Institute for Energy

European Framework Document for Risk-informed In-service Inspection

ENIQ Report nr. 23

ENIQ

European Network for Inspection and Qualification

EUROPEAN COMMISSION
DIRECTORATE-GENERAL
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EUR 21581 EN

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Luxembourg: Office for Official Publications of the European Communities, 2005

EUR 21581 EN

Printed in The Netherlands, Institute for Energy – JRC IE, PR & Communication

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European Commission

EUR 21581 EN - European Framework Document for Risk-informed In-service Inspection

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Luxembourg: Office for official Publications of the European Communities

2005 – 45 pp. – 21 x 29.7 cm

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European Framework Document for Risk-informed In-service Inspection

March 2005 ***ENIQ Report nr. 23*** ***EUR 21581 EN***

Approved by the Steering Committee of ENIQ

Directorate General
Joint Research Centre

FOREWORD

The present work is the outcome of the activity of the ENIQ Task Group Risk (TGR) on Risk-Informed In-service Inspection (RI-ISI).

ENIQ, the European Network for Inspection and Qualification, was set up in 1992 recognising the importance of the issue of qualification of NDE inspection procedures used in in-service inspection programmes for nuclear power plants. Driven by European Nuclear utilities and managed by the European Commission Joint Research Centre (JRC) of Petten, the Netherlands. ENIQ was intended to be a network in which the available resources and expertise could be managed at European level. It was also recognised that harmonisation in the field of codes and standards for inspection qualification would represent important advantages for all parties involved, with the ultimate goal of increasing the safety of European nuclear power plants.

The work of ENIQ generated an important milestone with the publication in 1995 of the first issue of the European methodology for qualification of non-destructive tests [1]. A second issue was then published in 1997 [2].

ENIQ also recognised the importance of tackling the issue of setting inspection priorities on the basis of risk, at a European level. Traditionally, stringent regulations and codes specify the locations, frequency and methods of inspection, based primarily on the type and safety class of the component. However, it has been recognised that much resource has often been spent inspecting sites of negligible risk for plant safety. On the other hand, practical experience and the use of probabilistic safety assessments have demonstrated that higher risk-importance failures can occur at locations not covered by the traditional inspection programme. As the costs of qualifying and performing such effective inspections are very high, the effort should be targeted at the most risk-significant locations.

For this reason, ENIQ set up a sub-group (called Task Group 4, or TG4) in order to help homogenise the different activities on risk-informed in-service inspection for nuclear reactor safety by promoting and rationalising a common position in the EU.

In 1999, the ENIQ TG4 produced a discussion document [3], which represented a first attempt at defining a European framework on RI-ISI. Such document then stimulated a discussion that has resulted in the publication of this work.

At the end of 2001, ENIQ members emphasised the need to strengthen the risk-related activities and to promote the full integration of RI-ISI into ENIQ. In connection with the reorganisation of ENIQ's working groups, TG4 became TGR. At the inaugural meeting of TGR, it was decided that the task group should aim at establishing a common European framework on RI-ISI.

The activities of TGR have been complementary to the Task Force of RI-ISI set up by the Nuclear Regulators' Working Group (NRWG). The NRWG has recently published a document summarising the common views of the European Regulators on RI-ISI [4].

This document is intended to provide guidelines to utilities both for developing their RI-ISI approaches and for using or adapting already established approaches to the European environment taking into account utility-specific characteristics and national regulatory requirements.

The voting members of the Steering Committee of ENIQ are listed hereafter: B. Chapman (Chairman, British Energy plc, UK), D. Couplet (Tractebel, Belgium), C. Faidy (EDF-Septen, France), K. Hukkanen (Teollisuuden Voima OY, Finland), P. Krebs, (Eng. Consulting, Switzerland), B. Neundorf, (HEW, Germany), J. Neupauer (Slovenské Elektrárne, Slovakia), U. Sandberg (Forsmark Kraftgrupp AB, Sweden), J. Shejbal (Dukovany NPP, Czech Republic), D. Szabó, (Paks NPP, Hungary), R. Van Beusekom, (EPZ, The Netherlands)

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This document was edited by O.J.V. Chapman, L. Gandossi, A. Mengolini, K. Simola, T. Eyre and A. E. Walker.

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1 INTRODUCTION

In-service inspection (ISI) is an essential element of the defence in depth concept. ISI consists of non-destructive examination as well as pressure and leakage testing. ISI helps to confirm that basic nuclear safety functions are preserved and that the probability of radioactive materials breaching containment is reduced.

Defence in depth is defined by two principles: accident prevention and accident mitigation. This high level definition of defence in depth has been traditionally broken down further into five levels, the first two being [5]:

1) Prevention of abnormal operation and failures.

2) Control of abnormal operation and detection of failures.

ISI plays an important role in the above-mentioned two levels of defence in depth. It contributes to the prevention of failures by monitoring plant status. Furthermore, it contributes to the control of abnormal operation by detecting degradation in mitigating systems.

Risk-informed in-service inspection (RI-ISI) reflects recent developments in Probabilistic Safety Assessment (PSA) technology, the understanding of degradation mechanisms (e.g. structural reliability modelling, root cause evaluations) and the experience gained from nearly 10,000 reactor years operating experience of NPPs.

RI-ISI aims at rational plant safety management by taking into account the results of plant-specific risk analyses. The fundamental idea is to identify high-risk locations where the inspection efforts should be concentrated. The objective is to provide an ongoing improvement in the overall plant safety, measured by risk, together with reduced doses for the inspection teams.

The development of a RI-ISI programme requires expertise from a number of different disciplines including inspection, maintenance, design, materials, chemistry, stress analysis, systems, PSA, operations and safety. It also requires a long-term co-ordinated management commitment through inspection qualification, inspection result analysis and final feedback to the risk analysis, in order to maintain a living ISI programme. Before embarking on a risk-informed programme it is essential, therefore, to obtain the backing and commitment of the utility and plant management.

This document is structured as follows. The scope is laid down in section 2. In section 3, the most relevant definitions that apply herein are given. The principal elements of RI-ISI are described in section 4. In particular, the RI-ISI scope is discussed in sub-section 4.3. Qualitative, quantitative and semi-quantitative methods for RI-ISI are examined in sub-section 4.4. Sub-section 4.7 deals with the assessment of probability of failure. The issues concerning consequence of failure assessment are examined in sub-section 4.8. Risk characterisation and ranking are discussed in sub-section 4.9. The definition of the ISI programme is considered in sub-section 4.10. In section 5 organization and responsibilities in RI-ISI applications are discussed. In particular, a management structure is suggested. Finally, section 6 deals with issues related to the documentation and archiving of the results produced in a RI-ISI application.

2 SCOPE

The scope of this document is limited to setting out the principles that a body carrying out RI-ISI should follow. The decision on whether a risk-informed approach should or should not be applied to devising an inspection strategy is a matter for agreement between the parties involved.

This document is intended to provide guidelines about the definition of an in-service inspection programme aimed at providing defence in depth as described in the introduction.

This document has been developed specifically for RI-ISI planning in the nuclear industry, but the general principles can be adapted to other industrial fields as well. Although the main application area is envisaged to be piping systems, RI-ISI of, for example, the reactor pressure vessel (RPV) and internals is not excluded. However, in the case of the reactor pressure vessel and internals, the use of PSA in consequence analysis is somewhat different from the applications related to other passive components and piping systems.

This document identifies the key principles that any RI-ISI approach needs to meet. This is regardless of the level of quantification in the assessment of failure probabilities and consequences. However, purely qualitative methods that do not use the PSA in order to define the consequences, or any form of structural assessment to determine the probability of failure, are not considered in this document.

The link between inspection qualification that complies with the European Qualification Methodology [2] and a RI-ISI programme, is identified as an essential step in ensuring the reliability of ISI.

This document is intended to be flexible so that different countries can use it to develop RI-ISI programmes which are consistent throughout Europe but which also meet their different national legal, regulatory and technical requirements.

3 DEFINITIONS

Most of the definitions that apply to this document are given in Appendix 1, whereas Appendix 2 lists the abbreviations used throughout.

The following concepts are critical within this document and are therefore discussed here in more detail.

3.1 Failure

Failure of a structural component is an event involving leakage, rupture or a condition that would disable its ability to perform its intended safety function. For piping, failure usually involves a leak or a rupture, resulting in a reduction or loss of the pressure-retaining capability of the element in question. The expected plant response depends on the severity of the failure, which is generally related to the leak size. Hence the consequences of the piping failure may vary significantly with leak size and it is often necessary to define and analyse different degrees of failure states of a single structural element.

3.2 Probability of failure (POF)

In Probability Safety Analysis (PSA), the probability of occurrence of failure events is defined either in terms of failure frequencies or failure probabilities per demand. The failure frequency is based on the assumption of a constant failure rate. However, when dealing with passive components subject to a time dependent degradation mechanism, the probability of failure is not constant with time. For situations where the probability of failure is low throughout the life of the plant this time dependency is probably of little significance and can be ignored. However, if the degradation mechanism is of an aggressive nature, it may not be suitable to spread the probability of failure over some given mission period into a constant failure frequency. Such situations require special attention and need to be treated separately from any mainstream RI-ISI programme. In this document the term ‘probability of failure’ is used as an abridged version of the full wording ‘probability of failure over a given time into the future’. Probability of failure is also used to refer to a probability of failure on demand in the case of components in a stand-by system actuated only on demand.

