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Benchmark of Risk-Informed In-Service Inspection Approaches when Applied to Non-Piping Pressure Boundary Components

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Executive Summary

This report is a product of the members of NUGENIA Technical Area 8 (TA8) – the European Network for Inspection and Qualification (ENIQ), and specifically the Sub-Area Inspection Effectiveness (SAE) group. The report summarises the results of a case study to review the applicability of existing Risk-Informed In-Service Inspection (RI-ISI) methodologies aimed at pipework to general pressure boundary components (e.g. valves, pressure vessels, and tanks). The objective of the work is to test the applicability of approaches considered when applied to a wider scope of components. The case study has been completed by taking several real components from a donor plant and completing a paper-based application of the RI-ISI methods up to the point of risk category assignment.

The case study results show that while non-piping components are more challenging than piping welds, existing methods can be extended to pressure boundary components, although such components are not strictly within scope.

Most of the methods considered already apply a semi-quantitative approach to the scoring of Probability of Failure (PoF) that does not rely on quantitative predictions (e.g. via probabilistic fracture mechanisms) and these methods can be extended most easily. Methods based on quantitative predictions can also be extended but the availability of appropriate models that can cater for a wide range of component types is lacking.

Existing Probabilistic Risk Assessment (PRA) data can generally be used where needed to assign Consequence of Failure (CoF) metrics for non-piping components. However, the PRA may be limited to the consequences of pressure boundary failure and resulting coolant leakage only. Where relevant, care must be taken that secondary, or in-direct, consequences of failure are captured and accounted for (e.g. missile and jet generation, flooding etc). The use of PRA data for CoF assignment is most applicable to components similar to piping (e.g. valves) and a more detailed review of Failure Mode, Effects, and Criticality Analysis (FMECA) is likely to be required for more complex components.

The review shows that there is a trend of simplifying methods, particularly simplification of PoF assignment, to reduce the resource requirements of applying RI approaches and balancing with conservatism. In general, this approach has been preferred by industry to allow more widespread application of RI-ISI and this approach is equally applicable to the scope of components considered. In some cases, it may be beneficial to invest more in the quantification of failure probabilities (e.g. for components close to a threshold). To aid identifications of these cases, a decision tree has been recommended for use to identify where more detailed consideration of failure modes and PoF analyses are warranted.

The review also highlights that several of the components considered are classified as high safety significance and would automatically be selected for ISI. Hence there is limited benefit from applying the methods considered relative to deterministic approaches. However, it is highlighted that RI principles can be more widely applied to these components by focusing on leveraging understanding of PoF, relevant degradation mechanisms, and fleet service experience to target ISI techniques and tailor inspection intervals. This approach is likely to be more acceptable to safety regulators and there are examples of this approach being applied in some design codes.

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1. Introduction

The current objective of the European Network for Inspection and Qualification (ENIQ) Sub-Area for Inspection Effectiveness (SAE) is the development of Risk-Informed (RI) methods for the application of non-destructive inspections of nuclear plants at various stages through the product life-cycle (e.g. In-Service Inspection (ISI), Pre-service Inspection (PSI)), and the production of guidance material and discussion documents to facilitate the application of such methods. Existing RI methods are widely applied for nuclear pipework systems, but the methods are less well established for non-pipework pressure boundary components (henceforth referred to as 'mechanical components').

In 2021, ENIQ SAE published ENIQ report no. 67 [1], titled 'Extending Risk-Informed In-Service to General Mechanical Components – Benefits and Challenges'. The report discusses the extension of RI-ISI methods for piping welds to cover other general mechanical components and identifies the benefits and challenges involved.

In 2023, SAE published ENIQ report no. 69 [2], which contains a review of SAE publications to date and provides recommended areas of work beneficial to extend RI-ISI methods to mechanical components, including those of highest safety significance.

To achieve the aims discussed and gain a better understanding in the gaps in knowledge and current methodologies, SAE elected to conduct a table-top case study on a selection of real components and associated data (e.g. Probabilistic Risk Assessment (PRA[1]) results) using a range of currently available RI-ISI methodologies, or codes/standards. Due to limited resources, the case study was limited to a comparison of the risk-ranking results from the methods compared.

Note - in 2006, ENIQ together with Office for Economic Cooperation and Development Nuclear Energy Agency (OECD NEA) launched a benchmark project RISMET [3] to compare RI-ISI approaches. This case study benefits from experience gained from the RISMET project, although the scope and extent of the current study are more limited.

2. Case Study: Objectives, Scope, and Approach

The objective of the case study is to test the applicability of RI-ISI methodologies to mechanical components and to gain a better understanding of the challenges identified in ENIQ report 67 and 69.

These challenges are summarised Table 1 and Table 2 with a specific focus on addressing the following topics:

- If different approaches lead to different risk rankings and identify the reasons.
- If there is currently sufficient data on mechanical components to apply risk-informed approaches based on quantitative methods.
- The challenges and benefits of applying semi-quantitative, or judgement-based, approaches to overcome the difficulties of gathering the data needed to fully apply quantitative methods. The effect of the application of risk-informed approaches on inspection scope and frequency of mechanical components relative to deterministic methods for ISI site selection.

[1] The terms PRA and Probabilistic Safety Assessment (PSA) are considered to have equivalent meanings for the purpose of this report.

Topic	Challenges
Collection of necessary data	Availability of documentation, especially for low safety class components.
Definition of scope and screening	Definition of component-adapted screening metrics and criteria.
Assessment of consequences	Evaluation of the consequence of general mechanical components.[2] Assessing other than reactor safety consequences.
Assessment of failure probabilities	Lack of reliability data or structural reliability model (SRM) tools to estimate the failure probabilities.
Risk characterisation and ranking	Choice of risk criteria, limits for failure probability and consequence categories, and treatment of risk outliers.
Proposal for new ISI programme	Requirements for new qualified inspection methods on new components and materials, definition of sample size for components of small population.
Effectiveness assessment	Uncertainties and lack of data; for many of the systems/components addressed, a formal in-service inspection programme does not exist.

Table 1 - Challenges in extending RI-ISI to non-piping components [1].

Topic	Challenges
Balancing ISI Programmes	Consideration of how to ensure a balanced RI-ISI programme can be achieved and the use of a risk cut-off.
High CoF SSCs	Development of a consensus position on assigning ISI based upon risk for high Consequence of Failure (CoF) SSC.
Semi-Quantitative RI-ISI Methods	Review of RI schemes primarily based upon operating experience, failure statistics and expert judgement.

Table 2 - Challenges in extending RI-ISI to mechanical components [2].

The scope of the case study is limited to management of risk associated with nuclear Systems, Structures and Components (SSC). The management of other risks is outside of the scope of the study (e.g. the risk to plant personnel from conventional hazards with no associated nuclear consequence).

The following approach was adopted for the case study:

- Select several exemplar components from an existing host Nuclear Power Plant (NPP) with results from a well-developed PRA. The components were selected to include a diverse range of component types, Degradation Mechanisms (DM), and consequences of failure.
- Apply available RI-ISI methodologies to the extent possible. Where necessary input information for each method is not available, make best use of available data and approximations to allow application.

[2] Adapted from the original aim of consideration of only tanks and heat exchangers.

- Compare the results of the risk ranking exercise between the methods and analyse the key differences and underlying reasons.
- Compare the results to deterministic methods for ISI sites selection to provide a benchmark to commonly used and established practice. Results from Swedish code AFS 2023:11 'Swedish Work Environment Authority: Use and control of pressurised equipment' [4] and ASME XI Division 1 [5] and are presented for this purpose. These codes were selected as they are both established, ASME XI is widely applied, and AFS 2023:11 has also been applied to the host plant.

Identify any developments or input data required to allow application of the methods to mechanical components.

3. Case Study Components

The components selected for the case study are a selection of pressure vessels, tanks, and valves at a host NPP. The host plant is a Westinghouse Nuclear Steam Supply System (NSSS) designed three-loop 1130 MWe Pressurised Water Reactor (PWR), in operation since 1983.

Two tanks, two valves and the Pressuriser shell were selected for inclusion in the case study. These components are briefly described in the following subsections[3]. ***Table 3*** summarises the Damage Mechanism (DM) and consequences of failure identified for each of the cases study components based on the plant PRA results, based on. The consequence of failure column represents the results for a postulated large failure or rupture. Values of Core Damage Frequency (CDF) are also listed based on PRA results to provide an indication of relative safety significance.

[3] The host plant has provided all the necessary information to perform the analyses, but the data is not published in detail in this report.

