Handbook of Ageing Management for Nuclear Power Plants

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FOREWORD

One of IAEA's statutory objectives is to *"seek to accelerate and expand the contribution of atomic energy to the development of peace, health and prosperity throughout the world"*. One way this objective is achieved is through the publication of the IAEA document series that include the IAEA Safety Standards and the IAEA Nuclear Energy Series (NES).

The IAEA Safety Standards establish "*standards of safety for protection of health, and minimization of danger to life and property"* as stated in Statute Article III, A.6. The safety standards consist of Safety Fundamentals, Safety Requirements, and Safety Guides. They represent the international reference for nuclear safety for the benefit of the Member States, their regulatory bodies and other national authorities. They are written in a regulatory style and are binding on the IAEA for its own operation.

Beside the safety standards, the IAEA produces a Nuclear Energy Series of documents designed to encourage and assist Member States in the development and application of nuclear energy for peaceful purposes. The NE series contains the best technologies and practices available as well as future advancements. They are written as guidelines, reports, handbooks and their content is exclusively based on input from leading subject matter experts for the benefit of Member States, government officials, nuclear power plant owners and operators, support organizations, academia and others.

Within the nuclear energy series, one prominent collection is the engineering documentation dealing with plant ageing management, with degradation mechanisms, failure prevention and mitigation programmes. They contain elements of material science and techniques as applied to ageing systems, structures and components ageing in nuclear power plants.

The same collection includes documentation on Plant Life Management (PLiM) that considers ageing management in the context of the nuclear power plant entire lifecycle even long term operation beyond the originally assumed operating period taking into consideration management objectives including economic aspects and the market in which the plant operates. At the same time the PLiM programme allows the optimization of monitoring, surveillance and maintenance programmes. It can achieve all this by relying on advanced deterministic and probabilistic techniques, the use of an integrated ageing knowledge base, backed by research and development and operation feedback.

Beyond the plant ageing documentation, the agency also organizes international conferences as well as working groups, coordinated research projects, technical and consultative meetings on systems, structures and component ageing in nuclear power plants as well as plant life management., This handbook on ageing management in nuclear power plants is intended as an easily accessible compendium of the ageing management knowledge base, including insights from Plant Life Management programmes, as they are being implemented by an increasing number of IAEA Member States.

The work contributed by the subject matter experts involved in the drafting and in the review of this handbook, is greatly appreciated. They are acknowledged at the end of this document. The IAEA officer responsible for this publication is K.S. Kang of the Division of Nuclear Power.

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1. INTRODUCTION

1.1. BACKGROUND

This Ageing management handbook has been developed in compliance with the requirements of the IAEA safety standards and the IAEA safety guide on ageing management for nuclear power plants [1] [2]. The primary safety goal of an ageing management programme should be to ensure the availability of all required safety functions throughout the service life of a nuclear power plant (NPP), taking into account the changes that occur with time and use. Consequently ageing management should be planned primarily with the safety principles in mind, not only during the operating phase of a plant, but from the very first basic conceptual design as stated in requirement 31 of IAEA Safety Standard on Design [3] and continuing into the operating period as stated in Requirement 14 of the IAEA Safety Standard on commissioning and operation [4].

The IAEA produced a comprehensive series of informative documents on ageing to capture the accumulated knowledge on the diverse aspects of ageing management and plant life management, including R&D findings, engineering support and operating experience. This Handbook has been written with the intent of providing both the underlying principles and the practical knowledge developed in the field on Ageing and Life Management of nuclear power plants. It is also a compendium and an update of the ageing information affecting structure, systems and components (SSC's) as collected by the Agency over the years.

The information is presented as far as possible in a concise manner, relying on flowcharts with a reference to the underlying principles in order that the handbook may also be used as a reference text. In addition, on-going research and development, innovative technology and procedures are also compiled by experts of participating IAEA member states in close collaboration with a dedicated organization in the field, known as the International Forum on Reactor Ageing Management (IFRAM).

The Member States and the IAEA have extensively re-examined the knowledge base on ageing management following the Fukushima accident [5]. The IAEA has concluded that it is difficult to entertain that any ageing processes may have contributed to the occurrence and consequences of the Fukushima Dai-ichi NPP accident, during the period between the time of the accident occurrence and the time when the accident developed beyond the plant design basis.

1.2. TERMINOLOGY AND DEFINITIONS

Ageing is 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 NPP documentation and of staff training.

Ageing Management Programme (AMP) is a set of policies, processes, procedures, arrangements, and activities for managing the ageing of structures, systems and components (SSCs) for a nuclear power plant (NPP)

Beyond Design Basis Accident (BDBA) comprises accident conditions more severe than a design basis accident that may or may not involve core degradation. They are not fully considered in the design process. However "beyond design-basis" accident sequences are analysed to the extent needed to fully understand their consequences and the robustness of the design

Condition assessment (CA) is defined as an ageing assessment methodology applied to systems, as well as components and structures, or groups of components with similar characteristics (commodities).

Control Rod Control System (CRCS) is a system capable of providing dynamic control of core power by actuating rod movements in and out of the ore. The system is distributed across several control groups.

Design Basis Accident (DBA) is defined as 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 (FEA) is a computational method used in engineering for finding an approximate solution to a problem involving a complex geometrical figure by using a simplified model 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

Institute of Electrical and Electronics Engineers (IEEE) is a professional association based in New York City that is dedicated to advancing technological innovation and excellence in the electrical and electronic fields

Integrated Life Cycle Management (ILCM) is a project integration management methodology utilized to enhance the viability and sustainability of localized technologies, namely the localization programme of a certain level of nuclear power generation technology

The IAEA International Generic Ageing Lessons Learned (IGALL) Safety Report provides a common internationally agreed basis on what constitutes acceptable ageing management programmes, as well as a knowledge base on ageing management for design of new plants, design reviews, safety reviews (such as periodic safety review), etc., and serves as a roadmap to available information on ageing management.

