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HANDBOOK ON AGEING MANAGEMENT FOR NUCLEAR POWER PLANTS

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The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world".

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FOREWORD

One of the IAEA's statutory objectives is to "seek to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world." One way this objective is achieved is through the publication of a range of technical series. Two of these are the IAEA Nuclear Energy Series and the IAEA Safety Standards Series.

According to Article III.A.6 of the IAEA Statute, the safety standards establish "standards of safety for protection of health and minimization of danger to life and property". The safety standards include the Safety Fundamentals, Safety Requirements and Safety Guides. These standards are written primarily in a regulatory style, and are binding on the IAEA for its own programmes. The principal users are the regulatory bodies in Member States and other national authorities.

The IAEA Nuclear Energy Series comprises reports designed to encourage and assist R&D on, and application of, nuclear energy for peaceful uses. This includes practical examples to be used by owners and operators of utilities in Member States, implementing organizations, academia, and government officials, among others. This information is presented in guides, reports on technology status and advances, and best practices for peaceful uses of nuclear energy based on inputs from international experts. The IAEA Nuclear Energy Series complements the IAEA Safety Standards Series.

IAEA Nuclear Energy Series publications contain guidance on a variety of topics, including plant ageing management, degradation mechanisms, failure prevention, materials science and mitigation techniques as applied to ageing structures, systems and components in nuclear power plants. The IAEA publication Plant Life Management Models for Long Term Operation of Nuclear Power Plants, NP-T-3.18, issued in 2015, considers ageing management in the context of the entire nuclear power plant life cycle, even beyond the originally assumed operating period, taking into consideration the plant owner's objectives, economic aspects and the market in which the plant operates. It also describes how plant life management techniques, such as monitoring of equipment and structures, in-service inspections, material surveillance and maintenance, allow for the optimization of all plant programmes. Plant life management relies on advanced deterministic and probabilistic techniques and the use of an ageing knowledge database integrated with a technology watch function, R&D and the international operation feedback system.

In addition to developing publications on plant ageing, the IAEA also organizes international conferences as well as working groups, coordinated research projects, and technical and consultancy meetings on plant life management and structure, system and component ageing in nuclear power plants. The present publication is intended as an easily accessible handbook on the ageing management knowledge base, including insights from plant life management programmes as they are being implemented by an increasing number of Member States.

The work contributed by the subject matter experts involved in the drafting and review of this publication is greatly appreciated and is acknowledged at the end of this report.

The IAEA officer responsible for this publication was K.S. Kang of the Division of Nuclear Power.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

This ageing management handbook has been developed in compliance with the requirements of the IAEA safety standards [1] and the safety guide Proactive Management of Ageing for Nuclear Power Plants [2]. The primary safety goal of an ageing management programme (AMP) is to ensure the availability of all required safety functions and the performance of the safety systems throughout the service life of a nuclear power plant, taking into account the changes that occur with time and use. Consequently, the basic conceptual design and safety goals, as stated in IAEA Safety Standards Series Nos SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design [3], and SSR-2/2 (Rev. 1), Safety of Nuclear Power Plants: Commissioning and Operation [4], should be upheld at all times. Other important goals of ageing management are to ensure the continuity of commercial operation, the target capacity and availability factors, and the readiness of the plant to engage in long term operation (LTO) if conditions warrant it.

The IAEA has produced a comprehensive series of informative publications capturing accumulated knowledge on the diverse aspects of plant ageing and life management, including R&D, engineering support and operating experience. This publication is an update on the management of ageing that affects structures, systems and components (SSCs) as collected by the IAEA over the years. This handbook presents both materials science and practical knowledge in the field of ageing management of nuclear power plants in a concise and accessible manner.

In order for this publication to be used as a handbook, the ageing behaviour of materials is presented using flow charts, with a reference to the underlying safety and performance principles. In addition, this publication includes the latest R&D contributions and knowledge on plant life management from participating Member States, in close collaboration with the International Forum on Reactor Ageing Management (IFRAM).

Following the accident at the Fukushima Daiichi nuclear power plant in Japan, the IAEA has extensively re-examined the knowledge base on ageing management to determine if any ageing process may have contributed, in any manner, to the accident [5]. The extensively re-examined results showed that it is highly improbable that ageing processes contributed to the occurrence and consequences of the Fukushima Daiichi accident, during the period between the time of the first occurrence of the accident and the time when the accident caused the plant to experience conditions beyond its design basis.

1.2. TERMINOLOGY AND DEFINITIONS

The term ‘ageing’, as used in this publication, refers to the continuous time dependent degradation of SSC materials during normal service, extending to standard power production and transient conditions. Postulated accident and post-accident conditions are excluded. The concept of ageing is also extended to the spheres of component obsolescence, to the state of update of the nuclear power plant documentation and to staff training.

A glossary and list of abbreviations are included at the end of this publication. Defined below are the fundamental terms that are used in this guide.

— *Ageing management programme:* A set of policies, processes, procedures, arrangements and activities for managing the ageing of SSCs for a nuclear power plant.
— *Beyond design basis accident (BDBA):* Accident conditions more severe than a design basis accident that may or may not involve core degradation. SSCs may not be fully designed to withstand BDBAs. However, BDBA conditions may be partially considered in the design process of particular SSCs. For example, an SSC may be designed to maintain its structural integrity and hold the fluid it contains, but not its original geometrical shape. This may allow, for example, large deformations in the plastic region. In general, BDBA sequences are analysed and taken into consideration in design to the extent needed to fully understand their consequences and the robustness of the SSCs in preventing their own collapse.
— *Condition assessment*: An ageing assessment methodology applied to SSCs, or groups thereof with similar characteristics (commodities).

— *Control rod control system (CRCS)*: A system capable of providing dynamic control of core power by actuating rod movements in and out of the core. The system is distributed across several core control groups.
— *Design basis accident:* Accident conditions against which a facility is designed to withstand, according to established design criteria, and for which the damage to the fuel and the release of radioactive material are kept within authorized limits.
— *Finite element analysis:* A computational method used in engineering for finding an approximate solution to a problem involving a complex geometrical figure, by using simplified models approximating the figure. The problem becomes solvable by a finite number of numerical operations called discretization, which is, in turn, characterized by a finite number of parameters, *N*, called degrees of freedom.
— *Integrated life cycle management (ILCM):* A project integration management methodology, namely a localization programme, utilized to enhance the viability and sustainability of an appropriate level of local technologies, in support of the life cycle of a nuclear power plant project.
— *Irradiation assisted stress corrosion cracking (IASCC)*: An ageing mechanism attacking materials such as austenitic steels in the core components of a nuclear reactor. Among the many factors characterizing susceptibility of these materials to IASCC is hardening, the dislocation of the material microstructure and changes in the grain boundary composition.
— *Large early release frequency (LERF)*: The frequency of those accidents leading to significant, unmitigated releases from containment with the potential for early health effects. Such accidents are generally the consequence of containment failure, containment bypass events and loss of containment isolation.
— *Life assessment:* An ageing assessment methodology applied to critical and/or complex components and structures that fulfil mainly passive functions and are typically not economically replaceable.
— *LTO:* Operation beyond an established time frame set forth by, for example, design assumptions, applied standards, licence requirements, licence term, regulations. LTO needs to be justified by safety assessments with consideration given to life limiting processes and SSC conditions.
— *Maintenance, surveillance and in-service inspection*: A programme in nuclear power plants designed to ensure that the levels of reliability and availability of all plant SSCs continue to meet the assumptions and the intent of the design, through the programme duration and the plant life.
— *Non-destructive examination (NDE):* Using non-destructive testing (NDT) methods to inspect and characterize materials and structures that are not destroyed by the testing medium or the form of energy used.
— *On-line monitoring:* The application of instrumentation and observation techniques to evaluate equipment performance by assessing its consistency with other plant or baseline indications. Industry experience at several plants, and R&D, has shown this approach to be very effective in identifying equipment that exhibits degrading or inconsistent performance characteristics.
— *Periodic safety review (PSR):* A comprehensive safety review of all important aspects of safety, carried out at regular intervals, typically every 10 years. A PSR may be used in support of the decision making process for licence renewal or LTO, or for the restart of a nuclear power plant, following a prolonged shutdown or lay-up period.
— *Plant life management (PLiM)*: The integration of ageing and economic planning to: maintain a high level of safety and an acceptable performance level; optimize operation, maintenance and service life of SSCs; maximize the return on investment over the service life of the nuclear power plant; and provide nuclear power plant utilities/owners with the optimum preconditions for LTO.
— *Proactive ageing management:* The management of SSC ageing implemented with foresight and anticipation throughout the lifetime using R&D results, feedback from operating experience, advanced monitoring and analysis tools, targeted in-service inspection (ISI) and walk downs, technology watch programmes, PLiM tools, prognostic techniques, among others.
— *Safety aspects of long term operation (SALTO):* An IAEA peer review service designed to assist operating organizations in adopting a proper approach to LTO including implementing appropriate interventions to ensure that plant safety and reliability will be maintained during the LTO period.

1.3. OBJECTIVE

This publication is intended as a general reference report on nuclear power plant ageing, covering topics such as material degradation mechanisms and ageing management techniques. It also serves as a common knowledge base to facilitate information exchange and to create a broader awareness of ageing management issues.

The AMPs for SSCs in nuclear power plants are aimed at preserving the current licensing basis envelope by adequately managing safety margins, by controlling performance degradation and by improving capacity factors and system availability. The publication intends to help Member States to start planning for LTO as early as possible in the life cycle management of their nuclear power plants.

Guidance provided here, describing good practices, represents expert opinion but does not constitute recommendations made on the basis of a consensus of Member States.

1.4. SCOPE

This publication covers ageing management processes and techniques that can be used to ensure the integrity and reliability of SSCs in compliance with a set of safety requirements described by a regulatory framework. It incorporates the knowledge base acquired by the Member States on component specific ageing issues and shows how to integrate this knowledge with PLiM techniques and insights for optimization of AMPs.

In the general economy of plant ageing management, the large non-replaceable and generally passive SSCs require the most attention and investment in managing their material degradation. Particular emphasis has been put on the advanced principles of proactive ageing management. The scope extends also to active components and subcomponents, as well as to classes of smaller devices and groups of similar constituents.

Obsolescence of SSCs that are important to safety should be managed proactively with foresight and anticipation throughout their service life. Obsolescence and upgrade management programmes are excluded from the scope of this publication, because these are not directly related to material degradation.

The important role that feedback from operating experience plays in proactive ageing management is emphasized throughout this publication, covering both successful practices and failures in the ageing management of SSCs in light water reactors (LWRs) and heavy water reactors (HWRs).

This publication is likely to be of particular interest to operators of Member States that are in the process of introducing nuclear power in newcomer countries to help them set up their AMPs. It can also be used in the training and support of nuclear power plant staff of both newcomer and established countries and as a reference tool for maintenance managers, vendors, research organizations and regulators, to assist them in their work on ageing management in operating nuclear power plants.

1.5. STRUCTURE

Section 1 is an introduction to the contents of this publication. Section 2 provides an overview of the basic concepts of ageing management and an analysis of the impact of ageing on plant safety and performance. It highlights common weaknesses of traditional AMPs and shows how operation feedback, R&D and other activities are used in ageing management. Section 2 also provides information on the general features of AMPs and on the proactive approach to ageing management.

Section 3 describes proactive ageing management in operating nuclear power plants. This section outlines ageing considerations during design, operation and decommissioning of nuclear power plants, and discusses effective AMPs.

Section 4 deals with the ageing mechanisms related to critical components, such as the reactor pressure vessel (RPV) and its parts, namely the vessel head, the vessel walls and the RPV internals in pressurized water reactors (PWRs) and boiling water reactors (BWRs). This section includes descriptions of the underlying science, the knowledge base, R&D and state of the art discoveries, particularly in the areas of radiation embrittlement, fatigue, stress corrosion cracking (SCC), general corrosion and erosion corrosion (also called flow accelerated corrosion (FAC)).

Section 5 describes the accumulated experience in material degradation and mitigation options, and shows how prioritization of all activities can be established through risk analysis and the definition of degrees of criticality.

Section 6 deals with inspection and evaluation tools, processes and regulations developed to measure the effectiveness of AMPs with regard to safety, safety margins, and reliability and performance management.

Section 7 deals with the regulatory framework concerning ageing management and describes the most commonly used approaches for ageing management. This section also shows how legal criteria and procedures for safety assurance, particularly in LTO, are to be dealt with and why, depending on local circumstances; different countries may have different approaches and priorities in achieving the same safety goals.

Section 8 discusses innovative techniques in ageing management, and shows how international cooperation on nuclear power plant ageing management is ongoing under various R&D programmes.

2. OVERVIEW OF AGEING MANAGEMENT

2.1. BASIC CONCEPTS

The primary objective of an AMP is to maintain safety and reliability and production functions throughout the service life of a nuclear power plant. This entails preserving SSC performance, and plant performance targets as a whole, and requires that the AMP addresses:

— Physical ageing: The ageing of SSCs due to physical, chemical and/or biological processes (degradation mechanisms);
— Non-physical ageing: The fact that some SSCs may become out of date (i.e. obsolete) owing to the evolution of knowledge and technology and/or changes in requirements, codes and standards.

'Degradation' and 'ageing' are terms used to describe SSC deteriorations under operating conditions. However, these terms are not interchangeable:

— Degradation is a gradual deterioration of one or more characteristics of an SSC that could, at one point, impair its ability to perform the required functions within acceptance limits;
— Ageing is a general deterioration process in which the characteristics of an SSC gradually change with time and use.

Maintenance activities are usually separated from ageing control activities, although the programmes may be integrated. Maintenance is applied mainly to active and replaceable SSCs. It also includes run to failure components, bulk materials and part inventories, which all need to be kept in good operable condition.

Ageing management should be considered at each stage of the life cycle of a nuclear power plant, beginning with the first conceptual design, on through detail design, construction and as-built changes, commissioning, commercial operation, outages, design changes, upgrades, refurbishments, shutdowns, lay-ups, restarts and LTO.

An efficient AMP requires active involvement in the monitoring, inspection and surveillance of critical SSCs to allow charting of their actual degradation and comparison to the predictions. When ageing processes are known, they can be monitored and mitigated through an appropriate AMP, which normally contains a set of policies, procedures and instructions, and a work scope with detailed planned activities, designed to successfully manage SSC ageing.

Ageing management is a key element in the safe and reliable operation of nuclear power plants, and is therefore the object of particular attention of regulators and industry auditors. The cumulative effects of ageing and obsolescence on the safety of nuclear power plants are re-evaluated periodically by licence holders and reviewed by regulators using PSR or equivalent processes for plant safety evaluations.

The PSR method is typically used in Member States with unlimited or continuing nuclear power plant licences. Most European States apply the PSR process periodically (e.g. every 10 years) as their main safety assessment method of the effects of ageing and obsolescence. Most regulators requesting the application of the PSR method call for updates to AMPs and LAs of the critical components as prerequisites to continuing the validation of the operating licences of nuclear power plants.

A second approach in evaluating the effects of nuclear power plant ageing on safety is the licence renewal process. It is applied in States where limited term licences are issued. At the end of the licence term, a licence renewal application (LRA) is submitted to the regulator, if longer operation of the plant is justified. In the United States of America, the LRA process is used as a safety assessment method against the effects of ageing and obsolescence. This does not mean that safety checks are ignored during the 40 year duration of the first licence. The regulator updates, whenever necessary, the regulation of the current licensing basis, through the promulgation of updated licensing requirements, as deemed appropriate, coupled with the mandatory implementation of explicit regulatory action items. Every implementation of a new requirement implies current licensing basis compliance checks, a thorough review of maintenance programmes, monitoring of safety performance parameters, and updates to the final safety analysis report (FSAR) and supporting analysis.

A third approach is a combination of the above two, such as that used in the Republic of Korea, where PSR is used every 10 years as the basic safety assessment method, complemented with updates to safety requirements when they become relevant and the regulator judges them appropriate for local plants.

Finally, it is to be noted that operators should apply a systematic SSC ageing review using screening techniques to categorize and prioritize SSCs. Operators can use advanced tools such as integrated R&D knowledge databases, operation feedback databases, elements of probabilistic, reliability and economic analysis, and other advanced engineering and financial tools to help manage plant life cycle in the most efficient manner.

2.2. GENERAL FEATURES OF AN AGEING MANAGEMENT PROGRAMME

Regulatory requirements for ageing management at the national level need to be developed before an operating licence can be released. A national regulatory authority may decide to adopt the ageing regulations of the technology of the originating State. However, even in this case, foreign regulations should never be adopted without a thorough applicability review. The national regulator should always update and adapt to local conditions regarding any foreign regulations on ageing management, including requirement and guidance documents. In addition, national regulators should require that all ageing issues be identified and documented in the safety analysis report (SAR) and periodically updated throughout the plant lifetime.

A systematic approach to ageing management is discussed in Section 3.4, where proactive ageing management principles are described in detail. This approach is an adaptation of the ‘plan–do–check–act’ (PDCA) cycle to the ageing management of an SSC.

2.2.1. Data collection and record keeping

Without a good data collection and record keeping system, the ageing management of nuclear power plants can only be reactive, which will eventually lead to decreasing plant performance, including safety performance. In order to prevent gaps in maintenance programmes and allow meaningful audits, regulators usually require three types of data: baseline data, operating history data and maintenance history data.

Baseline data refer to the initial, undegraded conditions, and should include:

- The component data sheet with descriptive and functional information including interfacing systems and components, boundary conditions and the functional relationship to the system and to the unit in which it operates;
- The operating and performance requirements including the design service conditions and any other operational limits;
- The initial, undegraded material conditions (e.g. the wall thickness profile of critical locations, especially critical pressure boundary piping and components subject to wall thinning such as multistage orifices and control valves, among others);
- Installation records, welding specifications used, manufacturing defects, repair records, accepted exceptions to non-conformances with sufficient description and justification of any deviations from the original design requirements, the component orientation, the component supports, and any particular as-built condition,

including specific modifications due to construction deviations, variances or field changes, their justifications as well as any design reconciliation statements approved by the local regulatory authority, if applicable;
— Functional capabilities as tested during commissioning;
— Component calibration data, if applicable.

The operating history data should include the actual service conditions experienced, inclusive of any evolution profile of relevant process parameters such as:

— Water chemistry;
— Environmental conditions inside and outside the SSC;
— The actual number and profile of the transients endured (e.g. pressure–temperature transients for pressure retaining components);
— Data on the component availability and performance;
— The results of tests conducted (if any), their reasons if unplanned or their context if planned.

The maintenance history data should include:

— Condition monitoring of critical components such as data collection, testing, inspection and surveillance activities aimed at understanding the degree, speed and modality of material and functional degradation;
— Repair or modification history (history dockets);
— Overall maintenance cost history (including any indirect costs, such as forced changes to the critical path and system downtime).

2.2.2. Continuing structures, systems and components reliability improvement programmes

A continuing SSC reliability improvement programme is of key importance in achieving high plant capacity factors and LTO. The main elements of such a programme are the implementation of a continuing operation feedback assessment programme, a corrective action optimization process and a technology watch programme to study and implement new technologies and new diagnostic tools, when warranted. Reliability improvement heavily depends on failure prevention, which, in turn, relies on the capability to:

— Measure the symptoms of incipient failures;
— Analyse operations feedback and, specifically, any relevant events and failures at other plants;
— Identify premonitory signs of operating anomalies.

In order to economically use these practices, operators should acquire the capability to map the locations that have higher failure probability. This is possible if they know the vulnerability in the degradation mechanisms of all their SSCs and if they effectively use inspection, surveillance, testing and monitoring equipment such as:

— Installed instrumentation capable of multiple use, for example, the measurement of process parameters that can also be indicators of degradation, as in the case of process temperature indicators coupled with recorders and analysers;
— Advanced monitoring systems, such as loose part, fatigue, vibration and acoustic monitoring systems;
— Oil sampling;
— Thermography;
— Electromagnetic signature monitors for motorized pumps and valves.

Figure 1 shows a process used to screen SSCs and apply monitoring where it can be effective in revealing degradation and ageing. The process begins with a review of the ageing monitoring techniques that are applied and an assessment of the limitations, compatibility and accuracy of these techniques when compared with the ageing observed in the field and recorded in the AMP. If gaps are found, the AMP needs to be upgraded, or, if necessary, a new AMP should be created. Any unexpected degradation observed in the field needs to be separately monitored, and a method to measure its growth and plan remediation needs to be implemented. In addition, the AMP for this

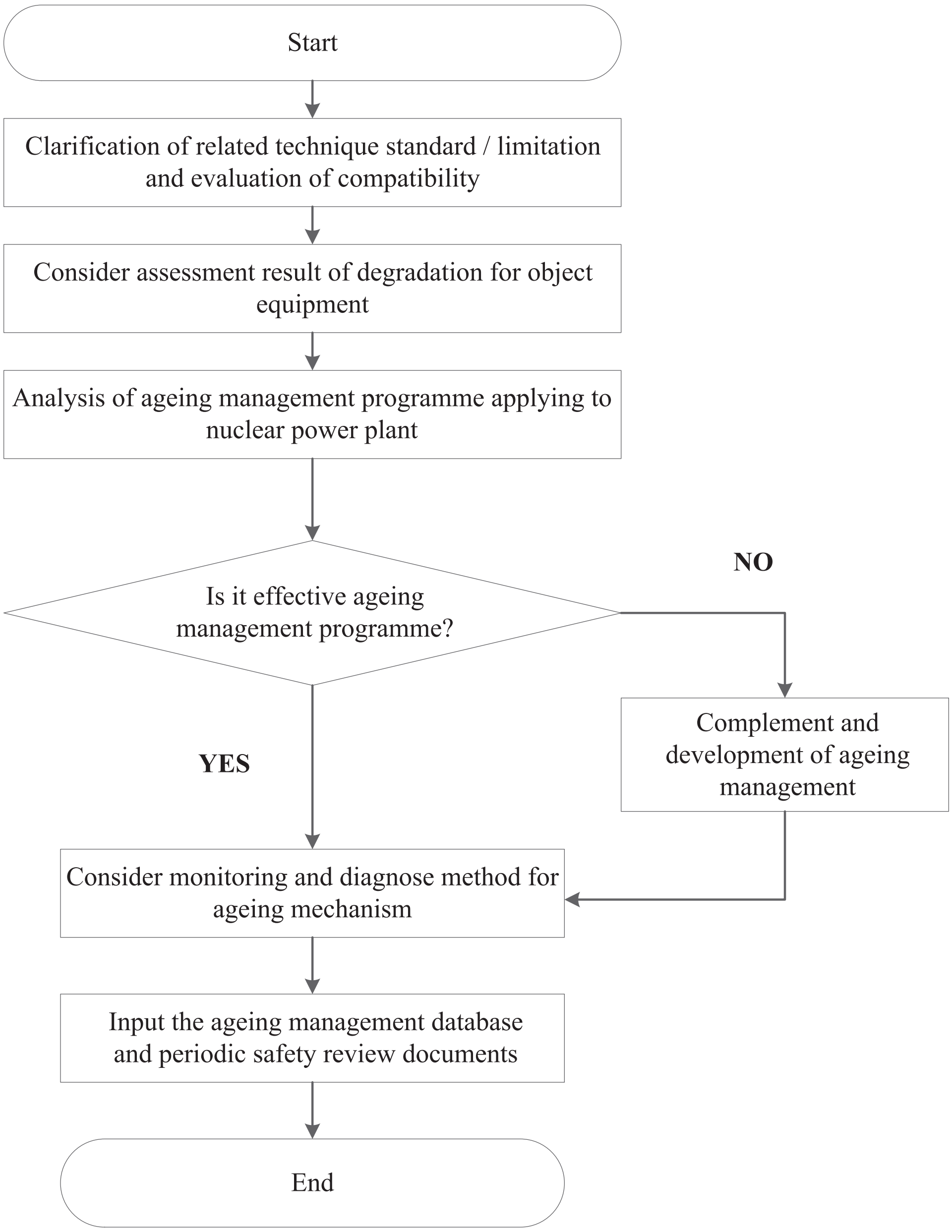

FIG. 1. Programme monitoring and managing the degradation and ageing processes [6].

specific mechanism needs to be taken into account in the design basis monitoring programme and recorded in the PSR documentation [6].

2.2.3. Relationships between ageing management and maintenance programmes of safety critical components

If a PLiM programme is not well established and systematically applied, it typically displays the following issues:

- Duplication of effort between ageing management and maintenance activities;
- Conflicts in planning activities;
- Avoidable rework;
- Deficiencies in the capacity to diagnose incipient failures and recognize age related degradation;
- Inadequacies in the knowledge of degradation mechanisms;

— Inadequacies in the administrative processes and procedures to monitor system and component performance, such as gaps in periodic walk downs, improper visual observations, missing detection of signs of degradation and non-conformances;
— Lack of effective inspection procedures during maintenance (e.g. inadequate instructions at the craft level to report ‘found’ and ‘repaired’ non-conformances in a component being maintained);
— Lack of a user friendly maintenance feedback system allowing, for example, a condition grading code system to efficiently characterize the equipment condition, or the lack of an SSC condition evaluation instruction to guide the diagnosis and the selection of adequate failure prevention barriers and ageing mitigation provisions, or the lack of an adequate approval process through an engineering function such as a system engineer or an area authority;
— Lack of an adequate equipment history and corrective action database that contains performance limits, which allow for the systematic prediction of failure trends of similar components across the plant and in similar plants, if it is a multiunit operator;
— Lack of a long term maintenance strategy to regulate, for example, how to proactively maintain components for as long as economically feasible;
— Lack of a systematic condition assessment programme for safety critical components, which makes use of diagnostic techniques to prevent failures;
— Lack of a systematic life assessment programme for critical irreplaceable and sometimes inaccessible SSCs.

2.2.4. Ageing management assessment and screening of structures, systems and components

In the past, active and less critical, and more easily replaceable components (e.g. rotating machinery, valves, etc.) have received the most attention, and more work was conducted with regard to diagnostic and prognostic techniques. Consequently, research on active components is more advanced and their ageing behaviour is better known than that of large and long life passive components such as pressure vessels, concrete structures, cables and piping. However, passive SSCs are the most critical in terms of safety and performance.

Even if operators have not implemented PLiM programmes and adopted probabilistic tools in support of their ageing management strategy, the first step they still need to take remains the screening of their SSCs to determine those that are going to be enrolled in their AMPs. By screening SSCs, operators can reduce their ageing management analysis efforts to a manageable scope. It is neither practical nor necessary to evaluate and quantify material degradation for all SSCs. The screening step allows operators to focus their resources primarily on those SSCs that may have a negative impact on the safe operation and economic viability of the plant. The selection should also include SSCs that may not have direct safety functions, but whose failure may prevent safety related and performance related SSCs from carrying out their intended safety functions. As a minimum, a simplified approach to screening is needed as a prerequisite to any (even the simplest) AMP. Figure 2 is a sample flow chart of a simplified SSC screening process in support of a nuclear power plant’s AMP.

The first step of an SSC screening exercise is to determine whether the ageing mechanisms and their effects on the safety, reliability and performance of SSCs are well known and understood. In this regard, post-service examination and testing of SSCs (including destructive testing) may be a good place to start to substantially improve the understanding of ageing mechanisms. An assessment of the component’s post-service examination and of the test results should be undertaken to characterize the environment and the stressors on the materials. The assessment should include a description of the ageing mechanisms, the type of degradation, the mapping of the degradation sites and a theoretical or empirical model, guided by test results. The model can be quantified and correlated to the knowledge base to predict future development of the degradation.

The identification of SSCs important to safety is usually conducted beginning at the system level by recognizing the system safety classification and by grouping all components belonging to it. Then, large components are analysed individually. Their internals are treated as subcomponents, each contributing to the safety functionality of the whole. Depending on the criticality of each subcomponent, they are categorized in the context of the component AMP or of the pragmatic AMP. The categorization of each subcomponent depends on whether or not its failure could lead to the loss or impairment of the component and of the system safety function. All subcomponents are analysed by maintenance type and entered in the AMP. As an example, a typical SSC and subcomponent screening process, as conducted in the Republic of Korea’s nuclear power plants, is illustrated in Fig. 3.

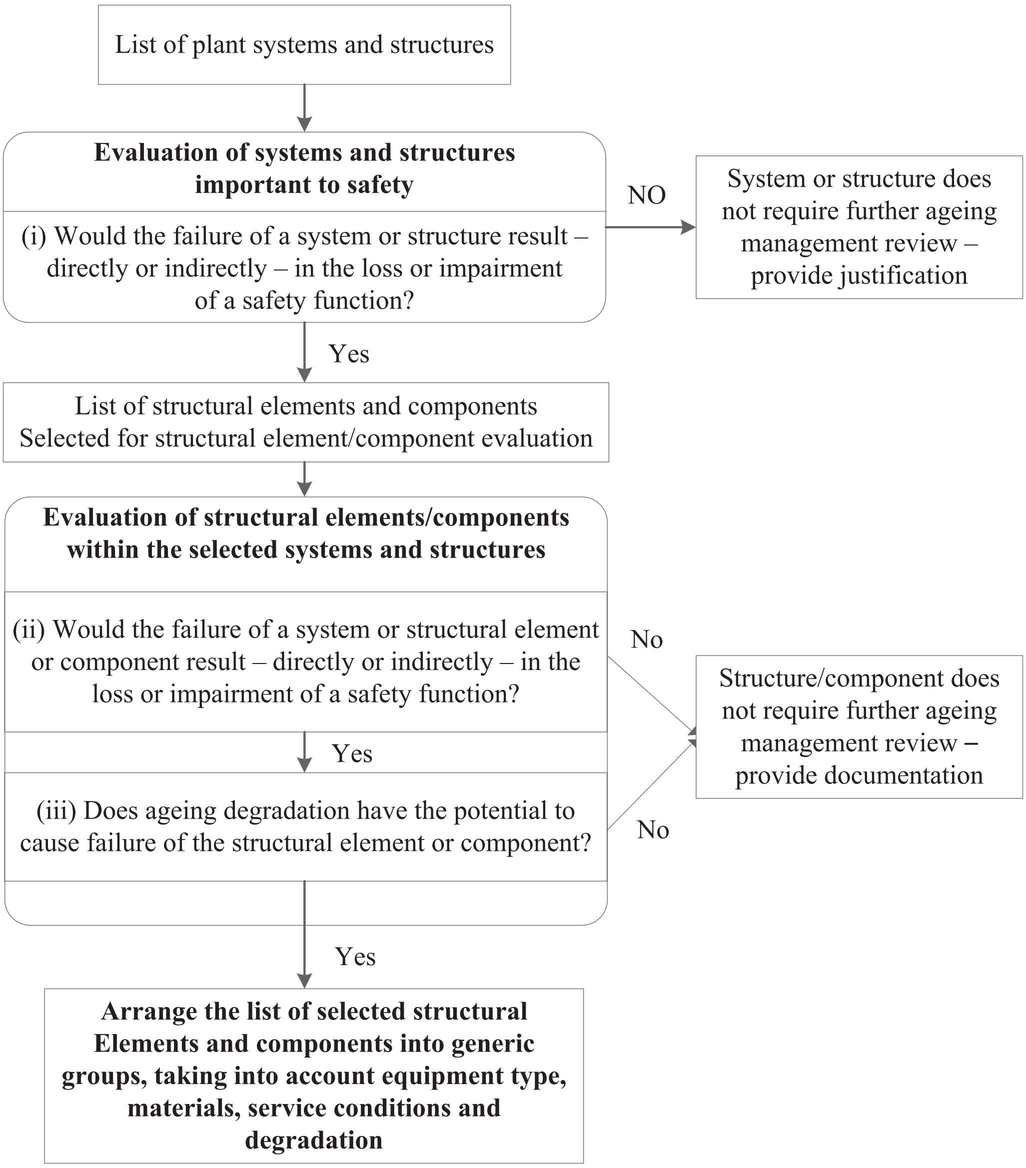

FIG. 2. Flow chart of structure, system and component screening for a nuclear power plant's AMP [6].

In a PLiM assessment, SSC screening is always conducted from the top down in a pyramidal model, beginning with the plant life limiting components, which require an extensive life assessment and a customized AMP, including R&D, if necessary. These components are usually mainly passive or structural in nature. The screening can then continue at the system level, where active, mostly safety critical and economically important components are identified. Two parallel phases of SSC prioritization are normally undertaken:

— A first phase identifying components that are critical to safety, where regulatory requirements regarding ageing management need to be considered;
— A second phase identifying SSCs with a failure rate threshold linked to economic requirements.

Exhaustive details on how to scope SSC screening for an ageing management assessment can be found in the Nuclear Energy Institute (NEI) guideline 95-10 [7]. The methodology recommended follows the requirements of the US Nuclear Regulatory Commission (NRC) Code of Federal Regulations (CFR) 10 CFR 54.5 [8]. A sample listing of potential information sources for the screening and a table listing typical structure, component and commodity groupings are provided as guidance. If probabilistic tools are available, a combination of probabilistic

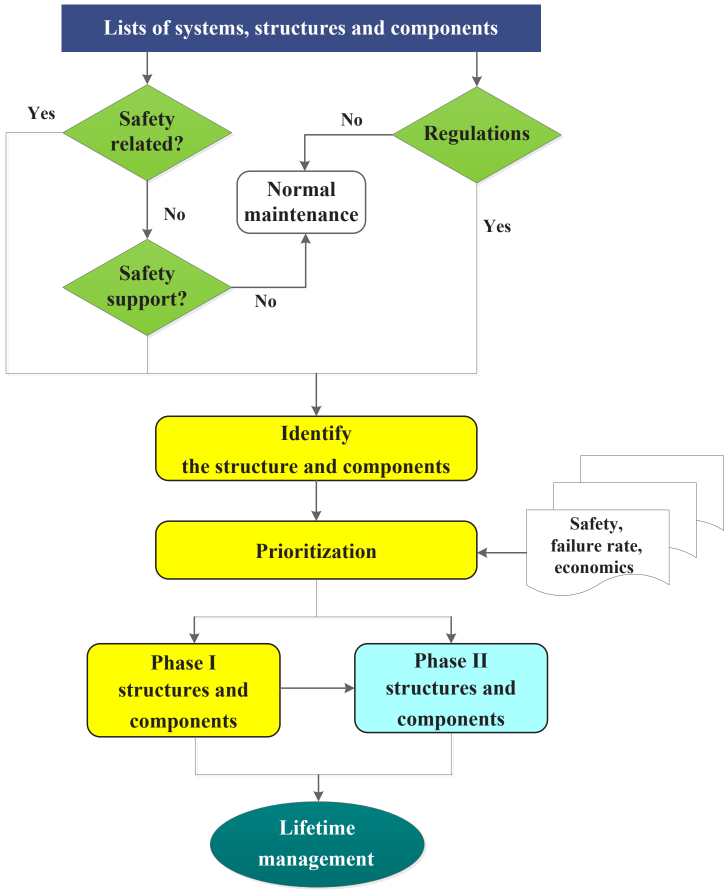

FIG. 3. Structure, system and component screening approach.

safety insights and deterministic methods is recommended. This approach is called 'risk informed' screening. It allows a prioritization of the ageing management activities on all SSCs on the basis of their safety significance.

Once the SSCs have been selected for ageing management, their programmes should be prioritized for the optimization of plant resources. The prioritization exercise consists of evaluating the importance of each SSC based on safety significance, plant life limiting significance, economics, management strategies, design criteria, plant performance, field data, failure rate, etc.

In order to carry out the SSC AMP optimization, a design and operation data collection activity is conducted for all screened components. Depending on the grade and criticality of the component or group of components, ageing studies may have to be conducted to help develop the most appropriate and cost effective AMP. In general, the following three degrees of SSC assessments can be used based on the complexity of the individual SSC or group of SSCs:

— Life assessment: This technique is the most complex and comprehensive of the three. It is typically applied to plant life limiting structures and components that are generally passive in nature and designed not to be replaced or whose replacement would render the plant uneconomical. The RPV, other large pressure vessels and certain inaccessible reactor structures, among others, belong to this category. Predicting the remaining life of these critical SSCs is essential and requires advanced prognostic tools and extensive studies. Life assessment begins with a rigorous assessment of the operating environment and of all plausible ageing related degradation mechanisms. The methodology entails a detailed review of the SSC commissioning and operating data in order to establish the component's current condition and ageing trends and to precisely evaluate all impact areas, even at a subcomponent level. Managing the ageing of passive and irreplaceable safety related SSCs is key to the planning of a successful PLiM programme for both the originally assumed design life, and for possible future longer term operation.

— Systematic assessment of maintenance: While life assessment activities are applied to large passive systems and structures, active systems and components (e.g. pumps and large valves) are generally replaceable, when necessary, to preserve system functions. The active systems and components need to be routinely monitored, analysed and restored with less onerous programmes. The analyst can make profitable use of systematic analysis methods, such as failure mode and effect analysis (FMEA). Input information for the FMEA is obtained from internal and external feedback and R&D findings. Results are processed utilizing streamlined reliability centred maintenance (RCM) techniques, as adapted to nuclear power plant applications.
— Condition assessment: This is a technical procedure, typically performed for smaller categories of components. The condition assessment analyst assigns components to commodity groups. Such components are usually instruments, fittings, loops, small valve types, small machinery, piping, tubing spools, etc. They are assigned to categories and evaluated in groups. The methodology entails a general statistical review of plant data in order to establish the current condition and to evaluate the material degradation of each category. The evaluation in a condition assessment normally includes the technical basis for the ageing management of the component category or the commodity, the periodicity and the type of maintenance activities envisaged. In particular cases, it may be necessary to identify the need for further or more rigorous assessments.

When dealing with a complex component such as a steam generator, its substructures and subcomponents, such as nozzles, heads, walls, tube bundles, tube supports, tube sheets, divider plates and other internal structures, are systematically examined and evaluated. The ageing related degradation mechanisms of all such subcomponents are identified.

The factors that impact the growth of a crack are, today, reasonably well understood, but the dynamics of incipient crack growth are less well known, as are the impacts of stressors and the sensitivity of diagnostic tools. These uncertainties affect the capability of determining the correct damage phase or its level, even if crack growth rate measurements are taken. If the failure mechanisms are unknown, failure data are sent to the R&D department for interpretation and research. Early detection of incipient degradation relies on the knowledge of precursors. Precursor characterization is key to a successful prognosis. Hence, the availability of instruments sensitive to precursors and of analysis tools capable of interpreting the evolution cycle of SSC degradation, from the precursor state to crack initiation and all the way to component failure, is essential.

The development of advanced measurement techniques, such as phased array ultrasound, acoustic emission and guided waves, can be deployed in ISI and in permanent monitoring programmes. The optimal selection of the sensor locations and the parameters to be monitored should be decided on the basis of a risk informed SSC analysis. The integration of conventional AMPs and PLiM insights provides the capability to customize the monitoring functions and support the design and installation of automatic on-line monitoring instrumentation and surveillance devices. These advanced techniques have proven crucial in indicating the crossing of thresholds and allow for the optimization of preventive maintenance activities.

With the implementation of a PLiM programme, the safety aspects of ageing management are also better served, because integration provides a better awareness of safety margins and a better management of pressure boundary integrity throughout the plant lifetime. If a PLiM programme is not implemented, degradation of safety critical components is usually only detected at periodic inspection time (every 10 years), and potentially only after significant degradation has already occurred. This practice may have safety implications if a serious failure occurs between periodic safety inspections. In terms of economics, the possibility of a forced reactive break plan in the maintenance schedule or an unpredicted corrective activity may result in higher maintenance costs and even extend the PSR outage, if such activities happen to affect the outage critical path.

Equipment qualification programmes also need to be reviewed in terms of their effectiveness. The best methods and practices should be identified and proposed to preserve and upgrade equipment qualification, where necessary. Recommendations can then be formulated, including the most appropriate organizational model for implementation.

It is possible to envisage a future transition of most plants from current labour intensive ISI to on-line monitoring and PLiM prognostics. Plant health monitoring and threshold alarm systems will become more automated, with higher levels of remote on-line access to SSC health parameters.

2.2.5. Component life evaluation

The data that are collected are imported into a PLiM database for analysis and SSC prognostics. A successful outcome depends on the quality of the information, the observed degradation state of the SSC and the projected development of the stressors over time, supported by their past history, if at all known. A useful prognosis needs to refer to the reading of the markings, typical of the material type and of the degradation state, and contain the thresholds of incipient component degradation, to establish early warning signs and calibrate the instruments.

The probability of failure of an SSC normally follows the 'bathtub curve' pattern shown in Fig. 4. At the beginning of life, when the new SSC enters service, it experiences the so called 'run-in' or 'breaking-in' period. The likelihood of failure during that time is relatively high, owing to a combination of factors that may originate from design flaws, fabrication issues or from the inappropriate intervention of inexperienced staff. There may also be the unintended legacy of some pre-operational or commissioning tests.

As operation of the plant proceeds, the probability of failure curve decreases as the SSCs enter a stabilization period, with a lower and somewhat flat failure rate.

The SSC ageing data, such as those of stressors, degradation mechanisms, time in service, etc., are used to calculate the likelihood of failure. The bathtub curve of an SSC is, in reality, the envelope of the probability of failure curves of each of the stressors and of the other degradation factors, acting independently.

The relatively flat stabilization period depends on quality and timely implementation of the preventive maintenance plan and the extent and quality of the condition monitoring programme. As the SSC ages, the stressors inevitably become more aggressive and the likelihood of failure is statistically bound to increase in what is known as the 'wear-out' period in the bathtub curve. If the maintenance programme is optimized, the flat stabilization period is extended.

In well managed maintenance programmes, the flat portion of the bathtub curve is low, and may even show declining probabilities of failure. If the maintenance programme is less than optimized, the wear-out period begins

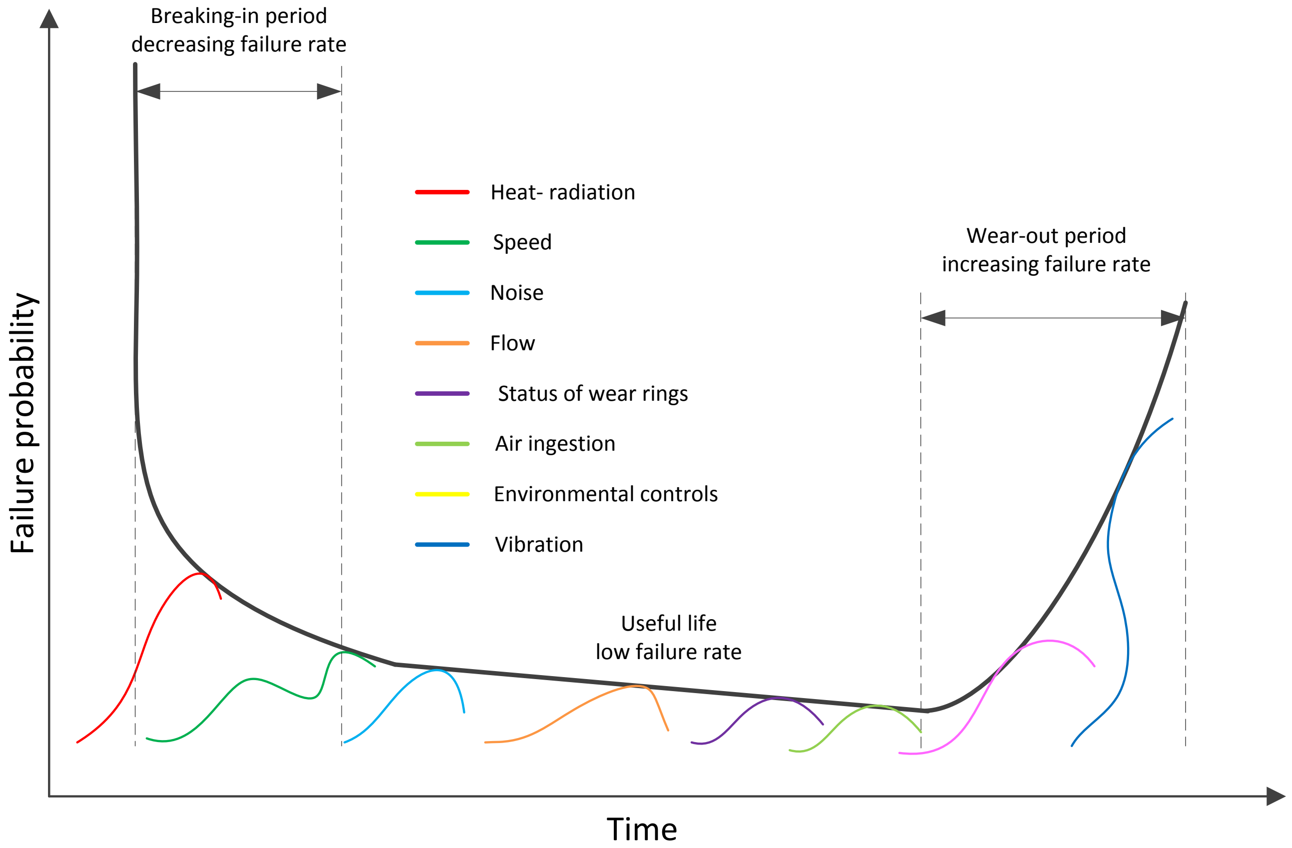

FIG. 4. Ageing bathtub curve (predicted failure rate versus time).

earlier and its slope is steeper. The starting point of the wear-out period and its rate of increase with time should be determined in order to plan corrective actions and allocate sufficient budget for its implementation.

As a long lived, irreplaceable component reaches its wear-out period, its probability of failure will increase, but if its life cycle is managed preventively and proactively, its failure rate can be controlled. Life cycle management techniques allow plant operators to correctly deal with this period. The aim is to prevent unexpected failures. Even if corrective actions and preventive maintenance interventions increase, the component's ageing remains manageable as long as it is economically viable to continue operation.

The next step toward life management is to undertake a critical component prognosis. This is best conducted by consulting PLiM databases containing the degradation mechanisms and related R&D data. In those plants where PLiM techniques have been adopted in support of the AMP, the PLiM analyst generates a credible prognosis for the critical SSCs, based on their bathtub curves, and recommends mitigating actions to extend their service lives.

Listed below are three of the most frequently used prognostic methods in PLiM analysis:

— Reliability data analyses, which use historical time to failure data to model time to failure predictions. These methods do not rely on actual plant operating data and may be less pertinent to the actual plant conditions.
— Stressor based methods, which take specific operating conditions into account. They allow trending of the stressor values and the development of a distribution curve: stressor values versus predicted failure times. These correlations are then used in life prognostics.
— Effects based methods, which compute a degradation or damage index and correlate this quantity to the probability of failure. In the prognosis produced by this method, the remaining useful life is typically estimated based on the time required by the damage index to exceed some predefined threshold.

The deployment of on-line condition based monitoring and the use of remote monitoring and maintenance planning and practices allow for a better prognosis capability and more reliable predictions of the remaining useful life of SSCs. Proactive approaches such as these are called either prognostic based health management or integrated health management. One such methodology developed by Korea Hydro & Nuclear Power Company for lifetime assessment is shown in Fig. 5, where a 'remaining useful life' or prognosis is conducted in six steps [6].

The life of a system is defined as the shortest life of a structure or component within the system boundary. In order to ensure the ageing mechanisms selected for the subcomponents are correct, the following information should be acquired:

— The ageing mechanisms that can occur in nuclear power plants;
— The susceptible operating environments that can cause material degradation;
— The functions, materials, design, fabrications and operating conditions of the subcomponents;
— The operating experience feedback and technical documents and reports issued on degradation experience worldwide.

In order to identify age related degradation mechanisms (ARDMs) and the ageing effects of the ARDMs, the analyst needs to evaluate the design and material data of structures and components and the operational and environmental conditions of the systems. Ageing mechanisms are a function of design characteristics and operating environments, and can be derived from the design documentation. Material characteristics and properties are normally taken from certified material test reports and from the FSAR. Operating environments can include working fluid, temperature, pressure, humidity, water chemistry and radioactive fluence, among others.

For LTO applications, quantitative life evaluations such as time limited ageing analysis (TLAA) are required, and are usually part of the regulatory requirements. In this case, SSCs are evaluated to determine whether they would maintain their intended functions and design integrity during the LTO period. TLAAs usually apply to relatively small components, but they are intense activities in terms of engineering effort and scientific investigation. In qualitative life evaluations, such as ageing management reviews, the plant data records since the first reactor criticality are assembled, and the SSC is systematically assessed for functionality and integrity. The guiding principle is that all intended functions should not be compromised by ageing effects during plant operation.

The generic ageing lessons learned (GALL) report in NUREG-1801 [9] was published in the USA to facilitate ageing assessments. Recommendations in the GALL report are a result of detailed life assessments and are applied to AMPs. These recommendations, such as component replacements, design modifications, the addition

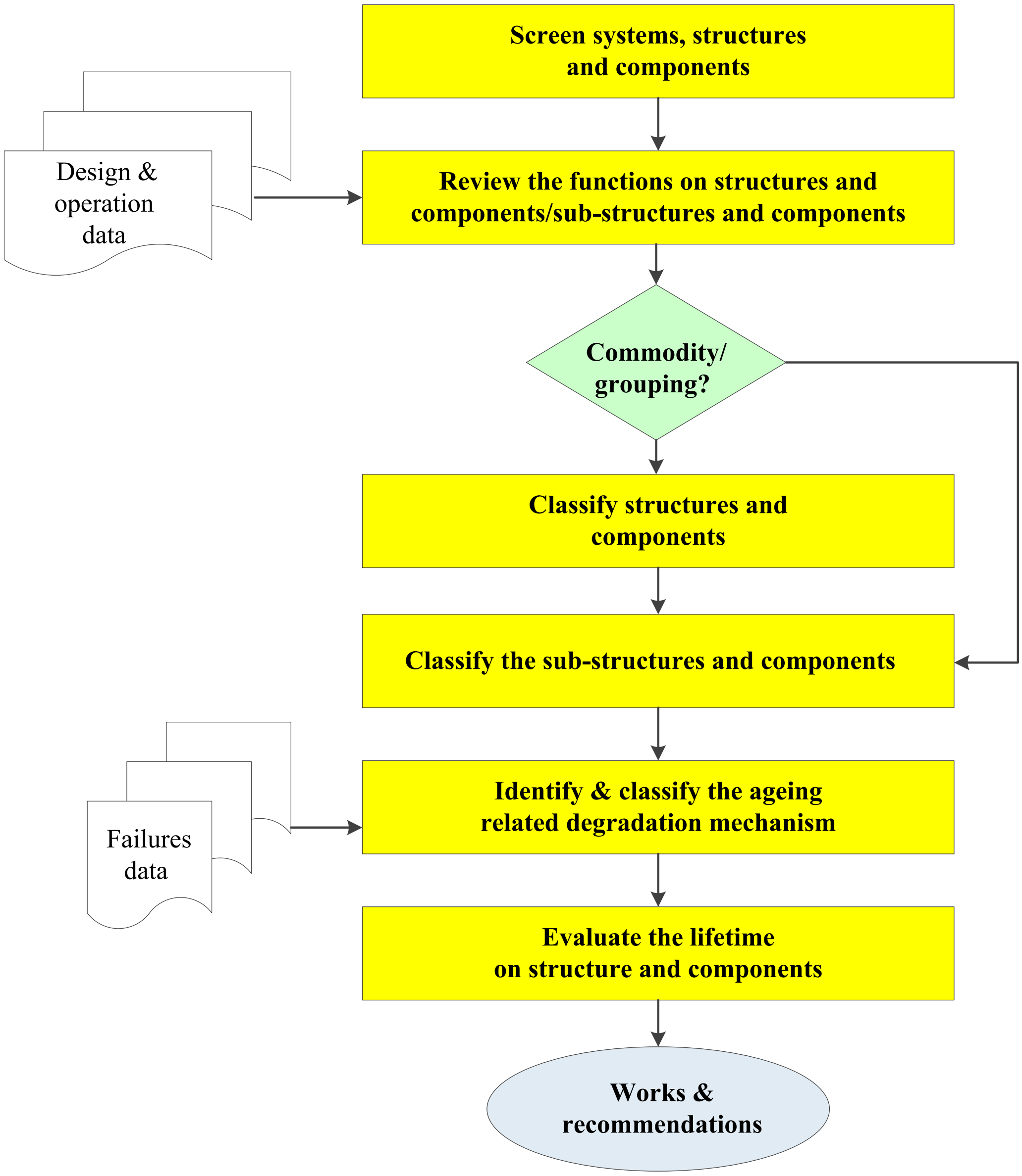

FIG. 5. Remaining useful life evaluation.

of performance monitoring functions and upgrading, among others, essentially aim at preserving the intended functions of the SSCs during the extended operation period. Before their implementation, all AMPs and derived maintenance activities should be compared with those of other similar national and offshore nuclear power plants, if available.

2.3. LESSONS LEARNED ON AGEING OF STRUCTURES, SYSTEMS AND COMPONENTS FROM THE FUKUSHIMA DAIICHI NUCLEAR POWER PLANT ACCIDENT

2.3.1. Background

Concern was expressed that material degradation in the Fukushima Daiichi nuclear power plant facilities might have played a role in the accident or even worsened its effects. In the report of the Japanese Government to the IAEA Ministerial Conference on Nuclear Safety [5], it is stated that the impact of ageing would be assessed, and the relationship between ageing and the causes of the accident would be examined in detail.

Utilizing the results of the technological assessment based on actual data collection of SSC material degradation in the Fukushima Daiichi nuclear power plant conducted in the past, prior to the accident, a theoretical assessment of safety critical equipment was conservatively conducted to understand, at least preliminarily, whether there could have been an impact of the ground motion on the equipment, as a result of the Fukushima earthquake. The equipment age was conservatively projected to a full 60 years of operation [10].

2.3.2. Results of the ageing assessment

2.3.2.1. Assessment of the upper grid plate and the reactor pressure vessel

Low cycle fatigue cracking and irradiation induced SCC were found, in the assessment of the upper grid plate damage, to exist before the accident, because they were the two ageing related phenomena that may have contributed to the damage observed after the Fukushima Daiichi accident. The assessment was carried out using the seismic ground motion of the earthquake and the results of the technological assessment of ageing conducted in the past. It was found that the impacts of the ageing phenomena were sufficiently small to preclude any contribution from them to the damage resulting from the accident. The impact on the margin to the acceptable limit was small, and the acceptable limit itself was originally selected with some additional margin to the expected damage limit. Therefore, the review showed that no appreciable contribution of the ageing phenomena was found on the observed upper grid plate damage following the accident.

Similarly, in the assessment of the damage to the RPV, conducted using the seismic ground motion of the earthquake and the results of the technological assessment of ageing resulting from neutron irradiation and from actual inspection data records, the impact of the irradiation related degradation on the margin to the acceptable limit was found to be small. Additionally, the acceptable limit itself was established with some margin to the actual damage limit. Therefore, it was again difficult to conclude that ageing phenomena may have had an impact on the consequences of the accident, such as the loss of safety functions.

2.3.2.2. Seismic impact assessment on major safety related equipment

During the seismic safety assessment, the surface corrosion of the basic volute of the reactor shutdown cooling system pump was assumed as the ageing phenomenon that might have had an impact on the damage caused by the earthquake. The assessment was carried out using the seismic ground motion of the earthquake and the results of the technological assessment of ageing from actual inspection data conducted in the past. Again, it was found that the impact on the safety margin to the acceptable limit was small.

Assessments were also conducted on the shroud support, on the main steam piping and on the nuclear reactor recirculation piping, using the seismic ground motion of the earthquake and the results of the technological assessment of ageing from actual inspection data and test result records, conducted in the past, which showed low cycle fatigue cracking. The results indicated that it was difficult to attribute a contribution to the consequences of the accident (such as the loss of function in safety related components) to these ageing phenomena, because their impacts on the safety margin to the acceptable limit were sufficiently small and the acceptable limit itself had a built-in additional margin to the actual damage limit.

2.3.3. Summary of the impact of ageing on the accident

As a result of the assessment, and based on the knowledge obtained to date, it is difficult to conclude that ageing had an impact on the loss of function of the safety related equipment in the accident. Also, until access to the critical SSCs near the core becomes possible and direct tests can be conducted on actual specimens from these SSCs, it is difficult to definitively state whether or not the ageing events contributed to the occurrence and aggravation of the accident during the period between its first occurrence and the time when the accident developed beyond the safety systems design basis.

The assessments conducted have all been of a predominantly theoretical nature, because, to date, it has not been possible to conduct an on-site inspection and testing of the equipment to confirm the theoretical results. As new field observations and findings become available in the future, additional confirmatory assessments will be conducted on the impact of ageing.

If further investigation leads to an update of safety requirements, ageing management processes would be adjusted to meet the new set of regulatory requirements. The ageing management processes described in this publication will remain valid to assure the reliability and performance of SSCs in compliance with any updated safety standards that may emerge from lessons learned from the Fukushima Daiichi nuclear power plant accident.

3. PROACTIVE AGEING MANAGEMENT IN OPERATING NUCLEAR POWER PLANTS

A proactive management strategy for ageing management in a nuclear power plant implies the adoption of two non-conventional guiding principles [2]:

— Recognizing the common weaknesses of conventional AMPs of nuclear power plants, such as the inability to trace root causes, to develop lessons learned and to prioritize the use of resources to mitigate the effects of ageing on the SSCs important to safety and reliability;
— Adopting a systematic ageing management process and pursuing continuing improvements to the AMP.

It has been common practice to prioritize work related to plant performance, while work related to long term ageing management lagged behind as a secondary goal. In this frame of mind, plants were operated as long as technical specifications remained valid. Corrective measures were taken only in reaction to component failures. This corrective or reactive management of plant ageing led to failures with safety consequences, as exemplified in the steam generator tube rupture events of the Palo Verde nuclear power plant and the failure of the control rod drive mechanism (CRDM) penetration nozzle at the Davis–Besse nuclear power plant, both of which occurred in the USA.

As a result, in 2004, the NRC took action to start a proactive materials degradation assessment programme, and issued NUREG/CR-6923 [11]. Since then, preventive maintenance methods have evolved to become more proactive, and AMPs are now better suited to prevent mishaps such as that at Davis–Besse, through the use of on-line plant condition monitoring systems, advanced analysis and prognostic techniques to anticipate failures. In addition, proactive SSC AMPs take into account not just current and anticipated events, but the whole life cycle of a plant. A proactive approach to ageing management also implies the use of systematic reviews of worldwide events and operating experiences during the whole plant lifetime. This starts at the design stage for new reactors or in modernization projects, uprates, upgrades and design changes, and continues through the component manufacture/fabrication phase, plant construction, system installation, commissioning and operations (including LTOs), and extends all the way to the decommissioning period.

A proactive AMP considers both safety and economic aspects. It also extends to all associated external activities, such as engineering, procurement, fabrication, transportation and preservation in storage, among others. A proactive attitude entails a continual search for improvement of methods and processes, and continual learning from the feedback of operating experience worldwide. This comprehensive approach helps to eliminate recurrent issues and prevents the plant from lagging behind in operation techniques and in achieving plant performance targets.

By its very nature, proactive ageing management implies accountability. The operating organization takes on the responsibility to address ageing concerns specific to the plant by issuing operating instructions, procedures and R&D to identify possible issues to be managed in the future. Similarly, it assumes the obligation of identifying generic ageing issues from national and international sources. Information is collected from operations feedback, R&D, vendor recommendations, maintenance and operating manuals, plant designers and expert insight, based upon shared knowledge and the fusion of knowledge. An operator needs to apply three approaches to qualify as being proactive in managing a plant:

— Analytical: Based upon the integration of existing knowledge with the records stored in databases;
— A priori: Based upon the understanding of the fundamentals of ageing science, which is then extrapolated to component behaviour in plant operation;

— Systematic data and knowledge elicitation from experts: Based upon the synthesis of various specialized contributions from experts in correlated fields.

Each known ageing issue is addressed with an adequate AMP [2]. To foster a proactive ageing management culture in a plant, it is important that the plant staff, at all levels, be familiar with proactive ageing management principles and that a protocol, such as the PDCA developed by the IAEA, be enforced in the plant based upon the three approaches mentioned above.

Figure 6 shows the relationship between proactive ageing management and an advanced maintenance programme with the classical two types of maintenance: corrective and preventive. The flow diagram above the maintenance box represents the steps followed by corrective maintenance. Clearly, in an advanced maintenance programme, failures during operation need to be minimized and confined to run to failure components. In fact, the corrective maintenance flow leads to the end box 'Not acceptable', which is how breakdowns and unexpected failures should be considered.

The line below the maintenance box on the left shows the path of preventive maintenance. The elliptical box marked 'Proactive ageing management' in the middle represents the team responsible for a proactive approach to maintenance (e.g. the PLiM team). The arrows in and out of it indicate where a proactive approach team can intervene. Preventive maintenance can be optimized with the use of directed proactive actions such as condition based maintenance, surveillance and testing on SSCs and with direction on where and how to collect actionable data in the field. The proactive approach team receives and analyses the data that is collected in order to identify possible new degradation modes.

If a new mode is identified, the team establishes acceptance criteria and thresholds and possibly new AMPs. The dotted box in the centre of the figure highlights all the possible triggers of maintenance activities in the field. The boxes on the right represent the outcomes of an advanced maintenance programme, resulting in either dead-ended threads or in legitimate maintenance activities. The box at the top shows how, in parallel, failure analysis and maintenance feedback analysis are needed to correctly diagnose symptoms and root causes of failures, to characterize trends and to decide corrective actions, such as additional or better monitoring, improvements to the maintenance strategy and continued training for maintenance staff.

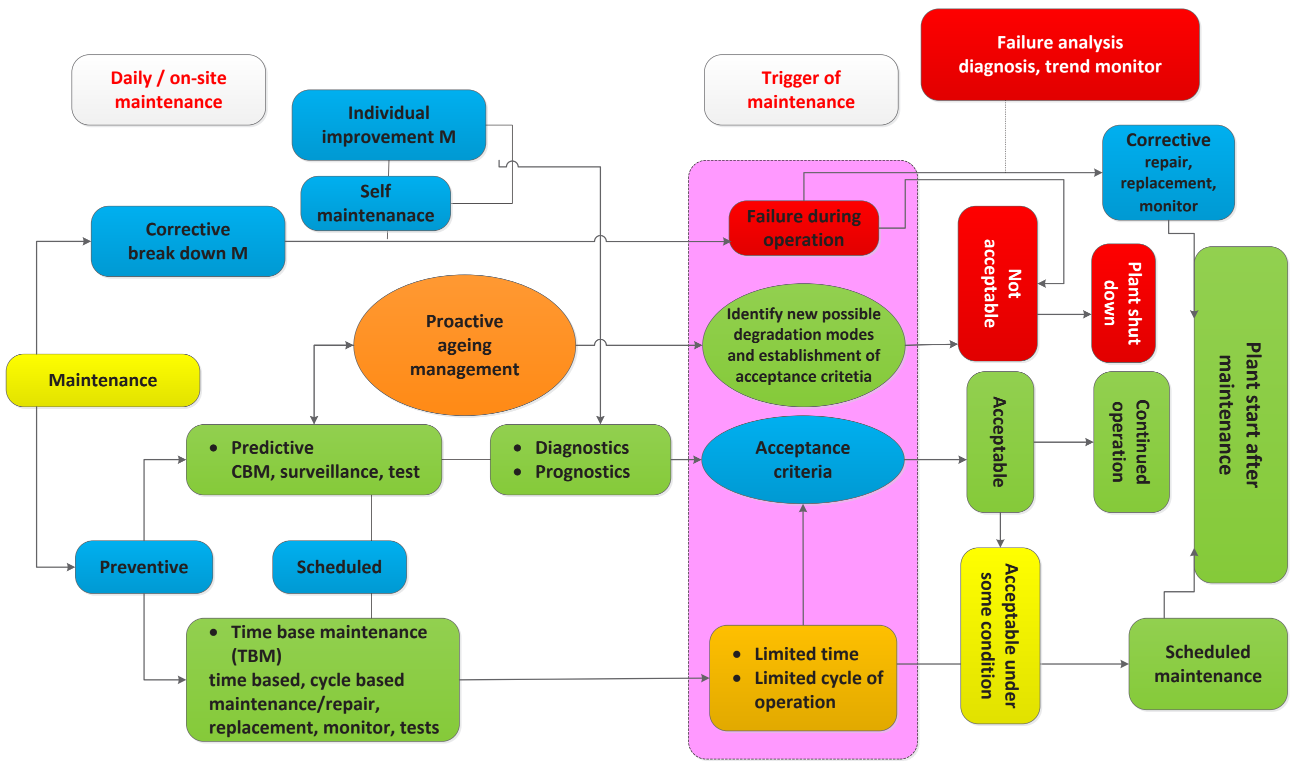

FIG. 6. Advanced maintenance approach incorporating proactive ageing management. CBM — condition based maintenance.

3.1. AGEING CONSIDERATIONS DURING DESIGN

3.1.1. Conceptual design

During the conceptual design phase of any nuclear power technology, ageing considerations of all SSCs need to be taken into account. They should be spelled out in the high level design objective documents, safety design guides, plant design specifications and SSC design requirement documents. During the design development phase, ageing should be taken into consideration in the basic assumptions and in the input data to the safety, thermohydraulic and stress analyses. All analysis reports should reference the design requirements on ageing, including those in the applicable codes and standards.

Known ageing phenomena are quantified and taken into account in the design of SSCs. The design should include the effects of wear and all other known age related degradation to ensure that safety and performance are maintained for the duration of the SSC's lifetime. If the component lifetime extends to the plant service life, as is the case for passive non-replaceable components, the design should consider all normal and transitory operating conditions, including testing stressors, maintenance interventions and the consequences of plant and system outages. Anticipated DBAs owing to all postulated initiating events are to be considered part of the operating life and hence part of the design calculations.

