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RISK-INFORMED IN-SERVICE INSPECTION OF PIPING SYSTEMS OF NUCLEAR POWER PLANTS: PROCESS, STATUS, ISSUES AND DEVELOPMENT

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IAEA NUCLEAR ENERGY SERIES No. NP-T-3.1

RISK-INFORMED IN-SERVICE INSPECTION OF PIPING SYSTEMS OF NUCLEAR POWER PLANTS: PROCESS, STATUS, ISSUES AND DEVELOPMENT

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FOREWORD

In-service inspection is an integral part of defence in depth programmes for nuclear power plants, to ensure safe and reliable operation. Traditional in-service inspection programmes were developed using deterministic approaches. However, as probabilistic approaches are being developed, risk insights are being used to optimize in-service inspection programmes by focusing in-service inspection resources on the most risk significant locations.

This publication was developed as part of the IAEA's activities in the area of plant life management. The aim is to develop and publish the best practices of the industry on structural integrity, ageing management, residual life assessment, and life management and long term operation.

This report is the result of a coordinated effort involving the participation of experts from nuclear organizations in several Member States. Its objective is to provide guidance on risk-informed in-service inspection technology and to describe the general process of developing and implementing methodologies, and the technological issues which lie behind the methodologies. It also provides information about the current status and ongoing research and development activities. It can be used by managers, ISI supervisors, and lead in-service inspection engineers of nuclear power plants and technical support organizations. It is also useful for regulatory staff reviewing risk-informed in-service inspection programmes.

This publication complements IAEA-TECDOC-1400 on Improvement of In-Service Inspection in Nuclear Power Plants. It was prepared with the participation and contributions of experts from Belgium, the Czech Republic, Finland, the Netherlands, Sweden, Switzerland and the United States of America. The IAEA wishes to thank all the participants and their Member States for their contributions. Special appreciation goes to P. O'Regan, USA, who led all the consultants meetings for drafting this report. The IAEA officers responsible for this publication were H. Cheng and J. Mandula of the Division of Nuclear Power.

EDITORIAL NOTE

CONTENTS

1. INTRODUCTION

During the design phase of the first nuclear power plants it was believed that the high standards used to design and fabricate passive components would allow problem free operation throughout their lifetimes. For this reason, the need for in-service inspection was not considered. However, when plants became operational it was discovered that components still degraded over time despite such high design standards, and the industry began to develop inspection programmes. The American Society of Mechanical Engineers (ASME) developed a standard, the ASME Boiler and Pressure Vessel Code, Section XI: Rules for In service Inspection of Nuclear Power Plant Components, which initially provided rules for inspection of class 1 systems only. Over thirty years the Code was revised to address many needs, including the inspection of class 2 and 3 systems [47–49].[1]

Traditionally, a number of commercial nuclear power plants implemented ASME Section XI to ensure the structural integrity of systems. Section XI was based on a sampling approach: 25% of class 1 and 7.5% of class 2 piping welds were examined to verify that no generic degradation existed. To search for generic degradation, Section XI required that piping be examined based on materials, configuration and potential stress levels. These criteria, although useful as inputs for determining possible examination locations, were not suited to be used alone as selection criteria. Because of these inadequacies, problems were typically identified via non Section XI activities, for example operator walkdowns or augmented programmes. Consequently, nuclear plants were devoting significant manpower, radiation exposure, and financial resources to examine locations with low failure potential and/or little safety significance.

In the past, while Section XI in service inspection was based on a sampling approach, some operating utilities followed vendor country rules, standards and experience of designers and equipment manufacturers. This approach was based in general on hydrostatic pressure tests due to the lack of reliable NDT volumetric methods. Too great a reliance on the experience of manufactures led to partially non-systematic development of in-service inspection (ISI) programmes, as different manufacturers and designer groups applied their specific philosophies, taking into account preferably their own practical manufacturing rules and experience not integrated into an overall ISI programme development approach. ISI programmes developed in such a way became mandatory ISI programmes approved by the regulatory bodies of individual countries. For this reason there were no accepted sampling rules for

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[1] **Class 1** — also known as reactor coolant pressure boundary (RCPB). Reactor coolant pressure boundary means all those pressure-containing components such as pressure vessels, piping, pumps, and valves, which are:

Part of the reactor coolant system, or

Connected to the reactor coolant system, up to and including any and all of the following:

The outermost containment isolation valve in system piping which penetrates primary reactor containment,

The second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment,

The reactor coolant system safety and relief valves.

Class 2 — Systems or portions of systems important to safety that are designed for:

Emergency core cooling,

Post-accident containment heat removal,

Post- accident fission product removal,

Reactor shutdown, or

Residual heat removal.

Those portions of the steam and feedwater systems of PWRs extending from and including the secondary side of steam generators up to and including the outermost containment isolation valves and connected piping up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic closure during all modes of normal reactor operation.

Class 3 — consists of other pressure vessels, heat exchangers, storage tanks, piping, pumps, and valves that are not class 1 or 2 and may include:

Cooling water systems,

Auxiliary feedwater systems

Post-accident containment heat removal,

Post-accident containment atmosphere cleanup,

Residual heat removal from the reactor and from the spent fuel storage pool (including primary and secondary cooling systems).

piping systems, and a variation in the percentage and locations of piping welds selected for examination could occur. In recent years, this situation has become more and more specific in different countries. On one side, there are ISI programmes that strictly follow vendor standards; on the other side, there are ISI programmes following new national standards and approaches that take into account worldwide experience and standards.

Traditional ISI requirements looked for generic degradation. Industry experience has shown that degradation is typically not a random occurrence. Degradation occurs where the conditions necessary for a particular mechanism exist. Over time, a better understanding of the degradation typically found in the systems at a nuclear power plant has been developed. A potential degradation mechanism can be assigned to those locations where the appropriate conditions may exist. Thus, locations that have a higher failure potential can be targeted.

Risk informed technology allows plants to take the next step and inspect those systems or portions of systems that are most risk important. Risk informed in-service inspection (RI-ISI) reflects recent developments in probabilistic safety assessment (PSA) technology, structural reliability as well as the experience gained from over 13 000 reactor years operating experience of nuclear power plants. Risk is defined in the engineering sense as the product of the consequences of a failure and the probability of that failure occurring. Using RI-ISI, the risk significance of a component and its failure potential are determined. This allows the plant to target its resources to examine locations that are truly risk significant, providing the ability to capture or minimize risk and thereby improving plant reliability while keeping radiation doses to workers as low as reasonably achievable.

This publication introduces the general approach of RI-ISI technology, application status in Member States, discussion of issues related to it, and on-going research developments including applications for new plants. The objective of this publication is to provide the guidance on RI-ISI technology and inform the readers of the current status and ongoing research and development (R&D) activities. In Section 2, a generic approach is given to RI-ISI. Section 3 introduces an overview of various RI-ISI methodologies and the current application status. Section 4 describes the process and organizational matters. Section 5 highlights a number of topical issues which should be addressed during any attempt to apply RI-ISI. The publication concludes with a summary of activities which are being undertaken in the world (Section 6).

2. GENERAL APPROACH TO RI-ISI

2.1. PROGRAMMATIC PERSPECTIVE

From a programmatic perspective, there are a number of issues that need to be dealt with in order to ensure an effective ISI programme. These include:

- Management support;
- A good understanding of the strengths and limitations of the existing ISI programme;
- Proper use of plant specific risk information (e.g. PSA);
- Multidisciplinary knowledge;
- A constructive interface with the country’s regulatory body.

The formation of an appropriate workforce structure is also an essential factor in devising and implementing a RI-ISI programme. Such a workforce will need to contain or have access to a large array of different disciplines including inspection, maintenance, design, materials, chemistry, stress analysis, systems, PSA, operations and safety.

In the context of a RI-ISI programme, the regulatory body should be involved at an early stage of the process, to either define or review the basic safety requirements that must be met.

2.2. TECHNICAL PERSPECTIVE

An overview of the fundamental aspects of most RI-ISI methodologies is shown in Fig. 1. This figure reflects the basic technical elements of the risk-informed concept as relevant to developing an ISI programme. From a technical perspective, the following main steps summarize a typical RI-ISI process:

- Definition of RI-ISI programme scope;
- Collection and analysis of the input data required;
- Evaluation of piping failure consequence;
- Identification and evaluation of piping failure potential;
- Risk ranking;
- Inspection element selection;

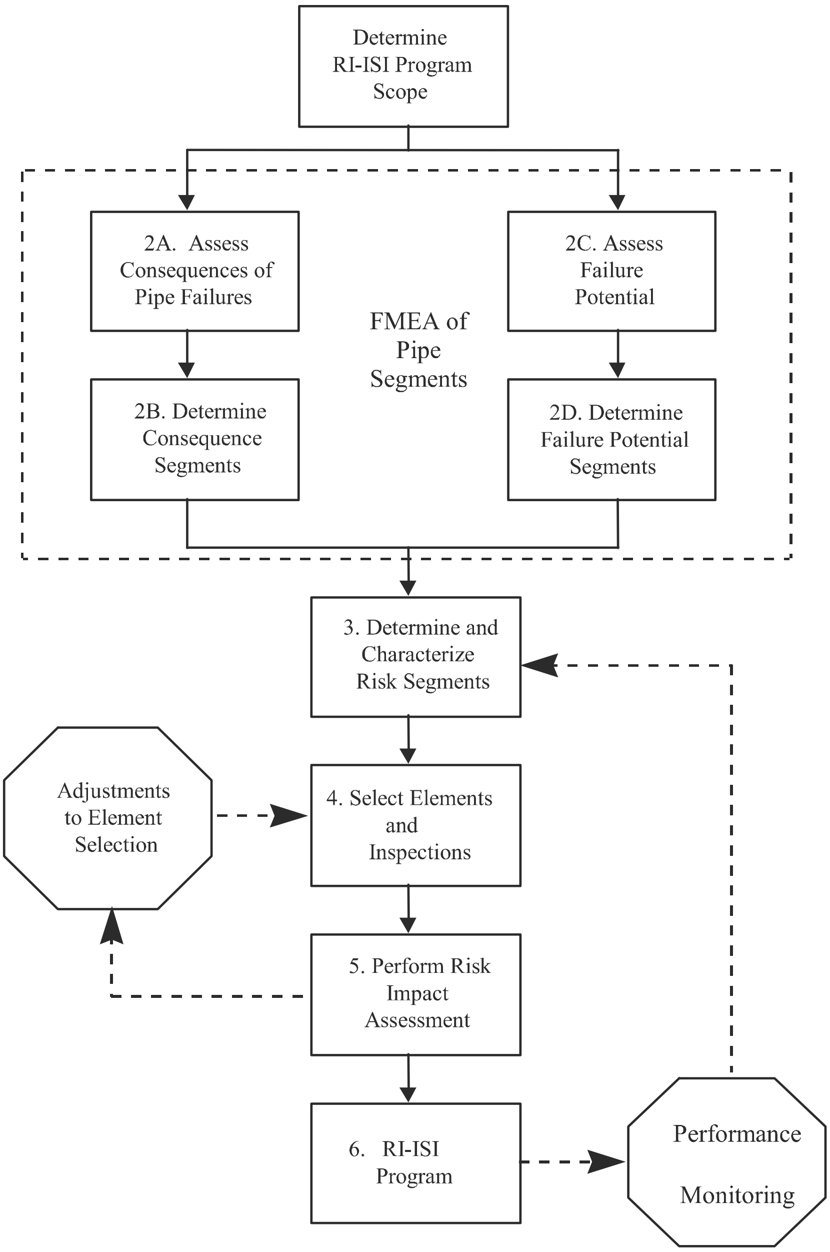


Fig. 1. RI-ISI methodology overview.

- Evaluation of risk impact of changes to inspection programme;
- Long term management of a RI-ISI programme;

Each step of the process is further discussed in the following sections.

2.2.1. Definition of RI-ISI programme scope

The first step is to decide the scope of the RI-ISI programme. As witnessed in various approved RI-ISI applications, the scope of a RI-ISI programme is subject to the goals of the plant operator as well as feedback received from the regulatory authority. Options include:

- Large scope applications including all safety class piping systems (e.g. ASME class 1, 2 and 3) and non-safety classified piping;
- Selection of individual piping system(s) (e.g. reactor coolant system), or alternative piping system scope (e.g. ASME class 1 only);
- A subset of a class of piping systems (see Sections 3.1 and 5.1).

Selection of the RI-ISI scope may strongly influence the results, and therefore the process used to define the scope should be properly understood by the operator and the regulatory body.