Irradiation Assisted Stress-Corrosion Cracking (IASCC) : Certain materials such as Austenitic steels for examples are known to degrade while in service as core components in a nuclear reactor. Among the many factors characterizing susceptibility of these materials to IASCC is hardening, and at the microstructural level, the dislocation of the material microstructure and changes in the grain boundary composition.

Large Early Release Frequency (LERF) is defined as 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 (LA) is defined as an ageing assessment methodology applied to critical and/or complex components and structures that involve mainly passive functions and typically are not expected to be replaced within the original design life of the plant.

Long Term Operation (LTO) of a nuclear power plant may be defined as operation beyond an established time frame set forth by, for example, licence term, design, standards, licence and/or regulations, which has been justified by safety assessment, with consideration given to life limiting processes and features of systems, structures and components (SSCs).

Maintenance, Surveillance, In-Service Inspection (MSI) is a programme in NPPs designed to ensure that the levels of reliability and availability of all plant structures, systems and components continue to meet the assumptions and the intent of the design through the programme duration and the plant life.

Non-Destructive Examination (NDE) is generally understood to mean using non-destructive testing 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 (OLM) is the application of instrumentation and observation techniques in order to evaluate equipment performance by assessing its consistency with other plant or baseline indications. Industry experience at several plants and R&D have shown this overall approach to be very effective in identifying equipment that exhibits degrading or inconsistent performance characteristics

Periodic Safety Review (PSR) is a comprehensive safety review of all important aspects of safety, carried out at regular intervals, typically every ten years. In addition, a PSR may be used in support of the decision making process for licence renewal or long term operation, or for restart of a nuclear power plant following a prolonged shutdown

Plant Life Management (PLiM) is the integration of ageing and economic planning to maintain a high level of safety, optimize the operation, maintenance and service life of SSCs, maintain an acceptable level of performance, maximize return on investment over the service life of the NPP; and provide NPP utilities/owners with the optimum pre-conditions for long term operation (LTO).

Probability of Detection (POD) of degraded portions of cables or of buried piping in nuclear power plants is a technique used to help optimize inspection and repair programmes. Curves of detection probability versus range of stressors and stressor fluctuations can help predict location of high degradation probability.

IAEA Safety Aspects of Long Term Operation (SALTO) peer review service is designed to assist operating organizations in adopting a proper approach to LTO including implementing appropriate activities to ensure that plant safety will be maintained during the LTO period. IAEA Service Series 26 SALTO Peer Review Guidelines provides details of this peer review service.

1.3. OBJECTIVE

The ageing management programmes for structures, systems and components (SSCs) in NPPs are aimed at preserving the current licensing basis envelope by adequately managing safety margins, by controlling performance degradation and improving system availability and capacity factors. Given the large interests at stake, the nuclear industry has established owners groups to share their operating experience and resources, to develop the best tools and R&D available in order to help individual plants successfully manage the ageing of their assets. This handbook is intended as a general reference document on NPP ageing, covering topics such as degradation mechanisms, and ageing management techniques. It should be used as a basic reference book by newcomer countries as they set up their ageing management programmes, as a learning tool for new staff, and as a common base at an international level to facilitate information exchange and awareness of ageing management issues.

The IAEA recognizes that plants in the future will be operated for longer periods of time and that advanced ageing management techniques will become even more important if operators are to improve and maintain the safety and reliability of NPPs, even beyond their original design life. To this effect, this handbook intends to help Member States to start planning for long term operation as early as possible in the life cycle management of their NPPs.

1.4. SCOPE

This handbook has been prepared in full compliance with the IAEA safety guides on ageing management and other IAEA publications on NPP ageing. In addition, it incorporates the knowledge base acquired by Member States on component specific ageing management and shows how to integrate this knowledge with overall plant life management insights for an optimization of the ageing management programme of all SSCs. Rather than taking each SSC in isolation, the manual shows how to give appropriate consideration to the associated and interfacing systems, to ultimately improve the unit overall capacity factor. The handbook shows why a correct hierarchy and sequence of all aging management interventions must be established to correctly plan the life management of plant assets. In this hierarchy large non-replaceable and difficult to replace, usually safety related, passive SSCs require the most attention and investment in managing their degradation.

Beyond long-lived passive components, the scope of this manual extends also to active components and sub-components as well as to classes of smaller devices and groups of similar constituents. Topics such as condition-based assessments and reliability-centred maintenance techniques, on-line monitoring data collection and processing techniques as well as databases with statistical data tracking capability are presented as modern tools capable of addressing plant life management and specific component ageing related issues. The information is always presented in a very practical easy-to-use consultative way, typical of handbooks.

1.5. END USERS

This IAEA handbook on ageing management of NPPs is designed for the training and support of nuclear power plant staff, maintenance managers, vendors, research organizations, and regulators to assist them in their work related to SSC ageing in operating NPPs. It should be of particular interest to operators of new Member States in the process of introducing nuclear power in their countries.

1.6. DOCUMENT STRUCTURE

The Handbook is derived from both the successful experiences and the failures in ageing management practices for SSCs in PWRs BWRs and PHWR plants with particular emphasis on the principles of proactive ageing management and plant life management.

Chapter 1 is an introductory chapter. It contains sections on the background, the circumstances under which the handbook is being published, on its scope and objectives and on the target audience. Section 1.3 contains a list of definitions used in AM, PLiM, MSI. It provides an overview of the basic concepts of ageing management, an analysis of the impact of ageing on plant safety and performance, common weaknesses of traditional ageing management programmes, operation feedback, R&D and other activities on ageing management including IAEA activities and member state initiatives on exploring the implications and drawing lessons learned, studying configuration changes and implementation plans derived from the fallout of the Fukushima accident.