Component	Component Information and Functions	Potential Degradation Mechanisms[4]	Consequence of Failure[5]	Relative Core Damage Frequency[6]
Boron Injection Tank	Provide the High-Pressure Safety Injection (HPSI) system with a supply of boronated water and provide a supply of negative reactivity at the beginning of a safety injection sequence.	None identified	Loss or degradation of HPSI system negative reactivity supply.	8.7e-3
Refuelling Water Storage Tank (RWST)	Storage of borated water and the provision of borated water during fuel handling operations. Supply tank for high- and low-pressure safety injection systems and the containment spray system.	None identified	Failure does not cause an initiating event, but a defence-in-depth barrier is lost if the function of the RWST is unavailable. Failures results in manual plant shutdown.	8.6e-3
Condensate System Valve	The system function is to increase the temperature of condensate water and transfer from the condenser to the main feed water system. The valve function is to control the level within the drainage tank via flow to the condenser.	Thermal Transient (TT), water hammer (WH) and Erosion-corrosion within connecting pipework segments	Failure results in plant/unit trip only.	1.6e-5
Feedwater System Valve	The system function is supply of heated water to the steam generators. The valve function is to isolate the system after the main feed water pump.	Erosion-corrosion	Failure results in a plant/unit trip only.	1.6e-5
Pressuriser	Maintenance of the Reactor Coolant System (RCS) pressure.	None for the vessel. Thermal Stratification, Cycling, and Striping (TASCS) for the connecting piping segment.	Large Loss of Coolant Accident (LOCA).	1

Table 3 - Degradation mechanisms and failure consequences

[4] Scope includes the component and connection pipework segments.
[5] Limited to postulated large LOCA. Potential for and consequences of internal hazards generated during failure (e.g. missile generation) is considered during plant PRA.
[6] PRA generated CDF values are presented relative to the bounding value from the Pressuriser. In each case values are based on postulated failure of connected pipework to each component.

4. Application of Risk-informed Methods

This section summarises the risk-informed methods applied to the case study components and the results for each method.

Several RI-ISI methodologies exist for the optimisation of ISI. Most of the methods specifically target ISI applied to piping and some of the methods also include other component types. However, the scope beyond piping is limited, and high safety significance components are typically excluded in favour of deterministic approaches (e.g. ASME XI Division I). In general, experience from application to mechanical components is limited.

The following risk-informed approaches were applied during the case study:

- Swedish Radiation Safety Authority's (SSM) regulations concerning mechanical devices in nuclear facilities (SSMFS) 2008:13 [6].
- EPRI traditional RI-ISI [7].
- CC N-716 (Streamlined EPRI RI-ISI methodology) [8].
- PWROG (Westinghouse RI-ISI) [9].
- 10CFR50.69 / RI-RRA (extension of EPRI RI-ISI to non-piping components) [10].
- CSA N285.7 (Canadian RI-ISI including non-piping comp) [11].
- ASME Code, Section XI, Division 2 Reliability and Integrity Management (RIM) [12].

The risk-informed approaches are applied up to the stage of identifying the risk-importance of the selected components. As most of the risk-informed approaches use a risk-ranking in three major categories (low, medium, high) these indicative risk categories are adopted in this study based around the EPRI traditional RI-ISI [7] definitions.

For defining the final inspection programme, the risk information is complemented with further technical evaluation for selecting the most appropriate inspection procedure. This phase is out of the scope of this case study.

The host NPP has implemented a full RI-ISI approach for all piping systems based on the EPRI traditional RI-ISI method [7]. Although mechanical components were not included, the results can largely support the analysis of the case study components.

For each of the components, analysis of failure consequences is limited to the consequences of connected piping. This is likely to be an accurate approximation for components such as valves but is less likely to capture all effects from the failure of components such as pressure vessels and tanks. Indirect consequences of failure (e.g. missile generation and impact with nearby components) are accounted for within the PRA to a certain extent by assuming concurrent failure in affected components where the initiating event is judged to be likely to result in the failure of nearby components. However, the CoF values from the Pressuriser and tanks are already around two orders of magnitude greater than the other components considered and so the approximation is unlikely to make a meaningful difference to the results.

For the connecting piping, the Probability of Failure (PoF) and CoF ranking is available for Westinghouse, EPRI and SSM methodologies.

4.1. Risk-informed approaches

In the following sub-chapters, the main principles of each methodology are briefly summarised and details of any adaptations necessary for application to the case study components is provided. The ranking of the case study component is then presented for each method.

Additionally, a description of the RIM approach according to ASME Section XI Division 2 [12] is provided. The results from applying this method to the case study components is not included as the required input data was not available.

4.1.1. SSMFS

Summary of the methodology

SSMFS 2008:13 [6] is a specific regulation of the SSM covering mechanical components. It contains, among other issues, the requirements for ISI and control, as well as requirements concerning repair, replacement and modification of structures and components.

According to the regulation, mechanical components shall be divided into Inspection Groups A to C to determine the scope of periodic inspection. The Inspection Group shall be determined by considering the relative risks as follows, based on judgement:

- A – Highest risk.
- B – The relative risks are judged to be lower than for group A but not minor.
- C – Minor risk.

In earlier regulations (SKIFS 1994:1 [13]), the assignment of the Inspection Group was determined in more detail, as a combination of a Consequence Index (CI) determined based on proximity (of piping) to the RPV and isolation valves, and a Damage Index (DI) determined based on Degradation Mechanism (DM) potential and mechanical fatigue (a more detailed insight into the SKIFS methodology can be found in [14]). As the present regulation (SSMFS 2008:13) is less detailed than before, the Swedish nuclear utilities have developed a common guidance document that interprets these regulations, and effectively, serves as an industry standard, PMT2004. This document fulfils the regulatory requirements and provides more details for assigning risk categories for components. The approach is simple, easy to use and conservative. A summary of the method is provided in Appendix 1.

Figure 1 shows the determination of the Inspection Group based on the CI and DI evaluations. Components with no CI are assigned to Inspection Group C.

	Consequence index		
Damage index	1	2	3
I	A	A	B
II	A	B	C
III	B	C	C
Inspection Group A = High Risk Inspection Group B = Medium Risk Inspection Group C = Low Risk			

Figure 1 - Risk matrix for determination of the Inspection Group according to SKIFS 1994:1.

Application to non-piping pressure boundary components

As the Consequence Index is simply determined based on the proximity of the RPV and isolation valves, the same consequences are used for mechanical components and piping.

Results

The application of this methodology to the case study components leads to the results in Table 4.

Component	Consequence Index (CI)	DM Identified	Degradation Index (DI)	Inspection Group
Boron Injection Tank	NA (note 1)	NA	NA	C
Refuelling Water Storage Tank	NA (note 1)	NA	NA	C
Pressuriser	I	None	III	B
Condensate system Valve	NA (note 1)	NA	NA	C
Feedwater System Valve	NA (note 1)	NA	NA	C

Table 4 - Results from applying the SSMFS 2008:13 methodology.

Note 1 - Based on *Figure 1* no CI assigned; DM screening not completed and, Inspection Group automatically assigned.

4.1.2. EPRI traditional RI-ISI

Summary of the methodology

The EPRI traditional RI-ISI approach [7] was first approved for use by the US Nuclear Regulatory Commission (US NRC) in 1999 and was codified in ASME Section XI nonmandatory appendix R, supplement 2 (Methodology B). It is the foundational methodology not only for ISI of piping welds but has been extended and adapted to other plant components and processes as discussed below.

Piping systems are divided into segments based both on the pipe rupture potential and its CoF.

A failure potential category is determined based on the identified DM. Known DMs are catered for within the method and a failure potential category is assigned based on industry experience of the likelihood of a large break developing in piping (e.g. flow-accelerated corrosion is assigned a high break potential whilst stress corrosion cracking is likely to result in leak-before-break behaviour, with a small corresponding leak, and is therefore assigned medium failure potential).

Finally, each segment, which includes all the elements within the segment, is placed in the appropriate place on the EPRI Risk Characterisation Matrix based on CoF and failure potential (see Figure 2). In this version the correlation between qualitative and quantitative PoF and CoF estimates is shown.

The consequence category is determined from the plant-specific PRA and calculation of the conditional core damage probability (CCDP) and the conditional large early release probability (CLERP) associated with the postulated failure. For SSC where a postulated failure does not result in an initiating event (e.g. LOCA), but the failure degrades or fails a system essential to plant safety, consequence category is assigned based on the expected frequency of unavailability, exposure time, and number of back-up trains. Where no back-up trains exist, plant safety systems are typically treated as High Safety significant (HSS).