In general, margins consist of design margins, operational margins and safety margins. They account for uncertainties, assumptions, instrument feedback tolerances and ranges, unexpected transitory peaks, contingencies and operating flexibility. Margins are mainly set in order to minimize the probability of component failure. Only the unquantifiable ageing effects should be included in the margin estimates. The plant operator should ensure, and the regulator should verify, that margins are adequately considered in the design and that uncertainties in the effects of ageing phenomena are included in the design margins of SSCs, particularly of those important to safety. The design documents should include as a minimum the following ageing management topics [1]:

— A recommended strategy for ageing management and prerequisites for its implementation;
— A list of all safety significant SSCs in the plant that could be affected by ageing;
— Proposals for appropriate materials monitoring and sampling programmes, where ageing may affect the capability of critical SSCs to perform their functions throughout the lifetime of the plant;
— Appropriate consideration of operating experience with respect to ageing;
— Recommendations for ageing management for critical SSCs (concrete structures, mechanical components, electrical and instrumentation and control (I&C) components, cables, etc.) and measures to monitor and mitigate their degradation;
— Equipment qualification requirements of SSCs important to safety, including equipment lists, functions required for normal operation and postulated initiating events;
— General principles stating how the environment of an SSC is to be maintained within specified service conditions (location of ventilation, insulation of hot SSCs, radiation shielding, damping of vibrations, submerged conditions and water chemistry, selection of cable routes, requirements for stabilized voltage centres, etc.).

Following the Fukushima accident, additional accident mitigating features, including portable SSCs, have been introduced in the context of increasing robustness, flexibility and/or resilience of the plant to mitigate the consequences of severe accidents such as BDBAs. They may or may not be on standby and energized during normal operation, and some of these may be in storage for long periods. In any case, they should all be included in the AMP and maintenance plan.

3.1.2. Equipment qualification of structures, systems and components

The equipment qualification process is illustrated in Fig. 7. The design vendor and installation documentation normally provide important information needed before an equipment qualification programme can be established for specific component and system applications. An AMP needs to include activities prescribed in the design documentation and in the equipment qualification programme, and should be particularly concerned with preserving the equipment qualification of SSCs requiring it, throughout their service period.

3.1.3. Contractual and bid evaluation phases

In the pre-contractual phase of a new nuclear power plant, the plant owner/operator needs to clearly communicate ageing and equipment qualification requirements in the bid specification to plant technology suppliers, design organizations and manufacturers, in order to receive designs that comply with all criteria and requirements for reliability, performance and even longer term operation.

Ageing and equipment qualification considerations are important aspects, complementary to each other in plant design, and all stakeholders should be informed of the owner's requirements in these areas in the bid specification. Correspondingly, in the subsequent bid evaluation phase, ageing and equipment qualification programmes should again be included on the list of items being evaluated for the technology or technologies under scrutiny.

During the contractual phase, once the nuclear power plant technology vendor has been selected, the owner should assign the division of responsibilities to the selected main contractors, the architect–engineer and the project manager for the detail design phase and the plant deployment phases.

While ageing requirements are normally well defined in the conceptual design phase, the owner/operator should ensure that ageing requirements are not underestimated, or compromised, by any of the parties involved during the subsequent phases. It is important that the technology vendors hand over to the operators all ageing considerations and all assumptions made, as well as all design and operating margins applied.

3.1.4. Procurement phase

Ageing considerations are equally important during the procurement phase. When the plant supplier and the plant operator select the equipment manufacturers, they should determine whether ageing considerations and equipment qualification tests are in compliance with their own equipment qualification programmes. In this respect, they should guard against non-conformances, interfacing disjointedness or material incompatibilities.

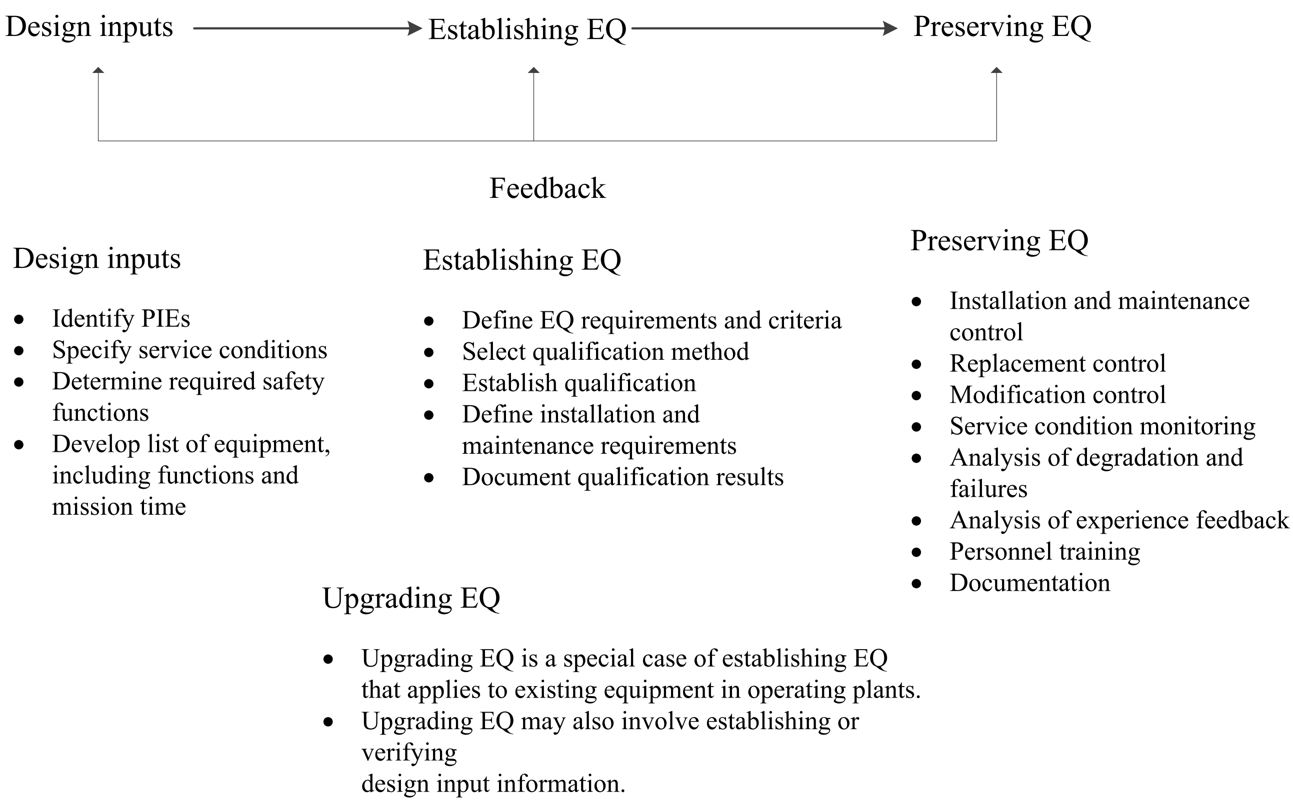

FIG. 7. Equipment qualification process. EQ — equipment qualification; PIE — postulated initiating event.

3.2. AGEING CONSIDERATIONS DURING FABRICATION AND CONSTRUCTION

The owner/operator should ensure that structural constructors, manufacturers, assemblers and installers do the following:

— Make current knowledge on the factors affecting ageing available to SSC manufacturers and properly take this knowledge into account during SSC fabrication;
— Take current knowledge about relevant degradation mechanisms and mitigation measures into account during the manufacture and installation of SSCs;
— Collect and document the appropriate amount of reference data (baseline data) at the point of origin;
— Correctly specify, make available and install surveillance specimens for ageing monitoring programmes are in accordance with design specifications.

The surface integrity of safety related components is of paramount importance in harsh environments during operation. It may not be considered critical during fabrication and installation, but surface integrity should strictly comply with the technical specifications in order to avoid premature crack initiation and accelerated material degradation during operation. During the component fabrication, construction and SSC installation phases, non-conformances or quasi-non-conformances may occur. Consequently, residual stresses may be induced that the design may not have considered. For example, during construction, a welder may apply a certain force to bring two parts (e.g. two pipe spools) closer together for welding. When the pipe spools are released, residual stresses may be induced in the weld, thus increasing the risk of crack formation during operation. Although the application of force to the parts in pipe welds is not forbidden, it is important that the plant operator be informed of the occurrences so that appropriate measures may be taken in managing the component usage factor and its ageing.

Other issues during construction could be:

— The use of inappropriate replacement materials;
— The inadvertent formation of loose parts in pipelines or vessels;
— Insufficient gap allowances for thermal expansion in pipe supports;
— Higher than normal temperatures in repairs of stainless steel welds or in multipass welding, inducing unacceptable or accumulated heat input;
— Geometrical discontinuities;
— Construction details that are not properly taken into account or modelled in stress analysis;
— Unaccounted for damage during transportation [2].

The operating organization or the contracting agency responsible for quality control and procurement should ensure that the suppliers adequately address all such factors affecting ageing management and that they provide sufficient information and non-conformance data to facilitate ageing management during plant operation [1].

3.3. AGEING CONSIDERATIONS DURING COMMISSIONING

During commissioning, the functional capability of SSCs is checked and baseline data are collected to record conditions at startup for later reference in ageing management during operation. In addition, environmental conditions in the various buildings during hot functional tests are checked for consistency with design and qualification assumptions. Commissioning records should include the mapping of the actual environmental conditions in each critical spot in the plant.

The operating organization should establish a systematic programme for the measurement and recording of ‘as-installed’ baseline data for critical SSCs. Baseline data facilitate usage factor assessments during operation or root cause investigations, should cases of premature ageing occur. The regulatory body, as part of its review and inspection programme, should ensure that the operating organization has collected all required baseline data.

Errors during commissioning may later induce accelerated ageing in some cases. For example, if the wrong resins or wrong chemicals were to be used, even though corrective action and a thorough cleanup may be undertaken during commissioning, residual aggressive agents may remain in the system and may cause early corrosion and cracks. Similarly, if abnormal temperatures occur during commissioning, the values and lengths of time that the high temperatures manifest themselves should be notified and discussed with the engineering department so that protective action may be taken to prevent systematic reoccurrence and accelerated ageing in adjacent SSCs such as concrete structures, cables, etc. Transients and extreme conditions during cold functional and hot functional tests should be recorded, and the engineering department should take measures to ensure that the system or structures involved will not be driven out of the bounds of their fatigue design assumptions during plant operation.

All parameters that can influence material degradation should be identified and recorded during commissioning. The information should be made available so that it can be tracked throughout the plant life. Special attention should be paid to the identification of hot spots in terms of temperature and dose rate, and, where relevant, vibration levels should be measured and recorded. As part of its review and inspection programme, the regulatory body should ensure that critical service conditions are in compliance with the design requirements and analysis assumptions.

3.4. AGEING MANAGEMENT DURING OPERATION

Nuclear power plant operators focus primarily on the SSCs of safety significance, and rightly so. However, they cannot ignore in their AMPs those SSCs that may not directly be of safety significance, but whose failure would have a significant financial impact on their operating budget, for example, those SSCs leading to long downtime and/or substantial repair or replacement cost penalties.

A systematic approach to managing ageing, such as the implementation of PLiM techniques, can help to produce an optimal mitigation plan of the effects of ageing. A PLiM implemented from the early stages of plant operation will automatically formulate a proactive approach to ageing management based on correctly diagnosing degradation mechanisms, and on preventing and mitigating the consequences of ageing, depending on circumstances, rather than applying a reactive approach that responds only to SSC failures, which inevitably results in increased costs and reduced performance.

A proactive approach to ageing management also means the implementation of a continuing education programme in ageing management and PLiM methods in order to increase staff awareness and familiarization with proactive ageing management, and provide the best motivation and sense of ownership of the operations, maintenance and engineering support departments. In order to ensure the correct application of proactive ageing management principles, appropriate procedures, tools, materials and qualified staff should be at hand, and sufficient spare parts and consumables should be made available to allow for preventive maintenance and proactive ageing management. In addition, the use of multidisciplinary task teams should be foreseen to deal with complex ageing management issues. The feedback from operating experience should be implemented and applied to the plant AMP. In parallel, the use of on-line monitoring and targeted NDE should be encouraged.

Figure 8 shows the typical elements necessary in the development of an ageing management strategy during the operation of a nuclear power plant.

Figure 8 was first presented by the Atomic Energy Society of Japan in 2009 and updated annually until 2011 [12]. It constitutes a strategy map for ageing management in that it points at the major principles that should govern the implementation of an AMP in nuclear power plants:

— Establishment of a material degradation database;
— Development of technical understanding;
— Evaluation of component degradation in the plant;
— Adoption of a systematic approach to maintenance and maintenance optimization;
— Use of advanced tools such as risk based guidance in the maintenance planning process.

The best manner in which to achieve proactive maintenance in an operating nuclear power plant is to integrate maintenance programmes with ageing management activities. If the plant has a PLiM programme, both maintenance and ageing management should be integrated with the PLiM programme. Figure 9 shows the interconnectivity between maintenance and ageing management and was presented at the Organisation for Economic Co-operation and Development (OECD/NEA) Workshop on Commendable Practices for the Safe, Long-term Operation of

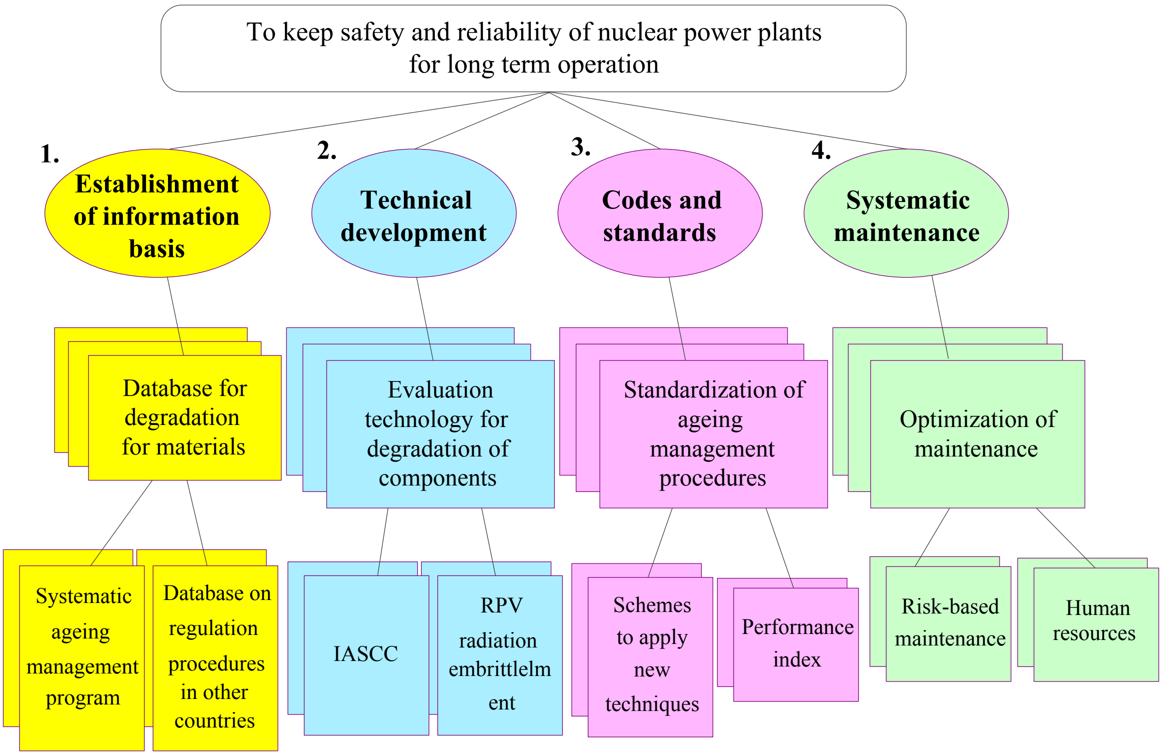

FIG. 8. Strategy map for ageing management [12].

Nuclear Reactors Nuclear Energy Agency (NEA) Stress Corrosion Cracking and Cable Ageing Project (SCAP) in Tokyo, Japan, in 2010 [13].

The white boxes in Fig. 9 indicate the interconnecting activities of the large maintenance envelope with the plant AMP. This connection gives rise to additional or modified programmes indicated by the yellow boxes, which eventually become part of the overall maintenance policies and goals and activities of the routine maintenance planning chart.

The IAEA has developed the PDCA cycle to help implement a systematic ageing management process [2].

Figure 10 shows a schematic diagram of a nuclear power plant's AMP, based on the proactive ageing management principles. The central box contains a list of typical proactive elements focused on obtaining a good understanding of material degradation, which is fundamental to a proactive approach to ageing management. Other typical proactive initiatives are:

— Review of the existing AMPs;
— Development of a susceptibility knowledge and confidence chart;
— Knowledge accumulation from the ageing management databases of similar nuclear power plants;
— Root cause analysis of failures incorporating lessons learned and corrective actions;
— Implementation of a general plant operating experience programme and of periodic independent reviews, including international generic ageing lessons learned (IGALL), the International Reporting System, IAEA Operational Safety Review Team SALTO peer reviews, World Association of Nuclear Operators (WANO) reviews, Western European Nuclear Regulators Association reviews and PSR results;
— Implementation of a configuration management programme;
— Implementation of a systematic programme of SSC scoping and screening with the inclusion of candidate SSCs into AMPs;
— Understanding the synergy of existing degradation mechanisms and new degradation;
— Implementation of a technology watch programme to capture all advances in the plant application fields.

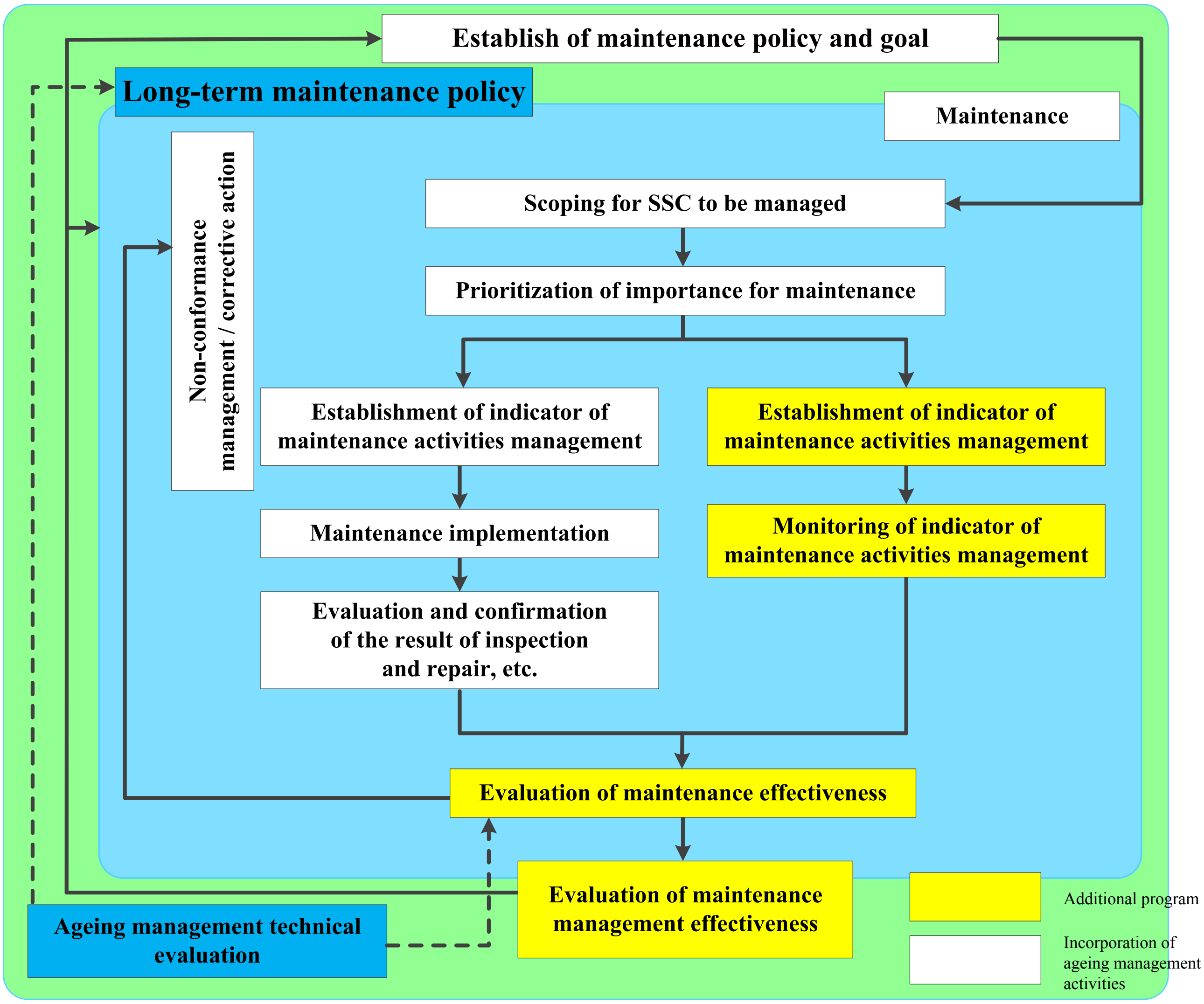

FIG. 9. Integration of a routine maintenance programme and ageing management activities. SSC — structure, system and component [13].

Once a good understanding of the SSC degradation mechanisms has been achieved, this knowledge is incorporated into the planning (PLAN), operation (DO), verification (CHECK) and corrective (ACT) phases of a nuclear power plant's proactive maintenance plan.

The items listed below are examples of important elements in the PDCA cycle:

(a) PLAN
- — National energy policy;
- — Regulatory requirements;
- — Public acceptance and communications;
- — Economic feasibility;
- — Codes and standards;
- — LTO planning;
- — Prioritized resources.

(b) DO
- — Mitigation;
- — Operation according to proposed mitigation measures;
- — Preventive maintenance.

(c) CHECK
- — Acceptance criteria;
- — Life evaluation;
- — TLAA;
- — Monitoring;
- — Field inspection and walk down;
- — Auditing and peer review.

(d) ACT
- — Regulatory licensing;
- — Licence renewal;
- — Continued operation;
- — Predictive maintenance;
- — Repair and replacement;
- — Spare part management;
- — Development of new AMPs based on results of ageing management reviews, if needed.

Figure 11 represents the key elements of an AMP for an RPV in a PWR utilizing the PDCA cycle.

The box in the middle is the first preparation step. It represents the identification of all the ageing mechanisms of a structure or component. If one of the mechanisms is new or unknown, then R&D may be required to develop an understanding. The well known mechanisms should be available in the PLiM database. Information on material properties, fabrication methods, stressors, operation experience and R&D results are also crucial to complement the understanding of the degradation mechanisms and should be available in the PLiM database.

The peripheral boxes in the PDCA cycle of Fig. 11 describe the four steps of a systematic implementation programme, as included in an AMP for a PWR RPV. The 'PLAN' box at the top is the first of these steps, and involves the planning of activities aimed at optimizing ageing management of the RPV and at improving its effectiveness. The second peripheral box to the right is the 'DO' box, which contains a collection of all activities aimed at minimizing the expected degradation of the component, such as:

- — Controlled operation of the component well within the limits of its technical specifications and in accordance with the applicable procedures;
- — Use of the best water chemistry and environmental control possible;
- — Careful recording of transients and historical events.

The box at the bottom is the 'CHECK' box, which requires periodic inspection, monitoring and condition assessment of the functional capabilities of the RPV to detect ageing symptoms. Instruments should be kept well calibrated and a good record keeping system of monitoring, inspection and assessment results should be maintained. When conditions grant it, a fitness for service investigation may be required.

The fourth peripheral box is the 'ACT' box, which deals with mitigating degradation. This involves preventive and corrective maintenance actions, spare parts inventory management and record keeping of all maintenance history.

At the end of the cycle, after going systematically through the four peripheral boxes, the loop is closed with the corrective action planning box at the top and an analysis of the loop, with a view to gaining feedback and further improving the effectiveness of the AMP.

The operating organization should also identify and address commonly encountered ageing issues in nuclear power plants such as:

- — Premature ageing of SSCs caused by more severe service conditions than assumed by design or by variances with the design;
- — Deviations in the fabrication, installation, commissioning and operation phases;
- — Errors in maintaining certain SSCs in the plant;
- — Lack of coordination in the execution of the AMP;
- — Unforeseen ageing phenomena;
- — An exclusively reactive AMP;

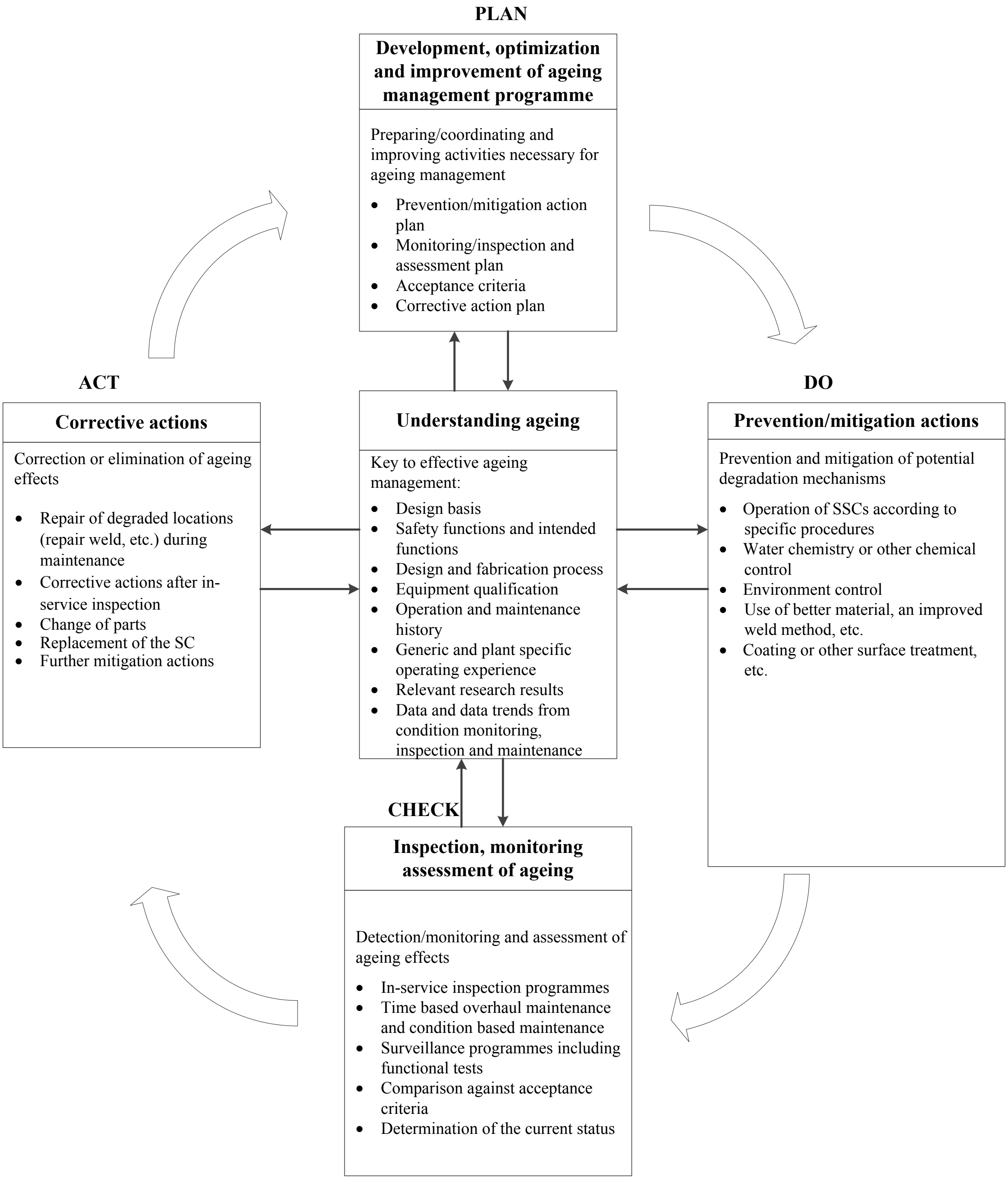

FIG. 10. Systematic approach to ageing management. SC — structure and component; SSC — structure, system and component [2].

— Lack of awareness of experience feedback, of new developments and of the latest R&D results;
— Unexpected stress loading to structures or components owing to extreme external events (e.g. earthquakes, floods, etc.).

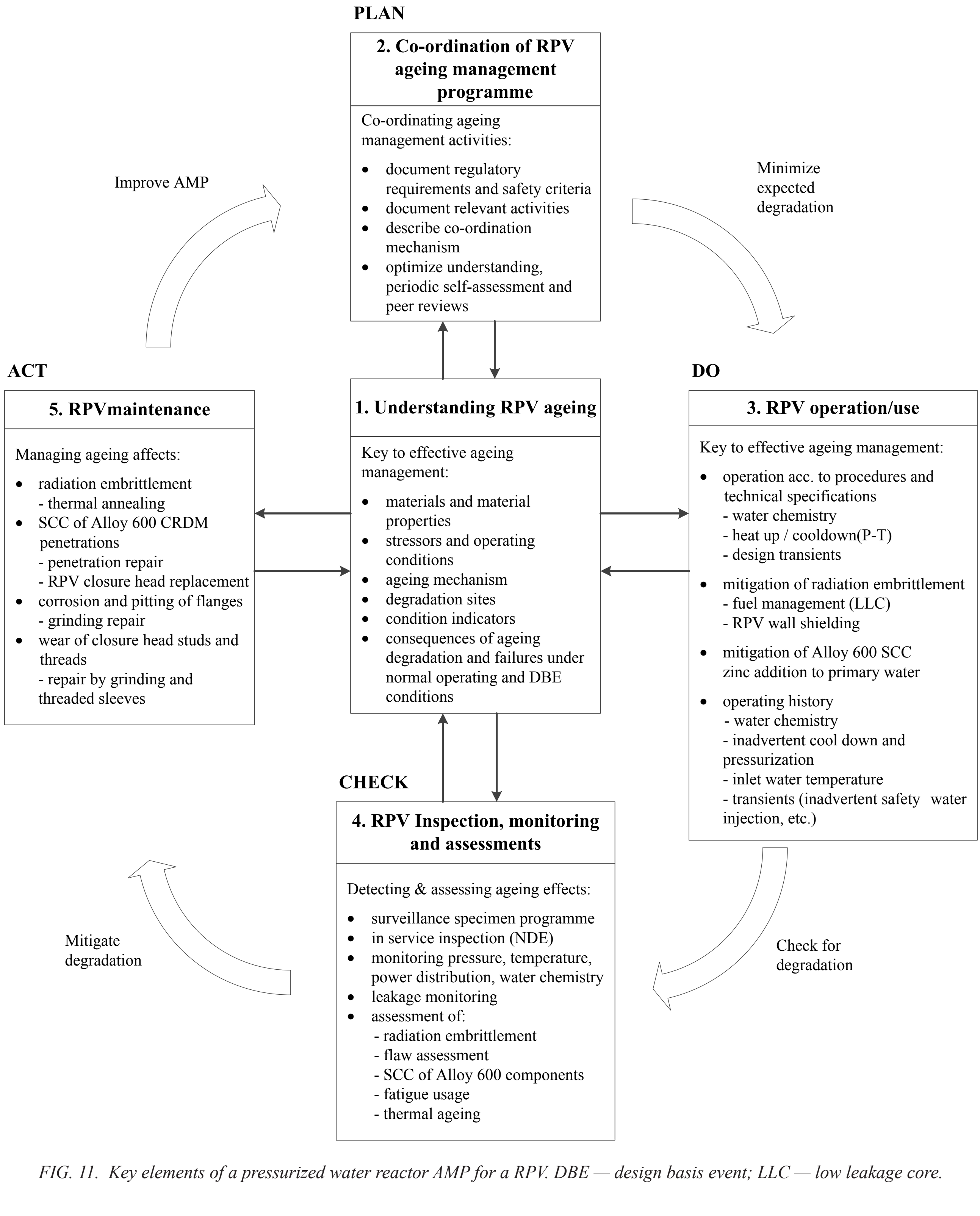

FIG. 11. Key elements of a pressurized water reactor AMP for a RPV. DBE — design basis event; LLC — low leakage core.

The cooperation and the beneficial relationships among all stakeholders in SSC ageing management of a nuclear power plant are essential. The cooperation should involve the regulatory authority, academia and engineering support, the industrial community providing services to the plant, the ageing management group and the maintenance department of the plant.

3.4.1. Refurbishment, modernization and power uprating

When dealing with improvement projects in a nuclear power plant, such as reactor power uprating, large plant refurbishments, modernizations or large equipment replacements, the design organization in cooperation with the operations organizations should identify all possible changes in process conditions (e.g. flow pattern, velocity or vibration) that could cause accelerated or premature ageing, or even failure of some components, and implement remedial action where necessary. In addition, if, because of obsolescence or for other reasons, a component needs to be replaced with another of similar but not identical specification, a thorough assessment needs to be conducted because material incompatibility issues may arise, even though the form and functions may remain the same. In addition, different specifications may produce discontinuities and accelerated ageing issues, from which safety consequences may arise.

A good example of this is the case of the guide tube pins in a reactor vessel's upper internals, typically made of a nickel based alloy. Very minor differences in the fabrication process have negatively affected the component sensitivity to stress corrosion. Another example is the case of the pressure tube feeder pipe thinning issue in Canada deuterium–uranium (CANDU)[1] type reactors, typically made of American Society of Mechanical Engineers (ASME) SA106 carbon steel. Thinning occurred in the feeders of early models. Operating experience and R&D evidence determined that feeder pipe bends in the reactor face region were much more susceptible to stress corrosion if fabricated with chromium content in the lower percentile (<0.2%) of the chromium band, than feeder pipes fabricated with the chromium content in the upper percentile (>0.2%) of the band. This evidence induced designers in refurbishment projects and in new build CANDU reactor projects to closely specify the chromium content for feeder pipes.

When a new ageing mechanism is discovered (e.g. through feedback of operating experience or research), affected nuclear power plant operating organizations should prepare contingency plans or modify their maintenance plans to deal with the issue.

Another area of concern is the management of spare parts. The availability of spare or replacement parts and their shelf life, and the availability of consumables, should be periodically monitored and protected from ageing related damage [1].

The rate of material degradation can often be reduced by optimizing operating practices and system parameters. For example, by optimizing the fuel pattern in the core and by reducing certain operating parameters, such as neutron fluence on PWR vessel walls, their embrittlement rate can be reduced, thus providing a potentially substantial mitigation of the effects of ageing [14]. Another area of possible operator intervention to reduce the rate of material degradation of the primary system components can be the careful management and recording of transients. In most cases, this allows the operator to improve stressors and fatigue usage factors. The lower usage factors can be subsequently used to replace those conservatively assumed by the original design organization in the system stress analysis and so allowing the operator to prolong the useful life of all system components.

3.4.2. Long term operation

The Member States that follow the PSR process to periodically take stock of the safety condition of the plant will find the task of justifying LTO easier if they have implemented a systematic approach to plant ageing management. A good AMP conducted from day one in the service life of a plant is an assurance that the plant will be in its best possible condition and will require a minimum number of changes at the time of its PSR submission for longer term operation, which usually requires more demanding justifications for continuing operation than regular PSRs. The plan of action shown in Fig. 12 illustrates a systematic phased approach developed by the Swedish Radiation Safety Authority for its operators to adequately justify continuing operation for a period longer than first postulated in the original design. This methodology has proven successful in the preparation of the PSR submission for longer term operation.

The blocks at the top of Fig. 12 represent the preparatory phase prior to the LTO safety review submission. The middle blocks denote the LTO assessment phase. In this phase, nuclear power plant operators are required to verify that their AMP is adequate. They conduct this verification through an ageing management review in which

[1] A commercial designation, indicating Canadian designed pressurized heavy water reactors.

they ensure that the original component design that has operating time limitations is verified for the new intended time of operation (e.g. 60 years). This verification is referred to as TLAA.

In keeping with IAEA Safety Guide publication on Ageing Management for Nuclear Power Plants (NS-G-2.12) [1], paragraph 2.5, and Safety Reports Series No. 57 [15], section 5.3, the regulator in Sweden requires that plant ageing be managed proactively, by coordinating existing programmes such as maintenance, ISI, surveillance, operation programmes, technical support programmes (including analysis of ageing mechanisms) and R&D. Ageing management should be a cross-cutting programme that provides a methodological process to detect and mitigate material degradation. This process should be used as part of the justification for safe LTO.

The blue block at the centre of Fig. 12 represents the proactive review effort of the AMPs of all passive SSCs, singled out by the screening process and grouped by major systems and engineering disciplines. All existing maintenance and surveillance programmes are checked for LTO acceptability, as represented by the yellow box on the left. The blocks to the right of the central blue box represent the typical major LTO projects being assessed for LTO and for which TLAAs are produced to qualify components, subcomponents and subsystems requiring it in each of the major LTO projects. Out of this assessment phase, a number of changes and upgrades are recommended and eventually committed.

The lower blocks in Fig. 12 represent the implementation phase of the committed changes. The figure shows how Sweden's nuclear regulatory framework integrates IAEA recommendations within the PSR process [12]. It should be noted that during the plant wide scoping and screening process, all relevant SSCs are considered. However, during the review of the plant AMP in the following stage of the process, only the long lived or hardly replaceable passive structures and components are considered.

Another model, shown in Fig. 13, was developed in the context of the Chinese licensing framework and national standard that requires a safety review submission for operation beyond the period first postulated in the original design.

Figure 13 represents a systematic ageing management model, provided by Qinshan phase 1 nuclear power plant in Zhejiang Province, China. The model includes an overall AMP and a series of topical and equipment specific AMPs. The overall AMP model deals with general aspects of ageing management at the plant level, such

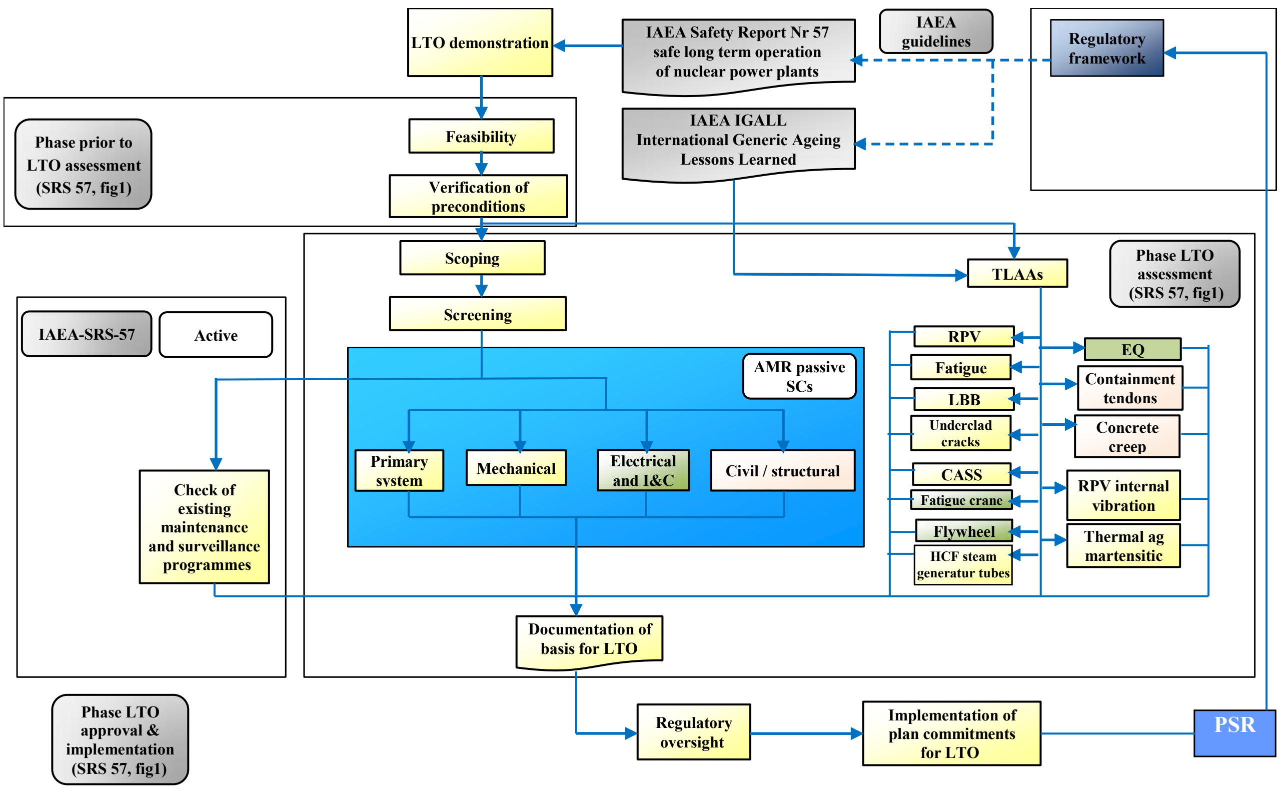

FIG. 12. Plan of approach for LTO projects (Swedish Radiation Safety Authority). AMR — ageing management review; CASS — cast austenitic stainless steel; EQ — equipment qualification; LBB — leak before break; SC — structure and component.

as plant policies, organizational structure, division of responsibilities, resource allocation, screening of ageing management equipment and identification of significant ageing mechanisms, among others. The equipment specific AMPs cover ageing management procedures for structures or components selected in the screening process. The AMPs define the ageing issues, the plan and the work scope, including the ageing mechanism analysis, the inspection programme, data collection, the ageing assessment, corrective actions for ageing mitigation and record keeping, among others. The topical AMPs focus on ageing issues across the plant by topic, such as FAC, SCC and obsolescence, among others. The systematic AMP model also includes the development of an ageing management database to be merged with the general plant information system. PSRs can be used to improve, as well as to evaluate, the plant AMP system. In the PSR cycles, the AMP system is reassessed based on the PSR guidelines, potential weaknesses are identified and recommendations are made for further improvements.

When a design upgrade is deemed necessary to enhance safety or performance for an LTO submission, there is a process to assess the project (see Fig. 14). From right to left, first a screening or clustering step is undertaken to review all issues, namely the native LTO issues, the pre-existing proposed improvement projects, which may or may not have LTO implications, and the non-LTO issues. A selection of the issues to be addressed in an LTO submission is undertaken based on a safety impact assessment and expert judgement supported by probabilistic insights (if available). The list of possible changes is consolidated and a short list of LTO issues is included in

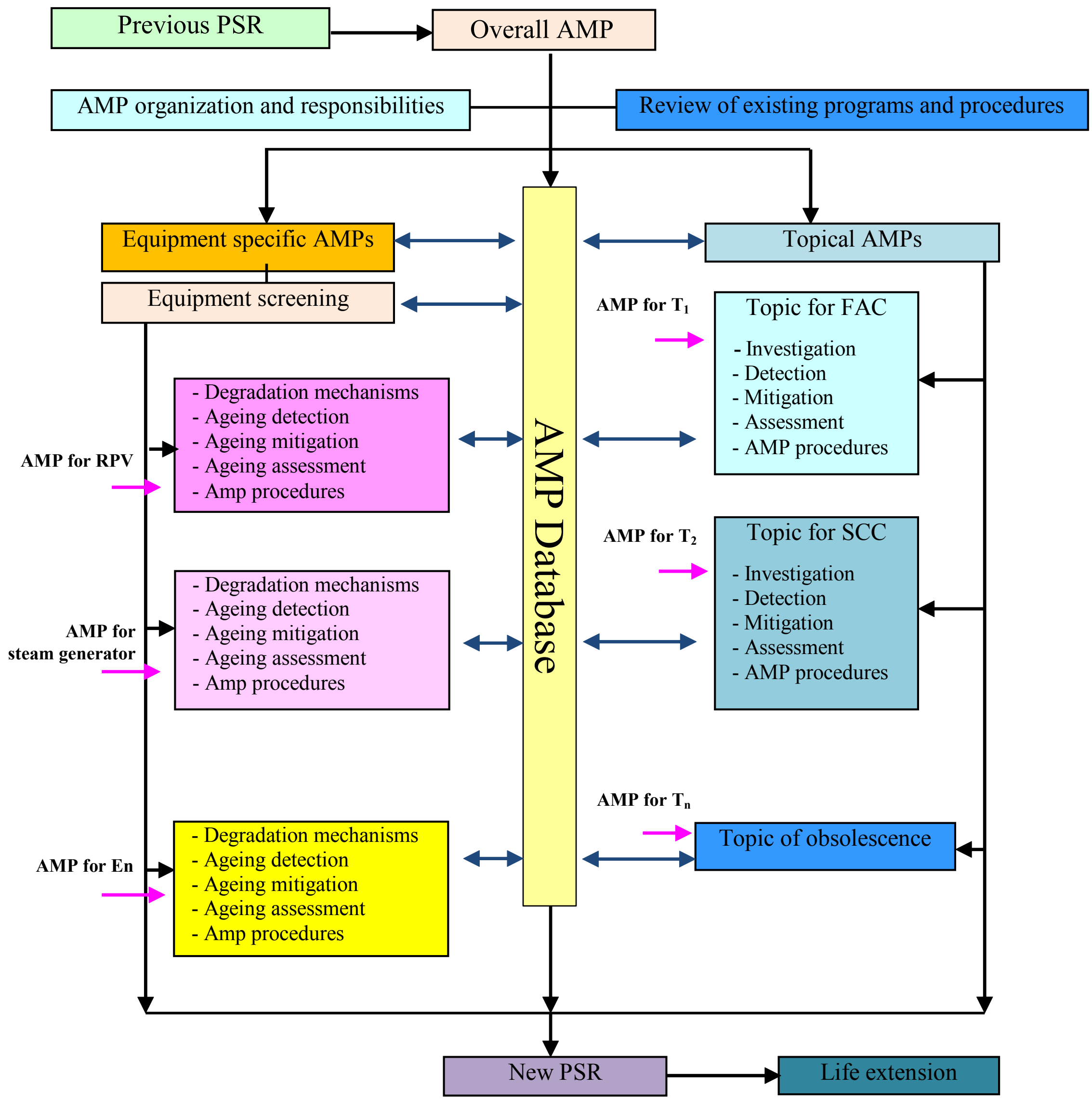

FIG. 13. Systematic ageing management model for Qinshan-1, China. En — engineering; SG — steam generator; T_1 — topic 1; T_2 — topic 2; T_n — topic n.

the PSR submission to the regulatory body for an LTO permit. A cost–benefit analysis is also conducted on all consolidated issues and is to include the proposed improvements in a feasibility study, which requires an estimate of the safety benefits. All inputs are then collected and an integrated assessment is conducted. The process eventually leads to an improvement plan, which includes both the short listed LTO drivers and the work plan containing the approved design upgrades.

3.5. AGEING MANAGEMENT PREREQUISITES FOR DECOMMISSIONING

Decommissioning activities occur in stages that can be increasingly financially onerous. To ensure these obligations will be met, regulators usually require, as early as the initial construction and operating licence submission(s), a full decommissioning plan with financial guarantees to execute it.

Once a licensee decides to decommission a plant, the regulator requires an application that includes an environmental assessment to show how the environmental footprint left by the decommissioned plant meets the environmental laws of the land for industrial operations and the international commitments of the State in which the decommissioning is taking place.

Appropriate arrangements, including proper ageing management, upgrades and replacements, among others, should be made for all SSCs required to support decommissioning (e.g. containment system, cooling equipment, lifting equipment and condition monitoring equipment) to ensure that they continue to remain available and functional during decontamination, dismantling and other decommissioning activities. In terms of safety of the decommissioning operations, the licensee is similarly bound to preserve the functionality of all those facilities and tools essential to carry out all decommissioning activities safely and to ensure the protection of the decommissioning staff, the public and the environment.

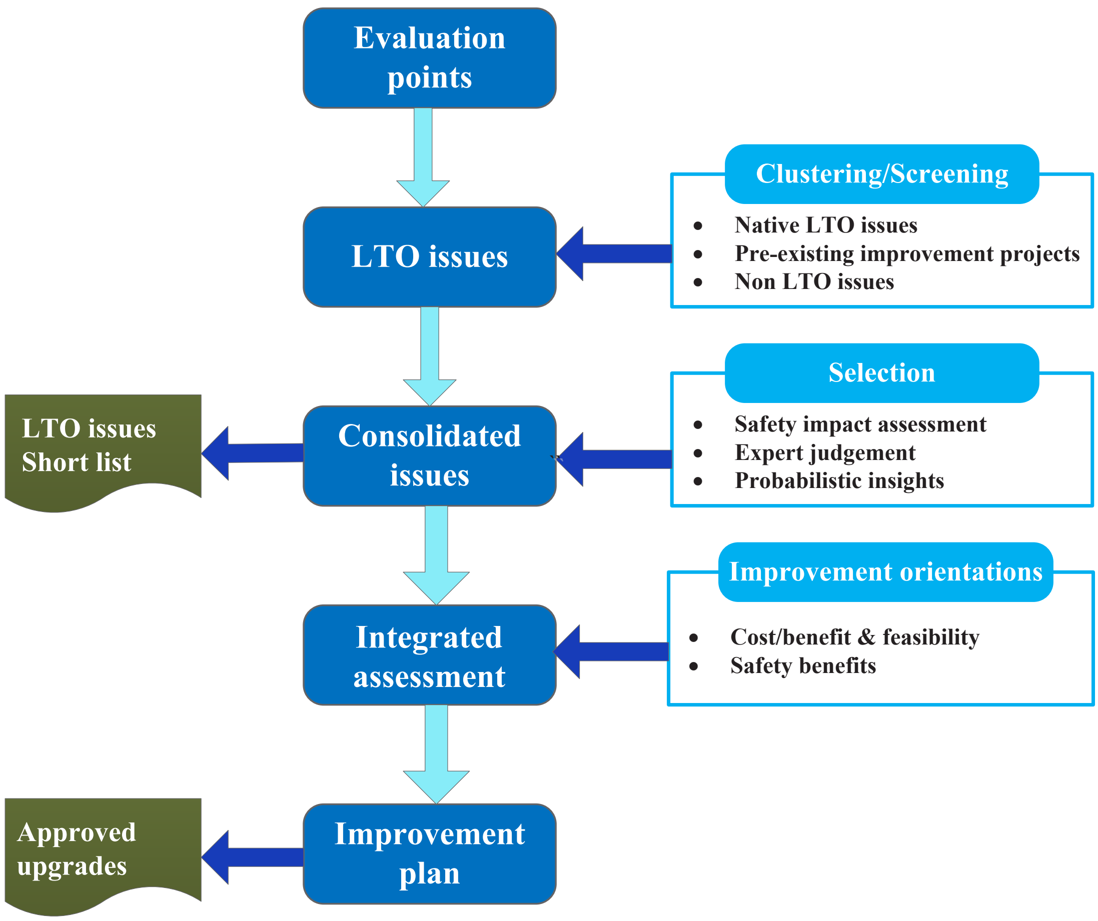

FIG. 14. Design upgrade assessment process.

Initially, the licensee may put the plant in a safe storage condition until sufficient radioactive decay has occurred in the plant and the residual radioactivity is manageable. All physical preparatory steps can be completed during this period, before decontamination operations are started.

Most parts of a nuclear power plant do not become radioactive during its service life, and the metal recovered from dismantling the plant can be recycled. Those components that become radioactive generate only low radiation levels. However, decontamination of such components may involve the construction of large decontamination systems and a storage tank facility to contain the by-products of decontamination activities. In addition, safe storage and safe burial space for low and medium radioactivity level waste may be required. Even before dismantling occurs, defuelling and removal of the reactor coolant and moderator need to be completed. Facilities and equipment necessary to implement defuelling and coolant/moderator removal, such as drums, storage facilities and their support systems, need to be maintained in a functional operating state throughout the life cycle of the plant to allow not only decommissioning at the end of the life cycle, but also to be used during major reactor refurbishments, repairs or upgrades, if necessary.

Approximately 99% of the radioactivity in a nuclear power plant to be decommissioned resides in the fuel and is removed from the plant when the reactor and the spent fuel bays are defuelled. Surface contamination is usually present in the systems containing the primary coolant and the moderator. The metal of the reactor vessel and associated structures may be activated by material mutations into active isotopes induced by years of high irradiation, but the half-lives of induced radioactive isotopes are relatively short. This characteristic is often the reason that operators, such as Électricité de France (EdF), have decided to adopt partial dismantling, with postponement of final demolition to later years to take advantage of radiation decay of these radioactive isotopes with time.

States without decommissioning experience may be faced with the following issues:

— Lack of a clearly defined regulatory requirements for the decommissioning of nuclear facilities;
— Lack of a waste management policy including waste transportation or routeing requirements;
— No long term storage and final waste disposal facilities.

In addition, fuel disposal and the management of spent fuel in general may also be an unresolved issue and should be addressed before the decommissioning of any nuclear power plant [16–18].

3.6. EFFECTIVE AGEING MANAGEMENT PROGRAMMES

The generic attributes of an effective AMP are summarized in Table 1 [1].

Administrative control of the development and execution of an AMP should not be neglected. It is an important activity aimed at maintaining quality control over the process. Figure 15 shows the overall ageing management procedure at the Paks nuclear power plant in Hungary. It illustrates the administrative control process of all ongoing ageing management activities at the plant, which can be grouped into four phases:

— Scoping or scope setting of the overall plant level AMP;
— Development and/or continual improvement of the component specific AMPs;
— Implementation of the AMPs;
— Feedback of the implemented AMP results into the overall ageing management documentation.

This procedure was designed to allow international and domestic experience, accumulated knowledge, new requirements and best existing practices to be taken into consideration, so that any relevant consequences on the plant could be integrated into the procedure and into the AMPs.

The basic rule for scope setting in Hungary is that all passive, safety related SSCs have to be included in the scope of the overall plant level ageing management activities. Also to be included are those non-safety related SSCs whose failure can jeopardize the safety functions of safety related SSCs. Based on the safety significance of the SSCs, the ageing management scope can be functionally split into two large groups:

TABLE 1. GENERIC ATTRIBUTES OF AN EFFECTIVE AGEING MANAGEMENT PROGRAMME [1]

Attribute	Description
1. Scope of the ageing management programme based on understanding ageing	● Structures (including structural elements) and components subject to ageing management ● Understanding ageing phenomena (significant ageing mechanisms, susceptible sites): — Structure/component materials, service conditions; stressors, degradation sites, ageing mechanisms and effects — Structure/component condition indicators and acceptance criteria — Quantitative or qualitative predictive models of relevant ageing phenomena
2. Preventive actions to minimize and control ageing degradation	● Identification of preventive actions ● Identification of parameters to be monitored or inspected ● Service conditions (i.e. environmental conditions and operating conditions) to be maintained and operating practices aimed at slowing down potential degradation of the structure or component
3. Detection of ageing effects	● Effective technology (inspection, testing and monitoring methods) for detecting ageing effects before failure of the structure or component
4. Monitoring and trending of ageing effects	● Condition indicators and parameters monitored ● Data to be collected to facilitate assessment of structure or component ageing ● Assessment methods (including data analysis and trending)
5. Mitigating ageing effects	● Operations, maintenance, repair and replacement actions to mitigate detected ageing effects and/or degradation of the structure or component
6. Acceptance criteria	● Acceptance criteria against which the need for corrective action is evaluated
7. Corrective actions	● Corrective actions if a component fails to meet the acceptance criteria
8. Operating experience feedback and feedback of research and development results	● Mechanism that ensures timely feedback of operating experience and research and development results (if applicable), and provides objective evidence that they are taken into account in the ageing management programme
9. Quality management	● Administrative controls that document the implementation of the ageing management programme and actions taken ● Indicators to facilitate evaluation and improvement of the ageing management programme ● Confirmation (verification) process for ensuring that preventive actions are adequate and appropriate and that all corrective actions have been completed and are effective ● Record keeping practices to be followed

— A group comprising the critical and safety related SSCs, such as the RPV, the reactor upper unit, the reactor internals and the main components of the primary coolant loops, whose ageing is individually managed through customized AMPs.
— The rest of the SSCs, divided into commodity groups. SSCs belonging to the same commodity group are identified by the similarity of materials, fabrication, form and function. These similarities result in the same ageing mechanisms for all components of each group, allowing also their ageing to be managed similarly.

Component specific AMPs comply with the 10 attributes of AMPs specified in NS-G-2.12 [1]. When an AMP practice is reviewed, the overall operational type ageing management support programmes, such as ISI programmes, maintenance procedures, water chemistry programmes, condition monitoring and/or other control programmes, are

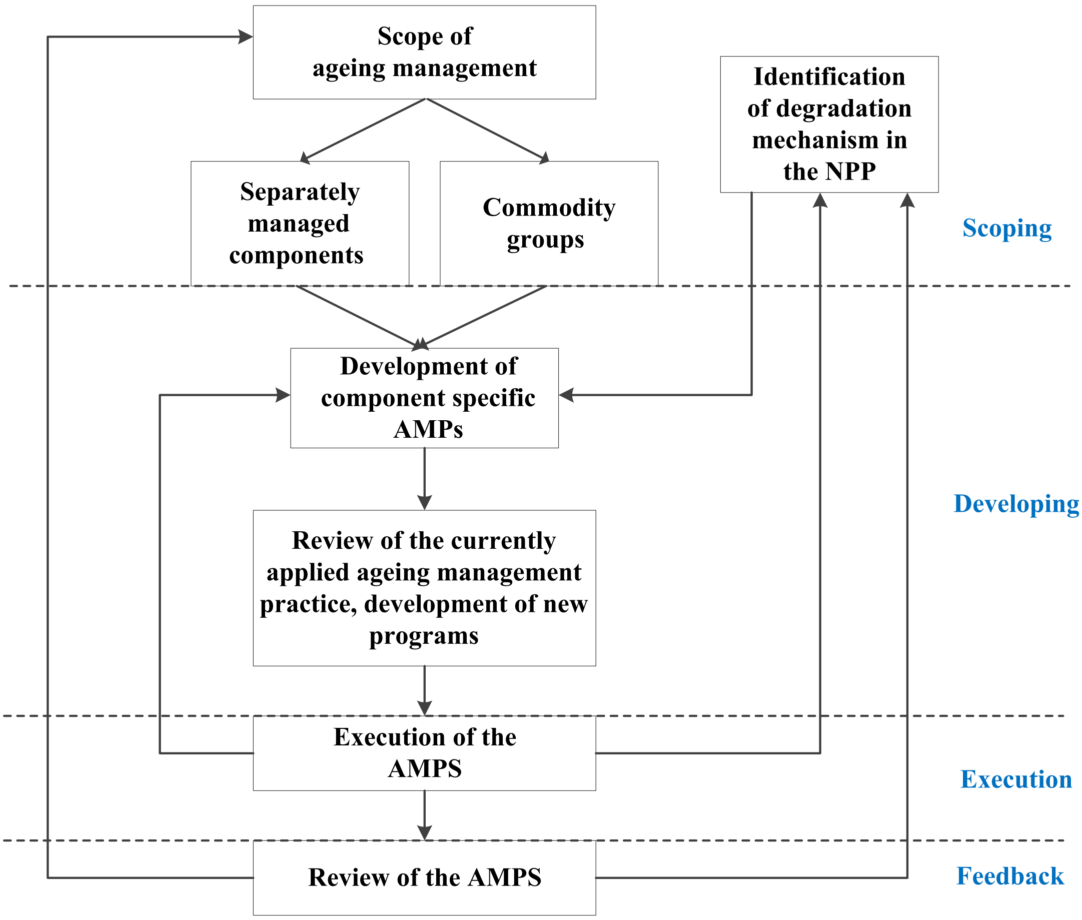

FIG. 15. Administrative control of the AMP at the Paks nuclear power plant in Hungary. NPP — nuclear power plant.

also reviewed and upgraded, as necessary, in accordance with the above mentioned 10 AMP attributes. This step leads to the harmonization of the content of any component specific AMP to the overall operational type AMP.

Any feedback from the ISI, the maintenance activities and other operational programmes is continually analysed, together with the information gathered from the component physical condition reports. When a previously unsuspected degradation mechanism, and its ageing effects, is discovered or a new degradation site is found on a component, the list of the potential degradation mechanisms is updated and a new degradation type AMP is developed. In support of these initiatives, the necessary R&D and scientific literature searches are conducted.

During the periodic review of the AMPs, relevant information is collected to decide whether the scope of the overall plant level operational type AMP or the content of a given AMP is still sufficient and effective or if any upgrades are needed. If updates are necessary, new full documentation is issued and the old versions are archived.

4. DEGRADATION MECHANISMS

Degradation mechanisms owing to ageing can be divided into two main categories:

— Those affecting the material internal microstructure or its chemical composition and thereby its ageing regression (e.g. thermal creep, radiation creep, irradiation damage, such as embrittlement, etc.);
— Those imposing macrogeometrical damage to the component either through metal loss (e.g. corrosion or wear) or through cracking or distortion (e.g. stress corrosion, deformation or cracking).

Figure 16 illustrates the concurrent action of typical stressors present on a material exposed to the typical environment of a water cooled nuclear reactor.

In the region closest to the core of water cooled nuclear reactors, metallic materials represented by the grey rectangle are subject to the concurrent influence of chemical stressors from the cooling medium, of internal and external stress fields and of stressors tied to the high radiation levels of the environment. The concurrent stressors are represented by the mixed coloured squares superimposed over the grey rectangle denoting the material. The square located at the centre of the diagram indicates a condition, the IASCC of austenitic stainless steel, to which LWR vessel internals are typically subjected.

The light green square in the lower left region represents high energy radiolysis of H_2O/D_2O molecules, turning them into H^+/D^+ and OH^- radicals. This is condition that the coolant, be it light or heavy water, undergoes as it is exposed to ionizing radiation. These H^+/D^+ and OH^- radicals are themselves chemically reactive, and as they recombine, they produce a series of highly reactive compounds such as superoxides (HO_2) and peroxides (H_2O_2), which inevitably produce oxidation damage to metallic surfaces. The squares to the right of the light green square represent the effect of radiolysis on the corrosion of stainless steels. The combination of stressors becomes particularly aggressive in the presence of oxygen, a condition conducive to intergranular stress corrosion cracking (IGSCC).

The effect of radiation and water radiolysis on activity transport in nuclear power plants has been the subject of numerous investigations. Radiation induced segregation is the consequence of irradiation at elevated temperatures, where the depletion of alloying elements (typically, chromium, molybdenum and iron) and the enrichment of others (typically, nickel and silicon) occur in regions near the component surface. In extreme cases, there can be dislocations, void creation, newly formed grain boundaries or phase boundaries of segregated elements within the alloy matrix.

Irradiation creep is the condition whereby changes occur in the physical properties of the metal owing to radiation exposure (namely its strength, ductility and elasticity moduli) that may lead to swelling, elongation, metal fatigue and even to creep related ruptures. Swelling is exacerbated by the effects of high temperatures, cold bending, impact damage and high stress levels.

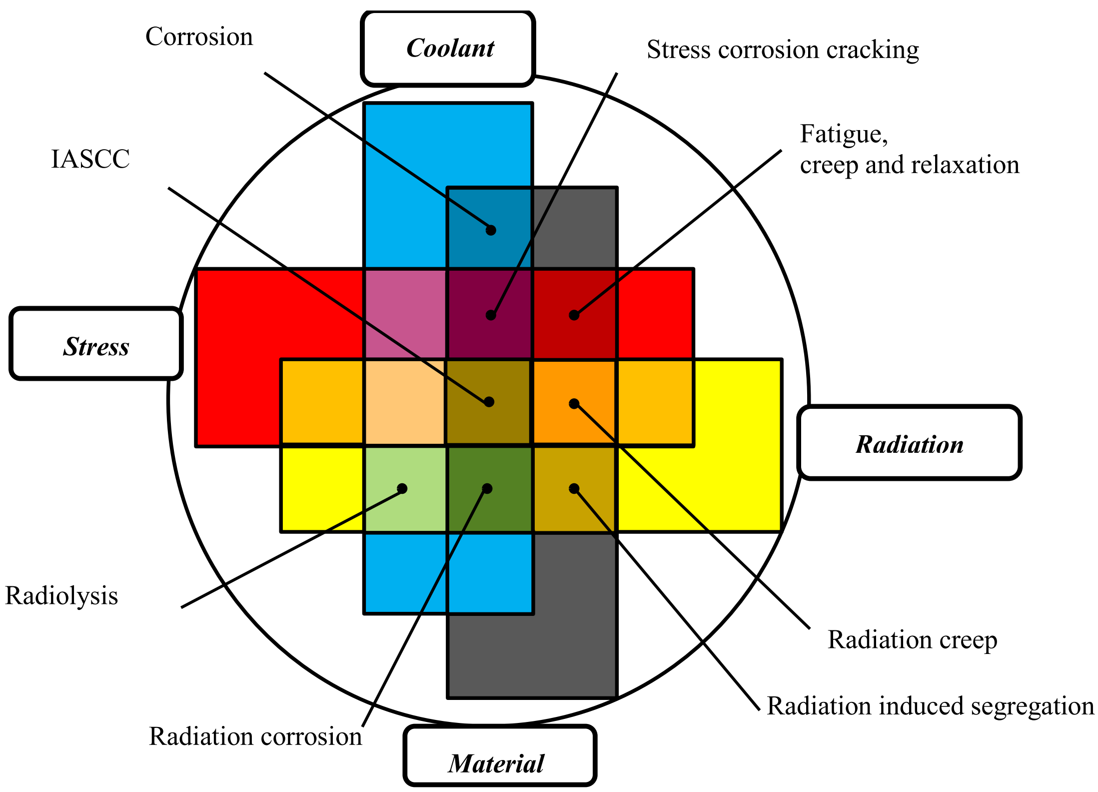

FIG. 16. Concurrent action of all stressors present on a material.

SCC is a condition typical of austenitic steels and aluminium alloys whereby, under corrosive environments (such as those induced by the presence of chlorides, alkalis, nitrates or ammonia), cracks form, which can lead to sudden failures, especially at elevated temperatures. The metal surface may appear shiny and unaffected, but the cross-section can harbour cracks that often go undetected. The condition can progress rapidly, producing devastating and unexpected failures. When cracks are detected and characterized through ISI, an analytical method, called fracture mechanics, is used, which applies the theories of elasticity and plasticity to the microscopic crystallographic defects and is able to predict mechanical failures with acceptable accuracy.