Chapter 2 provides upfront information on the general features of ageing programmes, on the proactive approach to ageing Management which includes, the proactive strategy for AM and the development and optimization of AM activities for structures, systems and components in a nuclear power plant.

Chapter 3 is intended to be a reference manual on material degradation and degradation mechanisms. It includes definitions, the underlying science, examples, the knowledge base, R&D and the state of the art discoveries particularly in the areas of monitoring and detection technologies applied to phenomena such as radiation embrittlement, fatigue, stress corrosion cracking, general corrosion, and specifics on erosion-corrosion and flow accelerated corrosion. Specific phenomena involving more particularly the SSCs of BWRs, PWRs and CANDU type reactors are also discussed.

Chapter 4 deals with the ageing phenomena and the degradation mechanisms related to critical components, such as the RPV and its parts, namely the vessel head, the vessel walls and the RPV internals in PWRs and BWRs. In parallel for PHWRs ageing phenomena typical of natural uranium fuelled reactors affecting pressure tubes, fuel channels, feeders and other components within the reactor face region are summarized together with the best recommendations from the extensive R&D conducted over the years in support of the reactor life cycle. Other major components discussed in this chapter are the steam generators, power and low voltage cables, I&C components, concrete ageing and ageing programmes for non-metallic components. The material is presented in a concise, but systematic way. In addition, guidelines are given in chapter 4 for the planning and implementation of ageing management programmes.

Chapter 5 describes the accumulated experience in material degradation and mitigation options made available through extensive international cooperation and conferences such as IAEA's conference on PLiM. Dissemination and identification of these phenomena and possible remedial measures facilitates decision-making on maintenance intervention and mitigation programmes.

Chapter 5 also shows how prioritization of all activities can be established through risk analysis and the definition of degrees of criticality. Prioritization can also be achieved by means of a single point of vulnerability (SPV) analysis. Depending on their categorization SSCs are assigned to a particular ageing assessment programme (time based assessment, condition assessment or life assessment). As a result of these assessments, a critical spare part list can be prepared and spare part inventories can be defined in support of repair/replace (R/R) strategies. It should be noted that the obsolescence in either design solutions or system configuration is also dealt with in Chapter 5 as part of ageing assessment programmes for SSC's.

Chapter 6 deals with the assessment of ageing and ageing management in operating plants using inspection and evaluation tools and the processes and regulations developed to measure the effectiveness of AM programmes with regards to safety, safety margins, reliability and performance management. Chapter 6 deals also with ageing assessments done specifically for LTO applications for example computational analysis to revalidate safety margins, Time Limited Ageing Analysis (TLAA) to consider the effects of actual SSC aging during the time limit assumptions of the initial licence, (for example 40 years) and provide the basis for conclusions related to the capability of the SSC to perform its intended function during LTO, predictive tools to determine ageing progression and remaining life prognosis, condition monitoring instrumentation, testing and engineering tools to proactively develop preventive maintenance programmes aimed at mitigating ageing degradation, Validating tools to confirm equipment qualification (EQ) programmes, pressure boundary programmes for LTO, probabilistic techniques and cost-benefit assessments to help design and optimize MSI.

Chapter 7 deals with the regulatory framework concerning ageing and ageing management in typical international jurisdictions and the common approach and best practices recommended by the IAEA and other international organizations. Legal criteria and procedures for the safety assurance of long-term operation are extensive and complex and different countries have different approaches and priorities in achieving the same safety goals. Therefore the readers of this handbook are well advised to pay particular attention in understanding the material provided in chapters 1-6 in order to better understand the vast diversities in the legal criteria described herein.

Chapter 8 treats of organizational structures necessary to manage ageing effectively, to implement guidance documents and methodologies, to establish data bases, periodic updates, and reporting systems, to prepare periodic safety reviews and to develop effective corrective action plans.

Chapter 9 discusses innovative techniques in ageing management and includes a survey of international experience. The various degrees of operator maturity and the wide distribution of nuclear power plants in the world is such that Plant Life Management, already established in advanced countries, is a relatively new field for traditional operators still reluctant to take the step. Regulators however are increasingly applying pressure on these Licensees to adopt more advanced Plant Life Management methods and the right mix of probabilistic and deterministic approaches. New international cooperation on NPP ageing and Plant Life Management is ongoing under various programmes including IAEA's PLiM programmes, OECD-NEA's SCAP, the International Forum on Reactor Ageing Management (IFRAM) as well as the European Commission's NULIFE, NUGENIA and related programmes.

Chapter 10 contains a summary of the main elements of modern NPP ageing management programmes and high level recommendations are also therein provided.

The Annex contains country reports where the various national approaches and experiences related in large part by the participating experts and authors of this handbook are summarized for the benefit of all users.

2. OVERVIEW OF AGEING MANAGEMENT

2.1. BASIC CONCEPTS

The Primary objective of an ageing management programme is to maintain the reliability and availability of the required safety functions throughout the service life of the NPP, in accordance with the licensing basis. This entails preserving SSC performance, and plant performance targets as a whole and requires that the plant ageing management programme addresses:

Physical ageing: the ageing of structures, systems and components 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.

An efficient ageing management programme (AMP) requires an active involvement in monitoring, inspection and surveillance of critical SSCs to follow and compare actual degradation against predictions. Critical SSCs mainly refer to long life, irreplaceable and passive components that, if not checked, may limit the useful service life of the plant. When ageing processes are known, they can be monitored and mitigated through an appropriate AMP which should contain a set of policies, procedures, instructions, and detailed planned activities designed to successfully manage SSC ageing in a NPP.

Ageing management knowledge should be applied at each stage of the life cycle beginning with first conceptual design, and on through detail design, construction, commissioning, including design changes, and commercial operations including refurbishments, shutdowns and re-starts, long term operation and even decommissioning.

