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Improvement of Effectiveness of In-Service Inspection in Nuclear Power Plants

IAEA

International Atomic Energy Agency

IMPROVEMENT OF EFFECTIVENESS OF IN-SERVICE INSPECTION IN NUCLEAR POWER PLANTS

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FOREWORD

Nuclear power plants require regular inspection and maintenance in order to maintain adequate safety and efficiency standards. In the nuclear energy field, even a minor critical component defect can cause undesirable production losses or potentially unacceptable reductions in safety. To address this, in-service inspection (ISI) techniques can be adopted as a means of verification of structural integrity and of safety relevant structures and components. ISI supports preventive failure analysis, maximizes safety and productivity, reduces production losses and improves the overall performance of nuclear power plants. Typically, ISIs are carried out periodically during maintenance outages at nuclear power plants.

During the operational lifetime of a nuclear power plant, especially as life extensions become more commonplace, the plant components will be exposed to influences whose individual or combined effect cannot be fully pre-empted with the level of accuracy needed to maintain nuclear safety standards. To date, the most important effects stem from component stress/strain, extreme temperatures, irradiation, hydrogen absorption, corrosive attack, vibration and fretting, all of which are dependent on the previously mentioned factors in regard to operational time and operating history. These influences may result in embrittlement, fatigue, formation and/or growth of cracks, corrosion or other changes in material properties commonly known as ageing phenomena, which reduce the overall safety of the plant.

Non-destructive examination (NDE), as a part of the ISI, is an important and essential action to ensure component integrity and avoid potential failures, and thus is a key tool in the management of nuclear power plant safety and productivity over the lifetime of a plant. Traditionally, inspection sites and volumes were based on prescribed codes and regulations. One of the identified ways to enhance ISI effectiveness is to optimize the inspections by selecting sites where potential occurrence of failure is relatively more probable and/or could lead to a more severe consequence, whereas the mitigation action as a result of a proper NDE, effectively reduces the probability of failure, thereby improving nuclear power plant safety and performance.

The ISI programme should promote maintenance requirements for nuclear power plants while they are in operation and when they are returned to service following plant outages and/or repair or replacement activities. These ISI activities require a mandatory programme of scheduled examinations and inspections to ensure evidence of adequate safety measures.

This publication represents a consensus among experts drawn from the IAEA and global practitioners to compile a set of common or good individual practices for use at nuclear power plants to improve ISI effectiveness. It sets forth a number of strategies and practices for improving the effectiveness of ISI and investigates the role of ISI in maintaining or improving safety at plants and the relationship of ISI improvement to cost. Strategies for improving ISI effectiveness are discussed, with consideration given to the entire framework of ISI, including effective selection of the proper inspection scope, inspection interval and NDE efficiency.

This publication is an update of the publication IAEA-TECDOC-1400, Improvement of In-service Inspection in Nuclear Power Plants, which was published in 2004. It takes into account new techniques and technologies for qualification processes for methodologies, equipment, procedures and personnel as well human factors in NDE inspections.

The IAEA wishes to thank all the experts involved and their Member States for their contributions. The IAEA officer responsible for the preparation of this publication was H. Varjonen of the Division of Nuclear Power.

CONTENTS

1. INTRODUCTION

The primary objective of an integrated life cycle management programme is to enable NPPs to compete successfully, without compromise of plant safety. Ideally, In–service inspection (ISI) must be done throughout the service life of a plant, to eventually facilitate life extension of NPPs and eventual decommissioning through improved engineering, technological, economic and managerial actions. The IAEA'S Technical Working Group on Nuclear Power Plant Life Management and other advisory groups, nominated by select IAEA Member States, provide recommendations of high priority needs of Member States in this area.

Several IAEA Member States that operate nuclear power programmes or aim to expand nuclear energy capabilities lend high priority to extending the operation of their NPPs beyond planned lifetime. During the operational lifetime of a nuclear power plant, the operator is required to examine the systems, structures and components (SSCs) for possible deterioration so as to determine whether components and materials acceptable to continue safe operation or whether remedial measures need to be taken during operation[1, 2]. In–service inspection, however, provides the systematic framework of these examinations; it is a means of verification for structural integrity of safety relevant SSCs and to conduct analysis of preventive failure and help maximize safety and productivity, all while reducing production losses and improving the performance of nuclear power plants.

1.1. BACKGROUND

An effective ISI programme ensures 1) the safety of the plant is not adversely affected following commencement of operation, and 2) the levels of reliability and availability of all plant SSCs remain in accordance with the basis, assumptions and intent of the design.

Outlined below are the main aspects and issues to be considered when developing and improving ISI effectiveness in NPPs. It also provides status of ISI practices in NPPs in select IAEA Member States, evaluating criteria for effective traditional ISI and introducing the concept of risk–informed in–service inspection (RI-ISI). RI-ISI uses an independent evaluation of consequence and failure probability. At present, RI-ISI is mainly used for pipes and, assesses the piping segments for potential damage, degradation and consequence of pipe failure.

1.2. OBJECTIVES

The objectives of this publication are to:

- Discuss and evaluate the status of ISI and its evolution in nuclear power plants in IAEA Member States;
- Discuss and evaluate the criteria for effective ISI and the constitutive elements, including human factors;
- Discuss and evaluate the implication of introducing RI-ISI concepts in operational NPPs;
- Generate a common recommendation as to how to benefit from RI-ISI concepts for further development and, possibly, improvement of ISI effectiveness;
- Discuss and evaluate the implication of inspection–, qualification or performance demonstration as part of ISI;

- To provide awareness of different qualification / performance demonstration methodologies in Member States;

The specific concept of this publication is to treat three key issues, including 1) inspection qualification / performance demonstration of ISI and RI-ISI, 2) to define fields and the logistics of the ISI and RI-ISI competing and complementary attributes, and 3) to assess the impact of the aforementioned concepts on NPPs life management.

The intention of this publication is to the disseminate information and increase harmonization in the areas of ISI effectiveness, ISI practices and ISI codes and standards, thereby achieving a higher level of safety and reliability in the operation of nuclear power plants around the world. The publication presents a rationale for ISI, accounting for risk informed assessment methodology (to be used for all nuclear power plants irrespective of their type or age). The benefit of the application of this methodology is seen by the enhanced safety and the increased competitiveness of the operation of nuclear power plants.

This publication is intended for all institutions and individuals involved in ISI activities and supporting organisations such as:

- Utilities / owners / operating organizations;
- Regulatory bodies;
- Qualification bodies;
- Research and academic organizations;
- Technical support organizations (TSOs);
- System vendors;
- ISI vendors; and
- Insurance companies.

In addition, other entities interacting with ISI and working in the following areas are also addressed:

- Structural integrity and component reliability;
- Maintenance, repair and replacement;
- Provision of parts and components for nuclear power plants; and
- Plant operation.

This publication is also suitable for providing briefing material for those making decisions regarding ISI, technically and financially.

1.3. SCOPE

The scope of this publication encompasses non-destructive examination (NDE) itself as a fundamental part of ISI, with regard to its feasibility and capability and its proof by qualification at NPPs. It also discusses technical disciplines which interact with NDE, as well as models and criteria that do not NDE, but influence the process of risk informed assessment on the whole:

- NDE methodology, potential, performance and effectiveness;
- Human factors related to NDE efficiency;
- Optimization of NDE methodology;
- NDE system qualification;

- ISI results and experience;
- Design, fabrication and operation;
- Operational experience;
- Code requirements;
- Assessment methodologies;
- Integrated structural integrity assessment;
- Assessment of the probability of occurance of failure (without considering ISI);
- Safety and / or cost consequences of failure including defence–in–depth consideration; and
- Code classification of components or component areas.

1.4. STRUCTURE

The publication is divided into eight main sections, where Section two provides an overview of the ISI programme and its requirements and different codes for use in nuclear power plants. Section three describes principles, methodologies and different codes. Section four focuses on ISI implementation and links to other plants programmes. Sections five and six describe ISI effectiveness and its correlation between the structural integrity assessment and mitigation, repair and replacement action, also touching upon human factors related to nuclear safety and reliability of NDE. Section seven focuses on risk–informed inspection from different perspectives and methodologies. Section eight concentrates on the inspection qualification process, performance demonstration and different roles and responsibilities through the qualification process.

Two safety standards that already exist in this field are IAEA Safety Standard Series No. SSR-2/2 entitled Safety of Nuclear Power Plants: Commissioning and Operation [1], which establishes, inter alia, the requirements about ISI that must be met to ensure safety; and IAEA Safety Standard Series No. NS-G-2.6 Maintenance, Surveillance and In–Service Inspection in Nuclear Power Plants, which recommends, actions, conditions and procedures for ISI to meet safety requirements [2].

Additionally, IAEA Nuclear Energy Series No. NP-T-3.1 Risk–Informed In–service Inspection of Piping System of Nuclear Power Plants: Process, Status, Issues and Development, provide inter alia guidance for risk–informed ISI technologies and its implementing methodologies [3].

2. ISI PROGRAMME AND REQUIREMENTS

2.1. BRIEF REVIEW OF PAST IN-SERVICE INSPECTION

The ISI is performed through destructive testing and NDE to detect flaws and other degradations that may exist particularly on the weld and base metal areas of the reactor pressure vessel (RPV), tubes and pipes, heat exchangers, other pressure vessels and components, key bolts and supports, which all bear direct and indirect implications on the safety of the NPP. These inspections identify appropriate repair or replacement measures to be taken to prevent potential future failures, so as to protect the public from possible releases of radioactivity and to support the efficient operation and maintenance of nuclear power plants.

It is appropriate to review the current status of ISI programmes and practices at nuclear power plants before discussing considerations and recommendations for ISI optimization and effectiveness. Typically, nuclear power plants around the world have established and implemented ISI programmes based on particular national regulations and technical standards applied to the design and operation of their nuclear power plants.

Most of the ISI programmes have the following characteristics:

- A well defined safety classification of the components;
- ISI intervals managed by a fixed periodicity, varying from ISI csode to code requirements;
- ISI scope encompasses all weldments of the RPV, within a fixed interval, and selected welds and other areas of the components classified as safety-related, according to information at that time.

Basic concepts of an ISI plan include randomly sampled, damage oriented and risk informed methodologies; some plants use any combination of these three above mentioned characteristics to establish an ISI or enhance it.

- The *randomly sampled* approach The American Society of Mechanical Engineers (ASME approach) is based on a sampling percentage of inspected welds, e.g. 25% for class 1 and 7.5% for class 2 piping. In this case, the ISI performance of the target detectable flaws is based on standard codified values (such as 10% of the wall thickness).
- The *damage oriented* approach considers only potential degradationmechanism. That is, the priority of inspection is on those components most susceptible to the degradation mechanism. Using a damage tolerance analysis approach, the flaw target can vary and not be fixed of 10% of wall thickness. Flow accelerated corrosion (FAC) is an example of this type of approach.
- The *consequence of a break* (or cost); using some component reliability to gauge potential damages and the consequences of a leak or break on overall plant safety.

With regard to the NDE, a prescriptive code regulates the sensitivity and verification of the calibration, recording and acceptance level via the application of amplitude criteria when using the ultrasonic testing (UT) method. The recording level is related to the reflectivity of the discontinuities detected in inspected components and sizing generally performed using the amplitude of echo dynamics from discontinuities.

International programmes such as the Programme for Inspection of Steel Components (PISC, previously known as the Plate Inspection Steering Committee), provided considerable assistance to objectively assess the reliability of NDE for different components by demonstrating the lack of capability of ‘traditional‘ methods and the benefits of advanced NDE practices. The results of these programmes triggered two major activities in conjunction with each other:

- Implementation of a more advanced NDE methodology;
- Implementation of systematic processes to prove the capability of the methodology to be employed.

These efforts were started in the early 1980s in the United Kingdom (called inspection validation) and subsequently in the USA (called performance demonstration according to The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI and Appendix VIII) [4].

The third phase of the PISC programme, Action 8 concerning Support to Codes and Standards, was founded in the late 1980s and concentrated on the complexity of ISI qualification. This decision was made due to NDE qualification being regarded as the true heritage of PISC, as it gathered all important results of the PISC. It also offered a major chance of the harmonization of NDE practices in various countries through generally valid principles of assessing the effectiveness of these practices.

Towards the end of the PISC, the European Bureau for Inspection Validation (EBIV) later became the European Network for Inspection Qualification (ENIQ), taking over the responsibilities of PISC. Shortly thereafter, the ENIQ drafted the first issue of the European Methodology for Qualification of Non–destructive Testing, followed by the second and third issue that is still valid today [5]. This initiative increased its pace and scope of work, as NDE qualification had also become an issue in France and Sweden.

In the meantime, the Electric Power Research Institute (EPRI) assembled most of the utilities of the performance demonstration initiative (PDI), rapidly progressing the qualification process according to ASME Code requirements.

The IAEA then merged the expertise of water cooled/water moderated power reactor (WWER) operating countries and Western Europe as well as the USA, generating the Methodology for Qualification of ISI Systems for WWER nuclear power plants [6]. This publication combined ENIQ and ASME approaches to provide a consistent, practical strategy for WWERs.

At present, in the USA and most ENIQ member countries, inspection qualifications were established, and more advanced NDE practices have received qualification or are in the process or planning stages of the qualification.

Member countries of the Canada deuterium–uranium reactor (CANDU) Owners Group are using qualification to meet Canadian Standards Association (CSA) N 285.4, where ENIQ is adopted as the basis for qualification methodology, and is supplemented by ASME Section XI for mock–ups defect distributions, procedure and personnel qualifications.

Some countries following the ASME Boiler & Pressure Vessel Code rule established their own performance demonstration and qualification system.

Recently, the progress achieved in terms of ISI effectiveness and optimization is under critical review, especially in the light of the following questions:

- Has the qualification process of NDE systems and the improvements of the achieved methodology led to NDE optimization for its own purposes?
- What realistic input, from empirical and engineering aspects, has been maintained in the process of establishing qualification e.g. the determination of the flaws to be used in the qualification?
- Is there consistency between the performance during the qualification test and the actual onsite examination performance?
- Are NDE experts aware of the need to interface with experts in other disciplines and vice versa, when ISI optimization strategies are being determined?
- How integrated models such as RI-ISI have affected implementation, which may give some relevant input to set the targets for ISI optimization?
- Has the long term inspection programme been established to the satisfaction of all relevant code requirements, encompassing all the equipment subject to inspection, and

have all the scheduled inspections performed faithfully as per the long term inspection programme?

As of 2017, the most widely used techniques for ISI include phased array ultrasonic testing (PA UT) and multi–array Eddy current testing (ECT), and the application of these technologies is expected to accelerate in future.

Recently, Nuclear Generation II & III Association (NUGENIA) Technical Area 8 (TA8 / ENIQ) and the sub-area on qualification (SAQ) launched a number of projects to improve ENIQ qualification methodology. A prominent methodology is the mutual recognition (MUREC) to facilitate qualifications between countries. Another project is developing a new recommended practice (RP) on Inspection Procedure. SAQ also launched a comprehensive study on the performance of computed and digital radiography, under the name COMRAD. The objective of the study is to identify the essential parameters that affect the result of computed radiography (CR) and digital radiography (DR).

Major technologies used in ISI are increasingly undergoing digitalization and computerization, and thus the massive data collected from the in–service inspections will allow more comprehensive understanding of ISI technology.

2.2. CODES AND STANDARDS

ISI requirements are usually defined and summarized in codes or standards, which are developed by engineering or industry associations, and are based on consensus of all stakeholders such as regulators, component manufacturers, operators, inspectors, insurance companies and researchers, among others. Each code is intended to be in line with the requirements of the given country's nuclear safety regulation, and their use may be mandatory by legislation.

The widely known ISI codes, e.g. those produced in the United States of America, France, Germany or the Russian Federation contain many similarities, but there are also some differences. A number of these similarities and differences are explored below.

ISI programmes typically follow ASME, Section XI, Rules for In-service Inspection of Nuclear Power Plant Components, except in the case of CANDU reactors, which follow the CSA, N285.4-05, and Periodic Inspection of CANDU Nuclear Power Plant Components. In the USA, 10 CFR 50.55a (Code of Federal Regulations) [7] In–service Inspection Requirement provides requirements that USA utilities need to follow during their in–service inspections. These codes stipulate specifications that must be applied for in–service inspections to be included in the final safety analysis report (FSAR), which must be submitted upon completion of nuclear power plant construction.

Other codes and standards that are followed and referred to during ISI include the following:

- ASME Code Section V: Non-destructive examination;
- ISO 9712, Non–destructive testing–Qualification and certification of NDT personnel
- The American Society for Nondestructive Testing, SNT-TC-1A, ASNT recommended practice: Personnel Qualification and Certification in Non–destructive testing;
- The American Society for Nondestructive Testing, SNT-CP-189: ASNT Standards for Qualification and Certification of Non-destructive testing personnel.

2.2.1. The American Society of Mechanical Engineer, ASME Code Section XI

ASME Section XI, Rules for In–service Inspection of Nuclear Power Plant Components, constitutes the ISI requirements for plant systems and components manufactured in accordance with ASME Code Section III. Since its first edition in 1970, numerous updates have been made; most recently, the 2015 edition was updated and published. The ASME Section XI is spread across three divisions: division 1 is for boiling or pressurized water reactor (PWR); division 2 is for gas–cooled reactor (GCR); and division 3 is written for liquid metal cooled reactor (LMCR) reactors. Each division contains subsections written for different parts and component classes, including the following key descriptions:

- *Areas subject to inspection*: Pressure boundary components and pipe welds are classified as class 1, 2, and 3; the components and welds are further divided into ISI sub-categories, based on weld material, geometry, stress level, applied load, environment, type of fabrication and other design criteria;
- *Provisions for accessibility and inspectability*: Appropriate testing space must be provided to allow access for the inspector, examiner and the inspection equipment as well as other testing conduct. This is one of the defined responsibilities of the plant owner;
- *Examination methods*: ISI inspection techniques are classified as visual examination, surface examination and volumetric examination, and testing procedures for each of the techniques are specified. When the application of examination techniques (as specified in the relevant code) is practically impossible or inadequate, the code allows the use of alternative examination methods;
- *Frequency of inspection*: Depending on the examination category, each component's inspection scope and frequency are specified in the examination table;
- *Record keeping and report requirements*: Requirements for storing and managing inspection documents are specified, including pre–service, in–service inspection records and repair history;
- *Repair requirements*: Repair or replacement requirements are defined based on the examination results.

2.2.2. Canadian Standards association, CSA code

The CSA 285.4 [8] was first published in 1978. Although its pressure boundary was designed in accordance with the ASME Code, the CANDU reactor does not follow ASME Section XI for ISI. The CSA N285.4 [8] approach is based on the application of fundamental principles of process–based risk management, as opposed to rule–based risk management. Due to the significantly large number of pressure retaining components in CANDU plants, as compared to LWRs, a risk–informed approach is used to establish the scope of the periodic in–service inspection document (PIPD). The overall structure of CSA 285.4 [8] is similar to that of the ASME Section XI [4] and defines inspection requirements for fuel channel pressure tubes, feeder pipes and steam generator tubes at CANDU plants. However, CSA 285.4 [8] stipulates that ISI inspection methods may be limited to non–destructive testing practices.

2.2.3. In-Service Inspection Rules for Mechanical Components of PWR Nuclear Islands, RSE–M code

In July 1990, the RSE–M Surveillance and Operation of PWR Mechanical Components Code were first published in France.

This code covers pressure-retaining components and their supports used in PWR nuclear islands, classified in safety classes 1, 2 and 3 of the safety analysis report for the nuclear power unit. It defines the rules applicable to maintenance operations in order to ensure the integrity and leak tightness of these components.

RSE–M rule classes 1, 2 and 3 apply to components of systems in safety classes 1, 2 and 3 respectively. For each main system, the licensee establishes the list of components falling within each RSE–M class. The rules stipulated in the RSE–M define requirements regarding operating documents such as procurement specifications, technical specifications, procedures, etc.

The licensee may need to supplement these requirements, in particular where component flaws are observed. Such supplements are needed when surveillance is required or where, in light of experience with similar components, it is thought that faults may be present.

2.2.4. The Nuclear Safety Standards Commission, KTA code

First published in 1982, the Part 4: In-service Inspections and Operational Monitoring of Components of the Primary Circuit in NPPs of the German Safety Standard KTA 3201.4 “Components of the Reactor Coolant Pressure Boundary of Light Water Reactors” [9] was issued. As its title indicates, it deals with the primary circuit components, i.e. mainly but not exclusively Class 1 components.

In the context of the standard, ISI is embedded in a comprehensive concept of ensuring component integrity during plant operation. ISI is a representative area to monitor the presumed consequences of potential operational damage mechanisms. The standard provides for extended inspection intervals of NDE for the various components; where the inspection interval is typically 5 years.

As for evaluation of the NDE results, it is suggested to draft a decision-making plan if indications are found. Acceptance levels are based on equivalent reflectors. If indications exceed the acceptance level, they are compared with the results of the previous ISI. By comparing measured values, it is then decided whether a first occurrence of the indication has taken place or whether an existing indication has grown larger. If discovered that it is a first occurrence of an indication or an existing indication has grown, then the cause must be determined and a safety analysis performed. The safety analysis may, for instance, comprise stress analysis, fracture mechanics evaluation, laboratory experiments, checks on similar components in the case of indication of systematic defects, and the evaluation of experience gained from other plants. Results of the cause determination and the safety analysis are decisive regarding the specification of the acceptance level, i.e. the decision whether or not a flaw may be left as it is.

2.2.5. Federal Nuclear and Regulatory Authority of Russia, PNAE G-7-008, 010 and NP-089-15 codes

The Russian and the former Soviet Union codes and standards (normative technical documents) maintain a separate structure, methodology of preparation, acceptance and validity in comparison with codes and standards of other countries. These documents were issued in two waves: the first series was released in the early 1970s, and the second, upgraded series from 1987–1990 [10]. No periodic upgrading / revisions was foreseen and realized since that time. The NPPs with WWER-440 reactors were practically designed in accordance with the first set of the documents; the Nuclear power plants with WWER-1000 reactors were also designed in accordance with the first set of documents, even though limited improvements were incorporated in accordance with the second set.

The normative technical documents contain requirements for the design of, manufacturing, commissioning and operation of NPPs, mainly from nuclear safety point of view. No document, however, deals with the evaluation of integrity of pressurized components and piping during operation. Consequently, no standalone Russian ISI code exists and this situation held true until March 2016,. The ref. [10] and [11] contains statements of inspection during operation, determining both the group of components to be inspected and the inspection intervals. The ref. [12] and [13] describes the NDE methods and the acceptance criteria. The preamble of the latter documents, however, states that the documents were not applicable to the operation period. The UT acceptance criteria consequently are given in equivalent reflectors.

Russian rules did not contain any requirements, procedures etc. for evaluation of behaviour of components and piping during the operation period in support of ISI evaluations, and so a document within the EU 5th Framework Programme on ISI was prepared by experts from Czech Republic, Slovak Republic, Hungary, Bulgaria and Finland entitled VERLIFE "Unified Procedure for Lifetime Assessment of Components and Piping in WWER type NPPs during Operation". It is in consistent with Russian normative technical documents for design and manufacturing, and some approaches applied for PWRs were incorporated. VERLIFE is now applied by some WWER operating countries. Other countries, including Finland and Hungary, follow ASME Section XI rules.

In early 2016, the Russian nuclear regulator Rostechnadzor, issued a new federal document for ISI, including: Rules for in–service inspection of base metal, welds and cladding of equipment, pipelines and other elements of NPPs, NP–084–15 [14]. This publication represents a new wave among Russian normative technical documents. Around that time, the document [10] was replaced by a new one: Rules for arrangement and safe operation of equipment and piping of nuclear power installations, NP–089–15 [15].

