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EC-JRC/OECD-NEA Benchmark Study on Risk Informed In Service Inspection Methodologies (RISMET)

CSNI Integrity and Ageing Working Group (IAGE)

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FOREWORD

The concept of risk-informed in-service inspection (RI-ISI) has been successfully implemented in several NEA member countries, as reported in the CSNI state-of-the-art report NEA/CSNI/R(2005)9 “Review of International developments and Cooperation on Risk-Informed In-Service-Inspection (RI-ISI) and Non-destructive Testing (NDT) Qualification in OECD-NEA member countries”. This state-of-the-art report is based on information on practices in NEA member countries collected in 2003 through a questionnaire and documented in the NEA report NEA/CSNI/R(2005)3, and on the proceedings (NEA/CSNI/R(2004)9) of the CSNI Workshop on International Development and Cooperation on RI-ISIand NDT Qualitifation held from in 2004in Sweden.

These reports noted that methodologies for risk-informed in-service inspection (RI-ISI) of nuclear power plants have been developed in several countries, although the only two widely applied methods are those developed by the Pressurised Water Reactor Owners Group (PWROG)/ASME and by the EPRI in the USA. To-date, there had not been any direct comparison of various RI-ISI methodologies applied to an identical scope of components (system, class, etc.). Recommendations and support for performing a benchmark study of various RI-ISI approaches had been given by several international groups and committees.

The idea of a benchmark study on RI-ISI was initially developed within a subgroup of the ENIQ Task Group on Risk. The preliminary JRC project proposal was endorsed by the CSNI Working Group on Integrity and Ageing of Components and Structures (IAGE), which resulted in the approval by the CSNI in December 2005.

The Benchmark Study on Risk-Informed In-Service Inspection Methodologies (RISMET) project eventually kicked-off at Leibstadt NPP (Switzerland) in January 2006 with the objective to apply various RI-ISI methodologies to the same case (namely, selected piping systems in one nuclear power plant). The comparative study was aimed at identifying the impact of such methodologies on reactor safety and how the main differences influence the final result (i.e. the definition of the risk-informed inspection programme).

This report documents the main results and conclusions of the RISMET project which constitutes a unique comparative study of selected approaches used to set up an ISI programme. As a result, the knowledge of different approaches and their impact on plant safety is likely to be enhanced and at the same time, the use of risk-informed ISI will be promoted.

ACKNOWLEDGEMENTS

EC-JRC and NEA/CSNI would like to thank the management of Ringhals NPP (Sweden) for agreeing to be the host plant, and for making all the required data available to the project participants.

This work represents the collective effort of the RISMET participants all of whom provided valuable time and considerable knowledge toward its production.

In offering thanks to these experts, the JRC and NEA wish to provide particular appreciation to Anders Leijon (Ringhals NPP) for supporting the project with his thorough knowledge of the plant, for providing all the required data and for leading the Application Groups “PWROG-SE” and “SKIFS”; Paul Stevenson (Westinghouse) for leading the Application Group “PWROG (original)” and the Evaluation Group “Scope of Application”; Patrick O’Regan (EPRI) for leading the Application Groups “EPRI” and “Code Case N-716”; Jos Slechten (Tractebel) and Carlos Cueto-Felgueroso (Tecnatom) for leading the Application Group “ASME section IX”; Bengt Lydell (Sigma-Phase, Inc) for leading the Evaluation Group “Failure Probability Analysis”; Göran Hultqvist (Forsmark NPP) for leading the Evaluation Group “Failure Consequence Analysis”; Björn Brickstad (SSM) for leading the Evaluation Group “Regulatory Aspects”; and Kaisa Simola (VTT) for chairing the project and for leading the Evaluation Group “Risk Ranking and Site Selection”.

The RISMET members contributing to this report were:

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EXECUTIVE SUMMARY

RI-ISI methodologies have been developed in several countries, although the only widely applied methods are those developed by the Pressurised Water Reactor Owners Group (PWROG)/ASME and by the EPRI in the USA. Recommendations and support for performing a benchmark study of various RI-ISI approaches has been given by several international groups and committees, including the CSNI.

By means of benchmarking the different methodologies it would be possible to identify how they impact reactor safety and whether they lead to significantly different results. The benchmarking could also result in the identification of critical paths (i.e. those phases in a methodology with the greatest potential to affect the outcome) and might suggest areas for further improvement.

The Benchmark Study on Risk-Informed In-Service Inspection Methodologies (RISMET) project was approved by the CSNI in December 2005 based on the interest of several organisations, including nuclear utilities, regulators, consultants and international bodies, in carrying out the benchmark exercise.

Objectives of the RISMET project

The overall objective of the project is to apply various RI-ISI methodologies to the same case (namely, selected piping systems in one nuclear power plant). More specifically, the following general objectives are defined:

1. To compare qualitative and quantitative RI-ISI methodologies (for instance: EPRI, PWROG, OMF Structures, SKIFS, etc.) with traditional ISI programmes, i.e. deterministic programmes that are based on the conventional, established safety classification of components (e.g. ASME section XI, NE-14) and augmented programmes developed in response to a particular issue (e.g. break exclusion regions, flow assisted corrosion, localised corrosion);

2. To study and compare these methodologies, namely:

   to compare the risk rankings obtained by different methodologies and evaluate the significance of differences in the results;

   to identify the critical paths in each applied methodology;

   to highlight good practices in each applied methodology;

3. To identify how the various approaches fulfil recommendations put forward in the NRWG document [4], in the ENIQ Framework Document for RI-ISI [5] and in the work of the NURBIM project.

4. To provide a basis for further development of existing or new methodologies, in order to achieve optimisation of the various factors involved (e.g. risk, radiation dose to workers, allocation of financial resources, simplicity and flexibility in application, fitness to regulation).

To achieve these objectives, the RISMET benchmark was composed of Application Groups, responsible for applying the methodologies, and Evaluation Groups, to analyse the information provided.

Host plant and systems included

The benchmark was limited to include four systems at Ringhals unit 4 (R4), a Westinghouse NSSS designed PWR nuclear power plant (NPP). Several criteria were identified for selecting systems to be included in the scope: all safety classes should be covered; a variety of degradation mechanisms should be covered; good coverage of risk categories should be achieved; systems with a significant increase or decrease in the new inspection programme (before/after applying RI-ISI) should be included; balance between initiating and mitigating systems should be ensured.

Based on these criteria, the following four systems were suggested by Ringhals and approved by the project team to be considered:

1. Reactor Coolant System (RCS)
2. Residual Heat Removal System (RHRS)
3. Main Steam System (MSS)
4. Condensate System (CS)

Applied methodologies

The following approaches to define the ISI programme were considered in the benchmark exercise:

Swedish regulatory requirements (“SKIFS”)

PWROG original methodology (“PWROG (original)”)

PWROG methodology adapted to Swedish regulatory requirements (“PWROG-SE”)

EPRI methodology (“EPRI”)

EPRI streamlined RI-ISI methodology (“Code Case N-716”)

ASME Section XI (deterministic) (“ASME section IX”)

SKIFS Methodology

The existing ISI programme of R4 is based on the Swedish regulations SKIFS1994:1. The approach in the Swedish regulations is based on assessing qualitatively the probability of cracking or other degradation (Damage Index) and what consequences (Consequence Index) this may have. Inspection groups are determined on the basis of these indexes as shown in the following table.

The SKIFS 1994:1 gives the following requirements: The majority of components within inspection group A shall be inspected. In group B, a well balanced sample inspection may be sufficient. For cases where there are no damage mechanisms, but inspections are motivated due to high consequences, the sample should contain at least 10 % of the components within inspection group B. Inspections by qualified NDE systems are required in inspection groups A and B. For the selection of sites for inspection group C (low risk), availability and occupational safety aspects are considered. SKIFS does not require pressure testing as part of the ISI.

Risk matrix for ranking of components according to SKIFS

	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			

PWROG methodology

In the PWROG methodology, formerly known as the Westinghouse Owners Group (WOG) methodology, the piping in the scope of application is divided into segments. The consequences (both direct and indirect) of the piping failure are postulated and evaluated with the plant PSA model. Failure probabilities are developed using the probabilistic fracture mechanics code (SRRA) for each of the consequences on the segments.

Based on the safety significance and failure importance, the segments are placed in the structural element selection matrix, shown in the next table. All High Safety Significant (HSS) elements affected by active degradation mechanism are inspected. For the remaining HSS elements, a statistical model is used to determine the minimum number of inspection locations. Pressure / leakage testing requirements remain in effect regardless of region in the structural element selection matrix. The acceptability of implementing the RI-ISI programme is demonstrated by calculating the change in risk for both CDF and LERF and by conducting a review for defence in depth philosophy.

PWROG structural element selection matrix

HIGH FAILURE IMPORTANCE	**3** **OWNER DEFINED PROGRAMME**	**1 (A) SUSCEPTIBLE LOCATION(S) (100%)** **1 (B) INSPECTION LOCATION SELECTION PROCESS**
LOW FAILURE IMPORTANCE	**4** **ONLY SYSTEM PRESSURE TEST & VISUAL EXAMINATION**	**2** **INSPECTION LOCATION SELECTION PROCESS**
	LOW SAFETY SIGNIFICANT	**HIGH SAFETY SIGNIFICANT**

PWROG-SE methodology

The PWROG-SE is an adaptation of the PWROG (original) methodology to the Swedish regulatory environment. The approach basically follows the PWROG (original) methodology in segmentation, failure probability and consequence analyses initial risk ranking. An expert panel is used to verify the initial risk ranking.

In the consequence evaluation, a different concept was applied with respect to loss of residual heat removal and loss of reactor coolant inventory. Also, the risk ranking is redone eliminating the impact of vibration fatigue. In the phase of structural element selection, the inspection sites are classified in the three inspection groups A, B and C according to the SKIFS (see the description above).

EPRI methodology

The EPRI methodology was developed to be implemented on a system by system basis. In order to conduct and document the analysis, the piping systems are divided into segments based both on the pipe rupture potential and its consequences. Each segment, which includes all the elements within the segment, is placed onto the appropriate place on the EPRI Risk Characterisation Matrix shown in the next table. The failure potential category is determined on the basis of identified degradation mechanism. The consequence category is determined from the plant-specific PSA by calculating the conditional core damage probability (CCDP) and the conditional large early release probability (CLERP).

The Risk Categories shown are combined into three risk regions for more robust and more efficient utilisation. For risk Category 1, 2, or 3, the minimum number of inspection elements in each category should be 25 percent of the total number of elements in each risk category. For risk Category 4 or 5, the number of inspection elements in each category should be 10 percent of the total number of elements in each risk category. Pressure / leakage testing requirements remain in effect regardless of risk category (i.e. risk category 1 through 7).

EPRI risk matrix

POTENTIAL FOR PIPE RUPTURE PER DEGRADATION MECHANISM SCREENING CRITERIA	CONSEQUENCES OF PIPE RUPTURE IMPACTS ON CONDITIONAL CORE DAMAGE PROBABILITY AND LARGE EARLY RELEASE PROBABILITY			
	NONE	LOW	MEDIUM	HIGH
HIGH FLOW ACCELERATED CORROSION	LOW Category 7	MEDIUM Category 5	HIGH Category 3	HIGH Category 1
MEDIUM OTHER DEGRADATION MECHANISMS	LOW Category 7	LOW Category 6	MEDIUM Category 5	HIGH Category 2
LOW NO DEGRADATION MECHANISMS	LOW Category 7	LOW Category 7	LOW Category 6	MEDIUM Category 4

Code Case N-716 methodology

Code Case N-716 is a streamlined process for implementing and maintaining RI-ISI, based upon lessons learnt from numerous approved RI-ISI applications. The Code Case N-716 approach differs from the traditional approaches in two respects. First, the consequence assessment is not required. 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 PSA directly. The second departure is that partial scope application, which is allowed by previous RI-ISI approaches, is not allowed by Code Case N-716.

According to the process, the inspection selection should equal to 10% of the high safety significant (HSS) welds, plus augmented programmes for flow accelerated corrosion, localised corrosion (e.g. MIC) and IGSCC in BWRs.

ASME Section XI

ASME Section XI is the most commonly applied approach to define ISI programmes, and is based on deterministic rules. The general philosophy of the ASME section XI aims at determining an inspection sample among the ASME classes 1, 2 and 3 piping. Surface and volumetric non destructive examinations are proposed for ASME classes 1 and 2 whereas class 3 are only submitted to visual examinations and pressure testing at operating conditions. The sampling from classes 1 and 2 is kept well balanced by several requirements which drive the selection process among the overall scope.

The ASME Section XI rules define criteria which allow the exemption of locations where a failure is unlikely or where consequences of a rupture are not severe. All the required examinations must be completed during every inspection interval (an inspection interval equals 10 years of plant service) and the selected samples for the first inspection interval are kept for the successive ones, to the extent practical.

Augmented Inspection Programs

Many risk-informed in-service inspection methodologies and programmes were originally developed as alternatives to deterministic in-service inspection programmes. At many plant sites there are other inspection activities being carried out by plant operators. Some of these inspections are as a result of a commitment to the regulatory body while others are a result of plant specific experiences and good practice initiatives. These inspections were developed to address a specific issue (e.g. break exclusion region, operative degradation) rather than provide a level of defence in depth like the deterministic ISI programme. These other inspection programmes have names such as "augmented" or "owner defined" programmes. Additionally, in some countries, these augmented inspections have been incorporated directly into the deterministic ISI programme while in other countries each inspection programme remains a separate programme onto itself. Because of these differences in intent, it is important that if these augmented programmes exist, that they should be integrated into (or coordinated with) the RI-ISI programme in manner that is logical and defensible.

Evaluation Groups

Evaluation Groups were composed of generalists and experts in specific areas to assess the safety impact and compare various aspects of the applications. Analyses to be considered by the Evaluation Groups included for instance: identification of differences in the analysis in all phases including results; analysis of the importance of identified differences; and comparison with more "traditional" inspection programmes. Four main Evaluation Groups were formed: (1) "Scope of Application"; (2) "Failure Probability Analysis"; (3) "Failure Consequence Analysis"; and (4) "Risk Ranking and Site Selection". A fifth group, "Regulatory Aspects", comprised of the RISMET members belonging to regulatory bodies was formed to address regulatory aspects.

Scope of RI-ISI Methodologies Application

RI-ISI methodologies can be used to develop full-scope as well as partial scope RI-ISI applications. Thus, the objective of this work group was to assess the impact of implementing various scopes of application on the ISI results and the technical basis for changes in ISI results, if any, for the various ISI methodologies.

Full and partial scope applications are acceptable for RI-ISI programmes. A full scope application addresses more of the risk and thus provides a more thorough programme with respect to plant safety, since all systems in the plant are considered. Although a partial scope application may not address as much risk as a full scope application, a partial scope application is still valuable based on:

> The application of RI-ISI methods could maintain and/or improve safety even within the partial scope application. In some cases there could be an acceptably small risk increase, and
>
> The safety level of the plant associated to the systems not included within the partial scope application remains the same.

The SKIFS and ASME Section XI offer a pre-determined, fixed scope of application. The Code Case N-716 methodology offers only a single scope of application, full scope. The EPRI and PWROG (original) methodologies offer a range of scopes from small to full scope applications. The PWROG-SE methodology scope of application is the same as the PWROG (original) methodology except that only full scope applications have been performed. Selection of the systems to include in the scope of application is similar for EPRI, PWROG (original), and PWROG-SE methodologies. The SKIFS, ASME Section XI, and Code Case N-716 methodologies all have different methods for determining the scope of application.

Changing the scope of application may impact the results of the RI-ISI programmes for the PWROG (original) and PWROG-SE methodologies since a relative ranking system is utilised. Based on the RISMET study, there is a greater difference between the single system scope and a four system scope than there is between a four system scope and a full scope application. Changing the scope of application in the EPRI methodology does not impact the results within any given system or in the total programme, because an absolute ranking process is used.

Analysis of Failure Probabilities

The objective of the evaluation of the role of pipe failure probability analysis in RI-ISI programme development includes consideration of the following technical issues and associated questions: use of Probabilistic Fracture Mechanics (PFM) and Structural Reliability Models (SRM); use of statistical models of pipe failure, including the role and use of service experience data; relationship (or interface) between service experience data and PFM/SRM; use of expert judgment/expert elicitation; treatment of uncertainties; definition and treatment of different structural failure modes; probability of flaw detection, and inspection intervals; reliability of leak detection and sensitivity of results to leak detection limits; treatment of different degradation mitigation strategies; updating of original failure probability analyses given new service experience, or implementation of piping design changes or new mitigation strategies; compatibility of pipe failure probability analysis approach with PSA requirements; and importance of POF in RI-ISI programme definition.

Results of the quantitative pipe failure probability analysis were available from PWROG-SE only. No other independent calculations were performed by any of the application groups.

Quantitative structural reliability analysis is not a pre-requisite for RI-ISI programme development. However, a successful implementation of any methodology is strongly dependent on an in-depth knowledge of structural integrity management and piping system degradation susceptibilities. The consideration of structural integrity (qualitatively or quantitatively) is one of several steps needed to develop a RI-ISI programme, and it is acknowledged iteratively throughout a programme development process.

Analysis of Consequences

The objective of this evaluation group is to assess how consequences of failures are treated in the different methodologies evaluated in the RISMET project. The work has been performed without detailed review of the Ringhals 4 PSA model.

Ringhals 4 PSA model includes a level 1 and level 2 evaluations for LOCA, transients, common cause initiators, internal flooding, steam releases, piping failures in CVCS, piping failure in auxiliary feed water system during hot standby, small LOCA during hot stand by and overpressure of RHRS.

A major difference between EPRI and PWROG is that EPRI uses an absolute ranking process while PWROG uses a relative ranking process. It follows that potential excessive conservatism in parts of the PSA can have different impact on the risk evaluation for EPRI and PWROG. For EPRI, potential conservatisms may add inspections compared to a best estimate population. For PWROG, potential excessive conservatism can, in addition to affecting the risk measures (RRWs) for those segments/systems whose PSA result is conservative, result in an underestimation of the RRWs for other segments and / or systems. This can affect the segments categorisation as HSS or LSS.

It is very important to check that the PSA model that is used for RI-ISI fulfils the demands for that application, i.e. the PSA is of high quality (high degree of realism) in those areas that are of importance for the risk evaluation.

Evaluation of Risk Ranking and Site Selection

This evaluation group was responsible of identify the main principles of the risk ranking and site selection process in each methodology, to assess the limitations of the benchmark and their effect on the evaluation, the evaluation of the qualitative differences of the results at a system level; and the assessment of the quantitative analyses to investigate the impact of the differences in the inspection site selection on the risk.

Risk and safety importance rankings of various applications cannot be straightforwardly compared, since the regions (safety class, risk region, inspection group) are determined differently. As an example, the PWROG methodology results in smaller number of segments ranked as HSS compared to segments in EPRI Medium and High Risk regions. On the other hand, at least one inspection is assigned to each HSS segment, while in the EPRI approach not even all High Risk segments are covered by inspections. The term “segment” is used in the EPRI approach strictly as an accounting tool (i.e. useful for streamlining the analysis and documentation process) and not as a technical component of the methodology itself. Despite the differences, the safety and risk classifications were considered useful in illustrating and comparing the results at system and segment level.

The RI-ISI methodologies allow a reasonable flexibility in selecting the inspection locations among the high risk sites. When comparing the different applications, it can be noted that all risk-informed approaches would result in significantly fewer (28-47) inspections in the Reactor Coolant System, for example, than the ASME XI application (113).

Sensitivity analyses were carried out to investigate the impact of some major different assumptions with respect to consequence assessment, vibratory fatigue, and FAC. For the various methodologies, these assumptions may affect in various ways the number of safety significant segments as well as the number of sites to be included in the inspection programme.

Conclusions

Risk-informed applications that base the consequence assessment on a plant specific PSA model have the capability of identifying risk important inspection locations that might otherwise be ignored. This is a clear benefit of the RI-ISI approaches, especially in full scope applications where also other than highest safety class systems are considered. In some cases reduction of inspections in primary circuit can be justified, and thus radiation doses can be significantly reduced. The economical benefit for the plant of moving to RI-ISI depends on the present ISI scope, rules and regulations.

In the RISMET project the selection of inspection sites was limited to identifying the number of inspections at segment level. It can be noticed that ASME XI application leads to significantly larger number of inspections in the Reactor Coolant System than any of the other approaches. In the case of the Residual Heat Removal system, there is large agreement on the low safety significance of most of the segments, but there are some significant differences in the ISI scope between the risk-informed applications. For the Main Steam System, there is generally a good accordance of the low safety significance of most of the segments. Some differences in failure potential assessment were identified, as well as differences in PSA interpretation. The Condensate System is excluded from the ASME XI and SKIFS scope. The system however has some risk-significant segments. The dominant degradation mechanism is flow accelerated corrosion (FAC), and in the US and a number of other countries the FAC susceptible piping would be a part of the augmented ISI programme and typically excluded from a RI-SI scope.

RI-SI process itself is a valuable exercise, since it forces the project team to review the piping degradation potential and identify both direct and indirect consequences of piping failures. Also this review may identify more efficient inspection procedures than are presently used. In order to benefit the most of a risk-informed approach, the plant specific PSA model should be of high quality. A lack of coverage can to some extent be compensated by expert judgement. Some of the results from the RISMET study indicate that a more PSA-based procedure, which accounts for the consequences of a pipe leak or break in a more consistent way, may represent an improved methodology.

It should be noted that RI-ISI evaluations often identify risk-significant segments or sites where other safety management measures than inspections may be more useful. For instance components subject to a fast degradation mechanism, such as vibratory fatigue, or components that are difficult to inspect because of materials or design may require alternative approaches.

Even if a ranking of segments or sites is initially done based on some risk measure, an important part of RI-ISI approaches is the consideratioin of other factors too. Availability aspects (high failure potential, low consequences), defence in depth (low failure potential, high consequences), radiation doses, accessibility, etc. affect the final definition of the ISI elements. In the PWROG methodology this is explicitly considered in the expert panels. In the EPRI methodology, these issues are addressed by requirements for inspections in various risk categories, and by the considerations of a multi-disciplinary Element Selection Team. In any RI-ISI application the decisions should be documented in a transparent way so that the bases for decisions can be traced and audited.

The effectiveness of the ISI programme depends on the choice of inspected elements, the inspection capability and ISI intervals. The benchmark was focused on the ranking and selection of ISI sites at segments level, and excluded the exact choice of welds or other sites to be inspected, the inspection methods and intervals. Thus it is impossible to judge how close the final selection of inspection locations would have been between applications, and whether the methodology would have had an impact on ISI method selection. For the US applications of EPRI and PWROG methodologies a 10 year ISI interval for inspections is an established practice. When the RI-ISI is applied to a wider scope including active degradation mechanisms, the ISI interval could also be a subject of optimisation.

ACRONYMS

AFWS	Auxiliary Feed Water System
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
CCDF	Conditional Core Damage Frequency
CCDP	Conditional Core Damage Probability
CDF	Core Damage Frequency
CI	Consequence Index
CLERF	Conditional Large Early Release Frequency
CLERP	Conditional Large Early Release Probability
CS	Condensate System
CST	Condensate Storage Tank
CVCS	Chemical & Volume Control System
DI	Damage Index
DMA	Degradation Mechanism Assessment
ECC	Emergency Core Cooling
ENIQ	European Network for Inspection and Qualification
EP	Expert Panel
EPRI	Electric Power Research Institute
FAC	Flow Accelerated Corrosion
HSS	High Safety Significant
IGSCC	Intergranular Stress Corrosion Cracking
ISI	In-service Inspection
LBB	Leak-Before-Break
LERF	Large Early Release Frequency
LOCA	Loss-of-Coolant-Accident
LSS	Low Safety Significant
MSS	Main Steam System
NDE	Non-Destructive Examination

NURBIM	Nuclear Risk-Based Inspection Methodology
NURBIT	Nuclear Risk Based Inspection Tool
PFM	Probabilistic Fracture Mechanics
POD	Probability of Detection
POF	Probability of Failure
PRA	Probabilistic Risk Assessment
PRAISE	Piping Reliability Analysis Including Seismic Events
PSA	Probabilistic Safety Assessment
PWR	Pressurised Water Reactor
PWROG	Pressurized Water Reactors Owners Group
PWSCC	Primary Water Stress Corrosion Cracking
RAW	Risk Achievement Worth
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RHRS	Residual Heat Removal System
RI-ISI	Risk-informed In-service Inspection
RISMET	Benchmark Study on Risk-Informed In-Service Inspection Methodologies
RIVAL	Ringhals RI-ISI project
RPV	Reactor Pressure Vessel
RRW	Risk Reduction Worth
RWST	Refuelling Water Storage Tank
SCC	Stress Corrosion Cracking
SER	Safety Evaluation Report
SG	Steam Generator
SKIFS	Swedish Regulatory Requirements
SRM	Structural Reliability Model
SRRA	Structural Reliability and Risk Assessment
SSM	Swedish Radiation Safety Authority (former SKI)
TF	Thermal Fatigue
V&V	Verification & Validation

1. INTRODUCTION

1.1 Background

RI-ISI methodologies have been developed in several countries, although the only widely applied methods are those developed by the Pressurised Water Reactor Owners Group (PWROG)/ASME and by the EPRI in the USA. The RIBA project [1] provided a summary and comparison of four different RI-ISI methodologies, namely PWROG, EPRI, EDF-OMF Structures and the Swedish regulation SKIFS (SKIFS 1994:1, and described in further detail in the utility procedure PAKT). However, the only case study in the RIBA project was the application of the PWROG methodology to the Reactor Coolant system and Auxiliary Feed Water system of Ringhals 4 PWR, and only a general comparison between the PWROG approach and the Swedish procedure for RI-ISI was made. So far there has not been any direct comparison of several RI-ISI methodologies applied to an identical scope of components (system, class, etc.).

Recommendations and support for performing a benchmark study of various RI-ISI approaches has been given by several international groups and committees:

- The USNRC advisory committee on reactor safeguards (letter, May 16 2003, [2]);
- Recommendations of the expert workshop on PSA in RI-ISI (Proceedings of workshop, JRC Petten, 2004, [3]);
- Nuclear Regulators Working Group – Task Force on RI-ISI ([4], 2004);
- ENIQ TG Risk (meeting in Petten, February 9-10 2005);
- OECD/NEA CSNI supported the proposal from the working group on Integrity and Ageing of Components to take initiative to a Benchmark study (meeting in Paris, December 2004).

By means of benchmarking the different methodologies it would be possible to identify how they impact reactor safety and whether they lead to significantly different results. The benchmarking could also result in the identification of critical paths (i.e. those phases in a methodology with the greatest potential to affect the outcome) and might suggest areas for further improvement.

In September 2005, at the Institute for Energy of the Joint Research Centre, Petten, The Netherlands, a preliminary meeting of the RISMET project took place. The meeting confirmed the interest of several organisations, including nuclear utilities, regulators, consultants and international bodies, in carrying out the benchmark exercise. The kick-off meeting of the project was held on January 30-31 2006 in Leibstadt NPP, Switzerland and the last project meeting (the 5th) was again held at JRC Petten in February 2008.

1.2 Project objectives

The overall objective of the project is to apply various RI-ISI methodologies to the same case (namely, selected piping systems in one nuclear power plant). The comparative study aims at identifying the impact of such methodologies on reactor safety and how the main differences influence the final result (i.e. the definition of the risk-informed inspection programme).

More specifically, the following general objectives are defined:

1. To compare qualitative and quantitative RI-ISI methodologies (for instance: EPRI, PWROG, OMF Structures, SKIFS, etc.) with traditional ISI programmes, i.e. deterministic programmes that are based on the conventional, established safety classification of components (e.g. ASME section XI, NE-14) and augmented programmes developed in response to a particular issue (e.g. break exclusion regions, flow assisted corrosion, localised corrosion);

2. To study and compare these methodologies, namely:

   - to compare the risk rankings obtained by different methodologies and evaluate the significance of differences in the results;

   - to identify the critical paths in each applied methodology;

   - to highlight good practices in each applied methodology;

3. To identify how the various approaches fulfil recommendations put forward in the NRWG document [4], in the ENIQ Framework Document for RI-ISI [5] and in the work of the NURBIM project.

4. To provide a basis for further development of existing or new methodologies, in order to achieve optimisation of the various factors involved (e.g. risk, radiation dose to workers, allocation of financial resources, simplicity and flexibility in application, fitness to regulation).

To achieve these objectives, the benchmark was composed of Application Groups, responsible for applying the methodologies, and Evaluation Groups, to analyse the information provided.

1.3 Organisations involved

Table 1 lists the organisations that participated in the project.

Table 1 RISMET project participants

Organisation	Country
Institute for Energy – Joint Research Centre	European Commission
Nuclear Energy Agency (NEA/OECD)	International Organisation
AREVA	France
Association Vinçotte Nucléaire (AVN)	Belgium
Canadian Nuclear Safety Commission (CNSC)	Canada
Électricité de France (EDF)	France

Electric Power Research Institute (EPRI)	USA
Forsmark Nuclear Power Plant	Sweden
Gesellschaft für Anlagen und Reaktorsicherheit (GRS)	Germany
International Atomic Energy Agency (IAEA)	International Organisation
Japan Nuclear Energy Safety Organisation (JNES)	Japan
Leibstadt Nuclear Power Plant	Switzerland
MPA Universität Stuttgart	Germany
US Nuclear Regulatory Commission (NRC)	USA
Nuclear Research Institute REZ (NRI)	Czech Republic
Ringhals Nuclear Power Plant	Sweden
Sigma-Phase, Inc.	USA
Swedish Nuclear Power Inspectorate (SSM, former SKI)[1]	Sweden
Tecnatom S.A.	Spain
Tractebel Engineering	Belgium
VEIKI	Hungary
VTT Technical Research Centre of Finland	Finland
VUJE	Slovakia
Westinghouse Electric Company LLC (Westinghouse)	USA

1.4 Implementation status of RI-ISI methodologies

RI-ISI methods development has been underway for quite a long time. As an example of early work, during 1987-94 the USNRC sponsored work at the Pacific Northwest Laboratory (PNL) which ultimately led to the development of some initial concepts for how to utilise probabilistic safety assessment (PSA) models and PSA insights coupled with structural reliability considerations to establish a more structured approach for in-service inspection (ISI) programme development [6]. In Sweden there has been a regulatory requirement to use a qualitative risk-informed methodology since the late 1980 according to [7]. Practical insights from NDE qualification, service experience data reviews, results from implementation of degradation mitigation techniques, and piping reliability evaluations contributed to the RI-ISI methods that have been implemented by plant owners and operators in the post-1995 timeframe. Active RI-ISI methods development has been pursued by many different organisations.

Except for the RI-ISI methods that have been developed by the Electric Power Research Institute (EPRI), the PWR Owners Group (formerly, Westinghouse Owners Group, or WOG) and the Swedish SKIFS methodology, the implementation status of some of the “alternative” RI-ISI methodologies have not yet evolved beyond the pilot application state. This means that conditional or unconditional regulatory approval for application by plant owner has not yet been obtained. However, it should be noted that in some instances these pilot projects involving application of “alternative” RI-ISI methodologies have produced analysis tools and databases that have found limited use by plant owners for internal follow-up of

[1] Note: As from July 1, 2008, the Swedish Nuclear Power Inspectorate (SKI) has been merged with another authority. The new regulatory authority is named Swedish Radiation Safety Authority (SSM).

inspection activities and for training and planning purposes. Equally important, the insights that have been generated through structural reliability model (SRM) R&D have proved to be valuable in the independent reviews of proposed RI-ISI programmes.

In the USA, the US-NRC has approved both the EPRI and PWROG methodologies as a valid alternative to ASME Sect. XI. RI-ISI is currently applied to the majority of the units in US nuclear power plants. Outside the USA, varying regulatory positions exist, ranging from country specific methodology development to adaptation of the US approaches. The number of applications is constantly growing and is briefly summarised in Table 2.

Table 3 lists the RI-ISI methodologies that are addressed in this report. Additional information on RI-ISI methodologies and the status of applications is documented in [8] and [9].

Table 2 Status of RI-ISI activities outside the USA

Country	Status
Bulgaria	Partial scope application of PWROG methodology
Canada	CANDU Owner's Group (COG) Sponsored Pilot Study Using EPRI Methodology
Czech Republic	EPRI pilot studies, several systems in Temelin (VVER-1000) and Dukovany (VVER-440)
Finland	Full scope RI-ISI projects under way (Loviisa VVER-440 & Olkiluoto BWR), using similar risk matrix as in EPRI methodology, but not following exactly the methodology
France	OMF-Structures methodology piloted to 12 systems
Japan	Activities taking place
Lithuania	DNV's RI-ISI approach pilot
Mexico	EPRI application in process (Laguna Verde BWR), Class 1&2
South Africa	Application of an enhanced EPRI methodology to Koeberg Units 1 and 2. The scope included Class 1, 2 and 3 piping. Full quantification of CCDPs and CLERPs, and development of failure frequencies for all inspection sites.
South Korea	Class 1 and 2 applications of PWROG methodology
Spain	Several applications for RI-ISI programmes have been approved for Class 1 and 2 piping systems (PWROG) in 4 PWRs and 1 BWR
Sweden	The Ringhals PWR units have applied the PWROG-SE methodology, approval process completed
	Oskarshamn pilot study using DNV's RI-ISI approach
	Forsmark pilot study using DNV RI-ISI approach Forsmark pilot study using EPRI Methodology
Switzerland	EPRI pilot study at Leibstadt, PWROG pilot study at Beznau
UK	Risk-based ISI applied for nuclear submarines, not for NPPs
Ukraine	EPRI pilot study at Khmelnitsky VVER-1000

Table 3 Implementation status of RI-ISI methodologies

<table>
<tr><th colspan="2">RI-ISI Methodology</th><th>Implementation Status</th><th>Considered in RISMET</th></tr>
<tr><td rowspan="2">EPRI</td><td>EPRI ^{Base} [10]</td><td>SER ([a]) issued, numerous approved applications.</td><td>Yes</td></tr>
<tr><td>EPRI ^{Quant} [10, 11]</td><td>SER issued, numerous approved applications.</td><td>No</td></tr>
<tr><td colspan="2">Code Case N-716 [12]</td><td>Three pilot applications (Grand Gulf & D.C. Cook U1 & U2) and six follow-on units approved by USNRC. A number of additional units undergoing USNRC review or in the course of preparation ([b]).</td><td>Yes</td></tr>
<tr><td colspan="2">SKIFS 1994:1 [13]</td><td>In use in Sweden since 1994. The precursor FTKA was introduced 1988.</td><td>Yes</td></tr>
<tr><td colspan="2">PWROG [14, 15, 16]</td><td>SER issued numerous approved applications.</td><td>Yes</td></tr>
<tr><td colspan="2">RIVAL([c]) / PWROG-SE</td><td>Conditional regulatory approval by SKI (SKI-2005/1401, [17]).</td><td>Yes</td></tr>
<tr><td colspan="2">OMF-S [18]</td><td>Applied by EdF Central Engineering Department.</td><td>No</td></tr>
<tr><td colspan="2">JNES [19]</td><td>PWR pilot application by JNES.</td><td>No</td></tr>
<tr><td colspan="2">INSPECTA NURBIT [20,21]</td><td>Oskarshamn-1 pilot application.</td><td>No</td></tr>
<tr><td colspan="2">VTT RI-ISI Methodology [22,23]</td><td>Olkiluoto pilot application by VTT.</td><td>No</td></tr>
</table>

Notes:

([a]) SER = Safety Evaluation Report. The US-NRC issues a SER to document its review findings. The respective EPRI & PWROG RI-ISI Topical Reports includes the SER as issued by NRC.

([b]) Code Case N-716 is being revised to incorporate lesson learnt from the pilot plant applications (e.g. addition of a LERF metric, PSA Technical Adequacy description) as well as extension to non-piping components.

([c]) RIVAL (= "Risk-Informerat provningsurVAL") is the Swedish acronym for a RI-ISI project carried out at Ringhals.

2. THE RISMET PROJECT

2.1 Organisation of the project

It was agreed that the underlying principle upon which the project operated was to be fair to all participants. In particular, all participants were responsible for leading the project and achieving consensus. The chairperson for the project, Kaisa Simola of VTT, was elected by the participants of the 1st working group meeting. NEA provided secretariat support and JRC was the project technical coordinator.

To achieve the project objectives, the benchmark was organised into Application Groups, responsible for applying the methodologies, and Evaluation Groups, to analyse the information provided.

Each Application and Evaluation Group had a task leader, with JRC acting as technical support. The technical work in the project (both application and evaluation tasks) was organised individually within each task group. Application groups were composed of participating organisations having knowledge and experience in one or more RI-ISI methodologies. Evaluation Groups were composed of generalists and experts in specific areas to assess the safety impact and compare various aspects of the applications. Analyses to be considered by the Evaluation Groups included for instance: identification of differences in the analysis in all phases including results; analysis of the importance of identified differences; comparison with more "traditional" inspection programmes, as well as the principles and recommendations of NRWG, ENIQ and NURBIM, etc. Four main Evaluation Groups were formed: (1) "Scope of Application"; (2) "Failure Probability Analysis"; (3) "Failure Consequence Analysis"; and (4) "Risk Ranking and Site Selection". A fifth group, "Regulatory Aspects", comprised of the RISMET members belonging to regulatory bodies was formed to address regulatory aspects.

The project started with a preliminary meeting in Petten, in September 2005. Five project meetings were held between February 2006 and February 2008. A final workshop was held in June 2008 in Madrid (Spain) to disseminate the outcome of the exercise.

The project was organised based on in-kind contributions. Each participant was asked to contribute to the work of one or several Application and/or Evaluation Groups.

2.2 Host plant and systems included

The benchmark was limited to include four systems at Ringhals unit 4 (R4), a Westinghouse NSSS designed PWR nuclear power plant (NPP). Ringhals NPP had applied the Pressurised Water Reactor Owners Group (PWROG) methodology to its units and had submitted a RI-ISI programme for unit 4 to a third part body for approval. Thus, preliminary results of the PWROG RI-ISI application were available for the benchmark study, which was a major advantage.

Several criteria were identified for selecting systems to be included in the scope: all safety classes should be covered; a variety of degradation mechanisms should be covered; good coverage of risk categories should be achieved; systems with a significant increase or decrease in the new inspection programme (before/after applying RI-ISI) should be included; balance between initiating and mitigating systems should be ensured. Based on these criteria, the following four systems were suggested by Ringhals and approved by the project team to be considered:

5. Reactor Coolant System (RCS)
6. Residual Heat Removal System (RHRS)
7. Main Steam System (MSS)
8. Condensate System (CS)

The Reactor Coolant System, RCS (313) is a Safety Class 1 system with high consequences (CCDP = $2{\cdot}10^{-2}$) and medium failure probability. The function of the Reactor Coolant System is to transfer heat generated in the core during normal operation to steam generators (SG) where steam is produced to drive the turbine generators. The Reactor Coolant System must also function as part of the residual heat removal system during shutdown and as part of the emergency core cooling system during a loss-of-coolant accident (LOCA). Identified degradation mechanisms are low cycle fatigue and thermal stratification.

Direct consequences associated with piping failures in the reactor coolant system during power operation are large LOCA, medium LOCA and/or small LOCA and depending upon the location of the postulated break, together with loss of some mitigating ability (e.g. one injection loop/train). No indirect effects were identified for the RCS. Primary Water Stress Corrosion Cracking (PWSCC) is not considered since the dissimilar metal welds (RPV nozzle-to-safe-end welds and SG nozzle-to-safe-end welds) are not part of the analysis scope.

The Residual Heat Removal System, RHRS (321) transfers heat from the RCS to the Component Cooling Water System (CCWS) to reduce the temperature of the reactor coolant to the cold shutdown temperature at a controlled rate during the second phase of normal plant cool down and maintains this temperature until the plant is started up again. During the first phase of cool down, the temperature of the RCS is reduced by transferring heat from the RCS to the steam and power conversion system through the steam generators. Parts of the RHRS also serve as parts of the Emergency Core Cooling System (ECCS) during the injection and recirculation phases of a LOCA. Finally, the RHRS is also used to transfer refuelling water between the refuelling cavity and the refuelling water storage tank (RWST) at the beginning and end of refuelling operations. Potential degradation mechanisms include vibration fatigue in small-bore piping and thermal fatigue (thermal stratification, cycling and striping). Failure probabilities can be high.

