that there is no possibility of reducing the leakage within 4 h. To achieve this RIM action, reliability of

the leak monitoring systems and human performance of operators in diagnosing and responding to the leak detection signal should be considered. The MWP module uses the integrated human event analysis system framework [I–37] to quantify the human error probability for operator action upon the leak detection signal from the leak monitoring systems.

— *Probability of identifying leakage as reactor coolant pressure boundary leakage (P_{RI}):* The operators may conduct an inspection to identify the source of unidentified leakage that does not exceed the unidentified leakage LCO limit. This top event highly depends on operator action, and the associated human error probability can be estimated using the integrated human event analysis system method.

The following equation is used to calculate the conditional probability of the leak of size x undergoing leak repair as in scenarios S_1, S_2 and S_4:

$$P(\text{Leak}[x] \rightarrow \text{New}) = P_{MD}P_{UL}P_{LL} + P_{MD}P_{UL}(1 - P_{LL})P_{RI} + P_{MD}(1 - P_{UL})P_{RI} \quad \text{(I–14)}$$

The corresponding RIM transition rate in the Markov process model (namely, the transition rate from Leak to New) is then computed by dividing the state transition probability in Eq. (I–14) by the mean-time-to-repair.

I–3.4 Calculating probabilistic failure metrics (step 4)

The renewal process model is solved numerically using the input parameters estimated based on the outputs from the PoF and MWP modules. Solving the Markov model equations produces the probabilistic failure metrics of interest, such as the cumulative and annual probabilities of the damage state at a specific time. As an example, for the MP Markov process model structure, a set of differential equations, Eqs (I–15) to (I–18), is used to calculate the state distribution over time. The Markov process model can be formulated as a time inhomogeneous Markov process with delta function terms representing the discrete maintenance events:

$$\frac{dN(t)}{dt} = -\varphi N(t) + \sum_{i=1}^{N_m}[L(t)P_{LD}\,\delta(t - t_i) + F(t)P_{FD}\,\delta(t - t_i)] \quad \text{(I–15)}$$

$$\frac{dF(t)}{dt} = \varphi N(t) - (\gamma + \lambda)F(t) - \sum_{i=1}^{N_m}[F(t)P_{FD}\,\delta(t - t_i)] \quad \text{(I–16)}$$

$$\frac{dL(t)}{dt} = \lambda F(t) - \rho L(t) - \sum_{i=1}^{N_m}[L(t)P_{LD}\,\delta(t - t_i)] \quad \text{(I–17)}$$

$$\frac{dR(t)}{dt} = \rho L(t) + \gamma F(t) \quad \text{(I–18)}$$

where:

$N(t)$ probability of ‘New’ state at time t;
$F(t)$ probability of ‘Flaw’ state at time t;

$L(t)$ probability of 'Leak' state at time t;
$R(t)$ probability of 'Rupture' state at time t;
P_{FD} probability of detection of 'Flaw' state;
P_{LD} probability of detection of 'Leak' state;
$\delta(\cdot)$ Dirac delta function;
N_m total number of periodic ISI events.

In the above equations, the physical degradation is modelled by the terms containing the degradation transition rates $(\varphi, \lambda, \gamma, \rho)$, while the ISI events are modelled by the terms including the Dirac delta function. Note that the transition only occurs if the component is detected to be in the Flaw or Leak state, so the transition is dependent on the value of P_{FD} and P_{LD}. These equations are solved from $t = 0$ until $t = t_l$ (the end of the component's life) to compute $\{N(t), F(t), L(t), R(t)\}, t \in (0, t_l]$.

I–3.5 Performing probabilistic validation (step 5)

To probabilistically quantify the degree of validity of the estimated probabilistic failure metrics, a probabilistic validation methodology is applied. Sakurahara et al. [I–38] proposed the probabilistic validation methodology as a new paradigm of validation strategy for modelling and simulation used in PSA. Beal et al. [I–39] showed how the probabilistic validation methodology can be implemented to support risk-informed analysis for advanced reactors using pipe reliability analysis for advanced WCRs as a case study. In probabilistic validation, the degree of validity of model predictions is measured by the epistemic uncertainty. When the epistemic uncertainty for the model output is small enough to derive risk-informed decision making with a sufficient level of confidence, the validity of the model prediction is considered to be acceptable; otherwise, the current degree of validity is considered to be not sufficient, and models and input data need to be refined to reduce their epistemic uncertainties. As an acceptance threshold in probabilistic validation, probabilistic acceptance criteria for pipe failure frequencies can be utilized. The probabilistic validation methodology is implemented in three steps: (i) identification, (ii) characterization, and (iii) propagation of epistemic uncertainties.

***Identification of the dominant sources of epistemic uncertainty*:** This sub-step consists of qualitative analysis (i.e. identification of potential sources of epistemic uncertainty) and quantitative screening (i.e. sensitivity analysis for the potential sources of epistemic uncertainty to screen non-influential ones). For the qualitative analysis, three categories of epistemic uncertainties are considered: (a) epistemic uncertainty related to the selection of probability distributions and estimation of the distribution parameters for random input variables, (b) epistemic uncertainty related to the model selection and assumptions, and (c) epistemic uncertainty related to the statistical convergence of probability estimation. Based on a careful review of the input parameters and the constituting models, potential sources of epistemic uncertainties existing in the I-PPoF framework are identified. For the quantitative screening, a sensitivity analysis is conducted at the level of the outputs from individual modules in I-PPoF. The Morris elementary effect method is suggested for quantitative screening of the potential sources of epistemic uncertainties [I–39]. Based on the results of the quantitative screening, the epistemic uncertainty sources that have negligible impacts on the model outputs are identified and excluded from further analysis.

***Characterization of each source of epistemic uncertainty*:** The epistemic uncertainty for each of the sources retained in sub-step 5.1 is characterized using probabilistic measures, such as a probability density function or upper and lower bounds corresponding to specific percentiles. For epistemic uncertainty related to the selection of probability distributions and estimation of the distribution parameters for random input variables, it is a common practice to develop continuous probability distributions based on various data sources, such as experimental data, field data and engineering judgement. For epistemic uncertainty related to the model selection and assumptions, one possible approach is to choose multiple alternative models (e.g. altering the key assumptions in the models) and assign a probability weight to each model based on the degree of confidence by an analyst. The set of the alternative models can then be characterized as a discrete probability distribution. For epistemic uncertainty related to the statistical convergence of probability estimation, the confidence intervals associated with the sampling based estimations of the statistical quantities (e.g. degradation transition rates in the renewal process model) are constructed in order to characterize the epistemic uncertainty associated with statistical inference based on a limited sample size.

***Propagation of the sources of epistemic uncertainty*:** The sources of epistemic uncertainty, identified and characterized in sub-steps 5.1 and 5.2, are propagated up to the level of the probabilistic failure metrics estimations. I-PPoF utilizes a nested Monte Carlo technique to propagate both aleatory and epistemic uncertainty in a coherent structure. The inner loop of the Monte Carlo simulation, conducted as a part of sub-step 2.2 (uncertainty quantification and transitions rate estimation), propagates aleatory uncertainties, while the outer loop of the Monte Carlo simulation, conducted in this sub-step for probabilistic validation, propagates epistemic uncertainties. For the outer loop, the sources of epistemic uncertainty characterized in sub-step 5.2 are randomly sampled, and the simulation runs for I-PPoF are repeated using each set of sampled random variables. The outputs from the nested Monte Carlo simulation are random samples of the estimates of the probabilistic failure metrics (e.g. frequencies of leak and rupture). These outputs can be used to compute the statistics characterizing the epistemic uncertainty associated with the probabilistic failure metrics estimations.

I–3.6 Conducting Bayesian updating of the simulation based estimation with OPEX

If any relevant OPEX is available, the simulation based estimations of the probabilistic failure metrics can be updated with the OPEX by conducting Bayesian parameter estimation. In this way, the resulting probabilistic failure metrics reflect the totality of data available including OPEX and the simulation data generated from the I-PPoF framework. A generic formulation of the Bayesian updating in this step is given as follows:

$$\pi(\mathbf{p}|E_{\mathrm{sim}}) = \frac{L(E_{\mathrm{sim}}|\mathbf{p})\pi_0(\mathbf{p})}{\int_{\mathbf{p}} L(E_{\mathrm{sim}}|\mathbf{p})\pi_0(\mathbf{p})\mathrm{d}\mathbf{p}} \tag{I–18}$$

where $\mathbf{p}$ is the probabilistic failure metrics being estimated, $\pi_0(\mathbf{p})$ is the prior distribution for $\mathbf{p}$, E_{sim} is the simulation based estimations of $\mathbf{p}$ obtained from the renewal process module integrated with the PoF and MWP modules, and $L(E_{\mathrm{sim}}|\mathbf{p})$ is the likelihood function for E_{sim}. $\pi_0(\mathbf{p})$ is constructed based on the relevant OPEX, such as the probability distribution for $\mathbf{p}$ obtained from the DDM analysis. The parameter $L(E_{\mathrm{sim}}|\mathbf{p})$ represents the uncertainty associated with the simulation based estimations and constructed based on the results of probabilistic validation. An additive error model (i.e. a normal distribution) or a multiplicative model (i.e. a lognormal distribution) can be used. The standard deviation of the additive or multiplicative error model is estimated based on the uncertainty bounds obtained from probabilistic validation, for instance, matching the 100(1–α/2)th and 100(α/2)th percentiles to the upper and lower bounds (where α denotes the level of significance).

I–3.7 Conducting global sensitivity analysis to rank the input parameters

In this step, global sensitivity analysis is conducted to study the sensitivity of the probabilistic failure metrics estimations to the input parameters of I-PPoF, including physical input (such as material properties, operating conditions and geometry) and RIM-related input (such as RIM policy, probability of detection, PIFs). The moment-independent, cumulative distribution function based method $S_i^{(cdf)}$ [I–40] is used to account for key characteristics of the integrated models, including uncertainties associated with the model input and output as well as non-linearity and interactions inside the model. As a measure of sensitivity, $S_i^{(cdf)}$ uses the expected difference between the unconditional cumulative distribution function of the model output and the conditional cumulative distribution function of the model output given each input parameter is fixed. $S_i^{(cdf)}$ is defined as follows:

$$S_i^{(\mathrm{cdf})} = E_{X_i}[A(X_i)]/|E(Y)| \tag{I–19}$$

where X_i is the ith input parameter of the model, Y is the model output, $E(Y)$ is the expected value of Y and assumed to be non-zero, while $A(X_i)$ is the area enclosed by the conditional and unconditional cumulative distribution functions and computed by:

$$A(X_i) = \int |F_{Y|X_i}(y) - F_Y(y)| \mathrm{d}y \tag{I–20}$$

In Eq. (I–20), $F_Y(y)$ is the unconditional cumulative distribution function of the model output, while $F_{Y|X_i}$ is the conditional cumulative distribution function of the model output, given that the input parameter X_i is fixed. The estimation of $S_i^{(cdf)}$ can be executed by using a two-loop Monte Carlo simulation. Examples of the global sensitivity analysis for I-PPoF are reported in [I–38] and [I–39]. The latter reference implemented the global sensitivity analysis to rank the RIM parameters in the MWP module based on their contribution to the cumulative rupture probability. The RIM parameters analysed in [I–40] include intervals and probability of detections of flaw and leak inspections, time to repair or replacement for a detected flaw and leak, and human error probabilities for RIM activities for a flaw and leak.

REFERENCES

[I-1] ATWOOD, C.L., et al., Handbook of Parameter Estimation for Probabilistic Risk Assessment, NUREG/CR-6823, U.S. Nuclear Regulatory Commission, Washington, DC (2003).

[I-2] OECD NUCLEAR ENERGY AGENCY, Operating Experience Insights into Below Ground Piping at Nuclear Power Plants, NEA/CSNI/R(2018) 2, Boulogne-Billancourt, France (2018).

[I-3] GUPTA, A.K., NADARAJAH, S. (Eds.) Handbook of Beta Distributions and Its Applications, Marcel Dekker, NY (2004).

[I-4] LYDELL, B., CHRUN, D., MOSLEH, A., "Enhanced Piping Reliability Models for Use in Internal Flooding PSA", (Proc. ANS PSA-2011 Conf., American Nuclear Society, LaGrange Park, IL, 2011), Paper #145.

[I-5] PEARSON, E.S., TUKEY, J.W., Approximate means and standard deviations based on distances between percentage points of frequency curves, Biometrika **52** (1965) 533–546.

[I-6] TREGONING, R., ABRAMSON, L. SCOTT, P., Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process, NUREG-1829, U.S. Nuclear Regulatory Commission, Washington, DC (2008).

[I-7] FLEMING, K., Markov models for evaluating risk-infomed in-service inspection strategies for nuclear power plant piping systems, Reliab. Eng. Syst. Saf. **83** (2004) 27–45.

[I-8] FLEMING, K., LYDELL, B., Database development and uncertainty treatment for estimating pipe failure rates and rupture frequencies, Reliab. Eng. Syst. Saf. **86** (2004) 227–246.

[I-9] LYDELL, B., “The Probability of Pipe Failure on the Basis of Operating Experience”, (Proc. ASME Pressure Vessel & Piping Division Conf.), PVP-26281, ASME, New York (2007).
[I-10] LYDELL, B., OLSSON, A., Reliability Data for Piping Components in Nordic Nuclear Power Plants: The ‘R-Book’ Phase I, SKI Report 2008:01, Swedish Radiation and Safety Authority, Stockholm, Sweden (2008).
[I-11] LYDELL, B., A review of the progress with statistical models of passive component reliability, Nucl. Eng. Technol. **49** (2017) 349–359.
[I-12] NUGENIA ASSOCIATION, ENIQ Recommended Practice 9: Verification and Validation of Structural Reliability Models and Associated Software to be used in Risk-Informed In-Service Inspection Programmes, Issue 2, ENIQ Report No. 52, Brussels, Belgium (2017).
[I-13] RACKWITZ, R., Reliability analysis – A review and some perspectives, Struct. Saf. **23** (2001) 365–395.
[I-14] SCHULZ, H., et al., Nuclear Risk-Based Inspection Methodology for Passive Components (NURBIM), NURBIM Final Report, FIKS-CT-2001-00172, Fifth Framework of the European Atomic Energy Community (EURATOM), Brussels, Belgium (2004).
[I-15] SIMOLA, K., et al., Studies on the Effect of Flaw Detection Probability Assumptions on Risk Reduction at Inspection, NKS-208, ISBN 978-87-7893-277-8, Nordic Nclear Safety Research, Roskilde, Denmark (2009).
[I-16] DILLSTRÖM, P., ZANG, W., ProLBB – A Probabilistic Approach to Leak Before Break Demonstration, SKI Report 2007:43, Swedish Nuclear Power Inspectorate, Stockholm, Sweden (2007).
[I-17] HECKMANN, K., MA, K., SIEVERS, J., “Probabilistic Aspects on Break Preclusion Assessment in Nuclear Piping”, Proc. 41st MPA-Seminar, Stuttgart (2015).
[I-18] BHIMANADAM, V., BLOM, F., “Probabilistic Leak Before Break Assessment Using Master Curve”, Transactions SmiRT23, Division II, Paper ID 312, Manchester, UK (2015).
[I-19] ELMAS, M., et al., Effectiveness of Measures to Ensure the Integrity of Pressure-Retaining Components in German Nuclear Power Plants, GRS-A-3700 (in German), GRS, Cologne, Germany (2013).
[I-20] HECKMANN, K., SIEVERS, J., “Integrity Assessment of Piping with the PROST Code”, Proc. 40th Enlarged Halden Programme Group (EHPG) Meeting, Lillehammer, Norway (2017).
[I-21] VEPSÄ, A., CROVALL, O., Comparison of Three Sampling Methods in the Context of Probabilistic Fracture Mechanic Analyses of NPP Piping Welds: Case Study, VTT-R-00152-10, VTT Technical Research Centre of Finland, Espoo, Finland (2010).
[I-22] HARRIS, D., DEDHIA, D., Lu, S., Theoretical and User’s Manual for pc-PRAISE, NUREG/CR-5864, U.S. Nuclear Regulatory Commission, Washington, DC (1992).
[I-23] RAHMAN, S., GHADIALI, N., PAUL, D., WILKOWSKI, G., Probabilistic Pipe Fracture Evaluations for Leak-Rate-Detection Applications, NUREG/CR-6004, Nuclear Regulatory Commission, Washington, DC (1995).
[I-24] SCOTT, P., KURTH, R., COX, A., OLSON, R., RUDLAND, R., Development of the PRO-LOCA Probabilistic Fracture Mechanics Code, MERIT Final Report, SSM-2010:46, Swedish Radiation and Safety Authority, Stockholm, Sweden (2010).
[I-25] DATTA, D., Development of an Advanced PFM Code for the Integrity Evaluation of Nuclear Piping System Under Combined Aging Mechanisms, Doctoral Thesis, Korea Advanced Institute of Science and Technology, Daedeok Innopolis, Daejeon, South Korea (2010).
[I-26] SCHIMPKE, T., Fatigue Benchmark Study, EURATOM FIKS-CT-2001-00172, NURBIM Final Report WP-4 Appendix B (2004).
[I-27] HECKMAN, K., SAIFI, Q., Comparative analysis of deterministic and fracture mechanical assessment tools, Kerntechnik **81** (2016) 484–497.
[I-28] TRUCANO, T.G., SWILER, L.P., IGUSA, T., OBERKAMPF, W.L., PILCH, M., Calibration, validation, and sensitivity analysis: What’s what, Reliab. Eng. Syst. Saf. **91** (2006) 1331–1357.
[I-29] STRÄHL, C., ZIEGEL, J., Cross-calibration of probabilistic forecasts, Electron. J. Stat. **18** (2017) 608–639.
[I-30] O’SHEA, N., MOHAGHEGH, Z., REIHANI, S., KEE, E., “Estimating loss-of-coolant accident (LOCA) Frequencies via Spatio-Temporal Methodology”, Proc. 13th Int. Conf. on Probabilistic Safety Assessment and Management (PSAM 13), International Association for Probabilistic Safety Assessment and Management (IAPSAM), Seoul, Korea, Oct. 2–7 (2016).
[I-31] BECKER, G., CAMARINOPOULOS, L., OHLMEYER, W., Discontinuities in homogeneous Markov processes and their use in modelling technical systems under inspection, Microelectron. Reliab. **34** (1994) 771–788.
[I-32] KAMAYA, M., Assessment of thermal fatigue damage caused by local fluid temperature fluctuation. Part I: characteristics of constraint and stress caused by thermal striation and stratification, Nucl. Eng. Des. **268** (2014) 121–138.
[I-33] DE, S., MANE 4240/ CIVL 4240: Introduction to Finite Elements, Abaqus Handout, Dept. Mechanical, Aerospace and Nuclear Engineering, Rensselear Polytechnic Institute, Troy, NY (2008).
[I-34] LAW, A. M., Simulation Modeling and Analysis, McGraw-Hill, New York (2014).
[I-35] EFRON, B., TIBISHIRANI, R.J., An Introduction to the Bootstrap. CRC Press LLC, Boca Raton, FL (1998).

[I-36] JANSSEN, H., Monte-Carlo based uncertainty analysis: Sampling efficiency and sampling convergence, Reliab. Eng. Syst. Saf. **109** (2013) 123–132.

[I-37] XING, J., CHANG, Y.J., SEGARRA, D.J., The General Methodology of an Integrated Human Event Analysis System (IDHEAS-G), NUREG-2198, U.S. Nuclear Regulatory Commission, Washington, DC (2021).

[I-38] SAKURAHARA, T., SCHUMOCK, G., REIHANI, S., KEE, E., MOHAGHEGH, Z., Simulation-informed probabilistic methodology for common cause failure analysis, Reliab. Eng. Syst. Saf. **185** (2019) 84–99.

[I-39] CHENG, W.C., et al., "Global Sensitivity Analysis to Rank Parameters of Stress Corrosion Cracking in the Spatio-Temporal Probabilistic Model of Loss of Coolant Accident Frequencies", Proc. Int. Topical Meeting on Probabilistic Safety Assessment and Analysis, PSA 2017, American Nuclear Society, Pittsburgh, PA, Sept. 24–28 (2017).

[I-40] BEAL, J., et al., "Coupling Degradation and Maintenance to Model Safety Risk and Financial Risk Under An Integrated Enterprise Risk Management Methodology for Nuclear Power Plants", Proc. 25th Int. Conf. on Structural Mechanics in Reactor Technology (SmiRT 25), Charlotte, NC (2019).

Annex II

ANALYSIS TOOLS AND TECHNIQUES

This Annex includes examples of analysis tools (i.e. computer applications) and techniques to support piping reliability analysis. Irrespective of the chosen method (DDM, PFM, I-PpoF), a piping reliability analysis includes the following common elements:

- Statistical analysis of experimental data and/or OPEX data on crack initiation and propagation.
- Propagation of uncertainty distributions through a model of piping reliability.
- Evaluation of the effect of different RIM strategies on the structural integrity of a piping pressure boundary.
- Specialization of piping reliability analysis results to account for plant specific application requirements. For example, determining the frequency of failure as a function of a certain through-wall flow rates or ranges of flow rates.

Open source software tools are available to facilitate piping reliability analysis. These software tools support statistical analysis, fracture mechanics analysis and multi-physics modelling. An analyst may prefer to use a combination of open source and proprietary software solutions to implement a piping reliability analysis framework. Examples of open source software tools include:

- WinBUGS. A statistical software for Bayesian analysis using Markov chain Monte Carlo methods. BUGS, which stands for 'Bayesian inference Using Gibbs Sampling', is concerned with a flexible software for the Bayesian analysis of complex statistical models. The project began in 1989 in the MRC Biostatistics Unit, Cambridge, and led initially to the 'Classic' BUGS programme, and then onto the WinBUGS software developed jointly with the Imperial College School of Medicine at St Mary's, London [II–1, II–2].
- Table II–1 provides the examples of PFM codes.

TABLE II–1. EXAMPLES OF PFM COMPUTER CODES

Name	Description	Year of software release	Access
PRAISE	PRAISE was initially developed for the assessment of the influence of seismic events on the failure probability of cracked piping in PWRs. The cracking mechanism originally considered by PRAISE was fatigue crack growth due to cyclic loading of pre-existing crack-like weld defects, introduced during the fabrication process. In the mid-1980s, the computer code was further enhanced to allow for the probabilistic treatment of the initiation and growth of intergranular SSC in sensitized weldments in Type 304 stainless steel piping in BWRs	1981	Open access, developed by the Lawrence Livermore National Laboratory with funding from the US NRC
pc-PRAISE	PRAISE for implementation on a personal computer	1992	Open access
Pipe Fracture Probabilities	Pipe FRActure Probabilities is meant for evaluation of the leak and rupture probabilities of a specific cross-section with a certain stress state and possibly containing a circumferential growing crack due to SCC. Detailed code description is found SKI Report 2000:48, The use of risk based methods for establishing ISI priorities for piping components at Oskarshamn 1 NPP, Swedish Radiation Safety Authority, Stockholm, Sweden, 2000	1997	Proprietary, originally developed by SAQ Kontroll AB of Sweden (now KIWA Inspecta AB)
Win-PRAISE	pc-PRAISE for Windows	1998	Developed by Engineering Mechanics Technology, Inc
SRRA	Structural Reliability and Risk-Assessment. A PFM code for assessing pipe component failure probabilities to support risk-informed ISI programme development analyses. Besides pipe components, SRRA can also be used for probabilistic integrity assessment of RPVs and RPV internals. According to the developer, SRRA has the capability to simulate the effect of a variety of time dependent material degradation mechanisms for components of carbon steel and stainless steels including low cycle fatigue crack growth of an existing (fabrication) flaw, crack growth of an existing flaw due to SCC, wall thinning due to material wastage (e.g. by flow induced corrosion), high cycle fatigue induced stresses exceeding the fatigue crack threshold.	1998	Proprietary, developed by Westinghouse
PROST	Probabilistic structural mechanics; developed by Global Research for Safety (GRS) to evaluate leak and failure probabilities of piping systems in NPPs. A graphical user interface supports the necessary data input. Leak and break probabilities from pre-existing semi-elliptical shaped inner surface cracks subjected to cyclic or static loading conditions can be estimated. The calculation of the subcritical crack growth and the final instability are based on deterministic fracture mechanics principles. The probabilistic nature is determined by the uncertainties of the input data entering the deterministic routines. Allows for probabilistic treatment of fatigue, intergranular SCC of austenitic stainless steel and strain induced corrosion cracking of ferritic steel. PROST allows for modelling of leakage rates	2002	Proprietary, developed by GRS

TABLE II–1. EXAMPLES OF PFM COMPUTER CODES (cont.)

Name	Description	Year of software release	Access
VTTBESIT	The code was originally intended for deterministic fracture mechanics based crack growth analyses, but it has been modified by adding probabilistic capabilities to the code. The probabilistically treated crack growth analysis input data parameters are depth of initial cracks; length of initial cracks; frequency of load occurrence	2007	Proprietary, developed by Fraunhofer-Institut für Werkstoffmechanik, Germany and the Technical Research Centre of Finland (VTT)
PASCAL-SP	As a part of research on the material degradation and structural integrity assessment for aged LWR components, a PFM analysis code PASCAL-SP (PFM Analysis of Structural Components in Aging LWR – SCC at Welded Joints of Piping) has been developed. This code evaluates the failure probabilities at welded joints of aged piping by a Monte Carlo method. PASCAL-SP treats SCC and fatigue crack growth in piping, according to the approaches of NISA and Japan Society of Mechanical Engineers fitness for service code. The reliability of flaw detection and sizing, and residual stress distributions are based on experimental data and introduced into PASCAL-SP	2009	Developed by Japan Atomic Energy Agency. Intended for a cross-check use by the regulatory body in Japan. This code can also be used for a research purpose by researchers in academia and industry
PRO-LOCA	Predicts the leak or break frequency for the whole sequence of initiation, subcritical crack growth until wall penetration and leakage, instability of the through-wall cracking (pipe rupture). The outcomes of the PRO-LOCA code are a sequence of failure frequencies which represent the probability of a surface crack developing, a through-wall crack developing and six different sizes of crack opening areas corresponding to different leak flow rates or LOCA categories. The code development was financed by an international consortium representing Canada, Sweden, Republic of Korea, USA and United Kingdom	2010	Restricted to the sponsoring organizations
PRAISE-CANDU	Developed by CANDU Energy Inc. for the CANDU Owners Group (COG), for structural integrity assessment of CANDU heat transport system piping. Degradation mechanisms that can be modelled include fatigue, SCC. The piping loads that are considered include normal and transient service loads, deadweight, environmental loads, seismic, vibratory and weld residual stresses	2012	Restricted to CANDU Owners Group Members
FAVOR	Failure Analysis of Vessels – Oak Ridge Version 16.1. A PFM code especially developed for the assessment of the structural integrity of reactor pressure vessels (FAVOR Users Guide, ORNL/LTR-2016/031 [II–3]). An overview of the code is found in PVP2017-65262. Developed at the Oak Ridge National Laboratory with funding from the US Nuclear Regulatory Commission. The most recent public release of FAVOR, v16.1, includes improvements in the consistency and accuracy of the calculation of fracture mechanics stress-intensity factors for internal surface-breaking flaws; special attention was given to the analysis of shallow flaws[a]	2016 (Version 16.1)	Publicly available, distribution is handled by NRC staff

Name | Description | Year of software release | Access

TABLE II–1. EXAMPLES OF PFM COMPUTER CODES (cont.)

