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Lessons Learned from the Application of Risk-Informed In-Service Inspection to European Nuclear Power Plants

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Authors

Robertas Alzbutas (LEI)
Otso Cronvall (VTT)
Carlos Cueto-Felgueroso (Tecnatom S.A.)
Eduardo Gutierrez Fernandez (Iberdrola)
Krešimir Gudek (Krško NPP)
Emil Kichev (Kozloduy NPP)
Anders Lejon (Ringhals NPP / Vattenfall)
Petri Luostarinen (Fortum)
Sorin Saulea (Cernavoda NPP)
Joakim Thulin (Forsmark NPP / Vattenfall)
Patrick O'Regan (EPRI; Ed.)
Oliver Martin (EC-JRC; Ed.)

NUGENIA Association

c/o EDF, Avenue des Arts 53, 1000 Bruxelles, BELGIUM

Email: secretariat@nugenia.org

Website: http://www.nugenia.org

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FOREWORD

This report was issued by NUGENIA Technical Area 8 (TA8) Sub-area on Risk (SAR). NUGENIA TA8 is the European Network for Inspection and Qualification (ENIQ), which is dealing with the reliability and effectiveness of non-destructive testing (NDT) for nuclear power plants (NPPs). ENIQ is driven by European nuclear utilities and is working mainly in the areas of qualification of NDT systems and risk-informed in-service inspection (RI-ISI). Since its establishment in 1992, ENIQ has performed two pilot studies and has issued 50 documents. Among them are recommended practices, technical reports, discussion documents and the two ENIQ framework documents, the "European Methodology for Qualification of Non-Destructive Testing" [1] and the "European Framework Document for Risk-Informed In-Service-Inspection" (RI-ISI) [2]. ENIQ is recognized as one of the main contributors to today's global qualification guidelines for in-service inspection. Its contributions are acknowledged, among others by the Western Nuclear Regulators Association (WENRA) [3].

The purpose of this report is to summarize the experience from Risk-Informed In-Service Inspection (RI-ISI) programmes and pilot studies of NPPs in Europe, in particular the experienced changes compared to previous deterministic ISI programmes. The reports covers the experience from countries, where RI-ISI is fully recognised by the nuclear regulator, but also countries that still follow a deterministic ISI approach, but which have performed RI-ISI pilot studies to get an idea of possible benefits or extra burdens when moving to RI-ISI. The report covers the experience from different reactor types, i.e. BWR, PWR, VVER, RBMK and CANDU reactor.

This document was formally approved for publication by the TA8 (ENIQ) Steering Committee (SC) and the following persons contributed to this report (in alphabetical order):

- Robertas Alzbutas — Lithuania Energy Institute (LEI)
- Otso Cronvall — Research Centre of Finland (VTT)
- Carlos Cueto-Felgueroso — Tecnatom S.A., Spain
- Eduardo Gutierrez Fernandez — Iberdrola, Spain
- Krešimir Gudek — Krško NPP, Slovenia
- Emil Kichev — Kozloduy NPP, Bulgaria
- Anders Lejon — Ringhals NPP / Vattenfall, Sweden
- Petri Luostarinen — Fortum, Finland
- Oliver Martin — European Commission – Joint Research Centre (EC-JRC)
- Patrick O'Regan — Electrical Power Research Institute (EPRI), USA
- Sorin Saulea — Cernavoda NPP, Romania
- Joakim Thulin — Forsmark NPP / Vattenfall, Sweden

This document was edited by Patrick O'Regan (EPRI) and Oliver Martin (EC-JRC).

EXECUTIVE SUMMARY

The purpose of this report is to summarize the experience from risk-informed in-service inspection (RI-ISI) programmes and pilot studies of nuclear power plants (NPPs) in Europe, in particular the experienced changes compared to previous deterministic ISI programmes. The report covers the experience from countries, where RI-ISI is fully recognised by the nuclear regulator, but also countries that still follow a deterministic ISI approach, but which have performed RI-ISI pilot studies to get an idea of possible benefits or extra burdens when moving to RI-ISI. The report covers the experience from Finland, Slovenia, Spain, Sweden, Bulgaria, Lithuania and Romania and thus covers different reactor types, i.e. BWR, PWR, VVER, RBMK and CANDU reactor.

1. Introduction

The purpose of this report is to compile lessons learned from the application of risk-informed in-service inspection (RI-ISI) to European Nuclear Power Plants (NPPs). While different approaches to developing RI-ISI methodologies have been developed, this report documents only the results and lessons learned from their application as the intended audience is programme owners and plant management as opposed to RI-ISI subject matter experts. Additionally, as the development of a RI-ISI programme requires expertise from a number of different disciplines outside of the inspection organization, this report will be of interest to maintenance, design, materials, chemistry, stress analysis, systems, probabilistic safety assessment (PSA), operations and safety personnel.

RI-ISI reflects recent developments in PSA technology, the understanding of degradation mechanisms (e.g. structural reliability modelling, root cause evaluations) and the experience gained from nearly 16,000 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 safety significant locations where the inspection efforts should be concentrated. The objectives are to provide ongoing improvements in overall plant safety (as measured by risk) together with optimized outage management practices, reductions in doses for the inspection teams and radwaste.

2. European Lessons Learned

Currently over thirty nuclear units in ten European countries have conducted full RI-ISI applications (i.e. with regulatory approval) or RI-ISI pilot studies. These applications cover a wide spectrum of plant designs (BWRs (Asea-Atom (later ABB-Atom), GE), PWRs (Framatome, Westinghouse), VVER, CANDU and RBMK) and scope of application (e.g. limited to the reactor coolant pressure boundary only or up to the full plant (i.e. safety related and non-safety related systems)). This chapter discusses most of these applications and presents lessons learned. This chapter is divided into two parts. Section 2.1 is devoted to applications that have received regulatory endorsement in their particular country while Section 2.2 presents the results for those applications that are at the pilot study stage.

2.1 Regulatory Approved RI-ISI Applications

2.1.1 Experience in Finland

In Finland, the use of risk-informed methodology in establishing the ISI program has been a regulatory requirement since 2004. The operating plants scheduled the moving from deterministic ISI to RI-ISI according to the renewal of their operating licenses. The operating NPPs have completed the RI-ISI applications, and started inspection according to the new ISI programs. Both Loviisa NPP and Olkiluoto NPP have now gained the regulatory approval (the latter plant during spring 2014).

System risk assessments were based on the existing extensive PSA model and the failure consequences to other systems were considered as they are taken into account in the PSA model. Conditional Core Damage Probability (CCDP) and Conditional Large Early Release Probability (CLERP) were used as selection criteria for the consequence assessment and classification of the systems (categorization/ segmentation). The work was focused on such systems and piping segments the failures of which can cause a core damage with a probability larger than 10^{-6} or a large early release with a probability larger than 10^{-7}. The other systems and piping have only minor effect on the core damage risk and are screened out. Failure assessments were performed on the qualitative basis by the panel of the company experts familiar with Loviisa NPP associated materials integrity and in-service activities.

The RI-ISI approach followed the main lines of the ASME XI Appendix R (Method B) [4][5], but with some unique different features. For instance the criteria in categorisation of the degradation potential were different from those in the US approach. The approach comprised also an independent expert panel to evaluate the risk ranking. The Independent Advisory Panel reviewed the basis of the risk classification in order to guarantee that the final RI-ISI program will be planned properly. The Nuclear Safety Authority STUK was invited as an observer to the panel sessions.

2.1.1.1 Loviisa NPP, Units 1 & 2

<u>Previous ISI Program</u>

The pre-service and ISI of the primary circuit components and piping of Loviisa NPP (VVER-440) have been carried out since the start-up of the plant following the ISI-programme, which is based on the requirements presented in ASME Section XI [6]. The last deterministic 10 years ISI programmes were completed in 2007 (Loviisa 1) and 2010 (Loviisa 2).

In addition to the ASME Code Section XI requirements, measurements and analysis of wall thinning have been carried out as part of maintenance activities. A control programme was developed in 1982-1983 to manage erosion-corrosion in the secondary system. The programme was widened and additional measures taken after guillotine pipe breaks of the feedwater system that occurred in 1990 and 1993 [7].

RI-ISI Results

The scope of the RI-ISI program evaluation covered not only safety class 1 and 2 piping as the existing deterministic ISI program, but the whole plant:

- Safety systems (emergency cooling, heat removal, fire protection etc.)
- Primary and secondary systems (primary piping, main steam lines, seawater cooling system etc.)
- Auxiliary systems (auxiliary water, waste water, pressurised air system, etc.)

These systems contain water, steam, nitrogen, hydrogen, air, oil, chemicals, and also empty systems are included. More than 100 systems were screened and around 50 were included in failure categorisation.

At the time of the writing of this report, not enough information was available and combined to allow producing detailed tables concerning those systems that were fully evaluated (i.e. not screened) for Loviisa 1 and Loviisa 2.

After completion of the consequence assessment and degradation potential evaluation a risk ranking is conducted and review and approval by the independent advisory panel is obtained.

In addition to the changes in inspection locations, the new RI approach has brought several new features [8] compared to the deterministic ISI program as listed below:

- New systems and / or portions of the systems are included.
- Small diameter instrumentation piping of the primary systems to be inspected.
- Consequence differences of parallel redundant safety system and system portions can often be discovered (mostly due to fire and flood).
- The new RI-ISI program includes a wide range of different inspection objects with different degradation mechanisms and inspection targets. This is managed by separate programs.
- The inspection system (procedure, personnel, equipment) shall be qualified.
- The systems with erosion corrosion, corrosion and outside surface failures are included in the piping condition monitoring program.
- The systems with possible impact of outside failures and leakages (typically small size piping) are included into the “walk down” program of the power plant with typical VT-2 and VT-3 visual examination.

Lessons Learned

Due to the fact that acceptance criteria for components belonging to safety classes 3 and 4 are not presented in the ASME Code, other criteria had to be defined to specify the acceptable sizes of flaws in these lower safety classes. So far there are no totally new inspection items but some inspections that are

now part of ISI have been performed in the past as a part of normal condition monitoring. Therefore, during the first years of RI-ISI implementation there has not been any remarkable change in the number of inspections performed during the annual shutdown. Consequently, the total radiation exposure of inspection personnel has not decreased. However, it is foreseen that the doses will be decreasing in the future due to the better focusing and timing of inspections. The most important benefit that is expected from the new RI-ISI program is the improvement of plant safety when more effort in ISI is placed on inspection items that are most important from the safety point of view.

Risk informed approach for all the piping systems of plant requires a huge amount of work. The co-operation of different organizations and professional areas inside and outside of the company has been fruitful for the future development of ISI program. The involvement of STUK in the process through the panel sessions was considered advantageous, since it helped the safety authority to follow and understand the RI-ISI process. The benefits of the work are not gained from the reduction of inspection scope, but aiming to reduce the total risk.

2.1.1.2 Olkiluoto NPP, Units 1 & 2

Previous ISI Program

The pre-service and in-service inspections of the primary circuit components and piping of Olkiluoto 1 & 2 BWR units (Asea-Atom) have been carried out since the start-up of the plant following the ISI-programme which is based on the requirements presented in the ASME Code Section XI [6].

Due to the intergranular stress corrosion cracking (IGSCC) findings in the 1980's and early 1990's, piping replacements as well as water chemistry changes were made and the inspection programs were extended to target IGSCC susceptible piping. Non-destructive inspection programmes have also been extended because of thermally induced fatigue.

RI-ISI Results

The initial scope of the RI-ISI program covered the entire plant, as required by the nuclear safety authorities. Safety classes 1,2,3,4 and EYT (non-nuclear) were included. Also the inclusion of bellows, hoses and flange connections was required by STUK.

Olkiluoto units both have 101 systems that include pipelines. Amongst them, 58 systems per unit were selected for RI-ISI analyses. Table 2.1.1-1 presents a listing of those systems that were fully evaluated (i.e. not screened) for Olkiluoto Units 1 & 2. Examples of systems that qualitatively screened out include many turbine side systems, scram, generator cooling, sealing, leakage and drainage systems. The degradation mechanism and failure potential of each separate system was reviewed by the expert panel. The erosion and flow-accelerated corrosion (FAC) inspection program was evaluated with RI approach, but kept to a large content as previously inspected.

The adopted RI-ISI approach followed largely the same principles as Loviisa RI-ISI: full scope, extensive use of plant PSA model, qualitative analysis of failure potential, and the use of independent expert panel to review the risk ranking. The classification of degradation potential is different from both Loviisa and ASME XI Appendix R (Method B) [4][5] approaches (e.g. assessment of IGSCC susceptibility).

Another difference in the approach is that degradation evaluation was done at weld level already in the risk ranking phase for reactor auxiliary systems. For these systems, the RI-ISI analyses utilised the plant

pipeline database, which includes e.g. piping dimension, material and loading information and allows various searches and analyses to support the evaluation and presentation of RI-ISI results. For other systems, the analysis was done at less detailed level [9][10].

Reactor auxiliary systems (300-systems), with SCC as potential degradation mechanism are evaluated on weld level. Weld information are fed into a 3D-program, which uses e.g. the following input from the database: diameter, base material carbon contents, temperature data, cumulative fatigue value (for systems where regulatory guide YVL requires), primary and secondary stress intensity ratio, flux. Based on this input data, the failure potential of each weld is determined, and together with the segment's consequence level the total risk of the weld is determined. For each system the minimum total weld amount for ISI is derived from the total amount of high/moderate risk class welds per system (> 25 % of high risk welds; > 10 % of moderate risk class welds).

Table 2.1.1-2 presents the results of the degradation evaluation for Olkiluoto 1 and Table 2.1.1-3 presents the results for Olkiluoto 2. Similarly as in the RI-ISI applications for the Loviisa plants, an independent expert panel reviewed the analysis results, and the Nuclear Safety Authority STUK took part as an observer to the panel sessions.

Table 2.1.1-4 presents a comparison between the previous deterministic ISI program and the RI-ISI program for Olkiluoto 1 and Table 2.1.1-5 presents the comparison for Olkiluoto 2. The main differences between the deterministic program and the results of the risk-informed analyses for the reactor auxiliary systems are following:

- The RI-ISI analyses support a reduction in the number of inspections in auxiliary feedwater system, core spray system, shut-down cooling system and reactor water clean-up system.
- A significant amount of inspections in reactor water clean-up system could be reduced from the risk perspective, due to low consequences. However, a large part of the inspections will be maintained because of the IGSCC susceptibility of the piping.
- The RI-ISI analyses indicate a need to increase inspections in the feedwater system and in the relief system.
- Recirculation system and boron system were added as new systems to the ISI program due to high consequences of pipe breaks in some segments.

At the turbine generator plant, the main degradation mechanisms are FAC and localized corrosion. Also possible vibration was confirmed. The RI-ISI analysis did not identify any major changes to FAC or component support programs.

As can be seen in Table 2.1.1-4 and Table 2.1.1-5, the RI-ISI analysis results for Olkiluoto 1 and 2 are to a large extent similar. The main differences are due to piping replacements, affecting the degradation potential.

<u>Lessons Learned</u>

The change from the deterministic ISI program to RI-ISI did not affect very much the total number of inspections. However, there are significant changes in inspected sites. More inspections are focused on

piping that may have a low potential for IGSCC, but are identified by PSA as having relatively high consequences. Inspections were reduced in the areas of low consequences.

When determining new inspection sites, one criterion has been the minimisation of doses to the inspection personnel. Significant changes in sites to be inspected were made (e.g. in the shut-down cooling system), where inspections were reduced in areas with high doses.

Table 2.1.1-1: Olkiluoto Units 1 & 2 – List of Systems Evaluated

System ID	System	Highest safety class of system piping				
		1	2	3	4	None safety
311	Steam lines in reactor building	x				
312	Feedwater system	x				
313	Recirculation system	x				
314	Relief system	x				
316	Condensation system	x				
321	Shut-down cooling system		x			
322	Containment vessel spray system		x			
323	Core spray system		x			
324	Pool water system		x			
326	Flange cooling system		x			
327	Auxiliary feedwater system		x			
331	Reactor water clean-up system			x		
332	Condensate clean-up system					x
336	Sampling system					x
345	Controlled area floor drain system					x
351	Boron system		x			
354	Scram system		x			
361	Containment over-pressurization protection system		x			
362	Containment filtered venting system			x		
412	Steam reheat system				x	
413	Turbine plant main steam system				x	
414	Lubrication oil system				x	
416	Control and trip oil system				x	
417	Seal and leakage steam system					x
425	Generator cooling system				x	
431	Condenser and vacuum system				x	
434	Condenser cooling system					x
436	Make-up water system					x
441	Condensate system				x	
445	Turbine plant feedwater system				x	
447	Steam extraction system					x
541	Process measurements		x			
632	Generator switching device				x	
652	Diesel engines auxiliary systems		x			
655	Start-up air system			x		
656	Diesel oil storage systems			x		
712	Shut-down service water system		x			
713	Diesel-backed normal operation service water system			x		

System ID	System	Highest safety class of system piping				
		1	2	3	4	None safety
714	Non-diesel-backed normal operation service water system				x	
715	Sea-water cooling system for generator					x
721	Shut-down secondary cooling system		x			
723	Diesel-backed normal operation secondary cooling system			x		
725	Containment gas cooling system					x
726	Brine system for air conditioning					x
727	Air cooling system					x
733	Distribution system for demineralised water			x		
741	Containment gas treatment system		x			
751	Diesel-backed compressed air system					x
753	Non-diesel-backed compressed air system					x
754	Compressed nitrogen system					x
755	containment inerting system					x
763	Heating system					x
861	Fire fighting water system				x	
862	Fire sprinkling system				x	
866	Carbon dioxide extinguisher system				x	
867	Haltron extinguisher system				x	

Table 2.1.1-2: Olkiluoto Unit 1 - Degradation Mechanisms Identified as Potentially Operative