According to the new standard NP–084–15[14], the interval of ISI was made more frequent – from the previous 4 years to 10 years (except the beginning and the end of the design life time). This standard allows distributing the examinations within the interval. Another development compared to PNAE G–7–010 [13] is the evaluation criteria which deals with acceptance levels of flaws. For some situations, flaw sizing is required, while in other situations, flaws are determined and compared to equivalent reflector areas. Reactor specific or even component area specific requirements are described in the annexes of the standard. In addition, one annex briefly describes the principles of RI-ISI, while another gives a detailed flow chart of the decision-making process based on ISI results. There are some similarities with ASME Section XI [4]; however, the content of this standard is not as detailed as ASME Section XI [4].

2.2.6. Korea Electric Power Industry Code

KEPIC published its first edition in 1995, developed on prevailing ASME codes and standards which are applicable to the electric power industry in Korea. KEPIC is applied to the construction of Korean NPPs since 1997, as per the endorsement of Korean government. It is also applied to the construction and operation of newly constructing NPPs in Korea following that time. KEPIC–MI "In–service Inspection" is a code for in–service inspection of nuclear power plants and the overall structure of KEPIC–MI is similar to that of the inspection requirements in ASME Section XI. It provides in–service inspection requirements for the safety–related components and describes guidelines for application of inspection standards, as well as a flaw evaluation scheme.

2.2.7. Japan Society of Mechanical Engineers Code

JSME published its first edition of a fitness–for–service (FFS) Code for NPPs in 2000, which provided rules of flaw evaluation for Class 1 pressure vessels and piping, referring to ASME Code Section XI. The second edition of the FFS code included rules for ISI requirements and was published in 2002. Individual inspection rules were prescribed for specific structures, such as the core shroud and shroud support for boiling water reactor (CONFREND) plants, in consideration of ageing degradation by stress corrosion or cracking (SCC). JSME published the third edition of the FFS Code in 2005, including requirements for repair and replacement methods and extended the scope of specific inspection rules for structures other than the BWR core shroud and shroud support.

2.3. EXAMPLES OF COUNTRY SPECIFIC REGULATIONS AND OTHER REQUIREMENTS

2.3.1. Regulatory endorsement of codes and standards

2.3.1.1. SSMFS 2008:13, The Swedish Radiation Safety Authority's regulations concerning mechanical components at certain nuclear facilities

The first regulation currently is known as "The Swedish Radiation Safety Authority's regulations concerning mechanical components at certain nuclear facilities" was published in 1978 as "*Regulations concerning inspection of components and system, significant for pressure boundary in NPP"*. In 1981, a new code was released, titled *"Code for periodic ISI in NPP"* and was approved by the Swedish Nuclear Regulator (SKI). The updated code was more detailed then the former code and is commonly known as AGÅB. Yet, a new regulation was released in 1984, called *"Regulation for pressure vessel safety in NPP"* (FTK) was considered a highly detailed regulation. It also handled the SCC issue and was influenced by the ASME Code. The next release of this regulation was titled "*Regulation for Pressure Vessel Safety in NPP and Facilities for Spent Fuel"* (FTKA) was released in 1988. In the FTKA, the first concept of a 'Consequence and Damage Index' was introduced.

The first SKIFS was issued 1995–01–01, known as SKIFS 1994:1 "*Regulations of mechanical components in nuclear installations"*. In SKIFS, a qualitative risk approach from three different control groups was introduced and the regulation was written at a detailed level. The next set of regulation was titled SKIFS 2000:2, which allowed the use of a probabilistic approach by two methods, either qualitative or quantitative. However, this regulation was not as detailed as the previous ones. SKIFS 2005:2 was issued in beginning of 2006 and the most important change (compared to previous regulations) was that inspection programmes included clauses related to accessible tension cables and sealing plates. The latest version of the regulation, SSMFS 2008:13, was issued the 19 of December 2008. A new version is currently being developed and it will probably be issued sometime in the 2018–2019 timeframe.

This Swedish regulation applies to the design, construction and in-service inspection of mechanical components belonging to the primary system or containment barrier, or as part of safety, operating and auxiliary systems of the following types of nuclear facilities:

- Nuclear power reactors;
- Reactors for research or materials testing;
- Facilities for the manufacture of uranium pellets and nuclear fuel assemblies; and
- Facilities for storage or other handling of spent nuclear fuel.

The regulations apply to technical and administrative measures and supplement the provisions contained in the Swedish Radiation Safety Authority's regulations (SSMFS 2008:1) concerning safety in nuclear facilities.

2.3.1.2. 10CFR 50.55a Codes and standards

With the launch of the US Nuclear Regulatory Commission (USNRC), codes and standards development became a major activity to ensure the safety of NPPs. Basic safety standards for NPPs were introduced to the 10 CFR part 50 by the NRC, which effectively became a federal regulation governing nuclear power plants.

10CFR 50.55a [7] requires the application of ASME Section XI and plant technical specifications to the operation of the ISI programme. 10CFR50 Appendix B (Quality Assurance Criteria for nuclear power plant & Fuel Reprocessing Plant–1970) [7] was subsequently established, and the background and structure of 10CFR50, Appendix B, is as follows:

- *Background:* As many nuclear power plants were planned, designed and constructed, a variety of problematic issues surfaced in the design and construction process. Therefore, a need for establishing an effective and systemic quality assurance plan also emerged to adequately identify the root cause of the problem and its resolution and to seek out latent problems at an earlier stage to allow for correction;
- *Basis:* 10CFR50 [7], Appendix B (Quality Assurance Criteria) was established based on the experience of US nuclear power plants under design or in the construction phase, or in operation, as well as the quality assurance (QA) programmes of the US Department of Defence (DOD) and the National Aeronautics and Space Administration (NASA);
- *Application:* 10CFR50 [7], Appendix B is applied to the design, manufacture and construction of structures, systems and components of nuclear power plants to prevent or minimize an event that could affect public safety.

2.3.2. Final safety analysis report

The FSAR describes matters to which the nuclear power plant operators, specified in the regulatory licensing documents must adhere to and contain requirements for limiting conditions for operation and in–service inspection.

2.3.3. Regulatory interactions

There are several types of communication that regulatory bodies use to communicate with the nuclear industry. For example, the USNRC uses Bulletins; Information Notices; and Generic Letters. These publications are written in response to an actual or latent abnormal operating event at NPPs, which are then reflected on the plant ISI plan.

Bulletins are used to transmit information or to request a specified action or response for significant issues related to safety and environmental issues. Bulletins require a written response by the addressees.

Information Notices are used to inform the nuclear industry of events that is related to safety or environmental issues. Sometimes USNRC conducts an additional investigation into the event and issues a Bulletin.

Generic Letters (GL) are published to notify the licensees of regulatory actions and schedules. GL is used to convey or request information from the licensees to clarify NRC polices. At times, Generic Letters is used to collect information to postpone or revoke a license.

2.3.4. Augmented inspection programmes—degradation specific inspections

In addition to the deterministic inspection programmes discussed above, many plant sites undergo additional inspection activities by plant operators. Some of these inspections result from a commitment to the regulatory body, while others are a result of plant specific experiences and good practice initiatives. These other inspection programmes based on experience and good practice initiatives have names such as 'augmented' or 'owner defined' programmes. Additionally, in some countries, these augmented inspections have been incorporated directly into the deterministic ISI programme while in other countries each inspection programme remains a separate programme onto itself.

2.3.4.1. Break exclusion region

Typical general design criteria for nuclear power plants requires that structures, systems, and components important to safety be designed to accommodate the effects of postulated accidents, including appropriate protection against the dynamic and environmental effects of postulated pipe ruptures [3].

Various 'regulation and standards' development bodies have issued documents that provide criteria for implementing the above requirement. These include the scope of applicable systems, postulation of break locations, methods for analysing pipe whip forces and displacements, design of rupture restraints and methods for evaluating the integrity of components subjected to the pipe rupture loads.

To determine where locations at which breaks are postulated in high–energy piping, the guidance provides special exclusion rules (e.g., containment penetration areas). These rules recognize that these areas may require extra protection (e.g., to ensure the integrity of the containment and the operability of the isolation valves). The rules provide the option of not specifying breaks in these regions, so that pipe break mitigation devices (e.g., pipe whip restraints) need not be installed in these areas.

Requirements for not specifying breaks in these regions may include special design requirements (e.g., minimize the length of piping, minimize the number of welded attachments) and additional inspections of welds in the plant area of concern. These 'additional' inspections are typically made part of the ISI plan and are identified as 'augmented' inspections [3].

In addition to having portions of the system classified as safety related, power conversion systems may be important to safety for other reasons such as their impact on reliable plant operation and personal safety.

In response to these concerns, a number of plants implemented programmatic activities to assure reliable system operation. These programmatic activities include developing a more robust understanding of system operating conditions that can adversely impact pressure boundary reliability (e.g. steam quality, corrosion potential, material selection), monitoring system and operational changes (e.g. throttling practices, operational changes, system modifications) as well as updating programmatic activities in response to more significant plant changes such as power up rates and extended fuel cycles.

Plant responses to these impacts include revised system operating practices, changes to system operation, strategic replacement of susceptible components with more resistant materials (e.g. chrome–molybdenum) and conducting inspections to confirm and / or calibrate predicted wear rates.

2.3.4.2. Localized and flow accelerated corrosion

Typical general design criteria for nuclear power plants require that provisions are installed for a system or systems that transfer heat from structures, systems, and components important to safety to an ultimate heat sink (e.g. service water systems). Per these design criteria, such systems may also allow for appropriate periodic inspection of important components to assure the integrity and capability of the system throughout plant lifetime. In addition, nuclear power plant facilities must meet corrective action programme requirements, as defined in their quality assurance programmes.

In some cases, operating experience with these systems has demonstrated that these systems or portion of the system are susceptible to localized corrosion, such as pitting and / or microbiologically influenced corrosion. The likelihood of degradation and accompanying degradation rates is a result of multiple factors including piping material, operating temperatures, flow conditions (stagnant, infrequent), water quality, water treatment (e.g. biocides, corrosion inhibitors). This experience has before resulted in the need for periodic maintenance, refurbishment, lining of components (inner or outer surfaces) as well as implementation of a visual and volumetric inspection programme to continue to assure reliable system operation.

3. PRINCIPLES, METHODOLOGIES AND ROLES

3.1. PRINCIPLES OF ISI

General ISI programme definition, ISI inspection scope and requirements are described in the IAEA publication NS-G-2.6 [2].

ISI is essential to ensure the safety and reliability of nuclear power plants following commencement of commercial operation. Those SSCs subject to ISI are selected based on various design considerations and regulatory requirements of the plant. The frequency of ISI is also determined in the ISI code. While the German code determines 5 years for the inspection interval regarding safety class 1 components, the ASME Code, the French and the new Russian codes have a 10–year interval. The 10–year interval was chosen based on historical failure rate data between non-nuclear steam power and petrochemical plant systems. Some of WWER operating countries either from the beginning (Finland) or from a certain stage of the operation introduced a longer ISI interval, as opposed to the 4–year interval determined by former Russian codes. Some countries, for example Sweden and UK use damage tolerance analysis to determine inspection interval.

ISI is performed during the plant refuelling and maintenance outages, using NDE methods adapted for service–induced flaws. These types of NDE methods differ from manufacturing inspections, due to the flaw types that could be detected. However, advanced NDE methods may implement during consecutive ISI instead of previous NDE method that was applied in PSI or in early stages of ISI.

ISI / NDE systems should be qualified in accordance to national regulations to prove its ability to provide reliable confirmation that a specific NDE–system is capable of achieving the required performance under real inspection conditions.

3.2. NDE METHODS FOR ISI

The three types of examinations used during ISI inspection are defined as visual, surface and volumetric examinations. Figure 1 illustrates, as an example the NDE methods used during ISI accordance with the Section XI of ASME code.

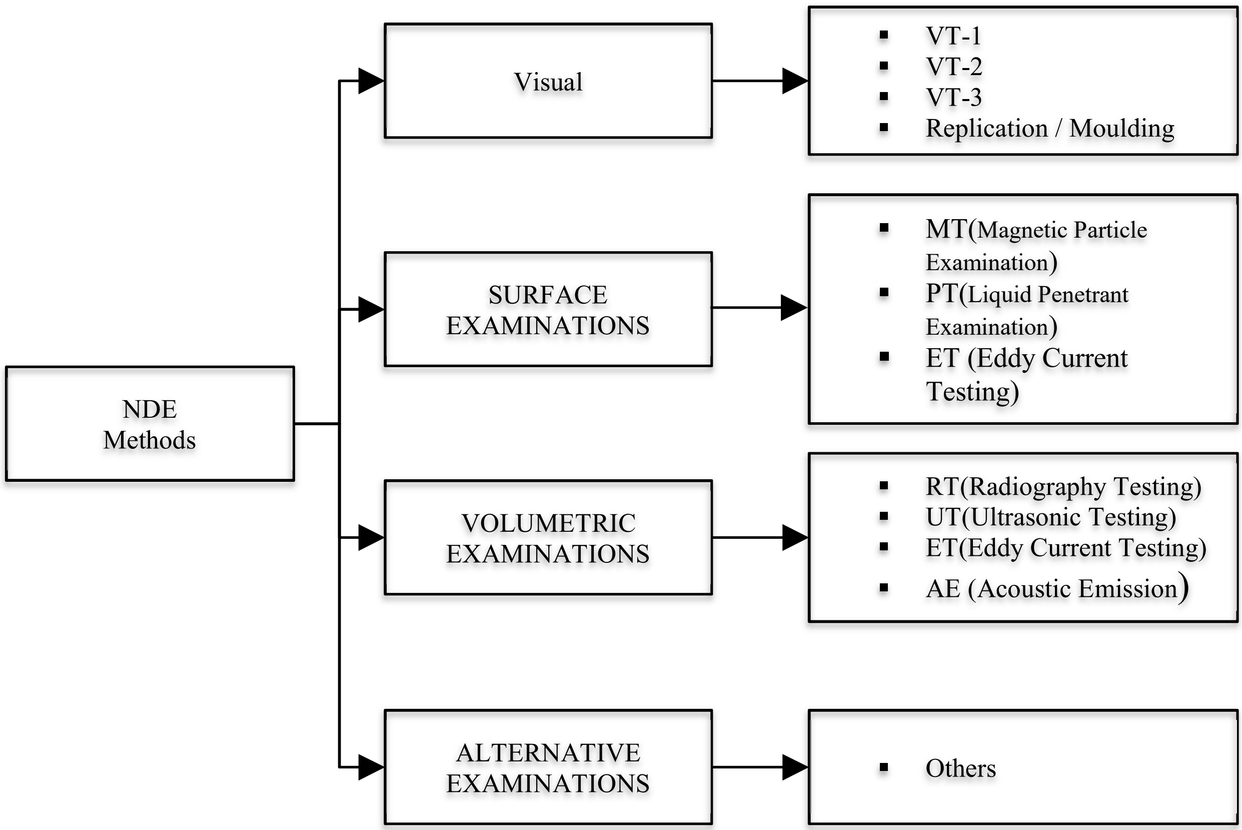

FIG. 1. NDE methods prescript by the ASME Code.

3.3. ROLES AND RESPONSIBILITIES

The roles and responsibilities of the main parties involved in nuclear power plants ISI are described in this section.

Reporting chains identifying the responsibilities of parties involved in a qualification process and in a RI-ISI process are further described in sections 3.4 and 3.5.

3.3.1. The owner / operating organization (licensee)

The overall responsibility for the ISI programme lies with the plant operator (licensee), these activities include:

- Classify plant SSCs into relevant code classes based on the regulator's classification standards, and to clarify or identify system boundaries;
- Develop and submit the ISI scope, schedule, and ISI summary reports to the regulatory body;
- Perform repairs or replacement according to approved procedures;
- Maintain records of NDE, testing, flaw evaluation, and repair & replacement;
- Ensure the implementation of the ISI programme; and
- Ensure adequate and qualified plant personnel involved in ISI programme.

It is recognized that utilities in different countries have different structures that vary in detail. Hence, what follows is a suggested management structure and interface that can serve as a requirement guide to implement an ISI programme.

The licensee also has responsibilities to ensure that an organization performing NDE (inspection organization) at its plant(s) has been previously qualified according to reference, legal system and regulatory requirements. The licensee has to provide such evidence, for the adequacy of examinations, to the regulatory body.

In European countries, the licensee typically provides input information for the qualification procedure that is prepared by the qualification body.

The licensee is also responsible in defining the technical specifications of the examinations required to be qualified and its required effectiveness for each particular case. Finally, the licensee has the responsibility to supervise the whole performance of examination activities, provide the logistics of the examination operations, evaluate the results and provide feedback from site examination either positive or negative to qualification body and regulatory body (as applicable).

3.3.2. The Regulatory Body

The regulatory body establishes ISI rules / regulations and guidelines consistent with national laws for the operation of nuclear facilities, monitors, inspects and evaluates compliance and ensure that the licensees fulfil the conditions of their site licenses. The regulatory body typically undertakes audits, periodic reviews and monitors the licensee's compliance with the safety requirements.

In some Member States, the Regulatory Body endorses the ISI programme and the results that affect nuclear power plant safety. If the reviews are not conducted by a regulatory body, then it is done by third party / authorized nuclear inspector (ANI). The Regulatory Body either defines or reviews the basic qualification requirements that must be met from a safety point of view. In some countries regulatory body also reviews and endorses (if applicable) the final qualification dossier.

3.3.3. The authorized nuclear inspector / independent third party organization

The ANI is responsible for verifying whether the materials used in NDE satisfy the applicable codes and their addenda, as well as code cases that the utility adheres to. ANI is also responsible for verifying that the required ISI and testing have been performed by examiners qualified in accordance with the code and qualification requirements. ANI verifies that inspection records are appropriately documented in accordance with documentation requirements. ANI checks whether the design specifications and reports are developed and maintained for repairs or replacement and whether the welding procedures meet applicable technical standards.

3.3.4. The qualification body

The qualification body is commercially and operationally an independent body that guarantees impartiality of ISI. This is normally done most of Member States, among other means, through a quality system approved by regulatory body, which guarantees its independence from commercial or operational influences and considerations. The qualification process managing, conducting, evaluating and certifying NDE systems is performed by this qualification body according to written qualification procedure and / or protocol.

The qualification body shall be competent, have adequate technical resources and facilities, and have an organization which ensures the quality of its work. Typically, there are three types of levels:

- *Type 1*: A qualification body which is an independent third party organization;
- *Type 2*: A qualification body which is an independent part of the utility's organization set up on a permanent or long-term basis; and
- *Type 3*: An ad hoc qualification body set up for a specific qualification.

The qualification body has the responsibility of developing detailed qualification procedure or protocol, and to identify and / or design the required test specimens. It is responsible for conducting, supervising and approval of the qualification in accordance to the agreed process. This includes assessment and approval of inspection procedure, technical justification (when applicable), invigilation of practical trials and evaluation of results, assembly of the qualification dossier and finally issuance and withdrawal of qualification certificates.

For example, the EPRI administers the PDI programme on behalf of all US licensees. This programme focuses on the implementation of the ASME Section XI, Appendix VIII Code requirements.

3.3.5. The inspection organization (vendor)

The inspection organization performs the examination with qualified NDE systems (equipment, procedure and personnel). Therefore, it is responsible for certifying its examination personnel, according to relevant national or international schemes. The inspection organization has the responsibility of developing the manipulator (if applicable) for inspection and qualification purposes, to develop an inspection procedure and technical justification, as applicable, for the proposed examination system.

The responsibility of the inspection organization is to participate in the qualification process in close cooperation with the qualification body, providing all the necessary information and documents according to the applicable qualification requirements.

The inspection organization is also responsible to provide feedback from site examination to the licensee.

3.4. THE QUALIFICATION PROCESS ORGANIZATIONS

The overall responsibility for the qualification lies with the plant operator (licensee), being responsible for the safety of the nuclear power plants. The main parties involved in the qualification process are as follows:

- The Regulatory Body;
- The Plant Operator (Licensee);
- The Inspection Organization (Vendor); and
- The Qualification Body (QB).

An example of a reporting chain of the responsible parties is shown in Figure 2.

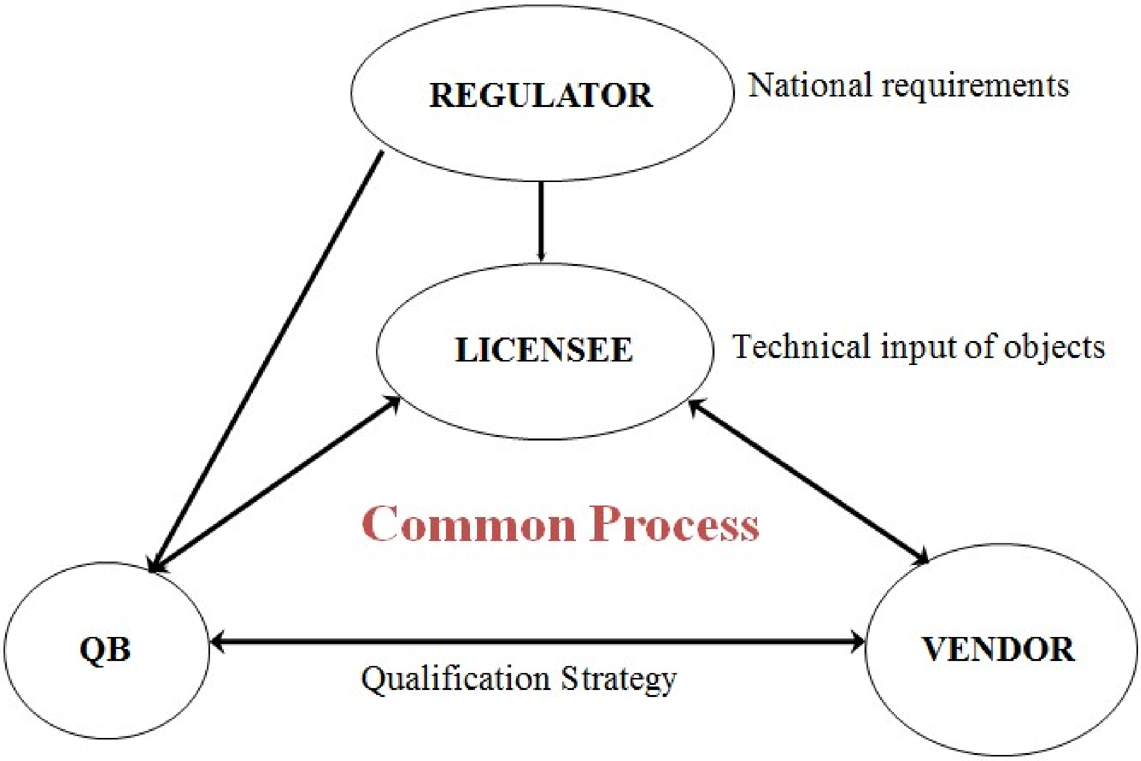

FIG. 2. Reporting chain, identifying the main responsibilities in a qualification process.

3.5. THE RI-ISI ORGANIZATION

Different setups for performing an RI-ISI project, depending on the national nuclear infrastructure, can be considered. The choice of management structure varies depending slightly of what methodology that has been chosen. In ASME XI appendix R, supplement 1 and 2, describes two different setups to develop an RI-ISI program. Below, the setup to perform an RI-ISI evaluation following the ENIQ approach is presented.

3.5.1. The outline of management structure

The main parties / personnel involved are as follows:

- The RI-ISI responsible person;
- The RI-ISI advisory panel;
- The RI-ISI team;
- The RI-ISI review panel;
- The inspection qualification team; and
- The regulatory body.

An example of a reporting chain of the responsible parties / personnel is shown in Figure 3.