PROMISE	Probabilistic Optimization of Inspection. An implementation of a probabilistic model of fatigue crack growth using linear elastic fracture mechanics (LEFM) methods, consistent with the flaw evaluation procedures of ASME XI	2018	Proprietary, developed by Structural Integrity Associates Inc
xLPR	The initial focus of the computer code development has been on evaluating pipe rupture probabilities within Alloy 82/182 dissimilar metal welds. For version 2, released in 2020, its core capabilities include modelling fatigue, SCC, ISI, chemical and mechanical mitigation, leakage rates and seismic effects. Developed at the Sandia National Laboratories with funding from US Nuclear Regulatory Commission and the Electric Power Research Institute[b]	2020	Open access, developed by the US NRC and the Electric Power Research Institute. Distribution is handled by NRC staff

[a] For more information, see: https://www.nrc.gov/about-nrc/regulatory/research/obtainingcodes.html#8
[b] For more information, see: https://www.nrc.gov/about-nrc/regulatory/research/obtainingcodes.html#8

REFERENCES

[II–1] LUNN. D. et al, The BUGS Book. A practical Introduction to Bayesian Analysis, CRC Press, Abingdon, UK (2013).

[II–2] KELLY, D.L., CURTIS, C.L., Bayesian inference in probabilistic risk assessment: The current state of the art, Reliab. Eng. Syst. Saf. **94** (2009) 628–643.

[II–3] DICKSON, T. L., WILLIAMS, P. T., BASS, B. R., KLASKY, H. B., Fracture Analysis of Vessels – Oak Ridge FAVOR, v16.1, Computer Code:User's Guide, ORNL/LTR-2016/310, Oak Ridge National Laboratory, Oak Ridge, TN (2016).

Annex III

SELECTED PIPING RELIABILITY DATA RESOURCES

This Annex discusses piping reliability data resources. A 'data resource' is a structured set of data held in a computer with a view to its utilization by subject matter experts (SMEs) in specific types of applications. Many different pipe failure data resources have been developed and with different objectives. The types of data resources range from piping OPEX databases ('event databases') to databases that include piping reliability parameters (e.g. tabulation of probabilistic risk metrics).

III–1 DIFFERENT TYPES OF DATA RESOURCES

There are three types of data resources:

(1) *Type 1*. Engineering databases that capture information on pipe failure events, including the underlying cause(s) of failure. Each event is classified according to a taxonomy that acknowledges chemical and mechanical properties of materials, environmental conditions, fabrication techniques, ISI technologies and material degradation mechanisms. The survey results are documented in the following tables:

— Nuclear industry (Table III–1)
  - Type 1.a: a database developed especially for piping reliability analysis tasks. Based on a detailed piping reliability taxonomy.
  - Type 1.b: a general purpose event database from which information on pipe failures can be extracted. The database includes information on pipe dimensions, material data and root cause of pipe failure.

— Non-nuclear industry (Table III–2).

(2) *Type 2*. General purpose OPEX databases from which information about pipe failures can be extracted. Typically, it requires a significant effort to analyse and classify the event information. The survey results are included in Table III–1.

(3) *Type 3*. Piping reliability parameter databases (or handbooks) that include tabulations of calculated pipe failure rates, including uncertainty distributions (Table III–3). This type of database is usually organized by the type of piping system, material, pipe size, degradation mechanism, etc. Application specific, reference parameter databases have been developed to support the following activities:

— Level 1 and Level 2 PSA;
— Internal Flooding PSA;
— Risk-informed ISI;
— RIM programme development; and
— Advanced WCR design certification PSA.

III–2 DATA ACCESS

Invariably the Type 1.a databases are either restricted or proprietary. As an example, CODAP is a restricted database. While CODAP project participation is open to any nuclear industry organization, access to the database is restricted to those organizations that actively participate in the data exchange process.

III–3 SOURCES OF PIPE FAILURE INFORMATION

The reporting of pipe failures, including degraded conditions, is done at different levels and in accordance with national regulations and codes and standards. Accessing the source data (or raw data) is subject to national regulations as well as the different national practices for the handling of restricted and proprietary information. Table III–4 provides an overview of different types of source (or raw) data in the USA.

TABLE III–1 SELECTED NPP PIPE FAILURE EVENT DATABASES

Database characteristic	Database Type and Name				
	Type 1.a	Type 1.a	Type 1.a	Type 1.b	Type 2
	OECD/NEA CODAP event database	PIPExp 'Pipe OPEX database'	KomPass	CHUG	IAEA/NEA incident reporting system
Accessibility	Restricted — database is located on the NEA secure server	Proprietary	Proprietary	Proprietary	Restricted database located on the IAEA secure server
Owner/ operator	CODAP project management board and with support of NEA-IT and CODAP operating agent	Sigma-Phase Inc., USA	GRS, Germany	Operated by the Electric Power Research Institute on behalf of the CHECWORKS User Group	Joint NEA and IAEA OPEX database
Database Scope	Piping (safety related and non-safety related) plus selected non-piping passive components (e.g. reactor internals)	Piping — safety related and non-safety related, water hammer event database, and piping component population database	Detailed information on the OPEX with pressure-retaining components in German NPPs	Sharing of OPEX involving flow accelerated corrosion. The CHUG web site enables uploading of event information, including photographs, drawings, etc.	General purpose OPEX database
Database content (no. data records)	5150 as of June 2021 (including ca. 500 non-piping passive component failures)	12 200 pipe failure events as of June 2021 Plus, a water hammer event database (736 events)	ca. 1100 DB records	Not a formal database; see below	Assumed to include >600 database records on pipe failures (this number is to be confirmed by IAEA)
Data submission protocol	Governed by an annual work plan, QA/QC Procedure, and Coding Guideline. CODAP is based on the principle that each participating organization shares OPEX data of interest to the material science community	Since 1993, continuously updated and maintained. A 1998 version of the database formed the basis of CODAP. PIPExp data mining and data processing are governed by a coding guideline	Continuously updated and maintained by GRS upon receipt of event reports from German NPPs, which have to be issued according to the reporting criteria defined in the German reporting ordinance (AtSMV)	Informal, data exchange twice yearly — verbal presentations and PPT presentations. The latter are available on the CHUG web site	The incident reporting system is based on the principle that each participant will provide timely information on its NPPs' OPEX so that it is available to all other participants
Taxonomy	Very detailed taxonomy	Very detailed taxonomy	Very detailed taxonomy	Flow accelerated corrosion phenomenology	Documented in the incident reporting system guidelines
Database type	Web based SQL	SQL – stored in MS-ACCESS	SQL – stored in MS-ACCESS	n.a.	Web based SQL
No. database fields	89 NEA/CSNI/R(2018)12 includes a description of the database structure	65 fields in main table, there are an additional 30 supporting (related) tables	61	n.a.	Mostly searchable narrative information. A user is to perform additional analyses w.r.t. identifying material, crack morphology, etc

TABLE III–1 SELECTED PIPE FAILURE EVENT DATABASES (cont.)

Database feature	**Database type and name**				
	Type 2	Type 1.a	Type 1.a	Type 2	
	US nuclear regulatory commission licensee event reports database[a]	STRYK	NESC thermal fatigue database	SAPHIR	
Accessibility	Public domain GP database[b]	Passive component damage database — access restricted to SSM staff	Access restricted to NESC project participants	Proprietary GP database	
Owner/operator	The Licensee Event Reports database is operated by the Idaho National Laboratory on behalf of the US NRC	Swedish Radiation Safety Authority (SSM)	[c]	EDF, France	
Database scope	General purpose OPEX database and limited to the US commercial NPPs	Any rejectable, recordable and reportable pressure boundary flaw or failure in the Swedish commercial NPPs	Selected thermal fatigue failures	Central web based repository for OPEX data from all EDF plants; similar to Institute of Nuclear Power Operations' ICES database	
Database content (no. data records)	>2300 records on pipe failure	>1350 events (piping and non-piping passive components)	About 40 events	>>1000 records on pipe failure	
Data submission protocol	NUREG-1022 R3 (Event Report Guidelines, 10 CFR 50.72 and 50.73)	Respective owner submits ISI/non-destructive examination results upon completion of each annual refuelling outage	Supplied by NESC participating organizations	Information management system for unexpected events in EDF plants	
	SAPIDE	ICES	NUCIA	EPRI NP-4394	EPRI TR-110102
Taxonomy	Documented in NUREG-1022 [III–1]	Database structure was developed by a materials scientist	33 database fields, 10 of which specifically relate to thermal fatigue phenomena		

TABLE III–1 SELECTED PIPE FAILURE EVENT DATABASES (cont.)

Database feature	**Database type and name**				
	Type 2	Type 2	Type 2	--	--
Database type	Web based access	SQL —stored in MS-ACCESS		Web based access	
No. database fields	Searchable narrative information	20			
Accessibility	Proprietary, GP	Proprietary	Public/Proprietary domains	Public domain	Public domain
Owner/operator	IRSN, France	Institute of Nuclear Power Operations (INPO), Atlanta, GA, USA	Japan Nuclear Technology Institute (JANTI), Tokyo, Japan	Electric Power Research Institute, Palo Alto, CA, USA	Electric Power Research Institute and Swedish Radiation Safety Authority
Database scope	Limited to ‘safety significant events’	Central web based repository for operating experience data from all US plants	Central web based repository for domestic and international operating experience	Small diameter (≤DN100) pipe failure data for BWR plants	Nuclear reactor piping failures at US commercial LWRs: 1961–1997
Database content (no. data records)	In 2017, this database contained >20 000 safety significant events[d]. The number of pipe failure events is unknown	Estimated to be >>1000 DB records on pipe degradation and failure	Estimated to be >>1000 DB records on pipe degradation and failure	Failure data – extracted from US NRC License Event Report database through mid-1982	4064 DB records
Data submission protocol	Includes events submitted to the ASN and NEA/IAEA IRS database	The amount of data provided varies by utility. The raw failure records in ICES can be inconsistent	OPEX submitted by the Japanese plant operators. NUCIA includes OPEX from IRS, WANO, INPO		n.a.
Taxonomy		[e]	[III–2]		Documented in TR-110102
Database type	Web based	Free-format web based database, mostly in narrative form	Web based		Microsoft Access (version 7.0 for Windows 95)
No. database fields	78	Not known	Not known	15	

Note: n.a. — not applicable.

[a] Additional pipe failure reports are available from https://adams.nrc.gov/wba/, which is the Agency-wide Documents Access and Management System (ADAMS). It is the official recordkeeping system, through which the US NRC provides access to the following ‘libraries’ or collections of publicly available documents: (1) The Publicly Available Records System (PARS) Library contains more than 730 000 full text documents that the US NRC has released since 1 November 1999, and several hundred new documents are added each day; and (2) the Public Legacy Library contains documents dating back to 1965.

[b] See: https://lersearch.inl.gov/LERSearchCriteria.aspx

[c] See: http://ie.jrc.ec.europa.eu/

[d] See: https://www.irsn.fr/fr/connaissances/installations_nucleaires/les-centrales-nucleaires/rex/pages/construire-regles-surete-demain.aspx#.Y6A7vtXMJaR

[e] See: https://na.eventscloud.com/file_uploads/aef761d3464553c9bb2cafadf428d0e0_18NSCSL_ICES_Final_Barnes.pdf

TABLE III–2 SELECTED NON-NUCLEAR PIPE FAILURE DATABASE

Name	Content	For more information
DOT/PHMSA Natural Gas Distribution Incident Data	Three subsets: (1) Safety related condition reports, (2) mechanical fittings failure database, and (3) gas distribution incident database. PHMSA's incident data access files can be downloaded free of charge	https://www.phmsa.dot.gov/data-and-statistics/pipeline/source-data
	Pipeline failure investigation reports	https://www.phmsa.dot.gov/safety-reports/pipeline-failure-investigation-reports
UKWIR National Mains Failure Database	The database contains ca. 500 000 water mains failure records covering over 95% of the UK companies for the period from 1995 onwards. The web based database includes lengths of piping for each material and diameter to allow the generation of comparisons of failure rates. The data sharing participants have access to and exchange failure data via a secure server	http://sewersandwatermains.ukwir.org/site/NMFD/home
CODAM, Corrosion and Damage Database	CODAM captures data on degraded and failed conditions in offshore oil and gas structures, risers and pipelines. The database content (in Norwegian) is publicly available	http://www.ptil.no/getfile.php/1345620/PDF/Roerledningsskader%20Oktober2017.pdf
CONCAWE (Conservation of Clean Air and Water in Europe) Database	CONCAWE has collected 46 years of spillage data on European cross-country oil pipelines. At nearly 37 500 km the current inventory includes the majority of such pipelines in Europe. Information on annual throughput and traffic, spillage incidents and in-line inspection activities are gathered yearly via on-line questionnaires. The results are analysed and published annually	https://www.concawe.eu/wp-content/uploads/2018/03/Rpt_18-6-2.pdf
HSE Hydrocarbon Release Database (HCRD)	The inquiry into the Piper Alpha accident in the North Sea (Cullen 1990) recommended that the Health and Safety Executive (HSE) should collect a database of hydrocarbon leaks from offshore installations in the UK sector, and provide it to operators to support QRA. The resulting hydrocarbon release database (HCRD) has collected all significant releases in the UK sector since October 1992. In addition, the HSE has estimated the exposed population of equipment items and from these has determined leak frequencies and size breakdowns for each equipment type. Incident details are available, free of charge, in the form of spreadsheet data from the HSE Offshore Statistics page	http://www.hse.gov.uk/offshore/hydrocarbon.htm

TABLE III–2 SELECTED NON-NUCLEAR PIPE FAILURE DATABASE (cont.)

Name	Content	For more information
International Association of Oil and Gas Producers (OGP)	(OGP) publishes the Risk Assessment Data Directory (RADD). The corresponding reports contain frequencies for different systems and subsystems of facilities from the oil and gas industry	https://www.iogp.org/
European Gas Pipeline Incident Data Group (EGIG)	In 1982, six European gas transmission system operators took the initiative to gather data on the unintentional releases of gas in their transmission pipeline systems. This cooperation was formalized by the setting up of EGIG (European Gas pipeline Incident data Group). Presently, EGIG is a cooperation of 17 gas transmission system operators in Europe and it is the owner of an extensive database of pipeline incident data collected since 1970	https://www.egig.eu/reports Published at the end of 2020 the 11th EGIG Report covers the period 1970–2019. Section 3 of this report summarizes pipeline failure statistics organized by: — Leak size; — Pipe diameter and leak size; — Cause of failure, pipe diameter and leak size; — Type of third party impacts (e.g. excavation damage) and leak size; — Ground movement, pipe size and leak size
Pipeline Performance in Alberta (Alberta Energy and Utilities Board — EUB)	The AER inspects pipeline construction and operations, reviews incidents, identifies non-compliances, and performs necessary enforcement. The AER documents pipeline failure data in the Field Inspection System and the Pipeline Registry System. At the end of 2012 the AER database included 17 605 failure events. The AER database is supported by a detailed taxonomy as well as pipe failure exposure term data	https://www.aer.ca/documents/reports/R2013-B.pdf
Major Hazard Incident Data Service (MHIDAS)	The database holds details of incidents which have occurred worldwide during the transport, processing or storage of hazardous materials which resulted in or it is considered had the potential to cause off-site impact. A 2010 version of the MHIDAS contained over 16 000 records corresponding to over 14 000 incidents, and was in support of the HSE Ageing Plant project. MHIDAS was an AEA Technology-hosted database. Details of plant and its function can be limited since much of the information is drawn from press reports and insurance databases	Research Report RR823, Plant Ageing Study, Phase 1; https://www.hse.gov.uk/research/rrpdf/rr823.pdf The database is no longer maintained. AEA Technology went into administration in 2012. HSE is the Health and Safety Executive of the United Kingdom

TABLE III–2 SELECTED NON-NUCLEAR PIPE FAILURE DATABASE (cont.)

Name	Content	For More Information
N.V. Nederlandse Gasunie	Pipe failure data from the Dutch natural gas industry. Gasunie has implemented the 'PiMSlider' software system in support of its pipe-line integrity management. This software system consists of a number of modules: pipeline-, environmental- and incident data, cathodic protection (CP) system monitoring data, analyses of ILI data, defect assessments and quantitative risk calculations	NACE-08141: The Utilization of a Pipeline Integrity Management System for ECDA Management within GASUNIE https://nace.org/home.aspx
Netherlands Oil and Gas Exploration and Production Association (NOGEPA)	NOGEPA keeps a database in which all natural gas accidents and incidents are reported by its members. This database is used for reporting to State Supervision of Mines. The available data from the database have a number of limitations that restrict their applicability to this project. One of the major issues is that there is no information is present about hole sizes. No distinction is also made between onshore and offshore and between above and below ground or water	https://www.nogepa.nl/?lang=en
Offshore and Onshore Reliability Database (OREDA)	The Norwegian Petroleum Directorate (now: Petroleum Safety Authority) initiated the OREDA Project in 1981. The primary objective was to collect reliability data for safety equipment. It was agreed that OREDA was to be run by a group of oil companies in 1983. The objective of OREDA was subsequently expanded to collect experience data from the operation of offshore oil and gas production facilities to improve the basic data in safety reliability studies. The OREDA project has since its start been run in phases normally lasting for 2–3 years	http://www.oreda.com/
Failure Rates and Event Data for Use within Risk Assessments (FRED)	This report includes tabulations of 'recommended' pipe failure rates for use in quantitative risk analysis (QRA)	http://www.hse.gov.uk/landuseplanning/failure-rates.pdf
ZEMA (Zentrale einfache Melderegisterauskunft) Germany	Central repository for reports on incidents involving corrosion related accidents in petroleum refineries. Concise technical summaries of chemical accidents. The on-line database covers accidents from 1981 onwards. Database queries produce detailed accident reports. The database includes >800 reports	http://www.infosis.uba.de/index.php/de/site/12981/zema/index.html

TABLE III–2 SELECTED NON-NUCLEAR PIPE FAILURE DATABASE (cont.)

Name	Content	For more information
ARIA (Analysis, Research and Information on Accidents), France	ARIA is tasked with compiling, analysing and disseminating information and experience feedback in the area of industrial and technological accidents, ARIA has inventoried over 46 000 chemical and oil and gas industry accidents and incidents occurring in France or abroad. Some 1200 new events are added to the database each year. The ARIA database includes significant accidents/incidents that showcase experience feedback as a risk prevention and mitigation tool. The criteria used to select the events to catalogue are continuously being updated	https://www.aria.developpement-durable.gouv.fr/the-barpi/the-aria-database/?lang=en
eMARS (electronic Major Accident Reporting System), European Commission	The Major Accident Reporting System (MARS and later renamed eMARS after going on-line) was first established by the EU's Seveso Directive 82/501/EEC in 1982 and has remained in place with subsequent revision to the Seveso Directive in effect today. The purpose of the eMARS is to facilitate exchange of lessons learned from accidents and near misses involving dangerous substances in order to improve chemical accident prevention and mitigation of potential consequences. eMARS contains reports of chemical accidents and near misses provided to the Major Accident Hazards Bureau (MAHB) of the European Commission's Joint Research Centre (JRC) from EU, EEA, OECD and UNECE countries (under the TEIA Convention). Reporting an event into eMARS is compulsory for EU Member States when a Seveso establishment is involved and the event meets the criteria of a major accident as defined by Annex VI of the Seveso III Directive (201218/EU). For non-EU OECD and UNECE countries, reporting accidents to the eMARS database is voluntary. The information of the reported event is entered into eMARS directly by the official reporting authority of the country in which the event occurred. As of August 2018, the eMARS database contains 917 event reports of which several hundred events involve pipe failure	https://emars.jrc.ec.europa.eu/en/emars/content **FOR PUBLIC USERS:** To access **eMARSpublic**, the access passes through an authentication in **EU Login** (formerly known as ECAS — European Commission Authentication System). https://webgate.ec.europa.eu/cas/eim/external/register.cgi

TABLE III–2 SELECTED NON-NUCLEAR PIPE FAILURE DATABASE (cont.)

Name	Content	For more information
FACTS (Failure and Accidents Technical Information System), The Netherlands	FACTS is an accident database which contains information on >25 700 industrial incidents/near-misses involving hazardous materials or dangerous goods that have happened all over the world during the past 90 years. For the most serious accidents, detailed background information is stored, most of it electronically and remains available for further research purposes. The FACTS chemical accident database was a product of TNO Industrial and External Safety. Cost of full on-line access to database is €1000.00 per year	http://www.factsonline.nl/
US Chemical Safety Board (CSB)	The CSB is an independent federal agency charged with investigating industrial chemical accidents. Headquartered in Washington, DC, the agency's board members are appointed by the President and confirmed by the Senate. The CSB conducts root cause investigations of chemical accidents at fixed industrial facilities. Root causes are usually deficiencies in safety management systems but can be any factor that would have prevented the accident if that factor had not occurred. Other accident causes often involve equipment failures, human errors, unforeseen chemical reactions or other hazards. The agency does not issue fines or citations, but does make recommendations to plants, regulatory agencies such as the Occupational Safety and Health Administration (OSHA) and the Environmental Protection Agency (EPA), industry organizations, and labour groups. Congress designed the CSB to be non-regulatory and independent of other agencies so that its investigations might, where appropriate, review the effectiveness of regulations and regulatory enforcement	https://www.csb.gov/ The CSB web site posts detailed root cause investigation reports, many of which address pipe failures

TABLE III–2 SELECTED NON-NUCLEAR PIPE FAILURE DATABASE (cont.)

Name	Content	For more information
Corrosion-Related Accidents in Petroleum Refineries: Lessons Learned from EU and OECD Countries	EUR 26331 EN (2013). This study of corrosion related accidents in refineries is based on 99 reports of important refinery accidents in which corrosion of an equipment part was identified or suspected as being the key failure leading to the accident event. Only reports listed in open sources and produced by or with the collaboration of parties directly involved in the accident investigation were used. Therefore, with a few exceptions, on-line government databases of accident reports were the main source of accident reports. In total, Moreover, since the study was conducted on refineries in a specific geographic area, reports that did not specify geographic location of the refinery could not be used	https://publications.jrc.ec.europa.eu/repository/bitstream/JRC84661/lbna26331enn.pdf
Battelle Memorial Institute	DTPH56-11-T-000003: Comprehensive Study to Understand Longitudinal ERW Seam Failures. Battelle's Experience with ERW and Flash Weld Seam Failures: Causes and Implications. This report presents an evaluation of the database dealing with failures originating in electric resistance welds (ERW) and flash weld (FW) seam defects as quantified by Battelle's archives and the related literature. Thereafter, the database was analysed and trended as the basis to determine the utility and effectiveness of hydrotesting and in-line inspection (ILI) to assess pipeline condition. Annex A is a tabulated database	http://www.carkw.com/wp-content/uploads/2013/10/9.20.12-Report-on-ERW-and-Flash-weld-seams.pdf
Failure Knowledge Database (FKD)	A failure 'mandalas' database (or failure knowledge database) developed by the Japan Science and Technology Agency (JST). The database includes detailed narratives of chemical and petrochemical accidents caused by pipe failure; about 25% of the total FKD content. The database covers the significant accidents worldwide and is managed by experienced academia. The accident reports are carefully reviewed by a committee and they contain detailed information on the accident often including process flow diagram, plant layout and fault tree analysis	http://www.shippai.org/fkd/en/index.html
Analysis of Equipment Failures as Contributors to Chemical Process Accidents	The paper by Kidama and Hurme includes the results of an analysis of pipe failure events in the FKD database	Process Safety and Environmental Protection, 91 (2013) pp 61-78

TABLE III–3 SELECTED PIPING RELIABILITY PARAMETER DATABASES

SOURCE OF PIPING RELIABILITY PARAMETERS		
R-Book	**EPRI 3002000079**	**EPRI 3002002787**
Reliability Data Handbook for Piping Components in Nordic Nuclear Power Plants, 1st Edition	Pipe Rupture Frequencies for Internal Flooding Probabilistic Risk Assessments, Revision 3	Piping System Failure Rates for Corrosion Resistant Service Water Piping
Availability: Restricted to the project funding organizations. A 'light version' is available by contacting Nordic PSA Group	Availability: Licensed Material. Restricted to EPRI Membership	Availability: Licensed Material. Restricted to EPRI Membership.
Prepared by: LR Energy Sweden AB	Prepared by: EPRI	Prepared by: EPRI
Funded by: Forsmark AB, OKG AB, Ringhals AB, Swedish Radiation Safety Authority	Funded by: EPRI Member Utilities	Funded by: EPRI Member Utilities
Content: Piping reliability parameters (failure rates and rupture frequencies) for the following systems: • Safety class 1 and 2 BWR Systems • Safety class 1 and 2 PWR Systems	Content: Piping reliability parameters (failure rates and rupture frequencies) for the following systems: • Circulating water; • Component cooling; • Fire Protection water; • Balance-of-plant; • Emergency core cooling system; • Service water (safety class 3 and non-safety)	Content: An expanded DDM with focus on how to derive piping reliability parameters for super-austenitic steel piping in raw water environments. Piping reliability parameters (failure rates and rupture frequencies) for service water systems
Date of Publication: January 2011	Date of Publication: April 2013	Date of Publication: August 2014
npsagsecretary@afconsult.com	askepri@epri.com	askepri@epri.com
The Nordic PSA Group has developed a piping reliability parameter handbook to be used in PSA. The Handbook includes pipe leak and rupture frequencies and conditional rupture probabilities on the basis of the OPDE pipe failure event database. A first version of the proprietary R-Book was released in 2011 and it was made available to the Nordic PSA Group member organizations [a]	This report updates a 2010 EPRI report (1021086) on piping system failure rates for use in PSA involving internal plant flooding and high energy line breaks and represents the third revision to this pipe failure rate handbook	This report provides failure rate estimates of corrosion resistant service water piping for use in internal flooding probabilistic risk assessments. The failure rates developed in this report supplement a more comprehensive piping system failure rate handbook, Pipe Rupture Frequencies for Internal Flooding Probabilistic Risk Assessments: Revision 3 (EPRI Product ID: 3002000079)

TABLE III–3 PIPING RELIABILITY PARAMETER DATABASES (cont.)