Direct consequences associated with a RHR piping failure could be a LOCA (non-isolable sections inside containment), or loss of RHR cooling during shutdown operations. Failure to isolate a failed pipe section could also result in a LOCA with containment bypass. Failure of RWST suction piping could lead to loss of RWST inventory and flooding of equipment areas in the Auxiliary Building. Consequence of failure was noted as extremely high (CCDF = $4{\cdot}10^{-1}$, see chapter 5).

The Main Steam System, MSS (411) represents Safety Classes 2 and 4, and has relatively high consequences and high failure probabilities (due to steam hammer). The main function of Main Steam System is to transfer steam from the SG to a steam header for further transfer to the turbine generators and secondary side steam dump system. The steam header consists of a pipe, four isolation valves with associated bypass valves and a number of drain valves. Potential damage mechanisms are low cycle fatigue and steam hammer.

Direct consequences associated with a piping failure during power operation are steam line break and reactor trip together with loss of mitigating systems. Mitigating systems of interest are by-pass feed water, turbine driven auxiliary feed water pump, steam supply to auxiliary feed water pump from one SG, dump operation, turbine stop valves, condenser dump valves and containment isolation. Indirect consequences associated with a piping failure during power operation are manual reactor trip together with loss of mitigating systems due to high temperature/humidity. Mitigating systems of interest are by-pass feed water and dump operation. Jet impingement has been identified for some piping segments.

The Condensate System, CS (414) represents Safety Class 4. The main functions of this system are to preheat, pressurise and transport condensate from the condenser to the feed water pumps. The potential degradation mechanisms include flow accelerated corrosion, low cycle fatigue and water hammer. The system has low consequences but high failure potential with respect to locations identified as potentially susceptible to FAC.

Direct consequences associated with a piping failure during power operation are manual reactor trip together with loss of mitigating systems. Mitigating systems of interest are by-pass feed water from one turbine train and dump operation to one condenser. Indirect consequences associated with a piping failure during power operation are manual reactor trip together with loss of mitigating systems due to high temperature/humidity. Mitigating systems of interest are by-pass feed water and dump operation.

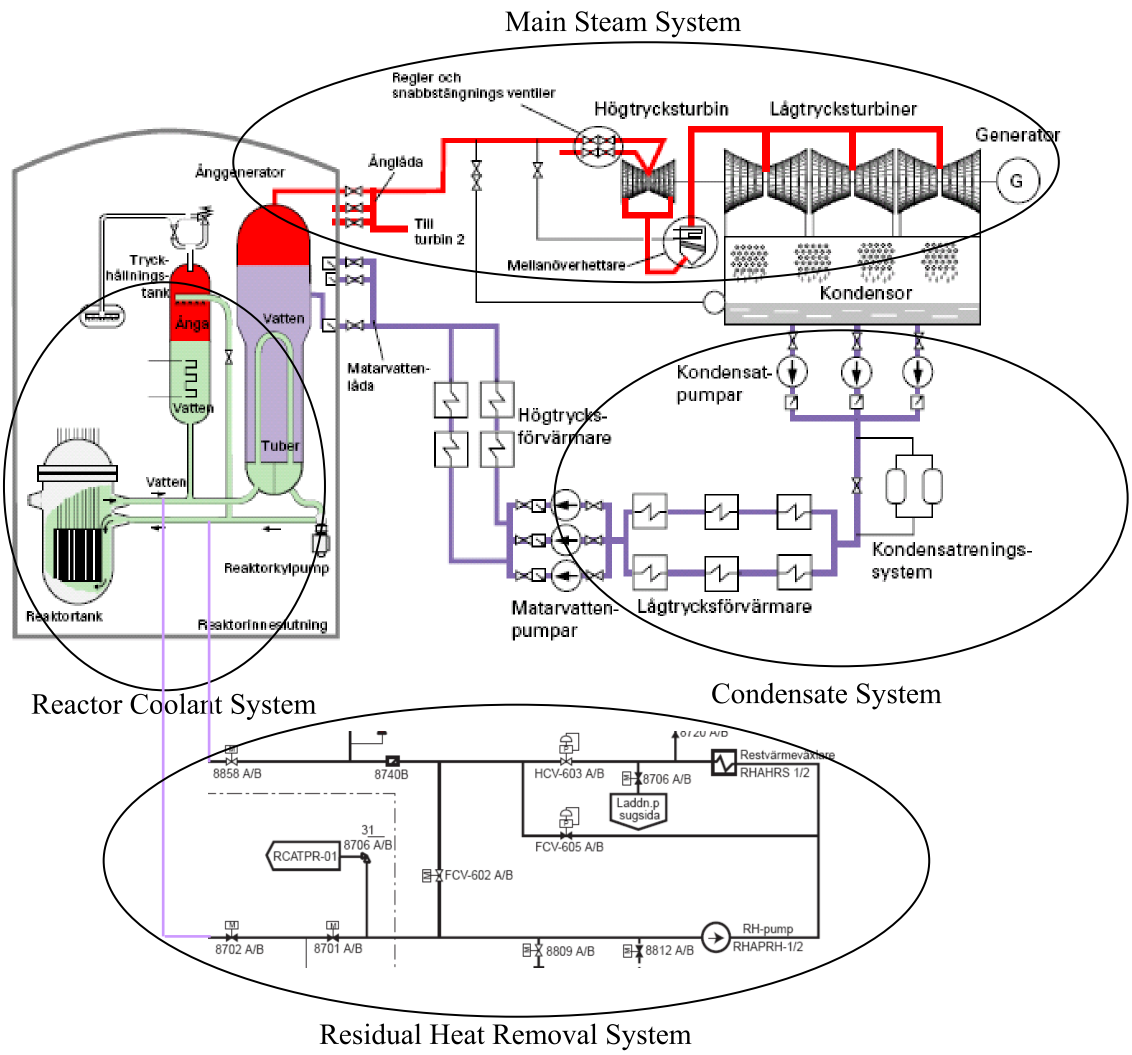

Figure 1 Schematic drawing of Ringhals systems included in the RISMET scope

2.3 Applied methodologies

The following approaches to define the ISI programme were considered in the benchmark exercise:

- Swedish regulatory requirements ("SKIFS")
- PWROG original methodology ("PWROG (original)")
- PWROG methodology adapted to Swedish regulatory requirements ("PWROG-SE")
- EPRI methodology ("EPRI")
- EPRI streamlined RI-ISI methodology ("Code Case N-716")
- ASME Section XI (deterministic) ("ASME section IX")

Augmented inspection programmes were also considered, i.e. for break exclusion regions, flow assisted corrosion, localised corrosion, etc.

These approaches are shortly summarised in the following.

2.3.1 SKIFS methodology

The existing ISI programme of R4 is based on the Swedish regulations SKIFS1994:1 [13]. The approach in the Swedish regulations is based on assessing qualitatively the probability of cracking or other degradation (Damage Index) and what consequences (Consequence Index) this may have.

The Damage Index is a qualitative measure of the probability that cracking or other degradation occurs in the specific component. The aim is to cover all relevant degradation, as no augmented programmes exist according to Swedish requirements. The Consequence Index is a qualitative measure of the probability of such cracking or other degradation will result in core damage, damage of the reactor containment, release of radioactive material or other damages. The Consequence Index is determined by: pipe position relative the core and valves that close automatically in the event of a break; pipe dimensions; and system and thermal technical margins. Inspection groups are determined on the basis of these indexes as shown in Table 4.

The SKIFS 1994:1 gives the following requirements: The majority of components within inspection group A shall be inspected. In group B, a well balanced sample inspection may be sufficient. For cases where there are no damage mechanisms, but inspections are motivated due to high consequences, the sample should contain at least 10 % of the components within inspection group B. Inspections by qualified NDE systems are required in inspection groups A and B. For the selection of sites for inspection group C (low risk), availability and occupational safety aspects are considered. SKIFS does not require pressure testing as part of the ISI.

Table 4 Risk matrix for ranking of components according to SKIFS.

	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			

2.3.2 PWROG methodology

In the PWROG methodology [14-16], formerly known as the Westinghouse Owners Group (WOG) methodology, the piping in the scope of application is divided into segments. The consequences (both direct and indirect) of the piping failure are postulated and evaluated with the plant PSA model. Failure probabilities are developed using the probabilistic fracture mechanics code (SRRA) for each of the consequences on the segments. The results of the PSA model and estimated failure probabilities are combined in the risk evaluation to develop risk metrics both with and without uncertainty. The risk reduction worth (RRW) is the main quantitative importance measure. This probabilistic and additional deterministic data is presented to and used by the expert panel along with their own expertise to determine the final safety significance of the segment.

Based on the safety significance and failure importance, the segments are placed in the structural element selection matrix, shown in Table 5. Structural element selection is based on placement in the matrix. All High Safety Significant (HSS) elements affected by active degradation mechanism are inspected. For the remaining HSS elements, a statistical model (Perdue Model) is used to determine the minimum number of inspection locations. Pressure / leakage testing requirements remain in effect regardless of region in the structural element selection matrix. The acceptability of implementing the RI-ISI programme is demonstrated by calculating the change in risk for both CDF and LERF and by conducting a review for defence in depth philosophy.

Table 5 PWROG structural element selection matrix

	LOW SAFETY SIGNIFICANT	**HIGH SAFETY SIGNIFICANT**
HIGH FAILURE IMPORTANCE	**3** **OWNER DEFINED PROGRAMME**	**1 (A) SUSCEPTIBLE LOCATION(S) (100%)** **1 (B) INSPECTION LOCATION SELECTION PROCESS**
LOW FAILURE IMPORTANCE	**4** **ONLY SYSTEM PRESSURE TEST & VISUAL EXAMINATION**	**2** **INSPECTION LOCATION SELECTION PROCESS**

2.3.3 PWROG-SE methodology

The PWROG-SE is an adaptation of the PWROG (original) methodology to the Swedish regulatory environment. The approach basically follows the PWROG (original) methodology in segmentation, failure probability and consequence analyses initial risk ranking and change in risk calculation, but differs in the structural element selection phases. An expert panel is used to verify the initial risk ranking.

In the consequence evaluation, a different concept was applied with respect to loss of residual heat removal and loss of reactor coolant inventory. Also, the risk ranking is redone eliminating the impact of vibration fatigue.

In the phase of structural element selection, the inspection sites are classified in the three inspection groups A, B and C according to the SKIFS (see the description above). The sample inspection procedure based on the so called Perdue model, as used in the PWROG (original) methodology is not used in the Swedish application. Instead in inspection group A, 100% of the susceptible location is to be inspected and in inspection group B, at least 10% of the structural element should be inspected. Segments that end up in inspection group C will be treated in the owner defined programme. This is in accordance with the Swedish regulations SKIFS 2005:2.

Realising that NDE will not reduce the risk for segments with vibration fatigue, segments with vibration fatigue and a failure probability $\geq 10^{-5}$ were identified and evaluated but then removed in the final risk evaluation in the PWROG-SE application. A special mitigation programme using other techniques than NDE was then implemented to reduce the risks for these segments. The impact of this on the inspection population is presented in chapter 6.

2.3.4 EPRI methodology

The EPRI methodology [10] was developed to be implemented on a system by system basis. In order to conduct and document the analysis, the piping systems are divided into segments based both on the pipe rupture potential and its consequences. While the analysis is conducted on a segment basis, it is for ease of use rather than being a technical component of the analyses. As such, differences in segment definition or segment boundary definition will have no impact on the final results for applications using the EPRI RI-ISI methodology. Each segment, which includes all the elements within the segment, is placed onto the appropriate place on the EPRI Risk Characterisation Matrix shown in Table 6. The failure potential category is determined on the basis of identified degradation mechanism. The consequence category is determined from the plant-specific PSA by calculating the conditional core damage probability (CCDP) and the conditional large early release probability (CLERP): High = CCDP > 10^{-4}; Medium = 10^{-6} < CCDP < 10^{-4}; Low = CCDP < 10^{-6}. For CLERP the boundary values are one order of magnitude smaller.

Table 6 EPRI risk matrix

POTENTIAL FOR PIPE RUPTURE PER DEGRADATION MECHANISM SCREENING CRITERIA	CONSEQUENCES OF PIPE RUPTURE IMPACTS ON CONDITIONAL CORE DAMAGE PROBABILITY AND LARGE EARLY RELEASE PROBABILITY			
	NONE	LOW	MEDIUM	HIGH
HIGH FLOW ACCELERATED CORROSION	LOW Category 7	MEDIUM Category 5	HIGH Category 3	HIGH Category 1
MEDIUM OTHER DEGRADATION MECHANISMS	LOW Category 7	LOW Category 6	MEDIUM Category 5	HIGH Category 2
LOW NO DEGRADATION MECHANISMS	LOW Category 7	LOW Category 7	LOW Category 6	MEDIUM Category 4

The Risk Categories shown are combined into three risk regions for more robust and more efficient utilisation. For risk Category 1, 2, or 3, the minimum number of inspection elements in each category should be 25 percent of the total number of elements in each risk category (rounded up to the next higher whole number). For risk Category 4 or 5, the number of inspection elements in each category should be 10 percent of the total number of elements in each risk category (rounded up to the higher whole number). Pressure / leakage testing requirements remain in effect regardless of risk category (i.e. risk category 1 through 7).

The following references give a good overview of the methodology: [10, 24-29].

2.3.5 Code Case N-716 methodology

Code Case N-716 [12] is a streamlined process for implementing and maintaining RI-ISI, based upon lessons learnt from numerous approved RI-ISI applications. The Code Case N-716 approach differs from the traditional approaches in two respects. First, the consequence assessment is not required. 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 PSA directly. The second departure is that partial scope application, which is allowed by previous RI-ISI approaches, is not allowed by Code Case N-716.

According to the process, the inspection selection should equal to 10% of the high safety significant (HSS) welds, plus augmented programmes for flow accelerated corrosion, localised corrosion (e.g. MIC) and IGSCC in BWRs. HSS welds are selected as follows: 1) a minimum of 25 % of the population identified as susceptible to each degradation mechanism and degradation mechanism combination; 2) for the Reactor

Coolant Pressure Boundary (RCPB), at least two thirds of the examinations shall be located between the first isolation valve (i.e. isolation valve closest to the RPV) and the reactor pressure vessel; 3) a minimum of ten percent of the welds in that portion of the RCPB that lies outside containment (e.g. portions of the main feed water system in BWRs) shall be selected, 4) a minimum of ten percent of the welds within the break exclusion region (e.g. high energy piping penetrating containment) shall be selected.

The following references give a good overview of the methodology: [12, 30-32].

2.3.6 ASME Section XI

ASME Section XI [33] is the most commonly applied approach to define ISI programmes, and is based on deterministic rules. The general philosophy of the ASME section XI aims at determining an inspection sample among the ASME classes 1, 2 and 3 piping (following the safety classifications A, B and C according to Regulatory Guide 1.26, [34]). Surface and volumetric non destructive examinations are proposed for ASME classes 1 and 2 whereas class 3 are only submitted to visual examinations and pressure testing at operating conditions. The sampling from classes 1 and 2 is kept well balanced by several requirements which drive the selection process among the overall scope.

ASME Section XI focuses on one degradation mechanism: failure by fatigue[2]. Therefore welds with significant stress intensity range or cumulative usage factor, dissimilar metal welds and terminal ends connected to vessels are selected first.

The sample number and selection criteria is class specific: 100% for B-F (pressure retaining dissimilar metal welds in vessel nozzles) and 25% for ASME class 1 piping and 7.5% but not less than 28 welds for class 2 piping. If the percentage is not met by terminal ends, dissimilar metal welds and structural discontinuities, additional welds are selected to reach the required percentage.

The ASME Section XI rules define criteria which allow the exemption of locations where a failure is unlikely or where consequences of a rupture are not severe. These criteria are mainly based on nominal pipe size or on the operating conditions of the fluid inside.

All the required examinations must be completed during every inspection interval (an inspection interval equals 10 years of plant service) and the selected samples for the first inspection interval are kept for the successive ones, to the extent practical.

The following references give a good overview of the methodology: [33-35].

2.3.7 Augmented Inspection Programs

Many risk-informed in-service inspection methodologies and programmes were originally developed as alternatives to deterministic in-service inspection programmes such as ASME Section XI, NE-14 and others as described above. In addition to these deterministic inspection programmes, at many plant sites there are other inspection activities being carried out by plant operators. Some of these inspections are as a result of a commitment to the regulatory body while others are a result of plant specific experiences and good practice initiatives. The key difference between these other inspection programmes and the deterministic ISI programme is that these inspections were developed to address a specific issue (e.g. break exclusion region, operative degradation) rather than provide a level of defence in depth like the

[2] Other degradation mechanisms which have appeared during operation of nuclear power plants, such as flow assisted corrosion (FAC) and Inconel 600 safe-end issues, are treated separately in augmented programmes (when enforced by the safety authorities) or owner defined programmes (when voluntary).

deterministic ISI programme. These other inspection programmes have names such as "augmented" or "owner defined" programmes. Additionally, in some countries, these augmented inspections have been incorporated directly into the deterministic ISI programme while in other countries each inspection programme remains a separate programme onto itself.

Because of these differences in intent, it is important that if these augmented programmes exist, that they should be integrated into (or coordinated with) the RI-ISI programme in manner that is logical and defensible. If these programmes do not exist at the plant site, due care should be taken in determining whether they need to be developed prior to, or in conjunction with, a risk-informed ISI application.

The following paragraphs provide a brief description of some of the more important so-called "augmented inspection programmes" (other augmented programmes not discussed here include for instance PWSCC and IGSCC in PWRs).

2.3.7.1 Break exclusion region

Typical general design criteria for nuclear power plants requires that structures, systems, and components important to safety be designed to accommodate the effects of postulated accidents, including appropriate protection against the dynamic and environmental effects of postulated pipe ruptures.

Various "regulation and standards" development bodies have issued document that provide criteria for implementing the above requirement. These include the scope of applicable systems, postulation of break locations, methods for analysing pipe whip forces and displacements, design of rupture restraints, and methods for evaluating the integrity of components subjected to the pipe rupture loads.

For the determination of the locations at which breaks are postulated in high-energy piping, the guidance provides special exclusion rules (e.g. containment penetration areas). These rules recognise that these areas may require extra protection (e.g. to ensure the integrity of the containment and the operability of the isolation valves). The rules provide the option of not specifying breaks in these regions, so that pipe break mitigation devices (e.g. pipe whip restraints) need not be installed in these areas.

Requirements for not specifying breaks in these regions may include special design requirements (e.g. minimise the length of piping, minimise the number of welded attachments) and additional inspections of welds in the plant area of concern. These "additional" inspections are typically made part of the ISI plan and are identified as "augmented" inspections."

2.3.7.2 Flow assisted corrosion

In addition to having portions of the system classified as safety related, power conversion systems may be important to safety for other reasons such as their impact on reliable plant operation and personal safety. For example, failures in portions of these systems can result in undesirable challenges to plant safety systems required for safe shutdown and accident mitigation, can result in complex challenges to operating staff and the plant, and/or can result in potential system interactions of high-energy steam or water with other systems, such as electrical distribution, fire protection, and security.

In response to these concerns a number of plants have implemented programmatic activities to assure reliable system operation. These programmatic activities include developing a more robust understanding of system operating conditions that can adversely impact pressure boundary reliability (e.g. steam quality, corrosion potential, material selection), monitoring system and operational changes (e.g. throttling practices, operational changes, system modifications) as well as updating programmatic activities in response to more significant plant changes such as power uprates and extended fuel cycles.

Plant responses to these impacts include revised system operating practices, changes to system operation, strategic replacement of susceptible components with more resistant materials (e.g. chrome-molybdenum) and conducting inspections to confirm and / or calibrate predicted wear rates.

2.3.7.3 Localised corrosion

Typical general design criteria for nuclear power plants require that provisions be installed for a system or systems that transfer heat from structures, systems, and components important to safety to an ultimate heat sink (e.g. service water systems). Per these design criteria such systems should also allow for appropriate periodic inspection of important components to assure the integrity and capability of the system throughout plant lifetime. In addition, nuclear power plant facilities must meet corrective action programme requirements as defined in their quality assurance programmes.

Operating experience with these systems has shown, in some cases, that these systems or portion of the system are susceptible to localised corrosion such as pitting or microbiological influenced corrosion. The likelihood of degradation and accompanying degradation rates are a result of multiple factors including piping material, operating temperatures, flow conditions (stagnant, infrequent), water quality, water treatment (e.g. biocides, corrosion inhibitors). This experience has resulted in the need for periodic maintenance, refurbishment, lining of components (inner or outer surfaces) as well as implementation of a visual and volumetric inspection programme to continue to assure reliable system operation.

2.4 Limitations and specific aspects of the applications

For the application of EPRI and ASME XI methodologies, the Application Groups obtained the necessary system and process information and drawings from Ringhals. For the EPRI application, Ringhals also provided the necessary PSA results, (e.g. the CCDP and CLERP values) system operating and design information (e.g. design basis documents, operating temperature / practices), system engineer input as well as SRRA inputs and results. In the EPRI application, the PWROG segmentation was reviewed, and segment boundaries changed if needed (e.g. dividing one segment into two segments).

In the ASME XI application, site selection was done following ASME rules as much as possible. For Class 1 piping, the ASME XI rules mandate that 100% of examination category B-F welds (pressure retaining dissimilar metal welds in vessel nozzles) be selected and 25% of examination category B-J welds (pressure retaining welds in piping) be selected. The B-J welds shall be selected based upon terminal ends connected to vessels, stress intensity and usage factors, and dissimilar metal welds. However, if the number of such welds is smaller than the required sample, precise guidelines for the choice of the additional welds are not defined (e.g. random selection). For Class 2 piping, 7.5% of the weld population is inspected and is distributed among terminal welds and structural discontinuities. The Application Group thus made judgments in the selection of sites to reach the required percentage, if this was not met by choosing the designated welds. Design stress analyses, previous NDE results, physical access, dose, etc. were not available, which was a limitation for the application.

In the EPRI application (traditional and streamlined approaches), the number of sites to be included in the inspection programme was determined following the respective methodology. However, for the final selection of inspection sites, an element selection meeting was not held according to the procedure, since it would have required additional plant resources and more detailed information. Instead, it was assumed that the site selection process would probably end up picking the same sites as have been chosen by Ringhals plant.

It was assumed that the exam locations for the EPRI and the Code Case N-716 applications would be the same as the exam locations selected for the PWROG–SE methodology. This is because, during the element selection process, each methodology uses inputs such as access, worker exposure, severity of postulated degradation and inspection history in selecting the final set of exam locations.

However, in practice this may or may not have been true if the selection of the exam locations had been done independently of the results of the PWROG-SE methodology. The principal reason being that in the EPRI and Code Case N-716 methodologies, the segments within a given risk category, are grouped together and a certain percentage of locations are selected across all segments in that category. In the PWROG-SE methodology, in general, at least 10% of the welds are selected for inspection on each HSS segment[3]. Thus it is possible that the site selection would have been different for the EPRI applications.

The PWROG (original) and PWROG-SE methodologies utilise an expert panel to determine the final categorisation of the segments as high or low safety significant. The expert panel could not be convened for the RISMET study. The categorisation process utilised by an expert panel is the same for full and partial scope applications. The only differences for the four systems between full and partial scope applications are associated with the RRWs. Therefore, the final categorisation of segments was estimated based on the full scope PWROG-SE methodology and discussions with Ringhals personnel where there were changes in RRWs that might impact the final categorisation. However, it is possible that more or less or different segments would have been categorised as high safety significant (HSS) for these methodologies but any differences would not be expected to be significant based on the method used to estimate the expert panel final categorisation of segments as low or high safety significant. Segments which were determined to be HSS for deterministic reasons for the full scope application were made HSS for the partial scope applications. Segment's whose RRWs increased to be quantitatively HSS for the partial scope applications were made HSS unless the full scope expert panel made the segment Low Safety Significant (LSS) based on operator actions.

The EPRI application had access to the postulated consequences and corresponding PSA values from the PWROG-SE methodology. As with any risk-informed application, it is possible that had EPRI developed the consequences independently (i.e. without prior knowledge of the PWROG-SE consequences) that different consequences and associated PSA values may have been identified. This may or may not have impacted the categorisation and structural element selection process. There were at least two cases (see section 5.4.1) where EPRI identified differences requiring further study.

The PWROG reactor coolant system (RCS) only application was added late in the study. Due to timing constraints and man-power limitations, the number of exams per segment with butt welds was assumed to be one. This is based on experience where in the vast majority of cases (excluding structural elements with active degradation mechanisms and thin wall piping) the statistical analysis (Perdue model) demonstrated that a single examination was required.

[3] In the PWROG-SE methodology, segments with the same failure potential and that have a limited number of elements (welds) may be combined for the element selection process.

3. SCOPE OF RI-ISI APPLICATION

As discussed in ENIQ Framework document on RI-ISI [5], RI-ISI methodologies can be used to develop full-scope as well as partial scope applications. Thus, the objective of this work group was to assess the impact of implementing various scopes of application on the ISI results and the technical basis for changes in ISI results, if any, for the various ISI methodologies. A second objective of the work group was to assess the definition of segment boundaries used in the various methodologies and to determine their impact on the ISI results and the technical basis for changes in the ISI results, if any. This section of the report documents the results and insights of the work by the RISMET evaluation group "Scope of Application".

This section focuses on the differences between the PWROG (original) methodology, the EPRI methodology, Code Case N-716 (EPRI streamlined methodology), SKIFS and ASME Section XI methodologies included in the RISMET analysis.

3.1 Impact of implementing various scopes of application on the ISI results

3.1.1 Description of RI-ISI methodologies with respect to scope of application

A brief description of the scope of application of each of the methodologies included in the RISMET project is provided in the following paragraphs.

Regardless of the scope of the application, the PWROG, EPRI, Code Case N-716, and ASME Section XI methodologies require that periodic pressure and leakage testing be conducted on Class 1, 2 and 3 piping systems.

PWROG (original) Methodology

For the PWROG (original) methodology, there are two general categories for the scope of application: full scope and partial scope applications. Full scope applications consider all fluid systems, but the systems included in the analysis are based on:

All Class 1, 2, and 3 systems within the ASME Section XI programme,
Piping systems modelled in the PSA; and
Various balance of plant fluid systems determined to be of importance (based on Maintenance Rule ranking if the Maintenance rule is used).

Additionally, piping systems that are identified to cause indirect effects which cause an initiating event or that impact equipment required for safe shutdown are included in the scope of analysis.

Utilising the above criteria, systems which may have piping identified as high safety significant (HSS) and would be inspected are included. Systems not meeting the above criteria are qualitatively screened out since the piping in these systems are unlikely to be identified as HSS and would not be inspected.

Partial scope applications include a subset of piping such as one or more ASME Classes of piping or one or more systems. Unlike full scope programmes, no screening criteria are applied. The piping included in the analysis is based on the ASME Class or Classes or system or systems included in the scope of application.

Per ASME Section XI, certain piping is exempt from ISI. Per the PWROG RI-ISI Topical Report, WCAP 14572 [14], piping that is exempt for ASME Section XI is not exempt for the PWROG RI-ISI methodology and is included in the scope of application. This is true for both full and partial scope applications.

After WCAP-14572 was approved by the United States Nuclear Regulatory Commission (USNRC), ASME developed Appendix R, a non-mandatory appendix for RI-ISI programmes. In Appendix R, for the PWROG methodology, the piping that is exempt for ASME Section XI can be exempted from the RI-ISI programme. Although it may change in the future, at this point in time, the PWROG methodology is based on WCAP-14572 and not Appendix R. Thus, piping that is exempt per ASME Section XI is not exempt for the PWROG methodology. This is based on the following:

WCAP-14572 has been approved by the USNRC.
ASME Appendix R has not been endorsed by the USNRC.
All PWROG RI-ISI applications done to date have been based on WCAP-14572 and not ASME Appendix R.

Most applications of the PWROG methodology are Class 1 only or Class 1 and 2 only applications.

PWROG-SE Methodology

The scope of application for the PWROG-SE methodology is the same as the PWROG methodology except that only full scope applications have been applied. From a practical perspective, a number of plant systems were qualitatively screened out from the RI-ISI application. When the screening was done, the full scope included 21 systems. Forty-four systems were excluded and not treated in the final scope. Examples of these systems include the drinking water system, the fresh water system including fresh water reservoir, the condensate polishing system, the sampling system, the liquid waste processing system, etc.

EPRI Methodology

The EPRI methodology was developed to be implemented on a system by system basis. As such, the methodology can be used to develop partial scope applications including a single system or Class of piping (e.g. Class 1 only), multiple systems or classes of piping (e.g. Class 1 and 2) up to a full scope application which would include all relevant plant systems.

Per the EPRI RI-ISI Topical Report [10] and ASME Appendix R, the EPRI methodology requires, as a minimum, that piping subjected to NDE per the existing ISI programme is included in the RI-ISI scope (i.e. piping that is exempt per ASME Section XI is exempt from the EPRI methodology). Exempt piping can be included in the scope of application per agreement between the plant operator and regulator. Including this exempt piping will not adversely impact the risk significance determination for non-exempt piping (e.g. larger bore piping).

In practice, the vast majority of applications of the EPRI RI-ISI methodology have been Class 1 only or Class 1 and 2 only piping subject to NDE requirements.

Code Case N-716 Methodology

Code Case N-716 is a full-scope application. It requires inspection of certain Class 1 and Class 2 piping as identified in IWB-1200 and IWC-1200 Components Subject to Examination from the ASME Boiler and Pressure Vessel Code, Section XI. Also, any other Class 2, 3 or non-safety related piping whose pressure boundary failure contributions to CDF is greater than 10^{-6} based upon a plant-specific PSA is required to be within the scope of the Code Case N-716 application[4].

The Code Case N-716 methodology is considered a full scope application since the criteria for determining high safety significance based on CDF and LERF is applicable to all piping regardless of size.

SKIFS Methodology

The SKIFS methodology has different criteria for pressurised water reactors (PWRs) and boiling water reactors (BWRs) for what piping is included in the scope of application. For this analysis, the SKIFS methodology is limited to the PWR applications. The SKIFS methodology has a pre-determined, fixed scope of application that is based on a consequence index. If piping meets the criteria for a consequence index of 1, 2 or 3, the piping is considered to be included in the scope of application.
A consequence index of 1 should be assigned to reactor vessel parts and devices with a nominal diameter greater than 150 mm in the main feed circulation system inlet lines, from the reactor pressure vessel to the second containment isolation valve that is automatically closed due to a pipe break. An index of 1 is also assigned to steam generator primary and secondary parts.

A consequence index of 2 should be assigned to devices with a nominal pipe diameter from 20 mm to 150 mm in the main feed circulation system, from the reactor pressure vessel to the second containment isolation valve that is automatically closed due to a pipe break. Corresponding exhaust lines and cross connection parts should be assigned a consequence index of 2 if the nominal diameter is greater than 70 mm. A consequence index of 2 should also be assigned to devices with nominal diameter greater than 150 mm in the main feed circulation system inlet lines after the second containment isolation valve that is automatically closed due to a pipe break and for internal parts in reactor pressure vessel that is important to cool the core, shutdown reactivity and maintaining of the core geometry.

A consequence index of 3 should be assign devices with nominal diameter of 20 mm to 150 mm in main feed circulation system inlet lines after the second containment isolation valve that is automatically closed due to a pipe break. Corresponding exhaust lines and cross connections parts should be assigned a consequence index of 3 if the nominal diameter is greater than 70 mm. A consequence index of 3 should also be assigned to other devices with a nominal diameter greater than or equal to 50 mm in parts of the system that are pressurised with reactor water or is a part of the containment integrity.

ASME Section XI

ASME Section XI has a pre-determined, fixed scope of application, which includes Class 1 and Class 2 piping as identified in IWB-1200 and IWC-1200 Components Subject to Examination. IWD for Welded Attachments to the pressure boundary and pressure / leak testing are also included in the scope. The following components or portions of components are exempt from NDE requirements.

For Class 1:

Piping whose failure results in a flow rate within the capacity of the makeup systems that are operable from on-site emergency power excluding emergency core cooling systems.
Piping with a nominal pipe size of 1 inch and smaller.
Reactor vessel head connections and associated piping with a nominal pipe size of 2 inches and smaller that are made inaccessible by control rod drive penetrations.
Welds or portions of welds that are inaccessible due to being encased in concrete, buried underground, located inside a penetration or encapsulated by guard pipe.

[4] Based on feedback from the USNRC, it is anticipated that the next revision of Code Case N-716 will include a provision that any piping segment whose pressure boundary failure contributions to LERF is greater than 10^{-7} based upon a plant-specific PSA is deemed to consist of HSS welds.

<u>For Class 2:</u>
For piping within residual heat removal, emergency core cooling and containment heat removal systems:

- Piping with a nominal pipe size of 4 inches and smaller for systems excluding the high pressure safety injection system in pressurised water reactor plants.
- Piping with a nominal pipe size of 1.5 inches and smaller within the high pressure safety injection system in pressurised water reactor plants.
- Piping of any size in statically pressurised, passive (i.e., no pumps) safety injection systems of pressurised water reactor plants.
- Piping of any size beyond the last shutoff valve in open ended portions of system that do not contain water during normal plant operating conditions.

For piping within systems or portions of systems other than residual heat removal, emergency core cooling and containment heat removal systems:

- Piping with a nominal pipe size of 4 inches and smaller for systems excluding auxiliary feed water systems in pressurised water reactor plants.
- Piping with a nominal pipe size of 1.5 inches and smaller within the auxiliary feed water systems in pressurised water reactor plants.
- Piping of any size in systems or portions of systems that operate (when system function is required) at a pressure equal to or less than 275 psig (1900 kPa) and at a temperature equal to or less than 200°F (95°C).
- Piping of any size beyond the last shutoff valve in open ended portions of system that do not contain water during normal plant operating conditions.

3.1.2 Scope of the RISMET project

The RISMET study compared four systems which are listed below.

1. 313, Reactor Coolant System
2. 321, Residual Heat Removal System
3. 411, Main Steam System
4. 414, Condensate System

For the PWROG (original) and PWROG-SE methodologies, the scope of application included all piping in the four systems in scope including the piping that would be exempt per ASME Section XI. The PWROG (original) and PWROG-SE methodologies utilise a relative ranking process. If the scope of application is changed, the relative risk ranking will be impacted which may or may not impact the final categorisation of the segments as high or low safety significant. To evaluate this impact of changing scope of application two additional evaluations were conducted. The results of RISMET scope of application were compared against the results from the PWROG-SE methodology full scope application conducted on Ringhals Unit 4. Additionally, a single system, reactor coolant system, scope of application was also conducted using the PWROG (original) methodology.

For the EPRI methodology, the scope of application was applied to the same four systems as the PWROG (original) and PWROG-SE applications. For risk ranking and element selection purposes, some ¾ inch lines were excluded due to assumed conservatism in the consequence analysis, the fact that the reliability of these locations are driven by vibratory fatigue, which is not amenable to periodic ISI and that the RI-ISI recommended inspection for these locations would be a VT-2 exam which is identical to that current ISI requirements and therefore would provide zero risk reduction potential. Other piping which can be exempted from the EPRI methodology was included in the scope of application. However, from an overall plant risk perspective, this piping is not a significant contributor to plant risk (i.e. contribution to CDF of less than 10^{-6} and LERF of less than 10^{-7}).

For Code Case N-716, the scope of application was applied to the four systems being analysed. Additionally, other Ringhals systems that are typically included in a Code Case N-716 application (e.g. CVCS, Safety Injection, Feed Water) were reviewed for consistency with past Code Case N-716 experiences. Based upon the Ringhals PSA [36], there was no piping identified (safety or non-safety related) that exceeded the CDF or LERF criteria of Code Case N-716 (> 10^{-6} to CDF and > 10^{-7} to LERF), Thus, per the criteria of Code Case N-716, all other piping not pre-defined by N716 as HSS is LSS.

For ASME Section XI, a pre-determined, fixed scope application, the scope was limited to just the Class 1 and 2 portions of the four systems being analysed. Towards the end of the study it was noted that how Class 2 piping is defined in Belgium is slightly different than how Class 2 piping is defined at Ringhals. The ASME Section XI analysis was conducted based on the ASME rules in Belgium. This resulted in one segment that is Class 4 at Ringhals being treated as Class 2 by the RISMET study and having examinations assigned to it for ASME Section XI methodology.

For the SKIFS methodology, a pre-determined, fixed scope application, the scope was limited to just its application to the four systems.

3.1.3 Impact of scope on the ISI results for various RI-ISI methodologies

Prior to presenting the results of this assessment, potential limitations are identified below. Three of these potential limitations are associated with steps done in the application analyses to streamline the project. These streamlined steps may or may not have affected the results.

1. To streamline the project for the EPRI and Code Case N-716 methodologies, it was assumed that the exam locations for these methodologies would be the same as the exam locations selected for the PWROG-SE methodology. This is because, during the element selection process, each methodology uses inputs such as access, worker exposure, severity of postulated degradation and inspection history in selecting the final set of exam locations. However, in practice the selection of the same exam locations may or may not have been true if the selection of the exam locations had been done independently of the results of the PWROG-SE methodology. In the PWROG-SE methodology, in general, exams are placed on each high safety significant segment. In the EPRI and Code Case N-716 methodologies, the segments within a given risk category, are grouped together and a certain percentage of locations are selected across all segments in that category, but there is no requirement that an exam be placed on each segment.

2. The PWROG (original) and PWROG-SE methodologies utilise an expert panel to determine the final categorisation of the segments as high or low safety significant. The expert panel could not be convened for the RISMET study. The categorisation process utilised by an expert panel is the same for full and partial scope applications. The only differences for the four systems between full and partial scope applications are associated with the RRWs. Therefore, the final categorisation of segments was estimated based on the full scope PWROG-SE methodology and discussions with Ringhals personnel where there were changes in RRWs that might impact the final categorisation. However, it is possible that more or less or different segments would have been categorised as high safety significant for these methodologies. Any differences are not be expected to be significant based on the following:

The categorisation was conducted by personnel who participated in numerous expert panel sessions including a Ringhals expert panel session.
Ground rules were established for the categorisation process based on the Ringhals 4 expert panel categorisation. Some of these ground rules included:

- Segments that are quantitatively HSS are assumed to be categorised as HSS unless the full scope panel categorised that quantitatively HSS segment as LSS.

- Segments that are quantitatively LSS are assumed to be categorised as LSS unless the full scope expert panel categorised the quantitatively LSS segment as HSS.

- Segments that were categorised as HSS by the full scope expert panel for deterministic reasons are assumed to be categorised as HSS.

- Similar to the full scope expert panel, segments that are not quantitatively HSS are assumed to be categorised as LSS unless there was a deterministic reason to categorise the segments as HSS.

The results of the assumed categorisation were reviewed by Ringhals personnel who participated in the Ringhals 4 full scope expert panel session.

3. An example of the potential effects of changing the scope of application for a relative ranking process, such as that used in the PWROG (original) and PWROG-SE methodology, is provided by comparing the PWROG-SE results for the full scope programme and the RISMET scope of application. However, late in the study, it was requested that a reactor coolant system (RCS) only application be added, since the potential differences might be greater for a smaller scope of application (i.e. a single system). The RCS only application was added, but due to limited time and resources, the number of exams per segment with butt welds was assumed to be one. This is based on experience where in the vast majority of cases (excluding structural elements with active degradation mechanisms and thin wall piping) the statistical analysis (Perdue model) demonstrated that a single examination was required.

4. In addition, it should be noted that because Code Case N-716, SKIFS and ASME Section XI are either a full scope application or a predefined scope, it is difficult to make a clear comparison between these approaches and the others with regards to the impact of the limited RISMET scope (i.e. the four systems).

