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Risk Informed In-service Inspection

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ABSTRACT

Safety of nuclear power plants is one of the most important conditions for their acceptance. Safety is being acheived by numerous methods and techniques in phase of design, manufacturing and maintenance of the nuclear power plants.

In-service Inspection (ISI) has a significant role in avoidances of failure in components of nuclear power plants just the same as in assurance of their integrity. Non-destructive examinations are performed periodically in accordance with 10 CFR 50 50.55a and ASME Boiler and Pressure Vessel Code section XI which is referenced by 10 CFR 50.55a. Non-destructive examinations provide information about a current condition of equipment at nuclear power plants and about any damage, defect or degradation mechanism. A lot of effort is often spent in situations in which the probability of failure and their effects on safety have a very low impact. Practical experience shows that failures can often occur at locations where the inspection has never been performed. Costs and expenses of in-service inspections are very high. Therefore, the accent has to be on locations with significant risk to safety. Many years of nuclear power plants' operation and maintenance have resulted in a more broad knowledge of degradation mechanism and the most susceptible locations and huge databases of different nuclear power plants' components.

U.S. Nuclear Regulatory Commission (NRC) and the nuclear industry have recognized that probabilistic risk assessment (PRA) has developed and changed to be more useful in improvement of traditional engineering approaches in nuclear power plants regulation. After the publication of its policy statement on the use of PRA in nuclear regulatory activities, the Commission ordered the NRC staff to develop a regulatory framework that incorporated risk insights.

The American Society of Mechanical Engineers (ASME) initiated Code Case N-560, N-577, and N-578 that address the importance of categorization and inspection of piping using risk insights. It is expected that licensees will ask for modifications of plant's design, operation, and other activities that require NRC approval to incorporate risk insights into their ISI programs. In-service inspection program in which is incorporated risk insights is known as risk-informed in-service inspection programs, RI-ISI. Until the RI-ISI is approved for generic use, the staff expects that licensees will ask for modifications of ISI programs by requesting NRC to approve alternative inspection programs. Alternative inspection programs must meet 10 CFR 50.55a (a)(3)(i) which says that "the applicant shall demonstrate that the proposed alternatives would provide an acceptable level of quality and safety". As always, licensees should identify how the chosen approach, methods, data, and criteria are appropriate for the decisions that have to be made.

RI-ISI can provide reduction of scope and frequency of inspections that may lead to higher risk. The higher risk means that a probability of failure or consequences are higher and

shutdown of nuclear power plants is more likely to occur. RI-ISI can also provide extensive scope of inspection or increased frequency of inspection.

1 INTRODUCTION

Plant operating experience and research in the fields of material sciences, degradation mechanisms, inspection techniques and methods combined with risk assessment techniques and associated data can be used to develop a more effective approach to ISI programs for piping which can focus inspection on the more important locations and reduce personnel exposure, while at the same time maintaining and improving public safety.

In this paper it will be given a status of RI-ISI in Europe and in U.S.A. and an overview of methodologies which have been used in Europe and in U.S.A.

2 NUCLEAR SAFETY

Nuclear fuel is placed in a nuclear reactor which therefore consists of a huge amount of radioactive materials. Core damage is the main and basic measure for nuclear safety as a core damage is the prelude to any nuclear release. An estimated frequency of occurrence of events which lead to core damage is known as core damage frequency (CDF). Core damage can vary from small to large damage and it is not directly an indicator of severity of the event or accident. The core damage may occur when there is no loss of radiation inside or outside of nuclear power plants and that's why the measure for radioactivity release is the large early release frequency (LERF) instead of core damage frequency (CDF). The large early release is "a radioactive release from the containment which is both large and early. Large is defined as involving rapid, unscrubbed release of airborne aerosol fission products to the environment. Early is defined as an occurrence before the effective implementation of the off-site emergency response and protective actions". Every release of radioactive materials in the environment can endanger the public health and safety not only next to nuclear power plants but also further away.

The defense-in-depth philosophy has been applied in nuclear power plants' design and operation. Their role is to prevent the release of radioactive material and to achieve a necessary level of safety. The defense-in-depth is a series of barriers such as a fuel matrix, a fuel cladding, a primary coolant circuit boundary and a containment structure. Operation of a nuclear power plant is allowed provided that the said barriers are not endangered and their function hasn't been lost. Technical systems protect the physical barriers from failure but also taken measures during design, construction and operation phase provide safety and also enhance reliability of physical barriers. All of the elements of defense-in-depth must be available at all times so that a nuclear power plant can operate normally.