System ID	System	Thermal Fatigue		Stress Corrosion Cracking	Localized Corrosion			Flow Sensitive		Vibration Fatigue	Other
		TASCS	TT	IGSCC	MIC	PIT	CC	E-C	FAC		
311	Steam lines in reactor building		x	x							Panel
312	Feedwater system		x	x						x	Inconel
313	Recirculation system					x			x		-
314	Relief system		x								-
316	Condensation system										-
321	Shut-down cooling system			x							Inconel
322	Containment vessel spray system										-
323	Core spray system									x	Inconel
324	Pool water system										-
326	Flange cooling system			x						x	Panel
327	Auxiliary feedwater system			x						x	
331	Reactor water clean-up system		x	x							
332	Condensate clean-up system								x		CS
336	Sampling system										-
345	Controlled area floor drain system										VT
351	Boron system		x	x							Panel
354	Scram system										-
361	Containment over-pressurization protection system										VT & supports
362	Containment filtered venting system										VT & supports
412	Steam reheat system								x		CS, coating
413	Turbine plant main steam system								x	x	CS, supports, VT
414	Lubrication oil system					x				x	Supports
416	Control and trip oil system					x				x	Supports
417	Seal and leakage steam system								x		
425	Generator cooling system									x	
431	Condenser and vacuum system					x					
434	Condenser cooling system										Coating
436	Make-up water system									x	Supports
441	Condensate system								x		Cs

System ID	System	Thermal Fatigue		Stress Corrosion Cracking	Localized Corrosion			Flow Sensitive		Vibration Fatigue	Other
		TASCS	TT	IGSCC	MIC	PIT	CC	E-C	FAC		
445	Turbine plant feedwater system					x			x		CS
447	Steam extraction system								x		
541	Process measurements					x					RPV nozzle, VT
712	Shut-down service water system						x				Coating
713	Diesel-backed normal operation service water system						x				Coating
714	Non-diesel-backed normal operation service water system						x				Coating
715	Sea-water cooling system for generator								x		Coating
721	Shut-down secondary cooling system										Water chemistry
723	Diesel-backed normal operation secondary cooling system										Water chemistry
725	Containment gas cooling system										-
726	Brine system for air conditioning										-
727	Air cooling system										-
733	Distribution system for demineralised water										-
741	Containment gas treatment system										-
751	Diesel-backed compressed air system						x				
753	Non-diesel-backed compressed air system										-
754	Compressed nitrogen system										-
755	containment inerting system										-
763	Heating system					x	x		x		
861	Fire fighting water system				x	x	x		x		
862	Fire sprinkling system										

Table 2.1.1-3: Olkiluoto Unit 2 - Degradation Mechanisms Identified as Potentially Operative

System ID	System	Thermal Fatigue		Stress Corrosion Cracking	Localized Corrosion			Flow Sensitive		Vibration Fatigue	Other
		TASCS	TT	IGSCC	MIC	PIT	CC	E-C	FAC		
311	Steam lines in reactor building		x								Panel
312	Feedwater system		x	x						x	Inconel
313	Recirculation system					x					-
314	Relief system		x								-
316	Condensation system										-
321	Shut-down cooling system			x							Panel, inconel
322	Containment vessel spray system										-
323	Core spray system			x						x	Panel, inconel
324	Pool water system										-
326	Flange cooling system			x						x	Panel
327	Auxiliary feedwater system			x						x	
331	Reactor water clean-up system			x							
332	Condensate clean-up system								x	x	CS
336	Sampling system										-
345	Controlled area floor drain system										VT
351	Boron system		x	x							Panel
354	Scram system										-
361	Containment over-pressurization protection system										VT & supports
362	Containment filtered venting system										VT & supports
412-816	Same degradation mechanisms as in OL1										

Meaning of abbreviation for degradation mechanisms used in Table 2.1.1-2 and Table 2.1.1-3:

TASCS – Thermal Striping, Cycling and Stratification,
TT – Thermal Transient,
IGSCC – Intergranular Stress Corrosion Cracking,
MIC – Microbiologically Influenced Corrosion,

PIT – Pitting,
CC – Crevice Corrosion,
E-C – Erosion – Cavitation,
FAC – Flow Accelerated Corrosion

There are also other owner defined programs (e.g. FAC) in place that were developed based on experiences from industry and plant specific operating. We should state how/if they are combined with the previous program and/or RI-ISI program or kept separate.

- Water chemistry: Sampling and analysis of process water. Inspections targeted accordingly.
- Erosion: FAC program is kept separate from RI-ISI. FAC ferritic pipelines are chosen according to plant specific operating experience. Chemical analysis of process medium and plant operation are the basis for the yearly re-evaluation of inspection target of erosion inspections.
- Panel: The expert Panel's main purpose is to identify inspection targets where failure potential and degradation mechanism is considered insufficient.
- Coating: coating failure.
- CS: Carbon steel piping included in the Erosion-program.
- VT: Visual inspections.
- Inconel: Inconel filler metal 82/182 at welds was classified as high failure potential due to IGSCC.

Table 2.1.1-4: Olkiluoto Unit 1 - Inspection Location Selections Comparison

System ID	System	Risk class	RI-ISI	Previous Program		RI-ISI Program		Remarks
		high (H) / moderate (M)	No. of welds	No. of Inspections	Portion of system only	No. of Inspections	Portion of system only	
311	Steam lines in reactor building	H M	8 174	0 27		8 22		
312	Feedwater system	H M	94 51	18 14		27 15		
313	Recirculation system	H M	0 153	0 0		0 26		FAC
314	Relief system	H M	1 116	0 0		1 12		
316	Condensation system	H M			x		x	
321	Shut-down cooling system	H M	217 108	61 15		60 11		
322	Containment vessel spray system	H M	0 0	0 0		0 0		
323	Core spray system	H M	14 32	5 12		6 6		
324	Pool water system	H M	0 0	0 0		0 0		
326	Flange cooling system	H M	14 18	4 3		6 3		
327	Auxiliary feedwater system	H M	18 26	5 6		8 6		
331	Reactor water clean-up system	H M	0 88	0 88	x	0 55	x	
332	Condensate clean-up system				x		x	CS, FAC
336	Sampling system				-		-	
345	Controlled area floor drain system				x		x	VT

System ID	System	Risk class	RI-ISI	Previous Program		RI-ISI Program		Remarks
		high (H) / moderate (M)	No. of welds	No. of Inspections	Portion of system only	No. of Inspections	Portion of system only	
351	Boron system	H M	2 45	2 0		2 7		
354	Scram system	H M	0 0	0 0		0 0		
361	Containment overpressurization protection system				x		x	
362	Containment filtered venting system				x		x	
412	Steam reheat system				x		x	FAC ,VT, supports
413	Turbine plant main steam system				x		x	FAC, VT, supports
414	Lubrication oil system				x		x	supports
416	Control and trip oil system				x		x	supports
417	Seal and leakage steam system				-		-	
425	Generator water cooling system							
431	Condenser and vacuum system				x		x	2
434	Condenser cooling system				x		x	2
436	Make-up water system				-		x	2
441	Condensate system				x		x	2
445	Turbine plant feedwater system				x		x	2
447	Steam extraction system				x		x	FAC

System ID	System	Risk class	RI-ISI	Previous Program		RI-ISI Program		Remarks
		high (H) / moderate (M)	No. of welds	No. of Inspections	Portion of system only	No. of Inspections	Portion of system only	
541	Process measurements				2		x	FAC, VT
712	Shut-down service water system				x		x	
713	Diesel-backed normal operation service water system				x		x	
714	Non-diesel-backed normal operation service water system				x		x	
715	Sea-water cooling system for generator				-		-	
721	Shut-down secondary cooling system				x		x	water chemistry
723	Diesel-backed normal operation secondary cooling system				x		x	water chemistry
725	Containment gas cooling system				-		-	
726	Brine system for air conditioning				-		-	
727	Air cooling system				-		-	
741	Containment gas treatment system				-		x	VT
751	Diesel-backed compressed air system				-		-	
753	Non-diesel-backed compressed air system				-		-	
754	Compressed nitrogen system				-		-	

System ID	System	Risk class	RI-ISI	Previous Program		RI-ISI Program		Remarks
		high (H) / moderate (M)	No. of welds	No. of Inspections	Portion of system only	No. of Inspections	Portion of system only	
755	Containment inerting system				-		-	
763	Heating system				-		-	
861	Fire fighting water system				x		x	Erosion/ Corrosion
862	Fire sprinkling system				x		x	Erosion/ Corrosion

There are also other owner defined programs (e.g. FAC) in place that were developed based on experiences from industry and plant specific operating. We should state how/if they are combined with the previous program and/or RI-ISI program or kept separate.

- Water chemistry: Sampling and analysis of process water. Inspections targeted accordingly.
- Erosion: FAC program is kept separate from RI-ISI. FAC ferritic pipelines are chosen according to plant specific operating experience. Chemical analysis of process medium and plant operation are the basis for the yearly re-evaluation of inspection target of erosion inspections.
- Panel: The expert Panel's main purpose is to evaluate inspection targets where failure potential and degradation mechanism is considered insufficient.
- CS: Carbon steel piping included in the Erosion-program.
- Coating: coating failure.
- Visual inspections.

Table 2.1.1-5: Olkiluoto Unit 2 - Inspection Location Selections Comparison

System ID	System	Risk class	RI-ISI	Previous Program		RI-ISI Program		Remarks
		High (H) / moderate (M)	No. of welds	No. of Inspections	Portion of system only	No. of Inspections	Portion of system only	No. of welds
311	Steam lines in reactor building	H M	1 176	0 25		1 24		
312	Feedwater system	H M	103 40	16 12		28 12		
313	Recirculation system	H M	0 152	0 0		0 26		Erosion
314	Relief system	H M	1 68	0 0		1 7		
316	Condensation system	H M			x		x	
321	Shut-down cooling system	H M	150 126	26 32		44 13		
322	Containment vessel spray system	H M	0 0	0 0		0 0		
323	Core spray system	H M	58 26	52 20		19 7		
324	Pool water system	H M	0 0	0 0		0 0		
326	Flange cooling system	H M	10 16	3 1		4 1		
327	Auxiliary feedwater system	H M	10 34	2 7		4 7		
331	Reactor water clean-up system	H M	0 81	0 57	x	0 34	x	
332	Condensate clean-up system				x		x	CS, FAC
336	Sampling system				-		-	

System ID	System	Risk class	RI-ISI	Previous Program		RI-ISI Program		Remarks
		High (H) / moderate (M)	No. of welds	No. of Inspections	Portion of system only	No. of Inspections	Portion of system only	No. of welds
345	Controlled area floor drain system				x		x	VT
351	Boron system	H M	2 45	2 0		2 7		
354	Scram system	H M	0 0	0 0		0 0		
361	Containment over pressurization protection system				x		x	
362	Containment filtered venting system				x		x	
412	Steam reheat system				x		x	FAC ,VT, supports
413	Turbine plant main steam system				x		x	FAC, VT, supports
414	Lubrication oil system				x		x	supports
416	Control and trip oil system				x		x	supports
417	Seal and leakage steam system				-		-	
425	Generator water cooling system							
431	Condenser and vacuum system				x		x	2
434	Condenser cooling system				x		x	2
436	Make-up water system				-		x	2
441	Condensate system				x		x	2
445	Turbine plant feedwater system				x		x	2

System ID	System	Risk class	RI-ISI	Previous Program		RI-ISI Program		Remarks
		High (H) / moderate (M)	No. of welds	No. of Inspections	Portion of system only	No. of Inspections	Portion of system only	No. of welds
447	Steam extraction system				x		x	FAC
541	Process measurements				2		x	VT/RT added
712	Shut-down service water system				x		x	
713	Diesel-backed normal operation service water system				x		x	
714	Non-diesel-backed normal operation service water system				x		x	
715	Sea-water cooling system for generator				-		-	
721	Shut-down secondary cooling system				x		x	water chemistry
723	Diesel-backed normal operation secondary cooling system				x		x	water chemistry
725	Containment gas cooling system				-		-	
726	Brine system for air conditioning				-		-	
727	Air cooling system				-		-	
741	Containment gas treatment system				-		x	VT
751	Diesel-backed compressed air system				-		-	
753	Non-diesel-backed compressed air system				-		-	
754	Compressed nitrogen system				-		-	

System ID	System	Risk class	RI-ISI	Previous Program		RI-ISI Program		Remarks
		High (H) / moderate (M)	No. of welds	No. of Inspections	Portion of system only	No. of Inspections	Portion of system only	No. of welds
755	Containment inerting system				-		-	
763	Heating system				-		-	
861	Fire fighting water system				x		x	Erosion/ Corrosion
862	Fire sprinkling system				x		x	Erosion/ Corrosion

There are also other owner defined programs (e.g. FAC) in place that were developed based on experiences from industry and plant specific operating. We should state how/if they are combined with the previous program and/or RI-ISI program or kept separate.

- Water chemistry: Sampling and analysis of process water. Inspections targeted accordingly.
- Erosion: FAC program is kept separate from RI-ISI. FAC ferritic pipelines are chosen according to plant specific operating experience. Chemical analysis of process medium and plant operation are the basis for the yearly re-evaluation of inspection target of erosion inspections.
- Panel: The expert Panel's main purpose is to evaluate inspection targets where failure potential and degradation mechanism is considered insufficient.
- CS: Carbon steel piping included in the Erosion-program.
- Coating: coating failure.
- Visual inspections.

2.1.2 Experience in Slovenia

2.1.2.1 Krško NPP

Previous ISI Program

The Krško Nuclear Power Plant (Krško NPP) is a Westinghouse design two loop pressurized water reactor type plant. It follows the USA regulation and American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Section XI Code, Inspection Program B [6]. Currently it runs ISI program for the fourth 10-year inspection interval. ISI programs for the first three intervals covered safety class 1, 2 and 3 systems and components and were based on deterministic approach with several additional augmented inspections which result from leak-before-break analyses of auxiliary lines greater than 6 inches, postulated rupture locations in fluid system piping inside and outside containment etc. The third ISI program was based on the requirements of ASME Sec. XI, edition 1995/A96 as amended by paragraph (b) of the Code of Federal Regulations, 10 CFR Part 50.55a.

RI-ISI Results

The fourth inspection interval started in June 2012 and it will end in June 2022. For the fourth inspection interval, the Krško NPP has decided to use a RI process to develop RI-ISI program for Class 1 and 2 piping which corresponds to ASME Sec. XI Examination Categories B-F, B-J, C-F-1 and C-F-2. Class 3 components were not included into the RI-ISI Program. The basis for the chosen RI process is described in Electric Power Research Institute (EPRI) Topical Report (TR) 112657, Rev. B-A "Revised Risk-Informed In-service Evaluation Procedure" [5]. The RI-ISI application was developed in accordance with ASME Code Case N-578-1 "Risk-Informed Requirements for Class 1, 2 and 3 Piping" and ASME Sec. XI, Non-mandatory Appendix R, "Risk-Informed Inspection Requirements for Piping, Method B" [4].

The Krško NPP's RI-ISI Program included the following systems:

1. Reactor coolant (RC)
2. Chemical and volume control (CS)
3. Residual heat removal (RH)
4. Safety injection (SI)
5. Auxiliary feedwater (AF)
6. Steam generator blowdown (BD)
7. Containment spray (CI)
8. Feedwater (FW)
9. Main steam (MS)

The Krško NPP's RI-ISI process went through the following steps:

Scope Definition: The existing ISI Program, instrumentation diagrams and additional plant information for all the above listed piping were bases to determine the Class 1 and 2 piping system boundaries. In the next step, the consequence piping segments were defined as continuous piping runs whose failure would result in the same consequence.

Consequence Evaluation: The consequence(s) of pressure boundary failures were evaluated and ranked based on their impact on core damage and containment performance (isolation, bypass and large early release). The impact on these measures due to both direct and indirect effects was considered using the guidance provided in EPRI TR-112657.

Failure Potential Assessment: Failure potential estimates were generated utilizing industry failure history, plant specific failure history and other relevant information. These failure estimates were determined using the guidance provided in EPRI TR-112657. Analyses showed that the following degradation mechanisms were potentially operative for the Krško NPP: thermal fatigue, intergranular stress corrosion cracking and primary water stress corrosion cracking. To supplement the degradation mechanism assessment, Krško NPP staff performed a service history and susceptibility review. This included a review of plant and industry databases and the NPP's documents to characterize operating experience with respect to the piping pressure boundary degradation. The review also included previous water hammer events as well as repair activities during components' manufacturing and erection of the plant systems.

Risk Characterization: In the preceding steps, each run of piping within the scope of the program was evaluated to determine its impact on core damage and containment performance (isolation, bypass and large early release) as well as its potential for failure. Given the results of these steps, risk groups are then defined as welds within a single system potentially susceptible to the same degradation mechanism and whose failure would result in the same consequence. Risk groups are then ranked based upon their risk significance as defined in EPRI TR-112657.

Element and NDE Selection: In general, EPRI TR-112657 requires that 25% of the locations in the high risk region and 10% of the locations in the medium risk region be selected for inspection using appropriate NDE methods tailored to the applicable degradation mechanism. In addition, EPRI TR-112657 requires, if the percentage of Class 1 piping locations selected for examination falls substantially below 10%, then the basis for selection needs to be investigated. As depicted below, a 10% sampling of the Class 1 elements has been achieved. No credit was taken for any Flow Accelerated Corrosion (FAC) or other existing augmented inspection program (e.g. high energy line break) locations in meeting the sampling percentage requirements. The following table summarizes the element selection for the Krško NPP:

Table 2.1.2-1: Krsko NPP Element Selection

Krsko NPP Element Selection								
Class 1 Piping Welds(1)			**Class 2 Piping Welds(2)**			**All Piping Welds(3)**		
Population	Selected	%	Population	Selected	%	Population	Selected	%
473	57	12.1	1551	62	4.1	1984	119	6

Notes:

1. Includes all Category B-F and B-J locations (welds).
2. Includes all Category C-F-1 and C-F-2 locations (welds).
3. All in-scope piping components, regardless of risk classification, will continue to receive Code required pressure testing, as part of the current ASME Section XI program. VT-2 visual examinations are scheduled in accordance with the station's pressure test program that remains unaffected by the RI-ISI program.

Very important consideration in the element selection process were inspectability, distribution of inspections among systems and segments, plant-specific inspection results and radiation exposure.