To evaluate the impacts that the scope and methodology have on the results of a RI-ISI programme, segments identified for inspection in one methodology or scope of application and not in another methodology or scope of application were assessed. Additionally, an assessment of each methodology was conducted to determine if varying the scope of application could impact the RI-ISI results. Based on the results of the applications analyses, as conducted, the following observations were made.

1. The ASME Section XI methodology is limited to Class 1 and 2 piping; therefore, the condensate system is not within the predefined scope of ASME Section XI. The SKIFS methodology did not identify any examinations on the condensate system either due to the piping being outside the scope (identified as not having a consequence index of 1, 2 or 3) or because the piping fell into Region C, an owner defined programme. The other methodologies, Code Case N-716, EPRI, PWROG (original), and PWROG-SE, all used the Ringhals flow accelerated corrosion (FAC) programme results to identify locations susceptible to FAC on the condensate system. The Code Case N-716 and PWROG-SE methodologies identified only FAC examinations. The EPRI methodology identified

three additional examinations on the condensate system beyond the FAC examinations. These were risk category 5 location (medium risk) and further analysis (see conservatisms discussed in other areas of this report, such as section 4 and 5) may obviate the need for these locations (i.e. shown to be low risk). The PWROG (original) methodology identified 24 additional examinations on the condensate system beyond the FAC examinations. The additional examinations are to address the welds in Region 1B (i.e. the welds not impacted by the active degradation mechanism).

2. The EPRI methodology identified one main steam segment (R4-411-35A, steam supply to turbine driven pump) as high safety significant that would normally be exempt per ASME Section XI. Similar to above, this segment was identified as risk category 5 (medium risk) and if additional analysis was performed (e.g. thermal fatigue susceptibility) this segment as well may have been determined to be low risk. If the ASME Section XI exemptions had been used for the EPRI methodology, this segment would have been exempt from the analysis.

3. The EPRI methodology was specifically developed so that changing the scope of application would not impact the result of the application. That is, each of the systems evaluated as part of the RISMET would have the same results whether they were evaluated as a single system, or group of four systems or as part of a full scope application.

4. As the SKIFS, ASME Section XI and Code Case N-716 methodologies have pre-defined scopes, it is not possible to conduct partial scope application. Therefore, varying the scope of application is not pertinent for these methodologies.

5. The PWROG (original) and PWROG-SE methodologies utilise a relative ranking process. When the scope of application is decreased, the amount of the overall plant piping CDF and LERF being addressed by the RI-ISI is decreased. Relatively speaking, this causes the risk reduction worths to increase potentially making the results for a partial scope programme for a given system more conservative than the same system results in a larger scope programme. Thus, as shown in the RISMET study, the RRW may increase above the threshold for identifying segments as quantitatively HSS in a number of cases. This may change the final categorisation of segments from low to high safety significant. The size of the change (i.e. number of new HSS segments) will be a function of what other systems are included in the larger scope as well as the conservatisms and assumptions used in the other systems' analyses. However, the final categorisation of the segment may not be affected by the change in scope because the final categorisation of the segment's safety significance is based on the expert panel. The expert panels can categorise a segment as HSS that is not quantitatively HSS based on deterministic insights or because the segment RRWs are close to being quantitatively HSS. Therefore, although decreasing the scope of application will impact the RRWs, the final categorisation of the segments as high or low safety significant may or may not be impacted.

   For example, comparing the PWROG-SE methodology for a full scope application versus the RISMET partial scope identified the following differences. Twenty-one segments had their risk reduction worth change from quantitatively medium safety significant to quantitatively high safety significant (six segments in RCS, eleven segments in RHRS and four segments in main steam system). However, based on the estimated expert panel categorisation, only three additional segments would be categorised as high safety significant. Comparing the PWROG (original) methodology for the RISMET partial scope (four systems) against a partial scope application of just the reactor coolant system identified the following differences. Thirty-nine segments had their risk reduction worth change from quantitatively medium safety significant or low safety significant (three) to quantitatively high safety significant. Based on the estimated expert panel final categorisation of the segments as high or low safety significant there was an increase of 34 HSS segments which was estimated to result in an increase of 13 volumetric examinations and 21 visual examinations (total of 34 examinations).

3.1.4 Discussion on full scope versus partial scope applications

Full and partial scope applications are acceptable for RI-ISI programmes. A full scope application addresses more of the risk and thus provides a more thorough programme with respect to plant safety, since all systems in the plant are considered.

Although a partial scope application may not address as much risk as a full scope application, a partial scope application is still acceptable based on:

- The application of RI-ISI methods could maintain and/or improve safety even within the partial scope application. In some cases there could be an acceptably small risk increase, and

- The safety level of the plant associated to the systems not included within the partial scope application remains the same.

Thus, the overall plant safety is maintained or improved.

3.2 Impact of the definition of segment boundaries

3.2.1 Summary of segmentation for RI-ISI methodologies

For the PWROG (original) methodology and PWROG-SE methodology, the piping is divided up into segments. Subsequent steps in the methodologies are performed at the segment level. Note that at the structural element / non-destructive examination selection step, the PWROG (original) and PWROG-SE methodologies diverge slightly. For both methodologies, several considerations are taken into account when defining segments but at a high level, segmentation of the piping is based on two primary considerations.

- Piping which has the same consequences of failure (i.e. a failure at any point in the segment results in the same consequence).

- Piping which has similar failure probabilities.

For the EPRI methodology, the segments are defined as portions of piping with similar consequences and susceptible to the same degradation mechanisms. Thus, segments are used in the process as a tool to streamline the documentation of the analysis rather than a technical component of the methodology.

For the Code Case N-716 methodology, portions of piping (segments) are placed into one of two bins. Thus segmentation is not really used in the Code Case N-716 methodology.

For the SKIFS methodology, there is no segmentation of piping. Individual welds are placed into the various bins based on their consequence and likelihood of failure.

For ASME Section XI, there is no segmentation of piping. Individual welds are placed into various bins based on the piping Class and type of weld.

3.2.2 Differences in segmentation

Three of the methodologies, PWROG (original), PWROG–SE and EPRI methodologies utilise segmentation. The other three methodologies, Code Case N-716, ASME Section XI and SKIFS do not utilise segmentation.

In the PWROG (original) methodology, the segments are used as the means to divide the piping for analysis and upon which the following steps are based. Each segment is evaluated individually for its potential consequences, potential degradation mechanisms, failure probability, piping CDF and LERF contribution, risk metrics, expert panel categorisation and NDE selection (i.e. the analysis is conducted on a segment basis and each high safety significant segment has at least one examination).

In the PWROG-SE methodology, the segmentation is utilised in the same manner as the PWROG (original) methodology except for NDE selection. The welds from the HSS segments are placed into one of three bins and certain percentages are selected. For segments with an active degradation mechanism, additional analyses are performed to determine if the welds not affected by an active degradation mechanism should be inspected. In general, each HSS segment has at least one examination. However, segments with only a few welds that have the same degradation mechanisms (i.e. SRRA inputs) may be combined into a single lot for determination of the number of welds to examine.

In the EPRI methodology, the piping is divided into sections of piping, segments based on the potential consequences and failure modes. These segments are assigned to a risk category based on the potential consequences and potential degradation mechanisms (i.e. sections of piping with similar consequences or degradation mechanisms are placed in the same bin). While the analysis is conducted on a segment basis, it is for ease of use rather than being a technical component of the analyses. The selection of individual inspection elements depends on risk ranking, the plant-specific service history, the degradation mechanism potentially present, physical access constraints, radiation exposure, and cost considerations. Other considerations that go into the element selection process are inspectability, distribution of inspections among systems and segments, and plant specific inspection results, repairs or remedial measures which have been implemented.

In the Code Case N-716, ASME Section XI and SKIFS methodologies, segmentation is not utilised. For Code Case N-716 and ASME Section XI, individual welds are placed into one of two bins. One bin is for welds that are considered for inspection and the other bin is for welds not considered for inspection. For SKIFS, individual welds are placed into one of three bins. Two of the bins are for welds considered for inspection while the third bin is for welds not considered for inspection as part of the SKIFS programme but that could be part of an owner defined programme.

3.2.3 Potential effect of differences in segmentation

In the PWROG (original) methodology, it is theoretically possible to impact the results of the analysis based on the segmentation. Decreasing the number of segments could increase the RRWs and the final categorisation of segments as high or low safety significant in a manner similar to the scope of application. Additionally changing the number of segments could impact the number of examinations in that a minimum of one examination per segment is required. A study was conducted outside of the RISMET project analysing the potential effects of differences in segmentation. The effects of having multiple pipe size segments versus single size segments (i.e. affecting the number of segments) were compared on five PWROG (original) RI-ISI programmes (one full scope and four Class 1 and 2 only scopes). The results of the study identified that there was no impact on the number of examinations. The results of this study are documented in WCAP-14572 Supplement 2 Revision 1-NP-A. Inspection of structural elements subjected to an active degradation mechanism are not impacted by the number of segments, since 100% of the high safety significant structural elements affected by an active degradation mechanism are inspected in the PWROG (original) methodology.

In the PWROG-SE methodology, it is theoretically possible to impact the results of the analysis based on segmentation in manner similar to the PWROG (original) methodology. Similar to the PWROG (original) methodology, the impact is expected to be negligible, if any. The potential impact is further reduced through the possible combining of smaller segments with the same SRRA inputs into a single lot for NDE selection.

In the EPRI methodology the number of segments has no impact on the number of examinations. For the other methodologies (Code Case N-716, SKIFS and ASME Section XI) there is no segmentation.

For the PWROG (original) and PWROG-SE methodologies, examinations are placed on each segment for the degradation mechanism of concern. In the EPRI, Code Case N-716, SKIFS and ASME Section XI, a certain percentage of the elements in a given bin is selected for inspection. Based on this, distribution of the inspections is an additional consideration for the elements selected for inspection. Each of these methodologies has certain criteria for selecting exam locations. It is theoretically possible that a segment within a given bin will not receive an inspection. For the EPRI, Code Case N-716, PWROG (original) and PWROG-SE methodologies, inspections are conducted to address every type of degradation as well as those locations with no degradation but a high consequence of failure in the systems with risk significant segments as determined by that methodology.

3.3 Conclusions

The SKIFS and ASME Section XI offer a pre-determined, fixed scope of application. The Code Case N-716 methodology offers only a single scope of application, full scope. The EPRI and PWROG (original) methodologies offer a range of scopes from small to full scope applications. The PWROG-SE methodology scope of application is the same as the PWROG (original) methodology except that only full scope applications have been performed. Selection of the systems to include in the scope of application is similar for EPRI, PWROG (original), and PWROG-SE methodologies. The SKIFS, ASME Section XI, and Code Case N-716 methodologies all have different methods for determining the scope of application.

The EPRI, SKIFS and ASME Section XI methodologies allow piping exemptions to varying degrees. The PWROG (original) and PWROG-SE methodologies currently do not allow piping exemptions. Depending on the scope of the application, implementation of RI-ISI methodologies may show that exempted piping may need to be inspected.

Changing the scope of application may impact the results of the RI-ISI programmes for the PWROG (original) and PWROG-SE methodologies since a relative ranking system is utilised. For these methodologies, decreasing the scope of application may increase the number of inspections in the remaining systems since the RRWs would increase. Based on the RISMET study, there is a greater difference between the single system scope and a four system scope than there is between a four system scope and a full scope application. The differences between the RISMET scope (four systems) and a full scope application were small because the increase in the RRWs did not cross the threshold for being quantitatively HSS. Additionally, segments whose RRWs crossed the threshold for being quantitatively HSS had been previously categorised by the expert panel as HSS. This is due in part to the four risk evaluation cases (CDF without operator action, CDF with operator action, LERF without operator action, and LERF with operator action) that are used to determine the quantitative safety significance of the segments. Another reason is that both the RISMET four system scope and the full scope applications included the condensate system (high failure potential due to FAC susceptibility) and the RHR system (high conditional core damage frequencies), which are two of the more but not most significant risk contributors. As expected, the RCS only application resulted in more conservative results (i.e. more inspections) due to lower total plant piping CDF and LERF being based on a single system. Based on a full scope application being acceptable, the relative ranking process is acceptable since decreasing the scope will provide more conservative results for segments in scope (i.e. higher RRWs).

Changing the scope of application in the EPRI methodology does not impact the results within any given system or in the total programme, because an absolute ranking process is used. That is, evaluating one or multiple systems in a RI-ISI application, at the same time or over a period of time, will produce consistent results for each individual system.

The process of segmentation (i.e. segment definition and segment boundaries) in the PWROG (original) and PWROG-SE methodologies can affect the programme results. Based on studies that were done outside of RISMET, the process of segmentation had no impact on the number of examinations.

The process of segmentation between the PWROG (original) and EPRI methodologies could theoretically result in differences in the structural element selection process. However, as discussed in section 3.1.3, this potential difference was neither confirmed nor disproved.

Full and partial scope applications are acceptable for RI-ISI programmes.

4. ANALYSIS OF FAILURE PROBABILITIES

4.1 Introduction

This chapter documents observations, insights and recommendations relevant to pipe failure probability analysis. Presentation material from the five RISMET project meetings, including work results produced by each Application Group, as well as relevant published technical reports on RI-ISI methodologies provided the input to the evaluation. The objective of this evaluation of the role of pipe failure probability analysis in RI-ISI programme development included consideration of the following technical issues and associated questions:

1. Use of Probabilistic Fracture Mechanics (PFM) and Structural Reliability Models (SRM) with specific requirements on computer codes (e.g. validation and verification), and including capability to account for different types of degradation mechanisms. What type of statistics do the SRM models generate?

2. Use of statistical models of pipe failure, including the role and use of service experience data. What are the quality requirements to be imposed on service experience data? Is it technically feasible to derive absolute measures of piping reliability based on service experience data?

3. Relationship (or interface) between service experience data and PFM/SRM. Specific issues addressed include the ranges of applicability of the two approaches. For example, how well does PFM/SRM predict failures in carbon steel piping subject to general corrosion, erosion-corrosion, or flow accelerated corrosion? How well do statistical models extrapolate stress corrosion cracking data to major structural failure?

4. Use of expert judgment/expert elicitation. Requirements on expert panels, including consistency issues in the case of RI-ISI programme updates. Requirements on the practical use of computer codes for calculating probability of pipe failure (POF).

5. Treatment of uncertainties;

6. Definition and treatment of different structural failure modes (as defined by peak through-wall flow rates);

7. Probability of flaw detection, POD, and inspection intervals. How does the respective methodology account for the effects of these factors on an inspection scope?

8. Reliability of leak detection and sensitivity of results to leak detection limits. Treatment of different leak inspection strategies (e.g. frequency of visual inspection and walkdown inspection);

9. Treatment of different degradation mitigation strategies (material, control of water chemistry);

10. Updating of original failure probability analyses given new service experience, or implementation of piping design changes or new mitigation strategies;

11. Compatibility of pipe failure probability analysis approach with PSA requirements (reverse-engineering). Can the pipe failure probability analysis results of RI-ISI directly support such PSA tasks as loss-of-coolant-accident (LOCA) frequency assessment, or internal flooding initiating event frequency assessment?

12. Importance of POF in RI-ISI programme definition.

RISMET perspectives on these technical issues are summarised in Section 4.5 of this chapter. The RI-ISI methodologies applied in RISMET by each Application Group have been subjected to extensive regulatory reviews, and in the USA regulatory approval has been obtained for nine plant-specific applications of Code Case N-716 and a much larger number of applications for the EPRI and PWROG approaches. The scope and depth of regulatory reviews have differed significantly from country to country, however. This also means that there are differences of opinion relative to certain technical issues. Published regulatory review insights and results are not repeated herein. The work by Evaluation Group “Failure Probability Analysis” was performed in two phases:

- Review of results as supplied by the RISMET Application Groups. In addition to the results supplied in the form of electronic data, the Evaluation Group "Failure Probability Analysis" had access to each Application Group for further clarifications as needed.

- Review of RI-ISI methodologies not addressed in the RISMET project. The basis for this review is the set of relevant documents and reports listed in the References section of this report, as well as the knowledge and insights of the Evaluation Group “Failure Probability Analysis” team members. The insights and results from this task are documented in a separate (unpublished) report.

4.2 Pipe failure probability analysis by the RISMET Application Groups

This section reviews the different approaches to pipe failure probability analysis by each RISMET Application Group. In a general sense, pipe failure probability analysis provides quantitative support to RI-ISI programme development. The extent of quantitative consideration varies extensively across available RI-ISI methodologies. The reader of this report is referred to Topical Reports and other relevant public domain literature for detailed descriptions of the underlying theories, techniques, assumptions, user guidance, and analysis tools for pipe failure probability analysis.

4.2.1 Scope of the evaluation

The scope of the evaluation of the various applications is limited to a review of source documents, and of presentation material directly related to the scope of work for each Application Group. In the case of PWROG-SE, the SRRA input and output files for the four systems in the work scope (Reactor Coolant System, Residual Heat Removal System, Main Steam System, and Condensate System) were made available by Ringhals NPP. These SRRA output files were converted to Microsoft Excel format to facilitate further data processing by the Evaluation Group members.

4.2.2 Convention

In this chapter the term “pipe failure probability analysis” is used in lieu of probabilistic piping reliability analysis using statistical models, SRM codes, or expert judgment with input from statistical analysis and/or SRM codes. The results of pipe failure probability analysis support the quantitative assessment of ΔRisk. That is, the evaluation of changes in core damage frequency (CDF) and large early release frequency (LERF) due to proposed changes in an inspection programme. An inspection programme “change” may involve adding inspection sites, eliminating inspection sites, or redistributing inspection sites from one system to another.

The pipe failure parameter of interest in RI-ISI applications is the frequency of pipe failure per "inspection site" and year. The output from some of the analysis tools that are used to support RI-ISI programme development is in the form of "cumulative failure probability at the end of an operating license," which can be 40 or 60 years. In order to facilitate ΔRisk evaluations, the cumulative failure probability is converted to a pipe failure frequency. In this report the notion of "pipe failure probability" refers to the likelihood of structural failure of certain magnitude, where "magnitude" refers to the leak rate or flow rate produced by a through-wall flaw.

4.2.3 SKIFS 1994:1 methodology

In Sweden consideration of pipe failure potential, cumulative usage factor and degradation mechanisms is required for ISI programme development. Regulation SKIFS 1994:1 documents the overall technical approach and implementation guidelines, which apply to all known degradation mechanisms. It is a qualitative approach and includes steps that address pipe failure potential through the assignment of Damage Indices (DI) and Consequence Indices (CI). Some R&D has been directed towards consideration of quantitative assessments into the pipe failure potential analysis and consequence analysis. The qualitative SKIFS 1994:1 has been in place since 1988. The SKIFS 1994:1 "DI-results" for the four systems in the scope were made available to the Evaluation Groups.

4.2.4 Code Case N-716 methodology

The methodology behind Code Case N-716 (Alternative Piping Classification and Examination Requirements, ASME Section XI, Division 1) is documented in [12]. Instead of performing calculations of pipe failure probability, the following bounding pipe failure frequencies may be used in the pipe failure potential estimation (Table 7):

- The failure frequency of $2 \cdot 10^{-6}$ per weld-year for welds in the high failure potential category.
- The failure frequency of $2 \cdot 10^{-7}$ per weld-year for welds in the medium failure potential category.
- The failure frequency of 10^{-8} per weld-year for welds in the low failure potential category.

It is noted that [12] does not provide a technical basis for the above bounding pipe failure frequencies, but rather they are founded on work performed during the development of the EPRI traditional RI-ISI methodology (e.g. ASME Code Cases N560, N578) in [24,25,26].

The objective of pipe failure potential estimation is to differentiate among the piping segments on the basis of the potential degradation mechanisms and the potential consequences of degraded conditions. Code Case N-716 does not include justifications for the given bounding pipe failure frequencies. However, the N-716 methodology builds on the EPRI-Base methodology (see below). According to Section 7 of Code Case N-716, a re-evaluation of RI-ISI examination selections needs to be performed given the plant-specific or industry occurrences of piping failures due to a new degradation mechanism, or a non-postulated mechanism.

Table 7 Code Case N-716 degradation mechanism categories

Failure Potential	Conditions	Degradation Category	Degradation Mechanism
High ([a])	Degradation mechanism likely to cause a large break	Large Break	FAC
Medium	Degradation mechanism likely to cause a small leak	Small Leak	Corrosion, SCC, Erosion-Cavitation, TF
Low	None	None	None

([a]) Segments having degradation mechanism listed in the "small leak" category shall be upgraded to the high failure potential "large break" category if the pipe segment also have the potential for water hammer loads.

4.2.5 EPRI-Base methodology

The EPRI-Base approach (also referred to as the "EPRI traditional" approach) does not require POF calculations to be performed. It is a qualitative approach to RI-ISI programme development, which builds on PSA insights and results. It also builds on the SKIFS 1994:1 methodology, and as explained in the EPRI Topical Report, Section 3.4 [11], the pipe failure potential evaluation step uses insights from service experience reviews (both plant-specific data and industry wide data) together with a detailed degradation mechanism analysis. Table 8 compares the "Damage Index" of SKIFS 1994:1 with the "pipe failure potential" of the EPRI traditional approach.

According to the Topical Report, the pipe failure potential evaluation is a cost-effective alternative to structural reliability analysis. In the EPRI approach, the degradation mechanisms in a pipe segment are identified by comparing actual piping design and operating conditions to a well-defined set of material and environmental attributes. Bounding pipe failure frequencies (1/Reactor-Year) are used in lieu of structural reliability analysis. The proposed bounding pipe failure frequencies are included in

Table 9, and a technical basis is documented in EPRI TR-110157 [37]. It is noted that these failure frequencies are not intended to be applied in a strict manner and there is documented recognition of the inherent uncertainty in the quantitative estimates.

Table 8 EPRI "Pipe Failure Potential" versus SKIFS "Damage Index"

Damage/Degradation Mechanism Criteria	
EPRI **Pipe Failure Potential**	**SKIFS 1994:1** **Damage Index (DI)**
3 - High FAC, Vibration Fatigue, Water Hammer	**DI-1** • Thermal sleeves susceptible to crevice corrosion • FAC • Vibration fatigue • Thermal fatigue of mixing tees with ΔT ≅ 100 °C • Welds incl. weld-HAZ susceptible to IGSCC, PWSCC • For a specified pipe segment, data on observed pipe failures is available (plant-specific or fleet-specific)
2 - Medium Corrosion (MIC, Pitting), Erosion-Cavitation, SCC, Thermal Fatigue	**DI-2** Service conditions (pressure, temperature, flow) that are such that a structural failure (e.g., large through-wall flow rate given a pipe flaw) is considered unlikely (e.g., moderate-energy piping systems)
1 - Low No Damage / Degradation Mechanism	**DI-3** No Damage / Degradation Mechanism

Table 9 Potential for pipe rupture according to EPRI-Base

Potential for Pipe Failure Per Damage/Degradation Mechanism	**Consequence Category (CCDP / CLERP)**			
	"None" CCDP < 10^{-8} CLERP < 10^{-9}	**Low** 10^{-8} < CCDP ≤ 10^{-6} 10^{-9} < CLERP ≤ 10^{-7}	**Medium** 10^{-6} < CCDP ≤ 10^{-4} 10^{-7} < CLERP ≤ 10^{-5}	**High** CCDP > 10^{-4} CLERP > 10^{-5}
High FAC, Vibration Fatigue, Water Hammer	**LOW** Category 7	**MEDIUM** Category 5	**HIGH** Category 3	**HIGH** Category 1
Medium Corrosion (MIC, Pitting), Erosion-Cavitation, SCC, Thermal Fatigue	**LOW** Category 7	**LOW** Category 6	**MEDIUM** Category 5	**HIGH** Category 2
Low No Damage / Degradation Mechanism	**LOW** Category 7	**LOW** Category 7	**LOW** Category 6	**MEDIUM** Category 4

4.2.6 PWROG methodology

For the pipe failure potential step, the PWROG methodology (ASME Code Case N-577) is based on industry service experience with safety-related piping systems and the Structural Reliability and Risk Assessment (SRRA) computer code to determine the failure probabilities of piping segments [15]. The SRRA code has been peer reviewed by Battelle-Columbus [15], and independent audit calculations have been performed by the Pacific Northwest National Laboratory for the USNRC [15]. The NRC safety evaluation report concluded that the probabilities calculated using the SRRA methodology is consistent with the industry failure experience for piping.

For the PWROG (original) methodology for piping risk-informed ISI, Reference [14,16], development of the SRRA input involves assigning all postulated degradation mechanisms present in a pipe segment to a single weld, and imposing the operating characteristics and environment to that weld. The failure probability developed for a single weld is subsequently used to represent the failure probability of the segment for the risk ranking, regardless of the number of welds in the segment, or the length of the segment. The initial flaw size and its uncertainty are calculated for typical welds using results from PRODIGAL [38]. The stress intensity factors for a semi-elliptical crack on the inside surface in a uniform stress field and the methods for calculating the effects of ISI (i.e. probability of detection) are the same as those in the pc-PRAISE code [39]. Monte Carlo simulation with importance sampling is used to calculate failure probabilities. According to the SRRA documentation [15], the following types of damage or degradation mechanisms can be modelled:

- Flow accelerated corrosion (FAC)
- Low-cycle fatigue

- Stress corrosion cracking
- Thermal fatigue
- Vibration fatigue

The SRRA code includes detailed consideration of different sources of uncertainty, and a basis for chosen uncertainty distributions is included in Supplement 1 of WCAP-14572 [15]. To develop the median values of the input parameters to the SRRA tool, which calculates the leak and break probabilities, a plant engineering team is trained by Westinghouse for several days and provides a guidance document, which devotes almost 30 full pages to developing the appropriate SRRA input. The expertise required by the team addresses all aspects of the potential degradation mechanisms that could result in piping failure due to a leak or break. This includes awareness of plant and industry piping failure experience because the calculated small leak probabilities must be compared with this experience to show that they are reasonable before the input can be accepted by the plant engineering team. This point is specifically emphasised in one of several hands-on training exercises. The SRRA results used for risk ranking and classification is further reviewed by an expert panel for the plant piping risk-informed ISI programme. Finally, the PWROG has a working group on piping risk-informed ISI that meets several times a year to discuss the effects of new industry experience and other emergent issues and their potential impact on RI-ISI including the PWROG input guidance for calculating failure probabilities.

In late 2000, another PWROG Working Group on Large Break LOCA Redefinition met with the NRC about changing their requirements in this area. The technical bases for this request included the SRRA calculated frequencies for all piping sizes that contribute to a LBLOCA (>5000 gallons per minute) in seven representative plants that had performed PWROG piping risk-informed ISI programmes. As a result, an expert panel was convened by NRC to develop LOCA frequencies for 6 sizes of breaks. This panel included PFM expertise as well as expertise fluent in the application of statistical models of piping reliability. After a number of independent reviews, including that by the US Advisory Committee on Reactor Safeguards, the final results of the expert elicitation are included in the final version of the USNRC Report NUREG-1829 [40]. Both the appropriate best-estimate median value and mean value in this final report are nearly identical to the corresponding average values for the seven PWR plants that performed PWROG piping risk-informed ISI programmes. These frequencies are based upon the piping large leak (>5000 gallons per minute) probabilities calculated by the seven plant engineering teams using the SRRA computational tool.

4.2.7 PWROG-SE application

The PWROG-SE application utilises the SRRA code for calculating pipe failure probabilities, but it is specialised to reflect the plant-specific service conditions as well as the scope of the application. Similar to the PWROG (original) methodology it includes consideration of flow accelerated corrosion, vibration fatigue and water hammer loads. Application-specific user guidelines have been developed for the Ringhals NPP RI-ISI project. These user guidelines have not been reviewed by the Evaluation Group.

4.3 Some selected results

A number of pipe segments were selected for a review of the estimated pipe failure probabilities. These segments are reported in Table 10.

Table 10 Selected pipe segments in Ringhals-4 RI-ISI application

Pipe Segment	Segment Description	Relevant Service Experience
R4-313-001-29"	Loop 1 Hot Leg piping (29") from RPV RCPCRV -01 to SG RCPCSG-01	No relevant service experience data available because this weld is not susceptible to PWSCC (not Alloy 82/182) and Ringhals study showed the cast stainless steel components were not susceptible to thermal ageing
R4-321-007-10"	8" and 10" line from discharge side of pump RHAPRH-01 to inlet side of heat exchanger RHAHRS-01 and AOV FCV-605A (train A)	No relevant service experience data available
R4-321-013-10"	10" line from AOV HCV-603A to check valve 8740A (train A); includes a tee fitting	Limited thermal fatigue data available for tees (including mixing tees). Reference [41] includes a summary of service experience
R4-411-019_2"	2" line from tee to reducers (2" to 3/4") – includes socket welds	Database includes 11 socket weld failures in Main Steam piping – all events have resulted in forced plant shutdown
R4-411-035B_8"	8" and 10" line from outlet side of pump AFAPST-01 to atmosphere	No relevant service experience data available
R4-411-035B_10"	8" and 10" line from outlet side of pump AFAPST-01 to atmosphere	No relevant service experience data available
R4-411-073_3/4"	Three 1/2" and 3/4" lines from main line to valve 1193 and FICA:s 474 and 475	OPDE Database includes 11 socket weld failures in Main Steam piping – all events have resulted in forced plant shutdown
R4-414-035B-10"	T41: A250 (10") line from SMOVs V141 and V142 to reducers (10" to 14")	FAC-susceptibility, where ample service experience data is available, is very dependent on local flow conditions
R4-414-037A-20"	T41: A500 line from tank 418 T103 to tee	High FAC-susceptibility, ample service experience data available
R4-414-037B-14"	T41: A350 (14") line from valve V131 to reducer (14" to 8")	High FAC-susceptibility, ample service experience data available
R4-414-041-8"	T41: A200 (8") line between two tandem pumps	High FAC-susceptibility, ample service experience data available
R4-414-045-10"	T41: A250 (10") line from discharge side of pump P105 to check valve V139	High FAC-susceptibility, ample service experience data available

Examples of estimated pipe failure probabilities (with and without consideration of ISI) for respective system and pipe segments are displayed in Figures 2 through 5.

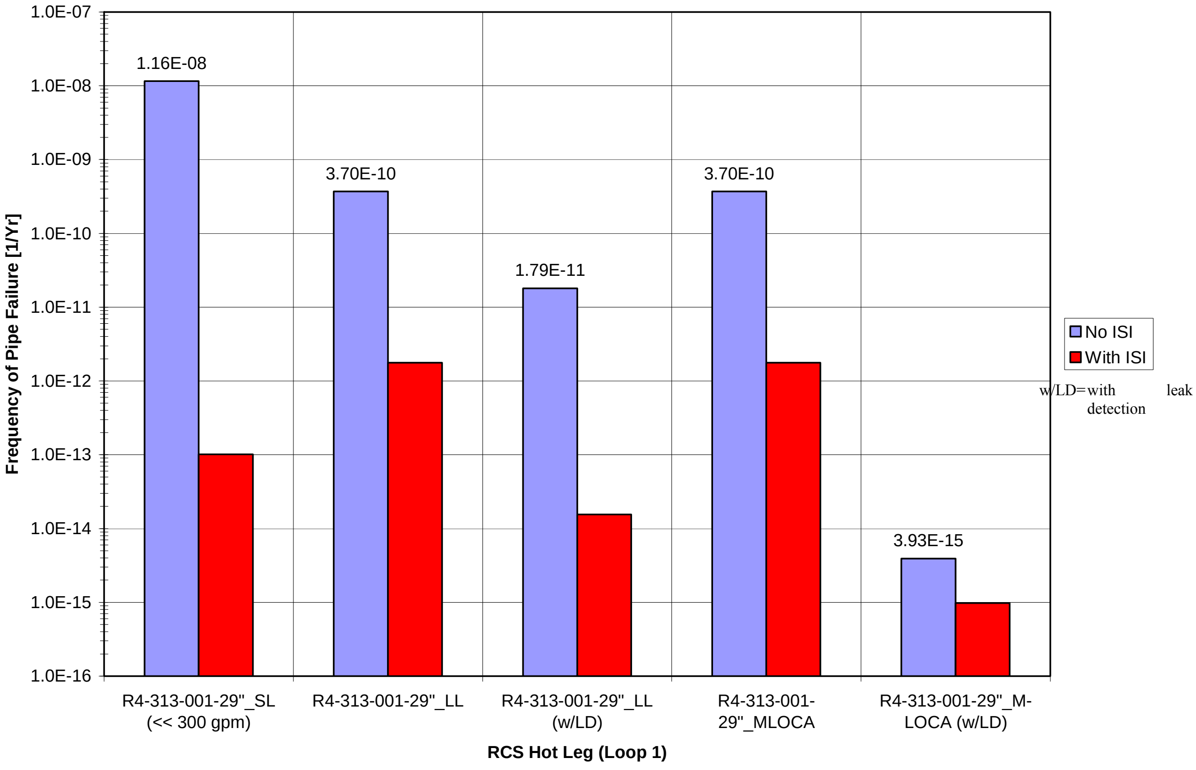

Figure 2 Selected results for reactor coolant system (System 313)

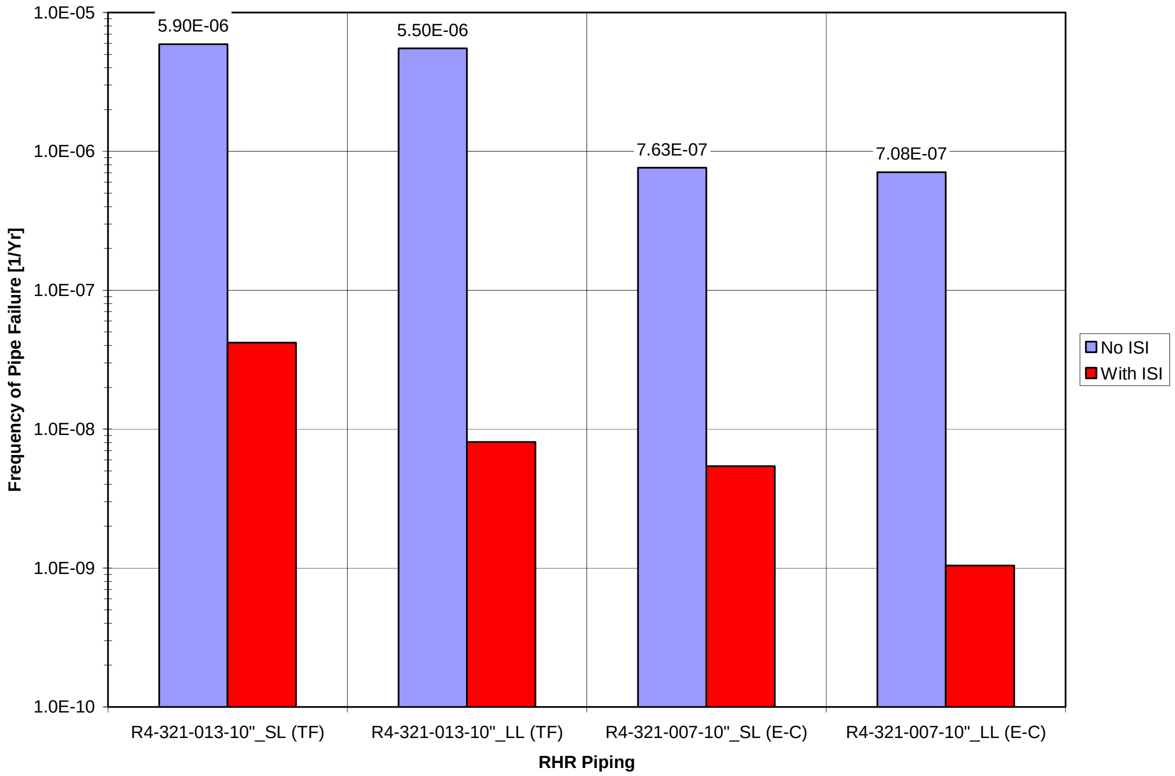

Figure 3 Selected results for Residual Heat Removal System (System 321)

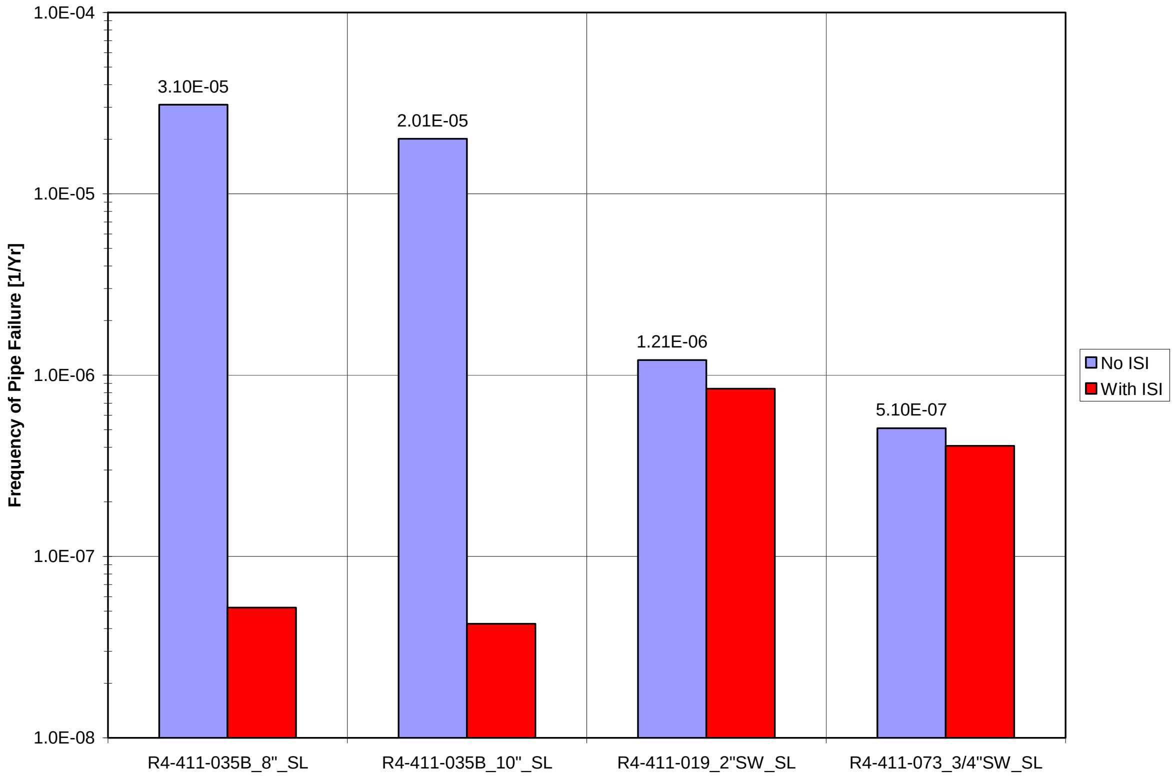

Figure 4 Selected results for Main Steam System (System 411)

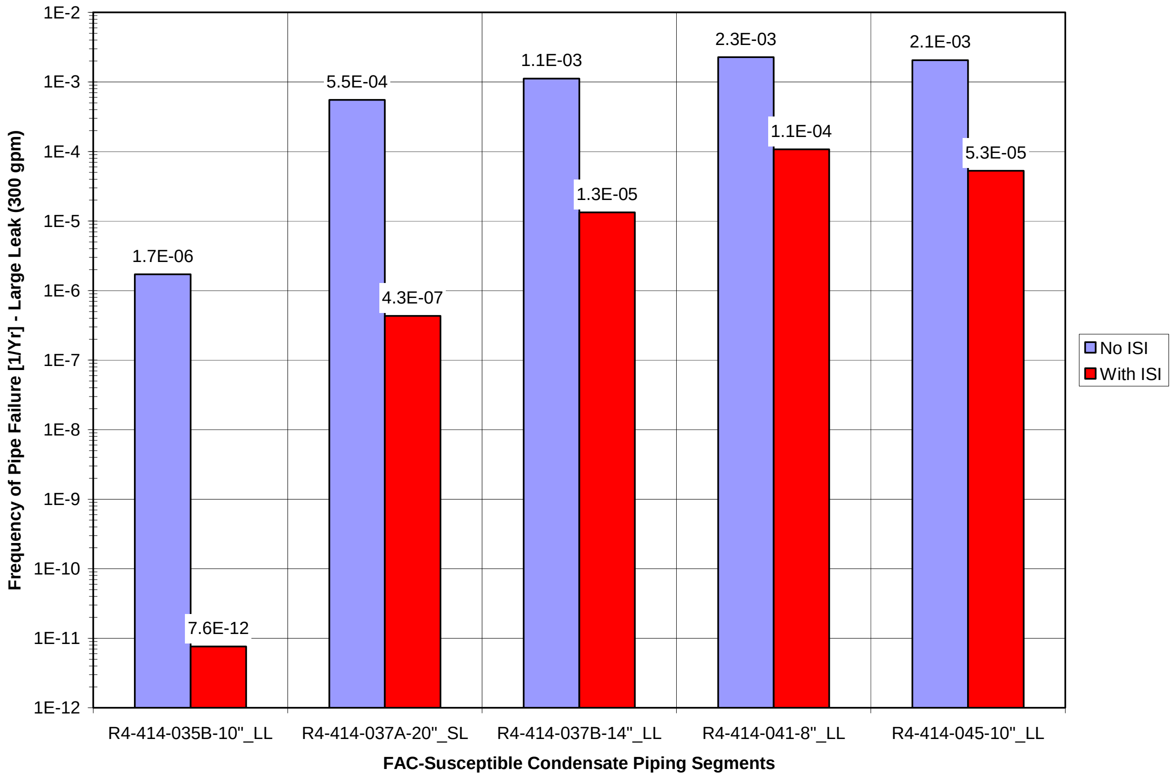

Figure 5 Selected results for Condensate System (System 414)

4.3.1 Observations

Results of the quantitative pipe failure probability analysis were available from PWROG-SE only. No other independent calculations were performed by any of the Application Groups. The following general observations apply:

- Except for socket welds, the influence of ISI on the calculated failure probability and failure frequency seems unrealistically strong. As shown in Figures 2 through 5, the impact of ISI on the calculated pipe failure probability is a reduction by several orders of magnitude relative to the no inspection cases. The assumptions about inspection effectiveness should be verified together with the procedure for calculating the risk reduction by ISI in the SRRA code. The Swedish regulatory review of PWROG-SE [17] includes detailed comments on the treatment of the influence of ISI on the pipe failure probability. The RISMET Evaluation Group concurs with the Swedish regulatory opinion. The assumptions about the reliability of non-destructive examination techniques appear to be unrealistic. However, it should be noted that Ringhals studied plant piping inspection capabilities and provided plant specific guidance for what the SRRA engineering team should use for inspection accuracy for different piping geometries and configurations. Also it should be noted that the effects of ISI are not used for segment risk ranking and classification. They are used for calculating the change of risk along with the effects of leak detection, if applicable. In many cases, the result of including these competing benefits is to significantly reduce the calculated benefit of the PWROG risk-informed programme relative to the existing ISI programme, which is very conservative. Most risk-informed evaluations are based upon more realistic calculations.