Degradation mechanisms include metallurgical phenomena such as irradiation embrittlement, fatigue, corrosion, interaction and any combinations thereof. Table 2 illustrates the main degradation mechanisms for nuclear power plant components.

Figure 17 correlates the stressors present in the material to the initiation of ageing mechanisms responsible for the observable damaging consequences.

A specific type of SCC is the cracking caused by radiation fields. Operating experience and R&D have shown that several austenitic stainless steels, such as types 304, 316 L, 316 CW and 347, are susceptible to IASCC, which is a degradation mechanism affecting mainly the reactor internals. In practice, it may be difficult to determine, simply through field observations unsupported by R&D, whether cracking is caused by IASCC or by other types of SCC.

4.1. RADIATION DAMAGE

Radiation induced microstructural changes significantly degrade material properties. Radiation damage in RPV steels is usually correlated with neutron fluence. Two different threshold energies have been identified for the characterization of irradiation conditions: neutrons with energies larger than 1 MeV for PWRs and BWR RPVs, and neutrons with energies larger than 0.5 MeV for water cooled water moderated power reactor (WWER) RPVs. Neutron fluences are not constant for all irradiation conditions; therefore, any data comparison should be analysed taking this into account. Radiation damage in the material of reactor internals is usually correlated to the displacement of atoms in the material in order to quantitatively characterize it easily. A physical unit called ‘displacements per atom’ (dpa) is used, which is a consequence based radiation exposure measurement unit and represents the number of atoms displaced from their normal lattice sites as a result of continuous subatomic particle hits. Although radiation damage cannot be fully characterized by a single parameter, dpa is well suited for correlating radiation to physical property alterations.

TABLE 2. MAIN DEGRADATION MECHANISMS IN PRESSURIZED WATER REACTOR COMPONENTS [19]

Materials	Phenomena	Subcomponents	Mechanisms	*S*	*K*	*C*
Carbon and low alloy steels base metal	Irradiation effects	Nozzle, intermediate and lower shell courses, inlet and outlet nozzles	Embrittlement	2.9	2.1	2.9
Carbon and low alloy steels welds and clad	Irradiation effects	Nozzle course welds, inlet and outlet nozzle welds	Embrittlement	2.8	2.1	2.8
Ni alloy base (A600)	Stress corrosion cracking	Bottom mounted nozzles and repair pieces	Intergranular/ transgranular	2.3	2.8	2.8
Ni alloy welds and clad (A82/182)	Stress corrosion cracking	Dissimilar metal welds for inlet and outlet nozzles, core flood nozzle, safety injection nozzles, bottom mounted nozzles, repair pieces	Intergranular/ transgranular	2.6	2.6	2.8

Note: *C* — confidence; *K* — knowledge; *S* — sensitivity.

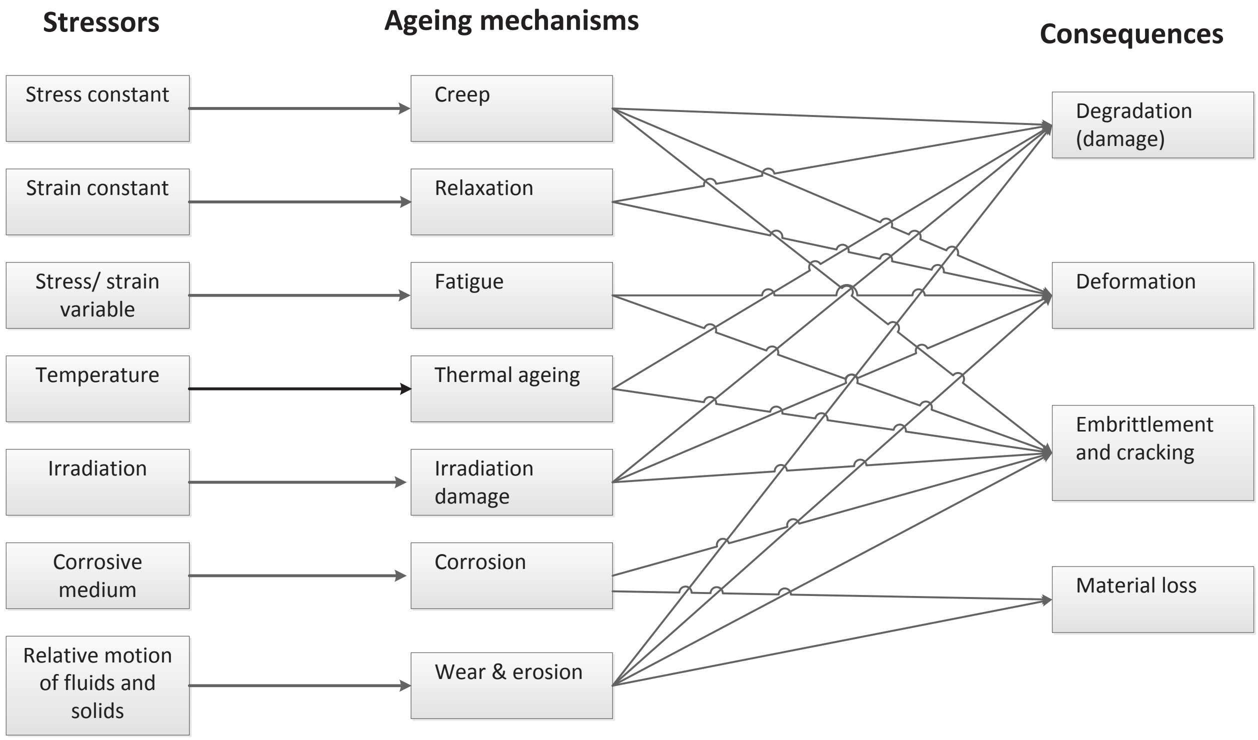

FIG. 17. Ageing factors (stressors), ageing mechanisms and consequences.

The RPV walls of PWRs and BWRs are fabricated from low alloy steels, of the manganese–nickel–chromium–molybdenum type and chromium–molybdenum–vanadium type, or of the nickel–chromium–molybdenum–vanadium type in WWER designs. The most damaging mechanisms are radiation embrittlement and radiation hardening, which determine the RPV lifetime. Embrittlement depends mainly on neutron fluence and on the material, specifically on the alloy composition and on the presence of detrimental elements (e.g. phosphorus and copper). Radiation embrittlement is characterized by the material transition temperature shift, which causes a change in the fracture type from brittle to ductile as the material moves to higher transition temperatures.

The RPV internals of BWRs and PWRs are mainly fabricated from austenitic stainless steels. Unlike the fuel elements, which are removed after a few years of service, the internals remain for the full service life of the plant. Consequently, they can be exposed to very high radiation doses, typically 5–10 dpa in a BWR and up to 80 dpa, depending on fuel management, in a PWR (assuming a 40 year service life). With such high radiation doses, the material microstructure and mechanical properties change considerably, affecting the susceptibility to stress corrosion in both reactor types. Other effects of radiation damage are swelling, creep and IASCC [20, 21].

Figure 18 shows the major degradation processes affecting austenitic steels under neutron irradiation and their effects on the material's physical and mechanical properties.

This diagram correlates the physical phenomena owing to neutron irradiation in the orange boxes at the top to the degradation mechanisms at the microstructural level, shown in the blue boxes, and to changes in the material physical properties, shown in the pink boxes.

In addition, neutron capture induces transmutations and, hence, changes in chemical composition. In summary, from a material viewpoint, the following radiation induced changes should be considered in relation to neutron irradiation:

(a) Microstructure:
- — High irradiation induced dislocation;
- — Loop density;
- — Cavities (bubbles and voids);
- — Partial $Fe_{\gamma} \rightarrow Fe_{\alpha}$ transformation;
- — Cluster formation;
- — Grain boundary segregation.

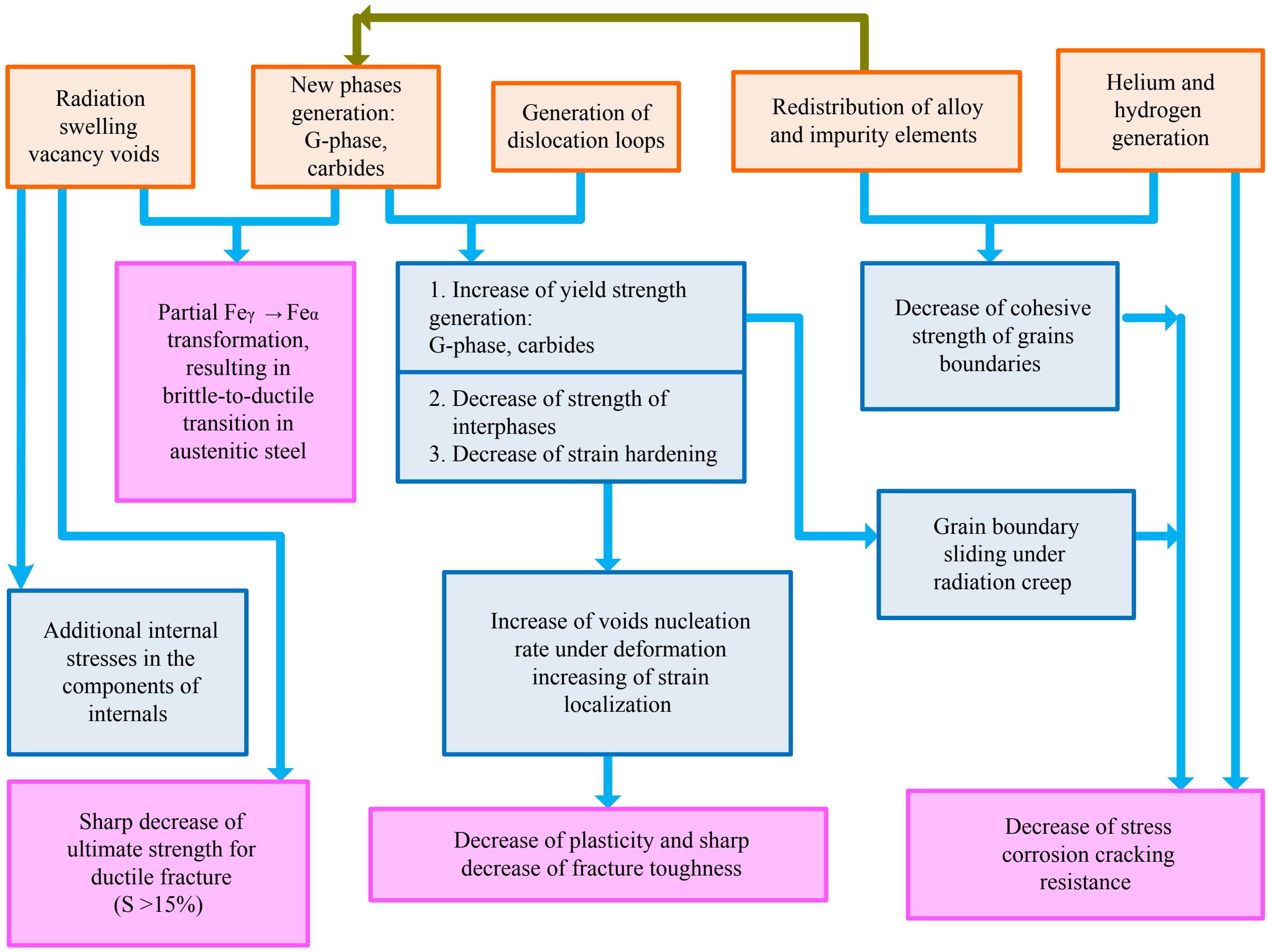

FIG. 18. Effects of neutron irradiation in austenitic steels.

(b) Mechanical properties:
- — Increased tensile properties (e.g. yield and ultimate tensile strengths of up to 1000 MPa);
- — Decreased uniform and total elongation (e.g. <1% uniform elongation);
- — Increased hardness;
- — Decreased fracture toughness (e.g. down to ~45 MPa·m$^{1/2}$).

(c) Chemical composition:
- — Radiation induced segregation at grain boundaries (mainly chromium, molybdenum and iron depletion, and nickel and silicon enrichment).

(d) Others:
- — Swelling as a result of cavity formation (at very high levels of neutron fluence);
- — Radiation induced creep leading to stress relaxation.

4.2. FATIGUE

Fatigue is the structural deterioration that can occur as a result of repeated stress-strain cycles caused by fluctuating mechanical or thermal loads. The first time the consequences of metal fatigue were understood was in the 1950s when fatigue failures caused multiple crashes of the world's first jetliner: de Havilland's Comet aircraft.

It has been determined that under repeated cyclic loading, if sufficient localized microstructural damage has been accumulated, crack initiation occurs in the most highly stressed locations. For metallic materials, a key factor in recognizing fatigue damage is the existence and the shifting of dislocations.

In a nuclear power plant, normal operation is the condition in which fatigue mechanisms more frequently manifest themselves, which can affect a variety of SSCs. Fatigue damage can be divided into two recognizable stages. In the first stage, fatigue causes local accumulation of irreversible plastic deformation with microcrack initiations. In a second stage, fatigue causes cyclic crack growth that continues until the crack reaches a critical size, beyond which the structure may suddenly collapse at any given time. With respect to the size of the deformation and to the number of cycles, fatigue conditions are phenomenologically divided into high cycle fatigue (typically producing low deformations) and low cycle fatigue (typically causing large deformations). When a very high number of cycles ($>10^9$) is involved, fatigue is referred to as 'vibration fatigue'.

Fatigue can be caused by alternating mechanical loading, alternating thermal loading (thermal transients) or a combination of both. When cyclic thermal loading is predominantly involved, fatigue is sometimes referred to as thermal fatigue. The mathematical characterization of irregular alternating loads in repeated and reversing stress cycles is normally conducted by means of the conversion of such irregular time series into a sequence of regular 'curved cycles' of stress versus time. This type of equivalent representation allows the development of predictive methodologies for high cycling and low cycling fatigue conditions in metals, including safe thresholds and material lifetime evaluations summarized by the development of usage factor criteria. Fatigue is one of the phenomena often considered in TLAA [21, 22].

4.2.1. Mechanical fatigue

Mechanical fatigue occurs often in locations of a structure or component affected by imperfections and features inducing high stress concentrations. These may be the result of technological manipulation (e.g. impact loads during fabrication) or they may be owing to the presence of geometrical discontinuities. Locations with structural defects, discontinuities and stress concentration may show early signs of localized deterioration of mechanical properties.

In the first stage of a fatigue evaluation, when physical signs have not yet developed, it is important to know where to focus attention. The key parameters for determining the locations most likely to develop fatigue damage are those with:

— Local excursions of stress tensor components in time;
— Reduced tensile properties of the components or materials under consideration;
— Environmental characteristics and residual stresses from fabrication and other pre-existing conditions.

In the second stage of a fatigue prone location in a component, crack growth occurs. At this point, it is important to characterize the damage to confirm a fatigue diagnosis. There are physical signs of the presence of fatigue damage that are clearly recognizable. It is necessary to measure and report the shape and dimensions of the cracks, the mechanical parameters characterizing the crack growth and the fracture toughness of the material.

Predictions about the evolution of fatigue damage are supported by well developed mathematical methods, often found in national codes and standards. The analyst needs to be given access to all material properties and to the loading history, the sequence of operational stress regimes or the conditions. The analysis methods take into account the necessary level of conservatism in order to correctly reflect the unavoidable uncertainty of the input data (number of cycles, stress level, etc.).

If a structure or component is designed to a known load spectrum, characterized by measurable representative parameters (e.g. temperature, pressure or flow rate), monitoring of such parameters is simple and necessary. Instrumentation to that effect should be provided and the SSC operating history should be recorded. Any deviations from the design assumptions should be recorded in the ageing management documentation and in the supporting PLiM databases (if available). In safety significant SSCs, deviations beyond a certain threshold need to be reported to the regulatory body.

Monitoring of fatigue damage, regardless of its development stage, is always a requirement. Where it is not possible to codify the operating spectra with known parameters and where the actual operating regime is random (e.g. vibration cases or acoustic resonances), it becomes necessary to provide continuous measurement of local critical operating parameters. In most cases, the local temperature distribution can be monitored; in more complex circumstances, it may be necessary to plan the periodic monitoring of deformations. If possible, a more advanced technology, such as direct on-line monitoring of the manifestations of these mechanisms, should be adopted.

Most fatigue tests are performed in air, while SSCs are under operating conditions in close contact with the coolant. In this case, design fatigue curves need to be adjusted to take into account the effects of the liquid coolant environment.

4.2.2. Thermal fatigue

Thermal fatigue is a major stressor in surge lines, spray lines and associated nozzles, and in lines with mixing tees. These components are subject to various stressors, such as thermal stratification, thermal shock, turbulent flow penetration and thermal cycling during normal operation, and operational transients such as reactor power manoeuvring and plant startup/shutdown. The original fatigue analysis of such components in older operating reactors covers only design basis thermal transients, not phenomena such as thermal stratification, typically found in surge and spray lines, and thermal cycling affecting mixing tees, among others. These are phenomena that have only recently been understood, and their effects are considered only in newer designs and whenever such systems and components are being replaced in older reactors.

4.2.3. Vibration fatigue

Vibratory fatigue is another type of fatigue of a mechanical nature that occurs in piping systems. Inadequate piping supports can induce fatigue in piping or tubing. The type and distance of piping supports greatly influence the natural frequency of the piping system. Tightly spaced restraints increase the natural frequency. Widely spaced restraints reduce the piping system's natural frequency. If an excitation source exists in the natural frequency range of the piping system, resonance may occur and high frequency fatigue damage may result. Failures have occurred at weldolets[2] and socket welds in small diameter pipe lines. They result from high cycle mechanical fatigue combined with low amplitude cyclic stress.

There are three main contributing factors to vibration fatigue: mechanical excitation mechanisms, cavitation and flashing.

An example of the mechanical excitation mechanism is the pressure pulsations in centrifugal pumps occurring at frequencies that are multiples of the vane passing frequency. They occur in the range of acoustic frequencies and are continuous and propagate through the coolant medium just as sound is transmitted through air. If pressure pulsations happen to coincide with a structural frequency of the piping system, severe vibratory fatigue damage may take place in the system.

The cavitation excitation mechanism occurs when the fluid pressure approaches its vapour pressure. Under these conditions, small fluctuations may induce the formation of vapour pockets, which rapidly collapse, generating intense shock waves. The resulting pressure pulsation can cause severe piping vibration downstream of the component, as may happen in the piping downstream of a multistage high energy orifice. Vibration measurements in the vicinity of control valves and high energy multistage orifices have revealed high frequency, broadband vibrations, with accelerations several hundred times the acceleration owing to gravity and with velocities at some locations as high as several metres per second. In addition to severe vibrations, the collapse of the cavities and the resulting impact on a solid surface removes material by mechanical erosion, damaging piping and adjacent components. One of the main differences between cavitation and mechanical excitation is the frequency content (the broadband) of the excitation.

Flashing is the third vibratory excitation mechanism, which occurs when the coolant temperature is higher than its saturation temperature at a given pressure and the coolant flashes into steam. This results in broadband pressure pulsations, causing vibration of the piping downstream of the flashing component and results in steam hammer phenomena. This is the reason that control valves regulating the flow of oversaturated coolant into a vessel kept at lower pressure are located as close as possible to the vessel entry nozzle, in order that flashing may occur in the larger volume of the vessel and not in the piping. A somewhat opposite phenomenon to flashing, the collapse of steam bubbles, may also cause water hammer, vibratory fatigue and eventual piping failures.

Acoustic modelling has only been introduced relatively recently. The onus rests, therefore, with older nuclear power plant operators to adequately monitor vibrations of this nature during plant operation and measure their

[2] Registered trade mark indicating a butt weld branch connection, designed to minimize stress concentrations and provide integral reinforcement.

characterizing parameters, such as velocity and displacements. These should then be fed back to the engineering department for adequate resolution or acceptable mitigation.

4.2.4. Fatigue in geometrical discontinuities

Geometrical discontinuities, such as those found in welds between thin piping runs and thick valve bodies or between vessel nozzles and piping, inherently induce stress concentrations in the weld geometry, which result in fatigue effects similar to those produced by cracks. The thickness change in the weld cross-sections intensifies the stress field in a localized area. The stress amplitude then decays rapidly to nominal values away from the discontinuity. In addition to geometrical discontinuities, imperfections and cracking, such as weld discontinuities in the weld metal or in the heat affected zone, also induce stress concentrations that drastically reduce the weld joint fatigue strength.

There can also be both geometrical and material discontinuities, such as welds between stainless steel piping and low alloy steel vessel nozzles, or stainless steel piping to steam generator nozzle welds or to pressurizer nozzle welds. These welds are called ‘dissimilar’ or ‘heterogeneous’ welds. The dominant degradation mechanism in such cases is carbon migration. This involves the decomposition of carbides, followed by carbon diffusion from the base steel into the butter material. The driving force in such migration is the carbon content gradient between the ferritic steel of the base metal and the stainless steel of the weld metal.

4.2.5. Environmental fatigue

In the last two decades, many international research institutions have actively studied environmental fatigue in LWRs, because it has been found that the LWR environment significantly accelerates the fatigue life of materials. Experimental data showed that fatigue life in an LWR environment was affected not only by the strain amplitude and medium, but also by the material type, strain rate, dissolved oxygen, temperature distribution and sulphur content, among others.

The ASME code [23] and other codes, as well as standards and guidelines, have adopted an environmental fatigue multiplier F_{en} to obtain a fatigue usage factor to account for the associated environment: $F_{en} = \frac{N_{air}}{N_{water}}$, where N_{air} is the number of cycles in air, at ambient temperature, and N_{water} is the number of cycles in light water, at ambient temperature. The ASME fatigue design curve takes into account the influence of the environment on fatigue by applying a safety factor of 2 for the strain amplitude and of 20 for the cycles to failure.

However, in the past two decades, experimental research has reported values below those shown in the ASME design curve, raising the issue of whether the ASME fatigue design curve should still be considered as valid.

Systematic and comprehensive fatigue monitoring systems have been developed by commercial companies for various structural materials in LWR environments (BWRs and PWRs), taking into account the effects of dissolved oxygen, dissolved hydrogen, strain amplitude, frequency (strain rate) and temperature, as well as the combined effects of all such factors. Similar experimental activities were also conducted in other countries to re-examine the effects of LWR environments on fatigue life. Some LWR materials, such as carbon steel in BWRs and stainless steel in PWRs displayed significantly higher values of F_{en} at lower strain rates, which meant that the operating life of the component subject to fatigue in a water environment could be reduced significantly, as compared with that in air.

Many theoretical calculations for various structures and components subject to fatigue during operation have been conducted, resulting in rather high usage factors. However, experimental measurements of actual strain amplitudes showed that most of these high usage factors could be reduced to acceptable values in several components. In terms of AMPs designed to take into account future LTO, there have been attempts to introduce this approach of fatigue analysis, guided by experimental data, into the flaw evaluation procedure for locations with fatigue usage factors close to 1 and above, where crack initiation and short crack growth characteristics were considered probable during the component lifetime.

4.3. GENERAL CORROSION

Corrosion is a form of material degradation caused by a chemical or electrochemical reaction with the environment. Corrosion results in a detectable change in the material surface aspect and characteristics, which can lead to a reduction or loss of function of a component or subcomponent. The severity of a corrosion attack is associated with the concentration of corrosive agents present in the environment or in the medium flowing through and reacting with the material of a given component.

There are many forms of corrosion associated with the material and the environment. They can be categorized into three major groups:

- Corrosion without mechanical loads (e.g. uniform corrosion, local corrosion attack or selective corrosion);
- Corrosion combined with mechanical loads (e.g. SCC, hydrogen cracking, irradiation assisted corrosion, corrosion cracking owing to deformation or corrosion fatigue);
- Flow assisted corrosion (e.g. erosion corrosion or FAC).

All forms of corrosion are characterized by loss of material and deterioration of mechanical properties. Corrosion reduces the wall thickness of affected structures and components, either locally or generally throughout the material on a larger scale.

When corrosion affects the material throughout the surface, it is called general corrosion. General corrosion of steel produces a slow generalized thinning and uniformly distributed loss of material, which involves a wide area without any appreciable localized attack. General corrosion degradation begins with the formation of a surface layer of loose material that increases in depth as long as the reactants are able to diffuse through the layer and sustain the reaction. In the absence of mechanical loading, only uniform deterioration on the material surface occurs. Its progression rate depends on several factors including oxygen content, temperature and flow rate of the medium, among others. General corrosion does not change the material microstructure underneath the surface layer.

Local corrosion has different characteristics. It usually appears in the form of localized pitting or small holes through the protective surface layer caused by the local concentration of certain chemical agents, such as oxygen, chlorides or ammonia, among others. For example, pitting often occurs on steam generator tubes in their attachment area.

Selective corrosion appears in certain areas, such as the area adjacent to the grain boundary, or it may affect only certain specific elements of an alloy. A typical example of selective corrosion is intergranular oxidation in some austenitic steels and some other nickel and iron based alloys. Intergranular oxidation is typically triggered by the combination of an aggressive environment and a sensitive material, such as stabilized austenitic steel. Austenitic steels may be sensitized by heat treatment or welding. Among the chemical agents that may cause intergranular corrosion are oxygen, chlorides, sulphates and hydroxides. If selective corrosion continues to a certain depth, even if no appreciable change of geometry occurs, loss of material and wall thickness reduction may lead to the weakening of the loading capacity of the structure or component.

Corrosion can be caused by either a chemical or an electrochemical reaction between the material and the environment. Electrochemical processes require an electrolyte. Normally, electrolytes form mainly during maintenance activities through leaks from flanges, pools or tanks, or in the presence of condensation phenomena or other humidity sources, coupled with the presence of oxygen. These conditions are conducive to the formation of acidic or caustic solutions, giving rise, when in contact with steel, to galvanic and stray currents that displace metallic ions.

Galvanic corrosion is an accelerated type of corrosion, occurring when a metal is electrically coupled with a more noble metal with a different surface potential and when they both have a common electrical path, for example, when they are both sharing the same electrolyte. A flow of electrons is generated by the difference of potential between the two metals, moving from the more active anode (less noble) to the less active cathode (more noble). Weld and base metal in dissimilar butt weld joints can sustain physical damage when the dissimilar metals are in contact with an electrolytic solution, such as in areas near vent line bellows, in flanged hatches in BWR wet wells, in penetration attachments to the containment wall or in pipe supports, ducts, grounding wire, among others.

Stray direct current (DC) is also known to cause accelerated corrosion. If a direct current flow (or part of it) strays and leaves its normal path, it may flow through the soil and into an electrolytic solution, as is often observed in underground piping or in other underground structures. Corrosion occurs at the point where the stray current leaves the metal structure and enters the electrolyte. Stray currents are generated predominantly in the cathode of cathodic protection systems, in high voltage direct current systems and in direct current welding operations.

Figure 19 shows the process used by Korea Hydro & Nuclear Power in the Republic of Korea to evaluate the extent of general and localized corrosion in SSCs operating in humid environments and in those containing corrosive fluids. It is the operator's responsibility to identify the form of corrosion that can occur. Evaluation of corrosion is based on an estimate of the corrosion rate. If only localized corrosion is found, the corrosion rate evaluation follows a different path to that of general corrosion cases. If the corrosion rate estimate is not acceptable, a second tier investigation is conducted on the component, which involves direct thickness measurements. NDE may be performed to obtain the measurements. If the remaining thickness is found to be unacceptable by code requirements, then mitigation of the condition should be planned via AMPs.

4.4. STRESS CORROSION CRACKING

SCC is a complex phenomenon driven by the synergistic interaction of mechanical, electrochemical and metallurgical factors. Stainless steels are particularly susceptible to SCC in the presence of chlorides and in neutral pH demineralized water containing oxygen, such as in the recirculating coolant, particularly in a BWR, although PWR components can also suffer from SCC.

A metal surface affected by SCC has a brittle appearance. SCC usually propagates perpendicular to the principal tensile stress. Cracks can vary in the degree of branching and propagate with little or no macroscopic plastic deformation. An alloy affected by SCC does not usually display abnormal mechanical properties (yield strength and tensile strength), although this may be observed in certain classes of alloys, such as precipitation hardened stainless steels or under certain environmental conditions as in radiation fields.

SCC does not occur in all environments. Conversely, an environment that induces SCC in one alloy above a certain temperature does not necessarily induce SCC in another alloy. Nickel–chromium–iron alloys can exhibit IGSCC at temperatures greater than 250°C when exposed to high purity water. Among the nickel alloys in weld metals, Alloy 52 is the only nickel based weld metal considered to be resistant to SCC, because no failures have occurred in service to date. On the other hand, other nickel alloys, such as Alloy 600, need to be evaluated for SCC resistance on a case by case basis.

SCC progression is usually divided into an initiation phase and a propagation phase. SCC initiates from sites with localized pitting or crevice corrosion. The initiation time can vary significantly and can take up to several decades. The propagation phase is itself often subdivided into two subphases: a slow propagation phase and a fast propagation phase. The latter is usually characterized by crack tip stress intensities, K_I, exceeding a characteristic apparent threshold value in pre-cracked type specimens, known as K_{ISCC} in fracture mechanics. Perhaps the most interesting fact concerning SCC is that three preconditions are necessary and need to be present simultaneously for SCC to take hold. The elimination of any one of these preconditions or the reduction of one of the three below some threshold level can, in principle, prevent the phenomenon from occurring. The three necessary preconditions for SCC to occur are given below and are shown in Fig. 20:

— A susceptible material;
— A tensile stress component;
— An aqueous environment.

Stainless steel pipes can suffer from SCC. Failure analyses have revealed that welding related thermal sensitization of non-stabilized stainless steels is the cause. Cracking can also occur as a result of severe cold work present in the metal and, in other cases, to moderate cold work in combination with crevice conditions, formed by excess penetration and shrinkage in the root areas of the welds.

At the grain level, two types of SCC can be recognized:

— SCC with transgranular (through the grains) stress corrosion cracking (TGSCC) morphology;
— SCC with intergranular (along the grain boundaries) IGSCC morphology.

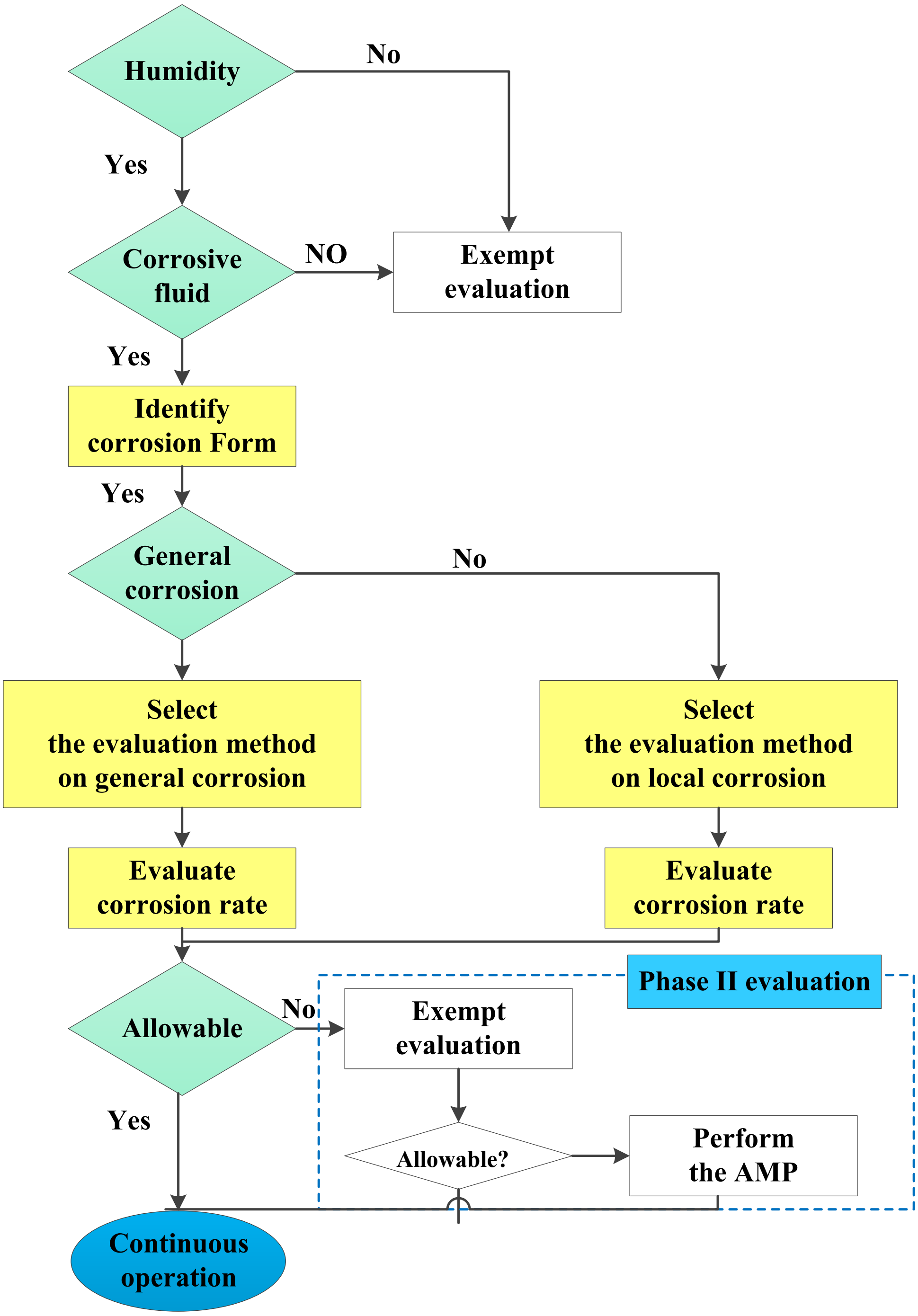

FIG. 19. Evaluation of general and localized corrosion by Korea Hydro & Nuclear Power (Republic of Korea).

Sometimes, the SCC types are mixed or the mode switches from IGSCC to TGSCC. Both can occur in the same alloy, depending on the environment, the microstructure or the stress-strain state.

4.4.1. Transgranular stress corrosion cracking

An SCC issue common to both types of LWRs is TGSCC of austenitic stainless steels. It occurs primarily owing to chloride contamination, although other halide anions, such as fluorides, can also cause TGSCC. The problem initiates on the outside surfaces of austenitic stainless steel components, mainly owing to a lack of adequate cleanliness. Wetting due to condensation or nearby water leaks can allow an aqueous environment to form and lead to TGSCC accompanied by pitting or crevice corrosion. The stress required for chloride induced TGSCC

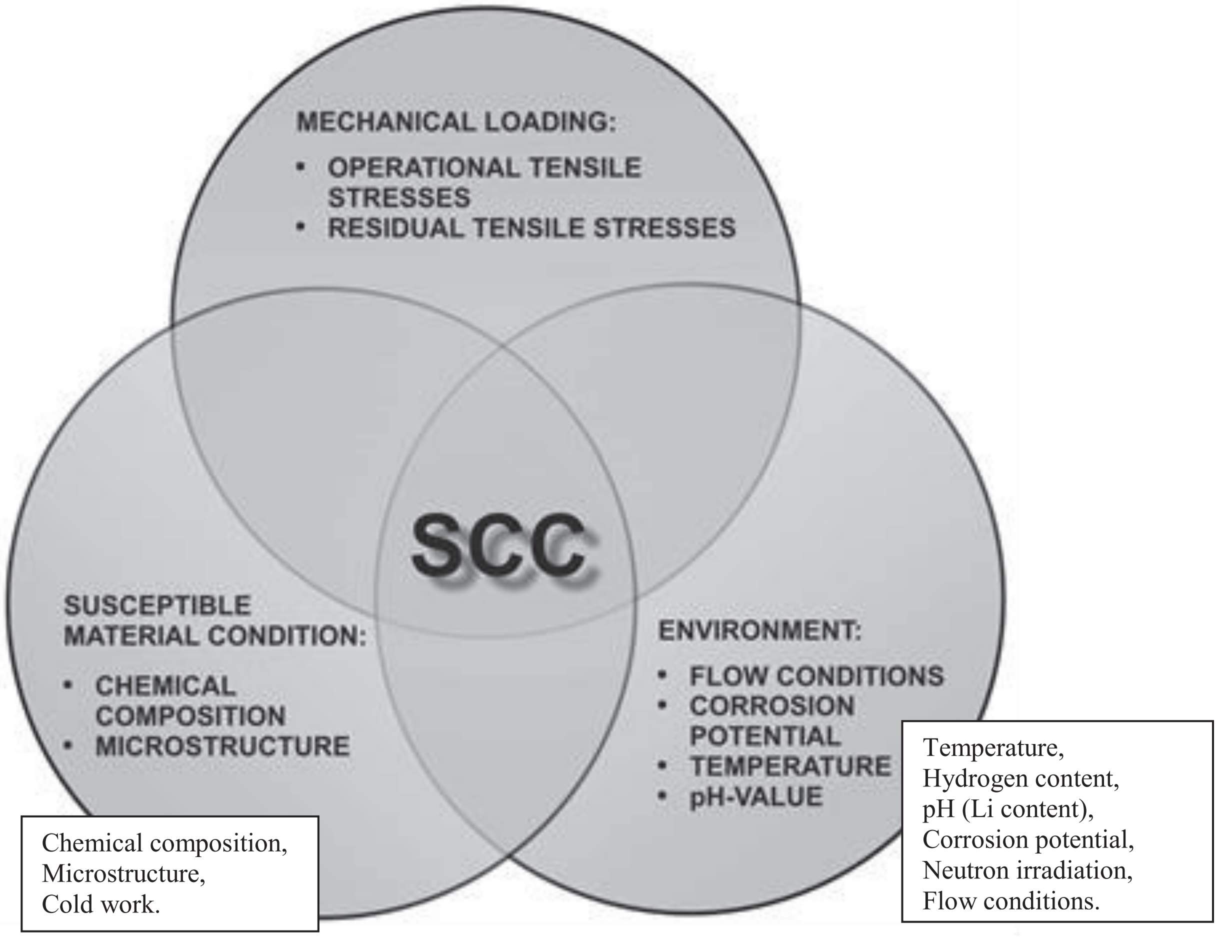

FIG. 20. SCC factors.

is relatively modest, the threshold being close to the proportional limit of solution annealed austenitic stainless steels. Implementation of known procedures that ensure adequate surface cleanliness is a continuing necessity that requires careful management attention at all stages of construction and operation of nuclear power plants.

One issue having an impact on the risks of chloride induced TGSCC of austenitic stainless steels is the choice and specification of thermal insulation materials. Fibreglass thermal insulation has been used in older plants. With its large concentrations of soluble silicate, it has a favourable buffering action in the presence of chloride contamination of austenitic stainless steel. The allowable limits for surface chloride contamination in combination with the soluble silicate content of thermal insulation are captured by the Karnes diagram shown in Fig. 21, taken from Ref. [24] (now updated as Ref. [25]).

In newer plants, many States are replacing fibreglass insulation with mineral wool insulation because of concerns about clogging of sump pump strainers, caused by debris from fibre glass shredding in the reactor building, during major loss of coolant accidents (LOCAs). Mineral wool insulation is less prone to clogging strainers, but has much less soluble silicates and is therefore less tolerant of surface chloride contamination, with obvious consequences on the management of surface cleanliness.

Chloride induced TGSCC can also occur in internal surfaces, generally in dead legs and stagnant regions due to the high probability of the simultaneous presence of chloride contamination and oxygen. One location that has been rather frequently affected by TGSCC in PWRs is at the canopy seals that ensure the pressure boundary of threaded connections in the control rod drive (CRD) housings that are located above the RPV upper head. Leaks from the canopy seals have caused serious boric acid corrosion of the upper head low alloy steel.

The root cause of TGSCC at the canopy seals is inadequate deoxygenation procedures after resealing of the primary circuit in an outage. Removing air bubbles from such locations is not a simple matter, given the complex pathway to the reactor vessel. Procedures during plant startup for eliminating these air pockets may vary, but a

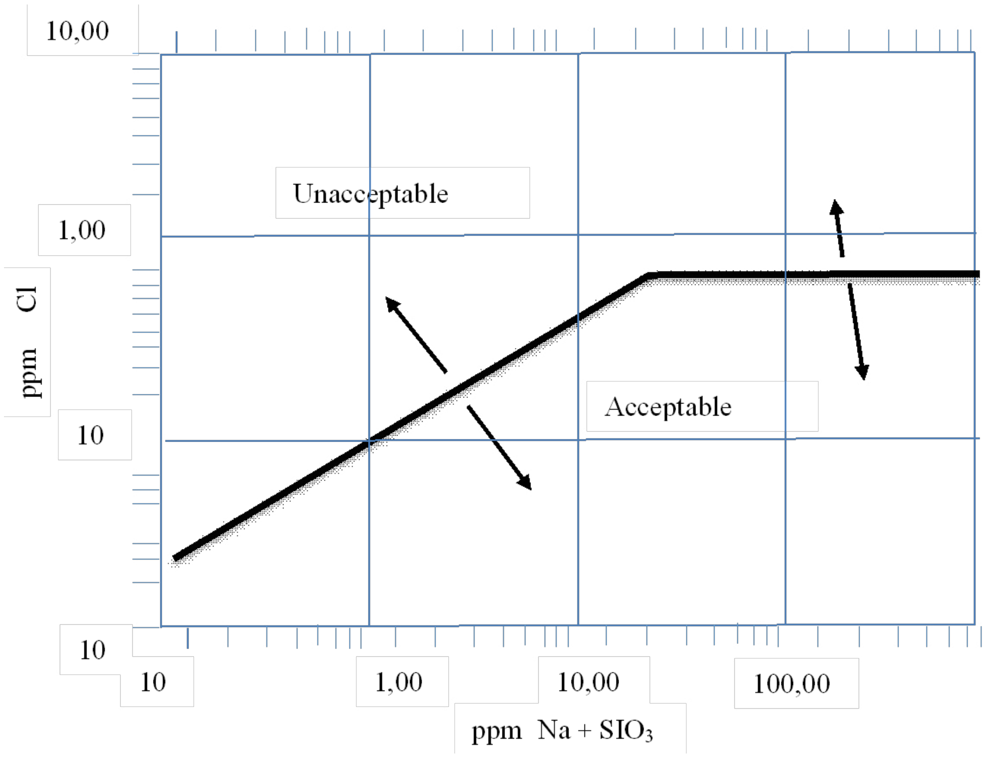

FIG. 21. Karnes diagram showing acceptable and unacceptable areas for chloride and soluble silicate in insulation material [24]. ppm — parts per million.

recognized reliable method consists of completing the final fill of the primary circuit after a vacuum pump has been connected to a penetration in the upper head.

4.4.2. Primary water stress corrosion cracking

Primary water stress corrosion cracking (PWSCC) refers to SCC specific to the primary water environment of PWRs. The term indicates the degradation that certain alloys undergo in contact with the reactor primary coolant. Another particularity is that PWSCC requires the simultaneous presence of high tensile stresses, of a conducive environment (i.e. high temperature water) and a susceptible microstructure. Inconel 600/182 and 82 are the materials more susceptible to this type of degradation among those that are used in the primary and interconnected systems in PWRs. There is no general consensus on the mechanism responsible for PWSCC, but two dominant hypotheses have been proposed:

— Hydrogen embrittlement, a mechanism involving hydrogen diffusion and its concentration at crack initiation tips;
— Dissolution/oxidation at the crack tip with dealloying of chromium and its surface diffusion.

It was found that Alloys 690/152/52 are more resistant to PWSCC than Alloys 600/182/82, possibly owing to their higher chromium content and to the protective influence of the carbides $Cr_{23}C_6$.

PWSCC of Alloy 600 was reported for the first time in the steam generator tubes at Obrigheim nuclear power plant in the Federal Republic of Germany in 1971. Similar degradation was observed worldwide and continued to grow until the tubes were replaced with thermally treated Alloy 690.

Significantly damaged areas included some of the U-bends of the tight row, roll transitions and tube support regions that had a high tensile stress due to secondary side induced tube denting. Most of the cracks were in the hot leg side, but there were also some cases of cold leg cracking. The cracks were mostly axial, but some plants showed circumferential cracks.

Another area of concern in steam generators was the weld between the stub and the divider plate in the channel head. The stubs and divider plates in most steam generators were made from forged plates of Alloy 600, which had final thermal treatment at the end of the forging sequence. The weld between the stubs and the divider plate was made of Inconel 600/182. It was the last weld made during the steam generator fabrication. It was conducted manually and was not followed by any thermal treatment.

Damage has also been reported to Inconel 600 material outside steam generators. In the USA in 1989, leakage was reported in 20 pressurizer heaters and instrument penetrations at the San Onofre Nuclear Generating Station, and in the pressurizer thermal sleeves and instrument nozzles at Calvert Cliffs nuclear power plant.

PWSCC has never been observed in Alloy 800 steam generator tubing or in replaced steam generators with Alloy 690 tubing. In contrast to the IGSCC problems experienced with stainless steels in BWR systems, the same materials used in PWR systems have suffered from relatively fewer problems, and those that have occurred have mainly been attributed to the presence of oxygen trapped in stagnant regions combined with thermal sensitization or pre-existing cold work [26].

4.4.3. Irradiation assisted stress corrosion cracking

IASCC has been extensively studied in research and test reactors and evaluated in actual plants. IGSCC and IASCC have significantly affected some components made of austenitic stainless steel or nickel based alloys in BWRs, namely the recirculation piping, core internals and some parts of the RPV, such as the in-core monitor housing and the CRD stub tubes.

Figure 22 shows a process developed by the Tokyo Electric Power Company in Japan to evaluate IASCC.

This systematic approach to an IASCC assessment begins with a selection of highly irradiated RPV internals in BWRs. Understanding of the phenomena is normally acquired through an analysis of the information collected and the characterization of surveillance specimen exposed in power reactors. Other sources of information are the results of R&D projects that make use of research reactors. A characterization step is followed by the establishment or the adoption of recommended acceptability criteria, such as flaw size, supported by an evaluation of the IASCC growth rate, residual stresses, fracture toughness and the harshness of the environment. ISI is then used at appropriate intervals to evaluate the SSC condition with respect to the flaw size criteria established in the previous step.

This work also allows an estimate of the remaining life of the reactor internals by interpreting the difference between the test reactor surveillance specimen data and the actual conditions in the reactor. Guidelines have been developed in Japan to directly benchmark the RPV internals in the whole reactor fleet. The IAEA guidelines (see Ref. [20]) also describe the procedure for evaluation of the effects of IASCC on the remaining lifetime of WWER reactor internals. The susceptibility of IASCC attack under various neutron fluences and irradiation doses is shown in Fig. 23.

For BWRs, two threshold values for the onset of IASCC have been reported, depending on the stress levels of the component. For components with high tensile stresses, the threshold is ~5×10^{24} m^{-2} (E>1 MeV), and for those with lower tensile stresses, the threshold is ~2×10^{25} m^2. Under BWR conditions, stress relaxation by irradiation creep can be expected at welds of near core components. Many experiments (including post-irradiated examination) have been carried out worldwide to acquire a better understanding and to establish a database of IASCC in austenitic stainless steels. In 2007, two correlations between IASCC and applied stresses were established from experimental results in simulated PWR primary coolant using irradiated samples removed from several operating plants. Although there are some differences in the threshold stresses reported, the threshold stress for IASCC initiation clearly decreases with increasing fluences. Highly irradiated 316 CW stainless steel at a neutron fluence of more than ~30 dpa showed IASCC susceptibility above relatively low stresses between ~0.4 Rp 0.2 and 0.6 Rp 0.2 (where Rp 0.2 is the as-irradiated yield strength of typically ~1000 MPa at such high fluence) [26].

4.5. FLOW ACCELERATED CORROSION

FAC, or erosion corrosion, is a chemical corrosion mechanism affecting metals such as carbon steels. Without flow and under static conditions, carbon steel owes its corrosion resistance to the formation of oxide layers. However, if a moving fluid is in contact with its surface, as in piping systems, the oxide layer protecting

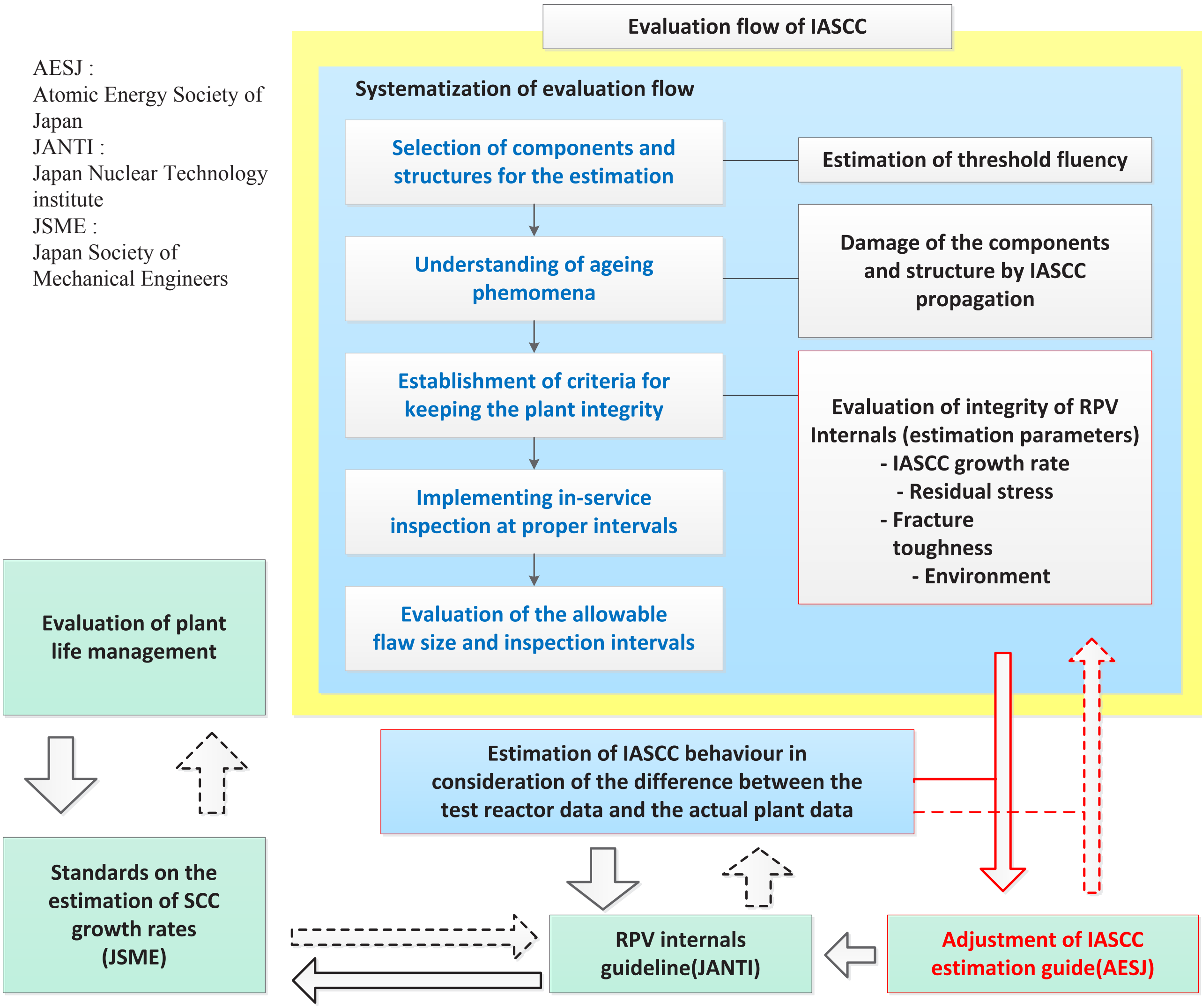

FIG. 22. IASCC evaluation scheme for RPV internals developed by the Tokyo Electric Power Company, Japan.

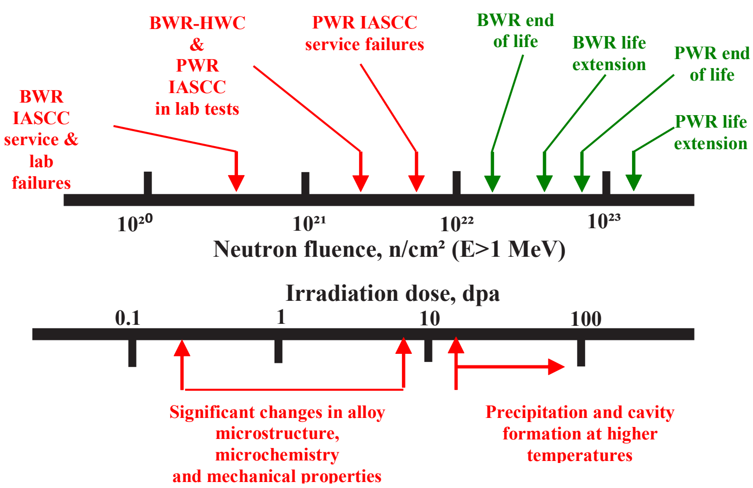

FIG. 23. IASCC susceptibility in relation to neutron fluence and irradiation dose. dpa — displacements per atom; HWC — hydrogen water chemistry.

the pipe internal surface is disturbed by the rubbing action of the moving fluid. In fact, an alternating process manifests itself in the continuing formation and dissolution of the protective surface oxide layer. This process is the mechanism responsible for wall thinning, which may ultimately cause perforation of the pipe wall if allowed to continue indefinitely. In addition, FAC may be aggravated by the impingement of liquid droplets and by the presence of cavitation.

FAC manifests itself in a generalized fashion. The attack is widespread and thinning occurs throughout the affected region, rather than in a localized area as in the case of pitting or cracking. The wall thinning rate is proportional to the:

— Solubility of the oxide layer;
— Material;
— Content of chromium, molybdenum and copper;
— Temperature;
— Flow velocity;
— Surface roughness;
— Pressure profile and flow.

FAC has been experienced worldwide and can occur in all types of nuclear and fossil fuel power plants. Although the FAC mechanism has been studied for a long time, it remains one of the major safety issues in terms of the potential for unexpected failures, personal injury and even loss of life. FAC may also affect plant availability.

FAC has caused deterioration to internal parts in steam generators such as tube supports and moisture separators and in balance of plant components, particularly piping in several plants in France, Japan and the USA. The Electric Power Research Institute (EPRI) has issued guidance documents dealing with FAC, notably:

— Flow-accelerated Corrosion in Power Plants [27];
— Pressurized Water Reactor Secondary Water Chemistry Guidelines [28].

A successful FAC programme should focus on reducing the FAC wear rate by improving the materials, water chemistry and design where possible. An example of an effective erosion corrosion or FAC evaluation process is shown in Fig. 24, courtesy of KHNP.

If the flow velocity and oxygen content in the water are beyond or below certain allowable values, the component is either exempt from evaluation or an erosion corrosion rate evaluation is carried out. If conditions warrant it, an AMP is applied and operation is allowed to continue under careful monitoring. The general approach to a technical evaluation of FAC/erosion corrosion in the context of an AMP is well described in an EPRI document (see Ref. [27]).

4.6. THERMAL AGEING

Thermal ageing is a non-reversible degradation mechanism that is heavily dependent on temperature, material microstructure and time. At high temperatures, the material may lose ductility and become brittle because of microstructural changes in the form of precipitates, whereby certain elements separate from the metal matrix. In the RPV steel, which contains appreciable copper impurities, most precipitates are copper rich. Thermal ageing depends on the time and temperature of exposure, together with the material type and its chemical composition. It occurs even without the presence of radiation related acceleration of element precipitation in the metal microstructure. Copper in RPV steel is initially trapped in a supersaturated state. With time, at PWR operating temperatures (~290°C), it may be ejected from the metal matrix, as the alloy strives toward a more thermodynamically stable state, thus forming agglomerates of stable precipitates.

Thermal ageing has no hardening effects in RPV steels, as only changes in the transition temperature are observed, without any measurable changes in yield strength. As this degradation is mostly accompanied by the segregation of phosphorus along grain boundaries, some intergranular fracture is usually observed. The effects of long term ageing at temperatures up to 350°C on the ductile to brittle transition temperature of RPV steels have

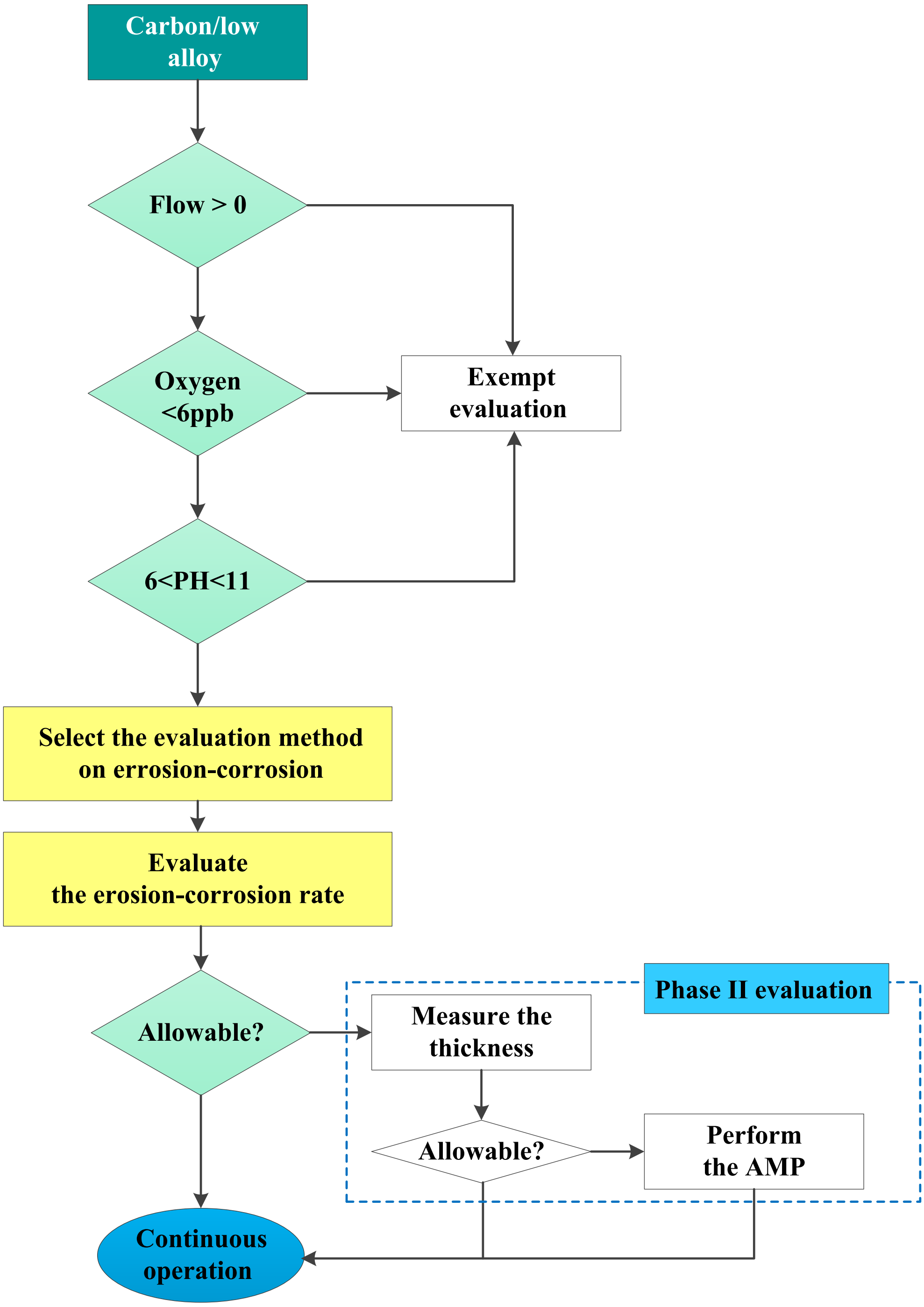

FIG. 24. FAC (erosion corrosion) evaluation at Korea Hydro & Nuclear Power, Republic of Korea. ppb — parts per billion.

been studied in several countries (France, Germany, Japan and the USA). No embrittlement has been found to occur in typical RPV steels at these temperatures for operating times as great as 100 000 hours.

The 15Kh2MFAA type steels used to fabricate most WWER-440 pressure vessels also do not appear to be susceptible to thermal ageing, even when they contain relatively high phosphorous impurity levels. Results from thermal ageing surveillance specimens located in the upper plenums of the WWER-440 pressure vessels, removed and tested after 30 operating years at about 300°C, indicate that the shift in the Charpy ductile to brittle transition temperature is small. These results are supported by Charpy ductile to brittle transition temperature measurements from RPV trepans removed from shutdown plants (Novovoronezh 1 and 2 in the Russian Federation), as well as boat samples (samples taken directly from the reactor pressure vessel) taken from operating plants.

The 15Kh2NMFA type steel used to fabricate the WWER-1000 pressure vessels is only slightly susceptible to thermal ageing at operating temperatures, owing to the high nickel and low vanadium content of this material. A shift in the ductile to brittle transition temperature of 10–30°C has been observed, due to some segregation of nickel and phosphorus at the grain boundaries, which results in an increase of susceptibility to intergranular cracking.

Figure 25 shows the thermal ageing evaluation approach for CASS developed by KHNP. CASS is a very ductile material that is resistant to SCC. It is widely used in safety related components, and is particularly suitable for the manufacture of components of complex geometries. However, complex geometries are difficult to inspect.

For static components of CASS with greater than 14% ferrite content, fracture analysis may be conducted. The input parameters such as the *J*-integral versus crack growth resistance (*J*–*R*) curves for polymeric materials with both elastic and plastic contributions can be obtained directly from load versus displacement records in the literature, without installing an on-line crack monitoring system. Once the input parameters are obtained, the crack

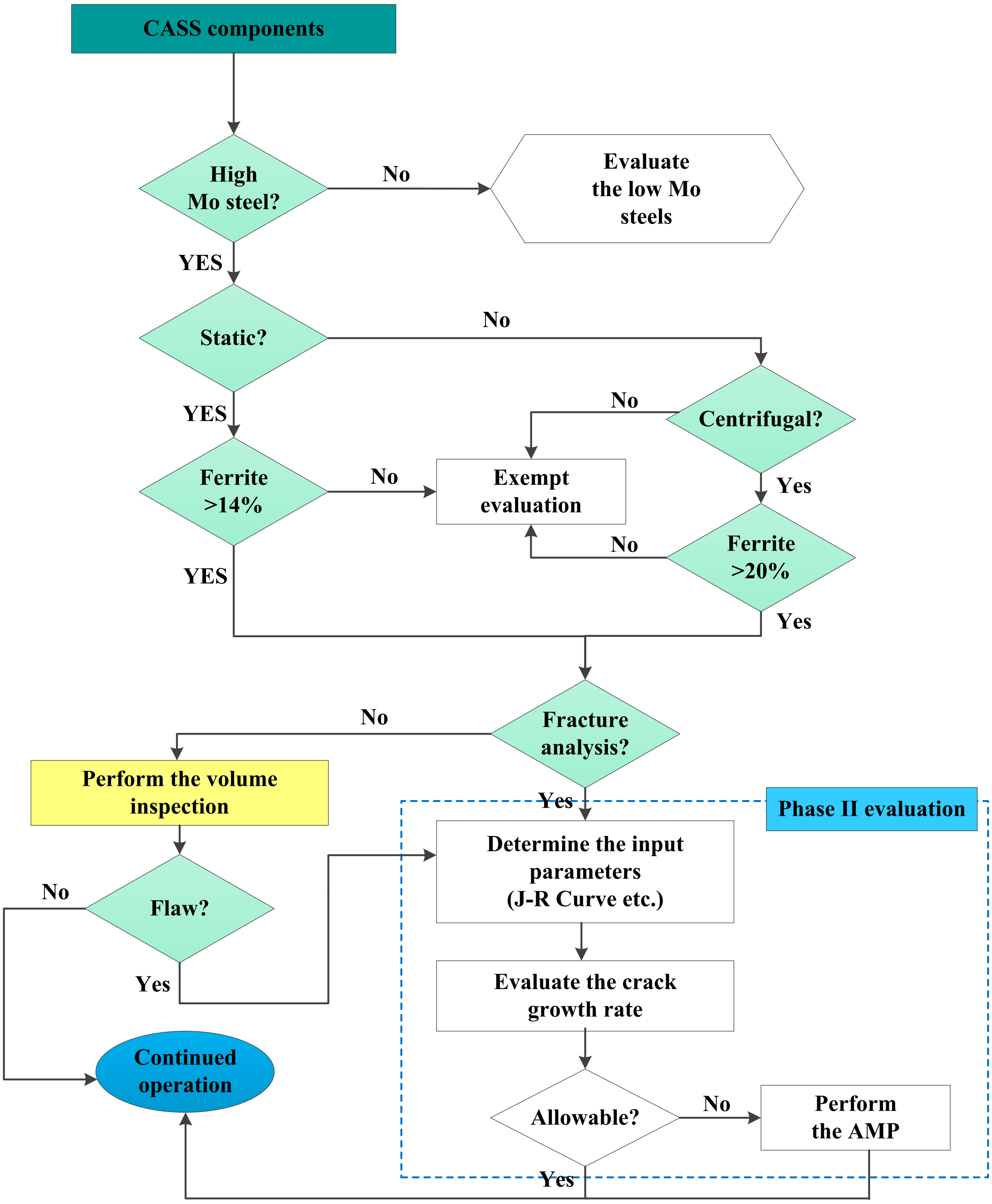

FIG. 25. Embrittlement evaluation of CASS at Korea Hydro & Nuclear Power, Republic of Korea.

growth rate can be evaluated. If it is within the allowable limit, the component is allowed to continue operation. If not, an AMP has to be developed before continuing operation.

For centrifugal machinery with casings made of CASS and with a ferrite content greater than 20%, fracture analysis can be conducted in the same manner as for static components. If the ferrite content is below 20%, the component can be exempt from evaluation. The process is well documented in the IAEA IGALL report [29].