In practice, any system not selected for inclusion in the RI-ISI programme scope is retained within the current (deterministic) in-service inspection programme. An additional decision has to be made on whether piping systems and degradation mechanisms covered within the plant's other inspection programmes, if any, will be incorporated into the RI-ISI programme. Examples of other inspection programmes that may or may not be incorporated into the RI-ISI programme include intergranular stress corrosion cracking (IGSCC), localized corrosion (e.g. MIC), and flow accelerated corrosion (FAC). This issue is further discussed in Section 5.2.

2.2.2. Collection and analysis of the input data required

The process of RI-ISI brings together a large amount of information from many different sources, which needs to be collected and analysed. This information can be classified in the following five categories:

(1) Equipment data;
(2) Plant operating data;
(3) General nuclear industry information;
(4) Safety analysis report and technical specifications;
(5) PSA data.

Data collection is an essential part of the RI-ISI process, as it constitutes the basis for the entire analysis and decision process. Data collection is likely to be a resource demanding phase in the RI-ISI process, but the data gathered should be of considerable value for many safety or reliability related activities.

2.2.3. Evaluation of piping failure consequence

The consequence analysis is normally performed on a system by system basis and leads to the preliminary definition of piping segments.

The term 'piping segment' may have different meanings for different RI-ISI methodologies and can include: common potential for failure, common consequence potential or both. In practice, at this stage a segment includes a region of the pressure boundary for which a failure would lead to the same consequences. Later, this classification can be refined to take into account insights from the failure potential analysis.

The consequences of pipe rupture are typically measured in terms of the conditional probability of core damage given a pipe rupture (CCDP) and the conditional probability of large early release given a pipe rupture (CLERP). These measurements require quantitative risk estimates which would be obtained from the plant specific

PSA models. This is accomplished by identifying the impacts of the pipe rupture in terms of initiating events, system mitigation and containment response. The use of PSA in RI-ISI is further discussed in Section 5.3.

2.2.4. Identification and evaluation of piping failure potential

The first step in the assessment of the probability of failure of a structural element or segment is the identification of the potential degradation mechanisms. This requires the qualitative evaluation of a range of influential parameters such as design and fabrication information, loadings, environmental conditions and inspection results. This analysis should be supported with a review of operating experience from the plant, its sister units and similar plants as well as insights from worldwide generic data. Such an analysis phase is very important in order to correctly classify or quantify the failure potential.

Piping failure potential can be assessed in different ways, ranging from purely qualitative assessment to quantification with either statistical analysis of service data or structural reliability models.

If the evaluation is carried out at the segment level, the initial consequence based segmentation could be refined at this stage to take into account differences in the degradation mechanisms or in the severity of the failure potential.

A discussion of the identification of the potential degradation mechanisms can be found in Sections 5.4 and 5.6. The use of qualitative versus quantitative pipe failure potential is further discussed in Section 5.5.

2.2.5. Risk ranking

The segments or structural elements are ranked according to their associated risk, which is determined from their failure potential and the severity of the failure consequences. The ranking or categorisation criteria can be expressed as thresholds in terms of CDF and LERF or pertinent importance measures such as risk reduction worth and risk achievement worth. Each segment is placed in the appropriate place of a risk characterization matrix or plot. See the examples in Figs 2 and 3.

Pipe elements within each segment are candidate locations to be selected for the inspection programme, based on the risk characterization of the segment to which each element belongs.

2.2.6. Inspection element selection

In this step, the revised set of inspection requirements is defined. Specific locations are selected for the inspection programme based on the segment's risk ranking and a set of practical considerations that bear on the feasibility and effectiveness of the specific inspection. The number of locations selected for inspection would be a function of the RI-ISI methodology selected for use. For those locations selected for NDE inspections, the inspections are focused on the type of degradation mechanism identified earlier. The ability to focus the examination on specific damage mechanism(s) enhances the effectiveness of the retained inspections. All locations, regardless of risk classification and element selection results are typically also subjected to pressure and leak testing requirements.

2.2.7. Evaluation of risk impact of changes to the inspection programme

The final step related to implementing a RI-ISI programme has to do with showing the impact of the proposed RI-ISI on plant safety (i.e. the change in risk). It should be confirmed that the initial selection of elements for the RI-ISI programme does not produce an unfavorable and unacceptable risk impact. Depending upon the RI-ISI application and its results, qualitative criteria, bounding estimates of risk impacts, or realistic estimates of risk impacts may need to be developed. If unacceptable risk impacts do occur, then adjustments to the selection of elements to meet the risk acceptance criteria may be needed.

The RI-ISI methodology used and the accompanying acceptance criteria will most likely be determined by the plant and its respective regulatory body. As an example, when using USNRC Regulatory Guide 1.174, it must be shown that the changes in risk due to changes in the inspection programme do not have a significant risk impact as determined by changes in CDF or LERF. In the Czech Republic, a Safety Guide has been developed for risk-

			Conditional Consequence				
			Very Low	Low	Medium	High	Very High
			$<10^{-6}$	10^{-6}–10^{-5}	10^{-5}–10^{-4}	10^{-4}–10^{-3}	$>10^{-3}$
Failure potential	Very High	$>10^{-4}$					
	High	10^{-5}–10^{-4}					
	Medium	10^{-6}–10^{-5}					
	Low	10^{-7}–10^{-6}					
	Very Low	$<10^{-7}$					

Fig. 2. Example of risk matrix (the qualitative and quantitative values are for illustrative purposes only).

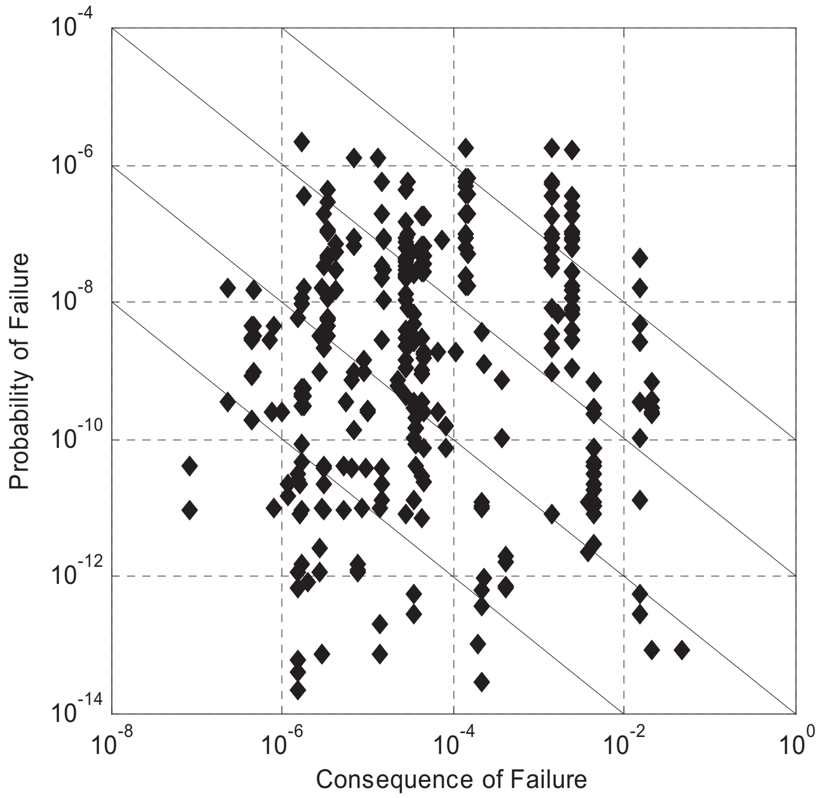


Fig. 3. Example of risk plot (the straight lines represent loci of constant risk).

informed applications. While similar to Reg Guide 1.174, it does have slightly different acceptance criteria. Other countries may have different acceptance criteria. This is further discussed in Section 5.10.

2.2.8. Long term management of a RI-ISI programme

The final step of the process is to document the RI-ISI programme and implement monitoring strategies so that the programme is managed on a long term basis. The frequency and content of these updates will most likely be agreed upon by the plant and its regulatory body. However, these updates may be more frequent if dictated by updates to the PSA or as new degradation mechanisms are identified.

Additionally, as changes to plant design are implemented, changes to the inputs associated with RI-ISI programme segment definition and element selections may occur. It may be important to address these changes to the inputs used in any assessment that may affect resultant pipe failure potentials used to support the RI-ISI segment definition and element selection. Some examples of these inputs would include:

- Operating characteristics (e.g. changes in water chemistry control);
- Material and configuration changes;
- Welding techniques and procedures;
- Construction and pre-service examination results;
- Stress data (operating modes, pressure, and temperature changes).

In addition, plant design changes could result in changes to a plant's CDF or LERF, which in turn could result in a change in consequence of failure for system piping segments.

3. OVERVIEW OF THE RI-ISI APPROACHES AND STATUS OF CURRENT APPLICATION

3.1. EPRI METHODOLOGY

The EPRI RI-ISI methodology consists of the following steps:

(1) Definition of RI-ISI programme scope;
(2) Failure mode and effects analysis (FMEA) of pipe segments;
(3) Characterization of risk segments;
(4) Inspection element selection;
(5) Evaluation of risk impact of changes to inspection programme
(6) Long term management of the RI-ISI programme.

Each of these steps is described in the following. The following references give a good overview of the methodology: [1–6].

3.1.1. Definition of RI-ISI programme scope

The first step is to decide the scope of the RI-ISI programme. Options include:

- Large scope applications including all safety class piping systems and non-safety classified piping;
- Selection of individual piping system(s), or alternative piping system scope;
- A subset of a class of piping systems (limited to examination category B-J welds as described in ASME code case N-560).

TABLE 1. CORRESPONDENCE OF CONSEQUENCE CATEGORIES TO NUMERICAL ESTIMATES OF CCDP AND CLERP

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

3.1.2. Failure modes and effects analysis

A failure modes and effects analysis (FMEA) of the piping systems within the RI-ISI programme scope is performed in order to identify the potential degradation mechanisms and the consequences of pipe breaks. This is the most resource consuming part of a RI-ISI application. The FMEA is normally performed on a system by system basis and leads to the definition of piping segments characterised by the same failure potential and consequence potential.

Based on criteria derived from service experience and pipe failure data, each piping segment is checked whether it is susceptible to a certain degradation mechanism, and accordingly assigned to a rupture potential category (high, medium or low).

The consequence analysis is a part of the FMEA phase. Consequences of pipe rupture are assessed in terms of the conditional probability of core damage given a pipe rupture (CCDP) and the conditional probability of large early release given a pipe rupture (CLERP). These assessments require quantitative risk estimates obtained from the plant specific PSA models available for the given plant. Such estimates can be used directly or used to calibrate tables that are applied to rank pipe rupture consequences for each location in the piping system. Consequence ranking of pipe ruptures can be determined by application of these tables, without extensive PSA computations. A consequence category is assigned to each piping segment according to Table 1.

3.1.3. Risk ranking

Each segment is placed onto the appropriate place on the EPRI risk characterization matrix shown in Table 2. This is based on three broad categories of failure potential and four broad categories of consequence potential. Several risk categories are identified, subsequently combined into three risk regions (low, medium or high) for more robust and efficient utilization, see Table 2. For risk category 1, 2, or 3, at least 25% of the elements belonging to each category should be inspected. For risk category 4 or 5, at least 10% of the elements should be inspected. The numbers resulting from taking the percentages are rounded up to the next integer.

3.1.4. Selection of the inspection elements

The selection of individual inspection elements depends on the degradation mechanism present, physical access constraints, radiation exposure, and cost considerations. An inspection for cause process is implemented at each inspection location. Therefore, examination methods, inspection volumes, and acceptance and evaluation criteria must be specifically designed for the degradation mechanism(s) active at the inspection location. The element selection is performed by a multi-disciplinary team, typically consisting of representatives of the teams that have performed the consequence and degradation analysis and experienced plant personnel.

3.1.5. Risk impact assessment

In the EPRI methodology, it must be shown that changes in risk due to changes in the number and locations of the inspections do not have a significant adverse risk impact as determined by changes in CDF or LERF. The risk impact assessment process may involve one or more of the following: application of qualitative criteria, bounding estimates of risk impacts, realistic estimates of risk impacts, and/or adjustments to the selection of elements to meet the risk acceptance criteria.

TABLE 2. EPRI RISK CATEGORIZATION 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

3.2. PWROG METHODOLOGY

The PWROG (PWR Owners Group) methodology is a quantitative RI-ISI approach that integrates an engineering analysis and a probabilistic safety assessment (PSA) for RI-ISI purposes [7–10]. The steps of the PWROG procedure are shown in Fig. 4, and briefly described in the following.

3.2.1. Definition of RI-ISI programme scope

The first step is to decide the scope of the RI-ISI programme. Options include:

- Large scope applications including all safety class piping systems and non-safety classified piping;
- Selection of individual piping system(s), or alternative piping system scope.