- For the FAC-susceptible pipe segments in the Condensate System the calculated pipe failure frequencies point to very high (assumed or observed) pipe wall wear rates. This raises questions about the service experience specific to the PWR units at Ringhals or about the assumption of not treating FAC as an augmented programme at Ringhals. RI-ISI may not be the appropriate structural integrity management approach for this system. FAC is a degradation mechanism for which a large volume of service experience data exists. It is recommended that results produced through structural reliability analysis are validated against empirical data. Current work on the development of PFM tools that address FAC is documented in Reference [42], and a recent application of a statistical model is documented in Reference [43]. Extensive insights are available from statistical analysis of service experience data on FAC. These insights should be considered when developing RI-ISI programmes for high-energy Turbine Building piping (e.g. condensate, extraction-steam, feed water, and moisture separator reheater piping).
- It is noted that an Owner-defined programme for FAC-inspections has been in place at Ringhals for some time. Reasons for including FAC-susceptible piping systems in the RI-ISI programme development project included verifying the reasonableness of the original inspections scope. That is, confirming that HSS segments are adequately addressed in the plant-specific inspection programme. Also, if the FAC examinations are not impacted by the RI-ISI programme, it is more reasonable to treat the FAC examinations as an augmented programme in the PWROG (original) and PWROG-SE methodologies. This will result in the ISI failure probabilities being used for the segment with FAC and may result in an increased number of examinations on other systems due to the relative ranking process used in the PWROG (original) and PWROG-SE methodologies.

4.4 Evaluation summary

The pipe failure probability analysis element of each RI-ISI methodology is summarised in Table 11. Additional evaluation insights are summarised via the plots in the two plots below. Figure 6 includes SRRA results and shows how the EPRI-Base application classified the pipe failure potential of Residual Heat Removal system pipe segments. For the majority of the segments the EPRI-Base application is consistent with the guidance in the EPRI Topical Report. Figure 7 compares the SRRA results with EPRI-Base application to the Condensate System (414). The following outliers are noted:

- In Figure 6, Segment 321-036 is classified as having “low” failure potential according to the EPRI-Base RISMET application. This particular segment (3/4” socket welded line) is susceptible to vibration fatigue and with a calculated failure frequency of 2.5×10^{-5} per year, which would correspond to “high” failure potential according to the EPRI Topical Report. It is noted that according to the EPRI Topical Report, vibration fatigue failure is not amenable to prevention via a periodic NDE inspection programme. As such, guidance is provided in developing other more effective plant response (e.g. design and operational changes).
- In Figure 6 segment 321-001 (12” line), which is determined to be susceptible to low cycle fatigue, is assigned “medium” failure potential. This is consistent with other EPRI-Base applications. However, according to the SRRA calculations for this segment, the failure frequency is approximately 3.0×10^{-8} per year, which is outside the upper bound range given by the EPRI Topical Report.
- Some outliers are found for the Condensate System (Figure 7). Three segments are assigned “low” failure potential although these segments have been determined to be susceptible to vibration fatigue and water hammer loads. All FAC-susceptible segments are assigned “high” failure potential.

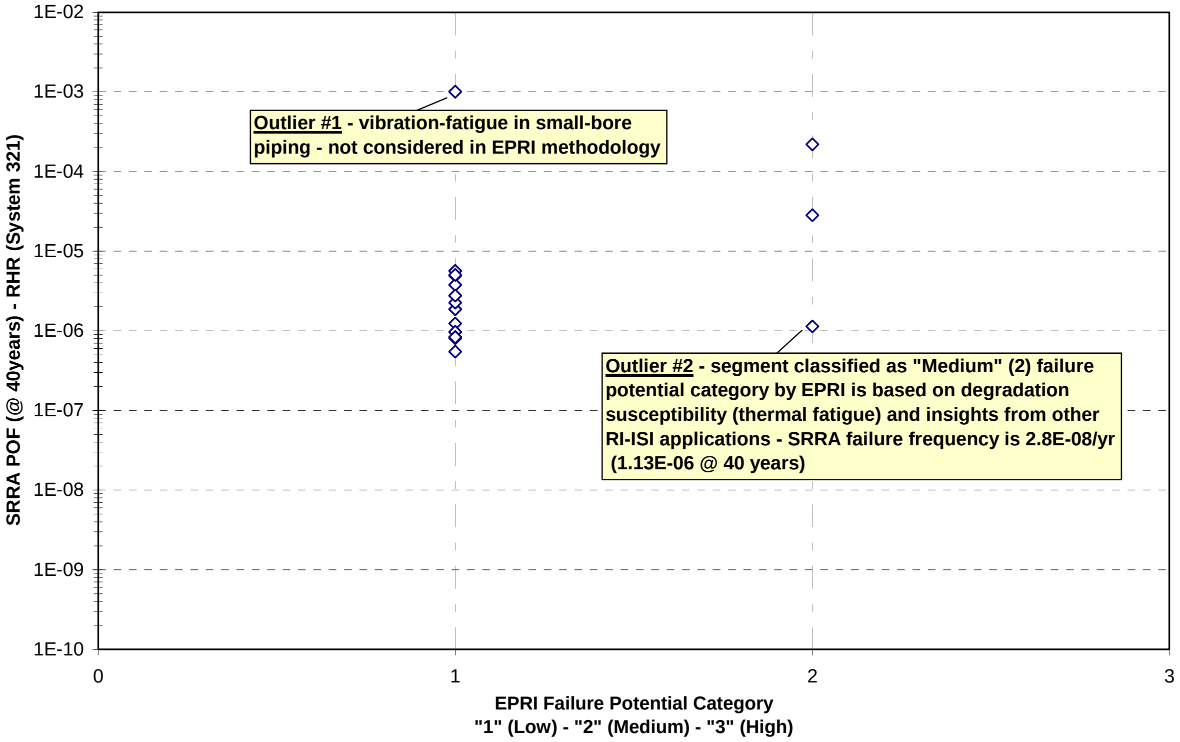

Figure 6 SRRA results for System 321 versus EPRI-Base failure potential categories

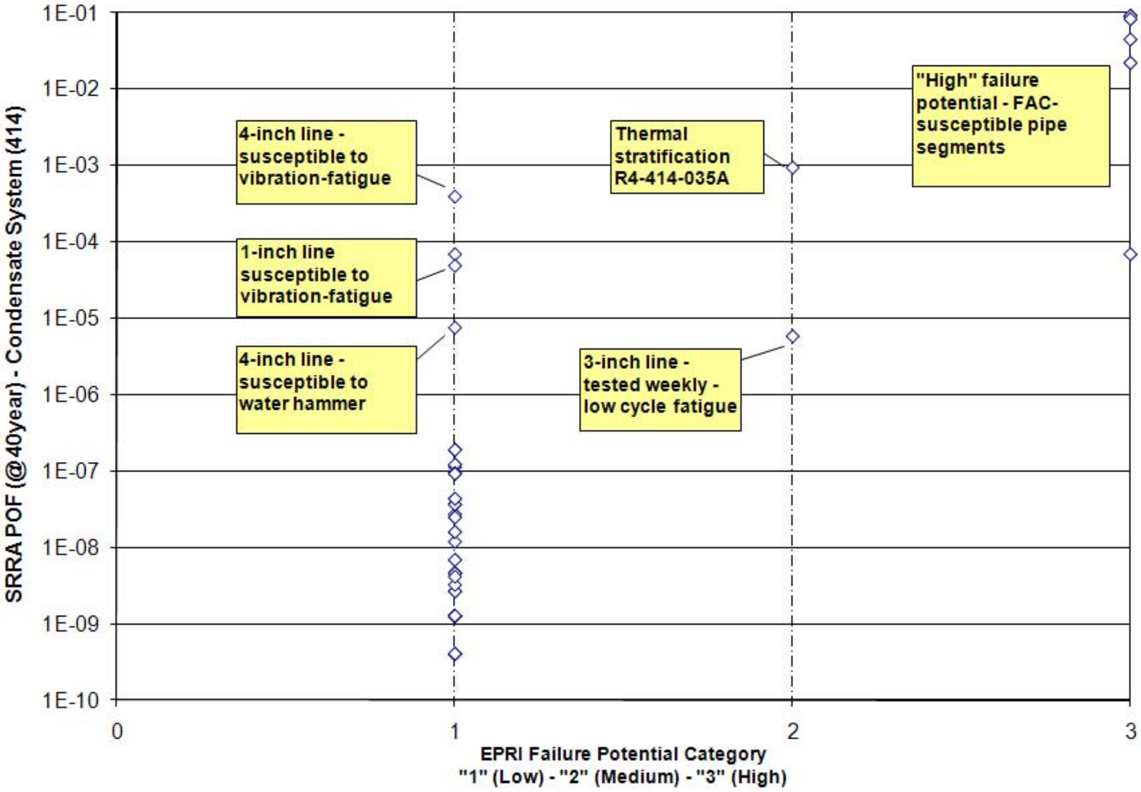

Figure 7 SRRA results for System 414 versus EPRI-Base failure potential categories

Table 11 Pipe failure data & pipe failure probability considerations by respective application group

Attribute	RISMET Application Groups			
	SKIFS 1994:1	**Code Case N-716**	**EPRI^{BASE}**	**PWROG**
Explicit consideration of industry experience	The degradation susceptibility is determined by comparing actual piping design & operating conditions to a well defined set of material & environmental attributes. Service experience considerations are an integral part of the DI assignment (see below)	The degradation susceptibility is determined by comparing actual piping design & operating conditions to a well defined set of material & environmental attributes. Service experience considerations are an integral part of the DMA task	The degradation susceptibility is determined by comparing actual piping design & operating conditions to a well defined set of material & environmental attributes. Service experience considerations are an integral part of the DMA task	Yes, the industry failure experience is explicitly considered and evaluated in developing the input to the SRRA tool in accordance with the requirements of the PWROG Guidance Document
Explicit consideration of plant-specific operating experience	Yes, the plant-specific service experience is collated and evaluated as part of DI assignment (see below)	Yes, the plant-specific service experience is collated and evaluated as part of DMA task	Yes, the plant-specific service experience is collated and evaluated as part of DMA task	Yes, plant-specific operating experience is explicitly considered and evaluated in developing the input to the SRRA tool in accordance with the PWROG Guidance Document
Quantitative assessment of pipe failure probability	No – the SKIFS procedure requires the assignment of a Damage Index (DI) (in lieu of pipe failure potential analysis; see EPRI^{BASE}) to each inspection site. SKIFS 1994:1 includes general guidance for DI assignment.	The failure frequencies of $2{\cdot}10^{-6}$ per weld-year for welds in the high failure potential category, $2{\cdot}10^{-7}$ per weld-year for welds in the medium failure potential category, and 10^{-8} per weld-year in the low failure potential category may be used as bounding failure frequencies.	Bounding estimates of pipe failure frequency. For RISMET the evaluation was limited to a review of PWROG-SE SRRA output and identification of any differences relative to EPRI methodology, and other applications to 3-loop Westinghouse PWR plants.	Yes, SRRA code – no new calculations were performed for RISMET

Attribute	RISMET Application Groups			
	SKIFS 1994:1	Code Case N-716	EPRI^{BASE}	PWROG
Assumption(s) about probability of detection (POD)	No explicit consideration of POD	Large collection of data resides at the EPRI NDE Center as well as engineering judgment; POD = 0.5 is used to represent the "with inspection" case [44]. Sensitivity studies are performed varying the improvement in inspection effectiveness based on work conducted at the EPRI NDE centre.	Large collection of data resides at the EPRI NDE Center as well as engineering judgment; POD = 0.5 is used to represent the "with inspection" case [44]. Sensitivity studies are performed varying the improvement in inspection effectiveness based on work conducted at the EPRI NDE centre.	The following equation defines the assumed POD-relation for UT-inspections: $POD(a) = (1-\varepsilon)\times[1-0.5\,erfc(\nu\times\ln(a/a^*))]$ Where a is crack depth (or wall thinning) and a^* is the value of "a" for which POD = 0.5 and erfc is the complementary error function. See [17] for further details.

(continues next page)

Table 11 (cont.) Pipe failure data & pipe failure probability considerations by respective application group

Attribute	RISMET Application Groups			
	SKIFS 1994:1	**Code Case N-716**	**EPRI^BASE**	**PWROG**
Structural failure mode(s) modelled	None explicitly modelled. However, the internal flooding requirement of Code Case N-716 requires that a spectrum of break sizes (e.g. small leak to large rupture) be evaluated.	None explicitly modelled. However, the consequence assessment portion of the methodology requires that a spectrum of break sizes (e.g. small leak to large rupture) be evaluated.	None explicitly modelled. However, the consequence assessment portion of the methodology requires that a spectrum of break sizes (e.g. small leak to large rupture) be evaluated.	• Small Leak (through-wall flaw with "minor" leakage"). • Large Leak (Through-wall flaw that leads to leakage beyond a user-defined value). • Full Break (complete severance of the pipe cross section).
Verification & validation (V&V) of methodology	Inspection programme subject to regulatory approval.	Pilot applications performed as part of Code Case development and USNRC Relief Request process.	SER in Topical Report [10].	SER in SRRA Topical Report [15] includes references to V&V.
Exceptions		Implementation requires that structural integrity management programmes be in place or developed for IGSCC, FAC and vibration fatigue. Criteria are provided within the methodology for what constitutes an acceptable programme.	Implementation requires that structural integrity management programmes be in place or developed for IGSCC, FAC and vibration fatigue. Criteria are provided within the methodology for what constitutes an acceptable programme.	None.

4.5 Summary of insights

This evaluation considered four RI-ISI methodologies. Quantitative structural reliability analysis is not a pre-requisite for RI-ISI programme development. However, a successful implementation of any methodology is strongly dependent on an in-depth knowledge of structural integrity management and piping system degradation susceptibilities. The consideration of structural integrity (qualitatively or quantitatively) is one of several steps needed to develop a RI-ISI programme, and it is acknowledged iteratively throughout a programme development process. With some exceptions, a comparison of the EPRI and PWROG methodologies points to similar POF ranges. The PWROG methodology appears to be more resource intensive than any of the other RI-ISI methodologies that are included in the RISMET scope of work. Respective topical report and implementation guidelines address the role and importance of service experience data in ensuring realistic results and as input to future RI-ISI programme updates. Within the scope of the PWROG-SE application the explicit roles of the plant-specific and industry wide service experience data could not be assessed, however.

In reviewing the results of the four applications the twelve technical issues concerning pipe failure probability analysis were considered by the members of the Evaluation Group on Pipe Failure Probability. Summarised below are the insights and recommendations resulting from the evaluation:

1. Use of PFM/SRM and specific requirements on codes (e.g. validation and verification), and including capability to account for different types of degradation mechanisms. What type of statistics do the SRM models generate?

   RISMET Perspective: As already stated, application of PFM/SRM codes is not a pre-requisite for successful RI-ISI programme development. The strengths and limitations of the currently available PFM/SRM codes are well documented in the NURBIM reports. First-and-foremost, the currently available structural reliability models are parametric models that support a broad range of sensitivity studies. The predictive power of these models is highly correlated with the underlying assumptions about flaw initiation, flaw growth (propagation), etc.

2. Use of statistical models of pipe failure, including the role and use of service experience data. What are the quality requirements to be imposed on service experience data? Is it technically feasible to derive absolute measures of piping reliability based on service experience data?

   RISMET Perspective: All four RI-ISI methodologies in the RISMET project included any explicit consideration of plant-specific or industry service experience data. However, it is noted that considerable progress has been made in developing statistical models of piping reliability that directly use service experience data as primary input. A large number of practical applications have been performed in support of risk-informed PSA applications. For degradation mechanisms such as vibration fatigue and flow accelerated corrosion the body of service experience data is extensive. Therefore, predictive piping reliability analysis is not only highly feasible but also recommended using statistical analysis tools. For a statistical analysis to be sufficiently robust it is necessary to have access to a validated and sufficiently complete pipe failure event data base that also includes data on cause and effect relationships for all remedial or mitigation actions that have been performed on the piping of interest.

3. Relationship (or interface) between service experience data and PFM/SRM. Specific issues addressed include the ranges of applicability of the two approaches. For example, how well does PFM/SRM predict failures in carbon steel piping subject to general corrosion, erosion-corrosion, or flow accelerated corrosion? How well do statistical models extrapolate stress corrosion cracking data to major structural failure?

   RISMET Perspective: As documented in the NURBIM reports, PFM/SRM is evolving. The results obtained are highly correlated with underlying assumptions as well as the knowledge and experience of analysts performing the calculations. It is recommended that analysts involved in

PFM/SRM applications familiarise themselves with the published reports on benchmarking of computer codes against service experience data, as it is for instance done in the training for the PWROG methodology.

4. Use of expert judgment/expert elicitation. Requirements on expert panels, including consistency issues in the case of RI-ISI programme updates. Requirements on the practical use of computer codes for calculating probability of pipe failure (POF).

   RISMET Perspective: Expert judgment (in various forms) is an inherent aspect of pipe failure probability analysis. Applications of the PWROG (original) methodology involve the application of PWROG user guidelines for the SRRA code. The user guidelines for the PWROG-SE application were not included in the evaluation scope, however. These plant-specific guidelines were developed because Ringhals personnel wanted to document any differences relative to the general PWROG guidance so that they were treating the SRRA input data in a consistent manner for all piping system segments.

5. Treatment of uncertainties; this subtask addresses the type of statistics that is generated by respective technical approach.

   RISMET Perspective: Only the PWROG (original) and PWROG-SE applications include explicit consideration of uncertainties. Detailed background is documented in WCAP-14572, Revision 1-NP-A. Depending on type of degradation mechanisms and loading conditions, the uncertainties in estimated pipe failure probabilities can be substantial and must be properly accounted for in the RI-ISI programme development process. In contrast to the quantitative uncertainty analysis conducted by the PWROG (original) and PWROG-SE approaches, the EPRI approach addresses uncertainty as part of the risk categorisation task. Three risk regions are used to account for uncertainties in the risk categorisation, and ensure that 1) high consequence segments are considered for all likelihoods of failure, and 2) segments with the potential for large leaks (high likelihood of failure) are considered for all consequence categories (except "none"). The consequence assessments are conducting postulating a range of break sizes from leaks up to and including full pipe rupture, irrespective of failure potential. The limiting case with respect to conditional core damage probability is used for the purpose of characterising the risk ranking of the pipe segment.

6. Treatment of different structural failure modes (as defined by peak through-wall flow rates); this subtask addresses failure modes definitions and how they are addressed in respective methodology.

   RISMET Perspective: Different structural failure modes are accounted for in the Code Case N-716, EPRI and PWROG applications. Although not specifically addressed by RISMET, for plants with a well developed internal flooding PSA model, the quantitative assessment of pipe failure probability needs to reflect the unique conditions that are addressed by the internal flooding scenarios. Where SRM tools are utilised it is essential to verify compatibility between the output and internal flooding initiating event frequencies.

7. Probability of flaw detection, POD, and inspection intervals. How does respective methodology account for the effects of these factors on an inspection scope?

   RISMET Perspective: The PWROG (original) and PWROG-SE applications account for POD and presence leak detection. As indicated elsewhere, the assumptions about the probability of flaw detection in the PWROG methodology appear unrealistic to the RISMET team that did not review the technical bases for the Ringhals plant-specific guidance on inspection accuracy

8. Reliability of leak detection and sensitivity of results to leak detection limits. Treatment of different leak inspection strategies (e.g. frequency of visual inspection and walkdown inspection).

RISMET Perspective: See #7. Leak detection is not used by any of the RI-ISI methodologies for determining the risk ranking of the segments. Leak detection is considered in the PWROG (original) and PWROG-SE methodologies only in the change in risk evaluation.

9. Treatment of different degradation mitigation strategies (material, water chemistry).

   RISMET Perspective: All RI-ISI methodologies include some consideration of different mitigation strategies. The ability of SRM models to explicitly account for mitigation (e.g. weld overlay, hydrogen water chemistry, stress relief) is addressed in the open literature but was not addressed further by the Evaluation Group.

10. Updating of an original failure probability analysis given new service experience, or implementation of piping design changes or new mitigation strategies.

    RISMET Perspective: All four RI-ISI methodologies include consideration of service experience data. The extent by which service experience data is directly applied to pipe failure probability analysis varies, however. It was outside the scope of RISMET to investigate the practical aspects of how to update an existing set of pipe failure probabilities given presence of new (generic or plant-specific) information about degradation susceptibilities. Furthermore, it is recognised that the practical implications (e.g. extent of analytical resources needed) are tied to specific aspects of a selected RI-ISI methodology (including absolute versus relative risk ranking). The ability of a quantitative approach to account for new service experience will determine whether an update of existing analysis is feasible. Reference [45] provides the results of updating eight different RI-ISI programmes.

11. Compatibility of a pipe failure probability analysis approach with PSA requirements (reverse-engineering). Can the pipe failure probability analysis results of RI-ISI directly support such PSA tasks as loss-of-coolant-accident (LOCA) frequency assessment, or internal flooding initiating event frequency assessment?

    RISMET Perspective: The “compatibility issue” is outside the RISMET scope of work. Potential for synergies between RI-ISI and PSA is related to technical issues #1, #2 and #3. Additionally, there are a number of significant flood sources beyond piping (e.g. tanks, vessels).

12. Importance of quantitative pipe failure probability analysis in RI-ISI programme development.

    RISMET Perspective: It is not a pre-requisite of RI-ISI programme development to perform quantitative assessments of pipe failure probability. As explained elsewhere in this report, many different factors determine what specific technical approach is selected for RI-ISI programme development.

5. ANALYSIS OF CONSEQUENCES

5.1 Ringhals PSA model

The Ringhals 4 project RIVAL is the basis for the RISMET study. Ringhals 4 PSA model includes a level 1 and level 2 evaluations for LOCA, transients, common cause initiators, internal flooding, steam releases, piping failures in CVCS, piping failure in auxiliary feed water system during hot standby, small LOCA during hot stand by and overpressure of RHRS.

5.2 Analysis of consequences by Application Groups

This section gives an overview of how consequences of failures are treated in the different methodologies evaluated in the RISMET project.

The work has been performed without detailed review of the Ringhals 4 PSA model. The information has been collected by discussions with persons involved in performing consequence analysis for the Ringhals plant and also for other plants. Certain selected documents have been studied delivered from the specialists of the methodologies.

The defence in depth levels are used in this chapter based on the IAEA INSAG 10 definitions as in Table 12.

Table 12 Levels of defence in depth

Levels of defence in depth	Objective	Essential means to fulfil the objective
Level 1	Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation
Level 2	Control of abnormal operation and detection of failures	Control, limiting and protection systems and other surveillance features (work within the technical specification limits)
Level 3	Control of accidents within the design basis	Engineered safety features (safety systems) and accident procedures
Level 4	Control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents	Complementary measures and systems to avoid further release of radioactivity and accident management
Level 5	Mitigation of radiological consequences of significant releases of radioactive materials	Off-site emergency responses

5.2.1 ASME Section XI

The ASME Section XI criteria are developed based on the safety level and quality classification of components and systems in the plant. ASME also requires specific programmes for inspection of ECC and containment systems. Defence in depth levels 1 and 2 are therefore included in the ASME programme classification.

The safety classification specifies – based on deterministic design requirements – different levels of importance for the different systems and components in the plant.

In principle, the following is valid for the safety classification:

> <u>Safety Class 1</u>: All piping forming the reactor coolant pressure boundary (RCPB) up to the second isolation valve
> <u>Safety Class 2</u>: Minor piping connected to the RCPB and important safety systems needed for emergency core cooling (ECC) and decay heat removal
> <u>Safety Class 3</u>: Service water systems and control systems needed for supporting the front line safety systems
> <u>Non-safety classified</u>: Systems outside containment for which leakages can be isolated from the reactor vessel.

There is no method in ASME to specify different risks for small or large breaks. Small and sometimes even medium sized breaks represent lower risk and larger safety margins compared with effects of large breaks.

The ASME methodology specifies the amount of inspections that shall be performed in each safety class. Even if an inspection in other safety class are more important than in Safety Class 1 the ASME methodology specifies no mechanism in reducing the amount of inspection in one safety class by increasing the inspections in other classes.

ASME does not take credit for safety system overcapacity or system interactions and the support from non-safety functions/systems to avoid core damage. This means that the inspection programmes are not chosen based on existing plant safety margins.

The ASME approach is based on the assumption that piping integrity is the last barrier (there is no credit for defence in depth level 3 or 4). Even though the piping integrity of the ECC systems and the containment are included in the scope the capacity and functions availability of the ECC system and the containment are not accounted for in the ASME methodology.

5.2.2 PWROG methodology

The consequence evaluation in PWROG consists of two parts, definition of consequences associated with piping failures and calculation of the CCDF/CCDP and CLERF/CLERP for the defined consequences using the plant PSA model.

In the definition of consequences associated with piping failures, both direct consequences and indirect consequences are considered. The direct consequences are the effects on the plant if the fluid medium (water or steam) in the pipe not reaching its intended destination. The indirect effects are the effects on the plant from the released medium (water or steam) following a piping failure which can give flooding, spray, jet impingement, pipe whip, high temperature and high humidity.

To ensure that the direct consequences are properly identified, two questions are asked :

- What are the consequences, if the piping failure occurs while the plant is operating normally (i.e. does the failure cause an initiating event)?

- What are the consequences, if the piping failure occurs as a secondary failure in a transient/accident?

Therefore, the direct and indirect consequences to be considered include:

- Failures causing initiating events such as LOCA or reactor trip (initiator).

- Mitigating system failures such as disabling a single train or system or multiple trains or systems (hidden failures or fails as a consequence of the initiating event).

- Failures that cause both the initiating event and failure of the mitigating system.

When defining direct mitigating system failures, two primary criteria are used: flow diversion and loss of inventory.

"*Flow diversion*" occurs when a certain percentage of flow is diverted from its intended flow path. Typically the flow diversion criterion used in the PSA model is used for the PWROG RI-ISI methodology. Typically this is a 1/3 pipe diameter. If a branch line fails that is 1/3 the diameter (or greater) of the main line, then the function of the main line is assumed to fail.

"*Loss of inventory"* occurs in a closed loop system. A piping failure may not cause sufficient diversion of flow to fail a train to perform its main function for a period of the sequence but over time, the water level in the closed loop system can lower to the point where there is pump cavitations or loss of net positive suction head and thus a loss of all flow in the system. Based on this, the affect of small bore piping failure after long time in a sequence can have the same effect as larger bore piping and give severe consequences.

In determining the consequences, especially in calculating the loss of inventory, a 24 hour mission time is used to be consistent with the PSA model. The actual mission time for a given function may vary based on the initiating event. Likewise the exact size piping failure is a range from a small leak to a full break with potentially different consequences. Thus to keep the process manageable, the potential direct consequences are typically based on the largest break size that could occur on the pipe and a mission time of 24 hours is used. Exceptions to basing the consequence on the largest break size include piping failures that can result in different size LOCAs, jet impingement or spray.

To determine the potential indirect effects existing documentation are reviewed and a plant walk through is conducted. The plant walk through are performed to evaluate potential flooding, spray, pipe whips and jet impingement. Typically a walk through inside containment is not necessary due to existing analyses and documentation.

Effects of piping failures are evaluated both with and without operator actions. "Without operator action" consequences assume that the operators take no action to isolate or mitigate the specific piping failure. "With operator action" consequences assume that the operators are perfect in taking the appropriate actions to isolate the leak (no human error probabilities are used). By evaluating consequences both with and without operator action, the range of potential effects of operator actions is bounded. The probability of a success for the operator action is not specified.

The following restrictions are specified to be able to credit operator action:

- There is an alarm or clear indication in the control room, to which the operator will respond.

- The operator response would be expected.

- The method for identifying the expected location of the piping failure is available such that a train or system can be recovered. Included in this is that there is sufficient time for the operators to identify, diagnose, and take the corrective actions.

- The isolation equipment is not affect by the failure.

- The action can be performed within the control room. As part of the expert panel process, the expert panel can credit operator actions that are taken outside the control room.

Core damage frequencies (CDF) and large early release frequencies (LERF) are evaluated with PSA, based on the prejudged effect of pipe failure in the different systems/segments over the specified mission time (24 hour). With a few exceptions (LOCAs, main steam piping failure etc.), piping failures are not normally included in the PSA model. Thus, a surrogate component, or a group of components, is (are) defined such that its (theirs) failure(s) would simulate the postulated consequences of the system/segment's failure. For failures causing initiating events such as LOCA or reactor trip (initiator) the PSA model generates a CCDP and a CLERP. For mitigating system failures such as disabling a single train or system or multiple trains or systems the PSA-model generates a CCDF and a CLERF.

The PSA results (CCDF/CCDP and CLERF/CLERP) along with the failure probabilities and test intervals are used as inputs to the risk evaluation for the quantitative risk ranking and expert panel categorisation of the segments. Refer to Section 6.

The PWROG methodology is able to addresses defence in depth at levels 1, 2 and, 3 and also to take into account the support from non-safety functions and diversified safety systems. In evaluating conditional large early release probability (CLERP) also defence in depth level 4 is addressed.

5.2.3 EPRI methodology

The purpose of the consequence evaluation phase of the EPRI RI-ISI methodology is to evaluate pipe failures in terms of their impact on Core Damage Frequency (CDF) and Large Early Release Frequency (LERF). The consequence evaluation focuses on the impact of a pipe section failure (loss of pressure boundary integrity) on plant operation. This impact can be direct, indirect or a combination of both:

- Direct Impacts - A failure results in a diversion of flow and a loss of the train and/or system or an initiating event (such as a LOCA).
- Indirect Impacts - A failure results in a flood, spray, or pipe whip, spatially affecting neighbouring structures, systems and components or results in depletion of a tank and loss of the systems supplied by the tank.

The consequences due to indirect effects and direct effects are treated explicitly.

Spatial effects are an example of indirect effects caused by pressure boundary failures. These include the effects of flood, spray, and pipe whip on equipment located in the vicinity of the break. Spatial consequences of the break are determined based on the location of the analysed break and the relative position of important equipment. The presence of important equipment in a specific location can be identified through existing analyses (e.g. internal flood analysis or fire analysis) and should be confirmed by a walkdown.

The possibility of isolating a break is also identified and accounted for as part of the consequence analysis. A break could be isolated by a protective check valve, a closed isolation valve, or it could be automatically isolated by an isolation valve that closes on a given signal. If not automatically isolated, a break can be isolated by an operator action, given successful diagnosis. Depending upon the scenario and other factors, operator action can be taken within or outside the control room. The likelihood of success of these actions depends on the availability of isolation equipment, a means of detecting the break, the amount of time available to prevent specific consequences (e.g. flooding of the room or draining of the tank), and human performance. If isolation is possible, the consequence assessment should be conducted for both cases: successful and unsuccessful isolation.

For each run of piping under evaluation, a spectrum of break sizes is evaluated. The break size ranges from a small leak to a rupture. Larger leaks and breaks have the potential to disable system or trains and to cause initiating events, flooding, or diversions of water sources. Typically, small breaks (minor leakage) would not render a train inoperable. They may, however, depending on the energy level of the system, spray onto adjacent equipment and cause equipment malfunction.

Pilot plant evaluations have shown that the large break scenarios (worst-case breaks) result in the most limiting consequences. However, the methodology was specifically developed to require that a spectrum of break sizes be evaluated so that, if smaller breaks can cause a measurable or the dominant consequence, they are identified and input into the risk ranking process.

The goal of the consequence evaluation is to establish a process that consistently ranks consequences caused by a pipe failure, based on its risk impact or safety significance. For example, in order to rank piping failures consistently, one needs to address the question of whether a pipe break that results in a loss of coolant accident (LOCA) is more safety significant than a pipe break that leads to a loss of feed water? Similarly, is a pipe break that disables one train of high pressure injection more safety significant than a pipe break that disables an auxiliary feed water train? In order to answer these questions consistently, consequences are categorised into different importance categories.

The consequences are ranked into those categories based on a combination of plant-specific PSA insights and results, and methodology lookup tables. The methodology lookup tables were developed, in order to standardise and streamline the consequence ranking process.

Four consequence importance categories have been defined based upon PSA evaluation. They are: high, medium, low, and none. The high category represents events with a significant impact on plant safety, while the low category represents events with a minor impact on plant safety. The none category defines those locations that have no impact on plant safety and are typified by “abandoned in place” piping.

These categories are defined by a range of Conditional Core Damage Probability (CCDP) or Conditional Large Early Release Probability (CLERP), associated with the impact of specific Pressure Boundary Failure (PBF). The ranges used to numerically define each category are shown in Table 13.

The CCDP and CLERP ranges are determined based on the estimates of the total risk associated with the piping failure. Risk is measured by Core Damage Frequency (CDF) or Large Early Release Frequency (LERF) as:

CDF [for a PBF] = [PBF frequency] x [CCDP]

LERF [for a PBF] = [PBF frequency] x [LERF]

Based on the above expression, and using a conservative estimate of the total PBF frequency for the plant (estimated in the order of 10^{-2} per year), CCDP and CLERP ranges are selected to guarantee that all pipe locations ranked in the low consequence category do not have a potential CDF impact higher than 10^{-8} per year or a potential LERF impact higher than 10^{-9} per year. The boundaries between the high and medium consequence categories, at CCDP and CLERP values of 10^{-4} and 10^{-5} respectively, are set to correspond with the definitions of small CDF and LERF values of 10^{-6} and 10^{-7} per year. The assumption that 10^{-6} and 10^{-7} represent suitably small CDF and LERF values is consistent with the decision criteria for acceptable changes in CDF and LERF found in RG 1.174. The medium category is selected to cover the area between high and low categories, and to address uncertainties in the CCDP and CLERP estimates.

Table 13 EPRI consequence categories

CCDP	CLERP	EPRI classification
CCDP > 10^{-4}	CLERP > 10^{-5}	High
10^{-6} < CCDP ≤ 10^{-4}	10^{-7} <CLERP≤ 10^{-5}	Medium
CCDP ≤ 10^{-6}	CLERP ≤ 10^{-7}	Low

The EPRI method addresses defence in depth at level 1, 2, and 3 and also the support from non-safety functions and diversified safety systems. In evaluating CLERP also defence in depth level 4 is addressed.

In conducting the RISMET applications, it is was noted that for two systems in particular (RHR and MS) results for the host plant application were different than what was expected based upon previous application of the EPRI methodology to a number of PWR plants. The results entitled "EPRI Base" represent assumptions made by the EPRI application team consistent with past experiences in implementing the EPRI RI-ISI methodology. While results entitled "EPRI R4" represent direct use by the EPRI application team of inputs provided by the host plant. Key inputs included crediting, or not crediting, operator action to isolate RHR (e.g. are existing plant procedures, timing, training and contingency plans sufficient?) and success criteria (e.g. alternative sources of inventory make-up to the condensate storage allowing longer time for secondary heat removal). For example, a number of plants have implemented risk management strategies to manage shutdown and plant refuelling activities. While initially developed to address shutdown evolutions, these plants changes (e.g. procedural changes, contingency plans and equipment availability) provide additional defence in depth measures for operational configurations as well as off-normal operation.

5.2.4 SKIFS methodology

The SKIFS methodology has a pre-determined, fixed scope of application that is based on a consequence index. If piping meets the criteria for a consequence index of 1, 2 or 3, the piping is considered to be included in the scope of application.

A consequence index of 1 should be assigned to reactor vessel parts and devices with a nominal diameter greater than 150 mm in the main feed circulation system inlet lines, from the reactor pressure vessel to the second containment isolation value that is automatically closed due to a pipe break. An index of 1 is also assigned to steam generator primary and secondary parts.

A consequence index of 2 should be assigned to devices with a nominal pipe diameter from 20 mm to 150 mm in the main feed circulation system, from the reactor pressure vessel to the second containment isolation valve that is automatically closed due to a pipe break. Corresponding exhaust lines and cross connection parts should be assigned a consequence index of 2 if the nominal diameter is greater than 70 mm. A consequence index of 2 should also be assigned to devices with nominal diameter greater than 150 mm in the main feed circulation system inlet lines after the second containment isolation valve that is automatically closed due to a pipe break and for internal parts in reactor pressure vessel that is important to cool the core, shutdown reactivity and maintaining of the core geometry.

A consequence index of 3 should be assign devices with nominal diameter of 20 mm to 150 mm in main feed circulation system inlet lines after the second containment isolation valve that is automatically closed due to a pipe break. Corresponding exhaust lines and cross connections parts should be assigned a consequence index of 3 if the nominal diameter is greater than 70 mm. A consequence index of 3 should also be assigned to other devices with a nominal diameter greater than or equal to 50 mm in parts of the system that are pressurised with reactor water or is a part of the containment integrity.

The SKIFS consequence index does not follow the safety classification boundaries. It is based on which demands a pipe break would put on the mitigation system to be able to avoid core damage. It is only partly dependent of the capacity of the existing safety functions.

SKIFS reflects different demands on mitigation systems in long and short term. On the other hand a plant with overcapacity for certain mitigation systems will not change the index.

The SKIFS method addresses defence in depth at level 1, 2 and to some extent level 3 according to the INSAG definition, but not the support from non-safety functions/systems and diversified safety systems to avoid core damage. The index does not include effects of the defence in depth level 4.

5.2.5 Code Case N-716 methodology

In the development of Code Case N-716 it was specified that it would be independent of consequence assessment (except for aspects indicated below in point 2).