3.3 Consequence of failure (COF)

The precise definition of the consequence is a matter of negotiation between the utility and the regulator. Whatever consequence is chosen, it must be measured as an outcome that is conditional on the probability of failure as defined earlier. This document is premised on the assumption that Conditional Core Damage Probability (CCDP) or Conditional Core Damage Frequency (CCDF) (depending on the impact of the failure on the plant), as defined by a Level-1 PSA, constitutes the minimum requirement for consequence evaluation.

3.4 Risk

The risk is defined herein in the engineering sense as the product of the consequences of a failure and the probability of that failure occurring, as follows:

$$\text{Risk} = \{\text{Probability of Failure}\} \cdot \{\text{Consequence of that Failure}\}$$

As probability is dimensionless, it follows that the metric of risk is the same as the metric of consequence of failure.

4 PRINCIPLES OF RISK-INFORMED IN-SERVICE INSPECTION

Risk-informed ISI aims at a rational plant safety management strategy by taking into account the results of plant-specific risk analyses.

The fundamental principles of any risk-informed inspection programme are:

1) The ability to define the consequence, probability and risk associated with structural failures so that the ISI programme can focus on an integrated defence in depth strategy.

2) The identification of an inspection programme that will decrease the risk from selected sites as far as it is practical and with due consideration of the costs and accumulated radiation dose to plant workers.

The advantage of such an approach is the optimisation of the inspection efforts. The term 'optimise' in this context is seen as a process that maintains defence in depth, whilst:

a) improving or at least maintaining the overall plant safety.
b) minimising the dose to personnel involved in the inspection activities.
c) providing improved plant reliability.

4.1 The process of risk-informed inspection planning

The following are the key elements constituting the process of risk-informed inspection planning:

1) Assurance of the long-term commitment of senior management to the risk-informed methodology.
2) Formation of the RI-ISI assessment team (sub-section 4.2).
3) Definition of the scope of the equipment/structures to be considered in the application (sub-section 4.3).
4) Collection and analysis of the information required to carry out the risk assessment (sub-section 4.5).
5) Definition of the level of the evaluation (sub-section 4.6).
6) Assessment of the probability of failure for all the components included in the scope of the application (sub-section 4.7).
7) Assessment of the consequences of failure for all the components included in the scope of the application (sub-section 4.8).
8) Ranking of the risks associated with all the components (sub-section 4.9).
9) Performing sensitivity studies to determine the impact of changes in key assumptions or data (sub-section 4.9.2).
10) Choice of the components to be inspected according to chosen criteria (sub-section 4.10).
11) Assessment of the implication on inspection qualification (sub-section 4.10.4).
12) Feed back of the obtained information (after completing the inspection) (sub-section 4.11).

Each element is further discussed in the following sub-sections.

4.2 Formation of the RI-ISI assessment team

The formation of an appropriate workforce structure is 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.

An example of a possible RI-ISI management structure, together with the roles, responsibilities and interfaces of the different parties involved, is described in section 5.

4.3 RI-ISI scope

The first practical step in developing a RI-ISI programme is to define the scope. The scope definition should clearly define the boundary of the programme, e.g. which systems and which structural elements (circumferential welds, longitudinal welds, socket welds, attachments, lugs, etc.) are to be included in the programme.

The scope of a RI-ISI programme can be either a full scope programme or, when an alternative programme is already in place, a partial scope programme. The scope should be clearly defined and documented.

A full scope programme can be defined as including all passive components and structures, including:

1) Those relied upon to perform a nuclear safety function during all design-basis plant conditions.

2) Those whose failure could compromise the function of safety-related structures, systems, or components or could cause a reactor trip or actuation of a safety-related system.

A partial scope programme is restricted to any subset of the systems or functions defining the full scope. The partial scope application can be justified, for instance, if an alternative (such as deterministic) programme is in place for the other passive components or piping not addressed by the RI-ISI programme.

A full scope RI-ISI programme is recommended because it treats all systems in a consistent and objective manner and a greater portion of the plant risk from pressure boundary failures is addressed. Nonetheless, it is recognised that in the application of RI-ISI, a partial scope programme could be justified.

A RI-ISI methodology should allow flexibility in determining the scope of application. Therefore, conducting the application 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) should still produce consistent and reliable results.

4.4 Qualitative, quantitative and semi-quantitative methods for RI-ISI

In principle, three different approaches to the use of risk concepts in developing ISI programmes can be envisioned: the purely quantitative, the semi-quantitative and the purely qualitative approach. These are defined in Appendix 1 under “Approaches to RI-ISI”.

In the nuclear industry the risks are assessed by means of PSA, which is an advanced quantitative approach (versus the more qualitative risk assessments used in other hazardous industries). The use of PSA is the foundation of risk-informed approaches in the nuclear industry, and thus a purely qualitative approach to RI-ISI that does not make use of PSA insights would be difficult to justify. Therefore, whilst it is recognised that useful risk insights can also be gained with qualitative assessments, this document does not cover the application of a purely qualitative approach to RI-ISI.

Ideally, a purely quantitative approach should be chosen whenever feasible, as only in this framework is it possible to quantify the risk change achieved when implementing the RI-ISI programme. However, it is recognised that the current state of knowledge and understanding of some of the degradation mechanisms and the availability of the required plant data can be insufficient to accurately quantify the probability of failure (see sub-section 4.7). Likewise, it is recognised that many current PSA analyses may not be detailed enough for the level required in an ISI application, and thus the consequence analysis must be complemented by some degree of qualitative assessment (see sub-section 4.8).

Thus, the most feasible approach to the development of any RI-ISI programme today would be the semi-quantitative one, based on a plant-specific PSA and with due cognisance of the current mechanistic understanding of any degradation mechanisms.

4.5 Information collection and analysis for a RI-ISI assessment

The process of risk-informed inspection planning 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) Historical plant operating data.
3) General nuclear industry information.
4) Safety Analysis Report and technical specifications.
5) PSA data.

The information required will depend on the approach adopted, but may include:

<u>Equipment data</u>

a) Design and manufacturing records.
b) As built information.
c) Design basis conditions.
d) Deterministic design stress and fatigue analysis.
e) Defined boundaries of plant items to be considered for inspection planning.

Historical plant data

a) Operational transient and condition monitoring data.
b) Plant failures and service experience data.
c) Pre-service and in-service inspection records.
d) Environmental conditions including temperatures, water chemistry and flow rates.
e) In-service degradation assessments (fatigue, SCC, erosion-corrosion, external effects).

General nuclear industry information

a) National and international databases.
b) National and international publications.

Safety analysis report and technical specifications

a) Test procedures and frequencies.
b) Emergency operating procedures.
c) System failure procedures.

PSA data

a) Failure frequencies.
b) Failure modes and effects analysis (FMEA).
c) Frequencies of initiating events (internal and external).
d) Data on human reliability and common cause failures (CCF).
e) System design.
f) Success criteria.

The availability and accessibility of this information will vary depending on the particular circumstances.

Data collection is an essential part of the RI-ISI process, as it constitutes the basis for the whole analysis and decision process. Due care and diligence should therefore be practised during this phase. 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, for example, periodic safety review. The RI-ISI team should therefore ensure that this data is also made available for other management planning activities.

4.6 Level of evaluation

The risk assessment in a RI-ISI programme can be carried out either at the element level (e.g. each weld and discontinuity) or for groups of contiguous elements where all relevant conditions are the same.

The advantage of evaluation at element level is that the output will be unique for every element and there will be a clear distinction between elements. However, to reach this distinction between the elements, a rather large amount of detailed input data is needed which might be prohibitive from a cost point of view.

The main advantage of grouping elements with the same conditions (a process termed segmentation) is that it is less time consuming. The initial segmentation is normally based on the postulated direct consequences of a failure, but it is important that indirect consequences are also considered. Piping segments are normally defined at major

components such as pumps, heat exchangers, check-valves, remote controlled valves or at pipe size changes. The possibility for operator action to mitigate the consequence can also be a reason for a segment boundary.

4.7 Probability of failure assessment

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 world-wide generic data. Such an analysis phase is very important in order to correctly classify or quantify the failure potential.

Ideally, the probability of failure of components or sites that are potentially in need of inspection should be calculated in a quantitative way, implying the use of structural reliability models (SRMs). However, two important facts are recognised concerning the use of SRMs. Firstly, such models do not exist for all the potential degradation mechanisms that currently affect nuclear power plants. Secondly, for degradation mechanisms that do have a viable SRM, there is only a limited acceptance that these estimates can be seen as representing some form of true or absolute value. This implies that the evaluation of the probability of failure for all potential ISI sites will necessarily yield a mixture of quantitative and qualitative assessments. Quantitative values, where they exist, may serve to quantify relative differences in the probability of failure from one site to another.

It is thus envisaged that the most likely way failure probabilities can be presently estimated for RI-ISI applications, is on the basis of a combination of quantitative and qualitative assessments. Such an approach is referred to as a 'semi-quantitative' analysis. This form of analysis would use all the potential knowledge available to derive an auditable ranking of the probability of failure.

A semi-quantitative analysis of the probability of failure can be obtained by:

1) Use of structural reliability models, where they exist, to provide a good estimate of the relative differences in the failure probabilities (see sub-section 4.7.1).
2) Statistical estimates based on both plant-specific and global databases in order to provide anchoring points for both the SRM analysis and the expert judgments (see sub-section 4.7.2).
3) Use of formal expert judgements using a combination of deterministic structural models and design insight (see sub-section 4.7.3).