All materials in a nuclear power plant can undergo aging and can lose partial or total design function *(Figure 2.1.).* Aging mechanisms of nuclear power plants' components may have an impact on the effectiveness of the defense-in-depth. Aging processes in nuclear power plants' components degrade and change property of components. It is possible that a degradation mechanism is not disclosed during operation or even testing and may lead to an accident.

In-service Inspection is one of the measures taken which provides safety with continuously monitoring and observing components' condition. In spite of conservative design, the owner of the nuclear power plant must obey Section XI, Rules for In-service Inspection of Nuclear Power Plant Components, of the ASME Boiler and Pressure Vessel

Code which consists of requirements for examination, testing, and inspection of components and systems, and repair and replacement activities in nuclear power plant.

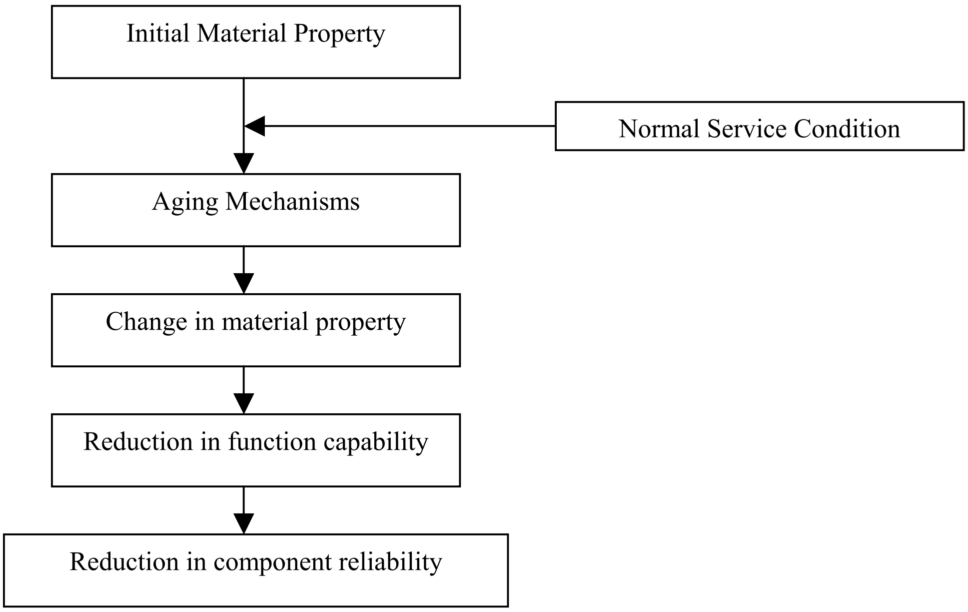

Figure 2.1. Impact of service conditions on material degradation and component reliability

3 WORLD RI-ISI STATUS

3.1 United States of America

ASME Boiler and Pressure Vessel Code section XI initiated Risk-informed in-service inspection programs as an alternative program to current inspection program. In-service Inspection is already risk informed due to ASME containing two basic risk elements, probability of failure and consequence. Probability of failure is incorporated in ASME Section XI and consequence in ASME Section III through categorization of the plant components depending on the probability of failure which leads to core damage. ASME approved Code Case N-560, Code Case N-577 and Code Case N-578 for risk-informed piping inspection. Code Case N-560 concerns Class 1, which is different from Code Case N-577,Westinghouse Owner Group Methodology (WOG), and Code Case N-578, Electric Power Research methodology as Class 1, 2 or 3 piping are concerned. ASME has developed Appendix X, which concerns Class 1, 2 and 3, for both EPRI and WOG methodology, which incorporates lessons learned from plant specific reviews. It is anticipated that Code Cases will be incorporated in ASME Code.

NRC PRA policy statement encourages great use of PRA to improve safety decision making and improve regulatory efficiency. Nuclear Regulatory Commission has developed and published several Regulatory Guides and Standard Review Plan sections which provide guidance on use of PRA findings and risk insights in support of licensee requirements for changing plant licensing basis.