Risk Characterisation Matrix		Consequence Category			
		None	Low CCDP < 1E-6 CLERP < 1E-7	Medium CCDP 1E-6 – 1E-4 CLERP 1E-7 – 1E-5	High CCDP > 1E-4 CLERP > 1E-5
Failure Potential/ Probability (eg pipe rupture)	High 2E-6 /y	Low (Cat 7)	Medium (Cat 5)	High (Cat 3)	High (Cat 1)
	Medium 2E-7y	Low (Cat 7)	Low (Cat 6)	Medium (Cat 5)	High (Cat 2)
	Low 1E-8 /y	Low (Cat 7)	Low (Cat 7)	Low (Cat 6)	Medium (Cat 4)

Figure 2 - EPRI Risk Characterisation Matrix

Application to non-piping components

Currently, the approach is limited to piping and piping welds with extension to mechanical components underway (e.g. welds in pressure vessels, nozzle welds in vessels, pressure retaining bolting > 2-inch diameter and welds in pumps & valves). When applying the methodology to mechanical components, the CoF step is conducted first. For those components with a low consequence, a low safety significance rank is assigned, and no further work is needed (e.g. no inspection required). For those components with a high or medium consequence rank, a failure potential evaluation like that contained in TR-112657 [7] is needed. This step is currently under development.

Results

For the case study components, the results in Table 5 were obtained.

Component	Consequence	Degradation Mechanisms Identified	Degradation Category	Risk Group
Boron Injection Tank	Medium	None Identified	Low	Low, category 6
Refuelling Water Storage Tank	High (defence-in-depth)	None Identified	Low	Medium, category 4
Pressuriser	High (LOCA potential)	None Identified	Low	Medium, category 5
Condensate system Valve	Low	TT combined with water hammer (WH)	High	Medium, category 5
Feedwater System Valve	Medium	None identified (but inspection would be required for Flow Accelerated Corrosion (FAC), if applicable)	Low	Low, category 6

Table 5 - Results from applying the EPRI traditional RI-ISI methodology

4.1.3. CC N-716

Summary of the methodology

Code Case N-716 [8][10] is a streamlined RI-ISI process based upon lessons learnt from numerous approved RI-ISI applications. Currently approved for use on piping, piping welds and non-piping pressure boundary components (e.g. welds in pressure vessels, nozzle welds in vessels, pressure retaining bolting > 2 inches in diameter and welds in pumps and valves). In the Code Case N-716 approach, the consequence assessment has been replaced with a pre-determined set of high safety significant locations (e.g. reactor coolant system, break exclusion area) and a plant-specific assessment of the impact of pressure boundary failure using the plant PRA directly.

Application to non-piping components

The methodology is developed for the treatment of non-piping pressure boundary components, so there is no specific limitation in the application.

Results

For the case study components, the results in Table 6 were obtained. It should be noted that the RWST is not currently within the scope of N-716. If it were included, it would be HSS, due to its importance from a defence in-depth perspective.

Component	Safety Significance	Degradation Mechanisms Identified
Boron Injection Tank	LSS* (noted to be high pressure)	None Identified
Refuelling Water Storage Tank	HSS	None Identified
Pressuriser	HSS	None Identified
Condensate system Valve	LSS	TT combined with WH
Feedwater System Valve	LSS	None identified (but inspection would be required for FAC, if applicable)

Table 6 - Results from Application of the CC N-716 Methodology

*LSS - Low Safety Significant

4.1.4. PWROG

Summary of the methodology

The Pressurised Water Reactor Owners Group (PWROG) methodology was developed by Westinghouse and is codified in ASME XI Appendix R Supplement 1 Method A [5]. During application of the method, piping is divided into segments, and the consequences of piping failure are postulated and evaluated with the plant PRA model. Failure probabilities are developed using the Westinghouse Structural Reliability and Risk Assessment (SRRA) probabilistic fracture mechanics code [15] for each segment. The results of the PRA model and estimated failure probabilities are combined in the risk evaluation to develop risk metrics both with and without credit for operator action. Based on the safety significance and failure importance, the segments are placed in the structural element selection matrix (see Figure 3). Structural element selection is based on placement in the matrix. All HSS elements affected by active degradation mechanisms are inspected. For the remaining HSS elements, a statistical model (Perdue Model) is used to determine the minimum number of inspection locations.

High Failure Impotance	Owner Defined Program **Region 3**	(A) Susceptible Location(s) (100%) ----------------- (B) Examination Location Selection Process **Region 1**
Low Failure Importance	Only System Pressure Test & Visual Examination **Region 4**	Examination Location Selection Process **Region 2**
	Low Safety Significant	High Safety Significant

Figure 3 - Structural Element Selection Matrix

The primary capability of the SRRA code is to estimate the probability of exceeding the specified limits for a given piping failure mode as a function of time due to the combined effects of the modelled degradation (aging) mechanisms and input uncertainties. The piping failure modes considered include:

- Small leak (through-wall crack).
- Large (system disabling) leak.
- Full break (exceed material flow stress in uncracked section) during a postulated design limiting event.

The piping materials supported are type 304 and 316 stainless steel and carbon (ferritic) steel. The degradation mechanisms that are modelled include:

- Low-cycle and high-cycle fatigue crack growth of existing (fabrication) flaws.
- High-cycle fatigue crack initiation (such as those due to vibration).
- Stress corrosion crack growth of an existing flaw.
- Wall thinning due to wastage (e.g. flow assisted corrosion).

The effects of flaws initiated by high-cycle fatigue or stress corrosion cracking are not fully modelled and are instead approximated.

There are two major challenges of all models based on PFM driven approaches in general. First, the code can only calculate probabilities for the failure modes, materials, degradation mechanisms, input variables and uncertainties modelled. Secondly, the calculated value of probability is the true failure probability of the modelled components if all the failure modes and degradation mechanisms are exactly as modelled and all the input variables, including their uncertainties, are correct, which is highly unlikely. Accordingly, it is necessary to validate input parameters and assumptions against service experience to validate the model outputs and generate accurate results. However, there is very limited data on failures available and therefore uncertainties are significant. When applying a PFM based approach to piping systems it can be argued that the piping system and segments within in it share many commonalities compared with distinctly different components (e.g. a pressure vessel and a valve). Therefore, while the absolute failure probabilities predicted via PFM may be subject to significant uncertainty, the relative probabilities of failure of different segments within a piping system are significantly more likely to be accurate. Thus, the risk significance of different locations can be compared and ranked.

Application to non-piping pressure boundary components

For the consequence analyses of the different component types considered within the case study, a variety of approaches must be applied. For valves, existing PRA data can be used as an approximation. In the case of tanks, some caution must be taken because the consequence of a sudden rupture of the tank could have a major indirect effect. The plant's flooding analyses can be used to estimate the consequences. Similarly for failure of the Pressuriser shell, it is conceivable that indirect effects are important. However, in this case consequence of failure can be approximated based on failure of the Pressuriser surge line pipework. This is reasonable as the consequence of failure is already significantly higher than the other components being considered and underestimating this metric will not affect the results meaningfully.

For the PoF analyses, the Westinghouse SRRA code is not designed to model anything other than pipework and hence it has not been possible to calculate quantitative PoF values. Instead estimated failure rates have been approximated as Low, Medium and High through the EPRI methodology.

Results

For the case study components, the results in Table 7 were obtained.

Component	Consequence	DMs Identified	PoF	Risk Group
Boron Injection Tank	Medium	None	Low	Low, category 6
Refuelling Water Storage Tank	High	None	Low	Medium, category 4
Pressuriser	High	None	Medium	High, category 2
Condensate system Valve	Low	FAC	High	Medium, category 5
Feedwater System Valve	Medium	None	Low	Low, category 6

Table 7 - Results from applying the PWROG methodology

4.1.5. 10CFR50.69 - Risk-Informed Repair/Replacement Activities

Summary of the methodology

10CFR50.69 [10] is a US NRC rule approved in 2004 presenting an opportunity to further enhance equipment reliability and plant safety by focusing on those critical systems and components with the highest safety significance. EPRI has developed an approach to implement this guidance for the treatment of SSCs [16].

10CFR50.69 is a voluntary rule that allows a new risk-informed categorisation process to be applied for a range of SSCs. In some cases, the resulting categorisation removes LSS, safety-related components from the scope of the special treatment requirements imposed by US NRC regulations. To distinguish what components are either HSS or LSS, components must be categorised using an US NRC approved process. The Nuclear Energy Institute (NEI) has published a comprehensive categorisation guideline (NEI-00-04) that factors in PRA model insights as well as deterministic insights from an integrated decision-making panel (IDP). By blending the PRA and deterministic insights, an appropriately balanced, technically sound categorisation result is produced.