5. DEGRADATION AND ITS MITIGATION IN STRUCTURES, SYSTEMS AND COMPONENTS

Integrated life cycle management programmes usually include periodic inspection programmes (PIPs) for components selected for preventive maintenance, water chemistry control programmes and the life cycle management of specific critical SSCs. Components under life cycle management programmes undergo health monitoring during normal operation and beyond operation during LTO. Figure 26 is a simplified diagram showing how all elements of an AMP fit together.

When integrated life cycle management programmes and PLiM techniques are adopted, maintenance interventions, mitigation activities and their prioritization can be optimized on the basis of risk and SSC criticality, with consideration given to global operating experience feedback and R&D. Prioritization can also be confirmed by means of single point of vulnerability analyses. Depending on their categorization, SSCs are assigned to a particular ageing assessment programme (time based assessment, condition assessment or life assessment). As a result, a critical spare part list can be generated, and spare part inventories can be built up to support these preventive maintenance programmes. Obsolescence should also be part of any ageing assessment programme, and its mitigation should be included in the prioritization effort. When the maintenance plan optimization effort is complete, actionable AMPs for each item can then be produced.

5.1. STEEL CONTAINMENTS

General corrosion of exposed metal surfaces, such as steel containments, begins with the formation of a thin corrosion layer, which continues to grow as long as the reactants can penetrate the layer. It is an electrochemical process that requires an electrolyte. For steel containments, the electrolyte can just be a humid environment, such as warm air with its natural aggressive constituents, oxygen and carbon dioxide, and any suspended salts, acids or caustic agents. These may be acquired from system leaks, particularly during maintenance, but also during normal operation through flanges, in BWRs through releases from the suppression chamber, the fuel pool, the refuelling operation, from groundwater intakes and a number of other sources.

The general, corrosion mechanisms affecting metal structures inside containments are as described in Section 4 and, for surfaces in contact with liquids, include:

— Atmospheric aggression;
— Aqueous corrosion;
— Galvanic corrosion;
— Stray current corrosion.

Similar to general corrosion, except for the much faster attack rates affecting steel containments, are the various forms of localized corrosion such as:

— Crevice corrosion;
— Pitting corrosion;
— Biologically and microbiologically induced corrosion.

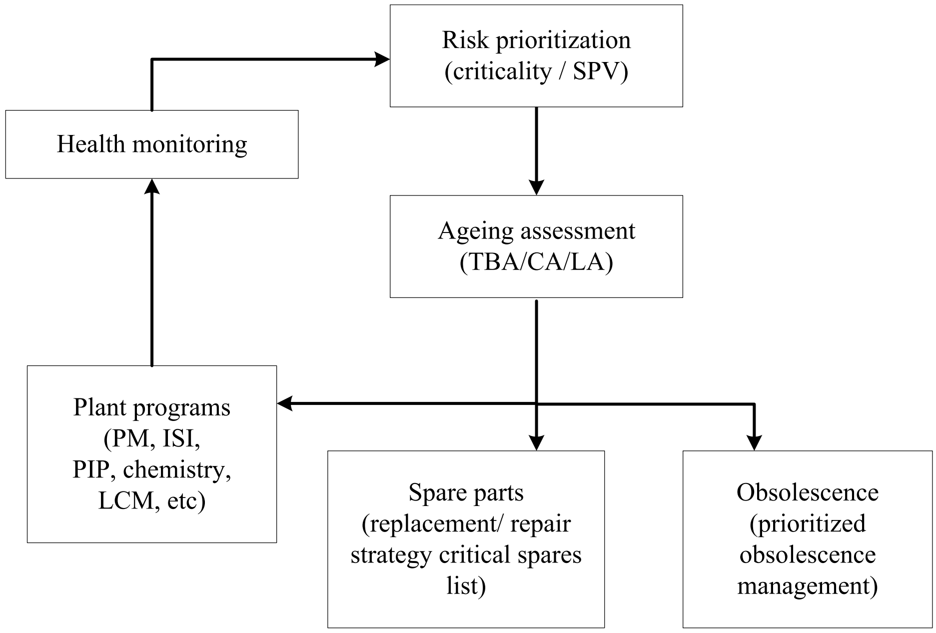

FIG. 26. Elements of a typical ageing and obsolescence programme (simplified diagram). CA — condition assessment; LA — life assessment; PIP — periodic inspection programme; PM — preventive maintenance; SPV — single point of vulnerability; TBA — time based assessment.

Flaws in the metal of steel containments can be corrosion initiation sites. Corrosion coincident with mechanical wear and/or metal fatigue can also accelerate the phenomenon. In this case, this type of corrosion is referred to as mechanically assisted or accelerated corrosion. Mechanical stressors in corrosion assisted degradation can be erosion, fretting wear, surface flaws, arc strikes, overload during ISI, repair work or missile impacts generated by equipment failure or pipe whip events. In terms of consequences, overload conditions can deform, bulge, bend, buckle or stretch the steel, and produce detrimental effects on the containment structural integrity and on its leak tightness. Mechanical loads of this kind are not expected to be frequent events affecting containment steels, as LERF assessments show. Possible exceptions are found in the following areas of BWR designs:

(a) Lubrite contact surfaces of the BWR mark I metal containment between the torus support column base plates and the base mat.
   (i) During thermal transients and pressure testing, these parts are subjected to relative motion and hence wear.
(b) Steam impingement during safety/relief valve discharge into the suppression pool.
   (i) Each discharge causes stresses in the suppression pool liner or in the torus shell at the vent/down-comer intersections, damaging and eventually removing the passive surface coating in the vent lines.

SCC and hydrogen induced cracking are phenomena not quite categorized as general corrosion, but in BWR containments, they can attack large portions of the containment envelope and hence they should be included among the general corrosion effects.

SCC in BWR containment steels is a rare event, but it can occur at relatively low tensile stresses, often well below the material yield strength. It may be caused by applied loads or residual stresses. It has appeared in the following areas:

— Heat affected zones of welds in the presence of corrosive agents;
— Stainless steel bellows;
— Electrical and piping penetrations;
— Surfaces exposed to solutions containing sulphates, hydroxides, chlorides, ammonia, fluorides, carbonates and decontamination fluids;
— The wet portions of the ABB Atom BWR design.

One of the dominant causes of cracking in BWR containments is fatigue. In all fatigue cracking, cyclic stresses are the main stressor. When an initial crack reaches a critical size and the remaining intact material can no longer sustain the load, sudden failure of the remaining cross-section occurs. Examples of cyclic stresses applied to BWR containment systems are:

— Temperature transients (e.g. startups/shutdowns, power manoeuvring, etc.);
— Piping loads at penetrations;
— Pressure testing and containment integrated leak rate testing;
— Safety relief valve discharge testing (including steam condensation loads).

Although BWR containments are designed with sufficient fatigue margins, local fatigue related issues may arise. Fatigue may affect the following examples:

— Locations with geometrical discontinuities, such as in the region connecting the cylindrical and spherical portions of the BWR mark I design;
— Locations with material discontinuities, such as in dissimilar metal welds) with different thermal expansion coefficients;
— Locations where installation misalignments occurred, such as those resulting from weld shrinkage or steel plate offsets that happened during construction;
— Locations with stress concentrations, such as in the embedded portion of the drywell base at the concrete to metal interface, where crevices can form and moisture containing corrosive substances can enter the crevice, thus initiating corrosion assisted fatigue;
— The suppression pool steel liner where the repeated condensation of steam bubbles produces pressure oscillations on the liner and on the spot welds of the anchor bolts.

Other fatigue susceptible components in BWRs are the containment penetration bellows, which are typically subject to low cycle fatigue and SCC during normal operation by virtue of their service requirements. Bellows are made of thin plies that are susceptible to indentation during fabrication. These, combined with operation related stress concentration, can reduce fatigue life.

In addition, TGSCC of bellows has occurred from high residual tensile stresses in coincidence with exposure to chlorides, sulphides or fluorides that may have accumulated during fabrication, installation or even operation.

Non-metallic materials are also subject to deterioration, which may lead to their destruction if they are not appropriately monitored and tested through AMPs. An important example of these is the non-metallic sealing material present in containment penetrations and airlock doors. Even low radiation doses and low temperatures may cause their embrittlement and failure. Leak tests should be conducted on doors and hatches at regular intervals, but also every time they are opened.

5.2. REACTOR PRESSURE VESSELS IN PRESSURIZED WATER REACTORS

5.2.1. Irradiation embrittlement in the region of the reactor pressure vessel beltline

The core beltline, located in the intermediate and lower shells directly surrounding the fuel element assemblies and includes additional wall volume both below and above the active core, is subject to irradiation related deterioration. The low alloy steels making up the beltline are subject to irradiation hardening (increases in yield strength and tensile strength) and embrittlement. These produce a shift in the transition temperature or a decrease in the upper shelf energy, which can lead to a loss of toughness and to a reduction in the fracture toughness (toughness being the ability of a material to absorb energy and plastically deform, namely elongate and reduce its cross-sectional area, without fracturing).

RPV steel toughness is significantly affected by the presence of copper, nickel and phosphorus. Vessels fabricated before 1972 contain relatively high levels of copper and phosphorous; hence, irradiation damage becomes a major consideration in the justification for their future operation. The sensitivity of welds to irradiation degradation and the degree of embrittlement is a function of chemical composition and the level of neutron exposure. In particular, the embrittlement of high copper and phosphorus and high nickel welds in RPVs plays a key role in the assessment of the significance of severe transients such as pressurized thermal shocks.

The main parameters governing the sensitivity of materials to radiation damage are:

- The flux spectrum, which may influence the degree of radiation embrittlement caused in ferritic steels;
- The microstructural characteristics, such as the grain size and the metallurgical phases, which can influence the severity of radiation damage;
- The alloy composition, which has a strong effect on radiation sensitivity, especially with elements such as nickel that increase the alloy sensitivity to radiation embrittlement with increasing fluence, lower flux levels, lower irradiation temperatures and increased manganese levels causing more damage [19];
- The presence of impurities, such as copper and phosphorus, that increase the sensitivity to radiation embrittlement;
- The presence of weld metals with high nickel content (up to 1.9 mass %) and high manganese content (over 0.8 mass %), which experience greater embrittlement than the base metal; therefore, in most cases, the weld metals remain the controlling materials for RPV embrittlement [19];
- A decrease in radiation embrittlement at temperatures above 310°C, attributed to the dynamic in situ annealing of the material (see Section 5.4.1.3).

Predictions of the evolution of degradation in the RPV can be made analytically in certain cases, which can be verified by testing the surveillance specimens that are inserted in special capsules in the RPVs and periodically withdrawn for testing. Surveillance programmes were originally designed for service lives of 40 years. Therefore, to allow surveillance to continue beyond the assumed service life, some corrective measures (e.g. additional capsules, prolongation of withdrawal schemes, etc.) need to be taken to ensure that data collection and continued testing remain possible for RPVs during LTO. As a general rule, materials with a tensile strength above 700 MPa at room temperature cannot be used for pressure boundary applications.

5.2.2. Under clad cracking in reactor pressure vessels

In the cylindrical portions of some of the first generation RPVs, fabricated from A508 Class 2 forging steel or equivalent European grades, cracks appeared in the under clad after RPV final stress relief heat treatment. In particular, post weld heat treatment of the claddings at elevated temperatures has resulted in decohesion of the grain boundaries that produced the cracks. Under clad reheat, cracks have been found to be approximately 2–3 mm in depth and can be detected during pre-service inspection through NDE, using special high angle beam transducers.

Cold cracking only occurred in the highly constrained nozzle regions when the second weld layer of the RPV cladding process was applied without preheating. The mechanism was hydrogen diffusion into the base metal during the application of the second cladding layer.

5.2.3. Primary water stress corrosion cracking in reactor pressure vessels

PWSCC requires a corrosive environment (in this case, high temperature water), a susceptible microstructure and the simultaneous presence of high tensile stresses. Alloy 600 has the structure most susceptible to this type of corrosion. PWSCC has caused cracking and leakage in reactor pressure vessel head (RPVH) penetrations and in dissimilar metal welds. There is no general consensus on the degradation mechanism of Alloy 600/182/82 in contact with primary water. Two dominant hypotheses, discussed in Section 4.4.2, are hydrogen embrittlement and oxidation with dealloying at the crack tip.

PWSCC attacks, particularly of the CRDM nozzles (see Fig. 27), are of safety concern because they infringe on the reactor coolant system (RCS) pressure boundary integrity. A CRDM break may also cause a control rod ejection accident.

CRDM penetrations in the pressure vessel head of RPVs in PWRs and BWRs are usually fabricated from forged stainless steel or rolled stainless steel bars. The ASME specifications used are SB-166 for bars and SB-167 for piping. The vessel head penetrations are shrunk fit into the vessel head openings by dipping them into liquid nitrogen and quickly inserting them into the openings. When the penetration returns to ambient temperature, a tight fit is achieved.

Vessel head penetration cracks were first found in France in 1991 at Unit 3 of the Bugey nuclear power plant. A year later, five 900 MW and four 1300 MW vessel heads also showed cracked CRDM penetrations, which occurred, at the earliest, after 30 000 operating hours. Owing to similarities in the materials and fabrication of their vessel heads, EdF decided to replace them all. By 2009, 50 vessel heads had been replaced with new upper heads, equipped with Alloy 690 CRDM penetrations.

In the USA, inspections first revealed vessel head penetration nozzle cracks in 2001 at Three Mile Island Unit 1, Crystal River Unit 3, North Anna Unit 1 and Oconee Unit 3. Inspections at other plants revealed nozzle or J-groove weld cracks and/or leaks at Oconee Unit 2, North Anna 2, Arkansas Nuclear Unit 1, St. Lucie Unit 2, Milestone Unit 2 and Beaver Valley Unit 1. In 2002, a near through wall corrosion accident of the RPV closure head occurred at Davis–Besse. A large cavity formed in the 152.4 mm thick low alloy carbon steel RPVH material with stainless steel cladding. The cavity was about 168 mm long and up to 127 mm wide, extending through the carbon steel all the way down to the 6.4 mm internal surface stainless steel, type 308 cladding of the RPVH. The root cause of the corrosion was not directly PWSCC, but was the result of boric acid interaction with the carbon steel of the RPVH base metal. The source of the boric acid was a primary water leak via a through wall crack in a CRDM nozzle due to PWSCC.

In WWER-440 and WWER-1000 reactors, RPVs are protected against corrosion with a relatively thick (8 mm) austenitic stainless steel cladding material. The welded joints between the head materials and austenitic tubing (such as the control rod instrumentation attachments for the CRCS) are made of austenitic materials of the same type as the cladding itself. The first layer/bead on the ferritic base metal is type 25/13 material and the upper layers are stabilized type 18/10 austenitic stainless steel.

As long as the water chemistry regime is controlled within its specified limits, general corrosion, pitting and selective corrosion on the inside surface are not a severe matter of concern for the ageing management of RPVs in WWER units.

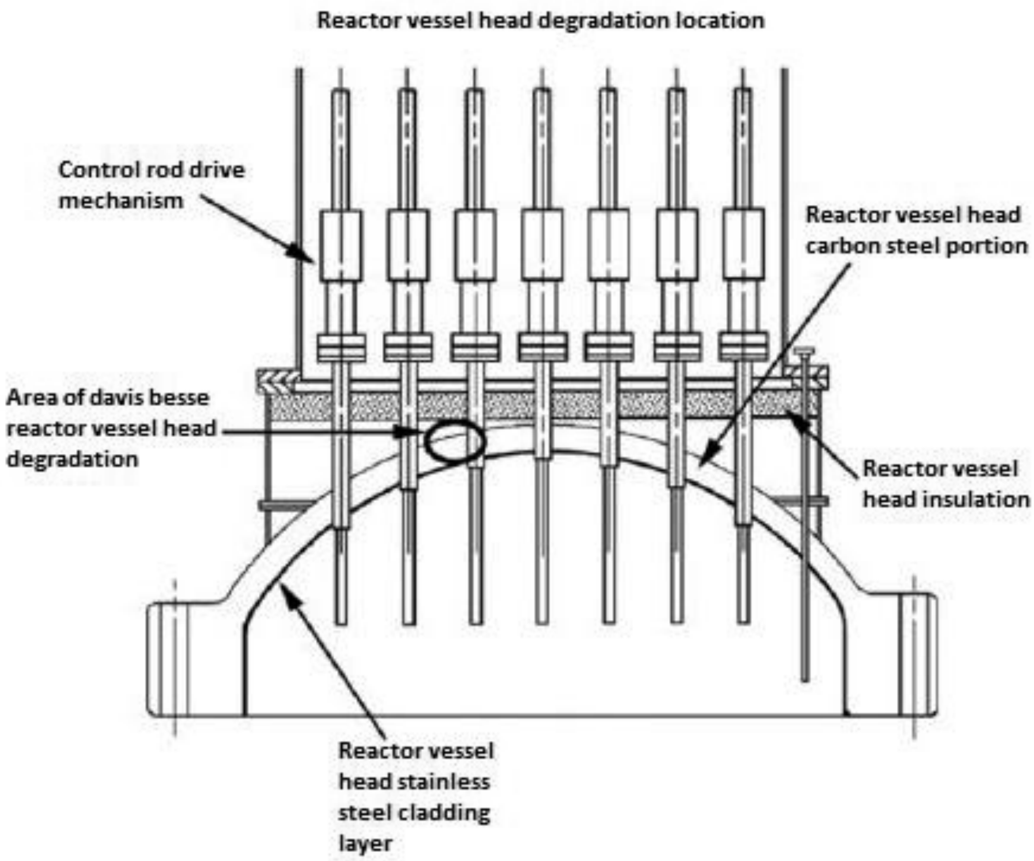

FIG. 27. RPVH degradation location.

5.2.4. Bottom mounted instrumentation penetration leaks in reactor pressure vessels

In the USA, in April 2003, small boron deposits around 2 of the 58 bottom mounted instrumentation penetrations (penetrations 1 and 46) were identified in the South Texas Nuclear Project Unit 1 PWR. In Japan, in January 2003, one small indication within the acceptance criteria (≤3 mm depth) was detected at the inner surface of one BMI penetration nozzle at Takahama PWR Unit 1 (see Fig. 28). Laser and water jet peening have been applied as mitigation measures at the Takahama unit and at other Japanese PWRs.

5.2.5. Thermal ageing in reactor pressure vessels

High temperatures in RPV materials produce a time dependent degradation mechanism. Thermal ageing does not appear to be generalized, but depends on the heat treatment, chemical composition and service time. The material may lose ductility and become brittle because of impurities precipitating and affecting its microstructure. Even before being aged in nuclear power plant service, RPV steels with a phosphorus content well above 0.02 mass% may be susceptible to tempering embrittlement during fabrication. However, the RPV materials of PWRs and BWRs normally contain less than 0.020 mass% phosphorus. Therefore, it is unlikely that any of these will exhibit tempering embrittlement.

The role of phosphorus in the overall embrittlement of RPV materials during operation has been the subject of much discussion over the years. The combination of thermal precipitation and radiation exposure accelerates the ductile to brittle temperature shift. The speed of degradation is also dependent on the type of microstructure of the metal. The effect of grain size on thermal ageing embrittlement may be due to grain boundary embrittlement by impurity segregation. Results from the thermal ageing surveillance specimens located in the upper plenums of the WWER-440/V-213 pressure vessels tested after 30 years of exposure at about 300°C indicate that the shift in the Charpy ductile to brittle transition temperature is small. If a 500°C thermal annealing of an irradiated RPV is required to recover fracture toughness, the possibility of tempering embrittlement should be evaluated.

The brittle to ductile transition temperature (critical temperature) of pressure vessel materials is time or use dependent. However, many damaging mechanisms can affect this temperature, primarily the temperature shift due to radiation embrittlement. A shift may also be caused by low cycle fatigue damage, corrosion and pitting.

5.2.6. Fatigue in reactor pressure vessels

In nuclear power systems, SSCs are mainly subject to low cycle fatigue (<1000 cycles), and the chief sources of cyclic stresses are temperature and pressure transients and, to some extent, vibration caused by some kind of abnormal operation (e.g. cavitation, acoustic resonance, wear and tear) if they remain undetected or unresolved. In RPVs, the closure studs have the highest usage factor (on the order of 0.66 for the 40 year design life of the vessel) of any other subcomponent. Unless an unexpected abnormal condition occurs that results in extreme vibration to any of the RPV subcomponents, fatigue damage is considered an insignificant degradation mechanism in RPVs.

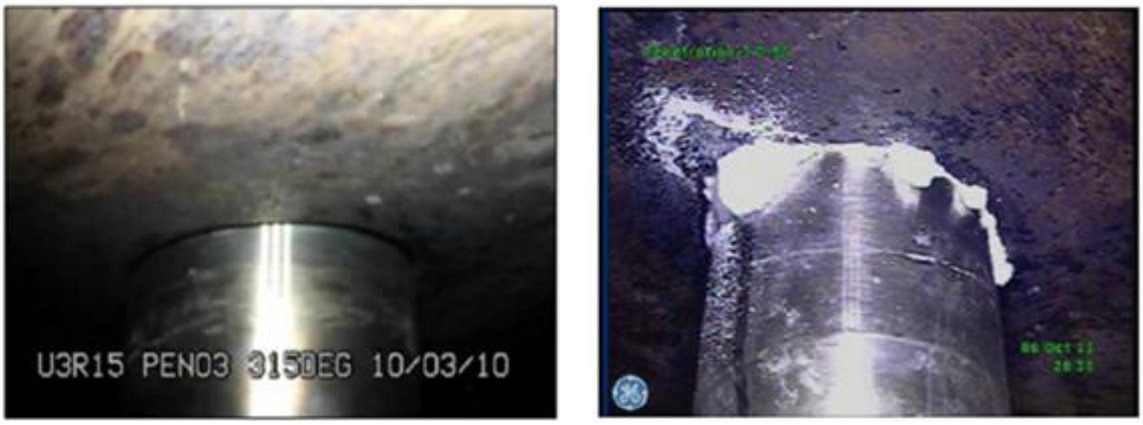

FIG. 28. Corrosion near the bottom mounted instrumentation penetration nozzle at the Takahama unit 1 plant, Japan.

The service life of the WWER-440 closure studs is limited to 15 years before their fatigue factor becomes a concern. Their replacement is regularly planned and follows a standardized maintenance procedure. In addition, in order to further reduce the probability of failure during their service life, these studs are tested every 4 years by ultrasonic and eddy current methods. The cladding of the primary nozzles on their inner radius is the second component subject to fatigue. However, its usage factor is below 1 for the total RPV service time. Additionally, it is subject to periodic inspections that would detect any abnormal fatigue signs.

5.2.7. Corrosion in reactor pressure vessels

Three major corrosion mechanisms have been identified in RPVs:

- Corrosion without mechanical loading (e.g. uniform corrosion/local attack and intergranular attack). This corrosion is characterized by material loss as metal ions dissolve in liquid electrolyte (anodic dissolution) and hydrogen is produced.
- Corrosion with mechanical loading (e.g. SCC and corrosion fatigue). The combined action of a corrosive environment and mechanical loading can cause cracking, even when no material degradation occurs under either the chemical or the mechanical conditions alone.
- Flow assisted corrosion attack (e.g. erosion corrosion, flow induced corrosion and cavitation).

Water chemistry control during operation, as well as during shutdown, is important with respect to avoiding corrosion problems. Thus, the content of all additives has to be carefully monitored and the ingress of impurities has to be strictly avoided (e.g. during stand-still periods and maintenance work).

5.3. PRESSURIZERS IN PRESSURIZED WATER REACTORS

Another vessel that is important to safety and reliability is the pressurizer. A life assessment is usually conducted on pressurizers for the development of a customized AMP in both PWRs and pressurized heavy water reactors (PHWRs).

5.3.1. Pressurizer relief line cracks

In Japan, during the 13th periodic inspection of Tsuruga Unit 2 in September 2003, cracks were found in the pressurizer relief line nozzle stub weld. The cracks were in the weld material, and the fracture surfaces were along the columnar grains. The cause was recognized to be PWSCC in the nickel based weld metal (Alloy 600 type). Subsequently, the weld metal materials were changed to nickel based Alloy 690 in PWRs.

The life assessment process for the pressurizer may vary in accordance with the regulatory practices of each country, and the type of plant and type of practice. The life assessment procedure applied to pressurizers in the Republic of Korea is shown in Fig. 29.

When identifying subcomponents within pressurizers of a specific type or class of plant, Korea Hydro & Nuclear Power makes use of the technical information in design guides and design manuals. The functions of each of the relevant subcomponents are then identified by reviewing the design specification, the technical specification, the bid specification, the manufacturing drawings and the FSAR. In order to identify and classify the degradation mechanisms, the operating and maintenance histories are reviewed. Finally, the appropriate technical codes and standards are used to evaluate the lifetime of the substructures and components.

5.4. MONITORING AND MITIGATION METHODS IN PRESSURIZED WATER REACTORS

5.4.1. Mitigation of radiation embrittlement

Radiation embrittlement in RPVs can be mitigated by either flux reductions (the most prominent being fuel management and RPV shielding from neutron flux exposure) or by thermal annealing of the RPV.

5.4.1.1. Fuel management methods of flux reduction

LLC flux reduction can be achieved by implementing a low neutron leakage core. An LLC is a core that utilizes spent fuel elements on the core periphery, which reflect neutrons back into the core, or absorb them, rather than allowing them to bombard the RPV wall. This LLC can decrease the neutron flux on the RPV wall to about 0.55–0.60 of the flux from the full core.

A more drastic reduction of neutron flux can be achieved by inserting dummy fuel elements (made of stainless steel) in the periphery of an active core, for example, in the corners of the WWER active core hexagons. The use of dummy elements usually results in a significantly different neutron balance in the core, reducing up to 2.5 times. The radial gradient is increased and thus the power distribution is disturbed in such a manner that the peak power may exceed certain limits. Thus, a reduction in the fuel cycle length or a reduction of the reactor output is often necessary.

5.4.1.2. Reactor pressure vessel wall shielding methods of flux reduction

Shielding of the RPV wall from neutron exposure can be accomplished by increasing the thickness of the thermal pads that exist on the thermal shield at locations where the fluence is high, or by placing shielding on the

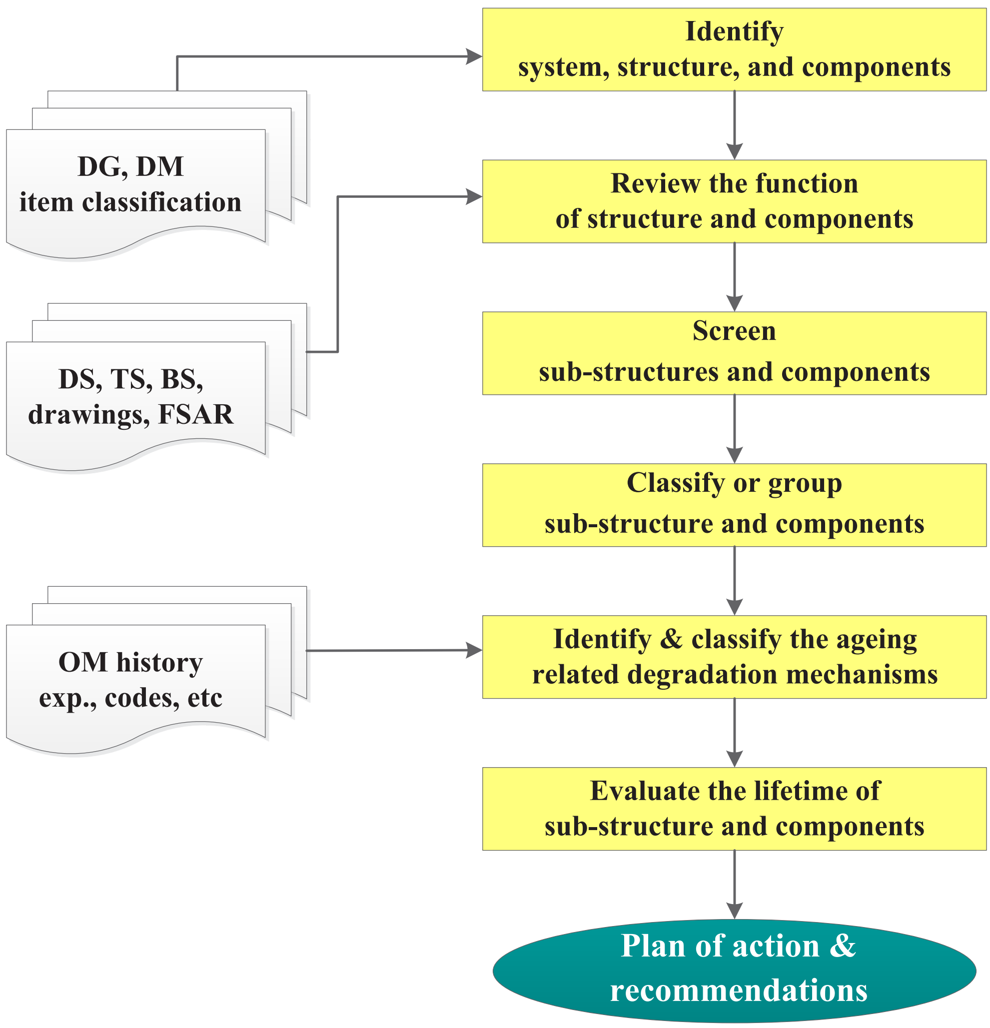

FIG. 29. Life assessment of a pressurizer by Korea Hydro & Nuclear Power, Republic of Korea. BS — bid specification; DG — design guide; DM — design manual; DS — design specification; exp. — experiences; OM — operation and maintenance; TS — technical specifications.

RPV wall. There are a number of alloys or elements that can provide shielding of the RPV wall by absorbing the high energy neutrons. A very effective shielding material is tungsten.

5.4.1.3. Thermal annealing

Once an RPV is degraded by radiation embrittlement (e.g. a significant increase in the Charpy ductile to brittle transition temperature or a reduction in the fracture toughness), thermal annealing of the RPV is the only method to recover the RPV material toughness properties. Thermal annealing is a method by which the RPV (with all internals removed) is heated to a high temperature by use of an external heat source (e.g. electric heaters or hot air), held for a given period and slowly cooled. In the USA, codes and standards that refer to annealing include 10 CFR 50 Appendices G [30] and H [31], 10 CFR 50.61 [32] and 10 CFR 50.66 [33] (the thermal annealing rule). They address the critical engineering and metallurgical aspects of thermal annealing and require, at the end of the process, a thermal annealing report. NRC regulatory guide 1.162 [34] describes the format and content of the required report.

In the Russian Federation, it was found that an annealing regime with a temperature at 475°C for a hold period of at least 100 hours resulted in acceptable mechanical property recovery (~90%) and a residual embrittlement, which was not caused by neutron fluence. Such annealing treatment was implemented in the RPVs of the WWER-440/V-230 type units in Bulgaria, Finland, Germany, the Russian Federation, Slovakia and Ukraine.

5.4.2. Mitigation of stress corrosion cracking in penetration of control rod drive mechanisms

The maintenance methods developed to correct the Alloy 600 RPVH penetration cracking problems include:

- Stress improvement methods.
- Grinding and rewelding or sleeving.
- Replacement of either individual CRDM nozzles or an entire head.
- Coolant additives. The most promising coolant additive is zinc, which has been shown to reduce the radiation activity of the primary coolant and to increase, at the same time, the resistance of the Alloy 600 to PWSCC. The zinc interacts with chromium in the oxide film on the Alloy 600 components and forms a protective and stable oxide coating, which delays initiation of PWSCC.
- Surface treatments, such as special grinding techniques, nickel plating, among others.
- Peening with shot or other methods that replace the high tensile residual stresses on the surface with compressive stresses (however, shot peening is not effective if cracks already exist).
- Grinding techniques to remove the surface layer where cracks, although undetected, might have initiated, followed by an application of compressive stresses on the regenerated surface.
- Nickel plating, which can protect treated surfaces from PWR coolant attacks, stops existing cracks from propagating and fills small cracks.

5.4.3. Vessel head repair and replacement

Repairs to the nozzle penetrations are possible by grinding out the stress corrosion crack and by filling the resulting cavity with a suitable weld metal or by inserting a thin liner of thermally treated Alloy 690 or austenitic stainless steel into the degraded nozzle penetration. The penetration is then pressurized, the liner expands onto the damaged penetration tube and seals the crack in the tube.

In France, EdF has renounced repairing vessel head penetrations and decided instead to replace all vessel heads as a preventive measure, because it is more economical and because current mitigation and repair techniques do not address the cracking issue of Alloy 182 weld metals.

5.4.4. Inspection and monitoring

RPVs in the USA are inspected in accordance with section XI of the ASME code [35]. There are three types of examinations used during ISI: visual, surface and volumetric. The three types of inspections always refer to pre-service inspection observations, which are required for all RPVs. Inspection plans are usually prepared, in which the first and the subsequent ISI intervals are decided.

All shell, head, shell to flange, head to flange and nozzle to vessel welds, as well as repair welds (if the repair depth is greater than 10% of the wall thickness) in the beltline region, are subjected to a 100% volumetric examination during the first inspection and at intervals of 3–10 years. All inspections require 100% volumetric examination of all these welds. The nozzle inside surfaces are all subjected to a volumetric examination, during each of the four inspections. Only 25% of the partial penetration nozzle welds (CRDM and instrumentation) are subjected to a visual examination, during each of the four inspections (amounting to an inspection coverage of all nozzles by the end of the fourth inspection). All nozzle to vessel butt welds with dissimilar metals (i.e. the ferritic steel nozzle to stainless steel or Alloy 600 safe end welds) are subjected to volumetric and surface examinations each time. All studs and threaded stud holes in the closure head are subjected to surface and volumetric examinations at each inspection. Any integrally welded attachments have surface (or volumetric) inspections of their welds at each inspection.

In the USA, all PWR RPVHs have closure bolts in compliance with ASME section III and are inspected according to ASME section XI [35]. NRC regulatory guide 1.65 [36] provides guidance on vessel closure bolting materials and inspections, and regulatory guide 1.150 [37] provides guidance on ultrasonic test procedures, which supplement those provided in ASME section XI.

In Germany, ISI includes all welds, nozzle surfaces, control rod ligaments in the top head, studs, nuts and threaded stud boreholes. The inspection interval for RPVs is 4 years (for conventional vessels, it is 5 years); the scope of an inspection may be subdivided into smaller scopes, carried out separately during the 4 year inspection interval.

In France, code RSE-M requires periodic hydro tests, acoustic emission NDE monitoring during outages, a material surveillance programme, loose parts (noise) monitoring during operation, leak detection during operation and fatigue monitoring. The code specifies a complete inspection programme, extending to both the utility and the regulatory agency. Areas of the RPV that need to be inspected include the beltline region of the shell, all the welds, the top and bottom heads, the nozzles, the penetrations, the CRD housings, the studs and the threaded holes. A hydro test at 1.33 times the design pressure (22.4 MPa) is required after the RPV fabrication is completed. A hydro test at 1.2 times the design pressure (20.4 MPa) is then periodically performed every 10 years of operation. The 10 year tests need to be performed at a reference temperature of a nil ductility transition temperature of 30°C.

For WWER plants, RPV ISI is carried out at least every 4 years (30 000 hours of operation) and includes NDE (visual, dye penetrant, magnetic particle, ultrasonic and eddy current testing), surveillance specimen evaluation and hydraulic testing. Parts and sections of the reactor to be inspected, as well as locations, volume and periodicity, are procedurally specified. Examinations of the RPV base and weld metal in the zones with stress concentrations or high neutron flux, such as the cladding/base metal interface, the nozzle transition areas, the sealing surfaces, the outer and inner surfaces of the vessel bottom and top heads, all bolts, nuts and threaded holes, are mandatory.

5.4.5. Surveillance programmes

In the USA, fracture toughness requirements are governed by Appendix G, Fracture Toughness Requirements, of 10 CFR 50 [30], which contains requirements for prevention of ferritic material fractures in the primary coolant pressure boundary of US nuclear power plants and by Appendix H [31], which contains a set of rules for the reactor vessel material surveillance programmes.

In Germany, regulations allow radiation embrittlement to be neglected when the neutron fluences are lower than 10^{21} n/m^2 (E > 1 MeV). The maximum allowed RPV fast neutron fluence in Germany is limited to 1.1×10^{23} n/m^2 (E > 1 MeV). The German nuclear safety standards, applies only to this fluence or lower values. The number of radiation surveillance sets and the withdrawal schedule, relative to the RPV end-of-life design fluence are fixed (two sets covering 50% and 100% of the RPV design life fluence). The KTA 3203 allows higher lead factors (>3) on the radiation capsules.

In France, the material surveillance programme is specified in RSE-M. The norms are similar to those of the US programme. Some changes in the French surveillance programmes (rearrangement of the capsules, new material in the capsules and laboratory tests, among others) are being studied to support a possible life extension from 40 to 50 years. The objective of these changes is to provide sufficient information early enough in the lifetime of a plant to help the owner/operator with the decision of whether to proceed with LTO planning at the 20 year mark of plant operation.

In the Russian Federation, the requirements for the WWER material surveillance programmes are given in the following documents:

— OPB-99/97, NP-001-97, General Regulations to Ensure Nuclear Power Plant Safety [38];
— NP-017-2000, Basic Requirements for Power Unit Lifetime Extension of Nuclear Power Plants [39];
— PNAE G-7-008-89, Rules for Arrangement and Safe Operation of Equipment and Piping of Nuclear Power Installations [40].

They were applied only to the WWER-440/V-213 and WWER-1000 nuclear power plants. The older design WWER-440/V-230 was not supplied with a material surveillance programme. Surveillance specimen programmes contain specimens from the base metal, weld metal and heat affected zone for tensile testing, Charpy V-notch impact tests and fracture toughness tests. In addition, some RPVs of the WWER-440/V-213 type reactors are supplied with supplementary programmes that also contain specimens from austenitic cladding and IAEA reference steel JRQ[3]. In addition to the specimens for monitoring radiation damage in the beltline materials, the WWER surveillance programmes include specimens for monitoring the thermal ageing damage in pressure vessel materials. Two complete sets of surveillance specimens are located well above the active core (receiving virtually no appreciable neutron flux), at approximately the elevation of the outlet nozzle ring. These sets are usually removed and tested after 5 and 10 (or 20) years of operation.

5.5. REACTOR PRESSURE VESSEL INTERNALS AND PIPING IN BOILING WATER REACTORS

The AMP materials typically subjected to material degradation phenomena are mapped in Fig. 30 in connection to three main factors: material type, environment and stressors. Two or three major degradation mechanisms are always shown as acting simultaneously.

The yellow boxes at the top of Fig. 30 represent the screening and examination of the various materials found in BWRs, such as low alloy steels, dissimilar metal weld materials, nickel based stainless steel alloys, duplex stainless steels with mixed microstructure of austenite and ferrite, carbon steels, martensitic stainless steels and others.

The dark blue boxes on the top left show the stressors causing fatigue and the computational codes in use, namely:

— CFD analysis to determine the constant loads, cyclic loading functions and time histories.
— Finite element method analysis to study the material response, namely fatigue calculations, stress intensity distributions and the cumulative usage factor (CUF) in response to the time histories of the mechanical strain rate, of the temperature distribution changes with their time history and of the evolutions of the dissolved oxygen content in time, in accordance with the relevant ASME code for pressure vessels in nuclear power applications. The CUFs are then duly augmented by the environmental effect correction factor (F_{en}) as a multiplier of the usage factor.

Piping calculations are similar, but somewhat simpler. Linear models can be used instead of three dimensional or axisymmetric finite element models, except in particular cases such as geometrical discontinuities (e.g. $t/3t$ or $3/5t$, where t denotes the material thickness) or special piping inserts such as thermal sleeves, special flanged connections, fluid heads as in line anchors, floating piping, metal expansion joints, bellows, etc.

The pink boxes show the stressors on the materials, namely the two phase flow (water–vapour and water–gas) and wall interaction, the irradiation distribution and the temperature distribution. The central part of the diagram

[3] American Society for Testing and Materials A-533-BCl.1 forged or rolled steel plate, first manufactured by Kawasaki, Japan, for the IAEA for its coordinated research projects on reactor pressure vessel integrity to provide a standardized RPV material for testing. It has since been internationally accepted as a base plate material for RPV irradiation damage studies.

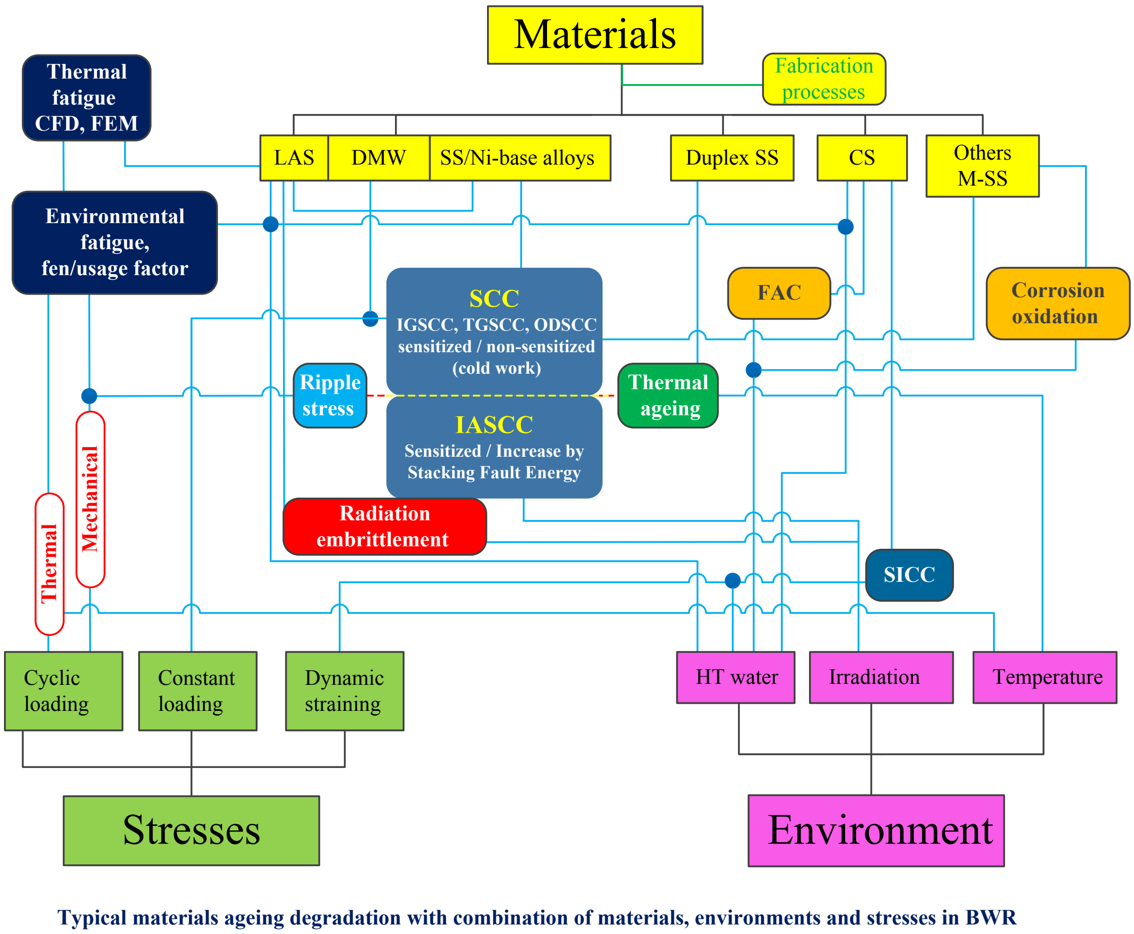

FIG. 30. Typical material degradation resulting from a combination of material types, environments and stressors. CFD — computational fluid dynamics; CS — carbon steel; DMW — dissimilar metal weld; FEM — finite element method; HT — high temperature water; M-SS — martensitic stainless steels; ODSCC — outside diameter stress corrosion cracking; SICC — steam induced corrosion cracking; SS — stainless steel; TGSCC — transgranular stress corrosion cracking.

shows the degradation mechanisms resulting from the superimposition of the main contributing factors and the degradation type to be expected, for example:

— Carbon steels acting coincidentally have residual stresses from fabrication processes (e.g. cold bending), or hydrogen injected coolant and metal interaction may be found, producing flow assisted corrosion and dynamic straining, producing strain induced corrosion cracking.
— Duplex stainless steels have residual stresses inherited from the fabrication processes (if any) may seemingly be the only main contributor to thermal material degradation.
— Martensitic and other types of stainless steels have residual stresses inherited from the fabrication processes and coolant to metal interaction, which produce different types of SCC, namely IGSCC and TGSCC, and contribute to material degradation.
— Nickel based alloys have residual stresses inherited from the fabrication processes, irradiation, coolant to metal interaction and constant loading, if present, which all contribute to material degradation. Degradation can take the form of radiation embrittlement, and, if in contact with the reactor coolant, SCC either as IGSCC or TGSCC.
— Dissimilar weld materials have residual stresses that are inherited from the fabrication processes and that contribute to material degradation. Constant loading, radiation embrittlement and contact with the reactor coolant can also cause SCC in the form of IGSCC and TGSCC, and ODSCC, that is either sensitized or non-sensitized.

— Low alloy steels have residual stresses inherited from the fabrication processes. Hydrogen injected coolant and metal interaction effects, ripple stresses, irradiation causing radiation embrittlement and SCC in the form of IGSCC and TGSCC contribute to material degradation.

For more information, see Refs [41–45] on materials degradation and inspections in BWR RPVs, RPV internals, nozzles and piping.

5.5.1. Radiation embrittlement in reactor pressure vessels and overlay cladding

Radiation embrittlement in ferritic materials is one of the key ageing management issues in BWRs. It is addressed by performing a surveillance programme, even though the estimated maximum dose at the end of a BWR vessel operational lifetime is lower than that of PWR vessels, owing to the larger vessel diameters in BWRs and the greater distance between the reactor core and the RPV wall. Fracture toughness and non-destructive radiographic testing evaluation provide a quantitative assessment of RPV integrity.

Another concern in BWRs is that the weld overlay is susceptible to irradiation related embrittlement. This cladding is typically made of 308 or 309 austenitic stainless steels with some ferrite. The combination of thermal ageing and irradiation may cause brittleness of the cladding in the presence of small flaws, or when ageing defects have built up in the welded joint affecting the cladding by mechanisms such as SCC. More recent vessels have been fabricated using knowledge gained from surveillance programmes and modern methods such as the use of large ring forgings to reduce the number of welds in the core beltline. The IAEA-TECDOC on BWR RPVs [42] discusses in detail the effects of neutron irradiation on the mechanical properties of the stainless steel cladded steels and welds of the RPVs of light water cooled reactors (PWRs, BWRs and WWERs).

5.5.2. Irradiation assisted stress corrosion cracking, intergranular stress corrosion cracking, transgranular stress corrosion cracking and outside diameter stress corrosion cracking in the core shrouds, piping and spent fuel pools

The water chemistry of BWRs is very different from that of PWRs, where there is no dissolved hydrogen under pressure, owing to the boiling conditions in the BWR vessel. On the other hand, water radiolysis is more chemically aggressive in BWRs than in PWRs and the electrochemical corrosion potential (ECP) in the reactor core region is rather high due to the formation of oxidants such as hydrogen peroxide (H_2O_2). Because of this rather high oxidizing environment, SCC phenomena were experienced in BWRs as early as the 1960s. The most common mitigation processes used individually, or in combination, to substantially reduce SCC in BWRs include:

— Design changes: Changing the design to implement internal jet pumps for water recirculation in order to eliminate recirculation pipes.
— Material changes: Using materials that are more IGSCC resistant, for example, changing the recirculation pipes from stainless steel type 304 to a lower carbon content stainless steel such as type 316L. It is to be noted that 304 stainless steel also showed SCC susceptibility when cold bent to form an elbow and when the surface is machined. Alternatively, cladding the surfaces of SCC susceptible regions with metal resistant deposit, such as stainless steel type 308L, Alloy 82 or a noble metal, have proven successful as isolation techniques. It should be mentioned that hardened surfaces tend to increase susceptibility to SCC as experienced in operation and predicted by crack tip strain model theory.
— Environment improvements: Changing the water chemistry to reduce the ECP is achieved via a drastic reduction of the oxidants by injecting hydrogen into the feedwater (0.3–2.5 parts per million depending on the core loading and the reactor fuel cycle as a function of burnup). This process is known as hydrogen water chemistry. One of the drawbacks to using hydrogen injection into the feedwater is an increase in the radiation levels in the main steam lines caused by the activation of nitrogen as ^{16}N. In order to reduce the dose increase in the system, platinum and rhodium are injected into the reactor water. This is called noble chemistry injection and is conducted either on-line or during an outage, to favour the catalytic recombination of the hydrogen and obtain the added bonus of further reducing the ECP. Another method to reduce radiation fields, fostered by hydrogen water chemistry in the system, even more effective than noble chemistry injection, is to clad (welding method) or to coat (plasma spray method) the reactor inner surface with noble metal.

Other interventions to reduce SCC include changing surface finishes to minimize the surface hardening layer, and changing surface roughness by polishing, such as buffing. In addition, changing tensile residual stresses in hardened surfaces to compressive stresses would be beneficial and could be achieved by surface peening (such as shot peening, water jet peening or laser peening). It is important to note here that peening remains effective only as long as compressive stresses are maintained. Otherwise, peening produces a hardened layer that may promote SCC. Hence it is important to ensure that compressive stresses remain such throughout the lifetime of the plant.

SCC has also been experienced in the spent fuel pools of BWRs. The fuel structural material consists mainly of zircaloy and stainless steel zircaloy. It has proven to be insensitive to any kind of corrosion phenomena (e.g. uniform corrosion, SCC or electrocorrosion) at temperatures below 60°C. Above this temperature, residual stresses in welds play a key role in promoting corrosion. Spent fuel pools have received particular attention since the Fukushima accident, where it was assumed that a shortage of cooling water may have produced spent fuel criticality. Consequently, the reliability of the cooling water system and its maintenance practices have recently been subjects of concern. One of the recommendations resulting from the lessons learned that followed the Fukushima accident is that any leaks of coolant, as well as the coolant level in spent fuel pools, need to be constantly monitored [46].

5.5.3. Stress corrosion cracking in dissimilar metal welds

The RPV base metal in BWRs is normally low alloy steel, internally protected against corrosion by means of a stainless steel cladding. The most common type of cladding is weld overlay cladding. This process creates a dissimilar metal weld situation at the interface between the stainless steel cladding and the low alloy steel base metal of the RPV. SCC in dissimilar metal welds is one of the key issues in a structural integrity assessment of a BWR reactor vessel. SCC can be initiated in nickel based weld metal in the dissimilar metal weld region, propagate in the weld and reach the fusion boundary. The safety concern is whether the SCC originated and propagated in the dissimilar metal weld can penetrate into the low alloy steel base metal. Laboratory observations in BWR plants have shown crack penetration crossing the fusion boundary and propagating into the low alloy steel, where impurities such as sulphur content can play an important role in the crack growth rate.

The first reactor to produce electricity in Japan was a 45 MW prototype BWR, the Japan Power Demonstration Reactor (JPDR), which reached criticality in 1963 and which provided a large amount of information for later BWR commercial reactors. In 1969, the JPDR reactor core was remodelled and converted to a 90 MW BWR and continued to run until 1976. The JPDR also later became the test bed for reactor decommissioning. After 13 000 hours of operation, SCC was found in the stainless steel weld overlay cladding of the JPDR RPV in the surface exposed to the reactor coolant. The corrosion cracks penetrated the interface between the stainless steel weld overlay cladding and the RPV base metal. Detailed analysis revealed that the cracks terminated at the fusion boundary, forming a large pit in the low alloy steel material, as it tended to relieve the stress-strain concentration at that location. To prevent similar conditions from repeating themselves in power plants that followed JPDR in Japan, the following corrective actions were applied:

— The residual stresses at the dissimilar material weld joint were reduced in order to lower the stress intensity factor K, and to increase the resistance to SCC, even if a crack reached the fusion boundary of the weld overlay cladding;
— Low sulphur RPV steels were used.

In most situations in commercial power plants, SCC stops at the fusion boundary, where the risk of crack penetration into the low alloy steel base metal is much lower. However, another possibility of crack propagation is at the low alloy steel to nickel based weld metal interface, where an SCC susceptible microstructure might form along the dissimilar metal weld fusion boundary. In such cases, a crack may very well propagate along the fusion line circumferentially, which may cause a sudden rupture of the fusion boundary when the residual stress field is favourable to such a crack propagation type.

5.5.4. Environmental fatigue in nozzles and piping

The ASME codes provide a design curve for fatigue calculations in pressure boundary components. The *S–N* curve shown in Fig. 31 represents fatigue limits in terms of alternating stress amplitude (*S*) versus number of cycles to failure (*N*), as demonstrated in extensive laboratory data obtained in Japan and in the USA [47].

To complicate the fatigue effects on BWR metals, environmental effects appear to accelerate metal fatigue as a time dependent phenomenon; hence, they are an important factor in material degradation studies. Environmental effects have been a concern because they resulted in prominent ageing factors in tests conducted in laboratory simulated BWR environments in Japan in the mid-1970s. Intensive research has revealed the possibility that cracks continue to penetrate into the low alloy steel base material, under specific environmental conditions. It became clear that environmental stressors have always acted with varying intensities in all materials used in BWR vessels (carbon steel, stainless steel, nickel based alloys and their weld metals).

Environmental effects affect the *S–N* fatigue curve and are more evident at lower frequencies or lower strain rates. However, it is not always simple to exactly define the endurance limit, in other words, the strain rate or the thermal fatigue load frequency at which the environmental effects are no longer a factor. The most prominent environmental effects in the *S–N* curve are observed in carbon steels. However, for other materials in use at nuclear power plants, working endurance limits correspond to large numbers of cycles. A project attempting to understand how environmental effects on thermal fatigue cause the reduction of fatigue life and the conditions under which fatigue cracks initiate and propagate, has been recently incorporated in a Japanese Nuclear Regulatory Authority (NRA) programme. In general, it has been found that thermal fatigue damage is limited only to the component surface in relation to the thermal stress distribution. However, if there are tensile and/or bending stresses in a piping system, small thermal fatigue cracks may appear, grow undetected and cause failure in pipes.

5.5.5. Flow accelerated corrosion in carbon steel piping

FAC is a material degradation phenomenon acting particularly in carbon steel components with high flow velocity. For FAC prone components, periodic ISI is required to ensure that sufficient pipe thickness, above the minimum requirement, remains available for the expected service life. Two ISI methods are used. One is to always monitor the same fixed points at each inspection to evaluate the FAC trend. The other is to inspect a wide area covering all possible FAC zones in a component, within a certain period of plant operation. A key factor in the management of FAC is the capability to make predictions on the trend of the FAC rate, based on periodic measurements of the minimum wall thickness reduction in pipe/elbow components.

5.5.6. Stress corrosion cracking and fatigue in pumps and valves

Several effects of SCC cases coupled with fatigue in pumps and valves have been reported in BWR plants. One important incidence is the cracking found in valve stems made of high strength martensitic stainless steel. In addition, pump casings and valve bodies made of cast stainless steel (duplex stainless steel) have shown cracking,

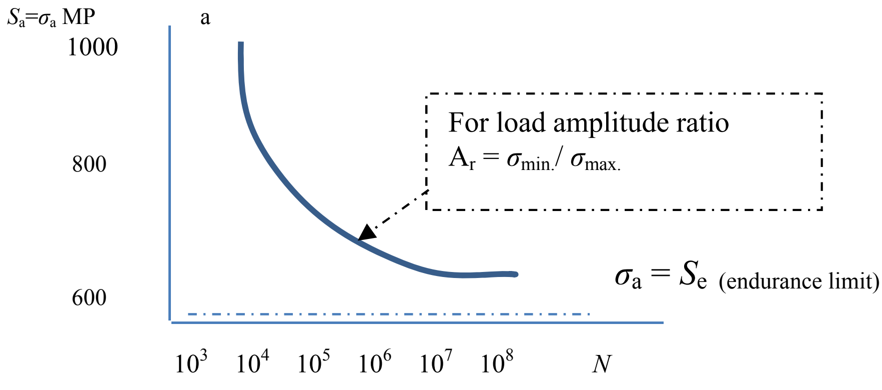

FIG. 31. S–N *fatigue limit curve for a specific loading amplitude.*

and a concern has been raised, in the context of proactive ageing management, about whether SCC and fatigue susceptibility is increased by thermal ageing.

5.5.7. Thermal ageing embrittlement in duplex stainless steels

Thermal ageing embrittlement may become a material degradation issue in cast stainless steel and austenitic stainless steel weld material. Both contain the delta ferrite phase of austenitic material. Thermal ageing takes place mostly in the delta ferrite phase, known as 'spinodal decomposition', which is a kinetic decomposition in a nanoscale pattern, forming a very highly chromium enriched zone. This results in a marked increase in hardness and embrittlement, which leads to the component displaying a fractural behaviour.

Evidence of this kind of embrittlement is the frequently observed increase in the ductile to brittle transition temperature in Charpy impact tests. Thermal material degradation can take place below 450°C; above this temperature, the ageing produces only a σ phase embrittlement, which is of no concern in LWRs.[4]

Under the BWR normal operating temperature of 288°C, thermal embrittlement can be of concern in the long term. This degradation should be monitored during periodic ISIs by measuring the metal hardness and determining the degree of embrittlement and its trend. Other NDE methods have been proposed in PWRs where this degradation can take place even earlier, due to the higher operational temperature of about 360°C. Among the methods in use at PWR plants, it is worth mentioning thermoelectromotive force measurements and calorimetric measurements. Further information is given in Section 5.8.

5.6. CANADA DEUTERIUM–URANIUM REACTORS

5.6.1. Assemblies: Calandria, shield tanks and end shields

In CANDU reactors, material degradation monitoring and maintenance activities on the reactor components, including fuel channel replacements, can be readily performed because of the extensive shielding provisions by design. Shielding in CANDU reactors can be divided into two parts:

— Shielding of the reactor faces or end shielding;
— Radial shielding.

End shield structures contain a recirculating light water cooling system, which, together with the fuel channel shield plugs, provides biological protection at the reactor faces, when the is reactor shutdown. The end shield cooling system also acts as a heat sink at both reactor ends. The end shield structures are made of two tube sheets supporting the fuel channel end fittings, as shown in Fig. 32. End shields and shield plugs together constitute the shielding of the two reactor faces.

Radial shielding is achieved in most cases by a shield tank surrounding the calandria. This tank also acts as a heat sink. In the CANDU 6 design, however, radial shielding from the calandria is provided by a steel lined concrete construction, the reactor vault, filled with light water, which also serves as a calandria support system. In the CANDU 6 design, the reactor vault offers the advantage of combining thermal shielding and biological radial shielding.

Together with the end shields, the reactor vault or the shield tank (depending on the specific design) provides full biological shielding on all sides of the reactor.

Encased in the end shields and protruding from the reactor faces are the specially shaped stainless steel end fittings. From a structural viewpoint, the end fittings support the zirconium alloy pressure tubes to which they are connected through rolled joints. The on-line refuelling machines latch onto the end fittings to insert fresh CANDU fuel bundles into the pressure tubes and remove burned up bundles from the pressure tubes.

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[4] The σ phase is a non-magnetic intermetallic phase composed mainly of iron and chromium, forming in ferritic and austenitic stainless steels when exposed to high temperatures well above the LWR operating range. Depending on the extent of the σ phase, loss of ductility may occur with little effect on material properties. However, cracking could occur if the components were impact loaded or incidentally stressed at lower temperatures (e.g. during maintenance work).

5.6.2. Ageing deformation of pressure tubes

Pressure tubes are exposed to temperatures up to 313°C, internal pressures of up to 11 MPa, neutron fluxes of up to 3.7×10^{17} $m^{-2} \cdot s^{-1}$ and fluences of up to 3×10^{26} m^{-2} (in 30 years of operation at a nominal 80% capacity). These conditions cause changes in dimensions and material properties through irradiation damage and microstructural evolutions.

Pressure tube deformation is a leading ageing factor in operating HWRs. In CANDU type reactors, horizontal pressure tubes undergo three types of deformations:

— Diametrical creep, leading to flow bypass of the fuel bundles with the consequent penalty to the heat flux;
— Longitudinal creep, which may lead, in time, to interference of feeder pipes among themselves and with the fuelling machine;
— Sagging of the pressure tubes, which may lead to interference between the pressure tubes and the calandria tubes, shown in Fig. 33, and eventually the fuel channel assemblies with regional power control components in the central core region.

Earlier CANDU fuel channels were not designed for a 30 year service life and had to be replaced before that length of time. The later generation of CANDU designs have deformation allowances of their fuel channels compatible with 30 years of operation.

In terms of ageing mechanisms, the presence of hydrides has been associated with crack initiation and propagation in zirconium alloy pressure tubes.

Although the overall performance of pressure tubes has been good, some pressure tubes in early CANDU reactors have leaked, and two pressure tubes have ruptured due to delayed hydride cracking (DHC), one during operation. As all crack propagation, and most crack initiation for CANDU pressure tubes, has been associated with DHC, the tube design and its specification have been modified to minimize the probability of DHC occurrence. It became imperative for CANDU reactors that the hydrogen content in new pressure tubes and the deuterium absorbed by the tubes during service be minimized. These measures have also increased the probability that a LBB, rather than a tube rupture occurs, should DHC still happen.

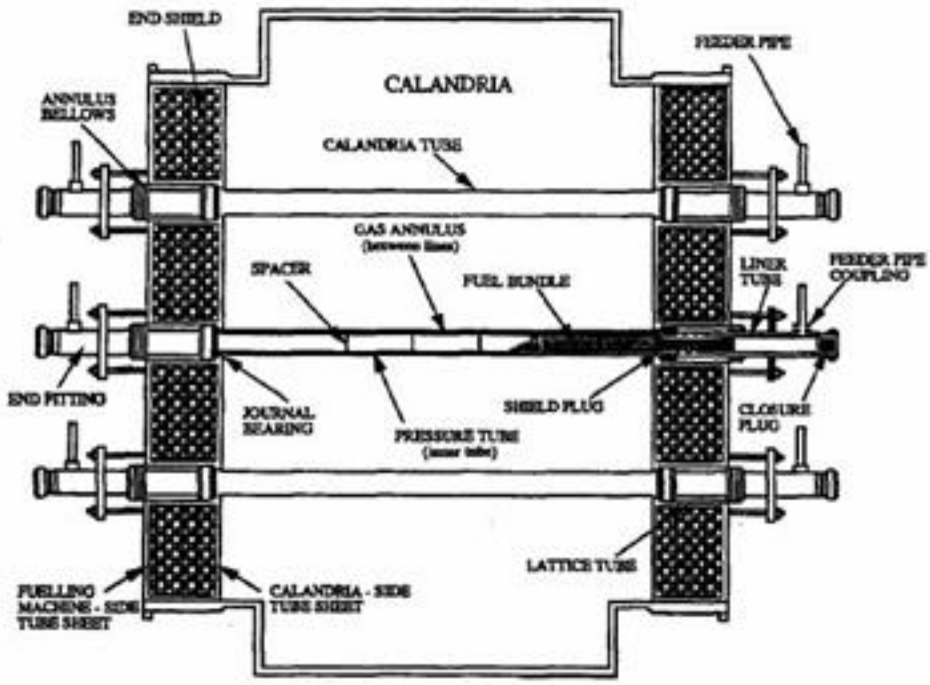


FIG. 32. CANDU reactor assemblies.

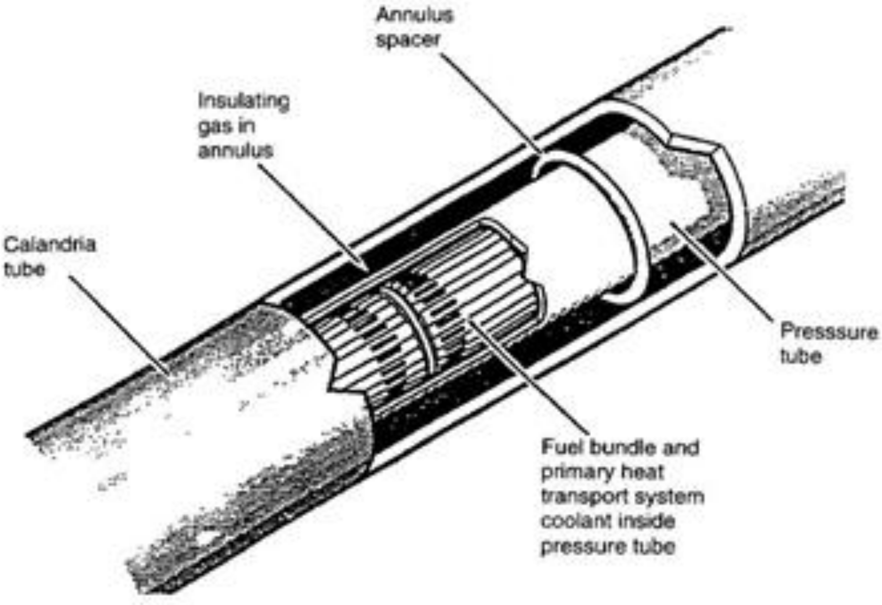


FIG. 33. CANDU: fuel channel schematic diagram.

5.6.3. Geometry monitoring of pressure tubes

Diametrical expansion of the pressure tubes results in flow bypass, which somewhat reduces fuel cooling, but this is not expected to cause a significant concern during the 30 year design life for CANDU fuel channels. In any case, if diametrical expansion becomes a concern, fuel management and design changes can be used to mitigate fuel cooling bypass. The channel inspection and gauging apparatus for reactors (CIGAR) inspection tool uses an ultrasonic gauging system that measures wall thickness and diameter at 60 positions around the tube and at 3 mm axial intervals along its length to confirm that these deformations continue to be acceptable. CIGAR is a multitasking tool. Other than diameter and wall thickness, CIGAR is also capable of flaw detection, pressure tube sag measurements, spacer location indication, inside surface profilometry, video visual inspection, gap measurement between the pressure and calandria tubes, and ultrasonic and eddy current flaw detection.