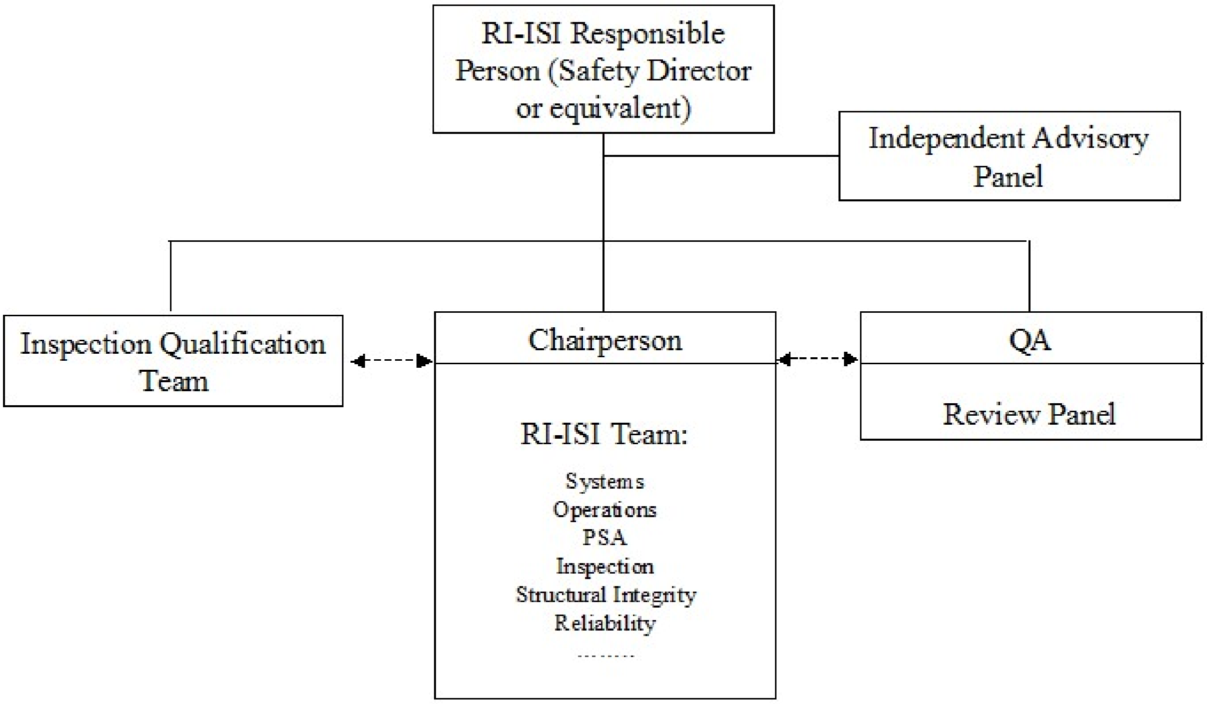

FIG. 3. Reporting chain identifying the main responsibilities in a RI-ISI process [16]

3.5.1.1. The RI-ISI responsible person

The RI-ISI responsible person oversees the setting of the boundary and scope of the RI-ISI programme. He or she is ultimately responsible for the acceptance of the final RI-ISI programme against the boundary and scope set, and for these reasons, the Responsible Person supposed to be a senior employee of the utility. Another way to address the setting for boundary and scope is that a proposal developed by the RI-ISI team is presented for an expert panel that will approve the final scope [16].

3.5.1.2. The RI-ISI independent advisory panel

The responsibility for the advisory panel is to counsel the RI–ISI responsible person with regard to any areas of the proposed RI-ISI programme that in its opinion is questionable, be it from the analytical modelling used, possible omissions, external considerations, etc. [16].

3.5.1.3. The RI-ISI team

The RI-ISI team should preferably be a multi-disciplinary team with different expertise, including the following expertise: quality, inspection, maintenance, design, materials, chemistry, stress analysis, systems engineers, probabilistic safety assessment (PSA), operations and safety.

"The RI-ISI team has the responsibility for developing the RI-ISI programme, including risk acceptance criteria and following the programme through to its implementation. It is responsible for coordinating the required effort into the utility, to produce the necessary documentation, compile the RI-ISI dossier and ensure that the relevant QA procedures are followed and records kept" [16].

When the new RI-ISI programme is developed, it is suitable to interact with the inspection qualification team to evaluate if the inspection locations could be inspected with available techniques and proper quality.

3.5.1.4. The RI-ISI review panel

"The purpose of the review panel is to provide an essential independent element in Quality Assurance process.

The Review Panel should contain experts in the relevant areas that are independent from those belonging to the RI-ISI team. Such experts could be from either inside or outside the utility. Their independence must be ensured in the sense that they are supposed to not have been at any stage involved in the generation of the basic probability of failure (POF) and consequence of failure (COF) data to be ranked" [16].

3.5.1.5. The inspection qualification team

"The inspection qualification team has the responsibility of advising the RI-ISI team with regard to the feasibility of achieving the specified outcomes from a proposed ISI programme. It ought to be clearly understood that the inspection qualification team cannot, at this time, guarantee that any subsequent inspection qualification against the specified requirements will be successful" [16].

3.5.1.6. The Regulatory Body

"The Regulatory Body typically undertakes audits, periodic reviews and monitors the licensee's compliance with the safety requirements. To these ends the Regulatory Body may wish to observe the development of any safety–driven RI-ISI programme. For instance, the Regulatory Body may wish to participate with the status of observer in the RI-ISI Advisory Body meetings [16].

3.6. ACCEPTANCE CRITERIA AND FLAW EVALUATION

3.6.1. Acceptance criteria

When ISI code usage was first developed, construction requirements were applied and, only later, were replaced with requirements appropriate for operating plants. From this time, the acceptance criteria were determined on the basis of fracture mechanics taking into account the detected flaw stability. The ISI acceptance criteria could be different in principle from that of used for component construction because their goal is to justify the component's fitness for service while the latter ones are quality control criteria.

In the ASME Code, e.g. the acceptance criteria are given in tabulated form. The tables contain the relations between flaw depth and wall thickness (a / t), flaw aspect ratio (a / 1) and relation to surface (Y). If the flaw size does not exceed the values in the relevant table of the code, the component is fit for service in the forthcoming ISI interval. If the flaw size however exceeds the criteria a fracture mechanics calculation can be performed to determine if sufficient safety margins exist.

3.6.2. Evaluation of flaw

When an indication is found during ISI examination, interpretation must be made to determine whether it is a relevant or non-relevant indication. When an indication is determined as relevant, which means it is caused by a condition or discontinuity; it is then evaluated to determine whether the condition is acceptable and does not affect the performance or serviceability of the material. The criteria used to determine the acceptance level is different depending on the plant's technical standards and the importance of the component or piping where the indication was detected. Sometimes numerical analysis, such as finite element analysis, can be performed on the flaw to predict the remaining life of the component, and necessary repair or replacement of the equipment is made accordingly. A general flow from ASTM E-1316 of flaw evaluation is described in Figure 4, as an example.

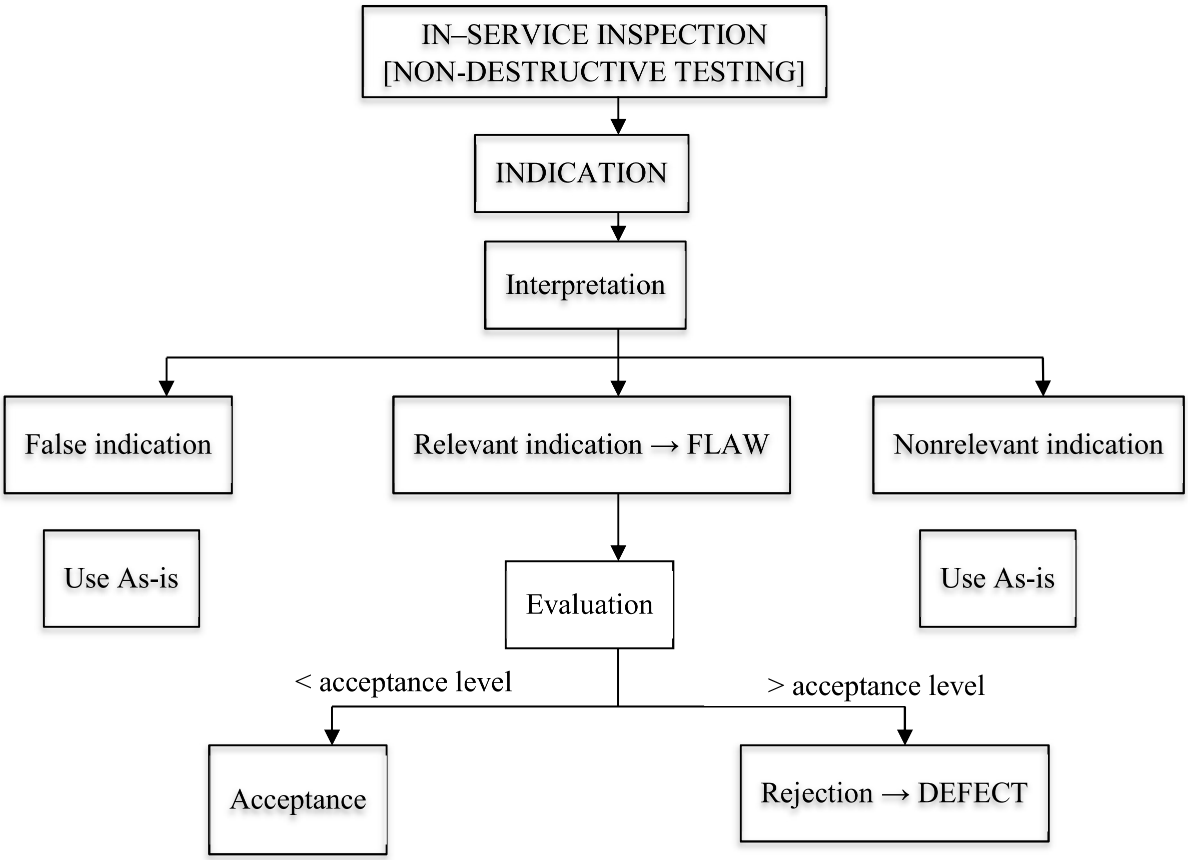

FIG. 4. Example of a flow chart for flaw evaluation. Reproduced courtesy of ASTM E1315[ASTM E1316—16a Standard Terminology for Non-destructive Examinations (2013)]

3.6.2.1. Flaw characterization and sizing

In order to determine whether the flaw is acceptable or not, the shape, length and depth of the flaw needs to be determined to estimate the severity of the flaws. The approach to flaw characterization can be different depending on the type of NDE technique to be used. Therefore, the characteristics and limitations of the NDE technique to be applied for flaw characterization needs to be fully understood. In order to raise the reliability of the characterization, more than one NDE technique can be used as well. Figure 5 below describes typical flow chart of flaw characterization and sizing of the indications detected, should a flaw arise.

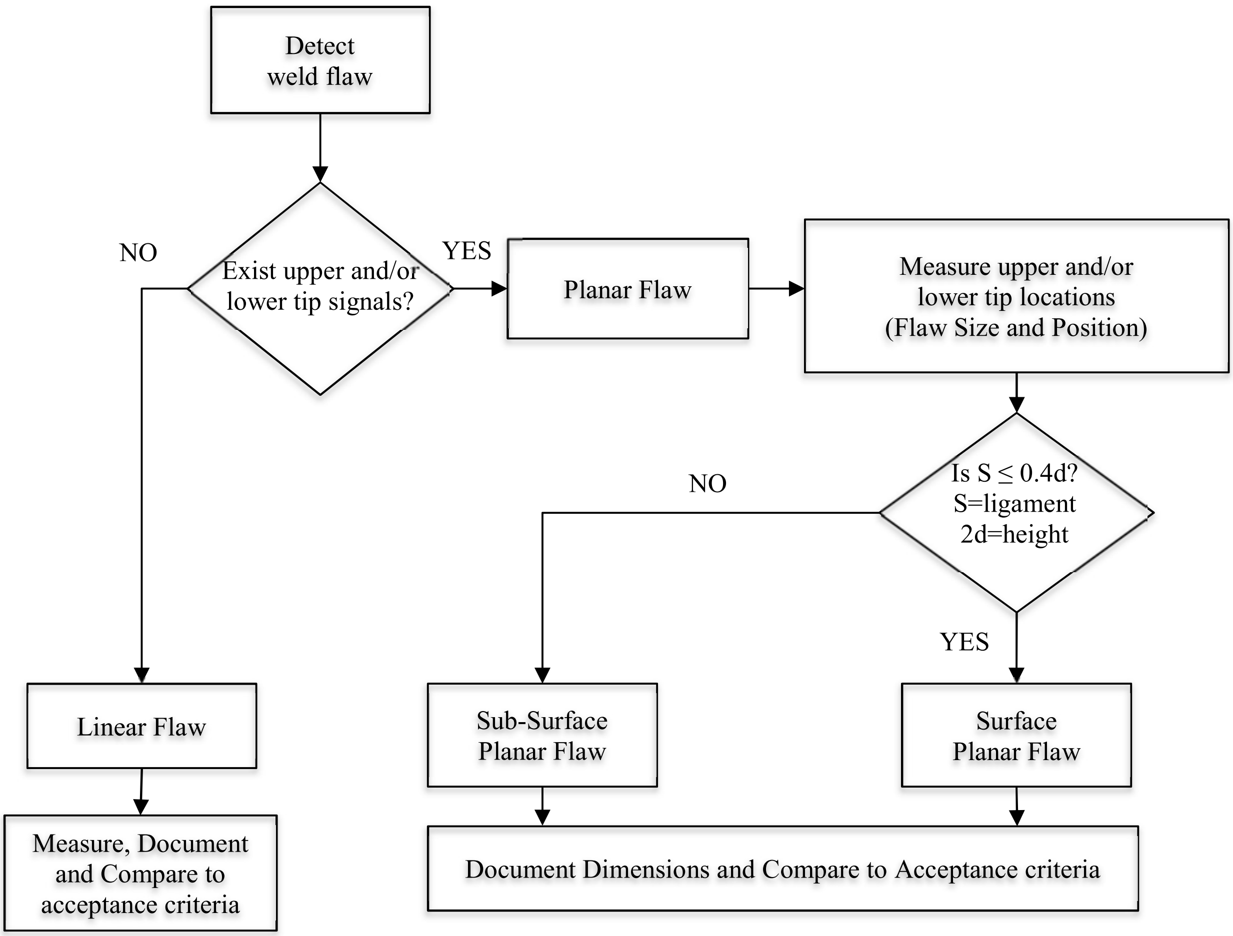

FIG. 5. Example of a flow chart for flaw characterization and sizing.

3.7. REPORTING

Performance of NDE generates different types of reports, which can be divided into two main types:

- Report of a single NDE;
- Summary report of a number of single NDE beyond a specific time period, for example:

– Outage reports used for evaluation of ISI activities planned and performed during an outage. Output of the report can be used as the one of inputs for permission of unit re–start after an outage;
– Yearly report used to evaluate ISI activities planned and performed during given year. This report adds to the above mentioned outage report evaluation of all ISI activities planned / unplanned and performed during a given year;

– Summary report about all ISI activities performed during a given year at all units in one locality / company.

Below are listed recommended items to be included in a NDE report:

- Unambiguous number of records;
- Unambiguous identification of inspected equipment within inspection areas;
- Identification of inspection type–planned, operative or conditional (performed if);
- Set up new inspection with previous attempts;
- Identification of Work order or prescribed work tasks for inspection performance;
- Specification of inspection method and references to inspection procedure;
- Information about inspection equipment used for inspection including due date of its calibration;
- Results of inspection covering unambiguous description of flaw indications exceeding the recording level with defined size, character, location and orientation of flaw indications;
- Comparison previous examination results with current ones;
- Comments to inspection – information about deviations from inspection procedure, accessibility deviations compared with inspection area drawings, supplemental information to performed inspection, status & results after repair / replacement of equipment, etc.;
- Name of inspector who performed inspection, name of inspector who evaluated results, date of inspection, date of record issue; and
- Attachment and distribution list.

Below are listed items of summary reports:

- List of inspected equipment and inspection areas;
- List of performed inspections (planned, operative or conditioned etc.);
- List of not–performed planned inspections;
- List of findings when flaw indications exceed the threshold specified in the acceptance standard with information how they were treated;
- List of vendors and attachments; and
- Final conclusion.

NDE is performed either by plant staff or the inspection organization (vendor). It is recommended that NDE reporting is supported and ensured by IT tools available at the plant. The more items from above lists that are put into the plant IT–system or database, the more useful the output reports would be. The IT tools would allow for the following:

- Control of report revisions (to have possibility to correct mistakes of issued records);
- Monitoring of fulfilment of outage / yearly ISI plan;
- Historical information e.g. information and results from previous performed inspections on the actual object;
- Reviewing and monitoring of trends in ISI (changes in flaw size, etc.); and
- Traceability of flaws and defects.

All flaws classified as defects ought to be reported and documented with the respective dimensions and positioning details.

All indications classified as irrelevant (e.g. geometry) ought to be reported on argumentation as to why they are irrelevant.

The content of the report should be consistent with the terms of the inspection contract (applicable standards, utility specifics, etc.). Recommended items are listed above.

A field report ought to be completed for each inspected area and turned over by the inspection vendor under the terms of the inspection contract (e.g. reportable indications should require immediate reporting).

Examination records, including calibration sheets, data sheets, inspection data etc. ought to be submitted to the utility for retention under the terms of the inspection contract and owner record retention requirements.

4. CONDITION FOR IMPLEMENTATION OF ISI PROGRAMME

Typically, the main parts of ISI planned activities are performed during refueling outages. It follows the description of conditions for implementation of ISI programme during pre–outage preparation and during outages. Furthermore, there are described activities based on operational experience.

Pre–outage preparation should include:

- Provision of information regarding unit type, system and equipment status before and during outage;
- Discussion on inspection conditions and readiness of equipment for inspection;
- Checking the readiness of work orders / work tasks for ISI inspection;
- Discussion about readiness of inspection instruments / manipulators;
- Discussion about job roles, responsibilities, and personnel qualifications;
- Risk evaluation of planned ISI activities which leads to effectiveness and safety of outage and decrease its potential risks. Here are examples of risks that ought to be evaluated:
  - Risks from conventional and radiation safety & environment (work at high radiation at the place of inspection, possible contamination with radioactive materials, limited space, high environment temperature etc.);
  - Risks from not performed planned scope of inspections: equipment is not prepared for inspection according to inspection procedure, failure of inspection equipment; etc.;
  - Risks from prolongation of time for inspection: influence to outage schedule etc.;
  - Risks from expected ISI inspection findings: Utility should be prepared to settle findings from ISI inspections;
  - Etc.

Implementation during the outages:

- Target of inspection, inspection procedure, inspection instruments, schedule for inspection;
- Previous inspection results (occurrence of discontinuities, register of findings etc.);
- Previous inspection experience;
- Specifics of the inspection at given unit / system / equipment;

- Acquaintance with conditions for inspection workplace handover and takeover;
- Interaction and communication between Utility and Vendor;

The aforementioned conditions should be discussed during the pre–job briefing, prior to start of the examination. The post–job debriefing should occur after the completion of the inspection activity, and should be based on feedback from examination personnel;

- Key performance indicators of inspections (examples);

  - Number of planned but not executed ISI inspections (supposed to be zero);
  - Number of ISI inspections with findings and non-corrective action (supposed to be zero);
  - Number of failed pressure / leak tests (due to ISI related weaknesses); and
  - Number of forced shutdowns (due to ISI related weaknesses).

  Note: All nonconformities should be identified, recorded and corrective action should be taken.

Licensees are encouraged to implement the following actions, based on operational experience:

- Implement an action plan to ensure guidance for and training of all roles and responsibilities of site NDE personnel assigned to monitor and oversee work activities is commensurate with the complexity of examination activities;
- Ensure that ISI programmes meet industry guidance and best practices;
- Communicate expectations to the supervisor who is responsible for NDE organization at NPPs and upcoming NDE evaluations;
- Perform and document observations of licensee oversight of NDE technicians during refueling outages including briefs, task practice, and field work to support effectiveness review;
- Engage training organization for consideration of adding lessons learned to leadership or other training products; and
- Develop procedural guidance to evaluate the complexity of examinations and identify the appropriate level of practice, briefs and licensee oversight.
- Ensure one or more sufficient human error prevention barriers, such as a preparation plan or checklist requiring licensee signature are utilized to document proper pre-job preparation prior to performance of the examination.

4.1. PERIODIC UPDATE OF ISI PROGRAMME

It ought to be ensured that ISI programme is consistent and reflective of all design and operational changes of SSC or regulatory requirements during the plant lifetime (usage of equivalents, installation of new SSC, etc.). Plants are assumed to have a systematic approach of evaluation these changes.

The licensee may assess the need to update its ISI programme and / or procedures due to the following considerations:

- Re-evaluation of the risk–informed selection process;
- Changes in standards and requirements;
- Improved inspection techniques;
- Inspection experience;

- Feedback on the qualification system;
- Domestic or international nuclear facility applicable operating experience; and
- Other related considerations.

Changes to ISI programme ought to be properly documented, appropriately justified and approved by utility, and as applicable by the Regulatory Body.

4.2. ISI LINKS TO OTHER PLANT PROGRAMMES

The ISI programme is just one of the many plant programmes, with some interconnection to other fields such as maintenance programmes, ageing management, long term operation and generation risk.

For example, availability of pre–service inspection data would include valuable input to ISI programme. Furthermore, design of inspected equipment has an impact on selection of NDE techniques and inspection procedures which are then implemented to ISI of the equipment [17].

On the other hand, the outputs from ISI programme can result in the need for immediate or planned activities in other areas (engineering, operation, maintenance etc.) to mitigate degradation of equipment.

It should be ensured the following plant programmes are accounted for, such as:

- Maintenance programme:

– ISI activities are planned, coordinated and managed within a set maintenance programme, which ensures proper planning, performance and recording of ISI activities in maintenance management system (or integrated plant information system);
– Supporting pre–outage readiness , ahead of potential unfavorable ISI results, can reduce unplanned outage maintenance activities.

- Ageing management programme or long term operation:

– ISI information is inevitable for the plant's comprehensive ageing management program and especially important for the long term operation of NPP (e.g. beyond the initial design life time) [18].

- Engineering:

– Modifications, replacements, new installations can lead to ISI programme update / revisions;
– Guidelines for inspectability for new Plant components should be developed and implemented.

5. OPERATING EXPERIENCES AND ISI EFFECTIVENESS

5.1. OPERATING EXPERIENCE WITH ISI EFFECTIVENESS CHALLENGES

ISI effectiveness facilitates an optimal balance between plant safety and inspection costs, in terms of fundamental ISI parameters (i.e. scope and interval) and the NDE system capability. ISI effectiveness has a strong correlation with efficient structural integrity assessment and mitigation, repair or replacement actions taken at a given facility. There is a two–way relation between the two: first, NDE delivers an essential input for integrity assessment (flaw size, orientation, position and other parameters), and second, the integrity assessment model establishes the requirements for NDE performance and reliability (worst case flaw to be detected, probability of detection and non–detection, accuracy of sizing and positioning).

Effective ISI is an integral part of plant life management / ageing management with its need for a reliable diagnosis of the condition of the components and the prediction of their future status. This is justified by the worldwide tendency of operating the NPPs beyond their initial design life i.e. long term operation (LTO). In the case of LTO, the eminent and gradually improving role of ISI / NDE is obvious due to the proportionally of the service period, where ageing effects may appear in more and more component areas, and sometimes unexpectedly accelerate. Consideration must also be taken when plants age and their service lives are extended, as unknown or unexpected ageing can occur. There is a clear correlation between the examination reliability and the intervals of subsequent inspections.

All these examples demonstrate ISI based evidence of the component's status on safety and cost of plant operation. If ISI techniques are able to characterize flaws in terms of correlation between specific types and growth mechanisms, the relevance of the ISI based evidence is even higher, as it may allow for a trend assessment. This trend assessment can be the basis of the determination of the time interval to the next inspection in which the component can still be safely operated.

Some operating experience examples of the past decade illustrate the importance of effective ISI, including:

5.1.1. Austenitic welds

Intergranular stress corrosion cracking (IGSCC) have been extensively discovered in stainless steel piping of BWR recirculation lines and was initially only detected by leakage. Yet the inspector qualification process, established in 1983, has allowed for subsequent improvement in over the last number of years. The results of these inspections have a direct impact on plant availability and cost by the need for mitigation activities like last pass heat sink weld (LPHSW) and augmented ISI.

