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Development and testing of VTT approach to risk-informed in-service inspection methodology

Final report of SAFIR INTELI INPUT Project RI-ISI

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Abstract

This report summarises the results of a research project on risk-informed in-service inspection (RI-ISI) methodology conducted in the Finnish national nuclear energy research programme SAFIR (2003–2006). The purpose of this work was to increase the accuracy of risk estimates used in RI-ISI analyses of nuclear power plant (NPP) piping systems, and to quantitatively evaluate the effects of different piping inspection strategies on risk. Piping failure occurrences were sampled by using probabilistic fracture mechanics (PFM) analyses. The PFM results for crack growth were used to construct transition matrices for a discrete-time Markov process model, which in turn was applied to examine the effects of various inspection strategies on the failure probabilities and risks.

The applicability of the developed quantitative risk matrix approach was examined as a pilot study performed to the Shut-down cooling piping system 321 in NPP unit OL1 of Teollisuuden Voima Oy (TVO). The analysed degradation mechanisms were stress corrosion cracking (SCC) and thermal fatigue induced cracking (in the mixing points). Here a new and rather straightforward approach was developed to model thermal fatigue induced cracking, which degradation mechanism is much more difficult to model than SCC. This study further demonstrated the usefulness of Markov analysis procedure development by VTT in RI-ISI applications. The most important results are the quantified comparisons of different inspections strategies. It was shown in this study that Markov models are useful for this purpose, when combined with PFM analyses. While the numerical results could benefit from further considerations of inspection reliability, this does not affect the feasibility of the method itself. The approach can be used to identify an optimal inspection strategy for achieving a balanced risk profile of piping segments.

Preface

This report has been prepared under the research project Risk Informed In-service Inspection (RI-ISI). The project is a part of the project SAFIR, which is a national nuclear energy research program. In the structure of SAFIR, RI-ISI is a subproject of a project INPUT, which is a part of a larger project system INTELI. INPUT stands for Reactor circuit piping, and INTELI stands for Integrity and lifetime of reactor circuits. The work was carried out at the Technical Research Centre of Finland (VTT). The project started in January 2003 and ended in January 2007. RI-ISI project was funded by the State Nuclear Waste Management Fund (VYR) and VTT.

Espoo, March 2007

Otso Cronvall, Ilkka Männistö & Kaisa Simola

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List of symbols

ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
CCDP	Conditional Core Damage Probability
CD	Core Damage
CDF	Core Damage Frequency
CLERP	Conditional Large Early Release Probability
CSNI	Committee on the Safety of Nuclear Installations
DNV	Det Norske Veritas
ECSCC	External Chloride Stress Corrosion Cracking
EDF	Electricité de France
ENIQ	European Network for Inspection and Qualification
EPRI	Electric Power Research Institute
EU	European Union
EURATOM	EURopean ATOMic energy community
FAC	Flow Accelerated Corrosion
FMEA	Failure Mode and Effect Analysis
HAZ	Heat Affected Zone
HFI	High Failure Importance
HSS	High Safety Significance
IAEA	International Atomic Energy Agency
IGSCC	Intergranular Stress Corrosion Cracking
ISI	In-Service Inspection
IWM	Fraunhofer-Institut für WerkstoffMechanik
JRC	Joint Research Centre
LERF	Large Early Release Frequency
LOCA	Loss Of Coolant Accident
LSS	Low Safety Significance
MIC	Microbiologically Influenced Corrosion
NEA	Nuclear Energy Agency

NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NRWG	Nuclear Regulatory Working Group
OECD	Organization for Economic Cooperation and Development
OMF	Optimisation de la Maintenance par la Fiabilité
OPDE	OECD Piping failure Data Exchange
PFM	Probabilistic Fracture Mechanics
POD	Probability Of Detection
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PWROG	Pressurized-Water Reactor Owners Group
PWSCC	Primary Water Stress Corrosion Cracking
RI-ISI	Risk Informed In-Service Inspection
RPV	Reactor Pressure Vessel
RRW	Risk Reduction Worth
SCC	Stress Corrosion Cracking
SRM	Structural Reliability Models
SRRA	Structural Reliability and Risk Assessment
STUK	The Finnish Radiation and Nuclear Safety Authority (SäteilyTUrvaKeskus)
TGR	Task Group on Risk
TGSCC	Transgranular Stress Corrosion Cracking
TVO	Teollisuuden Voima Oy
USNRC	U.S. Nuclear Regulatory Commission
WOG	Westinghouse Owners Group
WWER	Water-cooled Water-moderated Energy Reactor
VTT	Technical Research Centre of Finland (Valtion Teknillinen Tutkimuskeskus)
VYR	The State Nuclear Waste Management Fund (Valtion YdinjäteRahasto)

1. Introduction

In order to monitor the condition of the nuclear power plant (NPP) piping systems, they are subjected to in-service inspections (ISI). Inspections are performed during annual outages and only a portion of the piping components is inspected. The purpose of these inspections is to detect the possible degradation of the piping components. The risk of a pipe failure is thus minimised, assuming that corrective actions are taken if potential flaws are detected.

In this report the characteristics and development of risk informed in-service inspection (RI-ISI) methodology for nuclear piping systems are examined. In particular the aim is to develop a more accurate quantitative modification of the semi-qualitative EPRI (Electric Power Research Institute) RI-ISI risk matrix procedure /1/. This is carried out by applying the combination of probabilistic fracture mechanics (PFM) and Markov system analyses, and by refining the risk analysis procedure itself.

The applicability of the developed quantitative risk matrix approach was examined as a pilot study performed to the Shut-down cooling piping system 321 in NPP unit OL1 of Teollisuuden Voima Oy (TVO). As there does not exist enough applicable degradation data of the piping system in question to allow the use of statistical methods in quantifying the failure potential, structural reliability methods were resorted to. Probabilistic version of fracture mechanics based analysis code VTTBESIT, developed at VTT, was used to analyse the yearly leak and failure probabilities of a selection of circumferential piping welds. The analysed degradation mechanisms were stress corrosion cracking (SCC) and thermal fatigue induced cracking (in the mixing points).

A Markov system based approach to analyse the degradation potential and risks of the piping components was developed in the project. It includes also the effects of inspections, and thus the capability to analyse various inspection strategies, e.g. fixed vs. random. Besides forming the risk matrix concerning the piping system 321, with the application it was also possible to optimise it.

Through RI-ISI analyses remarkable benefits can be gained. Compared to traditional and conservative ISI approach, RI-ISI allows the energy utilities to optimise their inspection programs, so that the number of inspection locations and consequently the time the inspection team has to spend under radiation can be considerably reduced, while keeping the safety of the piping systems at least on the same level as earlier.

2. Overview of RI-ISI methodologies

Currently the two widely used RI-ISI methodologies are the EPRI methodology and the PWROG (Pressurized-Water Reactor Owners Group, formerly known as WOG, Westinghouse Owners Group) methodology. They both have been approved by the U.S. Nuclear Regulatory Commission (USNRC) as alternatives for the deterministic ASME (American Society of Mechanical Engineers) XI inspection procedure. These RI-ISI methodologies are described in Cases of ASME Boiler and Pressure Vessel Code /2, 3/, and their brief descriptions can be found in the work by Cronvall /4/. We summarise here the main features and differences of these two approaches. Other RI-ISI methodologies have been developed e.g. by EDF (Electricité de France) in France (Optimisation de la Maintenance par la Fiabilité (OMF) Structures), and DNV (Det Norske Veritas) in Sweden (NURBIT code), but their applications have been rather limited so far.

2.1 EPRI methodology

The EPRI RI-ISI procedure includes the following four major steps /1/:

- identification of the system and evaluation of boundaries, including the selection of the piping systems to be inspected,
- failure mode and effect analysis (FMEA), in which the potential failure modes of the chosen piping systems are determined and the consequences of a possible pipe rupture are estimated,
- division of selected piping systems into separate segments, where a piping segment consists of a continuous pipe run, the components of which have common rupture impacts and degradation/failure modes and
- risk assessment of each pipe segment.

Based on the risk assessment, pipe segments are divided into categories of high, medium and low risk, using as a risk parameter the conditional core damage probability (CCDP), which is the possibility that a rupture in a segment results in a core damage (CD), for a limiting pipe break size and the probability of a pipe break, where the probability of a pipe break is assessed on the basis of the determined degradation mechanism(s) for each pipe segment.

Figure 1 shows the EPRI Risk Characterization Matrix, which is based on three broad categories of failure potential (high, medium, or low) and four broad categories of consequence potential (high, medium, low, or none). In the EPRI methodology, each segment is placed to the appropriate place on the 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**

Figure 1. EPRI risk matrix.