It should be recognised that there is not a single, optimal method for assigning probability of failure. As such, each above mentioned approaches, or combination of them, needs to address the issues identified herein.

The issue of leak detection arguments within the RI-ISI framework is discussed in sub-section 4.9.4.1.

4.7.1 Structural reliability models

Whilst it is recognised that there are several degradation mechanisms not covered with the available analytical tools, structural reliability models (SRM) are essential tools in the evaluation of probabilities of failure for components of nuclear power plants.

It is essential to verify and validate any SRMs used in the evaluation of probabilities of failure. To this end, a number of steps can be defined [6]:

1) The basic programming should be shown to have suitable quality assurance documentation.
2) The scope, the analytical assumptions and the limitations of the modelling capability should be well defined.
3) The analytical assumptions should be well grounded and based on theory that is accepted as representative of the situations considered by the given SRM.
4) The model should be capable of reproducing the data on which its analytical assumptions are based. Examples should be provided that can demonstrate its general agreement with the available experimental data.
5) Attempts should be made to show how the model compares with the world or field data, accepting the inherent limitation of this data.
6) The model should be benchmarked against other SRM models within the same field or scope and any differences should be adequately explained.

Further key elements within a SRM are recognised to be:

a) The choice of statistical distributions. The reasons behind a specific choice of distribution should be made clear and, if possible, a mechanistic understanding of the choice should be made. If the choice is made based on a best-fit evaluation of the data, then a comparison with other distributions should be made. In many cases, especially when very small failure probabilities are evaluated, the tails of the distributions become important.

b) Sensitivity analyses should always be performed to assess the influence of different parameters and choice of statistical distributions. Note that it is recommended to use best estimate values of each set of input data, based on the best available knowledge at the given time. If in-built pessimistic design data is used, conservative estimates of the risk will be obtained, which may distort the relative risk ranking, especially when different types of components and failure modes are compared.

An advantage of many SRMs is the possibility to quantify the influence of inspections both in terms of inspection capability and frequency. This is a key factor in RI-ISI since it is desirable to select the most appropriate inspection capability for every risk site.

Further discussion on requirements and recommendations that should be set for SRM and associated software for RI-ISI applications are found in the reports produced within the NURBIM project [6].

4.7.2 Estimation from operating experience data

Operating experience data provides useful qualitative and quantitative information on the degradation of structural components. For example, Stress Corrosion Cracking (SCC) was discovered from field failures. Operating experience data covers not only leak and rupture data, but also other information on the presence of non-critical levels of degradation, such as small defects and wall thinning. The degradation information can be of considerable value in the development of SRMs and more generally in the assessment of structural failure probabilities.

In principle, operating experience data can, and indeed should be, considered in the evaluation of failure potential. The data should be broken down according to, for example, specific degradation mechanisms, pipe size classes, and major material and environmental characteristics. The data should be broken down as finely as possible without becoming too sparse. However, when parameters are estimated from structural component failure or degradation databases, the following shortcomings have to be taken into consideration:

1) Passive components usually have an increasing failure rate (ageing), and thus the exponential distribution does not correctly model the failure occurrence.
2) The data quality may be insufficient for obtaining reliable estimates due to:
    a. missing information related to the component population.
    b. uncertainties related to failure mechanisms and root causes.
3) Data is often very scarce.

Due to the shortcomings related to the quality and quantity of data, the estimates of passive component failure probabilities are subject to large uncertainties. For RI-ISI applications, probability of failure estimates obtainable from world-wide or generic data may not be sufficient. However, the data is extremely valuable in establishing prior probabilities. These values can then act as an anchor for either the SRM estimates or expert judgement, using plant-specific information, to identify the distribution of the probability of failure throughout the plant-specific sites.

4.7.3 Use of expert judgement through expert elicitation

The shortcomings in both SRM and operating experience data will sometimes limit a quantitative assessment of some of the active degradation mechanisms of interest. A possible alternative is to use expert judgement, preferably through the use of formal expert elicitation, to derive failure probabilities.

Well-structured expert elicitations can be a powerful tool for expanding the range of application of a RI-ISI. Such elicitations support and integrate individual expert judgements to provide an auditable set of probability of failure estimates. However, it is important to ensure that the use of this expert judgement is conducted within a structured expert elicitation process. The principal phases of a structured expert elicitation are:

1) Selection and training of experts.
2) Elicitation of judgements.
3) Modelling and combination of judgements.
4) Sensitivity analyses.
5) Discussion and feedback from experts.
6) Documentation.

It is recognised that experts are often not very familiar with probabilities, especially with subjective probability statements, and thus the training phase to give probabilistic estimates is important. The person leading the structured expert elicitation process should have proper knowledge in decision analysis, probabilities and statistics. This person is called the normative expert. His, or her, responsibility is to facilitate the process by giving training, conducting the elicitation and aggregating the expert opinions. A detailed discussion on the expert assessment approach within the nuclear industry can be found, for example in [7].

Using an expert judgement for all sites, including those for which an SRM and/or possible statistical data exists, can combine qualitative and quantitative probability of failure estimates. The SRM and/or statistical data then act as both an anchor for the rankings and as a form of cross correlation with the expert ranking. An example of such approach is described in [6].

4.8 Consequence of failure

The failure of a passive component in a nuclear power plant (NPP) can basically lead to one of the following classes of events of interest:

1) Initiating event: A pressure boundary failure occurs in an operating system resulting in an initiating event.
2) Loss of mitigating ability (stand-by): A pressure boundary failure occurs in a standby system and does not result in an initiating event, but degrades the mitigating capabilities of a system or train. After the failure is discovered (if discovered), the plant enters the Allowed Outage Time defined in the Technical Specification.
3) Loss of mitigating ability (demand): A pressure boundary failure occurs in a standby system when the system/train operation is required by an independent demand.
4) Combination: A pressure boundary failure causes an initiating event with an additional loss of mitigating ability (in addition to the expected mitigating degradation due to the initiator).

Furthermore, a pressure boundary failure that also affects the containment performance can be identified as a separate class.

The consequence analysis part of the RI-ISI process aims at evaluating the impacts of any of the above-mentioned events on plant risk. The consequence evaluation consists of the following primary steps:

a) A qualitative failure modes and effects analysis (FMEA) that determines the plant impacts of postulated failures of postulated sizes (e.g. small, medium, complete rupture). This step can consume the largest share of resources.

b) Qualification of the PSA for RI-ISI application.

c) Quantitative analysis with PSA.

PSA techniques have been developed since the 1970s [8] and PSA studies exist today for most types of nuclear power plants and individual power stations. Although the level of details and quality of these studies vary, the PSA methodology has become a well-known and frequently used tool in the analysis of nuclear safety.

The following items are considered critical if a robust interface between PSA and RI-ISI is to be developed:

- The levels and scope of PSA to be used in RI-ISI (sub-section 4.8.1).
- PSA quality, limitations and uncertainties (sub-section 4.8.2).
- Passive component failure treatment (sub-section 4.8.3).

4.8.1 Levels and scope of PSA to be used in RI-ISI

PSAs are performed at different levels, dealing with different types of consequences:

Level 1: Assessment of plant failures leading to core damage (CD) and the estimation of core damage frequency (CDF).

Level 2: Estimation of off-site fission product release. Consequences are usually expressed in terms of large early release frequency (LERF).

Level 3: Assessment of off-site consequences leading to estimates of the effects of fission product release on human health. Consequences are usually expressed in terms of human fatalities, public radiation doses and environmental pollution.

All modern NPPs have plant-specific PSA studies, usually at Levels 1 and 2. For this reason, it appears logical that they should form the basis for the consequence evaluation. Recent RI-ISI applications have mainly relied on the Level 1 consequence analysis. It is recognised that the use of Level 2 consequence (e.g. large early release) could be important for RI-ISI application, especially for reactors whose complete primary pressure boundary is not fully covered by the containment structure (for instance, RBMK and BWR reactors). In this case it should be borne in mind that Level 2 studies are based on assumptions and hypotheses that can be very difficult to verify in practice and thus are in general subject to higher uncertainties than estimates of core damage.

In view of the above, it is concluded that the Level 1 PSA forms the recommended (as well as the minimum) basis for analyses, but insights from Level 2 can be considered in handling priorities for elements with lower probability of failure but higher consequences.

The scope of a comprehensive PSA Level 1 study includes evaluation of the risk at power operation, start-up, shut down and cold shutdown. Among the initiating events that are usually considered are typical transients, loss-of-coolant accidents (LOCAs), fire, flooding, seismic and other external events.

The basic demand on the scope of the PSA is that all operating plant modes and initiating events must be addressed in the evaluation. It is however, not necessary that all modes and events are included in the calculations. A qualitative treatment of missing modes and events is sufficient when they have little influence on the result. This will differ from plant to plant.

4.8.2 PSA quality and limitations

It is important to develop results in the RI-ISI programme that are robust. Therefore the PSA study should be qualified for this purpose.

An overriding requirement is that the PSA should realistically reflect the actual design, construction, operational practices and operational experiences of the plant. The PSA should reflect the plant's different functions with the same accuracy and level of detail. The

evaluation of system demands should be done with the same level of realism and conservatism for all functions, and the input data used for PSA analyses should be verified to ensure that it reflects the state of the art.