Maintenance activities are usually separated from ageing control activities, although the programmes may be integrated. Maintenance is routinely applied mainly to active and replaceable components. It includes run to failure constituents, bulk materials and part inventories that must be kept in good operable conditions.

Degradation and ageing are terms used to describe SSC deteriorations under operating conditions. However, they should not be used interchangeably:

Degradation is a gradual deterioration of one or more characteristics of a SSC that could at one point impair its ability to perform its required functions within acceptance limits.

Ageing is a general deterioration process in which the characteristics of an SSC gradually change with time and use.

Ageing management is a key element of the safe and reliable operation of NPPs and as such it is the object of particular attention of regulators and industry auditors. The cumulative effects of ageing and obsolescence on the safety of NPPs are re-evaluated periodically by license holders and reviewed by regulators using for example PSRs or equivalent processes of safety evaluation.

The PSR method is typically used in Member States with unlimited or continuing licenses. Most European countries apply the PSR process periodically (for example every 10 years) as their main safety assessment method against the effects of ageing and obsolescence. Most regulators requesting the application of the PSR method call for updates to ageing management programmes and critical component life assessments as pre-requisites to continue validating the operating license of nuclear power units.

A second approach is the license renewal process. It is applied in countries where limited term licenses are issued (for example a term license of 40 years). At the end of the term license, a license renewal application must be submitted to the regulator if longer operation of the plant can be justified.

The USA use the License Renewal Application (LRA) concept. This does not mean that safety checks are ignored during the 40 year duration of the first licence. The regulator takes it upon itself to update whenever necessary the regulation of the current licensing basis (CLB) 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 CLB compliance checks, a thorough review of the maintenance programmes, the 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 former two as happens in Korea, where PSR is used as the basic safety assessment method every 10 years, complemented with updates to safety requirements when they become prominent on the international scene and the regulator judges then appropriate for the local plants.

Finally, it is to be noted that advanced operators use PLiM technology, overall plant management support technologies, that apply a systematic SSC ageing review using screening techniques to categorize and prioritize SSCs and a set of advanced tools including integrated R&D knowledge databases, operation feedback data bases, elements of probabilistic, reliability and economic analysis and other engineering and financial tools to help manage the whole plant life cycle. PLiM analysts study both passive and active SSCs to help optimize ageing management, maintenance, inspection and surveillance programmes and effectively support management decisions regarding normal operations, longer term operating extensions, large component replacements, modernization, refurbishment and decommissioning plans.

2.2. GENERAL FEATURES OF AGEING PROGRAMMES

Regulatory requirements for ageing management at the national level must be developed before an operating license can be released. It may be that the national regulatory authority adopted the ageing regulations from the regulator of the NPP technology originating country. If this were the case, the national regulator should nevertheless review and update and adapt ageing management requirements and guidance to the local conditions. The national regulator should require that all ageing issues be identified and documented in the safety analysis report and periodically updated throughout the plant lifetime.

2.2.1. Data collection and record keeping

Without a good data collection and a good record keeping system the AM of NPPs can only be reactive and is eventually bound to damage plant performance, including its safety performance. In order to prevent gaps and weaknesses in maintenance programmes, the data required is essentially of three types: baseline data, operating history data and maintenance history data.

The baseline data should include:

— Component data sheets with descriptive and functional information, its interfacing components, its boundary conditions, its relationship to the system and to the unit in which it operates.
— Operating and performance requirements including its design service conditions and any other operational limits.
— Initial, un-degraded material conditions such as its critical piping and component wall thickness profile especially in the case of critical pressure boundary components subject to wall thinning
— Installation records, including specific manufacturing defects and exceptions, any welding specs used, any deviations, the component orientation, as-built conditions, supports, and any specific modifications from the original design due to construction deviations or field changes, their justifications or their design reconciliation statements as accepted by the local regulatory authority.
— Functional capabilities as tested during commissioning,
— Component calibration, if applicable

The operating history should include: Actual service conditions experienced including any relevant evolution of process parameters such as,

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

The maintenance history records should include:

— Condition monitoring of critical components including 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 cost in terms of influence on critical path and system down time)

2.2.2. Continuing equipment reliability improvement programmes

A continuing SSC reliability improvement programme is of key importance to achieving high plant capacity factors and longer operation terms. The main constituents 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.

Reliability improvement heavily depends on failure prevention, which in turn relies on the capability to:

— Measure the symptoms of incipient failures,
— Analyse operations feedback, any events and failures at other plants
— identify premonitory signs of operating anomalies

In order to profitably use these techniques operators should acquire the capability to map the probability of failure locations and this is possible if they know the vulnerability to degradation mechanisms of all their SSCs, if they effectively use inspection, surveillance, testing and monitoring equipment as for example:

— Installed instrumentation assigned where possible to multiple use (e.g. Process parameter measurement and degradation monitoring) as for service and process temperature indicators, coupled with recorders and analysers to monitor not just temperatures but also infer fouling in critical component cooling heat exchangers
— The use or the addition of advanced monitoring systems such as loose part monitoring system, fatigue monitoring system, vibration monitoring system and acoustic monitoring system,
— Oil sampling,
— Thermography,
— Electro-magnetic signature monitors for motorized pumps and valves

Figure 1 shows a process used to screen the equipment and apply monitoring where it can be effective in managing the degradation and ageing processes. The process begins with a review of the ageing monitoring techniques applied, an assessment of their limitation, compatibility and accuracy versus the ageing observed in the field from the ageing programme in the plant. If gaps are found, the ageing management programme needs to be complemented or upgraded. Otherwise, a separate ageing management programme should be created. The degradation mechanism observed needs to be separately monitored and a method to diagnose its growth needs to be implemented. In addition, the ageing management programme for this specific mechanism needs to be incorporated into the design basis and into the periodic safety review documentation. [6]