The consequence category is determined in the form of the CCDP and the conditional large early release probability (CLERP), see Table 1. The failure potential category is determined on the basis of identified degradation mechanism, see Table 2. The Risk Categories shown are combined into three risk regions for more robust and more efficient utilization. Three risk regions also account for uncertainties in the risk categorization, 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”).

Table 1. Correspondence of Consequence Categories to Numerical Estimates of CCDP and CLERP /3/.

Consequence Category	Corresponding CCDP Range	Corresponding CLERP Range
HIGH	CCDP > 1E-4	CLERP > 1E-5
MEDIUM	1E-6 < CCDP < 1E-4	1E-7 < CLERP < 1E-5
LOW	CCDP < 1E-6	CLERP < 1E-7

Table 2. EPRI System for Evaluation of Pipe Rupture Potential /3/.

Pipe Rupture Potential	Expected Leak Conditions	Degradation Mechanisms To Which The Segment is Susceptible
HIGH	Large	Flow Accelerated Corrosion (FAC)
MEDIUM	Small	Thermal Fatigue Stress Corrosion Cracking (IGSCC, TGSCC, PWSCC, ECSCC) Localized Corrosion (MIC, Crevice Corrosion and Pitting) Erosion-Cavitation
LOW	None	No Degradation Mechanisms Present

Pipe elements within each segment are candidate locations to be selected for the inspection program. The number of elements to be examined as part of the RI-ISI program depends on the risk category for the risk-significant segments. For elements determined to have degradation mechanisms other than those included in the existing plant flow accelerated corrosion (FAC) and Intergranular Stress corrosion cracking (IGSCC) inspection programs, the following number of elements are to be volumetrically examined (beyond pressure/leak testing requirements) as part of the RI-ISI program:

- 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 next higher whole number).

When an application includes Class 1 piping, a check of the final number of inspections should be made. Class 1 inspection populations substantially below 10 percent should be reviewed.

The failure potential assessment in the EPRI approach is originally qualitative, but for the risk impact assessment process also quantitative measures can be applied. One possibility is to use bounding estimates. When bounding estimates are applied, EPRI suggests following values for the failure frequencies: 2×10^{-6}/weld-year for welds in the high-failure potential category, 2×10^{-7}/weld-year for welds in the medium-failure potential category, and 10^{-8}/weld-year for welds in the low-failure potential category /5/.

2.2 PWROG methodology

The PWROG methodology /2/ integrates an engineering analysis and a probabilistic safety assessment (PSA) for RI-ISI purposes. After the scope of the program has been determined, this combined procedure contains the following steps:

1. piping segment definition
2. failure mode and probability estimation
3. consequence evaluation
4. risk ranking evaluation and
5. structural element selection.

The PWROG methodology divides piping systems into segments having the same consequences of failure in terms of an initiating event and/or system failure included within the PSA model. Segment boundaries are primarily, but not exclusively, based on changes in consequences.

PFM based code SRRA (Structural Reliability and Risk Assessment) and Monte Carlo simulation are used to quantify the failure probability of the segment. The numerical output describes the relative estimate of the susceptibility of a pipe segment to failure. The PWROG methodology considers several sizes of pipe breaks as different failure modes including a small leak, a disabling leak, and a full break.

The direct consequences (initiating event occurrence, loss of system functions) and indirect consequences (spatial effects as flooding, water spray, pipe whip, jet impingement) of failures are evaluated in terms of core damage frequency (CDF) and large early release frequency (LERF). As the pipe segment failure events are not built individually into the PSA model, the consequences of surrogate events related to each piping segment are determined through re-quantifying the PSA results. The PSA model together with the evaluation of the failure probabilities is used to calculate the pressure boundary CDF or LERF, and the importance measures are calculated relative to that value.

The risk informed ranking of piping segments is based on the following considerations:

- The quantitative safety significance (HSS/LSS – high/low safety significance), which is defined as the risk reduction worth (RRW) measure is calculated for each segment by conventional PSA methods (see the third bullet).
- The final ranking of piping segments as HSS or LSS is conducted by a plant expert panel which combines the PSA and engineering information. The expert panel considers all the information described earlier, i.e. the system characteristics, the pipe segment characteristics, the risk-related information and other non-risk related deterministic information, too.

- The expert panel reviews all segments. Segments with RRWs greater than 1.005 are quantitatively HSS and are generally ranked as HSS by the expert panel. Segments with RRWs less than 1.001 are quantitatively LSS and are generally ranked as LSS unless there is some deterministic insight as to why the segment should be ranked as HSS. The expert panel spends most of the time reviewing segments with RRWs between 1.001 and 1.005 to determine the final safety significance.

In the selection of different structural elements (piping segments, locations) for examination (inspection), a set of criteria is used. All structural elements within HSS segments with high failure importance (HFI) are selected for examination. For the remaining structural elements in these HSS segments, a statistical process is used to determine the minimum number of additional examinations. Low safety significant segments with a HFI are not required to be examined as part of this process but should be considered for selection in an owner defined program (e.g. these segments may have small effect on safety, but higher effect on availability). For more details, see /2/.

2.3 Comparison of the methodologies

The main steps or structure in both of the above described methodologies are rather similar: scope definition, segmentation, probability of failure analysis and failure consequence analyses, risk ranking and finally selection of inspection sites.

The main differences arise from the way of performing the failure probability analyses. In the EPRI methodology, the approach is basically qualitative, and quantification is basically done using bounding values. In the PWROG methodology, a specific structural reliability analysis tool is used to quantify the failure probabilities. The use of risk-importance measures is somewhat different. In the PWROG methodology, the risk ranking is based on the RRW, while EPRI methodology uses the CCDP (and the CLERP).

There are also other minor differences related e.g. to the segmentation, and the final selection of inspection sites follows different rules. The impact of these differences is studied in an OECD-JRC co-ordinated RI-ISI benchmarking project RISMET.

3. VTT approach for RI-ISI

The EPRI and PWROG methodologies are approved by NRC and are widely used in the U.S. However, they are developed in the U.S. regulatory environment, and their use in other countries often needs adjustment due to local regulatory environment and different codes and standards. For instance in Finland, the regulatory position to RI-ISI is the adaptation of a full scope approach, reviewing all the piping systems without exclusion of any degradation mechanisms or lower safety classes. In the U.S., due to the augmented inspection programmes in their regulation, the SCC is excluded from the RI-ISI. Such exclusion is not possible in Finland, and at the Finnish boiling water reactor (BWR) units the SCC is considered as an important degradation mechanism.

The European Network for Inspection and Qualification (ENIQ) has published a Framework document for RI-ISI, see reference /6/. This document promotes the use of a semi-quantitative approach for RI-ISI. It recognises that, although the probability of failure should ideally be calculated in a quantitative way, implying the use of structural reliability models (SRMs), there are important facts concerning the use of SRMs. Firstly, such models do not exist for all the potential degradation mechanisms that currently affect NPPs. Secondly, for degradation mechanisms that do have a viable SRM, there is only a limited acceptance that these estimates can be seen as representing some form of true or absolute value. This implies that the evaluation of the probability of failure for all potential ISI sites will necessarily yield a mixture of quantitative and qualitative assessments. Quantitative values, where they exist, may serve to quantify relative differences in the probability of failure from one site to another /6/.

In the light of the above mentioned issues, and on the basis of earlier Finnish RI-ISI pilot studies by Mononen et al. /7/ and Tupala /8/, VTT has aimed at developing a RI-ISI approach which uses quantitative classification for both failure potential and consequence assessment, but recognising the above-mentioned limitations in failure probability quantification. As a starting point the EPRI risk matrix was adopted. However, the experience from Finnish pilot study indicated a possible need for more categories for the ranking of both failure potential and consequences in future applications.

In the VTT approach, the segmentation follows the basic principle, that within a segment, the degradation mechanism(s) and the consequence(s) of a piping failure remain unaltered. This is similar to the segmentation in the EPRI methodology. The VTT approach also includes an expert panel to review the initial risk ranking. This is in line with the ENIQ Framework Document and the Finnish regulations.

As a result, a more accurate and refined modification of the qualitative EPRI RI-ISI risk matrix procedure was suggested at VTT. The modifications include /9/:

- changing the degradation category in the risk matrix from qualitative to quantitative,
- assessing the piping segment failure probabilities with a PFM based analysis tool developed at VTT, instead of evaluating them roughly based just on operational/process conditions as is done in the EPRI procedure and
- assessing and optimising the piping segment risks with a Markov system based analysis tool developed at VTT.