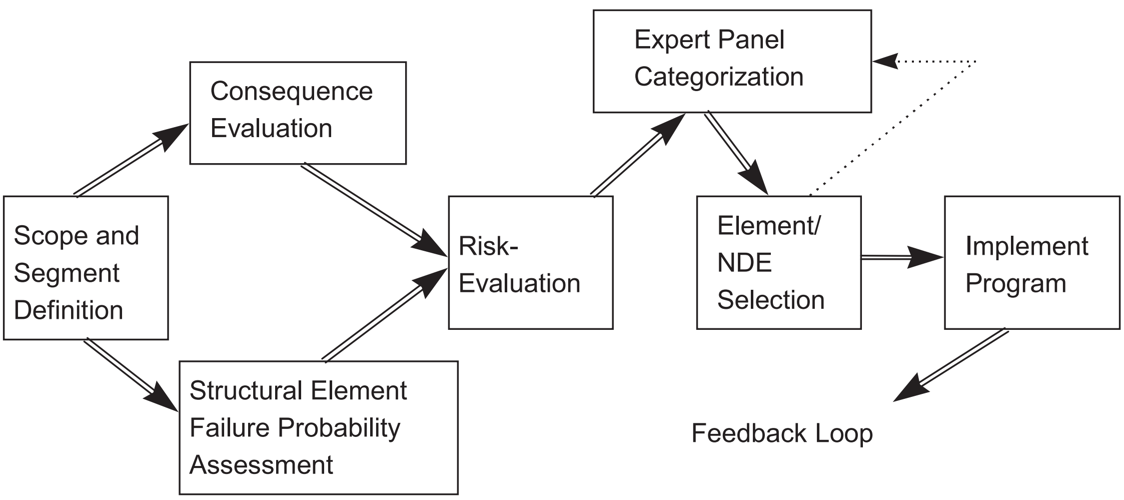


Fig. 4. PWROG RI-ISI process steps.

3.2.2. Piping segment definition

The piping systems are divided into segments having the same consequences of failure in terms of an initiating event and/or system failures included within the PSA model. Segment boundaries are primarily, but not exclusively, based on changes in consequences. The engineering aspects of determining segment boundaries include isolation points (motor and air operated values, check values), pipe size changes, flow splits and joins.

3.2.3. Failure mode and probability estimation

In the PWROG approach, all significant degradation mechanisms present in the segment are assigned on a single weld, considering the operational and environmental characteristics of that weld. For piping segments that have multiple pipe sizes, the failure probability for each pipe size may be determined and the highest probability used to represent the segment. A probabilistic fracture mechanics code (SRRA) implementing Monte Carlo simulation is 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 methodology considers several size pipe breaks as different failure modes including a small leak, a disabling leak, and a full break.

3.2.4. Consequence evaluation

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 conditional core damage frequency or probability (CCDF/CCDP) and conditional large early release frequency or probability (CLERF/CLERP). The pipe segment failure events are not individually built into the PSA model, instead the consequences of surrogate events related to each piping segment are determined through requantifying the PSA results. The PSA model, together with the evaluation of the failure probabilities, is used to calculate the pressure boundary core damage frequency (CDF) or LERF, and the importance measures are calculated relative to that value.

3.2.5. Risk ranking evaluation

The risk ranking of piping segments is first based on the quantitative safety significance (high safety significance, HSS, or low safety significance, LSS), which is defined by the risk reduction worth (RRW) measures that are calculated for each segment by conventional PSA methods. The final ranking of piping segments as HSS or LSS is conducted by a plant expert panel which combines the PSA results and engineering insights. The expert panel considers all the information described earlier, i.e. the system characteristics, the pipe segment characteristics, the risk related information, as well as other non-risk related (deterministic) information. In the expert panel review most of the time is dedicated to reviewing segments with RRWs between 1.001 and 1.005 to determine the final safety significance.

3.2.6. Structural element selection

Based on the risk ranking, the segments are placed in the structural element selection matrix, shown in Fig. 5. A set of criteria is used for selection of different structural elements (piping segments, locations) for examination (inspection). All locations that are HSS and highly susceptible to an active degradation mechanism are examined. A statistical model is used to identify the number of examinations for elements not affected by an active degradation mechanism. A minimum of one location is selected in each HSS segment. In addition, all postulated degradation mechanisms in the HSS segment must be addressed.

3.2.7. Risk impact assessment

The risk reduction achieved by the RI-ISI programme can be quantified by the PWROG methodology. The proposed RI-ISI programme is compared to the original inspection programme to determine if certain criteria are

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

Fig. 5. PWROG structural element selection matrix

met in going from the original programme to the RI-ISI programme. These criteria include an overall risk neutrality or risk reduction; in dominant systems, a risk neutrality or risk reduction; and in non-dominant systems, small defined increases in risk.

3.3. ASME CODE CASE N-716

ASME code case N-716 is a streamlined process, based on the EPRI RI-ISI approach, for implementing and maintaining a RI-ISI programme, based upon lessons learned from numerous approved RI-ISI applications [11]. The N-716 approach differs from the EPRI RI-ISI approach in two respects. First, consequence assessment is not required for every segment. The consequence assessment has been replaced by identifying generic HSS segments and a structured search for plant specific HSS segments. The second difference is that partial scope application, which is allowed by previous RI-ISI approaches, is not allowed by N-716.

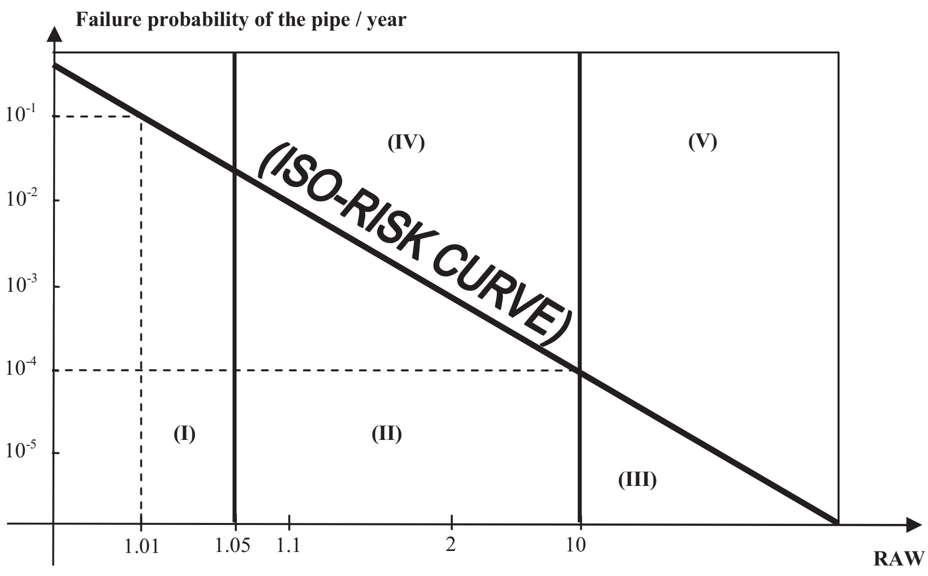
According to the process, the inspection selection should equal 10% of the HSS welds, which are selected as follows:

(1) Examinations are to be allocated according to the population identified as susceptible to each degradation mechanism and degradation mechanism combination;
(2) For the RCPB, at least two thirds of the examinations are to be located between the first isolation valve (i.e. isolation valve closest to the RPV) and the reactor pressure vessel;
(3) A minimum of 10% of the welds in that portion of the RCPB that lies outside containment (e.g. portions of the main feedwater system in BWRs) must be selected;
(4) A minimum of 10% of the welds within the break exclusion region must be selected.

3.4. OPTIMIZATION OF MAINTENANCE STRUCTURES

The L'Optimisation de la Maintenance par la Fiabilité (OMF)-Structures methodology was developed by Electricité de France (EDF). It focuses on the maintenance optimization of piping outside the reactor coolant system. The OMF approach includes similar steps as the US RI-ISI methodologies: the piping systems are broken down in segments, for which safety significance and degradation potential are evaluated. The aim is to optimize the maintenance of pipes and supports with respect to safety, availability and maintenance costs.

The contribution of structural failures to core damage frequency is evaluated using the PSA model, where possible, and the segments are classified as 'non-safety severe', 'safety severe' or 'very safety severe', based on the risk achievement worth (RAW). PSA analyses are complemented with deterministic operating analysis. For the



(I) (II) : non critical
(III) : non critical, but very safety severe
(IV) (V) : critical

Fig. 6. Component criticality evaluation in the OMF structures approach.

segments classified as ‘severe’ and ‘very severe’, a criticality analysis is done at component level to identify critical failure modes.

A detailed analysis of operating experience and information derived from degradation and reliability models is essential in selecting candidate maintenance scenarios and in the final choice of the maintenance programme (particularly task intervals).

Figure 6 defines risk regions considered in the OMF structures, in the case of a failure mode modelled in the PSA.

3.5. SWEDISH NUCLEAR POWER INSPECTORATE’S REGULATIONS (SKIFS) METHODOLOGY

The existing ISI programmes in Sweden are based on the Swedish regulations SKIFS1994:1 [12]. 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 approach does not make use of PSA results.

The damage index is a qualitative measure of the probability that cracking or other degradation occurs in the specific component. The consequence index is a qualitative measure of the occurrence of such cracking or other degradation resulting in core damage, damage to the reactor containment, release of radioactive material or other damage that may lead to injuries or affect health. The consequence index is determined by factors such as: pipe position relative to 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 Fig. 7.

The SKIFS 1994:1 gives the following requirements: The majority of components within inspection group A must 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 NDT 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.

Damage index	Consequence index 1	2	3
I	A	A	B
II	A	B	C
III	B	C	C

Fig. 7. Risk matrix for ranking of components according to SKIFS. Inspection group A = high risk; inspection group B = medium risk; inspection group C = low risk.

3.6. NUCLEAR RISK-BASED INSPECTION TOOL

The RI-ISI procedure, nuclear risk-based inspection tool (NURBIT) is a quantitative procedure (with associated software) for assessment of austenitic steel piping systems susceptible to intergranular stress corrosion cracking (IGSCC) and vibration fatigue. The procedure is developed by the Swedish company Inspecta Technology, formerly DNV.

Leak and failure probabilities are calculated for individual welds by means of a probabilistic fracture mechanics model, taking leak detection systematically into account. Initiation and growth of IGSCC is modelled, and the model takes into account effects of the complex crack shape that evolve during the wall penetration event. The effect of the crack shape, as well as plasticity effects, is important for accurate leak rate estimation. The procedure also models the effect of the probability of detection (POD curve). The code has been validated to other structural reliability models, for example in the NURBIM project, and it is concluded that the model accurately reflects the importance of different variables.

The quantitative assessment of consequence of failure is based on PSA results. Risk ranking is done in terms of the absolute risk measure CDF, assigned to the failure of individual welds.

NURBIT has a procedure to select safety significant sites for inspection, based on the evaluation of the plant specific relative risk levels. No expert panel has been used in the pilot studies. The effect of the inspection interval is systematically included in the quantitative optimization process. The NURBIT software package provides a means to optimize the inspection programme with respect to minimum risk, inspection costs, failure costs, and radiation dose for the inspection personnel.

3.7. RI-ISI APPROACH FOR THE LOVIISA NUCLEAR POWER PLANT

The principles in the Loviisa nuclear power plant RI-ISI follow to some extent the ASME XI Appendix R, Method B (‘EPRI methodology’), but with some important clarifications:

— The RI-ISI scope consists of the whole unit, and it is thus not limited only to piping systems of safety classes 1, 2 and 3. Screening is done on the basis of system CDF/LERF: all systems having a contribution larger than 10^{-6} to CDF or 10^{-7} to LERF are included in the scope.
— The consequence assessment is based on the results of the extensive PSA analysis, covering, e.g. flood analyses and low power states.
— The failure assessment is qualitative, but uses different criteria than the US approach.
— An independent expert panel evaluates the risk ranking, and the panel work is monitored by observers from the safety authority.

The experience from the RI-ISI application so far indicates that several new features will be brought to the future ISI programme:

— Many totally new systems and new portions of the system of the existing programme are included;
— Small diameter instrumentation piping of the primary systems is to be inspected, implying the development of new methods and techniques.

See Appendix II for a more detailed description of the RI-ISI approach implemented in Finland.

3.8. STATUS OF RI-ISI APPLICATIONS

In the USA, the NRC has approved both the EPRI and PWROG methodologies as valid alternatives to ASME Section XI. Further, it has approved plant specific applications relying on code case N-716 (i.e. EPRI Streamlined RI-ISI Approach). RI-ISI is currently applied to the majority of the units in US nuclear power plants. In other countries, varying regulatory positions exist, ranging from country specific methodologies to adaptation of existing approaches. The number of applications is constantly growing and is briefly summarised in the table below.