Source of piping reliability parameters			
Atomic Energy of Canada Limited -Misc-204	**Atomic Energy of Canada Limited -Misc-252**	**NUREG/CR-4407**	**TR-110161**
A Study of Piping Failures in US Nuclear Power Plants	Piping Performance in Canadian CANDU NGS	Pipe Break Frequency Estimates for Nuclear Power Plants	Piping System Reliability and Failure Rate Models for Use in Risk-Informed ISI Applications
No longer available (NLA)	NLA	https://www.osti.gov/biblio/6197523	Licensed Material. NLA, the report has been withdrawn from EPRI Products web site
Prepared by: Chalk River Nuclear Laboratories	Prepared by: Chalk River Nuclear Laboratories	Prepared by: Idaho National Laboratory	Prepared by: EPRI
Sponsoring Organization(s): Atomic Energy of Canada Limited, Ontario Hydro	Sponsoring Organization(s): Atomic Energy of Canada Limited, Ontario Hydro	Sponsoring Organization(s): US NRC	Sponsoring Organization(s): EPRI Member Utilities
Content: This study of piping failures was undertaken in support of a study of pipe rupture in the Primary Heat Transport System of CANDU stations	Content: Information on pipe failures in operating commercial CANDU plants was collected and analysed. Atomic Energy of Canada Limited -Misc-252 comprises failure rate calculations, classification of pipe failure events, and 'determination of significant correlations among the classifications'	Content: The objective of this study was: (a) to determine if sufficient pipe break failure data had been reported since the publication of WAH-1400 to improve the uncertainties of the pipe break failure frequencies in use by risk analysts; and (b) to determine if sufficient data exists to provide more specific pipe break frequencies for conditional factors, such as pipe failures for specific systems and failures per weld or per foot of pipe for these specific systems, and failures based on the operational mode of the plant	Content: This report establishes models and databases for piping system reliability assessment that utilize service experience from the first 2000 reactor-years of LWR OPEX. The approach that was followed was to employ Markov reliability models that permit the role of inspections and the time dependent issues associated with ageing processes to be addressed. Relationships are established between the time dependent pipe rupture frequencies and observable parameters that describe the failure, inspection and repair processes. A piping reliability database based on the cumulative OPEX of LWR piping systems was developed to support application of the models
Date of Publication: April 1981	Date of Publication: April 1984	Date of Publication: May 1987	Date of Publication: December 1998

TABLE III–3 PIPING RELIABILITY PARAMETER DATABASES (cont.)

Source of piping reliability parameters			
TR-111880	**TR-100380**	**EGG-SSRE-9639**	**INFO-0607**
Piping System Failure Rates and Rupture Frequencies for Use in Risk-Informed ISI Applications	Pipe Failures in US Commercial Nuclear Power Plants	Component external leakage and rupture frequency estimate	Failure Rates in Piping Manufactured to Different Standards
Licensed Material. The NLA report has been withdrawn from EPRI Products web site. A non-proprietary version[b]	Licensed Material. The NLA report has been withdrawn from EPRI Products web site	Web. Doi:10.2172/5461408	https://inis.iaea.org/collection/NCLCollectionStore/_Public/28/052/28052725.pdf
Prepared by: EPRI	Prepared by: EPRI	Prepared by: Idaho National Laboratory	Prepared by: G.D. Cooper Consultants Inc.
Sponsoring Organization(s): EPRI Member Utilities	EPRI Member Utilities	Sponsoring Organization(s): US Department of Energy	Sponsoring Organization(s): Canadian Nuclear Safety Commission
Content: A piping reliability database based on the cumulative OPEX of LWR piping systems was developed to support application of the models. Failure rates and rupture frequencies derived from this database are presented in this report. This database permits the application to all four LWR reactor vendors, all existing piping systems and all the observed pipe failure mechanisms. Practical application of the initial models and databases was demonstrated in a companion report EPRI TR-110161	Content: EPRI developed a methodology and database that uses actual experiences to support failure rate calculations on a plant or system specific basis	Content: In order to perform detailed internal flooding risk analyses of NPPs, external leakage and rupture frequencies are needed for various types of components – piping, valves, pumps, flanges and others. Based on a comprehensive search of Licensee Event Reports contained in Nuclear Power Experience, and estimates of component populations and exposure times, component external leakage and rupture frequencies were generated	Content: The approach taken in this study was to determine the causes of failure of non-nuclear piping subjected to service similar to that experience by piping in CANDU NPPs. The study examined information on carbon steel piping systems filled with water/steam which operate up to a maximum of 315°C and a maximum pressure of 1600 psi. The failure mechanisms were identified and analysed to determine whether application of the requirements of ASME Section III would have prevented the failure
Date of Publication: September 1999	Date of Publication: July 1992	Date of Publication: November 1991	Date of Publication: November 1995

TABLE III–3 PIPING RELIABILITY PARAMETER DATABASES (cont.)

Source of piping reliability parameters	
US NRC **OPEX results and databases**	**Failure rate and event data for use within risk assessments** **(06/11/2017)**
US OPEX data	Item failure rate 1.3 Pipework – Pipe Failure Rates
Public domain initiating event database. Data for all unexpected reactor trips during power operations at commercial NPPs were reviewed. Each event was reviewed and categorized according to the initial event and, additionally, was marked if certain other risk-significant events occurred, regardless of their position in the event sequence. The collected data were analysed for time dependence, reactor type dependence, and between-plant variance. Dependencies and trends are reported, along with the raw counts and the best estimate for initiating event frequencies; for example, initiating events due to piping pressure boundary failure[c]	Public domain database intended for risk analyses in the chemical process industries. Tabulations of pipe failure rates (per m per year) for different pipe sizes and hole sizes (3 mm diameter to 1/3 pipe diameter up to guillotine type failure) and for the following pipe diameters: • 0–49 mm • 50–149 mm • 150–299 mm • 300–499 mm • 500–1000 mm. The basis for derived pipe failure rates is included in the handbook[d]
Prepared by the US Idaho National Laboratory	Prepared by: Chemicals, Explosives and Microbiological Hazardous Division 5 of the UK Health and Safety Executive, Hazardous Installations Directorate
Sponsoring Organization(s): US NRC Office of Research	UK Health and Safety Executive, Hazardous Installations Directorate
Content: Presents an analysis of initiating event frequencies at US commercial NPPs from calendar year 1988 through 2018 as reported in licensee event reports. The current version includes estimates of very small LOCA frequencies	Content: A compilation of many of references ranging from proprietary study reports to textbooks comprising 96 pages of data tables and background information. A subset of this document includes pipe failure rates that have been derived from the chemical and oil and gas process industries
Date of Publication: A summary report issued as INL/EXT-19-54513 (Initiating Event Rates at US nuclear power plants 1988–2018)	Date of Publication: 6 November 2017

TABLE III–3. PIPING RELIABILITY PARAMETER DATABASES (cont.)

Source of piping reliability parameters	
Estimation of Failure Rates of Crude Product Pipelines, Proc, 11th Int. Conf. Applications of Statistics and Probability in Civil Engineering (ICASP11, Zürich, Switzerland, 1–4 August 2011)	**Lessons Learned from Oil Pipeline NATECH Accidents for NATECH Scenario Development**
A case study is presented to illustrate the application of the proposed methodology. The studied data was obtained from the Nigerian National Petroleum Company and consists of three different API 5L X42 pipelines. The number of pipeline failures due to corrosion from 1999 to 2009 was collected	The report summarizes US pipeline failures that were caused by natural hazards. NATECH = Natural Hazard Triggered Technological Accidents
Statistical methods for the reliability of repairable systems are applied to provide an estimate for the failure rate of cross-country crude product pipelines based on historical failures. The pipelines are assumed to follow minimal repair models, and the failure data are tested against the homogenous and the non-homogenous Poisson processes. Laplace and the MIL-HDBK 189 tests are used to test the null hypothesis that the process is a homogenous Poisson process against the alternative that the intensity is increasing, following a non-homogeneous Poisson process. The statistical tests revealed that homogenous Poisson process is an acceptable model describing the number of corrosions that occur in onshore crude product pipelines. The intensity function and the mean time between failures of the pipelines are determined to analyse the dynamics of failures between API 5L X42 pipelines installed at different periods. It is found that all other factors being equal, similar pipelines installed relatively at the same time would show similar mean time between failures and failure intensity[c]	Natural hazards can impact oil transmission pipelines with potentially adverse consequences on the population and the environment, causing as well a significant economic impact to pipeline operators. There is a limited historical information available regarding the impact of the dynamics of natural hazard on pipelines. This reference provides a summary on the collection and analysis of hazardous liquid and natural gas transmission pipeline incident data. European and US incident data sources were reviewed and imported into a specifically developed database-driven incident data analysis system. The total number of identified NATECH is 20. Recent NATECH are rare and there has only been one pipeline NATECH incident since 1995. Ninety per cent of the NATECH involve the pipe body, whereas the remainder involves pump stations. There are no reported NATECH at intermediate storage facilities. Geological hazards were the primary trigger (65%), followed by hydrological (20%) and climate hazards (10%). Meteorological hazards played a minor role. The main incident initiators among geological hazards were landslides and the rest was mostly subsidence events primarily affecting elements other than the pipe body [III–3]
Prepared by: Delft University of Technology, Delft, The Netherlands and Operations Department, Nigerian National Petroleum Company, Abuja, Nigeria	Prepared by: EC Joint Research Centre, Ispra, VA, Italy
Sponsoring Organization(s): No acknowledgements provided	Sponsoring Organization(s): The European Programme for Critical Infrastructure Protection, European Commission, Brussels, Belgium
Content: Tabulations of pipeline failure rates and plots of pipeline failure rates as a function of years of operation	Content: Tabulations of pipe failures organized by type of pipeline, diameter, etc. The information can be used to obtain pipeline failure rates
Date of publication: August 2011	Date of publication: 2015

[a] See: http://www.npsag.org/aboutus/

[b] See: https://www.nrc.gov/docs/ML0037/ML003776638.pdf

[c] See: https://nrcoe.inl.gov/resultsdb/InitEvent/

[d] See: https://www.hse.gov.uk/landuseplanning/failure-rates.pdf

TABLE III–4. EXAMPLES OF SOURCES OF PIPE FAILURE INFORMATION IN THE USA

Reporting level	Report/database type	Level of technical content from the point of view of material science, PSA applications, etc.	Third party accessibility for purposes of piping reliability analysis
		US NRC	
Regulatory requirements and guidance	Licensee event report	High level — requires interpretation and classification by an SME — high reporting thresholds. Since the 1990s only a few LERs of relevance to CODAP are issued on an annual basis; <10 per year.	Public domain; covers 1980 to date; https://lersearch.inl.gov/LERSearchCriteria.aspx. A large number of pre-1980 LERs are available from https://adams.nrc.gov/wba/
	Event notification (European standard) system	High level event tracking system for incoming notifications of significant nuclear events with an actual or potential effect on plant safety. An EN may result in an LER. On an annual basis, only a few ENs of relevance to CODAP are issued	Public domain; all ENs from 1 January 1999 onwards are available at: https://www.nrc.gov/reading-rm/doc-collections/event-status/event/en.html
	Generic communications — generic letters, information notices	Identifies potential generic issues that are safety significant and require technical resolution	https://www.nrc.gov/reading-rm/doc-collections/gen-comm/
	Generic letter 90-05 request for temporary repair	Applicable to moderate-energy piping, the operability evaluations that are part of the 90-05 relief request are comprehensive. Excellent source of pipe failure information	Public domain; https://adams.nrc.gov/wba/
	10 CFR 50.55a(z)(2) — Application of ASME code cases	The operability evaluations that are part of the ASME code case relief request are comprehensive. Excellent source of pipe failure information	See below under 'ASME code cases'. Any code case relief request that is submitted to the NRC for approval is also available for review by a third party
	Biannual problem identification and resolution inspection report	Regulatory review of a licensee's corrective action programme and the licensee's implementation of the programme to evaluate its effectiveness in identifying, prioritizing, evaluating, and correcting problems, and to confirm that the licensee is complying with NRC regulations and licensee standards for corrective action programmes	Public domain; https://adams.nrc.gov/wba/
	Quarterly integrated inspection reports	Significant repository for information on passive components failures. Requires interpretation and classification by an SME	Public domain; https://lersearch.inl.gov/IRSearchCriteria.aspx

TABLE III–4. EXAMPLES OF SOURCES OF PIPE FAILURE INFORMATION IN THE USA (cont.)

Reporting level	Report/database type	Level of technical content from the point of view of material science, PSA applications, etc.	Third party accessibility for purposes of piping reliability analysis
		US nuclear industry level	
Institute of Nuclear Power Operations	Institute of Nuclear Power Operations Consolidated Events Database. Participating plants submit Ars, CRs, WORs, etc. — mainly free-format narrative information — to a searchable database	Varying — data extraction, interpretation and classification can be time-consuming.	Access restricted to ICES participating organizations (i.e. nuclear plant operating organizations)
Nuclear Energy Institute (NEI)	NEI 07-07: Industry Ground Water Protection Initiative – Final Guidance Document, 2007	Reporting of below ground/buried pipe failures causing >0.38 m^3 unintended release	Annual radioactive effluent release report submitted by licensees to the US NRC
	EPRI-CHUG: Presentation material from the biannual CHUG meetings. Informal exchange of flow accelerated corrosion information (summary of flow accelerated corrosion failures, information on the use of non-destructive examination technology)	Varying – data extraction, interpretation and classification can be time-consuming	Access restricted to funding organization. Non-proprietary versions of EPRI-MRP reports can be downloaded from the NRC public web site
Electric Power Research Institute (EPRI)	EPRI-MRP, multiple programs – reactor internals, thermal fatigue, ageing management	NRC receives biennial summaries of the BWR and PWR reactor internals inspection results. These summaries are available on the NRC web site.	
	Plant engineering/long term operations/risk and safety management programs	Extensive source of material degradation and failure information organized by system, degradation mechanism, etc. These programs build on active technical input from the sponsoring US and international member organizations. Extensive repository of relevant PEO/LTO material degradation information	Access restricted to funding organization

TABLE III–4 EXAMPLES OF SOURCES OF PIPE FAILURE INFORMATION IN THE USA (cont.)

Reporting level	Report/database type	Level of technical content from the point of view of material science, PSA applications, etc.	Third party accessibility for purposes of piping reliability analysis
NRC/EPRI-MRP annual technical information exchange meetings	Presentation material summarizing R&D status, results of thermal fatigue inspections, etc	High level	All presentation material available through the NRC public document room
GE service information letter (SIL), and equivalent bulletins from the PWROG	Information on material degradation issues in response to recent OPEX	Chronological lists of events with some information on method of discovery, extent of degradation, flaw size data, etc. Plant identities not revealed	The SILs can be downloaded from the NRC public document room; for example, see https://www.nrc.gov/docs/ML0501/ML050120032.pdf
Plant-level			
During routine operation — shiftly walk-down inspection	Action request (AR), condition report (CR), corrective action request (CAR), incident report (IR), non-conformance report (NCR), problem investigation process (PIP), repair/replacement plant (RRP), work order request (WOR), Operability Determination, and Structural Evaluation in accordance with applicable ASME code case	Details on location of flaw/leak with reference to P&ID, isometric drawing — oftentimes accompanied by photographic records, root cause analysis results, non-destructive examination results, metallographic data, flaw size data, leak/flow rates, spatial effects, risk significance, safety significance. Piping fabrication data, operational parameters, piping design information (material, dimensional data)	Restricted/proprietary
Leak detection – containment /drywell	Shiftly (once every 8/12 h shift) RCS inventory mass balance calculation, Chem. Lab trending of inflows to aux. building/reactor building sumps and holding tanks	Logs, databases (various types)	
	Containment/drywell leak detection alarm (≥0.1 gpm/6.3E-3 kg/s)	Control room logs/computer printouts, Excel spreadsheets — holding tank inflow, details of through-wall leak rates as a function of time and location, etc	

TABLE III–4. EXAMPLES OF SOURCES OF PIPE FAILURE INFORMATION IN THE USA (cont.)

Reporting level	Report/database type	Level of technical content from the point of view of material science, PSA applications, etc.	Third party accessibility for purposes of piping reliability analysis
ISI per ASME XI — mandatory inspections	List of 'recordable indications', operability determinations/fitness for service evaluations, ASME XI relief requests (for more details see 'Regulatory' above. Within 90 days of the conclusion of a refuelling outage the plant owner is required to submit an ISI summary report ('Owner's Activity Report') to the US NRC. This report itemizes all inspections performed (Code Class 1, 2 and 3), any operability determination is documented	The ASME relief requests include all relevant information to meet CODAP event data input requirements. Included are 'Abstract of Repair/Replacement Activities Required for Continued Service' and 'Items with Flaws or Relevant Conditions That Required Evaluation for Continued Service'	The ISI summary reports are publicly available via the NRC Public Document Room; https://adams.nrc.gov/wba/
Owner defined ISI programs	Flow assisted corrosion, erosion-corrosion, microbiologically influenced corrosion and thermal fatigue programs	Non-destructive examination reports, 'system health reports'	Restricted/proprietary
ASME	Numerous code cases have been developed for the operability determination of degraded piping; for example, N-513 (Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 3 Piping), N-561 (Alternative Requirements for Wall Thickness Restoration of Class 2 and High Energy Class 3 Carbon Steel Piping), N-562 (Alternative Requirements for Wall Thickness Restoration of Class 3 Moderate Energy Carbon Steel Piping), N-666 (Weld Overlay of Class 1, 2,and 3 Socket Welds	Details on location of flaw/leak with reference to P&ID, isometric drawing – oftentimes accompanied by photographic records, root cause analysis results, non-destructive examination results, metallographic data, flaw size data, leak/flow rates, spatial effects, risk significance, safety significance. Piping fabrication data, operational parameters, piping design information (material, dimensional data). Structural integrity evaluation (inputs/results/interpretation) is part of the required documentation. The ASME relief requests include all relevant information to meet CODAP event data input requirements	The code case relief requests that are submitted to the NRC for approval are also available for third party review. Where a code case has been approved for plant specific use, an associated operability determination is kept on file on site and available for review by NRC Resident Inspector. Hence, a third part review is not possible unless the underlying technical information (e.g. fitness for service evaluations) has been placed in the NRC Public Document Room
System health reports (or System Engineers' Notebooks)	Access or Excel based databases that are part of the overall ageing management programme. The system health reports provide summaries of all through-wall leaks.	High level, intended for trending of material degradation issues, effectiveness of degradation mitigation programmes.	Restricted/proprietary

REFERENCES

[III–1] NUCLEAR REGULATORY COMMISION, Event Reporting Guidelines 10 CRF 50.72 and 50.73, Rep. NUREG-1022 Rev. 1, Office for Analysis and Evaluation of Operational Data, Washington, DC (1998).

[III–2] HIBINO, A., NIWA, Y., Graphical representation of nuclear incidents/accidents by associating network in nuclear technical communication, J. Nucl. Sci. Technol. 45 5 (2012) 369–377.

[III–3] GRIGIN, S., KRAUSMANN, E., Lessons Learned from Oil Pipeline Natech Accidents and Recommendations for Natech Scenario Development, JRC Science and Policy Reports, JRC92700, Publications Office of the European Union, Luxembourg (2015).

Annex IV
WCR PIPING OPEX SUMMARIES

IV–1 INTRODUCTION

Reviews and analyses of the WCR and advanced WCR piping OPEX with safety related and non-safety piping systems have been ongoing ever since the first commercial NPPs came on-line in the 1960s. Evaluations of the WCR piping OPEX data have been an integral element of regulatory and industry initiatives to address long term operation and nuclear plant license renewal. Relatively recent examples of such initiatives include:

- Proactive Materials Degradation Assessment [IV–1];
- Expanded Materials Degradation Assessment (EMDA) Expert Panels [IV–2, IV–3];
- Generic Aging Lessons Learned (GALL, NUREG-1801) for license renewal[1];
- Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR)[2];
- IAEA International Generic Ageing Lessons Learned (IGALL)[3].

Figures IV–1 through to IV–21 represent selected high level summaries of the WCR piping OPEX in the period 1970–2020. The summaries are but a very small excerpt from an extensive body of pipe failure data:

- Figure IV–1 shows how the piping OPEX data has evolved from end of calendar year 2016 to mid-June 2021. The data is organized by safety classification; primary system (safety class 1), piping connected to the primary system (safety class 2), safety related support systems (safety class 3), energy conversion systems (e.g. steam systems) and fire water systems. At the end of the calendar year 2020, the database contained 2401 records on primary system pipe failures. At the end of the calendar year 2019 and for the same category of piping the database content stood at 2281 failures; an increase by 120 failures and so on.
- Figure IV–2 shows pipe failures organized by a time period of occurrences and outer diameter. The OPEX is organized by the time period in which a failure was discovered.
- Figure IV–3 shows pipe failures organized by the time period in which a failure was observed and type of material. There are differences in piping material selections across the nuclear steam system supplier design types as well as the different nuclear steam system supplier design generations.
- Figure IV–4 is a summary of the piping OPEX organized by the different sources of raw data.
- Figure IV–5 shows the number of pipe failures by the time period in which a failure was observed and the operational impact (e.g. a controlled reactor shutdown, reactor trip). The term ‘expanded outage work’ means that in the process of performing a scheduled ISI a flaw was observed using non-destructive examination. In the course of evaluating the flaw, other similar locations were inspected and additional, rejectable flaws were discovered that required repair or replacement actions. The term ‘unplanned outage work’ means that during reactor heat-up

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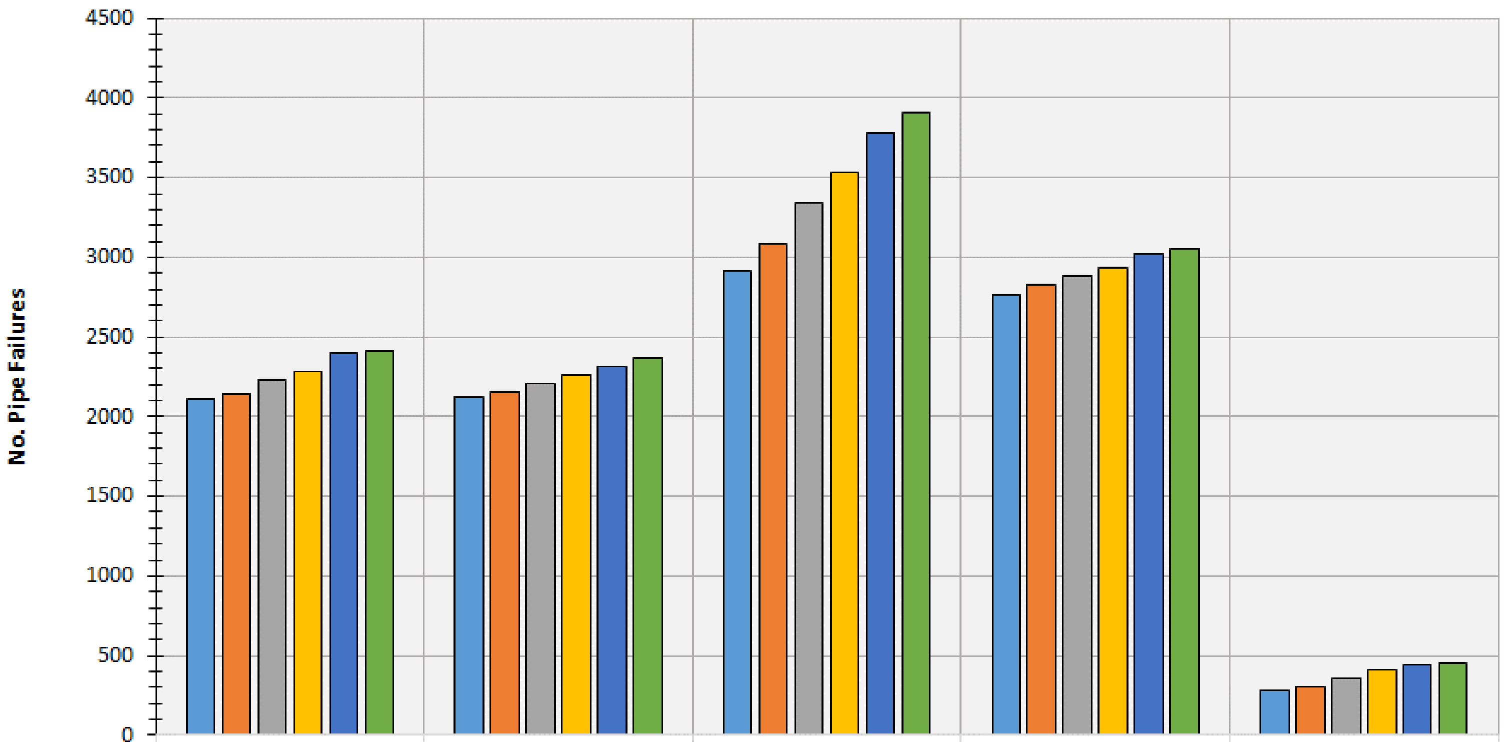
[1] https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1801/r2/index.html

[2] https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr2191/index.html

[3] https://gnssn.iaea.org/NSNI/PoS/IGALL/SitePages/Home.aspx

following a scheduled outage a leakage was discovered and in order to perform repairs the reactor was returned to cold shutdown.