Lessons Learned

Comparing the old Krško NPP's ISI Program for Examination Categories B-F, B-J, C-F-1 and C-F-2 and new RI-ISI Program, the following conclusions can be drawn:

- Looking at the total number of locations (welds) to be examined, there is a significant reduction. This reduction is obtained mainly because of elimination of all surface examinations - magnetic particle (MT) and penetrant (PT) for ASME Sec. XI Examination Categories B-F, B-J, C-F-1 and C-F-2. The total number of eliminated locations is 235. Additionally, the total number of ultrasonic (UT) examinations dropped from 152 to 131. In case of UT examinations it shall be mentioned that in the total number of 131 locations, 119 locations are the result of RI approach and the rest are coming from augmented programs like leak-before-break analysis, postulated rupture locations in fluid system piping inside and outside containment and Code Case N-770-1 examinations. All these items were also augmented in the old ISI Program. As far as the number of visual (VT) examinations is concerned, this number remains the same.
- Comparing the UT examinations, the most of items selected for the RI-ISI program were in the old ISI program, and only a few items have been changed in RI-ISI program. This confirms that also the old ISI program addressed positions with very high importance for safety. The RI-ISI program additionally added some very important welds what resulted in improvement of plant safety.
- Krško NPP was running the RI-ISI program only in the last two outages and is still collecting experience. In terms of collective radiation dose it can be said that the dose as a result of ISI activities related to the RI-ISI program is reduced for about 30%, total working hours for about 20% and number of manpower for 20% (mainly because of reduction of scaffolding, insulation dismantling and surface examinations).
- Slovenian Nuclear Safety Authority is involved in the RI-ISI process through its continuous presence during the development of the Krško NPP's RI-ISI program as well as implementation during outages and because it is "living program" the regulatory body will also be involved in future improvements and frequent adjustments as directed by NRC Bulletins or generic letter requirements or by industry and plant specific feedback.
- The Krško NPP's RI-ISI program gathered number of domestic and foreign experts who with different expertise contributed to the RI-ISI program. When finalized, it was evaluated by independent domestic and foreign eminent and competent organizations. The results of these evaluations were very positive statements which supported the Krško NPP's submission of the RI-ISI program to the Slovenian Nuclear Safety Authority for approval.

2.1.3 Experience in Spain

In Spain the RI-ISI methodology was initially defined in the CSN-UNESA RI-ISI-02 Guide [11] developed jointly by Nuclear Safety Council (CSN) and UNESA, an organization of electric utilities operators of Nuclear Power Plants. This guide was published in 2000, being of voluntary application.

In 2007, the CSN published the Safety Guide 1.17 [12], which incorporated learned lessons from the initial implementation of the guidelines at Ascó, Almaraz and Cofrentes NPPs. Its implementation, also voluntary, covers only piping inspection, including the possibility of partial or full scope, and requires to evaluate all degradation mechanisms that may affect the piping within the scope. The guide also includes the requirement for RI-ISI analysis updates at least in coincidence with each 10 year inspection interval, and whenever there are significant changes in plant configuration, the PSA, the emergence of not covered degradation mechanisms and when it is required by the CSN.

The RI-ISI analyses are based on dividing the piping system into "segments", which consists of portions of a pipeline with the same consequences in case of failure and affected by the same degradation mechanisms. To incorporate in a quantitatively way the shutdown PSA, a redefinition of those segments was carried out in an interactive process that was validated by the Expert Panel. The degradation mechanisms were identified for power operation and reviewed by a technical panel composed by experts in materials, ISI, and engineering. This panel also re-evaluated the degradation mechanisms for shutdown conditions in each of the plant operational states (POS), ruling out the non-applicable. Depending upon whether the plant was a BWR or PWR, the mechanisms that have been considered for power operation consists of fatigue, thermal stratification, erosion-corrosion (FAC), IGSCC, vibration and primary water SCC (PWSCC).

The risk significance of piping segments was performed by providing the expert panel information about the segments categorization based on their relevance related to the CDF and LERF, this last one only for power conditions, obtained from Risk Reduction Worth (RRW) measure of importance, supplemented by numerous sensitivity analysis, Risk Achievement Worth (RAW) values and deterministic defence-in-depth principles. The CDF of each segment is based on the Core Damage Conditional Probability (CCPD) according to Power PSA and also on the POSs, whereas the Conditional Large Early Release Probability (CLERP) was based on the Power PSA.

2.1.3.1 Ascó Units 1 and 2

Both Ascó Units are PWRs (Westinghouse) consisting of three reactor coolant loops, in which the pre-service and the ISI of nuclear piping had been carried out since start-up according to a program based on the requirements of the applicable Edition of Section XI of the ASME Code [6]. In addition, there were in place several augmented inspection programs. Specifically for Class 1 piping there was in place an inspection program for thermal fatigue according to US NRC Bulletin 88-08 [13].

<u>RI-ISI Results</u>

The Ascó Unit 1 Class 1 piping RI-ISI application was a pilot study undertaken in Spain in the frame of the development of the CSN-UNESA RI-ISI-02 Guide [11]. This pilot study was later submitted to the CSN and approved with slight modifications in September 2001. Application for Ascó Unit 2 was approved in 2005.

The scope of the RI-ISI program focused on the pipes that make up the reactor coolant pressure boundary, including all Class 1 piping, whatever their size, without excluding any of the systems having Class 1 piping. Table 2.1.3-1 provides a list of the systems included in the RI-ISI program scope for Units 1 and 2. The systems included in the RI-ISI were the same for both units. Unit 1 application was updated in 2006, because of entering in a new inspection interval and incorporating the plant modifications implemented since the approval of the original application.

Table 2.1.3-1: Ascó NPP Units 1 & 2 – List of Systems Evaluated

System	**Class**				
	1	**2**	**3**	**4**	**Non safety**
(10) Reactor coolant system (RCS)	✓			N/A	
(11) Chemical & volume control system (CVCS)	✓			N/A	
(14) Residual heat removal (RHR)	✓			N/A	
(15) Safety injection system (SIS)	✓			N/A	

The applications were based on the at power PSA Levels 1 and 2, and the shutdown PSA Level 1.

In order to determine the failure likelihood of the piping being evaluated, an assessment of whether different types of degradation could be operative in a particular system, or portion of a system, is conducted. Table 2.1.3-2 presents the results of this effort for Units 1 and 2. Thermal fatigue (TT) due to normal operation transients was postulated for all segments. In addition, other mechanisms such as thermal stratification, PWSCC in Ni-based alloy welds, SCC due to borated stagnant water and mechanical vibrations were also considered for the susceptible segments. Table 2.1.3-3 and Table 2.1.3-4 present a comparison between the previous deterministic ISI program and the RI-ISI program for Units 1 and 2, respectively.

Lessons Learned

A significant reduction in the number of inspections was achieved when changing from the deterministic ISI program to RI-ISI programs, while at the same time, slight reductions of risk or risk neutrality were obtained in comparison with the previous ISI programs, both in terms of the Core Damage Frequency (CDF) and of the Large Early Release Frequency (LERF). The success of this application facilitated the subsequent implementation of the RI-ISI methodology for piping in other Spanish plants.

Table 2.1.3-2: Ascó NPP Units 1&2 - Degradation Mechanisms identified as Potentially Operative

System ID	System	Thermal Fatigue		Stress Corrosion Cracking				Localized Corrosion			Flow Sensitive		Vibration Fatigue
		TASCS	TT	IGSCC	TGSCC	ECSCC	PWSCC	MIC	PIT	CC	E-C	FAC	
10	RCS	✓	✓	✓[1]			✓						✓
11	CVCS		✓										✓
14	RHR		✓										
15	SIS		✓										

Meaning of abbreviation for degradation mechanisms used in Table 2.1.3-2:

TASCS – Thermal Striping, Cycling and Stratification,
TT – Thermal Transient,
IGSCC – Intergranular Stress Corrosion Cracking,
TGSCC – Transgranular Stress Corrosion Cracking,
ECSCC -- External Chloride Stress Corrosion Cracking,
PWSCC -- Primary Water Stress Corrosion Cracking,
MIC – Microbiologically Influenced Corrosion,
PIT – Pitting,
CC – Crevice Corrosion,
E-C – Erosion – Cavitation,
FAC – Flow Accelerated Corrosion

[1] IGSCC in certain segments with borated water.

Table 2.1.3-3: Ascó Unit 1- Inspection Location Selections Comparison

System ID	System	Previous Program			RI-ISI Program		
		No. of Welds or Segments	No. of Inspections	Other[1]	No. of Welds or Segments	No. of Inspections	Other[Erreur ! Signet non défini.]
10	RCS	311	101	5	386	70	
11	CVCS	123	1		173	11	
14	RHR	143	34	5	143	22	
15	SIS	211	22	2	211	9	
Total		788	158	12	913	112	

Table 2.1.3-4: Ascó Unit 2- Inspection Location Selections Comparison

System ID	System	Previous Program			RI-ISI Program		
		No. of Welds or Segments	No. of Inspections	Other[Erreur ! Signet non défini.]	No. of Welds or Segments	No. of Inspections	Other[Erreur ! Signet non défini.]
10	RCS	307	101	5	382	56	
11	CVCS	126	1		176	6	
14	RHR	137	34	5	137	11	
15	SIS	207	22	2	207	7	
Total		777	158	12	902	80	

[1] The previous high cycle thermal fatigue augmented program according to US NRC Bulletin 88-08 is combined in the RI-ISI program.

2.1.3.2 Almaraz Units 1 and 2

Previous ISI Program

Both Almaraz units are PWRs (Westinghouse) consisting of three reactor coolant loops, in which the pre-service and the in-service inspections of nuclear piping had been carried out since start-up according to a program based on the requirements of the applicable Edition of Section XI of the ASME Code [6]. In addition, there were in place several augmented inspection programs. Specifically for Class 1 piping there was in place an inspection program for thermal fatigue according to US NRC Bulletin 88-08 [13].

RI-ISI Results

In 2003, a RI-ISI program was developed for the Class 1 piping of both Units. The scope of the RI-ISI program focused on the pipes that make up the reactor coolant pressure boundary, including all Class 1 piping, whatever their size, without excluding any of the systems having Class 1 piping. Table 2.1.3-5 provides a list of the systems included in the RI-ISI program scope for Units 1 and 2. The systems included in the RI-ISI program are the same for both units.

Table 2.1.3-5: Almaraz NPP Units 1&2 – List of Systems Evaluated

System	Class				
	1	2	3	4	Non safety
RCS	✓			N/A	
RHR	✓			N/A	
SIS	✓			N/A	
CVCS	✓			N/A	

RI-ISI analyses from Unit 2 were updated in 2005, because of entering in a new inspection interval, and, with this change, the commitments achieved with CSN in the license of the Unit 1 application were incorporated. In 2011 the RI-ISI analyses of both units were updated to quantitatively incorporate the shutdown PSA, the effect of power uprate, the mitigation by weld overlay of Alloy 82/182 welds between the pressurizer nozzles and safe-ends and the austenitic welds between the safe-ends and connected piping, the changes in plant configuration, the new operating experience of inspection, the updates of at power PSA Levels 1 and 2 as well as the incorporation of new requirements of CSN Safety Guide 1.17 [12].

In order to determine the failure likelihood of the piping being evaluated, an assessment of whether different types of degradation could be operative in a particular system, or portion of a system, is conducted. Table 2.1.3-6 presents the results of this effort for Units 1 and 2. Thermal fatigue (TT) due to normal operation transients was postulated for all segments. In addition, other mechanisms such as thermal stratification, PWSCC in Ni-based alloy welds and mechanical vibrations were also considered for the susceptible segments.

The RI-ISI program (i.e. inspection locations) was established by a group known as the technical panel ISI and they used the RI-ISI analyses (e.g. failure potential, consequence of failure, risk ranking) to determine the number and locations for inspections. These sets of inspections are validated by the expert panel, which in particular incorporates the principles of defence-in-depth into the final inspection population.

Table 2.1.3-7 and Table 2.1.3-8 present a comparison between the previous deterministic ISI program and the RI-ISI program for Units 1 and 2, respectively.

Important insights from these results include:

- The resulting CDF value in the categorization process encompassing power + shutdown was 6,127 E-06/year, whereas CDF due to shutdown only was 1,13 E-08/year.
- The risk increase associated to the RI-ISI program, compared to ISI program, in terms of CDF is -3.7 E-08, and in LERF terms is -3.6 is E-11 (only power), which meet the requirements of the CSN Safety Guide 1.14 [14].
- The RI-ISI program causes a 78% reduction in the locations to inspect allowing reducing the doses received by the inspection personnel and to focus the resources in the most risk significant locations and mechanisms.
- In terms of potential to failure, the mechanism with the highest potential of failure has resulted to be the PWSCC, with a range of 1.2 E-05 to 1.0 E-02 in 40 years, two orders of magnitude more than considering only the fatigue degradation mechanism.

<u>Lessons Learned</u>

From the analyses carried out it is necessary to emphasize the next points:

- RI-ISI analyses contribute to improve the knowledge about the safety importance of different degradation mechanisms allowing to focus efforts on those contributing more to the risk.
- The addition of the PSA analysis for shutdown in a quantitative way has shown that the risk associated with piping breaks is dominated by power operating conditions. The quantitative analysis for shutdown did not add any additional aspects that were not already considered in the earlier qualitative analysis.
- Most of the effort of the current review (i.e. 2011) of the RI-ISI analysis was due to the inclusion in a quantitatively way of the shutdown PSA. Therefore it is not considered efficient to include it in future revisions, unless new degradation mechanisms modify the current conditions.
- The inclusion in the 2011 update of the PWSCC degradation mechanism in a quantitative way showed that the probability of failure of segments susceptible to this mechanism increased two orders of magnitude and resulted in very high RRW values. This made it necessary to carry out a sensitivity analysis excluding this mechanism in order to provide information to the expert panel to identify other risk significant segments, which were not susceptible to PWSCC.
- The successive reviews of the analyses have shown the role of the expert panel to provide stability to the analysis in case of changes in the PSA and in the estimation of failure potential due to changes in models or the incorporation of operating experience and new mechanisms of degradation.

Table 2.1.3-6: Almaraz NPP Units 1&2 - Degradation Mechanisms identified as Potentially Operative

System ID	System	Thermal Fatigue		Stress Corrosion Cracking				Localized Corrosion			Flow Sensitive		Vibratory Fatigue
		TASCS	TT	IGSCC	TGSCC	ECSCC	PWSCC	MIC	PIT	CC	E-C	FAC	
RCS	Reactor coolant system	✓	✓				✓						✓
RHR	Residual heat removal		✓										
SIS	Safety injection system		✓										
CVCS	Chemical & volume control system		✓										✓

Meaning of abbreviation for degradation mechanisms used in Table 2.1.3-6**Erreur ! Source du renvoi introuvable.**:

TASCS – Thermal Striping, Cycling and Stratification,
TT – Thermal Transient,
IGSCC – Intergranular Stress Corrosion Cracking,
TGSCC – Transgranular Stress Corrosion Cracking,
ECSCC -- External Chloride Stress Corrosion Cracking,
PWSCC -- Primary Water Stress Corrosion Cracking,
MIC – Microbiologically Influenced Corrosion,
PIT – Pitting,
CC – Crevice Corrosion,
E-C – Erosion – Cavitation,
FAC – Flow Accelerated Corrosion

Table 2.1.3-7: Almaraz Unit 1- Inspection Location Selections Comparison

System ID	System	Previous Program			RI-ISI Program		
		No. of Welds or Segments	No. of Inspections	Other[Erreur ! Signet non défini.]	No. of Welds or Segments	No. of Inspections	Other[Erreur ! Signet non défini.]
RCS	Reactor Coolant	286	122	9	339	74	
RHR	Residual Heat Removal	16	4		16	-	
SIS	Safety Injection	176	83		176	3	
CVCS	Chemical and Volume Control	109	57		167	6	
Total		587	275	9	698	83	

Table 2.1.3-8: Almaraz Unit 2- Inspection Location Selections Comparison

System ID	System	Previous Program			RI-ISI Program		
		No. of Welds or Segments	No. of Inspections	Other[1]	No. of Welds or Segments	No. of Inspections	Other[Erreur ! Signet non défini.]
RCS	Reactor Coolant	298	136	9	352	75	
RHR	Residual Heat Removal	22	4		22	-	
SIS	Safety Injection	180	74		180	4	
CVCS	Chemical and Volume Control	111	60		169	7	
Total		611	274	9	723	86	

[1] The previous high cycle thermal fatigue augmented program according to US NRC Bulletin 88-08 is combined in the RI-ISI program.

2.1.3.3 Cofrentes NPP

Previous ISI Program

Cofrentes NPP is a single unit General Electric BWR/6 Mark III reactor, in which the pre- and in-service piping nuclear class inspections had been carried out since start-up according to the requirements of the applicable edition of Section XI of the ASME Code [6]. In addition, there were in place several augmented inspection programs. Specifically, for austenitic stainless steel piping there was in place an inspection program for IGSCC in the Category A and B welds according to NUREG-0313 Rev. 2 [15] existing at the plant, as well as a FAC or Erosion-Corrosion (E-C) program for low alloy and carbon steel piping carrying feedwater or steam.

RI-ISI Results

In 2007 the RI-ISI Program was incorporated for Class 1 and 2 piping, which emerged from the analysis started in 2004 and positively accepted by CSN in 2007, after a long and interactive evaluation program. Earlier in 2005 the RI-ISI Program for Class 1 piping was implemented in an interim way.

The RI-ISI program scope has been extended to all Classes 1 and 2 piping, with the exception of the pipes who carry air and who are under ¾" as long as their breakage does not affect to safety significant instrumentation, although all welds have been counted as belonging to the main piping. Table 2.1.3-9 provides a listing of the systems included within the RI-ISI program scope. Except as noted above, all Class 1 and 2 Systems have been included in the scope listed in Table 2.1.3-9 without exception, whether or not inspected for ASME XI.