Repair/Replacement Programme Pre-requisites:

- HSS - Must comply with all existing ASME Code and Quality Assurance requirements.
- LSS – Does not have to comply with ASME Section XI and Quality Assurance Manual requirements applicable to repair/replacement activities (RRAs). Therefore, RRAs can be performed in accordance with less costly/less burdensome Construction Codes and treatment requirements resulting in:
    - Reduced administrative/documentation requirements.
    - Reduced procurement costs.
    - Reduced fabrication and NDE costs for welds.
    - Owner-defined leakage tests instead of ASME Section XI pressure tests.

Application to non-piping pressure boundary components

The methodology consists of the application of the consequence of failure portion of the EPRI traditional RI-ISI methodology coupled with some additional considerations. Those components whose consequence rank is high are assigned to the HSS rank. Those components whose consequence rank is medium or low are assessed against the additional considerations and the result of that assessment determines whether the component is assigned LSS or HSS rank.

Results

For the case study components, the results in Table 8 were obtained.

Component	Safety Significance	Degradation Mechanisms Identified
Boron Injection Tank	LSS (noted to be high pressure)	None Identified
Refuelling Water Storage Tank	HSS	None Identified
Pressuriser	HSS	None Identified
Condensate system Valve	LSS	TT combined with WH
Feedwater System Valve	LSS	None identified (but inspection would be required for FAC, if applicable)

Table 8 - Results from Application of 10CFR50.69 - Risk-Informed Repair/Replacement Activities

4.1.6. CSA N285.7

Summary of the methodology

CSA N285.7 of the Canadian Standard Association (CSA) is a standard that defines requirements for the periodic inspection of the balance of plant pressure-retaining systems, components, and supports that form part of a CANDU NPP using a RI-ISI methodology.

The methodology is an adaptation of the EPRI traditional RI-ISI for use on CANDU reactors (balance of plant systems) which includes piping welds, piping, vessels, pump bodies, valve bodies, tanks, heat exchangers, supports, etc. The approach includes a formal pre-screening process to screen out low-safety components not requiring full RI-ISI evaluations.

One difference between CSA N285.7 and TR-112657 is that the CANDU design uses a large release frequency (LRF) metric rather than a large early release frequency (LERF, as used in the light water reactor (LWR) fleet) metric. Another difference is that the CANDU design contains some systems not found in the LWR fleet, and some of these have different operating conditions and fluids as compared to the LWR fleet, so the degradation mechanism evaluation methodology of TR-112657 needed to be updated to fit the CANDU situation.

Application to non-piping pressure boundary components

The methodology is developed for the treatment of non-piping pressure boundary components, so there is no specific limitation in the application.

Results

For the case study components, the results in Table 9 were obtained.

Component	Consequence	Degradation Mechanisms Identified	PoF Category	Risk Group
Boron Injection Tank	Medium	None Identified	Low	Low, category 6
Refuelling Water Storage Tank	High (defence-in-depth)	None Identified	Low	Medium, category 4
Pressuriser	High (LOCA potential)	None Identified	Low	Medium, category 4
Condensate System Valve	Low	TT combined with WH	High	Medium, category 5
Feedwater System Valve	Medium	None identified (but inspection would be required for FAC, if applicable)	Low	Low, category 6

Table 9 - Results from Application of the CSA N285.7 Methodology

4.1.7. ASME Section XI Division 2 – Reliability and Integrity Management Programmes for Nuclear Power Plants

Background

ASME Section XI, Division 1 was developed and evolved over more than 40 years but focused on existing PWR and Boiling Water Reactor (BWR) LWR technology. ASME Section XI, Division 2 RIM was developed as a technology neutral in-service code that can be applied to all reactor types with specific code supplements intended to account for different reactor designs.

Although ASME Section XI, Division 2 is a technology neutral code, it should be noted that at the time of writing that 10 CFR 50.55a(g) specifically requires the use of ASME BPV Section XI, Division 1 for boiling and pressurised water reactor designs.

Summary of the methodology

The ASME Code, Section XI, Division 2, provides requirements for creating RIM programmes, with the purpose of ensuring that the reliability and integrity of passive components are properly managed and that reliability targets are defined, achieved, and maintained throughout the plant lifetime. The RIM programme can cover implementation of traditional non-destructive methods, repair/replace activities, sample inspections or leak monitoring type activities to SSCs within scope.

RIM evaluates all SSCs for their impact to plant safety and reliability and establishes examination, tests, operation, monitoring, and maintenance requirements to ensure the SSCs meet the plant risk and reliability goals. This approach contrasts with the prescriptive approach used by Division 1 which uses Class 1, Class 2 and Class 3 approach to ISI with each Class having less rigorous criteria.

Development of the RIM programme follows a detailed, largely qualitative, process (supported by quantitative evidence) that has been mainly developed for application to new plant and different reactor designs. However, in principle, there is no technical aspect of the code that would prohibit its application to an existing plant design, and a section is marked within the Division for future development.

Establishment of a RIM programme requires the appointment of the RIM Expert Panel (RIMEP) and the Monitoring and NDE Expert Panel (MANDEEP).

- RIMEP is responsible for the technical oversight and direction of the risk-informed aspects of RIM programme development and implementation:
    - Establishes RIM Scope.
    - Establishes Reliability Targets.
    - Identifies RIM Strategies.
    - Addressing of uncertainties.
- MANDEEP is responsible for all aspects of NDE:
    - Develops Monitoring and NDE (MANDE) specifications.
    - MANDE qualification.
    - Inspection requirements.
    - Minimum acceptance level of the MANDE.

RIMEP and MANDEEP work together to establish a MANDE programme to be assigned to those SSCs that have been agreed to be within the scope. Justification for excluding SSCs from the MANDE scope must be documented. The MANDE is formulated based on expected credible degradation mechanisms in concert with ensuring an individual SSC's contribution to plant risk remains within the target reliabilities. The establishment of the RIM programme and the work of the RIMEP and MANDEEP requires a wide range of inputs which are summarised in Figure 4.

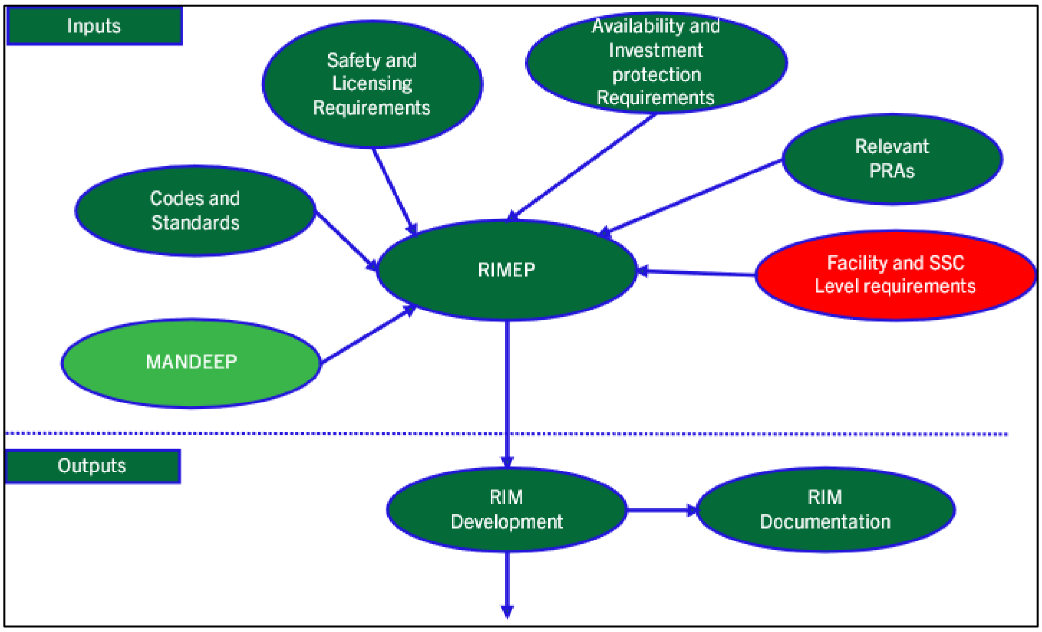

Figure 4 - Inputs to RIMEP for RIM Programme Development

Commentary

ASME Code, Section XI, Division 2, provides a comprehensive set of requirements for creating RIM programmes. It is a complex process that requires extensive inputs to the process and a wide range of technical expertise. ASME Section XI, Division 2 provides a largely risk-informed framework for the through-life monitoring and NDE of all SSCs.

Given the focus on application to new (non-PWR/BWR) reactor designs it has not yet reached a stage of maturity and industry understanding that facilitates ready application. Indeed, as far as SAE understands it has not been implemented or piloted on any plant design, and early efforts toward this goal are on-going. Although the process is logical and thorough there are aspects that are not clearly understood and would benefit from clarification as the code is developed and piloted.