Additional information on RI-ISI methodologies and the status of applications is documented in Ref. [13].

Country	**Status**
Applications	
Bulgaria	Partial scope application of PWROG methodology
Finland	Full scope RI-ISI projects under way (Loviisa WWER-440 & Olkiluoto BWR), using similar risk matrix as in EPRI methodology, but not following exactly the methodology
Mexico	EPRI application in process (Laguna Verde BWR), class 1&2
Republic of Korea	Class 1 and 2 applications of PWROG methodology
South Africa	Application of EPRI methodology (Koeberg PWR), class 1&2
Spain	Several applications for RI-ISI programmes have been approved for class 1 piping systems (PWROG)
Sweden	Ringhals has applied PWROG methodology, approval process not completed yet All Swedish plants have ISI programme based on SKIFS 1994:1
USA	EPRI methodology: 68 plants PWROG methodology: 17 plants EPRI & PWROG methodology: 5 plants Other methodologies: 2 plants No RI-ISI: 11 plants Note: of the above, 16 are transitioning to the EPRI Streamlined RI-ISI approach (5 EPRI, 5 Westinghouse, 6 None)
Pilot studies	
Czech Republic	EPRI pilot studies, several systems in Temelin (WWER-1000) and Dukovany (WWER-440)
France	OMF structures methodology piloted to 12 systems
Lithuania	NURBIT RI-ISI approach pilot
Slovakia	Application under way, future steps dependent on pilot study results
Sweden	Oskarshamn and Forsmark pilot studies using NURBIT RI-ISI approach Pilot Study under way at Forsmark, Unit 3 using EPRI methodology
Switzerland	EPRI pilot study at Leibstadt, PWROG pilot study at Beznau
Ukraine	EPRI pilot study at Khmelnitsky WWER-1000
Other	
Belgium	Participating in international activities (e.g. RISMET)
Japan	Some activities taking place (e.g. RISMET)
Taiwan (China)	Some activities taking place
UK	Risk-based ISI applied for nuclear submarines, not for nuclear power plants

4. RI-ISI IMPLEMENTATION PROCESS AND ORGANIZATION

Implementation of a successful RI-ISI programme requires the use and integration of inputs and analyses from multiple disciplines within the plant organization. This section discusses the various required inputs and provides suggestions for documenting these inputs for use in developing the RI-ISI programme, as well as supporting future updates and improvements.

4.1. ORGANIZATION

The overall responsibility for the RI-IS programme lies with the plant operator (licensee). It is recognised that utilities in different countries will have different organizational structures. The following provides an example of an organizational structure and interfaces that can be used as a guide to what is required in order to implement a risk informed ISI programme. A similar structure has been suggested in the ENIQ Framework Document on RI-ISI [14].

With reference to Fig. 8, the main parties/personnel involved are described in the following:

- **The RI-ISI responsible person:**
  The RI-ISI responsible person is responsible for the development and acceptance of the final RI-ISI programme.
- **The RI-ISI team:**
  The RI-ISI team has the responsibility of developing the RI-ISI programme and following it through to its implementation.

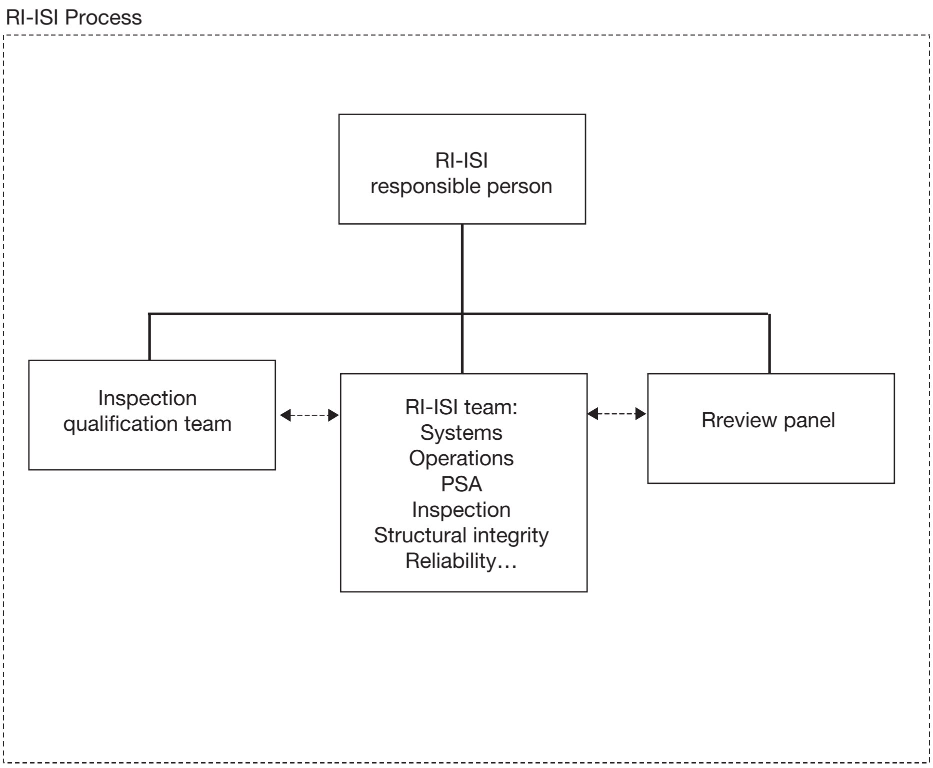


Fig. 8. Possible organizational structure for the RI-ISI process.

- **The RI-ISI review panel:**
  The purpose of the review panel is to provide an independent review function of the RI-ISI programme including supporting evaluations and calculations.
- **The inspection qualification team:**
  The inspection qualification team has the responsibility of identifying appropriate inspection locations, inspection techniques and qualification procedures for the proposed ISI programme.

4.2. PLANT INFORMATION REQUIREMENTS

The evaluation process begins by assembling information based on the scope of the application. This information is necessary for defining the systems to be analyzed and for identifying the elements to be evaluated within the system boundaries. The information and support that is needed for the evaluation process is summarized as follows.

Evaluation documentation:

- ISI programme plan;
- ISI isometrics;
- P&IDs (for applicable systems);
- Spatial databases;
- Flooding/spray studies;
- System training manuals;
- PSA information;
- Material specifications;
- Line lists;
- Valve lists;
- System design basis documents;
- Inspection programmes for active degradation (e.g. FAC, IGSCC);
- Plant service experience (i.e. cracking, water hammer, FAC event history, etc.);
- Normal, abnormal and emergency operating procedures;
- Alarm response procedures.

Evaluation personnel support:

- ISI personnel;
- PSA personnel;
- Operations personnel;
- System/design engineers;
- Material engineers.

Miscellaneous:

- Cost drivers (i.e. which systems, locations, etc.);
- Worker exposure (i.e. hot spots, time consuming examinations);
- Walkdown access.

4.3. DOCUMENTATION

Documentation of the RI-ISI application involves two general requirements:

(1) Documentation of the risk-informed evaluation;
(2) Documentation of the revised ISI programme.

This section discusses those attributes that will be useful in providing the necessary documentation. As ISI programmes for all plants must meet their country's specific documentation requirements, the information below is provided for illustration.

Two sets of documentation are normally required. The first set involves developing the RI-ISI application and the second set involves regulatory review and approval.

The first set of documentation is required to develop the analysis and support the plant review of the RI-ISI evaluations, as well as a means for future re-creation and/or modification of the risk-informed application. This type of documentation tends to consist of marked up drawings and supplemental calculations that will need to be retained in hard copy, microfilmed/fiched or scanned media. This is discussed in Section 4.3.1.

The second set of documentation is the actual submittal to the regulatory body and subsequent regulatory approval. An example of the type of information that may be contained in such a submittal is further described in Section 4.3.2.

Both sets of information would generally be expected to be retained and be retrievable on site.

4.3.1. RI-ISI evaluation documentation

This documentation is required to support the development and review of the RI-ISI application and to provide a means of recreating and/or modifying the results as part of maintaining a living RI-ISI programme. This information generally consists of marked up drawings and supplemental calculations that need to be retained in hard copy, microfilm or scanned media. It would typically contain the following types of information:

Scope:

- Definition;
- Basis for scope definition.

Piping system configuration:

- Current ISI programme;
- ISI isometric drawings;
- Piping and instrumentation diagrams (P&ID);
- Piping design specification;
- Material and fabrication specification;
- Inspection cost data.

Failure potential evaluation:

- Degradation mechanisms identified;
- System training manuals;
- Design basis documents;
- Operating conditions (e.g. operating temperatures, water chemistry);
- Plant specific and industry service experience;
- Segment properties (e.g. material, insulation).

Consequence evaluation:

- PSA analyses;
- FSAR;
- Spatial studies (e.g. internal flooding, pipe whip, spray);
- Spatial databases (e.g. component locations, plant layout);
- Results of direct and indirect effects analysis (e.g. initiating events, mitigating equipment failures).

Risk evaluation:

- Consequence and failure potential results;
- Risk ranking results;
- Expert panel results.

Element selection:

- Number and location;
- Examination methods and qualification requirements;
- Previous ISI programme;
- Integration with other inspection programmes.

Risk impact:

- Evaluation of results;
- Comparison with acceptance criteria.

4.3.2. Regulatory submittal documentation

The results of the RI-ISI analysis may be required to be submitted to the plant's regulatory body for approval. Although each regulatory body may require different information, the following list provides one example of the information contained in a so-called 'template submittal' [15] that has been used in the past:

- Justification for statement that PSA is of sufficient quality;
- Summary of risk impact;
- Summary of all inspection programmes that would be impacted;
- Revised FSAR pages impacted by the change, if any;
- Process followed and exceptions to methodology, if any;
- Summary of results of each step (e.g. number of segments in each risk category, number of locations to be inspected, etc.);
- Compliance with regulatory guides and principles (or any exceptions);
- Summary of changes from current ISI programme.

4.4. RI-ISI REVIEW AT THE REGULATORY BODY

Each country will need to determine how and what must be reviewed by the regulatory authorities, depending upon the ISI requirements in each regulatory framework. Early and close cooperation between licensees and the regulatory body is recommended to effectively and efficiently implement RI-ISI in each country. Prior agreement of several fundamental principles (scope of the application, risk acceptance criteria, applicability of other inspection activities) should be achieved before significant plant specific analyses are attempted. Once these agreements are in place, the plant operator and regulatory body can determine the appropriate review and approval process.

5. DISCUSSIONS ON IMPORTANT ASPECTS

This section presents a number of technical and programmatic topics that may be of interest to Member States wishing to implement a RI-ISI programme. While these topics have been addressed in some manner by Member States with existing RI-ISI programmes, there may be alternative or country unique aspects to each of these issues.

5.1. FULL SCOPE vs. PARTIAL SCOPE

In principle, a RI-ISI application can be conducted on a variety of scopes. That is, the application can be conducted on a large scale (e.g. a whole plant application), a system specific application (e.g. a single system) or a class of components (e.g. the reactor coolant pressure boundary). Care needs to be taken to ensure that regardless of the scope of application, the RI-ISI programme produces consistent and reliable results. The final decision on the scope of the application will be made between the plant operator and their regulatory body.

This inherent flexibility will allow the plant operator to determine the extent of the application from a cost–benefit perspective, while instilling confidence in the regulatory body and other stakeholders that the safety of the plant is maintained or improved. Of the RI-ISI programmes approved and implemented to date, the vast majority have been partial scope in nature.

When a partial scope application has been selected, it is recommended that the boundaries for the application be defined in a coherent manner. Examples of this include:

- A system, e.g.
  may include multiple classes of piping;
  limited to piping subjected to the current ISI programme.
- A pipe safety class, e.g.
  Class 1 only;
  Class 1 and 2 only.
- A safety function, e.g.
  Reactor coolant pressure boundary.

5.2. OTHER INSPECTION PROGRAMMES

Many risk-informed in-service inspection methodologies and programmes were originally developed as alternatives to deterministic programmes such as ASME Section XI, NE-14 or others. In some cases other inspection programmes have been implemented by plant operators. The key difference between these other inspections 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 deterministic ISI programme. These other inspection programmes have names such as ‘augmented’ or ‘owner defined’ programmes. In some countries, these augmented inspections have been incorporated into the overall ISI programme.

Because of these differences in intent, it is important that if these augmented programmes exist, they should be integrated into (or coordinated with) the RI-ISI programme in a manner that is logical and defensible.