Code Case N-716 has been developed to streamline the risk ranking process by using a pre-determined set of important locations and supplementing that with a plant-specific assessment of pressure boundary failure by utilising the plant's internal flooding study directly.

The scope of the RI-ISI-programme can only be a full scope covering all scenarios. No partial scope is accepted.

The programme includes the following:

1. High safety significant (HSS) welds:
   Portions of class 1 welds, shut down cooling flow path from the RPV up to the second isolation valve
   Portions of class 2 feed water system from steam generator to the to the outer containment isolation valve
   Piping within the "break exclusion zone"
   Other class 2, 3 and non-safety classified welds, irrespective of line size, if the core damage frequency is greater than 10^{-6} ([5])

2. Low safety significant (LSS) welds
   All other Class 1, 2, 3 or non-classed welds not classified as HSS per the above.

The Code Case N-716 methodology addresses the defence in depth at level 1, 2 and functions at level 3. In the LSS group there could be piping system that are from the non safety functions and diversified safety systems.

[5] Based on lessons learnt from USNRC approval at nine units, a revision of Code Case N-716 is currently underway which will include a provision that any piping segment whose pressure boundary failure contribution to LERF is greater than 10^{-7} based upon a plant specific PSA is deemed to consist of HSS welds.

5.2.6 Summary

Table 14 summarises the characteristics of the various applied methodologies.

Table 14 Summary of features of evaluation of consequence of failure in the RI-ISI methodologies

Characteristics	ASME	EPRI	PWROG	SKIFS	Code Case N-716
How are consequences handled?	Safety classes, excluding non-safety system failures	Based on PSA insights and results, qualitative descriptions, look-up tables and numerical ranges (CCDP / CLERP) to standardise the process.	Based on PSA outputs, qualitative descriptions, and deterministic insights	Based on deterministic view of safety system functions. No credit on Non- safety systems	Based on deterministic view of safety system functions and probabilistic insights from large number of RI-ISI applications. Credit on Non- safety systems used in defining additional HSS piping on a plant specific basis.
Are effects of non-safety class functions included?	NO	YES	YES	NO	YES(1)
Are integrated effects(4) of all safety functions included?	NO	YES	YES	NO	PARTLY
Is each safety class handled separately?	YES	DEPENDS(2)	DEPENDS(2)	NO	NO(5)
Are effects of breaks in small piping included?	NO	DEPENDS(6)	YES	NO	YES(5)
Are results depended on design of defence in depth level 3 (safety system design)	NO	YES	YES	NO	NO(7)
Is output affected by plant modifications in safety system design?	NO	YES	YES	NO	PARTLY(1)

Does methodology focus on the piping integrity and is it independent of designof defence in depth level 3?	YES	NO	NO	NO	PARTLY(3)

Table 14 – (cont.) Summary of features of evaluation of consequence of failure in the RI-ISI methodologies

Characteristics	ASME	EPRI	PWROG	SKIFS	Code Case N-716
Can plant modifications and/or procedures updates be a solution for higher CDF segments and other critical welds?	NO	YES	YES	NO	YES
Can the upgrading of PSA models (for instance removing conservatism) be a solution for higher CDF segments and other critical welds?	NO	YES	YES	NO	PARTLY(8)
Is it necessary to perform PSA technical adequacy (e.g. sensitivity studies, peer review, etc.)	NO	YES	YES	NO	YES
Is it important to evaluate "unknown failures(10)"?	YES	NO(9)	NO(9)	NO	NO
Is it needed to include "break exclusion zone"?	NO	NO	NO	NO	YES
Are indirect effects included?	NO	YES	YES	NO	YES

NOTES:

(1) Some piping may be classified as HSS even though its contribution to CDF/LERF is low. For example, a number of high consequence impacts have been based upon significant RI-ISI experience (e.g. LOCA, and Break Exclusion Regions have CCDPs/CLERPs greater than $10^{-4}/10^{-5}$). The impact of failure of all piping (safety related / non safety related, large bore / small bore) are assessed. Non safety related functions and systems are used to determine the final HSS / LSS scope.

(2) Each safety class within the scope of the RI-ISI programme is handled the same. Safety classes may be treated differently if one class is in scope and another class is outside the scope of the RI-ISI programme.

(3) Valid for the HSS part.

(4) Integrated effects means that the total effect of all safety functions are included in the evaluation of risk for core damage.

(5) Only full scope application is allowed.

(6) Scope (systems and pipe size limitations) are decided between the plant owner and the regulator.

(7) The full capacity of safety system is only used in those cases where the CDF is higher than 10^{-6}.

(8) Plant specific HSS piping (i.e. piping that exceeds 10^{-6} (CDF) and 10^{-7} (LERF) could potentially be reduced.

(9) These methodologies conduct the consequence assessment assuming the piping has failed, irrespective of cause and frequency.

(10) Unknown failures are failures that can not be quantified as there is total lack of data/knowledge. By using risk insights it is possible to exclude such as being outside of safety concerns. According to safety assessment praxis supported by regulations in different countries, events with a probability lower that 10^{-7} are specified as low risk and do not need specific attention based on safety concern. Thus, there is no need to look for less frequent events.

5.3 PSA demands for RI-ISI

This section discusses in a common level some issues that are important for consequence evaluations in a risk assessment applied to risk-informed in-service inspection.

As risk assessment means comparing different risks, it is important that the various modelling assumptions, input data and parameters influencing the way the risk is calculated are modelled with as high degree of realism as practically possible for the data that are important in the evaluation. It is important that any types of conservatism concerning the affected data used in (parts of) the studies are specified. The negative effect of uneven conservatism on the RI-ISI results has to be well understood.

The following PSA parameters, including understanding the impact of conservatism (lack of realism), are needed to obtain a sound consequence input to the RI-ISI ranking:

- Success criteria needs to be properly identified, i.e. how many trains of safety systems are needed to avoid core damage and the variation of system demands during the mission time.
- Identify break flows at different system pressures and different break sizes, e.g. different LOCA sizes.
- Including the appropriate mitigating systems/actions that can be used to bring the plant to a safe condition.
- Identify failure probabilities of components important to RI-ISI such as emergency core cooling (ECC), decay heat, auxiliary feed water, feed water and service water systems.
- Identify likelihood of loss of offsite power following a LOCA and time for restoration of power by diesels or off-site power where applicable and effects of house turbine operation.
- Common cause failure rates included for the applicable components.
- Identification of how human actions are developed and modelled in the PSA.
- All relevant dependencies among systems should be included the model.
- Event frequencies should be based on the latest years of plant operation which include plant upgrades on maintenance and operating procedures as well as modifications.
- The appropriate mission time is identified for system demands taking into account the various scenarios placing the demands.

Conservatism in parts of the above data can affect the risk profile which may result in bias towards those areas where the conservatism is largest. The effect of the above parameters is different in different type of reactors and it is of importance that the RI-ISI for a specific plant determines by sensitivity studies which parameters will most affect the ranking results if they are too conservative. Those parameters should to be assessed with higher degree of realism.

Most PSA studies are developed to get data on total core damage and large early release frequencies for the plant. To develop a fully realistic PSA is very expensive and requires many resources. For this reason, initial PSA models often include conservatism when data from SAR evaluations are used as basic input to the PSA model. Typically the most dominating sequences for CDF and LERF are identified, and conservatisms in the dominating sequences are replaced by more realistic data to generate a more refined PSA model. However, often there is still conservatism in the model where the conservatism has only a minor effect on the total CDF and LERF results. This conservatism may be difficult to quantify but different degrees of conservatism may affect ranking of welds in a RI-ISI programme.

It is very important to understand which data and models in the PSA studies affect the CDF and LERF from pipe breaks. These parts of the studies should be reviewed and if needed updated to reduce uneven conservatism in the model to get a correct risk ranking for a RI-ISI programme.

The best way to understand the degree of conservatism in PSA studies is to compare studies with sister plants and to evaluate the PSA study against quality guides as the Regulatory Guide 1.200 [42] which could involve a peer review of the PSA. Sensitivity studies are valuable tools in understanding the impact of these issues.

It is very important to check that the PSA model that is used for RI-ISI fulfils the demands for that application. There exist several guidance documents regarding PSA quality. These are specifying demands for different purposes of risk applications. There is still ongoing work to support development of PSA quality for different purposes. The bases for these demands are found in the R.G. 1.174 and 1.178. The ASME/ANS standard on PSA quality has recently been issued, replacing the prior ASME standard. It specifies different demands for different applications and it asks for a peer review of the PSA model. IAEA has developed the TECDOC-1511 on PSA Quality for Applications, 2006.

There is a need for qualifying the PSA studies against these standards and updating the PSA according to the demands for different applications. This qualification does not exist besides in US where the PSAs has been performed against the first version of the ASME quality standard for all US plants and are in a process of been updated against the latest version of ASME standard on PSA quality. EPRI and Westinghouse have developed guidance for their respective methodologies on PSA Technical Adequacy in support of RI-ISI programme development and maintenance.

5.4 Results of consequence evaluations in the RISMET study

The results of consequence analysis are presented for the four systems included in the scope of the RISMET project. The results are summarised in Table 15. This table is the main outcome of the consequence evaluation in RISMET. Table 15 presents some of the similarities and differences between the different methodologies based on the consequence analysis results.

The result from SKIFS methodology and the ASME method are independent of the PSA. The SKIFS methodology separates the different systems into more risk zones than the ASME methodology does. These methodologies include piping systems belonging to the Safety Classes 1 to 3. Therefore, systems and components that are non safety classified are excluded.

There are some similarities between the output from SKIFS and that of the PWROG and EPRI methodologies in separating the different systems into risk classes The PWROG and EPRI methodologies may be applicable to systems and components that are non safety classified. Another difference consists of the classification of system 321 outside the second isolation valve. In these cases however, PWROG and EPRI are sensitive to PSA modelling.

For the Code Case N-716 methodology, the consequence assessment is replaced with a pre-determined set of HSS locations. Additionally, the PSA is used directly as a complementary measure to identify other welds as HSS. For example, those welds whose failure would result in a contribution to CDF greater than 10^{-6} would be classified as HSS.

For EPRI, the following observations can be made regarding their sensitivity to the PSA quality:

- Conservatism, uncertainties or other weakness in the PSA may affect the risk ranking if it results in a change from one consequence category to another e.g. from high to medium or medium to high. Therefore, in the EPRI methodology a check has to be performed concerning existence of excessive conservatism on the segments in questions that can lead to added inspections that are not required. Also, large uncertainties or other weakness in the PSA that can affect the classification into the different consequence categories.

- In the consequence evaluation, the EPRI methodology uses an absolute ranking approach and any conservatism in the PSA will therefore at worst only add inspections compared with a best estimate population.

For PWROG, the following observations can be made regarding their sensitivity to the PSA quality.

- Conservatism, uncertainties or other weakness in the PSA may affect the risk ranking if they affect the PSA-results (CDF/LERF). The areas that are of importance in the PSA depend on e.g. the plant design but also on the inputs from the system failure probability for different pipe failure. Therefore PWROG must check that there exists no excessive conservatism in the PSA consequence assessment.

- The risk evaluation in PWROG uses a relative risk ranking. Excessive conservatism in any part of the PSA can therefore, in addition to affecting the risk ranking for those segments or system whose PSA result is conservative, result in an underestimation of the risk ranking for the other segments and other systems. This can affect the segments classification into HSS or LSS. This is a difference compared to EPRI.

- In the risk evaluation, an uncertainty distribution is placed on the PSA results and the failure probabilities to identify any segments that might become quantitatively HSS when reasonable variations are considered.

- As discussed in Section 6, the final categorisation of segments as HSS or LSS is determined by the expert panel based on a variety of inputs including risk metrics and consequences as well as other data. Part of the role of the expert panel is to account for limitations in the risk ranking process including PSA model limitations in identifying segments that should ranked as HSS.

5.4.1 Sensitivity analysis based on different assumptions in the consequence evaluation

EPRI were of the opinion that the CCDFs for RHRS and CCDPs for main steam system were very conservative and that operator action should be considered for main piping in RHRS based on their experience. Therefore, EPRI presents their results in this report for two different cases, one where EPRI uses their own experience (EPRI base) and one where EPRI uses the PSA results from Ringhals 4 (EPRI R4), refer to Section 5.2.3.

From that point of view it can also be of interest to see how corresponding changes affect the risk ranking for PWROG and Code Case N-716 for the RISMET scope as well as the PWROG full scope application due to the relative risk ranking. The other methodologies, ASME and SKIFS do not use the PSA and therefore are not included these sensitivity analyses. This section identifies sensitivity analyses where different assumptions in the consequence evaluation were assumed. This gives more information about how sensitive the risk ranking is for EPRI and PWROG regarding different assumptions in the consequence evaluation.

There are four assumptions in the consequence evaluation that can be of interest to analyse in more detail to see in what way they affect the outcome for EPRI, PWROG and Code Case N-716. The four assumptions are:

a. The assumptions for end state safe conditions within the mission time (24 hour) for transients between Ringhals 4 PSA-model and PSA in USA;

b. The assumptions for considering operator actions for piping failures in RHRS;

c. The assumptions resulting in potential conservatism in CCDP/CLERP for main steam system; and

d. The combination of assumptions a and c.

Evaluating the impact of these four sensitivity analyses requires the risk ranking evaluation to be conducted. Risk ranking evaluation is discussed in Section 6.4.1. Additional input for each sensitivity analysis is provided in the following paragraphs. Observations and conclusions for these sensitivity analyses are summarised after all sensitivity studies are presented.

Assumptions for End State Safe Conditions

In the USA, a safe condition is reach for transients with successful operation of auxiliary feed water system. This is based on having sufficient volume in the condensate storage tank (CST) to last for the full 24 hour mission time or by modelling the refilling of the CST in the PSA model. For Ringhals 4, RHRS must be in operation to get to a safe condition, even with successful operation of auxiliary feed water system. Due to the limited volume of water in the CST, it is not possible to use the CST during the whole mission time of 24 hours. In an emergency situation (i.e., inability to connect RHR), it is possible to connect the fresh water system to auxiliary feed water system and this is modelled in the PSA. However, since RHR can normally be connected, the modelling fresh water is not considered for all scenarios.

In the PWROG methodology failure of RHRS piping is assumed to lead to a loss of RC inventory. Since this loss of inventory will not occur unless the RHRS piping is placed in service, this consequence is typically modelled as a loss of both RHR trains. In the USA, postulating failure of large RHR piping, core damage only occurs if the RHRS is placed into service. For transients that end with successful operation of the auxiliary feed water system, core damage does not occur. For Ringhals 4, postulating failure of large RHR piping, core damage occurs for all transients since the RHRS is assumed to be placed in service. This results in the CCDF/CLERF of RHR postulated piping failures for the US plants being much smaller compared to Ringhals 4.

Based on the above, a sensitivity analysis was conducted comparing the following two scenarios for the EPRI, PWROG-SE[6], and Code Case N-716 methodologies.

- Scenario 1: Failure of RHRS piping where it is assumed that RHR must be in operation for a safe end state and loss of RC inventory is included. This scenario is represented in the base scenarios for the RISMET study (the EPRI R4 case, PWROG-SE cases for full scope and RISMET scope and Code Case N-716).

- Scenario 2: Failure of RHRS piping where successful operation of auxiliary feed water is considered a safe end state and the loss of RC inventory is modelled as loss of both RHR trains.

The piping associated with these scenarios is the large RHR piping. In general, failure of this piping leads to two general types of consequences, loss of RHRS and RCS inventory and loss of RHRS and RCS inventory along with loss of RWST. For scenario 1, the loss of RCS inventory combined with the assumption that RHR must be in operation for a safe end state dominates the contribution to CCDF/CLERF and is approximately 0.4 per year. For scenario 2, the CCDF/CCDP had to be estimated since calculating values would have required changing the Ringhals PSA model, which was not practical. For the piping whose consequences include the loss of RWST, the contribution to CCDF/CLERF is dominated by the loss of RWST and is approximately $3.86 \cdot 10^{-3}$ and $1.82 \cdot 10^{-4}$ for CCDF and CLERF respectively. For the piping whose consequences do not include a loss of RWST the CCDF and CLERF are estimated to be $9.64 \cdot 10^{-5}$ and $6.02 \cdot 10^{-7}$. These values may be slightly conservative but are consistent with the prior Ringhals 2 model where successful operation of auxiliary feed water was previously considered a safe end state.

No additional calculations or analyses were required for scenario 1 since it was included as part of the base scenarios. Since scenario 2 was not included as part of the base scenarios, additional calculations and analyses were required. For the EPRI methodology, it was assumed that the existing EPRI (base) case would be representative of scenario 2. For the PWROG-SE methodology, the risk ranking

[6] The PWROG-SE methodology is considered representative of the PWROG methodology for this sensitivity study.

evaluation was revised using the above data for scenario 2. For Code Case N-716, no additional calculations were required based on the following logic. No segments were added for CDF/LERF impact as part of the base study. Scenario 2 is expected to result in lower CDF and LERF values.

Assumptions for Considering Operator Action

A piping failure in RHRS after it has been pressurised by RCS and assuming that the piping failure is not isolated gives core damage because of the loss of RC-inventory. If operator action to isolate the piping failure is credited, typically by isolating a RHRS train, core damage only occurs if the other RHRS train were to also fail and there was an inability to go back to auxiliary feed water. This results in a significant reduction in the CCDF/CLERF of a postulated RHRS larger piping failure. At Ringhals 4, no operator actions were considered for RHRS larger piping failures because the available time for the operators to take action was considered to be too short based on a Ringhals 4 PSA-study for shut down operation and the expert panel. EPRI felt that operator should be credited. Refer to Section 5.4.1 for additional discussion on operator actions.

Based on the above, a sensitivity analysis was conducted comparing the following two scenarios for the EPRI, PWROG-SE[7], and Code Case N-716 methodologies.

- Scenario 1: Failure of RHRS piping where operator action is not credited. This scenario is represented in the base scenarios for the RISMET study (the EPRI R4 case, PWROG-SE cases for full scope and RISMET scope and Code Case N-716).

- Scenario 3: Failure of RHRS piping where operator action is credited.

No additional calculations or analyses were required for scenario 1 since it was included as part of the base scenarios presented in Section 6.

For the EPRI methodology, scenario 3 is represented by the EPRI (base) case.

The EPRI (base) case represents assumptions made by the EPRI application team consistent with past experiences in implementing the EPRI RI-ISI methodology for a large number of PWR plants. In particular, based upon risk applications and insights a number of plants have implemented risk management strategies to manage shutdown and plant refuelling activities. While initially developed to address shutdown evolutions, these plants changes (e.g. procedural changes, contingency plans and equipment availability) provide additional defence in depth measures for operational configurations as well as off-normal operation. For this scenario in particular, there are two configurations of note:

1. **RHR piping fails when the RHR system is initially aligned to the RCS**. This is a short time window and conservatively assumes that the RHR piping has not previously failed while the plant is at power and the RHR system is under RWST head or while the RHR system was subject to flushing and pre-conditioning prior to alignment to RCS. For this time window, operators are alert during this important transition and based upon training and procedures would immediately isolate the hot leg suction path and probably trip the RHR pump if there was a loss of inventory. If the operators did not react quickly enough, level would drop to the hot leg invert and start boiling off; and there is still another opportunity to isolate and provide RPV makeup. Also, the steam generators would be available given that isolation is successful during this early transition to RHR. The CCDP is typically assessed at approximately 10^{-4} considering the above.

2. **RHR piping fails after the RHR system is aligned to the RCS in the long term**. For this scenario, temperature, pressure and decay heat have been reduced. Given, the risk

[7] The PWROG-SE methodology is considered representative of the PWROG methodology for this sensitivity study.

management strategies discussed above and the longer time available for action and recovery, this scenario is bounded by scenario 1, above.

As part of the PWROG-SE methodology, two sets of consequences are developed without and with operator action. In scenario 3 operator actions are credited for isolating a train of RHRS. Since the without operator actions do not credit operator actions to isolate the failure, the results from Scenario 1 for the without operator action are also used for the without operator action for scenario 3. The "with operator action" consequences and risk ranking were altered to reflect the operator actions to isolate a train RHRS.

For Code Case N-716, no additional calculations were required based on the following logic. No segments were added for CDF/LERF impact as part of the base study. Scenario 3 is expected to result in lower CDF and LERF values.

Assumptions Resulting in Potential Conservatism in CCDP/CLERP for Main Steam System
In the Ringhals 4 PSA model, core damage is assumed if there is a failure to isolate more than one steam generator. This is based on design basis information in the Ringhals 4 safety analysis report. PSA models often are based on best estimate analyses. It may be possible based on best estimate analyses that blow down of two steam generators may not lead to core damage. Based on the above, a sensitivity analysis was conducted comparing the following two scenarios for the EPRI, PWROG-SE[8], and Code Case N-716 methodologies.

- Scenario 1: Failure of main steam piping where blow down of more than one steam generator is assumed to lead to core damage. This scenario is represented in the base scenarios for the RISMET study (the EPRI R4 case, PWROG-SE case for full scope and RISMET scope and Code Case N-716)

- Scenario 4: Failure of main steam piping where blow down of two steam generator is assumed to not lead to core damage.

No additional calculations or analyses were required for scenario 1 since it was included as part of the base scenarios presented in Section 6.

For the EPRI methodology, scenario 4 is represented by the EPRI (base) case.

Based on our experiences with a large number of PWR RI-ISI applications, we have seen CCDPs as high as $9\cdot10^{-5}$ and as low as $1.5\cdot10^{-6}$ for this piping. Because these values are bounded by the medium consequence range, a value of 10^{-4} was used.

For the PWROG-SE methodology, the CCDP and CLERP were approximated by reducing the CCDP and CLERP from scenario 1 by a factor of 100 for failure of the main steam piping. The risk ranking evaluation was then revised using the above data for scenario 4.

For Code Case N-716, no additional calculations were required based on the following logic. No segments were added for CDF/LERF impact as part of the base study. Scenario 4 is expected to result in lower CDF and LERF values for the main steam system segments.

Combination of the Assumptions for End State Safe Conditions and Assumptions Resulting in Potential Conservatism in CCDP/CLERP for Main Steam System
In this sensitivity study (scenario 5), the assumptions for the end state safe conditions and the assumptions resulting in potential conservatism in CCDP/CLERP for main steam system are

[8] The PWROG-SE methodology is considered representative of the PWROG methodology for this sensitivity study.

combined. This was done to see what impact it has on the PWROG-SE methodology results due to the relative ranking process. There was no need to run this scenario for Code Case N-716 as the results would not be different from scenarios 2 and 4.

For the PWROG-SE methodology, the risk ranking evaluation was revised using the revised data from scenarios 2 and 4.

Observations from Sensitivity Analyses
Based on the results of the sensitivity analyses (refer to Section 6), the following high level observations were made regarding the consequences analysis input to the various RI-ISI methodologies that use the PSA as input to the RI-ISI programme.

- Outputs from this sensitivity study indicate that the PWROG (original) and EPRI methodologies can be affected by the following parts of the consequence evaluatio

  – Realism in thermo-hydraulic evaluating system demands to avoid core damage, This includes issues as how to specify end-state (safe shut down conditions), mission time, safety system availability due to different demand during the mission time, and the realism in codes performing the evaluation of the LOCA-scenarios.

  – Probability of success in performing manual action as closing valves to isolate the breaks.

  – Assessment of system availability due minor loss of water inventory during long time (24 hours). This includes the assessment of operator actions to mitigate such failures.

- Areas of the PSA that are of importance regarding the demands of high quality can be different between PWROG and EPRI. For example potential conservatism in the CCDPs only adds inspections for the EPRI approach. While for the PWROG, conservatism in the CCDPs may add inspection to low risk areas and potentially remove inspections from other areas.

- For this sensitivity study, Code Case N-716 was not impacted by changes in the PSA model. This is because, even with the conservative CCDP values, the Ringhals PSA consistently showed this piping to be low risk (i.e. CDF less than 10^{-6} / LERF less than 10^{-7}).

- In a relative risk ranking, potential conservatism in parts of the PSA can have more or less impact on the risk evaluation (compare changes in assumption for RHRS and main steam system). The influence of potential conservatism is in general greater for partial scope compared to a full scope. The reason for that is that the total piping CDF/LERF is larger for a full scope compared to a partial scope and changes in the PSA have therefore a smaller impact on the total piping CDF/LERF for a full scope compared to a partial scope.

- The impact of crediting or not crediting operator action has different impacts on the results of the RI-ISI programme between the EPRI and PWROG (original) and PWROG-SE methodologies. For the EPRI methodology the only impact, if any, is on the segments for which the operator action is being considered. For the PWROG (original) and PWROG-SE methodologies, the “without operator action" results are not impacted. The “with operator action” results may change. The overall risk ranking is based on both the “with”- and “without operator action” results. Therefore, the impact, if any, is on the other segments due to the “without operator” actions not changing and the relative ranking process.

- Country specific analysis rules can influence the results of a RI-IS programme. For example, requiring a longer period of time to be able to credit operator action may result in not crediting certain operator actions. As discussed previously, risk assessment application should strive for best estimate conclusions, as supplemented with additional efforts (e.g. defence in depth). Thus, country specific analysis rules that can inadvertently impact the risk-informed results need to be well understood and measures taken to minimise their impact, if these impacts are adverse.

- In general, the ground rules for crediting of operator actions in a RI-ISI programme should be consistent with the ground rules for crediting operator actions in the PSA model. However it is important to understand the impact that the ground rules will have on the results of the RI-ISI programme and to make adjustments as necessary for a robust RI-ISI programme.

- If robust results (not depending on the evaluators) are to be developed by these methods, training, detailed guidance and assistance from knowledge personnel with prior experience in developing a RI-ISI programme are recommended.

Table 15 Results of failure consequence analysis for the four systems include in the scope of work of the RISMET project

System 313	**PWROG** (CCDP)	**EPRI Rank** (CCDP)	**SKIFS** (Consequence Index)	**ASME** (Safety Class)	**Code Case N-716 (CDF/LERF)**
Loop	10^{-2}	High (10^{-2})	1	1	HSS
Piping > 150 mm	10^{-2}	High (10^{-2})	1	1	HSS
Cold leg Piping 20 -150 mm	10^{-2}	High (10^{-2})	2	1	HSS
Hotleg piping > 70 mm	10^{-2}	High (10^{-2})	2	1	HSS
Piping 50-70 mm - Hot leg	10^{-3}	High (10^{-3})	3	1	HSS
System 321					
To second isolation valve	Depending on success time for isolation $10^{-1} – 10^{-4}$	Depending on success time for isolation High (10^{-1}) – Medium (10^{-4})	2	1	HSS
Outside second isolation valve	Depending on success time for isolation $10^{-1} – 10^{4}$	Depending on success time for isolation High (10^{-1}) – Medium (10^{-4})	3	2	LSS ($<10^{-6}$ / $<10^{-7}$)
System 411					
Steam generator to containment penetration	Depending on definition or realism in estimation of core damage $10^{-2} – 10^{-4}$	Depending on definition or realism in estimation of core damage High (10^{-2}) – Medium (10^{-4})	2	2	LSS ($<10^{-6}$ / $<10^{-7}$)
Containment penetration to main steam isolation valve	Depending on definition or realism in estimation of core damage $10^{-2} – 10^{-4}$	Depending on definition or realism in estimation of core damage High (10^{-2}) – Medium (10^{-4})	3	2	LSS ($<10^{-6}$ / $<10^{-7}$)
< 2" lines inside containment	10^{-5}	Medium (10^{-5})	3	2	LSS ($<10^{-6}$ / $<10^{-7}$)
< 2" lines outside containment and non safety classed piping	None	None	None	None	LSS ($<10^{-6}$ / $<10^{-7}$)
System 414					
Entire system	$10^{-5} – 10^{-6}$	Medium (10^{-5} - 10^{-6})	None	None	LSS ($<10^{-6}$ / $<10^{-7}$)

5.4.2 Outliers

Using PSA level 1 and 2 in the assessment of RI-ISI-programme will for some plants identify outliers. Outliers are components (welds or piping sections) that have a much larger risk for core damage (CD) or large early realease (LER). How this is handled in the RI-ISI-programme is described in section 6.

For the plant overall safety the identification of these outliers are very important complementary information. If efforts are made to eliminate these outliers with specific mitigation countermeasures, the plant safety will increase. In some cases such mitigation programmes (e.g. hardware, procedures, training) give more safety improvements than the specified ISI-programme.

Those methodologies/programmes that do not include PSA assessment will not get such important complementary information.

5.5 Conclusions

Based on the evaluation of how failure consequence is treated, the following has been found valid for the different methods (Table 16).

Table 16 General conclusions regarding failure consequence treatment

Different effect	**Valid for the following methods**
Estimation of total core damage risk for pipe failure (including all piping)	PWROG(1), EPRI(1), Code Case N-716
Optimisation of ISI for all piping failures, independent of classes	PWROG(1), EPRI(1), Code Case N-716
Inclusion of effects of ”Defence in depth level” 1, 2 and 3	PWROG, EPRI
Focus on ”Defence in depth level” 1 and 2	ASME, SKIFS, Code Case N-716
Inclusion of grouping of risk, with boundary limits effects	EPRI, Code Case N-716, SKIFS, ASME
Dependence on high degree of realism in the PSA in those areas that are of importance for the risk evaluation	PWROG, EPRI
Risk Ranking Results can be affected by or can be changed by plant modifications or procedure updates	PWROG, EPRI, (Partly in Code Case)
Risk Ranking Results can be affected by power up-rates	PWROG, EPRI, (Partly in Code Case)

(1) Only for full scope applications

A major difference between EPRI and PWROG is the following. EPRI uses an absolute ranking process while PWROG uses a relative ranking process. It follows that potential excessive conservatism in parts of the PSA can have different impact on the risk evaluation for EPRI and PWROG. For EPRI, potential conservatisms may add inspections compared to a best estimate population. For PWROG, potential excessive conservatism can, in addition to affecting the RRWs for those segments/systems whose PSA result is conservative, result in an underestimation of the RRWs for other segments and/or systems. This can affect the segments categorisation as HSS or LSS.

It is very important to check that the PSA model that is used for RI-ISI fulfils the demands for that application, i.e. the PSA is of high quality (high degree of realism) in those areas that are of importance for the risk evaluation. The important areas of the PSA for the RI-ISI evaluation depend on plant design and can be different between PWROG and EPRI.

The assumptions made about the safe condition (end state) for transients are important regarding the results of the consequence evaluation.

This study suggests that the PWROG and EPRI methodologies can be affected (changes from high level to low level) by the following parts of the consequence evaluation:

- Realism in thermo-hydraulic analyses of system demands to avoid core damage. This includes issues as how to specify end-state (safe shut down conditions), mission time, safety system availability due to different demand during the mission time, and the realism in codes performing the evaluation of the LOCA-scenarios.

- Probability of success in performing manual actions, such as closing valves to isolate breaks.

- Assessment of system availability due to minor loss of water inventory over longer times (24 hours). This includes the assessment of operator actions to mitigate such failures. (In Swedish BWR such leakages will give automatic isolations.)

As both deterministic and probabilistic safety assessments assume that there are no breaks in the "break exclusion zone", it is important to assess this area with complementary methods as it is done in the Code Case N-716 approach.

6. EVALUATION OF RISK RANKING AND SITE SELECTION

This chapter summarises the evaluation of risk ranking and inspection site selection. The main principles of the risk ranking and site selection process in each methodology were presented in section 2.3. First, the limitations of the benchmark and their effect on the evaluation are discussed in section 6.1. An overview of the results is given in section 6.2, with a discussion on risk measures used in the risk-informed approaches. In section 6.3, the qualitative differences of the results are presented at a system level. For each analysed system, the main differences are pointed out, and their reasons are discussed. In section 6.4 sensitivity analyses for PWROG, EPRI and Code Case N-716 are presented for different scenarios. In section 6.5 quantitative analyses are presented to investigate the impact of the differences in inspection site selection on the risk. Section 6.6 discusses specific features related to the various methodologies. Finally, section 6.7 summarises the main findings of the evaluation of risk ranking and site selection.

6.1 Effect of benchmark limitations on the risk ranking and site selection

The selection of inspection sites was limited to identifying the number of inspections at segment level. The exact inspection site selection exists for the PWROG-SE and SKIFS applications. The selection in the original PWROG application was made independently using the PWROG structural element selection process.

The selection in ASME XI application was made independently, following in RHR and MS systems the principles of Tractebel, and in RC and RH systems the inspection sites other than the terminal-ends were selected for this application on basis to the magnitude of the stress discontinuity (stress index).

In the case of the EPRI and Code Case N-716 applications, when there was consistency in risk ranking with PWROG-SE results, it was assumed that the exam locations for these methodologies would be the same as the exam locations selected for the PWROG-SE methodology. This is because, during the element selection process, each methodology uses inputs such as access, worker exposure, severity of postulated degradation and inspection history in selecting the final set of exam locations. However, in practice the selection of the same exam locations may or may not have been true if the selection of the exam locations had been done independently of the results of the PWROG-SE methodology.

In the PWROG-SE methodology, in general, exams are placed on each high safety significant segment. In the EPRI and Code Case N-716 methodologies, the segments within a given risk category, are grouped together and a certain percentage of locations are selected across all segments in that category, but there is no requirement that an exam be placed on each segment. This provides greater flexibility in inspection site selection (e.g. reduced worker exposure, fuller examination coverage). For the EPRI methodology there were inspection locations in high and medium risk categories which might have been selected for examination which were not high safety significant in the PWROG-SE methodology, and vice versa. Thus there might have been more differences in the exam locations between the PWROG (original)/PWROG-SE methodologies and the EPRI/Code Case N-716 methodologies.

It is important to understand the scope of systems chosen for the RISMET project are not what would be expected in a real plant RI-ISI application. For example, almost all real plant applications to date have been to a single class of system (e.g. Class 1 only), Class 1 and 2 systems, or a large fraction or the entire the plant. Because of the systems chosen, in this study the results of the RISMET scope are nearly identical to the results for a full scope analysis for both the relative ranking methodologies and the absolute ranking methodologies. However, drawing conclusions based on the RISMET scope of application may not be appropriate in all instances.

6.2 Overview of the results

6.2.1 General observations of the site selections

Table 17 summarises the number of inspection sites in each system in each application. It can be noticed that ASME XI application leads to significantly larger number of inspections in the Reactor Coolant System than any of the other approaches. In the case of the Residual Heat Removal system, SKIFS and Code Case N-716 applications result in very few inspections (Class 1 part of the system), while the other applications result in 20-40 inspections. The detailed analyses of the reasons for the selections are discussed in section 6.3 at system level.

The Erosion (E) and Flow Accelerated Corrosion (FAC) inspections are treated separately, because the plant has a separate programme for these inspections. The impact of the RI-ISI applications on Erosion and FAC susceptible systems is discussed in connection to the Condensate and Main Steam systems and the FAC sensitivity study.

Table 17 Summary of number of inspection sites in applications.

	RCS (313)	RHR (321)	MS (411)	CS (414)	Total
ASME XI	113	30	28	0	171
SKIFS 1994:1	40	1	29	0	70
PWROG orig 4	28(+4 VT-2)	21	10 + E	24+FAC	83(+4 VT-2 +FAC)
PWROG-SE full	28	35	3 + E	0+FAC	66(+FAC)
PWROG-SE 4	30	35	3 + E	0+FAC	68(+FAC)
EPRI base	47	26	2 + E	3+FAC	78(+FAC)
EPRI R4	47	40	40 + E	3+FAC	130(+FAC)
CC N-716	47	4	7 + E	0+FAC	58(+FAC)

6.2.2 Risk measures in the risk-informed approaches

One way to illustrate the results is to show the segments' risk as a function of failure frequency versus CCDP or CLERP. Figure 8 shows the PWROG evaluated segment specific risk values for initiating event failures for the case of vibration fatigue segments removed and no inspections or operator actions. The failure frequencies are calculated using the SRRA code. Each point in Figure 8 represents a pipe segment. The risk is constant along the diagonals and from the risk diagram it is easy to identify the high risk segments and whether the risk is dominated by high failure frequencies or high CCDP. Figure 9 shows the similar CDF-results for the pipe systems which only degrades a safety system. In this case the failure probability is shown as function of the CCDF.

The determination of the risk importance of a segment or weld is not unambiguous. As discussed in relation to the evaluation of failure potential and consequences, already in those phases different assumptions may lead to different rankings.

In the PWROG methodology, the safety significance is measured with the risk reduction worth (RRW) importance measure. In Figure 10 the correspondence of CDF and RRW is illustrated by using the data from SRRA failure probability evaluations and highest consequence estimates from the PSA runs. In the figure, for the consequence measure the highest value of calculated CCDP or 10xCLERP has been used. The reason for using CLERP multiplied by ten is that often in risk-informed applications a ten times smaller LERF is considered equivalent to a corresponding CDF value. Since the CLERP for a part of the system 321, if the operator actions are not credited, is as high as 0.4, the resulting consequence measure on the x-axis of the figure is 4. The plot indicates a good correlation between the CDF and the RRW.

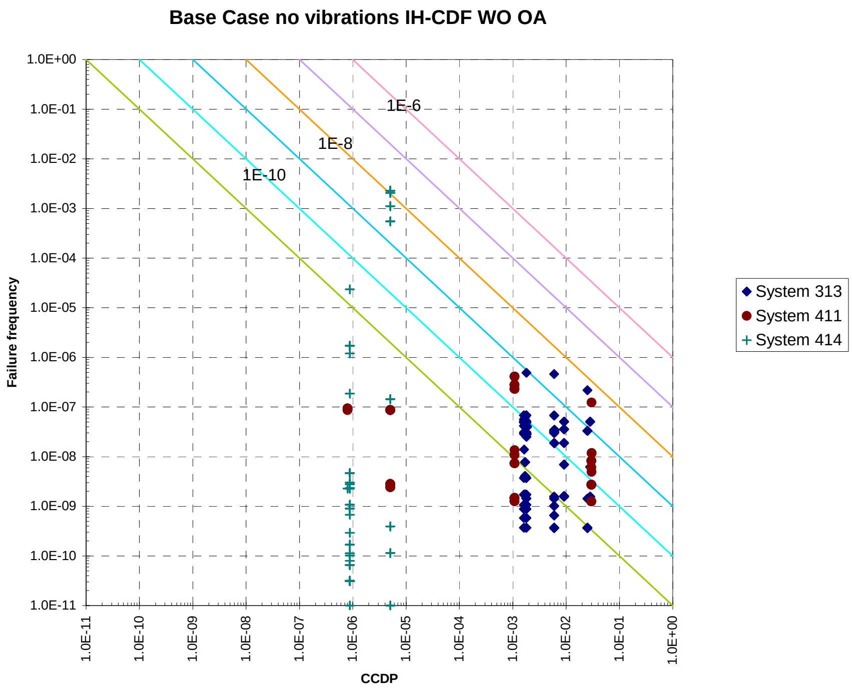

Figure 8 Segments causing initiating events shown on a risk plot based on SRRA results and Ringhals PSA results.

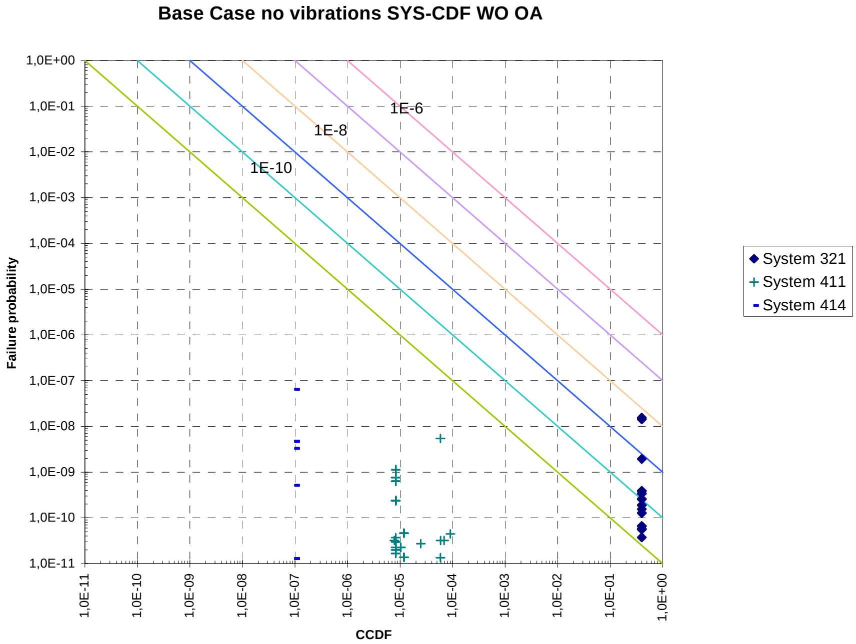

Figure 9 Segments causing the degradation of a safety function shown on a risk plot based on SRRA results and Ringhals PSA results.