Measurement of the pressure tube displacement profile and the maximum vertical deflection (sag) are also a periodic inspection requirement. CIGAR uses a servo inclinometer to measure the slope at any position along the pressure tube and calculates the displacement profile from a single integration. Although this is not expected to occur during the 30 year design life for fuel channels, measurements of fuel channel sag (and of the gap between fuel channels and mechanisms) are performed for lead units to ensure that contact will not occur. If necessary, the horizontal mechanisms can be moved downward to increase the gap between them and the calandria tubes above them.

Another radiation linked deformation is the elongation of the pressure tubes. Elongation measurements of channels without radiation exposure issues are achieved through a scanning tool for elongation measurements. This tool is attached to the fuelling machine and can be operated remotely. Measurements are made periodically, either with the reactor in the shutdown state or at power. Some units obtain the required data from the fuelling machines each time a channel is refuelled. They can be used to confirm that elongation of the fuel channel is within the design limits, but they can also provide an accurate elongation rate for each tube, which can be used, for example, to establish the correct timing for the mid-life channel reconfiguration, and provide a lead time for planning corrective action.

Spacer movement is an operation linked displacement of the garter springs, which are spacers that separate the hot pressure tubes from the cooler calandria tubes. Displaced spacers may cause two adjacent tubes to come into contact with each other. A spacer location and repositioning (SLAR) system has been developed, on behalf of the CANDU Owners Group, to allow repositioning of any garter springs that may have moved out of the original design position. SLAR has a fast scan blister detection system that uses six line focused ultrasonic transducers to inspect the outside surface of the bottom 60° of the pressure tube, during a single axial pass through the tubes. SLAR can also reposition spacers, if they are found to be out of place.

In terms of FAC, the small wall thickness reduction that occurs during reactor operation is of no concern, as it is more than compensated for by irradiation strengthening.

Deuterium and hydrogen ingress are monitored periodically. The periodic inspection programme specified in Canadian Standard Association (CSA) standard N285.4 [48] requires that 6 pressure tubes be scraped about 10 years after first power generation in the lead unit of a multiunit power generating station. A sampling tool is available that scrapes thin slivers of metal from operating tubes, leaving a smooth, rounded groove, with a depth much smaller than the tube corrosion and wear allowance. These samples are analysed for deuterium and hydrogen ingress using a specially developed vacuum extraction technique.

Fuel channels in CANDU reactors are equipped with a leak detection system. Dry carbon dioxide gas flows through the annulus between each pressure tube and its calandria tube. The moisture content in the annulus is constantly monitored during reactor operation, using very sensitive leak detection equipment. The annulus gas system almost immediately identifies if any pressure tube leakage starts to occur. If a leak occurs, the channel is inspected and replaced if necessary.

5.6.4. Flow accelerated corrosion in feeders

Feeder pipes carry the reactor coolant to and from individual fuel channels. Feeders are part of the pressure boundary and any leak in a feeder pipe would constitute a small LOCA. A case of FAC occurred in primary system feeder pipes of early CANDU designs (see Fig. 34). Considerable feeder thinning was observed for the first time at the Point Lepreau reactor in Canada in 1995, especially at the outlet elbows, close to the exit of the primary coolant from the reactor. The degradation mechanism identified was determined to be FAC. This prompted an inspection programme to assess the extent of the problem.

The Canadian industry developed fitness for service guidelines to provide evaluation procedures and industry standard acceptance criteria for assessing the structural integrity of feeder pipes. Advancements in wall thickness measurement techniques were made, such as the METAR Crawler developed by Hydro Québec, which scans feeder bends and uses 14 eddy current probes and motors to drive the wall thickness measuring bracelet along the feeder bend. Other feeder fitness for service tools called GAITS and Gravis, developed by Kinetrics and funded by the CANDU Owner’s Group, used eight transducers in a water wedge housing.

It was recognized that FAC of the outlet feeder piping was exacerbated by low chromium content in the feeder steel and by local hydraulic behaviour. In addition, the coolant becomes significantly unsaturated in iron as it traverses the non-ferrous zirconium alloy pressure tubes and is heated to ~300°C, which causes the feeders to dissolve more iron. This, combined with turbulence and high mass transfer as the coolant enters the feeder bends results in wall thinning rates at the bends that are high enough to reduce feeder life. Based on available solubility data for iron and Fe_3O_4, some reduction in rate can be achieved by tightening the coolant pH from the previously controlled range 10.2–10.8 to the reduced range of 10.2–10.4. For new reactors, and for replaced feeders during reactor refurbishments, it has been demonstrated by both in-service performance and in and out of reactor laboratory testing that specifying chromium contents of the feeder steel ≥ 0.3 mass % reduces feeder thinning rates by 50%. Typically, the chromium content of early CANDU reactor feeder pipes was very low, < 0.02 mass %, which is a consequence of the low cobalt content, originally imposed, in order to reduce corrosion product activation. Given that the feeder pipe FAC mechanism is well understood and predictable, this degradation mechanism can today be readily managed for current and extended operation [49].

5.7. STEAM GENERATORS IN PRESSURIZED WATER REACTORS AND PRESSURIZED HEAVY WATER REACTORS

5.7.1. Steam generator tubing materials

In the mid-1970s, PWRs around the world suffered from a series of SCC incidents that were mostly confined to Alloy 600 steam generator tubing, initially on the secondary side (ODSCC), then on the primary side (PWSCC)

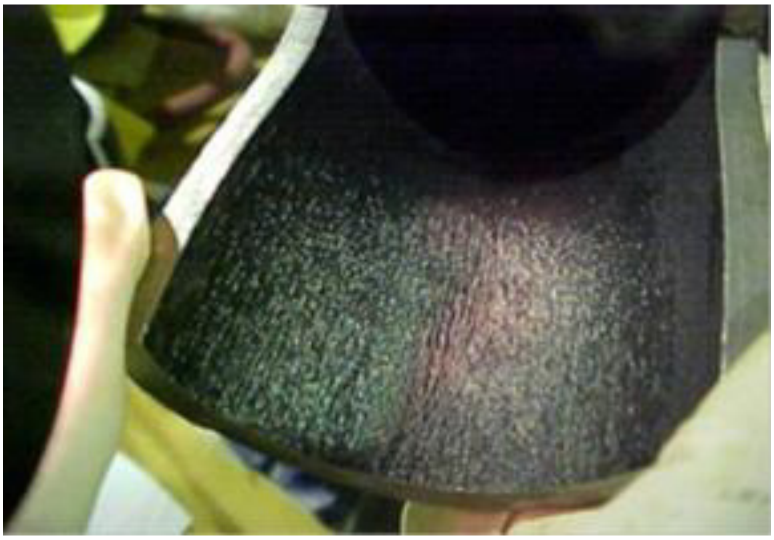


FIG. 34. FAC in primary system feeder pipes (~5 cm).

as well. The ensuing tube damage resulted in substantial economic loss for utilities in the 1980s and premature replacements of steam generators in the USA and elsewhere. ODSCC of mill annealed Alloy 600 steam generator tubing has also continued to the present day and has led to many steam generator replacements. ODSCC of thermally treated Alloy 600 steam generator tubing has also been observed. In the early 1980s, Alloy X-750 GT support pins also began to suffer from PWSCC, and many had to be replaced.

Steam generator tube bundles constitute a primary side pressure boundary and hence are of particular safety significance for regulators and operators alike. Degradation of the steam generator tubing is strongly dependent on the materials used, the tube support design and the tube installation. Initially, the steam generator tubing in PWRs was made of nickel based Alloy 600. The first German steam generators designed by Siemens developed leaks caused by SCC after only two refuelling cycles. They were replaced by steam generators with titanium stabilized Alloy 800 nuclear grade tubing. Since the early 1970s, most German and Canadian steam generators have been tubed with nuclear grade Alloy 800 (I 800) material, and have given good long term performance. Today, AREVA NP and Babcock & Wilcox Canada continue to supply steam generators with Alloy 800 nuclear grade tubing.

In France, Japan, the Republic of Korea and the USA, steam generators use thermally treated Alloy 690 tubes. This alloy has more chromium (29.5%) than Alloy 600 (15.5%) and proportionally less nickel. It is very resistant to PWSCC and has also improved corrosion resistance in secondary side environments. Newly fabricated, thermally treated Alloy 690 tubes are subjected to thermal treatment at 705°C for 15 hours to relieve fabrication stresses and improve their microstructure. No cracking on thermally treated Alloy 690 steam generator tubes has been reported in operating power plants since their introduction in 1989 [50].

In CANDU practice, the steam generator tube material used in the 1960s was Monel 400, a high nickel–copper alloy. This alloy had good corrosion resistance, but it was extremely sensitive to oxygen content. Its ferromagnetic properties made it difficult to inspect with standard eddy current tools. In later units, Babcock & Wilcox Canada manufactured CANDU steam generators with titanium stabilized Alloy 800 nuclear grade tubes and a manufacturing method that precludes any random heat addition to the tubing.

Russian WWER-440 and WWER-1000 steam generator tubing is made of 08Kh18N10T material, which is also a titanium stabilized austenitic stainless steel with excellent corrosion resistance. Outstanding performance was obtained with steam generator tubing made by Vitovice HM in the Czech Republic, Finland, Hungary and Slovakia. Studies have shown that tube plugging rates strongly depend on operational procedures and practices. In some countries, periodic blowdowns, regularly conducted, result in a much reduced sludge volume in steam generators, with significant impact on the corrosion rates of steam generator internals.

5.7.2. Degradation mechanisms in steam generators

General degradation that raises safety concerns in steam generators involves mainly steam generator tubes and divider plates. Steam generator tubes constitute a delicate primary to secondary side boundary, given their reduced thickness. When tubes crack and leak, a loss of coolant event to the secondary side occurs. At that point, the secondary side has no nuclear safety barriers and may contaminate the environment. A mitigating factor is that steam generator tubes have a small cross-section and they are unlikely to fail in great numbers all at once. It is important to note that there has been almost no damage in the free span sections of steam generator tubing, where the tubes are in contact with turbulent water. Degradation has occurred in bends and at the tubing terminal points, affecting primary and secondary sides.

To partially offset this vulnerability, steam generators are designed with 15–20% more tubes than the thermal design and design margins strictly require; therefore, when tubes degrade, they can be safely plugged and isolated from active service. If penetration of the crack in a tube is below 40% of the wall thickness, the tube can be sleeved instead of plugged.

Sleeving is a technique applied to the region near the tube sheet. It is practised by inserting a short section of a slightly larger tube (the sleeve) into the steam generator tube to cover the degraded zone. The sleeve is then welded to the tube.

Much R&D effort has been undertaken worldwide in order to improve the steam generator design and materials used, as well as the secondary side water chemistry. Although progress has been made, steam generator tube degradation remains a continuing challenge. Understanding degradation mechanisms has been the most important factor in finding solutions. It was found that steam generator tubing corrosion depended on the interaction of three factors: design, materials and water chemistry.

Flow assisted corrosion (at the tube supports) produced significant thinning of the tube support plate (TSP) ligaments with carbon steel TSPs. FAC was detected at the top TSPs of the Gravelines 2, 3 and 4 units in France. FAC was also diagnosed as the mechanism responsible for the degradation of peripheral egg crates in the ABB/Combustion Engineering steam generators at San Onofre 2 and 3 and Maine Yankee nuclear power plants in the USA. FAC has also been diagnosed as the main degradation mechanism in moisture separators, feedwater headers, nozzles, water boxes and blowdown headers, among others.

In WWER steam generators, the prevalent degradation mechanism is outer surface SCC, owing to the ingress of impurities, mainly iron and copper oxides, through the feedwater flow caused by the degradation of the main condensate system equipment. Leaks or degraded condenser tubes can contaminate the secondary loop water that circulates through the steam generator and lead to steam generator tube degradation. Improved condenser materials (e.g. titanium tubes), better leak detection devices and better water chemistry minimize condenser and steam generator problems.

Other examples of degradation factors in steam generators include:

— Make-up water quality (output of the water treatment plant);
— Condenser leaks (composition of cooling water);
— Consumables and auxiliary products, which may enter into contact with inner surfaces (e.g. maintenance tools and substances);
— Impurities in the conditioning chemicals.

Problems with steam generator tubes have often forced costly unscheduled and extended outages with high personnel radiation exposure and loss of power. Consequently, when repairs became too onerous, steam generator replacements became necessary.

5.8. REACTOR COOLANTS AND INTERCONNECTED SYSTEM PIPING

Reactor coolant and interconnected system designs vary with the technology, the vendors and the reactor model type. A number of primary coolant piping components, especially cast components in LWRs in contact with the reactor coolant, are made of duplex stainless steels. This material displays higher strength than austenitic steels and higher resistance to pitting corrosion and SCC. In addition, its good weldability and low thermal expansion coefficient allow the specification of thinner walls, lower weight and, hence, lower cost. This superior performance is due to equal amounts of austenitic and ferrite phases in the alloy.

Duplex stainless steels are subject to phase precipitation of many compounds at temperatures above 300°C, especially in welds, if not perfectly executed or if the material has been cold bent or subjected to other mechanical stressors during the fabrication processes. A well known phase separation mechanism is that of spinodal decomposition, already discussed in Section 5.5.7, consisting of chromium migration forming chromium rich and chromium poor segregated ferrite phases, causing the material to become brittle. As ageing proceeds beyond 3000 hours of operation at temperatures above 300°C, as happens in PWRs, a secondary precipitation occurs in the chromium rich alpha phase, with separation of ferrite and carbide.

Studies have indicated that duplex stainless steels generate an electron flux in the presence of a heat flux and a temperature gradient. This property has allowed the development of instrumentation capable of measuring the thermal electric phenomenon and correlating to ageing, in support of diagnostics on the state of the ferritic phase and prognostics on the ageing progression and on the remaining life of duplex stainless steel materials. Other techniques to evaluate ageing effects in duplex stainless steels have also been developed based on measurements of the lattice strain of the alpha ferrite. Predictions for the integrity of the duplex stainless steel components of LWRs are particularly important in the justifications for the extended service life of these reactors [51–53].

5.8.1. Thermal stratification

The stratification of layers of coolant at different temperatures and temperature gradients in piping components may result in thermal fatigue degradation, particularly in dead legs where there is no flow during normal operation and some flow patterns set in during transients. This condition is observable in components including:

— Branch pipes connected to the main process piping in feedwater lines connected to steam generators in PWRs;
— The pressurizer surge line connected to both the inventory in the pressurizer at one end and to the RCS main piping at the other end;
— The injection lines of the emergency core cooling systems.

Stratification may also be caused simply by leaking isolation valves.

The most discussed case of thermal stratification has been that of the surge line. The pressurizer surge line experiences stratified flow corresponding to the operation of the heaters during RCS transients. RCS inventory swellings and shrinkages cause hot RCS to be transported into the relatively stagnant pressurizer surge line, where stratification of cold and hot fluid streams occurs, resulting in thermal stresses on the pipe wall [47].

Several through the wall cracks attributed to thermal stratification have been signalled in operating experience reports in Europe, Japan and the USA from WWER plants. As this phenomenon was unknown when most of the currently operating nuclear power plants were designed, thermal stratification was not considered in design calculations.

5.8.2. Thermal striping

Thermal striping is defined as the fluctuating temperature field that is imposed on a structure when fluid streams at different temperatures mix in the vicinity of a metal surface. Thermal fatigue failure due to thermal striping in nuclear power plants is not negligible, and organizations such as EPRI and the European Union (EU) have funded research to quantify failure margins and predict behaviours in the field. Benchmark studies were conducted by comparing the CFD code FLUENT against experiments. Road maps and procedures are available, but a consistent methodology has not been reliably proposed so far.

5.8.3. Thermal fatigue management

Tee connections (mixing tees), in piping with flows at different temperatures mixing in the tee junction, are exposed to thermal fatigue due to low and high temperature cycling against the metal. Thermal fatigue cracks may occur in the welds, in the base material, in elbows, or in the piping at or downstream of the tee. For quantifying failure margins and to make reliable life predictions of the high cycling fatigue incidence on the piping, the use of conventional instruments such as thermocouples has proven useless, given the high turbulence and arbitrary nature of the phenomenon. CFD modelling can be used to determine the thermohydraulic behaviour of the coolant, assuming that measurements of the model's boundary conditions can be made available or inferred from operation records and R&D projects. Thermohydraulic parameters can then be used as inputs to advanced fatigue and fracture mechanics analysis, guided by experimental evidence, to develop a predictive procedure.

5.8.4. Fouling

In general terms, fouling is the accumulation of foreign material on surfaces, such as the insides of pipes, steam generators or heat exchangers. In in-service water pipes and component cooling heat exchangers, particularly in 'once through' raw water systems, the fouling material is carried by the water. It can be organic material such as algae or bacteria, in which case, it is referred to as biofouling, because it is mainly of the biological sort and is carried from outside the system, usually an open body of water, through an intake canal or through an open cooling tower.

Biofouling is further subdivided into two types, macro-fouling and micro-fouling, depending on the foreign material unit size, on the type of wall adherence and on the speed of adherence. The rate of fouling is normally taken to be proportional to a quantity called the fouling factor, which is a function of temperature and varies with the type of liquid or slurry.

The fouling material could also be dissolved in the liquid, in which case, the majority of the offending substances are inorganic compounds such as calcium carbonate and other salts. Very often, the water is heated or cooled during the process, which causes impurities in the water to precipitate and stick to the inner surfaces. In time, this material accumulates inside system components until fouling of plugs, pipes, heat exchanger tubes and other tubular components occurs.

Having less scaling or corrosion products in a pipe gives the fouling material less hold on the inner surface of a pipe. Steam generators, heat exchangers and steam turbine condensers need to have cleaning and ageing programmes of their own. The component cooling heat exchangers, especially those in open ended service water cooling applications, if serving safety related users, need to be checked periodically for fouling. Temperature measurements at the inlet and outlet streams need to be used periodically to infer the acceptability of the heat exchanger performance for the remainder of the maintenance cycle. Regulators usually require nuclear power plant operators to monitor heat exchangers and other components critical to safety and demonstrate that the accumulation of fouling is not going to affect the availability of the safety loads for the remainder of the operating cycle, which normally coincides with the fuelling cycle for LWRs. In the case of PHWRs, which use on-line refuelling, the operating cycle coincides with the inspection/maintenance cycle.

5.8.5. Water hammer and steam hammer

A water hammer is a pressure surge that can arise in any pressurized system that undergoes an abrupt change in its flow rate from starting and stopping of pumps, from the opening and closing of valves, from changes in flow direction or from water column separation and collapse. These events cause all or part of the fluid mass to undergo an abrupt momentum change. This change can produce a shock wave that travels back and forth between the cause that created it and a resistance point in the system such as a restriction orifice, a valve downstream or a number of elbows in between. If the intensity of the shock wave is high, physical damage to the system can occur. Water hammer can be interpreted as an application of the principle of conservation of energy because it results in the conversion of the flow of a fluid or momentum energy into pressure energy. As liquids have low compressibility, there is little attenuation and the resulting pressure energy tends to be high. Water hammer may affect piping networks in both conventional and nuclear systems.

A variation of the phenomenon is steam hammer. This is usually created by water slugs and steam bubbles entrapped and collapsing in piping systems containing subcooled liquid. Ideally, piping systems should be designed to absorb pressure transients or attenuate them.

Water, or steam, hammers have been experienced in PWR steam generator sparger feed lines, when voids formed during operational transients. Water/steam hammers have also been experienced in cooling water networks, after a system drain. A cooling water system typically serves heat exchangers and other heat sinks throughout the plant. If voids form at higher elevations and pumps are restarted after maintenance activities, air, vapours and other compressible gases form bubbles at high elevation. These bubbles suddenly collapse when the system is pressurized and a consequent water/steam hammer event occurs.

5.8.6. High energy pressure breakdown orifices

High energy multistage orifices are used where a flow reduction, but not full control over the entire flow range, is necessary. One typical application of high energy breakdown orifices is found in minimum flow recirculation lines to protect costly multistage centrifugal pumps. A typical application in nuclear power plants is the charging or feed pump in the primary coolant inventory control system. These pumps produce high discharge pressures, and their discharge line is usually fitted with a control valve that could be 100% closed in specific system alignments. Without overpressure protection, this would produce a condition that could damage the pump internals. In such cases, designers usually provide a pump recirculation line, drawing a minimum flow from the pump discharge line between the pump and the control valve. The recirculation line would typically contain a multistage pressure breakdown orifice to protect the pump and the system against overpressure, as orifices cannot be inadvertently closed. Pump manufacturers generally recommend a minimum recirculation flow between 15% and 40% of main flow, depending on the pump characteristics.

During the first fill-up of the reactor coolant circuit and interconnected systems, when the RCS being filled up is still at atmospheric pressure, these high energy multistage orifices are subjected to the maximum differential pressures of the system. Typically, the design of these orifices is optimized for normal operating conditions, not for maximum differential pressures. Under such severe conditions, the orifice and the line downstream may experience cavitation and vibration during the RCS system fill-up time. If the required minimum flow is small, designers tend to use small recirculation lines sizes, and if the line is much smaller than the main pump discharge line and ties into the main line by means of weldolets, vibration may damage the tie-in points, the small recirculation line and its supports.

Another even more destructive condition may be the onset of acoustic resonance. If the orifice emits an acoustic frequency that is too close to the natural frequency of the system, a Helmholtz resonator may set itself in, and the tie-in points of the recirculation line may break during pump startups or other operating transients. A permanent design remedy in such cases is to resize the recirculation line, provide more robust or tighter supports and replace weldolets with strong reducing tees.

High energy multistage pressure breakdown orifices can also become a problem as they age if erosion and corrosion affect their internal geometry. An AMP should be planned for these orifices and their recirculation lines. Monitoring of high energy orifices can take the form of velocity transducers and flow or pressure differential measurements. ISI of orifices during reactor shutdowns may include ultrasonic testing or even radiography [54].

5.8.7. Pump and valve cavitation

Cavitation can be an issue in pumps and control valves and can cause accelerated ageing, and damage the components and the piping in the region. If remedies are not applied, severe cavitation can render the component no longer repairable.

In order to characterize the risk of void creation and cavitation in a centrifugal pump, it is customary to define a parameter called the net positive suction head (NPSH). The NPSH is the difference between the total head on the suction side of the pump at the impeller and the liquid vapour pressure at the operating temperature. There are two classic ways to look at the NPSH: from the system viewpoint (the NPSH available), or from the viewpoint of the pump (NPSH required). With the NPSH quantities at hand, the cavitation prevention rule can provide insights into finding solutions to cavitation issues.

The original system design should have been delivered cavitation free, but also any design changes that may have been undertaken during construction, commissioning or operation should be verified for cavitation prevention. If inspection or monitoring during operation detects vibration, or other abnormal symptoms, the system engineer should review the system operating history and verify that the system configuration has not changed due to flow induced damage, ageing, fouling or clogging. In principle, any change to the operating parameters defining NPSH may cause a crossing of the cavitation threshold for the pump, which may, in turn, cause cavitation and induce vibrations and operating abnormalities.

Cavitation can similarly occur in a valve when the fluid static pressure drops below its vapour pressure at the operating temperature. The pressure can drop suddenly as the fluid is forced to accelerate in a control valve throat, where the fluid cross-section or its hydraulic diameter is reduced. The best approach to avoiding cavitation in valves is to size them for appropriate fluid velocities, typically less than ~400 mm/s. Control valves and pumps should, in general, be located in the lower regions of the systems to maximize the static head, especially in systems operating at higher temperatures such as the boiler feed pumps taking suction from deaerators and handling reheated condenser condensate mixed with preheated make-up streams. Pressure relief valves may be subject to chattering, which is the rapid opening and closing of the valve seat. This results in vibration, which may cause misalignment and valve seat damage and eventual mechanical failure of the valve internals and associated piping. Chattering may be caused by excessive inlet pressure drop and excessive backpressure. Chattering may also happen if the relief valve is oversized or if the valve is subjected to widely varying flow rates. Other causes could be inlet pipes being much smaller than valve inlets, obstructions in the inlet lines or inlet lines that are too long. Usually, valve chattering can be resolved with design changes.

5.8.8. Acoustic vibration

Early BWRs experienced significant acoustic resonance in their reactor and steam systems. In some cases, acoustic resonance resulted in damage to plant components in RPVs and steam lines.

In general, all nuclear power plant designs have, at one point, experienced excitation of acoustic standing waves in closed side branches, for example, safety relief valve side branches in the steam system, due to vortex shedding generated by the steam flow in the main steam lines. The amplitude of the acoustic pressure waves can be several times higher than the dynamic pressure present in the system. The acoustic waves propagate in the steam lines, eventually reaching sensitive components such as steam dryers and turbine stop valves. In addition, acoustic waves may cause vibration and lead to complications such as valve seat wear. Therefore, the structural components are subjected to high cycle fatigue loads, which may severely impact their functionality and safety.

As a result, nuclear power plant operators evaluate potential adverse flow effects during initial plant design and the startup of new reactors, or following major design changes, such as implementing power uprates of operating reactors. Turbulent steam flow in RPVs can excite large scale, low frequency acoustic modes. Steam flow in safety valves, safety relief valves and other branch lines can generate high frequency acoustic modes in the system. Acoustic resonance has caused severe vibration of steam line components.

In the USA, in June 2002, the steam dryer cover plate of a BWR failed after 90 days of power uprate operation (117% of original power) at the Quad Cities Nuclear Power Station, Unit 2. In June 2003, the steam dryer hood failed after an additional 300 days of power uprate operation. In March 2004, the steam dryer cracked after an additional 8 months of power uprate operation. In November 2003, the steam dryer hood failed in Unit 1 after 1 year of power uprate operation (117% of original power). The 150 mm × 230 mm plate of the steam dryer outer bank was lost in the reactor coolant and steam system, and damage was also found to several steam line components. The Unit 2 steam dryer was instrumented with pressure sensors, strain gauges and accelerometers. The steam lines were instrumented with strain gauges to calculate the pressure load on the Unit 1 steam dryer based on the Unit 2 benchmark.

In late 2005, the safety relief valves experienced short circuiting at Quad Cities Nuclear Power Station, Unit 2. Power was reduced for inspection and broken actuator parts were found. Both BWR units were shut down for inspection, and extensive damage was observed in several safety relief valves. It was found that the damage was caused by severe vibration from acoustic resonance. As a result, the owner/operator initiated a programme to eliminate acoustic resonance from the BWR steam lines. Analysis indicated that the safety valve and safety relief valve branch line length had not been reduced sufficiently to cause acoustic resonance to occur at steam velocities beyond the operating conditions. To address the issue, the owner/operator installed a T-connection and dead leg pipe in the safety valve and safety relief valve branch lines to effectively lengthen the branch line and cause acoustic resonance to occur at lower steam velocities with reduced pressure fluctuations. Upon reactor restart, measurements indicated that the sources of acoustic resonance had been eliminated.

The NRC regulatory guide 1.20 [55] was revised to provide guidance on potential adverse flow effects. Operating BWRs proposing a power uprate, and new BWR applications, are requested to evaluate potential adverse flow effects from acoustic resonance. Vendors have developed proprietary methods using data from the replacement steam dryers and steam lines to determine pressure loads and stresses on steam dryers. In addition, BWR licensees are requested to monitor their steam dryer and steam line during power ascension, and implement a steam dryer inspection programme during refuelling outages [54–58].

5.8.9. Emergency feedwater systems

In PWRs, the emergency feedwater system, also called the auxiliary feedwater system, is used to supply make-up water to the steam generators when operated as heat sinks to remove decay heat, when the reactor is shut down. The system is equipped with a number of throttling valves to adjust the flow to the steam generators during refuelling and reactor startup and to isolate the pumps and other components during maintenance. This system piping has displayed wall thinning in US plants. Throttled flow operation and FAC were determined to be the root cause. Through the wall erosion was found at the inlet end of the A106-B crossover pipe and at the tee. High crossover piping velocity of 11 m/s at 245°C and 4.74 MPa was postulated to be the root cause. It is believed that much of the damage may have occurred during early operation at reduced pH. Under high pH conditions with morpholine water treatment, erosion is greatly reduced. As a corrective action, the emergency water system and residual heat cooler headers and their thermal sleeves were replaced by more erosion resistant components (e.g. SA-335, 1-1/4 chromium material).

In Korean PWRs, wall thinning of the lower body of the main feedwater isolation valve was experienced. Using ultrasonic testing measurements, numerical analysis and CFD, it was determined that FAC was the root cause of the event.

5.8.10. Buried piping

Underground or buried piping and tanks are used in both safety and non-safety applications in nuclear power plants. They may contain liquids or gases. Buried piping or tanks are below ground and in direct contact with the soil. Underground piping and tanks, even though they are also below ground, are not kept in direct soil contact, as in the case of piping contained in an underground concrete duct.

Public concern was expressed in some jurisdictions following cases of radioactive or environmentally deleterious material leaks from buried piping systems at nuclear power plants (see Fig. 35). Although releases are typically within prescribed limits, such releases can negatively affect the reputation of the affected nuclear power plant. Additionally, leaks from buried piping can cause economic losses to plants, as they may require a unit outage and compensatory measures.

There are differences between underground piping in a nuclear power plant and those in conventional industries such as oil and gas, or municipal water systems. Pipelines used in conventional industries are typically straight, long and electrically isolated, with a much simpler infrastructure. In contrast, underground pipes in nuclear power plants are made of a variety of materials, are often shorter in length, and span a far greater range of diameters with overcrowded infrastructures. In addition, they are electrically interconnected and grounded for safety reasons. Owing to the distinct design complexities of underground or buried piping in nuclear power plants, in order to control ageing mechanisms, special considerations need to be given to the conditions of the materials and of their coatings [59], as well as to the necessity of scheduling periodic inspections, with suitable inspection techniques.

Owing to the high cost of digging up buried piping for open air inspections, nuclear power plants often take a risk informed approach to selecting inspection locations with high probability of detection, and utilize available commercial software tools for assistance in the selection. There is an increasing trend to leaving diagnostic equipment, such as guided wave collars, installed underground with test leads going to the surface to avoid repeat diggings over the operating life of a nuclear power plant.

5.8.11. Secondary side piping in pressurized water reactors

In FAC, as seen in Section 4, it is the constant creation and dissolution of the surface oxide layer that gives rise to metal wall thinning, which may ultimately cause perforations. The corrosion rate is proportional to the solubility of the oxide layer, and depends on:

- Chromium, molybdenum and copper content;
- Operating temperature;

FIG. 35. Occlusion of service water piping at Catawba nuclear power plant in the United States of America [60].

— Flow velocity;
— Surface roughness;
— Fluid pressure profile;
— Liquid droplet impingement;
— Presence of cavitation.

Carbon steel components on the secondary side, such as feedwater heaters, condensers, valve bodies and piping, have experienced severe wall thinning in nuclear power plants. Nuclear regulatory bodies of various countries require utilities to create wall thinning management programmes for carbon steel components. Pipe wall thinning can cause steam explosions in accessible areas of a nuclear power plant, and can cause fatal accidents that are severe in terms of human cost. A steam explosion following a condensate pipe break event occurred in the Surry plant in the USA in 1986, and four workers were killed (see Fig. 36).

Another quite similar fatal accident occurred at the Mihama plant in August 2004, in Japan. A pipe break occurred in a 56 cm carbon steel straight run of condensate pipe, ~30 cm downstream of a flow orifice (see Fig. 37). Steam released from the break killed five workers and injured six others.

In the period between 1986 and 2002, in the Russian Federation and in the Ukraine, 45 cases of operational failures in nuclear power plants were the result of FAC in pipelines:

— WWER-440 nuclear power plants: 19 failures;
— WWER-100 nuclear power plants: 17 failures;
— RBMK (high power channel type reactor) nuclear power plants: 7 failures.

Many operating plants have conducted reviews of their high energy piping systems. At the Finnish Loviisa nuclear power plant, which consists of two WWER-440 reactor units dating back to the late 1970s and early 1980s, an FAC programme review was conducted after the Surry accident in 1986 with the intent of incorporating the immediate feedback from that event. As the review was limited to feedback from the Surry accident, flow control orifices were not included in the review programme. A guillotine break of the feedwater system piping occurred in 1990 in Unit 1 of the Loviisa plant. A second guillotine break of the feedwater system piping occurred in 1993 in Unit 2. In order to reduce FAC in the feedwater system piping, the secondary water chemistry was changed from neutral to alkaline. Also, a piping material change from carbon steel to low alloy steel or stainless steel was carried out. Changing piping materials and water chemistry, coupled with on-line monitoring, have since improved resistance to FAC.

FIG. 36. A 460 mm elbow in a condensate line ruptured catastrophically in Surry Unit 2 nuclear power plant, United States of America [61].

FIG. 37. A 56 cm catastrophic rupture in the condensate line downstream of an orifice in the Mihama nuclear power plant in Japan [61].

EPRI first began researching FAC in response to the Surrey incident. Much has been learned over the years. Today, the main engineering tool used in FAC programme is CHECWORKS, an EPRI code that predicts the rate of pipe wall thinning due to FAC. Operators have made conservative decisions to move their inspections ahead and to add new inspection locations. Another popular code is COMSY, by AREVA, used in ageing and PLiM of mechanical components in power plants. As in the case of CHECKWORKS, the degradation analysis functions include lifetime prediction with respect to FAC, droplet impingement corrosion and cavitation corrosion, among others. The introduction of specialized computer codes has considerably increased the capabilities of the inspection programmes of FAC prone systems.

In Japan, utilities have studied the wall thinning phenomenon using the large amount of measurement data available on feedwater and steam systems. One of those utilities, Kansai Electric, reviewed data taken on about 30 000 components in its nine PWR plants. Upon completing a statistical evaluation of the data produced by Kansai Electric and other plants in similar reviews, the Japanese utilities issued, in 1990, guidelines for the management of pipe wall thinning in PWR secondary systems, in which were stipulated the inspection method, the inspection targets, their frequency, acceptance criteria and corrective actions (e.g. pipe replacement), among others. Wall thickness is normally measured by means of ultrasonic testing equipment. Pipes found to have a remaining service life of two years or less require corrective action.

Quality standards today recommend that all plants should have a modern FAC programme covering all high energy secondary side systems.

5.8.12. Low voltage cables and instrumentation and control systems

Cable ageing and condition monitoring of cables are important aspects of the PLiM of cables and I&C systems. Cables, especially their insulation and jacket material, are vulnerable to material degradation during normal operation. Means need to be established to ensure that cable ageing does not lead to unsafe operation. Cable fires, moisture intrusion, loss of functionality and sensitivity to electromagnetic noise/interference are examples of cable issues giving rise to current and future concerns about long term performance of cables in nuclear facilities.

Low voltage cables in nuclear power plants can typically be grouped into the following functional types:

- I&C cables (e.g. coaxial, triaxial, twisted and shielded);
- Low and medium voltage power cables (<1 kV);
- Specialty cables (e.g. articulating cables to move equipment close to the reactor or stationary cables subjected to high radiation levels);
- General service cables (e.g. ground cables, telephone cables, etc.).

Low voltage power cables are used to supply power to I&C safety grade components and low voltage auxiliary devices, such as small motors, motor control centres, heaters and small transformers. The main components of a low voltage power cable are a conductor, insulation, shielding and a jacket (see Fig. 38). Other components that may be present in a cable include:

- Filler or bedding materials, which occupy the gaps between insulated conductors in multiconductor or multicore cables to improve the mechanical stability of the cable;
- Tape wraps, which may provide additional electrical, mechanical and fire protection, or which simply signify the identification of the conductor grouping;
- Armouring layers below the outer jacket, which are sometimes used for mechanical protection.

Instrumentation cables (including thermocouple extension wires) normally carry low voltage and low current analogue or digital signals used by instruments. Resistance temperature detectors, pressure transducers and thermocouple circuits are normally made up of twisted pairs of shielded leads. Radiation detection and neutron monitoring circuits often use coaxial or triaxial shielded configurations.

Substantial research has been conducted by cable manufacturers, reactor vendors, laboratories and standard development organizations, such as the International Electrotechnical Commission (IEC), to understand the effects of ageing on cables and to establish an effective and reliable means of implementing cable qualification and condition monitoring.

Cable stressors in the field include exposure to radiation, environmental heat, ohmic heating, humidity, vibration, contact with chemicals and mechanical interference, which produce bending and squeezing. While cable degradation is typically associated with the embrittlement of the jacket or of the insulation material, the conductors can also be affected by degradation. In addition, ageing may induce the loss of equipment qualification requirements such as:

- Reduction in the insulation resistance at elevated temperatures (above 70°C);
- Gassing and interference of gaseous products with surrounding surfaces;
- Formation of conductive salts on the surface of insulated conductors.

In order to guard against the adverse consequences of cable ageing and degradation, a cable AMP should be included as part of the normal maintenance schedule in nuclear power plants. This should be mandatory for cables that serve safety related roles. A number of testing techniques have been developed to diagnose cable health conditions.

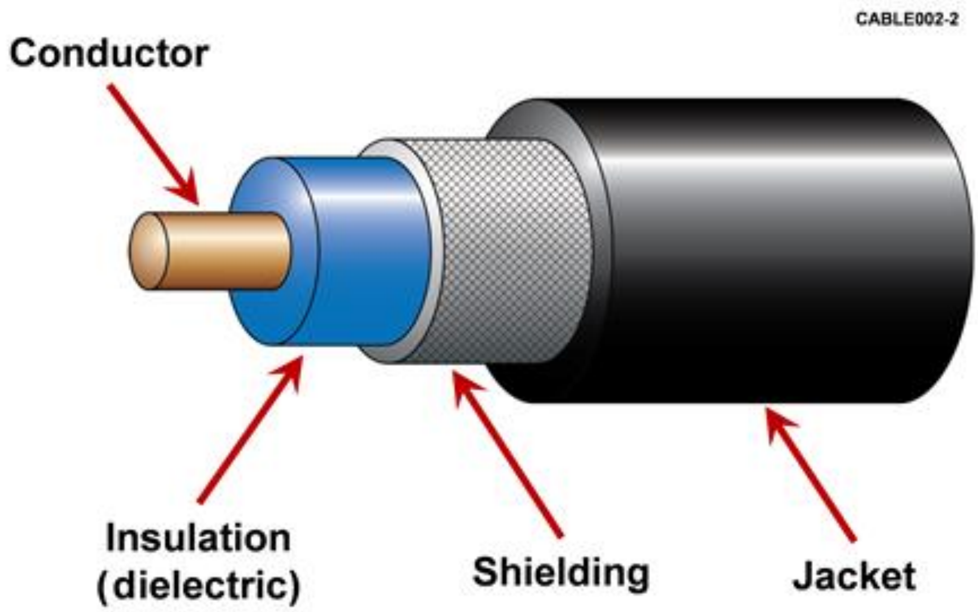


FIG. 38. Schematic of a typical cable.

5.8.12.1. Cable testing

The nuclear power industry has recognized that there are limitations in conventional ageing testing tools when applied to cabling such as pre-ageing techniques and the use of models, for example, the Arrhenius law for assessing qualified life. This is the main reason that hard wired periodic cable testing in nuclear power plants is still recognized as vital for troubleshooting. Periodic testing can easily identify problems such as signal anomalies. In addition, pre-operational testing is needed to establish baseline measurements for the evaluation of cable ageing and as a reference for predictive maintenance planning.

A wide range of cable testing methods has been developed for cable ageing management. Available cable testing techniques have been used to test all parts of a cable (insulation, jacket, shielding and conductor) as well as connections, penetrations, splices and terminations. In the broadest terms, there are four types of testing techniques (visual/tactile, electrical, mechanical and chemical) for cable condition monitoring, life management and troubleshooting, as shown in Table 3.

5.8.12.2. Cable acceptance criteria

Acceptance criteria for cables are needed for condition based qualification of insulating materials and for maintenance purposes. Considering that electrical properties do not change significantly with ageing, specific characteristics that do change with ageing should be determined and used to define acceptance criteria. It has been found that cable elongation, which is the measurement of how far the material will stretch before breaking, proportionately diminishes with ageing. Elongation is an example of a property that can be easily measured and correlated to the insulation resistance of ageing stressors and mechanical loading such as vibration, bending or other movements. This property has been used to set an acceptance criterion (e.g. a certain amount of minimum elongation expressed as a percentage of the ultimate elongation in relation to the years of continuous operation). An acceptance criterion of 50% of the ultimate elongation has been used for some materials. Elongation criteria effectively provide a quantifiable margin from unacceptable conditions, such as cable cracking on bending.

The IAEA published a nuclear energy series publication in 2012 containing information on state of the art cable qualification, condition monitoring and ageing management [62]. Both the Institute of Electrical and Electronics Engineers (IEEE) and the IEC, which produce the international standards and conformity assessment guidelines for all fields of electrotechnology, have developed many standards providing guidance on different methodologies for cable testing and diagnostics, including standards on:

TABLE 3. CABLE CONDITION TESTING TECHNIQUES

Technique	Monitoring type	Description	Destructive / Intrusive	Features	Remarks / Applicable materials
Walk downs	Visual and tactile	Capable of identifying cable conditions by appropriate manipulation and observation; this procedure should be considered for detailed testing	No / No	Simple and low cost method applicable to all materials with the benefit of obtaining immediate information	Needs to be carried out by skilled technicians / All materials
Thermal imaging / thermograph	Visual and tactile	Measurement used during walk downs for identifying environmental hot spots and unanticipated operating conditions	No / No	Supplements the plant walk downs to check cable degradation due to a localized hotspot or unanticipated operating condition and to provide a snapshot of the plant at intervals (e.g. every 10 years) through the plant life	Has significant limitations for use as a diagnostic tool as internal hot spots are not detectable and technique cannot be used for locating external hotspots during the plant full power operation / All materials
Elongation at break	Mechanical	Benchmark property of the cable by tensile testing	Yes / Yes	Although impractical as a routine condition monitoring technique, it generates optimal data for cable condition assessment and is particularly useful where cable samples have been placed in a sample deposit, specifically for condition monitoring	50% absolute elongation at break normally defines the end-of-life condition based on a conservative estimate of aged cable surviving a design basis accident / All materials
Indenter modulus	Mechanical	Measurement carried out by clamping the cable jacket or insulation while the probe penetrates the surface of the test material by a few hundred micrometres; the indenter modulus is a parameter associated with the specific compressive stiffness of the tested material calculated by the depth of penetration of the probe against the force exerted on the probe	No / Mainly no	Only provides information on the cable condition at the location being tested and could be used on operational cables	A probe with known dimensions should be driven through the surface of a sample at a carefully controlled speed / Polyvinyl chloride (PVC), chlorosulphonated polyethylene (CSPE), ethylene propylene rubber (EPR), ethylene propylene diene monomer
Recovery time	Mechanical	Measurement of the time to recover a set deformation resulting from prior indentation, made during the post-indentation phase, following a force relaxation phase, and upon retraction of the indenter probe	No / Mainly no	Higher sensitivity and stronger correlation with elongation than that of the indenter modulus	Proven to be very useful to assess the degradation of cables / All materials

TABLE 3. CABLE CONDITION TESTING TECHNIQUES (cont.)

Technique	Monitoring type	Description	Destructive / Intrusive	Features	Remarks / Applicable materials
Frequency domain reflectometry	Electrical	A non-destructive cable testing technique based on the transmission line theory that uses a swept frequency signal transmitted through an electrical cable circuit and analyses the reflected impedance changes in the circuit; these reflected signals are measured in the frequency domain and then converted into the time domain using an inverse Fourier transform	No / Disconnection needed	The reflected signal can travel through miles of cable without attenuation, provided the cable under test is shorter than the frequency domain reflectometry signal wavelength	The behaviour of a transmission line, part of an electric circuit, depends on its length in comparison with the wavelength of the electrical signal travelling through it / All materials
Time domain reflectometry	Electrical	Technique based on the transmission line theory just as frequency domain reflectometry; it involves sending a direct current pulsed signal through a cable circuit and measuring its reflection to identify the location of any impedance change in the cable and in the end device (load)	No / Disconnection needed	Allows cable diagnostics by providing information about the conductor and any connector or connection in the circuit, and to a lesser extent, about the cable insulation material and its insulation resistance	Allows diagnostics about a device at the end of the cable, such as a resistive temperature detector or thermocouple; the test depends largely on comparisons with baseline time domain reflectometry data / All materials
Inductance, capacitance and resistance (LCR)	Electrical	Impedance measurements, namely inductance (L), capacitance (C) and resistance (R), are made using an LCR instrument at specific frequencies to verify the characteristics of the cable conductor, insulating material and end device	No / Disconnection needed	For detecting imbalances, mismatches, or unexpectedly high or low impedances between the cable leads (due to cable degradation and ageing) to faulty connections, splices or to physical damage	Measurement results are evaluated to determine if they are as expected for the type of circuit being tested / All materials
Insulation resistance	Electrical	Measurements are made using an insulation resistance instrument at specific voltages to validate the cable insulating material characteristics	No / Disconnection needed	For detecting expected insulation resistance changes due to ageing phenomena such as oxidation either through moisture intrusion, or through other environmental effects	Typically, a voltage lower than the maximum rated voltage of the cable is applied to an inner conductor or to the cable shield (if applicable) and to a ground plane in contact with the cable Current in the cable is limited to avoid cable damage / All materials

TABLE 3. CABLE CONDITION TESTING TECHNIQUES (cont.)

Technique	Monitoring type	Description	Destructive / Intrusive	Features	Remarks / Applicable materials
Tan delta	Electrical	Monitoring integrity of the cable insulation by measuring the tangent angle (loss) between the resistive current and the capacitive current under alternating current voltage	No / Disconnection needed	Used to detect resistance current increases due to impurities or inclusions in the insulation	Methods include using a fixed frequency such as 60 Hz and stepping the alternating current voltage up to 1.2 times the rated cable voltage or applying a low voltage with a variable frequency up to 20 kHz / All materials
Density	Chemical	Measurement of density of polymer samples using the Archimedes principle, which states that the buoyant force that is exerted on a body immersed in a fluid equals the weight of the fluid that the body displaces; the polymer density can be obtained by measuring the weight of a specimen both in air and in a liquid of lower density than the specimen under test or by using a density gradient column (placing the specimen into a calibrated liquid column and deriving the polymer density from a known calibration curve once equilibrium is reached)	No / Micro-sampling required	Results are known to correlate with elongation at break	The higher the level of ageing, the greater the concentration of oxidation products and the higher the density of polymer samples / Cross-linked polyethylene (XLPE), other thermoplastics and some elastomer EPR
Ultrasonic velocity	Chemical	Measurement of the change in the velocity of sonic propagation in cable material due to changes to the elastic modulus and density as a result of ageing in cable materials	No / No	Still under development, but designed to measure the properties of the cable jacket over a small volume between the transducer probes	Strongly dependent on the cable construction and on the specific formulation of the jacket material; extensive baseline data may be required / PVC, polyethylene (PE), EPR
Oxidation induction time (OIT) / oxidation induction temperature in polymers (OITP)	Chemical	Measures the loss of antioxidants in a polymer as it degrades. OIT measures the time at which oxidation occurs when exposed to a constant temperature; OITP detects the temperature at which oxidation occurs when temperature is increased at a specified rate in an oxidizing environment	No / Micro-sampling required	Standardized for use in condition monitoring and has been shown to correlate well with degradation of certain polymers (e.g. OIT for PE and EPR and also CSPE and polychloroprene (PCP))	Degradation products from halogenated materials can damage expensive calorimeter cells, and continued multiple testing on this kind of material is impractical, unless specific corrosion resistant cells are used / EPR, PE, XLPE, PVC

TABLE 3. CABLE CONDITION TESTING TECHNIQUES (cont.)

Technique	Monitoring type	Description	Destructive / Intrusive	Features	Remarks / Applicable materials
Thermal gravimetric analysis	Chemical	Measures changes in weight relative to changes in temperature	No / Micro-sampling required	Particularly useful in diagnostics of cable insulation ageing in that it allows mapping of the degradation temperatures; sample preparation is similar to that in OIT / OITP testing; usually carried out on samples that involve corrosive degradation products as the sample chambers are chemically far more robust than those used in differential scanning calorimetry	The reference temperature is determined by one of two methods: determining the temperature at which 5% mass loss occurs or the temperature corresponding to the point at which the maximum rate of weight loss occurs / CSPE, PCP, polytetrafluoroethylene, PVC, EPR
Fourier transform infrared (FTIR) spectroscopy / near infrared reflectance (NIR)	Chemical	Polymers change their light absorption characteristics owing to the formation of new chemical bonds as they degrade; measurement of the FTIR is based on infrared radiation passed through a sample, such as a cable insulation cut; NIR is measured by means of a portable near infrared spectrometer, in which the infrared analysis in reflectance mode is carried out based on detecting the development of ageing induced infrared absorptions	No / No	A sample both absorbs and transmits parts of the spectrum; the spectral signature of a sample is used to quantify physical changes to the insulation characteristics during the ageing process, such as molecular quality, consistency and chemical composition; the measurement actually detects light absorbance due to ageing	Not applicable to polymers that contain heavily absorbent materials such as carbon black, e.g. CSPE and PCP

— Cable testing;
— Specific test methods for cable condition monitoring;
— Cable conductors;
— Cable insulation and jacket material;
— Qualification of equipment and cables used for Class 1E safety functions.

Any I&C cable installed in a nuclear facility for the purpose of supporting a safety function needs to be qualified to perform not only during normal service conditions but also during and after a design basis event.

5.8.13. Ageing management programmes for concrete and non-metallic structures, systems and components

5.8.13.1. Containment designs

The containment system is primarily designed to contain any radioactive material that may be released from the primary system in case of accident, to protect the nuclear systems from external threats such as missiles originated by earthquakes, tornadoes, wind and, in some cases, aircraft impact, and to act as a supporting structure for operational equipment (e.g. cranes). Most containment buildings are constructed of either steel or concrete, with a steel or epoxy liner for leak tightness. Typical metallic liners are made of carbon steel plate with thicknesses of up to 25 mm. The plates are joined by welding, and anchored to the concrete wall by studs, structural steel or other steel products. Containment structures are usually seismically qualified. They have a large, thick, base mat that supports the structures above it, and which helps to contain molten material, if a severe accident should ever happen.

The technical definition of nuclear power plant containments includes not only the structural reactor building shell and the base mat, but also all wall penetrations, air locks and any other component with a containment boundary function. Any evaluation of containment integrity needs to also include an evaluation of these.

Maximum allowable containment leakage rates are established in the SAR and in the plant technical specifications or equivalent documentation. Leakage rates are expressed in terms of a percentage of the total containment atmosphere mass (or weight) leaked over a 24 hour period.

5.8.13.2. Ageing management for concrete structures

The primary objective of an AMP for concrete structures is to ensure the timely detection and mitigation of degradation that could impact its safety functions. As for all safety related SSCs, a comprehensive understanding of concrete ageing, of the degradation mechanisms and of their impact on the structure's ability to perform design functions is fundamental in the AMP.

AMPs typically call for periodic inspections or structure monitoring with remedial measures being implemented, to deal with any observed degradation, before functionality is lost. For structural parts where detection of the degradation would be difficult, or repairs costly, monitoring of the environment or of potential stressors that could lead to degradation is mandated.

5.8.13.3. Ageing mechanisms for concrete structures

Most concrete structure problems are the result of design and construction errors. Quality control and quality assurance programmes at nuclear power plants are necessary to ensure production of high quality concrete (e.g. material selection, batching, mixing, placing and curing).

Cracks due to settlements, or other conditions causing the concrete to exceed its tensile strength or to lose surface material, may not be related to ageing. They are nevertheless significant, because they can hide major structural problems. Ageing mechanisms attacking concrete structures include:

— Salt crystallization produced by repeated wetting and drying cycles;
— Freezing and thawing attack in saturated conditions;
— Abrasion, erosion and cavitation causing progressive loss of material in concrete surfaces;
— Thermal exposure and thermal cycling (above 350°C there can be a decrease in strength);

— Irradiation (neutron and gamma rays);
— Fatigue and vibration;
— Chemical attack including efflorescence, leaching, sulphate attack and ettringite[5] migration.

The relationships among the root causes of concrete degradation and types of cracks before and after hardening are shown in Fig. 39. Primary causes of cracks are shown on the left and the relationships among primary causes and degradation types are listed in the table on the right.

Primary causes are grouped into two categories: cracks due to intrinsic (rheological) concrete properties and cracks caused by external loading [63, 64].

It is interesting to note that degradation may not always be due to ageing. Cracks originating from the rheological properties of reinforced concrete are either normally formed during the hardening process or they are due to the intrinsic discontinuities of the mixing elements of the aggregates and to the presence of totally dissimilar materials, such as the reinforcing steel bars. Cracks caused by the rheological properties can follow varying patterns. Some may be confined to the surface region, others may follow the reinforcing bar pattern and others may be influenced by the distribution of the aggregates.

5.8.13.4. Inspection and monitoring of concrete structures, liners, coatings and containment penetrations

Concrete is a durable material and its performance in nuclear power plants has been good. However, experience has shown that accelerated degradation of concrete may occur with poor quality construction, aggressive environments, excessive operational loads and extreme accident conditions. An impaired concrete structure increases the risks to public health and safety. Effective ageing management of concrete containment buildings and other concrete structures (including periodic inspection, testing, surveillance, preventive and corrective maintenance programmes) is therefore required to ensure their fitness for service throughout the plant life.

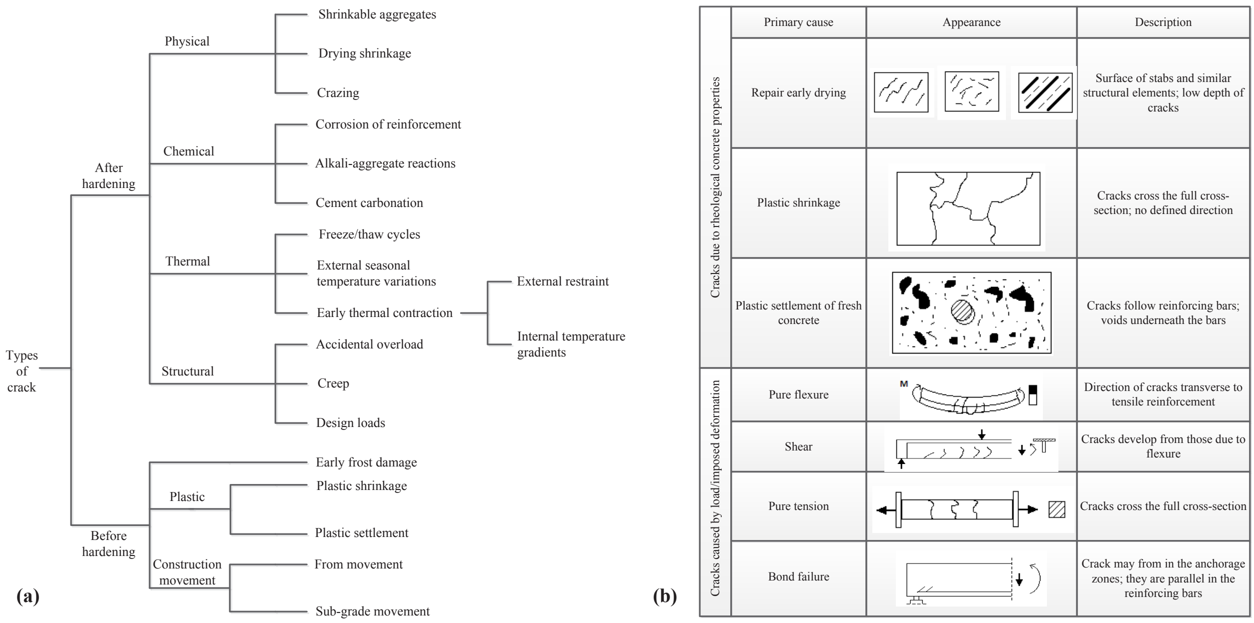


FIG. 39. Relationships among primary causes and types of cracks in concrete.

[5] Ettringite is the mineral name for calcium sulphoaluminate, which is normally found in concrete structures. Sulphates react with calcium aluminate in the cement to form ettringite within a few hours of making the concrete mix. Ettringite formation is the chemical reaction responsible for stiffening. Stressors on the concrete such as severe thermal transients and chemical attacks cause ettringite to dissolve and leave its original location to recrystallize in voids or cracks, causing concrete deterioration.

The scope of an AMP for concrete structures should cover the degradation of all potential ageing mechanisms through inspection and monitoring. The findings should lead to correct diagnostics and condition assessments supported by operating feedback and R&D, where available. Remedial action should be established and planned for the reactor building and all interconnected structures, such as:

— Mild steel reinforcing;
— Pre-stressing systems (where applicable);
— Containment penetrations;
— Liners (metallic and non-metallic);
— Anchor systems;
— Foundations and piles;
— Water stops, seals, gaskets and protective coatings.

Risk informed techniques can be applied to select particular areas or components that are representative of conditions in a predefined large target area [65, 66].

National codes and standards usually require a periodic assessment of the containment integrity and its leak tightness. This is carried out using visual inspections for corrosion damage, leakage rate and post-tensioning testing, where applicable. Visual inspection, other than the naked eye, may include remote devices such as still or video cameras, borescopes and periscopes. These allow detection of corrosion products, flaking, discoloration, peeling, abrasion and blistering. Some plants make use of in-service monitoring programmes and trend performance. Inspection associated with containment ageing becomes particularly difficult in inaccessible areas of the steel structures, and the detection of corrosion and its assessment requires robotics and advanced techniques.

Containment leakage rate testing is conducted by pressurizing the containment and then monitoring changes in pressure, temperature and vapour pressure. If the pressurization is conducted with a mixture of air and gas, then halogen gas detectors can be used to enhance leakage detection. Containment testing can only be conducted during plant shutdown. Local leak testing of penetrations, bellows expansion joints, airlock door seals, flexible metal seals and isolation valve leaks is also possible.

The ageing assessment of non-metallic seals and gaskets (which can threaten the containment leak tightness) is complex and it should be included in PIPs for evaluation. It is to be noted that non-metallic commodities are replaceable items. In case of uncertainty in the assessments, they can always be preventively replaced.

Protective coatings may also be difficult to inspect and diagnose. Several techniques are available, such as ultrasonic mapping, coating adhesion tests and dry film thickness measurements of polymer coatings, using microscopes with scale reticles or monopolar magnetic inductors. The ageing evaluation of coatings can be conducted by measuring their characteristics (e.g. thickness and bond to substrate). The economic life of a coating is determined by comparing maintenance costs with recoating costs (including personnel dose). The methods used to assess concrete conditions vary in relation to the concrete property to be assessed, and include the following:

— Visual inspection to assess bleeding channels, chemical composition, corrosive environments, cracking, delamination, honeycomb, structural performance, aggregate quality, uniformity and voids;
— Air permeability to check air content;
— Audio methods to check cracking;
— Break-off methods to check compressive strength;
— Carbonation depth to assess acidity and corrosiveness of the environment;
— Chloride testing to assess chloride content, acidity and corrosive environments;
— Core testing to assess concrete cover, cracking, creep, delamination, density, elongation, modulus of elasticity, modulus of rupture, moisture content, splitting tensile strength and voids;
— Infrared thermography to assess delamination;
— Instrumentation to assess, cracking, creep and elongation;
— Magnetic methods to assess concrete cover;
— Modal analysis to check structural performance;
— Petrographic methods to check alkali carbonate reaction, air content, alkali silica reaction, bleeding channels, cement content, chemical composition, aggregate content, mixing water content, cracking, delamination and frost damage;

- Probe penetration to assess compressive strength;
- Pull-out testing to assess compressive stress and pull-out strength;
- Radar detection to assess voids, concrete cover and embedded parts;
- Radiation/nuclear testing to assess cracking, creep, delamination, density, embedded parts, honeycomb, soundness and voids;
- Rebound hammer to verify compressive stress and uniformity;
- Stress wave transmission to check voids, cracking and delamination;
- Tomography to check cracking, delamination and voids;
- Ultrasonic pulse velocity to verify compressive strength, cracking, delamination, honeycomb, modulus of elasticity and voids.

In pre-stressed systems, primary ageing mechanisms are concrete shrinkage, creep, stress relaxation and corrosion of pre-stressing steel. These systems can be grouted or non-grouted. In non-grouted systems, tendons are available for inspection, testing and retensioning. For grouted pre-stressing systems, cementitious grout provides excellent corrosion protection for the tendons, when used properly. For non-grouted tendons, inspections and tests include:

- Visual inspection;
- Pre-stressing force measured at regular intervals;
- Mechanical tests on tendon materials;
- Tests on corrosion protection.

Other components of containment systems to be monitored and tested are:

- Liners and liner coatings: These can be directly monitored through ultrasonic, magnetic particle, liquid penetrant or electromagnetic testing. They are also tested for leak tightness, verified periodically by leakage rate testing. The integrated leakage rate test is evaluated by pressurizing the containment with air to a pre-established level. Liner coating thickness is also checked using magnetic pull-off and magnetic flux gauges.
- Penetrations: These are part of the containment pressure boundary and should be included in the ISI programmes.
- Structural anchorage.
- Foundations, piles and underground structures.

Additional guidance on inspection of post-tensioning systems is available in the ASME boiler and pressure vessel standards.

Other major concrete structures in nuclear power plants to be monitored at regular intervals are:

- Spent fuel pools;
- Cooling towers;
- Masonry structures;
- Water intake structures;
- Concrete pipes;
- Water stops, joint sealants and gaskets.

Usually, spent fuel pools are reinforced concrete structures supported by bearing walls and/or base mats. The spent fuel pools are used to store used fuel assemblies, fuel bundles or control and/or adjuster rods for a period of time until they are sufficiently decayed and cool to be transferred to dry cask or other storage. The reinforced concrete walls and floors are sufficiently thick for radiation shielding and may have a stainless steel or non-metallic liner to enhance leak tightness.

Cooling towers are used in closed cycle water systems to remove waste heat from the main condenser loop. They may also be made of concrete. Cooling towers are used as primary heat sinks. They are considered Class 1 structures and are designed to withstand natural phenomena (e.g. earthquakes and tornadoes).

During operation, the concrete inside a mechanical draft cooling tower shell is subjected to temperatures of about 40°C and 100% relative humidity. The natural cooling medium varies from one plant to another. Some plants use municipal retreated water containing chlorides or brackish water.

Other typical structures where concrete may be used are the water intake structures. A water intake structure is that portion of the circulating water system that facilitates the intake of cooling water to the plant main turbine condenser and to the component cooling heat exchanger. Concrete is also found in circulating pump supports, baffle/skimmer walls, concrete ponds and canals, as well as in large diameter underground water and sewage systems.

Equipment foundations in nuclear power plants are usually made of reinforced concrete. They serve as support structures for rotating machinery, such as turbines, turbogenerators, pumps or blowers, or as support structures for heavy static equipment such as electrical power transformers. Such equipment is often safety related or has crucial significance in power production. Foundations can be divided into table, spring, raft or floor foundations.

In conclusion, given the safety importance of concrete structures in nuclear power plants, concrete ageing has become one of the critical issues in relation to PLiM and LTO. More information on how concrete structures may be managed for long life operation and what needs to be considered to ensure that they are reliable throughout their service life can be found in Ref. [65].

5.8.13.5. Metallic structures

Steel containments are subject to the typical degradation mechanisms of all metallic constructions. They undergo degradation of their microstructure, which results in degradation of their mechanical properties. Stressors typically include exposure to high ambient temperatures, general and localized corrosion in their various forms causing wall thinning, pitting, intergranular precipitation of alloyed elements, embrittlement and void creation due to neutron radiation exposure and hydrogen embrittlement. All of these, in various degrees, but primarily pitting corrosion, may lead to through the wall penetrations and loss of leak tightness. In addition, imperfect welding and joining can, in time, produce cracks and void formation in deposited metal layers and mechanical loading can inflict damage such as buckling and flaws. Operating experience suggests that few of the instances of steel containment degradation have been discovered through planned inspections programmes. Most were discovered either by regulator audits or by the operator during ad hoc inspections, during preparations for leakage rate testing or during unrelated inspections of other components in the vicinity of the steel containment. As prevention is key to success in the ageing management of steel containments, it is important that operators increase their reliance on feedback and R&D in the preparation of their ISI programmes for steel containment and critical steel structures.

Steel anchors for attachment to concrete structures are required for heavy machinery, structural members, piping, ductwork, cable trays, towers and many other types of structures and components in nuclear power plants. In the USA, the requirements on concrete anchors for nuclear power plants are those of the NRC regulatory guide for the design of anchors and structural supports [67] complemented by the code requirements of the American Concrete Institute [68].

Embedded anchors are preferred wherever possible. Unfortunately, anchors are not standardized in operating plants around the world. In general, embedded anchors can be divided into two broad categories:

— Steel plates embedded in concrete at the time of concrete pouring for attachments of system supports to civil structures and for the supports to steel structures;
— Embedded anchor bolts to anchor equipment and steel columns.

Where embedded anchors have not been provided during the pouring of concrete, bonded type anchor systems can be used. For anchors using an embedded part emerging from the concrete surface, embedding depth, plate size and torque applied are critical. Bonded anchor systems use drilled holes in the concrete in which are inserted anchor bolts that are then grouted in place using cementitious grout or an adhesive that chemically bonds the anchor bolt to the concrete.

Metallic liners are thin plates joined by welding to each other and attached on the inside surface of many single wall containments to provide a barrier against gas leakage. Metallic liners usually consist of up to 25 mm thick carbon steel plates. They are anchored to the concrete walls by studs, structural steel shapes or other steel products. Containments in PWRs and dry well portions of BWR containments are typically lined with carbon steel plates. Liners of wet wells of BWR containments, and of fuel pool structures in most LWRs are instead made of stainless steel plates. The liner is never considered in stress analysis to contribute to the strength of the structure it protects.