Table 2.1.3-9: Cofrentes – List of Systems Evaluated

System ID	**System**	**Class**				
		1	**2**	**3**	**4**	**Non safety**
B21	Nuclear Boiler System	✓	✓		N/A	
B33	Reactor Recirculation Flow Control System.	✓	✓		N/A	
C11	CRD Hydraulic Control System	✓	✓		N/A	
C41	Standby Liquid Control System	✓	✓		N/A	
C61	Remote Shutdown System	✓	✓		N/A	
E12	RHR System (LPCI, RHR)	✓	✓		N/A	
E21	Core Spray System (LPCS)	✓	✓		N/A	
E22	High Pressure Core Spray System	✓	✓		N/A	
E31	Leak Detection System		✓		N/A	
E32	Main Steam Isolation Valve (MISIV) Leakage Control System	✓	✓		N/A	
E51	Reactor Core Isolation Cooling System (RCIC)	✓	✓		N/A	
G17	Radwaste System		✓		N/A	
G33	Reactor Water Clean-Up System	✓	✓		N/A	
G36	Filter/Demineralizer System		✓		N/A	
G41	Fuel Pool Cooling & Clean-Up system		✓		N/A	
G51	Suppression Pool Clean-Up System		✓		N/A	
N11	Main Steam System & Turbine Bypass System		✓		N/A	

System ID	System	Class				
		1	2	3	4	Non safety
N21	Condensate & Feedwater System		✓		N/A	
P11	Condensate Transfer Distribution & Storage System		✓		N/A	
P12	Demineralized Water Distribution System		✓		N/A	
P42	Reactor Building Closed Water System		✓			
P44	Non-Essential Chilled Water System		✓		N/A	
P64	Fire Protection		✓		N/A	
T70	Suppression Pool Make-Up		✓		N/A	

With this scope, the RI-ISI analyses started in 2004 and were completed in 2006. The RI-ISI program received a favourable appreciation from CSN in 2007 and was implemented during the re-fuelling outage that year. In the favourable appreciation, CSN had enforced as a condition the quantitative incorporation to RI-ISI analyses of the shutdown PSA. Once concluded and approved by the CSN the shutdown PSA, it was incorporated in a quantitative way in a review of the RI-ISI analyses started in 2010. This update also incorporated changes in the plant configuration, the experience in the results of inspections and upgrades PSA of Levels 1 and 2.

In order to determine the failure likelihood of the piping being evaluated, an assessment of whether different types of degradation could be operative in a particular system, or portion of a system, is conducted. Table 2.1.3-10 presents the results of this effort. Thermal fatigue (TT) due to normal operation transients was postulated for all segments. In addition, other mechanisms such as IGSCC and FAC were also considered for the susceptible segments.

The risk significance of piping segments was determined by providing the expert panel information about the segments categorization based on their relevance related to failure potential, consequence of failure, CDF and LERF as well as applicable risk importance measure (e.g. FV, RRW, RAW), supplemented by numerous sensitivity analysis. The expert panel establishes the final categorization.

Similar, to Almaraz NPP, the RI-ISI program is established by the technical panel ISI and validated by the expert panel. In particular in those aspects related to the inclusion of principles of defence-in-depth, such as not reducing the FAC inspection locations on basis to risk insights, to include inspections for fatigue in welds close to maximum FAC rate locations, and keeping all NUREG-0313 IGSCC Category B welds in the RI-ISI program.

Table 2.1.3-11 presents a comparison between the previous deterministic ISI program and the RI-ISI program.

Important insights from these results include:

- The resulting CDF value in the categorization process, without ISI, belongs to power + shutdown and was 4.227E-06 / year, whereas CDF associated to shutdown only, without ISI, was 3.230E-08/year. The number of quantifications of the PSA were 103 with the at power PSA and 499 with the shutdown PSA.
- The greater importance in relation to safety of Class 1 piping, 87% of total CDF, versus Class 2, 13% of total CDF.

- From a degradation perspective, IGSCC contributes to 44% of total CDF, versus 29% for FAC and 27% for fatigue. The FAC mechanism in the current revision (2011) has decreased its importance in comparison to the 2006 revision, as a result of adding to the deterministic FAC program, the locations that in 2006 were identified as risk-significant and until then had not been included in the FAC inspection program.
- The 87% of the locations susceptible to fatigue that resulted significant were not covered by the Section XI ISI program, because they belonged to locations in small pipes exempt from ASME Section XI. In the same way it should be noted that 209 segments that had locations included in the ISI program, have been included in the Region 4, Low-Risk Significance and Low Potential Failure.
- The increase in risk associated to the RI-ISI program, with respect to the deterministic ISI program, in terms of CDF is 6.3 E-08, which meet the criteria of the CSN Safety Guide 1.14 [14].
- The RI-ISI program entails a 48% reduction in the locations to inspect, a little bit higher in Class 1 than in Class 2, reducing the dose received by inspection personnel, a better use of resources by focusing efforts in the more significant locations and mechanisms.

Lessons Learned

From the analyses carried out it is necessary to emphasize the following points:

- RI-ISI analyses contribute to improve the knowledge about the safety importance of different degradation mechanisms allowing to focus efforts on those contributing more to the risk.
- The addition in the PSA RI-ISI analysis in shutdown has showed that the risk for the plant associated to piping breaks is dominated by power operating conditions, not including any additional aspect not covered by the qualitative analysis for shutdown.
- 90% of the effort of the current review of the RI-ISI analysis was due to the inclusion in a quantitatively way of the shutdown PSA and therefore it is not considered effective to include it in future revisions, unless new degradation mechanisms modify the current conditions.
- It has been necessary to develop new methodological aspects to estimate the stresses in pipes under 3" in which the design was made on basis to a guide and not with a formal stress analysis.
- It has been necessary to analyse consequences for piping in the "Break Exclusion Region (BER)". outside the containment. For these pipes the indirect effects of the breaks were analysed such as flooding, isolation caused by the action of the protection system for moderate and high energy pipelines, and the whip effect for high energy pipelines. Piping in this region belongs to B21, E12, E21, E32, E51 and G33 Systems.
- Analysis has shown the importance of inclusion in the pipes of ¾" associated to instrumentation. This piping is required to VT-2 exams at nominal pressure during each refueling outage.
- The 2011 revision, of the analyses performed in 2006 has shown the importance of the Expert Panel to provide stability to the analysis due to changes that occur in the PSA and in the estimation of stresses by changes in models or the incorporation of operating experience.

- The 2006 revision showed the importance of the interactive evaluation by the CSN that was beneficial for the quality and consolidation of the application the methodology that was in the beginning of its implementation for this type of plant.

Table 2.1.3-10: Cofrentes NPP - Degradation Mechanisms identified as Potentially Operative

System ID	System	PSA	ASME XI	Thermal fatigue		Stress corrosion cracking				Localised corrosion			Flow sensitivity		Vibration fatigue
				TASCC	TT	IGSCC	TGSCC	ECSCC	PWSCC	MIC	PIT	CC	E-C	FAC	
B21	Nuclear Boiler System	yes	yes		✓									✓	
B33	Reactor Recirculation Flow Control System.	yes	yes		✓	✓									
C11	CRD Hydraulic Control System	Yes	Yes												
C41	Standby Liquid Control System	Yes	No		✓										
C61	Remote Shutdown System	No	No												
E12	RHR System (LPCI, RHR)	Yes	Yes		✓									✓	
E21	Core Spray System (LPCS)	Yes	Yes		✓										
E22	High Pressure Core Spray System	Yes	Yes		✓										
E31	Leak Detection System	No	No		✓										
E32	Main Steam Isolation Valve (MISIV) Leakage Control System	Yes	No		✓										
E51	Reactor Core Isolation Cooling System (RCIC)	Yes	Yes		✓									✓	
G17	Radwaste System	Yes	No		✓										
G33	Reactor Water Clean-Up System	Yes	Yes		✓	✓								✓	
G36	Filter/Demineralizer System	Yes	No												
G41	Fuel Pool Cooling & Clean-Up system	Yes	No		✓										
G51	Suppression Pool Clean-Up System	yes	No												
N11	Main Steam System & Turbine Bypass System	No	Yes		✓									✓	
N21	Condensate & Feedwater System	No	Yes											✓	
P11	Condensate Transfer Distribution & Storage System	Yes	No		✓										
P12	Demineralized Water Distribution System	Yes	No												

System ID	System	PSA	ASME XI	Thermal fatigue		Stress corrosion cracking				Localised corrosion			Flow sensitivity		Vibration fatigue
				TASCC	TT	IGSCC	TGSCC	ECSCC	PWSCC	MIC	PIT	CC	E-C	FAC	
P42	Reactor Building Closed Water System	Yes	No		✓										
P44	Non-Essential Chilled Water System	Yes	No		✓										
P64	Fire Protection	Yes	No												
T70	Suppression Pool Make-Up	Yes	No		✓										

Meaning of abbreviation for degradation mechanisms used in Table 2.1.3-10:

TASCS – Thermal Striping, Cycling and Stratification,
TT – Thermal Transient,
IGSCC – Intergranular Stress Corrosion Cracking,
TGSCC – Transgranular Stress Corrosion Cracking,
ECSCC -- External Chloride Stress Corrosion Cracking,
PWSCC -- Primary Water Stress Corrosion Cracking,
MIC – Microbiologically Influenced Corrosion,
PIT – Pitting,
CC – Crevice Corrosion,
E-C – Erosion – Cavitation,
FAC – Flow Accelerated Corrosion

Table 2.1.3-11: Cofrentes NPP - Inspection Location Selections Comparison

System ID	System	Previous Program			RI-ISI Program		
		No. of Welds or Segments	No. of Inspections	Other	No. of Welds or Segments	No. of Inspections	Other
B21	Nuclear Boiler System		131	FAC	1539	122	FAC
B33	Reactor Recirculation Flow Control System.		68		237	54	
C11	CRD Hydraulic Control System				6	0	
C41	Standby Liquid Control System				268	0	
C61	Remote Shutdown System				3	0	
E12	RHR System (LPCI, RHR)		103	FAC		7	FAC
E21	Core Spray System (LPCS)		12		316	2	
E22	High Pressure Core Spray System		14		400	3	
E31	Leak Detection System				28	0	
E32	Main Steam Isolation Valve (MISIV) Leakage Control System				759	0	
E51	Reactor Core Isolation Cooling System (RCIC)		13	FAC	833	10	FAC
G17	Radwaste System				30	0	
G33	Reactor Water Clean-Up System		17	FAC	149	23	FAC
G36	Filter/Demineralizer System				8	0	
G41	Fuel Pool Cooling & Clean-Up system				27	0	
G51	Suppression Pool Clean-Up System				37	1	
N11	Main Steam System & Turbine Bypass System		1		12		
N21	Condensate & Feedwater System		1		7	1	
P11	Condensate Transfer Distribution & Storage System				47	0	
P12	Demineralized Water Distribution System				22	0	
P42	Reactor Building Closed Water System				79	0	
P44	Non-Essential Chilled Water System				31	O	
P64	Fire Protection				17	0	
T70	Suppression Pool Make-Up				20	0	
Total		2840	360		6848	223	

2.1.4 Experience in Sweden

Ringhals NPP consists of four units, one boiling water reactor, BWR (R1) and three pressurized water reactors, PWR (R2, R3 and R4). At this time, R1 remains with the original ISI program based upon the Swedish regulations SKIFS 1994:1 [16], while the PWR units have transitioned to a RI-ISI approach. The SKIFS 1994.1 approach in the Swedish regulations is based on assessing the probability of cracking or other degradation (Damage Index) and what consequences (Consequence Index) this may have. This is a somewhat qualitative risk ranking approach.

The damage index is a qualitative measure of the probability that cracking or other degradation occurs in a specific component. The aim is to cover all relevant degradation, as no augmented programmes exist according to Swedish requirements. The consequence index is a qualitative measure of the probability of such cracking or other degradation which will result in core damage, damage of the reactor containment, release of radioactive material or other damages. The consequence index is determined by: pipe position relative the core and valves that close automatically in the event of a break; pipe dimensions; and system and thermal technical margins. Inspection groups are determined on the basis of these indexes as shown in Table 2.1.4-1.

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

For each of the units, there is also an owner defined FAC program in place that was developed due to all the years of experience that have been gained from Ringhals NPP.

With respect to the RI-ISI methodology used at Ringhals NPP, the PWROG-SE methodology is an adaptation of the original PWROG methodology [17] to the Swedish regulatory environment. The approach basically follows the PWROG (original) methodology in segmentation, failure probability and consequence analyses initial risk ranking and change in risk calculation, but differs in the structural element selection phases. An expert panel is used to verify the initial risk ranking.

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

In the phase of structural element selection, the inspection sites are classified in the three inspection groups A, B and C according to the SSMFS 2008:13 [18] (see the description above about SKIFS 1994:1). The sample inspection procedure based on the so called Perdue model, as used in the original PWROG methodology is not used in the Swedish application. Instead in inspection group A, 100% of the susceptible locations are to be inspected and in inspection group B, at least 10% of the structural elements should be inspected. Segments that end up in inspection group C will be treated in an owner defined programme. This is in accordance with the Swedish regulations SSMFS 2008:13.

Table 2.1.4-1: Qualitative risk matrix for ranking of components according to SKIFS1994:1.

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

Inspection Group A = High Risk
Inspection Group B = Medium Risk
Inspection Group C = Low Risk

2.1.4.1 Ringhals Unit 2

Previous ISI Program

As discussed above, the previous ISI program for R2 was a full scope program and based on the Swedish regulations SKIFS 1994:1. The approach in the Swedish regulations is based on assessing the probability of cracking or other degradation (Damage Index) and what consequences (Consequence Index) this may have as previously discussed.

RI-ISI Results

The scope of the RI-ISI application is deemed a full scope application. That is, both safety related and non-safety related systems were evaluated. From a practical perspective, a number of plant systems were qualitatively screened out from the RI-ISI application due to the low impact of their failure on plant operations. This consisted on forty-four systems which were excluded and not treated in the final scope. Examples of these systems include the fresh water system including fresh water reservoir, the condensate polishing system, the sampling system, the liquid waste processing system, etc.

The following systems, or portions of systems, were evaluated in the risk-informed ISI program:

- (141) Containment Isolation system
- (313, 314, 315) Reactor Coolant system
- (321) Residual Heat Removal system
- (322) Containment Spray system
- (323) Safety Injection system
- (324) Spent Fuel Pit Cooling system
- (334) Chemical and Volume Control system
- (411) Main Steam system
- (414) Condensate system
- (415) Main Feedwater system
- (416) Auxiliary Feedwater system
- (417) Steam Generator Blowdown system
- (443) Circulating Water system
- (711) Component Cooling system
- (715) Salt Water system
- (735) Refueling Water system

- (751) Instrument Air (was excluded by expert panel)

Portions of the following systems are included in the program scope in order to consider the potential indirect (spatial) effects and flooding issues:

- (197) Drinking Water system
- (761) Service Water system
- (762) Fire Protection system
- (766) Auxiliary Steam system

In order to determine the failure likelihood of the piping being evaluated, an assessment of whether different types of degradation could be operative in a particular system, or portion of a system, is conducted. Table 2.1.4-2 presents the degradation mechanisms for R2. It should be noted that for some systems, no degradation mechanisms were identified and a default of thermal fatigue (TT) was used in the failure probability calculation

In Table 2.1.4-3, the classification results (i.e. the number of HSS (High Safety Significant), MSS (Medium Safety Significant) and LSS (Low Safety Significant) segments) for the base case are presented.

In Table 2.1.4-4, the classification results for the case with risk outliers excluded are presented. As the table shows it is important to investigate if risk outliers are in present in the scope of the RI-ISI application because they could have a big impact at the outcome of the analysis.

One example of a risk outlier is vibratory fatigue. As NDE will not reduce the risk for segments with vibration fatigue, segments with this damage mechanism and a failure probability $\geq 10^{-5}$ were identified and evaluated but then removed from the final risk evaluation. A special mitigation program using other techniques than NDE was then implemented to reduce the risks for these segments.

One of the final steps in the R2 RI-ISI application is the review and approval of the program by the plant expert panel (EP) which has a major contribution to the final RI-ISI program. The biggest effort is put into reviewing MSS segments regarding their final categorization and it is not uncommon that the EP re-defines several MSS segments as HSS due to deterministic insights, defence-in-depth etc. For the R2 RI-ISI application, the final number of HSS segments after that the EP review and approval was 1994.

Table 2.1.4-5 present a comparison between the previous ISI program (SKIFS 1994:1) and the new RI-ISI program. As can be seen from this table there were increases in inspections in some systems (e.g. residual heat removal, chemical volume and control, auxiliary feedwater, salt water) and reductions in inspections for other systems (e.g. reactor coolant, main steam, main feedwater).

<u>Lessons Learned</u>

Safety benefits (e.g. inspection of high risk sites, better inspection (e.g. expanded volumes))

One outcome of shifting from the old ISI program to the RI-ISI program is that inspection locations are moved from systems close to the reactor to system in the turbine building and the result of this is that the inspection staff gets a much lower radiation dose. The dose reduction has almost reached 50% compared to the former ISI program regardless of an increase of inspection locations.

The new RI-ISI program is more efficient compared to the previous program, because the inspection is at locations there we could see the highest risk contribution to the total risk at the plant.

Insights (impact of other inspection program, influence of plant-specific design or operating experience, e.g. failure history)

Historically RAB had a separate FAC program but when implementing the RI-ISI program this program was improved through the methodology and is now a part of RAB RI-ISI program. To perform this kind of big change in methodology and perform all different steps the knowledge of the plant is taken to a new level. That will impact the ordinary work in a positive way.

Regulatory interactions

The Swedish Regulatory document SSMFS 2008:13 allows you to use different methods when developing ISI programs. The relative risk can be estimated using quantitative or qualitative methods, or combination thereof. Independent of the method, or combinations of them, used the following should be considered:

- both direct and indirect consequences by a break or leakage in case of a formation of cracks, or by other degradation,
- all known damage mechanisms or degradations shall systematically be considered, taking into account existing loads and conditions in comparison with constructive design, dimensions and material characteristics of the equipment concerned.

If quantitative methods are used the analysis methods should be validated and the basic data being quality checked. Furthermore should the capability of the models and limitations be documented concerning the capability to an adequate detailed way in order to describe the damage development and damage consequences. Additionally the effect of insecurity in the models and in the basic data should be studied by using sensitivity analysis.

If you were to do this again from scratch, what would you do differently

If RAB should do this again one thing to mention is that you should have a larger working group. The group at RAB was quite small so we barely reach some form of critical mass to perform this kind of work. It is also very important that you have a larger working group to maintain the knowledge of the methodology when to performing updates and keeping the ISI program as a living program.

You should also think twice of what methodology that suits you. In our case the PWROG suits us because Ringhals NPP former program was quite close to the EPRI methodology except that we didn’t use any PSA analysis when to evaluating the consequences.

When RAB performed the job switching to a RI-ISI program the PWROG methodology had still some potentials for improvements i.e. the database that contains all the input data and results. From the beginning a lots of different work steps were performed in separate platforms and all results from these steps should be copied into the back end database. It was quite easy to make a mistake and copy data that wasn’t for instants the latest one. After a few years Westinghouse had built a new database that handle all the input data and output results, so today it is easier and safer to perform a yearly update of the ISI program.