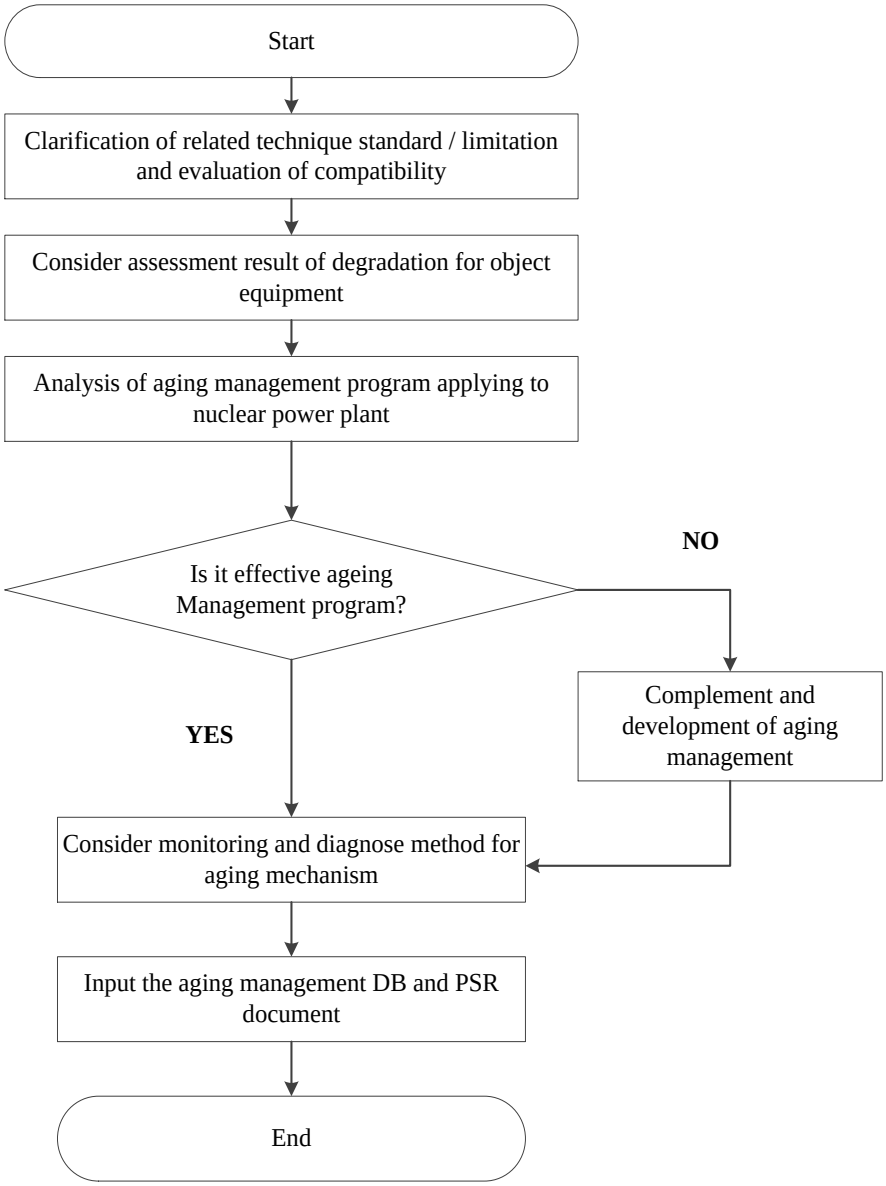


FIG.1. Programme Monitoring and Managing the Degradation and Ageing Process

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

If the plant life management programme is not well established and systematically applied, ageing and maintenance activities on safety critical components typically display the following issues:

— Duplication of effort between ageing management and maintenance activities,
— Conflicts in planning activities and re-work,
— Deficiencies in the capacity to diagnose incipient failures and recognize age related degradation (e.g. planning correctly routine walk-downs and on-line monitoring strategies). These deficiencies may be due to inadequacies in the administrative processes and procedures to monitor system and component performance, resulting in deficient or missing periodic walk-downs and improper visual observations for the successful detection of non-conformances.
— Lack of effective inspection procedures during maintenance. For example inadequate instructions at the craft level to report “as found” and “as-repaired” non-conformances in a component being

maintained.

- Lack of a user-friendly feedback system, which should include a condition grading code to characterize the equipment state and a component or system condition evaluation instruction to guide the evaluation and the selection of failure prevention barriers or ageing mitigation actions and an approval process by the system engineer or higher authorities.
- Lack of an adequate equipment history and corrective action database containing performance limits which does not allow the systematic prediction of failure trends of similar components across the plant and in other plants.
- Lack of a long-term maintenance strategy to regulate for example how to proactively maintain components for as long as it is economically feasible
- Lack of a systematic condition assessment programme for safety critical components making use of diagnostic techniques to prevent failures
- Lack of a systematic inaccessible equipment life assessment programme

2.2.4. Ageing Management Assessment and Screening of SSCs

Whether an operator has implemented or not a PLiM programme with probabilistic tools in support of its ageing management programme, still the first step all operators are forced to take remains that of screening all SSCs to determine those that are going to be enrolled in its ageing management programme. This is because it is neither practicable nor necessary to evaluate and quantify the extent of ageing degradation for every individual SSC. Operators must focus their resources on those SSCs that can have a negative impact on the safe operation and economic viability of the plant. This should include SSCs that do not have safety functions but whose failure could prevent other SSCs from performing their intended safety functions. As a minimum a safety based approach, such as the one outlined in [1], should be applied to the screening of SSCs as a pre-requisite to the outline of any (even the simplest) ageing management programme. See Figure 2 for flow chart of SSC screening for a NPP ageing management programme.