The principal idea of a refined risk matrix is presented in Figure 2.

		Consequence category [conditional core damage probability]				
Degradation category [pipe rupture frequency, 1/year]		10E-6	10E-5	10E-4	10E-3	10E-2
	10E-j	C	B	B	A	A
	10E-k	C	C	B	B	A
	10E-l	D	C	C	B	B
	10E-m	D	D	C	C	B
	10E-n	D	D	D	C	C

Figure 2. An example of a risk matrix that could be used in risk classification. A, B, C and D refer to different risk classes.

In improving the accuracy of the risk matrix, i.e. by including more matrix rows and columns, and by changing the classes of the degradation category from qualitative to quantitative, case studies are needed. In this connection it has to be clarified what data about conditions are needed (loads and other environment related factors) in addition to degradation mechanism when defining the degradation potential class. The idea is to define rather robust classes for the quantified degradation categories. Based on structural reliability calculations of selected cases, an understanding of the suitable categorisation should be achieved. In order to be able to develop a refined classification of consequences, magnitudes of the CCDP values resulting from the consequence analyses should be examined.

In the following we summarise the main principles for the quantification of failure probabilities and consequences, and inspection strategy evaluations. In Chapter 4, the practical application in a case study is presented.

3.1 Assessment of failure probability

The implementation of RI-ISI methodology calls for more accurate estimates of pipe degradation and rupture probabilities related to various degradation mechanisms, which information is also needed to enhance the bases for decision making. The degradation potential can be evaluated qualitatively, as is done in the EPRI approach /1/, where the categorisation of piping segments is carried out according to the suspected degradation mechanisms. However, in order to obtain estimates about the change in the risk due to changes in inspection procedures, the effects of various degradation mechanisms as well as inspection intervals and quality to failure probabilities should be quantified.

The quantification of leak and failure probabilities of piping components can be performed with several approaches, which can also supplement each other. These approaches are statistical models, structural reliability models and expert judgement.

If a lot of failure data is available, statistical methods can be applied. However, piping failures in NPPs are typically rare events, meaning that they have a low frequency of occurrence (e.g. less or much less than one failure per plant and year). The major piping failures are not only rare occurrences from the viewpoint of occurrence frequency, but they are also rare when viewed against the population scale of passive components /10/. Thus statistical methods alone are not a sufficient and accurate enough approach to quantify leak and failure probabilities of piping components.

Having only little failure data available, one can apply structural reliability methods. These methods consider the physical reality of the examined components through applying physical models in a probabilistic way. A good example of this approach is probabilistic fracture mechanics. The results of PFM analyses include leak and failure probabilities of piping components. When considering complex components that are exposed to several load cases, the structural reliability based analyses are usually computationally heavy and uneconomical. However, in the case of piping components geometry is simple, the number of different materials is relatively small, the number of different transient load cases which the components are exposed to is usually quite small as well and the loading is stationary for most of the time.

The most troublesome locations of piping components are welds due to often unknown weld residual stress distribution, mismatch between weld and base material, manufacturing defects, heat affected zones (HAZ), etc. The PFM approach requires quite specific information about e.g. the stresses, and it may be argued that the approach may be too laborious for some applications, e.g. to define PSA loss of coolant accident (LOCA) frequencies. However, in RI-ISI applications where the procedure requires a detailed analysis of the piping system, the quantification of leak and break probabilities may be realised with relatively reasonable additional efforts. Despite of large uncertainties related to the quantification of these probabilities, a PFM approach in the selection of potential degradation locations can be considered as an appropriate decision support.

The idea in the VTT approach is to study with a PFM approach a set of welds in order to get the understanding of their (relative) failure potential. These results can later be used as anchoring points for judging the order of magnitude of failure probability without analysing all the welds.

3.2 Consequence assessment

The consequence assessment is based on the plant-specific PSA model. In Finland, the PSAs are developed and maintained at the NPPs, and they are extensively used in the daily safety management work.

As seen from various RI-ISI approaches, different risk importance measures can be used to evaluate the consequence of a pipe failure. The VTT approach has adopted the use of the CCDP for the consequence classification of pipe segments. The CCDP gives a comparable consequence measure for all pipe segments, both to those that cause an initiating event and to those that do not. PSA studies already conducted do not necessarily directly provide the CCDP values, but they can be calculated with reasonable effort. Higher CCDP values mean more important segments of piping in terms of safety. Pipe ruptures can have different roles in PSA: they can cause an initiating event as well as the unavailability of certain safety systems relating to that initiating event. Some pipe ruptures do not cause initiating events, but rather just the unavailability of a system. In general this means three types of pipe faults /11/:

- Rupture causes a LOCA initiating event.
- Rupture does not cause an initiating event, but is noticed immediately and shutdown is required.
- Latent fault in the piping is only noticed and apparent when functioning of the system is demanded.

Initiating events caused by pipe ruptures are of the LOCA type. Current PSA study of Olkiluoto NPP has six different categories for LOCAs based on the size and location of the rupture. Location is only defined as being either inside or outside of the containment building. LOCAs inside the containment are categorised as small LOCA (S2), medium LOCA (S1) and large LOCA (A0). LOCAs outside the containment building are categorised respectively as Y2, Y1 and Y0. The definitions for different LOCAs are given in Table 3.

Table 3. LOCA categories in OL1/2 PSA.

LOCA category	Medium	Diameter (mm)		Flow (kg/s)		Area (cm^2)	
		min D	max D	min m	max m	min A	max A
A0	water			0	0	0	0
	steam	200	268	314	564	314	564
S1	water	41	400	53	5027	13	1257
	steam	95	200	71	314	71	314
S2	water	16	41	8	53	2	13
	steam	16	95	2	71	2	71

When modelling a pipe rupture another consideration is which systems might become unavailable because of it. For purposes of PSA an analysis of affected systems is required for each segment of the piping system. System failure is modelled in PSA by setting the unavailability probability of the system equal to 1.

For segments of piping that cause a LOCA initiating event both the probabilities of different size LOCAs and the systems made unavailable need to be considered. Figure 3 illustrates the modelling of pipe ruptures that cause an initiating event in PSA.

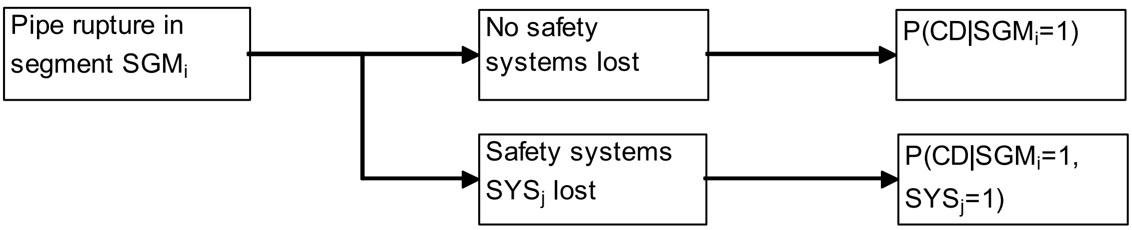

Figure 3. Modelling of pipe ruptures.

Calculations are carried out using the accident sequences already defined in the PSA. The Olkiluoto PSA study also gives the frequencies of these sequences. The CCDP for a certain piping segment is calculated as follows by assuming that the segment is ruptured or leaking so that it causes one of the initiating events listed in Table 3:

$$\mathrm{CCDP}\left(\mathrm{SGM}_i\right) = P\left(\mathrm{CD}\middle|\mathrm{SGM}_i\right) = \sum_j P\left(\mathrm{CD}\middle|\mathrm{LOCA}_j, \mathrm{SGM}_i\right) \cdot P\left(\mathrm{LOCA}_j \middle| \mathrm{SGM}_i\right), \quad (1)$$

where CD is the event that the core damage occurs, SGM_i is the event that the *i*:th segment of the piping system ruptures and LOCA_k is the *j*:th LOCA event, caused by the SGM_i. Equation (1) also allows for a single segment to break into different size LOCAs, and the probabilities correspond to the possible ways in which the pipe can rupture according to Table 3.

In cases where the break is noticed immediately, shutdown is usually required and the event tree for reactor scram or planned shutdown is used. The frequencies of sequences leading into core damage are calculated as in equation (2) and there is only one type of an initiating event to consider:

$$\mathrm{CCDP}(\mathrm{SGM}_i) = P(\mathrm{CD}|\mathrm{SHUT}, \mathrm{SGM}_i), \tag{2}$$

where SHUT stands for the shutdown, which is also an initiating event in the PSA model.