- Figure IV–6. Pipe material degradation mechanism propensity normalized against intergranular SCC in the unmitigated BWR operating environment (i.e. normal water chemistry as opposed to hydrogen water chemistry).
- Figure IV–7. BWR, CANDU and PWR thermal fatigue OPEX organized by the time period in which a failure occurred.
- Figure IV–8. Selected PWR specific thermal fatigue OPEX organized by affected system; RCS, chemical and volume control system, and emergency core cooling system [IV–4].
- Figure IV–9. Selected primary water SCC OPEX [IV–4]. Limited to large diameter piping, the primary water SCC failure data is organized by through-wall depth as a percentage of the pipe wall thickness. To date there have been 35 failures, most of which have involved relatively shallow, surface-connected cracks that required repair before resumption of operation.
- Figure IV–10. Primary water SCC failures in small and medium diameter PWR primary system piping. The primary water SCC experience is organized by the nominal pipe diameter and time period in which a failure was discovered. With few exceptions, the small diameter primary water SCC experienced is associated with instrument line penetrations in RCS cold legs and hot legs of Babcock and Wilcox and Combustion Engineering PWR plants. In these reactor types the cold legs and hot legs consist of stainless steel clad carbon steel piping, and therefore dissimilar instrument line dissimilar metal welds.
- Figure IV–11. WCR primary system piping OPEX organized by in-line location of a weld (or weld heat-affected zone) in which a degraded condition was discovered (e.g. elbow to pipe, pipe to safe-end, reducer to pipe).
- Figure IV–12. Socket weld OPEX summary. Socket welds are used extensively for small diameter piping. However, this type of fitting is prone to high cycle fatigue failure. Some national codes and standards no longer allow socket welds in primary system applications.
- Figure IV–13. Summary of significant isolable primary coolant system pipe failures.
- Figure IV–14. Summary of significant non-isolable primary coolant system pipe failures.
- Figure IV–15. Summary of selected WWER piping system OPEX.
- Figure IV–16. Water hammer experience. The data is organized by system group and the time period in which significant water hammer was observed.
- Figure IV–17. Service water (raw water cooling) piping OPEX.
- Figure IV–18. The piping OPEX organized by method of detection.
- Figure IV–19. Selected WCR OPEX organized by the observed (i.e. calculated or measured) through-wall mass flow rate as a function of the equivalent diameter of the hole in the pipe wall.
- Figure IV–20. Summary of pipe failures that are attributed to failure of ISI/NDE to identify a pre-existing pipe flaw. This OPEX is detailed in [IV–5].
- Figure IV–21. Summary of the safety class 1 pipe failure data organized by the plant mode of operation (cold shutdown, hot standby, hot shutdown, power operation) at the time of discovery, and method of detection.



	Reactor Coolant Pressure Boundary - Class 1	Safety Class 2	Safety Class 3	Non-Safety - Balance of Plant	Fire Water System
2016	2114	2117	2915	2768	285
2017	2146	2152	3087	2823	305
2018	2231	2205	3343	2877	352
2019	2281	2260	3534	2932	407
2020	2401	2319	3781	3017	437
2021 (June)	2411	2372	3904	3056	451

2016 2017 2018 2019 2020 2021 (June)

FIG. IV–1. Evolution of the WCR piping OPEX.

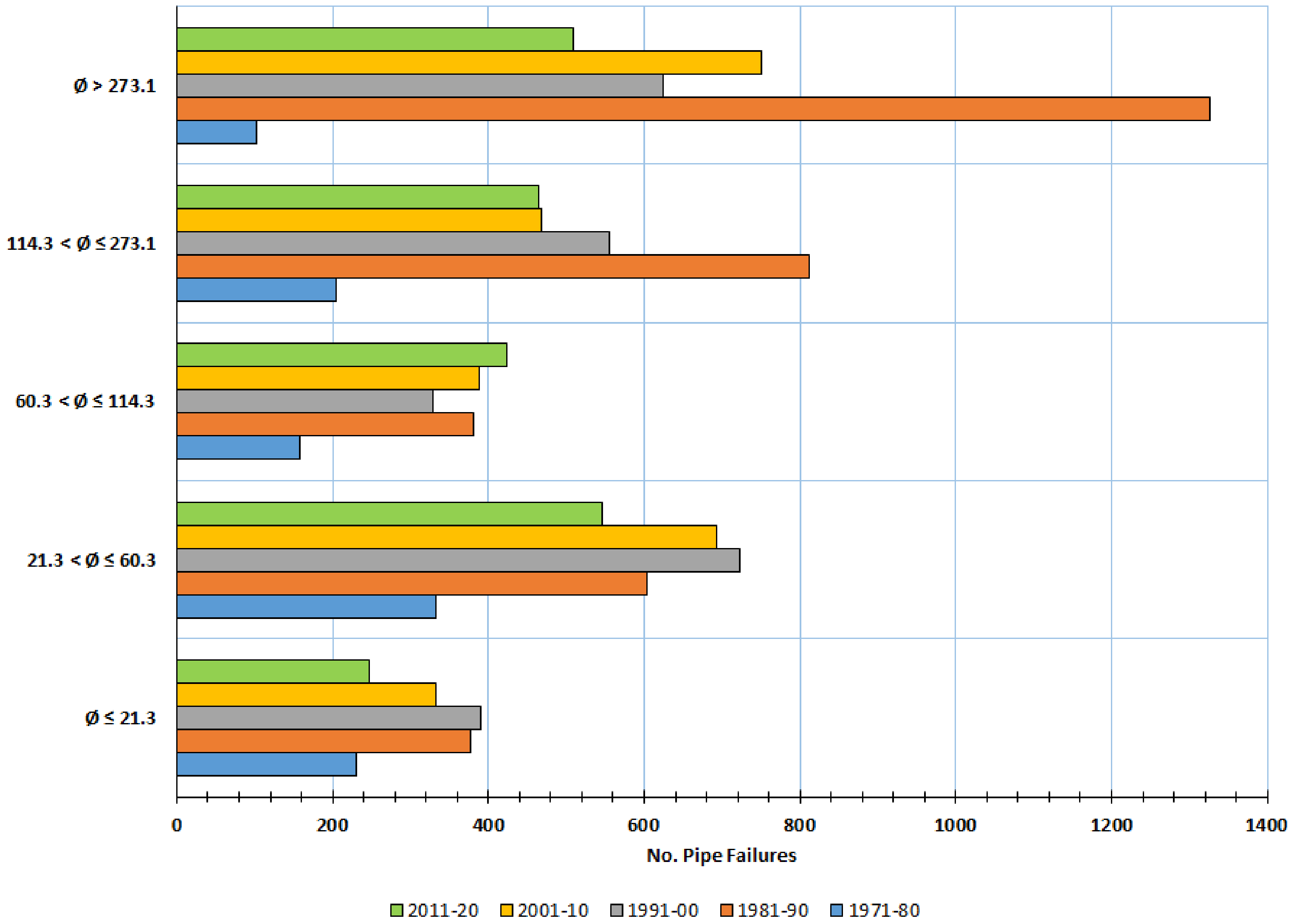


FIG. IV–2. Number of pipe failures by time period and outside diameter (∅ in mm).

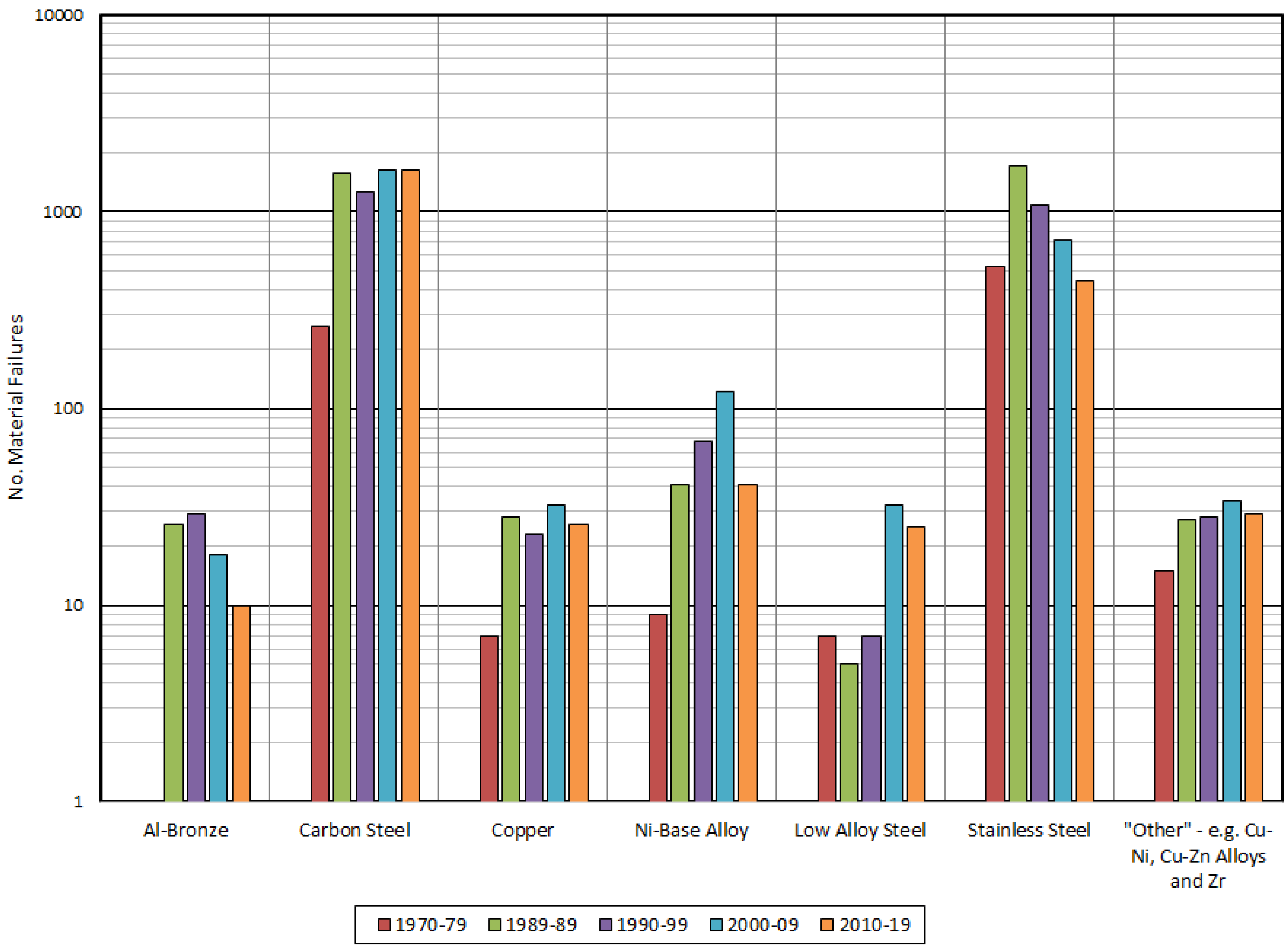


FIG. IV–3. WCR pipe failure experience by material type and time period.

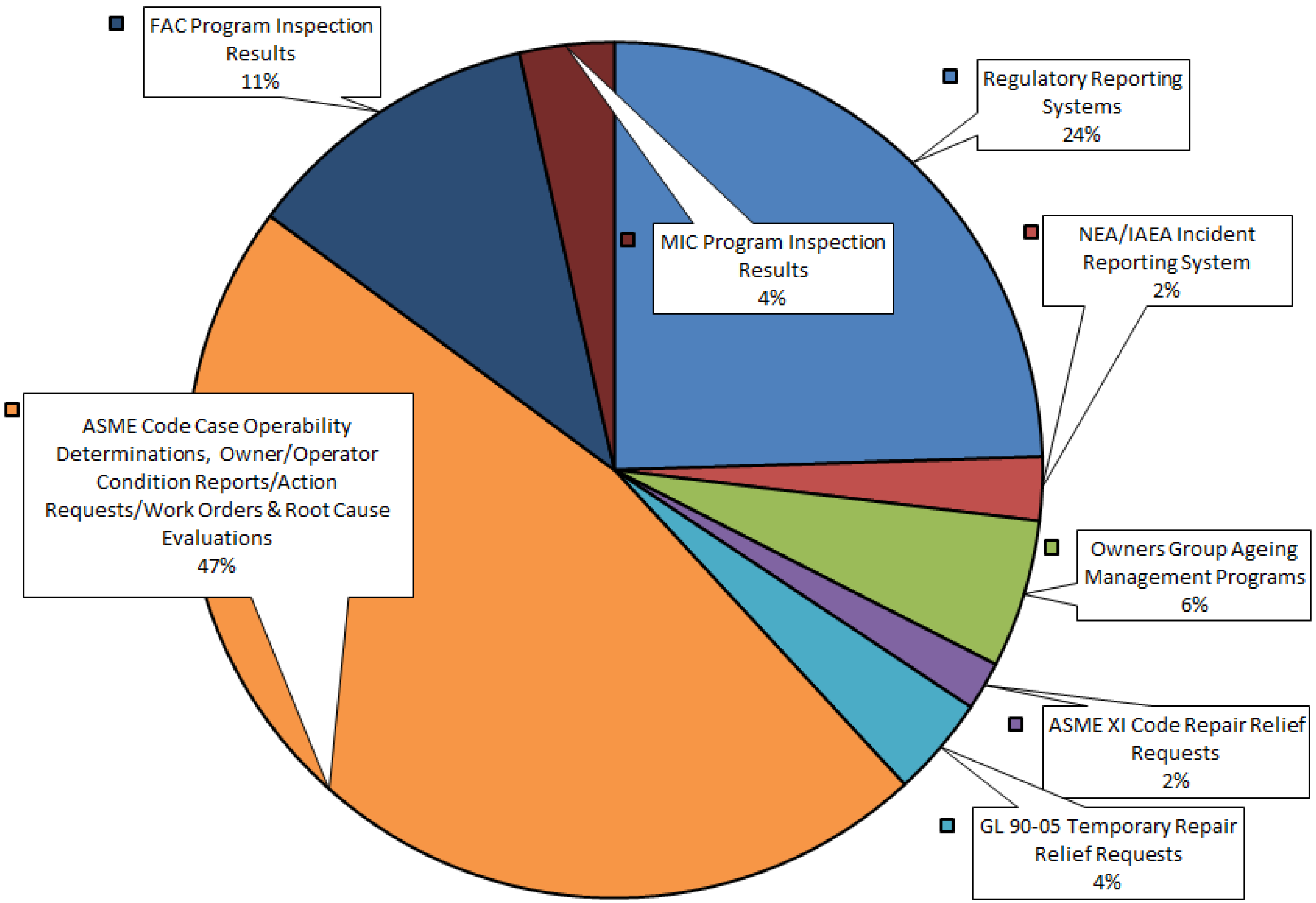


FIG. IV–4. Pipe failure data by source[4].

[4] Percentage of the total database content displayed in Figure III.1.

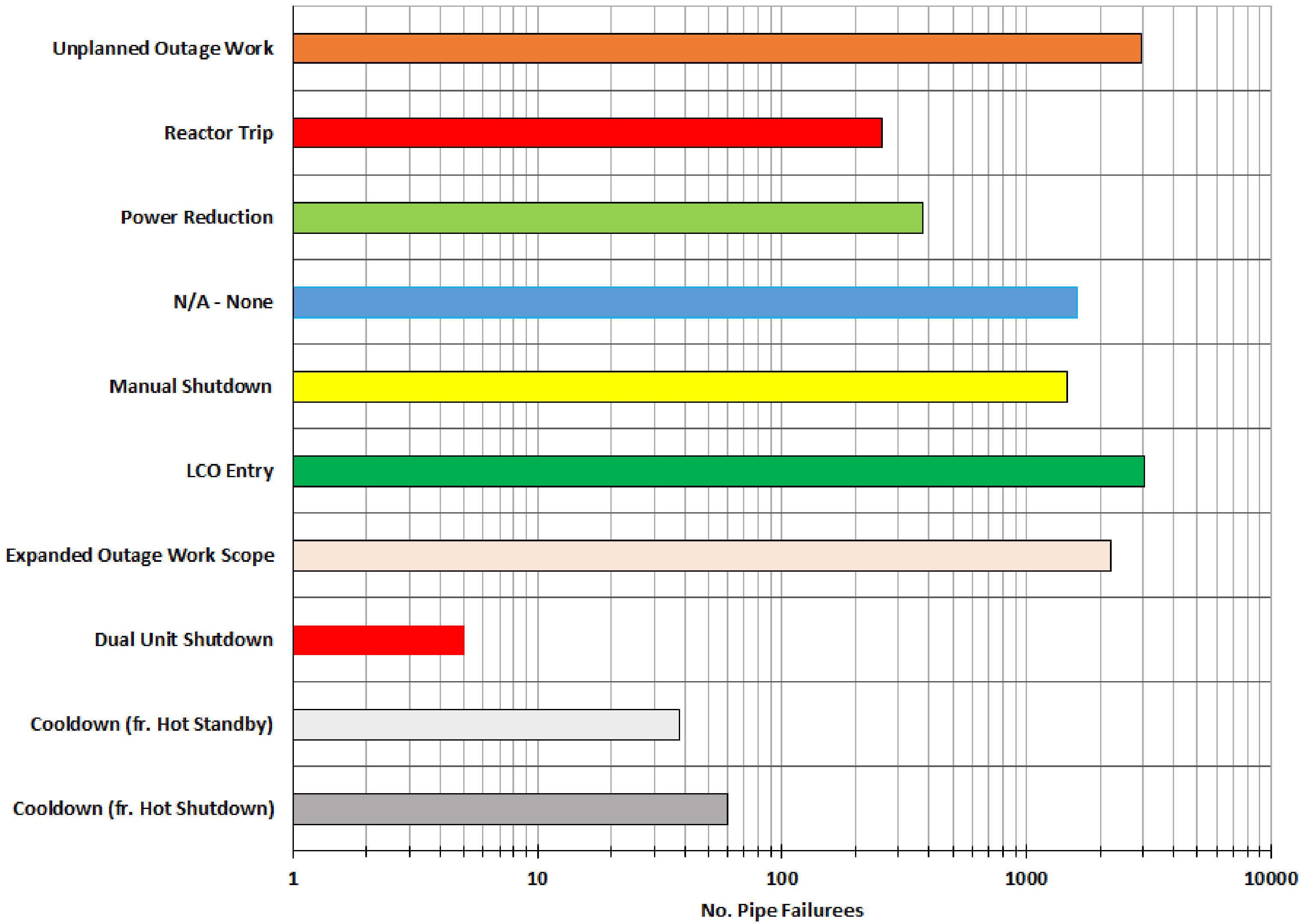


FIG. IV–5. Impact of pipe failure on plant operation.

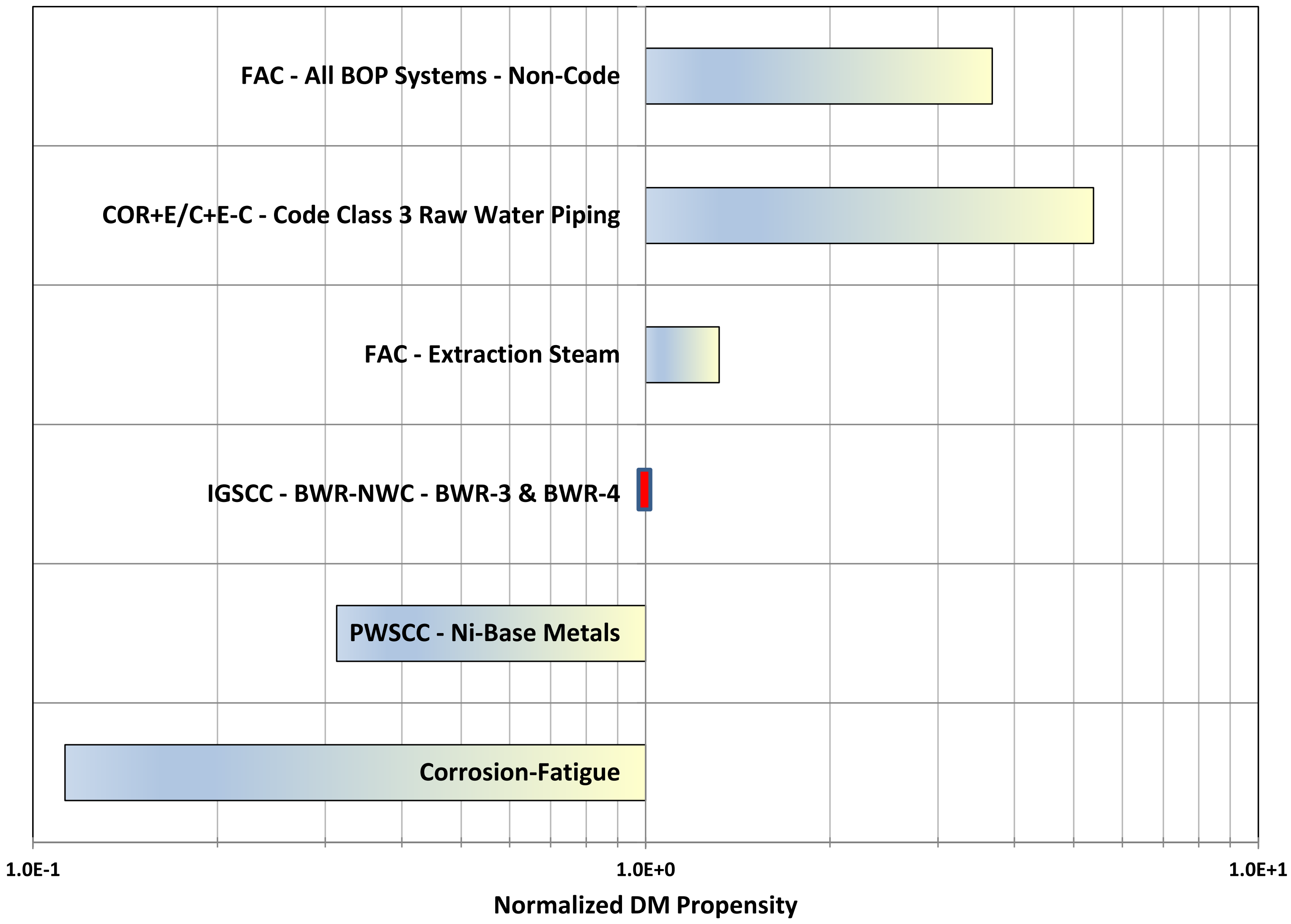


FIG. IV–6. Normalized pipe degradation propensity.

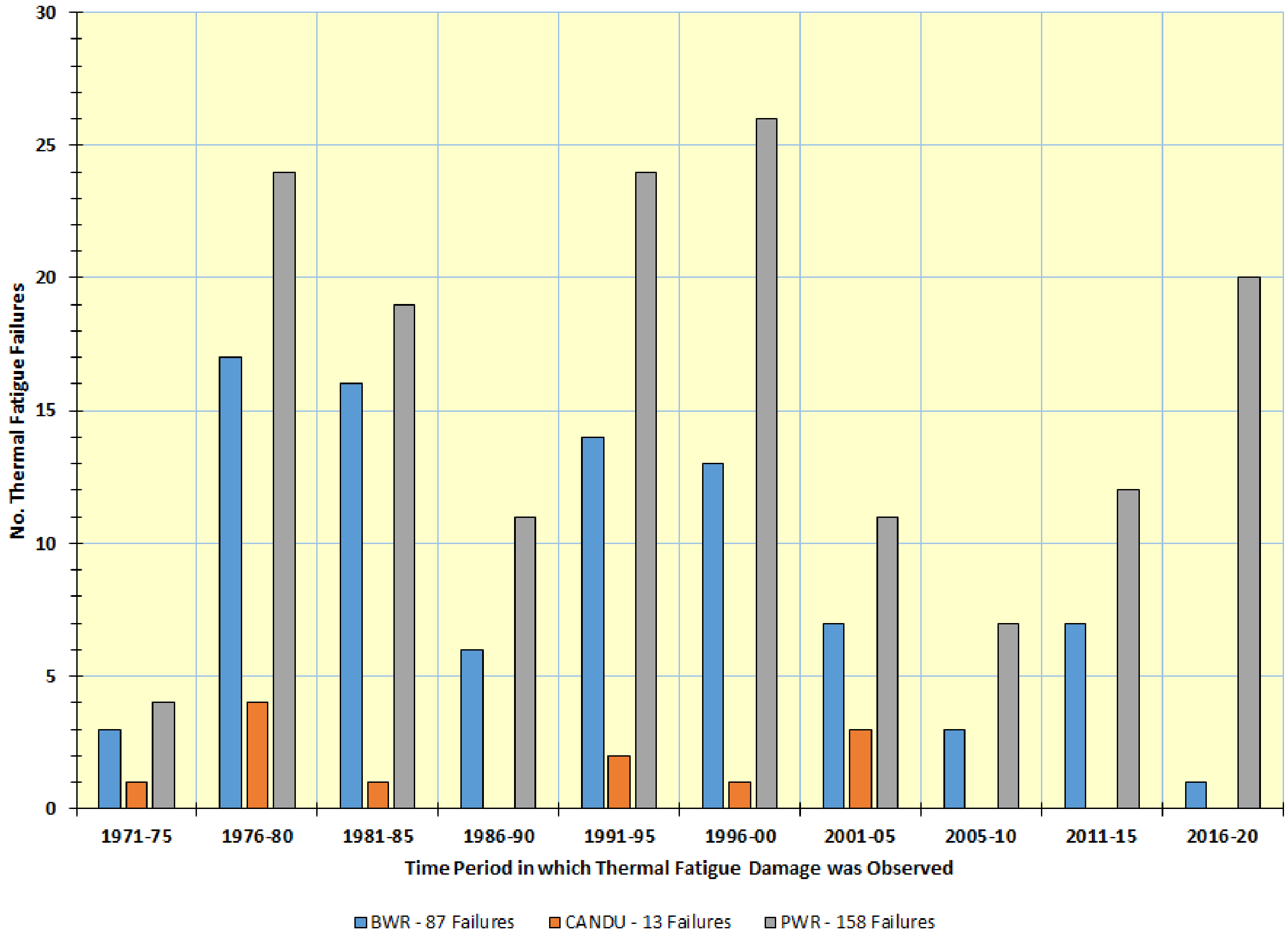


FIG. IV–7. Thermal fatigue OPEX.