Personnel and emergency airlocks are metallic structures that are fairly similar in principle. They typically have two pressure seating doors arranged in series. These doors are supported from flat, often stiffener reinforced bulkheads, which are, in turn, welded to a circular sleeve, either embedded in the reactor building concrete wall or welded to the steel shell if they penetrate steel containments. Occasionally, a personnel airlock sleeve, concentric with the equipment hatch spherical cap, is provided as an integral part.

Pipe penetrations are another category of metallic structures that may be either embedded when rigidly attached to concrete containment walls or provided with bellows in the case of steel containments:

— Rigid penetrations of concrete containments typically include a cylindrical sleeve with a concrete shear anchor. The sleeve is welded to the steel containment vessel. Closure plates may be either a forged head or a flat plate welded to the sleeve and to the penetrating pipe spool.
— Flexible containment piping penetrations provided with bellows are typically constructed of stainless steel and are primarily used in steel containments, with only limited use in reinforced and pre-stressed concrete containments. An example of flexible penetration is shown in Fig. 40. The bellows are part of the pressure boundary and allow the penetrating pipe(s) to move relative to the containment wall. They are used when the piping design requires flexibility at the penetration. Bellows minimize the loading imposed on the containment structure, caused by differential movements between the steel containment and the penetrating pipe [66].

Stainless steel bellows in some penetrations are subject to fatigue and SCC, and they can be difficult to inspect.

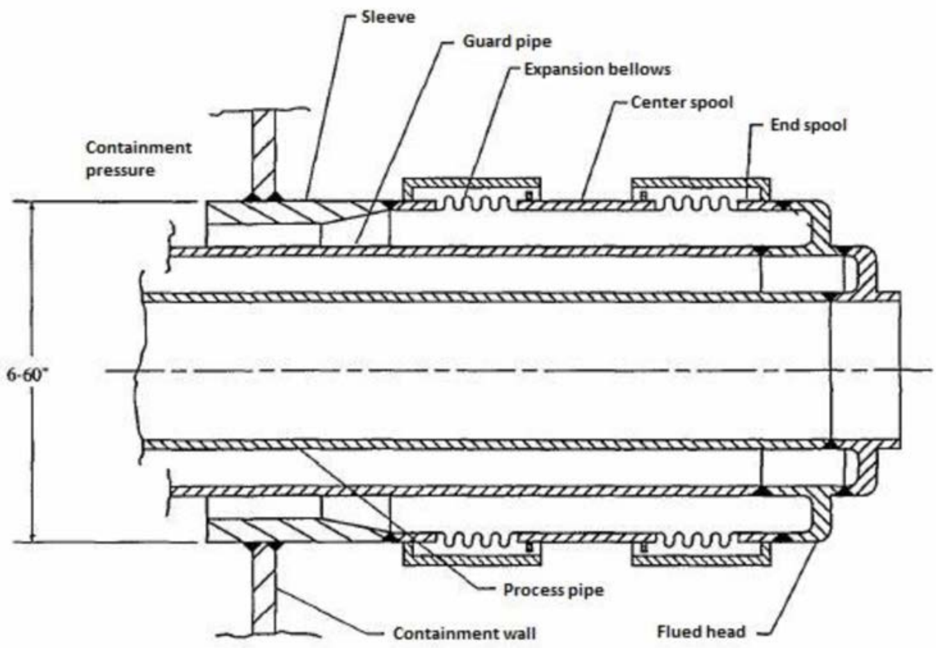


FIG. 40. Typical containment piping penetration bellows.

Waterstops are static seals used between joints in concrete structures to prevent the passage of water or other fluids. Waterstops are embedded in the concrete or run across and/or along joints. They are used in containment, spent fuel pools, wastewater treatment plants, secondary hazardous fluid containment structures, tunnels, ponds and water reservoirs, among others.

Joint sealants are used for caulking and sealing joints subject to concrete movements to provide a firm, flexible, weather tight seal. They may be used to enhance leak tightness within containment structures.

Inspections and assessments of accessible metallic structures and components normally conducted in a nuclear power plant are summarized in Table 4. Inaccessible structures are usually assessed using NDT techniques, including non-contact NDT, deep penetrating NDT and embedded sensors.

TABLE 4. METHODS TO ASSESS METALLIC STRUCTURAL COMPONENTS AND PROTECTIVE MEDIA PROPERTIES OR CHARACTERISTICS

Property or characteristic	Evaluation method												
	Coating measurement	Cross-cut test	Four electrode method	Grease tests	Half-cell potential	Lift-off test	Liquid penetrant	Local leak test	Magnetic particle test	Rate of corrosion probes	Tendon mechanical tests	Ultrasonic tests	Visual inspection
Aggressive ions				X									
Coating bond performance		X											X
Broken wires													X
Coating thickness	X												
Weld cracks							X		X				X
Coating distress													X
Elongation											X		
Free water quantity				X									
Leakage								X					X
pH value				X									
Pre-stressing force loss						X							
Reinforcement corrosion			X		X					X			X
Structural degradation													X
Surface cracks							X		X				X
Ultimate strength											X		
Wall thickness										X		X	
Yield strength											X		

6. INTEGRITY ASSESSMENT OF STRUCTURES, SYSTEMS AND COMPONENTS

6.1. INSPECTION AND ASSESSMENT METHODS

National codes and standards usually require a periodic assessment of the containment integrity and its leak tightness. This is carried out using either non-destructive or destructive testing methods.

6.1.1. Non-destructive testing

NDE methods are widely used to check cracking and material degradation phenomenon. They can be either limited to the surface of the SSC being examined or they can be volumetric, where they provide a volumetric representation of the SSC body being examined. The most common non-destructive inspection methods are:

- Visual examinations, which are heavily dependent on the skills and experience of the examiner. Without special visual aid tools or dismantling the SSC to gain accessibility, they are limited to accessible areas. Three types of visual examinations are classified in the ASME boiler and pressure vessel code:
  - Visual test-1, to detect discontinuities and imperfections on the surface such as cracks and corrosion, regardless of the complexity of the surface;
  - Visual test-2, to detect leakage from pressure retaining components;
  - Visual test-3, to determine the general mechanical and structural condition of components and their supports.
- Liquid penetrant inspections, which are particularly recommended to detect cracks, porosity, laminations and capillary discontinuities in solid components. This method makes use of the capillary effect in which the liquid is absorbed by the surface defect.
- Eddy current inspections, which are very useful for detecting cracks, porosity and inclusions of electrically conductive materials. This method is based on the principle that alternating currents through a coil close to a conductor induce a current in the conductor and create a magnetic field of the opposite sign to that of the field created by the original current in the coil. Any surface discontinuity alters the field and can be detected in a secondary coil in the eddy current probe. Flaw size is indicated by the extent of the response change, as the probe senses the object being examined.
- Magnetic particle inspections, which are only used to detect surface and shallow subsurface discontinuities in ferromagnetic materials. A magnetic field is induced into the material, and then its surface is dusted with iron particles that are either dry or in coloured or fluorescent liquid suspension. The lines of the magnetic flux are locally disrupted in a pattern that corresponds to the flaw.
- Ultrasonic testing using high frequency sound waves, which is used to detect surface and subsurface flaws. Echoes are scattered or reflected back to a receiver from any internal imperfections. However, signal scattering may be misguiding. The travel time of the wave and the shape of the spectral response of the transmitted versus the reflected signal provide an indication of the flaw. The method requires interpretation by operators familiar with patterns, signal magnitude, timing and probe positioning to determine the flaw size, distance and reflectivity. The measurement is volumetric and is suitable for up to 6 m of axial penetration. It is extremely sensitive and can determine very small flaws. Ultrasonic testing is commonly used in the pulse echo mode to monitor wall thinning. The transducer transmits waves, and the signals are reflected from the front and back surfaces of the wall. The difference in arrival times of the two signals can indicate the current thickness. Ultrasonic imaging techniques are used to detect and localize thickness reductions in the metallic containment of BWRs. Ultrasonic testing can also be used to detect and monitor corrosion.
- Electromagnetic acoustic transducers, which can be used in various contexts. They have been used to measure thickness, to detect flaws and to characterize materials. They are a form of ultrasound examination technique that does not require contact with the metal surface. High energy ultrasound shear waves, which can travel a good distance, are generated directly in the metallic component being examined, rather than in the transducer,

and the method is non-dispersive. Shear waves can be generated even through surface coatings of thicknesses of up to 1.5 mm.

— Magnetostrictive sensors, which are used to inspect piping. These are devices that launch guided waves, which are more sensitive than the shear waves used in the electromagnetic acoustic transducer system. These waves can follow the contour of a structure, can accommodate thickness discontinuities, can travel considerable distances and can be picked up at a remote location. This method can be used to characterize corrosion damage and to inspect structures whose surfaces are not directly accessible, as in the case of painted surfaces.
— Radiographic techniques, which are used in the detection of both surface and deep flaws in materials. These techniques use penetrating gamma or X ray radiation. Flaw detection is based on the differential absorption of radiation. Radiation is routed through the material and is captured on film on the opposite side of the source. Two dimensional defects are indicated as the density changes (i.e. various shades of grey) on the film, which remains as a permanent record. Gamma ray sources are portable, while X ray machines are usually bulky, so the choice depends on the application. Instead of a film, gamma radiometry uses a radiation detector and a counter. The count or count rate is used to characterize the geometry. The following is a list of limitations of this testing technique:
  • Cracks are hard to size and shape because the sizing capability is highly dependent on the angle of incidence of the rays with respect to the crack orientation;
  • Results are not immediate;
  • Access from both sides is mandatory;
  • Accessibility to the test specimen may be a problem;
  • Heavy biological shielding is usually necessary and even total area evacuation may be required.
— Acoustic emission inspections, which are useful for the detection of crack propagation. This technique uses small amplitude stress waves that are emitted by kinetic energy, as the material is locally strained beyond its elastic limit. Piezoelectric transducers placed on the material surface detect stress waves as small displacements. Triangulations can be used to identify the exact location of the emerging crack, even in very large structures. This NDE method is best used as a continual monitor of critical SSCs for first detection of crack propagation. The system is extremely sensitive and fast, but also obstructive. Applications include SCC, hydrogen cracking and evolution of gas inclusions. Conducting the test and interpreting results requires considerable experience, as background noise often interferes with signals.
— Thermographic inspections, which are recommended for the detection of property variations at interfaces of layered materials. The materials need to be thermally conductive, because a pulse of thermal energy is applied. A thermographic scanning camera with infrared spectrum detection capability monitors the thermal state of the object. The time dependent temperature gradient to the internal condition of the material is used to produce the results. It is less effective in the detection of subsurface flaws in thick objects and resolution is lost at the edges. The higher the maximum temperature difference, the better the flaw detection capability.
— Electrochemical corrosion monitoring, which is used to measure corrosion rates. The direct current is directed from a counter electrode to the working electrode and the change in potential is measured between the two electrodes. Alternatively, alternating current impedance changes can be used and converted into corrosion rate information.

6.1.2. Destructive testing

Tests altering the shape, size or structure of a material are considered destructive. Two types of destructive methods are applied:

— Planned destructive tests. The most common planned destructive test is that of the RPV surveillance programme, in which specimens are inserted into the RPV, periodically withdrawn and destructively tested; these tests serve for periodic evaluation of RPV material embrittlement and are a necessary part of PSRs. Some special cases, such as a rule in the Russian Federation [40], still require one weld from the primary piping in each plant to be cut out after 100 000 operating hours to produce specimens to determine any potential changes in the mechanical properties of the material that has seen all actual stressors during operation. Of particular interest are the tensile properties. These tests are usually omitted nowadays, as other non-destructive methods are available with equivalent reliability.

— Unplanned tests (e.g. material ageing, integrity, etc.). These tests are usually conducted when some information about material properties is missing or it is necessary to obtain new data.

Destructive tests require that some material be removed from the component surface, by mechanical or electroerosive machining, to be used as a test sample. The removed material can be either negligible (e.g. taking scraps for chemical analysis or for neutron dosimetry measurements) or substantial in size (e.g. samples for metallographic tests, mechanical tests, tensile tests, subsize Charpy impact specimens, fracture toughness tests, small punch tests for determination of tensile properties and/or transition temperature tests, etc.). In all these cases, stress analysis of the component with the material removed (e.g. effect of sampling on stresses and potential notch effect on fatigue) is necessary to confirm that the safe operation of the component will not be affected. In all such cases, previous authorization/approval from the regulatory body needs to be obtained.

Not all destructive tests require complete destruction of the specimen. A test is called semidestructive when it has negligible effects on the component integrity, shape and size. Examples of tests of this nature are hardness tests, instrumented hardness tests, automated ball indentation tests and replicas for metallographic examination.

6.1.3. Leak before break

The LBB criterion allows the relaxation of design requirements of safety related pipes and headers. The concept has been used to implement safety upgrades of existing plants that do not meet current regulatory requirements. When restrictive regulatory rules are issued to operating plants, applying the LBB concept allows these plants to obtain regulatory approval without the implementation of large design changes that would have not been economically feasible.

Such is the case of postulated double ended guillotine breaks (DEGBs). Where DEGBs are postulated and the LBB concept can be justified, an LTO case can become licensable. The LBB criterion, when applied to the main primary circuit piping, suggests that DEGBs in the primary circuit piping are extremely unlikely. This conclusion has been based on several tests which have shown that the failure mode of large pipes and headers has a high probability to begin with leaks through small cracks before they break, and that the crack grows only gradually until it reaches its critical crack length. Another important point is that the leak does not immediately challenge the pipe’s capability to withstand its design loads. The LBB concept also relies on the fact that leaks can be easily detected, by using either existing or newly added monitoring techniques.

The LBB concept applied to DEGBs has been interpreted by some as a proactive and preventive approach to large LOCA issues. In order to validate the concept and turn it into a design criterion, fracture mechanics methods have been applied and models validated by experiments that have been proven capable of predicting the evolution of postulated DEGBs, under all operational conditions, including accident loads. This capability has allowed operators to use the time interval between the first leak and the critical crack length to control, inspect and monitor their critical piping systems. These defensive measures, together with conservatively selected critical crack parameter thresholds, has allowed the development of administrative procedures for the prevention of catastrophic failures, without having to introduce massive design changes and the risks that these involve.

If a leak grows large enough to start producing a void in a pipe, a power pulse may follow. To avoid such an occurrence, mitigating action should be taken before voiding starts; this needs to be long before a crack has grown to the critical crack length.

The probability of leaks that would propagate quickly to DEGBs has been estimated for CANDU reactors and gave failure rates of 7.7×10^{-10} per reactor-year for 460 mm pipes and 8.8×10^{-10} per reactor-year for 560 mm headers. Failure rates for smaller pipes were much higher. By crediting LBB, all pipes greater than 180 mm in diameter would have extremely low failure rates. By not crediting LBB, only pipes greater than 460 mm would have extremely low failure rates [69].

Other concepts somewhat similar to LBB, notably the low break probability and break preclusion concepts have also been used. Implementation of the break preclusion concept ensures that the probability of piping breaks and/or leakage cracks, when occurring for unknown and/or hard to predict reasons, is lowered to an acceptable level. This concept is used, for example, in PWRs in France and Germany to prove the quality of the primary coolant piping systems through fracture mechanics. The break preclusion concept originally applied to the main piping has been developed into an integrated concept for the whole pressure boundary within the containment and is also used in the PSR of operating nuclear power plants.

6.2. INTEGRITY MONITORING

Existing monitoring methods should be evaluated, with account taken of relevant operating experience and research results, to determine whether they are effective for timely detection of material degradation before failure of the structure or component. Sampling checks of equipment should be applied to detect precursors of material degradation when possible. In the evaluation of existing monitoring methods to identify the most effective and practical technologies, the following points should be addressed and documented before committing to a specific monitoring technology:

- The technology should make use of functional parameters and condition indicators for detecting, monitoring and trending material degradation of the structure or component;
- The technology should be capable of measuring these parameters and indicators with sufficient sensitivity, reliability and accuracy;
- The technology should offer data evaluation techniques for recognizing instances of significant degradation, failure rates and their tendencies, and for predicting future integrity and functional capability of the structure or component.

Half-cell potential measurements are a typical example of an integrity monitoring system. They are used as monitors of corrosion in the reinforcement steel of concrete structures. One electrode is connected to the reinforcement steel and the potential is measured at several locations on the concrete surface. Half-cells can be mounted on a roller bar with an automated data acquisition system.

6.3. NON-DESTRUCTIVE EXAMINATION/IN-SERVICE INSPECTION OF REACTOR PRESSURE VESSELS

The integrity of RPVs is of critical importance for stable and reliable operation of nuclear power plants. Therefore, during the plant service life, periodic ISIs of RPVs need to be performed. Through NDE, critical information can be gained about the condition of the RPV.

The scope, the frequency and the type of NDE techniques in RPV inspections are defined by various regulations and standards. In addition, certain European States follow their own national standards that are based on the European Network for Inspection and Qualification (ENIQ) methodology [70].

The most common NDE methods applied during RPV inspection are ultrasonic, visual and eddy current testing. Ultrasonic inspection is used to assess the entire wall material of the RPV combined with visual inspection of the inner surface, and, optionally, eddy current examinations of the reactor vessel inside surface. Most inspections are performed by remotely controlled robots from the inner surface. However, certain jurisdictions also require and perform inspections from the outside surface to obtain further information about the condition of the vessel. A visual example of a robotic inspection system is given in Fig. 41.

The scope of the RPV examination differs for different plant types:

- PWRs: Circumferential and longitudinal welded joints of PWRs, inlet/outlet/safety injection nozzle welds, nozzle inner radii, nozzle to shell welds and baffles to former plate bolts;
- WWER-440s: Circumferential welded joints, inlet/outlet nozzle welds, nozzle inner radii and vessel material in the active region of the RPVs (zones exposed to highest neutron fluxes);
- WWER-1000s: Circumferential welded joints, nozzle welds, nozzle inner radii and vessel material in the active region; during a special outage, 100% of the RPV surfaces are examined using the ultrasonic and eddy current methods;
- BWRs: Inspection of nozzles (internal diameter and outside diameter), of certain welds and of RPV supports.

Inspection of the RPV is an activity on the critical path of every PWR and BWR outage and needs to be performed with maximum reliability. Robots for RPV inspection have been developed in such a manner that maximum efficiency and inspection speeds can be achieved in order to directly and efficiently reduce the total outage time. New technologies, such as highly automatized robots, associated software packages, ultrasonic

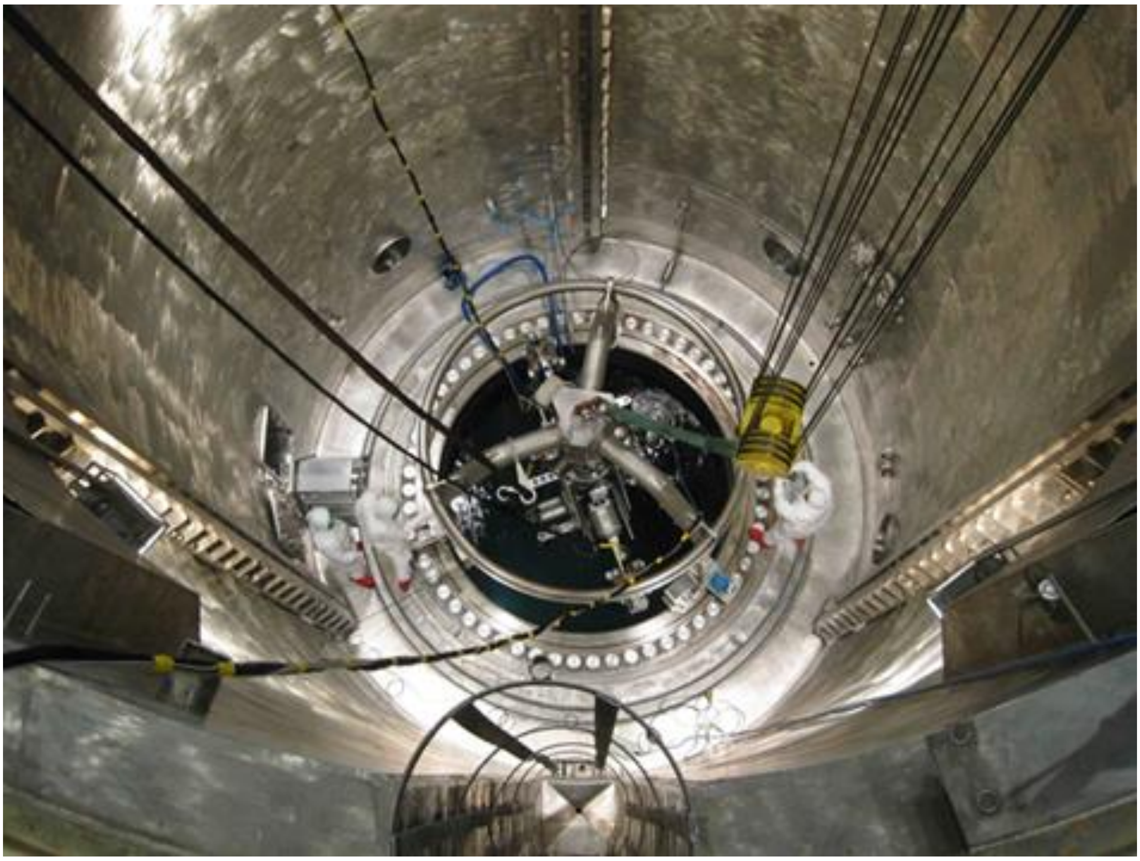


FIG. 41. Example of a RPV inspection manipulator.

phased array technology and high resolution visual inspection cameras, are used to provide enhanced inspection capabilities.

The need for speed should never compromise the quality and technical requirements of an RPV inspection. A common requirement in all types of power plants is the qualification process. There are a number of qualification approaches (e.g. ENIQ and ASME) that differ in technical and formal aspects, but all approaches ensure that the inspection system and the technique used can meet the code requirements.

This is valid also for structures inside the RPV that need to be subjected to NDE inspections. These structures include:

— Core barrel support lugs;
— Bottom mounted instrumentation;
— RPV threaded flange holes, nuts and bolts.

The NDE technique is required for bottom mounted instrumentation structures (see Figs 42 and 43). This has been emphasized in the last 10 years following the ageing damage incurred in plants that use bottom mounted instrumentation (see Section 5). Ultrasonic testing is used to inspect the bottom mounted instrumentation tube inner surface, and eddy current testing is used to inspect the tube J-weld.

Recent operating experience from WWER power plants revealed issues with the integrity of the RPV sealing components, such as bolts, nuts and threaded holes. The threaded holes of the RPV flanges and studs can be examined using ultrasonic and eddy current NDE methods.

FIG. 42. Examples of bottom mounted instrumentation inspection equipment: (a) bottom mounted instrumentation and (b) inspection equipment.

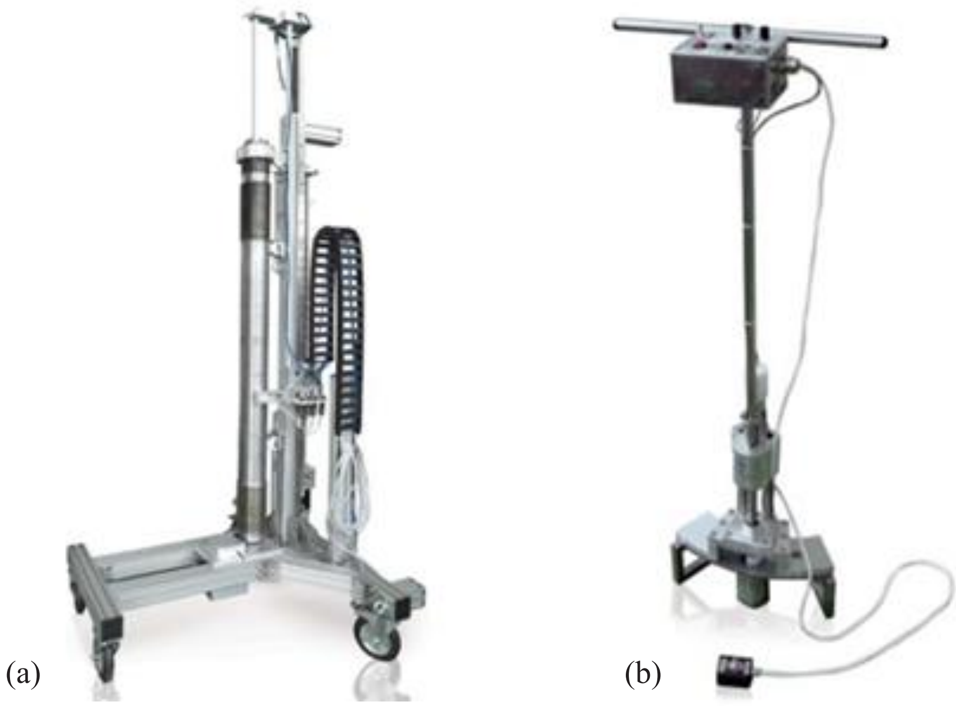


FIG. 43. Examples of RPV: (a) stud and (b) hole inspection manipulators.

6.4. NON-DESTRUCTIVE EXAMINATION/IN-SERVICE INSPECTION OF STEAM GENERATORS

Various NDE techniques are used in ISIs of steam generators. They are designed to provide vital information about the steam generator conditions. The scope, the frequency and the type of NDE techniques to be used are defined in various regulations and standards.

The main steam generator inspection activity is eddy current internal diameter examination of the steam generator tubing. These inspections are performed with remotely controlled robots installed inside the steam generator primary channel heads while the eddy current probe is positioned on the desired tube. Figure 44 shows examples of steam generator inspection manipulator for WWERs and PWRs/PHWRs.

The differences between the steam generator tubing inspections of WWER and PWR/PHWR plants are mainly owing to the different steam generator configurations. The PWR/PHWR steam generators have a vertical tube bundle configuration. They use robots attached to the tube sheet that can easily move from tube to tube across

(a)

(b)

FIG. 44. Examples of steam generator inspection manipulators: (a) water cooled water moderated power reactor inspection manipulator and (b) pressurized water reactor/pressurized heavy water reactor inspection manipulator.

the tube sheet. Such designs offer the possibility of multiple robots per steam generator channel head; hence, the work can be distributed efficiently.

The R&D on eddy current testing has developed advanced inspection methods that offer details about the condition and degradation of the tubing. Advanced technologies use a probe array that enables the examination of the entire tube circumference, allowing the determination of the size (length and depth) and morphology of the detected degradation, the source of the defect initiation and its orientation. This is valuable information for further condition assessments, operational assessments and ultimately for efficient steam generator ageing management (see Fig. 45).

Newer robots offer higher inspection speeds and faster data collection. This results in large amounts of data, collected in short periods of time, that require advanced software capable of reliable and efficient eddy current data processing and automatic report generation.

In addition to data collection on steam generator tubes, eddy current hardware and software are used to collect, process and evaluate data on other subcomponents such as:

- TSPs;
- Antivibration bar gap measurements;
- Independent tube internal diameter validation;
- Deposit mapping.

Deposit mapping is a method that provides significant benefits to the ageing management of steam generators. It allows for measurements of the height of sludge deposits on the tubes and automatic mapping of the sludge distribution for the entire tube bundle. Deposit mapping allows better sludge maintenance, resulting in more stable steam generator operation.

(a) (b)

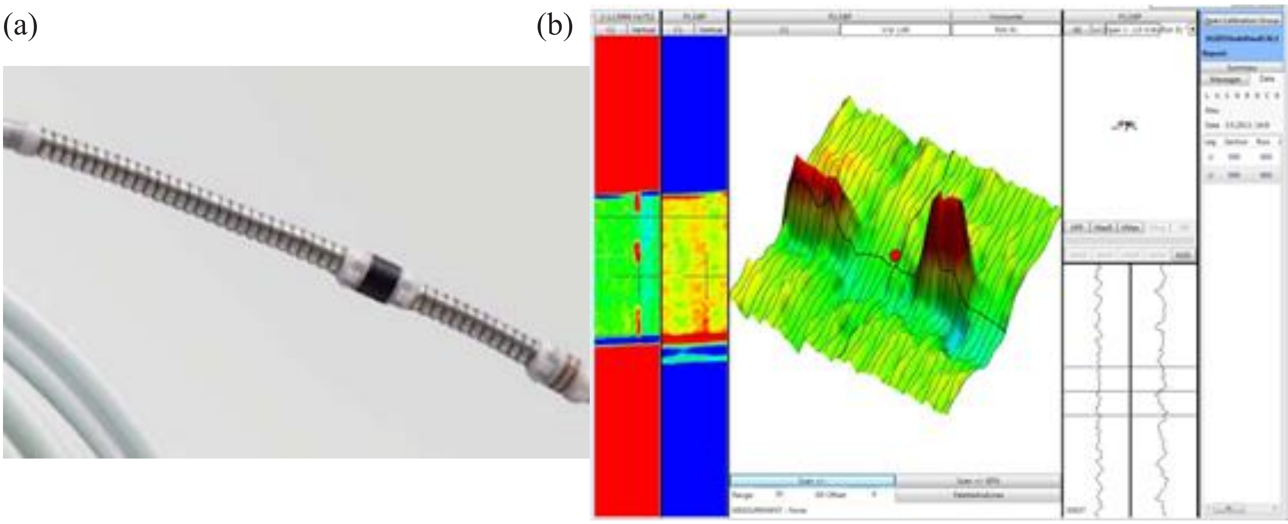


FIG. 45. Steam generator eddy current array probe: (a) example of probe and (b) example of data.

Each new steam generator inspection system and technique needs to be qualified prior to site implementation to verify that all selected and newly developed inspection methods meet code requirements. Currently, most PWR utilities require qualification in accordance with EPRI guidelines [71], while others apply their own qualification procedures.

6.5. NON-DESTRUCTIVE EXAMINATION/IN-SERVICE INSPECTION OF REACTOR PRESSURE VESSEL HEADS

Inspection of RPVHs and subcomponents involves ultrasonic, eddy current and visual examination. Remotely controlled robot systems automatically position the inspection devices on the desired RPVH penetration.

In the WWER fleet, RPVH inspections are performed automatically only in certain plants. In PWR plants, after the well known events (see Section 5.2.3) involving RPVHs in 2001 and 2003, inspection became mandatory and resulted in a number of RPVH replacements. Normally, RPVH inspections are performed from under the RPVH, where an inspection robot is deployed (see Figs 46–48).

Inspections of PWR RPVHs include:

— Bare material visual examination: The top of the RPVH is visually inspected using a crawler robot fitted with a camera.
— RPVH penetration examination: The penetration welds through the RPVH are inspected from the inside surface of the penetration. Ultrasonic examination is used for the entire material volume, while eddy current examination is used for the inside surface of the RPVH penetrations. Additionally, ultrasonic testing is used for the detection of leak paths.
— J-weld inspection: Surface inspection needs to be performed on the J-welds and the outside surface of the penetration tubes. Inspection is conducted using eddy current array probes with a large number of coils (16 or more).
— CRDM joint weld examination: The bimetal weld is examined by ultrasonic and eddy current inspection. This is similar to the RPVH penetration tube inspection.

Inspection methods applied in RPVH examinations include time of flight diffraction (TOFD) ultrasonic inspection techniques, which provide good results for small material thicknesses. Inspection probes for RPVH penetration examination combine both ultrasonic TOFD crystals and eddy current coils. J-weld eddy current inspections are conducted with array probes that utilize multiple coils, enabling coverage of large inspection volume.

FIG. 46. Example of a RPVH inspection probe manipulator.

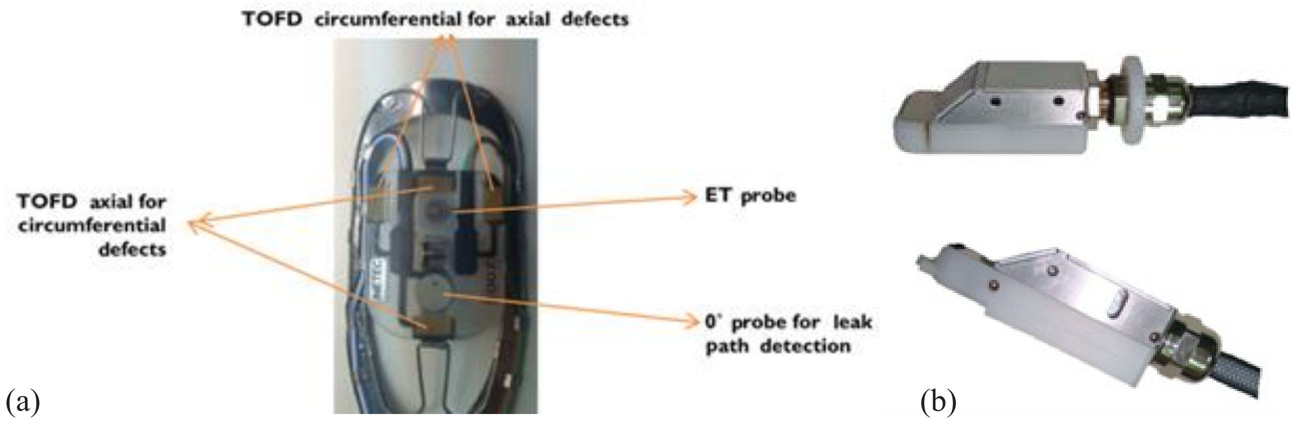


FIG. 47. Examples of RPVH inspection probes: (a) ultrasonic probe for RPVH penetration inspection and (b) eddy current array probes for J-weld examination. ET — eddy current testing; TOFD — time of flight diffraction.

Each inspection system and technique needs to be qualified prior to its site implementation to verify that the selected technique, and any newly developed inspection method, meet all requirements. Currently, most PWR utilities require qualification of the ultrasonic inspection of RPVH penetrations.

As degradation was found in a number of RPVHs, corrective actions were developed, qualified and implemented, without impairing the structural integrity of the components. Examples of such corrective actions are:

— Automated surface repair, not involving heat treatment;
— Embedded (under the surface) flaw repair, involving heat treatment.

An example of a commercially available repair tool is presented in Fig. 48.

FIG. 48. Example of a RPVH repair tool.

7. REGULATORY FRAMEWORKS ON AGEING MANAGEMENT

7.1. INTRODUCTION

The ageing management regulatory frameworks in most States are designed to provide the licensees of nuclear power plants with a common approach for demonstrating the adequacy of the plant critical SSCs and their own continuing capability to operate the plant safely and reliably throughout the licensing period. In general, each Member State sets its own ageing management requirements based on internationally recognized licensing practices. However, all are guided by common principles and essential safety control features.

All ageing management regulations aim at confirming, throughout the plant life, that the licensing basis is upheld. This entails that operators demonstrate that the plant is capable of maintaining the required dose limits and releases to the environment prescribed by the plant operating licence under all operating conditions, including accident conditions, even as SSCs continue to age. Therefore, at regular intervals, the plant safety analysis is updated and the SSCs important to safety are evaluated with regard to the effects of ageing.

7.2. GENERAL CONCEPTS OF LEGAL AND REGULATORY FRAMEWORKS

7.2.1. Legal framework

Each Member State establishes its own governmental and legal framework with the purpose of ensuring the effective regulatory control of nuclear facilities and related activities in the State. The framework defines the following:

— Responsibilities and functions covering the necessary liaison with the global safety regime;
— The necessary support services to implement the safety requirements (including radiation protection);
— The emergency preparedness and response strategy;
— A nuclear security regime including the State control and accounting of nuclear material;
— The requirements governing a nuclear regulatory body and a regulatory framework.

The regulatory body develops strategies and promulgates regulations to administer the State laws and policies. These include rules within the regulatory framework that set up the appropriate organizations to deal with the operation of a nuclear power plant, including SSC ageing management.

Figure 49, taken from Ref. [6], shows the steps used during PSR preparation to ensure that the nuclear power plant organizations, the licensing procedures and related safety policies comply with the regulatory framework criteria.

7.2.2. Regulatory framework for ageing control

From the licensing standpoint, there are three conceptual approaches that Member States use to periodically verify the safety and adequacy of their nuclear power plant structures and the capability of their operators to continue operating their nuclear power units, even beyond the originally assumed service life of their units. All three approaches are based on plant configuration control and on a good understanding and awareness of SSC ageing:

— The LRA process;
— The PSR approach;
— A combined approach of the above two.

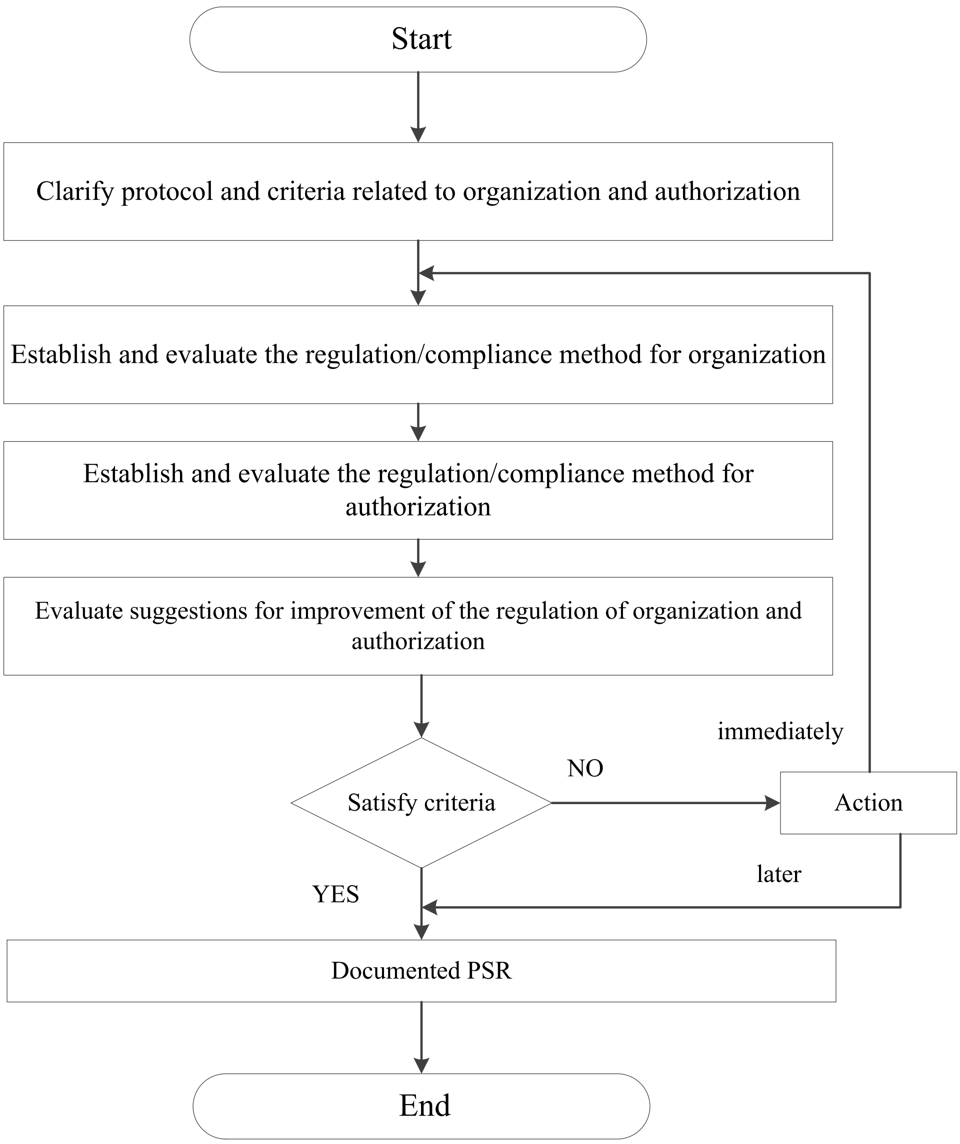


FIG. 49. Compliance with the regulatory framework [6]. PSR — periodic safety review.

The USA developed and practises the LRA process, while most European States and Japan use the second approach, PSR, to obtain their authorization to continue operation even beyond the originally assumed design life. Other Member States utilize the third approach, a combination of the first two concepts and related regulatory approaches, encompassing elements of both LRA and PSR to meet local conditions and practices.

7.2.3. Standard periodic safety review approach

The IAEA has standardized the PSR approach and issued recommendations to that effect [6]. This is because most newcomer countries to nuclear power are likely to adopt PSR, or a mixed approach, to control the effects of ageing on the safety and licensing of their facilities and the related licensing rules on ageing and ageing management safety. In States where the safety performance of nuclear power plants is monitored through PSRs, if the PSR results are satisfactory, the regulator releases an authorization to continue operation to the end of the PSR cycle (usually 10 years). This regulatory system does not limit the number of PSR cycles, even beyond the originally assumed service life of a nuclear power generation unit. The fundamental requirement is for the licensee to demonstrate a good understanding of the plant condition and capability to operate the plant safely for the duration of the PSR cycle. If the new operating period reaches or approaches the end of the originally assumed service life, the essence and the main focus of the LTO authorization process becomes that of determining whether the ageing of critical SSCs has been effectively managed, whether its current conditions are well documented and whether its remaining life prognosis is justified and acceptable. The regulator seeks sufficient assurance that all required safety functions will be maintained throughout the LTO period. Regulators may also use PSR as a tool to identify and resolve regulatory gaps and incorporate new generic safety issues. The PSR process is shown in Fig. 50.

The selection of SSCs for the safety review of the LTO period is a critical step. It determines which plant SSCs will be included in the safety review scope. Some national regulators also require that all SSCs, with an essential function that mitigates certain types of events and which is credited in a safety analysis, be included in the scope. Such events may include:

- Fires and floods;
- Extreme weather conditions;
- Earthquakes;
- Pressurized thermal shocks;
- Anticipated transients without scram;
- Station blackouts.

A scoping exercise may follow the decision making process (see Fig. 51).

Safety analysis reviews incorporate deterministic and probabilistic methods, which, to a large extent, are complementary. A probabilistic safety assessment provides insight into the risk items related to time dependent variables affecting safety, performance and defence in depth. This capability is particularly useful in the optimization and prioritization of inspections and preventive maintenance tasks, while deterministic methods provide insights into the integrity of the SSCs critical to safety and into the plant capability to respond to the various operational conditions.

A PSR includes a review of the plant operating and emergency procedures, its maintenance, surveillance and ISI programmes, human performance, staff knowledge management and of adequacy of processes and procedures to monitor and manage ageing mechanisms.

In addition to the PSR effort, every time a major change is introduced into the plant configuration, the plant safety analysis models (including probabilistic safety assessment) are updated to demonstrate continued compliance with the plant safety envelope, which comprises:

- The plant defence in depth;
- The adequacy of the plant protection and its control functions, including set points and control parameters;
- The capability to mitigate the effects of internal and external events including extreme conditions.

A prerequisite for a PSR submission is that the following plant programmes have been well established from the beginning of plant operation:

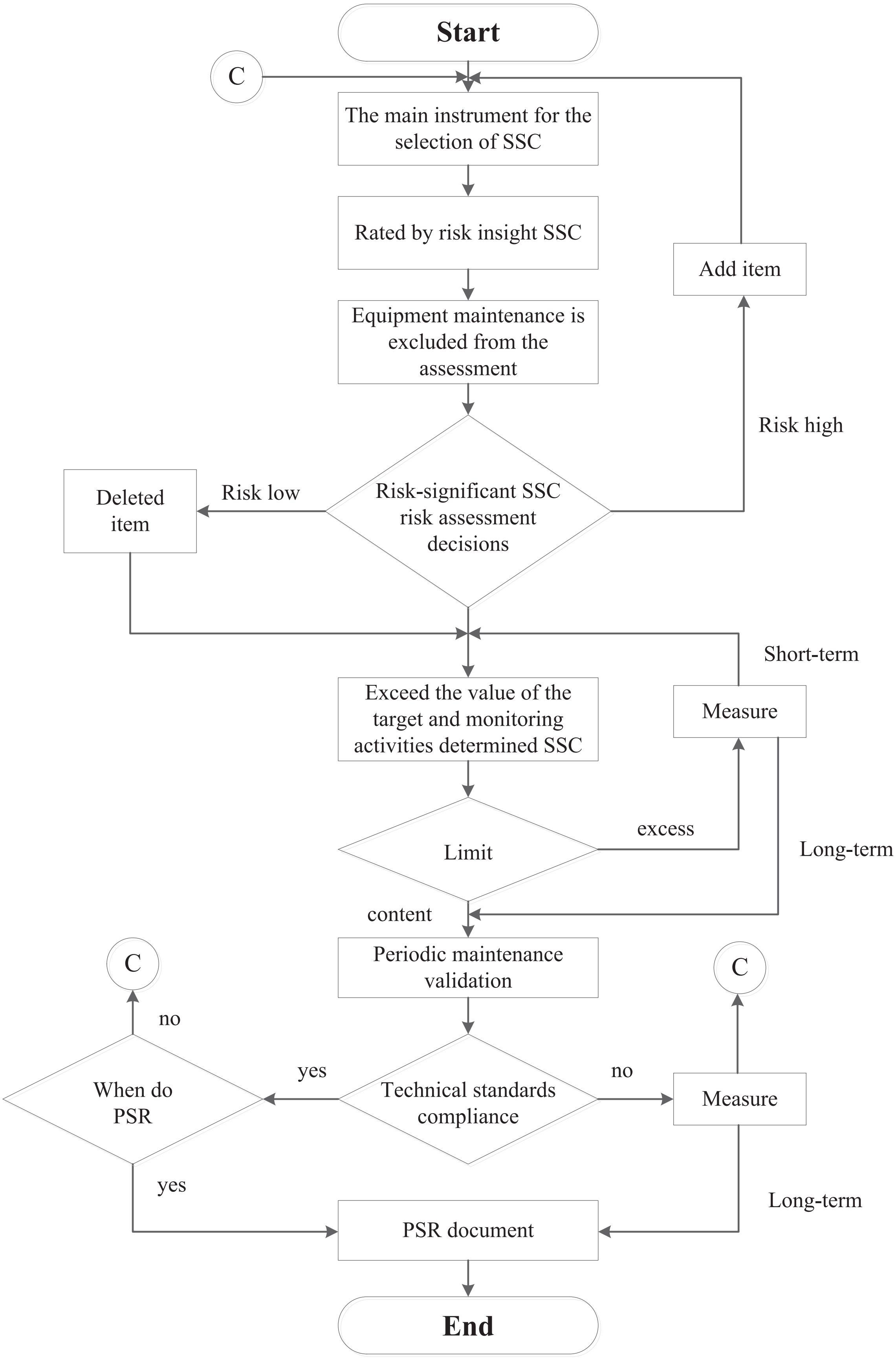


FIG. 50. PSR report preparation process [6]. SSC — structure, system and component.

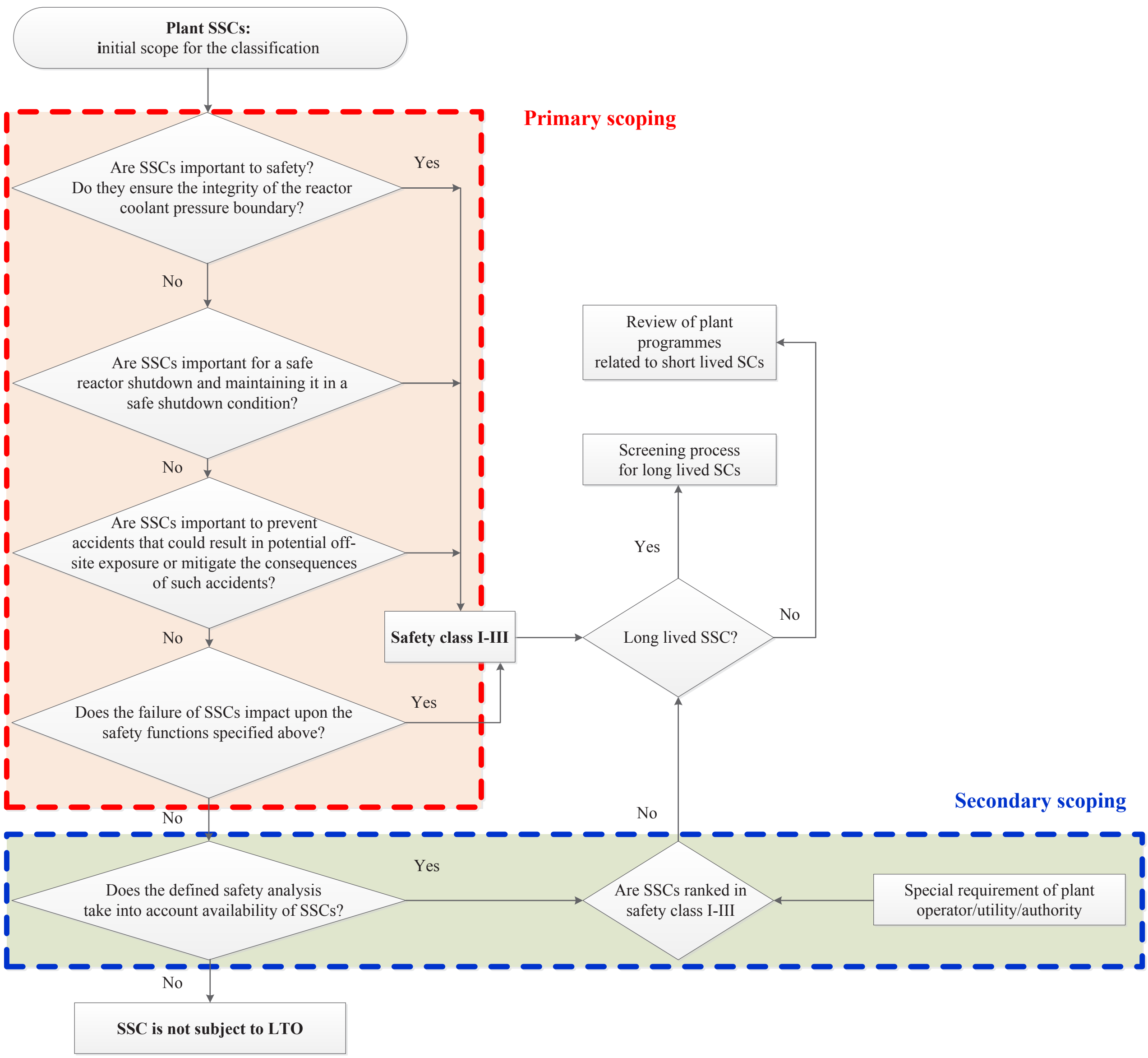


FIG. 51. Scope setting process or 'scoping'. SC — structure and component; SSC — structure, system and component.

— Maintenance;
— Equipment qualification;
— ISI;
— Surveillance and monitoring;
— Monitoring of chemical regimes;
— Operational diagnostics;
— Configuration management.

Other relevant programmes at the plant need to include:

— Resolution of safety issues;
— Probabilistic safety assessment;
— PSR;
— Regular updating of the SAR.

Depending on the licensing framework in force, regulators may require an evaluation of TLAAs, particularly for PSR reports submitted for LTO permits.

As shown in Fig. 52, long lived critical components are selected through a screening process to identify SSCs whose degradation is managed by AMPs and evaluated by TLAAs. The screening process results in the development of:

— A list of AMPs to be verified;
— A list of SSCs for which AMPs are to be developed;
— A list of TLAAs whose validity need to be reviewed;
— A list of SSCs with no ageing effects (no TLAA needed).

Short lived non-critical SSCs are managed by plant programmes, such as maintenance and equipment qualification reviews, and may be subject to regular replacement.

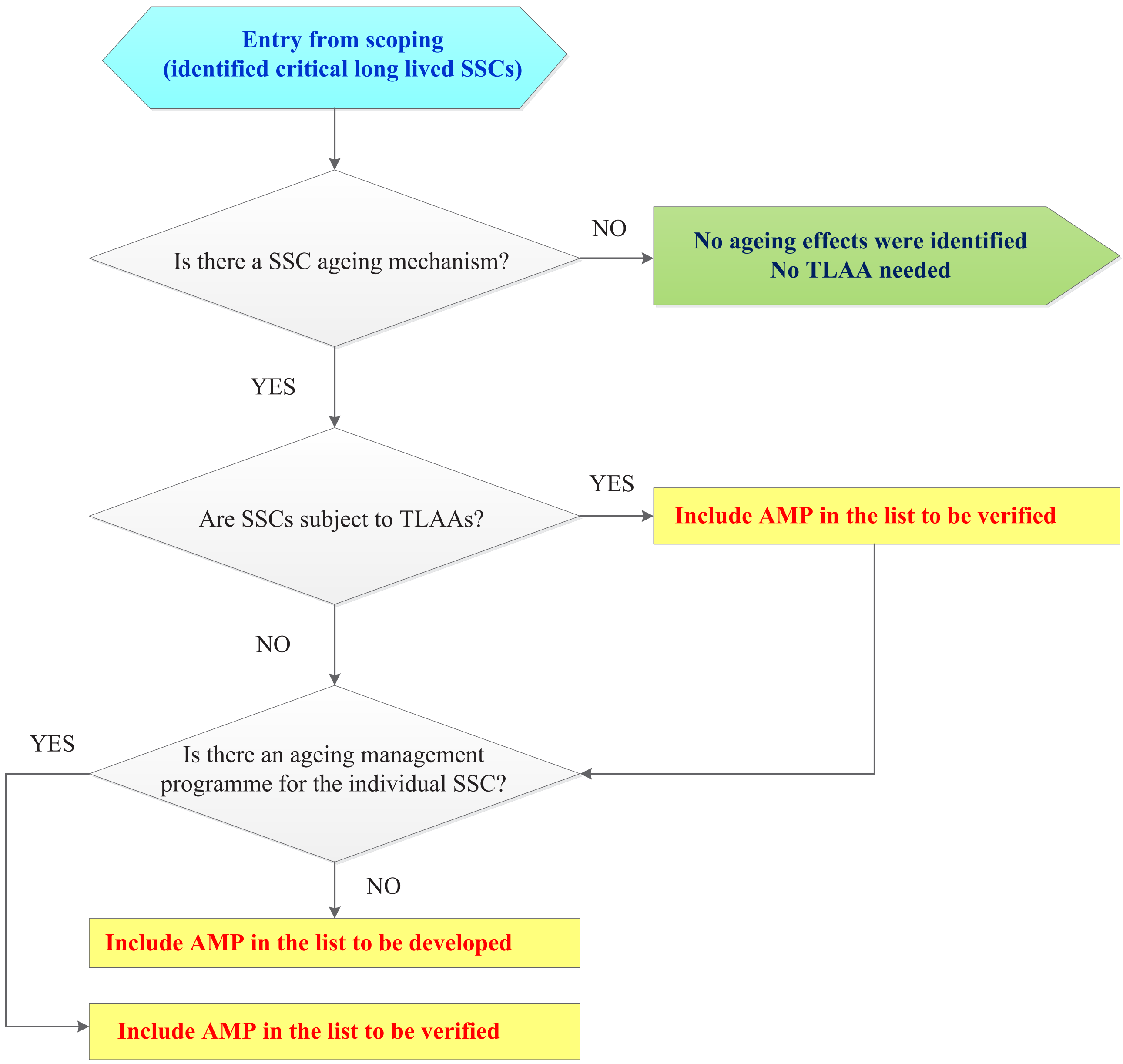


FIG. 52. Screening process for long lived SSCs in a PSR for LTO. SSC — structure, system and component.

The TLAAs are plant specific analyses for the time period under consideration. The TLAA time period may cover a limited number of loading cycles or the even more limited duration of a specifically defined event. If a component has reached its originally intended service life and is required to operate for a longer period, then its TLAA needs to be reviewed and revalidated with respect to the longer service period. It is also necessary to verify whether the conditions or limits imposed on the component, as a result of the analysis, do not generate new TLAAs. If new TLAAs are required, the review process should identify all measures needed for a successful outcome of the new analyses. The new TLAAs will ultimately verify whether such measures will be effective and valid throughout the entire extended service period. In the case of unfavourable results, corrective action(s) need to be taken to ensure that the effects of ageing on the intended functions will be adequately managed throughout the entire period.

The anticipated minimum scope of TLAAs to be revalidated (or of new TLAAs) is:

- Crack growth analysis (flaws in structures and components that were found during ISI);
- Analysis of the RPV resistance against brittle/fast fracture (PTS analyses);
- Determination of the RPV pressure–temperature dependence curves (e.g. pressure–temperature curves for normal conditions, interference modes, pressure tests and auxiliary modes);
- Low cycle thermal and mechanical fatigue of the main primary mechanical structures and components;
- Thermal stratification of pressurizer surge line;
- Thermal stratification of steam generator feedwater inlet nozzles;
- Consequences of postulated high energy piping ruptures (with respect to fatigue);
- LBB analysis;
- Vibrations of reactor internals caused by the primary coolant flow;
- Decrease of fracture resistance of reactor internals;
- Decrease of fracture resistance of ferritic cast steel structures and components caused by thermal ageing;
- Decrease of fracture resistance of heterogeneous welds of austenitic primary piping caused by thermal ageing;
- Acceptability of wall thinning of primary components due to corrosion with respect to LTO (corrosion allowance);
- Flying fragments and their effects causing possible damage to primary and secondary active devices;
- Environmental impact on electrical and I&C equipment, structures and components (Safety Classes I–III);
- Impact of ageing effects on confinement penetrations;
- Ageing of confinement structures (welds and transition welds);
- Ageing of cranes, which may affect safety function performance;
- Ageing of spent fuel pit paint coats;
- Changes in properties of heavy concrete materials due to their ageing;
- Impact of ageing on structures and the important functions of buildings.

Requirements regarding the TLAA documentation submissions are:

- Terms of reference;
- Computational model(s), tests and measurements;
- Operating parameters, history of operational loadings, etc.;
- Material properties/parameters;
- Computational analyses, tests, measurements and their evaluation;
- Results of calculations, tests, measurements and their evaluation;
- Conditions for validity of each analysis with respect to structure and component ageing;
- Conclusions, recommendations and proposed corrective actions.

7.2.4. Generic organizational structure of an ageing management programme

The organization of an AMP, in terms of the main elements making up a generic programme, is shown in Fig. 53. The central column shows the main sequence of activities; to the left and to the right are shown the supporting data. Specifically, the boxes on the left show the data from the ageing knowledge base and the technology and engineering tools used to implement the programme. The boxes on the right show the direct input data from

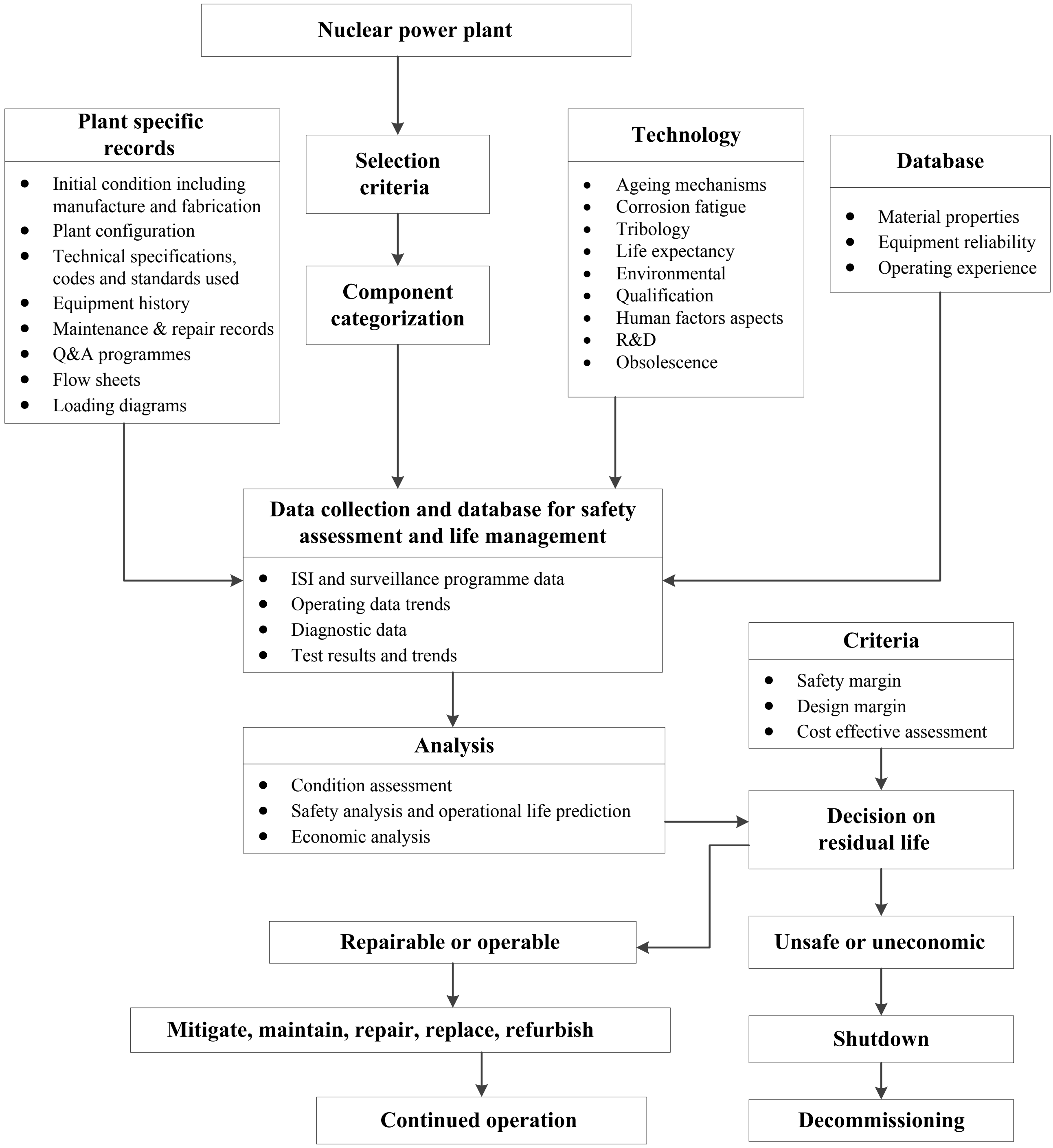


FIG. 53. Main elements of an AMP.

the plant specific operation records. The boxes at the bottom show the acceptance criteria and the decision making processes.

An example of a specific AMP for a nuclear power plant is shown in Fig. 54. This AMP was developed for Ringhals, the Swedish nuclear power plant south of Gothenburg, comprising one BWR and three PWRs. In the top row, under AMP, the division of responsibilities in handling SSC ageing is based on a traditional grouping by discipline. The subprogrammes of the overall AMP are shown in the rows below.

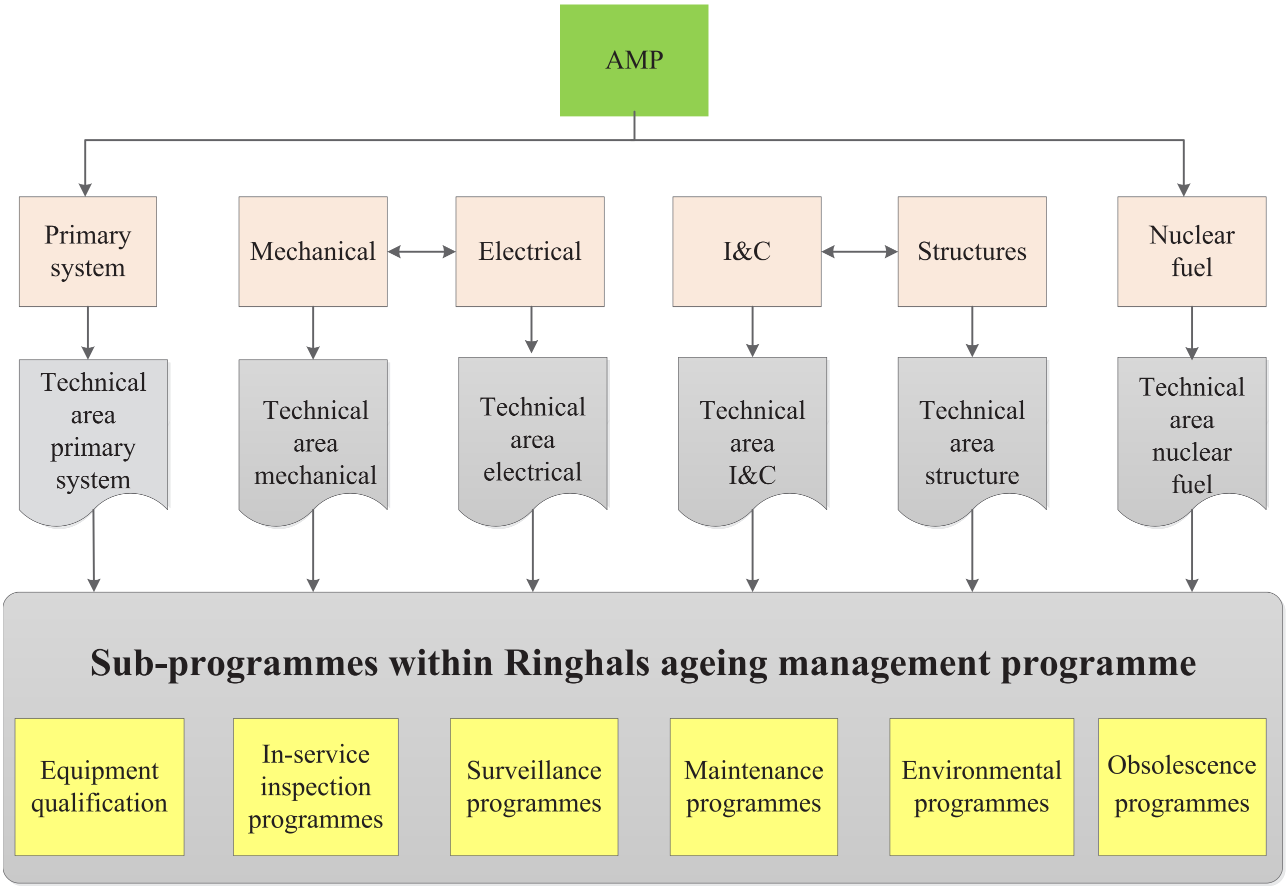


FIG. 54. AMP structure of Ringhals nuclear power plant, Sweden.