Specific challenges to the application of ASME XI, Division 2 include:

- Complexity of the framework for RIM.
- Lack of maturity of and experience with developing RIM.
- Regulatory position regarding application to LWR designs.
- Interface between quantitative metrics and qualitative processes.
- Resource burden.
- Determination of frequency/periodicity of inspection.

As such, it is the view of SAE that it is currently not viable to perform a useful assessment to allow a reasonable comparison to be made to the other methodologies.

5. Deterministic Approaches

Two deterministic approaches were applied to the case study components:

- Occupational safety of pressurised vessels/piping at Swedish NPPs based on the Swedish code AFS 2023:11 [4].
- ASME Code Section XI Division 1 [5].

5.1. ISI requirements according to the Swedish regulations AFS 2023:11

The Swedish Work Environment Agency's regulations and general advice (AFS 2023:11) on work equipment and personal protective equipment-safe use [4] classify pressure-retaining components according to their volume, pressure, temperature and media. The regulation is not specific for NPPs but SSMFS stipulate that the regulation should be used to handle occupational safety for component with no radiation hazards.

According to the code, pressure-retaining components are classified into two classes, A and B. Components in class A require inspection and system checks, while in class B only system checks are required. The inspection interval is typically 4 years but can be extended to 6 years, and up to 10 years for vessels and piping, depending on the previous inspection results and overall status. Inspections are typically visual (inside and outside) and could be supplemented with NDE.

Application to the case study

The following results were obtained with the AFS 2023:11

- Boron Injection Tank – 10§, Fluid group 2a, P×V>1000 bar litres gives Class A, Inspection interval every 4 years and system check every 2 years.
- Refuelling Water Storage Tank – 12§, Fluid group 2a and t>65°C, V>50000 litres gives Class B, system check every 3 years.
- Pressuriser – Not included - follows SSMFS 2008:13.
- Valve in the Condensate system – 11§, Connecting to DN=500, Fluid group 2a, 3500 < P×DN<5000 barlitres gives Class B, system check every 4 years.
- Valve in the Feedwater System – 11§, Fluid group 2a, DN350, P×DN>5000 bar litres gives Class A inspection interval every 4 years and system check every 4 years.

5.2. ISI requirements according to the ASME Section XI

The ASME codes are followed in many countries. The ASME Code Section XI, Division 1 provides rules for in-service testing and inspection, as well as repair and replacement of NPP components, pressure vessels and piping. The inspection requirements are given in a series of articles. To highlight, some of the most important ones include:

- IWA-2000 Examination and Inspection, which covers e.g. defining the inspection programme and extent of examination.
- IWB-2000 Examination and Inspection, which covers e.g. for Class 1 Components of LWR plants the determination of the inspection schedule.
- IWC-2000 Examination and Inspection, which covers e.g. for Class 2 Components of LWR plants the determination of the inspection schedule, to supplement IWB-2000.

- IWC-3000 Acceptance Standards, which covers e.g. necessary supplemental examinations and applicable acceptance standards.
- Mandatory Appendix III Ultrasonic Examination of Vessel and Piping Welds.
- Mandatory Appendix IV Eddy Current Examination.

It can be seen, that ASME XI, Division 1, covers inspection requirements for NPP components very thoroughly, including the typically used inspection techniques.

Application to the case study

Table 10 summarises the inspection requirements of case study components according to the ASME Section XI. Of the components included in the case study, only the Pressuriser and the boron injection tank would be included in the inspection programme.

Component	**Inspections**	**Interval/years**
Pressuriser	• Ultrasonic Testing (UT) of both (bottom and top) shell-to-head welds and 300 mm of the two longitudinal welds intersecting shell-to-head welds. • UT of circumferential head weld. • UT of nozzle-to-shell welds (six welds). • Liquid Penetrant Testing (PT)/Magnetic Particle Testing (MT) of welded attachments.	10
Boron injection tank	• UT of shell circumferential welds. • UT of head circumferential welds (top and bottom shell-to-head welds). • UT and PT/MT of nozzle-to-head welds (top and bottom heads). • PT/MT of welded attachments.	10
Refuelling water storage tank	ASME Section XI exempts atmospheric tanks from non-visual NDE.	NA
Valve in condensate system (414)	ASME Section XI does not cover examination requirements for Class 4 components.	NA
Valve in feedwater system (415)	ASME Section XI does not cover examination requirements for Class 4 components.	NA

Table 10 - Inspection Requirements within the Scope of Deterministic ASME Section XI

6. Analysis of results

Results of the risk ranking of the case study components using the Risk-Informed and deterministic approaches considered during the case study are presented in Table 11.

The results from the RI methods are compared using the EPRI risk categories, which are common to the majority of the methods. The CSA and RI-RRA methods have been developed from the EPRI methodology, and the CSA risk characterisation matrix is the same as the EPRI risk characterisation matrix. The RI-RRA process uses the "consequence portion" of the EPRI RI-ISI methodology supplemented with "additional considerations" to define final high and low safety significance. High consequence rank is considered HSS with no further review and medium/low consequence rank has been subject to additional considerations and then categorised as HSS or LSS. The same approach has been applied to the N-716 method based on links to the EPRI RI-ISI methodology. The PWROG results are strictly quantitative, therefore, to allow comparison, the PRA/Failure probability values are matched to the thresholds values that are behind classification of the CCDP potential and pipe rupture potential in the EPRI risk characterisation matrix (see Figure 1).

The main differences in the rankings shown in Table 11 are connected to the methods and assumptions/decisions made within each methodology. For example, for the Boron Injection Tank, requirements invoked via ASME XI Division I is strictly connected to the safety class and the Boron Injection Tank is a Class 2 component. When the requirements of the applicable part of the code are applied, IWC-2000, ISI is required. In contrast the RI-ISI methods produce a low-risk significance categorisation. For AFS 2023:11, the high pressure and volume of the BIT require inspection based on the conventional risk posted by the tank.

In the case of SSMFS, the scope of the method only includes systems up to the second isolation valve from the RPV, so systems beyond that boundary will always be placed in Inspection Group C (Low). For this reason, the RWST receives a low risk ranking and leads to the greater variation in result for this component.

When comparing the results, it should also be noted that the CC N-716 and RI/RRA methods are based on simplifications of the EPRI traditional methodology. The outcome of these methods is limited to HSS or LSS and so it is not possible to assign the medium risk category, which explains some of the variation in the categories.

Overall, the results demonstrate general agreement between the methods considered, noting that some simplification has been necessary. This outcome should not be unexpected as many of the methods are linked to the EPRI traditional methodology.

Method	**Boron Injection Tank**	**Refuelling Water Storage Tank**	**Pressuriser**	**Valve in Condensate System**	**Valve in Feed Water System**
SSM	Low (C)	Low (C)	Medium (B)	Low (C)	Low (C)
EPRI	Low	Medium	Medium	Medium	Low
N-716	Low	High	High	Low	Low
PWROG	Low	Medium	High	Medium	Low
CSA	Low	Medium	Medium	Medium	Low
RI/RRA	Low	High	High	Low	Low
ASME XI Div 1	Volumetric Inspection	Visual inspection only	Volumetric Inspection	No inspection	No inspection
Source of Current ISI Requirements	AFS 2023:11 (A)	AFS 2023:11 (B)	SSMFS 2008:13 (B)	AFS 2023:11 (B) Maintenance programme	AFS 2023:11 (A)

Table 11 - Comparison of the results of applying RI-ISI risk ranking and deterministic (grey shading) ISI site selection approaches to the case study components

7. Discussion

7.1. Review of Case Study Findings

The main purpose of the case study was to analyse whether there is sufficient data on non-piping components to apply a RI approach for a wide range of mechanical components. The applications show that for methods based on semi-quantitative approaches informed by expert judgement, the need for data is very limited. In these cases, the risk assessment is based mainly or entirely on the possible consequences of the component failure, often combined with an assessment of the likelihood of a degradation mechanism being active. These approaches are heavily influenced by service experience from existing reactor fleets to promote sites within the risk ranking where experience suggests degradation is more likely, and thus the relative risk is higher.

One area where data was lacking was CoF values for some components. For these components, it has been possible to apply approximations to estimate CoF by taking PRA results of the largest connecting piping (or highest consequence piping) as a method of assigning CoF via data that is typically already available. As discussed earlier in the report, this is likely to be an accurate approximation for many component types (e.g. valves). For large tanks or pressure vessels, consequences of failure may be more complex that simply loss of coolant, and secondary consequences may be more severe than the basis for PRA evaluation of connecting piping. For more complex components, the impact of such approximations must be carefully considered. In these cases, the application of a methodology such as Failure Mode, Effects, and Criticality Analysis (FMECA) is likely to be required to better quantify CoF. In this case study the approximations were not thought to be significant as the components affected are already HSS and toward the top of the risk ranking and comfortably above the threshold where ISI would typically be applied. Additionally, more accurate consideration of secondary consequences is likely to further increase CoF, rather than decrease risk and remove the components from ISI scope.