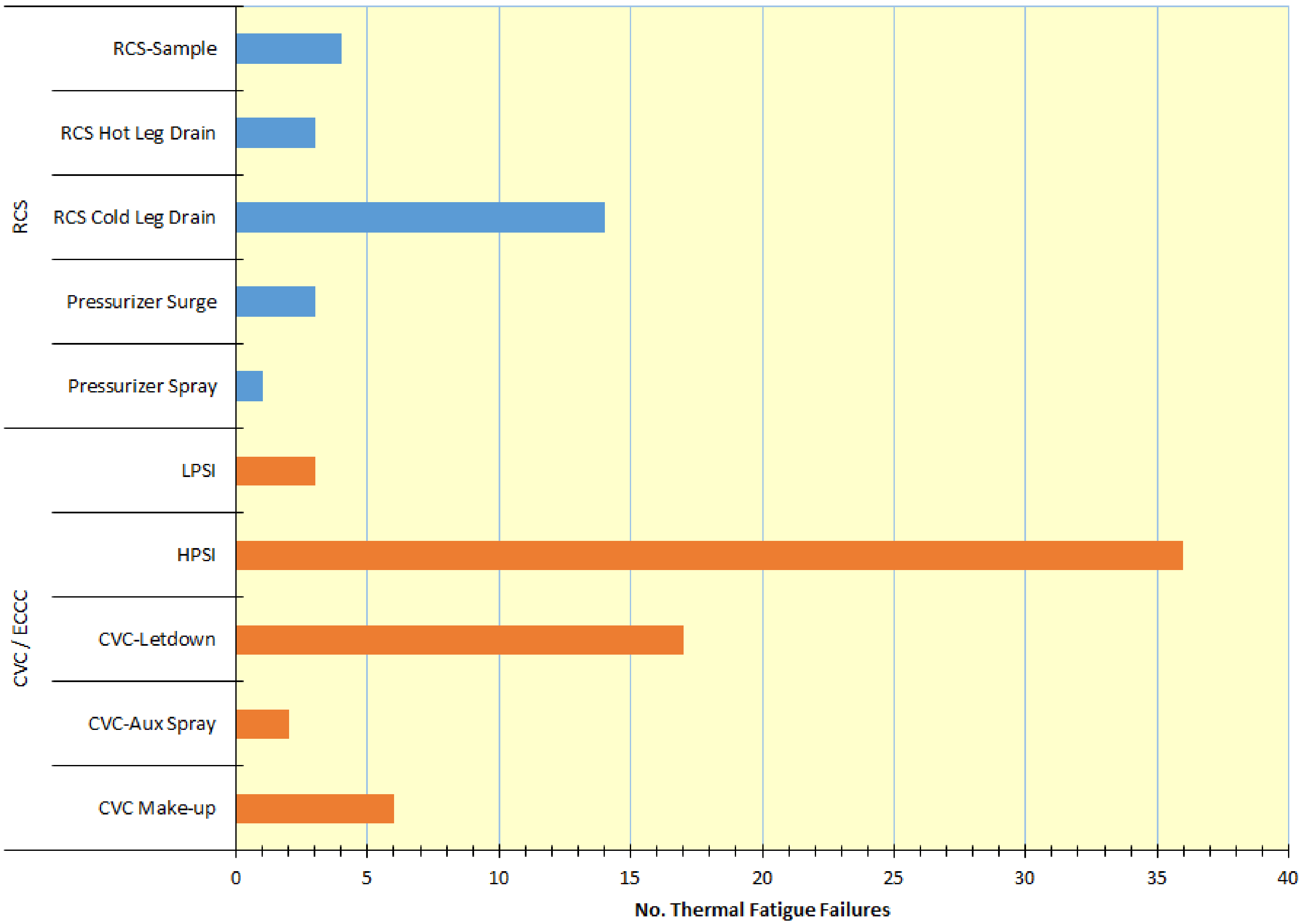


FIG. IV–8: Selected PWR thermal fatigue OPEX organized by affected system.

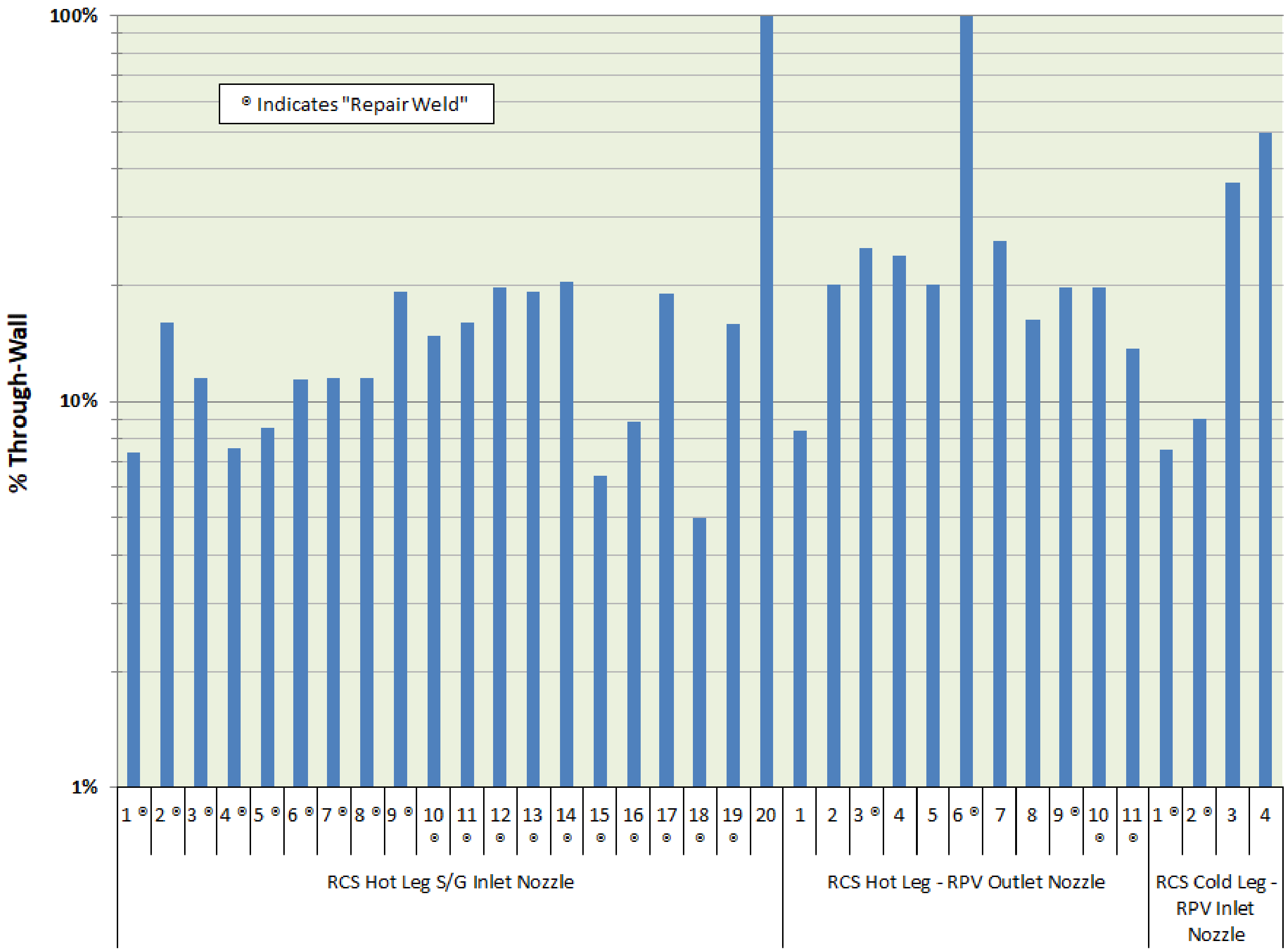


FIG. IV–9. Primary water SCC OPEX by the location of a failed component[5].

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[5] Superscript ® indicates a repair weld (indicating high WRSs). This chart is a summary of specific primary water SCC events. As one example, to date there have been 20 events (1 through to 20) involving primary water SCC in PWR RCS hot leg steam generator inlet bimetallic welds.

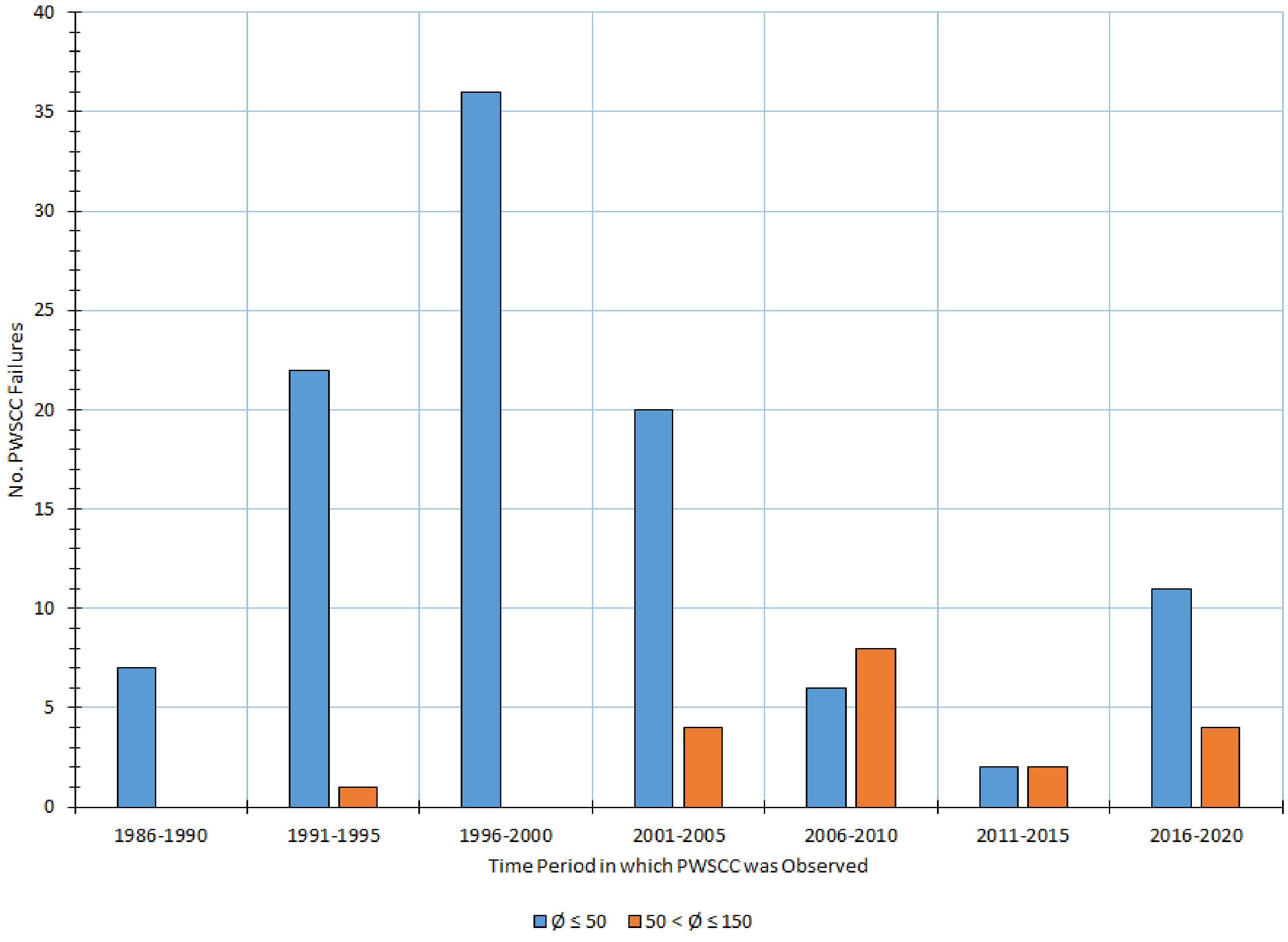

FIG. IV–10. PWSCC failures in small and medium diameter (∅) PWR primary system piping segments.

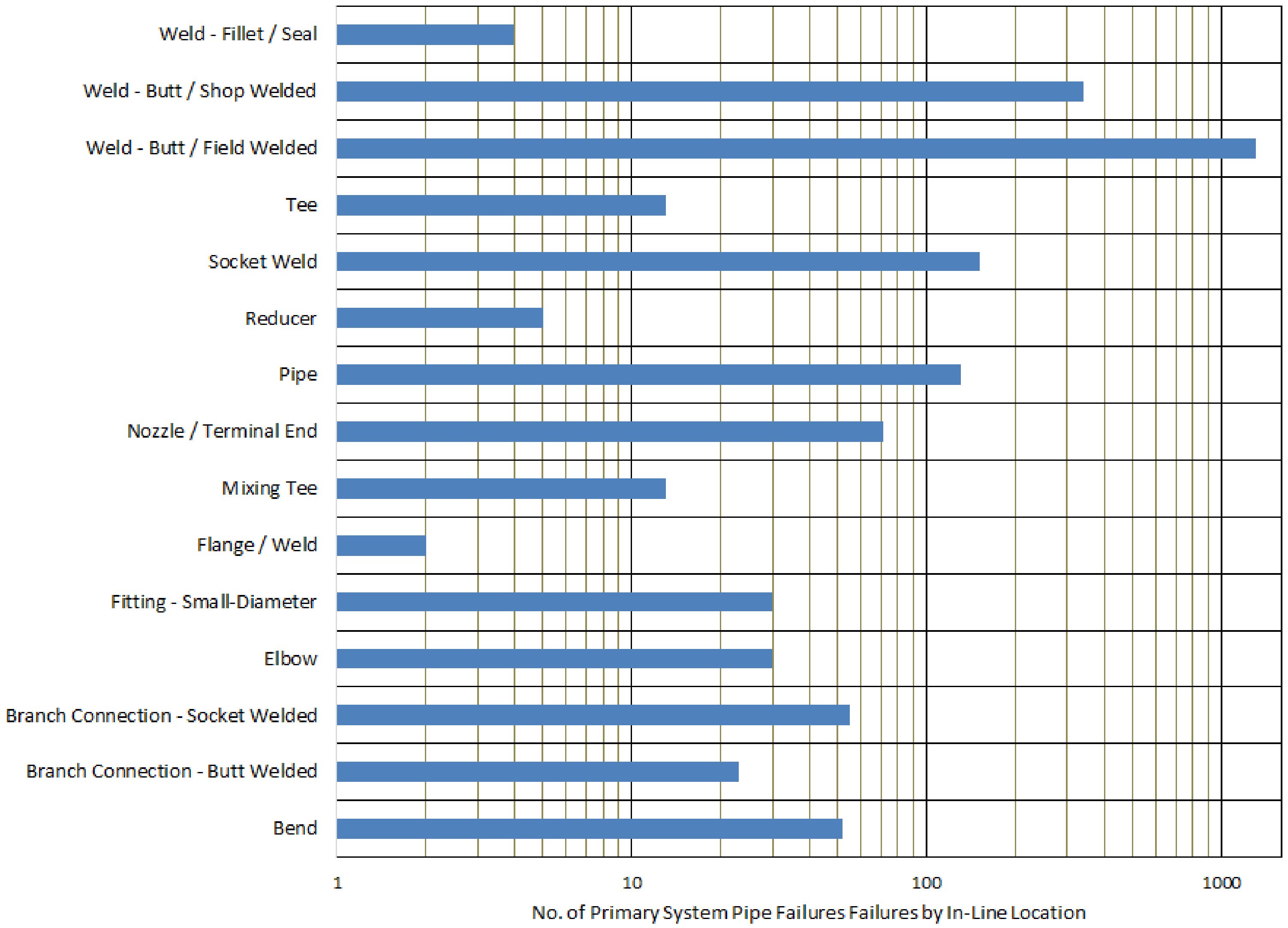

FIG. IV–11. Primary system piping failure data organized by in-line damage location.

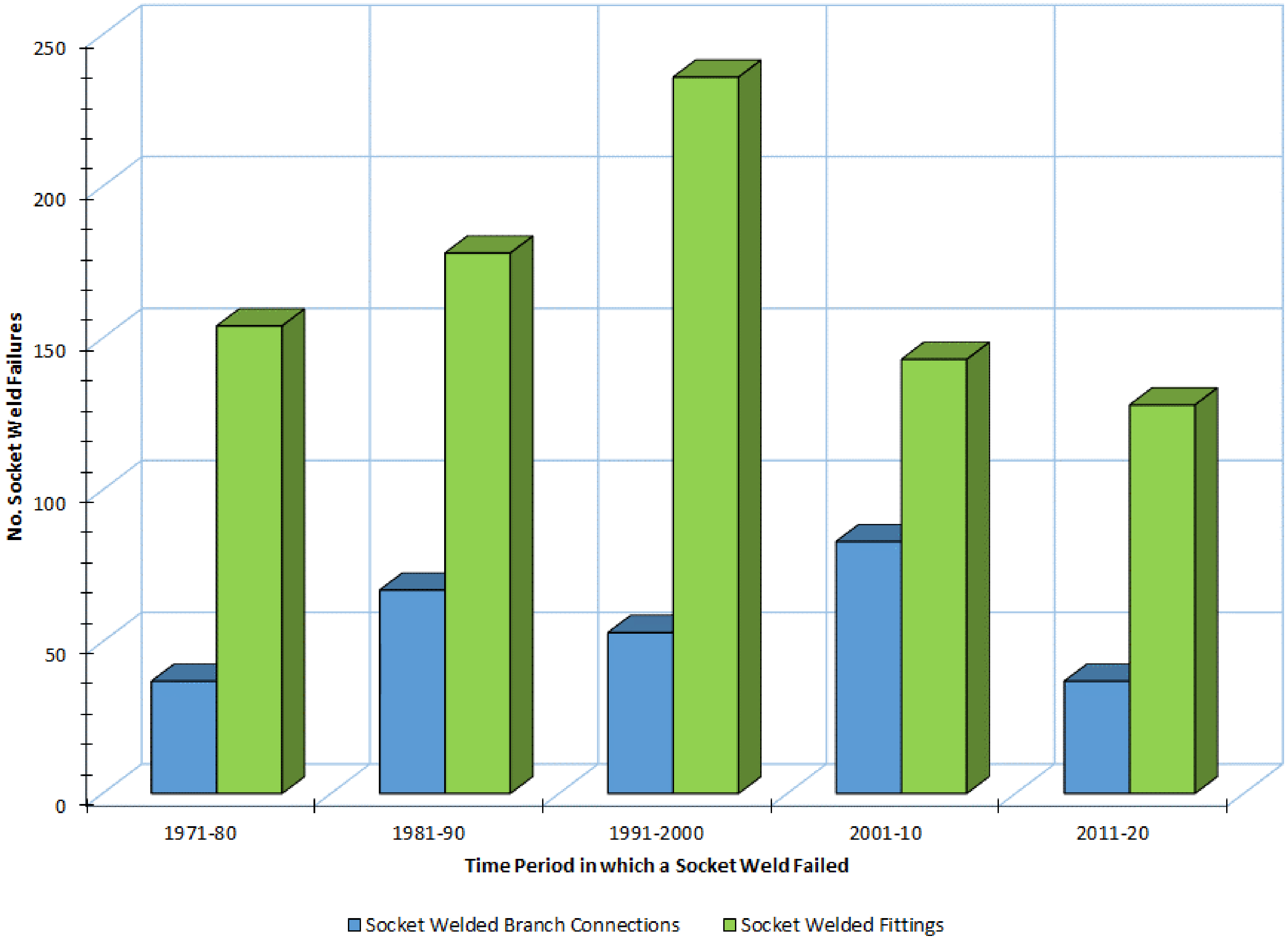

FIG. IV–12. Socket weld OPEX.

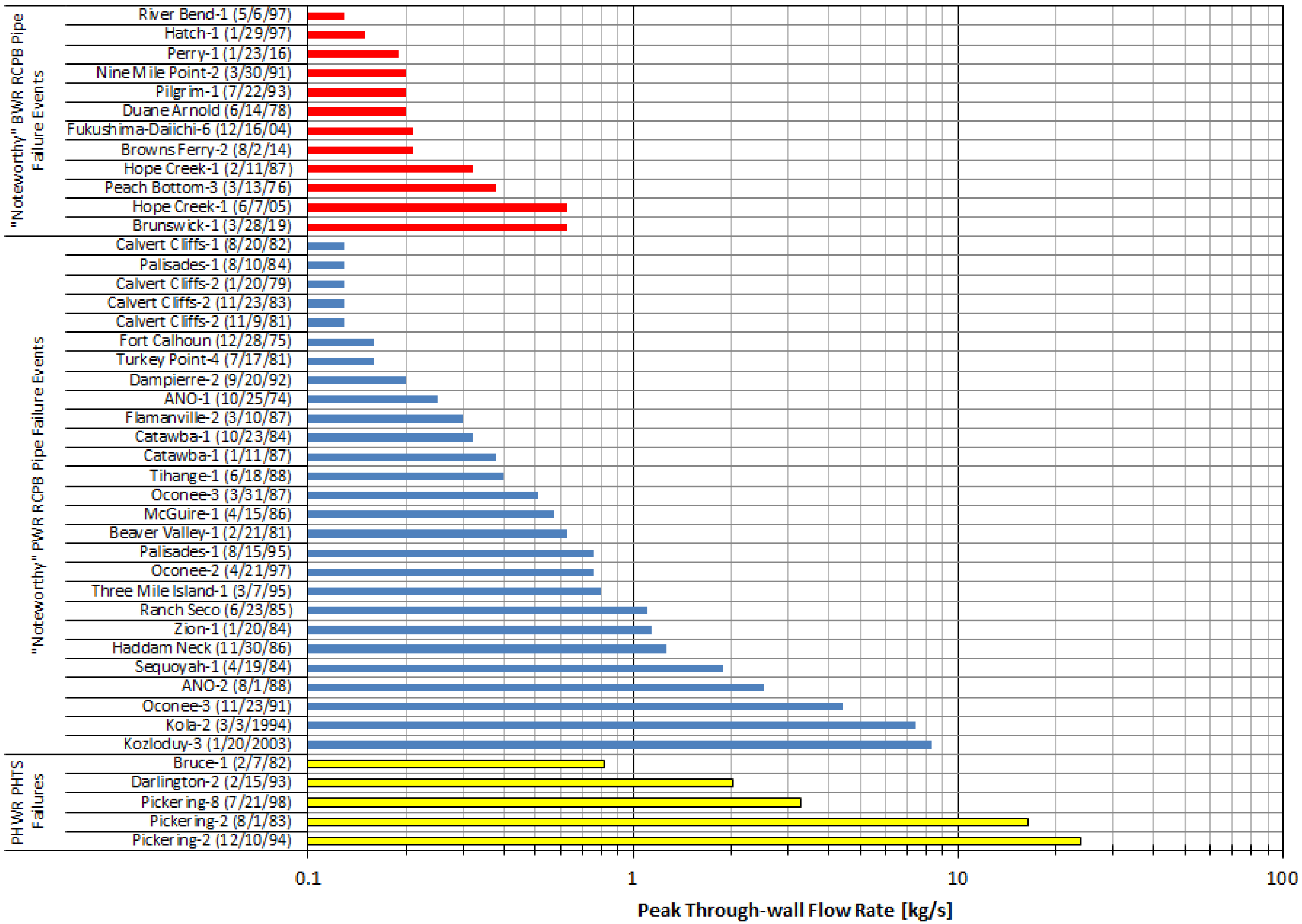

FIG. IV–13. Summary of non-isolable primary coolant system pipe failures (1970–2020).

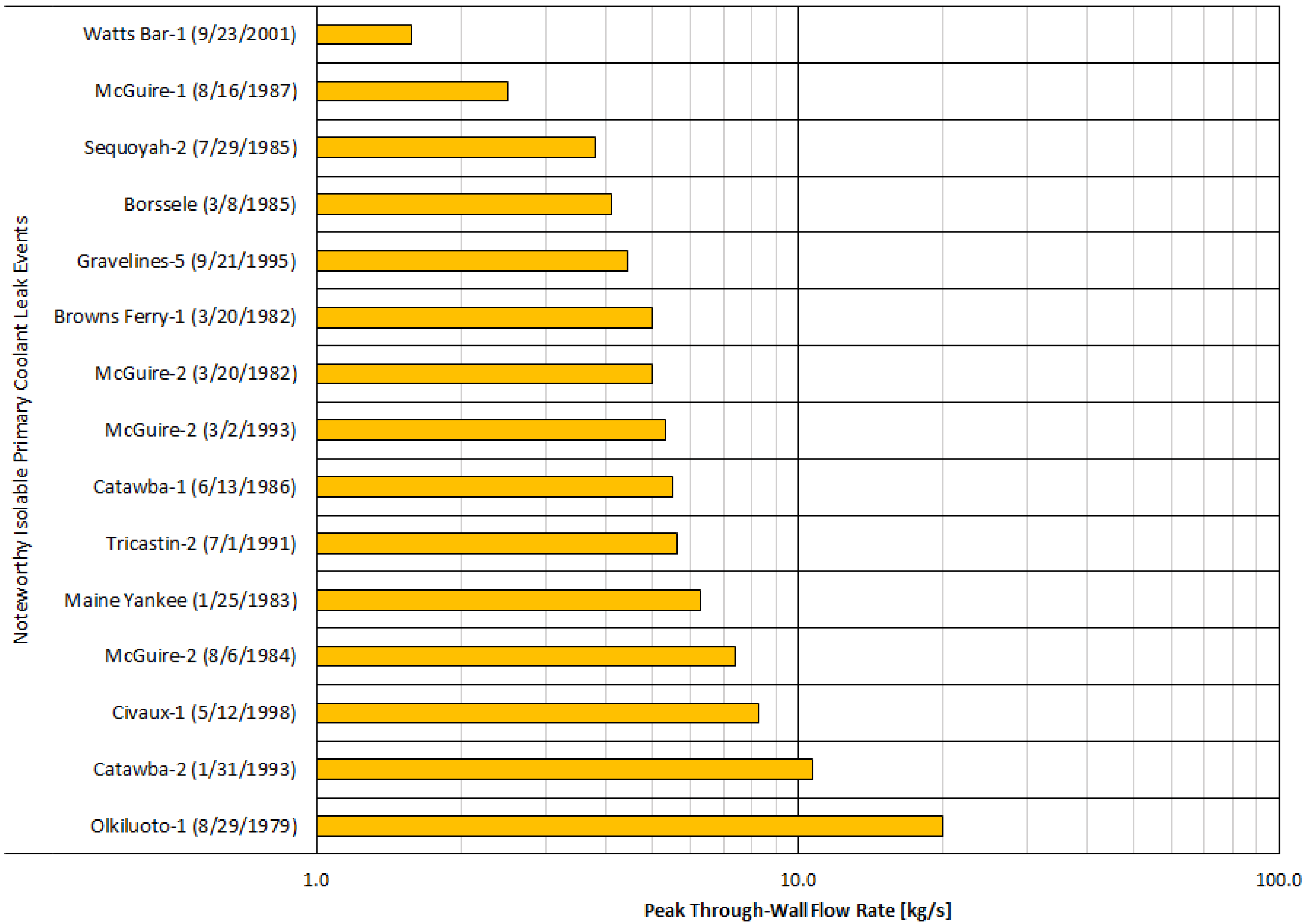

FIG. IV–14. Summary of isolable primary coolant system pipe failures (1970–2020).