7.3. VARIOUS NATIONAL APPROACHES TO LICENSING IN AGEING MANAGEMENT

This subsection contains the licensing approaches on ageing management of three Member States: the USA, France and Hungary. Each represents one of the three licensing approaches discussed in Section 7.2.2:

— The USA representing the LRA approach;
— France representing the PSR approach;
— Hungary representing the combined approach.

7.3.1. Licensing requirements in the United States of America

In the USA, the NRC issues initial licences for commercial power reactors to operate for a maximum term of 40 years and allows these licences to be renewed for an additional 20 years through the LRA system. The maximum licence term of 40 years was selected on the basis of economic and antitrust considerations, not technical limitations. There is no limit on the number of licence renewals, as long as the plant can be shown to continue to be run safely and in accordance with environmental requirements. The decision whether to seek a licence renewal rests entirely on the nuclear power plant owners, and is typically based on economic factors and safety considerations.

An LRA may be submitted to the NRC as early as 20 years after first operation and no later than 35 years of operation. This 15 year window of opportunity was set by the NRC to allow enough time for the owner/operators to develop their long term business plans and to allow the NRC staff enough time, at least 5 years, to perform a proper review of the LRA. A typical NRC review takes anywhere from 22 to 30 months. When significant intervention is involved, the review may take even longer than 5 years, due to the adjudicatory process, but the plant is allowed to continue operation. The licence renewal process proceeds along two tracks: one focused on the review of safety issues (10 CFR 54) [72] and the other on the review of environmental issues (10 CFR 51 [22]).

Code 10 CFR 54.21 [73] describes the licence renewals, and requires each renewal application to include:

— An integrated plant assessment;
— An evaluation of the changes to the current licensing basis;
— An evaluation of TLAAs;
— An FSAR supplement.

The NRC has issued regulatory guides and NUREG documents to address details and questions regarding the regulatory process for an LRA. In addition, the NEI has developed, in support of its members, an industry guidance document, NEI 95-10, Revision 3 [7], on how to best prepare an LRA. The NEI report has been endorsed by the NRC in its regulatory guide 1.188 [74]. All regulations, guidance documents and background information on the licence renewal process are available to the public on the NRC web site at www.nrc.gov. The standard review plan for the licence renewal sections is keyed to the regulatory guide 1.188 format [74], namely the sections are numbered according to the section numbers of the guide.

In the LRA, the applicant provides the information needed to demonstrate that any changes to the safety and performance envelope, including the effects of ageing, will meet the licence renewal requirements during the entire LTO period. In order to fulfil this goal, an applicant should analyse the effects of ageing on the plant SSCs to ensure that actions have been or will be taken to:

— Show the effectiveness of the long term management of ageing during the period of extended operation. This determination should be based on the functionality of all SSCs subject to an ageing management review and on the effectiveness of their ageing programmes during the LTO.
— Perform and incorporate any changes (either in hardware or in administration, programmes and procedures) implied by the TLAA on all components requiring it.

The GALL report [9] is treated as an approved topical report. The NRC reviewers do not repeat their review of a matter described in the GALL report. They usually find an application that references the GALL report and confirm its acceptance if the evaluation of the matter in the GALL report applies to the plant. If the applicants rely on the report for licence renewal, the staff seeks only to ensure that the material presented in the GALL report is applicable to the specific plant and that the applicants have identified specific programmes as described and evaluated in the GALL report. The results of the GALL effort are presented in a table format in the GALL report, Revision 2. The table column headings include:

— Item;
— Link;
— Structure and/or component;
— Material;
— Environment;
— Ageing effect/mechanism;
— AMPs;
— Further evaluation.

The terms of how the NRC reviews are conducted are spelled out in the NRC standard review plan for LRAs (SRP-LR) of nuclear power plants (NUREG-1800) [75]. The SRP-LR is divided into four major topics:

— Administrative information;
— Scoping and screening methodology for identifying structures and components subject to ageing management review, and implementation results;
— Ageing management review;
— Time limited ageing analyses.

The SRP-LR addresses various site conditions and plant designs, and provides complete procedures for all of the areas of review pertinent to each of the SRP-LR sections. For any specific application, NRC reviewers may select and emphasize particular aspects of each SRP-LR section, as appropriate for the application. For these and similar reasons, the NRC reviewers need not carry out in detail all of the steps listed in each SRP-LR section in the review of every application. The subsections are:

— Areas of review: This describes the scope of review, that is, what is being reviewed by the branch that has primary review responsibility. It contains a description of the SSCs, analyses, data or other information that are reviewed as part of the LRA. It also contains a discussion on the information needed or the review expected from other branches to permit the primary review branch to complete its review.
— Acceptance criteria: This contains a statement on the purpose of the review, an identification of applicable NRC requirements, and the technical basis for determining the acceptability of programmes and activities within the area of review of the SRP-LR section. The technical bases consist of specific criteria, such as NRC regulatory guides, codes and standards, and branch technical positions. Consistent with the approach described in NUREG-0800 [76], the technical bases for some sections of the SRP-LR can be provided in branch technical positions or appendices as they are developed and can be included in the SRP-LR.
— Review procedures: This discusses the manner in which the review is accomplished. It is generally a step by step procedure that the reviewer follows to provide reasonable verification that the applicable acceptance criteria have been met.
— Evaluation findings: This presents the type of conclusion that is sought for the particular review area. For each section, a conclusion of this type is included in the safety evaluation report, in which the reviewers publish the results of their review. The safety evaluation report also contains a description of the review, including:
  • Which aspects of the review were selected or emphasized;
  • Which matters were modified by the applicant, or required additional information, will be resolved in the future, or remain unresolved;
  • Where the applicant's programme deviates from the criteria provided in the SRP-LR;
  • The bases for any deviations from the SRP-LR or exemptions from the regulations.
— Implementation: This discusses the plans of the NRC staff in using the SRP-LR section.
— References: This gives a list of the references used in the review process.

These analyses need to include:

— The effects of ageing on the SSCs;
— A consideration of the effects of time limited assumptions, defined by the current operating licence and determined to be relevant to safety;
— A justification for the presumed capability of the SSCs to perform their intended functions;
— All calculations contained or incorporated by reference in the current licensing basis.

A screening of SSCs involving TLAAs needs to be conducted. The TLAAs then need to be evaluated to ensure the following:

— The analyses remain valid for the period of extended operation;
— TLAAs are projected to the end of the licence renewal period;
— Ageing can and will be managed adequately for the period of extended operation.

An example of TLAAs submitted by Exelon to the NRC is shown in Table 5.

7.3.1.1. Organizational structure

The organization of PLiM in the USA varies widely from plant to plant. For example, some plants may have a dedicated PLiM organization that coordinates the PLiM activities and provides the PLiM study results to the appropriate plant organizations (e.g. maintenance, engineering or operations) for implementation. Other plants may not have a dedicated PLiM organization, but rely on various organizations (e.g. design engineering, system engineering or maintenance) to conduct the required PLiM studies for LTO on a case by case basis, for example,

TABLE 5. EXAMPLE OF TIME LIMITED AGEING ANALYSIS TABLE PREPARED BY EXELON

TLAA	Description	Disposition category
1.	Reactor vessel neutron embrittlement	
	10 CFR 50 Appendix G reactor vessel rapid failure propagation and brittle fracture considerations: Charpy upper shelf energy reduction and RTND increase, re-flood re-shock analysis	Revision of the analysis and validation of the analysis for the period of extended operation
	Reactor vessel thermal limit analyses: operating pressure-temperature limit (P-T limit) curves	Revision of the analysis
	Reactor vessel circumferential weld examination relief	Revision of the analysis
	Reactor vessel axial weld failure probability	Validation of the analysis for the period of extended operation
2.	Metal fatigue	
	Reactor vessel fatigue	Management of the ageing effect
	Reactor vessel internals fatigue and embrittlement	
	Reactor vessel internals fatigue analyses	Validation of the analysis for the period of extended operation and management of the ageing effect
	Effect of fatigue and embrittlement on EOL re-flood, thermal shock analysis	Validation of the analysis for the period of extended operation

Note: RTND — reference temperature of nil ductility transition; EOL — end-of-life.

by component groups (e.g. piping, cables or transformers) or by individual components (e.g. turbine generators or steam generator).

The organizational structure for preparing an LRA varies from utility to utility, but a typical structure is based on creating a project team to address each major discipline of the needed studies. The team members may be from utility organizations, contracted consultants who specialize in preparing LRAs or a combination of the two groups, which is most common. A typical organizational structure is shown in Fig. 55.

The team typically consists of 10–20 full time members, with several other part time support members from various expert organizations (utilities and contracted consultants) needed to perform the engineering, ageing management and environmental studies, and to prepare the supporting documentation for the LRA project.

7.3.2. Licensing requirements in France

In France, the regulatory authority is the Autorité de sûreté nucléaire (ASN). Its technical support organization (TSO) is the Institut de radioprotection et de sûreté nucléaire, the Institute for Radiological Protection and Nuclear Safety. The ASN does not issue a licence for a specified period of time. The safety authorities give an authorization to restart each unit after reloading at the end of each cycle (approximately every 12–16 months, depending on the reactor design series and the fuel cycle).

With the intent to minimize the need for long outages, an agreement was reached between EdF and the safety authorities, to implement modifications at the 10 year safety review milestone, during which a complete safety check of the unit is performed, as prescribed by regulations.

All modifications are defined, taking into account the results of the PSR, performed not unit by unit, but at the same time for the entire series of the same design. This occurs before the 10 year outage of the first unit of the series. According to the PSR results and to the context and the implementation plan of the modification batch

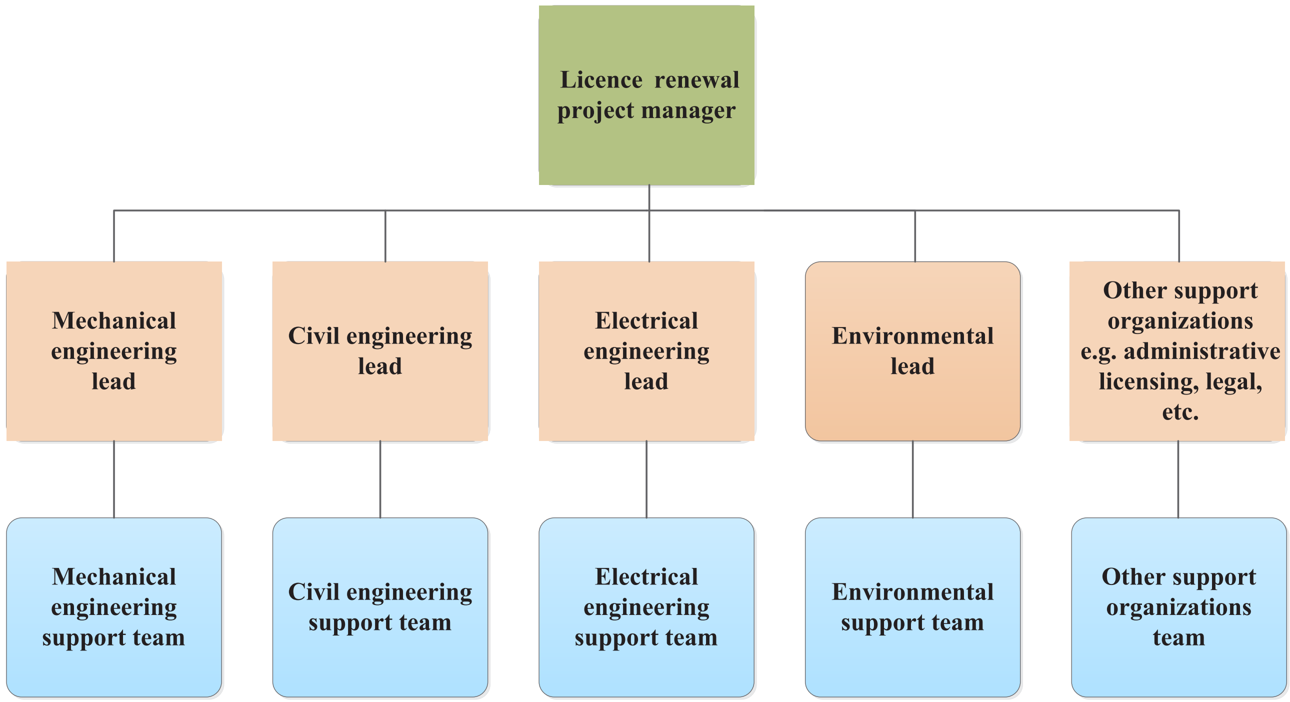


FIG. 55. Typical licence renewal project team organization chart.

proposed (the same for all the units of the series), the series is allowed operation for 10 more years (except if a specific problem on one unit becomes a common case and a generic action item).

Presently, the oldest 900 MW units of the CP0 and CP1 series, have been authorized to operate for 30 years, and a justification file has been submitted for extending their operation to 40 years. The safety authority has publicly expressed the opinion that it will not consider a life extension request before the third PSR cycle, and not for more than 10 years at a time.

7.3.2.1. Organizational structure

In preparation for the 10 year outages, an AMP is in force at EdF in order to justify that all SSCs that may be affected by an ageing mechanism remain within the applicable design and safety criteria. The AMP procedure is carried out in three main steps, in agreement with French regulations and with IAEA Safety Guide NS-G-2.12 [1]:

— Selection of safety related SSCs affected by an ageing mechanism;
— Review of all SSCs subjected to a degradation mechanism, as selected by the experts using ageing analysis, in which maintenance resource requirements, difficulty of repair and replacement, and obsolescence are taken into account;
— Preparation of detailed ageing management reports required for sensitive components (e.g. the RPV, reactor internals, civil structures, I&C or electrical cables).

Each nuclear power plant provides a plant ageing management report to the nuclear safety authority at least 12 months before the 10 year outage. Relying on a thorough ageing analysis of the plant safety components, the report needs to justify the ability of the plant to operate for 10 more years.

A detailed, systematic procedure is available in France to review the consequences of ageing on safety components. A key condition for the success of the ageing management review is to ensure the licensee has an effective understanding of ageing mechanisms and an efficient integration system of operating feedback. The LTO strategy is illustrated in Fig. 56, and includes the following:

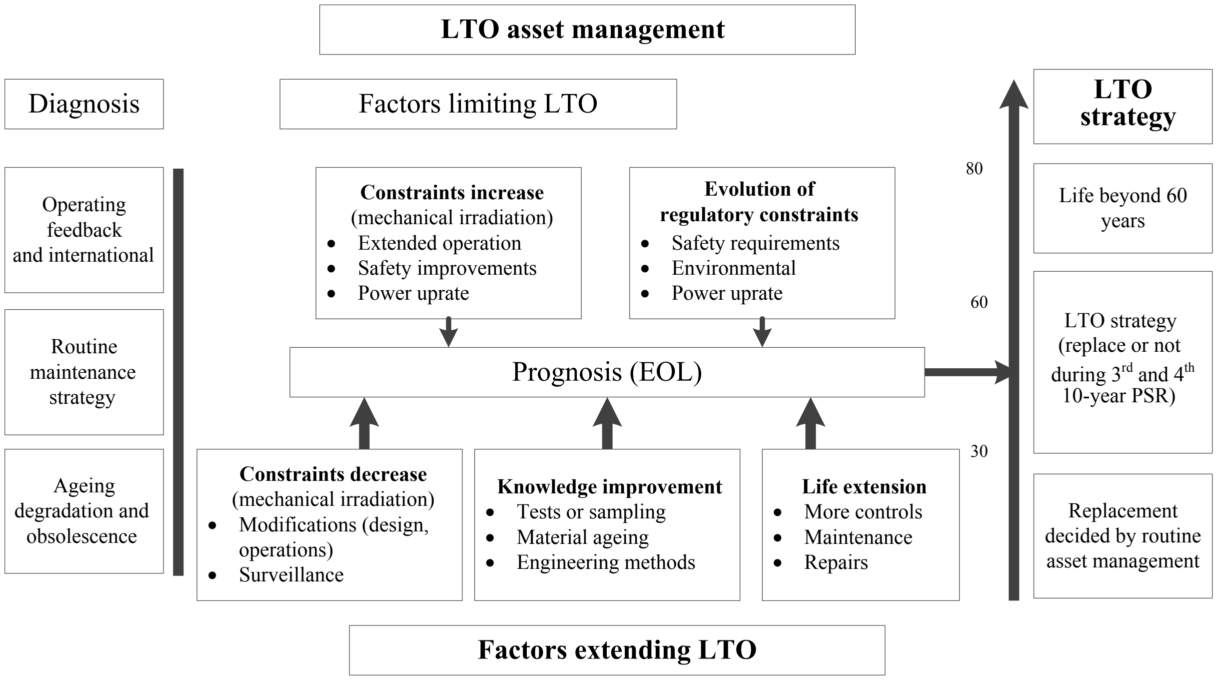


FIG. 56. LTO decision making process for major nuclear power plant components. EOL — end-of-life.

— A diagnosis of the state of the plant based on ageing analysis and operating feedback;
— A prognosis on the ability of the main components to continue operation (estimated end-of-life criteria), taking into account LTO limitations as well as all factors facilitating LTO;
— A strategy document (asset management actions), including a special maintenance programme specifically focused on LTO requirements.

The strategy is defined according to the estimated end-of-life, namely:

— If the estimated end-of-life is beyond 60 years, the strategy is periodically updated by the Executive Committee. At the same time, a component replacement feasibility dossier is prepared to cope with obsolescence and spare parts and with the repair/replacement process. The dossier is then used as guidance to adequately plan strategic modifications and cope with unexpected demands for replacements. Moreover, an R&D programme on issues covering material research, evolution of methods, NDT evaluation and configuration modifications, among others, is established in support of the strategy.
— If the estimated end-of-life is between 40 and 60 years, the LTO strategy includes a special maintenance programme, periodically revised by the Executive Committee. In order to justify an LTO investment from a technical and economic point of view and to help in planning it correctly (this usually happens between the third or fourth 10 year outage), a decision making tool is used and specific methods are implemented to test various schedules and assess the consequences on safety and operating conditions.
— If the estimated end-of-life is less than 40 years, the replacement/refurbishment of components is decided by the Executive Committee as part of the routine asset management programme, aiming at an extension of operations of up to 60 years.

7.3.3. Licensing requirements in Hungary

In Hungary, a comprehensive regulatory system has been specifically developed to oversee the safety of the four WWER-440/V-213 reactors at the Paks nuclear power plant for the LTO period. Compliance with the current licensing basis is periodically verified, via an annual updating of the FSAR and routine regulatory inspections and approvals. The FSAR updating needs to reflect the actual plant configuration and demonstrate compliance with the current licensing basis and include a design basis update.

The PSR process is used as a self-assessment tool by the licensee, and is reviewed and approved by the regulator. However, PSR is not considered a licensing tool in Hungary. It is performed every 10 years, primarily to assess the overall ageing of the SSCs on a timescale broader than the routine yearly checks. The broader timescale allows the reviewers to better take into account the development of science and technology in relation to safety and ageing. The content of the PSR is very similar to that described in IAEA's related safety guide [77].

Licence renewal is the formal process used in Hungary to apply for an operating licence extension beyond the original design term. This process is similar to the one governed by the licence renewal rule in the USA, with some notable deviations.

The licensee has to prepare and submit the LTO programme to the regulator, no later than 4 years before the licence expiration, but also not before having completed 20 years of unit operation. During the 4 years prior to the licence expiration date, the regulatory authority exercises continuous oversight over the licensee's LTO programme to ensure all tasks in the programme are performed as planned.

A formal LRA has to be submitted 1 year before the licence expiration date. The application needs to include an assessment of the 3 years of operation conducted in accordance with the LTO programme and demonstrate that the LTO programme is effective, that it ensures continued safe operation, and that the observed trends match the forecasts made to justify the safety of the plant throughout the extended operating period.

7.3.3.1. Organization of plant life management for long term operation

The PLiM organization for LTO is of the functional type, drawing on all PLiM related disciplines at the Paks nuclear power plant. The specialty items required for LTO are supported by the following specialized departments in the plant:

- Design;
- Fabrication;
- Procurement (including the rating);
- Erection (construction, assembly and installation);
- Commissioning;
- Operation covering;
- Operational tests;
- Monitoring and surveillance;
- ISI;
- In-service testing;
- Condition monitoring;
- Preventive maintenance;
- Corrective maintenance;
- Maintenance effectiveness monitoring (Hungarian maintenance rule);
- Spare parts management;
- Configuration management;
- Ageing management;
- Equipment qualification;
- Replacement and reconstruction;
- Education;
- Asset management and economy planning;
- Control of the plant safety.

7.3.3.2. Organization for developing an operating licence extension application

A formal project team (of eight employees) was set up for the preparation of the Paks LTO programme and the LRA. A project manager and technical deputy directed the execution of the project tasks prepared by the project team.

The project team followed a project plan, approved by management, which included the technical tasks and the budget details. The project team relied on the technical support department in the role of in-house TSO, on dedicated Hungarian engineering firms and universities, as well as on independent organizations for review tasks. The relationship between the project team and other in-house teams can be seen in Fig. 57. The project staff is responsible for several tasks, including the following:

- — Implementation of the project technical tasks;
- — Coordination between the internal and external experts and the organizations involved;
- — Recording and filing of the analyses and of other project related documentation;
- — Preparation of regular progress reports for project management and for the nuclear authority;
- — Organization of technical meetings;
- — Adherence to budget and deadlines.

An in-house expert team helps with the project technical tasks. It includes dedicated experts from most of the technical sections or divisions, with responsibilities in their respective areas of expertise. This team also helps with the in-house verification process and the approval phases. These activities and relationships are represented in Fig. 58.

Management regularly reviews progress against the project plan. Weekly meetings chaired by the head of the technical support department are held to discuss progress and future activities, and a monthly meeting chaired by the plant general manager is held to reconcile issues that may have arisen, to fill gaps and to resolve issues. The formal application for licence renewal is reviewed and commented upon independently by foreign expert firms (e.g. Entergy Ltd) and by a formal SALTO mission conducted by the IAEA.

7.3.4. Member States following the periodic safety review approach

Those countries that have no limited licensing term have adopted a PSR process that is usually conducted every 10 years. For each PSR, the operator reviews the plant configuration in light of operating experience, new R&D findings and lessons learned. Each safety review is intended to prove that plants have upheld their safety and the terms of their operating licence over the years. Each PSR is an extensive undertaking.

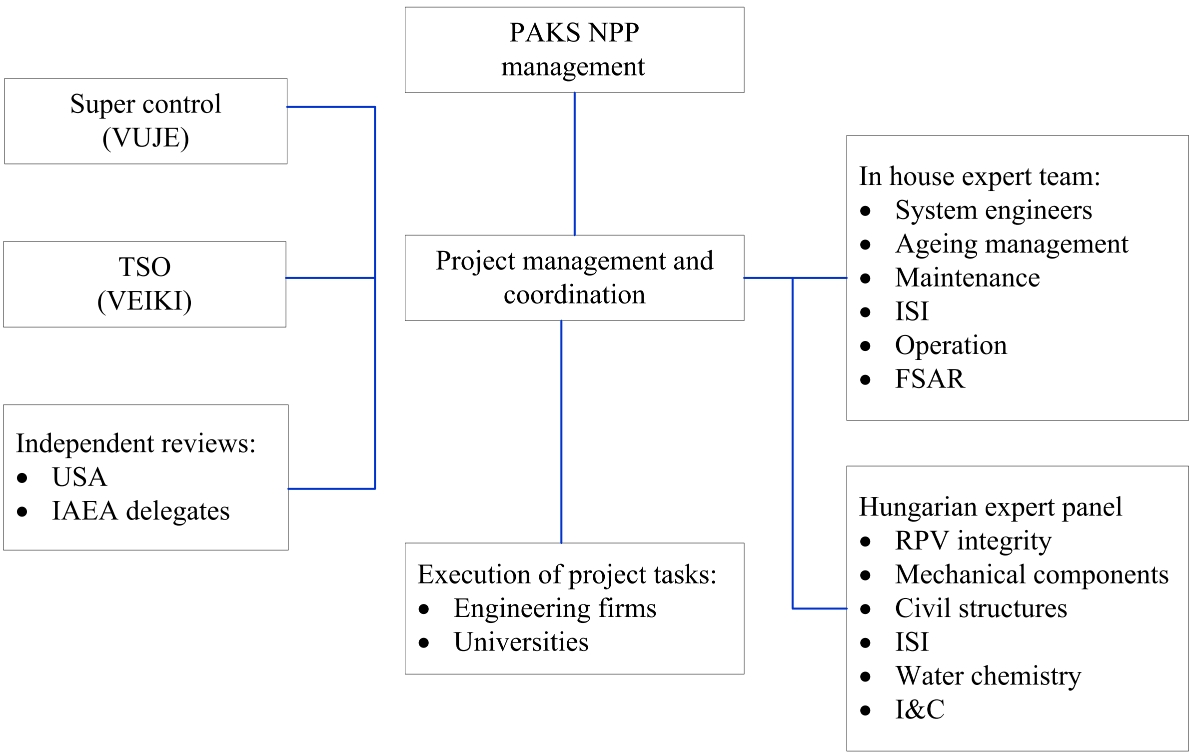


FIG. 57. Project organization for developing a LTO programme. VEIKI — an independent testing laboratory and product certification body; VUJE — Nuclear Power Plant Research Institute Trnava.

The countries applying the PSR approach are Argentina, Armenia, Belgium, Bulgaria, France, India, Japan, the Republic of Korea, the Netherlands, Pakistan, Russian Federation, Slovakia, Sweden, Switzerland, South Africa and Ukraine. These Member States continue to conduct PSRs even for their LTO period, although some regulators may impose additional requirements for that period.

7.4. CODES AND STANDARDS

Standards are norms, principles and methods whose purpose is to ensure that within the standard area of implementation, products and services are provided in a safe, reliable and quality manner. Standards are usually live documents that are periodically updated and maintained, and new ones are developed as needs arise.

Organizations that produce standards also issue certifications and stamps of approval to product and service providers. Doing so ensures that certified products conform to standards.

From the product and service provider's viewpoint, standards help to minimize errors and waste, foster public confidence and acceptance of their products, and ultimately help to increase productivity and profits. If a producer follows international standards, it can be easier to penetrate new markets. From the consumer's viewpoint, standards guarantee product quality and safety, and allow direct comparisons of different suppliers and different markets to help in the search for value. From the viewpoint of government and local authorities, standards help to protect public health and safety, and facilitate trade and market fairness.

7.4.1. International

The International Organization for Standardization (ISO) is an independent, non-governmental developer of international standards. The ISO standards cover every facet of productivity and services including manufacturing, technology development and implementation, agriculture, food, product distribution and healthcare.

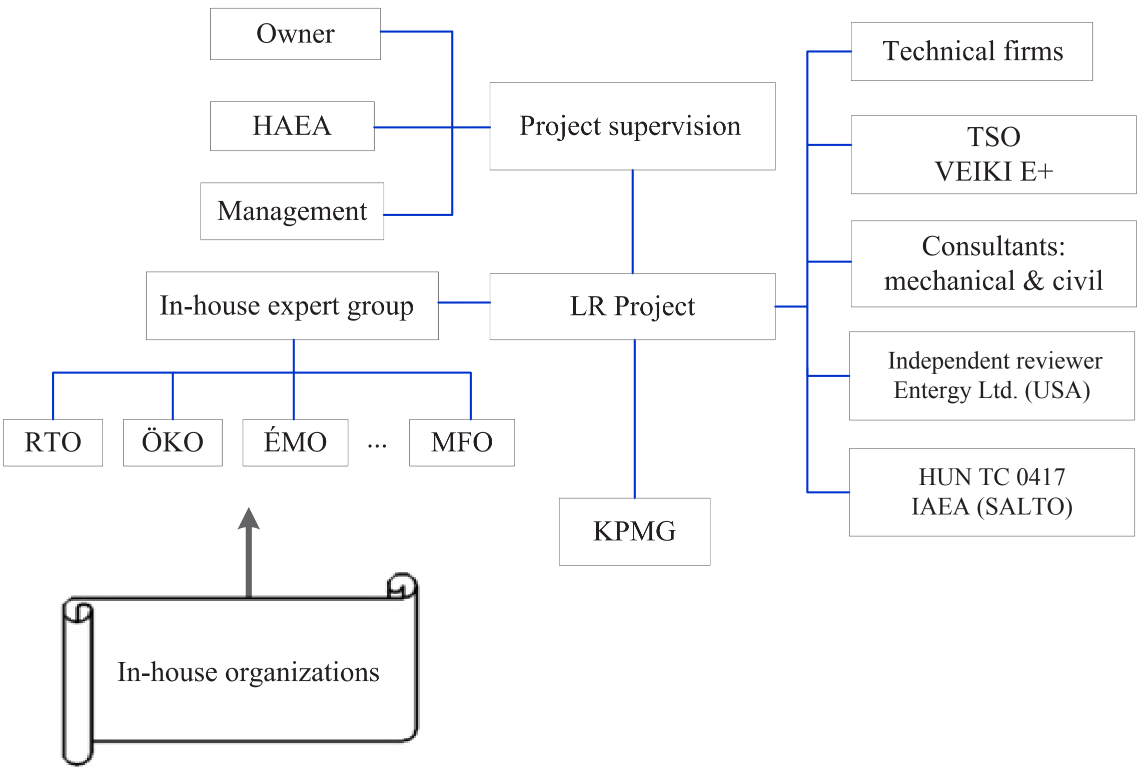


FIG. 58. Project organization for developing an LRA report. HAEA — Hungarian Atomic Energy Authority; RTO — system engineering section; ÖKO — ageing management section; ÉMO — architect engineering section; MFO — technical support division.

The ISO standards are used both in conventional industry and in the nuclear industry. The most popular ISO standards in use in the nuclear industry cover quality management, environmental management, social responsibility, energy management, risk management, information security management and sustainable events.

Other international organizations produce norms and guidance on ageing management. Among these are the IAEA (see Section 8.6.1), OECD/NEA and NEI.

Even before international standards appeared, many national standards had been developed to address local or technology related needs to foster quality, consistency, uniformity and various other national interests. The organizations developing national standards also issue certifications. The national codes and standards most used and recognized in the nuclear industry are those of Canada, France, Germany, the Russian Federation and the USA.

7.4.2. Canada

In Canada, the Nuclear Safety and Control Act establishes the regulatory framework for nuclear power. The Canadian Nuclear Safety Commission develops the regulatory documents that contain requirements and guidance for the industry. Beyond its own requirements, the Canadian Nuclear Safety Commission relies also on the CSA nuclear standards and on other international standards on design, materials, fabrication, construction and performance of nuclear power plants to evaluate licence applications and the licensee's qualifications.

Depending on the province or territory in which nuclear power plants are located or nuclear power related activities are conducted and pressure retaining components are designed, manufactured, fabricated and installed, provincial legislation requires owner/operators of such equipment to comply with the provincial or territorial technical standards and safety act. The task to ensure the act is followed, and the operational safety of pressure retaining components and their supports is promoted and enforced, is assigned to the provincial Technical Standards and Safety Authority, which reviews the design of new technology, new installations, alterations and modifications to existing equipment for compliance to codes and regulations. It inspects/audits trades people, contractors, boilers and pressure vessels and their support systems and installation for compliance with codes and regulations. It trains and certifies equipment operators and monitors developing safety related trends or issues and supports code development and standard setting.

7.4.3. France

In France, regulatory requirements and applicable codes and standards are mandatory references for the design and construction of nuclear power plants. Construction codes and associated standards in use in France are the European and ASME codes, American Society for Testing and Materials (ASTM) and national standards.

The first French 900 and 1300 MW reactors were manufactured under the Westinghouse licence, and their design was based on ASME Section III. National codes and standards began to be produced in 1978.

National standards have been developed by representatives of the nuclear industry for the French association for nuclear codes and standards (AFCEN). Their aim was to minimize risks, address safety concerns, integrate feedback and experience, and take into account national economic interests.

7.4.4. Germany

In Germany, the nuclear industry is governed by the Atomic Energy Act. A series of ordinances detail the requirements of the act by sector. Both the Act and the ordinances are legally binding throughout the federal lands in Germany and include administrative provisions binding only in local jurisdictions. Relevant to the nuclear industry are:

— Regulatory documents of the Federal Ministry of the Environment;
— Recommendations of the Reactor Safety Commission;
— Nuclear technical standards issued by the Nuclear Safety Standards Commission (KTA).

The lowest level below the administrative provisions are the technical.

KTA consists of 50 members representing the German nuclear industry including the manufacturers, utilities, and safety and licensing authorities, among others. The safety standards contain the opinion and the experience accumulated from the licensing, construction and operation of nuclear power plants. KTA safety standards are officially issued by the Federal Ministry for the Environment, Nature Protection and Nuclear Safety. Every 5 years, or earlier, KTA standards are reviewed and updated if they no longer represent modern practices or if they have been challenged.

7.4.5. Russian Federation

The Russian codes and standards for nuclear power plant ageing management and for longer term operation are part of the regulatory framework that is governed by the legislative acts of the president and government implemented in the federal norms and rules for nuclear energy. Article 9 of the federal laws of the Russian Federation deals with the use of nuclear energy. Within its mandate, the president and the Government of the Russian Federation issue the regulatory legislative acts.

7.4.6. United States of America

In the USA and in some of States operating US designed nuclear power plants, in the context of ageing management and safety management, the NRC approach has been that of developing regulations and rule setting based on state of the art R&D, operations feedback and current knowledge in support of design, construction, operation, certification and licensing.

The NRC often refers to the ASME code in the nuclear power plant pre-certifications of new designs, in the combined licence application reviews, in licence renewals and virtually in all its other regulatory activities. ASME is the author of the universally accepted boiler and pressure vessel code, which has become an international code of reference in nuclear and conventional industries. The ASME code is a live collection of design, fabrication, construction and inspection rules and methods in support of the life cycle of SSCs in conventional and nuclear power plants. It is divided into sections, divisions and subsections. In addition, a very extensive set of appendices contains details on specific subject matters.

In cooperation with the American National Standards Institute, ASME has issued a series of complementary codes for the benefit of the nuclear industry. ASME also maintains various working committees to coordinate, promote and foster the development of its standards. One such committee is the ASME Verification and Validation (V&V) Standards Committee, which has developed procedures for assessing and quantifying the accuracy and credibility of computational models and simulations:

- V&V 10: Verification and Validation in Computational Solid Mechanics;
- V&V 20: Verification and Validation in Computational Fluid Dynamics and Heat Transfer;
- V&V 30: Verification and Validation in Computational Simulation of Nuclear System Thermal Fluids behaviour;
- V&V 40: Verification and Validation in Computational Modeling of Medical Devices.

In support of the industry, ASME issues certification, stamps and accreditations to companies in the nuclear industry in over 75 countries, in addition to the USA. Organizations in the nuclear industry with quality assurance programmes in accordance with ASME section III can apply for ASME certification and stamps, namely:

- NQA-1 for quality certification for an organization supplying components or services providing a safety function for nuclear facilities. ASME recognizes the organization’s quality assurance programme to be in conformance with the requirements of the NQA-1 standard. However, this certificate does not prequalify or exempt the organization from a qualification audit by the purchaser of the items or services it provides.
- N for vessels, pumps, valves, piping systems, storage tanks, core support structures, concrete containments and transport packaging.
- NA for field installation and shop assembly of all items.
- NPT for parts, appurtenances, welded tubular products and piping subassemblies.
- NS for supports.
- NV for pressure relief valves.
- N3 for transportation containments and storage containments.
- OWN for nuclear power plant owner.

In order to deal with nuclear power plant ageing issues and to prepare for licence renewal, it became necessary to establish a framework of regulations, codes and standards to manage SSCs. In this respect, the ASTM and the American Nuclear Society updated their material standards referenced by the NRC in their regulations to better serve the nuclear industry. ASTM members delivered test methods, specifications, guides and practices to support the nuclear industry and the regulator. Today, ageing in nuclear power plants is regulated in the USA by the Maintenance Rule, 10 CFR 50.65 [78]. In addition, guidance documents provide assistance regarding implementation of the Maintenance Rule.

In the USA, operating licences are not subjected to PSRs as in other jurisdictions, and a nuclear power plant licence is valid to the end of the licence period (usually 40 years). Instead, the NRC regularly reviews the lessons learned from nuclear power plant operating experience and, whenever needed, issues a new regulation (NUREG) that is enforceable within a certain time frame and eventually forms part of nuclear power plant licences. At the end of the licence period, a licensee can submit an LRA to the NRC.

The licence renewal process is governed by 10 CFR Part 54 [72]. Operators are required to provide an integrated plant assessment, an evaluation of TLAAs, a supplement to the FSAR and any necessary changes to the plant technical specifications (along with related justifications) in accordance with the NRC regulatory guide 1.188 [74].

A systematic tool used by the NRC is the GALL report [9], which contains the NRC staff evaluation of the existing plant programmes. The information in the GALL report has been incorporated into NUREG-1800 [75], as directed by the NRC, to improve the efficiency of the licence renewal process.

At the time of the application for the construction licence, an operator is required to submit a report on the potential environmental impact of the proposed nuclear power plant and associated facilities. This report is regulated by regulatory guide 4.2 [79] and regulatory guide 4.2S1 [80].

Renewal of a nuclear power plant operating licence requires an update to the original environmental assessment through the preparation of an environmental impact statement, regulated by NUREG-1437 volumes 1–3 and supplements [81–83]. These regulatory guides are periodically reviewed and supplements are issued as the NRC gains experience from plant sites that have undergone environmental reviews in the context of their LRAs.

7.5. SUMMARY AND RECOMMENDATIONS

This subsection gives a summary of and recommendations on the ageing management of a nuclear power plant previously discussed. The most important issues with respect to safe operation, and in view of preparing the plant for LTO, are the understanding of the ageing mechanisms of the plant SSCs and the proactive management of the impact of ageing on plant safety. The operator's first concern is to ensure, at all times, the integrity of the plant defence in depth barriers and the safety system performance, because these elements are the foundation of any nuclear power plant operating licence. This is achieved by gaining a good understanding of the ageing processes of critical non-replaceable structures and of other SSCs essential to safety and plant viability, through the implementation of a suitable and proven plant ageing management methodology. The tools usually required to reach this understanding are:

— Analysis of operating experiences;
— Ranking of parts and components most subject to failures;
— Classification of failure severity (incipient, degraded or catastrophic);
— Identification of failure mechanisms;
— Early detection methods through plant surveillance testing and maintenance programmes.

The in depth reviews of ageing management should include plant programmes and practices that will be used to support the management of ageing effects, even during LTO. The reviewers should ensure they are consistent with the generic attributes of an effective AMP such as those given in Table 1 of this publication.

The review process should involve the following main steps:

— An appropriate screening method to ensure that structures and components important to safety are evaluated for LTO;

— Demonstration that the effects of ageing will continue to be identified and managed for each structure or component even during the planned period of LTO;
— Revalidation of safety analyses that were developed using time limited assumptions, to demonstrate that they continue to be valid and that the ageing effects will be well managed. This implies that the intended function of a structure or component will remain within the design safety margins even throughout the planned period of LTO.[6]

In order to consider time limited ageing phenomena, it is necessary to evaluate, for each of the risk significant components, the increase in the likelihood of their unavailability with time as described in Ref. [84].

A time limited analysis method should provide a risk sensitivity curve of all essential components as they age. Individual components should then be grouped by type and by system, and the component groups ranked from those with the highest to those with the lowest potential risk impact. The ranking should not be simplified on the assumption that all components age at the same rate. The analysis results need to be coupled with time dependent failure rate models. The total rate of risk increase for the plant is obtained by adding the contributions of individual components.

The application of probabilistic techniques provides an understanding of the relative importance to plant risk of the ageing of critical plant components. In combination with the evaluation of operating experience and the contribution of expert opinion, understanding of the time dependant risk factors is normally required in PSRs or LRAs. It also allows the continuing refinement of general surveillance and maintenance programmes during operations, and guides investment decisions to implement design, reliability, safety and operability improvements throughout the plant service period. Particular attention should be paid to improving on-line monitoring, surveillance, testing and maintenance interventions to help achieve the early detection of material degradation symptoms. The data obtained from surveillance or inspection in various plants worldwide should be collected in an in-plant reliability data system. The data reporting system should include key information needed to evaluate ageing concerns, such as the materials involved, the environmental and operating conditions contributing to non-conformances or failures, instrument readings and an assessment of the capability to detect component degradation in the incipient state before failure. In preventing component failure, it is essential to achieve a correct understanding of the causes of any instrument drifts. Drift errors are caused by deviations in the performance of the measuring instrument (measurement system) that occur after calibration. Major causes are the thermal expansion of connecting cables and thermal drift of the frequency converter within the measuring instrument.

In addition, equipment qualification and the results of safety and other technical, administrative and independent topical reviews and audits, deemed to be related to ageing management, should be addressed in any AMP.

8. INNOVATION TECHNIQUES AND RESEARCH AND DEVELOPMENT

The selection of R&D tasks, the development of new knowledge, and the collection and commercialization of innovative techniques and new information are all activities that usually imply large investments and hence require detailed technical, economic, financial and budgetary justification. New technologies and ongoing R&D activities in the framework of proactive initiatives categorized under the PDCA approach are summarized below.

The centre box in Fig. 59 illustrates the process that facilitates an improvement of the predictive capabilities and the understanding of future evolutions in SSC degradation. This box includes all the proactive initiatives designed to acquire additional knowledge on SSC degradation and predictive capability about possible new

[6] Revalidation of safety analyses with time limited assumptions is an assessment of degradations under normal service conditions applied to safety analyses conducted at the time of first design, on the basis of a specified length of plant operation. It would include fatigue calculations, PTS analysis and equipment qualification of electrical, instrumentation and control cables [1, 2].

degradation and its rate of advancement, particularly in LTO applications. Improvement of the predictive capability is achieved using:

- — The susceptibility, knowledge and confidence chart method, which involves an indexing methodology comprising both theoretical and historical components; it uses 'smart' defaults to account for missing data and a decision model to grade the risk of SSC failure;
- — Foresight acquired through R&D and synergy observations, and by inferring cascade degradation;
- — Knowledge accumulated and recorded in ageing management databases.

In the 'Plan' box are the elements that are needed to appear in the operational plan of a proactive nuclear power plant AMP. These elements certainly include all new regulatory requirements and directives resulting from advancements in R&D understanding and in collective knowledge, as well as ageing mitigating activities and corrective actions resulting from PSRs and LTO studies. As resources are not infinite, priorities should be established with respect to the resources available and these should be reflected in the plan.

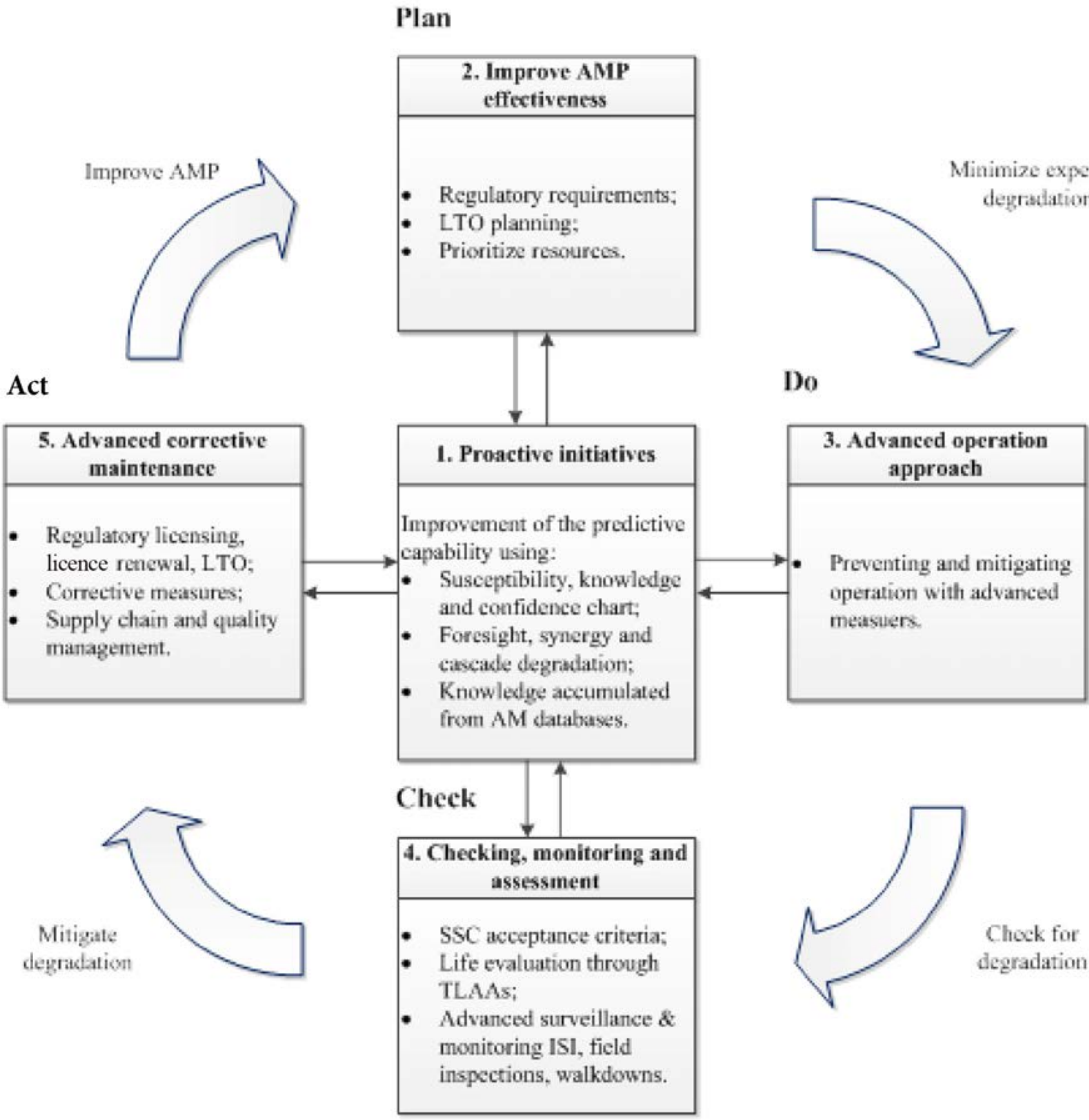


FIG. 59. Proactive plan–do–check–act cycle in a nuclear power plant operating environment. AM — ageing management; SSC — structure, system and component.

The ‘Do’ box lists the operational changes that need to be implemented in the conduct of operation. These may be the adoption of more advanced filters and more effective resins in water treatment, maintaining a more appropriate pH in the primary coolant circuit to slow down wall thinning or corrosion products, or changing materials with high antimony or cobalt content to reduce hot spots and general activity levels in the circuits, among others.

In the ‘Check’ box, all verification and testing activities should be listed. The installation of advanced surveillance and monitoring facilities should be complemented by acceptance criteria for ageing SSCs. In addition, all time limited SSCs should be evaluated for remaining life through TLAAs, and corrective action programmes should be established based on the analyses. An improvement to the ISI programmes as recommended by the proactive initiatives is also desirable, as well as targeted walk downs to areas where SSCs are found to be particularly susceptible to increased degradation rates or to new degradation.

In the ‘Act’ box, preventive and corrective maintenance activities should be listed. The studies conducted in accordance with regulatory and licensing requirements for the LTO and licence renewal programmes normally produce a list of corrective actions and improvements for SSCs. Licensing requirements can include PSRs, and the results of these may yield corrective actions in the plant. Another area of proactive involvement is the ever evolving supply chain and quality management programme. New supplies and new suppliers may not always meet quality standards and specifications. If not corrected at the source, flaws in the materials supplied may be problematic in the management of SSC ageing in the plant.

8.1. RESEARCH AND DEVELOPMENT AND MITIGATION TECHNOLOGIES

Ageing management studies in support of operating nuclear power plants have been conducted in many States as plants approach their first assumed service life. Many insights have been gained and knowledge gaps filled for large safety related components, such as steam generators. Wear assessment programmes have been developed, and results have been incorporated into thermohydraulic codes, FAC and wear assessment codes. In order to develop material reliability programmes, check material integrity and manage material ageing, R&D has been conducted by EPRI and others. For example, R&D has developed a good understanding of the degradation mechanism of Alloy 600, based on which nuclear power plant operators were able to develop appropriate AMPs for Alloy 600 nozzle penetrations and drain nozzles of steam generators in many PWRs.

In the context of a PSR, R&D may be needed to acquire supplementary information or to generate new understanding of degradation effects and their mechanism before designing and implementing any mitigation or corrective actions. The R&D results lead to technical solutions, which can be incorporated together with their justification into the PSR safety documentation. The process to turn the R&D understanding into engineered solutions and operational practice is shown in Fig. 60.

The procedure starts with an examination of the R&D results to acquire a proper understanding and derive adequate corrective or mitigating measures. If further studies are not required, the information applied to the ageing issue is developed and properly incorporated into the outage plan and described in the PSR documentation. If further studies are required, before being able to engineer solutions and incorporate them into the outage plan, R&D develops the concept capable of adequately mitigating or correcting the consequences of the degradation. Practical measures then need to be engineered and justified, and their implementation appropriately planned. The R&D study results and the engineered solution with its justification and verification can be reported in the PSR documentation. If the engineered solution requires supplementary vendor information, the latter needs to be obtained, before submitting the PSR documentation and the implementation plan to the regulatory body.

8.2. CHINA

The RPV ageing management documentation system used for Qinshan phase 1 is shown in Fig. 61. At the centre in yellow is the PDCA cycle, a systematic approach to ageing management spearheaded by the IAEA. Beginning with the planning phase, the AMP coordination group, after receiving feedback from the field and from operating experience, and after building a sufficient understanding of all RPV degradation mechanisms, develops the RPV AMP and the ageing management procedures. For the ‘DO’ step, operation, represented by the yellow

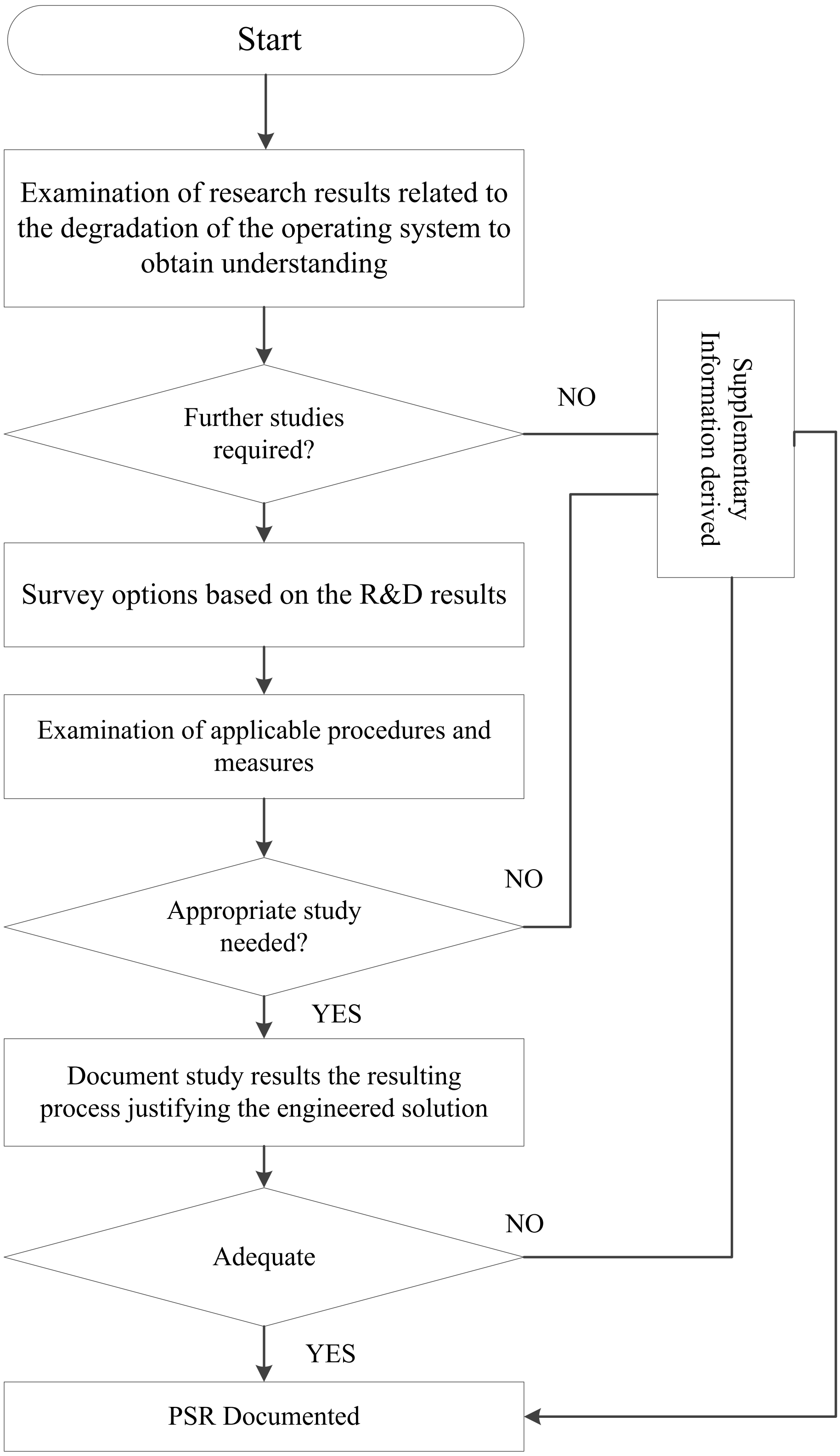


FIG. 60. Procedure to incorporate research and development findings into the PSR documentation.

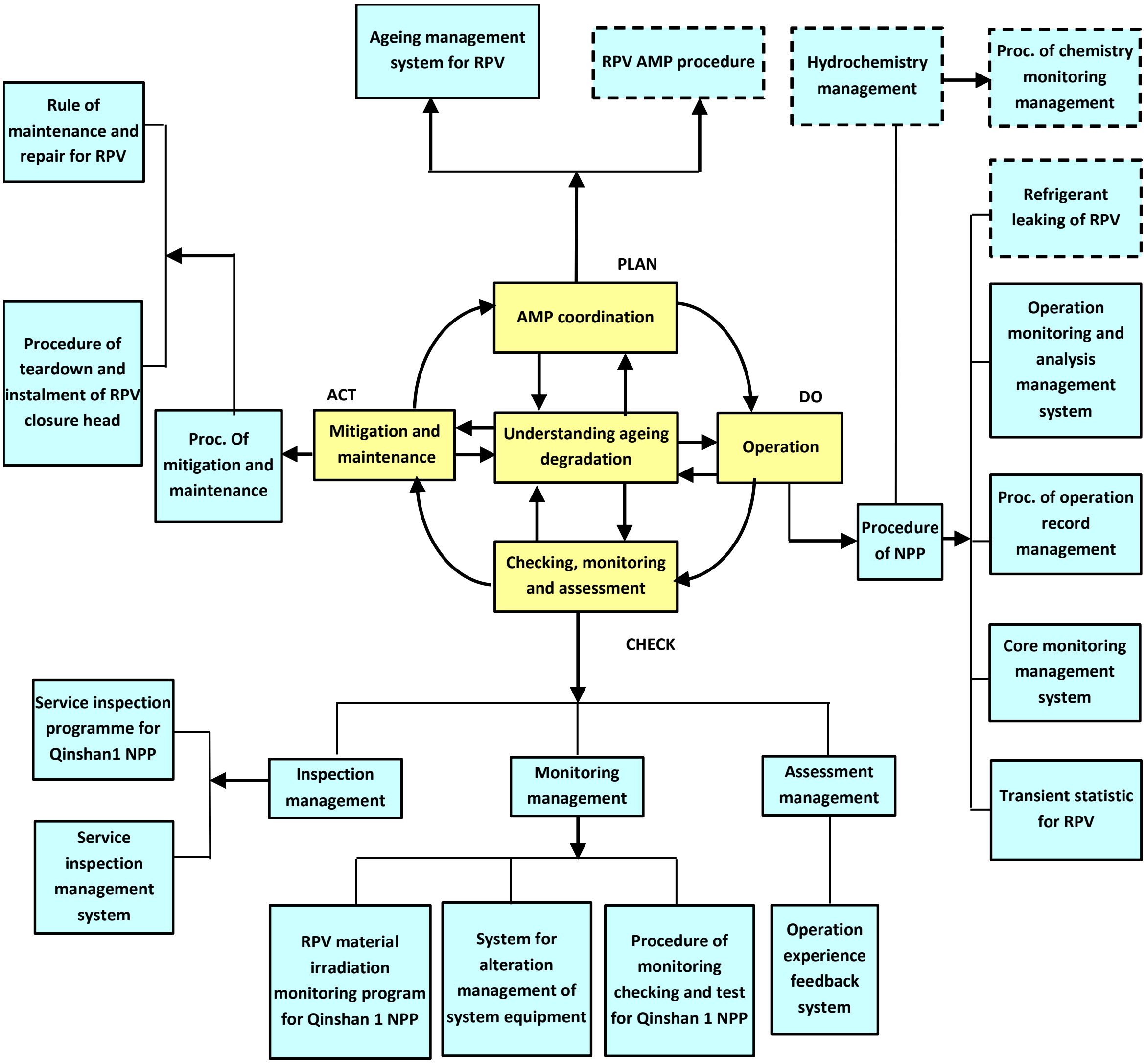


FIG. 61. RPV ageing management documentation system for Qinshan phase 1, China.

box to the right, applies the nuclear power plant procedures and ensures that water chemistry is correctly managed. The yellow box at the bottom, representing the group in charge of the 'CHECK' step, manages the inspection, the monitoring, the surveillance programme and the condition assessment and sends its output to the operation feedback system. For the 'ACT' step, the yellow box to the left represents the group in charge of the mitigation and maintenance aspects.

Figure 62 shows the ageing management activities and the milestone achievements since 2009 in the Chinese plants of Daya Bay and Ling Ao.

The pyramidal process begins at the top with a screening of all SSCs significant to safety, reliability and electricity production in the nuclear power plant.

The SSCs that are non-significant to safety, reliability and production can be run to failure if they seldom require maintenance and they can be replaced easily without shutting down the plant. If, on the other hand, their operability can be ensured through regular maintenance, a simple, periodic maintenance schedule is all they require.

The SSCs that are significant to safety, reliability and electricity production are under a major AMP or are replaced within the operating lifespan. Of these, the SSCs without a clear AMP or replacement plan require topical R&D and the development of a life cycle management programme. The critical SSCs with a major AMP can have their life cycle management simply scheduled for implementation.

Figure 63 represents the generic process, common to all nuclear power plants in China, for preparation, implementation, assessment and development of a corrective action plan and reporting in a typical PSR submission

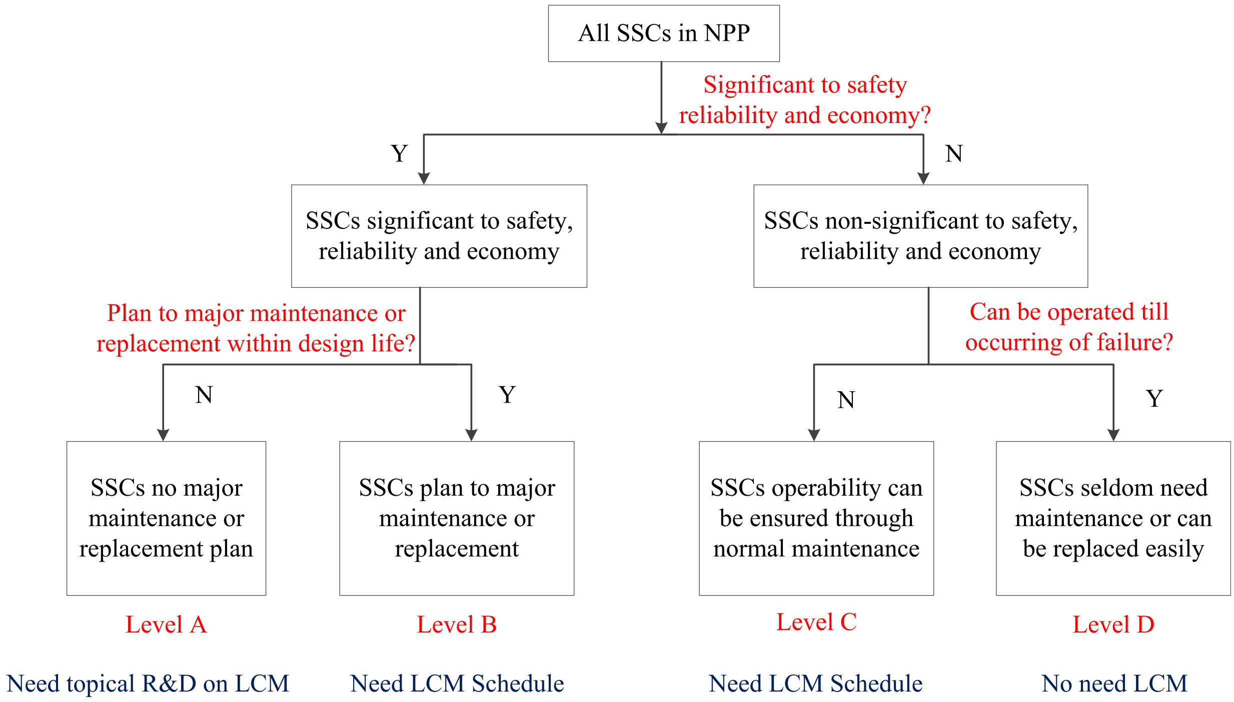


FIG. 62. Screening, life cycle management and preventive maintenance activities in Daya Bay and Ling Ao nuclear power plants, China. LCM — life cycle management.

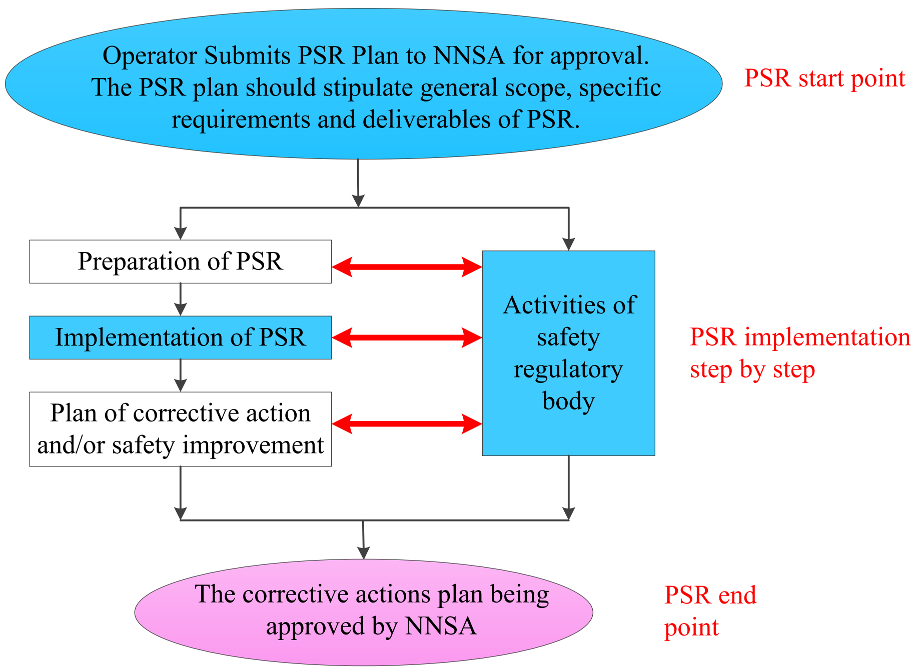


FIG. 63. Generic flow chart for PSRs in Chinese nuclear power plants. NNSA — National Nuclear Safety Administration.

for nuclear power plants. As a starting point, the nuclear power plant owner/operator, as the licensee, submits the PSR plan that includes the scope, the specific requirements and the deliverables to the regulator for approval. The activities included in the submission are those requested by the regulatory body and those coming from the owner's own plant diagnostics and corrective action plan. The PSR ends with the regulatory approval of the PSR corrective action plan.

8.3. GERMANY

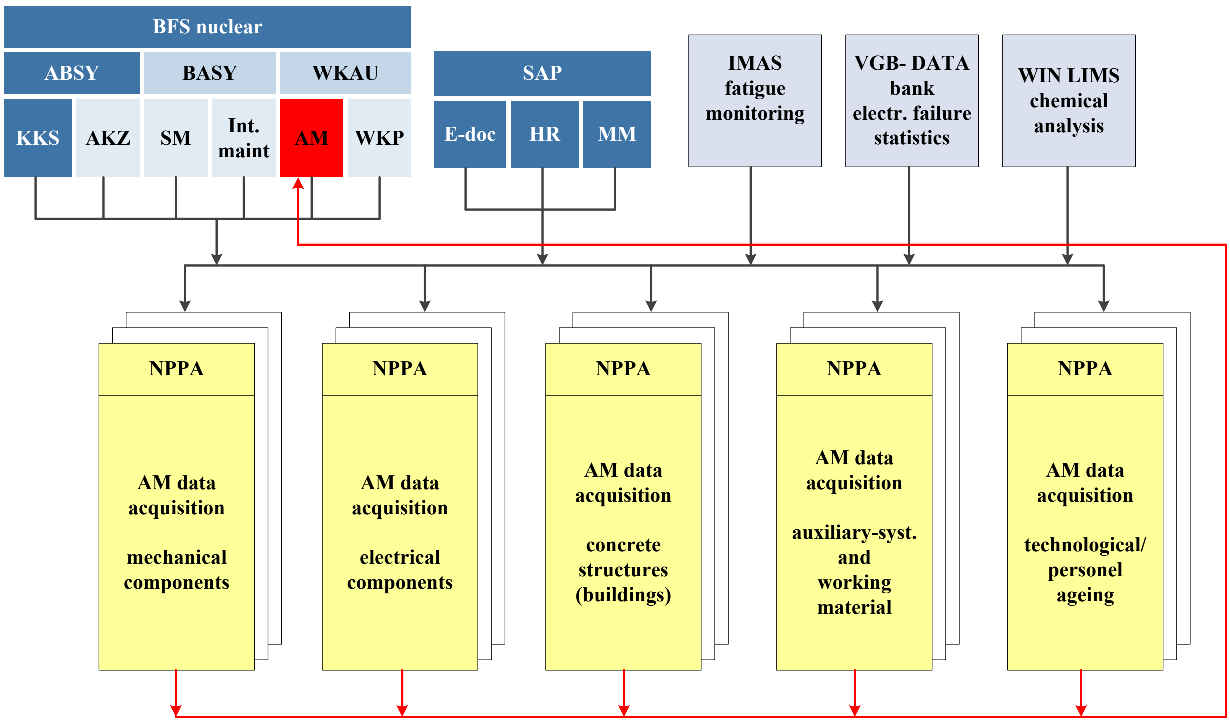
Figure 64 shows the data acquisition system used in German nuclear power plants [85].