In a more difficult case the fault in a piping system is unnoticed and causes the unavailability of a system(s) under demand. Then the CCDP is /11/:

$$P(\mathrm{CD}|t, \mathrm{SGM}_i) = 1 - \exp[-f(\mathrm{CDF}|\mathrm{SGM}_i) \cdot t] \ , \tag{3}$$

where t is an assumption for the time required to notice the fault. The term $f(\mathrm{CDF}|\mathrm{SGM}_i)$ is the core damage frequency when the unavailability of systems affected by segment SGM_i is taken into account. This is accomplished by setting the unavailability of those systems to 1 in the PSA calculations. In equation (3) it is assumed that the occurrence of initiating events is a Poisson process.

3.3 Investigations of inspection strategies

Based on the failure probability and consequence assessments, the piping segments are placed in a risk matrix, see Figure 2. This risk matrix presentation corresponds to the situation where inspections are not taken into account. It is of interest to investigate the effect of inspection strategies to the risk reduction of individual segments or the whole system.

As the inspections do not affect the failure consequences, but only the failure probability, the risk reduction due to inspections can be examined by combining the PFM calculations with assumptions on inspection efficiency and intervals. The inspection strategies are studied with Markov system based analyses, and this approach is described in the following.

Markov models are used widely in modelling reliability problems. For an introduction see for example references /12/ and /13/. Different discrete states in the Markov model correspond to different configurations of the inspected system. In this application concerning pipe degradation and inspections, these different states correspond to crack growth in the pipe walls. Wall thickness is divided as a function of crack depth into states according to detection probabilities and assumed repair policies.

A discrete time Markov procedure was chosen for piping risk analyses. The overall method can be summarised in six steps /9/:

1. crack growth simulations based on PFM,
2. construction of degradation matrix transition probabilities from PFM simulations and database analysis of crack initiation frequencies,
3. model for inspection quality, which used to construct inspection matrix transition probabilities,
4. Markov model to calculate pipe rupture probabilities for chosen inspection schemes,
5. assessment of pipe rupture consequences from plant specific PSA and
6. comparison of results for different inspection strategies. Measures of interest include yearly rupture probability, yearly core damage probability and average values for both of these over plant lifetime.

The basic discrete time Markov equation is /9/:

$$\overline{\mathbf{p}}_{\mathbf{t}} = \overline{\mathbf{p}}_{\mathbf{t-1}} \times \mathbf{M}, \tag{4}$$

where $\overline{\mathbf{p}}_{\mathbf{t}} = \begin{bmatrix} p_0 & p_1 & p_2 & p_3 & \cdots & p_{n-1} \end{bmatrix}$ is a probability vector, the elements of which contain the probability for each system state at time t, n is the total number of such system states and $\mathbf{M}$ is the transition matrix that contains the transition probabilities to each state.

Due to Markov property, the probability vector can be calculated after any number of steps with equation /9/:

$$\overline{\mathbf{p}}_{\mathbf{t}} = \overline{\mathbf{p}}_{\mathbf{0}} \times \mathbf{M}^{\mathrm{T}}, \tag{5}$$

where $\overline{\mathbf{p}}_{\mathbf{0}}$ is the vector containing the probabilities of different degradation states in the initial condition. It is assumed that the probability of detectable flaws or other degradation conditions is initially zero, i.e. the pipe is in as good as new condition.

In this study Matrix $\mathbf{M}$ is calculated from the results of the PFM simulations. The degradation-inspection process was modelled with two matrices: degradation matrix $\mathbf{M}_{\mathbf{deg}}$ and inspection matrix $\mathbf{M}_{\mathbf{ins}}$. Applying the Markov process, the state probabilities for any weld when the inspection strategy is known can be calculated as:

$$\overline{\mathbf{p}}_{\mathbf{t}} = \overline{\mathbf{p}}_{\mathbf{t-1}} \times \mathbf{M}_{\mathbf{deg}} \times \left(I(t) \cdot \mathbf{M}_{\mathbf{ins}} + \left(1 - I(t)\right) \cdot \mathbf{I} \right) \tag{6}$$

where $\mathbf{M_{deg}}$ is the degradation matrix, $\mathbf{M_{ins}}$ is the inspection matrix, *I(t)* is a Boolean function with value 1 if inspections are performed at time step *t* and 0 if no inspections are performed at time step *t*, and **I** is unit matrix.

The degradation matrix models the declining condition of the piping due to any identified degradation mechanisms. The inspection matrix models the activity of inspecting the piping. To achieve this the degradation matrix contains yearly probabilities for transition from one state into a more degraded state, while the inspection matrix contains yearly probabilities for transition from a degraded state into state zero (flawless state, assuming that damaged welds are replaced with essentially new ones).

Equation 6 allows comparing the effects of different inspection strategies on rupture probability, which is one of the main advantages of using the Markov process. This is achieved by varying the Boolean function *I(t)*, by defining it as 1 for any year *t* when the weld is inspected and as 0 for any year no inspections are performed. Effects on the overall risk can also be compared, by weighting the rupture probabilities with CCDP values calculated for rupture in each weld, as taken from PSA analysis.

4. Pilot study: RI-ISI analysis of an existing Finnish nuclear piping system

The piping system examined in this study is the Shut-down cooling system 321 located in BWR unit OL1 of energy company TVO. The main task of system 321 is the cooling of the reactor when it is shut down during fuel change or during cold shutdown. During all operational conditions system 321 feeds water to the Reactor water clean-up system 331, after which it returns this water to the reactor pressure vessel (RPV). The consequence of losing the integrity of system 321 would be a LOCA either inside or outside of the containment. The system 321 contains four pipe lines that penetrate the containment, each of which contains an isolation valve both inside and outside the containment /8/.

The total length of the piping components of the system 321 is approximately 400 m, joined with a few hundred circumferential welds. Most piping components run in horizontal or vertical directions, and a minor part of the piping components runs in oblique directions.

The segmentation of the system 321 used in the pilot study follows the one defined in the TVO pilot study in 2002 /8/, since there are no new or other information that would necessitate the altering of this segmentation. The segmentation is based on four main criteria:

- isolation actions with which the consequences of a leak can be restricted
- degradation mechanisms in question
- direct consequences and
- indirect consequences.

The location of the leak holds significance for the plant response, and through that to systems the use of which will be prevented as a consequence of the leak. Another criterion for this categorization is the possibility to isolate the leak using isolation valves.

The most significant, i.e. potentially most dangerous, degradation mechanisms affecting the system 321 are assumed to be SCC and thermal fatigue induced cracking in the mixing points. There are eight segments in which either of these degradation mechanisms is assumed to act as the prevailing one. All in all the system 321 is divided to 20 segments. There are also several segments in the system 321 which are assumed to be affected by no degradation mechanism /8/.

16 piping welds were chosen to be analysed, twelve of them are SCC cases, and four of them are thermal fatigue induced cracking cases. They cover all those eight segments which are assumed to be susceptible to thermal fatigue induced cracking or SCC.

The technical characteristics of the analysed piping system were taken mostly from original design drawings and documentation. However, according to the support team from TVO, many piping components have been replaced there with new ones since the start of the operation, which was back in 1978. Of these replacements no data were available to the VTT project team.

4.1 Analyses of failure probabilities for selected welds in system 321

The PFM analyses were carried out in two phases. First, analyses were performed with analysis code PIFRAP, which was developed by DNV, and is an early version of the NURBIT code. The code was developed to model especially the SCC, but is not suitable for the thermal fatigue induced cracking analyses. Thus in the second phase, failure probabilities were calculated with VTTBESIT code. In this report we present only analyses and results obtained with the VTTBESIT code.

VTTBESIT code has originally been developed by the Fraunhofer-Institut für Werkstoffmechanik (IWM), Germany and by VTT. With the VTTBESIT it is possible to quickly compute mode I stress intensity factor values along the crack front as well as crack growth /15/. The modifications concerning VTTBESIT and performed within this study deal with the addition of probabilistic capabilities to the code, which is originally intended for deterministic fracture mechanics based crack growth analyses.

The probabilistic treatment of some of the crack growth analysis input data parameters is described first in the following. Other crack growth analysis input data parameters than those selected here as probabilistically distributed were considered to have single deterministic values.

In general, several of the input data parameters relevant in fracture mechanics analyses have markedly scattered characteristics, which can be observed e.g. from laboratory test results. These include /16/:

1. initial crack dimensions: depth and length
2. formation frequency of initial cracks
3. certain material properties: e.g. fracture toughness, tensile strength, and
4. service conditions: e.g. frequencies of load cycles.