It is recommended that the PSA study be qualified/certified for use in RI-ISI application by fulfilling demands specified by ASME standard or IAEA standards/requirements (to be published). The qualification/certification could also be performed by peer review of the PSA for RI-ISI application. The qualification of the PSA should be documented [9-14].

Due to the small probabilities of failure of passive components in comparison with active components, the former usually only contribute to a small proportion of the total plant risk evaluated in the PSA study. Moreover, because of low probabilities of failure, the data available regarding passive failures is usually limited. This has naturally led to very limited treatment of such failures within PSA studies. Due consideration must be given to this fact and how the passive components should be treated in the consequence analysis. The next sub-section (4.8.3) discusses this issue in more detail.

If it is considered that the PSA does not fully meet the quality requirements for RI-ISI application, specific attention should be paid to its use in the consequence evaluation. The PSA may still provide useful information for the analysis, but in this situation it should be supported by complementary analyses that may be of qualitative nature.

Each PSA study is also subject to unavoidable uncertainties, such as parameter uncertainties, model uncertainties and completeness uncertainties. Parameter uncertainties related to statistical data are usually quantified and a PSA study provides uncertainty boundaries as a part of its output. Model and completeness uncertainties are normally not included in the PSA study, but they have to be evaluated in the process of PSA qualification.

Quantified PSA uncertainties, such as parameter uncertainties, could be transferred to the RI-ISI procedure to increase its robustness significantly, but it is recognised that there are no straightforward algorithms to do so. The RI-ISI evaluation should specify the effects of the parameter uncertainties in connection to sensitivity analyses (sub-section 4.9.2).

4.8.3 Passive component failure treatment

Pipe breaks are modelled directly as initiating events such as Loss of Coolant Accidents (LOCAs), main steam or feed-water line breaks or flooding. PSA studies may spatially subdivide the pipework into zones and then consider various types of LOCAs occurring within the zone as initiating events but the studies do not normally model initiating events at segment or element level. Furthermore, the components are normally modelled in the PSA with two states: full function or total failure. In the case of piping, these states are ‘no leakage’ and ‘complete pipe break’. Thus, a ‘partly failed’ component (such as a crack or a small leak) that is handled by a manual action is not modelled in the basic PSA. All events leading to an automatic initiation of safety systems are included in the PSA.

As the modelling of structural components in the base PSA may be coarse and deficient for many systems, additional analysis is required to determine the consequences at the degree of detail needed in RI-ISI. A complementary FMEA should be conducted in order to define both the direct and indirect impact of failure on plant operation. Indirect effects include failure consequences affecting other systems, components or piping segments, such as:

1) Pipe whip.
2) Jet impingement.
3) Decompression waves.
4) Flooding.
5) High environmental temperatures, etc.

It is recognised that indirect effects of passive component boundary failures may have a significant influence on the evaluation of core damage frequency and it is therefore recommended that such effects be explicitly taken into account. Spatial consequences are determined based on the location of the failure and relative position of important equipment, and it is recommended that the analyses be confirmed by a walk-down.

The FMEA should include the evaluation of consequences of a spectrum of leak sizes with considerations of leak detection (see sub-section 4.9.4.1), and the analysis of the possibility to isolate the leak or break. Both automatic and manual isolation should be considered.

The extent to which the findings of the FMEA can be incorporated in the PSA model for the quantitative consequence evaluation depends on the PSA and plant-specific issues. Issues not explicitly included in the PSA model should be judged qualitatively and be taken into account in the final review and adjustment of the consequence ranking.

The base PSA does not usually include models of passive component failures that could result in the loss of mitigating ability (e.g. safety system pipe break). A way to address such situations is to identify basic events or groups of events already modelled in the PSA, whose failure captures the effects of a piping segment failure. This method is sometimes called the 'surrogate component approach'.

The uses of the plant-specific PSA in the RI-ISI analysis can be summarised as follows:

a) The PSA model and success criteria are used to define safety functions and backup trains.
b) PSA results for all initiators are applied directly for relevant consequence impacts.
c) PSA system and/or train unavailability are used to determine the reliability of mitigative equipment given a pressure boundary failure.
d) Internal flood results are used in the analysis of spatial effects.
e) External events studies (e.g. fire analysis) results are used in the evaluation of external events.
f) Shutdown PSA, if available, is used in the evaluation of other modes of operation.
g) Level 2 PSA results are used to identify event sequences that provide the dominant contribution to LERF with respect to pipework failures.

4.9 Risk characterisation and ranking

This sub-section provides guidance on how the risk characterisation and ranking may be developed to support establishing the ISI programme. As already stated, risk is defined in engineering terms 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. Combining the information from the probability of failure assessments and the consequence analyses forms the risk ranking. The risk ranking can be carried out at either element level or segment level.

4.9.1 Graphical representation of risk

Each segment or element can be ranked from highest to lowest according to its risk. A useful way to evaluate the risk of failure and clearly represent it in a graphical way is to develop risk plots, risk matrices or Pareto risk diagrams.

In risk plots, each component is represented as a point on a log-log plot. The consequence of failure is represented on the x-axis (the abscissa of the point). The probability of failure is represented on the y-axis (the ordinate of the point). Refer to Figure 1 for an example of a risk plot.

A risk plot provides a clear picture of how the risk is distributed over the range of consequences. Given the nature of the risk plot, log-log axes, sites of constant risk are identified by straight lines. This fact greatly aids risk visualisation and ranking for the given parameters and assumptions. Parallel lines of constant risk can be drawn at fixed distances apart, identifying risk bands (for example, decades).

In a semi-quantitative approach to risk, probability of failure and consequence of failure are not numerically evaluated in absolute terms, but are ranked using either a qualitative scale such as high, medium, low, or broad categories such as 10^{-3} to 10^{-4} etc. In this case, a risk matrix can be used to represent the rankings in the form of subsets as shown in Figure 2.

Another way to graphically represent the risk is by means of the Pareto risk diagram. A Pareto risk diagram shows the risk from individual components ranked in a descending order in the form of histograms. See examples of Pareto diagrams (linear and logarithmic) in Figure 3. The Pareto plot is closely related to the Risk Reduction Worth (RRW), commonly used in PSA technology. This relationship between the Pareto diagram and the RRW is discussed in Reference [6]

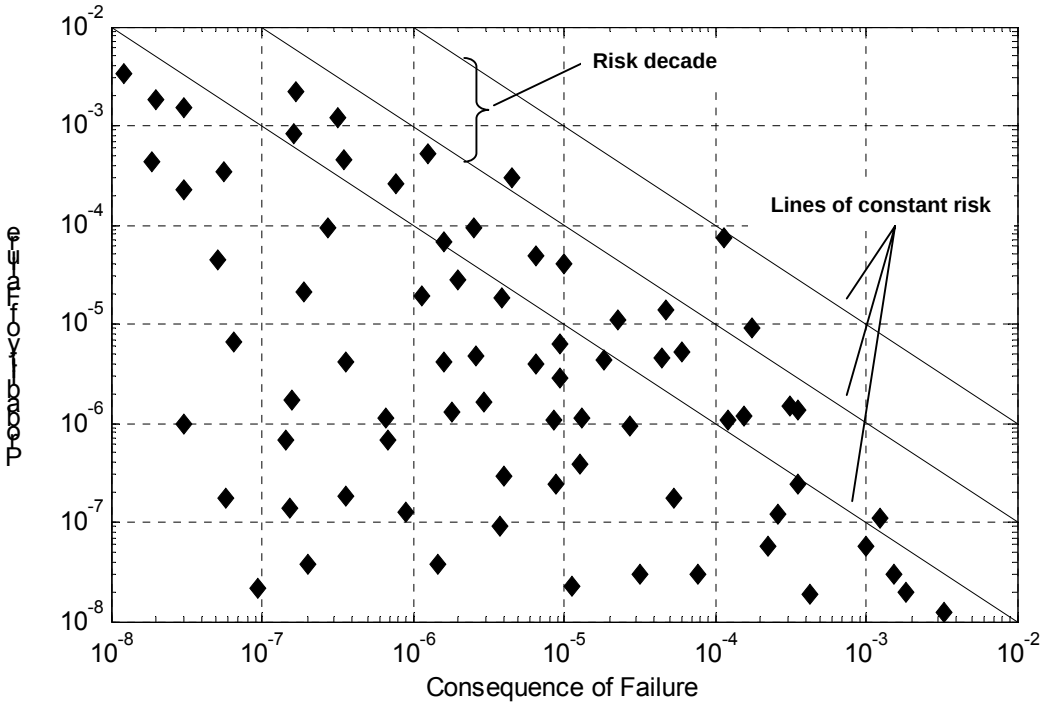

Figure 1 Risk plot (the plot is purely illustrative)

Whilst the Pareto plot gives a ready impression of the risk distribution and the amount of risk being addressed by a given inspection, the risk plot and risk matrix are probably better graphical representations with regards to inspection planning. These are more informative since they also show if the risk is governed by the probability of failure or by its consequence. 'High consequence–low probability' sites require different considerations than 'high probability–low consequence' sites even if they have an equal total risk. This issue is further discussed in sub-sections 4.9.4.2 and 4.9.4.3.