5.1.2. Dissimilar metal welds

Similar problems have emerged from the inspection of dissimilar metal welds (DMWs) in both PWR and BWR type reactor units, where NDE techniques simply missed flaws, and as a result there were leakages in the subsequent operation period. In the last decade, steam generators (SGs) in WWER–440 units have shown DMW problems, too. These DMWs are situated between the steam generator shell (non-alloyed steel) and the interim piece to the primary collector and reactor coolant pipe (stainless steel).

A common experience today based on results from international round robin tests on DMW [19] show:

- Inside diameter (I.D.) procedures provide superior performance over outside diameter (O.D.) procedures as measured by probability of detection (POD), depth and length sizing root mean square (RMS) error;
- Flaw orientation has an influence on detection performance, i.e. circumferential flaws being easier to detect than axial flaws;
- I.D. procedures that include eddy current testing (ET) performed better at length sizing than procedures that do not include ET;
- The diversity of techniques uses tend to improve performance for detection, depth sizing and length sizing;
- The advances in the use and deployment of phased array ultrasonic testing (PA UT) are significant and procedures including this technology tend to perform better than those relying on conventional UT using one or only a few inspection angles; and
- Most of procedures exhibited length sizing performance would meet ASME Boiler and Pressure Vessel Code, Section XI requirement of RMS error within 19 mm.

Only a few number of procedures exhibited depth sizing performance that would meet ASME Boiler and Pressure Vessel Code, Section XI requirement of RMS error within 3.12 mm.

5.1.3. Cast austenitic stainless steel

Cast austenitic stainless steel piping was used in the primary pressure boundary of Westinghouse PWRs due to its relatively low cost and high corrosion resistance. Cast austenite stainless steel in PWR primary system has had an incident-free service record of over 35 years. However, as noted in the literature, there remains a concern of possible thermal embrittlement and thermal fatigue crack. But, the coarse-grained anisotropic structure of cast material makes it difficult to inspect reliably. Several research studies have been conducted since 1980s into the matter. Yet, conventional UT inspections are challenging due to the anisotropy and inhomogeneity of the coarse microstructures of cast materials. To overcome above metallurgical characteristics, low-frequency transmitter-receiver longitudinal technique (with synthetic aperture focusing technique (SAFT)) was employed. Since the 2000s, PAUT is extensively adopted in cast material piping weld inspections and inspection results have improved. But detection and sizing of flaws below 30% through wall show uncertainty and it is a widely accepted limitation by the nuclear industry. ASME Code Case N-824 [20] was developed using the improved information on how to detect flaws in cast material. From the code case, development and implementation of ASME supplement 9 [4] is apparently possible soon.

5.1.4. Steam generator tubing

Several tubing degradation mechanisms have been discovered since the 1970s, each with differing characteristics of the eddy current signals. If optimized or advanced probes (e.g. with motorized rotating coils) are being used and the adequate data analysis algorithms and logistics are qualified and applied, these mechanisms can be identified, sized and a trend in degradation can be established. The results have considerable relevance, and therefore, are double or even triple checked in the data analysis process. Due to the direct impact on plant power output by the number of plugged tubes and on plant energy availability (number and duration of unplanned outages) and cost by the eventual need for SG repair or replacement. From the point of safety, rupture of SG tubing could initiate small break loss of coolant accident (LOCA) type accident, which involves RPV emergency core cooling and associated thermal transients for the primary system. To mitigate this, some member states adopt multi–array ET technique for SG tube inspection. It provides more accurate sizing result and reduces overall inspection time.

5.1.5. Reactor pressure vessel head penetrations

PWR RPV head penetrations have emerged as a problem area of considerable significance in recent decades. Boric acid corrosion initiated by primary water stress corrosion cracking (PWSCC) on penetration tubes has led to significant decrease in structural integrity. Due to high radiation levels and complicated geometrical or clearance conditions, modular tools with complicated sensor carriers were developed using sophisticated UT technique (e.g. TOFD), qualified for the different inspection tasks and successfully applied on-site. Similar to SG results, the results of these inspections demonstrate a direct impact on plant availability and cost by the need for repair or vessel head replacement and augmented inspections. Drastic damage, which has occurred, demonstrates the importance of properly interpreting the results of ISI and taking appropriate corrective action.

The RPV head penetrations in WWER-440s differs from the PWR design. The problems identified in WWER-440s do not jeopardize the vessel head integrity. However, due to deformation of the cladding tube, it could lead to stuck control rods. Monitoring of the WWER penetration requires advanced UT technique.

5.1.6. Reactor internals

Reactor internal flaws have occurred in bolt and weld areas. In view of high radiation levels and the minimal chances for repair due to the severe access conditions, replacement appears to be the only alternative for effectively fixing the problem. However, as load and flaws growth rate is generally small, a certain degree of defective areas can be tolerated, providing assurance by evaluations.

Decisions on whether to replace or trend, impose a large responsibility for the ISI, in one case to give the final criterion for reactor internals replacement, in the other to supply a solid basis for trend analysis of the defective areas and the severity of the individual flaws. The most representative example here is cracking of baffle bolts.

5.1.7. Small bore piping

Experiences in the United States of America or the Republic of Korea and other nuclear power plants demonstrate that failure of socket weld at small bore piping (Ø < 50mm) is a recurring problem and, in association with LTO, requires increasing attention for safe operation. Mechanical or thermal fatigues, weld flaws (lack of fusion), SCC are the usual ageing effects. Although ASME BPVC Section XI requires surface examination only, UT was introduced but due to limited accessibility the conventional UT was later replaced by PA UT.

In WWER plants, similar degradations appeared in small bore piping welds. The ageing effects here also include mechanical fatigue, thermal fatigue and SCC. In some cases, DMWs (level measurement and blowdown pipes on SG) were affected, where bimetallic corrosion on the carbon steel side was detected. To mitigate, the ISI programme was amended by tradition of volumetric examination (RT).

6. BASIC ELEMENTS OF ISI EFFECTIVENESS

The ISI requirements are usually summarized in codes or standards developed by engineering associations and are based on consensus of all stakeholders such as regulators, component manufacturers, operators, inspectors, insurance companies, researchers, etc. Each code may be in line with the requirements of the given country's nuclear safety regulation, and their use may be mandatory, according to the law or regulation. The codes usually contain the general (including administrative) requirements, the requirements against NDE personnel, the NDE methods, the ISI programme (inspection interval or cycle), the evaluation of NDE data (acceptance standards) and the documentation rules; some codes also include repair and replacement activities.

In the 1970s, it was assumed that failures can occur randomly, and are only slightly influenced by service or design conditions (e.g. radiation, fatigue, local stresses, DMWs). Also, only the half of portions of components examined fell into welds, while the remaining portion into other areas like cladding, supports, bolts, casting surfaces. Over time it became clear that failures did not occur randomly in the determination of areas to be examined, but instead resulted from degradation in specific areas. Currently, inspection is primarily concentrated on welded joints, but some codes also require base metal inspection. NDE is often carried out on fatigue sensitive areas, though ISI intervals vary in the different codes (between 4 and 10 years). The 10–year interval was chosen based on historical failure rate data for non-nuclear steam power and petrochemical plant systems. In the first years of ISI code usage the construction requirements were applied for the ISI and, only later, they were replaced with requirements appropriate for operating plants. From this time, the acceptance criteria were determined on the basis of fracture mechanics taking into account the detected flaw stability.

In the context of effectiveness, the following questions arise:

- What kind of degradation processes are active in the component examined and in which part of the component do they take place?
- What kind of NDE methods and techniques are able to detect, characterize and size it reliably?
- How often does it need to be inspected?
- What kind of capability demonstration is required from the NDE system (equipment, procedure, personnel)?

Answering each of these questions has its own cost that spans the investigation period, specialists involved, investments, etc.. The considerations in cost calculations are discussed more detailed in subsection 6.3.

The development of ISI requirements bears the marks of those changes, which characterize the changes taking place in approach from the regulatory bodies of recent decades. This was largely influenced by the results of international research, round–robin programmes and network activities such as the ASME NDE Task Group, PISC I, II and III, EBIV, ENIQ, etc. Formerly, requirements using deterministic methods and detailed standards composed the fundamentals for ISI; nowadays, these methods are substituted by optimized processes reflecting safety, reliability and risk in an integrated manner. The inspection philosophy placed the focus on performance based and risk based / informed approach instead of detailed regulation. As a result of these activities, the fundamentals of an 'effective ISI' have been laid down, i.e.:

- The capability demonstration of ISI / NDE systems was grown; and;

- The application of PSA[1] for passive components (pressure boundary components) opened the door for risk informed ISI (RI-ISI).

Moving towards an effective ISI is not an episode but an evolutionary process which can always be improved, no matter the type of facility or unit involved. Figure 6, summarizes this evolutionary process and the main features of ISI effectiveness as described above.

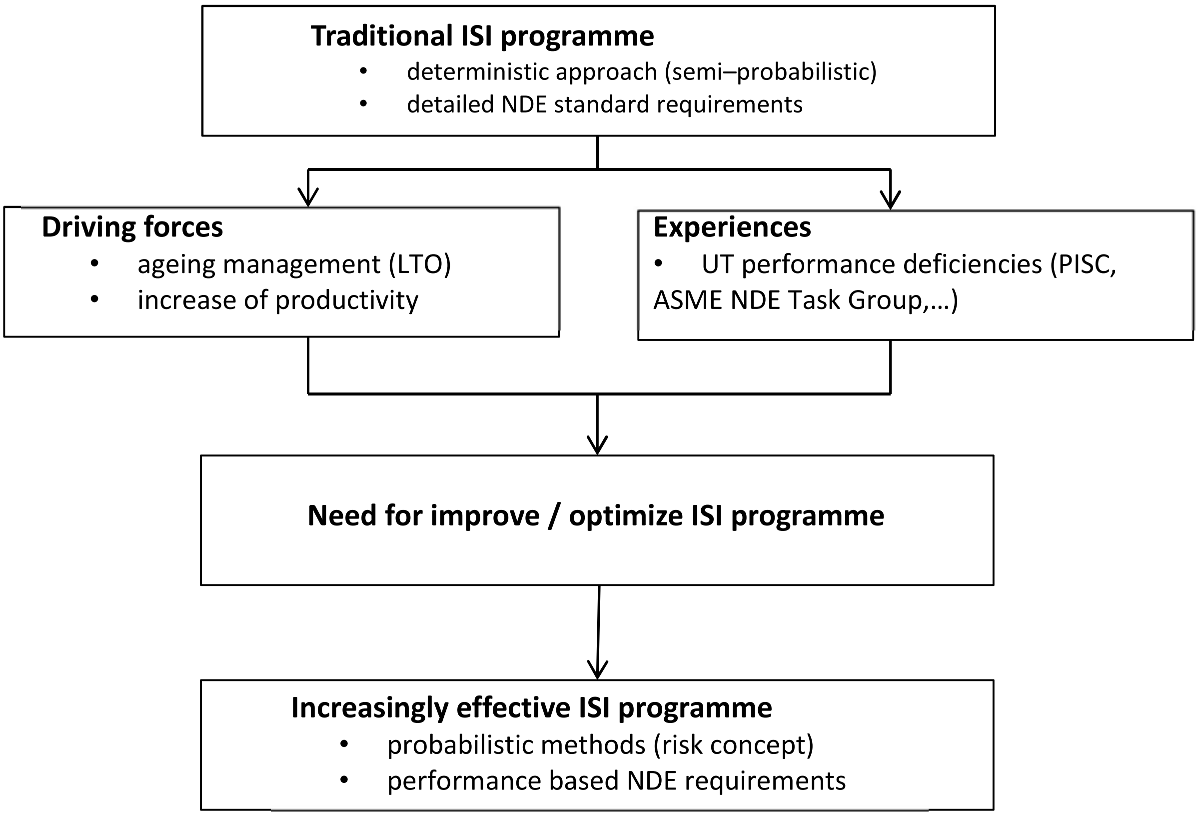

FIG.6. Overview of ISI effectiveness features.

Effective ISI is constituted of the following fundamental elements, and can be continuously improved on by following the general directives and considerations above (though details are explored below):

6.1.1. Risk informed ISI

RI-ISI is a type of ISI that focuses the inspection efforts and resources on the high-risk locations elements and by this means increases or at least maintains the overall plant safety, as measured by risk.

[1] In USA: probabilistic risk assessment.

6.1.2. Qualification or performance demonstration of NDE systems

Inspection qualification or performance demonstration is an organized process to establish confidence by a systematic and independent assessment that the NDE procedure, equipment and personnel are capable of meeting the inspection requirements in real circumstances.

6.2. RELATION OF THE BASIC ELEMENTS OF ISI EFFECTIVENESS TO STRUCTURAL INTEGRITY ASSESSMENT

ISI is a substantial tool of structural integrity assessment. Structural integrity assessment of pressure boundaries means the evaluation of their resistance to strength and fracture. Since the energy requirement for ductile failure is far greater than that required for failure in the brittle mode, the basic tool of the structural integrity assessment is the fracture mechanics. Fracture mechanics allows calculation of the limit condition of the material, complete with intrinsic flaws (crack) without unstable crack propagation. The assessment method can be deterministic or probabilistic; its scheme is shown in Figure 7. It is visible from Figure 7 that the awareness of loading and environmental conditions, material properties, size and position of the existing flaws is necessary for assessing the structural integrity. All of these parameters are subject to changes during plant operation due to ageing mechanisms, and consequently a continuous decrease in safety margin has to be taken into account. This is primarily important in light of LTO.

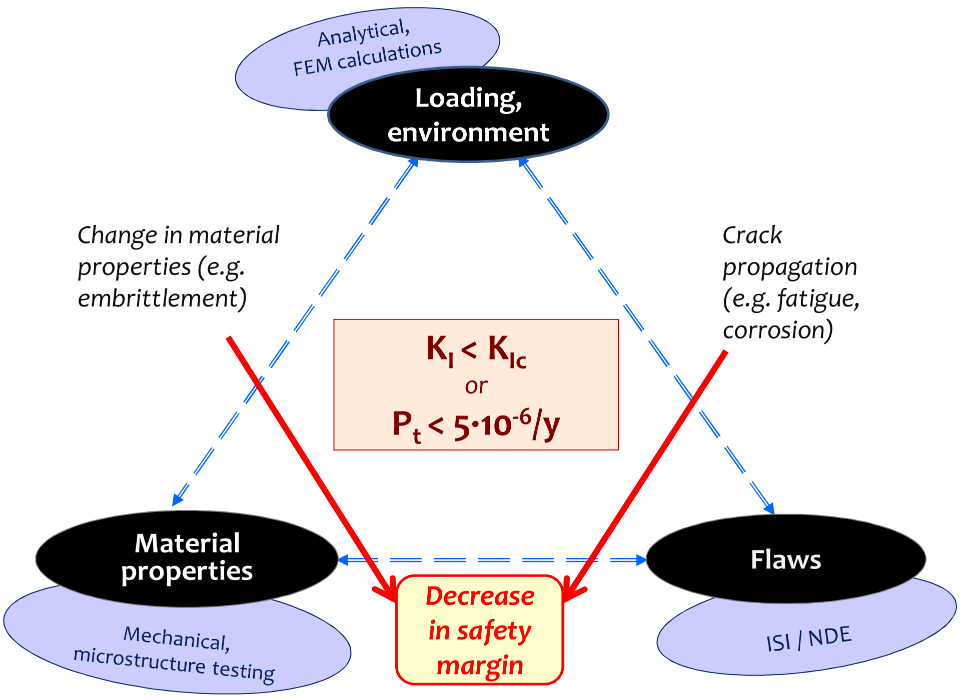

FIG. 7. Scheme of structural integrity assessment.

In Figure 7, K_I is the stress intensity factor (fracture mechanics parameter) while K_{IC} is its critical value named critical stress intensity factor or fracture toughness (material feature); P_f is the probability of failure.

Structural integrity assessment models and ISI are related in both directions:

- ISI is to supply most reliable data regarding:

– Presence of flaws in a given component in terms of dimensions;
– Flaw location within the wall including ligament dimension;
– Flaw characteristics including changes from previous inspections; and
– Proximity to other flaws, etc.;

All these data are among the most important input data for the subsequent structural integrity assessment.

- Structural integrity assessment is formulating the requirements for the level of ISI performance, such as:

– Scope and inspection volume;
– Flaw evaluation process;
– Target detectable flaw size;
– Sizing accuracy, at least indirectly;
– Accuracy of the location of the flaw;
– Need for more detailed characterization of flaws beyond the sizing capability if necessary;
– Accuracy of the determination of the ligament between flaw and closest component surface; and
– Inspection interval determined from the ISI information and its assessed quality.

6.3. RELATION OF THE BASIC ELEMENTS OF ISI EFFECTIVENESS TO COST

As it was introduced, ISI effectiveness is that the fundamental ISI parameters and NDE system capability are in optimal balance between safety and cost. The discussion of the aspects of ISI effectiveness within the preceding sections was oriented towards the principles of safety. In the interest of the safe and, at the same time, cost effective operation of NPPs, it is also of importance to investigate the relation of ISI effectiveness and overall cost of the plant operation when employing effective ISI and to identify potential benefit of combining safety and cost aspects when considering ISI effectiveness improvement. This integrated consideration may also present some answers to the questions posed at the end of section 3.2.

6.3.1. Radiation dose considerations

Radiation doses accumulated by workers is a significant element of ISI planning and influences the cost in several ways. In some countries, explicit dose targets are set and financial penalties assigned when accumulated dose exceeds established levels.

Dose considerations play a role in determining what type of NDE equipment and process (for example, automated–versus–manual inspection) is to be used and how many inspection personnel are to be used. Thus, dose contributes to the cost of ISI in an explicit way.

In the case of RI-ISI, benefits from dose reduction can be calculated. Based on a ten year inspection interval, 4,537 inspections have been eliminated at 24 plants, which projects to about 19,509 inspections eliminated at 103 plants in US [21]. For this example, projected benefits (cost savings) are determined in the Table 1 below using various cost models.

TABLE 1.COST MODEL PROJECTIONS DUE TO RI-ISI IN US NPPS

Number of inspections	Cost per inspection	Inspection savings	Dose savings Sv	Dollars per 10 Sv	Total savings
19 503	USD 1000	19 503 000	40	10 000	59 503 000
19 503	USD 3000	58 509 000	50	10 000	108 509 000
19 503	USD 5000	97 515 000	60	10 000	157 515 000

6.3.2. Relation of selection of ISI-scope with cost

The subsequent considerations, shown in Figure 8, are merely qualitative, but quantitative data points can be projected, if data is available. The assumption in the following figure is that the level of ISI performance is fixed, as in case of ineffective ISI, and the scope is irrelevant.

The cost of ISI is increasing with increasing scope of the inspection (blue dotted line). It is visible that in case of both effective scope (black dotted line) and ineffective one (round red dotted line) the scope has an insufficient level which is associated with an almost infinite or very high cost due to consequence of failure. This level is however differing from each other in the two different cases. The total feasible cost of the effective ISI scope is lower than that of the ineffective one, and this lower absolute cost value is corresponding with a lower percentage of scope. This can be regarded as the optimum selection of inspection areas from the mere standpoint of cost.

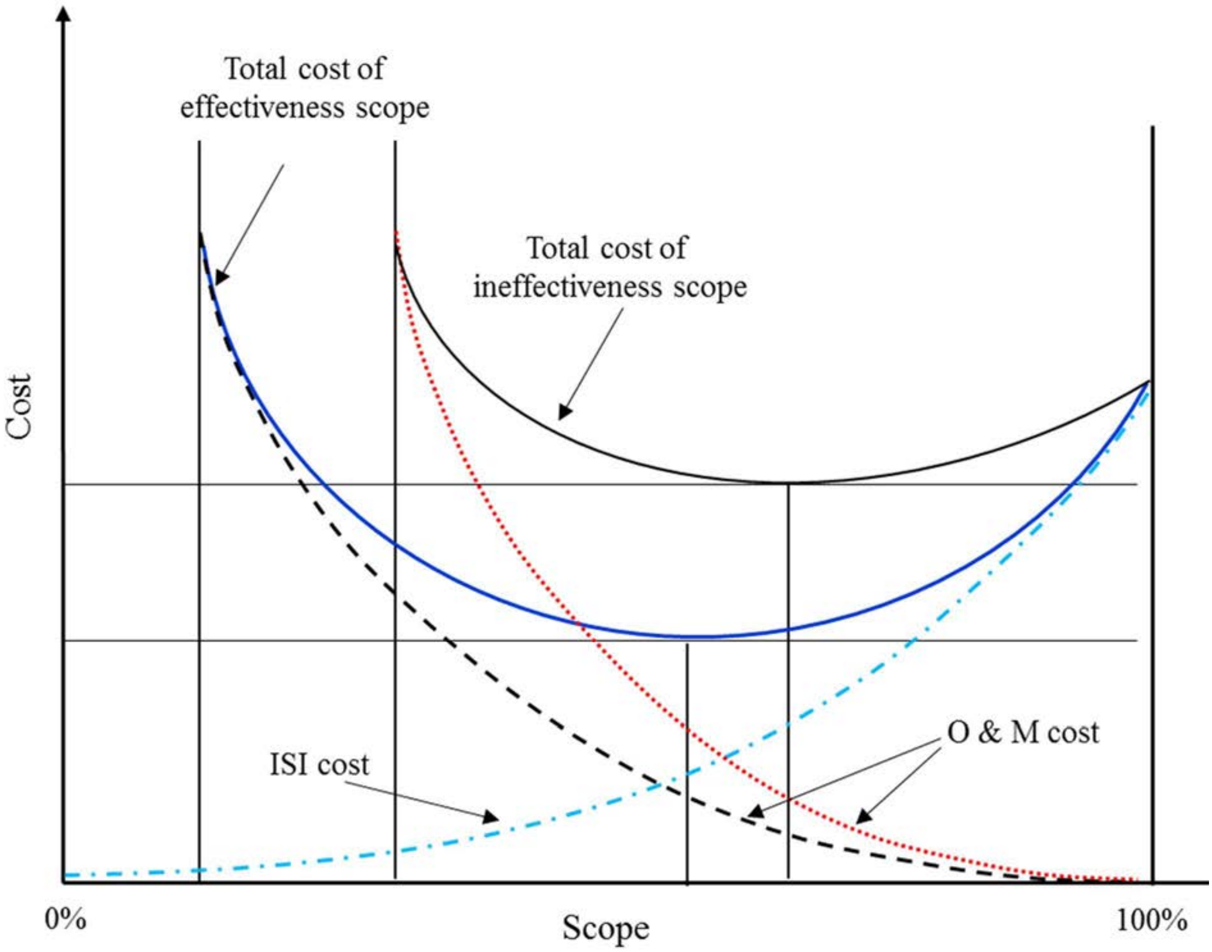

FIG. 8. Qualitative relation of ISI scope with cost.

The diagram does not account for the probability of the non–failure of the component or area, if certain components or areas have not undergone inspection. However, this consideration can still be implemented, but is not relevant to the conclusion that a systematic selection of the scope of inspection is necessary to support cost effective plant operation. This systematic selection should be based on models, which regards ISI in its entire field of interaction with other features of the structural integrity assessment.

6.3.3. Relation of NDE efficiency with cost

Similarly, to the selection of ISI scope, qualitative considerations for the relation of NDE efficiency with cost can be calculated (Fig. 9).

Making improvements to NDE systems that are already at a high level of efficiency is more costly than making similar incremental improvements to less effective systems. If NDE systems with a lower level of efficiency are used, there is a ‘penalty’ of increased plant cost due to the failure of the component. The total cost reaches a minimum level at a relatively high efficiency. This cost of the improvement could be further subdivided by the number of components or plants e.g. with the same or very similar design and materials, but which benefit from the same improvement effort.