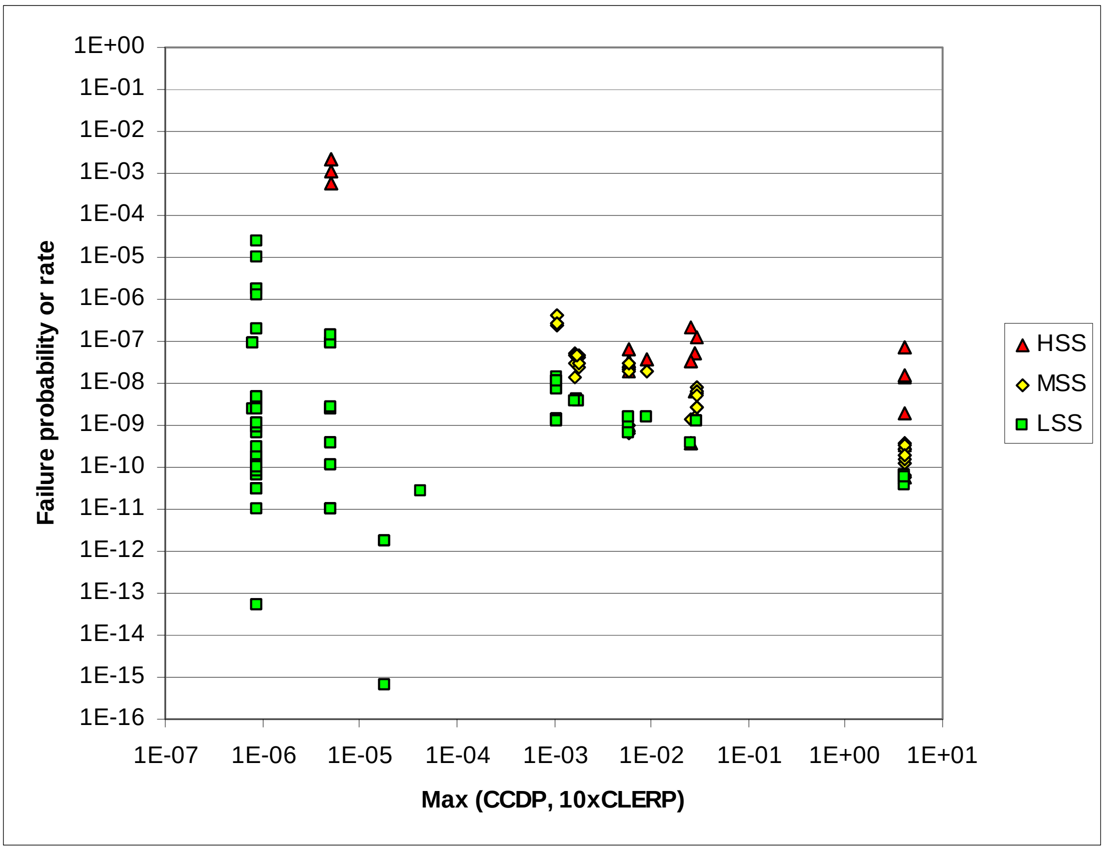

Figure 10 Risk plot based on SRRA results and Ringhals PSA results showing initial safety significance classification based on RRW values.

In the SKIFS and ASME XI methodologies, the piping importance is evaluated qualitatively without the use of PSA results. The division of segments in SKIFS inspection groups are plotted in Figure 11 using the same data from the PWROG application as in the previous figure. A similar plot for the ASME XI classification is presented in Figure 12. There is a clear disagreement of consequences of the RHR system between the R4 PSA analyses and SKIFS, since following the SKIFS principles most of the system falls in inspection class C due to low consequences.

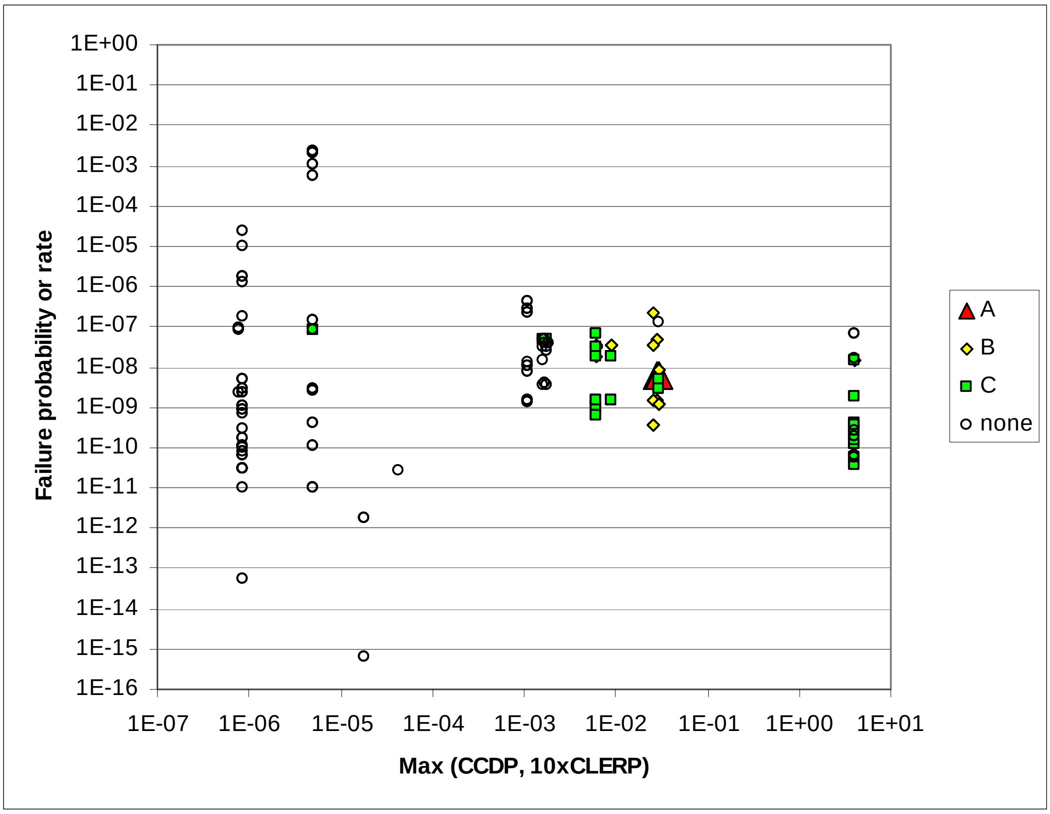

Figure 11 Segments according to SKIFS inspection groups shown on a risk plot based on SRRA results and Ringhals PSA results.

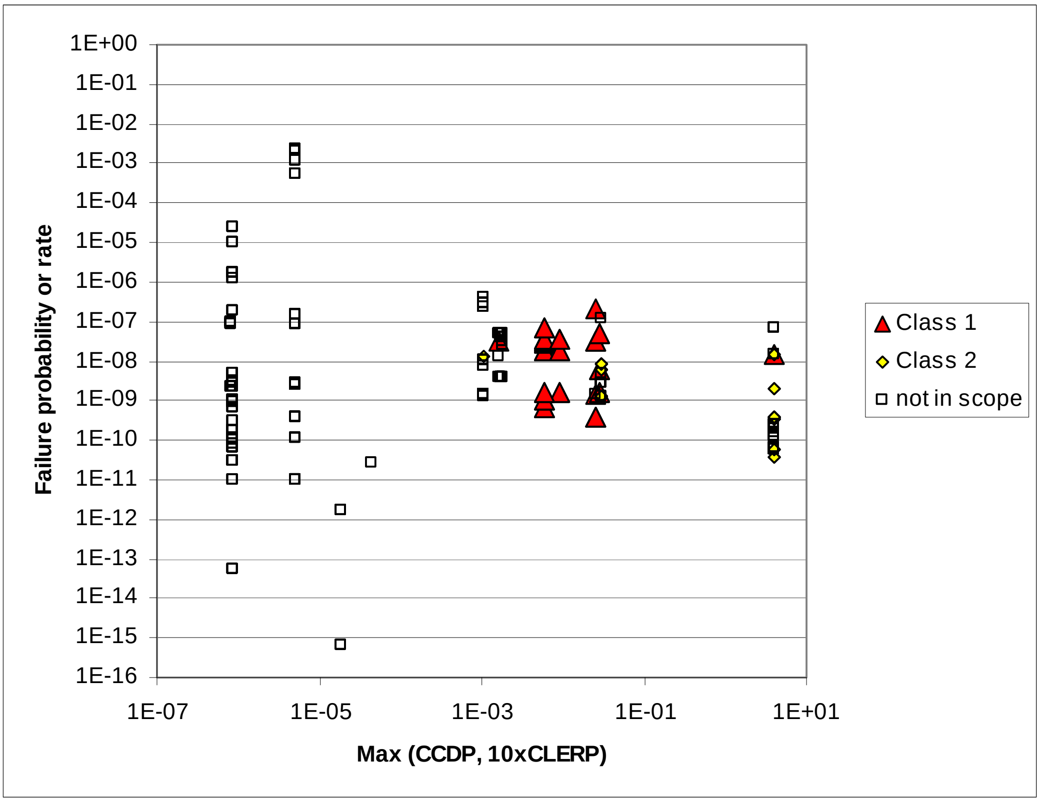

Figure 12 Segments according to ASME XI classification shown on a risk plot based on SRRA results and Ringhals PSA.

6.3 Evaluations at system level

In the following, the main findings are presented at system level. For each system, a summary of the segment classification is presented with a colour coding. This is to give an overview of the differences without details on the segments. The main differences are then discussed in more detail.

The nomenclature and colour coding using in the following tables are summarised in Table 18. A symbol “X” in the tables means that the segment is grouped with an adjacent segment. If a segment is left blank (white cell in Tables 19 to 22), the segment is not within the scope of the methodology.

Note that the classification is not always done at segment level. A segment can for instance contain welds belonging to two different inspection groups in the SKIFS application. For simplicity, the whole segment is coloured according to the highest inspection group present.

It should also be noted that the colour coding is not directly comparable between different applications. In the PWROG methodology a segment ends up in region 1 if there is an active degradation mechanism present. The whole segment is marked “red”, even if only the elements having an active mechanism are ranked to region 1A, and other elements within the segment belong to region 1B. The highest EPRI risk region includes not only segments with high failure potential, but also segments ranked as medium failure potential if they have high consequences. Thus it is quite natural that the EPRI methodology results in more “red” segments than the PWROG applications in a system with high consequences.

Table 18 Nomenclature and colour coding using in risk ranking tables.

	Legend entry	Explanation	Colour coding
AMSE	**ASME XI**		White: Not in the scope of the methodology
			Red: Class 1, coverage required 25%
			Yellow: Class 2, coverage required 7.5%
SKIFS	**SKIFS**		White: Not in the scope of the methodology
			Red: Inspection group A
			Yellow: Inspection group B
			Green: Inspection group C
EPRI	**EPRI base :**	EPRI base case	Red: Risk region High
	EPRI R4:	EPRI application using R4 PSA results as such for consequence ranking	Yellow: Risk region Medium
			Green: Risk region Low
PWROG	**W4 :**	PWROG 4 systems, ranking based on RRW (**prior** to expert panel)	Orange: High Safety Significant
	W-S full :	PWROG-SE, full scope, ranking based on RRW (**prior** to expert panel)	(light) Yellow: Medium Safety Significant
	W-S4 :	PWROG-SE, 4 systems, ranking based on RRW (**prior** to expert panel)	(light) Green: Low Safety Significant
	W4 EP :	PWROG 4 systems (ranking **after** expert panel)	Red: Region 1A[1]
			Yellow: Region 1B & 2[1]
			Green: Other (3 and 4)[1]
	W-S full EP :	PWROG-SE, full scope (ranking **after** expert panel)	Red: Region/inspection group A[2]
	W-S4 EP :	PWROG-SE, 4 systems (ranking **after** expert panel)	Yellow: Region/inspection group B[2]
			Green: Other (C or none)[2]
	Code Case N-716	Code Case N-716 application	Red: High Safety Significant
			Green: Low Safety Significant

Notes: ([1]) For the regions, see the PWROG Structural Element Selection Matrix. For the red cells, not all elements in the segment do necessarily belong to region 1A, but at least one does.
([2]) Inspection groups according to SKIFS methodology.

6.3.1 Reactor coolant system (313)

The summary of segment classification for the Reactor Coolant System is presented in Table 19.

As a general observation it can be noted, that ASME XI and Code Case N-716 have the same criteria for ranking the high safety significant segments. Most of these segments are considered of low or medium importance in the other applications. However, there is an important difference between the ASME XI and Code Case N-716 applications: while ASME XI requires 25 % inspection coverage, for Code Case N-716, which addresses failure potential and consequence of failure in its inspection site selection process, 10 percent of the population is sufficient. This reduces the number of inspections from 113 to 49.

In the EPRI applications, ¾ inch lines were excluded from the analysis. This exclusion was justified by an assessment that the reliability of these locations is driven by vibratory fatigue, which is not amenable to periodic ISI (see section 3.1.2 on scope).

There are not any differences between EPRI base and EPRI R4 applications for this system. Also the various PWROG applications are in accordance with each other. The EPRI applications resulted in slightly larger selection than SKIFS. The PWROG applications result in the smallest number of inspections.

First, we comment the expert panel decisions of the PWROG methodologies. Segments 1-3 have high consequences, and the expert panel has moved them to HSS category because of deterministic reasons (defence in depth). Most of the segments originally ranked as Medium Safety Significant (MSS) according to the RRW have been lowered to LSS. The categorisation of MSS segments depends on many factors including risk metrics, deterministic insights, PSA values, failure probabilities, etc. Segments with RRWs in lower portion of the MSS range are typically ranked LSS unless there is some deterministic reason to rank the segment HSS.

The segments that EPRI has ranked as high risk (13-15, 19-20, 23-26, 28, 30, 37 and 39) all have a possibility for thermal fatigue, according to the criteria used in the EPRI methodology. The segment 30 has been identified as the most important segment in the PWROG applications. This is a surge line from a hot leg loop to the pressuriser, and the failure probability is considered high due to thermal stratification.

The SKIFS application identifies two welds (in segments 7 and 9) as of highest safety significance because there the charging lines are connected to the cold legs.

Table 19 Ranking of segments and number of inspections in the Reactor Coolant System.

313	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
ASME	3	4	2	3	3	3	6	5	5	1	1	1	2	2	2
SKIFS	1	1		1			2	1	1	1	1	1	1	2	2
EPRI base	1	1	1				2	2	2				2	2	2
W4															
W-S full															
W-S4															
W4 EP	1	1	1				1	1	1				1	1	1
W-S full EP	1	1	1				1	1	1				1	1	1
W-S4 EP	1	1	1				1	1	1				1	1	1
EPRI R4	1	1	1				2	2	2				2	2	2
CC N-716	1	1	1				2	2	2				2	2	2
	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30
ASME	1	1	1	1	1			1	1	3	1	3	4		2
SKIFS	1	1	1	1	1			1	2				2		1
EPRI base	1	1	1					1	1				1		5
W4															
W-S full															
W-S4															
W4 EP	1	1	1					1	1				1		1
W-S full EP	1	1	X					1	1				1		5
W-S4 EP	1	1	X					1	1				1		5
EPRI R4	1	1	1					1	1				1		5
CC N-716	1	1	1					1	1				1		5
	31	32	33	34	35A	35B	36A	36B	37	38	39	40	41	42	43
ASME	1	1	2	1					3	1	1	1	1	1	4
SKIFS	2	2							1		1	1	2		1
EPRI base	1	2	1	1					1		1		2		1
W4															
W-S full															
W-S4															
W4 EP	1	1	1	1					1		1				1
W-S full EP	1	2							1		1				
W-S4 EP	1	2	1	X					1		1				1
EPRI R4	1	2	1	1					1		1		2		1
CC N-716	1	2	1	1					1		1		2		1
	44	45	46	47	48	49	51	53	54	55	56	57	58	59A	59B
ASME	1													6	
SKIFS	2	1	1												
EPRI base	2	1	1											1	
W4															
W-S full															
W-S4															
W4 EP														1	
W-S full														X	

EP															
W-S4 EP														X	
EPRI R4	2	1	1											1	
CC N-716	2	1	1											1	

Table 19 (cont.) Ranking of segments and number of inspections in the Reactor Coolant System (313).

313	60A	60B	61A	61B	62A	62B	63A	63B	64A	64B	65A	65B	66A	66B	67A
ASME	6		5		1	3	1	3	1	3	1		1		1
SKIFS															
EPRI base	1		2		1		1		1						
W4															
W-S full															
W-S4															
W4 EP	1[a]		1[a]		1		1		[a]						
W-S full EP	1		X		1		1		1						
W-S4 EP	1		X		1		1		1						
EPRI R4	1		2		1		1		1						
CC N-716	1		2		1		1		1						

	67B	68	69	70	71	72	73	74	75	76	77	78
ASME												
SKIFS												
EPRI base												
W4												
W-S full												
W-S4												
W4 EP												
W-S full EP												
W-S4 EP												
EPRI R4												
CC N-716												

([a]) VT-2 examination is performed on all or part of the segment.

6.3.2 Residual heat removal system (321)

The summary of segment classification for the Residual Heat Removal System is presented in Table 20.

The Residual Heat Removal System is a class 2 system, with the exception that segments 1, 2, 18 and 19 belong to class 1. Several segments are exempt from ASME XI examinations due to their small size. The SKIFS and Code Case N-716 selection in this system are remarkably small compared to the other applications. According to SKIFS approach, the piping after the second valve are classified as low consequence and are thus outside SKIFS scope.

For the entire system, the PSA CCDF and CLERF values are very high, when the operator actions are not credited. The reason for these high values is the analysis of the shutdown state. As discussed in Sections 5.2.3 and 5.4.1, there was a difference of opinion as to whether operator action should be credited for failures of the main RHR piping and whether ISI would be the most appropriate risk management strategies for these potential scenarios. The time windows for accepting the crediting of operator actions may be different in different countries, a fact which also affects the interpretation. These topics were discussed in connection to the consequence evaluation. The difference between the EPRI base and EPRI R4 results arises from crediting operator action in EPRI base but not in EPRI R4.

The Code Case N-716 examinations are not impacted by the decision to credit or not credit these operator actions, and the Code Case N-716 results (CDF less than 10^{-6}, LERF less than 10^{-7}) are based on PSA model not crediting the operator actions. Crediting the operator action in the PSA model would lower the CDF/LERF results and not add any exams.

In this system, we can see differences between the PWROG applications in the classification prior to expert panels. In general, the classification should be the same for PWROG (original) 4 systems and PWROG-SE 4 systems applications. In the PWROG-SE methodology, segments with vibratory fatigue are removed from the risk evaluation to identify if any other segment would become quantitatively HSS. For the RISMET scope of application, four RHR segments with vibratory fatigue (36-39) were removed from the risk evaluation. Due to these four segments being dominant contributors for the CDF and LERF "without operator action" cases, ten other segments increased above the threshold for being identified as quantitatively HSS.

The four RHR segments with vibratory fatigue (36-39) were identified as quantitatively HSS based on the "without operator action" risk ranking results. The expert panel ranked these four segment LSS based on the lower "with operator action" risk results due to operator action. These segments are therefore placed in inspection group C (owner defined inspection programme).

The differences between PWROG-SE full scope and PWROG-SE 4 systems applications classification prior to expert panels are due to the change of scope, which was discussed in Section 3.1.3.

There is a good agreement between PWROG, EPRI R4 and ASME XI applications on the importance of most of the segments 1 -17, and segments 50-51. Segments 13 and 14 are thermal mixing points after the heat exchanger, and thus placed in category 1A (highest safety significance) in PWROG applications.

In system 321, the EPRI Base methodology has ranked as Medium risk four segments and the EPRI R4 application 12 segments that would typically be exempt if following the ASME Section XI exemptions. In the Ringhals PSA, all of these segments were identified as contributing less than 10^{-6} (CDF) and less than 10^{-7} (LERF).

Table 20 Ranking of segments and number of inspections in the Residual Heat Removal System (321).

321	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
ASME	1	1			5	2	4	2		1			5		2
SKIFS	1														
EPRI base	1				4	4	4	3	2	2			1	1	
W4															
W-S full															
W-S4															
W4 EP	1	1	1	1	1	1	1	1	1	1	1	1	2	2	1
W-S full EP	X	1	2	1	4	4	4	4	1	1	1	X	1	2	2
W-S4 EP	X	1	2	1	4	4	4	4	1	1	1	X	1	2	2
EPRI R4	1	1	2	2	4	4	4	3	2	2	1	1	1	1	2
CC N-716	2	2													

	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30
ASME	5														
SKIFS															
EPRI base															
W4															
W-S full															
W-S4															
W4 EP	1	1													
W-S full EP	3	2													
W-S4 EP	3	2													
EPRI R4	2	2													
CC N-716															

	31	32	33	34	35	36	37	38	39	40	41	42	43	44	45
ASME															
SKIFS															
EPRI base															
W4															
W-S full															
W-S4															
W4 EP															
W-S full EP															
W-S4 EP															
EPRI R4															
CC N-716															

	46	47	48	49	50	51	52	53	54	55	56	57	58
ASME					1	1							
SKIFS													
EPRI base					2	2							
W4													
W-S full													
W-S4													
W4 EP					1	1							
W-S full EP					1	1							
W-S4 EP					1	1							
EPRI R4					2	2	1						
CC N-716													

6.3.3 Main steam system (411)

The summary of segment classification for the Main Steam System is presented in Table 21.

As can be seen from the summary table, there is generally a good accordance of the low safety significance of most of the segments. The segments of interest are 1-12 and 35-44.

For segments 1-18 and 31-35 the R4 PSA model gives CCDP $3{\cdot}10^{-2}$ and for segments 36, 41-68 and 86-89 the R4 PSA model gives CCDP $1.08{\cdot}10^{-3}$. No operator actions were considered for these segments. In these cases the PSA model was considered very conservative by EPRI, and smaller CCDP values based on EPRI own experience from many RI-ISI applications were used instead in the EPRI base application. In EPRI R4 application the PSA results were use as such, which raises these segments from Low to Medium risk region. Code Case N-716, which is based on the same PSA model as the EPRI R4 application, resulted in significantly fewer exams than EPRI R4. The reason for this is that this piping contributes less than 10^{-6} (CDF)/10^{-7} (LERF).

Both SKIFS and ASME place a majority of inspections in segments 1-9. For ASME, these are Class 2 piping locations, while the remainder of the piping is typically classified as non-safety related. The SKIFS methodology assumes that there is no consequence after the main steam valves based on the ability to isolate leaks downstream of the main steam valves.

In the PWROG applications, segments 1-3 are initially estimated to have a low failure probability, but the expert panel has identified them HSS due to deterministic insights related to vibrational loads in the past for R3, some repairs and that there is a high velocity of the steam in this piping Segments 5 and 6 are categorised HSS because of RRW>1.005.

Segments 7-9 have been placed in the highest risk region in the SKIFS application. The reason for this is that the Fessenheim 1 plant in France has had damage problems in a corresponding segment. Also Code Case N-716 and ASME XI applications would have inspections in these segments.

Segment 35A is judged to be subject to thermal transients. EPRI application has also ranked segment 34 in Medium risk region due to the failure potential (thermal fatigue).

For segments 41-48, the PWROG applications have evaluated a relatively high failure ("medium"!) probability. The EPRI analyses do not support this assumption. Segments 41-44 are judged to be of high safety significance due to erosion in PWROG applications. In the EPRI and Code Case N-716 applications, it is assumed that these segments 41-44 (as well as segments 1-3) are included in the plant erosion programme. A methodological difference can be noted here between EPRI and PWROG applications: the PWROG methodology requires additional inspections of welds to these HSS erosion susceptible segments.

Segments 411-1 through 3 and 41 through 44 are identified as being part of an erosion programme in the EPRI and Code Case N-716 methodologies. Prior to the PWROG RI-ISI programme, these seven segments were not identified as part of the Ringhals erosion programme. Due to the expert panel's perceived potential for erosion and due to the relatively high velocity steam, the expert panel made these segments HSS and erosion examinations were added to these segments. Since erosion examinations were identified for these segments for the PWROG-SE methodology, the segments were identified as having erosion exams for the EPRI and Code Case N-716 methodologies. However, since the EPRI and Code Case N-716 methodologies do not change existing erosion programmes (modifications, additions or deletions), these segments probably would not have been identified as having erosion exams if the results of the PWROG-SE methodology had not been known. In the PWROG (original) and PWROG-SE methodologies, existing erosion exams are not changed, but additional exams may be identified as part of the process, if the scope of RI-ISI application includes the applicable system. This scenario also demonstrates how the expert panel can make segments HSS and add examinations based on deterministic reasons.

There is a particularity in segment 36: six ASME selections have been placed to this segment, while there are no inspections for that segment in any other methodology. The reason for inclusion this segment in the ASME XI analysis is that in the Belgian practice the class break between Safety Classes 2 and 4 occurs downstream of the second isolation device.

Table 21 Ranking of segments and number of inspections in the Main Steam System (411).

411	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
ASME	2	2	2	3	3	3	5	1	1								
SKIFS	4	4	2	2			1	2	3	4	4	3					
EPRI base	[a]	[a]	[a]														
W4																	
W-S full																	
W-S4																	
W4 EP	1[a]	1[a]	1[a]		1	1											
W-S full EP	[a]	[a]	[a]		X	1											
W-S4 EP	[a]	[a]	[a]		X	1											
EPRI R4	4[a]	4[a]	2[a]	2	3	3	4	2	3	4	4	3					
CC N-716	[a]	[a]	[a]	1	1	1	1	1	1	1							

	18	19-21	22-24B	25-27	28-30	31	32	33	34	35A	35B	36	41	42	43
ASME												6			
SKIFS															
EPRI base										2			[a]	[a]	[a]
W4															
W-S full															
W-S4															
W4 EP										1			1[a]	1[a]	1[a]
W-S full EP										2			[a]	[a]	[a]
W-S4 EP										2			[a]	[a]	[a]
EPRI R4										2			[a]	[a]	[a]
CC N-716													[a]	[a]	[a]

	44	45	46	47	48	49 to 56	57 to 60	61 to 68	69 to 85	86 to 89	90	91
ASME												
SKIFS												
EPRI base	[a]											
W4												
W-S full												
W-S4												
W4 EP	1[a]											
W-S full EP	[a]											
W-S4 EP	[a]											
EPRI R4	[a]											
CC N-716	[a]											

([a]) Inspections added for erosion

6.3.4 Condensate system (414)

The summary of segment classification for the Condensate System is presented in Table 22.

The condensate system is a class 4 system, and thus not included in the ASME XI inspections. Also it is excluded from the SKIFS methodology.

One of the main degradation mechanisms is erosion-corrosion (Flow accelerated corrosion), and in most countries the balance of plant requires a special, so called augmented programme (e.g. USNRC Generic Letter 89-08), to address the degradation mechanism. These augmented programmes address observed degradation and the inspections tend to be targeted at locations where the most severe effects are expected. According to the US Regulatory Guide 1.178, selected augmented programmes, or parts of the programmes, may be incorporated into a RI-ISI programme provided that the licensee identifies and the staff approves the specific programmes and programme changes.

In practice, FAC is excluded from the RI-ISI process in most countries, and the analysis only points to the plant's augmented programme which should be in line with the existing regulations. In other countries, there are differing practices as to how FAC is integrated with the RI-ISI programme. In the case of Ringhals 4 plant, the FAC susceptible segments are included in the RI-ISI analyses.

A prerequisite for applying the Code Case N-716 methodology is the existence of augmented FAC inspection programme. Given the FAC augmented programme and that all Condensate piping contributes less than 10^{-6} (CDF) and 10^{-7} (LERF), the Condensate System is determined to be Low Safety Significant.

In the EPRI application, it is assumed that the Ringhals FAC programme is equivalent to an augmented programme according to industry good practices. Thus, the RI-ISI analysis is assumed not to have any impact on the inspection programme for the FAC susceptible piping.

The difference between EPRI/PWROG-SE and PWROG (original) applications related to the FAC susceptible segments appears here. For the segments 37A-48, having a high failure probability due to FAC, the PWROG application results in at least one additional inspection per segment for defence in depth on these HSS segments.

Thermal fatigue has been identified by EPRI in two segments (35A and 36A), and three welds from these segments are included in the EPRI Base and EPRI R4 RI-ISI programmes. In PWROG applications, the failure probability of these segments has been identified as high, but the RRW criterion categorises them as LSS. Based on requirements from SSM, inspections were added to these segments due to high failure potential and potential for personnel injury.

In the Swedish application the FAC susceptible segments are placed in the inspection group A. The consequence of this classification is that a qualified inspection will be required for these inspections. This is a notable difference compared to the earlier inspection programme, (as well as ASME, EPRI and PWROG (original) approaches), where the FAC susceptible piping was in the Owner defined programme, and a qualified inspection method was not necessarily used.

Table 22 Ranking of segments and number of inspections in the Condensate System (414).

414	1 to 34	35A	35B	35C	36A	36B	36C	37A to 48
ASME								
SKIFS								
EPRI base		2			1			FAC
W4								
W-S full								
W-S4								
W4 EP		[a]			[a]			24+FAC
W-S full EP		[a]			[a]			FAC
W-S4 EP		[a]			[a]			FAC
EPRI R4		2			1			FAC
CC N-716								FAC

	49 to 100	101 to 104	105 to 132
ASME			
SKIFS			
EPRI base			
W4			
W-S full			
W-S4			
W4 EP			
W-S full EP			
W-S4 EP			
EPRI R4			
CC N-716			

([a]) Based on requirements from SSM, inspections were added to these segments due to high failure potential and potential for personnel injury

6.4 Sensitivity studies

Several sensitivity studies were conducted. The first set of sensitivity studies is associated applying the different assumptions used in the EPRI Base consequence evaluation to the PWROG–SE RISMET scope, Code Case N-716 and PWROG-SE full scope applications (section 6.4.1). The other sensitivity studies are related to the effect of excluding certain segments or systems on the risk ranking in PWROG-SE application. The effect of removal of the segments with vibration fatigue from the PWROG-SE RISMET scope is summarised in subsection 6.4.2, and sensitivity to removing erosion–corrosion segments is presented in 6.4.3. Subsection 6.4.4 summarises the effect of limiting the scope to RCS system only in PWROG application.

6.4.1 Sensitivity studies based on consequence assumptions

As part of the consequence evaluation, EPRI were of the opinion that the CCDFs for RHRS and CCDPs for main steam system were conservative and that operator action should be considered for main piping in RHRS based on their experience. The results for EPRI are presented for two different cases, one based on the EPRI experience (EPRI base) and one based on the PSA results from Ringhals 4 (EPRI R4).

As a comparison, sensitivity studies were conducted to see the effects of applying these EPRI assumptions, both individually and in combination on the EPRI, PWROG-SE RISMET scope, Code Case N-716, and PWROG-SE full scope applications. The ASME Section XI and SKIFS methodologies are not included in these sensitivity studies as their methodologies are not impacted by these different assumptions. The PWROG-SE methodology is considered representative of the PWROG (original) methodology for the purposes of these sensitivity studies. Table 23 presents the different scenarios and the assumptions (variations in parameters/actions) associated with each.

Table 23 Scenarios for Sensitivity Studies Based on Consequence Assumptions.

Assumptions (Parameter or Action)	**Scenario**				
	1 (Ringhals Assumptions)	**2**	**3**	**4**	**5 (EPRI Assumptions)**
RHR CCDF	$4.00 \cdot 10^{-1}$	$1.00 \cdot 10^{-3}$ (1, 2)	$4.00 \cdot 10^{-3}$ (1, 3)	$4.00 \cdot 10^{-1}$	$1.00 \cdot 10^{-3}$
Crediting operator action to isolate leaks on RHR piping greater than 1 inch	No	No	Yes	No	Yes
Main Steam CCDP	$2.98 \cdot 10^{-2}$	$2.98 \cdot 10^{-2}$	$2.98 \cdot 10^{-2}$	$<1.0 \cdot 10^{-4}$	$<1.00 \cdot 10^{-4}$

(1) For the EPRI methodology, the RHR CCDF value is combined with the applicable exposure time (from TR-112657, a value of $1.9 \cdot 10^{-2}$ is used for a long allowed outage time) to determine the CCDP. In this case, a CCDP value of less than 10^{-4} is obtained.

(2) Represents the RHR CCDF prior to the application of the exposure time.

(3) Represents “with operator action” RHR CCDF prior to the application of the exposure time and any human error probabilities

Scenario 1 represents the PSA model used at Ringhals and is the scenario to which all others are compared. Scenario 5 represents the assumptions used in the EPRI base case. Assumptions 2, 3 and 4 represent the individual changes.

Additional information on the assumptions and their scenarios can be found in Section 5.4.1. Although these assumptions are associated with the consequence evaluation, evaluating their impact requires the risk ranking evaluation to be conducted. The risk ranking evaluation for these sensitivity studies is presented in the following paragraphs.

Due to timing and man-power availability constraints, this sensitivity study was limited to identifying the segments that would be selected for inspection (i.e. HSS segments for the PWROG-SE methodology and high or medium risk for the EPRI methodology).

Table 24 and Table 25 present the results of the consequence assumptions sensitivity studies for the PWROG-SE full scope and PWROG-SE RISMET scope, respectively, for the five different scenarios. Since it was impractical to convene the Ringhals expert panel for the various sensitivity studies, the expert panel categorisation was estimated based on experience from Ringhals earlier expert panel meetings. In Table 24 and Table 25, the quantitative risk ranking results (From Risk) and the final expert panel categorisation (From EP) are presented.

Table 26 presents the results of the consequence assumptions sensitivity studies for the EPRI methodology for the five different scenarios. The EPRI methodology does not use an expert panel for the final categorisation of the segments as high, medium or low.

Applying the various consequence assumptions for the five scenarios had no impact on the results for Code Case N-716.

Table 24 Results of consequence evaluation assumption sensitivity study for the Ringhals 4 full scope application for PWROG-SE methodology.

Plant System	Number of HSS Segments									
	Scenario 1 (Ringhals Assumptions)		Scenario 2		Scenario 3		Scenario 4		Scenario 5 (EPRI Assumptions)	
	From Risk	From EP	From Risk	From EP	From Risk	From EP	From Risk	From EP	From Risk	From EP
RCS	18	26	27	31	27	31	19	26	27	31
RHRS	12	18	0	4	12	19	12	19	0	4
Main Steam system	3	10	7	10	7	10	0	7	0	7
Condensate system	20	20	20	20	20	20	20	20	20	20
Containment spray	0	0	0	0	0	0	0	0	0	0
Safety injection	10	13	14	13	10	13	10	13	14	13
Auxiliary feed water	4	0	4	0	4	0	4	0	4	0
CVCS	33	12	38	12	33	12	33	12	38	12
Blow down system	0	0	0	0	0	0	0	0	0	0
Feed water system	0	3	0	3	0	3	0	3	0	3
Reheating system	0	0	0	0	0	0	0	0	0	0
Bleed steam system	6	6	6	6	6	6	6	6	6	6
Component cooling	2	2	2	2	2	2	2	2	2	2
Demineralised water	0	0	0	0	0	0	0	0	0	0
RWST	0	0	0	0	0	0	0	0	0	0
Service water	0	2	0	2	0	2	0	2	0	2
Fire protection	0	0	0	0	0	0	0	0	0	0
Auxiliary steam	0	0	0	0	0	0	0	0	0	0
Total	108	112	118	103	121	118	106	110	111	100

Note: Systems presented in black represent the systems in the RISMET scope of study. Systems presented in grey represent the other systems included in the PWROG-SE full scope application.

Table 25 Results of consequence evaluation assumption sensitivity study for the RISMET scope of application for PWROG-SE methodology.

Plant System	Number of HSS Segments									
	Scenario 1 (Ringhals Assumptions)		Scenario 2		Scenario 3		Scenario 4		Scenario 5 (EPRI Assumptions)	
	From Risk	From EP	From Risk	From EP	From Risk	From EP	From Risk	From EP	From Risk	From EP
RCS	24	29	41	44	41	44	32	35	41	44
RHRS	23	21	0	4	23	21	23	21	0	4
MS	7	10	11	14	11	14	0	7	0	7
CS	20	20	20	20	20	20	20	20	20	20
Total:	74	80	72	82	95	99	75	83	61	75

Table 26 Results of consequence evaluation assumption sensitivity study for the RISMET scope of application for EPRI methodology

Plant System	Number of Segments Identified as High or Medium Risk				
	Scenario 1 (Ringhals Assumptions) [1]	Scenario 2	Scenario 3	Scenario 4	Scenario 5 (EPRI Assumptions) [2]
RCS	73	73	73	73	73
RHRS	31	16	16	31	16
MS	56	56	56	2	2
CS	24	24	24	24	24
Total:	184	169	169	130	115

([1]) For the EPRI methodology, Scenario 1 is equivalent to the EPRI R4 case.
([2])For the EPRI methodology, Scenario 5 is equivalent to the EPRI Base case.

Some observations regarding the results of the consequence evaluation assumption sensitivity analysis are presented below.

Comparison of Scenarios 1 and 2 (assumption a from Section 5.4.1)
As noted in Section 5.4.1, the difference in the PSA values can be attributed to whether credit is given for auxiliary feed water systems (AFW) success being a safe end state and whether loss of RCS inventory is considered or can be restored.

- For the PWROG-SE methodology, a change in the safe condition end state for successful operation of auxiliary feed water in the PSA-model results in no quantitatively HSS segments in RHRS; however, the expert panel identified four RHRS segments as HSS for deterministic reasons. This is valid for both the RISMET and full scope applications. As part of the PWROG (original) methodology, a review for defence in depth is conducted and if the expert panel had not made any segments HSS, exams would have been added to RHRS for defence in depth. With the exception of RHRS, the RCS is most affected system by this change due to the relative ranking process. The impact of the relative ranking is somewhat offset by the expert panel categorisation process. In this instance, some of the segments that became quantitatively HSS when changing the safe condition end state had been selected as HSS by the expert panel for deterministic reasons prior to the change. This is valid for both the RISMET and full scope application.

- For the EPRI methodology, 15 segments in the RHR system are reduced from medium to low risk. This is consistent with the EPRI base case and shows the reduced importance of RHR when AFW is considered a safe end state or other compensatory measures are in place.

Comparison of Scenarios 1 and 3 (assumption b from Section 5.4.1)
The difference in these scenarios is based on whether the operator action to isolate a piping failure in RHRS piping greater than 1 inch is considered is feasible prior to uncovering and damaging the core.

- For the PWROG-SE methodology, considering operator action for RHRS has no impact on the number of quantitatively HSS segments in RHRS from the risk evaluation because these segments are quantitatively HSS due to high RRWs for the CDF/LERF “without operator action” cases. This is valid for both the RISMET scope and full scope application. The changes to the “with operator action" results lead to an increase in the number of quantitatively HSS segments in RCS and main steam system for both the RISMET scope and full scope applications due to the relative ranking process with RCS being the most impacted. Again the impact of the relative ranking is somewhat offset by the expert panel categorisation process.

- Similar to Scenario 2, for the EPRI methodology, consideration of operator actions in RHRS results in a change of 15 segments in RHRS from medium to low risk.