Regulatory Guides:

- 1.174 An Approach for using Probabilistic Risk Assessment in Risk- Informed Decisions On Plant-Specific Changes to the Licensing Basis
- 1.175 An Approach for Plant-Specific, Risk informed Decision-making: Inservice Testing

- 1.176 An Approach for the Plant Specific, Risk-Informed Decision-making: Graded Q
- 1.177 An Approach for the Plant-Specific, RI-Decision Making: Technical Specifications
- 1.178 An Approach for the Plant-Specific, Risk-Informed Decision-making: ISI of Piping
- 1.182 Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants

Standard Review Plans

- Chapter 19 Use of PRA in Plant-Specific, Risk-Informed Decision-making: General Guidance
- Chapter 3.9.7. Risk Informed In-service Testing
- Chapter 16.1. Risk-Informed Decision-making: Technical Specifications
- Chapter 3.9.8. Trial Use for the Review of Risk-Informed In-service Inspection Piping.

Recent activities on Risk-Informed In-service Inspection include the extension of RI-ISI methodology to Break Exclusion Zone Region Piping (BER) through the 10 CFR 50.59 which is in process. Both EPRI and WOG have developed methodologies for said application.

3.2 Europe

Different initiatives on risk-informed in-service inspection have been done at a national level in many European countries. In order to gain a common approach regarding RI-ISI within European Network for Inspection Qualification (ENIQ) Task Group 4 (TG4) was established. The TG4 is complementary to Task Force(TF) set up by Nuclear Regulators' Working Group. Most of the members of ENIQ TG4 joined the European Network of Risk-Informed In-Service Inspection (EURIS) set up by Directorate General RTD of the European Commission.

Task Force (TF) was established to agree with the principles and philosophy governing risk-informed in-service inspection and in-service testing of mechanical components of nuclear power plants in order to maintain sufficient margins against leakages and failures.

The TF was made of representatives from several European regulatory bodies or their representatives such as ANV (Belgium), SONS (Czech Republic), STUK (Finland), BfS (Germany), CSN (Spain), SKI(Sweden) and HSK (Switzerland).

The Task Force went through existing approaches of risk informed in-service inspection and through results of testing which were done in different countries. Their work was completed in 1999 with document ”Report on risk-informed in-service inspection and in-service testing” (EUR 19153), which contains conclusions and it gives recommendations for the future. It was believed that PSA methodology would contribute to optimization of in-service testing and in-service inspection and that pilot studies were useful to develop methodology and guidelines for implementing risk-informed in-service inspection and in-service testing proposals. Pilot studies have beed performed at NPP in Finland, Spain, Sweden and Switzerland.

The need for review of initiatives and approach of RI-ISI resulted in the set up 24-month Study Contract named RIBA by European Commission. The participates were Serco Assurance (project co-ordination), EDF, Ringhals AB, Tecnatom SA and Westinghouse Electric Europe.

4 OVERVIEW OF RISK-INFORMED IN-SERVICE INSPECTION

4.1 United States of America

WOG Methodology

WOG Methodology Code Case N-577 *(Figure 4.1.)* uses PSA analysis to divide plant piping systems into piping segments where failure has the same consequence as measured by core damage frequency. Each piping segment is categorized as low to high safety significance. Assessment of degradation mechanism and failure probability are used to ascertain failure importance of each segment from low to high. Segment has high failure importance if its probability of failure is greater then 10^{-4} per 40 years of plant's operation. The final safety significance category is a 2x2 matrix which is made by an expert panel review using the PSA and deterministic and design insights.