This diagram does not consider the cost of the inspection itself. Initial calculations consider employment of a more efficient NDE methodology/technique, which could be regarded as costlier. However, the experience e.g. with the implementation of UT phased array technique shows that simpler probe systems (together with a combination of techniques and with the reduction of scanning areas and steps) have led to a better coverage and consequential savings of inspection time.

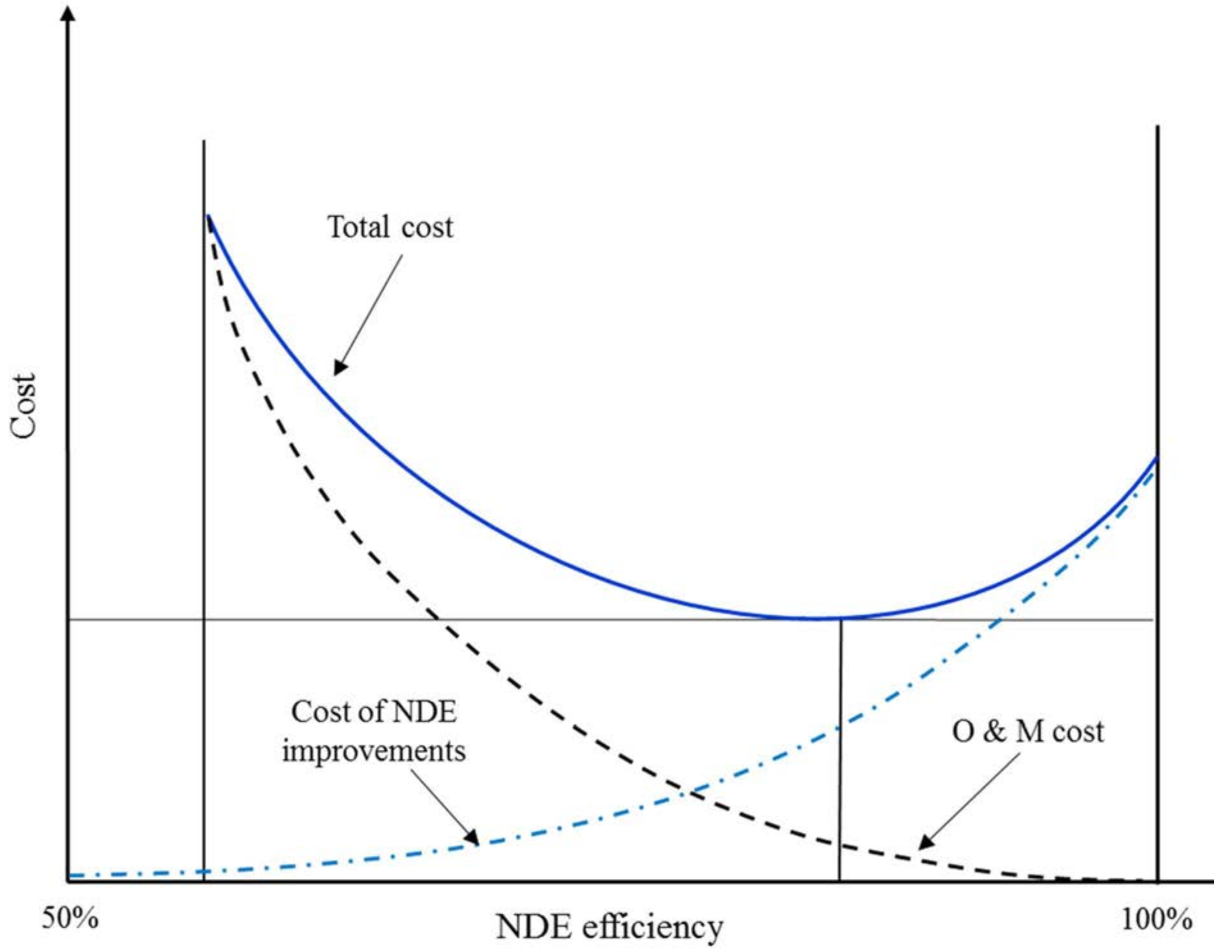

FIG. 9. Qualitative relation of NDE efficiency with cost.

6.3.4. Relation of NDE efficiency with inspection interval

Again, the consideration of the relation of NDE efficiency with the inspection interval is purely qualitative.

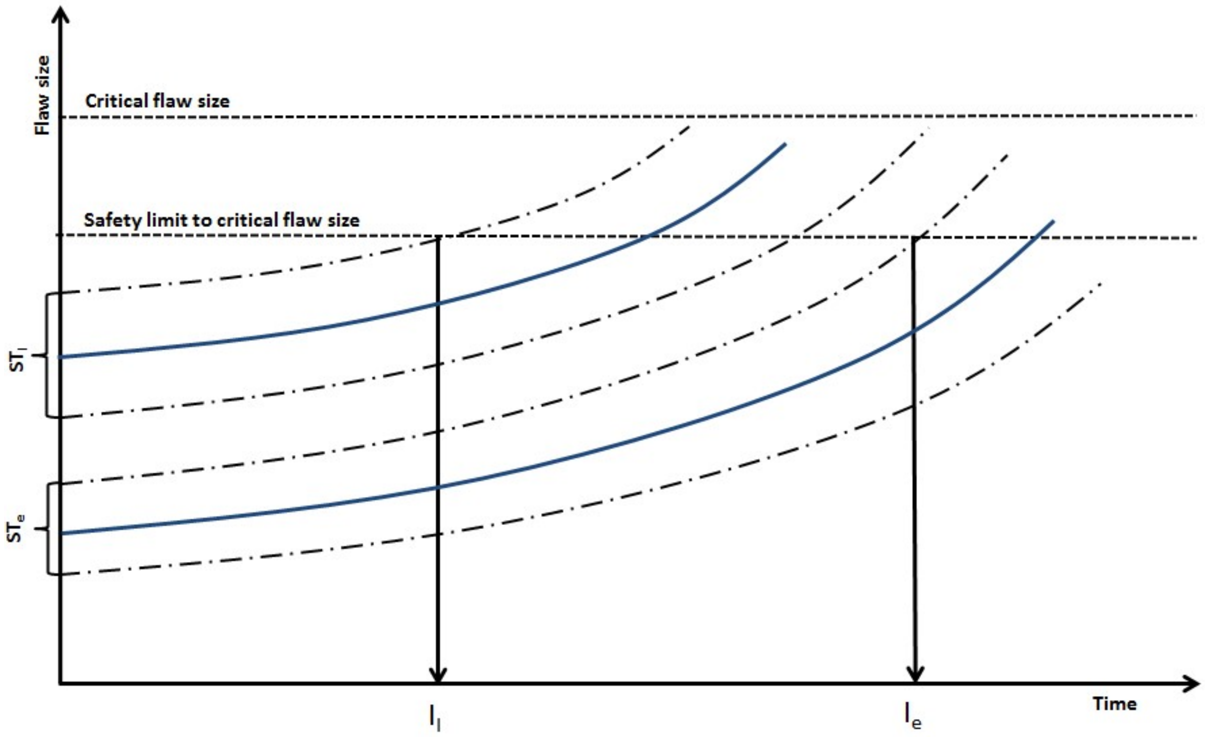

FIG. 10. Qualitative relation of NDE efficiency with ISI interval

The efficient NDE methodology is capable of reliably detecting flaws in relatively small dimensions (D_e being the through wall dimension of the flaw detectable with the efficient NDE). The accuracy of sizing for this case is within a small sizing tolerance being ST_e. The inefficient NDE methodology is capable only to reliably detect flaws in relatively large dimensions (D_i, and the accuracy of sizing are within a relatively large sizing tolerance (ST_i)). For the purpose of simplification, the sizing tolerance to over– and under–sizing is assumed to be the same.

A 'defect free' condition, in which there is an absence of defects, and the condition is larger than the defect corresponding to the minimum detectable plus the upper sizing tolerance, can be established at the 'time zero'. Over the ensuing time intervals, the defects are expected to grow, depending on original size and loading conditions. The subsequent inspection has to be scheduled for the time when the maximum defect present has grown to the size corresponding to the safety limit (allowable size).

Figure 10 demonstrates that the potential extension of the inspection interval between I_i to I_e can be accomplished, thereby reducing cost, while maintaining plant safety if an efficient NDE methodology is being employed. This consideration could be demonstrated using traditional deterministic principles or risk–informed methodologies.

6.4. RELATION OF NDE EFFICIENCY TO NDE CAPABILITY

The greater the need for risk reduction by the NDE, the greater the need for NDE system reliability and for demonstration of its capability. In general, the following elements create NDE reliability:

- Applicability—a proper signal / noise ratio;
- Reproducibility—a correct system calibration;
- Repeatability—stability of the NDE system; and
- Capability—probability of detection.

POD is the fraction of detected flaws out of a total number of flaws, as a function of flaw size. POD was established to support The National Aeronautics and Space Administration (NASA) and other programmes in 1970s and became the fundamental element of quantitative NDE in other industries. Figure 11, shows a POD and PND (probability of non-detection) curve for a NASA space programme where new points were generated from a limited number of original ones by a special simulation programme, and thus it was possible to draw up the confidence interval as a function of crack size.

POD is a possible measure of NDE capability. To determine POD, a reproducible calibration and appropriate acceptance level are necessary. POD methods are useful for development of repeatable NDE procedures, but a precondition for their application is a stable NDE procedure [22].

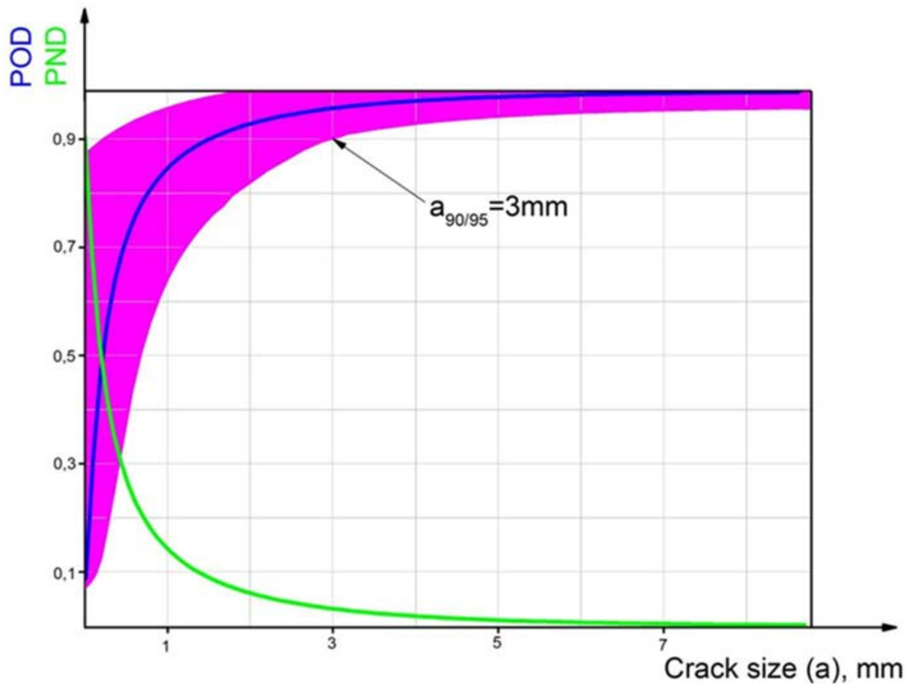

FIG.11. POD curve (for a=3mm: POD = 90%, confidence = 95%) [23].

POD is in strong correlation with NDE efficiency. Characterization of flaws is needed in order to draw conclusions of the mechanism of flaw initiation and of its subsequent growth. For this reason, we will differentiate between ‘status or momentary assessment’ and ‘trending assessment’.

6.4.1. Criteria for status assessment

There are well known criteria for detection and sizing, which are both affected by the influential parameters of the component and the influential parameters of the flaw sizes and the flaw characteristics as well as of the technique. Under these considerations, POD as a criterion is applied.

It is often assumed that a technique able to detect small flaws will also be able to detect the large flaws, which is not self–evident. This means there is a need to add some larger flaws (e.g. an unfavourable orientation or larger composite flaws) to the catalogue of flaws in order to be able to exclude a systematic lack of performance in the presence of larger flaws.

Another criterion for status assessment is the probability of correct acceptance or rejection (PCA or PCR) of a detected flaw, which is represented by the sizing tolerance of the technique. In this case, the question of eventual non–consistency of the sizing tolerances must be considered.

Conventionally, it is expected that NDE can detect the smallest flaw. This form of the question is in relation to the NDT techniques, and in terms of fracture mechanics it may even be the initial flaw size. Important is, on the contrary, the size of the 'largest flaw which is not detected' by NDT.

In many cases the assessment of the performance concentrates on these criteria, which are displayed in the traditional POD diagrams (Fig. 11) and are useful for status assessments. However, at least in case of subsurface flaws, the correct determination of the ligament from the adjacent flaw tip to the surface is of importance as well. An assessment model accounting for this influence was already presented within the PISC programme.

6.4.2. Criteria for trending assessment

This consideration asks for more information concerning the flaw detected. For this assessment, the methodology must be able to supply criteria, which allows reliable conclusions to be drawn about the type of flaw, whether composite and/or faceted, which orientation, if planar or partly voluminous; its location within the weld cross-section, etc. This information is not the only source of information, as there are contributions from other disciplines (e.g. welding metallurgy), which help to narrow the possible variety of flaw types. This characterization—within certain limits—allows for a trending assessment of flaw growth.

In this respect, the criteria for NDE efficiency are determined even more carefully, as well as the strategy for the qualification of such methods. As an example, the selection of flaw parameters as well as their realistic simulation for the experimental evidence poses considerable problems.

6.5. STATE–OF–THE–ART NDE TECHNIQUES AND EQUIPMENT

In other respects, developments in the fields of information technology and microelectronics has contributed significantly to NDE techniques development in recent decades. Among NDT techniques applied for ISI many examples could be selected to illustrate the development, some of them briefly discussed below.

Conventional UT usually refers to UT techniques based on application of single or dual element pulse-echo transducers applied for flaw detection or characterization through information provided by reflection, attenuation and / or velocity. Selection of the incident angle may depend on several factors including the material type, component thickness and purpose of the examination.

Crack depth sizing can be performed based on both detection of a corner reflection signal and a phenomenon known as crack tip diffraction. Crack tip diffraction refers to the emission of a weak ultrasonic signal from the tip of the crack.

The time-of-flight diffraction (TOFD) technique is a two–probe method using one probe for transmitting and the other probe for receiving. The transmitter introduces an L–mode beam at an angle and a so–called ‘lateral wave’ that propagates along the component surface. Depth sizing may be accomplished by detection and transit time analysis of a tip–diffracted signal from the flaw tip.

Phased–array ultrasonic techniques have been gaining increased acceptance for performing ISI of nuclear power plants. PA UT uses a transducer consisting of multiple piezoelectric or piezocomposite elements. Electronic beam steering and focusing is achieved by careful time delay sequencing of excitation signals to the individual elements in the PA UT transducers to create complex constructive and destructive interference patterns to intensify the sound field in a desired location.

A linear array transducer is only capable of steering the beam over a range of refraction angles within a single plane while a 2–D matrix array is capable of providing adjustments to both beam refraction angle and beam skew.

One of the significant capabilities facilitated by the use of PA UT is that of sectorial scanning. Sectorial scanning refers to sweeping of the sound beam over a range of refraction angles. This allows data obtained from many angles to be collected quickly, enhancing flaw detection and characterization.

The Full Matrix Capture (FMC) technique is a promising application of the phased array technology. FMC is a specific data acquisition process that allows for the capture of every possible transmit-receive combinations for a given ultrasonic phased array transducer. This new technique aims to increase the resolution and S / N ratio of conventional phased array UT.

An eddy current probe consists of one or more coils with the axis alignment most often perpendicular or parallel to the inspection surface normal. An alternating current source is applied to one or more coils, generating magnetic fields. These magnetic fields induce eddy currents in the conducting materials when the probe is positioned nearby.

In general, the advantage of eddy current techniques over ultrasonic techniques is that they are usually more sensitive to small defects and the probes do not require coupling to the test material surface. As noted, a significant disadvantage of eddy current techniques is that they are often relegated to surface inspections and are not very useful for characterizing the depth of flaws.

In addition to the development of NDE techniques, inspection organizations (vendors) involving research organizations and academic institutions are also improving the inspection equipment / manipulators for automated examinations.

The innovative approaches and solutions provided by the automated examinations have to be in line with the NDE technique developments. The inspection manipulators also have to overcome inspection challenges that presented by the design of some of the nuclear power plant components having regions that are difficult to access (e.g. long small diameter buried pipelines with one side access only, narrow gap between the reactor vessel penetration and the thermal sleeve, etc.).

To improve the effectiveness of the NDE, the state–of–the–art equipment / manipulators are designed and developed in such a way to, among other benefits, reduce the examination time, increase the safety of the examination personnel, reduce radiation doses accumulated by workers, allow inspection and increase the inspectability (e.g. inspection volume) of difficult to access components.

6.6. HUMAN FACTORS

Increasingly, human factors play a significant role in the design, operation, maintenance and decommissioning of nuclear power plants. According to the fundamental safety principles of the [24]), *"[a]n important factor in a management system [for safety] is the recognition of the entire range of interactions of individuals at all levels with technology and with organizations. To prevent human and organizational failures, human factors have to be taken into account and good performance and good practices have to be supported"*.

According to some statistics [25], the proportion of human performance related problems in maintenance, testing and calibration (42–65%) exceed those in normal nuclear power plant operations (8–30%) and abnormal and emergency operations (1–8%).

In 2009, an event in the US North Anna nuclear power plant [26], in which NDE in–service inspection personnel failed to identify five PWSCC indications in the steam generator safe-end weld, further highlights the need for human factors consideration in NDE activities. Two through–wall and three partial through–wall indications exceeding the acceptance criteria were detected in a subsequent ISI in 2012 and judged to be within the inspectors' ability to detect during previous ISI activities. The post-event root cause analysis revealed inadequate practices of the on–site NDT organization towards the supplemental NDE personnel and their inadequate briefing assigned to insufficient consideration of human factors.

Even though human factors are far less investigated than in other domains, the research on human factors in NDE has over the years provided evidence of the variability between NDE–personnel in the inspection results and of the variety of human and organizational factors affecting the inspection performance [27–30].

There is not one *human factor* but a variety of *factors* that affect NDE inspection performance from within the NDE–personnel (intrinsic factors) and, more predominantly, from the environment (social, physical or the organizational, it is recommended to develop strategies to include NDE (including external subcontractors) into general nuclear power plant considerations.

Improvements to NDE reliability, and consequently, the effectiveness of ISI could be achieved through including human factors considerations into the design of technology, inspection procedures and working practices. Managers and supervisors should be aware of the factors negatively affecting performance and develop strategies, together with human factors experts, to mitigate potential negative effects. Human factors in NDE should also find its place in the personnel training. And finally, continuous learning through acquiring new knowledge, risk assessment, transfer from other disciplines and from operational experience is a suggested path to follow.

6.7. CONCLUSIVE REMARKS TO THE ASPECTS OF ISI EFFECTIVENESS

In the above considerations, criteria and guidelines were discussed for achieving true improvements of ISI effectiveness. The ongoing implementation of innovative techniques demonstrates the potential to improve the effectiveness of NDE in terms of safety and of cost. However, as already mentioned, ISI effectiveness relies also on the quality of the selection of the scope together with the correct determination of the inspection interval. Most of these criteria lie in the field of interaction between ISI with its three major aspects determining its effectiveness and the other disciplines contributing to structural integrity assessment. However, the factors, which have become the most important ones, call for integrated conceptions being both: safety conscious and cost effective, such as risk informed inspection. For this reason, the next chapter will discuss an overall concept in detail allowing for substantial conclusions to be drawn in terms of criteria and recommendations for ISI effectiveness improvement.

7. RISK-INFORMED ISI

7.1. GENERAL APPROACH TO RI-ISI

7.1.1. Programmatic perspective

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

- Management support;
- A good understanding of the strengths and limitations of the existing ISI programme;
- Proper use of plant–specific risk information (e.g. PSA / PRA);
- Multidisciplinary knowledge;
- A constructive interface with the regulatory body;
- "The formation of an appropriate workforce structure is also an essential factor in devising and implementing a RI-ISI programme. Such a workforce will need to contain or have access to a large array of different disciplines, including" [16]: inspection; maintenance; design; materials; chemistry; stress analysis; systems; PSA; operations and safety; and
- In the process of developing a RI-ISI programme, the regulatory body may be involved at an early stage of the process, to either define or review the basic safety requirements that must be met.

7.1.2. Technical perspective

The concept of RI-ISI consists of ranking the elements for inspection, such as welds in piping systems, according to their risk significance and developing an inspection strategy commensurate with their risk significance. Prior experience in a multitude of facilities has shown that RI-ISI provides a framework for effective allocation of inspection resources and helps to focus the inspection activities where they are most needed. As part of the RI-ISI process, an understanding of the most likely degradation mechanisms is developed, which is used to focus required inspections to use the most appropriate inspection methods for the anticipated damage mechanisms;

To date, RI-ISI has primarily focused on piping system inspections, but in principle can be applied to any passive component covered in the ISI programme. These passive components are normally not explicitly modelled in a 'base case PSA'. Hence, special analyses may need to be performed to estimate component (e.g. weld) level failure rates and consequences of component failure due to the loss of function and secondary flooding and other consequences of system pipe breaks. These special analyses are used to develop the risk rankings which are used to help prioritize candidate changes to ISI programmes.

"An overview of the fundamental aspects of most RI-ISI methodologies is depicted in Figure 12. Figure 12 reflects the basic technical elements of the risk-informed concept as relevant to developing an ISI programme" [31].

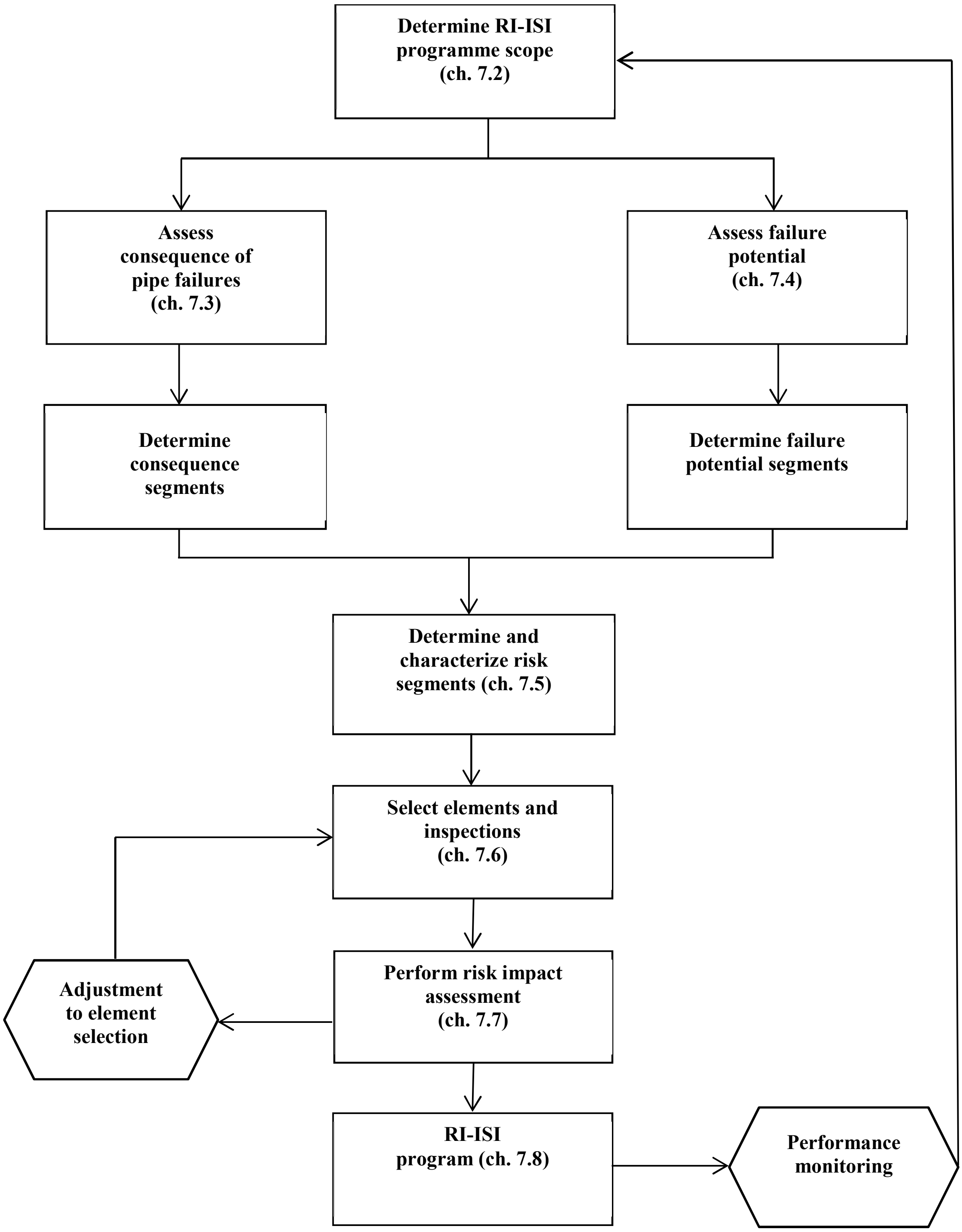

FIG. 12. RI-ISI methodology overview.