The method most affected by lack of input information was the PWROG approach as the method relies on a quantitative estimate of PoF. The is challenging for a component that is more complex than a simple piping weld. There may be several plausible degradation mechanisms, the geometry is likely to be complicated and varied at different locations around the component, and there may be several relevant failure modes to consider. Existing structural reliability risk assessment models are not typically flexible enough to cater for all potential component types and degradation mechanisms. There is also a sparsity of data to provide confidence that the output of such models would be reliable when used to compare diverse components subject to different degradation mechanisms, failure modes and consequences. The development of new, or evolution of existing codes, to cater for these challenges is a significant undertaking and a potential barrier for these methods that must be overcome.

Overall, the results demonstrate a general trend of the simplification of methods (e.g. ranking only as LSS or HSS) to enable more widespread application of RI-methods as a means of delivering more targeted inspection and replacement activities. Effectively, granularity within the risk assessment output has been traded for conservatism, for ease of application and as a method of overcoming lack of data. This approach means that it is not necessary to produce a refined estimate of CoF and PoF to varying degrees. In most cases it is therefore not possible to complete a refined ranking of the risk significance of sites.

As has been discussed in earlier reports [17], increased use of RI methods typically results in significant benefits for plant safety and therefore greater application of the methods should generally be welcomed. A potential drawback of the simplification of methods is that information is lost on the relative risk significance of components, particularly HSS components, which can be used in some circumstances to make more informed decisions.

For example, due to the difficulty of PoF estimation, the RI-RRA approach developed by EPRI is fully consequence-based. It has been considered by industry to be more cost-effective not to evaluate failure probabilities. However, in some cases, a simplified modelling of PoF could be beneficial in some circumstances to better understand the likelihood of failure. Such cases include components with high or medium consequences, especially if replacing the component is very costly.

Improved data on the likelihood of degradation mechanisms being significant to PoF and associated data (e.g. potential growth rates) allows ISI to be targeted for greatest risk reduction whilst maintaining an appropriate balance with the burdens of ISI (e.g. dose). Examples of activities that would further benefit from additional data includes:

- Intelligent ISI site selection, or sampling of potential ISI sites and inspection volumes. For example, where circumstances mean it is not possible to inspect all sites because of access limitations or dose burden etc.
- Selection of site-specific inspection periodicity based on understanding of the relevant degradation mechanisms. References [18][19] provide an example of the application of this type of approach to a HSS location which can be applied more widely. In this example the period between ISIs is extended based on understanding of the degradation mechanism targeted. This type of approach is likely to be more acceptable to regulators for HSS locations.
- Ensuring ISI procedures are targeted and validated for the most relevant degradation mechanisms.
- Ensuring that ISI programmes are appropriately balanced between site selection due to known degradation mechanisms (inspection for cause) and sites selected based on CoF with no known degradation mechanisms of concern.

It is likely that the limitations discussed could be addressed where required via several ways, such as:

- Judgement from the expert panels typically assembled during the application of most RI-ISI methodologies; the approach of using expert panels to refine the outputs from RI-ISI methodologies is well established.
- Targeted application of more refined studies of PoF and CoF where there is reason to believe that relative risk, or safety significance, is being under or overestimated or there is actionable benefit from more data.
- Greater use of semi-quantitative methods for CoF and/or PoF estimation.

7.2. Future Methods Development

To help select the cases where more refined CoF or PoF estimate would be needed, or beneficial, a decision flowchart is presented in Figure 5.

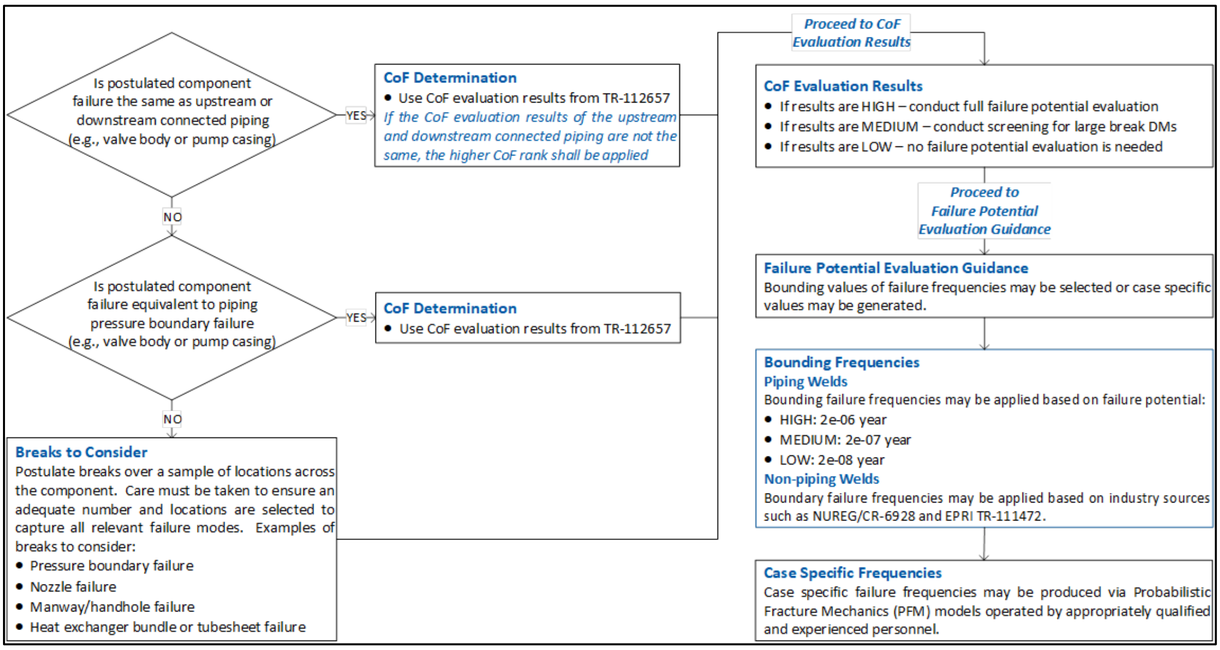

Figure 5 - Flowchart for the Decision Support when Selecting Components for POF Analysis

For guidance, general PoF values for selected representative components from several literature sources are provided in Table 12 – the basis of the values presented can be found in Appendix 1.

Data Source	Boron Injection Tank	Refuelling Water Storage Tank	Pressuriser	Valve in Condensate System	Valve in Feed Water System	Failure Modes Considered	
						Leak	Break
NUREG/CR-6181 rev. 1	2.76E-05	2.34E-05	1.6E-06	NA	N.A.	✓	✓
PNNL-16625	N.A.	N.A.	2.01E-05*	NA	N.A.	✓	×
AERB/NPP/TD/O-1	9.46E-05	9.46E-05	NA	8.76E-05	8.76E-05	✓	✓
INL 2015	3.31E-07	2.57E-07	NA	NA	NA	✓	×

Table 12 - Quantitative PoF estimates (failures per annum) from different literature sources

*Average value based on three scenarios (cases): case 1 accounts for existing service experience with bi-metallic welds in surge line; case 2 assumes equal PWSCC susceptibility for bi-metallic welds in surge line, welds in the Pressuriser spray line and relief line; case 3 is as case 1 except that 1 of 2 flaws in the service experience is near or through-wall.

As discussed, application of purely quantitative approaches to PoF estimation is very challenging due to the complexity of many mechanical components. As a result, an attractive option is to adopt a semi-quantitative approach based on expert judgment and simplified quantification. Such an approach relies on the use of service experience and fundamental knowledge of degradation mechanisms as per most RI-ISI approaches. Hence there is a need to agree on the governing degradation mechanisms. The IAEA

International Ageing Lessons Learned (IGALL) [20] documentation provides useful up-to-date information on this issue and the Ageing Management Program (AMP) series of documents, which are freely available [21].

Component screening could be based on material type, loads and process environment to define one or two governing damage mechanisms for each component, or component region. The Swedish PMT approach applies this kind of logic, with three levels of degradation potential. There are slightly different opinions about the optimal number of classes for degradation potential. Normally three is adequate, as the approach needs to be simple. On the other hand, the limitation of the three categories may be considered too limited, as the differences in PoF values can be several orders of magnitude. One possibility could be to adopt a methodology given in American Petroleum Institute's Risk Based Inspection approach (API RP 581) [22] where PoF is calculated based on the chosen generic failure frequency being multiplied with so-called Damage Factors that are time-dependent and specific for each degradation mechanism.