It is often sufficient from the viewpoint of the quality of probabilistic analysis results to consider only two or three of the most relevant scattered input parameters as distributed /16/.

In this study probabilistic distributions were assessed for the following three input data parameters /9/:

1. depth of initial cracks
2. length of initial cracks and
3. frequency of thermal loads in mixing points.

A statistically sufficient amount of data of detected flaws concerning the 321 piping system was not available. However, to a reasonable extent applicable and publicly available flaw data from nine Swedish BWR units was used instead /17/. According to this data the initiation frequency of cracks is 4.08E-04 initial cracks per year per weld. Main part of this data was used in this study.

The uncertainties in the estimation of initial crack dimensions caused by the quality, amount, origin and type of the crack data include:

- *Quality*. There is uncertainty concerning the degradation process that caused the initiation and growth of a crack, i.e. the diagnosis concerning the degradation mechanism that actually caused the cracking may not be correct, or the cracking may have been caused by a combination of two or more degradation mechanisms.
- *Amount*. The amount of available degradation data concerning piping components are in most/all cases so scarce, that the accuracy of statistical estimates based on them is often poor/insufficient.
- *Origin*. Due to differences in loading histories, component replacements, etc., the degradation characteristics of even twin NPP units differ from each other, and obviously more so when compared to units in other plants, and thus if no degradation data concerning the piping system in question is available, it is to varying extent uncertain to use those from other plants of the same type.
- *Type*. As the available piping degradation data contain only information concerning detected cracks/leaks, the unknown sizes of initial cracks have to be somehow estimated recursively, which is in several ways an uncertain procedure.

Exponential probabilistic density functions for estimated initial crack depth and length distributions were fitted to the above mentioned degradation data /9/.

The probabilistic treatment of thermal loads in the mixing points is described in the following. In general, there exist many uncertainties in the estimation of the thermal load cycles in the piping mixing points, which include:

- difficulties and inaccuracies in modelling correctly the turbulent mixing phenomena and
- difficulties in defining the heat transfer coefficient values between the fluid and inner surfaces of the piping walls.

A reasonably robust and conservative approach was applied in the assessment of thermal load cycle distributions /9/:

1. The range of the thermal loads was taken as the total temperature difference of the two mixing fluids in cases it was not large, and as slightly decreased in more severe cases.
2. The shape of the load cycles was assumed as sinusoidal and the assessed amplitude of the thermal load cycles was case specifically assumed as constant.
3. A reasonably conservative value was assumed for the heat transfer coefficient.
4. The thermal stress distributions in the pipe walls were analyzed over a sufficiently wide range of load cycle frequencies with an analysis code that applies finite difference method in axially symmetric geometry.
5. Of the analyzed load cycle frequencies those resulting with highest stress distribution amplitudes were chosen to be used in the following fatigue analyses.
6. When assessing the yearly number of load cycles the frequencies used for thermal loads were assessed case specifically in relation to so called turn over and transit times and assuming that the distribution of all realistically possible load cycle frequencies is even.
7. Load cycle frequencies used in the analyses were assumed as Poisson distributed with the highest point (i.e. mean value) being the above mentioned frequency inducing the highest stress distribution amplitudes.

The thermal stress distributions in the pipe walls were analysed with analysis code DIFF which has been developed at VTT, see reference /18/. With DIFF it is possible to analyse the stresses caused by pressurised thermal shocks in straight cylinders.

The analysis procedure of the probabilistic version of VTTBESIT is as follows /9/:

- Reading of the deterministic input data is performed.

- Random sampling of certain input data parameters from the specified distributions is performed, including:
  1. thermal fatigue induced cracking; probability distributions for initial crack depth, length and for load cycle frequency, and
  2. SCC; probability distributions for initial crack depth and length.
- Crack growth analysis is performed: the amount of crack growth in each time step is calculated with the respective crack growth equation.
  - ⇒ The ending criterion of the analysis is that crack depth reaches the outer pipe surface.
  - For each analysed circumferential piping weld 5000 separate simulations were performed, and for each of these values of the above mentioned distributed input data parameters/variables are sampled from the respective assessed probabilistic distributions.
- The degradation state to which the crack has grown is calculated for each year of the considered time of operation and for each simulation.
  - ⇒ These results are used in the consequent Markov system probabilistic degradation analyses performed with a Matlab application developed in the project.

The SCC analyses were performed for quasi-stationary operational conditions, and the thermal fatigue induced cracking analyses for fluctuating high frequency thermal loads, which were discussed above. The considered operational lifetime of the plant was taken as 60 years.

4.2 Consequence analyses for 321

CCDP values have been calculated in the RI-ISI pilot study of the Finnish Radiation and Nuclear Safety Authority, STUK /7/ and in Tupala's Masters Thesis /8/. The CCDP values for STUK study concerning the system 321 are listed in Table 4.

Table 4. STUK RI-ISI pilot study CCDP values /7/.

Location	CCDP	EPRI Category
Segments inside containment building	2.2E-04	High
Segments outside containment building	4.61E-05	Medium
Segments between inner and outer isolation valve	4.60E-05	Medium

Tupala /8/ analysed CCDP values differently, and achieved greater resolution for different segments. The effect of rupture in each segment was analysed in greater detail, taking into account the effect of each segment on other systems and subsystems besides 321. STUK pilot study assumed that a rupture in system 321 does disturb the functioning of other systems. Tupala's study did not assume this, and thus different segments have different CCDP values based on this assumption. CCDP values ranged from 2.5E-04 to 2.1E-06. Even though Tupala's study is more detailed, the results presented in it are based on the same foundation as in the STUK study and the Olkiluoto PSA study.

4.3 Inspection strategies

In order to study the effect of inspections on failure probabilities, Markov analyses were carried out as described in Section 3.3. The degradation matrices were constructed from the PFM simulations performed with VTTBESIT. For each analysed weld the probabilities of the crack state are calculated with the Markov equation (equation 5). The Markov calculations are similar for all welds – all the differences are taken into account in the PFM stage of this methodology. Probabilities were calculated for the whole of the planned 60 years of operational lifetime of the NPP unit in question. The Markov analyses were performed with a Matlab application developed in the project. Six Markov states are used in the model in this study, see Table 5.

Table 5. Markov system states /9/.

State	Crack depth	Description
0	0	New piping section falls into this category.
1	0–1mm	Small flaw – very unlikely to detect
2	1 mm–50% of wall thickness	Progressed crack in the segment. Possibility of detection, but no repair.
3	50–99% of wall thickness	Grown crack. Possibility of detection and then segment is repaired.
4	99%–<100%	Leak-before-break. Repaired if detected.
5	100%	Rupture.

In the case study calculations, a constant probability of detection was used both for the crack and the leak detection. This detection probability was set to 0.9.

Three inspection strategies were examined:

1. no inspections
2. fixed inspection strategy and
3. random inspections strategy.

In the fixed inspection strategy high inspection frequency is assigned to welds in higher risk classes, and if the risk is very low, no inspections are performed. In the random strategy, inspection targets are chosen randomly each year.

4.4 Results from the case study

We summarise here some results of the case study. More detailed results can be found in the work by Simola et al. /19/ and Cronvall et al. /9/. A special report concerning the case study and the methodology developed in the RI-ISI project is presented in Cronvall et al. /14/.

Each weld starts at a flawless state when the NPP starts operation - year 0 in this study. When the NPP is operated, the yearly rupture probabilities start to climb towards a steady-state rupture probability. This means that as the time advances, the significance of the initial states slowly decreases. The tendency towards steady-state rupture probability is shown in Figure 4.