Comparison of Scenarios 1 and 4 (assumption c from Section 5.4.1)
The difference in these two scenarios is associated with the results from Ringhals assumption of whether blowdown of more than one or more than two steam generators will lead to core damage and allowing additional recovery actions not credited in the Ringhals PSA model.

- For the PWROG-SE methodology, a lower value for CCDPs for main steam system results in no quantitatively HSS segments for the main steam system. The expert panel made seven segments HSS for deterministic reasons. This is valid for both the RISMET scope and full scope application. Potential conservatism in CCDPs for main steam has a small impact on the other systems in Ringhals 4 full scope application by adding one quantitatively HSS segment for RCS. This change has a larger impact on RCS in the RISMET scope (eight additional segments are identified as quantitatively HSS).

- For the EPRI R4 case, as expected changing the CCDP value from above the threshold of $1.0 \cdot 10^{-4}$ to below the threshold changes 54 segments from medium to low risk.

Comparison of Scenarios 1 and 5 (assumption d from Section 5.4.1)
The difference in these two scenarios is the combination of the different assumptions used between Ringhals and EPRI.

- In the PWROG-SE methodology, for the case where successful operation of auxiliary feed water results in a stable end state, operator action is considered feasible for isolation of RHRS piping greater than one inch, and eliminating potential conservatism in the CCDPs for main steam results in adding 15 HSS segments to the RCS and changing 17 segments in RHRS and 3 segments in MS from HSS to LSS.

- For the EPRI methodology, comparison of scenarios 1 and 5 is a comparison of the EPRI R4 versus the EPRI base cases and results in a change of 69 segments from medium to low risk.

General observations
Some general observations regarding the results of the sensitivity analysis are provided below.

- For the PWROG-SE methodology, changes in the assumptions for the consequence evaluation are less sensitive for Ringhals full scope application compared to RISMET scope. The full scope application has a greater total piping CDF/LERF and the individual changes in a segment's CDF and LERF represent a smaller portion of the total CDF/LERF in a full scope application compared to a partial scope.

- In this sensitivity analysis for the PWROG-SE methodology based on the expert panel categorisation, the different assumptions in the consequence evaluation only affected the system where the assumption had an impact and the RCS (due to relative ranking).

- For EPRI methodology, a change in the consequence assumptions only affects those systems where the assumption had an impact. The classification of other segments (in other systems) as high, medium or low risk according to EPRI risk matrix is not impacted.

- Consideration of operator action can have different impacts on the final outcome between EPRI and PWROG-SE methodologies. When EPRI considers operator action it may result in a lower number of medium or high risk segments in that particular system and has no impact on the other systems. When PWROG considers operator action it is not always true that it results in a lower number of quantitatively HSS segments in that particular system because "without operator action" consequences are still considered and the expert panel provides the final categorisation of the segment as HSS or LSS considering both probabilistic and deterministic insights. Furthermore, in the PWROG methodology, considering operator actions in one system can affect other systems due to the relative ranking process.

- A change in the assumption in the consequence evaluation can be more straightforward in EPRI because it can only impact the system affected by the change in assumption. For PWROG-SE it can be a little bit more complicated because a change in an assumption in the consequence evaluation can affect the system impacted by the assumption but also other systems due to the relative ranking process. For the RISMET and full scope application, changes in the assumption for RHRS and main steam system had a small impact on the other systems and only RCS was affected by these changes. The magnitude of the impact, if any, of an assumption on other systems is dependent upon the relative importance of the system where the change (assumption) is being made.

- The PWROG-SE is not affected by small changes in the output from the PSA due to the relative ranking process. In the presented sensitivity study, the change in output from the PSA is on the order of two magnitudes (i.e. a very big change in the output from the PSA).

- In the EPRI approach, if a segment or number of segments are near the consequence rank threshold (e.g. close to 10^{-4}, 10^{-6}), there is the potential that small PSA changes could result in changes to the consequence rank (e.g. high goes to medium, medium goes to high). To minimise this potential impact, the EPRI methodology contains look-up tables (see section 3.3.3 and 3.3.6 of TR-112657) that are used to stabilise the initial RI-ISI results as well as the results from RI-ISI updates conducted in support of living programme requirements.

- Fairly significant changes in assumptions in the consequences and corresponding piping specific PSA results had no impact on Code Case N-716 results where potential impacts were identified for the EPRI and the PWROG-SE methodologies. This is because as stated earlier, piping classified as low safety significant (LSS) per Code Case N-716 contributes less than 10^{-6} (CDF) and 10^{-7} (LERF) per the Ringhals PSA.

- In the EPRI methodology, the exposure time is applied to the CCDP. Exposure time is typically associated with the likelihood of an event occurring and has no real impact on the potential consequences if the piping failure occurs. Application of the exposure time to the CCDPs results in lower CCDPs when comparing to other risk-informed programmes.

6.4.2 Sensitivity studies based on removal of segments with vibratory fatigue

In the PWROG-SE methodology, the quantitative risk ranking is done in a two step process. The first risk ranking is done with all segments. In the second risk ranking, the segments with vibratory fatigue and a failure probability $\geq 10^{-5}$ are removed. Refer to Section 2.3.3.

Due to the relative ranking process used in the PWROG-SE methodology, the removal of segments from the second risk ranking may result in additional HSS segments (prior to the expert panel categorisation). In the PWROG-SE RISMET scope of application, four RHRS segments (36-39) with vibratory fatigue are removed from the risk evaluation. Due to these four segments being dominant contributors for the CDF and LERF “without operator action” cases, ten other RHRS segments increased above the threshold for being identified as quantitatively HSS. No segments from the other three systems increased to HSS.

The four RHRS segments with vibratory fatigue were identified as quantitatively HSS based on the “without operator action” risk ranking results. The expert panel ranked these four segment LSS based on the lower “with operator action” risk results due to operator action. These segments are therefore placed in inspection group C.

NDE provides limited benefits for addressing vibratory fatigue. By conducting the quantitative risk ranking in a two step process, the PWROG-SE applications has the benefit of identifying segments that are HSS due in part to vibratory fatigue. This provides a valuable insight and allows the plant to use alternative measures to address the risk such as a vibration monitoring programme. Conducting the second quantitative risk ranking without the vibratory fatigue identifies segments that may become quantitatively HSS when the vibratory fatigue is removed thus, potentially addressing more of the plant risk. It is worth noting that removal of the segments with vibratory fatigue is a conservative means to address this as the risk contribution from due to non-vibratory considerations are not included in the risk ranking.

The PWROG methodology includes vibratory fatigue in the failure probability calculations used in the risk ranking. The EPRI, Code Case N-716, ASME and SKIFS methodologies do not include the consideration of vibratory fatigue in their risk ranking. As previously discussed, experience has shown the appropriate risk management action for locations potentially susceptible to vibratory fatigue is to evaluate design and operation changes to address this issue.

6.4.3 Sensitivity studies based on removal of segments with erosion-corrosion

As part of the Ringhals full scope RI-ISI programme, a sensitivity study was conducted to see the impact of erosion/corrosion on the risk ranking and categorisation of segments. For the sensitivity study, the segments with erosion/corrosion and that were major contributors to the risk (quantitatively HSS) were removed from the risk evaluation. In reality the segments would still have some contribution to risk if just the erosion/corrosion potential were removed. Thus the sensitivity study is conservative and demonstrates a maximum potential effect. The study was only been performed on Ringhals 3 but it is assumed to apply to Ringhals 4, too.

Nine segments in system 334 (Chemical and Volume Control System, outside RISMET scope) and 20 segments in system 414 were identified as having erosion/corrosion and were removed from the risk evaluation. All the nine segments identified in system 334 are so-called risk outliers. These segments were removed from the other risk evaluations and have also been removed in the sensitivity study.

As a result of the removal of the erosion/corrosion segments, totally 19 new segments showed up as quantitatively HSS segments, compared to the sensitivity study with vibrational segments excluded: 5 segments in system 313, 4 segments in 321, 6 segments in 323 (Safety Injection System) and 4 segments in 411.

Of the five system 313 segments that became quantitatively HSS, two of the segments had already been made HSS by the expert panel. Two of the segments were categorised LSS by expert panel based on the original quantification results, the number of exams already identified on system 313 and perceived limited additional gains in risk reduction by adding exams to these two segments. Thus, one additional segment was made HSS by the expert panel.

Of the four system 321 segments that became quantitatively HSS, all four segments were made LSS based on crediting operator action. Overall seven additional segments were made HSS by the expert panel (one in 313, two in 323 and four in 411).

6.4.4 Sensitivity studies based on RCS only scope of study

The PWROG (original) and PWROG-SE methodologies utilise a relative ranking process. When the scope of application is decreased, the amount of the overall plant piping CDF and LERF being addressed by the RI-ISI is decreased. Relatively speaking, this causes the RRWs to increase potentially making the results for a partial scope programme for a given system more conservative than the same system results in a larger scope programme (i.e. potentially more quantitatively HSS segments and potentially more segments categorised as HSS by the expert panel).

As part of the scope of application portion of the project, comparisons were made between the RISMET scope of application and a full scope application and a single system (RCS) scope of application. As expected, the difference between the RISMET scope and full scope applications was relatively small. Only three additional segments would be expected to be categorised as HSS. The difference between the RISMET scope and the RCS only scope was larger. The resulting number of inspections for the RCS only scope would be about the same as in SKIFS. 34 additional segments would be expected to be categorised as HSS, resulting in an increase of 13 volumetric examinations and 21 visual examinations. Refer to Section 3.1.3 for additional details.

6.5 Quantitative delta risk evaluations

An approval criterion for RI-ISI applications is typically that the total change in piping risk should be neutral or a reduction. The US NRC Regulatory Guide allows a small risk increase, in some other countries this may not be the case. In any case, RI-ISI applications to existing plants should include a comparison of the old and the risk-informed inspection programme to make sure that the RI-ISI application leads to an acceptable inspection programme from a safety point of view.

For the RI-ISI applications in RISMET, the acceptance criterion was set as no overall increase in risk (for the 4 systems in RISMET scope) in reference to the existing SKIFS inspection programme. The criteria for the system risk are per the individual methodology. The delta risk analyses were carried out by both EPRI and PWROG applications to verify that this criterion is met. In addition, delta risk evaluations were done against the ASME XI selection.

The criteria used by PWROG and EPRI, as well as the way to calculate the delta risk, are slightly different. Thus, it is of interest to investigate the risk impact from both perspectives. In the following we first summarise the delta risk evaluations based on the PWROG methodology, and second the approach based on EPRI principles and bounding values is applied.

6.5.1 Summary of delta risk evaluations performed by Westinghouse

Westinghouse has compared the various site selections using the PWROG methodology/delta risk decision criteria as starting point. Each ISI site selection strategy has been assumed as the “old” ISI programme in turn, and the acceptance of the selection based on other approaches has been evaluated. The acceptance criteria are basically:

- For the total plant - no risk increase
- For the dominating systems[9] – no risk increase
- For the non-dominating system – small risk increase is allowed

If these criteria are not met with the original selection, additional sites are selected until the criteria are met. The criteria have to be met for all the cases: CDF, LERF, with and without operator actions. Note that the calculations are based on PWROG methodology assumptions. Thus the failure probabilities calculated with the SRRA code are used.

Table 27 shows the number of additional inspections for each of the analysed combination.

[9] A dominating system is defined as a system that contributes 10% or more to the total plant piping CDF or LERF per the RI-ISI program.

Table 27 Number of additional inspections to meet the delta risk criteria when moving from one inspection strategy to another.

		New ISI programme					
		ASME XI	SKIFS	PWROG	EPRI base	EPRI R4	N-716
Old ISI programme	ASME XI	-	3	7	0	7	4
	SKIFS	0	-	0	0	0	0
	PWROG	29	42	-	15	0	19
	EPRI base	24	27	3	-	0	5
	EPRI R4	33	46	10	22	-	26
	CC N-716	20	22	1	0	0	-

As an example, if plant is moving from ASME XI to PWROG, after having made the base selection of sites, delta risk evaluation indicates that 7 additional sites have to be included to the inspection programme to meet the delta risk criteria.

If plant has EPRI (base) RI-ISI as “old” inspection programme and wants to move to an inspection programme based on the SKIFS selection, 27 additional inspections should be added to the current SKIFS selection to meet the delta risk criteria.

SKIFS is the only methodology that when moving to another ISI programme requires no additional inspections. The primary reason for this is how the consequences are defined in SKIFS. At a high level, the consequence index is based on the size of the pipe and how far the pipe is away from the reactor vessel. For the other risk informed methodologies, the consequence is based on the CDF and LERF for which line size and proximity to the reactor vessel does not necessarily result in higher consequences. Since the delta risk is based on the results from the PWROG methodology which uses CDF and LERF, it is only natural that the SKIFS exams would appear to address less risk. For ASME XI, while the Class of piping somewhat correlates to proximity to the reactor vessel, a higher percentage of inspections are required for the RC system and more of the RHR system is included in the scope.

As a conclusion, using the PWROG delta risk criteria and the EPRI and PWROG applications as reference, both the ASME XI and SKIFS selections should be completed with about 24-42 additional inspections to meet the criteria. Again, it must be kept in mind that these evaluations are based on Ringhals PSA assumptions and the SRRA analyses and assumptions (e.g. the inspection accuracy) may have an impact on the results.

6.5.2 Comparison based on EPRI risk matrix

Another comparison is made based on the EPRI risk matrix and bounding values used by EPRI. In this comparison, PWROG (original) and PWROG-SE full are excluded.

Table 28 shows how many inspection sites have been selected by each of these applications in each of the EPRI risk categories. The interpretation of the results for the Condensate System (414) is not straightforward due to the typical exclusion of the FAC programmes from many (e.g. Mexico, South Africa, Spain, US) RI-ISI analyses. Thus in this evaluation the system 414 has been left out of the analysis, and the results are calculated based on the three systems only. Also the erosion inspections of the Main Steam System (411) have been excluded.

From the EPRI base perspective, ASME XI, SKIFS, PWROG and EPRI R4 applications are placing many inspections in the low risk region (Medium consequence, low pipe rupture potential). These cases are related mainly to the Main Steam System, where there is a remarkable disagreement of the safety significance of many segments. The PWROG and EPRI R4 results are based on different set of assumptions than EPRI base regarding the consequence potential where the EPRI base assumed a much lower consequence potential. Thus it only stands to reason when exams identified by PWROG and EPRI R4 that many of the results would be in a low risk region. Had the delta risk evaluation been conducted on the basis of the EPRI R4 assumptions, many of these exams would have moved to the medium category and the overall results would be different.

For a quantitative comparison of the results, the EPRI bounding values for failure probabilities ($2{\cdot}10^{-6}$, $2{\cdot}10^{-7}$ and 10^{-8}) and consequence values (10^{-4} for MEDIUM and 10^{-6} for LOW) are used. For the consequence class HIGH, a value $5{\cdot}10^{-2}$ is used, since this is the highest CCDP value in the PSA results (for some segments in the system 313).

In Table 29, the total risk of sites selected for inspections for each methodology is presented, and they and then compared to the EPRI-base selection. The percentage indicates how much the ISI selection of each application carries risk in reference to the EPRI application.

The difference between EPRI base and R4 is negligible when seen from EPRI base perspective, since 50 out of the 51 additional inspections are in Medium consequence – Low failure probability class.

Even if the ASME XI selection carries 50% more risk than the EPRI selection, and even more compared to the other approaches, the delta risk criterion (change in CDF less than 10^{-7} per system) is fulfilled. In the case of PWROG selection, the difference to the EPRI selection is a CDF increase of $4{\cdot}10^{-8}$/y.

Table 28 Number of inspections in six applications shown in the EPRI risk matrix, CS system and FAC inspections excluded.

Potential for Piping Failure	Consequence Category (CCDP/CLERP)			
	NONE	LOW	MEDIUM	HIGH
HIGH				
MEDIUM			EPRI base 5 EPRI R4 6 PWROG-SE4 6 ASME 7 SKIFS 1 CC N-716 4	EPRI base 16 EPRI R4 16 PWROG-SE4 13 ASME 24 SKIFS 15 CC N-716 16
LOW			EPRI base 0 EPRI R4 51 PWROG-SE4 12 ASME 35 SKIFS 29 CC N-716 7	EPRI base 54 EPRI R4 54 PWROG-SE4 37 ASME 105 SKIFS 25 CC N-716 31

Table 29 Total risk of sites selected for inspections for each methodology and comparison with EPRI-base

	Total CDF contribution of sites included in ISI programme	Total CDF contribution compared to EPRI base
PWROG-SE4	$1.5 \cdot 10^{-7}$	79 %
ASME	$2.9 \cdot 10^{-7}$	156 %
SKIFS	$1.6 \cdot 10^{-7}$	87 %
Code Case N-716	$1.8 \cdot 10^{-7}$	94 %
EPRI R4	$1.9 \cdot 10^{-7}$	100 %
EPRI base	$1.9 \cdot 10^{-7}$	100 %

6.6 Specific issues

The risk ranking and site selection procedures and principles differ between the various approaches. Table 30 summarises a list of issues and how they are considered in the methodologies.

Table 30 Specific issues related to risk ranking and site selection.

Attribute	RISMET Application Groups			
	EPRI^{BASE}	**Code Case N-716**	**PWROG (original)/PWROG-SE**	**SKIFS-1994:1**
Risk measures used for risk ranking	CCDP and CLERP for consequence, combined with qualitative assessment of degradation susceptibility to determine risk category. 7 risk categories grouped into three risk regions (High, Medium, Low).	Initial consequence assessment mainly based on experience (order of magnitude of CCDP and CLERP), supplemented with plant-specific CCDP and CLERP information from flooding analyses. Qualitative assessment of degradation susceptibility.	Risk Reduction Worth used as primary quantitative risk measure. Risk Achievement worth (RAW) is also calculated. RRW and RAW are based on piping CDF and LERF with and without operator action. Segments are classified as high or low safety significant by expert panel based on probabilistic and deterministic insights.	Risk ranking into three inspection groups (A, B and C) based on qualitatively determined consequence index and damage index.
Definition and treatment of risk outliers	No specific treatment needed with the methodology	Same as in EPRI methodology	No specific requirement in the original PWROG process, but plants have checked for outliers. Risk outliers can be defined as segments contributing 20% or more to total piping CDF/LERF, or RRW > 1.25. Possible outliers can be excluded from the analysis and new results be recalculated. Results are evaluated by the expert panel. PWROG-SE uses the risk outlier guides as a requirement.	No
Treatment of uncertainties	Methodology considered robust (large threshold margins for risk regions), not needing specific uncertainty analyses. In delta-risk evaluation alternate ways to treat failure probabilities (best estimates, bounding values, POD considerations).	Same as in EPRI methodology	Uncertainties are considered in calculation of failure probabilities and in risk evaluation by considering variations in pipe failure probabilities and CCDP/CLERP.	No
Use of expert panels	Risk ranking: Not required due to the process driven approach. Site selection: the element selection function uses people from relevant disciplines to choose the elements to be inspected.	Same as in EPRI methodology	Expert panel is a part of the methodology. Reviews the initial risk ranking and makes the final determination of HSS or LSS. In addition the expert panel reviews the elements to be inspected and concurs on the scope of the programme.	No

Table 30 (cont.) Specific issues related to risk ranking and site selection.

Attribute	RISMET Application Groups			
	EPRI[BASE]	**Code Case N-716**	**PWROG (original)/PWROG-SE**	**SKIFS-1994:1**
Treatment of Low Probability / High Consequence sites/segments	These segments fall in the risk category 4, being part of the risk region Medium, and are subject of 10 % inspection coverage.	Based on generic insights a number of sites/segments are included in the identification of high safety significant locations (e.g. LOCA, LOCA outside containment, Break Exclusion Regions). The element selection process requires that a representative sample of these locations be selected for inspection.	Segments can be categorised as HSS based on having a high consequence by the expert panel, if they originally are not HSS based on the RRW. Statistical model is used to determine the minimum number of inspection locations on HSS segments with low probability/high consequence. PWROG-SE does not use statistical model instead a number of written rules are used to place the segments into right category (as an example segments in inspection group B will have at least 10 % of the structural elements inspected).	For devices in a range of 5 m from containment penetrations and/or isolation valves there is a requirement for special consideration (Consequence index 2)
Treatment of High Probability / Low Consequence sites/segments	10 % of High Probability segments with Low Consequence are inspected from practical perspective. If there are no consequences identified, no inspections are needed. Usually for locations susceptible to FAC 100% inspection required.	Requires that programmes be in place for FAC and localised corrosion and inspections conducted accordingly. Also, requires that plant specific service experience (flaws, cracking, failures) be reflected in the final inspection population.	Segments can be categorised as HSS based on having a high probability by the expert panel, if they originally are not HSS based on the RRW. 100% of the HSS structural elements that have a high probability are selected for inspection. LSS segments with high probability are recommended for owner defined programme or may be inspected as part of an augmented programme. For PWROG-SE low safety significant, high failure importance segments should be considered for selection in inspection group B (10% sample) if in addition to a high failure frequency there is high risk for personnel injuries.	No requirements or guidance

Table 30 (cont.) Specific issues related to risk ranking and site selection.

Attribute	RISMET Application Groups			
	EPRI^BASE	Code Case N-716	PWROG (original)/PWROG-SE	SKIFS-1994:1
Site selection process	25 % of high risk welds and 10 % of medium risk welds are selected for inspection. Selection shall include all relevant degradation mechanisms and degradation mechanisms combinations. Augmented programme can be integrated with the RI-ISI selections, where appropriate. System leakage testing required for high, medium and low risk welds.	10 % of high safety significant welds are selected for inspection. Selections preferentially allocated to welds susceptible to a degradation or degradation mechanism combination as well as high consequence. Augmented programme can be integrated with the RI-ISI selections, where appropriate. System leakage testing required for HSS and LSS welds.	Segments are placed into five regions (1a, 1b, 2, 3, 4) based on expert panel ranking and failure probability. 100 % inspection for region 1a, and a statistical number of inspections for regions 1b and 2. Owner defined programme recommended for region 3. Pressure tests maintained for region 4. In PWROG-SE, the SKIFS inspection groups are used instead of PWROG (original) regions. 100% of group A, 10% of group B and owner defined programme for group C.	Use the guide in appendix 3 of SKIFS 1994:1 to decide inspection location and the extent of inspection.
Consideration of other criteria, e.g. radiation dose, accessibility, inspectability	The element selection function uses several criteria to select the inspection elements within the risk region.	Same as in EPRI methodology	Not considered in risk-ranking. These are secondary considerations for the selection of the inspection locations.	Dose and availability are considered when it comes to defining the extent of inspections.
Leak detection	Leak detection not credited	Same as in EPRI methodology	Leak detection is only considered when conducting the delta risk evaluation. For those segments where leak detection is credited, the failure probabilities with leak detection are used.	Leak detection not credited
Risk acceptance and other criteria for the application	Delta risk acceptance criteria following NRC regulation (in US applications) or country specific acceptance criteria. Base methodology limits increases in risk to less than 10^{-7} (CDF) and 10^{-8} (LERF) per system.	Delta risk criteria following NRC regulation (in US applications) or country specific acceptance criteria. Base methodology limits increases in risk to less than 10^{-7} (CDF) and 10^{-8} (LERF) per system.	Overall change in risk should be negative or neutral. No risk increase allowed for risk dominant systems (contributing >10% to the total risk). For non-dominant systems a small increase is allowed.	No numerical risk criteria. The majority of components within inspection group A shall be inspected. In group B, a well balanced sample inspection may be sufficient. The sample should contain at least 10 % of the components within inspection group B.

6.7 Concluding remarks on risk ranking

Risk and safety importance rankings of various applications cannot be straightforwardly compared, since the regions (safety class, risk region, inspection group) are determined differently. As an example, the PWROG methodology results in smaller number of segments ranked as HSS compared to segments in EPRI Medium and High Risk regions. On the other hand, at least one inspection is assigned to each HSS segment, while in the EPRI approach not even all High Risk segments are covered by inspections. The term “segment” is used in the EPRI approach strictly as an accounting tool (i.e. useful for streamlining the analysis and documentation process) and not as a technical component of the methodology itself. Despite the differences, the safety and risk classifications were considered useful in illustrating and comparing the results at system and segment level.

The RI-ISI methodologies allow a reasonable flexibility in selecting the inspection locations among the high risk sites. Since the element selection meeting was excluded from the EPRI applications (see limitations of applications in section 2.4), it is impossible to judge whether the final selection of inspection locations would have been as close to each other between EPRI and PWROG applications as it is in this benchmark exercise.

The results of the RI-ISI applications were provided as tables with a limited amount of other documentation, so it is difficult to judge or compare the transparency of the approaches. In principle, the PWROG expert panel records should provide explanations and additional information, but often the records are very brief. In the case of EPRI application, the assumptions on e.g. operator actions seem to have sometimes a significant effect on risk ranking. EPRI TR-112657 provides direction on how and when operator action can be credited and guidance on the documentation of analyses. However, it could not be reviewed how the assumptions are documented in practice.

When comparing the different applications, it can be noted that all risk-informed approaches would result in significantly fewer (28-47) inspections in the Reactor Coolant System than the ASME XI application (113). The PWROG applications result in the lowest number of inspections in the RCS. As the PWROG methodology uses a relative ranking, it was of interest to study how much the results do change if the scope is limited to RCS only. When the application is limited to this single system, the number of volumetric inspections increases from 28 to 41, which is close to the number of inspections in both SKIFS and EPRI applications.

For the results of the Residual Heat Removal System, there are some significant differences in the ISI scope between the risk-informed applications, even if there is large agreement on the low safety significance of most of the segments. The SKIFS and Code Case N-716 applications would result in only few inspections. In other applications, inspection needs are mainly driven by the high consequences associated with the underlying assumptions (see section 5.4.1).

For the Main Steam System, there is generally a good accordance of the low safety significance of most of the segments. There is however a significant difference in the number of inspections: SKIFS, ASME and EPRI R4 applications result in about 30-40 inspections, while the other applications assign not more than 10 inspections to the system (erosion inspections excluded).

The Condensate System is excluded from ASME XI and SKIFS inspections. Most of the applications would require only FAC inspections. The PWROG application makes an exception, since in the methodology additional inspections for defence in depth are required for the segments with high failure probability (due to FAC in this case).

Sensitivity analyses were carried out to investigate the impact of some major different assumptions with respect to consequence assessment, vibratory fatigue, and FAC. For the various methodologies, these assumptions may affect in various ways the number of safety significant segments as well as the number of sites to be included in the inspection programme.

The delta risk analyses showed that the risk impact assessment is not an unambiguous process. The results are dependent on the assumptions made on failure probabilities and consequence measures. On the other hand, the contribution of pipe failures to the CDF and LERF are very small, and a risk criterion allowing a small risk increase is relatively easy to fulfil.

The drawback of having small contributions from pipe failures on CDF and LERF is that often, in basic PSA studies, there has not been a particular drive to avoid unnecessary conservatism. To reduce the resources needed to carry out a PSA study, conservatism is accepted provided that this does not affect the main result. Therefore, to assure a correct risk ranking in plant with low CDF and LERF for pipe failures, it is of great importance that the PSA study is assessed to avoid all such conservatism affecting the consequences of pipe failures.

If absolute risk scales are based on getting a "reasonable" number of inspections with the current PSA, developing a more realistic PSA or newer plants with enhanced safety features may have significantly fewer inspections due to the lower CDF and LERF values. For example, at least one next generation nuclear power plant has a total plant CCDF less than 10^{-6}. Using a change in risk criterion of CCDF less than 10^{-6} as acceptable would mean that no inspections would ever be added since performing no inspections would still meet the criterion. It is not the aim of this report to judge whether this is acceptable or not.

The risk impact is only one aspect of the RI-ISI process, and the benefit of a RI-ISI application is not straightforward to measure. It should be understood that the RI-ISI process itself is a valuable exercise, since it forces the project team to review the piping degradation potential and identify both direct and indirect consequences of piping failures. The process typically results in reduction of doses, and may identify more efficient inspection procedures. If the possible savings in inspection costs are directed to safety management activities of other, more risk-important targets, a better balance in plant risk management is achieved.

7. REGULATORY ASPECTS

7.1 General aspects

As reactor facilities continue to age, it becomes more important that adequate inspections are conducted to ensure that components are capable of performing their function and thus, that safety is sufficiently maintained. Experience in the US has shown that risk-informed methods can reduce costs and resources through the elimination of low-risk, low-consequence inspections and concentrating resources on pipe systems associated with greater risks where damage mechanisms have been identified. Experience in Europe has also been positive where risk-informed methods have been used, but the industry has been more measured, proceeding in stages as the methods are further developed and the requirements of the regulatory bodies change.

In-service inspection is one of the primary tools in the management of age-related degradation in nuclear power plants. This effort has become increasingly critical to the safe operation of nuclear plants as the units continue to age and are being approved for license renewal. Materials degradation of components in nuclear power reactors has been experienced since the inception of nuclear power production. History and operating experience have shown that new forms of degradation, or variants of existing forms, continue to arise as time progresses. These degradation problems can result in significant operational and potential safety problems which may require substantial investments in research, mitigation, repair, replacement, and subsequent frequent inspection with correspondingly strong regulatory involvement and actions. Dependable in-service inspection programmes are one tool that can be used to deal with materials degradation in the management of resources.

In the United States, the Probabilistic Risk Assessment (PRA) Policy Statement [46] formalised the US Nuclear Regulatory Commission (NRC) commitment to risk-informed regulation through the expanded use of probabilistic risk assessment (PRA). The PRA Policy Statement states, in part, the use of PRA technology should be increased in all regulatory matters to the extent supported by the state of the art in PRA methods and data, and in a manner that complements the NRC deterministic approach and supports the NRC traditional defence in depth philosophy. Methodologies were developed by the Westinghouse Owners Group (WOG) and the Electric Power Research Institute (EPRI). The NRC reviewed and has approved these methodologies. Nearly all of the 104 nuclear power reactors in the US have now implemented one of these methodologies for the risk-informed selection of components for inspections. Nuclear plant owners have chosen to implement the methodologies selectively on a system basis or Class of piping basis (e.g. Class 1 only).

In Europe, work on risk-informed selection of components for inspections has progressed more slowly. For example, pilot studies have been performed in Sweden, Finland, Spain, Switzerland, Czech Republic and Ukraine. The Nuclear Safety Council (CSN) in Spain has approved the use of the PWROG methodology, and it has been applied in 4 PWRs and 1 BWR. In Sweden, Ringhals AB uses an application of the PWROG methodology for the majority of the pipe systems in Ringhals 2-4 (known as the RIVAL project). Recently, SSM (former SKI) approved the application for Ringhals Unit 2-4, but with some additional requirements compared with the original PWROG procedure. It should be noted that relative to the scope of a risk-informed ISI programme, SSM is recommending a full scope (Class 1, 2, 3 and many Class 4 systems) programme since it has been shown that even Class 4 systems (with FAC) can represent high risk segments. The basis for this position is that Sweden does not rely on augmented programmes (programmes implemented by licensees to address particular degradation mechanisms) as is the case in the US for example.

The nuclear power plant owners have formed the Task Group on Risk within the European Network for Inspection and Qualification (ENIQ). This group is charged with formulating its view of risk-informed selection of components for inspections. Recommendations from the task group are reported in a Framework Document [5]. In Europe the Nuclear Regulators' Working Group (NRWG) decided in November 1996 to set up a task force (TF) to agree on the philosophy and principles governing RI-ISI of mechanical components of Nuclear Power Plants in order to maintain sufficient margins against leakages and failures, considering dose exposures to the public. The TF produced in 2004 a "Report on the regulatory experience of risk-informed in-service inspection of nuclear power plant components and common view" (EUR 21320 EN). The common views, as stated in this document, are agreed in general by all the participating regulatory bodies of the European Union, or their representatives. They express what should be considered by the regulatory body when establishing guidelines on RI-ISI and assessing licensees' RI-ISI applications. They are not to be understood as a regulatory commitment to the introduction of RI-ISI programmes. Whether RI-ISI programmes are introduced is and will be decided at the national level.

Frameworks have been developed for applying PSA in reactor regulation. The purpose of a framework is to provide a general structure to ensure consistent and appropriate application of PSA methods in regulating nuclear reactors. The objectives of a framework are: (1) enhance safety by focusing regulator and licensee resources in areas commensurate with their importance to health and safety; (2) provide a model for using risk information in all regulatory matters; and (3) allow use of risk information to provide flexibility in licensing and operational areas.

Policies have been established by several regulators to address issues related both to risk assessment [i.e. the variety of assessment methods that are now in use or could be used (and how these are or might be used) in risk-informed regulation] and risk management (i.e. the establishment of metrics and goals for risk to appropriate individuals or groups). For example, it is important to recommend a quantitative inspection goal. This means not just requiring risk reduction (or risk neutrality) when compared to a previous inspection programme but also recommending a risk reduction factor (perhaps 5-10) compared to not performing inspections. This ensures that the highest risks are reduced to a level as low as can be reasonably achieved. Addressing issues relative to risk assessment and risk management are important to meet policy objectives in regulating nuclear power plants (i.e. safety goals for the operations of nuclear power plants).

Policies have also been established providing general guidance on the regulatory use of risk assessment methods in a risk-informed regulatory framework. Implementation of this general guidance can be accomplished by a variety of approaches involving regulatory staff and licensee use of risk insights and risk assessment in regulatory decision-making. In each case, there are two principal considerations:

(1) What specific use is the regulatory staff expected to make of risk insights and risk assessment in development of regulations and guidance, licensing, inspection, assessment, and enforcement?

(2) What specific use is the licensee expected to make of risk insights and risk assessment in planning and conducting its operations?

A number of factors are important to these two considerations. They relate primarily to what can be gained in terms of safety and reduction of regulatory burden versus the cost of implementation. These factors include: hazard and complexity of the activity, degree of human involvement in the activity, technical sophistication of the licensee community, regulatory staffing, and training issues.

In any regulatory decision, the goal is to make a sound safety decision based on technically defensible information. Therefore, for a regulatory decision relying upon risk insights as one source of information, there needs to be confidence in the PSA results from which the insights are derived. Consequently, the PSA needs to have the proper scope and technical attributes to give an appropriate level of confidence in the results used in the regulatory decision-making. The main goal of the PSA review is to identify weaknesses in the PSA that could be relevant to a decision so that they can be properly considered.

The transition to a risk-informed regulatory inspection framework has been incremental. It has taken time to develop alternatives to the deterministic and prescriptive requirements. Also, the estimated resource savings must be balanced against the programme development costs. The purpose of in-service inspections is to verify that no known damage mechanisms are compromising reactor safety, operational safety and personnel safety and also to monitor known degradation before it is a safety problem. In addition, in-service inspections are used to verify that no unknown damage mechanisms (or known damage mechanisms that appear in unexpected locations) are compromising safety. Risk-informed in-service inspection (RI-ISI) approaches are increasingly being used to determine which areas are the most important from the point of view of safety and where inspection resources should primarily be focused. The principle of RI-ISI programmes is to focus the selection on components that are risk significant per the respective methodology but to also take into account other aspects such as defence in depth. Results from the application of RI-ISI in the US have shown that, from the standpoint of risk, inspections of secondary systems where active damage mechanisms such as vibration fatigue or erosion corrosion exists can be more important than inspections of some primary coolant reactor components. Thus, it has been possible to eliminate the inspection of many components determined to be low-risk. The analyses show that this has been possible while maintaining or reducing the level of risk. In addition, significant cost reductions have been realised.

7.2 Recommended guidelines

From a regulatory perspective for the RISMET project, the US, Swedish and Canadian regulatory bodies are in agreement on the following seven items that an RI-ISI programme should:

1. Use the risks of core damage (CDF) and release of fission products (LERF), ranked in relation to one another, to form the basis of inspection groups. Inspections should be specially focused on areas with active damage mechanisms, i.e. mechanisms where damages can initiate and grow during operation. This means that the highest risks should correspond to active damage mechanisms.

2. Take into account risk outliers so that these do not distort the selection of components. This means that risk outliers should be separated from the risk profile before making the risk ranking on which the selection of components is based (see Figure 13).

3. Take into account areas with high consequences regardless of risk. This means that areas with high consequences should be included in the selection of components for inspection even if they are not associated with the highest risks (see Figure 13).

4. Take into account areas with a high probability of leak and rupture if they also may result in serious personnel injuries. These areas should be included in the selection of components for inspection even if they are not associated with the highest risks (see Figure 13).

5. Consider deterministic insights, for example, by using expert panels. This means that other aspects, which cannot be expressed through numerical risk values, should also be used to guide the selection of components for inspection, e. g. assessment of susceptibility to damage and consequences on the basis of operational experience and the benefit of certain operator actions.

6. Propose non-destructive examination methods (NDE) which give a significant risk decrease for the damage mechanisms present, especially for the most risk-significant areas. Be integrated with performance demonstration or qualified inspection procedures. For example, probability of detection curves and sizing error estimations are key inputs to the probabilistic fracture assessment. The effectiveness of inspections and POD play a significant role when it comes to assess whether the proposed risk-informed ISI programme will reduce risk sufficiently. For this reason, it is important that the quality of the NDE should be demonstrated. This means that if high risks in certain areas mainly depend on the occurrence of vibration fatigue as the only damage mechanism and where NDE is not expected to give any decrease in risk, such areas should be separated from the risk profile before making the risk ranking on which the selection is based. For such areas special mitigation programmes should be developed using methods other than NDE to effectively decrease risks. This may also apply to areas subjected to steam-hammer or water-hammer loads.

7. Preferably result in a decrease in accumulated radiation exposure to the inspection personnel compared with the present inspection programme. This means, for example, that when choosing between different inspection areas associated with similar risks, factors such as radiation exposure may be decisive.

There is a slight difference, however, between the US and European approaches on an eighth point regarding an RI-ISI programme.

8. **(a)** In the US, Regulatory Guide 1.174 [47] discusses the risk-acceptance guidelines to be used. Acceptance guidelines for CDF are divided into three regions (see Figure 3 in the guide). The guidelines are intended for comparison with a full-scope assessment of the change in risk. Guidelines are provided to address full-scope and partial-scope PSAs. Application of RI-ISI programmes resulting in a decrease in CDF or where the calculated increase in CDF is very small (less than 10^{-6} per reactor year) are acceptable. Small increases in CDF, up to 10^{-6} per reactor year, are allowed if the total CDF is less than an approximate value "substantially above" 10^{-4} per reactor year. Figure 3 in Regulatory Guide 1.174 shows that this is around 10^{-3} per reactor year. When the calculated increase in CDF is in the range of 10^{-6} to 10^{-5} per reactor year, the NRC will only consider RI-ISI programmes if it is shown that the total CDF is less than 10^{-4} per reactor year. RI-ISI programmes that would result in increases in CDF above 10^{-5} per reactor year will not normally be considered.

   (b) European regulatory bodies, on the other hand, are in most cases only allowing an overall risk decrease or risk neutral change. This is discussed in the Nuclear Regulators Working Group (NRWG) report [4]. Under this approach, the RI-ISI programme should result in a decrease in risk or at least be risk neutral when compared to the present inspection programme. This means that the risks (sum of CDF and LERF for all piping systems) should decrease risk or be risk neutral. Locally at a system level, small risk increases can be acceptable for low risk systems. Note that even if the present inspection programme is generally assessed as being reasonably efficient, more detailed analyses may demonstrate that the earlier inspection programme and the inspection methods have not given a sufficient decrease in risk. In such cases, the RI-ISI programme should give a substantial decrease in risk compared with the present programme.

There is another item of note. Degradation of small diameter piping has been reported in US pressurised water reactor designs as well as in CANDU feeder piping. The reported mechanisms have included flow accelerated corrosion, stress corrosion cracking, and thermal and fatigue cracking. Small diameter piping should be included in a RI-ISI programme as it is exempted from ASME non-destructive examination requirements. Also, the programme should consider the consequences of loss of inventory. For example, an un-isolated small diameter pipe failure on a closed loop system may cause a significant loss of inventory, resulting in the loss of a train in the system or the entire system, or resulting in the failure of sump recirculation and a direct release path outside containment if the piping failure is outside containment.