			Conditional Consequence				
			Very Low	Low	Medium	High	Very High
			$<10^{-6}$	10^{-6}-10^{-5}	10^{-5}-10^{-4}	10^{-4}-10^{-3}	$>10^{-3}$
Probability of failure	Very High	$>10^{-4}$					
	High	10^{-5}-10^{-4}					
	Medium	10^{-6}-10^{-5}					
	Low	10^{-7}-10^{-6}					
	Very Low	$<10^{-7}$					

Figure 2 **Risk Matrix (the values used in this table are purely illustrative and should in no way be taken as a requirement).**

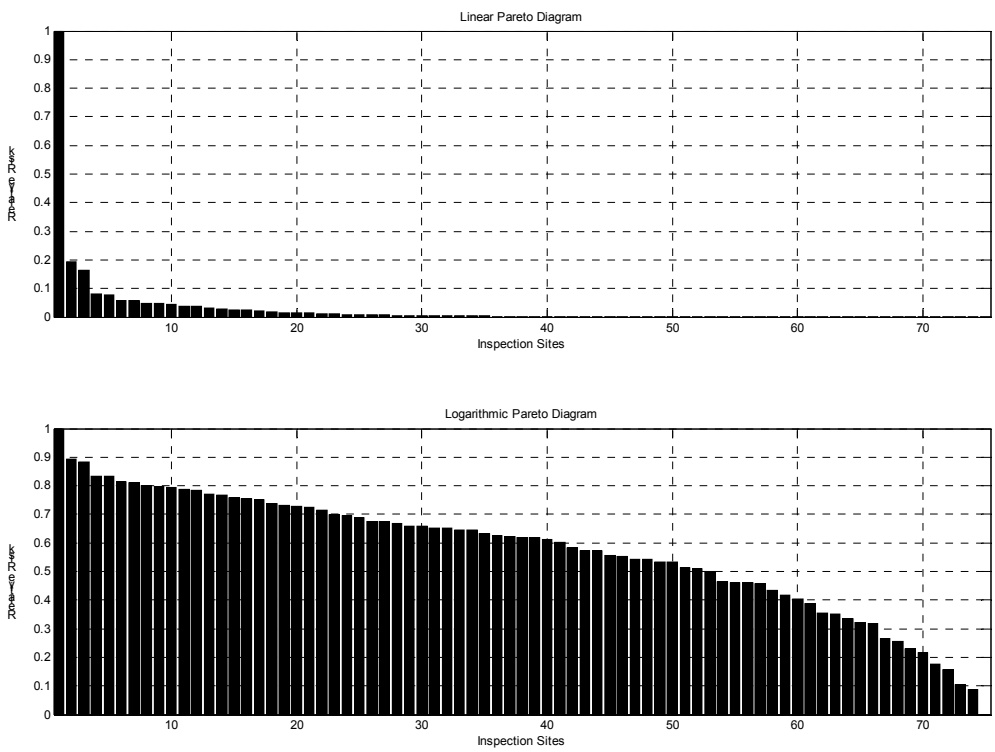

Figure 3 **Pareto diagram (the diagram is purely illustrative)**

4.9.2 Sensitivity analysis

Sensitivity studies should be performed to determine if changes in key assumptions or data could have any significant impact on the rankings. These sensitivity studies should address the potential changes in component ranking by varying the estimates of pressure boundary failures and estimates of the consequence of failure. Also, crediting the effect of leak detection on the results should be investigated. These results should then be integrated in the decision-making process.

The sensitivity studies can identify potential risk outliers by identifying ISI components that could dominate risk for various operational modes, PSA assumptions, and data and model uncertainties.

4.9.3 Safety-significant sites

The development of a risk plot or ranking does not in itself identify sites that could be said to be safety-significant. Such a choice is subjective. What follows is only advisory but provides a possible structure for discussion between the utility and the regulator.

In addressing this question it must be borne in mind that in-service inspection leads to radiation exposure to the inspection personnel. Each combination of ISI programme is associated with a certain radiation exposure. In principle, it is possible to develop ISI programmes with the same risk reduction but with different total radiation exposure. When faced with such choice, the RI-ISI programme that give as little radiation exposure as possible should be chosen, according to the As Low As Reasonably Practical (ALARP) principle.

The first step in the process of determining risk-significant sites is the identification of risk outliers. Risk outliers are sites that have a much higher risk than the overall mean risk level for all sites. Outliers should be treated as recommended in sub-section 4.9.3.1.

The second step consists of defining a risk value, relative to the highest risk (excluding any outliers) that can be considered as the level separating potentially safety-significant sites from those that can be considered as non safety-significant. Sites falling above this level are considered as potentially safety-significant. No specific relative risk level is given here since different factors may need to be considered for different utilities and different regulatory bodies. Among such factors are for example the risk distribution of the plant, the definition of risk outliers, the nature of risk associated with each site, the ambition the utility has with its RI-ISI programme and national regulatory requirements.

Having identified the potentially safety-significant sites for a RI-SI programme, it is recommended that an expert panel is used to review the proposed sites. This panel should review the information, analysis and insights that have been used to identify the safety-significant sites. It is also important to investigate alternative possibilities for mitigation against the risk. It is, therefore, necessary to look at the nature of the risk associated with each site. It may be possible to identify ways other than inspection to address the risk.

If the RI-ISI analysis is done at segment level, the evaluation of safety significance is also done first at segment level. In this case, the probability of failure and consequence are evaluated for a representative element within the segment. The risk associated with such elements also approximates the risk of the other elements within the segment. Naturally, potentially safety-significant sites are identified among the most safety-significant segments. The selection of sites should then be reviewed by the expert panel as described above.

4.9.3.1 Risk outliers

A risk outlier is a site with an associated risk much higher than the overall risk level for most of the sites. Risk outliers, the few sites dominating the overall risk, should be treated carefully to avoid giving a skewed view of the risk distribution.

The occurrence of risk outliers should be seen as a situation requiring special treatment. Inspection alone may not be considered sufficient to address the risk from such sites and consideration should be given to other complementary methods such as continuous monitoring, improved leak rate detection, load reduction or component replacement.

4.9.3.2 Flat risk distribution

There are situations where the principle of identifying a specific risk band to determine which sites are safety-significant is not applicable. One such situation occurs when the risk distribution across the plant is very flat. In this case the Pareto diagram, see sub-section 4.9.1, would show the flat distribution across all the proposed inspection sites whilst the risk plot would show that a great number of the sites having the highest risk fall in the same narrow risk band. Risk may then no longer be a very meaningful way of differentiating between sites for inspection. One logical option would be to weight the inspection in terms of the consequence, with possibly some random inspections.

4.9.4 Specific issues

4.9.4.1 Leak detection

For several degradation mechanisms known to act in nuclear power plants, extensive experience accumulated over several years of operation has shown that failure does not occur catastrophically (double edge guillotine break) but rather in the form of a leak. This implies that any crack that grows to a through-wall configuration is likely to be detected by plant monitoring systems before reaching a size that would result in a rupture.

The leak detection functions are part of the defence in depth strategy, level 3 - Control of accident within the design basis, as defined in INSAG 10 [5]. Such functions must be designed to handle situations when other steps in the defence in depth chain have failed.

Leak detection functions in some plants are part of the reactor protection system as they automatically initiate the plant safety function when the leak rate exceeds a specified value. Furthermore, the technical specifications and operating procedures give the operators guidance on how to act when the leakages increase above certain rates.

In the case of degradation mechanisms with an anticipated leak before break, the probability of large leak or rupture is affected by the detection probability of the (small) leak before it develops to a larger leak. Leak detection thus is an issue related both to the probability and the consequence of failure.

RI-ISI analyses should, in principle, include the effects of leak-detection both in the case of actuation of automatic safety functions and of manual actions. The reliability of leak detection and related actions could be quantified in the consequence evaluation, SRM

models or a combination of both. The RI-ISI documentation should identify in which part of the study these issues are modelled.

Within the RI-ISI framework, some studies [6] have shown that neglecting leak rate detection in the assessment of probability of failure may cause an unjustified focus on low-risk components for inspection selection. Since the rupture and large leak probabilities will approach the small leak probabilities, it will be more difficult to identify the highest risk locations. On the other hand, experience has shown that crediting leak detection may result in RI-ISI programmes with almost no inspection required for piping located inside containment. Whilst this could be perfectly justified from a risk assessment point of view, it is envisaged that such result could present a problem of acceptability.

When crediting leak detection, care should be taken to ensure that conservative or non-conservative assumptions are not imposed on the analysis. Leak rate detection should only be credited when evaluating the probability of failure if it is commensurate with the leak detection capabilities, operating procedures and licensing basis of the plant. In this context it is required that adequate models of crack opening areas, leak flow rates and leak flow rate detection are available. The assumed reliability of leak detection capability, its basis and its impact on the results should be documented.

In areas without specific leak detection systems, leaks may be detected (for example) by means of other alarms or by walk-around. To be consistent in the analysis of failure probabilities, the probability of leak detection should be estimated also for elements or segments at such locations. It is however recognised that a reliable and consistent modelling of leak detection is a difficult task that carries large uncertainties.

When investigating the probability of failure, it is recommended that both the effects of crediting and not crediting leak detection be investigated.

4.9.4.2 High consequence – low probability of failure sites

It is recognised that sites characterised by high consequence and low probability of failure can be of concern even if the risk at such sites is not high in absolute or relative terms. Such sites would fall in the lower right hand corner of a risk plot (see Figure 4). Note that these sites are not identified directly by the Pareto plot or the RRW measure.