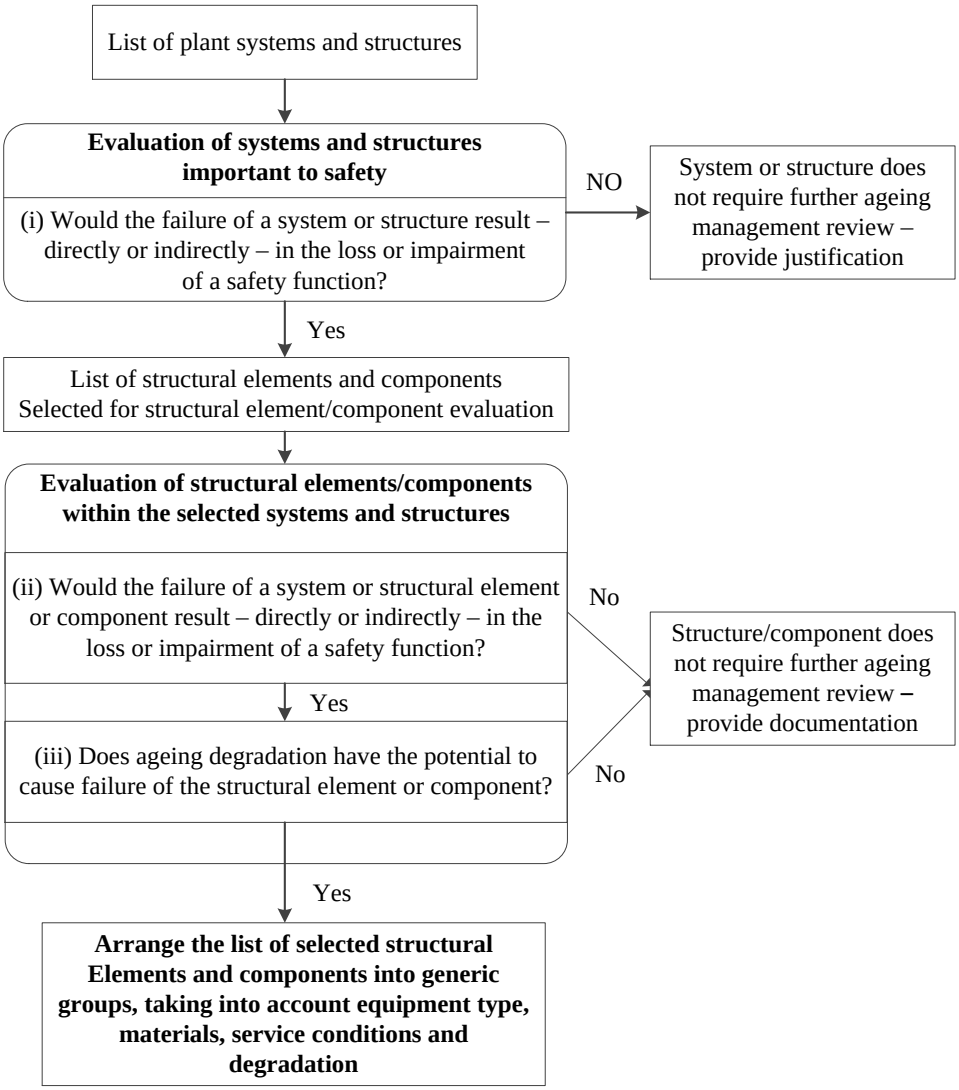


FIG.2. Flow chart of SSC screening for a NPP Ageing Management Programme

The first step of a SSC screening exercise is to determine whether the ageing mechanisms and their effects on the SSC safety, reliability and performance are all 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 post-service examination and of the results of any test that may have been run 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, if tests have been conducted. Results can be quantified and correlated to predict future development of the degradation.

Essentially, in the screening of all SSCs, those that are important to safety are identified on the basis of whether or not their malfunction or failure could lead to the loss or impairment of a safety function. Also the structural elements and sub-components of safety sensitive SSCs whose failure could lead to the loss or impairment of a safety function should be identified and grouped by type. As an example, the outline of a typical screening process conducted in Korea's nuclear power plants is illustrated in Figure. 3.