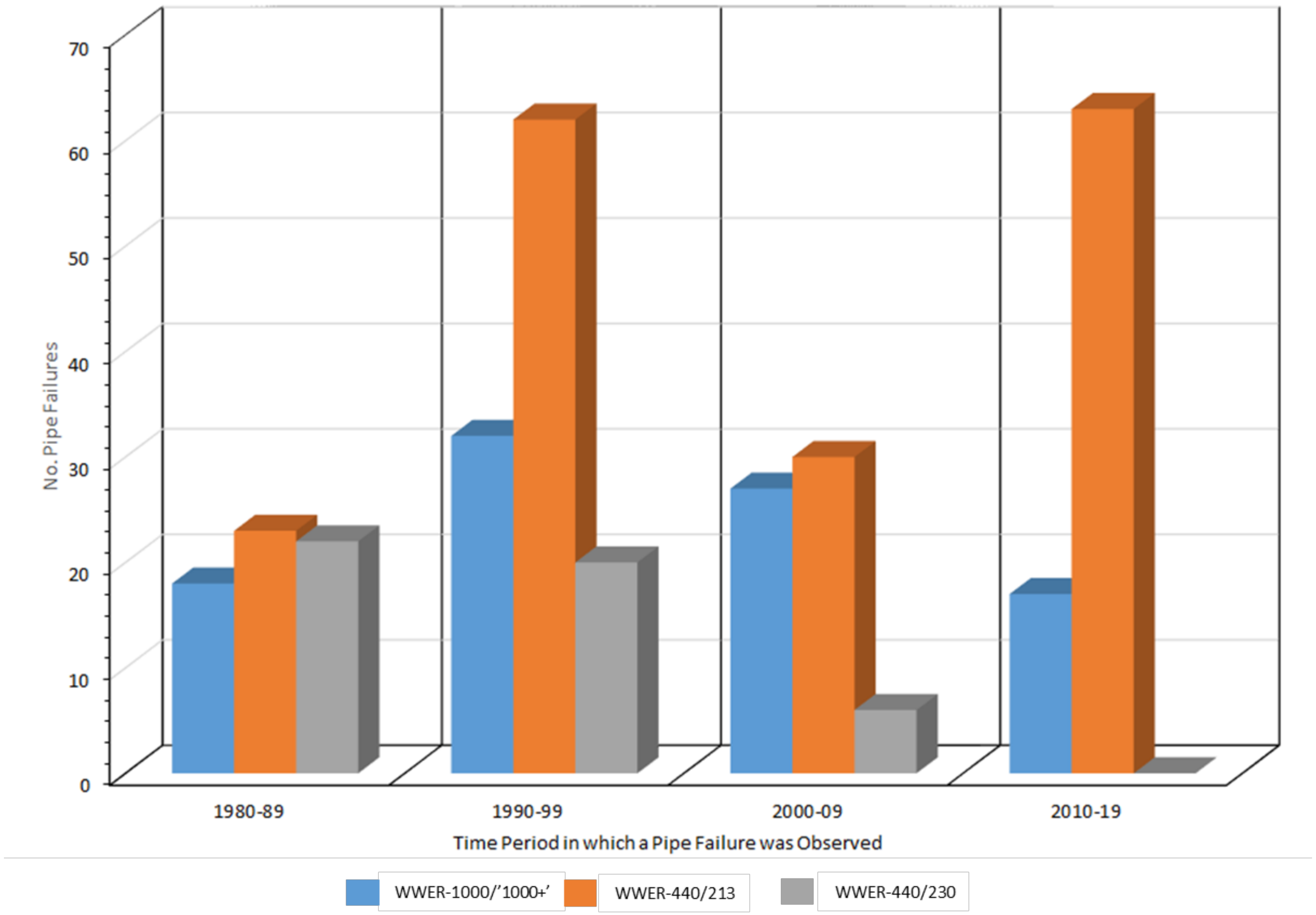

FIG. IV–15. WWER piping OPEX (selected representative events).

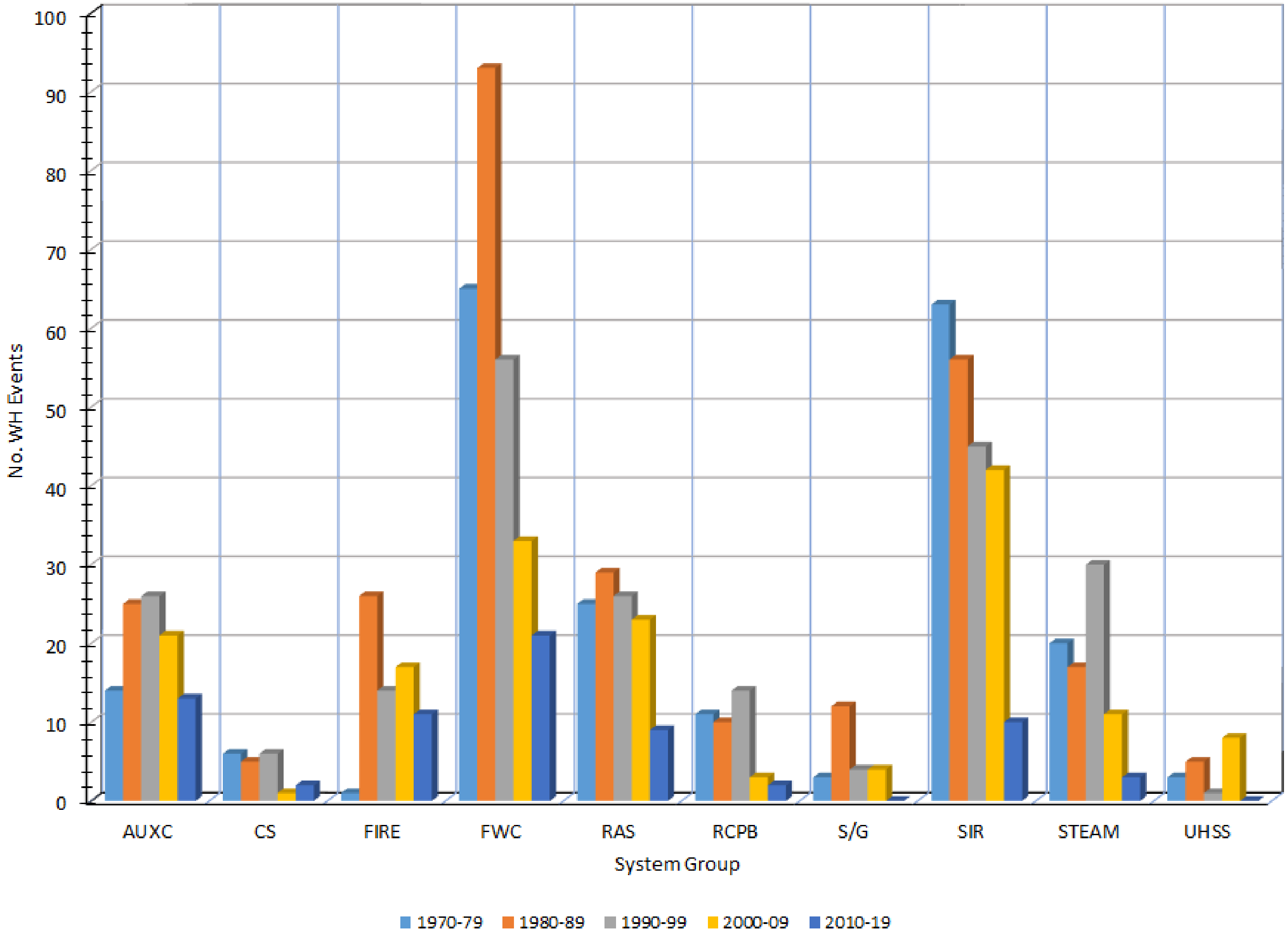

FIG. IV–16. Water hammer experience by system group and time period.

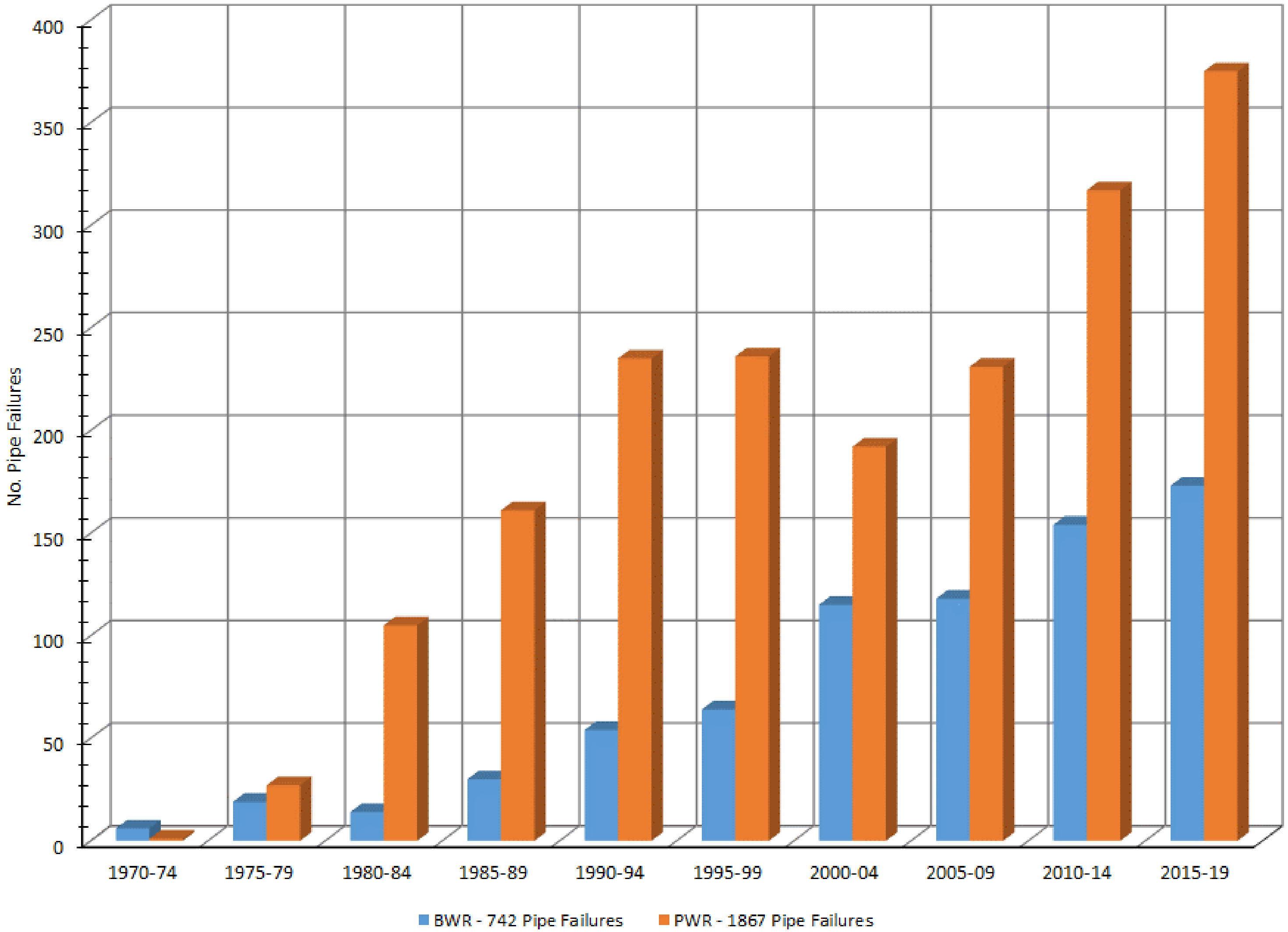

FIG. IV–17. Evolution of the raw water cooling piping system OPEX.

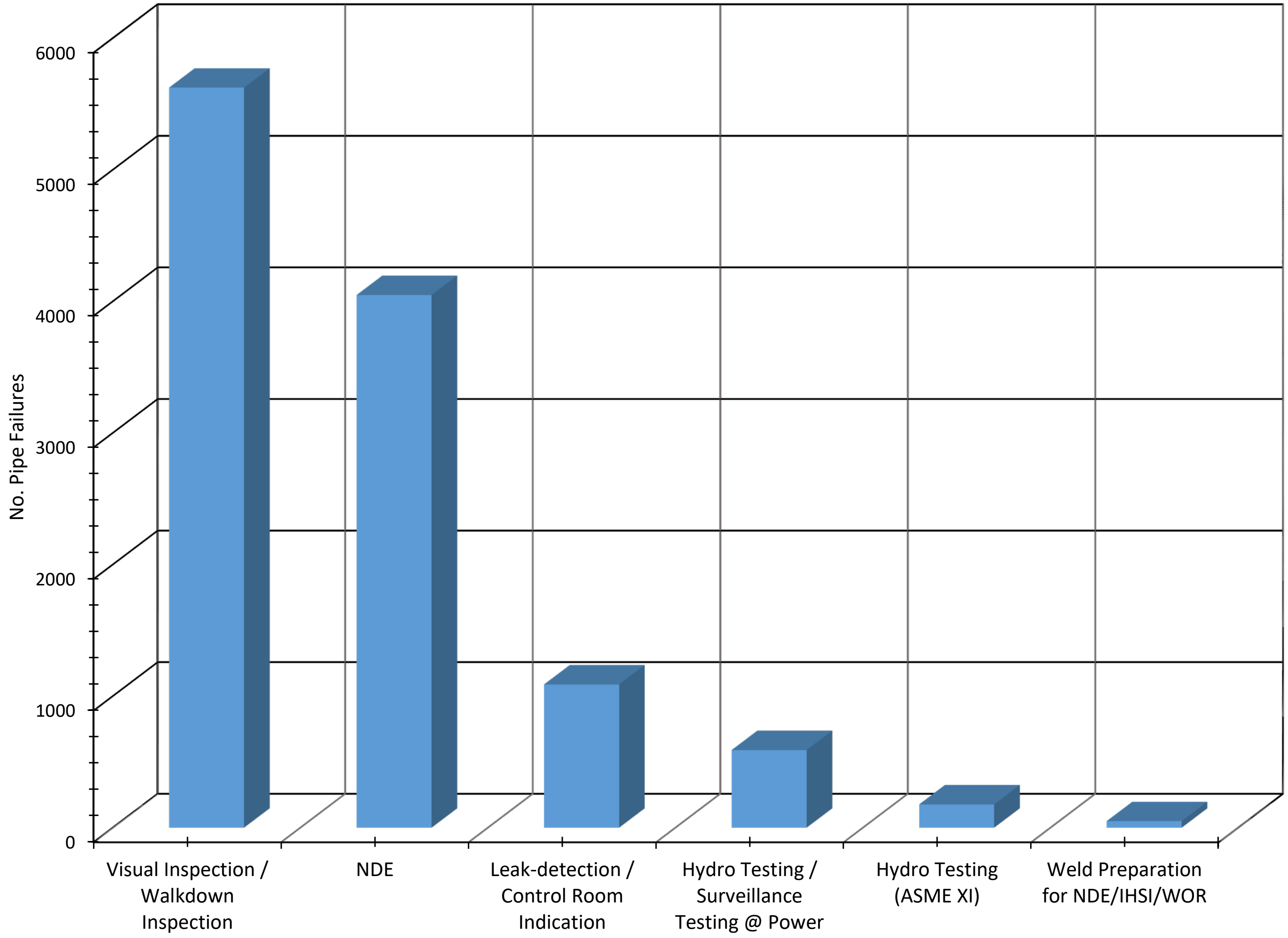

FIG. IV–18. Pipe failure OPEX by method of detection.

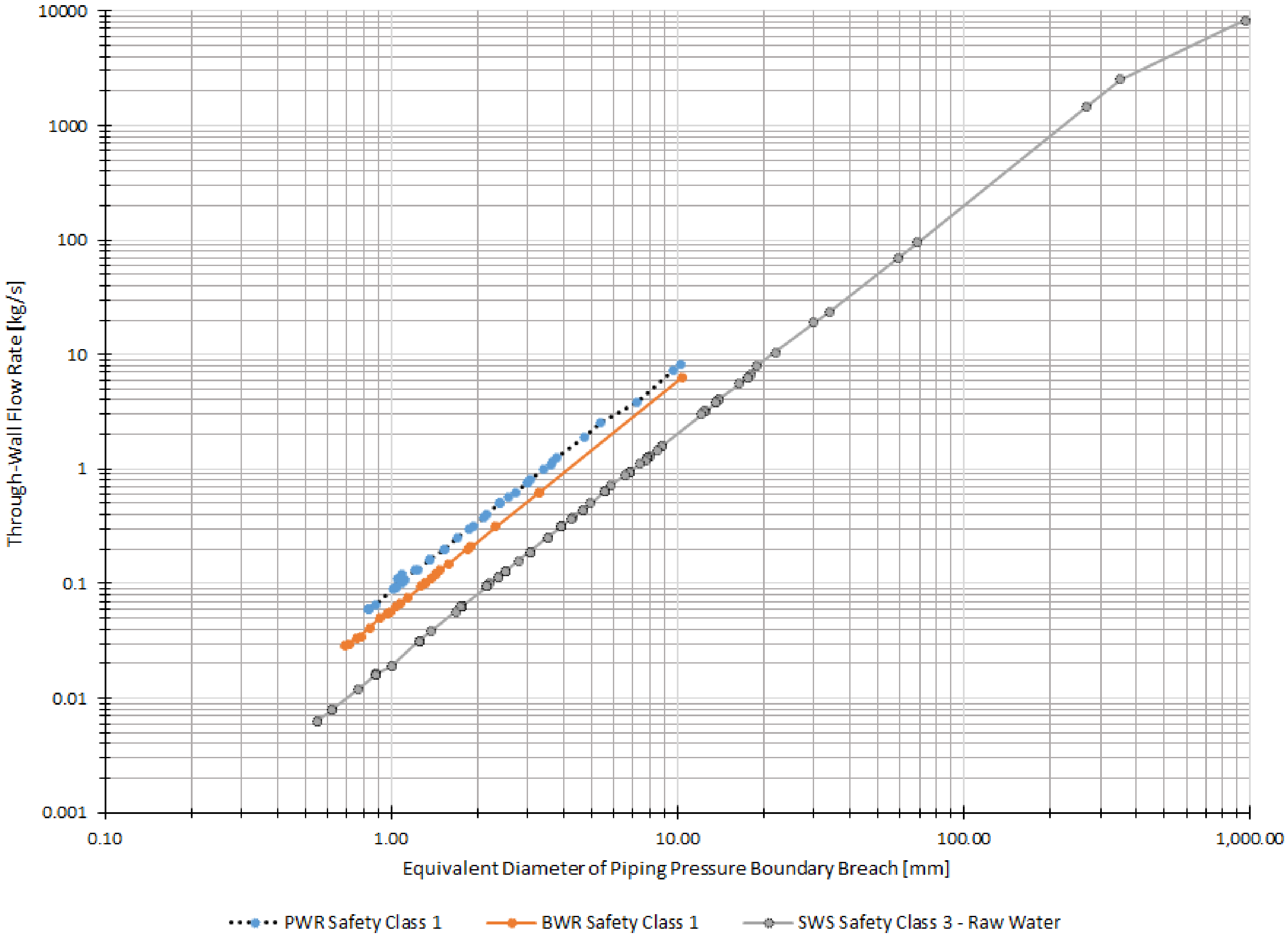

FIG. IV–19. Pipe failure experience by through-wall mass flow rate vs. equivalent diameter of pressure boundary breach.

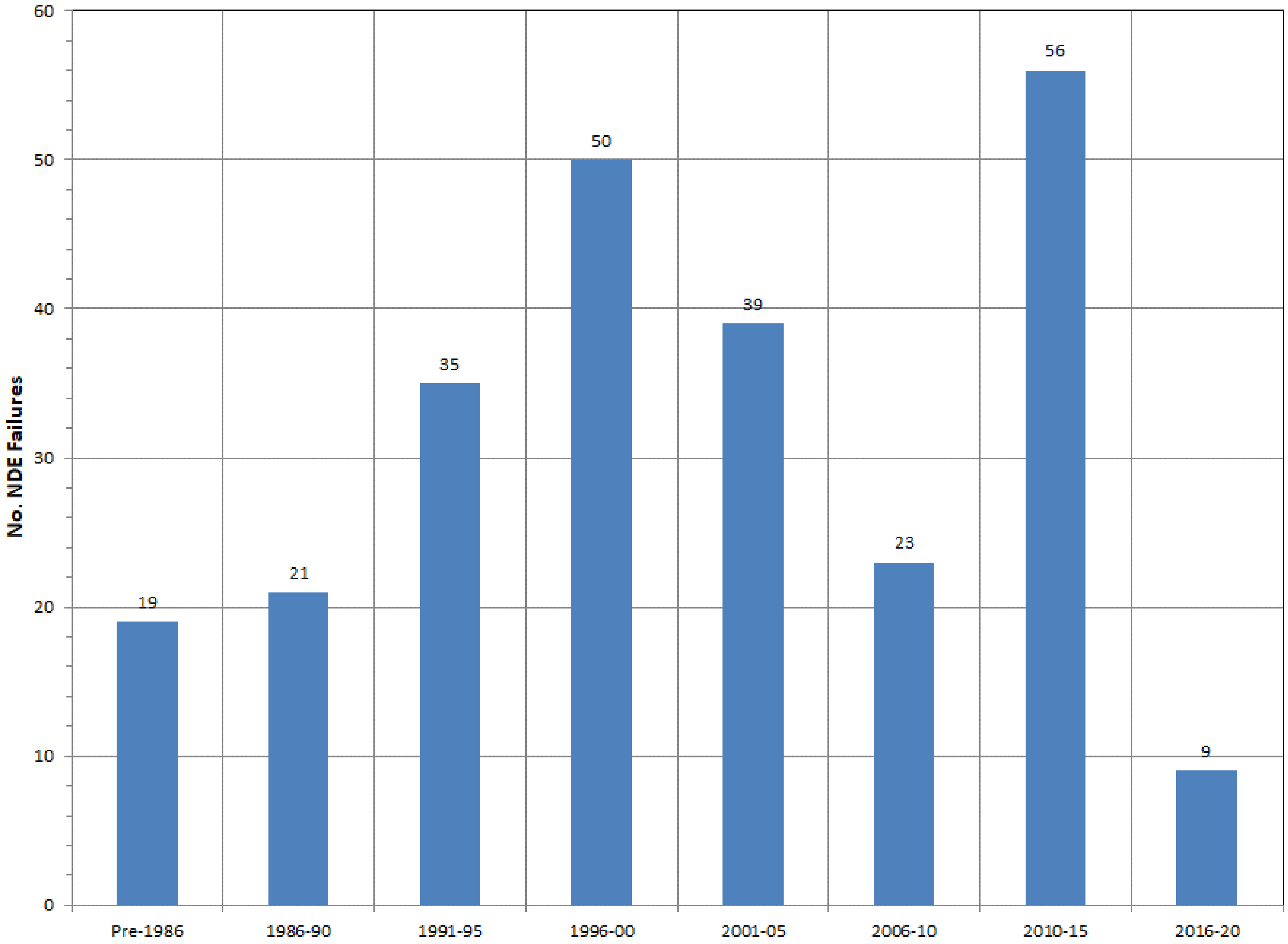

FIG. IV–20. Number of pipe failures attributed to RIM programme failure.

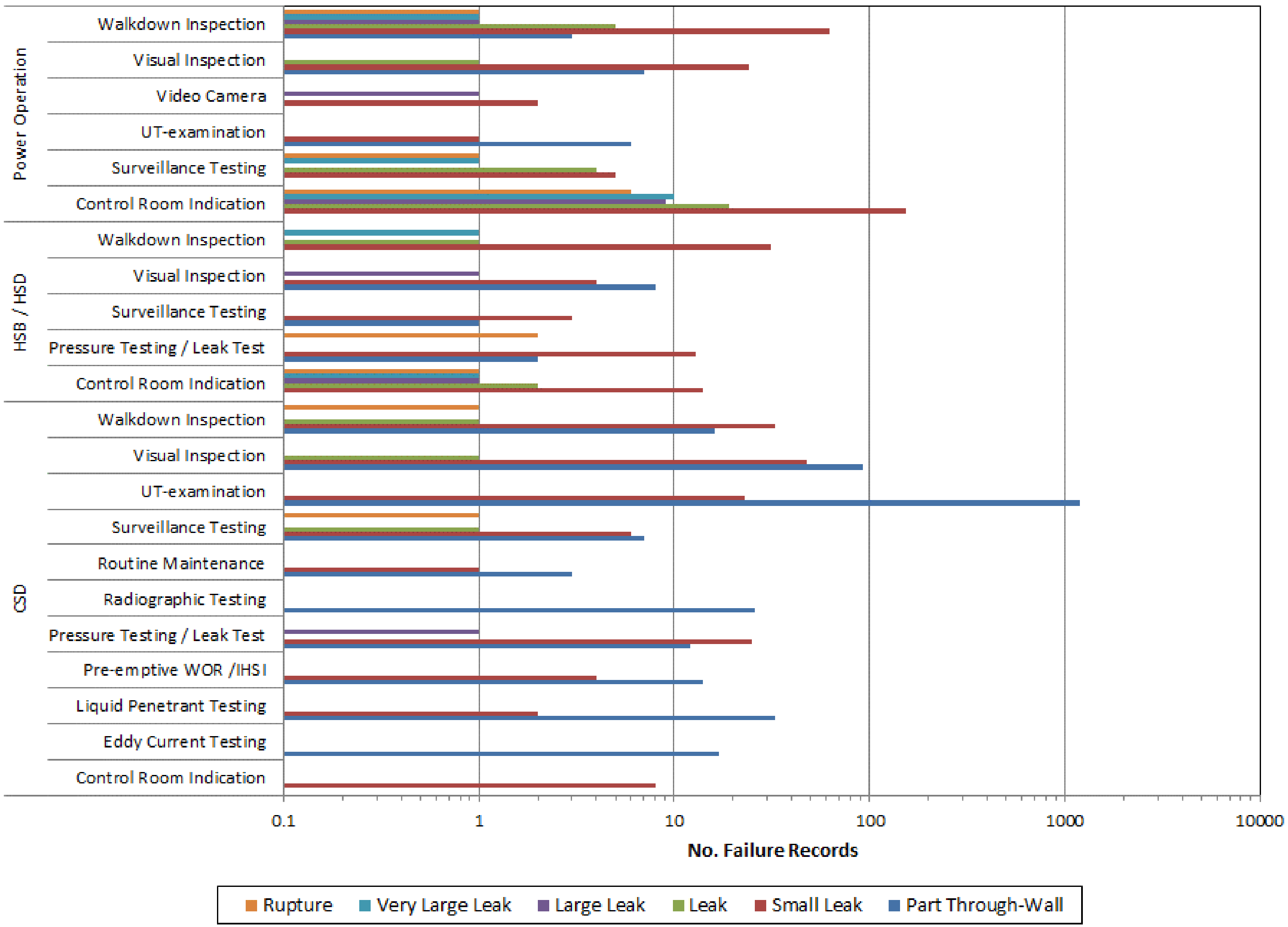

FIG. IV–21. Safety class 1 pipe failures by method of detection.