Table 2.1.4-2: Ringhals Unit 2 - Degradation Mechanisms identified as Potentially Operative

System ID	System	Thermal Fatigue		Stress Corrosion Cracking				Localized Corrosion			Flow Sensitive		Vibratory Fatigue
		TASCS	TT	IGSCC	TGSCC	ECSCC	PWSCC	MIC	PIT	CC	E-C	FAC	
141	Containment Isolation		✓										
313, 314, 315	Reactor Coolant	✓	✓										✓
321	Residual Heat Removal	✓	✓								✓		✓
322	Containment Spray		✓										✓
323	Safety Injection		✓										
324	Spent Fuel Pit Cooling		✓										✓
334	Chemical and Volume Control		✓								✓		✓
411	Main Steam		✓									✓	
414	Condensate	✓	✓									✓	✓
415	Main Feedwater	✓	✓									✓	✓
416	Auxiliary Feedwater		✓										
417	Steam Generator Blowdown		✓										
443	Circulating Water												
711	Component Cooling		✓									✓	✓
715	Salt Water												
735	Refueling Water												
197	Drinking Water[1]												
761	Service Water[1]												
766	Auxiliary Steam[1]		✓										

Meaning of abbreviation for degradation mechanisms used in Table 2.1.4-2:

TASCS – Thermal Striping, Cycling and Stratification,
TT – Thermal Transient,
IGSCC – Intergranular Stress Corrosion Cracking,
TGSCC – Transgranular Stress Corrosion Cracking,
ECSCC -- External Chloride Stress Corrosion Cracking,
PWSCC -- Primary Water Stress Corrosion Cracking,
MIC – Microbiologically Influenced Corrosion,

[1] Portion of system only.

PIT – Pitting,
CC – Crevice Corrosion,
E-C – Erosion – Cavitation,
FAC – Flow Accelerated Corrosion

Table 2.1.4-3: Ringhals Unit 2 - Results - Base Case

System ID	System	Total no. of Segments	No. of HSS Segments	No. of MSS Segments	No. of LSS Segments
141	Containment Isolation	52	0	0	52
313	Reactor Coolant	75	2	22	51
321	Residual Heat Removal	48	0	0	48
322	Containment Spray	79	4	4	71
323	Safety Injection	115	10	4	101
324	Spent Fuel Pit Cooling	6	0	0	6
334	Chemical and Volume Control	129	12	6	111
411	Main Steam	106	0	12	94
414	Condensate	96	2	14	80
415	Main Feedwater	78	3	3	72
416	Auxiliary Feedwater	41	0	0	41
417	Steam Generator Blowdown	31	0	0	31
443	Main Cooling Water	12	0	0	12
711	Component Cooling	111	14	7	90
715	Salt Water	76	43	0	33
735	Refueling Water	1	0	0	1
751	Instrument Air, The EP excluded this system from the scope	0	0	0	0
197	Drinking Water[1]	2	0	0	2
761	Service Water[1]	4	2	0	2
766	Auxiliary Steam[1]	4	0	0	4
Total		1066	92	72	902

[1] Portion of system only.

Table 2.1.4-4: Ringhals Unit 2- Result – Outliers Removed

System ID	System	Total no. of Segments	No. of HSS Segments based on numerical value	No. of HSS Segments after EP-meeting	No. of LSS Segments
141	Containment Isolation	52	0	0	52
313	Reactor Coolant	75	14	17	58
321	Residual Heat Removal	48	0	4	44
322	Containment Spray	79	4	0	79
323	Safety Injection	115	17	18	97
324	Spent Fuel Pit Cooling	6	0	0	6
334	Chemical and Volume Control	129	4	12	117
411	Main Steam	106	0	10	96
414	Condensate	96	14	18	78
415	Main Feedwater	78	6	28	50
416	Auxiliary Feedwater	41	0	11	30
417	Steam Generator Blowdown	31	0	0	31
443	Circulating Water	12	0	0	12
711	Component Cooling	111	14	15	96
715	Salt Water	76	0	58	18
735	Refueling Water	1	0	0	1
751	Instrument Air, The EP excluded this system from the scope				
197	Drinking Water[1]	2	0	2	0
761	Service Water[1]	4	2	0	4
766	Auxiliary Steam[1]	4	1	1	3
Total		1066	**76**	194	872

[1] Portion of system only.

Table 2.1.4-5: Ringhals Unit 2- Inspection Location Selections Comparison

System ID	System	Previous Program			RI-ISI Program		
		No. of Welds or Segments	No. of Inspections	Other[1]	No. of Welds or Segments	No. of Inspections	Other[1]
141	Containment Isolation						
313	Reactor Coolant		80		160	42	
321	Residual Heat Removal		2		90	20	
322	Containment Spray						
323	Safety Injection		43		137	35	
324	Spent Fuel Pit Cooling						
334	Chemical and Volume Control		8		83	19	
411	Main Steam		54	✓	126	19	✓
414	Condensate			✓	474	31	✓
415	Main Feedwater		59		70	12	
416	Auxiliary Feedwater				113	27	
417	Steam Generator Blowdown			✓			✓
443	Circulating Water						
711	Component Cooling				164	19	
715	Salt Water					43	
735	Refueling Water						
751	Instrument Air						
197	Drinking Water[2]				2	2	
761	Service Water[2]						
766	Auxiliary Steam[2]			✓			✓
Total			246		1419	269	

[1] There are also other owner defined programs (e.g. FAC) in place that were developed based on industry and plant-specific operating experience. Today the FAC part is included in the RI-ISI program, but we also have an owner defined program for small bore piping where FAC/EC currently could exist.
[2] Portion of system only.

2.1.4.2 Ringhals Units 3 + 4

For Ringhals Units 3 & 4 the same RI-ISI methodology was used. The results are quite similar but one thing which is different between Unit R2 compared to R3 & R4 is the flooding pathways. RAB also did a small change in scope between the units because some systems at R2 didn't contribute to the CDF and LERF so it wasn't meaningful to include these systems in the analysis for R3 and R4.

The final scope of the RIVAL project R3/4 contains the following systems:

- 313 Reactor coolant system
- 321 Residual heat removal system
- 322 Containment spray system
- 323 Safety injection system
- 327 Auxiliary feedwater system
- 334 Chemical and volume control system
- 337 Blow down system
- 411 Main steam system
- 414 Condensate system
- 415 Main feedwater system
- 418 Reheating system
- 419 Bleed steam system
- 443 Main Cooling Water System
- 711 Component cooling system, reactor part
- 715 Salt water system, reactor part
- 718 Salt water system, turbine part
- 733 Demineralized water system
- 735 Refueling water storage
- 761 Service water system
- 762 Fire protection system
- 766 Auxiliary steam system

The difference in scope between R2 and R3/4 is in following systems, 141 (R2), 197 (R2), 324 (R2), 751 (R2), 418 (R3/4), 419 (R3/4), 718 (R3/4) and 733 (R3/4). Within the next update of R2 RI-ISI program the system that is present for R3/4 should be added into the scope of R2.

Table 2.1.4-6: Ringhals Unit 3/4 - Degradation Mechanisms identified as Potentially Operative to Table 2.1.4-9: Ringhals Unit 3/4- Inspection Location Selections Comparison provide the results for the R3/4 RI-ISI programs.

Table 2.1.4-6: Ringhals Unit 3/4 - Degradation Mechanisms identified as Potentially Operative

System ID	System	Thermal Fatigue		Stress Corrosion Cracking				Localized Corrosion			Flow Sensitive		Vibratory Fatigue
		TASCS	TT	IGSCC	TGSCC	ECSCC	PWSCC	MIC	PIT	CC	E-C	FAC	
313	Reactor Coolant	✓	✓										
321	Residual Heat Removal	✓	✓										✓
322	Containment Spray		✓										✓
323	Safety Injection		✓										✓
327	Auxiliary Feedwater		✓										✓
334	Chemical and Volume Control		✓										✓
337	Steam Generator Blowdown		✓									✓	
411	Main Steam		✓										✓
414	Condensate	✓	✓									✓	✓
415	Main Feedwater	✓	✓									✓	✓
418	Reheating system		✓									✓	
419	Bleed steam system		✓									✓	
443	Main Cooling Water system												
711	Component Cooling		✓									✓	✓
715	Salt Water, reactor part												
718	Salt water, turbine part												
733	Demineralized water system												
735	Refueling Water												
761	Service Water[1]												
762	Fire Protection[1]												
766	Auxiliary Steam[1]		✓										

Meaning of abbreviation for degradation mechanisms used in Table 2.1.4-6: Ringhals Unit 3/4 - Degradation Mechanisms identified as Potentially Operative

TASCS – Thermal Striping, Cycling and Stratification,
TT – Thermal Transient,
IGSCC – Intergranular Stress Corrosion Cracking,

[1] Portion of system only.

TGSCC – Transgranular Stress Corrosion Cracking,
ECSCC -- External Chloride Stress Corrosion Cracking,
PWSCC -- Primary Water Stress Corrosion Cracking,
MIC – Microbiologically Influenced Corrosion,
PIT – Pitting,
CC – Crevice Corrosion,
E-C – Erosion – Cavitation,
FAC – Flow Accelerated Corrosion

Table 2.1.4-7: Ringhals Unit 3/4 - Results - Base Case

System ID	System	Total no. of Segments	No. of HSS Segments	No. of MSS Segments	No. of LSS Segments
313	Reactor Coolant	87	7	35	45
321	Residual Heat Removal	58	12	15	31
322	Containment Spray	91	0	0	91
323	Safety Injection	139	4	26	109
327	Auxiliary Feedwater	72	2	2	68
334	Chemical and Volume Control	184	20	26	138
337	Steam Generator Blowdown	44	0	0	44
411	Main Steam	87	1	18	68
414	Condensate	146	16	4	126
415	Main Feedwater	145	0	0	145
418	Reheating system	28	0	0	28
419	Bleed steam system	24	2	4	18
443	Main Cooling Water system	0	0	0	0
711	Component Cooling	192	0	2	190
715	Salt Water	0	0	0	0
718	Salt water, turbine part	0	0	0	0
733	Demineralized water system	7	0	0	7
735	Refueling Water	5	0	0	5
761	Service Water[1]	7	0	0	7
762	Fire Protection[1]	4	0	0	4
766	Auxiliary Steam[1]	25	0	0	25
Total		1345	64	132	1149

[1] Portion of system only.

Table 2.1.4-8: Ringhals Unit 3/4- Result – Outliers Removed

System ID	System	Total no. of Segments	No. of HSS Segments based on numerical value	No. of HSS Segments after EP-meeting	No. of LSS Segments
313	Reactor Coolant	87	16	26	61
321	Residual Heat Removal	58	12[1]	18	40
322	Containment Spray	91	0	0	91
323	Safety Injection	139	10[1]	13	126
327	Auxiliary Feedwater	72	4	0	72
334	Chemical and Volume Control	184	33[1]	11	173
337	Steam Generator Blowdown	44	0	0	44
411	Main Steam	87	3	10	77
414	Condensate	146	20	20	126
415	Main Feedwater	145	0	0	145
418	Reheating system	28	0	0	28
419	Bleed steam system	24	6	6	18
443	Main Cooling Water system	0	0	0	0
711	Component Cooling	192	0	2	190
715	Salt Water	0	0	0	0
718	Salt water, turbine part	0	0	0	0
733	Demineralized water system	7	0	0	7
735	Refueling Water	5	0	0	5
761	Service Water[1]	7	0	0	7
762	Fire Protection[1]	4	0	2	2
766	Auxiliary Steam[1]	25	0	0	25
Total		1345	104	108	1237

[1] Portion of system only.

Table 2.1.4-9: Ringhals Unit 3/4- Inspection Location Selections Comparison

System ID	System	Previous Program			RI-ISI Program		
		No. of Welds or Segments	No. of Inspections	Other[1]	No. of Welds or Segments	No. of Inspections	Other[1]
313	Reactor Coolant		75			**87**	
321	Residual Heat Removal		2			72	
322	Containment Spray		0			0	
323	Safety Injection		23			34	
327	Auxiliary Feedwater		6			6	
334	Chemical and Volume Control		2			14	
337	Steam Generator Blowdown		0			1	
411	Main Steam		63			13	
414	Condensate		0			36	
415	Main Feedwater		37			0	
418	Reheating system		0			12	
419	Bleed steam system		0			16	
443	Main Cooling Water system		0			16	
711	Component Cooling		0			0	
715	Salt Water		0			8	
718	Salt water, turbine part		0			8	
733	Demineralized water system		0			0	
735	Refueling Water		0			0	
761	Service Water[2]		0			0	
762	Fire Protection[2]		0			0	
766	Auxiliary Steam[2]		0			0	
Total			208			323	

[1] There are also other owner defined programs (e.g. FAC) in place that were developed based on industry and plant-specific operating experience. Today the FAC part is included in the RI-ISI program, but we also have an owner defined program for small bore piping there FAC/EC currently could exist.
[2] Portion of system only.

2.2 Pilot Plant RI-ISI Applications

While in Section 2.1 the results of RI-ISI application that have been approved by the country specific regulator, are presented this section contains the results of lessons learned from pilot plant activities. While some of the results and lessons learned are similar to those found in Section 2.1, this section also includes insights from application of the technology to other countries (industry and regulatory viewpoints) as well as additional plant designs (NSSS and balance of plant).

2.2.1 Experience in Bulgaria

The current ISI program of Units 5 and 6 of Kozloduy NPP (both VVER-1000/B320) is based on the requirement of the Bulgarian regulations.

Current ISI Program

There are general requirements of the Bulgarian Nuclear Regulatory Agency (BNRA) related to ISI of NPPs. They are described in the following documents:

- Regulation for Providing the Safety of Nuclear Power Plants (published in May 2004, last amendment in June 2007);
- Regulation on the Procedure for Issuing Licenses and Permits for Safe Use of Nuclear Energy (published in July 2004, last amendment in October 2012).

A program for ISI of the base and weld materials of the equipment and pipes of a NPP unit must be developed by the licensee and shall be part of the documentation. It must be submitted to the regulator in the case of issuing an operating license for a NPP unit.

The components in the primary circuit shall be designed, manufactured and located in a way allowing periodical testing and inspection during the whole operating period of the NPP. The programme for control of the primary circuit shall ensure monitoring of the influence of radiation, initiation of cracks due to stress corrosion, embrittlement and ageing of the materials especially at locations with high radiation levels and/or environmentally unfavourable conditions.

The condition of the base metal and the welded joints of systems, structures and components (SSC) important to safety shall be periodically controlled by qualified NDT regarding the locations, methods, detection of defects and effectiveness according to the developed procedures.

Some additional requirements concerning ISI inspection intervals are defined based on Russian designer and manufacturer's documents (e.g. PNAEG-7-008-89 "Rules for structure and safe operation of equipment and pipelines of NPPs"). E.g. periodic NDE shall be performed in the following periods:

- Not later than 20,000 hours of operation of the equipment and pipes.
- All subsequent - for equipment group A and equipment and pipelines group B, made of tubes or casings with longitudinal welding not later than every 30,000 hours of operation from the previous periodic control; all other equipment and pipes subject of control every 45,000 hours of operation from the previous periodic testing.

- Implementation of the scheduled control (after the first) can be distributed in stages within the required period, but not less than 5,000 hours.

There are no specific requirements of the BNRA to RI-ISI. NDT of the base metal and welded joints of the components of Units 5 and 6 of Kozloduy NPP is performed in accordance with ISI program developed by Kozloduy NPP and approved by the BNRA. The ISI program describes the compulsory scope and period of the ISI testing of each SSC important to safety. A deterministic approach is used for development of the ISI program. It is based on the Russian ISI requirements. The ISI program is developed for each unit and for every outage. An example of some equipment and pipe-work testing periods is given in Table 2.2.1-1. Examples for degradation mechanisms for equipment of the primary and secondary circuit are given in Table 2.2.1-2.

Table 2.2.1-1: Component / equipment subject to inspection

Component / Equipment	**Time between consecutive inspections**
RPV, RPV inner equipment, control rod drive system, RPV upper unit, RPV equipment, SG, Pressuriser, Pressuriser pipe-work, MCP Dy850, surge-line Dy300 from RPV to the hydro-accumulators.	30,000 operating hours
MCP, Barbotage tank, filters, heat exchanger and additional cooling of the blow down system. Pipe-work: Blow down line, feeding water, drainage and bypass cleaning of primary circuit; emergency and planned core cooling; emergency injection of boric solution, emergency feeding water inside the confinement, SG air ducts, blowers from the pressuriser; SG main steam and feedwater pipe-works, non-isolatable section within the containment.	45,000 operating hours
High and low pressure cylinders; OK-12A turbine	
Outlet pipe-work between LPC and condenser; Low pressure heater drainage lines; condense from 'fresh' steam pipe-line to low pressure heater;	30,000 operating hours
High pressure Deaerator; low and high pressure heaters; main steam lines and fresh steam lines; feedwater lines RL; welded joints, elbows of systems *RC, RQ, RM, RD, RB, RN* pipe work; Pipe lines from low pressure heater to high pressure Deaerator; Pipe line from high pressure heaters to safety valves; Steam lines to separators I and II; Pipe lines for the condensed hot steam, from 'fresh' steam pipe-line to high pressure heater; High pressure heaters drainage lines;	45,000 operating hours

Table 2.2.1-2: Ageing mechanisms of some components

Component	**Ageing mechanism**
Steam Generator	• Corrosion • Fatigue • Thermal Ageing
Main Primary Coolant and Surge Pipes	• Thermal Ageing • Corrosion • Fatigue
Main Coolant Pump	• Thermal Ageing • Corrosion • Wear
Pressurizer	• Thermal Ageing • Corrosion • Fatigue
Pressurizer Safety Valves and Relief Pipe	• Fatigue

Component	Ageing mechanism
	• Corrosion • Wear
Piping / supports of MSL, MFWL, EFWL	• Stress and fatigue • Erosion and corrosion
Valves (secondary circuit)	• Stress and fatigue • Erosion and corrosion

RI-ISI Results

RI ISI Pilot Study

Kozloduy NPP has performed a pilot project, based on the PWROG methodology [17] for RI-ISI. The RI approach, as defined in the U.S. NRC RG 1.174 – "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Nov 1998) [19] and RG 1.178 - "An Approach for Plant Specific Risk-Informed Decision making for In-service Inspection of Piping" (Sept 1998) [20] was used in this project. The basic steps to develop a RI-ISI programme are shown in Figure 2.2.1-1.