7.2. SCOPE OF RI-ISI PROGRAMME

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

The scope of a RI-ISI programme can be full scope programme or, when an alternative programme is already in place (e.g. PMT–2004, ASME Section XI and CSAN285.4 [8]), a partial scope programme. However, the scope should be clearly defined and documented as the programme continues.

A full scope programme can be defined to the point as to include all passive components such as:

- Those relied upon to perform a nuclear safety function during all design–basis plant conditions;
- Those whose failure could compromise the function of safety–related systems or components or could cause a plant trip or actuation of a safety–related system.

A partial scope programme is restricted to any subset of the systems or functions defining the full scope. The partial scope application can be justified, for instance, if an alternative (such as deterministic, augmented) programme is in place for the other passive components or degradation mechanisms.

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

A RI-ISI methodology may allow flexibility in determining the scope of application. Therefore, conducting the application on a large scale (e.g. a whole plant application), a system specific application (e.g. a single system) or a class of components (e.g. the reactor coolant pressure boundary) may still produce consistent and reliable results" [19].

7.3. CONSEQUENCE ASSESSMENT

"The failure of a passive component in a NPP can basically lead to one of the following classes of events of interest:

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

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

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

- A qualitative failure modes and effects analysis (FMEA) that determines the plant impacts of postulated failures of postulated sizes (e.g. small, medium, complete rupture). Both the direct consequences (initiating event occurrence, loss of system functions) and indirect consequences (spatial effects as flooding, water spray, pipe whip, jet impingement) of failures are evaluated. This step can consume the largest share of resources;
- Qualification of the PSA for RI-ISI application; and
- Quantitative analysis with PSA.

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

- The levels and scope of PSA to be used in RI-ISI;
- PSA quality, limitations and uncertainties; and
- Passive component failure treatment.

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

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

Level 2:Estimation of off–site fission product release. Consequences are usually expressed in terms of the combination of small, large, early and late containment failures (e.g. large early release frequency (LERF));

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

All modern NPPs have plant–specific PSA studies, usually at Level 1 or Levels 1 and 2. For this reason, it appears logical that they may form the basis of the consequence evaluation. Current RI-ISI applications have mainly relied on CDF and LERF as the consequence metrics of interest.

It is recognized that the use of other Level 2 consequence metrics (e.g. large early release) could be important to RI-ISI application, especially for reactors whose complete primary pressure boundary is not fully covered by the containment structure (for instance, RBMK and CANDU reactors). In this case it may keep in mind that Level 2 studies are based on assumptions and hypotheses that can be very difficult to verify in practice and thus are in general subject to higher uncertainties than estimates of CD.

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

The scope of the most comprehensive PSA Level 1 studies includes evaluation of the risk at power operation, start-up, shutdown and cold shutdown. Among the initiating events that are usually considered are transients, loss–of–coolant accidents (LOCAs), support system failures, internal fire, flooding, seismic and other external events.

The basic demand on the scope of the PSA is that all relevant operating plant modes and initiating events must be addressed to the evaluation. It is however, not necessary that all modes and events are included in the calculations. A qualitative treatment of missing modes and events is sufficient when they have little influence on the result. This will differ from plant to plant" [16, 31].

"With respect to external hazards, it is important to understand the purpose of the RI-ISI development. That is, the purpose of a RI-ISI application is to develop a periodic inspection programme that maintains or improves plant safety. Therefore consideration of other hazards outside the baseline PSA (e.g. external hazards) is not needed if they would not significantly impact the decision making process (e.g. selection of inspection location)[32].

The following provides a summary on why some hazards need not be included in the PSA used to develop COFs and a RI-ISI programme for piping. However, one or more hazards can be included at the option of the RI-ISI programme developer:

- *Internal fire events:* The potential contribution of piping failure of internal fire risk are insignificant because the failure probability of piping is insignificant compared to the failure probability of other SSC, such as pumps, valves and power supplies. Fire events are also not likely to present significantly different challenges to the piping in the scope of this application. Meeting defence in depth and safety margin principles provides additional assurance that this conclusion will remain valid. ISI is an integral part of defence in depth, and the RI-ISI process will maintain the basic intent of ISI (that is, identifying and repairing flaws) and therefore provide reasonable assurance of an ongoing substantive assessment of piping condition. In addition, there are no changes to design basis events and therefore safety margins are maintained;

- *Seismic events:* Well–engineered systems and structures (for example, piping systems) are seismically rugged. Individual plant examination of external events (IPEEE) and other industry and NRC studies (for example, EPRI report TR-1000895, NUREG/CR-5646) has shown piping systems to have seismic fragility capacities greater than the screening values typically used in seismic assessment and are not considered likely to fail during a seismic event. ISI is not considered in establishing fragility of such SSCs. As with the internal fire hazards discussion, meeting defence in depth and safety margin principles provides assurance that this conclusion will remain valid. ISI is an integral part of defence in depth, and the RI-ISI process will maintain the basic intent of ISI (that is, identifying and repairing flaws) and therefore provide reasonable assurance of an ongoing substantive assessment of piping condition. In addition, there are no changes to design basis events and, therefore, safety margins are maintained;

- *High winds, external floods, and other external hazards:* As described previously, the purpose of developing an RI-ISI programme is to define an alternative ISI strategy for piping systems. Other hazards (for example, high wind or external floods) need not be considered in the development of an ISI programme for piping. The reasons include the structural ruggedness of the piping systems, location (because relevant systems are typically inside well-engineered structure), and the consequence assessment for internal events already includes the consideration of spatial impacts. In addition, the substantial industry experiences with plants implementing RI-ISI programmes have not identified changes based on insight from the evaluation of these other external hazards. The small potential impacts on the potential for piping failure of a RI-ISI process, and the approaches to maintaining defence in depth and safety margins summarized previously, provide confidence in this conclusion;

- *Conclusion:* Quantification of other hazard groups will not change the conclusions derived from the RI-ISI process. As such, EPRI report 1021467 [32], provides guidance on meeting RG 1.200, revision 2 and RG 1.174 is sufficient for developing RI-ISI programmes. Based on RG 1.174:

– The magnitude of the potential risk impact is not significant;
– Traditional engineering arguments including defence in depth and safety margin are applied;
– Including other hazard groups would not affect the decision; that is, they would not alter the results of the comparison with the acceptance guidelines" [16].

7.3.1. PSA quality and limitations

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

An overriding requirement is that the PSA realistically reflect the actual design, construction, operational practices and operational experiences of the plant. The PSA should reflect the plant's different functions with the same accuracy and level of detail. The evaluation of system demands could be done with the same level of realism and conservatism for all functions, and the input data used for PSA analyses may be verified to ensure that it reflects the state of the art.

"It is recommended that the PSA study was qualified / certified for use in RI-ISI application by fulfilling demands specified by ASME standard or IAEA standards / requirements. The qualification/certification could also be performed by peer review of the PSA for RI-ISI application. The qualification of the PSA could be documented [32, 33].

Due to the small probabilities of failure of passive components in comparison with active components, the former usually only contribute to a small proportion of the total plant risk evaluated in the PSA study. Moreover, because of low probabilities of failure, the data available regarding passive failures is usually limited. This has naturally led to very limit treatment of such failures within PSA studies. Due consideration must be given to this fact and how the passive components may be treated in the consequence analysis" [31]. The next sub–section (7.3.2) discusses this issue in more detail.

If it is considered that the PSA does not fully meet the quality requirements for RI-ISI application, specific attention should be paid to its use in the consequence evaluation. The PSA may still provide useful information for the analysis, but in this situation it may be supported by complementary analyses that may be of qualitative nature" [16, 31].

7.3.2. Passive component failure treatment

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

- Pipe whip;
- Jet impingement;
- Decompression waves;
- Flooding; and
- High environmental temperatures, etc.

It is recognized that indirect effects of passive component boundary failures may have a significant influence on the consequence evaluation and it is therefore required that such effects be explicitly taken into account. Spatial consequences are determined based on the location of the failure and relative position of important equipment and it is recommended that the analyses are confirmed by a walk-down" [31].

The FMEA could include the evaluation of consequences of a spectrum of leak sizes and the analysis may be addressed the possibility to isolate the leak or break. Both automatic and manual isolation need to be considered.

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

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

- The PSA model and success criteria are used to define safety functions and backup trains;
- PSA results for all initiators are applied directly for relevant consequence impacts;
- PSA system and / or train unavailability are used to determine the reliability of mitigative equipment given a pressure boundary failure;
- Internal flood results are used in the analysis of spatial effects;
- Shutdown PSA, if available, is used in the evaluation of other modes of operation; and
- Level 2 PSA results are used to identify event sequences that provide the dominant contribution to containment performance (e.g. LERF) with respect to pipework failures, as applicable" [16, 31].

7.4. FAILURE POTENTIAL ASSESSMENT

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

Ideally, the probability of failure of components or sites that is potentially in need of inspection may be calculated in a quantitative way, implying the use of structural reliability models (SRMs). However, two important facts are recognized concerning the use of SRMs. Firstly, such models do not exist for all the potential degradation mechanisms that currently affect nuclear power plants. Secondly, for degradation mechanisms that do have a viable SRM, there is only a limited acceptance that these estimates can be seen as representing some form of true or absolute value. This implies that the evaluation of the probability of failure for all potential

ISI sites will necessarily yield a mixture of quantitative and qualitative assessments. Quantitative values, where they exist, may serve to quantify relative differences in the probability of failure from one site to another.

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

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

- Use of SRM, where they exist, to provide a good estimate of the relative differences in the failure probabilities;
- Statistical estimates based on both plant–specific and global databases in order to provide anchoring points for both the SRM analysis and the expert judgements; and
- Use of formal expert judgements using a combination of deterministic structural models and design insight.

It supposed to be recognized that there is not a single, optimal method for assigning probability of failure. As such, each above mentioned approach or combination of them, needs to address the issues identified herein" [16, 31].

7.4.1. Structural reliability models

"Whilst it is recognized that there are several degradation mechanisms not covered with the available analytical tools, SRM are essential tools in the evaluation of probabilities of failure for components of NPPs" [16, 31].

The objective of structural reliability analysis is to determine the probability of an event occurring during a specified reference period. It is essential to verify and validate any SRMs used in the evaluation of probabilities of failure ENIQ RP 9 [34].

The results from SRM provide a relative risk ranking appropriate for the purpose of developing a RI-ISI programme. It is important to recognize that the absolute values developed by SRMs need to be used with caution when used for other applications.

"An advantage of many SRMs is the possibility to quantify the influence of inspections both in terms of inspection capability and frequency. This is a key factor in RI-ISI since it is desirable to select the most appropriate inspection capability for every risk site" [16, 31]. The POD functions may be used to describe the efficiency or reliability of the inspections [35].

"Further discussion on requirements and recommendations for SRMs and associated software for RI-ISI applications are found in the reports produced within the NURBIM project" [16, 36].

7.4.2. Estimation from operating experience data

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

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

- Passive components usually have an increasing failure rate (ageing), and thus the exponential distribution does not correctly model the failure occurrence;
- The data quality may be insufficient for obtaining reliable estimates due to:

– Missing information related to the component population;
– Uncertainties related to failure mechanisms and root causes;

- Data is often very scarce.

Due to the shortcomings related to the quality and quantity of data, the estimates of passive component failure probabilities are subject to large uncertainties. For RI-ISI applications, probability of failure estimates obtainable from world–wide or generic data may not be sufficient. However, the data is extremely valuable in establishing prior probabilities. These values can then act as an anchor for the SRM estimates or expert judgement, using plant-specific information, to identify the distribution of the probability of failure throughout the plant–specific sites" [16, 31].

7.4.3. Use of expert judgement through expert elicitation

"The shortcomings in both SRM and operating experience data will sometimes limit a quantitative assessment of some of the active degradation mechanisms of interest. A possible alternative is to use expert judgement, preferably through the use of formal expert elicitation, to derive failure probabilities." A process how this expert elicitation could be performed is described in" [37].

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

It is recognized that experts are often not very familiar with probabilities, especially with subjective probability statements, and thus the training phase to give probabilistic estimates is important. The person leading the structured expert elicitation process supposed to have proper knowledge in decision analysis, probabilities and statistics. This person is called the normative expert. His, or her, responsibility is to facilitate the process by giving training, conducting the elicitation and aggregating the expert opinions. A detailed discussion on the expert assessment approach within the nuclear industry can be found, for example in" [38].

"Using an expert judgement for all sites, including those for which an SRM and / or possible statistical data exists, can combine qualitative and quantitative probability of failure estimates. The SRM and / or statistical data then act as both an anchor for the rankings and as a form of cross–correlation with the expert ranking. An example of such approach is described in" [16, 36].

7.5. RISK RANKING

This sub–section discusses risk characterization and risk ranking that is usually "developed to support establishing the ISI programme. Risk is defined in engineering terms as the product of the measure of the consequence resulting from a failure and the probability of that failure occurring within a given period of time. Combining the information from the probability of failure assessments and the consequence analyses forms the risk ranking. The risk ranking can be carried out at either element level or segment level. Guidance for conducting the risk ranking can be found in"[16, 31, 37, 38].

7.5.1. Graphical representation of risk

"Each segment or element can be ranked from highest to lowest according to its risk. Useful ways to evaluate the risk of failure and clearly represent it in a graphical way include the development of risk plots and / or risk matrices.

In risk plots, each component is represented as a point on a log-log plot. The consequence of failure is represented on the x–axis (the abscissa of the point). The probability of failure is represented on the y–axis (the ordinate of the point). Refer to Figure 13 for an example of a risk plot.

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

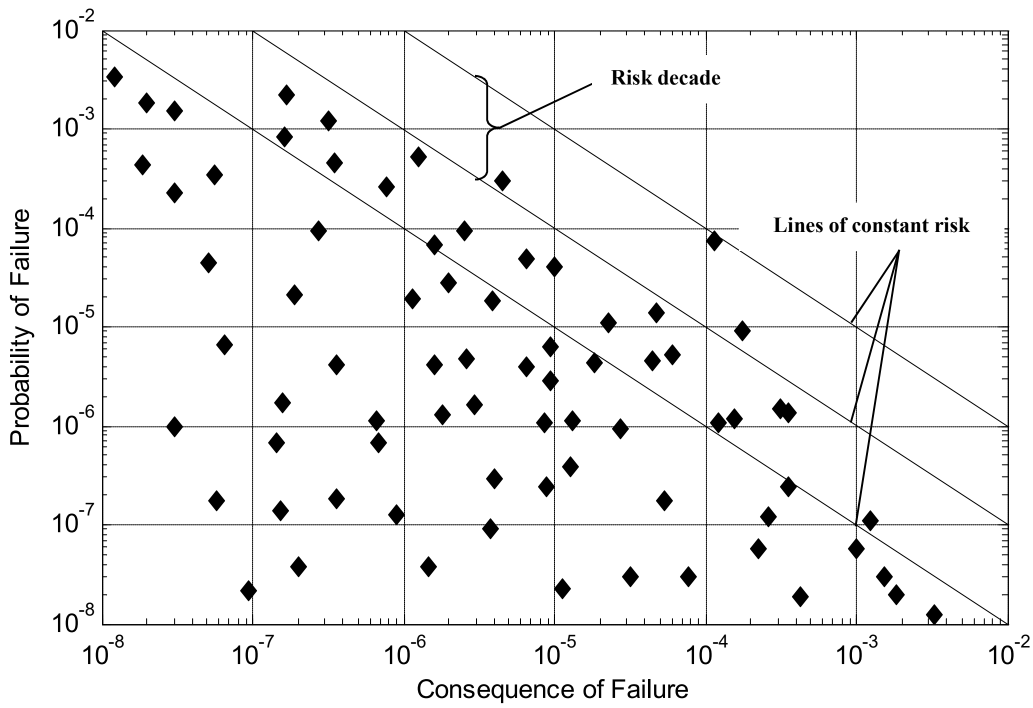

FIG. 13. Risk plot (the plot is purely illustrative) [12].

In a semi–quantitative approach to risk, probability of failure and consequence of failure are not numerically evaluated in absolute terms, but are ranked using either a qualitative scale such as high, medium, low or broad categories such as 10–3 to 10–4 etc. In this case, a risk matrix can be used to represent the rankings in the form of subsets as shown in Figure 14".

The risk plot and risk matrix are informative since they show if the risk is governed by the probability of failure or by its consequence. 'High consequence–low probability' sites require different considerations than 'high probability–low consequence' sites even if they have an equal total risk.

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

FIG. 14. Risk matrix.

The values used in this table are purely illustrative and should in no way be taken as a requirement"[16, 31].

7.5.2. Sensitivity analysis

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

The sensitivity studies can identify potential risk outliers by identifying ISI components that could dominate risk for various operational modes, PSA assumptions and data and model uncertainties" [16].

7.5.3. Safety-significant sites

"The development of a risk plot or ranking does not in itself identify sites that could be said to be safety–significant. Such a choice is subjective.

The first step in the process of determining risk-significant sites is the identification of risk outliers. Risk outliers are sites that have a much higher risk than the overall mean risk level for all sites.

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

Having identified the potentially safety–significant sites for a RI-SI programme, an expert panel could be used to review the proposed sites. This panel may review the information, analysis and insights that have been used to identify the safety–significant sites. It is also important to investigate alternative possibilities for mitigation against the risk. It is, therefore, necessary to look at the nature of the risk associated with each site. It may be possible to identify ways other than inspection to address the risk" [16, 31].

In determining the high safety significant sites needed for inspection, other important aspects supposed to be taken into consideration to include, sites with a consequence of failure but with very low failure potential and vice versa (see Fig. 15 below), as well plants with relatively flat risk profiles.

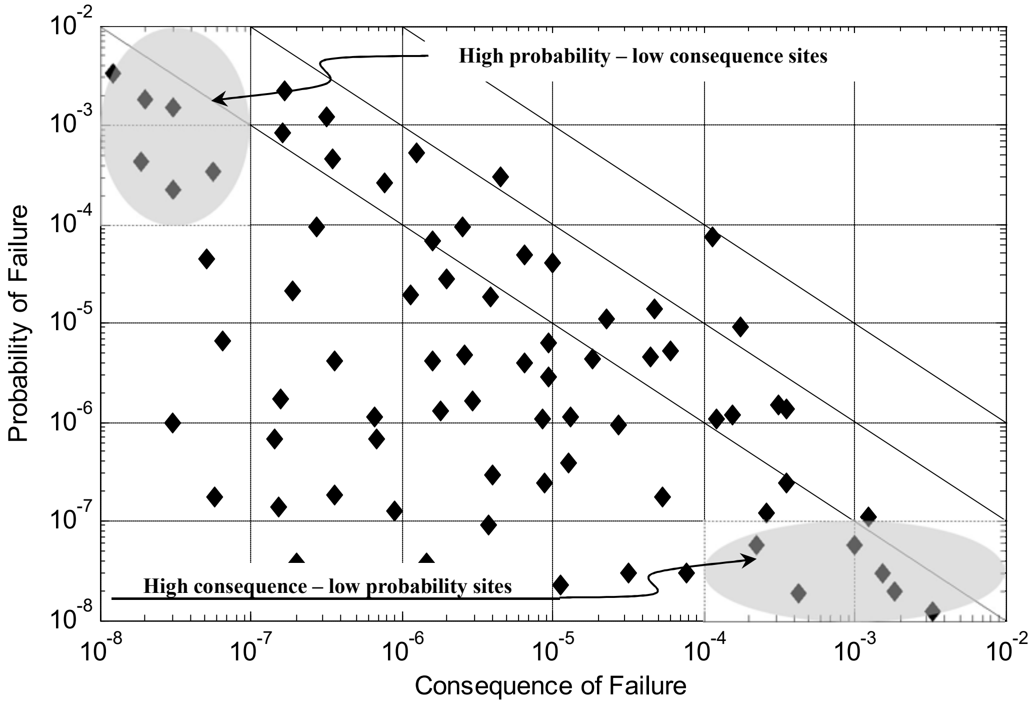

FIG. 15. Risk plot showing the high probability–low consequence region and the high consequence–low probability region [16].

"The plot is purely illustrative; the way the two regions are represented should in no way be taken as a requirement" [16, 31].

7.6. STRUCTURAL ELEMENT AND NDE SELECTION

“The overall principle underlying the definition of the RI-ISI programme, e.g. the identification and selection of individual sites for inspection, is that the proposed inspection programme supposed to provide defence in depth.

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

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

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

Sites that have a low probability of failure but high consequence could be considered for inclusion. Also, due consideration could be given to the low consequence but high probability of failure sites.

The scope of the RI-ISI programme may also need to be completed with sites requiring inspection in order to meet other legal requirements, for instance in relation to the safety protection of workers” [12].

7.7. RISK IMPACT ASSESSMENT

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

To gain confidence that the proposed new ISI programme is at least as effective as the current ISI programme in reducing risk, it is recommended that the two programmes are compared by using the inputs to the risk estimates” [16, 31] (e.g. failure frequency, conditional CD probability), the inspection intervals, inspection efficiencies and as applicable probability of detection. A reference [34, 39, 40] provides additional information on these types of assessments.

The sensitivity of the risk results to the above inputs could also be studied. This could include sensitivities on assumed failure frequencies, inspection intervals, inspection efficiencies and probabilities of detection. “These analyses could also be used the other way around to identify the level of inspection capability required for achieving a certain risk reduction.

When moving to a RI-ISI programme, at least risk neutrality or, better, risk reduction need to be achieved. The risk reduction achieved depends not only on the risk addressed, but also on the capability and frequency (intervals) of the inspections” [16, 31].

Risk reduction through implementation of RI-ISI is achieved through a variety of means. Examples include increasing the number of inspections at high risk sites, tailoring the inspection techniques to the degradation mechanism of interest (i.e. inspection for cause), identifying the appropriate examination volume (e.g. counter bores), and altering the inspection interval for aggressive degradation and improvements to the reliability of the NDE method.

“The overall principle underlying the definition of the RI-ISI programme, e.g. the identification and selection of individual sites for inspection, is that the proposed inspection programme must provide defence in depth” [16, 31].

7.8. PERIODIC UPDATE OF THE RI-ISI PROGRAMME

“The risk assessment provides a ‘snap–shot’ of the risk distribution within the ISI boundary at a given point in time. The determination of an effective risk-informed inspection strategy requires the development of a feedback procedure based on the idea of updating the risk ranking after plant changes affecting the probabilities of failure or consequences of failure have been made.