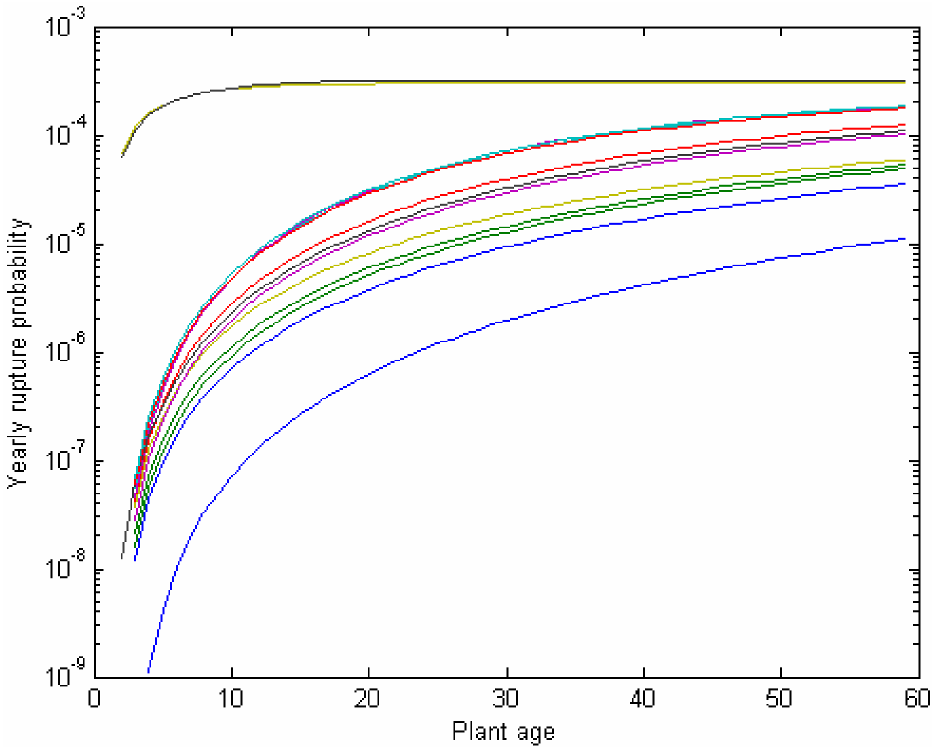

Figure 4. Yearly rupture probabilities with no inspections for the analysed 16 welds.

It is shown in Figure 4 that, depending on the weld and degradation mechanism, the 60 years operational lifetime of the NPP is enough for the welds to approximatively reach steady-state rupture probability. More precisely, the measure used for pipe degradation in this study is the average yearly rupture probability.

Table 6 shows the obtained average yearly rupture probabilities and consequence values (CCDP) for the 16 analysed welds. In the weld identification code, three first numbers specify the segment. Thus, for instance welds 2–6 belong all to the same segment. Here we can see significant scatter in failure probabilities within the segment.

Table 6. Computed average yearly rupture probabilities and consequence values (CCDP) for the 16 analysed welds.

Weld number	Weld identification	Damage mechanism	Average yearly rupture probability	CCDP
1	3-1-2--1	SCC	1.23E-5	6.8E-5
2	4-1-0--2	SCC	3.20E-6	2.1E-6
3	4-1-0--3	SCC	2.22E-5	2.1E-6
4	4-1-0--4	SCC	4.73E-5	2.1E-6
5	4-1-0--5	SCC	1.68E-5	2.1E-6
6	4-1-0--6	SCC	7.81E-5	2.1E-6
7	4-1-3--7	SCC	1.86E-5	2.5E-6
8	4-1-3--8	SCC	4.04E-5	2.5E-6
9	4-3-0--9	Thermal fatigue	2.78E-4	2.1E-6
10	4-3-2--10	Thermal fatigue	2.99E-4	6.8E-5
11	4-4-0--11	Thermal fatigue	0	2.1E-6
12	4-5-0--12	Thermal fatigue	0	2.1E-6
13	4-8-0--13	SCC	7.81E-5	2.1E-6
14	4-8-0--14	SCC	7.46E-5	2.1E-6
15	4-8-0--15	SCC	7.41E-5	2.1E-6
16	4-8-0--16	SCC	3.71E-5	2.1E-6

It is notable that two welds with thermal fatigue as the main degradation mechanism show a rupture probability of zero in Table 6. This is because the PFM simulations did not include any transfers to higher states from the first states – i.e. the cracks did not grow, due to the low load and consequent stress levels.

In Table 7 the above mentioned analysis results are presented in a risk matrix with categories of one order of magnitude. It is worth noticing, that in the EPRI classification all the welds would end up in the same *Medium* failure probability class. On the other hand, if the calculated failure probabilities are compared to the bounding values used in EPRI delta-risk evaluations, one notices that for all the welds except the two thermal fatigue cases with low loads the values exceed the bounding estimate for *High* failure probability. Due to the large uncertainties in some assumptions, one should not focus too much on the absolute risk values, but rather on their relative order.

Table 7. Risk classification without accounting for inspections.

			Consequence category			
			None	**Low <1E-5**	**Medium 1E-5...1E-4**	**High >1E-4**
Degradation category	**> 1E-4**	**High**		4-3-0--9	4-3-2--10	
	1E-5..1E-4	**Medium**		4-1-0--5, 4-1-3--7, 4-1-0--3, 4-8-0--16, 4-1-3--8, 4-1-0--4, 4-8-0--15, 4-8-0--14, 4-1-0--6, 4-8-0--13	3-1-2--1	
	<1E-5	**Low**		4-4-0--11, 4-5-0--12, 4-1-0--2		

The effect of inspections was illustrated with Markov analyses applying both fixed and random inspection strategies. In the fixed inspection strategy, the inspection frequencies were assigned to the welds according to the risk importance (see Table 7) in the following way. Risk class:

- A (red) – one inspection per 1 year
- B (orange) – one inspection per 3 years
- C (yellow) – one inspection per 5 years
- D (green) – one inspection per 10 years and
- E (white) – no inspections ever.

Note that in this example the inspection coverage is 100% for all the classes except 0% for E. Only the inspection interval is varied. In practice, also the inspection coverage would depend on the risk class.

In the random inspection strategy a weld is chosen randomly to be inspected regardless of its risk importance. The total number of inspections during the observation time is same as in the fixed inspection scheme.

Figure 5 shows the behaviour of rupture probability in system 321 with different inspection strategies for the considered 60 years of time in operation. The visible serrated edge with peaks at 10 year intervals results from the ten welds in the low risk class, which are inspected at 10 year intervals. Every 10 years the inspections take place they cause a drop in rupture probability. Even though the highest rupture probabilities were observed in welds classified to risk classes medium and high, which are inspected at 3 and 5 year intervals, no serrated edges with those intervals are visible. Upon further analysis it was noticed that the segments in those classes have thermal fatigue as a degradation mechanism, which causes the welds to deteriorate quickly once a crack is initiated. This means that inspections are of limited use in noticing cracks caused by this degradation mechanism, and no serrated edges are visible.

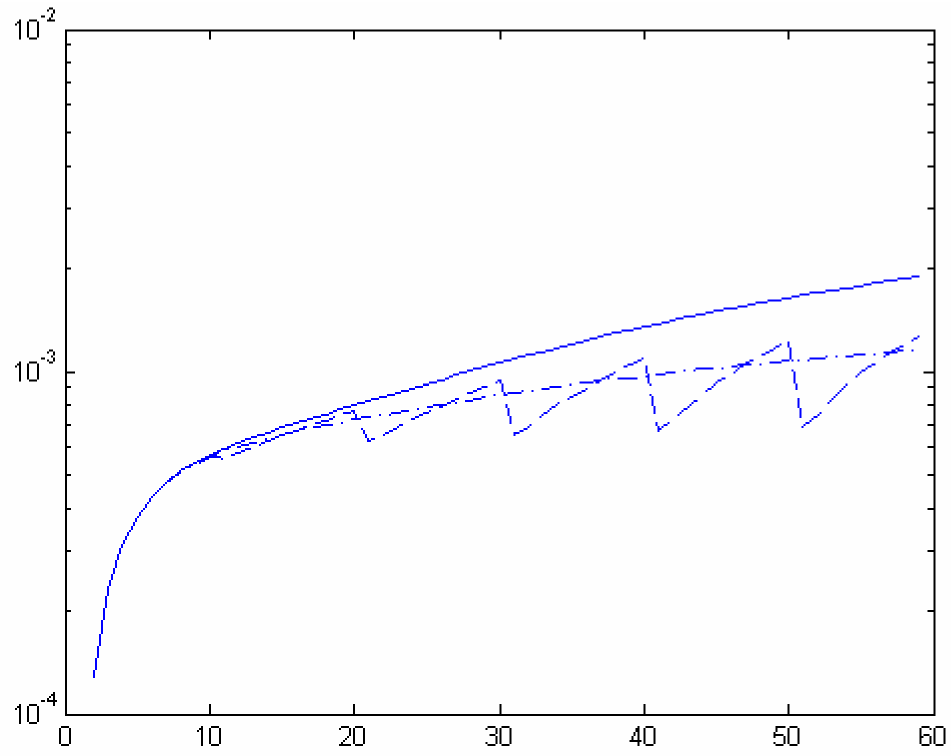

Figure 5. Yearly probability of a rupture in system 321 for no inspections (solid line), fixed inspection strategy (dashed line) and random inspections strategy (dash-dotted).