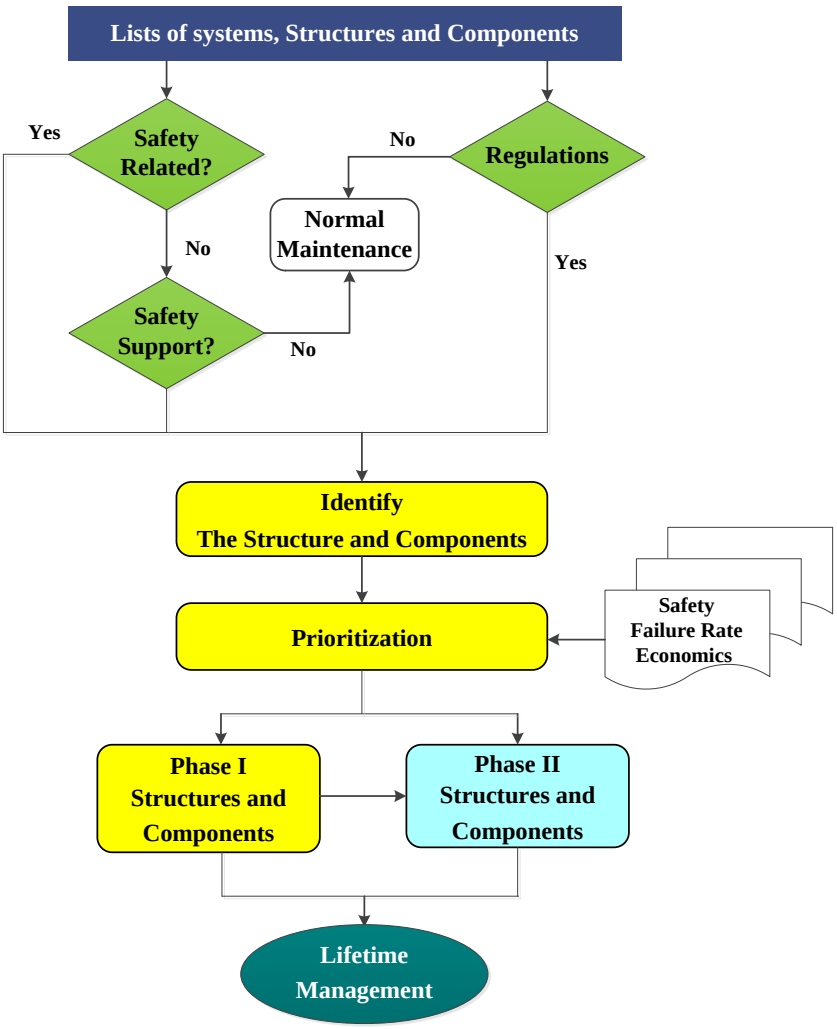


FIG.3. Structure & Component Screening Approach (Korea Nuclear Power Plant Approach) [6]

Details on how to scope the SSCs to be screened for an ageing management assessment in accordance with US 10CFR54.5 is well described in NEI 95-10 [7]. Its appendix includes a sample listing of potential information sources for the screening and a table listing typical structure, component and commodity groupings.

Screened SSCs for aging management should be prioritized for resource usage optimization. The prioritization exercise consists in evaluating the importance of each SSC to plant life based on field data and information such as safety significance, plant life limiting significance and economics, failure rate, , management strategies, etc.

If probabilistic tools are available, a combination of probabilistic safety insights and deterministic approaches is recommended. This method is called a risk informed screening. It allows a prioritization of the ageing management activities on all SSCs on the basis of their safety significance. Essentially, the method yields a SSC grading based on the SSC impact on the core damage frequency. When conducting the probabilistic safety analysis, attention should be paid to common cause degradation across SSCs with similar vulnerabilities.

Following the SCC screening step, a design and operation data collection activity is conducted for all screened components. The selected components are then classified in terms of importance to safety and production. When the SSC selected is a complex component, such as a steam generator, sub structures and sub-components such as for example the steam generator nozzles, its heads, walls, tube bundles, tube supports, tube sheets, divider plate and other internal structures are systematically examined and classified. Then the ageing related degradation mechanisms of all subcomponents are identified. If the failure mechanisms are unknown, failure data are sent to R&D for interpretation and research. As this step is completed, all the elements are now available to undertake a prognosis.

It is to be noted that in the past, active and less critical, more easily replaceable components (such as rotating machinery, valves, etc.) have received more attention with regard to diagnostics and prognostics and their ageing behaviour research is more advanced than that of large pressure vessels, concrete structures, reactor structures and piping. The factors that impact the growth of a crack are reasonably well understood, but the dynamics of incipient crack growth are less well known, as are also the impact of stressors, the sensitivity of the diagnostic tools which impacts the capability of determining the correct damage phase or level from crack growth rate measurements. 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 states, to crack initiation 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 today in ISI programmes, and these methods can also be used for online monitoring. Data from ISI are then imported into the AMP to obtain SSC prognostics. Accurate SSC life prognosis is never easy. It is a function of the degradation state of the SSC and of the assumed development of the stressors in time, but also of their past history, if at all known. They must contain the thresholds of incipient component degradation for early warning and be capable of reading the marks of the material type and degradation state. The optimal selection of the sensor locations and the parameters to be monitored must be decided on the basis of a risk-informed SSC analysis.

If the maintenance programme is optimized, the flat stabilization period of what is known as the SSC ageing bathtub curve (see figure 4) can be extended. The reliability or probability of failure of a SSC in this stabilization period depends on the 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 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 less than optimized, the wear out period begins earlier and its slope is steeper. In well managed maintenance programmes at nuclear power plants, the flat portion of the bathtub curve is low and may even show declining probabilities of failure. With time however it will eventually increase and enter the wear-out period. The starting point of the wear out period and its rate of increase with time should be determined to plan corrective action, insert it into the maintenance schedule and allocate budget for it.