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

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

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

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

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

Even if no evidence of flaws is found after the performance of a certain number of risk-informed ISI inspections programme is completed, it is still very important to ponder the meaning of these results. The critical issue then becomes the capability of the inspection technique. Another could be the conservatism in the failure potential evaluation (e.g. for defence in depth purposes).

Some guidance regarding living PSA can be found in [41, 42]. Reference [43] provides eight examples of plants that have conducted updates to their RI-ISI programmes" [12].

Plants entering LTO (beyond initial plant design lifetime) may need to re-evaluate the RI-ISI programme (e.g. supporting analyses) to incorporate any new considerations that may be warranted caused by extended operation. Examples include additional cycles, environmental effects on fatigue life, thermal ageing and embrittlement.

7.9. STATUS OF RISK-INFORMED INSPECTION IN MEMBER COUNTRIES

In the USA, the USNRC has approved both the EPRI and pressurized water reactor owner group (PWROG) methodologies as valid alternatives to ASME Section XI. Further, it has generically approved Code Case N-716-1 [44] (i.e. EPRI Streamlined RI-ISI Approach) in Regulatory Guide 1.147, revision 17 [45] thereby eliminating the need for plant-specific submittals and regulatory review. RI-ISI is currently applied to all the units in US NPPs. In other countries, varying regulatory positions exist, ranging from country–specific methodologies to adaptation of existing approaches. The number of applications is constantly growing and is briefly summarized in the Table 2.

Additional information on RI-ISI methodologies and the status of applications is documented in [3].

TABLE 2. STATUS OF RI-ISI, WORLDWIDE

Country	**Status**
Applications	
China	Application of EPRI traditional methodology (Revised Risk-Informed In-service Inspection Evaluation Procedure, TR-112657,2000) at Tianwan NPP(Unit 1&2, PWR, WWER V-428), Daya Bay NPP(Unit 1&2,PWR, CPR-1000) and Ling Ao NPP(Unit 1&2,PWR, CPR-1000)
Finland	All operating plants (Loviisa WWER-440, Olkiluoto BWR & soon to be in operation Olkiluoto PWR) have implemented RI-ISI for all piping systems as is required by Finnish authority. Pilot project for RI-ISI in Finland was approved by the authority in 2006. Full scope RI-ISI projects under way (Loviisa WWER-440 & Olkiluoto BWR), using ASME Section XI, Appendix R (EPRI traditional methodology), but not following exactly the methodology
Mexico	Application of EPRI traditional methodology (Laguna Verde BWR), Class 1&2
Republic of Korea	Class 1 and 2 applications of PWROG methodology
Slovenia	Application of EPRI traditional methodology (Revised Risk-Informed In-service Inspection Evaluation Procedure, TR-112657) has been approved at the Krško Nuclear Power Plant (Krško NPP)
South Africa	Application of EPRI traditional methodology (Koeberg PWR), Class 1&2

Country	Status
Sweden	• Ringhals has approval for use of the PWROG-SE methodology • All Swedish BWR plants have ISI programme based on PMT-2004
USA	• EPRI methodologies: 78 plants • PWROG methodology: 3 plants • EPRI & PWROG methodology: 1 plant • Transitioning to EPRI methodologies: 18 plants
Pilot studies	
Bulgaria	Partial scope application of PWROG methodology
Canada	Pilot application to CANDU nuclear systems (CSA N285.4) and new standard for balance of plant systems (CSA N285.7) using EPRI traditional methodology
Czech Republic	EPRI pilot studies, several systems in Temelin (WWER-1000) and Dukovany (WWER-440) Risk-Weld methodology is applied by Czech NPPs Operator (CEZ). This methodology enables to determine actual level of risk and real condition of risk parts of technology systems with weld joints. Depending on the degree of risk, the necessary measures will be taken to ensure that all welded joints considered are expected to have an acceptable risk of degradation at the required time.
France	OMF-Structures methodology piloted to 12 systems
Lithuania	NURBIT RI-ISI approach pilot
Slovakia	Application under way, future steps dependent on pilot study results
Sweden	Oskarshamn and Forsmark pilot studies using NURBIT RI-ISI approach Pilot Study at Forsmark, Unit 3 using EPRI methodology
Switzerland	EPRI pilot study at Leibstadt, PWROG pilot study at Beznau
Ukraine	EPRI pilot study at Khmelnitsky WWER-1000
Other	
Belgium	Participating in international activities (e.g. RISMET)
Japan	Some activities taking place (e.g. RISMET)
Taiwan	Some activities taking place
UK	Application of EPRI traditional methodology (Sizewell B, PWR), Class 1 and 2 Risk–based ISI applied for nuclear submarines, not for NPPs

8. QUALIFICATION OR PERFORMANCE DEMONSTRATION OF NDE SYSTEMS

8.1. GENERAL

8.1.1. Scope and objectives

Inspection qualification or performance demonstration, by definition, is a process of systematic and independent assessment, by all those methods that are needed to provide reliable confirmation, of specific NDE system to ensure it is capable of achieving the required performance under real inspection conditions.

Reliable results of NDE in nuclear power industry are of utmost importance and fundamental for the safe operation of any nuclear power plant, therefore a failure to detect a flaw that may threaten nuclear power plants primary circuit integrity, or to declare flaw detection in an unflawed component of nuclear power plant is undesirable. Through the process of qualification (performance demonstration) an assessment of the capabilities and limitations of NDE systems is performed. An objective of qualification is to ensure that the detection, characterization and sizing of flaws, if presents, are reliably achieved throughout components of NPPs, hence resulting in effective NDE that contributes to the overall ISI effectiveness.

Consequently, the scope of qualification of NDE systems consists of three main elements: the equipment by which the examination is implemented; the personnel that perform the examinations, and the procedure according to which personnel properly perform NDE tasks using the applicable equipment.

8.1.2. Acronyms

Inspection Qualification is the term adopted by ENIQ and is now universally accepted in Europe whereas the performance demonstration is the term more commonly adopted in the U.S. and is required by Appendix VIII of ASME Section XI [4]. However, ASME XI refers to inspection qualification as well.

Both terms describe the process of independent assessment of NDE system, through performance demonstration is the term most commonly used to describe the practical assessment of NDE by using test pieces.

Appendix VIII of ASME Section XI describes the requirements for performance demonstration of ultrasonic examination systems using blind test pieces that integrate personnel, equipment and procedures into a single entity.

ENIQ describes a methodology of how an inspection qualification can be performed and used for all of NDE techniques [5]. The inspection procedure, equipment and personnel may to be qualified separately, including a technical justification, on open and blind test pieces.

8.2. QUALIFICATION PROCESS

8.2.1. General principles

Rules and regulations that define the requirements for the ISI besides defining the scope, frequency and methods to be applied also prescribe the requirements for the qualification of NDE systems. In currently existing approaches to qualification of NDE systems, the qualification flow chart, see Figure 16, is generally very similar. However, depending on applied rules and regulations there can be differences in certain stages.

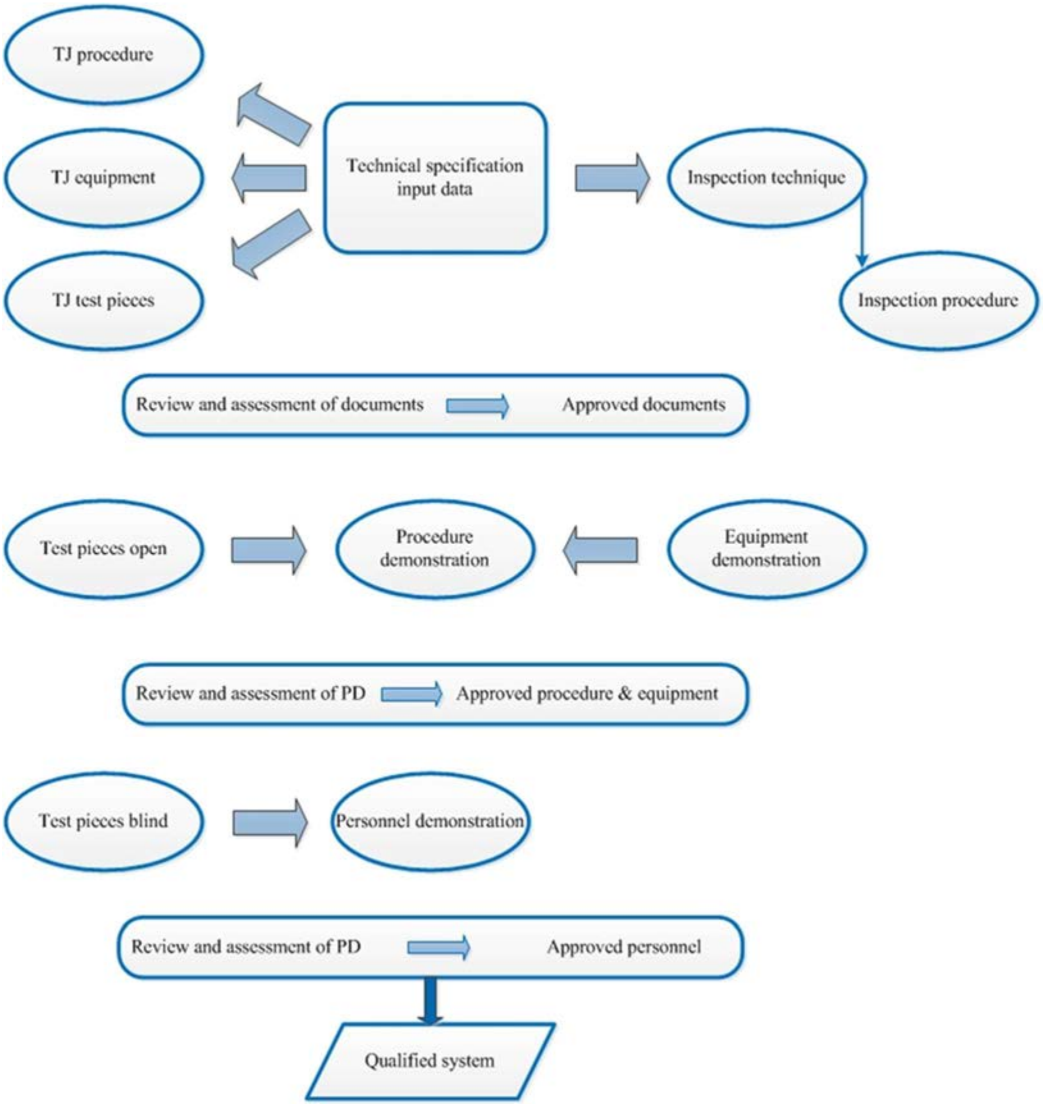

FIG. 16. Example of a Qualification Process

The detailed scope of a qualification process, in terms of required inspection area(s) and NDE method(s) as well as flaws being sought and required examination effectiveness is defined in written form before starting any qualification process. This information is provided by the reference codes and standards or in form of key qualification document typically entitled technical specification. This publication is produced by the licensee.

The qualification procedure or protocol is produced taking into account applicable rules and regulations. All the input information is set out at the beginning and typically contains an orderly sequence of steps describing how a specific combination of NDE procedure, equipment and personnel applied to a specific inspection area has to be qualified. This includes generation of the technical justification, required test specimens, type and location of flaws, conditions of the practical trials, grading and success criteria for practical trials and any other special requirements, where applicable.

Based on the detailed input information, the inspection organization develops equipment (manipulator, if applicable for automated examinations) for inspection and qualification, a technical justification and an inspection procedure. According to the qualification process, relevant data onto a specific inspection situation typically presented in a document called a technical justification. The technical justification contains the combined presentation of all the work carried out and all the information produced to substantiate and justify that the inspection system satisfies the requirements stipulated in conjunction with a specific inspection situation.

An open test block can be used for the initial preparation and adequacy of inspection technique and procedure. The same open test piece(s) is normally also used through the performance of the procedure qualification.

Following successful open practical trials, blind trials are performed for qualification of personnel. In some countries the qualification of personnel maybe accepted through certification scheme per national or international codes and standards. In some qualification approaches blind test pieces can also be used for qualification of procedure. Depending on the qualification approach, an inspection manipulator used for practical trials does not need to be a complete inspection manipulator, but can be just a simple scanner device that has the elementary functions of full scale robot. Parts of the inspection manipulator that have an important influence of the inspection result, for instance the probe arrangement, needs to be included in the simplified qualification equipment. Control of inspection organization equipment (hardware and software) with regard to changes, additions, revisions, security can be achieved.

Upon successful qualification of the complete system, the qualification body issues a qualification report / protocol and certificates to the inspection organization for all elements of NDE system (equipment, procedure and personnel) and assembles all relevant information into a qualification dossier, which is open for review and assessment by the regulatory body.

Following qualification, the degree of post qualification support to be provided by the qualification body is agreed between the parties, e.g. provisions for feedback from site experience, re–qualification.

The qualification of procedure will be valid as long as:

- The essential variables are within the tolerances of the qualified procedure;
- There are no changes to the qualification requirements; and
- Practical experience does not reveal any failure to detect or correctly sentence those flaws for which it has been qualified.

Personnel qualifications may have a limited validity and one way is to use the same period as for the basic NDE certification according to ISO 9712:2012 [46] (a maximum of five years), provided the personnel work regularly with the equipment and procedures for which they have been qualified, and that they receive appropriate annual training.

Nevertheless, training before inspection with the actual inspection procedure is important independently of requalification of personnel or not.

8.2.2. Qualification methodologies

Two main qualification methodologies for performance of inspection qualification exist, ASME / PDI and ENIQ. A third approach, which was primarily developed for WWER nuclear power plants is the IAEA methodology which combines the ENIQ and ASME approaches. The following paragraphs focus on the two main qualification methodologies.

8.2.2.1. ASME methodology

This methodology is highly based on practical trials conducted on full scale representative test pieces, with large number of flaws required, resembling the component to be inspected. Types, locations and sizes of flaws that have to be detected, including assessment criteria and tolerances are based on the ASME Code [4].

The ASME approach is a generic qualification, not specific to any particular plant or to any specific flaw type (with some exceptions). Accordingly, the approach easily lends itself to a collaborative performance demonstration administrator (PDA) for the qualification programme which is the recognized qualification body. In the United States, the NPP licensees established collaboration called the PDI to provide the independent services of a PDA. This organization is managed by the Electric Power Research Institute (EPRI). The qualification certificates issued by EPRI / PDI are called Performance Demonstration Qualification Summary (PDQS), are valid for all members (all U.S. utilities and some non US utilities).

8.2.2.2. ENIQ methodology

ENIQ is based on documentation preparation in combination with practical trials on representative open and blind test pieces, with a large number of flaws required, resembling the component to be inspected.

ENIQ as a qualification methodology is a recommendation of how a qualification of an inspection system can be performed, and the approach typically results in qualified NDE system specific to particular NPP. Two main distinctions between ASME and PDI are the requirement of a technical justification and the qualification of procedure on open test pieces. Types, locations and sizes of flaws that have to be detected, including assessment criteria and tolerances, can be taken from a code and standard or be based on structural analysis and fracture mechanic calculations. The qualification procedure is written by the qualification body, and is submitted to the plant operator for acceptance.

8.2.2.3. The IAEA methodology

This methodology combined the ENIQ and ASME approaches to provide a consistent and practical strategy in methodology for WWER NPPs entitled “Methodology for Qualification of In–Service Inspection Systems for WWER Nuclear Power Plants” [6].

8.2.3. Input information

8.2.3.1. ASME code

Input Information is not a standard requirement for qualification by the methodology specified in ASME code. Instead, the general requirements for qualification of non–destructive examination personnel contained in Section XI, IWA-2300, are amended by Section XI, Appendix VIII. Appendix VIII describes the additional requirements for performance demonstration of ultrasonic examination systems that integrate personnel, equipment, and procedures and also includes 14 Supplements that contain specific instructions for the conduct of performance demonstrations, including: specimen requirements; conduct of performance demonstration and acceptance criteria.

The requirements for components not included in the above are given in ASME Section V or in Appendix I, III and VII of ASME Section XI, and in applicable rules and regulations

(10CFR50.55a and Request for Relief or Code Cases (Alternative Requirements) that the NRC has approved). In addition, for some qualifications EPRI prepares qualification protocols (a document which is very similar to ENIQ's / IAEA's Input Information) for specific component examination that contains all required information about component designs, examination objectives, test pieces descriptions, examination methods, qualification programme including demonstration process and flaws, review of procedure, acceptance (procedure and personnel demonstration, detection and false calls, location tolerance, length and depth sizing, orientation, essential variables, etc.), reporting criteria, etc.

8.2.3.2. ENIQ method

In order to create good prerequisite requirements for vendor and qualification body, reliable and correct data about the test object is required. This data comprises both basic information from the inspection documentation, as well as a series of different object–specific details and working environment factors. These jointly makeup the object description.

It is licensee's responsibility to ensure that the necessary information is produced. In order to facilitate the production of inspection objectives, information can be summarized in an inspection datasheet.

The licensee releases the draft inspection datasheet for comment by the qualification body (QB) and any other relevant involved parties, to obtain consensus that the requirement is properly described.

In defining the content and format of an inspection data sheet, there are a number of key points:

- A full description of the component to be tested including material, surface finish and access;
- Type, dimension, orientation and location of flaws to be detected and / or sized, depending on the flaw situation considered;
- The inspection performance (detection, characterization, sizing and location) to be achieved;
- NDT procedure, equipment and personnel requirements; and
- Environmental consideration if applicable.

The ENIQ document "Guidance on the Specification of Inspection and Inspection Qualification Requirements" can be found at NUGENIA webpage and to be used as a template [47].

8.2.4. Technical Justification

8.2.4.1. ENIQ method

According to the qualification process, relevant data for a specific inspection situation must be presented in a document called a technical justification. The technical justification contains the combined presentation of all the work carried out and all the information produced to substantiate and justify that the inspection system satisfies the requirements stipulated in conjunction with a specific inspection situation. The structure of the technical justification essentially conforms to that recommended by ENIQ in "Recommended Practice 2" (RP2) [48]. The accredited testing laboratory normally prepares the technical justifications. However, some information that must be included in the technical justification may be supplied by other sources. For example, the licensee normally provides information concerning the object as well as the requirements that the inspection system must satisfy.

It is important to understand that the structure of a technical justification is chosen to ensure that also the reader will understand what is being dealt with, and not only for the author. When preparing the data that will be described in the technical justification, this work will probably be carried out in a different order for the subdivision shown below.

A technical justification may be structured in accordance with the RP 2:

1) Summary
2) Section 1: Introduction;
3) Section 2: Summary of Relevant Input Information;
4) Section 3: Overview of Inspection System 6;
5) Section 4: Analysis of the Influential and Essential Parameters;
6) Section 5: Physical Reasoning (Qualitative Assessment);
7) Section 6: Prediction by Modelling (Quantitative Assessment);
8) Section 7: Experimental Evidence;
9) Section 8: Parametric studies;
10) Section 9: Equipment, Data Analysis and Personnel Requirements 8;
11) Section 10: Review of Evidence Presented;
12) Section 11: Conclusions and Recommendations; and
13) References

An analysis of the conditions will be presented, in order to see which individual parameters they influence. These parameters will then be listed and an assessment will be conducted as to whether they are considered to be solely influential or essential parameters.

8.2.4.2. ASME method

The technical justification is required on a case by case basis when qualification is performed according to the ASME requirements. It is required, for example, for performance demonstration of one side examination of reactor vessel nozzle to shell welds or reactor upper head penetration ultrasonic examinations for volumetric leak path procedures qualification such cases, the technical justification is prepared to meet the requirements of ASME Section V, Article 14.

8.2.5. Test pieces

The ASME and ENIQ approach relies highly on performance demonstration by using full–scale test pieces. The design of test pieces is based on the information taken from the technical specification.

Practical trials may involve test pieces replicating the component being inspected in size and geometry. The defective condition may also be accurately replicated. If metallurgical flaws are involved, the test pieces may be designed to contain flaws in the type judged to be possible in appropriate positions and include the ‘worst case’ flaws judged the most difficult to detect, characterized and sized for the given inspection situation.

Such test pieces will produce realistic results but are expensive to manufacture and can usually only replicate a small fraction of the flaws which might occur.

Test pieces are essential for open and blind trials of qualification process, together with the technical justification. Open trial is a practical demonstration in which the inspection personnel is previously informed on the type, number and characteristics of the test pieces as well as on the type, morphology, position and dimensions of the flaws to be detected and sized.

Open trial test pieces are used in the course of preparation and verifying adequacy of NDE procedure and equipment, training for personnel, etc.

Blind trial test pieces are used for practical demonstrations in which the inspection personnel have no detailed knowledge of the number, position and size of any flaw. ASME section XI also specifies blind test pieces are to be used for procedure qualification, allowing the qualification of equipment, procedure and personnel to be carried out simultaneously.

When and where practically possible, a representative full–scale test pieces are fabricated. Original dimensions and production fabrication methods, including welding processes, are used. A large number of flaws with well–controlled and well–known sizes and locations are placed in test pieces included flaws with different depths, lengths, positions and orientations. The flaw simulation technique which is used to implant flaws in test pieces has a very important role. The signal response with actual inspection technique may similar to the corresponding real flaws.

8.2.6. Equipment Qualification

Equipment, as per ASME Section XI Appendix VIII approach, is qualified together with the procedure through the blind trials. Equipment essential parameters with allowable values and tolerances are identified within the procedure, and verified/measured, as appropriate, during the practical demonstration.

Within the ENIQ methodology, equipment (manipulator) can be qualified together with the procedure on the open demonstration or be qualified separately on an object specific mock–up. If equipment is qualified by itself, a technical justification is presented for review by the QB, together with a practical demonstration on a mock–up. A certificate is valid as long as no modification has been done of the equipment.

8.2.7. Procedure qualification

The purpose of an inspection procedure is to be an important instruction for inspectors. Therefore, all inspection procedures should be written in an unambiguous way, such that different inspectors will do the same and come to a similar result when they follow the procedure, i.e. a clear instruction describing what and how to perform the inspection, and not why. The "why" will be described in the technical justification?

The qualification within ENIQ is to demonstrate the inspection procedure step–by–step on open test pieces. It must be demonstrated the procedure instructions guide the examiner to detect, characterize and size the flaws within stipulated criteria and tolerances.

The certificate is valid unless no changes have been done of the technique.

ASME appendix VIII provides a list of the essential variables whose value must be specified in the inspection procedure to ensure that there are no unspecified variables which could cause the performance to vary from that established by qualification.

Qualification requirements as per Appendix VIII are based on a number of flaws in blind test pieces. For procedure qualifications the required number of flaws is at least equal to three personnel performance demonstration test sets from the blind trials. At least one successful personnel qualification may be performed and successful personnel qualification might be combined to satisfy requirements for procedure qualification.

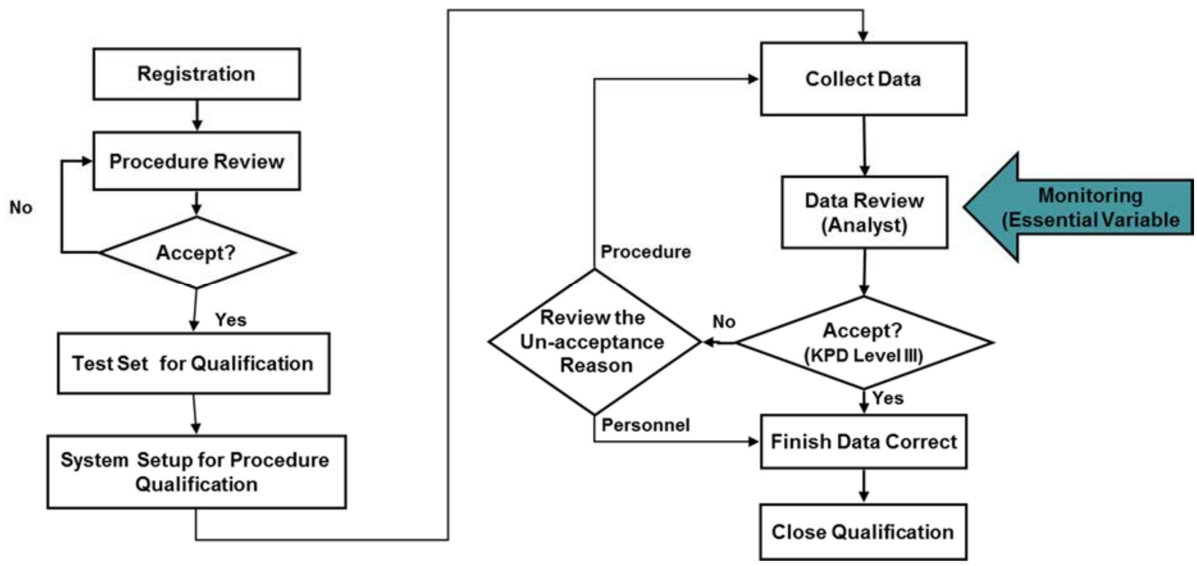

FIG. 17. ASME PD procedure process.