Yearly core damage probability caused by potential ruptures in system 321 is seen in Figure 6. The overall risk was in most cases dominated by thermal fatigue cases, mainly due to rapid growth of cracks in the simulations, which in turn was caused by large number of significant thermal load cycles. By increasing the inspection frequency it was possible to lower the risk levels of the examined welds in the analyses. It was also noticed that inspections are not as efficient against the thermal fatigue induced cracking as against the SCC, because the cracks tend to advance more rapidly once initiated in the case of the former degradation mechanism. This means that inspections have a smaller chance of detecting the crack before it proceeds into a rupture. Because the core damage probability for system 321 is dominated by segments susceptible to thermal fatigue, the risk decrease gained by switching from random strategy to fixed one is small.

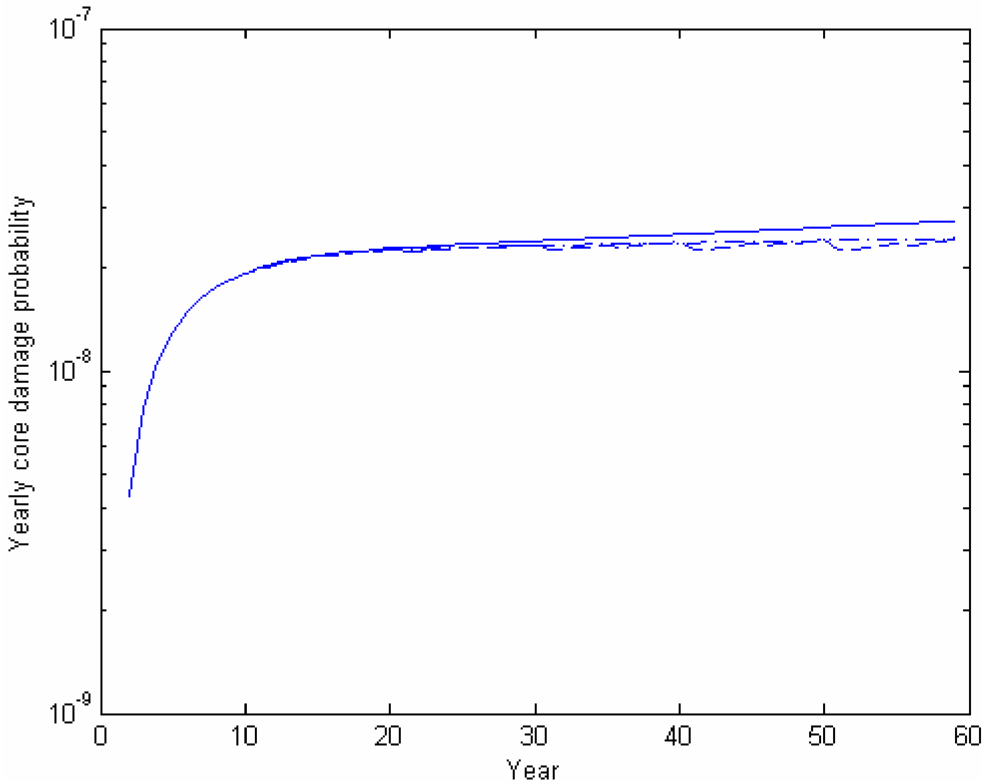

Figure 6. Yearly probability of core damage caused by a rupture in system 321 for no inspections (solid line), fixed inspection strategy (dashed line) and random inspections strategy (dash-dotted).

4.5 Conclusions from the case study

The applicability of a quantitative RI-ISI approach was examined as a pilot study performed to the Shut-down cooling system 321 of TVO. The aim of the pilot study was to gain experience on the quantification of the piping failure probabilities with to some extent simplified probabilistic calculations. Further, the study aimed at comparing the effects of different piping inspection strategies on risk in NPPs. This was carried out by applying the combination of PFM and Markov process system analyses, and by refining the risk analysis procedure itself.

The analysed degradation mechanisms were SCC and thermal fatigue induced cracking (in the mixing points). Due to quasi-stationary loading conditions and other characteristics, SCC analyses were not difficult to perform. However, difficulties were encountered in analysing thermal fatigue induced cracking, due to many involved physical phenomena, e.g. turbulent mixing of fluids of differing temperatures. Here a new and rather straightforward approach was developed to model this degradation mechanism. However, it needs to be developed further due to several involved uncertainties.

The degradation state analysis results varied for the 16 analysed welds quite a lot. This was an expected outcome, as the pipe dimensions, loads and overall conditions of the covered locations varied quite a lot as well. Even though all the inspection targets, typically pipe welds, within a segment should have approximately the same failure potential, according to analysis results here this is not nearly at all the case.

The study demonstrated the usefulness of the integrated use of Markov processes and PFM analyses in comparing the effects of different inspection strategies on risk. The developed method yields quantitative results for pipe break and core damage probabilities under any inspection strategy. The risk matrix can be used for inspection resource allocation by concentrating the inspections on high-risk segments of the piping system, thus reducing the risk in the most efficient way. However, due to the nature of the dominant degradation mechanisms and risk profile in system 321, the reduction in risk was slight in the case study. This indicates that for a more efficient resource allocation the risk matrix should take into account the characteristics of the degradation mechanism, in addition to the two risk measures used here.

The analysis of this single system was sufficient for demonstrating the comparison of different inspection strategies. However, analysis of further systems would be required for a thorough comparison of different risk matrix approaches. As the case study was focused on one system with little variance in the CCDP values and only two types of identified degradation mechanisms, the optimal form and size for the risk matrix for a more generic use could not be fully defined. Based on the experience from the case study, the accuracy of one order of magnitude both in the failure probability and consequence evaluations seems feasible. The upper and lower limits, or verbal characterization (very low, low, medium, high, very high…), may need to be plant specific. Although we recommend the quantitative approach with a precision of the order of magnitude, one should not focus too much on the absolute risk values, but rather on the relative importance of the piping welds or segments.

5. RI-ISI related international activities

A part of the research project has been the close follow-up and participation in international activities in the field of RI-ISI. In this chapter, the major RI-ISI related international activities are summarised, and the participation and contribution of VTT is highlighted.

RI-ISI methods are widely applied in the U.S., where the U.S. NRC has approved both the EPRI and PWROG methodologies as a valid alternative to ASME Section XI. In Europe the situation is somewhat different since there are roughly speaking as many regulatory environments as there are countries with NPPs in operation. This implies a variety of ISI codes and standards and national guidelines. In most cases, the U.S. methodologies cannot be adopted as such, since they have been originally developed to the U.S. regulatory environment. In this light, it is understandable that both the European regulators and utilities have funded research activities and established working groups to discuss RI-ISI related issues, to identify some common views, and agree on recommendations and good practices. Examples of past activities are European Union (EU) -funded projects RIBA /20/, /21/, /22/, and NURBIM /23/. The Nuclear Regulatory Working Group (NRWG), which is an advisory expert group to the European Commission, consisting of representatives from nuclear safety authorities and technical support organizations, has published a document summarizing the common views on RI-ISI of the European Regulators /24/. Several issues where work is recommended to be done are identified in the report. A recent EURATOM (The European Atomic Energy Community) FP6 project GAIN (Gap Analysis for Long Term Inspection Needs of Nuclear Plant) has also identified research needs related to inspection optimisation /25/. In the following, a brief summary is given on current international activities coordinated by the European Commission, OECD Nuclear Energy Agency (NEA) and the International Atomic Energy Agency (IAEA).

5.1 The European Network for Inspection and Qualification (ENIQ)

The ENIQ is a network working towards a harmonized European approach on reliable and effective ISI. Driven by European nuclear utilities and managed by the European Commission’s Joint Research Centre (JRC) of Petten, ENIQ was meant to be a network in which the available resources and expertise could be pooled at European level. More specifically ENIQ works on qualification of ISI systems and on risk-informed in-service inspection (RI-ISI) within a European context. To reflect the latter, the steering committee has set up a specific Task Group on Risk (TGR). Currently, TGR has about

20 members representing European nuclear utilities, research organizations and consultants. The chairperson of the TGR is from VTT.

As a result of the work of the TGR, the European Framework Document for Risk Informed In-Service Inspection /5/ was published in 2005. It is intended to serve as guidelines for both developing own RI-ISI approaches and using or adapting already established approaches to European environment taking into account utility-specific characteristics and national regulatory requirements.