5.3. CONSEQUENCE EVALUATION

Because pressure boundary failures tend to be of low risk significance, there are very few pressure boundary failures modelled in the PSA. As such, a robust RI-ISI programme requires that a comprehensive assessment for a spectrum of piping failures, from pipe leaks to ruptures, be explicitly considered. The impacts of these failures can be direct, indirect or a combination of both, as discussed below:

- Direct consequences — A failure resulting in a diversion of flow and a loss of the train and/or system or an initiating event (such as a LOCA).
- Indirect consequences — A failure resulting in a flood, spray, jet impingement or pipe whip, spatially affecting neighboring structures, systems and components.

Spatial effects are an example of indirect effects caused by pressure boundary failures. Spatial consequences of the break are determined based on the location of the analyzed break and the relative position of important equipment. The evaluation of the environmental and spatial effects is performed by the following two methodologies, the mechanistic approach and the effect oriented approach. In the mechanistic approach, leak and break locations are postulated in every piping system and the indirect consequences are evaluated for each location. In the effect oriented approach the focus is placed on the equipment which needs to be protected from the effects of pipe failure and evaluations are only performed for locations where there is a real potential for interaction (e.g. causes initiating event or loss of mitigating ability). In order to streamline the consequence evaluation, the analyzed locations of the break should be consistent with locations analyzed in other spatial analyses performed for the plant (e.g. internal flood analysis or fire analysis). The effect oriented approach requires a comprehensive and robust analysis. Some older screening analysis may not have the required technical adequacy to support RI-ISI programme development. The presence of important equipment in a specific location can be identified through these analyses but may need to be confirmed by additional walkthroughs.

Another key component of the consequence evaluation is identifying and assessing the possibility of isolating a break. 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 could be isolated by an operator action, given successful diagnosis, sufficient time, direction and availability of equipment.

Once the consequence of the piping failure has been identified, its importance needs to be determined. This is usually done by using the plant specific PSA to identify a corresponding consequence ranking (e.g. CCDP, CCDF, CLERP, CLERF). The ranking also needs to reflect other important considerations such as external events and other modes of operation (e.g. low power, shutdown). The technical adequacy of the plant specific PSA should be commensurate with how the PSA is used in the consequence evaluation and ranking.

5.4. DEGRADATION MECHANISMS

Service experience has not shown a strong correlation between actual failure probability and design stresses. Failures typically result from degradation mechanisms and loading conditions (i.e. IGSCC, flow accelerated corrosion, thermal stratification, etc.) not anticipated in the original design stress reports.

As such, operating experience data provides useful qualitative and quantitative information on the degradation of structural components. Operating experience data covers not only leak and rupture data, but also other information on the presence on non-critical levels of degradation, such as small defects and wall thinning. This information can be of considerable value in the development of and assessment of structural failure probabilities.

Since the likelihood of a piping failure is strongly dependent upon the presence of an active degradation mechanism, an important goal is to identify and evaluate the type of degradation mechanism present in a pipe segment. This can be accomplished by comparing actual piping design and operating conditions to a well defined set of material and environmental attributes.

Table 3 provides a high level example of this approach.

TABLE 3. EXAMPLE OF SOME DEGRADATION MECHANISM CRITERIA AND SUSCEPTIBLE REGIONS

Degradation Mechanism		Criteria
TF	TASCS	• Potential exists for low flow allowing mixing of hot and cold fluids; • Potential exists for leakage flow past a valve allowing mixing of hot and cold fluids; • Potential exists for convection heating in dead-ended pipe sections; • Potential exists for two phase (steam/water) flow; • Potential exists for turbulent penetration into a relatively colder branch pipe connected to header piping containing hot fluid.
	TT	• Potential for relatively rapid temperature changes including: • Cold fluid injection into hot pipe segment; • Hot fluid injection into cold pipe segment.
SCC	IGSCC (BWR)	• Sensitized materials (e.g. depletion of chromium in regions adjacent to the grain boundaries in weld heat-affected zones); • High stress applied (e.g. residual welding stresses); • A corrosive environment (e.g. high level of oxygen or other contaminants);
	IGSCC (PWR)	• Certain austenitic stainless steels; • Function of operating temperature; • Tensile stress (e.g. Residual stress); • Initiating contaminants.
	TGSCC	• Certain austenitic stainless steels; • Function of operating temperature; • Tensile stress (e.g. residual stress); • Initiating contaminants.
	ECSCC	• Austenitic stainless steel; • Tensile stress (e.g. residual strees); • Outside piping surface is exposed to wetting from concentrated chloride bearing environments; • Function of operating temperature.
	PWSCC	• Piping material is Inconel (Alloy 600), and • Primary water at high temperatures; • Residual stress (e.g. weld repair).
LC	MIC	• Function of operating temperature; • Low or intermittent flow; • Presence/intrusion of organic material; • Water source is not treated with biocides.
	PIT	• Potential exists for low flow; • Oxygen or oxidizing species; • Initiating contaminants.
	CC	• Crevice condition; • Function of operating temperature; • Oxygen or oxidizing species.

TABLE 3. EXAMPLE OF SOME DEGRADATION MECHANISM CRITERIA AND SUSCEPTIBLE REGIONS (cont.)

Degradation Mechanism		Criteria
FS	E-C	• Cavitation source; • Function of operating temperature; • Flow and sufficient velocity present.
	FAC	• Velocity; • Dissolved oxygen; • Ph; • Fluid quality; • Chromium content.
OL	Various	• Waterhammer; • Vibratory fatigue; • Support malfunction.

5.5. QUANTITATIVE vs. QUALITATIVE ASSESSMENT OF FAILURE POTENTIAL

There are alternative ways to assess the probability of failure, ranging from purely qualitative assessment to quantification with either statistical analysis of service data or structural reliability models.

A purely qualitative assessment is based on the ranking of the segments according to their susceptibility to degradation mechanisms based on influential parameters such as material properties, loadings, environmental conditions, etc. The disadvantage of a qualitative approach is that the relative importance/severity of the degradation potential is not addressed, and thus a quantitative comparison of alternative inspection strategies cannot be done.

In a quantitative approach, failure probabilities are developed based on their susceptibility to degradation mechanisms. A disadvantage with quantitative approaches based on structural reliability modelling (SRM) is that validated models do not exist for all the potential degradation mechanisms that currently affect nuclear power plants. 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. Quantitative values may however serve to quantify relative differences in the probability of failure. Quantitative approaches can also be used to conduct sensitivity studies, for instance to assess the impact of various POD values on the ISI results.

The evaluation of the probability of failure for potential ISI sites will necessarily yield a mixture of quantitative and qualitative assessments. It should be kept in mind that in a RI-ISI process, very exact estimates are not needed, and a rather coarse evaluation provides sufficiently good results for identifying risk significant locations. This can be done by assigning, based on service experience, a plausible order of magnitude of the failure probability to the qualitatively assessed degradation potential.

5.6. SERVICE EXPERIENCE REVIEW

Each plant should perform a service history review to look for degradation. The goal of this review is to confirm that the failure potential analysis and assumptions used in the analysis are consistent with the plant and how it is operated. This is a two step process. First, an in-depth review of industry databases is conducted to characterize each plant's operating experience with respect to piping pressure boundary degradation. The system conditions, which cause or promote the existence of a degradation mechanism, are in general very similar from plant to plant. As such, determination of the potential presence of a degradation mechanism should primarily be based on industry service experience. This can be done on a plant specific basis or embedded into the RI-ISI methodology itself.

The second step of the service history review is to collect plant specific data. This review is conducted for each in-scope system not to replace, but to supplement the industry review. This step is considered necessary due to the uniqueness of particular plant designs, operating configurations and service conditions that may have resulted in the manifestation of a damage mechanism in such a manner as to not be identified in the industry review. Additionally, the site specific review will identify any mechanisms or events potentially resulting in piping failures as well as actual through wall failures. Plant specific service history is considered a key element in identifying degradation mechanism susceptibility. This information can also be utilized in the element selection process. Collection of this data allows fine tuning of the element selection process where applicable, and provides additional confirmation of the appropriate assignment of damage mechanisms. Table 4 provides an example of such a review.

5.7. SAFETY SIGNIFICANCE DETERMINATION (e.g. RISK RANKING)

As mentioned before, risk is defined as the product of the measure of the consequence resulting from a failure and the probability of that failure occurring within a given period of time. The risk associated with a failure can be illustrated by a risk matrix or a risk plot (see Figs 2, 3). The graphic representation is useful with regards to inspection planning and shows in a straightforward manner if the risk of a component is governed by the probability of failure or by its consequence. The components are evaluated with respect to specified risk criteria and critical components (for example high consequence, low probability sites) can be identified and treated appropriately.

When the consequence of failure and failure potential evaluations have been completed, they are then combined to determine the safety significance of the piping components (i.e. the risk ranking). The method of categorizing the piping can vary. For example, if the pipe failure event frequency or probability is estimated and the events are incorporated directly into the PSA logic model, importance measure calculations may be performed. Alternatively, if a conditional core damage probability (CCDP), or conditional large early release probability (CLERP) is estimated for each segment from the PSA, a CDF and LERF caused only by pipe failures may be developed by combining the conditional consequences and segment failure frequencies external to the PSA logic model. Importance measures can also be developed using these results and these measures compared to appropriate threshold criteria to support the determination of the safety significance. The calculations used in such a process should yield well defined estimates of CDF, LERF, and importance measures.

Alternatively, it is possible that the consequence of failure be represented by qualitative categories instead of quantitative estimates for each segment. In this case, the potential for pipe failure would also be developed as categories ranging from high to low depending on the degradation mechanisms present and the corresponding likelihood that the segment will fail. These consequence and failure likelihood categories are then systematically combined to develop categories of safety significance. As with the above, such a process should be developed so that there are well defined criteria for relating the consequence and failure likelihood categories to the safety significant category assigned to each combination.

The safety significance category of the pipe segment will then help determine the level of inspection effort devoted to each system. In general, safety significant areas will receive more inspections and more demanding inspections than low safety significant areas. Irrespective of the method used in the analysis, the final categorization process should result in a robust and sound safety significance determination.

5.8. ELEMENT SELECTION AND INSPECTION METHODS

The purpose of this step is to determine specifically which elements are to be inspected. An inspection for cause approach should be implemented at each inspection location. Therefore, examination methods and applied techniques, inspection volumes and acceptance and evaluation criteria are to be designed specifically for the postulated degradation mechanisms at the inspection location. The selection of individual inspection elements depends on the degradation mechanism present, physical access constraints (including a variety of aspects such as complex piping geometry, existence of counter bores and surface shape changes, weld root and weld crown limitations), radiation exposure, availability of (qualified, if required) automatic and/or manual inspection procedures and cost considerations. The element selection process is conducted by a multidisciplinary team. The team usually consists of representatives of the groups that have performed the consequence and degradation analysis and experienced plant personnel such as:

TABLE 4. EXAMPLE OF SERVICE HISTORY AND SUSCEPTIBILITY REVIEW RESULTS

Source Documents/Databases reviewed for	*Damage mechanisms*												Additionally considered	
Historical piping pressure boundary	Thermal fatigue		Stress corrosion cracking				Localized corrosion			Flow sensitive		Mechanical	Water	Other
degradation occurrences	TASCS	TT	IGSCC	TGSCC	ECSCC	PWSCC	MIC	PIT	CC	E-C	FAC	VF	hammer	findings
Station information management system	None	None	None	None	None	None	None	None	None	None	None	PBF(3)	None	(1), PD(2)
Paperless condition reporting system	PE(4)	None	None	None	None	None	None	None	None	None	None	PBF(3)	None	None
Licensing research system	PE(4)	None	None	None	None	PBF(5)	None	None	None	None	None	PBF(3)	None	PD(2)
Nuclear plant reliability database system	None	None	None	None	None	None	None	None	None	None	None	None	None	None
ISI program records	None	None	None	None	None	None	None	None	None	None	None	None	None	(1)
Control room station log	None	None	None	None	None	None	None	None	None	None	None	None	None	None
System upper level documents	None	None	None	None	None	None	None	None	None	None	None	None	None	None
Other station documents	P(6)	P(6)	None	None	None	P(6)	None	None	None	None	None	None	None	None

Legend:

P (Precursor) — This category includes identification of postulated damage mechanisms and loadings through knowledge of operating parameters, water chemistry, etc. No physical evidence of pressure boundary degradation currently exists. This category includes postulated mechanisms identified as a result of this review.

PE (Plant event) — This category includes identification of postulated damage mechanisms and loadings as a result of an observed or potential plant event (e.g. water hammer). No physical evidence of pressure boundary degradation currently exists.

PD (Physical damage) — This category includes identification of observed pressure boundary degradation as evidenced by cracking, pitting, wastage, thinning, physical deformation or other deterioration.