TABLE IV–1. PWSCC OPEX BY NUCLEAR STEAM SYSTEM SUPPLIER VENDOR AND AFFECTED LOCATION

Nuclear steam system supplier vendor	Primary pressure boundary location	Time period			
		1980–89	1990–99	2000–09	2010–19
Babcock & Wilcox	PZR-INSTR		1	3	1
	PZR-surge line			1	
	RCS hot leg — small diameter instr. penetrations			8	
	RHR			2	
Combustion Engineering/Korea Electric Power Company	CVC make-up				2
	PZR-INSTR	3	22	3	2
	PZR-PORV		1		
	PZR-SPRAY				1
	PZR safety relief valve				2
	PZR-surge line			1	
	RCS cold leg — small diameter instr. penetrations		1		4
	RCS hot leg — small diameter instr. penetrations		15	10	
	RCS-sample		1		`
	RPV-head vent			2	
	S/G-system – small diameter nozzles		5	2	
Framatome	BMI				1
	PZR-INSTR	3			
	PZR-surge line			1	
	RCS Hot Leg			1	
Westinghouse/Mitsubishi heavy industries	BMI			3	
	PZR-INSTR			1	
	PZR-PORV			2	
	PZR-SPRAY			2	
	PZR safety relief valve			6	
	PZR-surge line			2	
	RCS cold leg			2	
	RCS hot leg			8	
	S/G inlet			19	
	S/G-system — small diameter nozzles	1		6	5

Note: PZR — pressurizer; PORV — power operated relief valve; RCS — reactor coolant system.

TABLE IV–2. SUMMARY OF WATER HAMMER EVENTS RESULTING IN MAJOR PRESSURE BOUNDARY FAILURE

Water hammer mechanism	Major pressure boundary failure caused by water hammer	
	Impacted component	No. events
Valve mis-alignment/operation	Elbow	2
	EXJ M (metal expansion joint)	3
	Flange	1
	Nozzle	1
	Pipe	12
	Reducer	1
	Heat exchanger tube	1
	Subtotal:	21
Steam bubble collapse	Elbow	1
	End cap	1
	EXJ-M	5
	Pipe	8
	Rupture disc	2
	Valve body	1
	Weld	4
	Subtotal:	22
Valve_C (valve controller failure)	EXJ R (rubber expansion joint)	1
	Pipe	2
	Pump casing	1
	Reducer	1
	Vacuum breaker	1
	Subtotal:	6
Valve_M (mechanical failure of internals)	EXJ R	1
	Pipe	1
	Pump casing	1
	Weld	1
	Subtotal:	4
Air entrapment/voiding	Rupture disc	3
	Weld	2
	Subtotal:	5
Water column separation	EXJ R	1
	Pipe	1
	Subtotal:	2
	Total no. M-PBF events (out of a total of 945 events):	60

Note: EXJ — expansion joint.

IV–2 SELECTED PIPE FAILURE EVENTS OF HISTORICAL SIGNIFICANCE

This section includes selected pipe failures of historical significance. The term 'significance' is a reflection of combinations of operational impact, and the extent by which industry and regulatory agencies responded in terms of finding acceptable solutions to the long term management of pipe material degradation. Listed in chronological order below, are summaries of selected significant pipe failures that were attributed to flow accelerated corrosion, severe overloading of a section of piping, SCC and thermal fatigue:

- **Flow accelerated corrosion event #1.** On 9 March 1985, a main feedwater isolation following a turbine trip at the Trojan NPP produced a pressure pulse that reached a maximum total pressure of approximately 6.0 Mpa in the heater drain and feedwater system. The pressure surge ruptured a 368 mm outer diameter carbon steel pipe in the feedwater heater drain pump discharge piping and released a steam-water mixture into the turbine building. The system flow velocity was 6.1–7.3 m/s, and the normal operating pressure and temperature at the time of the break were about 3.1 Mpa and 177°C, respectively. The ruptured portion of the piping had been thinned from a nominal thickness of 9.5 mm to about 2.5 mm. Before this rupture, it was believed that only piping carrying two-phase fluid was susceptible to flow accelerated corrosion. Because the ruptured drain pipe carried single-phase fluid, it was not inspected.
- **Flow accelerated corrosion event #2.** On 24 April 1986, Hatch Unit 2 was in steady state operation at approximately 85% of rated thermal power, plant personnel were investigating the report of a large steam leak in the condenser bay area. At that time, the generator or exciter field ground detection relay actuated tripping the main turbine. Consequently, the reactor scrammed and both recirculation pumps tripped due to a trip of the main turbine at greater than 30% power. The relay, which is located in the generator exciter housing, actuated due to moisture buildup from steam condensing in the area around the main silicon control rectifier bridges which supply the main generator field. The steam was from the steam leak under investigation when the scram occurred. The leak was due to a through-wall failure in the sixth stage feedwater heater extraction steam line at the downstream reducer leading to the sixth stage feedwater heater. The through-wall failure was 457 mm long by up to 25 mm wide. The failure was caused by wet steam erosion. The piping in the extraction steam lines was carbon steel with 0.3–0.6% copper.
- **Flow accelerated corrosion event #3.** On 9 December 1986, a main steam isolation valve failed to close at Surry Unit 2, and the resulting increased pressure in the steam generator collapsed the voids in the water. This caused the system pressure to surge beyond the normal operating pressure and led to a catastrophic failure of a 90° carbon steel (SA-234 Grade WPB) DN450 elbow in the suction line to the main feed pump. At the time of the event, the reactor was at full power and the feedwater was single phase, with a flow velocity of about 4.3 m/s, a pH level in the range of 8.8–9.2, an oxygen content of about 4 ppb, and a coolant temperature and pressure of approximately 188°C and 3.1 Mpa, respectively. Ammonia was used for the feedwater treatment. The examination of the ruptured elbow showed that the wall thinning was relatively uniform except in some local areas. The wall thickness of the elbow was reduced from a nominal 13 mm to 0.38–1.22 mm in small local areas and to 2.3 mm in larger areas. Eight workers were burned by flashing feedwater, four of whom subsequently died. The flashing feedwater interacted with and disrupted the fire protection, security and electrical distribution systems.
- **Flow accelerated corrosion event #4**. On 28 May 1990, Loviisa Unit 1 experienced a main feedwater system pipe break in the turbine building. The break location was downstream of a flow control orifice (Fig. IV–22). About 70 m^3 of water from the secondary circuit was discharged into the turbine hall. The reactor was scrammed manually (and shut down to cold condition) and the leak was isolated in 17 min. The pipe break dynamic caused damage to electrical cables and some small diameter piping. Non-destructive examinations to detect pipe wall thinning had been carried out since 1982. However, during the first years of the pipe wall thinning inspections the programme mainly addressed two-phase flow systems (e.g. steam lines). After the feedwater pipe rupture at Surry 2 in 1986 the number of inspections of single-phase flow systems were increased. For some reason flowmeter flanges and corresponding structures were not covered by the programme.

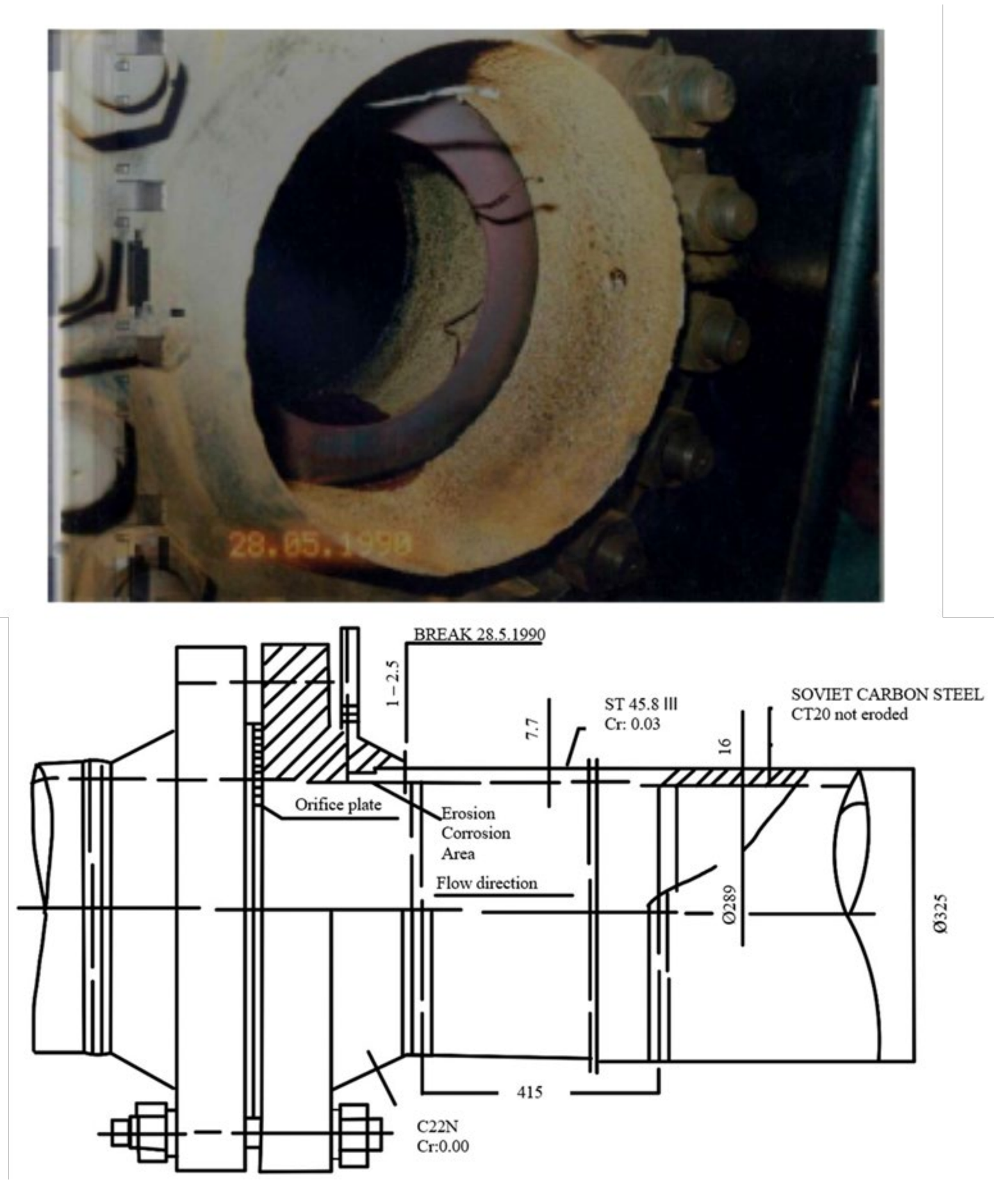

FIG. IV–22. Double-ended guillotine break of feedwater line (adopted from STUK, 2005).

— **Flow accelerated corrosion event #5.** On 21 April 1997, Fort Calhoun NPP experienced a DN300 extraction steam line rupture in the turbine building. The rupture occurred in the fourth stage extraction steam piping, in a DN300 sweep elbow (radius equal to five times the pipe diameter). When operators heard a loud noise from the turbine building, the reactor was manually tripped. The rupture (estimated by the plant owner to be approximately 0.9 m long) occurred at the outer edge of a large radius bend in the extraction steam line. Significant steam impingement damage to balance-of-plant motor control centres 4C3 and 4C5 occurred. Additionally, collateral damage was experienced in several cable trays and pipe hangers, and insulation containing asbestos was blown throughout the turbine building. The fire suppression system actuated in the area and was subsequently isolated. Intermittent electrical system grounds occurred during the event. Insulation, containing asbestos, was blown throughout the turbine building. No automatic safety system actuations occurred during the event. However, portions of the fire protection system were actuated throughout the turbine building due to the heat and temperature rise associated with the steam rupture. The steam from the rupture caused seven wet pipe sprinkler heads to actuate in the basement level of the turbine building. These sprinkler heads were in the immediate vicinity of the steam leak and were designed to actuate at 71.1°C. The team noted that the steam in the vicinity of these sprinkler heads exceeded the actuation temperature of the sprinkler heads. The deluge system for the turbine lube oil reservoir also actuated at the time of the event.

— **Flow accelerated corrosion event #6.** While Mihama Unit 3 was in operation at the rated thermal output, a fire alarm sounded in the central control room on 9 August 2004. The control room operators determined that the alarm-generated area was on the second floor of the turbine building and checked the area to find that the building was filled with steam. Thus, it was judged that there was a high possibility of steam or high temperature water leakage from the secondary piping. The operator started emergency load reduction. While those operations took place, a 3A *Steam Generator Feedwater < Steam Flow Inconsistency Trip1* alarm was generated and the reactor and then the turbine shut down automatically. The operator made an inspection in the turbine building and confirmed a ruptured DN500 feedwater pipe from the fourth feedwater heater to the de-aerator running near the ceiling on the de-aerator side at the second floor of the turbine building as shown in Fig. IV–23.

FIG. IV–23. Ruptured feedwater pipe (JNES, 2005).

— **Hydrogen assisted cracking event.** In March 2019 a brittle fracture of a DN25 pipe coupling in the reactor vessel level indication system occurred at a US BWR. At the time of the event the reactor had been in commercial operation for 43 years. On 28 March 2019, while operating at 100% reactor power, the narrow range reactor water level instrument failed. It is an instrument to tap off the steam space of the reactor vessel. Drywell pressure and drywell floor drain leakage increased. Investigation inside the containment determined that a DN25 pipe coupling on the steam side of a reactor level condensing chamber experienced a 360° circumferential separation at the approximate centre of the coupling. This failure resulted in a primary system leak rate of approximately 0.5 kg/s. The post-event metallurgical report determined that the coupling showed no evidence of localized plastic deformation. The coupling experienced hydrogen embrittlement and did not exhibit a 'leak before break' failure mechanism. The coupling was made of a shape memory alloy material composed primarily of nickel-titanium-iron (Tinel material). Examination of the failed coupling was conducted at a metallurgical laboratory. Microhardness testing, visual microscopy and scanning electron microscopy were used to characterize the failed material. The examinations confirmed that the failure was caused by hydrogen embrittlement. This was supported by the transgranular cleavage on the fracture surface, high hardness values in the region exposed to the process fluid, and a hydrogen rich environment, which are all consistent with hydrogen embrittlement. The root cause of this event was that the selection of Tinel was inappropriate for long term application in a high temperature process that contains elevated levels of hydrogen.

— **Overloading Event #1:** On 28 April 1970, during hot hydraulic testing of the unfuelled H.B. Robinson-2 reactor, a pipe nozzle, counter-bored and tapered from Schedule 80 to Schedule 40, between a pressurized main steam line and a safety valve failed completely; a 360° 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 15.24 cm pipe nozzle which connected the safety valve to the 66.04 cm main steam line. Plant recovery was accomplished in an orderly manner. The secondary system pressure was about 6.2 Mpa with the main steam isolation valve and bypass valves closed. The RCS temperature decreased in minutes from 282–159°C. In a little over 9 min, the pressurizer pressure dropped from 15.3–12.8 Mpa, and the steam generator boiled dry. The section of pipe which failed was insulated to a point approximately 75 mm 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° wide and was directed about 45° 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.

— **Overloading Event #2:** On 2 December 1971, during hot functional testing of the unfuelled 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 350 × 350 × 200 mm thick walled 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 a significant volume 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.

— **Overloading Event** #3. On 24 September 1996, Oconee Unit 2 experienced a DN450 heater drain steam line rupture when maintenance workers were making adjustments to valves in the basement of the turbine building. The unit had just been restarted following a routine maintenance outage. The NRC Augmented Inspection Team reached the conclusion that the pipe rupture was caused by water hammer that may have resulted from inadequate procedures for system startup. The operators may have followed procedures as written, but the procedures might have been inadequate to prevent water hammer from causing a rupture, even in a new pipe. A precursor event occurred in May of 1996 when a water hammer event in the second stage reheater drain system caused a support to fail in the area of the current pipe rupture. During the July 1996 startup, the facility engineers spent considerable time working with operations to find a better way of realigning flow from the second stage reheater drain tank from the condenser to the feedwater heaters. The general procedure for feed-forward of the drains was used, and it was decided to close two manual valves prior to startup in an attempt to eliminate possible water hammers. During the 24 September 1996 startup, the same procedure was used; however, the procedure had not been modified to include the specific guidance about system pressures and valve opening timing. The pipe failure mode was a 100% ductile rupture. There was no evidence of wall thinning due to flow accelerated corrosion. The failure investigation team's conclusion was that severe water hammer was the immediate cause of the pipe rupture

— **Overloading Event #4:** On 16 April 2005, Kori Unit 1 was in the early stages of returning to power following completion of the twenty-third refuelling and maintenance outage when manual operation of the main steam power operated relief valve (PORV) was performed at a low speed to reduce the pressure of the main steam line with a main steam isolation valve and five main steam safety valves, which remained closed. When the pressure in the main steam line reached about 7.12 Mpa 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 the 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 refuelling water storage tank 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. Although the release of radiation did not occur, it would be meaningful to find out the root cause of such incident by calculating the transient hydraulic loads resulting in the unwelcome failure of the pipe line in the downstream of the PORV for the purpose of establishing the appropriate regulatory action. 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, and (4) the crushed region of the refuelling water storage tank and adjacent part were confirmed to be safe through the non-destructive test. During the incident, including safety injection and reactor trip, the key safety functions were maintained via prompt operator actions and the plant entered hot standby condition.

— **SCC Event #1**. At Duane Arnold NPP a slowly increasing drywell unidentified leakage had been monitored since 1 May 1978. On 14 June 1978 an increase in unidentified leakage was observed. The leakage increased from approximately 0.06 kg/s to approximately 0.2 kg/s during the day of 14 June 1978. The plant initiated an investigation to determine the possible cause of the increase in leakage. The investigation progress was discussed in meetings during the week. A decision was made late on Friday, 13 June, to wait until Monday to see if the leakage would remain steady. The leakage at that time was about 0.21 kg/s in comparison to the technical specification limit of 0.3 kg/s. At 00:55, 17 June 1978, during weekly control valve testing, an automatic reactor scram occurred due to problems in reactor protection system relays associated with the testing. The Chief Engineer was notified approximately 30 min after the shutdown. The Chief Engineer made the decision to reduce reactor pressure and to enter the containment and investigate the leakage. Based on the week's observation of drywell leakage and the fact that the plant was shut down, it seemed prudent to look for the leak rather than wait until the leakage approached the technical specification limit of 0.32 kg/s. The survey of the drywell revealed a leak in the area of the N2A nozzle (1-of-8 recirculation inlet nozzle-to-safe-ends). The next several hours were spent removing shield blocks and insulation to determine the actual source of the leak. Preparations then began for possible repairs. After the leak in the safe-end was discovered, the leaking safe-end and the other seven safe-ends were examined in place by UT. All but two safe-ends showed indications large enough to be considered rejectable according to Code Standards. Subsequently, all safe-ends were removed. Destructive examinations of the affected safe-ends revealed that all eight safe-ends had inside surface cracks which extended essentially completely around the circumference of the design; these cracks were located in the creviced region of the safe-end design. The depth of the cracks typically ranged from 50–75% of the wall thickness, except for the leaking safe-end where a through-wall cracking was present at an 80° segment of the circumference.

— **SCC Event #2**. In response to a ruptured DN400 extraction steam pipe, the Santa Maria de Garona NPP was manually shut down on 9 February 1980. Following the repair, the plant startup was initiated on 11 February 1980 at 17:00 hours. On 12 February 1980, at 04:00 hours, with a reactor pressure of 2.9 Mpa, a routine leakage inspection was carried out, discovering several leaks in valve packings located inside the drywell. The pressure was lowered, the leaks were repaired and pressure was restored. At 19:00 hours on 12 February 1980, with a reactor pressure of approximately 5.9 Mpa, a significant increase of the leakage into the drywell was noted. The pressure was lowered and the origin of the leakage was investigated. On 13 February the leak was finally identified as a through-wall cracking in the safe-end of the nozzle N2-D of the recirculation system. The crack measured 35 mm in length and was located in the lower part of the nozzle. The leakage through the crack of the nozzle N2-D safe-end was 0.05 kg/s. The failure had been caused by intergranular SCC.

— **SCC Event #3**. On 7 October 2000, plant personnel at V.C Summer NPP identified an accumulation of boric acid near the 'A' loop of the reactor vessel on the 125.6 m elevation of the containment. Unidentified RCS leakage had been measured during the cycle in the range of 0.02 kg/s. It was subsequently confirmed that the boric acid deposits originated from a through-wall cracking located in the alpha loop reactor vessel nozzle weld. It was estimated from isotopic and tritium analyses that the through-wall leak had initiated about one year prior to discovery. The reactor vessel nozzles, made from low alloy steel (SA508 Cl 2), had been 'buttered' at the manufacturer during fabrication using Alloy 182 material. The butter was stress-relieved along with the vessel, and a J-groove was machined as the weld preparation for field welding from the outside surface. The stainless steel recirculation piping (SA376 Type 304N) was welded to the buttered nozzle using nickel base alloys, Alloy 82 and Alloy 182. Many repairs were made during field welding to repair numerous in-process defects. The weld essentially became a double V design because welding and grinding were performed from both the inner diameter and outer diameter surfaces. The weld was pre-service inspected (radiographic, ultrasonic, liquid penetrant and visual) and was later ultrasonically inspected in 1987 and 1993. No surface connected flaws were discovered during these inspections. Significant inspection, evaluation, root cause and destructive examination of the weld crack were performed. Outer diameter dye penetrant examinations, outside and inside visual examinations, and inner diameter ultrasonic and eddy current inspections were all performed. A 0.3 m segment containing the nozzle-to-pipe weld joint was removed for both non-destructive and destructive examinations. The metallurgical examination confirmed the presence of an axial crack located 7º clockwise from the top of the pipe (as viewed from the centreline of the reactor vessel). It was determined that multiple crack initiation sites occurred on the inner diameter in the original Alloy 182 butter deposit. The crack extended approximately 65 mm along the inside surface and intersected a shorter (about 40 mm in length) circumferential crack. The axial crack was bounded on the pipe side by the heat affected zone of the stainless steel pipe and by the low alloy steel reactor vessel nozzle on the opposite end. The axial crack, therefore, was contained entirely within the Alloy 82 weld metal and the Alloy 182 nozzle butter with small ligaments that reached the pipe outer diameter surface as a single small weep hole. The circumferential crack intersecting the axial crack at the 7º location was found to be contained within the Alloy 182 cladding/buttering in the bore of the low alloy steel nozzle. Crack depth was about 5.08 mm and resulted in some minor corrosion pitting at the interface of the carbon steel.

— **Thermal Fatigue Event #1**. On 29 August 1979, Olkiluoto Unit 1 experienced a DN150 pipe fracture in the reactor water cleanup system. This failure resulted in a spill of 5000 kg primary water outside containment. The through-wall cracking in the pipe was 150 mm long and 2 mm wide. The unit was in the commissioning phase of operation, and the pipe fracture resulted in reactor trip and containment isolation. The rupture occurred in a mixing tee between the reactor water cleanup and residual heat removal systems. The apparent cause was due to thermal stratification, but the underlying cause was due a deficiency in the system operating procedure. The fatigue crack evolved as a consequence of plant operation with an intermediate basic state of reactor water cleanup valve 331-V19 leaving two flows having considerably different temperatures mix incompletely.

— **Thermal Fatigue Event #2**. An unidentified 3.8-L/min primary coolant leak was detected when Crystal River Unit 3 was in operation on 21 January 1982. A visual inspection revealed that the leak was associated with the MU/ high pressure safety injection line. As the leak was unisolable, the plant promptly proceeded to cold shutdown. Inspection of the safe-end revealed that a through-wall, circumferential crack was present in the safe-end-to-check valve weld. The circumferential extent of the crack at the outside surface was 140°. The crack consisted of two separate cracks; one initiated at the outer diameter and another one initiated at the inner diameter. The one on the outside surface was initiated and propagated by mechanical fatigue caused by pipe vibrations. The one on the inside surface was initiated and propagated by thermal fatigue caused mainly by turbulent mixing of hot reactor coolant and cold make-up water. Thermal shocks during periodic make-up water additions could have played some role in causing the fatigue damage.

— **Thermal Fatigue Event #3**. In 1987, a leak occurred inside the containment of Farley 2 during normal power operation. The leak was found in an unisolable location of a safety injection line. The crack was on the inside surface of the weld and extended approximately 120° circumferentially around the underside of the pipe. About 25 mm of this crack was through wall. The crack was caused by thermal fatigue and had developed slowly. The leak rate was 2.7 L/min. The monitoring of circumferential temperature distribution at the failed weld at Farley 2 carried out after the leak event showed spatial and temporal fluctuations in the temperature. The circumferential temperature difference at the weld varied from 3°C to as high as 120°C. Based on these measurements, it was assumed that the temporal variations resulted from intermittent action of the check valve. There were, however, no test results supporting this assumption. Experiments performed in Japan simulating the Farley event showed that the temperature fluctuation in the safety injection line was not caused by the intermittent action of the check valve (i.e. fluctuation in the flow rate) but by the mixing of low temperature leakage flow with high temperature turbulent flow in the pipe downstream of the check valves. The Japanese test results also concluded that the thermal cycling is severe enough to cause high cycle fatigue failure of the piping material when the leak flow rate is equal to or larger than 100 kg/h as measured at Farley 2.