The yellow boxes show the data acquisition systems for the various disciplines, the auxiliary systems, the system ageing and the personnel training and qualification data system.

Figure 64 in its entirety shows a systematic and comprehensive computer assisted process, which produces a reduced volume of information that is then assessed in detail and the results input into the AMP. Its basic features are completeness, traceability and independence from individual judgement. It is built on the knowledge base of the operator at the particular site. The source of data is the BASY system, an operating requirements and non-conformance reporting system. Malfunction messages and the information about the planned and unplanned operational activities are all merged into the BASY system. From the BASY system, non-conformance reports are prepared.

The following step involves an assessment of whether the SSC non-conformance is due to an ageing effect, and if so, the relevant events are then analysed by a team of experts. Any repercussions on the AMP are assessed and all possible common mode failures affecting other SSCs are identified. The results of this analysis are documented and presented in an annual status report. The output from all data acquisition systems goes to the ageing management database.

Figure 65 represents an integrated maintenance management programme as applied in Germany. The concept follows the PDCA cycle and describes a concept designed to guarantee the remaining lifetime of a critical SSC.



BFS: Betriebsprüfungssystem = nuclear commissioning system
ABSY: Anlagenbeschreibungsystem = equipment specification system
BASY: Betriebsauftrags- und Störmeldesystem = operating requirements and non-conformance reporting system
AKZ: Anlagenkennzeichnungssystem = equipment tagging system
KKS: Kraftwerk-Kennzeichensystem = station designation and identification system
MM: material management
SM: Störmangelmeldungen = alarm annunciation and recording system
WKAU: Wiederkehrende Prüfungen Ausgabe = periodic testing result system
WKP: Wiederkehrende Prüfungen = periodic testing
HR: human resources
WIN LIMS: Windows labour-information-management system

FIG. 64. Data acquisition and management system in German nuclear power plants [85]. AM — ageing management; E-doc — electronic documentation management system; HR — human resource training and qualification database; IMAS — integrity management system; NPPA — nuclear power plant A; SAP — systems, applications and products in data processing; VGB-DATA — the electrical failure statistical databank .

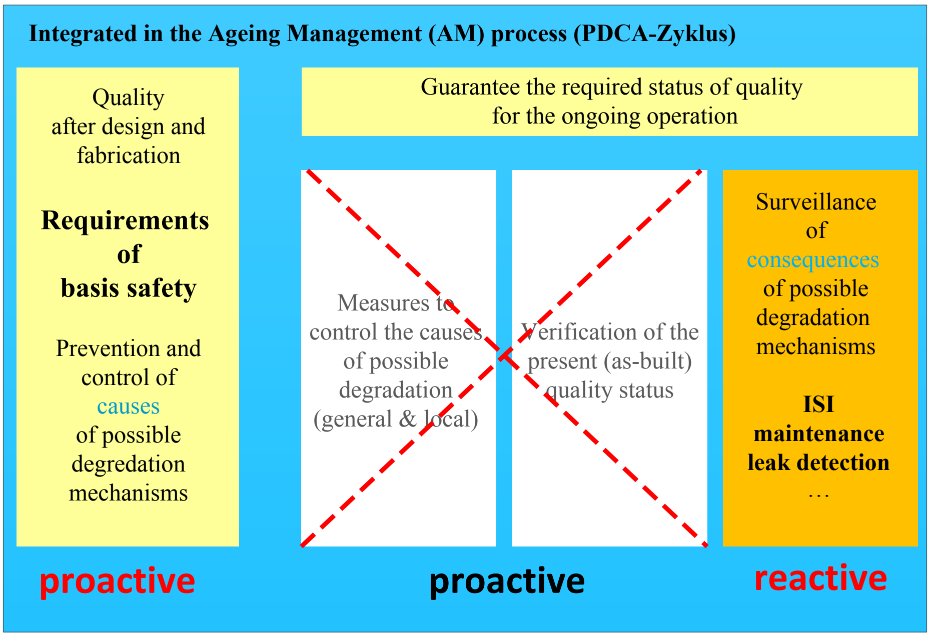


FIG. 65. Integrated ageing maintenance management programme as applied in Germany. PDCA — plan–do–check–act.

Starting from the left, the yellow box represents the pre-operational steps that should be taken proactively to guarantee design and fabrication quality and ensure that all basic safety requirements are met. After fabrication, the integrated ageing management process includes those steps necessary to prevent and control the cause of degradation mechanisms that may affect the component. During the operation period, represented by the boxes to the right, the evaluation of the plant condition consists of two parts:

— A proactive maintenance part, shown shaded and crossed out, with monitoring devices and mitigation measures to control degradation. In parallel, the SSC condition verification steps are conducted, involving analytical evaluation of operating experience feedback, stress analysis and fracture mechanics, among others. It is shaded because it is built into the system as time based and frequency based elements.
— A reactive maintenance part, represented by the dark yellow box on the right, consisting of the reactive components to cover unexpected events, for which reactive maintenance should always be available to intervene, even for low frequency occurrences. This maintenance component, although reactive by definition, can rely on indicators such as surveillance of the symptoms (or possible consequences) of postulated degradation mechanisms. This is achieved by making use of ISIs, leak detection and exploratory investigations.

8.4. JAPAN

Figure 66 shows validation of the diagnostic and prognostic process for concrete structures in use in Japanese nuclear power plants.

The primary integrity evaluation of the concrete structure is conducted by gathering information on the SSC condition. This includes a clarification of the degradation mechanisms by studying the effect of radiation on the concrete structures and by predicting the progress of the degradation. It also includes an adequacy assessment (degree of sophistication) of the evaluation methods, particularly when it comes to identifying the degree of corrosion of the reinforcing steel owing to cracking in the concrete structure.

A secondary integrity evaluation is conducted by investigating the degradation at the microscopic level. This investigation allows the evaluation of the remaining strength of the concrete structure. Shown on the right of the secondary evaluation diamond is the development of the process used in the microscopic evaluation. This is conducted by surveying and conducting experimental studies on the evaluation methods used in the secondary integrity evaluation of the concrete structures.

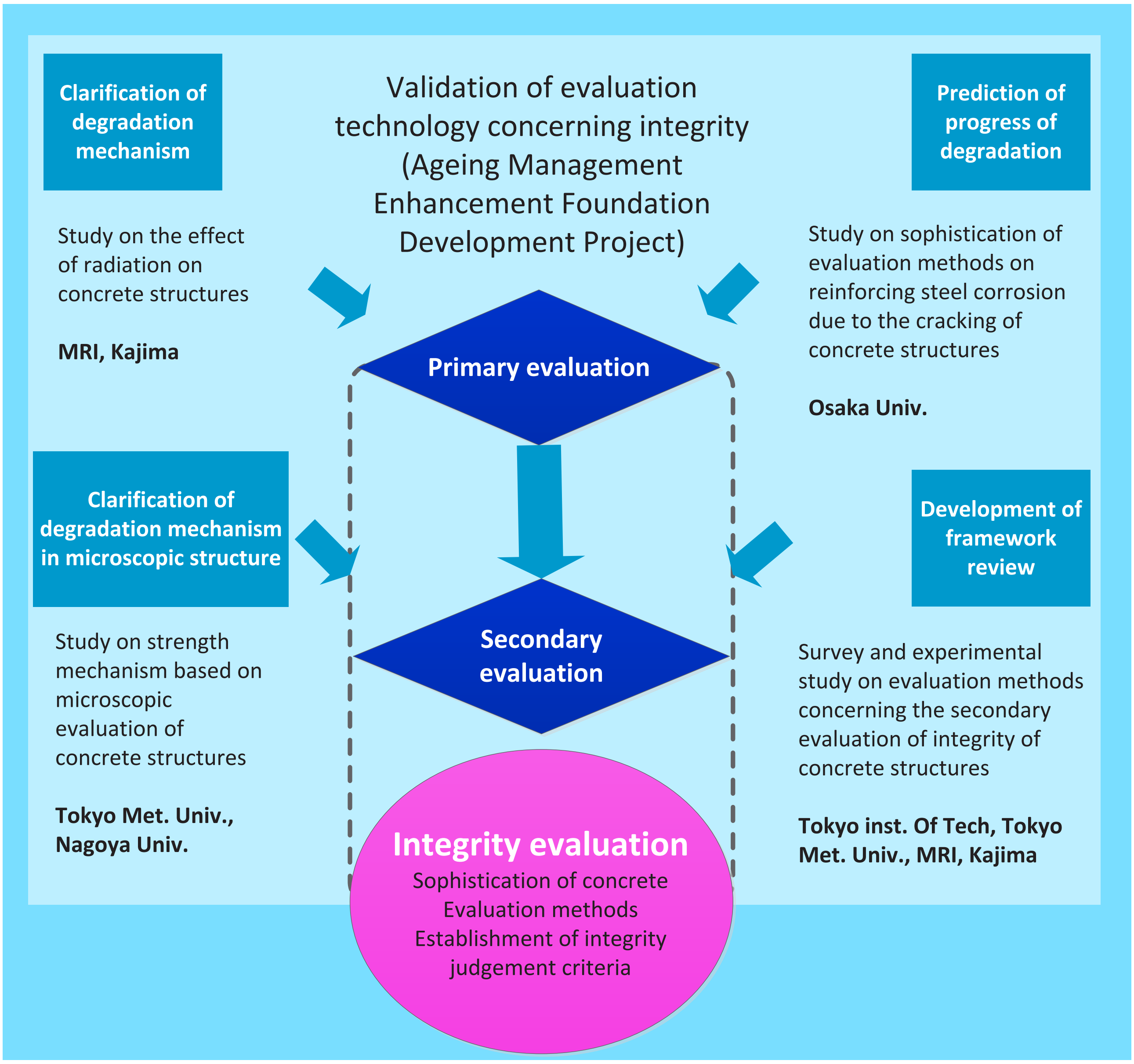


FIG. 66. Validation of ageing management validation technology at Tokyo Metropolitan University, Japan. Inst. — Institute; Met. — Metropolitan; MRI — Mitsubishi Research Institute; Univ. — University.

To develop estimates of the accident progression and the status inside the Fukushima Daiichi reactors to support the planning and conduction of decommissioning, a Japanese research institute, Japan Nuclear Energy Safety Organization, is conducting fundamental studies on the thermal hydraulic behaviour in the reactor, fuel damage and degradation processes, behaviour of structural materials and pressure vessels, and release and migration of fission products. Specifically, the activities include:

— Study on the effects of sea water on coolant heat transfer and other coolant properties;
— Study using a test fuel rod heated under simulated LOCA conditions to examine phenomena such as core melting and degradation;
— Experimental efforts to investigate the behaviour of molten fuel falling into coolant.

Several research programmes are under way at this research institute to provide technical support to Japan's NRA. The ROSA-SA (severe accident) project is one such programme focusing on the thermohydraulics of the containment. The project consists of integral tests, separate effects tests and analytical studies to improve and validate the prediction methods. The technical issues addressed in this programme are related to over temperature containment failure, hydrogen risk and source term behaviour as affected by thermohydraulics.

8.5. INTERNATIONAL COLLABORATION ON AGEING MANAGEMENT

The international standardization of AMPs is achieved, as much as possible, through international collaboration on ageing management and through the cooperation of international agencies operating in the nuclear power field. They include the Nuclear Energy Agency (OECD/NEA), the IAEA and organizations such as WANO and EPRI. Collaboration and cooperation are achieved through working groups and international meetings. The OECD/NEA released an LTO guide called the ‘green booklet’ (because of the colour of its cover), which is a general guide for regulatory bodies regarding application for operation beyond the nominal design life of a plant [86]. Cooperation initiatives include the OECD/NEA Piping Failure Data Exchange (OPDE) project, which includes events of interest with regard to piping failures. It covers the main safety systems’ piping and those non-safety piping systems that have failures that could lead to common cause initiating events such as internal flooding of vital plant areas. Steam generator tubes are excluded. Two meetings of the OPDE working group are held annually.

The SCAP has defined and refined the database performance requirements, the data format and coding guidelines, populated the database and assessed the data. This work, together with the knowledge base and commendable practices, has provided a tool for assisting member countries and supported regulatory activities specifically in the fields of SCC and cable insulation.

The OECD Component Operational Experience, Degradation and Ageing Programme [50] is an umbrella programme that combines the follow-up of both the OPDE and the SCAP initiatives.

International cooperation meetings fostering collaboration on nuclear power plant ageing management include the following programmes:

— PARENT: Cyclotron produced radionuclides for medical diagnostics and therapies;
— International Group for Radiation Damage Meeting: Dealing with radiation damage;
— Zorita project: Cooperative research project on ex-plant materials between the José Cabrera (Zorita) nuclear power plant in Spain (a site under decommissioning) and the Spanish Consejo de Seguridad Nuclear;
— IFRAM: Global cooperation initiative on identifying and addressing technical issues in the ageing of nuclear power plants.

8.6. INTERNATIONAL ACTIVITIES ON AGEING MANAGEMENT

The IAEA has long been involved in activities aimed at supporting Member States with their nuclear power plant AMPs, both in terms of publications as safety standards and safety guides, and also as review services of ageing programmes and programme implementation guidelines.

8.6.1. Ageing management initiatives in IAEA activities

8.6.1.1. Safety standards, guides and services for ageing management programmes

The IAEA publication SSR-2/2 (Rev. 1) [4], contains requirements related to nuclear power plant ageing management, namely:

— Requirement 14: The operating organization shall ensure that an effective AMP is implemented to ensure that required safety functions of SSCs are fulfilled over the entire operating lifetime of the plant.
— Requirement 16: Where applicable, the operating organization shall establish and implement a comprehensive programme for ensuring the long term safe operation of the plant beyond a timeframe established in the licence conditions, design limits, safety standards and/or regulations.

The IAEA publication NS-G-2.12 [1] provides a set of guidelines and recommendations for managing ageing of SSCs important to safety in nuclear power plants. It mainly focuses on physical ageing, but also includes the management of obsolescence.

Other IAEA activities related to ageing management are peer review missions, which are services to Member States in support of their LTO programmes. These missions are conducted by international expert teams and are led by IAEA representatives. Missions related to LTO are regulated by the IAEA extrabudgetary programme (EBP) SALTO guidelines. The mission report, Safety Aspects of Long Term Operation of Water Moderated Reactors [87], contains recommendations on the scope and content of a typical programme for safe LTO of water moderated reactors, and deals with the framework necessary to embark on LTO. It covers, in detail, mechanical components and materials, electrical components and cabling, I&C equipment and also structures and structural elements.

The IAEA safety report entitled Safe Long Term Operation of Nuclear Power Plants [15] contains the underlying guiding principles of the SALTO peer reviews and the experience accumulated during the SALTO missions, including a summary of the key elements extracted and generalized from the EBP SALTO reports. It deals with LTO feasibility, scoping and screening, the assessment and management of structures and components for material degradation, safety analysis revalidation, using time limited assumptions, the documentation required and regulatory oversight. This report can be used as preventive guidance before nuclear power plant operators begin the development of LTO programmes for their nuclear power plants.

8.6.1.2. International generic ageing lessons learned

Another IAEA ageing management initiative relates to the development of the IGALL database. The IAEA sponsored this database through an EBP established in September 2010 (see Ref. [30]). Its first phase was completed in 2013 and its second phase in 2014–2015. The IGALL programme developed and maintains documents and a database to provide a technical basis and practical guidance on managing ageing of mechanical, electrical and I&C components and civil structures of nuclear power plants important to safety. The IGALL database supports the application of the IAEA requirements on design [3], on commissioning and operation [4], on ageing management [1], on PSR [74] and on safe LTO [15].

The IGALL database contains:

— A generic sample of ageing management review tables;
— A collection of proven AMPs;
— A collection of typical TLAAs.

The itemized table of recommended AMPs is particularly aimed at facilitating the safe LTO of either new or operating nuclear power plants. The IGALL database was developed by over 100 key international experts on ageing management issues from 13 Member States. The European Commission was represented by regulators, utilities, vendors, manufacturers and the originators of codes and standards. The IGALL report [30] was based on the results of the application of a systematic approach to ageing and maintenance techniques for a variety of nuclear power plant structures and components as described in NS-G-2.12 [1]. It contains consolidated international information on SSC degradation mechanisms and a compilation of the state of the art methodologies to evaluate the extent of component degradation for the various technologies under consideration. The IGALL report constitutes a model establishing common ground for discussions between regulators and owner/operators to facilitate the implementation of acceptable AMPs. It can be used as a guide for the implementation of AMPs for nuclear power plants, and it serves the main nuclear power technologies in use by Member States (i.e. PWRs, BWRs, WWERs and CANDU/PHWRs). The IGALL report is intended as a live document, to be updated at least every 5 years.

8.6.1.3. Ageing management initiatives in coordinated research projects

The IAEA's Division of Nuclear Power sponsored a number of CRPs and international workshops. These are important IAEA mechanisms for organizing international research work to achieve specific research objectives, consistent with the IAEA work programme.

A primary objective of a CRP is normally the formulation and preparation of a research project for a specific application. Other objectives are often related to the availability of the developed results through electronic media and to the production of relevant technical documents or guides containing detailed descriptions of the research subject.

Examples of CRPs are:

- Qualification, Condition Monitoring, and Management of Aging of Low Voltage Cables in Nuclear Power Plants: The goal of this CRP was to provide the current and next generation of nuclear facilities with information and guidelines on how to qualify new cables, monitor the performance of existing cables, and establish a programme of cable ageing management for both the current fleet of reactors and the next generation of nuclear facilities.
- Advanced Surveillance, Diagnostics, and Prognostics Techniques used for Health Monitoring of Systems, Structures, and Components in Nuclear Power Plants: The specific objective was to define and coordinate research to support the development of new surveillance, diagnostic and prognostic techniques as the singular point detection techniques for the monitoring of SSC functions and integrity in nuclear power plants [88].
- Review and Benchmark of Calculation Methods of Piping Wall Thinning due to Erosion-Corrosion in Nuclear Power Plants: The overall objective was to provide references and boundary conditions for the use of the available prediction tools, and to provide a document to support the development of a sold technical base and serve as a guideline for use in the control of FAC in piping systems in nuclear power plants.

8.6.1.4. Coordinated research projects on integrity of reactor pressure vessels

The IAEA sponsored a series of CRPs that focused on RPV structural integrity, by measuring the best irradiation fracture parameters, using relatively small test specimens.

The engineering judgement approach used in TLAAs and in the design of SSCs establishes safety margins built into safety coefficients (safety factors), which, in the final outcome, display a considerable degree of conservatism. Experience gained from testing materials extracted from the plants and from conducting structural analyses showed that the margins obtained through engineering judgement, in numerous cases, do not reflect the real rate of ageing damage and the exact conditions of a given structure or component. A typical example of this gap between conservative design and actual material test results and advanced analysis is evidenced in fatigue evaluations, conducted using advanced engineering tools (e.g. finite element analysis or other computational codes), and using worst (most conservative) material properties, as listed in material standards. This approach provided a more realistic report on usage factors, based on the actual operating history (actual transients experienced by the SSCs). Infinitely small theoretical increments were used in the analysis models, in order to more closely match the input curves. These structural analyses, carried out at locations that should have manifested a usage factor value close to 1, showed instead much lower values and a similar gap with the safety factors.

It is always extremely helpful and beneficial to compare different round robin exercises and comparative benchmark studies, for example, through the mechanism of the IAEA CRPs. The impact of different evaluation methodologies, applied to the testing of identical materials, has been studied many times over the past 20 years in the nuclear industry. It has quickly been recognized that it is very important to use harmonized methodologies, and to apply codes and standards that are universally accepted for the investigation of similar materials and SSCs, and for structures that have been designed by the same designer, technology supplier or produced by the same manufacturer.

The first CRP (or CRP phase 1) on irradiation embrittlement of RPV steels, focused on the standardization of methods for measuring embrittlement in terms of both the mechanical properties of materials subject to irradiation and the neutron irradiation environment itself. The main results are published in the IAEA Technical Reports Series No. 163 [89].

CRP phase 2, Analysis of the Behaviour of Advanced RPV Steels under Neutron Irradiation, involved testing and evaluation of advanced RPV steels that had a reduced amount of residual elements (copper and phosphorus). The results are summarized in IAEA Technical Reports Series No. 265 [90].

CRP phase 3, Optimizing RPV Surveillance Programmes and Analyses, addressed the direct measurement of fracture toughness using irradiated surveillance specimens. A key achievement of this CRP was the acquisition of a series of RPV steels for radiation embrittlement research. The IAEA reference material JRQ is documented in IAEA-TECDOC-1230 [91].

The main emphasis during CRP phase 4, Assuring Structural Integrity of RPVs, was the experimental verification of the Master Curve approach for surveillance size specimens [92]. The application included a large test matrix using the Japanese JRQ forging steel (ASTM A-533-BCl.1) and other national steels including WWER materials. No differences among laboratories were identified, and results obtained from dynamic data also followed the Master Curve.

Guidelines were developed and additional Master Curve testing was performed under CRP phase 5, Surveillance Programme Results Application to Reactor Pressure Vessel Integrity [93]. The large working group of this CRP consisted of 20 testing laboratories representing 15 Member States. This CRP had two main objectives:

— To develop a large database of fracture toughness data using the Master Curve methodology for both pre-cracked Charpy size and 25.4 mm thick compact tension specimens to assess possible specimen bias effects and any other effects due to the range of temperatures used to determine the reference fracture toughness temperature, T_0, (either using single temperature or the multitemperature assessment methods);
— To develop international guidelines for measuring and applying Master Curve fracture toughness results for RPV integrity assessments; the results were published in IAEA Technical Reports Series No. 429 [92] and IAEA-TECDOC-1435 [93].

CRP phase 6, Effects of Nickel on Irradiation Embrittlement of Light Water RPV Steels, comprised the procurement of materials, the determination of mechanical properties, the irradiation and testing of specimens and microstructural characterization. The results clearly show a significantly higher radiation sensitivity of the high nickel welds (1.7 mass %) compared to the lower nickel base metal (1.2 mass %), as documented in IAEA-TECDOC-1441 [94].

CRP phase 7, Evaluation of Radiation Damage of WWER-440 RPV Materials using IAEA DB, was focused on WWER-440 steels and the need for an improved predictive embrittlement correlation. In this study, a group of eight representatives from seven Member States developed new correlations for WWER-440 RPVs that provided better predictive capabilities based upon the steel composition and neutron exposure. These new correlations were developed in a framework that better simulates the known embrittlement mechanisms for these steels. The work was published in IAEA-TECDOC-1442 [95]. The CRP was carried out through the completion of four tasks:

— Collection of WWER-440 surveillance and other relevant data that were subsequently input into the IAEA International Database on RPV Materials (IDRPVM);
— Analysis of radiation embrittlement data of WWER-440 RPV materials using the IDRPVM;
— Evaluation of predictive formulas depending on material chemical composition, neutron flux and fluence;
— Guidelines for the prediction of radiation embrittlement of operating RPVs of WWER-440 units, including the methodology used for the evaluation of surveillance data of a specific operating unit.

CRP phase 8, Master Curve Approach to Monitor the Fracture Toughness of RPV, is an ongoing extension of CRP phase 5, in that some of the outstanding issues associated with the use of the Master Curve fracture toughness methodology are being studied in more detail. The overall objectives of CRP phase 8 include:

— Better quantification of fracture toughness issues relative to testing surveillance specimens for application to RPV integrity assessments.
— Development of approaches for addressing Master Curve technical issues in integrity evaluation of operating RPVs. As the Master Curve approach is applicable to all nuclear power plant ferritic steel components, including the RPV, the scope of materials to be addressed will include both RPV and non-RPV materials.

The overall objective of CRP phase 9, Review and Benchmark of Calculation Methods for Structural Integrity Assessment of RPVs during PTS, was to perform benchmark deterministic calculations of a typical PTS regime with the aim of comparing the effects of individual parameters on the final RPV integrity assessment, and then to recommend the best practice for their implementation in PTS procedures.

At present, several different procedures and approaches are being used for RPV integrity assessments. Differences exist not only between WWER and PWR reactor types, but also within each group. These differences exist, on the one hand, because different codes and rules are being used in design, manufacture and material selection for the various reactor types, and on the other hand, because of the different implementation levels of recent developments in fracture mechanics. Benchmark calculations have been performed to improve the user qualification and to reduce the user effect on analysis results. These were conducted for generic PWR and WWER vessel types and included sensitivity analyses to check the influence of several factors. The studies showed that the factors that most significantly influenced the assessments were: the flaw size, shape, location and orientation; the thermohydraulic assumptions and the material toughness. Applying different national codes and procedures to the benchmark cases produced significant differences in allowable material toughness. This was mainly related to the safety factors used and to the various approaches to postulated defects, to postulated transients and to the characterization of material toughness.

8.6.2. Ageing management related initiatives of other international organizations

International initiatives regarding nuclear power plant ageing management have also been undertaken by other international organizations such as IFRAM, OECD/NEA, the EU Joint Research Centre, etc.

IFRAM was formed to facilitate the sharing of information among participants worldwide on advances in material ageing management (including diagnostics and prognostics) for nuclear power plants.

Individual activities related to diagnostics and prognostics are under way at the Joint Research Centre (and in EU member nations), Canada, China, India, Japan, Republic of Korea and the USA, where EPRI, the Department of Energy and the NRC are engaged.

The US federally funded activities, under the Department of Energy Office of Nuclear Energy, are organized in the Light Water Reactor Sustainability (LWRS) programme. The industry organizations addressing the technical challenges related to LTO, including universities and national laboratories, are proposing the integration of advanced diagnostic tools into the new builds and into the design of the next generation of nuclear power plants.

8.6.2.1. NUGENIA

The Nuclear Generation II & III Association (NUGENIA) is a European international non-profit association founded under Belgian legislation in November 2011, and launched in March 2012, dedicated to the R&D of nuclear fission technologies, with a focus on generation II and III reactors. NUGENIA gathers stakeholders from industry, research organizations, safety organizations and academia who are committed to developing joint R&D projects. It builds on past successes of a European Commission supported network called NULIFE (NUclear plant LIFE prediction), and integrates a working group from the Sustainable Nuclear Energy Technology Platform.

Similar to the US LWRS programme, NUGENIA's mission is to be the integrating structure between industry, small and medium enterprises, research organizations, academia and technical safety organizations, for the development of safe, reliable and competitive generation II and III reactors. NUGENIA facilitates harmonized approaches at the European level for safety requirements and an advanced scientific and technical base for the development of generation II and III technology.

The main mandate of the NUGENIA foundation is to coordinate the development of a strong project portfolio with the following specific objectives:

— Become the EU reference platform on generation II and III technology;
— Support national research co-programming and facilitate public–private partnerships;
— Launch surveys on experimental facilities;
— Facilitate sharing infrastructures and co-investing;
— Pursue the objective of representing all nuclear power plant technologies in Europe;
— Develop an international dimension, both within Europe and internationally.

NUGENIA has therefore developed a roadmap of eight technical focus areas, namely:

— Plant safety and risk assessment;
— Severe accidents;
— Improved reactor operation;
— Integrity assessment of SSCs;
— Fuel development, waste and spent fuel management and decommissioning;

— Innovative LWR design and technology;
— Harmonization;
— ISI and NDE.

In the context of the European Commission Code for European Severe Accident Management project, which is an R&D project in the seventh European Commission Framework Programme, efforts are dedicated to improving the European Severe Accident Management Code, ASTEC, to model BWR specifics (e.g. in-vessel degradation) and to apply the code for severe accident management analysis of different types of nuclear power plants, with special emphasis on those operating in Europe.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Ageing Management for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-2.12, IAEA, Vienna (2009).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Proactive Management of Ageing for Nuclear Power Plants, Safety Reports Series No. 62, IAEA, Vienna (2009).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power Plants: Design, IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), IAEA, Vienna (2016).

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power Plants: Commissioning and Operation, IAEA Safety Standards Series No. SSR-2/2 (Rev. 1), IAEA, Vienna (2016).

[5] PRIME MINISTER OF JAPAN AND HIS CABINET, Report of Japanese Government to the IAEA Ministerial Conference on Nuclear Safety — The Accident at TEPCO's Fukushima Nuclear Power Stations (2011), http://japan.kantei.go.jp/kan/topics/201106/iaea_houkokusho_e.html

[6] HWANG, I.S., Development of Periodic Safety Review Plan for Operating Nuclear Power Plants, Seoul National University, Seoul (2000).

[7] NUCLEAR ENERGY INSTITUTE, Industry Guideline for Implementing the Requirements of 10 CFR Part 54 — The License Renewal Rule, Rep. NEI 95-10, Rev. 6, Washington, DC (2015).

[8] NUCLEAR REGULATORY COMMISSION, Requirements for Renewal of Operating Licenses for Nuclear Power Plants, 10 CFR 54.5, US Govt Publishing Office, Washington, DC (2001).

[9] NUCLEAR REGULATORY COMMISSION, Generic Ageing Lessons Learned (GALL) Report, Final Report, Rep. NUREG-1801, Rev. 2, Office of Nuclear Reactor Regulation, Washington, DC (2010).

[10] NUCLEAR AND INDUSTRIAL SAFETY AGENCY, Impact of Ageing on the Accident of TEPCO's Fukushima Daiichi Nuclear Power Station, Tokyo (2012).

[11] ANDRESEN, P.L., et al., Expert Panel Report on Proactive Materials Degradation Assessment, NUREG/CR-6923, Brookhaven National Laboratory, Upton (2007).

[12] NAOTO SEKIMURA, Ageing Management and Safe Long Term Operation, Proc. Int. Symp. on Nuclear Power Plant Life Management, Shanghai, 2007, IAEA, Vienna (2008).

[13] ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT, Workshop on Commendable Practices for the Safe, Long-term Operation of Nuclear Reactors Nuclear Energy Agency (NEA) Stress Corrosion Cracking and Cable Ageing Project (SCAP), OECD/NEA, Tokyo (2010).

[14] INTERNATIONAL ATOMIC ENERGY AGENCY, Power Uprate in Nuclear Power Plants: Guidelines and Experience, IAEA Nuclear Energy Series No. NP-T-3.9, IAEA, Vienna (2011).

[15] INTERNATIONAL ATOMIC ENERGY AGENCY, Safe Long Term Operation of Nuclear Power Plants, Safety Reports Series No. 57, IAEA, Vienna (2008).

[16] INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning of Nuclear Power Plants and Research Reactors, IAEA Safety Standards Series No. WS-G-2.1, IAEA, Vienna (1999).

[17] INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning of Facilities, IAEA Safety Standards Series No. GSR Part 6, IAEA, Vienna (2014).

[18] INTERNATIONAL ATOMIC ENERGY AGENCY, Policies and Strategies for the Decommissioning of Nuclear and Radiological Facilities, IAEA Nuclear Energy Series No. NW-G-2.1, IAEA, Vienna (2011).

[19] BUSBY, J.T., et al., Expanded Materials Degradation Assessment (EMDA): Aging of Reactor Pressure Vessels, Rep. NUREG/CR-7153, Vol. 3, Oak Ridge National Laboratory, Oak Ridge (2014).

[20] INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: PWR Pressure Vessels, IAEA-TECDOC-1556, IAEA, Vienna (2007).

[21] INTERNATIONAL ATOMIC ENERGY AGENCY, Integrity of Reactor Pressure Vessels in Nuclear Power Plants: Assessment of Irradiation Embrittlement Effects in Reactor Pressure Vessel Steels, IAEA Nuclear Energy Series No. NP-T-3.11, IAEA, Vienna (2009).

[22] NUCLEAR REGULATORY COMMISSION, Regulations, Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions, 10 CFR 51, US Govt Publishing Office, Washington, DC (2013).

[23] AMERICAN SOCIETY OF MECHANICAL ENGINEERS, ASME Boiler and Pressure Vessel Code, Section III, Division I, Rules for Construction of Nuclear Facilities Components, 2007 Edition, ASME, New York (2007).

[24] ATOMIC ENERGY COMMISSION, Nonmetallic Thermal Insulation for Austenitic Stainless Steel, Regulatory Guide 1.36, Directorate of Regulatory Standards, Washington, DC (1973).

[25] NUCLEAR REGULATORY COMMISSION, Nonmetallic Thermal Insulation for Austenitic Stainless Steel, Regulatory Guide 1.36, Rev. 1, NRC, Washington, DC (2015).

[26] INTERNATIONAL ATOMIC ENERGY AGENCY, Stress Corrosion Cracking in Light Water Reactors: Good Practices and Lessons Learned, IAEA Nuclear Energy Series No. NP-T-3.13, IAEA, Vienna (2011).
[27] ELECTRIC POWER RESEARCH INSTITUTE, Flow-accelerated Corrosion in Power Plants, Rep. TR-106611-R1-01, EPRI, Palo Alto, CA (1998).
[28] ELECTRIC POWER RESEARCH INSTITUTE, Pressurized Water Reactor Secondary Water Chemistry Guidelines, Rev. 7, Rep. 1016555, EPRI, Palo Alto, CA (2009).
[29] INTERNATIONAL ATOMIC ENERGY AGENCY, Approaches to Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL) Final Report, IAEA-TECDOC-1736, IAEA, Vienna (2014).
[30] NUCLEAR REGULATORY COMMISSION, Domestic Licensing of Production and Utilization Facilities, 10 CFR 50, Appendix G, Fracture Toughness Requirements, US Govt Publishing Office, Washington, DC (2015).
[31] NUCLEAR REGULATORY COMMISSION, Domestic Licensing of Production and Utilization Facilities, 10 CFR Part 50, Appendix H, Reactor Vessel Material Surveillance Program Requirements, US Govt Publishing Office, Washington, DC (2008).
[32] NUCLEAR REGULATORY COMMISSION, Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events, 10 CFR 50.61, US Govt Publishing Office, Washington, DC (2015).
[33] NUCLEAR REGULATORY COMMISSION, Requirements for Thermal Annealing of the Reactor Pressure Vessel, 10 CFR 50.66, US Govt Publishing Office, Washington, DC (2015).
[34] NUCLEAR REGULATORY COMMISSION, Format and Content of Report for Thermal Annealing of Reactor Pressure Vessel, Regulatory Guide 1.162, NRC, Washington, DC (1996).
[35] AMERICAN SOCIETY OF MECHANICAL ENGINEERS, ASME Boiler and Pressure Vessel Code, Section XI, Division 1, Subsection IWA, Rules for Inservice Inspection of Nuclear Power Plant Components, Federal Requirements, ASME, New York (1995).
[36] NUCLEAR REGULATORY COMMISSION, Materials and Inspections for Reactor Vessel Closure Studs, Regulatory Guide 1.65, NRC, Washington, DC (2010).
[37] NUCLEAR REGULATORY COMMISSION, Ultrasonic Testing of Reactor Vessel Welds During Pre-service and In-service Examinations, Regulatory Guide 1.150, Rev. 1, NRC, Washington, DC (1983).
[38] GOSATOMNADZADOR, General Regulations to Ensure Nuclear Power Plant Safety, Rep. OPB-99/97, NP-001-97, Gosatomnadzador, Moscow (1997).
[39] GOSATOMNADZADOR, Basic Requirements for Power Unit Lifetime Extension of Nuclear Power Plants, Rep. NP-017-2000, Gosatomnadzador, Moscow (2000).
[40] ENERGOAMIZDAT, Rules for Arrangement and Safe Operation of Equipment and Piping of Nuclear Power Installations, Rep. PNAE G-7-008-89, Energoamizdat, Moscow (1990).
[41] INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: Metal Components of BWR Containment Systems, IAEA-TECDOC-1181, IAEA, Vienna (2000).
[42] INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: BWR Pressure Vessels, IAEA-TECDOC-1470, IAEA, Vienna (2005).
[43] INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: BWR Pressure Vessel Internals, IAEA-TECDOC-1471, IAEA, Vienna (2005).
[44] INTERNATIONAL ATOMIC ENERGY AGENCY, Improvement of In-service Inspection in Nuclear Power Plants, IAEA-TECDOC-1400, IAEA, Vienna (2004).
[45] ELECTRIC POWER RESEARCH INSTITUTE, Effect of Environment on Fatigue Usage for Piping and Nozzles at Oconee Units 1, 2, and 3, Rep. TR-110120, EPRI, Palo Alto, CA (1999).
[46] INTERNATIONAL ATOMIC ENERGY AGENCY, Storage of Spent Nuclear Fuel, IAEA Safety Standards Series No. SSG-15, IAEA, Vienna (2012).
[47] CHOPRA, O.K., SHACK, W.J., Review of the Margins for the ASME Code Fatigue Design Curve — Effects of Surface Roughness and Material Variability, Rep. NUREG/CR-6815 and ANL-02/39, Argonne National Laboratory, Argonne (2003).
[48] CANADIAN STANDARD ASSOCIATION, Periodic inspection of CANDU nuclear power plant components, CSA N 285.4, CSA, Toronto (2014).
[49] STRATTON, K.P., Life Extension at the Point Lepreau Generating Station — Powering the Future, IAEA Conf. Proc., IAEA-CN-164-4S02, IAEA, Vienna (2012).
[50] SNCT PUBLICATIONS, CODAP — French Code for Construction of Unfired Pressure Vessels and Division 2 for the Construction of Complex Vessels, Rep. 92038, SNCT Publications, Paris (2005).
[51] MASSOUC, J.-P., et al., “Thermal ageing of PWR duplex stainless steel components”, Proc. 7th Int. Conf. Nuclear Engineering, Tokyo (1999).
[52] KIRIYAMA, K., et al., Ageing effects on the lattice strain behaviour of duplex stainless steel, Journal of Strain Analysis for Engineering Design **46** 7 (2011) 697.
[53] BUYN, T.S., BUSBY, J.T., Cast Stainless Steel Ageing Research Plan, Rep. ORNL/LTR-2012/440, Oak Ridge Natl Lab., TN (2012).

[54] NUCLEAR REGULATORY COMMISSION, Flow Induced Vibration Failures After Power Uprate, Rep. ML040080392, Information Notice 2002-26, and Supplement 2, NRC, Washington, DC (2004).
[55] NUCLEAR REGULATORY COMMISSION, Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing, Regulatory Guide 1.20, NRC, Washington, DC (2006).
[56] HAMBRIC, S.A., "Flow induced vibration effects on nuclear power plant components due to main steam line valve singing", Proc. 9th NRC/ASME Symp. on Valves, Pumps and In-service Testing, Washington, DC (2006).
[57] DEBOO, G., et al., "Identification of Quad Cites main steam line acoustic sources and vibration reduction", Proc. 2007 ASME Pressure Vessels and Piping Division Conf., San Antonio, TX (2007).
[58] ZIADA, S., Flow Excited Acoustic Resonance in Industry (ZOLOTAREV, I., HORACEK, J., Eds), Institute of Thermomechanics, Flow Induced Vibration, Prague (2008).
[59] ELECTRIC POWER RESEARCH INSTITUTE, Guidelines on Nuclear Safety Related Coatings, Rep. TR-109937, EPRI, Palo Alto, CA (1998).
[60] LEFLER, S., Catawba Nuclear Station Use of High Density Polyethylene Piping, American National Standards Institute, Washington, DC (2010).
[61] INTERNATIONAL ATOMIC ENERGY AGENCY, Material Degradation and Related Managerial Issues of Nuclear Power Plants (Proc. Tech. Mtg Vienna, 2005) IAEA, Vienna (2006).
[62] INTERNATIONAL ATOMIC ENERGY AGENCY, Assessing and Managing Cable Ageing in Nuclear Power Plants, IAEA Nuclear Energy Series No. NP-T-3.6, IAEA, Vienna (2012).
[63] COMITE EURO-INTERNATIONAL DU BETON, Durable Concrete Structures, Bulletin D'information No. 183, Design Guide published for CEB by Thomas Telford, London (1992).
[64] LITZNER, H.U., BAKER, A., Design of concrete structures for durability and strength to Eurocode 2, Materials and Structures **32** (1999), 323.
[65] INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: Concrete Containment Buildings, IAEA-TECDOC-1025, IAEA, Vienna (1998).
[66] LAMBERT, L.D., PARKS, M.B., Experimental Results from Containment Piping Bellows Subjected to Severe Accident Conditions, Rep. NUREG/CR-6154 and SAND94-1711, Sandia National Laboratories, Albuquerque (1995).
[67] NUCLEAR REGULATORY COMMISSION, Anchoring Components and Structural Supports in Concrete, Regulatory Guide 1.199, NRC, Washington, DC (2003).
[68] AMERICAN CONCRETE INSTITUTE, Code Requirements for Nuclear Safety Related Concrete Structures and Commentary, Rep. 349-13, ACI, Farmington Hill, MI (2014).
[69] CANADIAN NUCLEAR SAFETY COMMISSION, Assessment of LBB Applicability to CANDU Primary Heat Transport Piping, Rep. RSP-0197, CNSC, Ottawa, ON (2014).
[70] INSTITUTE FOR ENERGY, European Methodology for Qualification of Non-destructive Testing — Third Issue, European Network for Inspection and Qualification Rep. 31, European Commission Joint Research Centre Rep. EUR 22906 EN, Petten (2007).
[71] ELECTRIC POWER RESEARCH INSTITUTE, Performance Demonstration Initiative (PDI) Program Description, EPRI-TR-1020593, EPRI, Palo Alto, CA (2010).
[72] NUCLEAR REGULATORY COMMISSION, Requirements for Renewal of Operating Licenses For Nuclear Power Plants, 10 CFR 54, US Govt Publishing Office, Washington, DC (2015).
[73] NUCLEAR REGULATORY COMMISSION, Contents of Application-Technical Information, 10 CFR Part 54.21, US Govt Publishing Office, Washington, DC (2015).
[74] NUCLEAR REGULATORY COMMISSION, Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses, Regulatory Guide 1.188, Rev. 1, NRC, Washington, DC (2005).
[75] NUCLEAR REGULATORY COMMISSION, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants – Final Report, Rep. NUREG-1800, Rev. 2, Office of Nuclear Reactor Regulation, Washington, DC (2010).
[76] NUCLEAR REGULATORY COMMISSION, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition, NUREG-0800, Office of Nuclear Reactor Regulation, Washington, DC (2007).
[77] INTERNATIONAL ATOMIC ENERGY AGENCY, Periodic Safety Review for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-25, IAEA, Vienna (2013).
[78] NUCLEAR REGULATORY COMMISSION, Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, 10 CFR 50.65, US Govt Publishing Office, Washington, DC (1991).
[79] NUCLEAR REGULATORY COMMISSION, Preparation of Environmental Reports for Nuclear Power Stations, Rep. NUREG-0099, Regulatory Guide 4.2, Rev. 2, NRC, Washington, DC (1976).
[80] NUCLEAR REGULATORY COMMISSION, Preparation of Supplemental Environmental Reports for Applications to Renew Nuclear Power Plant Operating Licenses, Regulatory Guide 4.2S1, Supplement 1 to Regulatory Guide 4.2, NRC, Washington, DC (2000).
[81] NUCLEAR REGULATORY COMMISSION, Generic Environmental Impact Statement for License Renewal of Nuclear Plants, Rep. NUREG-1437, Vol. 1, Rev. 1, Office of Nuclear Regulatory Research, Washington, DC (2013).

[82] NUCLEAR REGULATORY COMMISSION, Generic Environmental Impact Statement for License Renewal of Nuclear Plants — Final Report, Rep. NUREG-1437, Vol. 2, Rev. 1, Office of Nuclear Regulatory Research, Washington, DC (2013).
[83] NUCLEAR REGULATORY COMMISSION, Generic Environmental Impact Statement for License Renewal of Nuclear Plants — Final Report, Rep. NUREG-1437, Vol. 3, Office of Nuclear Regulatory Research, Washington, DC (2013).
[84] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Aspects of Nuclear Power Plant Ageing, IAEA-TECDOC-540, IAEA, Vienna (1990).
[85] HERTER, K.H., "Effective utilization of technical information and development for ageing management in Germany", Int. Symp. of the Ageing Management and Maintenance of Nuclear Power Plants, Tokyo (2008).
[86] OECD-NUCLEAR ENERGY AGENCY, The Economics of Long-term Operation of Nuclear Power Plants, NEA-7054, OECD, Paris (2012).
[87] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Aspects of Long Term Operation of Water Moderated Reactors, IAEA-EBP-SALTO, IAEA, Vienna (2007).
[88] BOLZONI, F., CABASSI, R., PARCO A., "Review of singular point detection techniques by 43010", Proc. 7th Int. Symp. on Research in High Magnetic Fields, Parma (2004).
[89] INTERNATIONAL ATOMIC ENERGY AGENCY, Neutron Irradiation Embrittlement of Reactor Pressure Vessel Steels, Technical Reports Series No. 163, IAEA, Vienna (1975).
[90] INTERNATIONAL ATOMIC ENERGY AGENCY, Analysis of the Behaviour of Advanced Reactor Pressure Vessel Steels Under Neutron Irradiation, Technical Reports Series No. 265, IAEA, Vienna (1986).
[91] INTERNATIONAL ATOMIC ENERGY AGENCY, Reference Manual on the IAEA JRQ Correlation Monitor Steel for Irradiation Damage Studies, IAEA-TECDOC-1230, IAEA, Vienna (2001).
[92] INTERNATIONAL ATOMIC ENERGY AGENCY, Guidelines for Application of the Master Curve Approach to Reactor Pressure Vessel Integrity in Nuclear Power Plants, Technical Reports Series No. 429, IAEA, Vienna (2005).
[93] INTERNATIONAL ATOMIC ENERGY AGENCY, Application of Surveillance Programme Results to Reactor Pressure Vessel Integrity Assessment, IAEA-TECDOC-1435, IAEA, Vienna (2005).
[94] INTERNATIONAL ATOMIC ENERGY AGENCY, Effects of Nickel on Irradiation Embrittlement of Light Water Reactor Pressure Vessel Steels, IAEA-TECDOC-1441, IAEA, Vienna (2005).
[95] INTERNATIONAL ATOMIC ENERGY AGENCY, Guidelines for Prediction of Irradiation Embrittlement of Operating WWER-440 Reactor Pressure Vessels, IAEA-TECDOC-1442, IAEA, Vienna (2005).

GLOSSARY

The following definitions have been taken from the IAEA Safety Glossary, Terminology Used in Nuclear Safety and Radiation Protection, 2007 Edition, IAEA, Vienna (2007).

ageing

General *process* in which characteristics of a *structure, system or component* gradually change with time or use.

- ⓘ Although the term *ageing* is defined in a neutral sense — the changes involved in *ageing* may have no effect on *protection* or *safety*, or could even have a beneficial effect — it is most commonly used with a connotation of changes that are (or could be) detrimental to *protection and safety* (i.e. as a synonym of *ageing degradation*).

non-physical ageing. The *process* of becoming out of date (i.e. obsolete) owing to the evolution of knowledge and technology and associated changes in codes and standards.

- ⓘ Examples of *non-physical ageing* effects include the lack of an effective *containment* or *emergency* core cooling *system*, the lack of *safety design* features (such as *diversity*, separation or *redundancy*), the unavailability of qualified spare parts for old equipment, incompatibility between old and new equipment, and outdated *procedures* or documentation (e.g. which thus do not comply with current regulations).
- ⓘ Strictly, this is not always *ageing* as defined above, because it is sometimes not due to changes in the *structure, system or component* itself. Nevertheless, the effects on *protection and safety*, and the solutions that need to be adopted, are often very similar to those for *physical ageing*.
- ⓘ The term ***technological obsolescence*** is also used.

physical ageing. *Ageing* of *structures, systems and components* due to physical, chemical and/or biological *processes* (*ageing* mechanisms).

- ⓘ Examples of *ageing* mechanisms include wear, thermal or *radiation* embrittlement, corrosion and microbiological fouling.
- ⓘ The term ***material ageing*** is also used.

ageing degradation

Ageing effects that could impair the ability of a *structure, system or component* to function within its *acceptance criteria*.

- ⓘ Examples include reduction in diameter due to wear of a rotating shaft, loss in material toughness due to *radiation* embrittlement or thermal *ageing*, and cracking of a material due to fatigue or stress corrosion cracking.

ageing management

Engineering, *operations* and *maintenance* actions to control within *acceptable limits* the *ageing degradation* of *structures, systems and components*.

- ⓘ Examples of engineering actions include *design*, *qualification* and *failure analysis*. Examples of *operations* actions include surveillance, carrying out operating *procedures* within specified *limits* and performing environmental measurements.
- ⓘ ***Life management*** (or ***lifetime management***) is the integration of *ageing management* with economic planning: (1) to optimize the *operation*, *maintenance* and *service life* of *structures, systems and components*; (2) to maintain an acceptable level of performance and *safety*; and (3) to maximize the return on investment over the *service life* of the *facility*.

design

1. The *process* and the result of developing a concept, detailed plans, supporting calculations and specifications for a *facility* and its parts.[1]

2. The description of *special form radioactive material*, *low dispersible radioactive material*, *package* or *packaging* which enables such an item to be fully identified. The description may include specifications, engineering drawings, reports demonstrating compliance with regulatory *requirements*, and other relevant documentation.[2]

ⓘ This is a much more restricted definition than (1), and is specific to the Transport Regulations.

design basis

The range of conditions and *events* taken explicitly into account in the *design* of a *facility*, according to established criteria, such that the *facility* can withstand them without exceeding *authorized limits* by the planned *operation* of *safety systems*.

ⓘ Used as a noun, with the definition above. Also often used as an adjective, applied to specific categories of conditions or *events* to mean 'included in the *design basis*'; as, for example, in *design basis accident*, *design basis external events* and *design basis* earthquake.

licence

1. A legal document issued by the *regulatory body* granting *authorization* to perform specified *activities* related to a *facility or activity*.

ⓘ The holder of a current *licence* is termed a ***licensee***. Other derivative terms should not be needed; a *licence* is a product of the *authorization process* (although the term ***licensing process*** is sometimes used), and a *practice* with a current *licence* is an authorized *practice*.

ⓘ *Authorization* may take other forms, such as *registration*.

ⓘ The *licensee* is the person or organization having overall responsibility for a *facility* or *activity* (the ***responsible legal person***).

2. [Any *authorization* granted by the *regulatory body* to the *applicant* to have the responsibility for the *siting*, *design*, *construction*, *commissioning*, *operation* or *decommissioning* of a *nuclear installation*.][3]

3. [Any *authorization*, permission or certification granted by a *regulatory body* to carry out any activity related to management of *spent fuel* or of *radioactive waste*.][4]

licensing basis

A set of regulatory *requirements* applicable to a *nuclear installation*.

ⓘ The *licensing basis*, in addition to a set of regulatory *requirements*, may also include agreements and commitments made between the *regulatory body* and the *licensee* (e.g. in the form of letters exchanged or of statements made in technical meetings).

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[1] The terms *siting*, *design*, *construction*, *commissioning*, *operation* and *decommissioning* are normally used to delineate the six major stages of the lifetime of an *authorized facility* and of the associated *licensing process*. In the special case of *waste disposal facilities*, *decommissioning* is replaced in this sequence by *closure*.

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Regulations for the Safe Transport of Radioactive Material — 2005 edition, IAEA Safety Standards Series No. TS-R-1, IAEA, Vienna (2005).

[3] Convention on Nuclear Safety, INFCIRC/449, IAEA, Vienna (1994).

[4] Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management, INFCIRC/546, IAEA, Vienna (1997).

life/lifetime

design life. The period of time during which a *facility* or *component* is expected to perform according to the technical specifications to which it was produced.

operating life/lifetime. 1. The period during which an *authorized facility* is used for its intended purpose, until *decommissioning* or *closure*.
ⓘ The synonyms ***operating period*** and ***operational period*** are also used.

2. [The period during which a *spent fuel* or a *radioactive waste management facility* is used for its intended purpose. In the case of a *disposal facility*, the period begins when *spent fuel* or *radioactive waste* is first emplaced in the *facility* and ends upon *closure* of the *facility*.][4]

qualified life. Period for which a *structure, system or component* has been demonstrated, through testing, *analysis* or experience, to be capable of functioning within *acceptance criteria* during specific *operating conditions* while retaining the ability to perform its *safety functions* in a *design basis accident* or earthquake.

service life. The period from initial *operation* to final withdrawal from service of a *structure, system or component*.

life cycle management

Life management (or *lifetime management*) in which due recognition is given to the fact that at all stages in the lifetime there may be effects that need to be taken into consideration.
ⓘ An example is the approach to products, *processes* and services in which it is recognized that at all stages in the lifetime of a product (extraction and processing of raw materials, manufacturing, *transport* and distribution, use and reuse, and recycling and *waste* management) there are environmental and economic impacts.
ⓘ The term 'life cycle' (as opposed to lifetime) implies that the life is genuinely cyclical (as in the case of recycling or *reprocessing*).
ⓘ See *cradle to grave approach*.
ⓘ See *ageing management*.

life management (or lifetime management)

See *ageing management*.

maintenance

The organized activity, both administrative and technical, of keeping *structures, systems and components* in good operating condition, including both preventive and corrective (or *repair*) aspects.

corrective maintenance. Actions that restore, by *repair*, overhaul or replacement, the capability of a failed *structure, system or component* to function within *acceptance criteria*.
ⓘ Contrasted with *preventive maintenance*.

periodic maintenance. Form of *preventive maintenance* consisting of servicing, parts replacement, surveillance or testing at predetermined intervals of calendar time, operating time or number of cycles.
ⓘ Also termed ***time based maintenance***.

planned maintenance. Form of *preventive maintenance* consisting of refurbishment or replacement that is scheduled and performed prior to unacceptable degradation of a *structure, system or component*.

predictive maintenance. Form of *preventive maintenance* performed continuously or at intervals governed by observed condition to monitor, diagnose or trend a *structure, system or component*'s *condition indicators*; results indicate present and future functional ability or the nature of and schedule for *planned maintenance*.
ⓘ Also termed ***condition based maintenance***.

preventive maintenance. Actions that detect, preclude or mitigate degradation of a functional *structure, system or component* to sustain or extend its useful life by controlling degradation and *failures* to an acceptable level.
ⓘ *Preventive maintenance* may be *periodic maintenance*, *planned maintenance* or *predictive maintenance*.
ⓘ Contrasted with *corrective maintenance*.

reliability centred maintenance (RCM). A *process* for specifying applicable *preventive maintenance* requirements for *safety related systems* and equipment in order to prevent potential *failures* or to control the *failure modes* optimally. *RCM* utilizes a decision *logic* tree to identify the *maintenance* requirements according to the *safety* consequences and operational consequences of each *failure* and the degradation mechanism responsible for the *failures*.

periodic safety review

A systematic reassessment of the *safety* of an existing *facility (or activity)* carried out at regular intervals to deal with the cumulative effects of *ageing*, modifications, operating experience, technical developments and *siting* aspects, and aimed at ensuring a high level of *safety* throughout the *service life* of the *facility (or activity)*.

qualification

equipment qualification. Generation and *maintenance* of evidence to ensure that equipment will operate on demand, under specified *service conditions*, to meet *system* performance requirements.
ⓘ See footnote 5.
ⓘ More specific terms are used for particular equipment or particular conditions; for example, ***seismic qualification*** is a form of *equipment qualification* that relates to conditions that could be encountered in the event of earthquakes.

qualified life

See *life*.

safety function

A specific purpose that must be accomplished for *safety*.
ⓘ Footnote 6 lists 19 *safety functions* to be fulfilled by the *design* of a nuclear power plant in order to meet three general *safety requirements*:

(a) The capability to safely shut down the reactor and maintain it in a safe shutdown condition during and after appropriate *operational states* and *accident conditions*;
(b) The capability to remove *residual heat* from the reactor core after shutdown, and during and after appropriate *operational states* and *accident conditions*;
(c) The capability to reduce the potential for the release of *radioactive material* and to ensure that any releases are within *prescribed limits* during and after *operational states* and within *acceptable limits* during and after *design basis accidents*.

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[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment and Verification for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-1.2, IAEA, Vienna (2002).

[6] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power Plants: Design, IAEA Safety Standards Series No. NS-R-1, IAEA, Vienna (2000).

ⓘ This guidance is commonly condensed into a succinct expression of three ***main safety functions*** for nuclear power plants:

(a) *Control* of *reactivity*;
(b) Cooling of *radioactive material*;
(c) *Confinement* of *radioactive material*.

In earlier *IAEA publications*, ‘basic *safety function*’ and ‘fundamental *safety function*’ were also used.

safety issues

Deviations from current *safety standards* or *practices*, or weaknesses in *facility design* or *practices* identified by plant *events*, with a potential impact on *safety* because of their impact on *defence in depth*, *safety* margins or *safety culture*.

screening

A type of *analysis* aimed at eliminating from further consideration factors that are less significant for *protection* or *safety* in order to concentrate on the more significant factors. This is typically achieved by consideration of very pessimistic hypothetical *scenarios*.

ⓘ *Screening* is usually conducted at an early stage in order to narrow the range of factors needing detailed consideration in an *analysis* or *assessment*.

structures, systems and components (SSCs)

A general term encompassing all of the elements (items) of a *facility* or *activity* which contribute to *protection and safety*, except *human factors*.

ⓘ ***Structures*** are the passive elements: buildings, vessels, shielding, etc. A ***system*** comprises several *components*, assembled in such a way as to perform a specific (active) function. A ***component*** is a discrete element of a *system*. Examples of components are wires, transistors, integrated circuits, motors, relays, solenoids, pipes, fittings, pumps, tanks and valves.

ABBREVIATIONS

AMP	ageing management programme
ARDM	age related degradation mechanism
ASME	American Society of Mechanical Engineers
ASN	Autorité de sûreté nucléaire
ASTM	American Society for Testing and Materials
BASY	Betriebsauftrags- und Störmeldesystem (operating requirements and non-conformance reporting system)
BDBA	beyond design basis accident
BMI	bottom mounted instrumentation
BWR	boiling water reactor
CA	condition assessment
CANDU	Canada deuterium-uranium reactor
CASS	cast austenitic stainless steel
CFD	computational fluid dynamics
CFR	Code of Federal Regulations
CIGAR	channel inspection and gauging apparatus for reactors
CRCS	control rod control system
CRD	control rod drive
CRDM	control rod drive mechanism
CRP	coordinated research project
CSA	Canadian Standard Association
CUF	cumulative usage factor
DBA	design basis accident
DEGB	double ended guillotine break
DHC	delayed hydride cracking
DMW	dissimilar metal weld
dpa	displacements per atom
EBP	extra budgetary programme
ECP	electrochemical corrosion potential
EdF	Électricité de France
ENIQ	European Network for Inspection and Qualification
EOL	end-of-life
EPRI	Electric Power Research Institute
EQ	equipment qualification
EU	European Union
FAC	flow accelerated corrosion
FMEA	failure mode and effect analysis
FSAR	final safety analysis report
GALL	generic ageing lessons learned
GOST	Gosudarstvennyy standard (Russian national standard)
HWC	hydrogen water chemistry
HWR	heavy water reactor
I&C	instrumentation and control
IASCC	irradiation assisted stress corrosion cracking
IDRPVM	International Database on Reactor Pressure Vessel Material
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
IFRAM	International Forum for Ageing Management
IGALL	international generic ageing lessons learned
IGSCC	intergranular stress corrosion cracking

ILCM	integrated life cycle management
ISI	in-service inspection
ISO	International Organization for Standardization
JPDR	Japan Power Demonstration Reactor
KHNP	Korea Hydro & Nuclear Power
KTA	Kerntechnischer Ausschuss (German nuclear safety standards)
LA	life assessment
LAS	low alloy steel
LBB	leak before break
LERF	large early release frequency
LLC	low leakage core
LOCA	loss of coolant accident
LR	licence renewal
LRA	licence renewal application
LTO	long term operation
LWR	light water reactor
LWRS	light water reactor sustainability
MC	Master Curve
NDE	non-destructive examination
NDT	non-destructive testing
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NPSH	net positive suction head
NRA	Nuclear Regulation Authority
NRC	Nuclear Regulatory Commission
ODSCC	outside diameter stress corrosion cracking
OECD	Organisation for Economic Co-operation and Development
OL	operating licence
OPDE	OECD/NEA piping failure data exchange
PDCA	plan–do–check–act
PHWR	pressurized heavy water reactor
PIP	periodic inspection programme
PLiM	plant life management
PM	preventive maintenance
PSR	periodic safety review
PTS	pressurized thermal shock
PWR	pressurized water reactor
PWSCC	primary water stress corrosion cracking
RCM	reliability centred maintenance
RCS	reactor coolant system
RPV	reactor pressure vessel
RPVH	reactor pressure vessel head
SALTO	safety aspects of long term operation
SAR	safety analysis report
SC	structure and component
SCAP	Stress Corrosion Cracking and Cable Ageing Project
SCC	stress corrosion cracking
SER	safety evaluation report
SG	steam generator
SLAR	spacer location and repositioning
SRP-LR	standard review plan for licence renewal applications
SS	stainless steel
SSC	structure, system and component

TGSCC	transgranular stress corrosion cracking
TLAA	time limited ageing analysis
TOFD	time of flight diffraction
TS	technical specification
TSO	technical support organization
TSP	tube support plate
V&V	Verification and Validation
WANO	World Association of Nuclear Operators
WWER	water cooled water moderated power reactor

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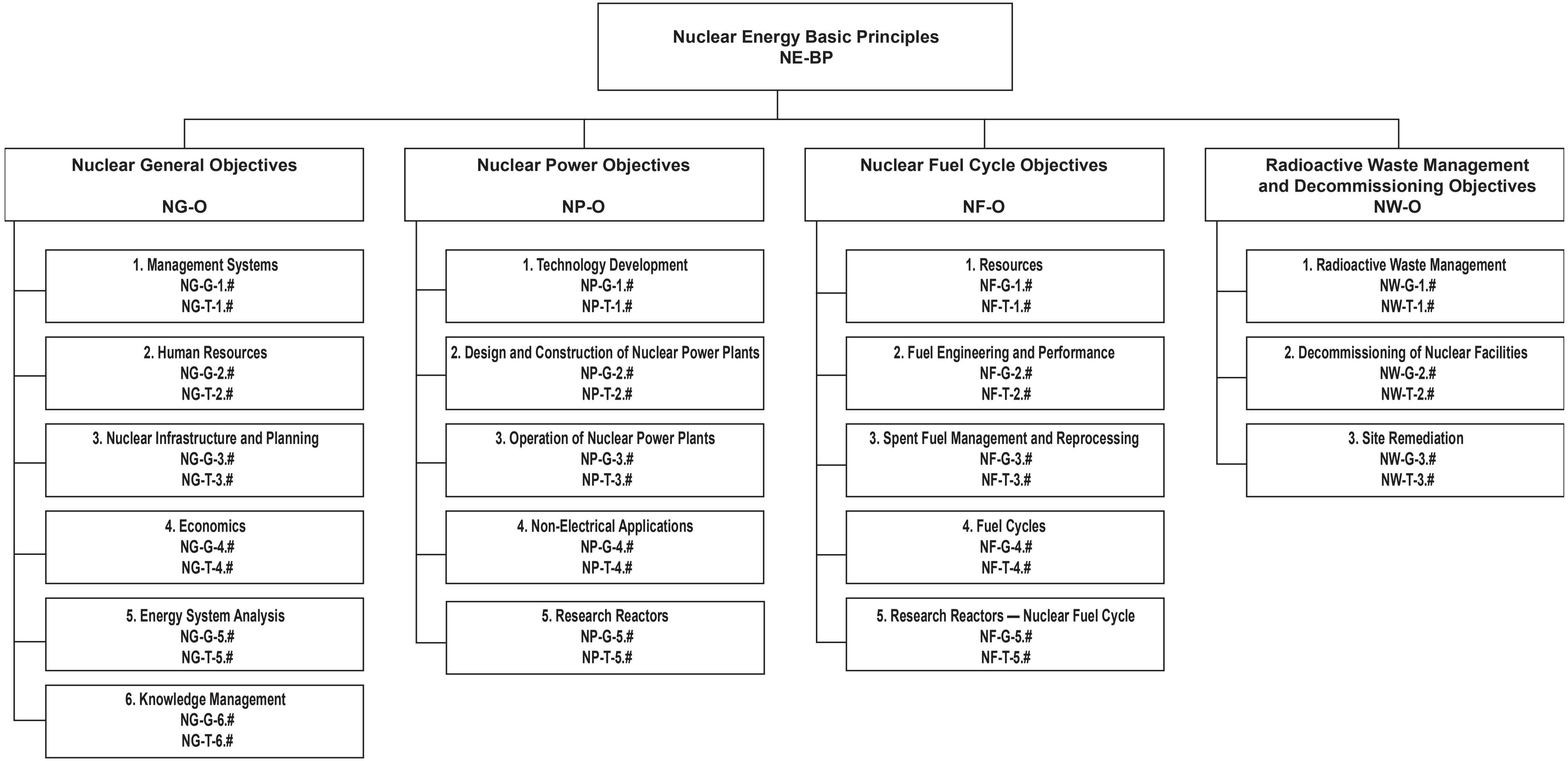

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