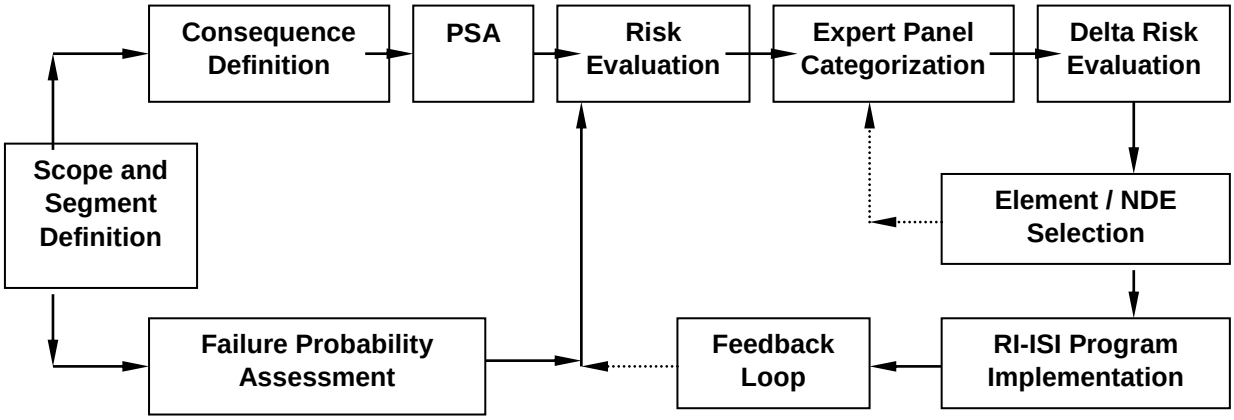

Figure 2.2.1-1: Basic Steps for Development of RI-ISI Programme

Scope definition – the system addressed in this programme were based on safety significance, part of the original ISI programme, impact on outage, and exposure. The systems selected were main steam pipe (RA), steam generator main feedwater pipe (RL); primary make-up and blow-down (TK); low-pressure system for emergency and planned core cooling (TQ2); primary circuit main coolant pipes (YA); protection system of the primary circuit against overpressure (YP), part pressurizer system, emergency core cooling passive part - hydro-accumulators (YT).

Economic impact – an analysis was performed using two methods: Method 1 – average flows incomes/expenses and Method 2 – Monte Carlo simulation. The impact of unplanned reactor trips, implementation costs, operational costs, outage impact, risk impact and personnel exposure and used generic data were taking into account. A number of recommendations for changes in ISI activities of the systems addressed were provided.

Results – the analysis for optimization of the original ISI program is applied only to main steam pipe (RA), steam generator feedwater pipe (RL), primary make-up and blow-down (TK), low-pressure system for

emergency and planned cooling (TQ2); main coolant (YA), pressurizer system (YP), emergency core cooling passive part (YT).

For Unit 5 the original ISI program covers 2191 segments for testing and RI-ISI program covers 223 segments. It is a reduction by 89.82% in the number of tested segments. The summary of the results is presented in Table 2.2.1-3 and Figure 2.2.1-2.

Table 2.2.1-3: Summary of the Result based on Changes in ISI to RI-ISI for Unit 5

System	Number of Testing in ISI	Number of Testing in RI-ISI
TK	113	40
TQ2	207	34
RA	475	35
RL	816	52
YA	105	28
YP	238	28
YT	237	6
Total	2191	223

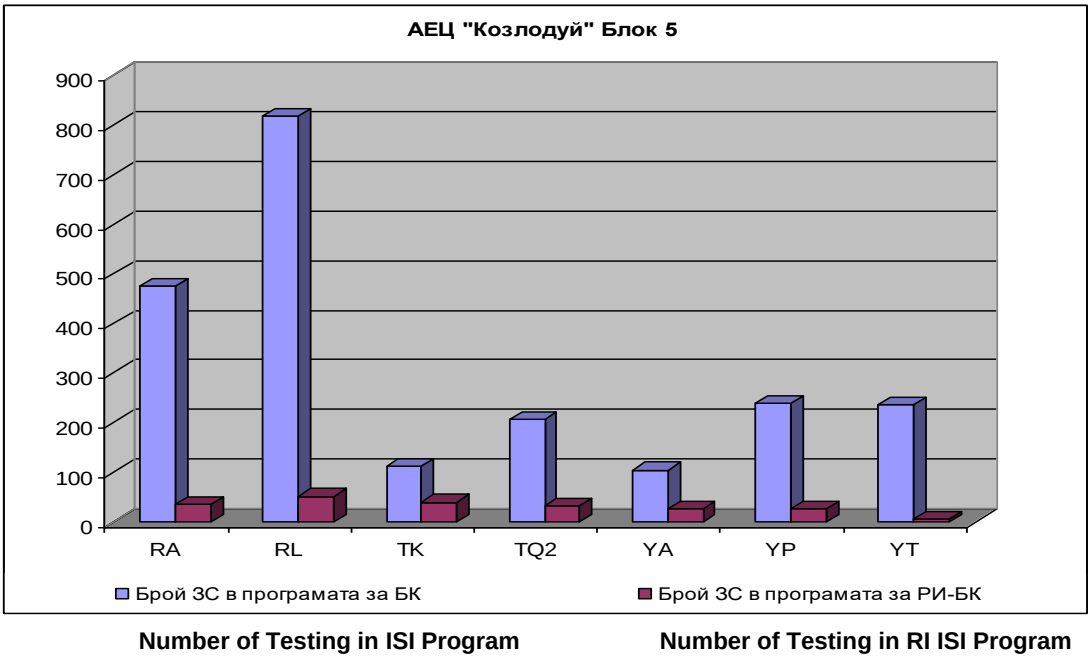

Figure 2.2.1-2: Correlation between Number of Testing in ISI and RI-ISI for Unit 5

For Unit 6 the original ISI program covers 2102 segments for testing and RI-ISI program covers 224 segments. It is reduction of 89.34%. The summary of the results is presented in Table 2.2.1-4 and Figure 2.2.1-3.

Table 2.2.1-4: Summary of the Result based on Changes in ISI to RI-ISI for Unit 6

System	No. of Testing in ISI	No. of Testing in RI-ISI
TK	125	40
TQ2	196	34
RA	466	35
RL	751	52
YA	117	28
YP	230	29
YT	217	6
Total	2102	224

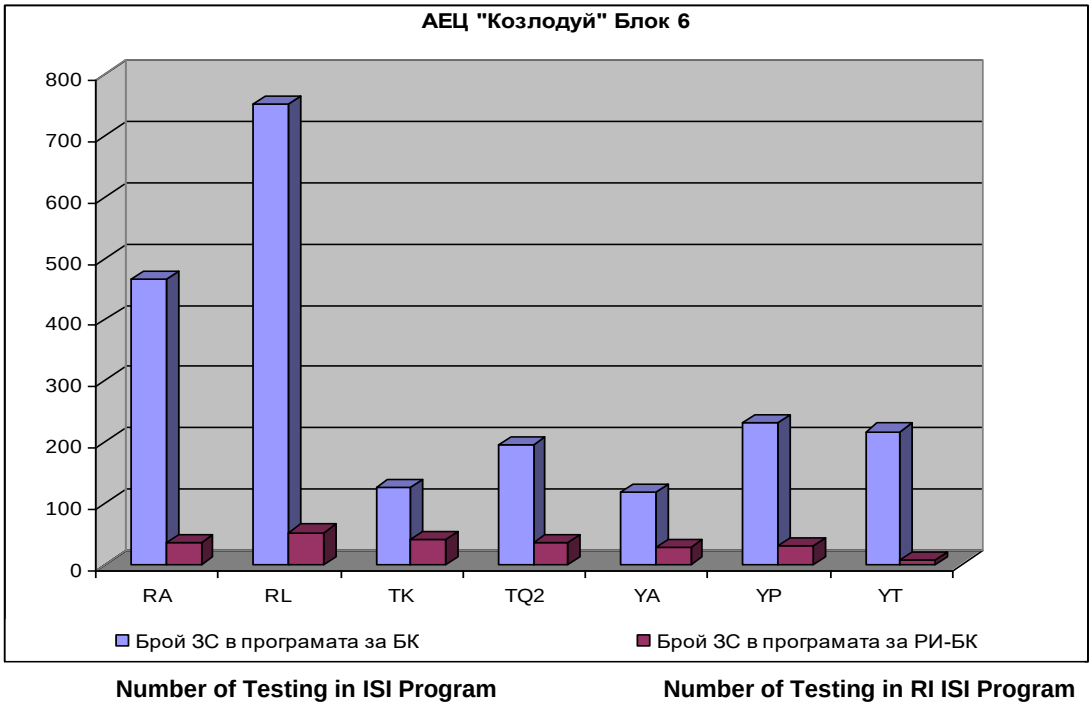

Figure 2.2.1-3: Correlation between Number of Testing in ISI and RI-ISI for Unit 6

The results of this RI-ISI pilot study were used for development of a revision of the Kozloduy NPP ISI program for Units 5 & 6 in 2011. The main areas re-considered were the scope and period of the ISI testing of the following equipment:

- Primary circuit main coolant pipes Dy850 (YA system);
- Primary circuit pipes Dy300 (emergency core cooling passive part – hydro-accumulators – YT System);
- Pipes of protection system of the primary circuit against overpressure, part pressurizer system (YP);
- Pipes of the primary make-up and blow-down system (TK);

Table 2.2.1-5 provides an example page of the new procedure.

Table 2.2.1-5: Example procedure for ISI at Kozloduy NPP

№	Name of Equipment and Segments for ISI	Type of Testing	Reference	Scope of Testing [%]	Period	Note
4.8	**Pipes of protection system of primary circuit against overpressure (YP), part pressurizer system, group B, safety class 2H**					
4.8.1	**Pipeline for emergency cold injection to pressurizer (5,6YP-2)** Control scheme No. 11					
4.8.1.1	Base metal and welds of pipe bends with two parts	VT	6	100	Not later than 60 000 operating hours	
		PT (with color)	8	100		Band 100mm wide on outside surface around welds
4.8.1.2	**<u>For Unit 5</u>:** Welds of the main pipelines Dy Ø 219x19 – No 12, 13, 17, 18, 20, 21, 30, Control scheme No35.PO.YP.IC.007-1 And Welding No10 with Control scheme 35.PO.YP.ИС.007-2 **<u>For Unit 6</u>:** Welds of the main pipelines Dy Ø 219x19 - No 5a, 5б, 5н, 6, 7, 8, 10, 19, 20 Control scheme No 36.PO.YP.ИС.007-01.	VT	6	100	Not later than 30 000 operating hours	
		PT (with color)	8	100		
		UT	7	100		
4.8.1.3	Welds of the main pipelines Dy Ø 219x19, Ø 159x18, without welds mentioned in без изброените в т. 4.8.1.2.	VT	6	100	Not later than 60 000 operating hours	
		PT (with color)	8	100		
		UT	7	100		
4.8.1.4	Welds of the main pipelines Dy Ø 57x5, Ø 38x3,5, Ø 18x2,5.	VT	6	100	Not later than 60 000 operating hours	

<u>Lessons Learned</u>

Resulting Number of Examinations

As shown in Table 2.2.1-3 and Table 2.2.1-4 and corresponding figures there is a significant reduction in the number of examinations compared to the original ISI program for both units.

Economic Impact

An economic analysis was performed using two methods: (Method 1) Average flows of incomes & expenses, (Method 2) Monte Carlo simulation. The economic analysis considered the impact of unplanned reactor trips, implementation costs, operational costs, outage impact, risk impact, and personnel exposure and used generic data that is representative of Kozloduy NPP. The economical impact of the three areas (ISI, maintenance and testing) is presented in Table 2.2.1-6. The negative share represents the probability that the task could have a negative economic impact based on the results of Method 2. Both methods are in agreement on a relative basis and indicate that changes to maintenance will provide the largest economic benefit with a benefit of approximately 52 million Euros. Implementation of proposed changes for testing activities shows a potential 22% chance that a positive economic impact will not be realized. Based on these results the implementation of all three tasks was recommended.

Table 2.2.1-6: Results of Economic Impact (in Euros)

	Method 1	**Method 2**			
	Mean	**Low**	**Mean**	**High**	**Negative share, %**
RI-ISI	320 000	108 000	313 000	540 000	0.46
Testing	140 000	-247 000	90 000	328 000	22.00
Maintenance	49 000 000	33 000 000	52 000 000	76 000 000	0
Combined Cost	49 460 000	32 861 000	52 403 000	76 868 000	0

Some specific issues were found during the development of the version of the ISI program with elements of RI-ISI:

- **Group definition of SSCs that are included in the scope of the RI-ISI programme**: The testing period must be defined for each piece of equipment in that group of SSCs. E.g. two testing periods are defined for the standard welded joints of the main pipelines with Dy Ø 219x19.
- **Definition of physical boundaries for each segment/equipment included in the RI-ISI program**: The definition process includes a detailed study of the design scheme for each piece of equipment and the developed and applied control scheme (how to control the performance of the ISI testing).

These issues can be discussed for clarification and can be considered as an area for exchanging operating experience.

Up to now the version of the ISI program with elements of RI-ISI is considered as a pilot program only and it is not applied. The deterministic (conservative) method for ISI is applied. Thus e.g. the ISI testing period remains at 30,000 operating hours for all pipes of the protection system of the primary circuit against overpressure, part pressurizer system (YP).

2.2.2 Experience in Lithuania

In Lithuania a RI-ISI pilot study was performed at Unit 2 of Ignalina NPP in 2001. Ignalina NPP consists of two RBMK-1500 type reactors. Unit 1 was in operation from 1983 until 2004 and Unit 2 from 1987 until 2009. The so called project IRBIS (Ignalina Risk-Based Inspection Pilot Study) investigated optimal ISI strategies with respect to risk and required resources. The main project objectives were:

- Perform a RI-ISI of 300 mm pipe systems and to define a new risk-informed ISI program with focus on the high risk locations;
- Account for the cumulative radiation exposure of the inspection personnel when suggesting a new inspection program;
- Recommend improvements and further investigations that may increase the operation safety of Ignalina NPP.

In total 1240 stainless steel welds were analysed, assuming IGSCC to be the main damage mechanism. Pipe break frequencies were estimated by probabilistic fracture mechanic methods and combined with safety barriers, provided by PSA studies.

Practical experience indicated that IGSCC was the most important damage mechanism in RBMK plants. However, very few failures have actually occurred and this situation precluded any estimation of the failure probability based on observed data, other than perhaps small leak probabilities. To estimate the failure probability analytical methods have been used instead. All steps of the RI-ISI program development basically correspond to ENIQ recommendations [2].

After 3 years of operation, updating of RI-ISI was performed by taking into account new statistical data on pipe defects. Comparison with previous RI-ISI program was performed.

<u>Previous ISI Program</u>

At Ignalina NPP inspection of relevant 300 mm diameter austenitic piping of the main circulation circuit (MCC) were scheduled with an interval of 4 years, i.e. during a 4 years interval 100% of the welds of that piping system were inspected.

<u>RI-ISI Results</u>

At Ignalina NPP Unit 2 a RI-ISI pilot study was performed for the part of the MCC. All considered pipes were made of Ti-stabilised stainless steel with nominal diameter of 300 mm and nominal thickness of 16 mm. The primary damage mechanism in these piping systems is IGSCC. Until 2000 a total number of 278 cases of IGSCC have been found in Unit 1 at 17 years of operation. The corresponding number for Unit 2 is 57 cases of IGSCC at 13 years of operation. The deepest cracks reached a depth of about 12 mm. However, no leaks have been experienced so far for this type of piping. The mean value of the initial defect length has been about 57 mm, corresponding to 6% of the inner circumference. In some rare cases defect lengths exceeding 30% of the inner circumference were detected.

The piping with 1240 welds considered for RI-ISI includes the following MCC pipe systems:

- WEP - water equalizing piping;
- DC - downcomers;
- SH-PH - bypass between suction header (SH) and pressure header (PH);

- PP - pressure piping, connecting PH and group distribution header (GDH);
- GDH - group distribution headers;
- BCS - blowdown and cool down piping system.

IGSCC cracks appeared at the pipe inner surface in the heat affected zone near the weld root and propagated to the outside surface along the fusion line. Hundreds of IGSCC damages were detected during reactor operation, but only a few of them led to pipe breaks or pipe leakage. The root cause for IGSCC and crack growth in general were not clearly identified. The development of a weld-repair method that significantly reduces or even avoids the occurrence of IGSCC proofed to be challenging.

The consequences of the selected pipe systems are determined and quantified by PSA studies. The consequences of the analysed pipe systems were given by a Level 1 PSA study and quantified by conditional core damage frequency (CCDF) entitled as safety barriers, i.e. probability of core damage given pipe failure. The safety barrier for a medium loss-of-coolant-accident (LOCA, 298–2685 kg/s leak rate of water) was used in most cases and this barrier was between 1.4E-3 and 3.3E-3, depending on the location of the rupture. Calculations were also performed for leaks (leak rates below 27 kg/s). Automatic scram is not generated in case of leaks and safety barrier equals to 2.9E-07, which is primarily associated with CDF due to manual scram event.

The starting-point for the risk ranking is the individual risk of a weld, which is evaluated according the outcomes of earlier inspections. At this point a decision is made, whether the weld will be selected for future inspections or not. The following table shows the total number of welds over all systems divided into different risk levels.

Table 2.2.2-1: Number of welds divided into different risk intervals.

Risk level per weld (earlier ISI)	Total number of welds	Number of welds, current ISI selection	Number of welds, RBI-1 selection
Very high risk VH $CDF \geq 1 \cdot 10^{-8}$	3	3	3
High risk H $1 \cdot 10^{-9} \leq CDF < 1 \cdot 10^{-8}$	202	202	202
Medium risk M $1 \cdot 10^{-10} \leq CDF < 1 \cdot 10^{-9}$	238	238	238
Low risk L $1 \cdot 10^{-11} \leq CDF < 1 \cdot 10^{-10}$	546	546	57
Very low risk VL $CDF < 1 \cdot 10^{-11}$	251	251	27

Table 2.2.2-1 shows that the current ISI program covers the welds of highest risk levels by 100%, but also 797 locations of low or very low risk, which do not significantly affect the total CDF. This indicates potential for optimization of the ISI program.