These sites are considered not to be safety-significant due to the very small associated probability of failure. However, it is recognised that a problem of confidence can arise if the component's probability of failure is well below the area for which there is any practical experience. Any inspection in this area is intended to provide additional confidence in the assessed probability of failure. In this way the inspection can be considered to cover an unknown or unexpected degradation mechanism that could challenge the integrity of the component and thus provides an element of conceptual defence in depth.

Very low probability of failure implies that there is no identifiable damage mechanism, and a failure over the lifetime of the component is considered incredible. The setting of inspection qualification goals is therefore not a straightforward matter and it is likely that the approach may vary, depending, upon circumstances and regulatory requirements.

4.9.4.3 High probability of failure – low consequence sites

Despite their appreciable probability of failure, components falling in the upper left corner of the risk plot in Figure 4 are not risk-significant due to their low impact in terms of nuclear safety.

For such items it may be possible that a failure over the lifetime of the plant is reasonably foreseeable. In these cases, inspection could be justified in order to maintain plant availability, improve workers' safety and provide an indication of a good nuclear safety culture.

Specification of a defect type and size for ISI is likely to be relatively straightforward as such sites have a clearly identifiable damage mechanism. Since this region of the risk plot is not safety-significant the utility could consider carrying out a financial risk assessment to identify those risks that would return the greatest financial benefit in terms of improved plant availability. This could then form the basis for a financially optimised inspection programme. Furthermore, since this area is not risk-significant, the level and form of any inspection qualification should be a matter for the utility to decide. However, there is no intent here to undermine the value of these inspections. Any failures in nuclear power plant influence the safety culture and the perception of risk by the public.

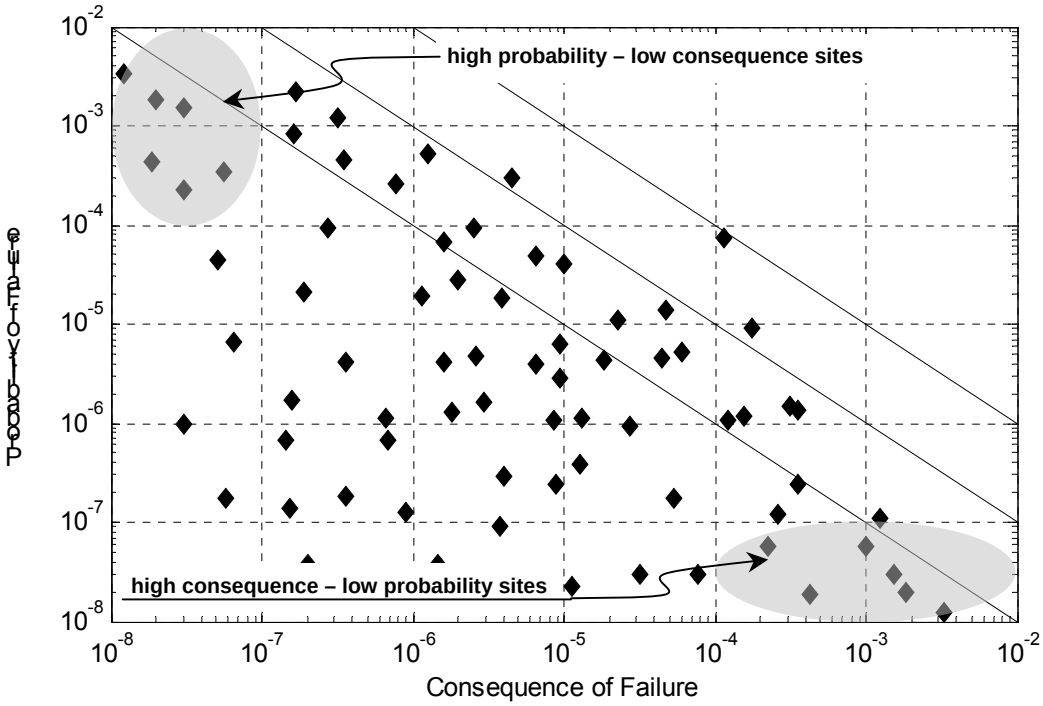

Figure 4 **Risk plot showing the high probability – low consequence region and the high consequence – low probability region (the plot is purely illustrative; the way the two regions are represented should in no way be taken as a requirement).**

4.10 Definition of the ISI programme

The overall principle underlying the definition of the RI-ISI programme, e.g. the identification and selection of individual sites for inspection, is that the proposed inspection programme should provide defence in depth as discussed in section 1.

It was recognised in sub-section 4.4 that even when a purely quantitative analysis has been performed, it is very difficult to demonstrate that the assessed levels of risk are true in absolute terms. It is thus also very difficult to compare the total risk assessed for one plant with that calculated for another and therefore this document does not give recommendations based on absolute risk levels. For this reason, the proposed approach is based only on the relative risk ranking.

4.10.1 Minimum valid RI-ISI programme

In sub-section 4.9.3 it was suggested that the first step to define a valid RI-ISI programme consists of identifying the possible existence of risk outliers (and treat this separately). The second step is to define which sites should be identified as potentially safety-significant. These should then be seen as the primary candidates for inclusion in a risk-informed inspection programme.

After having identified the potentially safety-significant sites, the next step is to select the sub-set to be included in the inspection programme. In selecting these sites, other criteria than risk can also be considered. Such criteria are, for example, the severity of the degradation mechanisms, radiation dose, accessibility of the site and the inspection costs.

Sites that have a low probability of failure but high consequence should be considered for inclusion, as discussed in sub-section 4.9.4.2. Also, due consideration should be given to the low consequence but high probability of failure sites, as discussed in sub-section 4.9.4.3.

After having established an initial selection of candidate sites based on these criteria, the risk associated with such elements should be compared with the risk associated with the sites forming the scope of the current (deterministic) ISI programme. The total risk being addressed by the selected sites should address at least as much but preferably more than the total risk addressed by the current ISI programme.

When moving to a RI-ISI programme, at least risk neutrality or, better, risk reduction should be achieved. The risk reduction achieved depends not only on the risk addressed, but also on the capability and frequency (intervals) of the inspections. In the following sub-section, the issue of risk reduction is discussed in more detail.

The scope of the RI-ISI programme may also need to be completed with sites requiring inspection in order to meet other legal requirements, for instance in relation to the safety protection of workers.

4.10.2 Risk reduction achieved through ISI

An accurate estimate of the risk reduction can be calculated only if a quantitative measure of the inspection capability (e.g. a Probability of Detection, or POD curve) is known. When such a measure of the inspection capability is available, a structural reliability model could be used to quantify the risk reduction, see sub-section 4.7.1. However, it is recognised that in the majority of cases, this measure is not available and would require a disproportionate use of resources to obtain it.

To gain confidence that the proposed new ISI programme is at least as effective as the current ISI programme in reducing risk, it is recommended that the two programmes be compared simply by using as input the risk estimates of each site, the inspection intervals, and a hypothetical probability of detection.

The sensitivity of the results to the assumed inspection capability should be studied by varying this parameter. In this comparison, the time interval considered should correspond to the completion of the ISI scope (e.g. 10 years) or may even be the expected remaining lifetime of the plant. These analyses could also be used the other way around to identify the level of inspection capability required for achieving a certain risk reduction.

4.10.3 Inspection intervals

Currently, most in-service inspection intervals are either set by a code, such as ASME section XI [15], by some consensus between the regulator and the utility or by some form of deterministic principle, usually based on design considerations and general technical experience.

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

Keeping in mind the fundamental principle of achieving risk neutrality or possibly risk reduction, it is beyond the scope of this document to provide more explicit requirements regarding inspection intervals.

In principle, optimal inspection intervals could be determined by means of, for example, validated SRMs coupled with the knowledge of POD curves.

4.10.4 Inspection qualification

Within Europe, ENIQ has an established methodology for inspection qualification [2] and it is recommended that this approach be adopted.

An essential input to the ENIQ qualification process is the identification of appropriate inspection qualification requirements defining the defects of concern for which highly reliable detection and sizing is to be demonstrated. The ENIQ qualification process, if followed to a successful outcome, provides high confidence that the inspection system can achieve the inspection qualification requirements for defect sizing and detection. An important point to note is that it does not provide a quantitative measure of inspection efficiency. Currently the output statement from the ENIQ qualification process is expressed in linguistic terms such as ‘highly reliable’ or ‘highly efficient’.

If a validated SRM is available, the ENIQ qualification defect size could be determined so that the proposed inspection aims at reducing the probability of failure of the inspected site by, for example, one decade. Implicit in this would be the need to establish a simplified but still realistic quantitative estimate of the POD for defects greater than the qualification size. Consideration could also be given to the possibility of setting a lower qualification size in order to reduce the specific site risk even further if this is practical, without incurring unreasonable costs.

4.11 Concept of living RI-ISI

The risk assessment provides a ‘snap-shot’ of the risk distribution within the ISI boundary at a given point in time. It is recommended that the risk assessment be kept ‘live’ so that it reflects as accurately as possible, the distribution of risk within the ISI boundary at the time of

the ISI. To ensure that this is the case, due cognisance should be taken of all currently available plant knowledge. The determination of an effective risk-informed inspection strategy requires the development of a feedback procedure based on the idea of updating the risk ranking after plant changes affecting the probabilities of failure or consequences of failure have been made.