The NRC has published a report, NUREG/CR-6813 [48]. The report was prepared by Karl N. Fleming and was published in April 2003. Based on the author extensive experience and information gathered from PSA practitioners, NRC staff, and selected industry representatives, a set of recurrent issues that arise in the use of PSAs for risk-informed decision making was identified. The NUREG/CR report grouped the issues as following:

- Use of limited-scope PSAs in risk-informed applications submitted in accordance with Regulatory Guide to quantify full-scope metrics.
- Lack of completeness within the specified scope.
- Model-to-plant fidelity issues.
- Lack of, or inadequate, treatment of uncertainties.
- Quantification issues (e.g. error due to cut-set truncation).
- Multi-unit site modelling issues.
- Lack of treatment of ageing effects.
- Issues with the use and interpretation of risk metrics.
- Lack of coherence between probabilistic and deterministic safety approaches.

An incomplete technical basis was cited as the most important area for improvement. Issues identified included:

- Lack of criteria and inconsistent criteria for evaluating the impact of missing elements in scope on the application of RG-1.174.
- Lack of acknowledgment or consideration of limitations in the PSAs used in submittals.
- Inadequate justification and documentation for screening events from a PSA.
- Lack of incorporation of operating experience in PSAs.
- Inadequate treatment of common-cause failures.
- Lack of detailed review by plant personnel to ensure fidelity with plant systems, operator actions, etc.

Fleming noted that while the exclusion of portions of a full-scope PSA model can be technically valid, resources must be continually expended by both NRC and its licensees to determine the validity of decisions based on an incomplete model. This raises the question whether, in the end, it might be less burdensome to develop a full-scope PSA.

Once the issues were identified, actions were taken to begin addressing them (e.g. Regulatory Guide 1.200 [49]).

As previously discussed, in Europe the Task Group on Risk within the European Network for Inspection and Qualification (ENIQ) formulated its view of risk-informed selection of components for inspections which were reported in [5]. The NRWG Task Force on RI-ISI that developed guidelines for risk-informed selection of components for inspections was mentioned previously. The two groups are in agreement on most of the basic principles that should apply to risk-informed selection of components for inspections. NRWG final report specifies five main principles. These are:

1. The introduction of risk-informed in-service inspection must be in accordance with the legal and regulatory framework in the European countries.

2. A risk-informed inspection programme must ensure that sufficient account is taken of the defence in depth argument.

3. A risk-informed inspection programme must maintain sufficient safety margins against leakage and rupture.

4. Risks should be reduced via inspection to a level determined by national regulations. In connection with a transition to risk-informed selection of components for inspections, a risk reduction or at least risk neutrality should be obtained.

5. A risk-informed inspection programme must be constantly kept up-to-date with respect to new information that is gained.

7.3 Specific issues

7.3.1 Scope of application

The scope of an application of RI-ISI should preferably be a full scope which is defined in section 3. The reason for this is that important components, which could be risk significant, may be excluded in a partial scope. Experiences when using a full scope, e.g. from the Swedish PWR Ringhals 2 [17], have shown that certain segments in class 4 with FAC were among the highest risk significant segments in the entire scope. These segments would not have been identified as high risk significant with a partial scope.

However, it is realised that a partial scope can be justified, especially if it is used for a well-defined class of components for which one wants to investigate the usefulness of different inspection alternatives. But every restriction of the scope of application should be well justified. The fact that a specific pipe system is not included in the previous (non risk-informed) inspection programme or that a certain pipe system is not included in the plant specific PSA, does not in general represent a sufficient condition to exclude it from the scope. Exempting piping system from the scope can be questionable especially when there are no augmented inspection programmes in place, such as for FAC and IGSCC in the US. Moreover, segments with vibration fatigue were identified as belonging to the top risk significant segments in the Ringhals 2 study. Realising that NDE will not reduce the risk for such segments, these high risk segments with vibration fatigue were then removed from the risk profile before the final risk ranking was performed. A special mitigation programme using other techniques than NDE was then implemented to reduce the risks for these segments. However, it is important to include such segments in the original risk evaluation in order to be able to identify segments which are high risk segments with vibration fatigue as the dominant damage mechanism. For the RISMET study, vibration fatigue was included in the PWROG (original) and PWROG-SE methodology. In the latter methodology, the high risk segments with vibration fatigue were indeed identified, by evaluating the failure frequencies and the consequences from the PSA, but then removed from the risk profile before the final risk ranking was performed. This was a regulatory requirement. In the EPRI methodology, vibration fatigue is not considered, also with reference to the fact the NDE will not be effective to reduce risks in such segments. However, it then important that high risk segments with vibration fatigue can be effectively identified and mitigated by a special mitigation programme. It is not clear how high risk segments with vibration fatigue are identified outside the EPRI methodology

In the RISMET study, a partial scope with 4 systems was used. For the PWROG (original) methodology this resulted in more conservative results compared to a full scope application. However, this is not necessarily true in all cases. Imagine a segment in a partial scope being a low risk segment which in a full scope would be classified as having a higher relative risk. This would be the case if the full scope would include many segments representing even lower risk values than the partial scope segments.

7.3.2 Expert panels

RI-ISI programmes can enhance overall safety by focusing inspections of piping at HSS locations and locations where failure mechanisms are likely to be present. These programmes also can improve the effectiveness of inspection of components through a focus on personnel qualifications and inspection for cause. Finally, RI-ISI programmes can enhance overall safety through the use of the expert panel. The expert panel process compensates for the inherent weaknesses of PSA implementation approaches resulting from the PSA structure and limitations in the meaning of the importance measures. The expert panel, which makes many of the final determinations in the RI-ISI process, plays an integral role in determining the quality of an RI-ISI programme.

Documents are publicly available discussing the regulatory issues raised during the review of proposed RI-ISI programmes. A number of these issues specifically address the expert panel process. Following is a list of some of the more important regulatory issues relative to the expert panel process:

- The panel members should have been subjected to a training programme under the supervision of people with long experience of the role of expert panels in RI-ISI.

- The structure of people within the expert panel is such that every person's vote is equal and that the panel members should represent all types of competence necessary for RI-ISI.

- It is important that the expert panel is allowed to make an unconditional assessment of all the data without any biases against e.g. cost restrictions.

- Adequate justification is required by the expert panel to lower the classification of HSS segments, and the justification is to be documented as part of the programme. The expert panel should be focused primarily on adding piping segments to the higher classification.

- The expert panel's records must be retained on site and be available for review by the regulatory authorities having jurisdiction at the site.

- The Perdue Model is intended to be used on highly reliable piping to establish a statistically relevant sample size and verify the condition of the piping. In cases where an active degradation mechanism exists, particularly where there is an ongoing augmented programme, it is inappropriate to use the Perdue Model for element selection. In these cases, the expert panel must apply other rationales for selecting the number of elements to examine. Objections have been raised against the Perdue Model, see [17] for the PWROG-SE application of the Ringhals PWR unit 2. In the PWROG-SE application, the Perdue Model is not used. Instead a statistical sample of at least 10 % of a well defined group of segments should be used.

- Plant functions are considered in the expert panel review process during the consequence evaluation. The safety functions of the system and piping segments to be reviewed are to be presented to the expert panel to ensure that the expert panel specifically addresses the relationship between the systems and piping being evaluated and their associated plant safety functions. The expert panel is expected to consider the importance of these functions for scenarios not modelled in the PSA so that the categorisation of safety significance of the pipe segments reflects all plausible accident scenarios.

- Structural reliability and risk assessment models or other probabilistic fracture mechanics codes are used to estimate the failure probabilities for important components. These codes are used because they can provide a higher level of detail than estimates based on historical data or expert judgment. Limitations in the use of the codes relative to the modelling of IGSCC, lack of benchmarking of erosion-corrosion models compared to existing erosion-corrosion programmes, and lack of modelling of complex geometries have been noted. Thus, the expert panel must pay particular attention to ensure that the results from the codes seem appropriate. It should also be noted that the ability of such codes to estimate failure frequencies is limited by the quality of the input data and modelling limitations inherent in the code itself. Providing bounding or conservative inputs to the model or relying on the conservative nature of certain aspects of the code can potentially lead to inappropriate conclusions regarding the relative susceptibility to failure of various piping segments and components. Therefore, it is extremely important that these limitations be recognised by the user of the code and by the expert panel and that the results of the analyses are carefully scrutinised to assure that they make sense when compared to engineering knowledge of degradation mechanisms and plant specific and generic operating experience.

- The expert panel review of risk information should address both active and passive functions and structures, systems, and components (the review ensures that the element(s) will still perform their intended safety function during subsequent operation).

- The expert panel should consider: spatial effects as well as direct; the failure of the structure, system, or component on its safety significant function, and aspects modelled in the PSA and not limit its review to only those aspects not modelled in the PSA. Thus, the failure of the structure, system, or component should not fail a safety significant function and failure of the function or structure, system, or component will not directly or indirectly (e.g. spatial effects) fail another safety significant function or structure, system, or component, including those that are assumed to be inherently reliable (e.g. piping and tanks) and those that may not be explicitly modelled in the PSA (e.g. room cooling systems and instrumentation and control systems). Also, the expert panel should consider functions and structures, systems, and components that are necessary for significant operator action required to mitigate accidents and transients, regardless if they are in the PSA or not.

- Following are some other items relative to functions and structures, systems, and components (function/SSC) that an expert panel should review to ensure adequate resolution of issues: 1) failure of function/SSC will not prevent or adversely affect the plant's capability to reach or maintain safe shutdown conditions and is not significant to safety during mode changes or shutdown; 2) the function/SSC does not act as a barrier to fission product release during severe accidents; 3) the function/SSC does not support a significant mitigating or diagnosis function for accidents and transients; and 4) failure of the function/SSC will not result in releases of radioactive material that would result in the implementation of off-site emergency response and protective actions.

7.3.3 Inspection effectiveness and inspection intervals

When the risk significant segments have been identified and a selection of components for inspection has been made, the next step is to choose an appropriate inspection method and inspection interval in order to obtain a sufficient risk reduction, especially for the high risk locations. In this process, the concept of POD is of key importance. POD is normally given as the probability to detect a crack as function of crack depth. The risk reduction factor can be defined as 1/ (1-POD). Reliable POD-values are not easy to obtain. It involves a series of many blind tests for test blocks with different crack sizes and a statistical treatment to produce the POD with a certain degree of confidence. An example of a study to obtain POD for manual ultrasonic testing of piping can be found in the report by Jelinek et al [50]. Other efforts aimed at producing POD curves using expert elicitation methods are under way, e.g. Shepherd et al [51]. In the RISMET study, only the PWROG methodology involved assumptions of POD. No details have been released within RISMET about the features of the POD curves used. However, the selected results presented in section 4.3 indicate that unrealistic POD-curves have been used. A more detailed discussion about the regulatory aspects of the POD-assumptions used for the Ringhals PWR units is given in the Swedish regulatory review of PWROG-SE [17]. From a regulatory point of view, it is important to use realistic POD-curves since they provide the input to how much the risk will be reduced with the proposed inspection techniques and this is one aspect of judging the effectiveness of the inspection programme. On the other hand, all other RI-ISI methodologies besides the PWROG methodology do not require detailed POD-information and it is then more difficult to assess the risk reduction capabilities. This is a disadvantage of these more qualitative procedures.

The other factor besides POD which has an impact on the risk reduction is the inspection interval. This has not been a variable in the RISMET study. For example, it has been assumed in the PWROG methodology that inspection is performed every 10 years but with the intent that more detailed inspection intervals will be determined by using deterministic evaluations. In Sweden so far only deterministic evaluations have been performed. In these evaluations a starting crack, large enough to be reliably detected with the inspection method used, is postulated in the piping component. Then a crack growth evaluation is made and the inspection interval is determined such that the crack size at the next inspection is still acceptable with appropriate safety factors. This deterministic procedure has been successful in Sweden in the sense that no unexpected leaks have been experienced in components where this procedure has been applied. In general, the inspection interval can be included in a quantitative risk evaluation which means that it is determined by a probabilistic evaluation. However, with only a global risk acceptance criterion for the risk between the new and old inspection programme, there is a risk that individual high failure probability components will have large inspection intervals with a probabilistic evaluation and still fulfilling the global risk acceptance criterion, whereas a deterministic evaluation would result in a much shorter interval. This means that there may be an increased risk for undesirable leakages which in turn would violate one of the NRWG key principles, to maintain sufficient safety margins against leakage and rupture. Thus from a regulatory point of view, for the present it is recommended to use deterministic evaluations for determining inspection intervals. However, it is important to further explore the role of probabilistic methods to determine inspection intervals.

7.3.4 Living RI-ISI programme

Regulatory requirements require the periodic update of RI-ISI programmes. This is to ensure that the performance monitoring and feedback provisions are addressed. The update feature has also been referred to as a living programme. Implementation requires feedback of new relevant information to ensure the appropriate identification of safety significant piping locations. It is not clear that the periodic updates have always been implemented in a consistent adequate manner.

The performance monitoring and feedback provisions should evaluate changes in safety-significance and inspection requirements due to plant design feature changes, plant procedure changes, equipment performance changes, new information on damage mechanisms, probabilistic models, inspection effectiveness and examination results including flaws or indications of leaks. The primary goal is to ensure that no adverse safety degradation occurs because of changes. This periodic review and adjustment of the risk-ranking of segments ensures that changes to the PSA that the licensee will make will also be incorporated into the RI-ISI programme as necessary. Documentation of programme updates should be kept and maintained by the Owner.

7.3.5 Other observations

The selected results presented in section 4.3 indicate that for certain segments, unrealistic values of the failure frequency for FAC have been used. The evaluations are produced by the PWROG methodology and in these cases the result does not appear to be consistent with failure statistics of the plant. In the Swedish regulatory review of PWROG-SE [17], it was recommended to provide further evidence of the validation of the structural reliability results for FAC or to make further developments of the model itself. Conservative inputs to the model or relying on the conservative nature of certain aspects of the code can potentially lead to inappropriate conclusions regarding the relative susceptibility to failure of various piping segments and components. In the specific application of PWROG-SE for the Ringhals plant, this is (temporarily) resolved by removing all the high risk segments with FAC and then performing a new risk ranking and adding new risk significant segments to the selection of inspection sites.

Many segments in systems 411 and 414 are outside the scope of the SKIFS methodology. This is mainly due to the definition of the consequence index, see the description in section 3.1.1. A lot of pipe segments on the secondary side, such as in systems 411 and 414, which are not pressurised with reactor water or is part of the containment tightness function, will not be assigned any consequence index and are then not within the scope of the SKIFS methodology. However, many segments within systems 411 and 414 belong to the most risk significant segments using the PWROG methodology. This is also consistent with some of the EPRI-results, at least in system 414. Thus it is recommended to make a thorough review of the SKIFS methodology regarding its further application to Swedish Nuclear Power Plants. This is valid even with the regulatory changes and utility interpretations made since the SKIFS methodology [13].

7.4 Future work

The application and consideration of RI-ISI in the US and Europe has resulted in a number of similar questions. Further research efforts are needed in the field of risk-informed selection of components for inspection. Some of the issues are:

- Consistent criteria are needed to determine when a potential for a certain damage mechanism occurs.
- The effect of leak flow rate detection is still unclear with respect to aspects such as measurement uncertainty for leak detection and human error in operator response to detected leaks.

- It is difficult to carry out reliable probabilistic analyses for some damage mechanisms. This is the case for stress corrosion, erosion corrosion and vibration fatigue. More efforts should be devoted to further develop probabilistic models which better correlate to service experience data.

- There are no detailed guidelines for how PSA analyses should be designed for the application of risk-informed inspection programmes. Such analyses are used to calculate the risk values that make up the risk profile (see Figure 13) for the components included in the risk analysis.

- More and better information is needed regarding the efficiency of inspection for various NDE methods that are used for risk reduction. In this context it is also important to further explore the role of probabilistic methods to determine inspection intervals.

- Many plants have increased the power or are in the process of performing power uprating. It is important that RI-ISI are able to identify possible new areas which may be risk significant as a result of plant modifications or changes in the plant operation. This is also valid for plant life extension.

It is anticipated that future research and development efforts will provide answers to some of these questions. It is also important that accredited third party review bodies update their competence so that they can review inspection programmes based on risk-informed selection of components to ensure that the inspection programmes comply with procedures and methods approved by the regulatory bodies.

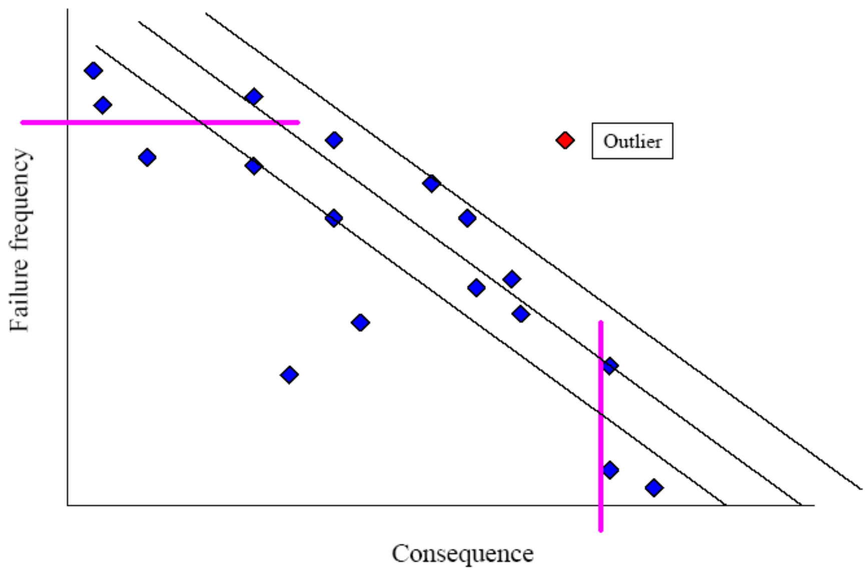

Figure 13 Risk profile

8. SPECIFIC UTILITY ASPECTS

8.1 US Experience

Section 8.1 represents the views of two US utilities. The authors were Kevin Hall (Entergy, for the EPRI methodology) and Alex McNeill (Dominion, for the PWROG methodology).

8.1.1 Background

Traditional, deterministic ISI requirements developed by Code and Standards bodies (e.g. ASME Boiler and Pressure Vessel Code) were developed to look for generic degradation to confirm the robustness of the as-designed plant. Inspection locations were random or based upon high design stress locations in the piping stress analysis, or component configuration. Operating experience has shown that degradation is typically not a random occurrence, and the designed high stress locations may not be the same location as the failure event. Degradation occurs where the conditions necessary for a particular mechanism exist. As operational experience has been obtained, the industry has discovered locations susceptible to new degradation mechanisms and developed augmented programmes to supplement the deterministic Code requirements.

Additionally insights have also been gained with respect to plant safety and the consequence of failure of certain components. These insights as well, have at times resulted in the development of augmented inspection programmes (e.g. break exclusion requirements/high energy line break). Additionally plant specific Probabilistic Safety Analysis (PSA) evaluations have been developed providing a tool to identify high consequence of failure locations within the plant.

With this experience, industry has developed a better understanding of the consequence of failure of components and the degradation typically found in the systems at a nuclear power plant. We can assign potential degradation mechanism to those locations where the known conditions may exist. Thus we can target our selections to locations that have a high failure potential and high consequence of failure. Thus allowing the RI-ISI programme to integrate and where appropriate subsume its inspection requirements with that of the augmented inspection programmes. This provides plants the ability to capture or minimise risk and thereby improve plant reliability while keeping dose as low as reasonably achievable. However, even with this focus on known degradation, it is essential that a reasonable amount of inspections continue so that defence in depth is maintained. For example, as plants remain in service longer than originally planed (e.g. long term operation, license renewal), it is essential for RI-ISI to address new degradation mechanisms that may be discovered while not invalidating the previous inspection programme. This would be done as part of the living programme that requires an evaluation update on a periodic basis when new information is obtained from plant or industry experience. It is this type of approach that will be used in managing generic ageing issues discovered in piping systems.

8.1.2 Lessons learnt

From a practical perspective, RI-ISI programmes can only be successful if they are cost-effectively integrated into the plants engineering processes and procedures. As more and more plants have adopted risk-informed programmes the following lessons learnt have been identified:

1. Having a risk-informed programme with an immature PSA or use of a methodology that is too sensitive to PSA changes is not recommended. Both these negative attributes can lead to large changes in required examinations as a result of the required evaluation update process. Having frequently changing inspection selections can result in a large expenditure of plant resources with little significant risk captured. Changes to the inspection population as a result of the updating process should be minimal, unless there is a substantial physical change to the plant or a new type damage mechanism is identified.

2. Documentation of the project must be developed and maintained in a method that supports updating the programme. Adopting a RI-ISI programme at a utility should not add significant burden to the station. Once the programme is in place functioning, procedures and processes must be put in place to allow maintaining the programme current without frequent expenditures of manpower and resources.

3. Plant staffs need to have sufficient training and involvement in the development of the RI-ISI programme. They must fully understand, at all levels of station management, what is expected of the programme and what is necessary to maintain the programme valid.

4. To manage resources the industry must strive to identify new mechanisms, aggressively investigate, and quickly develop criteria allowing them to be subsumed by the RI-ISI programme, as appropriate.

Further, experience has shown that RI-ISI evaluations of piping systems often identify segments or structural elements as safety-significant that cannot be inspected effectively or that other measures or means may be more useful in managing plant risk. One example concerns components which are subject to degradation mechanisms such as vibratory fatigue, which can develop very fast, and usually in components that are difficult to inspect because of materials or design (e.g. socket welds). Another example could be piping failures that result in high consequences due to lack of, or insufficient, plant procedures. A third example could be use of plant modifications to address risk significant sites (e.g. barriers, thermal transient monitoring) in lieu of inspections. Such occurrences suggest that plant owners and regulatory bodies should adopt a wider perspective risk management versus in-service inspection alone.

To further emphasise, it is also important to investigate alternative possibilities for mitigation against the risk caused by pressure boundary failures. Thus, it is necessary to look at the nature of the risk associated with each segment or system. It has been shown in the past, that is possible (and more beneficial) to identify ways other than inspection to address the risk, such as monitoring, improved water chemistry control, new or revised procedures and / or hardware modification.

Recent experience has also shown that many plants are increasing their capacity factors. While utilities have nuclear safety as their top priority, they also have a commitment to ratepayers and regulatory agencies to spend their resources in a responsible manner. This increase in plant capacity factors has been accompanied by decreases in outage duration. This presents a challenge to the plant operator in that it is necessary for a plant to maintain defence in depth programmes (e.g. ISI) without these becoming critical path outage activities.

8.1.3 Conclusion

RI-ISI is an unquestionable success. It has targeted inspections to critical locations while allowing utilities to decrease resources expended on inspection of non risk significant locations. As the industry continues to improve (i.e. safety, reliability) it will be necessary to build on the risk-informed ISI success and expand risk-informed technologies into other inspection and testing programmes.

The next challenge facing the industry is development of RI-ISI programmes for new reactor designs. The new reactor designs have advanced PSA analysis and software as part of the technical documentation. However, current RI-ISI processes used by the industry have not been written to accommodate these new reactor designs where operational experience is limited. Overcoming these issues and allowing the use of RI-ISI programmes for new reactor designs is the next industry goal in RI-ISI technology.

8.2 Ringhals experience

Section 8.2 represents the views of Ringhals NPP. The author was Anders Leijon.

8.2.1 Ringhals journey to reach an approved RI-ISI programme

Ringhals NPP started to develop the new RI-ISI programme in the year 2000. Previously, Ringhals had performed a pilot study at unit 4, including the RCS (reactor coolant system) and AFWS (auxiliary feed water system) with the PRWOG methodology. One of the reasons for choosing this methodology was that its seams to improve the former ISI programme to a new dimension. The major differences were the use of the plant PSA study in conjunction with calculated failure probabilities for all piping in the scope. The PRWOG methodology itself is not strictly risk-based. Rather, it is a risk-informed methodology which also pays attention to deterministic insights by using an expert panel at the end of the process. Ringhals started the work with Ringhals unit 2 and it took about 2 years to complete the work to develop the new RI-ISI programme. When the programme was completed Ringhals submitted the new RI-ISI programme to the Swedish Nuclear Inspectorate for final approval. Because this was a new methodology in Sweden, the regulator performed a very thorough review of the work and Ringhals personnel spent about one and a half man-years to answer questions and to improve the methodology so that it should fit the Swedish regulations.

The review also include the methodology itself and some part of it had to be excluded, the Swedish regulator did not give their approval to use the structural element selection matrix and the Perdue model. That forced Ringhals to develop a new methodology that should replace the original part. After this was done Ringhals finally got the approval, but still it had a lot of conditions that must be fulfilled before Ringhals could complete the new RI-ISI programme. Some examples of the conditions, which had to be solved, are shown below.

- Improve the FAC model used in the Structural Risk Reliability Assessment, SRRA-code or demonstrate that the method Ringhals has used is conservative and do not disturb the RI-ISI programme.
- Authorised the inspection intervals for each HSS segment with damage tolerance analysis for postulated defects.
- Develop new POD curves that are more accurate, due to the assumed damage mechanism.
- Develop other methods to handle segment that has a damage mechanism there ISI does not help, i.e. vibrations.

Ringhals AB have developed a work plan for SSM on how to handle the remaining issues from the above bullet list and received an approval from the safety authority in December 2009. When Ringhals performed the work for unit 3 and 4 it was easier and not so time consuming, it took about one year to develop the new RI-ISI programmes for unit 3 and 4. For unit 3 and 4 the third part body should review the outcome from all this work. The methodology was already approved by SSM but the third part body was forced to understand how Ringhals had come up with all these numbers. Again Ringhals had a quite long process to get the final approval and still we have some actions to fulfil due to the methodology approval.

The journey to develop a new RI-ISI programme has not been an easy one. The outcome of the work is however a good RI-ISI programme that let Ringhals perform inspection where it does most good. The new RI-ISI programme also reduces by about half the radiation dose to the personnel involved and this is a great benefit compared to the old programme. Ringhals had a risk informed inspection programme already from the beginning, similar to the EPRI model, but with a 3×3 matrix instead of 3×4. For Ringhals, the total number of inspections is not reduced. The numbers of inspections have slightly increased with the introduction of the new RI-ISI programme.

The time to introduce a full scope RI-ISI programme for piping at unit 2-4 took all together 8 years and a tremendous amount of work has been done to achieve the present status. Still there are some more actions to be completed before we are finished with all regulatory aspects. If Ringhals had known how much effort it would take to obtain approval for a RI-ISI programme regarding to time, money and resources because of the extensive regulatory review of the first application of methodology, Ringhals may not had not started this work

8.2.2 Conclusion

RI-ISI has targeted inspections to critical locations while allowing decrease resources expended on inspection of non risk significant locations. In which degree depends on which kind of method for selection you had before the change to RI-ISI. For Ringhals no reduction in number of inspection where achieved but a relocation to other inspection areas with lower radiation doses.

Because the rules in Sweden stipulates that you should go through and if necessary update your ISI programme ones per year, it is very important to put effort to streamline or simplify the RI-ISI process. One step in the right direction is a new RI-ISI database software that will help us to reduce the work due to the yearly updates.

9. CONCLUSIONS

Despite the limitations of the benchmark exercise (see Section 2.4), several conclusions could be drawn in each phase of the evaluation. This section summarises the main findings of the evaluation groups and presents some additional generic observations from the benchmark study and RI-ISI approaches.

9.1 Evaluation of the scope

In the evaluation of the scope of the ISI selection approach, attention was paid to the flexibility of the approach; and if the scope can be modified, what effects does it have on the results. Also the differences in division of the piping into segments were analysed.

The PWROG (original), PWROG-SE and EPRI methodologies allow a range of scopes from partial to full scope applications. In practice, only full scope applications have been applied for the PWROG-SE methodology while most PWROG (original) and EPRI methodology applications have been partial scope (Class 1 only or Class 1 and 2 only piping). The SKIFS, ASME Section XI and Code Case N-716 methodologies offer basically one scope.

The EPRI, SKIFS, and ASME Section XI, methodologies allow piping exemptions to varying degrees, e.g. based on small size. The PWROG (original), PWROG–SE and Code Case N-716 methodologies currently do not allow piping exemptions. Analysis may show exempted piping to be high safety significant, especially if augmented inspection programmes are not utilised.

There is a significant difference between EPRI and the PWROG methodologies concerning the change of scope. If new systems are added in the scope, it does not affect the inspection site selection for the other systems in the EPRI methodology. In the PWROG methodology, the risk ranking is done for the entire scope, and addition of new systems may influence the ranking of segments in other systems. Decreasing the scope of application, if it has an impact, typically results in an increase in the number of inspections in the remaining systems. Based on the RISMET study, there is a greater difference between the single system scope and a four system scope than there is between a four system scope and a full scope application of PWROG and PWROG-SE methodologies.

The EPRI and the PWROG methodologies both divide the piping systems into segments in their analyses. There are two main differences in the segmentation principle. The first is that in the PWROG methodology the segmentation is based primarily on the consequences and inspection site selection is done at the segment level. The second is that in the EPRI methodology the segmentation is more for ease of use in documenting the consequence of failure and failure potential analyses. Segments are then combined into bins in support of the risk ranking and inspection site selection tasks. The process of segmentation between the PWROG and EPRI methodologies could theoretically result in differences in the structural element selection process. However, this potential difference was neither confirmed nor disproved in this study.

9.2 Evaluation of failure probabilities

The objective of this evaluation work was to contrast-and-compare the different technical approaches to pipe failure probability analysis that were used by the application groups. In the evaluation of the failure probability assessment, the following attributes were addressed and evaluated: explicit consideration of industry experience and of plant-specific operating experience, quantitative assessment of pipe failure probability, structural failure mode(s) modelled, and verification and validation of methodology.

There are basically three main approaches to estimate the leak and rupture frequencies of piping: 1) use of probabilistic fracture mechanics / structural reliability models, 2) statistical estimation from experience data, e.g. large databases, and 3) use of expert judgement. The structural reliability models are parametric models that support a broad range of sensitivity studies, and they are useful for estimating relative failure frequencies. These models can also provide the ability to model combinations of degradation mechanisms, specific configurations and operating conditions, and explicitly account for mitigation actions (e.g. weld overlay, stress relief). These methods are especially valuable when service (failure) data is lacking, as in many Class 1 systems. One disadvantage is that the calculations tend to be complex and costly, and include large uncertainties, although the uncertainties in using industry data for piping where there is no failure data available are expected to be just as large.

The quantification of piping failure probabilities is not a pre-requisite for RI-ISI programme development. If a qualitative ranking is used, a delta risk evaluation may however require the use of some bounding values for failure potential in different categories.

A successful implementation of any methodology, independently of its level of quantification, is strongly dependent on an in-depth knowledge of structural integrity management and piping system degradation susceptibilities. The results obtained are highly correlated with underlying assumptions and the knowledge and experience of analysts.

With some exceptions, a comparison of the EPRI and PWROG methodologies points to similar POF ranges. The PWROG methodology appears to be more resource intensive than any of the other RI-ISI methodologies that are included in the RISMET scope of work. Respective topical report and implementation guidelines address the role and importance of service experience data in ensuring realistic results and as input to future RI-ISI programme updates. Within the scope of the PWROG-SE application the explicit roles of the plant-specific and industry wide service experience data could not be assessed, however.

The effectiveness of inspections plays a significant role when the risk reduction by inspections is evaluated. The probability of detection (POD) values used in the assessment should be well justified. This holds both for the models and assumptions built in the structural reliability models, as well as for POD improvement factors used together with the bounding failure probability values.

9.3 Evaluation of consequences

The characteristics of various approaches concerning consequence analyses were reviewed and the results of consequence assessment were compared. In particular, the following aspects were considered.

- Estimation of total core damage risk for pipe failure
- Inclusion of all safety classes
- “Defence in depth” coverage
- Consequence grouping
- Conservatism and uncertainties in the PSA model

- Effects of plant modifications, procedure updates and power uprates on risk ranking

An important conclusion of this study was that the PSA model should have a high degree of realism in those areas that are of importance for the risk evaluation. These areas depend on plant design and can be different between PWROG and EPRI. Potential excessive conservatism can result for instance in an increased number of inspections, in the system where the conservatism exist, in the EPRI methodology. In the PWROG methodology, such conservatism can lead to overestimating the RRWs in those systems whose PSA result is conservative and underestimating the risk importance of other systems which may impact the amount or distribution of inspections across systems.

The assumptions made about the safe condition (end state) for transients and about crediting operator actions are important. The following parts of the consequence evaluation were identified as having an important role:

- Success Criteria for the different initiating events.

- Realism in thermo-hydraulic analyses of system demands to avoid core damage. This includes issues as how to specify end-state (safe shut down conditions), mission time, safety system availability due to different demand during the mission time, and the realism in codes performing the evaluation of the LOCA-scenarios.

- Probability of success in performing manual actions, such as closing valves to isolate breaks.

- Assessment of system availability due to minor loss of water inventory over longer times (24 hours). This includes the assessment of operator actions to mitigate such failures.

The evaluation group recommends further research activities. First, guidance on the technical quality of PSA is needed to reliably support risk ranking in RI-ISI. This should include guidance on the PSA scope, on elimination of excessive conservatism, on leak detection and isolation modelling, and on completeness in pipe rupture consequence modelling, with a particular focus on the three issues identified in the list above. Second, guidance on handling inspections in the “break exclusion zone” should be developed based on its specific importance and the fact that both deterministic and probabilistic analyses assume that no leakages occur there.

9.4 Evaluation of risk ranking and site selection

The main tasks of this evaluation work were to identify and analyse the differences in the process of risk ranking and inspection site selection, and in the results. A comparison was made on specific issues such as:

- Risk measures used for risk ranking.

- Definition and treatment of risk outliers.

- Treatment of uncertainties.

- Use of expert panels.

- Treatment of low probability/high consequence and high probability/low consequence segments.

- Site selection process.

- Consideration of other criteria, e.g. radiation doses, accessibility and inspectability.
- Consideration of leak detection.
- Risk acceptance and other criteria for the application.

The risk-informed inspection site selections were compared to a selection based on the existing SKIFS inspection programme and to a selection based on ASME XI rules. Specific attention was given to the importance of these differences. The differences in failure probability and consequence assessment, identified in corresponding steps of the evaluation, are reflected in the risk ranking. However, the rankings cannot be straightforwardly compared, since the risk regions are determined differently in various methodologies.

From the comparison of resulting ISI site selection in the applications, following main observations can be made. All risk-informed approaches would result in significantly fewer inspections in the RCS (313) than the ASME XI application. This is an expected result and seen in all prior RI-ISI applications. In the case of the RHR System (321), there is large agreement on the low safety significance of most of the segments, but there are some significant differences in the ISI scope between the risk-informed applications. The SKIFS and Code Case N-716 applications result in only few inspections, while in other applications high consequences imply a larger ISI selection. For the Main Steam System (411), there is generally a good accordance of the low safety significance of most of the segments. Some differences in failure potential assessment were identified, as well as differences in PSA interpretation. The Condensate System (414) is excluded from the ASME XI and SKIFS scope. The system however has some risk-significant segments. The dominant degradation mechanism is flow accelerated corrosion (FAC), and in the US and a number of other countries the FAC susceptible piping would be a part of the augmented ISI programme and typically excluded from a RI-ISI scope.

Delta risk analyses comparing all methodologies against each other were performed by Westinghouse using the PWROG approach. Also EPRI performed risk impact analyses against SKIFS and ASME XI selections. The analyses show that the results of the risk impact assessment are dependent on the assumptions made on failure probabilities and consequence measures, as well as on the inspection capability assumptions.

9.5 Other remarks

Risk-informed applications that base the consequence assessment on a plant specific PSA model have the capability of identifying risk important inspection locations that might otherwise be ignored. This is a clear benefit of the RI-ISI approaches, especially in full scope applications where also other than highest safety class systems are considered. Reduction of inspections in primary circuit can be justified, and thus radiation doses can be significantly reduced. The economical benefit for the plant of moving to a RI-ISI depends on the present ISI scope, rules and regulations.

The RI-ISI principles in the SKIFS methodology have been used for more than 20 years in Sweden. Although inherently simple, it has been successful in identifying many of the high risk locations in the plants. However, some of the results from the RISMET study indicate that a more PSA-based procedure, which accounts for the consequences of a pipe leak or break in a more consistent way, may represent an improved methodology. This is also supported by the recently completed pilot study, [52].

RI-ISI process itself is a valuable exercise, since it forces the project team to review the piping degradation potential and identify both direct and indirect consequences of piping failures. Also this review may identify more efficient inspection procedures than are presently used. In order to benefit the most of a risk-informed approach, the plant specific PSA model should be of high quality. A lack of coverage can to some extent be compensated by expert judgement.

The EPRI and PWROG RI-ISI methodologies have originally been developed in the US regulatory environment as alternatives to ASME XI, and it is assumed that in addition so-called augmented programmes are in place to address some specific degradations. Even if according to the US Regulatory Guide 1.178 selected augmented programmes, or parts of the programmes, may be incorporated into a RI-ISI programme, they are in practice treated separately. The regulations are not the same all over the world, and even if plants typically have owner defined additional inspections, they are not necessarily directly comparable to the US augmented programmes. In some countries, plant owner defined additional inspections may have been incorporated into the overall ISI programme. When applying RI-ISI, it is important that the owner defined or augmented programmes are integrated into or coordinated with the RI-ISI programme in a logical manner.

Beside the expected presence augmented programmes, the RI-ISI methodologies may have other assumptions not necessarily valid outside US. When adapting PWROG, EPRI or Code Case N-716 methodologies in different regulatory environment, one should be careful and make sure that the procedure does not exclude important piping from consideration due to such assumptions.

It should be noted that RI-ISI evaluations often identify risk-significant segments or sites where other safety management measures than inspections may be more useful. For instance components subject to a fast degradation mechanism, such as vibratory fatigue, or components that are difficult to inspect because of materials or design may require alternative approaches. Other ways to address the risk may be e.g. continuous monitoring, improved leak rate detection, improved water chemistry treatment and follow-up. Also in some cases of high consequences, plant modifications or changes in plant procedures may be a rational way to address the risk. Thus, it is also important to investigate alternative possibilities for mitigation against the risk, and the utilities and regulatory bodies should adopt a wider perspective of risk management than RI-ISI alone.

Even if in some cases ISI is not the best safety management solution of some risk-significant segments or structural elements, and they would be excluded from a further evaluation from a NDE perspective, it is important to assign such items a “high risk” status in the risk ranking phase. The plans for further treatment of those items should be clearly documented.

For the plant overall safety the identification of risk outliers is very important complementary information. If efforts are made to eliminate these outliers with specific mitigation countermeasures, the plant safety will increase. In some cases such mitigation programmes (e.g. hardware, procedures and training) give more safety improvements than the specified ISI-programme. Methodologies that do not include PSA assessment (SKIFS, ASME) will not provide such important complementary information.

Within the RISMET benchmark it was not possible to judge or compare the full documentations that would come as output of a RI-ISI application, but a transparent and traceable documentation of the RI-ISI should be highlighted. Even if a ranking of segments or sites is initially done based on some risk measure, an important part of RI-ISI approaches is the consideration of other factors too. Availability aspects (high failure potential, low consequences), defence in depth (low failure potential, high consequences), radiation doses, accessibility, etc. affect the final definition of the ISI elements. In the PWROG methodology this is explicitly considered in the expert panels. In the EPRI methodology, these issues are addressed by requirements for inspections in various risk categories, and by the considerations of a multi-disciplinary Element Selection Team. In any RI-ISI application the decisions should be documented in a transparent way so that the bases for decisions can be traced and audited.

The effectiveness of the ISI programme depends on the choice of inspected elements, the inspection capability and ISI intervals. The benchmark was focused on the ranking and selection of ISI sites at segments level, and excluded the exact choice of welds or other sites to be inspected, the inspection methods and intervals. Thus it is impossible to judge how close the final selection of inspection locations would have been between applications, and whether the methodology would have had an impact on ISI method selection. For the US applications of EPRI and PWROG methodologies a 10 year ISI interval for inspections is an established practice. When the RI-ISI is applied to a wider scope including active degradation mechanisms, the ISI interval could also be a subject of optimisation.

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