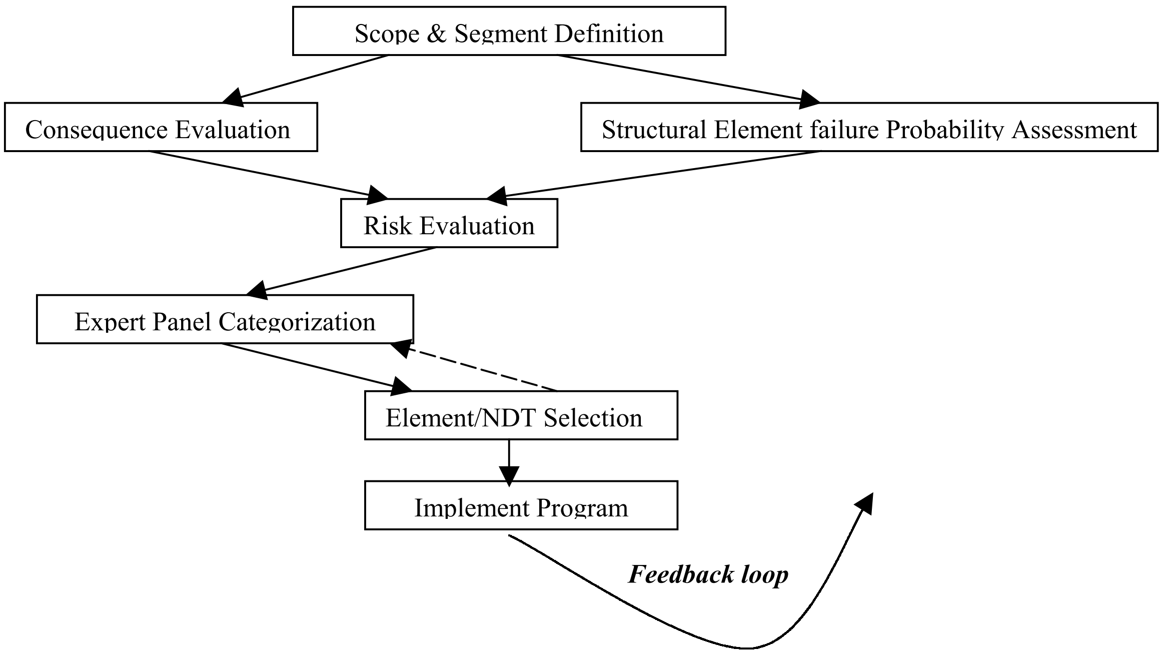

Figure 4.1. WOG RI-ISI Methodology

EPRI Methodology

EPRI Methodology *(Figure 4.3.)* refers to Class 1, 2 or 3 piping and commence with Failure modes and effects analysis (FMEA) which consists of direct and indirect consequence evaluation and engineering review of possible degradation mechanisms and associated failure modes. Pipings system are divided into segments in the way that a piping segment has the same degradation mechanisms and the same failure consequences for the given mode of failure.

Piping segments are categorized into one of seven risk region of 4X3 risk matrix *(Table 4.2.)*. Categorization is done according to consequence category and relative potential for large leak.

Table 4.2. Risk Matrix

Relative Potential for Large Leak	***Consequence Category***			
	None	*Low*	*Medium*	*Large*
Large	**Low CAT7**	**Medium CAT5**	**High CAT3**	**High CAT1**
Small	**Low CAT7**	**Low CAT6**	**Medium CAT5**	**High CAT2**
None	**Low CAT7**	**Low CAT7**	**Low CAT6**	**Medium CAT4**

Proportion of the welds that shall be inspected during 10 years hinges on the risk region that piping segment belongs to.