The affected portions of the risk-informed in-service programme should be re-evaluated as new information affecting the implementation of the programme becomes available (component system design change, plant PSA changes, plant operating condition changes, industry-wide failure notifications, etc).

Also very relevant is the information gathered after the inspection exercise has been completed (even if no acting damage mechanism is found) as it increases the knowledge of the plant and should be carefully fed back into the process. This information clearly influences the assessment of the site probability of failure as the uncertainty concerning the presence or absence of a degradation mechanism is changed.

This active (or living) process is one of the strengths of the risk-informed approach, as it leads to an enabling process that is both flexible and responsive to emerging problems.

If evidence of significant damage is found by inspection it is assumed that actions are taken to reduce the increased risk. These actions include substitution, repair, or fitness-for-purpose assessments to justify maintenance in service coupled with prescriptive follow-up inspections of the affected locations at subsequent outages. An assessment should also be carried out during the current outage to determine whether the defect is due to particular conditions at the affected location(s) or if it is the consequence of a more widespread damage mechanism. In the latter case, additional examinations should be carried out to determine the possible extent of the condition.

From the point of view of a risk-informed methodology, the question must be posed as to whether the occurrence of the degradation was in line with that expected when the risk was assessed prior to inspection. If the answer to this question is negative, then the models that were used to evaluate the probability of failure need to be reassessed.

Even if no evidence of defects is found after the risk-informed ISI programme is completed, it is still very important to ponder the meaning of this result. The critical issue then becomes the capability of the inspection technique (see sub-section 4.10.4 for a discussion on the issue of inspection qualification).

Some guidance regarding living probabilistic safety assessment can be found in [16]. Reference [17] provides eight examples of plants that have conducted updates to their RI-ISI programmes.

5 ORGANIZATION AND RESPONSIBILITIES

5.1 Introduction

The overall responsibility for the RI-IS programme lies with the plant operator (licensee).

In this section the roles, responsibilities and interfaces of the different parties involved in the creation of a RI-ISI programme are described.

It is recognised that utilities in different countries have different structures that vary in detail. Hence, what follows is a suggested management structure and interfaces that can be used as a guide to what is required in order to implement a risk-informed ISI programme.

5.2 Outline of management structure

The main parties/personnel involved are as follows:

1) The RI-ISI Responsible Person.

2) The RI-ISI Advisory Panel.

3) The RI-ISI Team.

4) The RI-ISI Review Panel.

5) The Inspection Qualification Team.

The reporting chain of the responsible parties/personnel is shown in Figure 5.

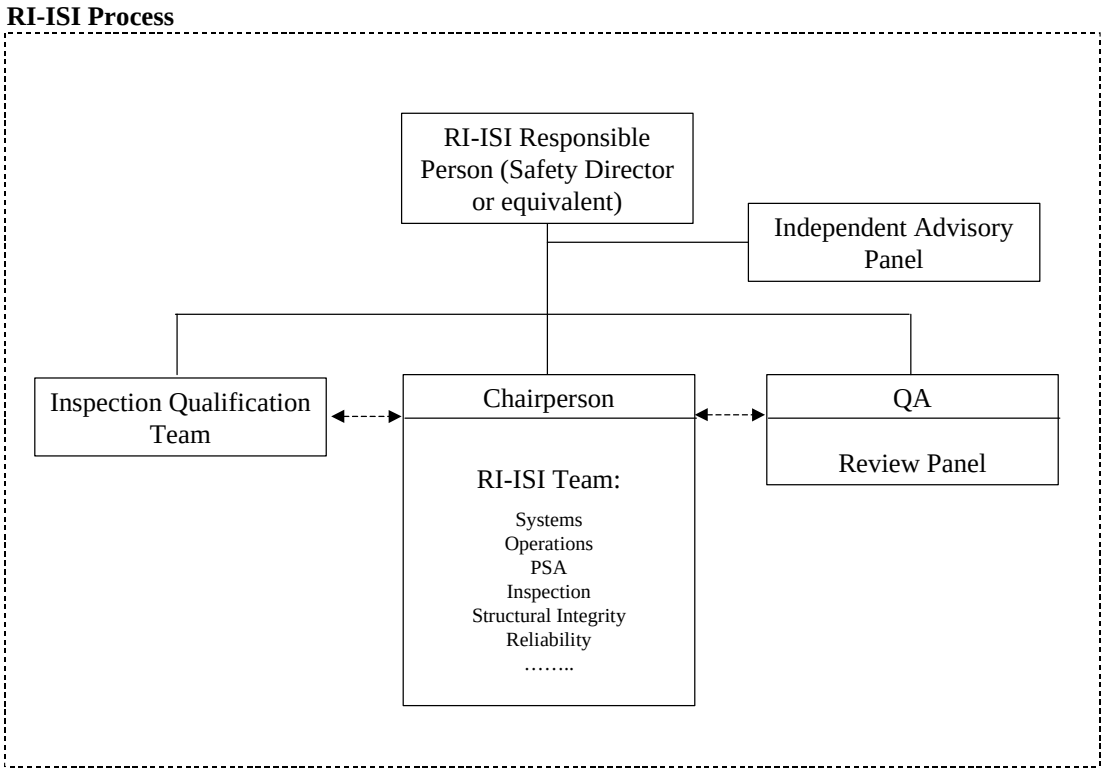

Figure 5 Reporting Chain Identifying the Main Responsibilities in a RI-ISI Process

5.3 Definition of responsibilities

5.3.1 The RI-ISI responsible person

The RI-ISI Responsible Person is responsible for setting the boundary and scope of the RI-ISI programme. He or she is ultimately responsible for the acceptance of the final RI-ISI programme against the boundary and scope set, and for these reasons, the Responsible Person should be a senior employee of the utility.

The Responsible Person will ensure that sufficient resources are made available for the full RI-ISI procedure to be followed through to the final Inspection Qualification and data archiving of the proposed inspections. The Responsible Person will form the RI-ISI team and appoint its chairperson. It will then be the RI-ISI team responsibility to recommend a RI-ISI programme to the Responsible Person. In order to assist the Responsible Person in accepting the proposed RI-ISI programme, he or she may wish to form a RI-ISI Advisory Panel, appointing senior advisors.

5.3.2 The RI-ISI independent advisory panel

The responsibility of the Advisory Panel is to counsel the Responsible Person with regard to any areas of the proposed RI-ISI programme that in its opinion are questionable, be it from the analytical modelling used, possible omissions, external considerations, etc.

The Advisory Panel will not have the power to change the risk rankings unilaterally or, more specifically, change how the probabilities or consequences of failure and hence the risk are derived. It may recommend changes, verification of the work carried out, etc.

The panel should raise any questions or concerns associated with ISI that may not be directly described in terms of the parameters used to measure the failure or consequence aspects of the risk-informed methodology adopted. This would include political or public concerns that cannot be uniquely described in terms of the consequence.

The RI-ISI Advisory Panel should be independent from commercial or operational considerations. Representatives from the regulator could attend meetings of the panel as observers.

5.3.3 The RI-ISI team

The RI-ISI team will need to be a multi-disciplinary team, with expertise including quality, inspection, maintenance, design, materials, chemistry, stress analysis, systems, PSA, operations and safety.

The RI-ISI team has the responsibility of developing the RI-ISI programme and following it through to its implementation. It is responsible for co-ordinating the required effort within the utility, to produce the necessary documentation, compile the RI-ISI dossier and ensure that the relevant quality assurance (QA) procedures are followed and records kept.

It is the responsibility of this team, via the authority of the RI-ISI Responsible Person, to ensure that the relevant departments within the utility provide the necessary support to generate a RI-ISI programme. This support will be required to produce a ranking of the probabilities of failure and the consequences for all the probable inspection sites within the

defined ISI boundary. The team will be responsible for setting up expert elicitation exercises when required and ensuring an adequate interchange between the various experts required for a RI-ISI programme.

The RI-ISI Team will present for review its probability of failure (POF) and consequence of failure (COF) rankings to the RI-ISI Review Panel. It also has ultimate responsibility for the finalisation of the rankings of probability of failure and consequence of failure, and for integrating such rankings in order to construct a risk ranking for all the locations within the scope of the exercise.

The RI-ISI team will establish a risk acceptance criterion, agreed with the Regulatory Body.

The RI-ISI team will outline an inspection programme based on such risk ranking, Risk acceptance criterion and taking into due account:

1) Risk outliers.
2) High consequence-low probability sites.
3) Defence in depth considerations.
4) Uncertainties associated with the models, etc.

In doing so, the team will also interface with the Inspection Qualification team to assess that the proposed inspection sites are both practical in the sense of carrying out the inspection and that there is a realistic chance of providing an inspection qualified to the ENIQ methodology.

The RI-ISI team will present the RI-ISI programme for approval to the Responsible Person.

5.3.4 The RI-ISI review panel

The purpose of the Review Panel is to provide an essential independent element in the Quality Assurance process.

The Review Panel should contain experts in the relevant areas that are independent from those belonging to the RI-ISI team. Such experts could be from either inside or outside the utility. Their independence must be ensured in the sense that they should not have been at any stage involved in the generation of the basic POF and COF data to be ranked.