7.3. Review of Challenges Identified for Expansion of RI-ISI

Several challenges have been identified for the extension of RI-ISI to mechanical components, presented in Table 1 and Table 2 in Section 2.

Table 13 reviews the challenges and provides a commentary based on the experiences gained during the case study.

Topic	**Challenge**	**Findings and Comments**
Collection of necessary data	Availability of documentation, especially for low safety class components.	Based on the components reviewed, this case study confirms that detailed data may not always be available for low safety class components (e.g. assembly drawing available with bill of material (BOM) but detailed information lacking). This presents an overhead for the application of RI methods where the data isn't available, but this challenge does not preclude application. Lack of available data on PoF for some components is also a limitation but can be overcome via adoption of the conservative approximations discussed in this section.
Definition of scope and screening	Definition of component-adapted screening metrics and criteria	As the case study was limited to a small set of components instead of e.g. entire systems, this topic was not fully covered. Where CoF evaluation exists for piping, at least those components that are connecting to piping will have data available to inform CoF.
Assessment of consequences	Evaluation of the consequences of general mechanical components	Lack of data can be overcome via approximations using piping PRA data but does result in loss of data associated with secondary consequences that could be significant in some situations. For this case study it was not judged to be a limitation. For investigation of the flooding effects the plant flooding PRA could be used.
	Assessing other than reactor safety consequences.	Outside of the scope of the case study.
Assessment of failure probabilities	Lack of reliability data or SRM tools to estimate the failure probabilities.	As discussed earlier in this chapter, the estimation of PoF values can be limited to a small group of components where it is worthwhile. For the assessment, simplified qualitative / semi-quantitative expert judgements could be used in the absence of reliability data or SRMs. This is not currently a barrier but there

		are many opportunities available to take benefits from improved PoF estimation.
Risk characterisation and ranking	Choice of risk criteria, limits for failure probability and consequence categories, and treatment of risk outliers.	In principle, similar criteria and treatment of outliers as in piping RI-ISI should be applied. At present HSS mechanical components, which would be risk outliers, are automatically subject to inspection and efforts to apply risk-informed thinking currently focus on modification to the inspection requirements (e.g. tailoring of inspection volumes or extension of inspection periodicity) based on understanding of the relevant degradation mechanism. There are opportunities to apply these types of approach more widely where supporting data is available.
Balancing ISI Programmes	Consideration of how to ensure a balanced RI-ISI programme can be achieved and the use of a risk cut-off.	
Proposal for new ISI programme	Requirements for new qualified inspection methods on new components and materials, definition of sample size for components of small population.	The case study was limited to risk ranking and did not consider the inspection methods. This could be a topic for further studies. In general, the effect of inspections can be modelled with probability of detection (POD) curves. These curves are dependent on inspection method and material type. This solution is difficult to implement without supporting data which can be cost prohibitive to obtain. Greater use of the principles discussed in RP8 [23] may provide a more accessible route forward.
Effectiveness assessment	Uncertainties and lack of data; for many of the systems / components addressed, a formal ISI programme does not exist.	When there is no existing ISI programme to compare with for LSS components, the effectiveness assessment cannot be based on comparison. However, it can be assumed that the application of a systematic approach consistent with piping RI-ISI analyses results in a more balanced and coherent inspection programme.
High CoF Structures, Systems, and Components (SSCs)	Development of a consensus position on assigning ISI based upon risk for high CoF SSC.	As discussed, risk-informed principles can be used to further tailor requirements for high CoF SCC and there are already examples of developing bespoke requirements for such components (e.g. [18][19]).
Semi-Quantitative RI-ISI Methods	Review of RI schemes primarily based upon operating experience, failure statistics and expert judgement.	Several of the methods reviewed are heavily influenced by current light water reactor fleet experience and expert judgement. Greater use of these methods has been highlighted as a route to allow RI-ISI methods to be more widely deployed, as discussed in Subsection 7.2.

Table 13 - Review of the Case Study Findings against the Challenges raised in Tables 1 and 2

8. Conclusions

The case study results show that it is possible to apply RI approaches to mechanical components. Challenges exist but as demonstrated they can largely be overcome.

Most of the methods considered already apply a semi-quantitative approach to the scoring of PoF that does not rely on quantitative predictions (e.g. via probabilistic fracture mechanisms) and these methods can be extended most easily. Methods based on quantitative predictions can also be extended but the availability of appropriate models that can cater for a wide range of component types is lacking.

Existing PRA data can generally be used where needed to assign CoF metrics for non-piping components. However, PRA may be limited to the consequences of pressure boundary failure and resulting coolant leakage only. Where relevant, care must be taken that secondary, or in-direct, consequences of failure are captured and accounted for (e.g. missile and jet generation, flooding etc).

This approach is most applicable to components similar to piping (e.g. valves) and more detailed review of the FMECA is likely to be required for more complex components.

The review shows that there is a trend of simplifying methods, particularly simplification of PoF assignment, to reduce the resource requirements of applying RI approaches and balancing with conservatism. In general, this approach has been preferred by industry to allow more widespread application of RI-ISI and this approach is equally applicable to the scope of components considered. In some cases, it may be beneficial to invest more in the quantification of failure probabilities (e.g. for components close to a threshold). To aid identifications of these cases, ENIQ SAE recommends a graded approach to the analysis of the risk-importance of non-piping pressure boundary components, and a decision tree has been recommended for use to identify where more detailed consideration of failure modes and PoF analyses are warranted.

The review also highlights that several of the components considered are classified as high safety significance and would automatically be selected for ISI. Hence there is limited benefit from applying the methods considered relative to deterministic approaches. However, it is highlighted that RI principles can be more widely applied to these components by focusing on leveraging understanding of PoF, relevant degradation mechanisms, and fleet service experience to target ISI techniques and tailor inspection intervals. This approach is likely to be more acceptable to safety regulators and there are examples of this approach being applied in some design codes.

9. Recommendations

ENIQ SAE recommends that there would be benefits from further work in the following areas:

- Integration of the principles discussed within ENIQ Recommended Practice 8 [23] on graded levels of inspection qualification into RI-ISI methodologies.
- Consideration of greater use of POD curves, or similar approaches, to take a more quantitative approach for accounting for the effect of inspection reliability during RI-ISI site selection.
- Further consideration of the integration of RI principles for ISI requirements for HSS components within deterministic codes such as ASME XI Division I. For example, using fleet experience and knowledge of relevant degradation mechanisms to tailor requirements such as inspection periodicity.

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Appendix 1: Background Information on PMT2004

As discussed in 4.1.1 the assignment of the Inspection Group was determined in more detail, as a combination of a Consequence Index (CI) determined based on proximity (of piping) to the reactor vessel and isolation valves, and a Damage Index (DI) determined based on degradation mechanism potential and mechanical fatigue.

The Consequence Index (1 – 3) is determined based on proximity (of piping) to the RPV and isolation valves. A summary is illustrated in Figure 6, where the value 1 indicates the greatest consequence. Where special arguments can be made based on deterministic and/or probabilistic safety analyses, the impact index can be changed one step (raised or lowered) in relation to these guidelines. Where components fall outside of the scope of Figure *6* (e.g. due to the protection provided via isolation valves, or non-return/check valves), no CI is assigned.

The Damage Index (I – III, I being the most severe) is determined based on degradation mechanism potential (e.g. mechanical fatigue) using simplified screening criteria that are similar in intent to those included within other methodologies (e.g. ASME XI Division II Mandatory Appendix VII). DIs are largely based on experience from operating reactors. If no significant cause of degradation can be identified the damage index is set to III.