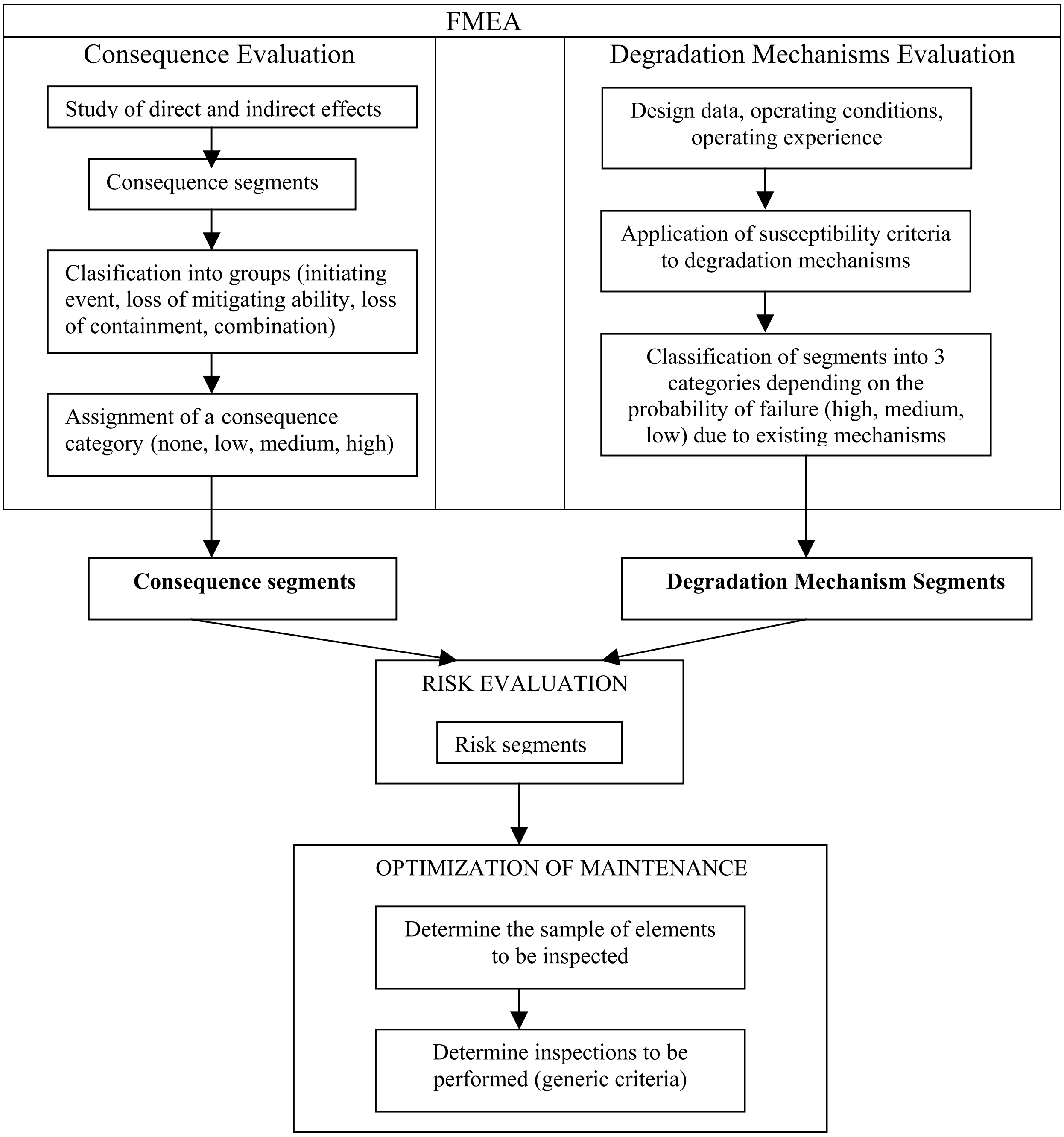

Figure 4.3. EPRI RI-ISI Methodology

4.2 Europe

Spain

The Spanish Utilities (UNESA) and the Spanish Nuclear Safety Council (CSN) have been developing a Risk Informed In-service Inspection Piping Guide for applying US RI-ISI Methodology to the Spanish Nuclear Power Plants. ISI Programs in Spain are mainly based on ASME Section XI. UNESA and Spanish regulatory body have an interest in a possible optimization of the ISI Programs. The objective of above said project was a guide for RI-ISI of piping that incorporates quantitative and qualitative methodologies. Intention of the guide is to extend the ASME scope in order to include other specific degradation programs, suggesting new inspection frequencies and methods.

Sweden

The Swedish approach is orientated to design safety class but not to probability safety analysis. Risk informed principles are used to assign components partially or entirely as inspection group A, B or C. Probability of cracking or other degradation and potential consequences are considered for classification components or their parts into inspection group A, B or C *(Table 4.4.)*.

Table 4.4. SKI Risk Evaluation Matrix

		Consequence Index		
		1	*2*	*3*
Damage Index	*I*	A	A	B
	II	A	B	C
	III	B	C	C

A – the highest risk
B – the intermediate risk
C – the lowest risk

Although the described Swedish approach is qualitative, there is a tendency to quantitative approaches using PSA.

France

The French utility EDF has developed the method called OMF Structures which uses risk-informed principles. The goal is to identify critical components but taking safety, availability and maintenance costs into account.

The OMF Structures process consists of the following:

1. Functional analysis at system level
2. Consequence evaluation (FMEA) at component segment level
3. Analysis of critical state (FMECA) at component level
4. Definition of preventive maintenance programs
5. Preventive maintenance programs and/or corrective maintenance programs.

Each segment of component or entire component is classified into safety categories such as very safety severe, safety severe and not safety severe. *Table 4.5.* shows critical state of components that is based on severity category of segment/component and potential degradation mechanisms.

Table 4.5. EDF's OMF Structure evaluation of critical state of components

Degradation mechanisms	***Severity Category***		
	Non severe	*Severe*	*Very severe*
Relevant mechanism and high probability	I	IV	V
Relevant mechanism and low probability	I	II	V
Non relevant mechanism	I	II	III

5 CONCLUSION

There are three main RI-ISI Methodologies that are used WOG, EPRI and EFD's OMF Structures process. WOG, EPRI and OMF Structures methodologies are based on the similar principles which are segment definition, consequence evaluation, probability assessment, risk evaluation and decision-making inspection selection. The main difference is in "qualitative" and "quantitative" approaches. Recent activities include the extension of RI-ISI methodology to BER piping via 10 CFR 50.59. EPRI and WOG have developed methodologies for the above mentioned application.

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