PBF (Pressure boundary failure) — This category includes identification of through-wall flaws resulting from the effects of an identified damage mechanism.

Notes:

1. Reference JO 00770489 and ISI programme records. Multiple indications (surface and subsurface) have been identified over time in the reactor coolant system. These indications were either removed (e.g. gouges or linear surface flaws) or evaluated (e.g. laminar or planar subsurface flaws) and determined to be Code acceptable. None of these indications were attributed to an in-service damage mechanism and are believed to have been non-service induced (i.e. fabrication or other origin).
2. Reference JO 00723183, LER 86-006 and IE Information Notice 86-108 which document corrosion wastage (boric acid) on the exterior of the P-32A discharge cold leg HPI nozzle region due to leakage from an above HPI isolation valve (body-to-bonnet leak).
3. Reference JO 00776670, JO 00776959, JO 00786058, CR 1-89-0029, CR 1-89-0312, CR 1-90-0010, LER 89-002, LER 89-010 and correspondence to the NRC which document small diameter (1½ and 1 inch NPS) cold leg drain line leaks primarily attributable to vibrational fatigue.
4. Reference NRC Bulletin 88-08, NRC Bulletin 88-11, CR C-88-0047, CR 1-92-0327, CR 1-93-0164, CR 1-98-0117 and correspondence to the NRC which address the potential for thermal stratification in the reactor coolant system.
5. Reference LER 90-021, which documents a small diameter (1 inch NPS) pressurizer level tap nozzle leak attributed to primary water stress corrosion cracking.
6. Reference Calculation No. EPRI-116-310 of the RI-ISI pilot application submittal, which identifies the potential for TASCS, TT and PWSCC in the reactor coolant system.

— ISI programme representative;
— System engineers responsible for associated systems;
— PSA representative;
— Operations representative;
— Non-destructive examination representative;
— Materials engineer with degradation mechanism and piping operation and repairs experience;
— Health physics representative;
— Personnel knowledgeable on scaffolding, insulation and craft requirements.

For all high risk locations, a highly effective inspection appropriate to the damage mechanism should be performed. Thus, the inspection method should be determined based on the degradation mechanism(s) identified by the RI-ISI evaluation. The coverage or grid spacing should be specified in relation to the degradation mechanism, in order to ensure an effective inspection. Inspection locations and coverage should be identified. Examination techniques that would reliably detect the degradation mechanisms of interest should be used.

5.9. INSPECTION INTERVAL

When sites are selected for inspection based on risk criteria, the inspection intervals should be reviewed considering the information available on the acting degradation mechanisms, the inspection capability and the risk importance of each location.

An accurate estimate of the change in risk due to the inspection interval and method can be calculated only if a structural reliability model (SRM) is available and a quantitative measure of the inspection capability (e.g. a probability of detection, or POD curve) is known. Optimal inspection intervals could be determined by means of a validated SRM coupled with POD curves. However, due to lack of validated SRMs for some degradation mechanisms and to the relatively large effort required for SRM analyses, the inspection intervals are in many cases determined by deterministic principles such as ASME Section XI [16] or by general technical experience.

5.10. RISK IMPACT EVALUATION

One of the final steps in ensuring that the RI-ISI process provides a reasonable and substantive inspection programme is the risk impact evaluation. This evaluation compares the risk associated with the new RI-ISI programme to that of the previous (typically deterministic) ISI programme. As discussed in various guidance documents [17], the required sophistication of the evaluation depends to some extent on the magnitude of the potential risk impact. For changes that may have a substantial impact, an in-depth and comprehensive PSA analysis (one appropriate to derive a quantified estimate of the total impact of the proposed change) will be necessary to provide an adequate justification. In other applications, calculated risk importance measures or bounding estimates may be adequate. In still others, a qualitative assessment of the impact of the change on the plant's risk may be sufficient.

It is expected that plant safety will be maintained or improved by implementing a RI-ISI programme. Additionally, there are other benefits beyond CDF and LERF impacts that would warrant pursuing a RI-ISI programme. These include the positive impacts on personnel safety, worker exposure and radwaste. Taken together, all these considerations should show that overall plant safety is improved.

5.11. INSPECTION QUALIFICATION

Implementation of a RI-ISI programme requires the use of NDE techniques that are designed to be effective for specific degradation mechanisms and examination locations. This ‘inspection for cause’ approach involves identification of the specific damage mechanisms that are likely to be operative, identification of the locations where they may be operative, and the selection of appropriate examination methods and volumes. This approach provides assurance that risk significant locations selected for examination will be examined using effective methods.

Throughout the NDE community, there are various qualifications requirements [16, 18] for demonstrating the effectiveness of ultrasonic examination procedures and personnel. The scope of these requirements may not include all damage mechanisms and locations relevant to RI-ISI such as FAC, MIC, and thermal fatigue that can be reliably detected even without these additional qualification requirements.

Irrespective of the above, plants implementing a RI-ISI programme maintain their responsibility to ensure that reliable examination methods and appropriate personnel are utilised in every case.

5.12. CONDUCT OF MULTIDISCIPLINARY EXPERT PANELS AND INDEPENDENT REVIEWS

The risk measure defined with the aid of the PSA models and failure potential assessments is not the only decision criterion for selecting inspection sites, as insights from several other disciplines should be integrated in the decision making process. The decision making process should address such issues as the treatment of high consequence — low failure probability sites/segments, risk outliers, radiation doses, inspection accessibility, deterministic insights, etc. Also, issues related to the quality and coverage of the PSA model, and assumptions related to degradation and consequence evaluations, can be taken into account at this stage.

Such a decision making process would be typically carried out by a group of plant experts and engineers, usually called an ‘expert panel’, meeting to discuss and take such insights into consideration. Expert panels can have different roles and compositions, depending on the organization and resources of the RI-ISI project. For example, an expert panel could be formed for an internal review of the failure probability and consequence analyses. It could also act as a forum to ensure a systematic review of the analyses and a balanced utilisation of information and expertise from several disciplines in the decision making process. A team representing various expertises, especially including NDE experts, is typically formed to make the final selection of inspection sites among a number of possible locations. An expert panel could also be an independent review body, largely consisting of members external to the RI-ISI project. See for instance the ENIQ Recommended Practice on Expert Panels for more guidance on the subject [19].

5.13. RI-ISI IN THE CONTEXT OF THE PLANT’S RISK MANAGEMENT PHILOSOPHY

PSAs were originally developed to quantify plant risk and, where applicable, to identify cost effective means of reducing plant risk. Over the years, an increase in industry and regulatory body attention to PSA technology has increased its fidelity as well as its acceptance in day to day plant operations. The use of PSA and applicable regulatory guidance varies from country to country. Established regulatory guidance and acceptance criteria enhance the successful use of this technology.

RI-ISI is one of many risk-informed applications that form part of a plant’s overall risk management strategy. It is also the one risk-informed application that uses the PSA the least directly, as many of the components evaluated in a RI-ISI application (i.e. piping components) are not included in the plant PSA model. Other risk-informed applications range from simple application of the PSA model (e.g. a quantification run) to detailed analyses supporting the application (e.g. technical specifications optimization, on-line maintenance, configuration control).

5.14. OTHER MITIGATION STRATEGIES

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. When investigating alternative mitigation strategies, it is necessary to look at the nature of the risk associated with each site (or segment). It may be possible to identify ways other than inspection to address the risk, such as continuous monitoring, improved leak rate detection, improved water chemistry treatment and follow-up, load reduction or component replacement.

For example, in lieu of inspection, components could be modified to mitigate susceptibility to degradation mechanisms such as vibratory fatigue, which can develop very fast in components that may be difficult to inspect because of materials or design (e.g. socket welds). Another example could be to develop plant procedures for piping failures that result in high consequences due to lack of, or insufficient, procedures. A third example could be the use of plant modifications to address risk significant sites (e.g. barriers, thermal transient monitoring) in lieu of inspections. These examples suggest that plant owners and regulatory bodies should adopt a wider perspective risk management, rather than in-service inspection alone.

5.15. RI-ISI PROGRAMME MAINTENANCE AND IMPROVEMENT

An integral part of implementing a RI-ISI programme is the maintenance of a living RI-ISI programme, including a periodic update at some time interval.

The results of examinations performed should be reviewed for indications of leakage or flaws. If the failure is due to a degradation mechanism postulated in the original RI-ISI evaluation, the inspection (type, frequency) specified for that degradation mechanism should be re-evaluated for adequacy. If the failure results from a degradation mechanism that is new, or different from that postulated in the original RI-ISI evaluation, all segments potentially affected by the new mechanism should be identified and the RI-ISI analysis updated accordingly.

PSA changes should be evaluated for their impacts on RI-ISI. PSA maintenance generally involves changes resulting from updated initiating event data, updated equipment performance data, modifications to plant equipment or procedures, etc. PSA upgrades involve changes to the methodology of the model (e.g. changing model platforms, changing initiating event grouping, making major changes to known significant areas), incorporating external events and other modes of operation (e.g. low power and shutdown operation). For cases when there are changes in the PSA model, the effects of these changes to the results of the RI-ISI programme have to be evaluated.

Plant design changes can be physical, programmatic or procedural, and can have impacts beyond the PSA model. Physical changes can include new piping or equipment installation, or modification of existing equipment, and should be evaluated for impact on the scope of application, failure potential, consequence or segment definition. Programmatic changes can include for instance: power uprating, change in fuel cycle, implementation of plant chemistry changes, etc. These changes should be identified by the design control process or by monitoring the licensing basis, and should be evaluated for impact on consequence evaluations or failure potential. Procedural changes can include modifications to surveillance tests or operating procedures and can be identified by the procedure change review process. Procedural changes can affect consequence evaluations and failure potential.

All segments potentially affected by any change should be identified and a determination made on whether a re-evaluation is necessary. If necessary, the re-evaluation should be performed by inserting the new information at the appropriate level of the analysis. It may not be necessary to perform the entire risk-informed selection process again, but the evaluation for the changes to the piping selections that do occur need to be documented.

5.16. LINK TO AGEING MANAGEMENT PROGRAMMES

While existing ISI programmes or more recent programmes may provide robust levels of quality and safety, the question of licence extension and/or renewal is also pertinent. Even within the current operating period, the question of component ageing is valid, to some extent. Several countries have addressed this issue by conducting a comprehensive evaluation of existing inspection programmes and plant ageing management programmes [20, 21].

Risk technology could be used to supplement this work as it could enhance the effectiveness of these reviews by enabling plant management to focus on safety significant critical components from a core damage and containment performance perspective. Also, as both processes assess critical components for their susceptibility to degradation and then identify appropriate treatment to manage this degradation, there is the possibility to streamline each effort and minimize duplication of effort. Examples of this synergistic effort include EPRI 1015138, Nondestructive Evaluation: License Renewal — Small Bore Piping Evaluation Process [22]. Additional information is provided in Refs [23, 24].

5.17. TIMING TO START RI-ISI WITHIN ISI CYCLE

The safety, cost and worker exposure benefits of this process suggest that it be implemented in a timely fashion. In fact, plants that are not currently implementing a RI-ISI programme are missing safety benefits and may be expending undue resources.

For plants that implement at the beginning of an inspection interval, the frequency of inspections is typically consistent with the deterministic inspection interval supplemented with augmented inspections (and inspection intervals) for specific degradation mechanisms. For plants that implement the RI-ISI programme during the inspection interval (e.g. at the beginning of the second period), examinations should be allocated between the previous deterministic scope and the RI-ISI scope. For example, if two thirds of the deterministic inspections have been completed, then one third of the RI-ISI inspections should be conducted in the current interval.

6. RI-ISI DEVELOPMENT ACTIVITIES

While a number of plants and countries have implemented some form of RI-ISI programme, there is also additional work completed or under way to extend this technology. Some of these activities, presented in chronological order, are summarised in the following sections.

6.1. IGSCC INSPECTION REQUIREMENTS IN BWRs

Reference [25] uses risk and performance insights to define alternative requirements to those contained in NUREG-0313. This NUREG previously defined the extent and frequencies for piping inspection requirements for stainless steel piping in BWR environments. Reference [25] provided the technical basis for revision to the NUREG requirements and is based on the consideration of inspection results and performance data over a number of years. The new requirements are also based on additional knowledge regarding the benefits of improved BWR water chemistry. Risk insights are then used to select the final set of inspection locations.

6.2. RIBA PROJECT

The need for a European review of a Risk-Informed Approach for In-Service Inspection of Nuclear Power Plant Components (RIBA) was identified in 1998. The RIBA Project was established and carried out between November 1999 and 2001, and was funded by the European Commission. The purpose of the project was to review existing risk-informed methodologies, carry out a comparative study of RI-ISI applications and finally to draw conclusions and express recommendations for the application of RI-ISI to European nuclear power plants [26–29].