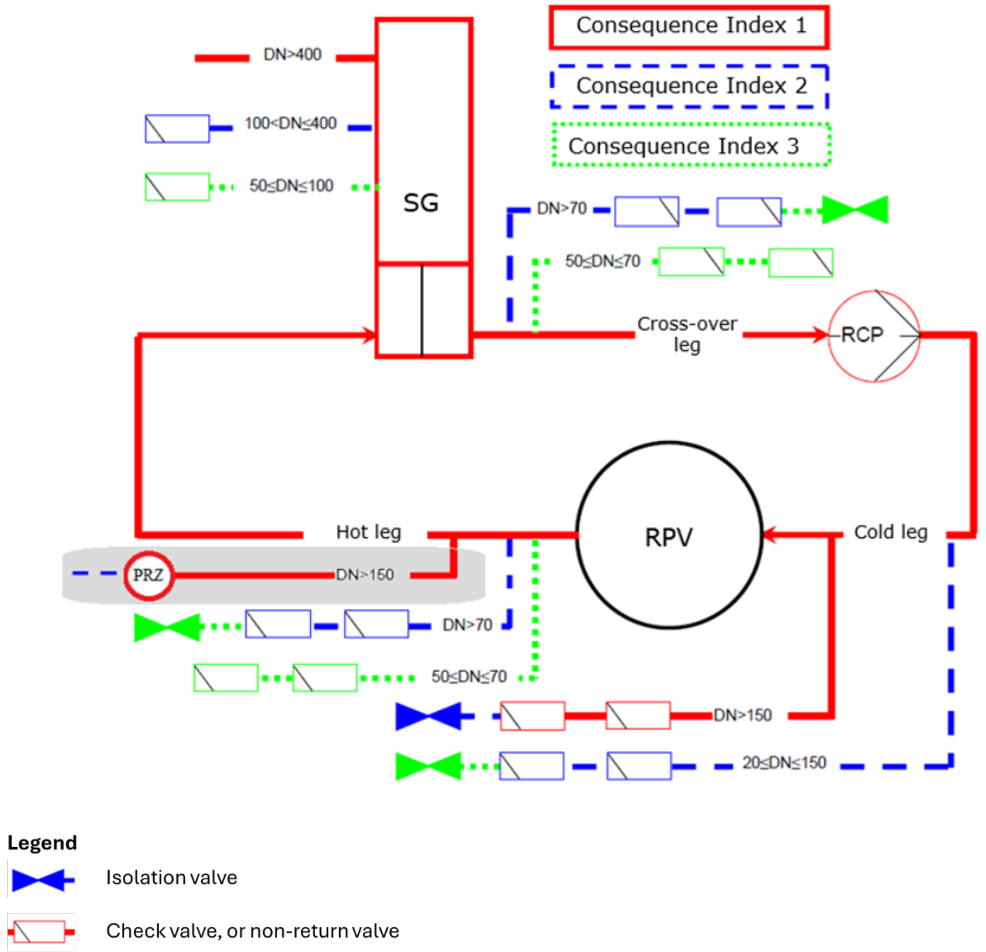

Figure 6 - Criteria for consequence index in piping and pressure-retaining components of a PWR unit according to PMT2004.

Appendix 2: Basis of PoF Values (Table 12)

The presented PoF values for the tanks and the Pressuriser according to NUREG/CR-6181 (1997) are based on estimated rupture frequencies (see [24]). These probability of failure (PoF) estimates have been obtained from expert judgement elicitation analyses performed in the early 1990s. Therein rupture corresponds to break or large leak. The presented PoF data does not show which of these two failure modes is specifically considered. Expert judgement elicitation was applied due to lacking statistical rupture data from nuclear power plants (NPPs). Because of this, the presented PoF data is conservative to varying extents.

Table 12 also presents the PoF value for the Pressuriser surge line nozzle according to PNNL-16625 (2007) [25]. The PoF value provided is the average estimate based on three cases (see footnote in Table 12). The PoF values from PNNL-16625 are based on evaluation of service experience for the selected components. The reported PoF or rather failure data has been collected between the early 1980s and 2006 based on worldwide experience available. This data has been obtained mainly from articles and NUREG reports, such as NUREG/CR-4792, Vol. 3, NUREG/CR-6335, NUREG/CR-6674 and NUREG-1829. PNNL-16625 notes that there were only two reported events for the surge nozzle location, linked to cracks detected within weld/repair welds, but no failures that involved through-wall cracks. A failure frequency was assessed with consideration of the welds in the relevant population of plants and the corresponding number of plant years of operation.

The PoF values for the tanks and valves from AERB/NPP/TD/O-1 (2006) are also provided in Table 12 - Quantitative PoF estimates (failures per annum) from different literature sources These PoF values are based on recommended leak and break frequencies for these components. The former PoF values are used here. The latter PoF values were not used, as there was no statistical data to back them up (e.g. leak frequencies with small break consequence of failure and separately large break frequencies with consequence of failure of large breaks). The AERB/NPP/TD/O-1 report refers to several background documents including many NUREG reports, such as NUREG/CR-1205, NUREG/CR-1331, NUREG/CR-1363, NUREG/CR-1740, NUREG/CR-2728 and NUREG/CR-2815. These reports have been published between 1980 and 1990. In most cases the PoF values in them are based on statistical analysis of data from licensee event reports (LERs), but in some cases expert judgement has been applied as well. Other notable background references include the WASH-1400 report from 1975, IEEE 500 report from 1984 and RKS 85-25 Reliability Data Book from 1985. In these documents the presented PoF values are based on operating experience statistics.

PoF data from INL 2015 (2017) generally represents reliability data for a period between 1998 to 2015, directly obtained using the Reliability and Availability Data System (RADS). The original set of component reliability data sheets were extracted from NUREG/CR-6928 and generally contain data from the period spanning from 1998 to 2002. Here industry-average failure rates taken from INL 2015 correspond to the applied PoF values. The considered failure state is a large leak. The PoF values for the unpressurised tank are used for refueling water storage tank (RWST) and those for the pressurised tank for BIT. The PoF values for the valves could not be used as the considered failure state is failure to open/close the valve, i.e. not any actual degradation state. The PoF values shown in Table 5 have been taken directly from the INL 2015 report. Failure to open/close and internal leakage are not relevant to this case study.

PNNL-16625, AERB/NPP/TD/O-1 and INL 2015 do not provide consequence values corresponding to the given PoF data. Therefore, assessing quantitative or qualitative risk would be difficult to do. NUREG/CR-6181, Rev. 1 provides core damage frequency values together with the PoF values. To apply these in risk assessment, the core damage frequency values would first need to be converted to conditional core damage probability (CCDP) or conditional large early release probability (CLERP) values.

Abbreviations

API	American Petroleum Institute
ASME	American Society of Mechanical Engineers
BIT	Boron Injection Tank
BWR	Boiling Water Reactor
CCDP	Conditional Core Damage Probability
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CI	Consequence Index
CLERP	Conditional Large Early Release Probability
CoF	Consequence of Failure
CSA	Canadian Standards Organisation
DI	Damage Index
DM	Degradation Mechanism
ENIQ	European Network for Inspection and Qualification
EPRI	Electric Power Research Institute
FAC	Flow Accelerated Corrosion
FMECA	Failure Modes, Effects and Criticality Analysis
HPSI	High-Pressure Safety Injection
HSS	High Safety Significant
IGALL	International Ageing Lessons Learned
ISI	Inservice Inspection
LERF	Large Early Release Frequency
LOCA	Loss Of Coolant Accident
LRF	Large Release Frequency
LSS	Low Safety Significant
LWR	Light Water Reactor
MANDE	Monitoring and Non-Destructive Examination
MANDEEP	MANDE Expert Panel
MT	Magnetic Particle Testing
NPP	Nuclear Power Plant
OECD NEA	Nuclear Energy Agency of the OECD
PFM	Probabilistic Fracture Mechanics

PoF	Probability of Failure
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PSI	Preservice Inspection
PT	Liquid Penetrant Testing
PWR	Pressurised Water Reactor
PWROG	Pressurised Water Reactor Owners Group
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RI	Risk Informed
RIM	Reliability and Integrity Management
RIMEP	RIM Expert Panel
RPV	Reactor Pressure Vessel
RWST	Refuelling Water Storage Tank
SAE	Sub-Area Inspection Effectiveness
SG	Steam Generator
SRM	Structural Reliability Model
SRRA	Structural Reliability and Risk Assessment
SSC	Systems Structures and Components
SSM	Swedish Radiation Safety Authority
TASCS	Thermal Stratification, Cycling, and Striping
TT	Thermal Transient
USNRC	US Nuclear Regulatory Commission
UT	Ultrasonic Testing
WH	Water hammer

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ABOUT ENIQ AND NUGENIA

The **European Network for Inspection and Qualification (ENIQ)** is a utility driven network working mainly in the areas of qualification of non-destructive testing (NDT) systems and risk-informed in-service inspection (RI-ISI) for nuclear power plants (NPPs). Since its establishment in 1992 ENIQ has issued over 70 documents. Among them are the "European Methodology for the Qualification of Non-Destructive Testing" and the "European Framework Document for Risk-Informed In-Service Inspection". ENIQ is recognised as one of the main contributors to today's global qualification guidelines for in-service inspection.

ENIQ is the technical area 8 of NUGENIA, one of the three pillars of the Sustainable Nuclear Energy Technology Platform (SNETP) that was established in September 2007 as a R&D&I platform **to support technological development for enhancing safe and competitive nuclear fission in a climate-neutral and sustainable energy mix.** Since May 2019, SNETP has been operating as an international non-profit association (INPA) under the Belgian law pursuing a networking and scientific goals. It is recognised as a European Technology and Innovation Platform (ETIP) by the European Commission.

The international membership base of the platform includes industrial actors, research and development organisations, academia, technical and safety organisations, SMEs as well as non-governmental bodies.

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