The critical task of the Review Panel is to review the work carried out by the utility (RI-ISI team) in deriving the estimates of POF and COF.

As a part of the RI-ISI process, the RI-ISI team should provide the Review Panel with draft POF and COF rankings, plus details regarding the assumptions made and the calculations carried out. The members of the Review Panel should be in dialogue with the RI-ISI Team to agree a final set of ranked values. The RI-ISI team will ultimately be responsible for the finalisation of the ranking of the probabilities of failure and consequences of failure.

The Review Panel should use all the pertinent probabilistic and deterministic data provided by the RI-ISI team plus any world data available, together with their own expertise and experience.

5.3.5 The Inspection Qualification team

The Inspection Qualification team has the responsibility of advising the RI-ISI team with regard to the feasibility of achieving the specified outcomes from a proposed ISI programme. It should be clearly understood that the Inspection Qualification team cannot, at this time, guarantee that any subsequent inspection qualification against the specified requirements will be successful.

5.3.6 The Regulatory Body

In all countries the Regulatory Body is assigned the task of monitoring and evaluating safety and of ensuring that the licensee fulfils the conditions of its site licenses.

In the context of a safety-driven RI-ISI programme, the Regulatory Body either defines or reviews the basic safety requirements that must be met.

The Regulatory Body also undertakes audits, periodic reviews and monitors the licensee's compliance with the safety requirements. To these ends the Regulatory Body may wish to observe the development of any safety-driven RI-ISI programme. For instance, the Regulatory Body may wish to participate with the status of observer in the RI-ISI Advisory Body meetings.

6 DOCUMENTATION AND ARCHIVING

6.1 RI-ISI programme dossier

The RI-ISI team will prepare the RI-ISI dossier which should contain all information related to the whole process of risk-informed inspection planning and should comprise at least the following:

1) Nomination of the personnel involved in the process.

2) Scope of the RI-ISI programme.

3) Input data.

   a) All data used in calculations (design specifications, historical plant data, etc.).
   b) Assumptions made.
   c) Models used.
   d) PSA data.
   e) Risk criteria used.

4) Output data.

   a) Records of meetings of panels, etc..
   b) List of sites to be inspected.
   c) Requirements for inspection qualification under ENIQ (including qualification level).
   d) Subsequent analysis.
   e) Further action required.
   f) Risk change for sites inspected.

If parts of this information cannot be included, precise references should be made in the RI-ISI dossier.

6.2 Obsolescence of storage medium

Problems associated with the future obsolescence of the medium used for storing the RI-ISI Dossier information should be properly addressed.

7 REFERENCES

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[10] Probabilistic Risk Assessment Peer Review Process Guidance, Revision A3, Nuclear Energy Institute, NEI-00-02, March 2002.

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[15] ASME Boiler and Pressure Vessel Code, Section XI: Rules for In-service Inspection of Nuclear Power Plant Components, Document Number: ASME Section XI, American Society of Mechanical Engineers, 2001.

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APPENDIX 1 – Definitions

ALLOWED OUTAGE TIME. The number of hours the plant may operate under a pre-defined configuration (e.g. inoperable equipment) as controlled by the limiting condition(s) for operation in the Technical Specifications.

APPROACHES TO RI-ISI: Three different approaches are used: the purely quantitative, the purely qualitative and the semi-quantitative approach.

Purely quantitative approach
In a purely quantitative approach, numerical estimates are determined for probabilities of failures and consequences of failures. Fully validated and verified Structural Reliability Models are used to determine POFs and the plant Probabilistic Safety Assessment is used to calculate COFs.

Purely qualitative approach
In a purely qualitative approach, both the POFs and COFs are characterised by qualitative terms, such as extremely low, low, high, etc. In this framework, expert elicitation is used to rank the POF and COF of individual components. The purely qualitative approach does not use the quantitative risk assessment (PSA) results in the consequence evaluation. A qualitative analysis can be used to allow systems to be quickly prioritised for further more detailed (quantitative) risk analysis.

Semi-quantitative approach
A semi-quantitative approach is one where the COFs are based on PSA results, but the PSA information may be expressed equally well in quantitative or qualitative terms. The failure probabilities may be partly evaluated using fully validated and verified SRM, but partly also using experience data and expert elicitation. The resulting risk plot or risk matrix can be expressed on a qualitative scale but with quantitative links aiding a realistic assessment of the true risk.

CONDITIONAL CORE DAMAGE FREQUENCY (CCDF): Conditional frequency of core damage, given a loss of a mitigating ability.

CONDITIONAL CORE DAMAGE PROBABILITY (CCDP): Conditional probability of core damage, given an initiating event.

CONDITIONAL LARGE EARLY RELEASE PROBABILITY (CLERP): Conditional probability of a large early release, given an initiating event.

CONSEQUENCE OF FAILURE*:* See section 3.

CORE DAMAGE: Heating-up of the reactor core (typically following loss of coolant) to the point where damage to reactor fuel elements or cladding takes place.

CORE DAMAGE FREQUENCY: An estimated frequency of occurrence of events leading to core damage.

DEFENCE IN DEPTH: A design and operational philosophy with regard to nuclear facilities that calls for multiple layers of protection to prevent and mitigate accidents. It includes the use of controls, multiple physical barriers to prevent release of radiation, redundant and diverse key safety functions, and emergency response measures.

EXTERNAL EVENT: An event that initiates outside of plant systems and results in the perturbation of steady-state plant operation (e.g., seismic event, storm, etc.).

FAILURE: See section 3.

FAILURE MODES AND EFFECTS ANALYSIS (FMEA): A detailed technique specifically designed to identify the failure of an analysed component, the impacts of the failure on operations, the system and surrounding components, and controls for limiting the probability of such failures.

INITIATING EVENT: An event that perturbs steady-state plant operation or normal shutdown evolution resulting in a plant transient and challenge to control and safety systems. Based on its origin, an initiating event can be an internal or external event.

IN-SERVICE INSPECTION (ISI): An inspection performed after commissioning inspections and test runs are satisfactorily completed and the system or component has been certified or accepted for normal service operation. The objective of such inspections is to detect degradation that might have occurred during plant operation.

INTERNAL EVENT: An event that initiates within plant systems and results in the perturbation of steady-state plant operation (e.g., loss of coolant, loss of heat sink, etc.)

LARGE EARLY RELEASE: A radioactive release from the containment which is both large and early. Large is defined as involving the rapid, unscrubbed release of airborne aerosol fission products to the environment. Early is defined as occurring before the effective implementation of the off-site emergency response and protective actions.

PIPING SEGMENT: Continuous length of piping with the same degradation mechanism and failure consequence.

PIPING SYSTEM: An assembly of piping segments with defined functions. A piping system might include one or more AMSE Code classes.

PRESSURE BOUNDARY FAILURE. Piping element failures involving ruptures or leakage that result in a reduction or loss of the element pressure-retaining capability.

PROBABILISTIC SAFETY ASSESSMENT (PSA): A quantitative assessment of risk. For nuclear power plant application, the risk is associated with plant operation and maintenance. Risk is measured in terms of the frequency of occurrence of various events, leading to a consequence of interest (e.g., core damage or release of radioactive material).

PROBABILITY: A numerical measure of the state of confidence about the outcome of an event.

PROBABILITY OF FAILURE: See section 3.

RISK: See section 3.

RISK- INFORMED IN-SERVICE INSPECTION (RI-ISI): In-service inspection optimised with risk insights.

RISK REDUCTION: The risk reduction achieved by the ISI programme is the amount by which the total risk is reduced by undertaking ISI.

RISK REDUCTION WORTH: An importance measure indicating the maximum possible reduction factor in risk, if the item under consideration (e.g. a pipe segment) is assumed perfectly reliable.

SEGMENTS: See piping segment.

SPATIAL EFFECTS: The indirect impact of an event affecting other systems and components in the spatial vicinity, including flooding, spray, pipe whip, jet impingement, etc.

STRUCTURAL RELIABILITY MODELS: Models concerned with the calculation and prediction of the probability of safety limit state violations (failure) for engineering structures.

APPENDIX 2 – Abbreviations

ALARP:	As Low As Reasonably Practicable
ASME:	American Society of Mechanical Engineers
BWR:	Boiling Water Reactor
CCDP:	Conditional Core Damage Probability
CCDF	Conditional Core Damage Frequency
CCF:	Common Cause failures
CDF:	Core Damage Frequency
CLERP:	Conditional Large Early Release Probability
COF:	Consequence of Failure
ENIQ:	European Network for Inspection and Qualification
FMEA:	Failure Modes and Effects Analysis
IAEA:	International Atomic Energy Agency
ISI:	In-Service Inspection
LERF:	Large Early Release Frequency
LOCA:	Loss of Coolant Accident
NPP:	Nuclear Power Plant
NRWG:	Nuclear Regulators' Working Group
POD:	Probability of Detection
POF:	Probability of Failure
PSA:	Probabilistic Safety Assessment
QA:	Quality Assurance
RBMK:	Reactor Bolshoy Moshchnosty Kanalny (Russian designed high-power channel reactor)
RI-ISI:	Risk-Informed In-Service Inspection
RPV:	Reactor Pressure Vessel
RRW:	Risk Reduction Worth
SCC:	Stress Corrosion Cracking
SC	Steering Committee
SRM:	Structural Reliability Model
TGQ	ENIQ Task Group Qualification
TGR	ENIQ Task Group Risk