An important recommendation was expressed in the desire to have a harmonised approach within Europe. Such an approach should have represented a consensus of utilities, regulatory bodys and other stakeholders and be flexible enough to be adaptable to the different regulatory environments of different countries. The project recommended drawing guidelines that would take into account the results of European RI-ISI pilot studies, and other European risk-informed initiatives relating to activities such as maintenance, testing and surveillance.

6.3. BREAK EXCLUSION REGION OF HIGH ENERGY PIPING INSPECTION REQUIREMENTS

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

In determining the locations at which breaks are to be postulated in high energy piping, the guidance provides special rules for excluding postulated breaks (e.g. containment penetration areas). These rules on the one hand recognize that these areas may require extra protection (e.g. to ensure the integrity of the containment and the operability of the isolation valves). On the other hand, the rules provide the option of not specifying breaks in these regions, so that pipe break mitigation devices, such as whip restraints, need not be constructed in these areas.

Requirements for not specifying breaks in these regions may include specific design requirements (e.g. minimize the length of piping, minimize 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 in-service inspection plan and are identified as ‘augmented’ inspections. Alternative means of determining the number and location of these inspections have been developed using risk-informed techniques [30, 10].

6.4. IAEA-TECDOC-1400

IAEA-TECDOC-1400 on Improvement of In-Service Inspection in Nuclear Power Plants, published in 2004, describes strategies for improving the effectiveness of ISI [31]. The role of in-service inspection in maintaining or improving safety and the relationship of ISI improvement to cost are examined. The strategy aimed at improving ISI effectiveness considers the entire framework of in-service inspection, including effective selection of the proper scope, inspection interval and NDE effectiveness. Both inspection qualification and risk-informed ISI are offered as two important examples of ISI improvement efforts.

6.5. NUCLEAR REGULATORY WORKING GROUP

The Nuclear Regulatory Working Group (NRWG) was an advisory expert group to the European Commission made up of representatives from nuclear safety authorities and technical support organizations from EU member and candidate countries, with Switzerland participating as observer. In 2004 the NRWG published a document summarizing the common views of the European Regulators on RI-ISI [32]. This report identified several issues where further research would be beneficial.

6.6. NUCLEAR RISK-BASED INSPECTION METHODOLOGY PROJECT

Nuclear Risk-Based Inspection Methodology (NURBIM) was an EU funded project in the 5th Framework Programme (2001–2004). The objective of NURBIM was to progress the recommendations of EURIS (European Network of Risk informed In-service Inspection, [33]) and subsequent work of ENIQ Task Group 4 (now Task Group Risk) to develop improved procedures to identify where the highest probability of failure is located in passive systems, structures and components and to provide quantitative measures of the associated risk [34].

The following issues were addressed in NURBIM [35]:

— Active and potential damage mechanisms of piping;
— Verification and validation of structural reliability models (SRMs);
— Integration of qualitative and quantitative analyses;
— Review and benchmarking of SRMs and associated software;
— Interface between probability of failure and consequence;
— Acceptance criteria and cost–benefit;
— Relationship between inspection capability and qualification;
— Case study.

6.7. IAEA RI-ISI PILOT STUDY

In 2005-06, the IAEA co-organised a pilot study of RI-ISI in the Czech Republic, which covered the primary piping and pressurizer surge lines of WWER-440 at the Dukovany nuclear power plant under TC project RER4027.

This pilot study showed a possible significant reduction of inspection sites from more than 60% to 12% of the total number of welds.

6.8. IAEA-TECDOC-1511

PSA of nuclear power plants complements the traditional deterministic analysis and is widely recognized as a comprehensive, structured approach to identifying accident scenarios and deriving numerical estimates of risks dealing with nuclear power plant operation and associated plant vulnerabilities. Increasingly, during the last years, PSA has been broadly applied to support numerous applications and risk-informed decisions on various operational and regulatory matters. The expanded use of PSA in the risk-informed decision making process requires that PSA possess certain features to ensure its technical consistency and quality.

IAEA has developed a report [36] that provides a comprehensive list of PSA applications, including RI-ISI, and describes technical features (termed ‘attributes’) of a PSA that may be useful in developing PRA application. Consideration has been also given in Reference [36] to the basic set of attributes characterizing a ‘base case PSA’ that is performed with the purpose of assessing the overall plant safety.

6.9. SMALL BORE PIPING INSPECTION FOR LICENCE RENEWAL

As a condition of gaining approval for long term operation (i.e. beyond 40 years) a number of plants have made a commitment to volumetrically inspect small bore piping. This piping had not been previously volumetrically inspected. Risk technology was used to define which locations, and the number of locations, that would need to be inspected in order to fulfil this commitment in a cost-effective manner [22].

6.10. INSPECTION REQUIREMENTS FOR NEW PLANTS

In the USA, work was conducted to determine the feasibility and appropriate timing for application of risk technology to new plant design, construction and operation. Reference [37] identified the applicability of RI-ISI to pre-service and in-service inspection of new plants. Work is under way to define any needed changes to existing RI-ISI technology to meet the unique aspects of these new plant designs as compared to the operating fleet., This work includes case studies for the revised methodology and determining a strategy for regulatory approval.

In Finland, as part of its tender for a new plant, TVO requested that a RI-ISI programme be included. Work is under way with the owner, vendor and regulatory body to define the RI-ISI programme requirements.

6.11. REACTOR PRESSURE VESSEL WELD INSPECTION INTERVAL EXTENSION

Nuclear power plants following ASME Section XI conduct reactor vessel weld inspections at 10 year intervals. The industry has expended significant cost and radiation exposure but has identified no serviced induced flaws in the reactor vessel for ASME Section XI Category B-A (reactor vessel welds) or B-D (reactor vessel to nozzle welds).

The PWROG conducted a study for decreasing the frequency of inspection by extending the inspection interval from the current 10 years to 20 years for ASME Section XI Category B-A and B-D welds for pressurized water reactors. The intent of the report is to provide a bounding assessment to be used by individual plant operators. Results of the study demonstrated that the reduction in frequency of inspection of these welds can be accomplished with an acceptably small change in risk according to RG 1.174 acceptance guidelines. US NRC approval for the topical report, WCAP-16168 [38], was granted in 2008.

6.12. OECD/NEA PROJECTS

In 2003, the OECD/NEA published a status report on developments and cooperation on RI-ISI and NDT qualification in the OECD/NEA member countries [13].

The OECD/NEA is co-ordinating with EC/JRC on the RISMET project. The NEA is also co-ordinating the OECD Piping Failure Data Exchange (OPDE) project, where piping failure event data are collected and analysed to promote a better understanding of underlying causes, impact on operations and safety, and prevention. Further, the NEA has recently supported a benchmark study on the probabilistic assessment of the structural integrity of a pressurised water reactor pressure vessel (PROSIR project).

6.13. ENIQ TASK GROUP ON RISK

The European Network for Inspection and Qualification (ENIQ) is a utility driven network that works towards a harmonized European approach to reliable and effective ISI. In particular, ENIQ task groups work on issues related to the qualification of ISI systems (task group on qualification) and to RI-ISI (task group on risk, TGR). TGR has about 20 members representing European nuclear utilities, research organizations and consultants [18].

ENIQ TGR published the European Framework Document on RI-ISI in 2005 [14]. This publication provides guidelines about the definition of risk-informed in-service inspection programmes. More recently, ENIQ TGR has been working at producing more detailed recommended practices and discussion documents on several RI-ISI issues, such as defence in depth [39], guidance for expert panels (ENIQ5), verification and validation of structural reliability models [40] and RI-ISI application to pressure vessels.

6.14. RISMET PROJECT

In January 2006 ENIQ TGR, together with the European Commission Joint Research Centre and OECD/NEA, initiated a project aimed at benchmarking RI-ISI methodologies (RISMET). The overall objective of the project was to apply various RI-ISI methodologies to the same case, i.e. selected piping systems in one nuclear power plant. The comparative study aimed at identifying the impact of the differences in methodologies on the final results, i.e. the definition of the risk-informed inspection programme. The benchmark was also intended to provide a basis for further development of existing or new methodologies. The Swedish nuclear power plant Ringhals 4 (PWR) was the host plant. The project is in the final phase at the time of writing.

7. CONCLUSIONS AND RECOMMENDATIONS

This report has provided an overview of various RI-ISI methodologies and their application in a number of Member States. Although somewhat new for some Member States, risk technology has been in use for over thirty years. The development of RI-ISI, in particular, began almost twenty years ago, and a number of plant applications have been in effect for over ten years. The benefit of this maturity in technology for other Member States lies not only in the lessons learned with respect to developing a RI-ISI programme but also in implementing and maintaining the RI-ISI programme as it moves forward.

As documented in this report, over one hundred nuclear power plants have implemented some form of RI-ISI. These RI-ISI applications have ranged from small scope applications (e.g. reactor coolant pressure boundary), to medium scope applications, up to large scope applications (e.g. large portions of safety and non-safety piping).

Because of the flexibility of RI-ISI methodologies to address various scopes of application, plant owners have the option to learn and apply this technology in a measured manner reflecting the limited resources that may be available at any one time. In support of this, the report also reviews the different disciplines that are needed to develop and implement a RI-ISI programme (e.g. PSA, system engineers, ISI personnel). One of the conclusions of this report is that most plants already have the basic infrastructure (PSA, system engineers, ISI personnel) necessary to develop and implement a RI-ISI programme.

As witnessed by the large number of plants that have implemented RI-ISI programmes, RI-ISI has the ability to focus finite plant resources on those components most important from a public health and safety perspective. Proper integration of various inspection activities (e.g. flow accelerated corrosion) also has the benefit of addressing personnel safety. In addition, experience with already approved and implemented RI-ISI programmes has confirmed that refocusing resources can reduce burden as compared to traditional deterministic approaches. As such, it is the opinion of the authors of this report that plants that have not adopted a RI-ISI programme should actively pursue developing and implementing RI-ISI programmes. These programmes would be particularly useful during licence renewal of existing plants, and could potentially be beneficial for future plants.

Appendix I

RI-ISI COST BENEFIT

I.1. Challenge

Traditional ISI programmes are based upon inspecting locations selected using deterministic criteria including design stress analyses, structural discontinuities and/or randomly. As such, this approach does not take into consideration the potential for real plant operating conditions, service experience, and potential causes of component degradation. In addition, many of these inspections are located in the containment, drywell or steam tunnel and are, therefore, only accessible during plant outages. Because of their physical location, these inspections also impact outage planning and substantially increase worker exposure.

I.2. Response

The RI-ISI methodology is an alternative to the deterministic requirements. The goal was to use risk insights to establish a piping integrity management programme, which reduces industry and regulatory burden while concurrently improving plant safety. The net effect of the effort is to reduce cost while improving plant safety. The RI-ISI programme focuses the inspection effort on piping segments representing greater risk relative to other segments. This focus on safety important piping segments distinguishes it from the deterministic programme. Specifically, in the RI-ISI programme, piping segments, which are more likely to fail or which have greater impact on safety should they fail, are inspected more frequently than other piping. This effort included developing a generic methodology, its trial application to both BWR and PWR plants, submittal of the generic methodology to the NRC and finally, review and approval of the generic methodology by the NRC.

I.3. Benefits

RI-ISI programmes have been approved at over eighty plants in the USA, and the majority of the remaining units are expected to implement RI-ISI in some form in the near future. Based on a review of 24 approved plant specific applications, reductions in the number of inspections is running at 71 per cent and totals almost five thousand inspections. When completely implemented across the US industry more than 20 000 inspections will have been eliminated while safety is enhanced. Coincident with the burden reduction associated with eliminated inspections is a substantial decrease in worker exposure and radwaste. Dose reductions of 3000 to 7000 REM per ten year inspection interval have been conservatively estimated. Using typical ranges of costs per inspection and an avoided cost per REM of $10 000, the total estimated cost savings associated with RI-ISI is $60 to 160 million. In addition, as many of these inspections would only be done during an outage, this effort also has had positive impacts on outage duration and management.

I.3.1. Basis for benefits

Based upon a ten year inspection interval, 4537 inspections have been eliminated at 24 plants, which works out to 19 503 inspections eliminated at 103 plants. Project benefits for the industry are determined in Table 5 using various cost models.