The procedures must have clear criteria for reporting and the analysis of detected indications. The procedure may define the responsibilities for resolution and disposition of all indications reported.

8.2.8. Personnel Qualification

This ENIQ recommended practice 10 (RP10) [49] provides recommendations for the qualification of inspection personnel where this is required. The recommended practice does not give guidance of when personnel qualification could be performed—this is an issue to be agreed with the relevant organizations.

The RP10 [49] is relevant to any non–destructive testing method. It is emphasized that the general principles given in this recommended practice can also be used for qualification of manufacturing inspections.

The principal objective of personnel qualification is to ensure that those carrying out an inspection are appropriately trained, experienced and examined to ensure it is applied correctly and effectively. Automated inspections usually involve several stages which may be performed by different personnel: for example, manipulator operators and data collectors and data analysts. It may be necessary to qualify some or all of the personnel undertaking these roles in different ways to demonstrate that they are capable of performing the tasks required of them.

For instance, when qualifying automated inspection and techniques, the need to qualify manipulator operator's decrees, because the positioning is checked by the data acquisition and data analyst operators.

It is necessary, when an inspection procedure is developed, to determine the requirements for personnel who will carry out the inspection. These supposed to be clearly defined and will be determined by a number of factors:

- Whether the inspection is manual or automated and the different roles fulfilled by different groups of personnel in the latter case;
- If the inspection is a manual one, whether the inspection imposes technical demands beyond those examined through a national certification scheme such as those discussed above; and
- If the inspection is automated, whether it has features which require particular skills beyond those normal for automated inspections.

In the ENIQ methodology personnel qualification is done through one or any combination of the following:

- Theoretical and / or open practical examination; or
- Blind trials.

In some cases, personnel are approved through a national NDT personnel certification scheme, but this is not the same as qualification on an object specific inspection procedure.

Qualification of personnel as per ASME Section XI Appendix VIII approach, for both manual and automated examinations, is exclusively through the blind trials. For automated examinations the qualification only applies to data analysis personnel. An initial requirement for personnel qualification is that candidate is certified to at least Level II through a national NDE certification scheme. Successful personnel qualification might be combined to satisfy requirements for procedure qualification. Other personnel (data operators / acquisition, supporting personnel, etc.) requirements including their training requirements are specified in the examination procedure. Qualification of personnel for manual examinations is often performed using generic qualified procedures and equipment, however many vendor organizations also qualify their own proprietary manual procedures. Criteria for successful personnel qualification with regard to detection and false calls, location tolerance, length and depth sizing are given in respective supplements of Appendix VIII and qualification protocols.

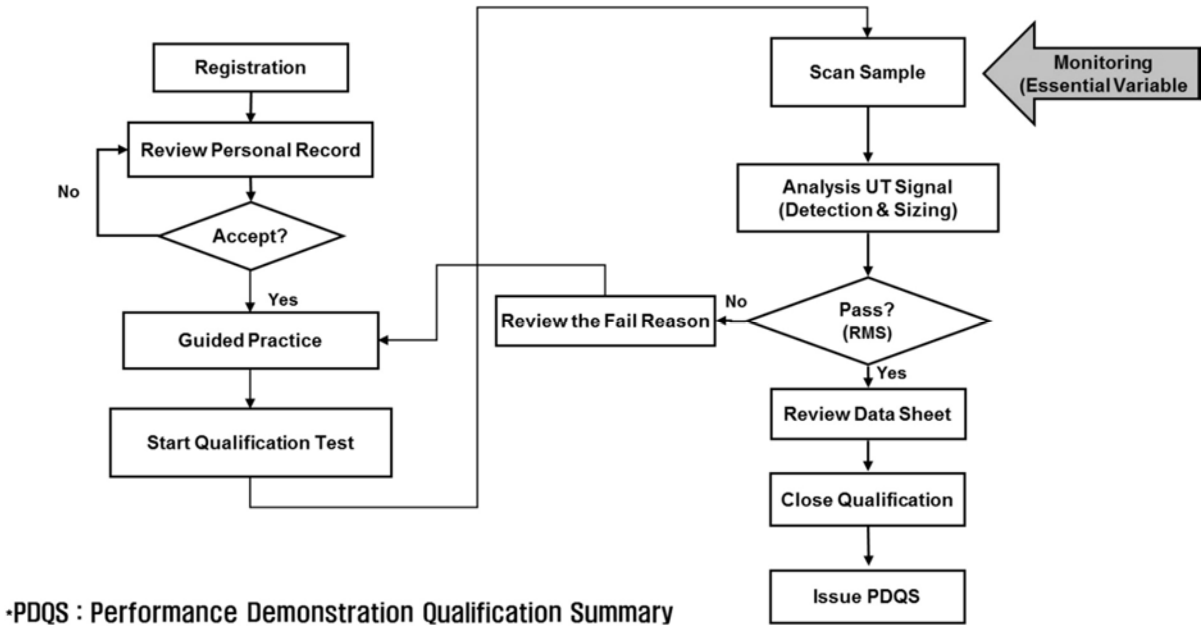

FIG. 18. ASME PD personal manual qualification process.

The IAEA methodology for qualification is very similar to the ENIQ methodology. The main differences are in the personnel qualification requirements. In the IAEA methodology is personnel qualified only through practical trials under 'blind' conditions.

8.2.9. Personnel re–qualification

Following successful qualification, the QB issues a qualification report or protocol and certificates to the inspection organization for all elements of NDE system (equipment, procedure and personnel).

Qualification certificates for procedures and equipment is valid indefinitely for all qualification approaches unless changes are made to the procedures or equipment or to any mandatory code whose requirements must be met.

Per ENIQ / IAEA approach the validity of personnel qualification certificates is limited in time and is complementary to national certificates (typical 3–5 years). The ASME approach is not limited in time, except Appendix VII of ASME Section XI requires annual training, at least 8 hours per year, to maintain personnel examination skills. Appendix VII requires that personnel practice UT techniques by examining or by analysing pre-recorded data from material or welds containing flaws similar to those that may be encountered during in-service examinations.

Operator qualification can be extended through demonstration or through technical justification, and could also be valid for inspection with other procedures. Validation of these other procedures may be based on equivalent technology, related equipment, calibration and evaluation instructions and be judged on operator's ability to make the same demands on data collection, detection, characterization and sizing.

8.3. QUALIFICATION DOSSIER

Records of performance demonstration as per ASME Section XI Appendix VIII and ENIQ approach and their results are maintained by QB in accordance with their QA Programme and internal procedures for document control and security of information. This collection of documents and records is called qualification dossiers. The QB has a secrecy agreement with all utilities and vendors to secure that all confidential information is handled in a correct way.

8.4. OPERATIONAL FEEDBACK OF NON CONFORMANCE

Operational feedback is an important element of qualification and may be both positive (to justify the qualification) and negative. However, if feedback from site examination results in evidence which is not in conformance with what was demonstrated during the qualification, it consequently requires change in the qualification dossier.

Example include, but are not limited to insufficient scope of examination due to previously unknown obstacles, incorrect location of the object which was subject of examination, environmental conditions (temperature, humidity, noise, radiation, etc.) that affect both the examination and examination results, etc. In such a case, qualification results are reanalysed, qualification certificates updated or withdrawn and qualification repeated.

8.5. STATUS OF QUALIFICATION OF NDE SYSTEMS IN MEMBER COUNTRIES

8.5.1. ASME / PDI methodology

The ASME methodology for qualification of NDE systems is applied in all U.S. nuclear utilities (U.S. Participants) and in several Member States worldwide (Non–U.S. Participants). Some Member States (Non–U.S. Participants) involved in PDI programme include, Brazil, Taiwan, Spain, South Korea, South Africa and Switzerland.

The PDI program maintains a controlled web site that contains a variety of documents and information that allows authorized users to explore and retrieve information at their convenience. Utility members, PDI staff, vendors and candidates can view and download numerous information and PDI products and documents. As applicable, both past and present revisions of these products and other information are available.

8.5.2. ENIQ and IAEA methodology

The ENIQ is a framework and not a code or standard, and it’s up to each country regulatory body to require the scope of qualification. In Europe, and some countries outside Europe, ENIQ is chosen as the methodology for qualifications of NDE systems. Similarly, the IAEA methodology is guideline developed specially for WWER nuclear power plants and it’s up to each country regulatory body with installed WWER nuclear power plant to require the scope of qualification in accordance with it.

Table 3 presents a brief summary of how ENIQ and IAEA methodologies are applied in some Member States, where data was available.

TABLE 3. BRIEF SUMMARY METHODOLOGIES APPLIED IN SOME MEMBER STATES

Country	**Status**		
	Regulatory Requirements	*Test pieces and Practical trials*	*Personnel Qualification*
Canada	The periodic inspection standard CAN/CSA N285.4 defines the requirement. Compliance with the standard is an operating license requirement imposed by the nuclear Regulator. The law is indirect – the federal law gives the regulator the authority to set requirements. There is disconnecting within the standard of what's included in the scope. The qualification clause makes the requirements universal i.e. any NDE method, any component, any application, but both procedure and personnel. There is a limitation to volumetric methods.	Requirements on test pieces for material, welding, geometry and defects has to match those objects in plant. Deviation from postulated mechanism has to be justified. Information of blind test pieces is confidential.	Personnel qualification is based on blind testing – either physical test pieces or recorded data. No information of validity period.
Czech Republic	There are a legal requirement for qualification of ISI, but usually connected with changes depending on operating requirements or prolongation of inspection periods. The scope is to fulfil Regulatory Body requirements, but mainly volumetric techniques are required to be qualified. The qualification follows ENIQ methodology.	Requirements on test pieces for material, welding, geometry and defects has to be the same as the inspected one or close as much as possible. No specific requirements for confidentiality.	Consistent with the ENIQ methodology, qualification of the personnel related to the inspection system is not required
France	A French ministerial order requires the qualification of all NDE-systems performed for ISI of the main components of primary and secondary circuits (PWR reactor).	Test pieces are representative to the real component related to the physical phenomenon used, and defects are representative out from the licensee requirement description. No confidential requirements are necessary when only open test pieces are used.	Consistent with the ENIQ methodology, qualification of the personnel related to the inspection system is not required.
Finland	ISI are qualified according to Finnish qualification rules YVL E5, closely following ENIQ-RP. It includes open and blind test blocks and also an assessment of technical justifications as described in ENIQ. Evaluation criteria for qualification assessments are based on ASME Code and they use RMS	Over the years, Fortum has gathered experience on inspection qualifications and fabrication of its own test blocks and flaws. Both open	All personnel for data analysis shall be qualified through practical demonstrations on blind test pieces or through technical justification. The

	instead of confidence level for sizing analysis. Finland also has a different inspection interval on reactor vessels between Loviisa and TVO. The interval for Loviisa is 8 years and for TVO it is 10 years, the same as in Sweden. Inspecta Certification is the qualification body accredited by Finnish Accreditation Service (FINAS) and approved by the Finnish regulatory body STUK, but qualifications are more like an ad-hoc, then QB involve personnel from inspection laboratories (Dekra) and from licensee on case by case. The scope of qualification activities are very similar to what is done in Sweden.	and blind test blocks are used. The same approach is also used by TVO, but they are not manufacturing own test blocks. Sometimes they offer this manufacturing from AREVA Uddcomb and also implant true flaw cracks. Practical trials are essential part of qualification of inspection systems. Qualification of procedures and equipment can depend on the inspection method be performed by technical justification and either by blind or open trials or a combination of these.	last information I got was that no qualification of personnel for data acquisition is required. Evaluation is based on ASME rules. The validity period is 5 years. All personnel involved in ISI by NDT shall have as minimum Level 2 certificate according to ISO 9712 in the method concerned. For UT personnel, additional training and theoretical exam on crack detection is required. In the qualification of inspection systems, personnel qualification is performed by blind trials. For data acquisition, necessity for qualification is decided case by case.
Hungary	Hungarian Nuclear Safety Rules require the qualification of ISI / NDE systems used for inspection of safety relevant components and of pressurized components under regulatory supervision. The qualification will follow ENIQ methodology.	Material, geometry and flaws of test pieces should simulate that of the components to be inspected. Special test pieces are usually borrowed from other WWER operators.	One pilot personnel qualification project was completed so far for butt weld inspection on austenitic stainless steel pipe. Blind test piece and consultancy service for the pilot qualification was provided by SQC.
United Kingdom	Regulatory requirements are relatively complex and not easily described. Its split between personnel and Equipment and mainly UT and RT are included, with occasionally ET.	Requirements on test pieces for material, welding, geometry and defects has to be simulated out from the actual component. Only approved personnel have access to the confidentiality of test piece information.	Personnel are qualified for different tasks such as data acquisition and data analysis. Criteria for pass or fail are based on successful blind trial together with evidence of approved training. The validity period is 5 years.

Sweden	The Swedish regulations require that ISI on nuclear safety related mechanical systems has to be qualified by an independent and impartial organization, the QB. The regulation strongly recommends the ENIQ methodology and corresponding RP and should include all aspects of the NDE-system. Procedure, Equipment and personnel for inspection of RPV´s and other primary systems in inspection groups A and B will be performed with a qualified system. All NDE-methods used for crack detection, characterization and sizing are included. The technical justification has an important role, more than in many other countries.	Practical trials on test pieces are an important part and have a very high influence of the decision of inspection qualification, together with the technical justification. The defect simulation techniques, which are used to implant flaws in test pieces, have a very important role, and the signal response with actual inspection technique ought to be similar as from a corresponding real flaw. Open test pieces are used to demonstrate the Inspection Procedure and Blind test pieces to demonstrate personnel ability to detect, characterize and size flaws.	All personnel for data acquisition and data analysis ought to be qualified through practical demonstrations on blind test pieces or through technical justification. Successful blind trials with an 80% hit rate for detection and characterization, and within stipulated tolerances with a 70% confidence level for sizing. The validity period is 5 years.

9. CONCLUDING REMARKS

This report describes the principles for improving ISI effectiveness. Improvements in ISI effectiveness will result in increases in plant safety and reductions in inspection program costs, worker exposure and radioactive wastes. Concepts for improving the fundamental elements of ISI are examined, namely, improving the selection of inspection scope and interval as well as improving the effectiveness of NDE (e.g. increased NDE reliability, incorporation of human factors considerations).

As an example, the use of risk-informed insights to improve the scope, selection and the role of performance demonstration (or inspection qualification) in improving the effectiveness of NDE is discussed. While there are several risk–informed ISI methodologies that have been developed and are in use worldwide, each has particular features that may make one or the other more appropriate for a particular plant application. Accordingly, a lesson learned is that plant managers could use practical guidance for evaluating these methodologies to determine the approach that would best fit their particular situations. For example, the available methodologies require somewhat different skill sets and different levels of support required from the plant staff. A particular plant manager may therefore need to evaluate the resources available to support implementation. As a future effort, scoping tools could be developed to aid plant staff to easily determine the most practical and effective selection of a scope of application.

In parallel, inspection qualification / performance demonstration is also discussed in detail. Through the process of qualification (or performance demonstration) and incorporation of human factors, an assessment of the capabilities and limitations of NDE systems can be determined. The objective is to ensure that the flaw detection, characterization and sizing are reliably achieved throughout NPP components, hence resulting in effective NDE that contributes to the overall ISI effectiveness and thus to plant safety and cost-effectiveness use of limited plant/regulatory resources.

The relationship between ISI effectiveness and cost are discussed in this report. Inspection scope selection and NDE effectiveness (including performance demonstration / inspection qualification, human factors) have strong and complex influences on total cost. These complex relationships make it difficult for specialists to evaluate how best to optimize the scope-cost-NDE effectiveness relationship. As plant managers require practical tools and models that can be used to guide them to make informed decisions about approaches to be taken. New tools and models could potentially take generic inputs related to technical and economic issues as well as particular requirements for each specific situation an provide consistent guidance on a path forward.

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GLOSSARY

Amplitude. The vertical height of a signal (measured from base to peak) on a screen with its numerical value representing the energy received from a reflector.

Consequence. The impact or the ultimate result of an event. Consequences can be measured in terms of impact on public safety, impact on the environment and cost or damage to the plant.

Core damage. Uncovery and heat-up of the reactor core to the point where damage to reactor fuel element or cladding is anticipated.

Core damage frequency. An estimated frequency of occurrence of events leading to core damage.

Damage. See degradation.

Defect. Macroscopic imperfection. Includes flaws as well as other macroscopic imperfections like over penetration in welds that exceed acceptance standards.

Degradation. Phenomena or process that attacks (wear, cracking etc.) the component material and might result in a reduction of component integrity.

Distance amplitude curve. A curve constructed from the peak amplitude responses from reflectors of equal area at different distances in the same material. This techniques are important because of the amplitude of ultrasonic pulses varies with distance from the probe, and this needs to be compensated in order to perform the evaluation on a constant sensitivity level.

Fission product release frequency. An estimate of the likelihood of radioactivity release involving release of airborne fission products.

Flaw. An imperfection or discontinuity that may be detectable by NDE and is not necessarily rejectable.

Indication, The response or evidence from the application of a NDE.

Incredibility of failure. A systematic compound of measures from the design to the operational life of a component ensuring that its failure frequency is less than 10^{-7} per year.

Inspection procedure. A document specifying all essential parameters and setting out the precautions to be observed when applying an inspection technique for a specific inspection.

Inspection qualification. The systematic assessment, by all those methods that are needed to provide reliable confirmation, of an inspection system to ensure it is capable of achieving the required performance under real inspection conditions.

Inspection system. All parts of the non-destructive examination including equipment, inspection procedure and personnel which can influence the outcome and quality of inspection.

ISI. A periodic non-destructive examination of nuclear power plant components in order to provide information about their current condition and any damage, flaw or degradation that might occur.

Iso-risk lines. Straight lines in the risk matrix connecting data points representing the same risk level.

Inspection validation. Term used to describe the qualification of the inspection system carried out for Sizewell B in the UK.

Large early release. A radioactivity release from the containment involving the rapid unscrubbed release of airborne fission products to the environment.

Large early release frequency. An estimate of the likelihood of severe accident associated with a radioactive release from the containment occurring before the effective implementation of off-site emergency response and protective actions.

Last pass heat sink weld. A specific welding process which imposes compressive stresses on the inner layer of the wall thickness of stainless steel piping.

Ligament. Distance between the flaw and closest component surface.

Linear flaw. A flaw having finite length and narrow uniform width and depth.

Model reflector. Well defined reflectors, used to establish amplitude levels in order to compare detected indications with these levels.

Modelling. The use of mathematical models of NDE to predict quantitatively the outcome of the inspection.

Performance demonstration. The process of qualification of an inspection system according to ASME Section XI, Appendix VIII.

Phased array technique. Application of ultrasonic transducers subdivided into a number of elements. The timing of their excitation can be individually controlled in order to produce beam steering or focusing.

Planar flaw. A flat two-dimensional flaw oriented in a plane other than parallel to the surface of the component.

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

Qualification. See inspection qualification.

Qualification body. Organisation that are approved to conducts inspection qualification.

Reflector. Interface at which an ultrasonic wave encounters a change in acoustic impedance.

Risk. The product of the measure of the (generally undesirable) consequence of an initiating event, and the probability of that event occurring within a given period of time.

Scanning. Systematic movement of the probe over the material to be tested. It can be performed manually or automatically.

Structural reliability model. Prediction of the performance of a component or system based on probabilistic input data e.g. probabilistic fracture mechanics and flaw distribution.

Worst case consideration. Defined as those cases of flaws, component geometry etc., which are likely to present the greatest challenges for detection and / or sizing within the framework of specific situation considered for inspection.

ABBREVIATIONS

AB	Aktiebolag / limited company
ALARA	as low as reasonably achievable
ANI	authorized nuclear inspector
ASME	The American Society of Mechanical Engineers
ASNT	The American Society for Non-destructive Testing
BWR	boiling water reactor
CANDU	Canada deuterium-uranium reactor
CD	core damage
CDF	core damage frequency
COF	consequence of failure
COFREND	Confédération Française pour les Essais Non Destructifs
CR	computed radiography
CSA	Canadian Standard Association
DMW	dissimilar metal welding
DOD	Department of Defence
DR	digital radiography
EBIV	European Bureau for Inspection Validation
ECT	Eddy current testing
ENIQ	European Network for Inspection and Qualification
EPRI	The Electric Power Research Institute
FAC	flow accelerated corrosion
FFS	fitness for service
FMC	full matrix capture
FMEA	failure mode and effect analysis
FSAR	final safety analysis report
FTKA	Regulation for Pressure Vessel Safety in NPP and Facilities for Spent Fuel
GCR	gas cooled reactor
GL	generic letter
IAEA	International Atomic Energy Agency
IGSCC	intergranular stress corrosion cracking
Ii	inspection interval
IPEEE	individual plant examination of external events
ISI	in-service inspection
ISO	International Organization for Standardization
JSME	Japan Society of Mechanical Engineers
KEPIC	Korean Electric Power Industry Code
KTA	Kerntechnischer Ausschuss
LERF	large early release frequency
LMCR	liquid metal cooled reactor
LOCA	loss of coolant accident
LPHSW	last pass heat sink weld
LTO	long term operation

LWR	light–water reactor
MUREC	Mutual Recognition
NASA	National Aeronautics and Space Administration
NDE	non–destructive examination
NDT	non–destructive testing
NPP	nuclear power plant
NRC	Nuclear Regulatory Commission
NUGENIA	Nuclear Generation II & III Association
NURBIM	nuclear risk based inspection methodology for passive components
PA	phased array
PAUT	phased array UT
PCA	probability of correct acceptance
PCR	probability of correct rejection
PDA	performance demonstration administrator
PDI	performance demonstrations initiative
PDQS	performance demonstration qualification summary
PIPD	periodic ISI document
PISC	Programme for Inspection of Steel Components
PNAE	Rules and Standards in Atomic Energy Industry of Russian Federation
PND	probability of non-detection
POD	probability of detection
POF	probability of failure
PSA	probabilistic safety analysis
PT	liquid penetrant examination
PWR	pressurized water reactor
PWROG	Pressurized Water Reactor Owner Group
PWSCC	primary water stress corrosion cracking
QA	quality assurance
QB	qualification body
RBMK	Reactor Bolschoi Moschtschnosti Kanalny
RI-ISI	Risk–Informed In–Service Inspection
RMS	root mean square
RP	recommended practice
RPV	reactor pressure vessel
RT	volumetric examination
SAFT	synthetic aperture focusing technique
SAQ	sub–area on qualification
SCC	stress corrosion cracking
SG	steam generator
SKI	Swedish Nuclear Power Inspectorate
SKIFS	Swedish Nuclear Power Inspectorate Regulations
SRM	structural reliability model
SSC	systems, structures and components
TECDOC	technical document
TOFD	time-of-flight diffraction
TSO	technical support organization

TVO	Teollisuuden Voima Oyj
USA	United States of America
USNRC	US Nuclear regulatory commission
UT	ultrasonic testing
VERLIFE	Unified Procedure for Lifetime Assessment of Components and Piping in WWER NPPs
WWER	water cooled, water moderated power reactor

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