The following criteria are suggested for selection of inspection locations at Unit 2 of Ignalina NPP:

1. Select 100% of all locations with CDF≥1E-10 (VH, H and M risk).
2. Select 10% of all locations with CDF < 1E-10 (L and VL risk).

A very large number of RI-ISI selections (especially if one considers different choices of inspection intervals) exist in practice that still can be regarded as a reasonable choice. In this study, one particular RI-ISI selection out of many is presented, which is referred to as RBI-1.

The last column of Table 2.2.2-1 gives information on the number of selected welds for inspection for RBI-1, using the above presented criteria. A considerably smaller number of welds is selected for inspection at lower risk levels. The 10% selection in the lower risk levels has been done in a way that at least 10% of all welds have been selected (starting with the highest risk) in each pipe system, if there are welds that have risks in the considered risk level. The primary purpose of 10% selection is to avoid totally un-inspected piping segments. The reasons for this approach are that unknown failure mechanisms cannot be definitely excluded and that uncertainties in risk estimates do not guarantee the risk level to be as low as estimated. These inspections can also be considered as for defence-in-depth purposes.

Table 2.2.2-2 shows suggested inspection intervals for RBI-1 for each risk level. It is desirable to reduce the highest risks more than the lower risks and that is why a shorter inspection interval is suggested for the two highest risk levels. This is also one way to reduce the total CDF compared to the current ISI program, although a lower number of welds are selected for inspection in the RBI-1 case.

Table 2.2.2-2: Inspection intervals suggested for RBI-1 program.

Risk Category	Inspection interval
VH	1 year
H	2 years
M	4 years
L	4 years
VL	4 years

Table 2.2.2-3 shows the comparison between the current ISI program and the RBI-1 program in terms of core damage risk (per pipe system) and the number of selected welds of each pipe system. A considerable reduction of inspected welds is achieved for the DC and PP-system. These piping systems contain a large amount of low and very low risk welds. Despite this reduction in number of inspected welds, all pipe systems except GDH, will have a reduced CDF in case of RBI-1. Table 2.2.2-3 also shows the relative changes in CDF. The main reason for these changes is that all the high risk welds are inspected with a shorter inspection interval. The relative net reduction of the total core damage risk is 35% compared to the current ISI program.

Table 2.2.2-3: Sum of Core Damage Frequency in each pipe system with different ISI programs.

Piping System	CDF current ISI program	CDF RBI-1 program	⍰CDF
BCS	5.2E-8 (59 welds)	1.6E-8 (21 welds)	-3.6E-8 (-70%)
DC	6.3E-8 (604 welds)	4.9E-8 (279 welds)	-1.4E-8 (-22%)
PP	4.8E-8 (375 welds)	3.9E-8 (183 welds)	-9.0E-9 (-20%)
SH-PH	4.6E-9 (88 welds)	4.2E-9 (21 welds)	-3.8E-10 (-8%)
WEP	1.6E-9 (34 welds)	1.6E-9 (15 welds)	0 (0%)
GDH	1.6E-10 (80 welds)	4.8E-10 (8 welds)	+3.2E-10 (+207%)
All systems	1.7E-7 (1240 welds)	1.1E-7 (527 welds)	-6.0E-8 (-35%)

With a quantitative RI-ISI analysis for Ignalina NPP Unit 2, it is possible to combine a 44% reduction in the number of future inspections and a 35% reduction in risk. This is possible due to proposed shorter inspection intervals for high risk welds. Shorter inspection interval is suggested for 205 welds in the higher risk locations. Less than 100% extent of inspection in the lower risk levels is well compensated by the choice of a shorter inspection interval for the higher risk locations.

Many low risk locations are suggested not to be included in the new ISI selection. This means that the radiation exposure to plant personnel can be reduced and resources can be redirected to other safety related issues. The reduction of accumulated future radiation exposure for the suggested RBI-1 program is more than 3300 mSv from 2001 to 2017 compared to the current ISI program.

The highest risk is observed for some of the welds in the BCS piping system whereas some of the GDH-welds have the lowest risk. The highest risk per weld is 6.4E-8 per year. There are basically three reasons that explain differences in risk estimates:

- **Operating stresses**: The three BCS welds with the highest risk have exceptionally high operating stresses. Maximum dead weight stress is Pb = 53 MPa and maximum thermal expansion stress is Pe = 87 MPa. On the other hand, the GDH cap welds have very low operating stresses with zero thermal expansion stresses due to no restraint at the end cap.
- **Defect occurrence rate**: The occurrence frequency for IGSCC is generally higher (up to one order of magnitude) for the high risk welds in the BCS system compared to the GDH system. This is reflecting the number of cracks that have been observed in the respective pipe systems.
- **Weld residual stresses**: They are relatively large (max. 201 MPa at the inside of the pipe weld) for the high risk welds and lower for the low risk welds. However, weld residual stresses are quite similar for components within specific piping systems. Therefore, this parameter does not influence component ranking within the piping system.

The results of IRBIS update study indicate that some welds have moved from low and very low risk categories to higher risk categories. The result of this change was an increase in number of inspections and personnel radiation exposure by approximately 7%. The overall changes in the RI-ISI program after updating are not very significant. However, these changes raise some questions with respect to future updates of the RI-ISI program. In particular is there a need to periodically update the RI-ISI program and if yes, would development of a RI-ISI update procedure be beneficial?

<u>Lessons Learned</u>

The performed study with quantitative risk evaluations can be used for the following benefits:

- Optimise the selection of inspection locations.
- Optimise the inspection interval.
- Give quantitative information of the changes in risk and costs due to plant modifications, for example when:
    - A qualified inspection technique is introduced.
    - An inspection interval is changed.

- In case of an inaccessible location that makes it impossible to inspect.
- A replacement or repair is made.
- The water chemistry is changed.

Sensitivity studies performed within the IRBIS project have shown that:

- An effective way to reduce the overall risk is to improve the inspection detection efficiency for the high risk welds.
- There is in general little benefit to use very long inspection intervals (for example 10 years) unless very good detection efficiency can be achieved. This means that unless the detection efficiency can be proved to be better than assumed in this study, a substantial risk reduction can only be achieved by using inspection intervals between 1 and 4 years.
- The existence of high cycle vibrations in combination with IGSCC will have a quite harmful effect on the core damage risk.
- Improving leak rate detection capabilities may be an efficient way to reduce the risk of core damage for leaking cracks that have not been detected by inspections.
- Different crack growth rates will generally cause a preserved risk ranking order between different welds even if the absolute risk values are changed.
- Including dynamic effects with different safety barriers will increase the total CCDF by nearly a factor of 3 for the currently planned inspection program, mainly originating from the influence of the welds in the pressure piping system.
- Basing the RBI selection on release frequency from PSA, Level 2 puts more focus on welds located outside the confinement ALS.

After completion of the IRBIS project Ignalina NPP took advantage of the pilot study results and prepared a new inspection program focusing on the highest risk locations. The number of inspections was not reduced, but the risk was reduced significantly.

The Lithuanian regulatory body (VATESI) in general agrees to use a RI-ISI program for austenitic pipelines and waits for the proposal of Ignalina NPP. VATESI stated that if the number of inspections is reduced, then compensating actions should be taken, i.e. if some of the low risk welds are not periodically inspected in the future, a more precise leak detection system should be applied.

The “Requirements for Safety Assessment of Austenitic Components with IGSCC Cracks” includes the procedures for safety assessment and the procedures for determination of ISI extent and frequency. According to the requirements the extent of the inspection should be 100% or defined according to the risk ranking of the system under consideration. Risk is determined by multiplying PSA consequences with damage indexes (defect occurrence frequencies).

Before taking full advantage of the results of the IRBIS project for Ignalina NPP Unit 2, it was recommended to:

- Clarify the possible existence of high cycle vibrations in addition to IGSCC as a damage mechanism.

- Clarify the root cause of why the SCC cracks have not lead to any leak so far.
- Clarify the detection efficiency (POD) of the implemented inspection techniques. The best way to clarify this is probably to introduce the concept of inspection qualification.
- To further develop the PSA study with respect to dynamic effects and release barriers. It is important to base a future RBI selection on the release frequency.
- To form an expert panel with the task to review new proposed RBI programs and suggest possible changes and additions which are not always covered by the elements of the RBI program. This would include plant feedback as new plant information is obtained.

2.2.3 Experience in Romania

Romania has two CANDU-6 reactors in operation at Cernavoda at the Danube River. Unit 1 of Cernavoda NPP is in commercial operation since December 1996, Unit 2 since October 2007. Both units are operated by the state nuclear power corporation Societatea Nationala Nuclearelectrica (SNN), established in 1998.

As all operators of CANDU reactors also SNN is a member of the CANDU Owners Group (COG). The COG utilities have established a working group (WG) to promote the implementation of common RI-ISI methodologies for application to pressure boundary components. The objectives of the RI-ISI WG are to establish a common utility position on application of RI-ISI to CANDU stations, develop a common approach for CANDU stations to meet ongoing pressure boundary component challenges, and provide a forum for discussion of common RI-ISI issues.

To achieve the above objectives, COG conducted a pilot study applying an existing RI-ISI methodology [5] to a CANDU unit. After review of the CANDU design, supporting PRA analyses and the existing RI-ISI methodology, the methodology was updated to be CANDU specific (e.g. large release frequency metric was used in lieu of the large, early release frequency metric) to provide a CANDU best fit RI-ISI methodology for piping welds.

Along with the development of the COG RI-ISI project and in support of development of a new CSA N285.7 Standard, “Periodic Inspection of CANDU Nuclear Power Plant Balance of Plant Systems and Components”, the scope of the COG RI-ISI study was extended beyond piping welds to include other pressure boundary components such as tanks, vessels, pumps, valves, supports, mechanical couplings and rotating machineries.

Given the nature of a NPP and the new “balance of plant” systems scope, there are a large number of possible systems that could be subject to RI-ISI evaluation, and as such a system-by-system evaluation has the potential of becoming quite resource intensive. Therefore, a formal pre-screening process was developed that efficiently identifies those systems, or portions of systems, that need to be subjected to the full extent of the RI-ISI methodology while providing technical justification for excluding other (less important) systems.

The pre-screening process consists of a three track assessment of the potential for pressure boundary failures causing direct or indirect effects on plant operation as well as the plant’s mitigative ability. The pre-screening process also provides for identifying and defining plant practices that can be used to further define those systems (or portions of system) requiring detailed RI-ISI analyses. This progressive approach utilises plant resources more efficiently and allows for integration in existing (or improved) plant practices. Table 2.2.3-1 and Table 2.2.3-2 provide examples of the pre-screening process, once where the raw water reliability program (RWRP) is integrated into the pre-screening process and once where it is not.

As discussed above, the RI-ISI methodology has been expanded to include consequence of failure assessment and potential of failure assessments (e.g. new types of degradation and susceptibility criteria) for all piping systems (e.g. balance of plant systems) and related components. This included an exhaustive literature search and evaluation of plant-specific and CANDU fleet operating experience.

Table 2.2.3-1: Pre-screening Process – Results (with RWRP Credit)

System	Consequence	Method[1]	Comment
Main Steam Supply (36110)	High	Track # (RI-ISI)	Large main piping (JP-4369)
	Low	Track 3	Smaller piping (JP-4369)
Steam Generator Blow off (36410)	Low	Track 3	
Boiler Feed (43000)	High	Track # (RI-ISI)	Large main piping between LCV and pump (JP-4369)
	Low	Track 3	All other piping (JP-4369)
Boiler Feed Pump Gland Seal Supply (43230)	Low	Track 3	
Condensate (44000)	Low	Track 3	
Extraction Steam (48100)	Low	Track 3	
Recirculated Cooling Water System (72200)	Low	Track 3	
Powerhouse Upper Level Service Water (72300)	Medium	Track 2	MIC/E-C evaluation required
Auxiliary Service Water (72500)	Low	Track 3	
Emergency Service Water (72800)	High	Track # (RI-ISI)	Detailed evaluation required

Table 2.2.3-2: Pre-screening Process – Results (without RWRP Credit)

System	Consequence	Method[2]	Comment
Main Steam Supply (36110)	High	Track # (RI-ISI)	Large main piping (JP-4369)
	Low	Track 3	Smaller piping (JP-4369)
Steam Generator Blow off (36410)	Low	Track 3	
Boiler Feed (43000)	High	Track # (RI-ISI)	Large main piping between LCV and pump (JP-4369)
	Low	Track 3	All other piping (JP-4369)
Boiler Feed Pump Gland Seal Supply (43230)	Low	Track 3	
Condensate (44000)	Low	Track 3	
Extraction Steam (48100)	Low	Track 3	
Recirculated Cooling Water System (72200)	Low	Track 3	
Powerhouse Upper Level Service Water (72300)	Medium	Track # (RI-ISI)	PS service experience per NK38-MAN-721 00-1 0001-R006, detailed evaluation required
Auxiliary Service Water (72500)	Low	Track 3	
Emergency Service Water (72800)	High	Track # (RI-ISI)	Detailed evaluation required

[1] Track # (RI-ISI) means the system (or portion of system) does not pass the pre-screening process and will be further evaluated.
[2] Track # (RI-ISI) means the system (or portion of system) does not pass the pre-screening process and will be further evaluated

Results for the four systems that required the full RI-ISI evaluations through this effort are provided in Table 2.2.3-3 to Table 2.2.3-6[1].

Table 2.2.3-3: Main Steam Summary

Component Type	RC	Selections
Piping	4	21
Piping Supports	4	38
Valves	4	1
Mechanical Couplings	4	1

Table 2.2.3-4: Boiler Fee Summary

Component Type	RC	Selections
Piping	4	26
Piping Supports	4	47
Valves	4	4
Mechanical Couplings	4	4
HX Welds	4	12
HX Supports	4	2

Table 2.2.3-5: Powerhouse Upper Level Service Water Summary

	Case 1 DM Assessment		Case 2 DM Assessment	
Component Type	RC	Selections	RC	Selections
Piping	5	73	6	0
Piping Supports	5	36	6	0
Valves	5	6	6	0
Mechanical Couplings	5	19	6	0
HX Welds	5	2	6	0
Pumps	5	1	6	0
Pump Supports	5	3	6	0

Table 2.2.3-6: Emergency Service Water Summary

	Case 1 DM Assessment		Case 2 DM Assessment	
Component Type	RC	Selections	RC	Selections
Piping	2	309	4	124
Piping Supports	2	339	4	170
Valves	2	25	4	10
Mechanical Couplings	2	76	4	30
Tank Welds	2	11	4	5
Tank Supports	2	2	4	2

[1] Note: Case 1 and 2 and Table 2.2.3-5 and Table 2.2.3-6 reflect a sensitivity study relative to DM susceptibility. Also, note that the original RI-ISI methodology sampling rules (e.g. 25% for Risk Category 1, 2 and 3 and 10% for Risk Category 5 and 6) were used to determine the number of inspections in Table 2.2.3-3 through Table 2.2.3-6.

Pumps	5	1	6	0
Pump Supports	5	1	6	0

Insights from this effort were identified that included:

- The base RI-ISI methodology [5] could be efficiently adapted to the CANDU design and operating regime.
- There are unique aspects of the CANDU design as compared to the light water reactor fleet (e.g. large release frequency, potential for multi-unit impact).
- Due to the scope and breadth of systems included, additional risk metrics are warranted (e.g. irradiated fuel bays).

Benefits witness by conducting this RI-ISI pilot study include:

- Improvements in plant safety by identifying and focusing resources on those components that can initiate or mitigate important plant events.
- The updated methodology cost-effectively identifies systems/components requiring further RI-ISI evaluation thereby reducing analysis burden.
- Reductions in worker exposure and inspection cost by identifying and eliminating low value added inspection activities.

The CANDU Best Fit RI-ISI methodology and the results of the pilot plant application have been adopted in a new CSA N285.7 Standard. The original RI-ISI sampling rules (25% and 10%) were modified and updated in CSA N285.7 in order to better fit the CANDU balance of plant design.

2.2.4 Experiences in Sweden

2.2.4.1 Forsmark Unit 3

This project was sponsored by the Swedish regulator (SSM) with the objective of a pilot plant demonstration of the EPRI RI-ISI Methodology [5] to selected systems at Forsmark, Unit 3 (F3). As described in Reference [22], five systems were selected for evaluation. These systems were selected because they allow the project to focus on a number of issues of interest in developing a RI-ISI methodology and RI-ISI program. This includes the following:

- Several different types of degradation may be identified,
- Several different types of "consequence of failure" may be identified,
- Different types of safety systems are evaluated, and
- Non-safety systems are evaluated

Using the results of this application, insights and comparisons between SKIFS and the EPRI methodologies' were developed including the following:

- Consequence of pressure boundary failure (PBF),
- Degradation mechanism evaluation,
- Risk ranking,
- Elements (welds) selected for inspection, and
- Risk impact of the new ISI program versus the proposed ISI program.

<u>Previous ISI Program</u>

The SKIFS inspection approach [16] is similar to the EPRI RI-ISI approach [5] in that both consequence of failure and degradation potential are evaluated and then welds are ranked in a risk matrix (see Figure 2.2.4-1). The following summaries the SKIFS evaluation steps:

- Consequence index (KI) is determined based on a pipe's role in keeping reactor fuel covered by water, i.e. proximity of piping to the reactor vessel and isolation valves, a pipe's role in the reactor cooling cycle and a pipe's role in performing emergency core cooling and scramming the reactor.
- Degradation potential or damage index (SI) is determined based on degradation mechanism potential and mechanical fatigue.
- Inspection groups are determined using the risk matrix in Figure 2.2.4-1.
- Elements (welds) are selected for inspection.

This study was based upon F3 implementation of SKIFs guidance as well as other consideration as documented in the PMT program.