As the Framework Document provides general principles without going into details in RI-ISI implementation, the TGR recognized the need to produce more detailed recommended practices and discussion documents on several RI-ISI related issues. The TGR identified a list of issues that would need further consideration within the group, and preliminary work plans for more than ten tasks were developed.

The benchmarking of various RI-ISI methodologies was identified as one of the top priorities. As the project for the benchmark was successfully launched in co-operation with JRC and OECD NEA, the TGR decided to integrate several work plans to the benchmarking. At the end of 2006, TGR had activities in the following tasks:

- benchmarking RI-ISI methodologies (RISMET project) /26/,
- interaction between RI-ISI and inspection qualification,
- guidelines for expert panels,
- defence in depth issues,
- SRM codes verification and validation and
- RI-ISI application for internals, RPV.

5.2 OECD-JRC co-ordinated RI-ISI Benchmark

So far there has not been any direct comparison of the several existing RI-ISI methodologies applied to an identical scope of components (system, class, etc.). Recommendations and support for performing a benchmarking of various RI-ISI approaches has been given by several international groups and committees:

- the United States Nuclear Regulatory Commission (USNRC) advisory committee on reactor safeguards /27/,
- recommendations of the expert workshop on PSA in RI-ISI organized by the JRC /28/,
- Nuclear Regulators Working Group – Task Force on RI-ISI /24/,

- ENIQ Task Group on Risk /29/ and
- OECD/NEA committee on the safety of nuclear installations (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) /30/.

Recently a project for benchmarking RI-ISI methodologies was initiated by TGR together with the European Commission Joint Research Centre and OECD/NEA. The project funding is based on in-kind contributions. The project has more than twenty participating organizations from Europe, the U.S., Canada and Japan. Also the IAEA is participating in the project. The chairperson of the project is from VTT, and VTT is also leading one evaluation group.

In the benchmark exercise, RI-ISI methodologies are applied to four piping systems of Ringhals 4 PWR plant, and the resulting risk ranking and inspection site selection are compared with each other, and with the deterministic ASME XI inspection programme. The selected systems represent a variety of safety classes and degradation mechanisms.

By means of benchmarking the different methodologies it would be possible to identify how they impact on reactor safety and whether they lead to significantly different results. Also, the benchmarking could 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.

5.3 International Atomic Energy Agency (IAEA)

The IAEA has several on-going and planned RI-ISI related activities. The IAEA has coordinated a pilot study on RI-ISI for water-cooled water-moderated energy reactor (WWER) NPPs, and the results were presented in a Workshop in Vienna in June 2006. IAEA is also launching a Coordinated Research Project on Development of a Methodology for RI-ISI Based on PSA. The aim is to develop a guidance document on consistent use of PSA in the development of RI-ISI programmes. Further, in 2007 IAEA is starting the development of a technical document on Guidance for Risk Informed In-Service Inspection of Piping Systems of Nuclear Power Plants. VTT has been invited to participate in the preparation of this document.

6. Discussion and conclusions

The aim of the study was to develop a robust quantitative risk matrix approach suitable for risk ranking of pipe segments in RI-ISI applications. Earlier pilot studies carried out in Finland have indicated that the principles of EPRI categorisation would not allow enough accuracy for RI-ISI application in the Finnish regulatory environment and considering the plant specific PSAs. The applicability of a refined risk matrix with rough quantitative categorisation of the piping failure probability combined with conditional core damage probabilities obtained with a plant specific PSA was examined as a pilot study. The study included development of a PFM analysis procedure, and an approach to combine PFM and Markov system analyses to investigate inspection strategies.

The pilot study was performed to the Shut-down cooling system 321 of TVO, from which 16 welds were selected for detailed PFM analyses. The PSA results originated from an earlier pilot study performed at the power plant. The analysed degradation mechanisms were SCC and thermal fatigue induced cracking (in the mixing points). The analysis of one system is too limited to fix an optimal risk matrix, with the ideal number of failure probability and consequence categories. However, several conclusions can be drawn from the project.

It was identified that within a segment, the failure probability may vary significantly from weld to weld. Some variation is inevitable, and in principle it is recommended to classify the segment according to its "weakest" weld. But if inspection coverage and intervals are defined based on the worst conditions, the segment – especially if it includes a high number of possible inspection volumes – may get more attention than would be optimal in a balanced risk-informed inspection programme. Thus the segmentation should be done carefully, and locations with clearly deviating loadings, stresses and/or material properties should be treated as separate segments.

The study demonstrated the integration of PFM calculations with a discrete time Markov process analysis to model piping degradation states at inspections, and accounting for flaw and leak detection probabilities. The risk reduction gained with different inspection strategies was investigated by altering the inspection intervals of the analysed welds. While the principles and applicability of the integrated PFM-Markov analysis were successfully demonstrated in this study, some further development is envisaged. In the case study, a single value was used for the detection probabilities of cracks and leaks. In the future one should aim at defining realistic but still robust probability distributions by accounting for the quality of the inspections, and e.g. different detection probabilities as a function of the crack size. Further, inspection history could be incorporated into the detection probabilities and even inspection strategies.

Further development is needed also concerning the assessment of failure probabilities. In the study a new and rather straightforward approach was developed to model the thermal fatigue, but it needs to be developed further due to several involved uncertainties. Besides the development of useful PFM approaches, the possibility to support the failure probability analyses with operating experience should be reviewed. This is a topical issue, since the OECD OPDE (OECD Piping Failure Data Exchange) project has resulted in an advanced piping failure database with a good quality.

As stated before, the EPRI RI-ISI risk matrix is not seen applicable as such in Finland. Instead we are promoting the use of a risk matrix having a quantitative categorisation for the failure probability with a precision of one order of magnitude. The CCDP is seen as a suitable consequence measure, but also there the precision of one order of magnitude is recommended, while in the EPRI risk matrix the *Medium* consequence category covers two orders of magnitude.

The highest absolute risk values depend both on the PSA model and the assumptions made in the failure probability calculations. In our case study the highest CCDP values were less than 10^{-4} /y, but for another system or power plant the values can even be of the order of 10^{-2} /y. In such cases some precision is needed above the limit CCDP>10^{-4}. The failure probabilities are highly dependent on some assumptions, and thus one should be very careful in drawing conclusions from the absolute values. The relative order of the failure probabilities is of importance, and care should be taken when analysing various degradation mechanisms or using several calculation tools, to make sure that the assumptions are consistent.

In the light of the above, we recommend that instead of sticking to a predetermined risk matrix, the categorisation is tailored so that it is fit for the plant specific RI-ISI purpose. The aim is to obtain a risk ranking of the piping segments which is at the same time robust and accurate enough.

The study included also close follow-up and participation in international activities in the field of RI-ISI. It can be concluded, that the developed approach is in line with the recommendations of ENIQ. The further development needs identified in the course of the study can benefit from continued international co-operation, especially in the field of benchmarking the RI-ISI methodologies and studying the interaction between RI-ISI and inspection qualification.

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Development and testing of VTT approach to risk-informed in-service inspection methodology

Final report of SAFIR INTELI INPUT Project RI-ISI

Abstract

This report summarises the results of a research project on risk-informed in-service inspection (RI-ISI) methodology conducted in the Finnish national nuclear energy research programme SAFIR (2003–2006). The purpose of this work was to increase the accuracy of risk estimates used in RI-ISI analyses of nuclear power plant (NPP) piping systems, and to quantitatively evaluate the effects of different piping inspection strategies on risk. Piping failure occurrences were sampled by using probabilistic fracture mechanics (PFM) analyses. The PFM results for crack growth were used to construct transition matrices for a discrete-time Markov process model, which in turn was applied to examine the effects of various inspection strategies on the failure probabilities and risks.

The applicability of the developed quantitative risk matrix approach was examined as a pilot study performed to the Shut-down cooling piping system 321 in NPP unit OL1 of Teollisuuden Voima Oy (TVO). The analysed degradation mechanisms were stress corrosion cracking (SCC) and thermal fatigue induced cracking (in the mixing points). Here a new and rather straightforward approach was developed to model thermal fatigue induced cracking, which degradation mechanism is much more difficult to model than SCC. This study further demonstrated the usefulness of Markov analysis procedure development by VTT in RI-ISI applications. The most important results are the quantified comparisons of different inspections strategies. It was shown in this study that Markov models are useful for this purpose, when combined with PFM analyses. While the numerical results could benefit from further considerations of inspection reliability, this does not affect the feasibility of the method itself. The approach can be used to identify an optimal inspection strategy for achieving a balanced risk profile of piping segments.

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