— **Thermal Fatigue Event #4**. On 9 September 1995 at Three Mile Island Unit 1, a primary coolant leak occurred in an RCS cold leg drain line. The through-wall cracking was in the weld between the first elbow downstream of the reactor coolant loop nozzle and the horizontal pipe run. At the time of the leak, the NPP had been operating for 21 years and was at 0% power, beginning a cooldown. The leak rate was 20 drops per second. The pipe routeing is vertically down ca. 350 mm from the cold leg, then ca. 2225 mm horizontally to the first valve. The vertical run is DN40 diameter pipe, the horizontal run is DN50 and the elbow between the pipe runs is a reducing elbow. The elbow and horizontal run are Type 316 stainless steel, and the vertical run included an Inconel safe-end. The location of the crack was in the weld between the elbow and the horizontal pipe, near the top of the pipe. The crack centred at the 11 o'clock position was circumferential, 50.8 mm long on the inside and 14 mm long on the outside surface. The drain line was not insulated. The cause of the cracking was thermal fatigue, attributed to turbulence penetration of the hot RCS fluid extending into the horizontal pipe. The uninsulated horizontal pipe allowed heat to escape to the surroundings, and when the turbulence penetration reached the horizontal run, this produced thermal stratification in the line. Fluctuations in the extent of turbulence penetration caused local thermal cycling at the elbow weld, a point of stress concentration. There were other contributing causes: two improperly installed pipe support U-bolts restricted the free thermal expansion of the pipe and produced a 255 Mpa stress at the elbow; the placement of the pipe supports caused the horizontal pipe to slope upward away from the elbow, which facilitated thermal stratification and cycling in this pipe run; and the toe of the cracked weld had a pre-existing notch.

— **Thermal Fatigue Event #5**. On 12 May 1998, with Civaux Unit 1 in intermediate shutdown mode, the pressurizer level decreased and a fire alarm occurred in the Reactor Building. A large (ca. 8.3 kg/s) primary leak had developed, and it was compensated by the charging system. After rapid cooldown of the reactor, the leak was identified to come from residual heat removal train 'A'. The source of the leakage was a cracked longitudinal weld on an elbow downstream of a control valve. The crack length was approximately 180 mm. The observed cracking was located immediately downstream of the mixing tee in a DN250 304L stainless steel seam-welded elbow (Fig. IV–24).

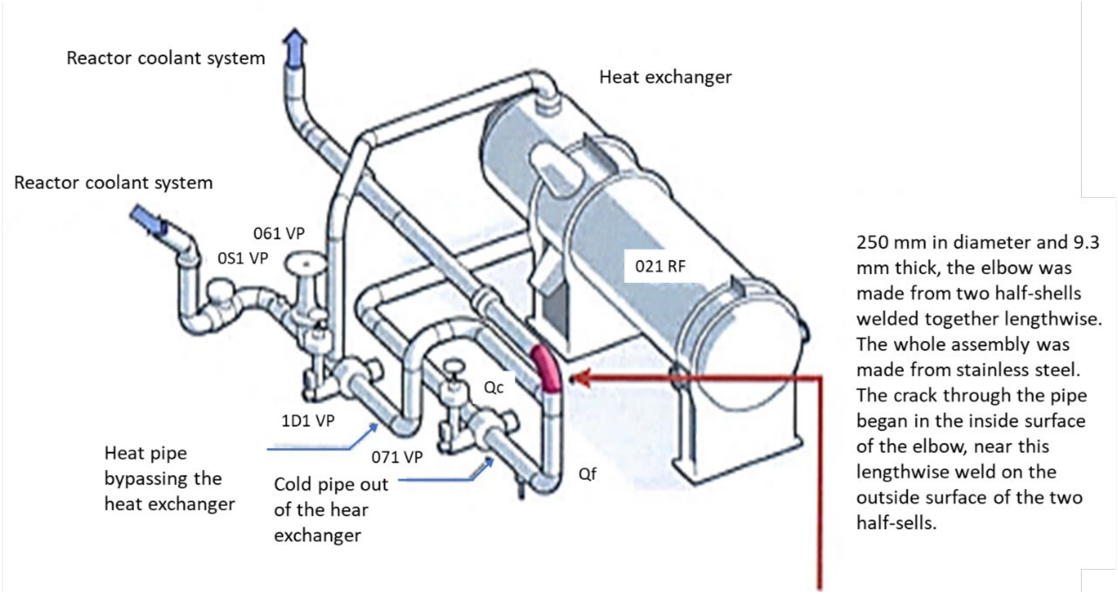

FIG. IV–24. Civaux-1 thermal fatigue failure (from [IV–6].

The surrounding area of the elbow and locations between the elbow and the mixing tee showed widespread crazing type cracking characteristic of thermal fatigue. Examination of this elbow and similar ones from other plants confirmed that high cycle thermal fatigue was the mechanism. In general, the deep cracks developed along the longitudinal and circumferential welds, starting at the root of the weld on the inner surface. There were many small cracks, mainly in the weld counter-bore regions and where there was evidence of grinding.

REFERENCES

[IV–1] LEISHEAR, R.A., "Hydrogen Ignition Mechanisms for Explosions in Nuclear Facility Pipe Systems", (Proc. ASME 2010 Pressure Vessels & Piping Division Conf.), PVP2010-25261, ASME, New York (2010).

[IV–2] CHOPRA, O.K., Effects of Thermal Aging and Neutron Irradiation on Crack Growth Rate and Fracture Toughness of Cast Stainless Steels and Austenitic Stainless Steel Welds, NUREG/CR-7185, U.S. Nuclear Regulatory Commission, Washington, DC (2015).

[IV–3] WÜTHRICH, C., Crack opening areas in pressure vessels and pipes, Eng. Fract. Mech. **5** (1983) 1049–1057.

[IV–4] SCHULZ, H., Comments on the probability of leakage in piping systems as used in PRAs, Nucl. Eng. Des. **110** (1988) 229–232.

[IV–5] BUSH, S.H., "Statistics of pressure vessel and piping failures", Pressure Vessel and Piping Technology: A Decade of Progress, ASME, New York (1985) Ch. 8.9.

[IV–6] OECD NUCLEAR ENERGY AGENCY, A Review of the Post-1998 Experience with Thermal Fatigue in Heavy Water and Light Water Reactor Piping Components, NEA/CSNI/R(2019)13, Boulogne-Billancourt, France (2022).

Annex V

ABSTRACTS OF EARLY PIPE RELIABILITY STUDIES (1970–1990)

V–1 INTRODUCTION

Since the 1960s the evolution of nuclear safety principles and risk and reliability analysis methodologies has been influenced by the consideration of various types of rare events, from common cause failures of redundant active or passive components to catastrophic failures of highly reliable piping components. The question of how to quantitatively demonstrate the structural integrity of pressurized components received significant attention during the early years in the evolution of NPP safety practices (i.e. 1960–1975 time frame). Numerous studies were completed to address the feasibility of applying fracture mechanics as well as classical statistical methods to the estimation of pipe failure frequency and pipe failure probability, conditional on some type of pre-existing condition.

The purpose of this Annex is to summarize selected early published works on piping reliability analysis that were based on direct estimation on the basis of limited OPEX data or through expert judgement. These direct estimation studies are not to be confused with the DDM technique as it is described in Annex I. The abstracts are provided for reference only and to assist in placing in perspective the significant progress that has been made in the collection and analysis of piping OPEX data, methods development in general, development and application of material degradation mitigation processes and non-destructive examination technologies. The inclusion of the selected abstracts does not constitute an endorsement of direct estimation type assessments of piping reliability.

V–2 PIPING RELIABILITY CONSIDERATIONS IN THE US REACTOR SAFETY STUDY

A limited evaluation of nuclear pipe reliability was performed as part of the Reactor Safety Study (WASH-1400; [V–1]–[V–4]). The evaluation was based on actual failures in nuclear systems related to the early operating period of NPPs. The objective was to derive order-of-magnitude LOCA frequencies for input to event tree analysis (Table V–1) and pipe failure rates for input to system fault trees. WASH-1400 examined several different sources to obtain failure rates for small diameter and large diameter pipe. The reason for using several data sources was the interest in pipe ruptures (complete pipe severances) resulting in reactor coolant loss, and none had occurred in the 150 US commercial nuclear ROYs considered by the study. Therefore, other pipe failure data sources were sought for extrapolating pipe failure rates for use in the WASH-1400.

TABLE V–1. WASH-1400 LOCA FREQUENCIES

LOCA category	LOCA initiating event frequency [1/ROY]	
	Median	Range factor
Small	1.0×10^{-3}	10
Medium	3.0×10^{-4}	10
Large	1.0×10^{-4}	10

A criticism of the approach of WASH-1400 was that the database on significant pipe failures only included eleven (11) significant events. Therefore, the statistical uncertainties of the failure rate estimates and LOCA frequencies were considerable. As a further criticism, several inconsistencies existed in the failure rate estimation as well as in the interpretation of estimates within WASH-1400. In Appendix III of WASH-1400 the failure rates were calculated so as to provide estimates having the dimension of [1/h.ft], while in the systems analyses the same failure rates were assumed having the dimension of [1/h.section]. Although inconsistently applied, in WASH-1400, a pipe section was assumed to correspond to about 12 linear feet (3.6 m) of piping.

For small diameter piping (≤75 mm), the failure rate λ was derived from the following:

- In 1972 the accumulated US WCR OPEX was approximately 150 (ROYs) and 11 significant pipe failures were recorded in 1960–1972. They all occurred in small diameter piping. Based on this information a point estimate was calculated as:

$$\lambda = 11/(150 \times 8760) = 8.37 \times 10^{-6}/\text{h} \qquad \text{(V–1)}$$

In Appendix XI of WASH-1400 (Comments on the Draft Report, page 14-3) information is given on the amount of LOCA-sensitive piping in a typical commercial, US NPP: "...5% or 8,500 feet of piping is large LOCA-sensitive..." This information would imply the total amount of piping to be on the order of 170 000 ft (51,816 m). On page III-75 of Appendix III it is stated that 4.7% of total plant piping is small LOCA-sensitive (i.e. about 7990 ft (2,435 m)). Therefore, the failure rate of small diameter piping would be on the order of:

$$\lambda = 8.37 \times 10^{-6}/7990 = 1.0 \times 10^{-9}/\text{h.ft} = 2.9 \times 10^{-5}/\text{ROY.m.} \qquad \text{(V–2)}$$

In the fault tree models this failure rate was interpreted as being valid for each section of piping. While there were inconsistencies within WASH-1400, there have also been inconsistencies in how later piping reliability studies interpreted WASH-1400.

V–3 PIPE FAILURE RATES ACCORDING TO PNNL (1975)

After the publication of WASH-1400 in 1975, the Pacific Northwest National Laboratories (PNNL) performed an independent assessment of piping reliability based on US light water reactor OPEX and non-nuclear OPEX [V–4]. Differences between WASH-1400 and PNNL results are attributed to how the limited failure data was interpreted. The study by PNNL addressed the role of periodic inspection, and addressed failures due to intergranular SCC. Among the conclusions were:

- Failure probabilities for larger sizes of nuclear piping were considered to be in the range of 1×10^{-4} to 1×10^{-6} per reactor-year (exclusive of intergranular SCC). Note that the PNNL study addressed initiating event frequencies [1/ROY].
- Smaller pipe sizes, of lesser safety significance, have much higher failure rates.
- In BWRs, intergranular SCC can cause failure rates much higher than 1×10^{-4} in piping of size DN 100 to DN 250.
- Catastrophic failures would appear more likely from operator error or design and construction errors (water hammer, improper handling of dynamic loads, undetected fabrication defects) rather than conventional flaw initiation and growth by fatigue.

V–4 PIPE FAILURE RATES ACCORDING TO ATOMIC ENERGY OF CANADA LIMITED (1981)

The Atomic Energy of Canada Limited (AECL) performed a study on the US WCR piping OPEX for the period 1959 through 1978, representing 409 ROYs of experience [V–5]. The study was initiated in support of an analysis of the consequences of pipe rupture in the Primary Heat Transport System for CANDU power stations. Another objective was to establish whether the additional OPEX that had accumulated since publication of WASH-1400 warranted new pipe failure rates to be used in PSA applications. The pipe failure events were classified according to: (1) severance, (2) leak, and (3) defect. Of the total 840 failure events considered by the study, 87 pipe failures were interpreted to be severances (8 events in small diameter primary system piping). Statistical analysis was limited to the estimation of confidence limits for failure rates using the chi-square distribution. Because of uncertainties in the pipe failure event database and assumptions in interpretation of the data, the order of magnitude failure rate estimates by WASH-1400 were viewed by the AECL as representative of true failure rates. Table V–2 summarizes failure rate estimates for primary system pipe breaks. In retrospect, a problem with these estimates is the lack of specificity with respect to how they relate to safety class, system, operating environment, etc.

TABLE V–2. PIPE FAILURE RATES ACCORDING TO AECL (1981)

Pipe size, nominal diameter (DN) [mm]	Failure rate [1/ROY] upper boundary at 95% confidence
≤25	3.9×10^{-2}
25 < DN < 150	7.3×10^{-3}
DN ≥ 150	7.3×10^{-3}

V–5 THOMAS MODEL OF PIPING RELIABILITY (1981)

In 1981, H.M. Thomas of Rolls Royce & Associates Ltd published an extended technical paper on a generalized approach to an interpretation of pipe failure data with recommendations on how to adjust or convert generic industry data to plant specific data [V–6] and [V–7]. Among the reliability influence factors acknowledged in updating generic data were: design learning curve, pipe diameter, plant age, fracture toughness, pipe length, number of load cycles, base material versus weld material, fatigue stress, crack dimensions and wall thickness. On the subject of pipe length Thomas stated that: "... It is known that a typical [nuclear power] plant contains about 16,500 feet of pipe less than 4 inch diameter and about 18,500 feet of pipe greater than 4 inch diameter, making a total of 35,000 feet ..." [V–6].

Thomas referenced WASH-1400, Appendix III. However, there is a discrepancy between WASH-1400 and the Thomas paper. One can speculate how the information on pipe length was derived. Some insights can be gleaned by assuming that Thomas arrived at a number of 350 000 ft (96 012 m) being the total length of piping in a typical NPP. By multiplying this length by 4.7% and 5.3%, respectively, one would (and consistent with WASH-1400) get the total length of small diameter, LOCA sensitive piping and large diameter, LOCA sensitive piping, respectively (i.e. together about 35 000 ft (96 012 m) of pipe). It is presumed that Thomas was influenced by the paper of Spencer Bush published in 1975 in which a typical BWR was stated as having 315 000 linear feet of LOCA-insensitive piping. Under the set of assumptions there would be consistency between Thomas and Bush (i.e. 315 000 + 35 000 = 350 000 ft (96 012 m). Presumably, the term 'LOCA sensitive' meant safety class 1 piping and if so, the given estimate is incorrect (cf. Section VI.7). At the time (early 1980s), the Thomas correlation was seen by many as the only practical approach to estimate pipe leak and rupture frequency. From time to time it has been proposed that the single most important advancement in piping reliability would be to simply validate the original correlation using today's failure statistics.

IV–6 PIPE FAILURE RATES ACCORDING TO RISØ NATIONAL LABORATORY (1982)

Within the framework of the SÄK-1 (Probabilistic Risk Assessment and Licensing) project sponsored by the Nordic Liaison Committee for Atomic Energy, Risø performed a pipe failure study [V–8] and [V–9]. The derived failure rates were based on Swedish and Finnish nuclear plant OPEX in 1975–1981, corresponding to 43 ROYs. A total of 62 pipe failures were recorded in Swedish plants for the study period, of which two events represented crack or rupture. A summary of the pipe failure rates is given in Table V–3. These pipe failure frequencies were based on reports of degraded conditions that required some form of pipe repair to maintain structural integrity. It would require considerable reinterpretations of the analysis in order to make these results useful.

TABLE V–3. PIPE FAILURE RATES ACCORDING TO RISØ (1982)

Pipe rupture category	Pipe failure frequency [1/ROY] (90% range)	
	Water pipe	Steam pipe
Small	5.8×10^{-1}–1.0	6.1×10^{-2}–2.0×10^{-1}
Medium	7.6×10^{-2}–2.8×10^{-1}	8.0×10^{-2}–1.1×10^{-1}
Large	8.2×10^{-3}–1.1×10^{-1}	$\leq 5.4 \times 10^{-2}$

V–7 PIPE FAILURE RATES ACCORDING TO S.H. BUSH (1985)

In a series of technical papers, S.H. Bush[6] summarized insights from analyses of WCR piping OPEX obtained during a 20 year period (1965–1985). As one example, in [V–10] failure rates are presented on the basis of events that have occurred in locations other than welds. These failure rates are in terms of number of events per ROY and metre of pipe [1/ROY.m]; Table V–4. It is not straightforward to interpret these failure rates, however. The failure counts appear to be related to catastrophic events (i.e. pipe failures attributed to water hammer or major design and construction errors).

TABLE V–4. S.H. BUSH PIPE FAILURE RATES

Data	BWR	PWR	Combined
Number of plants × number of components (= length of piping)	315 000 m	280 000 m	595 000 m
Number of failures	4	4	8
Failure rate (>DN100)	4.3×10^{-5}	4.6×10^{-5}	4.3×10^{-5}

V–8 PIPE FAILURE RATES ACCORDING TO IDAHO NATIONAL LABORATORY (1987)

Published in 1987, the objective of the work performed by the Idaho National Laboratory (INL) was to update the failure rate estimates of WASH-1400 by utilizing the accumulated US nuclear OPEX available as of December 1984 [V–11]. About 800 ROYs were considered. The proposed pipe rupture frequencies are summarized in Table V–5. Relative to WASH-1400 an additional 650 reactor-years were accounted for to improve the uncertainties of the pipe failure rates. Whereas WASH-1400

[6] S.H. Bush was an international authority (and a 'walking database') on metallurgy and piping reliability, and a driving force in the development of nuclear power plant codes and standards (in particular ASME Section XI). He passed away in 2005. For more details, see Recent Advances in Non-destructive Examination and ASME B&PVC Section XI — a Memorial Symposium in Honour of Spencer H. Bush, 22–26 July 2007.

accounted for eleven (11) significant pipe failures, the Idaho National Laboratory Study identified twenty (20) significant pipe failure events.

TABLE V–5. PIPE RUPTURE FREQUENCIES ACCORDING TO IDAHO NATIONAL LABORATORY (1987)

EBS [mm]	**Pipe rupture frequency [1/ROY]** 5%-tile	Median	95%-tile
		BWR	
12 – 50	2.6×10^{-3}	9.6×10^{-3}	2.5×10^{-2}
50 – 150	1.1×10^{-3}	6.4×10^{-3}	2.0×10^{-2}
>150	6.4×10^{-3}	1.6×10^{-2}	3.3×10^{-2}
		PWR	
12 – 50	7.0×10^{-4}	4.1×10^{-3}	1.3×10^{-2}
50 – 150	2.8×10^{-3}	8.3×10^{-3}	1.9×10^{-2}
>150	1.7×10^{-3}	6.2×10^{-3}	1.6×10^{-2}

V–9 PIPE FAILURE RATES ACCORDING TO GRS (1989)

In support of Phase B of the German Risk Study (1981–1989), GRS performed piping reliability studies in recognition of the significant limitations of the then available pipe reliability estimation techniques. GRS proposed two general analysis approaches: (1) statistical evaluation of relevant OPEX data, and (2) PFM studies. The former approach has to be applied to small diameter piping for which failure experience existed, while the latter approach supported analysis of piping for which some experimental data existed. Table V–6 summarizes the German approach [V–12]–[V–14][7].

TABLE V–6. PIPING RELIABILITY MODEL ACCORDING TO GERMAN PSA PRACTICE

Pipe diameter [DN]	**Method**	**Comment**
≤50	λ_L can be estimated from OPEX	λ_L = frequency of leak [1/CB.ROY]. CB is component boundary (e.g. weld)
	CFP = λ_B / λ_L = 2.5/DN if analyst does not have an access to relevant OPEX data	λ_B = pipe break frequency. D = pipe diameter in mm. The origin of this method is documented in [V–13]
e50 < DN ≤ 150	$\lambda_L = C_x(L_D D)/t_D^x$	D = pipe diameter in mm. The other parameters are obtained using the Thomas model
150 < DN ≤ 250	$\lambda_L = C_x(L_D D)/t_D^x$ $C_x = \frac{\sum_{50}^{150} N_{L,D}}{\sum_{50}^{150} L \times D/t^x \times T}$	D = 150, the other parameters are obtained using the Thomas model [V–7] and [V–8] L_D = Number of susceptible locations t_D = Pipe wall thickness [mm] x = a dimensionless parameter N = number of leaks T = ROYs
>250	Assumption: λ_L< 1e-7/ROY	For large diameter piping, the German practice recommends using a PFM methodology to estimate the CFP

V–10 PIPE FAILURE RATES ACCORDING TO EPRI (1992)

Co-sponsored by Northeast Utilities Service Company (now Dominion Generation) and the EPRI, Jamali developed a methodology and database for pipe failure rate estimates [V–15]. This study

[7] Guidelines for how to implement this method are found in GRS Fachseminar: Ermittlung der Häufigkeiten von Lecks und Brüchen in druckführenden Systemen für probabilistiche Sicherheitsanalysen, Cologne, Gemany, 18–20 September 1995.

was done in order to create a US nuclear plant pipe failure database reflecting the additional experience generated since the publication of WASH-1400. The principal sources of pipe failure information were Licensee Event Reports, Nuclear Power Experience published by S.M. Stoller Corporation (the Blue Books and the Red Books) and the Nuclear Plant Reliability Data System operated by the Institute of Nuclear Power Operations. An excerpt from this work is summarized in Table V–7.

TABLE V–7. PIPE FAILURE RATES ACCORDING TO EPRI (1992)

Inside pipe diameter (ID) [mm]	Pipe failure rate [1/ROY.pipe.section] EPRI	WASH-1400
$12 \leq ID < 50$	5.3×10^{-6}	3.2×10^{-5}
$50 \leq ID < 75$	$2.6 \times 10^{--6}$	3.2×10^{-5}
$75 \leq ID < 150$	2.6×10^{-6}	3.2×10^{-6}
$ID \geq 150$	6.1×10^{-6}	3.2×10^{-6}

For estimation of pipe failure rates from OPEX, a new EPRI methodology was developed. A parameter referred to as failure severity code was introduced as a key element of the methodology. This parameter accounted for the fact that the effective break area can be significantly smaller than the area calculated using the pipe inner diameter. It was used to estimate the conditional probability of having a given effective break size for a given pipe size. The EPRI methodology also accounted for factors that can be quantified from the database and that may significantly affect the values of the failure rates. These included the nuclear steam system supplier, system type, pipe size and plant age. Variance analysis techniques were used to estimate the effect of system types on leakage failure rates. Key aspects of the EPRI methodology are summarized below:

— Piping component boundary definition. A pipe section is a segment of piping, between major discontinuities, such as valves, pumps, reducers, tees, etc. as defined by WASH-1400. While the EPRI reports do not give specific guidance on how to apply this definition, the report includes typical pipe section counts for BWRs and PWRs [V–16].
— Pipe failure attributes. These are factors believed to significantly impact pipe the failure rate. The EPRI methodology accounts for failure mode, pipe size, system type and time (i.e. age of piping). Four size categories were used: $12 < ID \leq 50$, $50 < ID < 150$, $ID \geq 150$ and unknown size; where ID is an inner pipe diameter in mm.
— Pipe failure mode definitions. Three failure modes were considered: (1) cracking, failures with no seepage of process fluid to the outside of the pressure boundary, (2) leakage, loss of fluid in amounts of less than 3 kg/s, and (3) major piping breakage.

REFERENCES

[V–1] BARTEL, R., WASH-1400, the Reactor Safety Study. The Introduction of Risk Assessment to the Regulation of Nuclear Reactors, NUREG/KM-0010, U.S. Nuclear Regulatory Commission, Washington, DC (2016).

[V–2] U.S. NUCLEAR REGULATORY COMMISSION, Failure Data. Appendix III to the Reactor Safety Study, NUREG-75/014, Washington, DC (1975).

[V–3] KELLER, W., MODARRES, M., A historical overview of probabilistic risk assessment development and its use in the nuclear power industry: A tribute to the late Professor Norman Carl Rasmussen, Reliab. Eng. Saf. Anal. **89** (2005) 271–285.

[V–4] BUSH, S.H., Reliability of piping in light water reactors, Nucl. Saf. **17** (1976) 568–579.

[V–5] JANZEN, P., A Study of Piping Failures in U.S. Nuclear Power Reactors, AECL-Misc-204, Atomic Energy of Canada Limited, Special Projects Division, Chalk River Nuclear Laboratories, Chalk River, Canada (1981).

[V–6] THOMAS, H.M., 1981. Pipe and vessel failure probability, Reliab. Eng. **2** (1981) 83–124.

[V.7] LYDELL, B.O.Y., Pipe failure probability: The 'Thomas Paper' revisited, Reliab. Eng. Syst. Saf. **68** (2000) 207–217.

[V–8] PETERSEN, K.E., “Pipe Failure Study”, Probabilistic Risk Analysis and Licensing, NKA/SÄK-1-D(82)9 (Risø-M-2363) (Proc. of Seminar 2, Helsingør, Denmark, March 29–31, 1982) 129–149.

[V–9] PETERSEN, K.E., “Analysis of Pipe Failures in Swedish Nuclear Plants” (Proc. of the 4th EuReDatA Conf., Venice (Italy), March 23–25, 1983).

[V–10] BUSH, S.H., “Statistics of pressure vessel and piping failures”, in SUNDARARAJAN, C. (Ed.) Pressure Vessel and Piping Technology — A Decade of Progress, ASME, New York, (1985) 875–893.

[V–11] WRIGHT, R.E., STEVERSON, J.A., ZUROFF, W.F., Pipe Break Frequency Estimation for Nuclear Power Plants, EGG-2421 (NUREG/CR-4407), Idaho National Engineering Laboratory, Inc., Idaho Falls, ID (1987).

[V–12] BELICZEY, S., SCHULZ, H., The probability of leakage in piping systems of pressurized water reactors on the basis of fracture mechanics and operating experience, Nucl. Eng. Des. **102** (1987) 431–438.

[V–13] BERG, H-P., GERSINSKA, R., SIEVERS, J., Improved approach for estimating leak and break frequencies of piping systems in probabilistic safety assessment, Reliab. Risk Anal.: Theory Appl. **4** (2009) 69–81.

[V–14] BUNDESAMT FÜR STRAHLENSCHUTZ, PSA Data Resource Document, BfS-SCHR-38/05 (in German), Salzgitter, Germany (2005).

[V–15] JAMALI, K., Pipe Failures in U.S. Commercial Nuclear Power Plants, TR-100380, EPRI, Palo Alto, CA (1992).