<u>RI-ISI Results</u>

The systems selected for the F3 pilot study and reasons for selecting these systems are provided below:

- 311 “Main Steam” from the reactor pressure vessel (RPV) to the 421 system in the Turbine Building
  - Postulated failures may result in a LOCA event;
  - Postulated failures may result in a plant transient;
  - Piping located inside and outside containment;
  - High pressure / temperature steam environment;
  - Normal operating system;
  - Safety related and non-safety related system.
- 312 “Feedwater Lines” from the 463 system in the Turbine Building to the RPV
  - Postulated failures may result in a LOCA event;
  - Postulated failures may result in a plant transient;
  - Piping located inside and outside containment;
  - High pressure / temperature water environment;
  - Normal operating system;
  - Safety related and non-safety related system.
- 321 “Residual Heat Removal”, a closed loop system that takes suction from the RPV and returns to the reactor through the 312 system injection path
  - Postulated failures may result in a plant transient;
  - Postulated failures may impact mitigative (standby low pressure portion of system) equipment;
  - Piping located inside and outside containment;
  - Portions of system experience a high pressure / temperature water environment;
  - Portions of system experience a low pressure / temperature water environment;
  - Portion of system normally operating;
  - Portion of system normally in standby;
  - Safety related and non-safety related system.
- 323 “Low Pressure Injection” takes water from the suppression pool and injects into the RPV
  - Postulated failures may impact mitigative (standby ECCS function) equipment
  - Piping located inside and outside containment
  - Low pressure / temperature water environment
  - System normally in standby
  - Safety related system

- 462 “Condensate” takes water from the condenser hot well and supplies the feedwater system (312) via system 463
    - Postulated failures may result in a plant transient
    - Piping located outside containment
    - Moderate pressure / temperature water environment
    - System normally operating
    - Non-safety related system

From a consequence of failure perspective, some of the insights gained are listed below:

- None of the F3 scope piping has the highest Consequence Index, KI=1 (all piping segments have KI = 2 or 3 or Blank). Yet, the EPRI methodology identifies several pipe segments as a high consequence. This includes piping in the 311, 312, 321 and 323 systems directly connected to the reactor vessel that is not isolable. The conditional core damage probability (CCDP) in the F3 PSA is relatively high for this piping in comparison to all other piping in the scope of this evaluation and the CCDP exceeds the EPRI criteria for a high consequence rank. Also, certain piping outside containment between the penetration and the first isolation valve outside containment in the 311, 321 and 323 systems was determined to have a high consequence as a result of exceeding the EPRI conditional large early release probability (CLERP) criteria. There is one isolation valve inside containment and CCDP becomes medium, but this piping failure causes containment bypass and the CLERP results in a high consequence. Note that the 312 system has 2 check valves inside containment, which reduces the probability of an unisolable LOCA outside containment such that the CLERP for this system is medium.

- At the other extreme is piping with a KI = 3 or blank (blank fields in the database were assigned a none consequence), which is similar to a low consequence rank, which was also reviewed for comparative insights.

  <u>311 System</u> – all piping segments identified as low using the EPRI methodology were also identified as KI=3 or none, which is considered the same as low.

  <u>311 System</u> – a KI=3 or blank (none) is assigned to segments identified as medium consequence using the EPRI methodology. It appears that piping considered small LOCA (S2) is assigned a low consequence per SKIFS versus medium consequence per EPRI.

  <u>312 System</u> – no segments were identified as low using the EPRI methodology. Three segments assigned a medium consequence have a KI = 3 or blank assigned to portions of these segments.

  <u>321 System</u> – three segments identified as low using the EPRI methodology were assigned a KI=2.

  <u>321 System</u> – several segments assigned a medium consequence have a KI=3 or blank assigned to portions of these segments.

  <u>323 System</u> – the low consequence segments using the EPRI methodology have a blank KI (None) with the exception of one weld.

323 System – several segments assigned a medium consequence have a KI = 3 assigned to portions of these segments.

462 System – the system was assigned a medium consequence using the EPRI methodology; this system in not in SKIFS requirements scope.

Both the SKIFS and EPRI methodologies rank failure potential according to likelihood of the piping being exposed to some type of stressor (e.g. degradation). In simple terms, low stress results in a low failure potential (e.g. SI III) for SKIFS while high stress results in a high failure potential (SI I). In the EPRI approach, FAC is typically the reason piping is assigned to the high failure potential rank. In contrast, FAC is typically not assessed as part of the SKIFs process as systems susceptible to FAC are not within the scope of SKIFs.

To further explain the insights gained from this effort, Table 2.2.4-1 has been developed. In this table, the degradation mechanisms identified for each system is listed for each methodology and the final column identifies insights from this comparison.

Degradation Severity

As mentioned above, SKIFS utilizes the three degradation categories (I, II, III similar to high, medium, low) for SCC whereas the EPRI approach would categorize this as medium and if the piping is resistant material it would be categorized as low. However, the EPRI approach also points to the IGSCC augmented program; differences between this program (NUREG-0313/BWRVIP-075) and SKIFS were evaluated and are briefly discussed below. SKIFS also has three degradation categories for mechanical fatigue (MF), which is also discussed below.

Mechanical Fatigue

One of the key insights from this review is that SKIFs methodology includes MF in its failure potential ranking and also its risk ranking. MF evaluations in these instances are based upon design basis loadings and stresses. This is somewhat consistent with the original ASME philosophy.

The EPRI RI-ISI does not include MF, as part of its failure potential assessment scheme. This is documented in [5] and is based on three factors. First, a review of service experience has shown that failures do not occur at locations of high stress per the design stress reports. Secondly, by the very nature of meeting the allowable stress values, these locations are not expected to fail due to loading conditions contained in the design stress reports. Finally, failures typically occur due to phenomena not accounted for in the design stress reports (e.g. SCC).

However, there is a point of consistency in that during the element selection process, absent other considerations (e.g. severity of degradation, dose, and access), stress report results and stress discontinuities can be used to preferentially select inspection locations.

NUREG-0313/BWRVIP-075

Although somewhat consistent, NUREG-0313 and SKIFS have somewhat different philosophies with respect to program scope and selection of locations for inspections. The scope of piping contained within the NUREG-0313 program is stainless steel piping ($\geq$ 4 NPS) exposed to reactor water at operating temperature greater than 200°F (93°C). From a F3 perspective, this would allow smaller bore piping to be excluded. This would also allow most of the RHR low pressure circuit to be considered not susceptible to

SCC as it is in standby and isolated from reactor coolant during normal power operation. Another difference is that NUREG-0313 classifies in-scope piping as either resistant or not resistant to SCC. For weldments identified as susceptible, NUREG-0313/BWRVIP-075 assigns a category ranging from A to G. While SKIFS ranks the piping as susceptible to SCC with three damage indices (SI I, II or III).

The EPRI and SKIFS risk matrices are shown below for comparison (Figure 2.2.4-2). As shown, they are very similar except the SKIFS consequence rank is left to right High (1), Medium (2), Low (3) and None versus the EPRI consequence from left to right is None, Low, Medium and High. This has no technical impact, only visual effects. For example, the SKIFS high risk cells (Inspection Group A) is in the top left corner versus the top right for the EPRI matrix (H).

Risk Ranking Results

Differences in the criteria used to assign consequence and degradation potential result in differences in risk ranking where fully evaluated during this project. The following summarizes key differences and their impact:

- Mechanical Fatigue (MF): One of the key insights from this risk ranking review is that SKIFS methodology includes MF in its failure potential ranking and thus its risk ranking. For the reasons discussed above, the EPRI RI-ISI does not include MF, as part of its failure potential assessment scheme. To investigate this difference, a sensitivity case was considered where the SKIFS risk ranking was completed without MF (SI was revised to III for all MF).
- IGSCC: Although somewhat consistent, NUREG-0313 and SKIFS have somewhat different philosophies with respect to program scope and selection of locations for inspections. The scope of piping contained within the NUREG-0313 program is stainless steel piping ($\geq$ 4 NPS, 100DN) exposed to reactor water at operating temperature greater than 200°F (93°C). From a F3 perspective, this would allow smaller bore piping to be excluded. This would also allow most of the RHR low pressure circuit to be considered not susceptible to SCC as it is in standby and isolated from reactor coolant during normal power operation. Another difference is that NUREG-0313 classifies in-scope piping as either resistant or not resistant to SCC. SKIFS ranks the piping as susceptible to SCC with three damage indices (SI I, II or III).
- Sensitivity cases were conducted to assess the impact of these (and other) considerations on plant risk.

The Table 2.2.4-2 summarizes the results and as shown, the number of High (A) risk welds was reduced in the Feedwater (312) system and the number of Medium (B) risk welds was reduced in both the Main Steam (311) and Feedwater (312) systems. There was no change to the RHR (321) and Low Pressure Injection (323) systems. Note that this would impact element selection and risk impact for the 311 and 312 systems.

Table 2.2.4-1: Failure Potential Insights

System	EPRI	SKIFS	Insights
311	None = 353	SI-I = 3 welds SI-II (SCC) = 6 welds SI-II (MF) = 53 welds None = 291 welds	• See discussion on MF. • Appendix A provides a comparison to NUREG-0313 for IGSCC
312	IGSCC = 22 welds None = 124 welds	SI-I (SCC) = 11 welds SI-I (MF) = 10 welds SI-II (SCC) = 6 welds SI-II (MF) = 18 welds None = 101 welds	• See discussion on MF. • Appendix A provides a comparison to NUREG-0313 for IGSCC
321	IGSCC = 138 welds IGSCC, TT = 2 welds TT = 28 welds None = 250 welds	SI-I (SCC) = 102 welds SI-II (SCC) = 119 welds SI-II (MF) = 2 welds SI-III (SCC) = 2 welds None = 193 welds	• See discussion on MF. • Appendix A provides a comparison to NUREG-0313 for IGSCC
323	IGSCC = 8 welds TT = 14 welds None = 516 welds	SI-I (SCC) = 10 welds SI-II (SCC) = 2 welds None = 526 welds	• See discussion on MF. • Appendix A provides a comparison for IGSCC • TT identified on drain line used to respond to vessel level excursions
462	FAC = 10 segments None = remaining segments	FAC = 10 segments None remaining segments	• Consistent with previous experiences

Table 2.2.4-2: Risk Ranking Results - Summary

System	EPRI			SKIFS with MF			SKIFS without MF		
	High	Med	Low	High (A)	Med (B)	Low C	High (A)	Med (B)	Low C
311	0	82	271	0	32	321	0	0	353
312	8	34	102	21	22	103	11	4	129
321	10	189	219	15	107	296	15	107	296
323	10	41	487	10	2	526	10	2	526
462	Yes	0	Yes	0	0	0	0	0	0
Total	**28**	**346**	**1079**	**46**	**163**	**1246**	**36**	**113**	**1304**

		Consequence Index (KI)		
		1	2	3
Degradation Index (SI)	I	Inspection Group A	Inspection Group A	Inspection Group B
	II	Inspection Group A	Inspection Group B	Inspection Group C
	III	Inspection Group B	Inspection Group C	Inspection Group C

Figure 2.2.4-1: SKIFS Risk Matrix

Note that in practice there is also a “None” consequence index utilized by SKIFS similar to EPRI.

EPRI Risk Matrix

		Consequence Rank			
		None	Low	Med	High
DM Potential	High	L	M	H	H
	Med	L	L	M	H
	Low	L	L	L	M

SKIFS Risk Matrix

		Consequence Index (KI)			
		1	2	3	None
Damage Index (SI)	I	A	A	B	C
	II	A	B	C	C
	III	B	C	C	C

Figure 2.2.4-2: EPRI and SKIFS Matrix

3. Generic Lessons Learned

Due to the number of RI-ISI applications, the different countries in which they were conducted and the variety of plant designs involved, there were a number of lessons learned. Some of the lessons learned are generic in nature while some are unique to a particular country. Chapter 2 provides a detailed description of these lessons learned from each application.

Below is a summary of these lessons learned partitioned into three main areas: benefits of applying RI-ISI, technical and process considerations, and regulatory considerations.

Benefits of applying RI-ISI

Plant Safety: For each application, the RI-ISI program developed was able to be shown to improve or at least maintain plant safety as compared to the previous ISI program.

Worker Exposure: Each application was able to show a reduction in worker exposure and radwaste.

Outage Operations: Even though some RI-ISI applications did not show a large reduction in the number of inspections, outage activities were simplified. That is, as compared to the previous ISI program, the RI-ISI methodologies provide more flexibility in picking inspection locations. E.g. when using RI-ISI, often one set of scaffolding can be used to inspect multiple locations.

Improved Focus: RI-ISI allows plant operators and regulators to focus finite resources on those components most important to safety.

Technical and process considerations

Current Practices: A thorough understanding of current plants practices (e.g. codes/standards followed, regulations, owner defined inspections) is important prior to making decision about the scope of the RI-ISI application and the makeup of the project team.

Project Team: As RI-ISI is a multiple discipline technology, the project team must be equipped with the appropriate disciplines and level of technical expertise. Additionally, a team leader with plant management sponsorship greatly ensures the chance of success.

Ownership by Plant Management: As alluded to above (Project Team), ownership by plant management is vital to assuring a successful RI-ISI application. Not only is needed to support the technical project team, but regulatory interactions at the appropriate levels is very important to an efficient review and approval process.

Status of the PRA: Successful application of RI-ISI does not require a full-scope, state of the art PRA. While a more complete PRA would certainly streamlined the process, the RI-ISI methodologies were developed to be used with PRAs having a varying degree of “completeness”. Were developed with the estate of PRA A, thorough understanding of current plants practices (e.g. codes/standards followed, regulations, owner defined inspections) is important prior to making a decision about the scope of the RI-ISI application and the makeup of the project team.

Regulatory considerations

Scope: In some countries a "partial scope" application was deemed acceptable (e.g. Class 1 only, Class 1 and 2 only) while in other countries a "full scope" were required. However, even in "full scope" applications, it was seen that a large number of systems were subject to the full RI-ISI methodology while a number of system were either qualitatively, or semi-quantitatively (e.g. PRA importance measure) screened out from the RI-ISI application.

Plant/Regulatory Interactions: While the relationship between each plant operator and its regulator is country specific, experience has shown that having early engagement with the regulatory body is very beneficial. This is particularly important if risk technology has not been widely used in the past.

Risk Acceptance Criteria: While all RI-ISI methodologies have processes for determining the "change-in-risk" associated with the RI-IS application, the acceptance criteria used to determine if that change is acceptable, is at times country-specific. As such, during the early interactions with the regulator, this would be a key discussion topic.

4. Overall Conclusions

This report documents the European experience with the use of risk-informed inservice inspection (RI-ISI) technology. While this report documents the use of RI-ISI 19 units in seven countries, additional countries have or are conducting related work that will be captured in a future revision to this report.

In Europe, RI-ISI has been applied to a number of different piping systems (e.g. reactor coolant, condensate) as well as a wide spectrum of plant designs including boiling water reactors (GE, Asea-Atom), pressurized water reactors (CANDU, VVER, Westinghouse) and the RBMK design.

The conclusion from this report is that RI-ISI has and can be used to cost-effectively improve plant safety, reduce worker exposure and radwaste and allow plant operators to focus limited resources to the areas of greater benefit.

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LIST OF ABBREVIATIONS

ASME	American Society of Mechanical Engineers
BNRA	Bulgarian Nuclear Regulatory Agency
BWR	Boiling water reactor
CC	Crevice corrosion
CCDF	Conditional Core Damage Frequency
CCDP	Conditional Core Damage Probability
CDF	Core Damage Frequency
CLERF	Conditional Large Early Release Frequency
CLERP	Conditional Large Early Release Probability
CLRF	Conditional Large Release Frequency
CLRP	Conditional Large Release Probability
CNE	Canadian Nuclear Executives
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners Group Inc
CS	Carbon steel
CSA	Canadian Standards Association
CSN	Nuclear Safety Council (Spanish nuclear regulator)
E-C	Erosion-Cavitation
ECSCC	External chloride stress corrosion cracking
ENIQ	European Network for Inspection and Qualification
EPRI	Electric Power Research Institute
FAC	Flow-accelerated corrosion
FMEA	Failure Modes and Effects Analysis
FS	Flow Sensitive
HSS	High Safety Significance
IEfrequency	Initiating event frequency
IGSCC	Intergranular stress corrosion cracking
ISI	In-service inspection
LC	Localized Corrosion

LERF	Large Early Release Frequency
LERP	Large Early Release Probability
LOCA	Loss-of-coolant-accident
LRF	Large Release Frequency
LRP	Large Release Probability
LSS	Low Safety Significance
MCC	Main circulation circuit
MIC	Microbiologically influenced corrosion
NDE	Non-Destructive Evaluation
NDT	Non-Destructive Testing
NPP	Nuclear Power Plant
NUREG	US Nuclear Regulatory Commission Regulation
OPEX	Operating experience
P&IDs	Pipe and instrumentation drawings
PBF	Pressure Boundary Failure
PIT	Pitting
POD	Probability of detection
POS	Plant operational states
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PT	Penetrant testing
PWR	Pressurized water reactor
PWSCC	Primary water stress corrosion cracking
RAW	Risk achievement worth
RCPB	Reactor Coolant Pressure Boundary
RIBA	Risk-Informed Approach for In-Service Inspection of NPP Components
RI-ISI	Risk-informed in-service inspection
RPV	Reactor Pressure Vessel
RRW	Risk Reduction Worth
RWRP	Raw water reliability program
SCC	Stress corrosion cracking
SDC	Shutdown cooling

SRM	Structural Reliability Modelling
SSC	Systems, structures and components (SCC)
TASCS	Thermal stratification cycling and striping
TF	Thermal fatigue
TGSCC	Transgranular stress corrosion cracking
TT	Thermal transient
UNESA	Assembly of Spanish NPP operators
US NRC	US Nuclear Regulatory Commission
UT	Ultrasonic testing
VT	Visual testing
VVER	Russian-type PWR
WENRA	Western European Nuclear Regulators Association

ABOUT NUGENIA and ENIQ

NUGENIA is an international non-profit association under Belgian law established in 2011. Dedicated to the research and development of nuclear fission technologies, with a focus on Generation II & III nuclear plants, it provides scientific and technical basis to the community by initiating and supporting international R&D projects and programmes. The Association gathers member organisations from industry, research, safety organisations and academia.

The activities of NUGENIA cover plant safety & risk assessment, severe accidents, reactor operation, integrity assessment and ageing of systems, structures & components, development of fuel, waste & spent fuel management & reactor decommissioning, innovative light water reactor design & technologies, harmonisation and in-service inspection & their qualification.

The European Network for Inspection and Qualification (ENIQ) is a utility driven network working mainly in the areas of qualification of non-destructive testing (NDT) systems and risk-informed in-service inspection for nuclear power plants. Since its establishment in 1992 ENIQ has issued nearly 50 documents. Among them are the “European Methodology for Qualification of Non-Destructive Testing” and the “European Framework Document for Risk-Informed In-Service Inspection”. ENIQ is recognised as one of the main contributors to today’s global qualification guidelines for in-service inspection. ENIQ became Technical Area 8 of NUGENIA in 2012.

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