TABLE 5. INFORMATION ABSTRACTED FROM EPRI INNOVATOR, ‘INDUSTRY APPLICATION OF RISK INFORMED INSERVICE INSPECTION (RI-ISI) RESULTS IN SUBSTANTIAL COST AND WORKER EXPOSURE REDUCTIONS

Number of inspections eliminated	Cost per inspection (US $)	Inspection savings (US $)	Dose savings	Dollars per dose (US $)	Total savings (US $)
19 503	1000	19 503 000	4000 Rem	10 000	59 503 000
19 503	3000	58 509 000	5000 Rem	10 000	108 50 000
19 503	5000	97 515 000	6000 Rem	10 000	157 515 000

Appendix II

CURRENT STATUS AND EXPERIENCE OF APPLICATION OF RI-ISI IN FINLAND

This appendix describes the RI-ISI activities in Finland. The regulatory requirements related to risk-informed in-service inspections are summarised, and the status of the Finnish RI-ISI projects at the end of 2008 are reported.

II.1. Regulatory requirements

The regulatory requirements are expressed in the Finnish Regulatory Guides, so called YVL-guides. The regulatory guides that are related to the risk-informed decision making and in-service inspections are:

- YVL 2.8: Probabilistic Safety Analysis in Safety Management Of Nuclear Power Plants;
- YVL 3.8: Nuclear Power Plant Pressure Equipment. In-Service Inspection With Non-Destructive Testing Methods.

These two guides require the use of risk-informed approaches in development of inspection programmes.

YVL 2.8, that came into force in May 2003, states the following [41]: “The results of PSA shall be used in the drawing up and development of the inspection programmes of piping as per Guide YVL 3.8. Combining the information from PSA and the damage mechanisms of pipes and the secondary impacts of damages, the inspections are focused in such a way that those are weighted temporarily and quantitatively on those pipes whose risk significance is greatest. While drawing up the risk-informed inspection programme, the systems of classes 1,2,3,4 and EYT (not safety related) must be regarded as a whole. Similarly how far the radiation doses can be reduced by focusing inspections and optimising inspection periods shall be regarded.”

YVL 3.8, in force since September 2003, states [42]: “In the drawing up of inspection programmes for safety class 1, 2, 3 and 4 piping and class EYT (non-nuclear) piping as well as in the development of inspection programmes for operating plants, risk-informed methods shall be utilised to ascertain the inclusion in the inspection scope of those components posing the highest risk.” Further, the same regulatory guide states: “When changing an operating plant’s in-service inspection programme to a risk-informed inspection programme, the changes shall be made in entities such that a better safety is achieved than with the inspection programme in use”.

II.2. Pilot studies to prepare for RI-ISI applications

In order to test the applicability and resource needs of RI-ISI, pilot studies have been conducted on limited scopes both at the regulatory body and at the utility. For instance insights from the STUK pilot study [43] were utilised in the revision of the above-mentioned Finnish regulatory guides. Also, the Technical Research Centre of Finland (VTT) has conducted its own pilot studies. The pilot studies tested approaches resembling to some extent the EPRI methodology [1]. Typical of the studies is the quantitative use of plant specific PSA models. STUK and utility pilot studies have used a qualitative evaluation of piping failure potential, while VTT has also tested quantification with structural reliability models.

II.3. RI-ISI project for Loviisa

A plan for utilising RI-ISI for Loviisa unit 1 was prepared in 2005 and accepted by STUK in July 2006. The principles follow to some extent the ASME XI Appendix B (‘EPRI methodology’), but with some important clarification [44]:

- The RI-ISI scope is the whole unit, not only for piping systems of safety classes 1, 2 and 3. Screening is done on the basis of system CDF/LERF: all systems having a contribution larger than 10-6 to CDF or 10-7 to LERF are included in the scope.

— The consequence assessment is based on the results of the extensive PSA analysis, covering e.g. flood analyses and low power states.
— The failure assessment is qualitative, but using different criteria than the US approach.
— An independent expert panel is evaluating the risk ranking, and the panel work is monitored by STUK's observers.

The plan for the new RI-IS programme has been submitted to STUK for review. For the new RI-ISI programme more than 100 systems were identified and around 50 systems were segmented. Fourteen systems have segments of risk category 1 or 2 with the number of inspection locations at least 25% and 17 systems risk categories 4 or 5 with the number of inspection locations at least 10% in accordance with ASME XI Appendix R, Method B. The new risk-informed approach brought several new features into the ISI programme [44, 45]:

— Inspection locations of the primary circuit piping are reduced, but new systems are included in the new RI-ISI-programme (radiation exposure will probably be reduced);
— Small diameter instrumentation piping of the primary systems has to be inspected, implying the need of development of new methods and techniques;
— Many totally new systems and new parts of old systems are included;
— Consequence differences of redundant safety systems and system portions (mostly due to fire and flood) are taken into account;
— Exceptionally high consequence targets need special attention;
— Empty and pressure free piping need reasonable new methods and techniques.

In parallel to RI-ISI programmes there are other condition monitoring programmes, e.g. for secondary circuit and seawater cooling piping. The in-service inspections according to the RI-ISI programme of the Loviisa 1 unit were first carried out in 2008.

Documentation was an important part of the RI-ISI project work and was found to be useful for many purposes. All the old technical drawings of the piping are now documented in electronic form. The co-operation between experts of different organizations and professional areas has been fruitful for the development of ISI. The benefits of the work are maybe not gained from the reduction of inspection scope, but by hopefully reducing the total risk. The radiation exposure seems to be clearly reduced.

II.4. RI-ISI activities at Olkiluoto

TVO has recently started the RI-ISI application for the Olkiluoto 1 BWR Unit. The adopted RI-ISI approach follows largely the same principles as Loviisa RI-ISI: full scope, extensive use of plant PSA model, qualitative analysis of failure potential, and the use of an independent expert panel to review the risk ranking. The classification of degradation potential is different from both Loviisa and EPRI approaches. Since IGSCC one of the main expected degradation mechanisms in BWRs, more resolution in classification of IGSCC susceptible piping is adopted. Another difference in the approach, as compared to Lovisa, is that degradation evaluation is already done at weld level in the risk ranking phase. The RI-ISI analyses utilise the plant's pipeline database, which includes, e.g. piping dimension, material and loading information and allows various searches and analyses to support the evaluation and presentation of RI-ISI results [46].

A RI-ISI programme is also required for the Olkiluoto 3 EPR plant under construction. This RI-ISI programme is developed by the vendor (AREVA), and generally follows the EPRI RI-ISI methodology.

II.5. Public research and international co-operation

RI-ISI is also a research topic in the Finnish Research Programme on Nuclear Power Plant Safety 2007–2010, SAFIR2010 [50]. The ongoing research work, carried out by VTT, includes studies related to structural reliability modelling, inspection reliability and RI-ISI follow-up and updating. Many research activities are connected to international projects and networks. VTT participates actively in the European Network for Inspection and Qualification (ENIQ) Task Group Risk [18] work and to the OECD/NEA-JRC coordinated RI-ISI benchmark project RISMET.

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ABBREVIATIONS

CC	crevice cracking
CCDF	conditional core damage frequency
CCDP	conditional core damage probability
CDF	core damage frequency
CLERF	conditional large early release frequency
CLERP	conditional large early release probability
COL	combine construction and operating license
E-C	erosion-cavitation
ECSCC	external chloride stress corrosion cracking
ENIQ	European Network for Inspection and Qualification
EURIS	European Network of Risk-informed In-service Inspection
FAC	flow-accelerated corrosion
FMEA	failure modes and effects analysis
FS	flow sensitive
FSAR	final safety analysis report
HSS	high safety significance
IGSCC	intergranular stress corrosion cracking
ISI	in-service inspection
LC	localized corrosion
LERF	large early release frequency
LOCA	loss of coolant accident
LSS	light safety significance
MIC	microbiologically influenced corrosion
NDE	non destructive evaluation
NDT	non destructive testing
NRWG	Nuclear Regulators Working Group
NURBIM	nuclear risk-based inspection methodology
NURBIT	nuclear risk-based inspection tool
NUREG	Nuclear Regulatory Commission Regulation
OL	operating loads
OMF	optimization of maintenance for structures
OPDE	OECD piping failure data exchange
R&IDs	pipe and instrumentation drawings
PIT	pitting
POD	probability of detection
PSA	probabilistic safety assessment
PWSCC	primary water stress corrosion cracking
RAW	risk achievement worth
RCPB	reactor coolant pressure boundary
RIBA	risk-informed approach for in-service inspection of nuclear power plant components
RI-ISI	risk-informed in-service inspection
RISMET	RI-ISI methodology benchmark project
RPV	reactor pressure vessel
RRW	risk reduction worth
SCC	stress corrosion cracking
SKIF	Swedish Nuclear Power Inspectorate's Regulations
SRM	structural reliability modelling
SRRA	structural reliability and risk assessment
TASCS	thermal stratification cycling and striping
TF	thermal fatigue
TGR	task group on risk
TGSCC	transgranular stress corrosion cracking

TT	thermal transient
TVO	Teollisuuden Voima Oy
US NRC	US Nuclear Regulatory Commission

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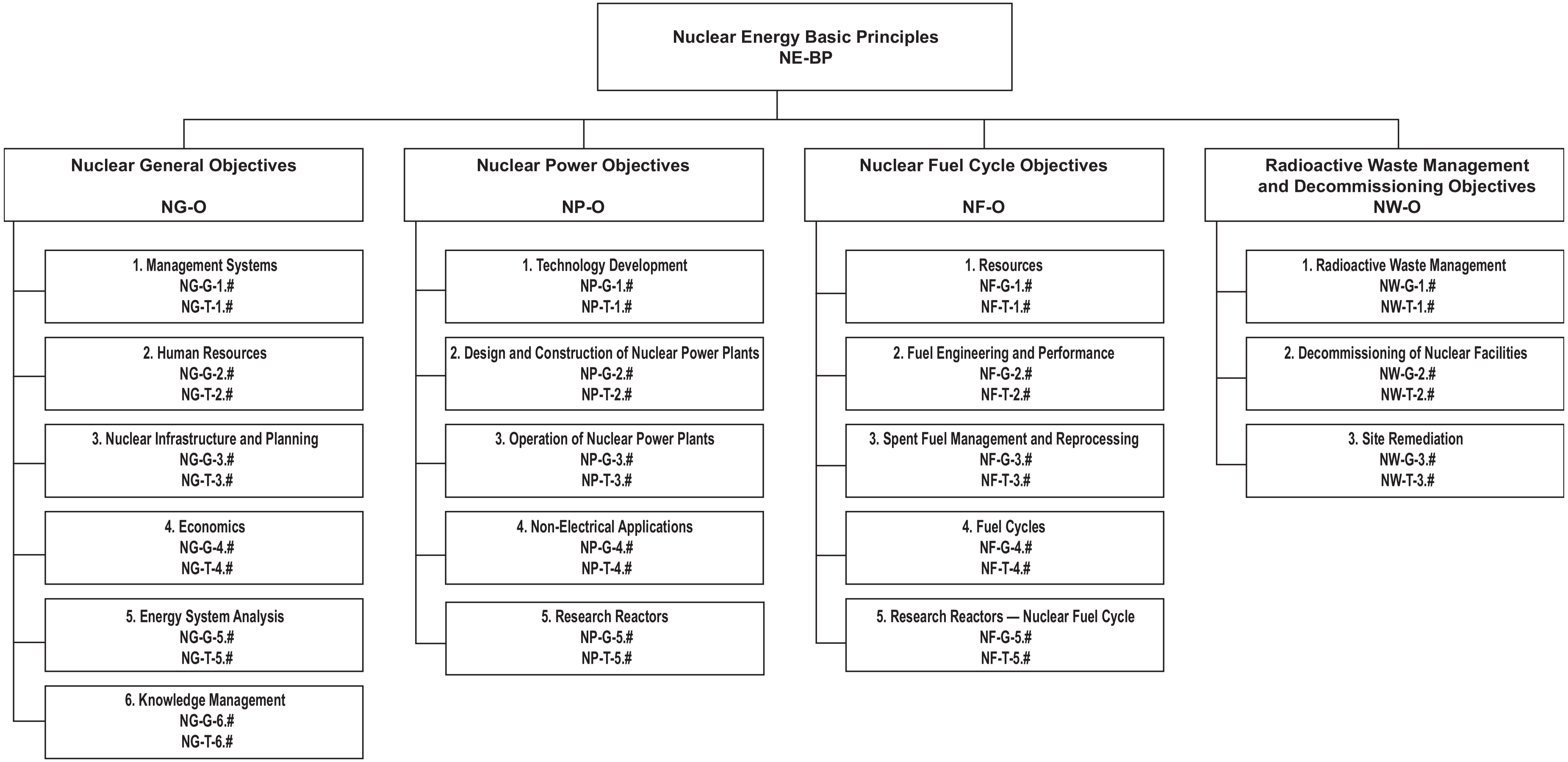

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