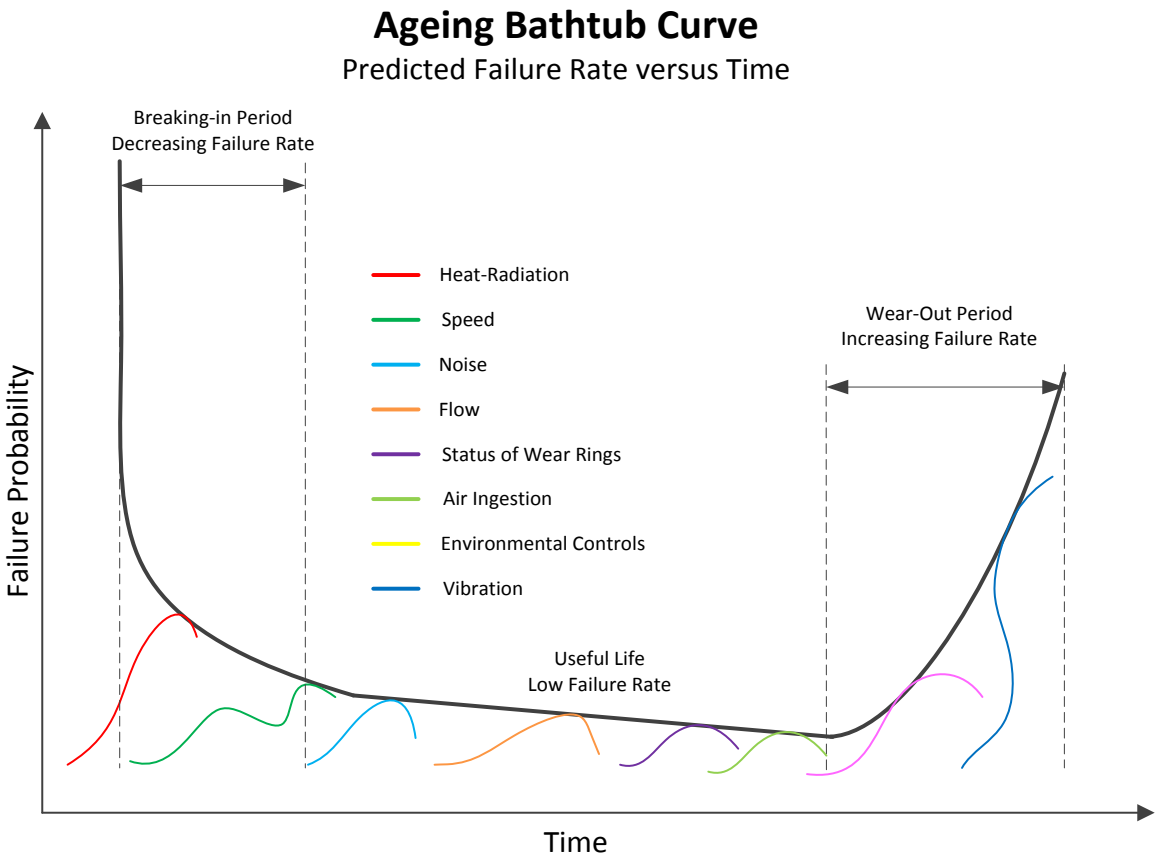


FIG.4. Ageing Bathtub Curve

The probability of failure normally follows the well-known "bathtub curve" pattern as shown in Figure 4. At the beginning of life, when the new SSC enters service it experiences the so-called "run-in" period, the likelihood of failure during that time is relatively high, but the failure curve decreases as the SSC enters a stabilization time with a lower and somewhat flat failure rate.

The SSC ageing data such as 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 acting independently. In order to determine the beginning of the SSC wear-out period, one method is to determine the dominant stressors and their degradation mechanisms. This is usually done by consulting PLiM data bases containing the degradation mechanisms and related R&D data. At this point the PLiM data bases will generate a credible prognosis for the SSCs, based on their bathtub curves and plan mitigating actions to extend the SSC lives. Below are listed three of the most used prognostics methods in PLiM analysis:

— Reliability data analysis uses historical time-to-failure data to model time to failure predictions. This method does not rely on actual plant operating data and may be less pertinent to the actual plant conditions.
— Stressor-based methods that take specific operating conditions into account. They allow trending of the stressor values and the development of a distribution: stressor value versus predicted failure times. This correlation is then used in the life prognostics.
— Effects-based methods compute a degradation or damage index and correlate this quantity with the probability of failure. The RUL is typically estimated based on the time for the damage index to exceed some predefined threshold
— The integration of conventional ageing management programmes and PLiM insights provides the capability to select the ad hoc monitoring programmes and support the design and installation of automatic on-line monitoring instrumentation and surveillance devices. These programmes are crucial to indicate the crossing of thresholds and hence allow the optimization of preventive maintenance activities. With the implementation of a PLiM programme, the safety aspects of

ageing management are also better served, since integration provides a better awareness of safety margins and a better management of pressure boundary integrity throughout the plant life cycle. If PLiM techniques are not implemented, degradation in safety-critical components is usually only detected at periodic inspection time (every 10 years), and potentially only after significant degradation has already occurred. This has potential implications in terms of safety if a serious failure occurs and in terms of economics, possibly resulting in higher maintenance costs and even extending an outage when a defect is found.

In addition, to physical ageing, the equipment qualification (EQ) programmes are reviewed in terms of their effectiveness. Up to date methods and practices are proposed to preserve and upgrade, where necessary, equipment qualification. Recommendations are 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 more online monitoring and prognostics for passive components as they become more capable, automated, digitized and with higher levels of remote management.

2.2.5. Components Life Evaluation

In a plant life management assessment, a preliminary screening is done from the top down, beginning with plant life limiting components requiring a life assessment and a customized ageing management programme, including R&D where necessary. These components are usually mainly passive or structural in nature. The screening can then continue at the system level, where active mostly safety and economically critical components are considered and AMPs developed. Two parallel phases of SSC prioritization are normally undertaken:

— A first phase covering components critical to safety. In this instance regulatory requirements regarding ageing management must be considered as part of the regulatory framework.
— A second phase for SSCs with a failure rate threshold linked to economic requirements.

By screening systems, structures, and components, owner/operators can reduce their ageing management analysis effort to a manageable scope. Depending on the grade and criticality of the component or group of components, ageing studies can be optimized to help develop the most appropriate and cost effective ageing management programme. An example can be the following three assessment methods for three criticality dependent categories of SSC:

— Life assessment (LA): this technique is typically applied to the 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 exploitation uneconomical. Predicting the remaining life of these critical systems requires prognostics tools and methods. This begins with a rigorous assessment of all plausible ageing related degradation mechanisms. The methodology entails a detailed review of plant data in order to establish current condition and to evaluate ageing degradation at a sub-component level. While life assessment activities are focused on passive systems and structures, active systems (such as pumps and valves) are, and can continue to be, well managed and routinely monitored, diagnosed, analysed and upgraded with less onerous practices. However, it must be said that passive systems and structures such as large pressure vessels and reactor structures are not easily or economically upgraded when degradation is detected during for example an in-service inspection (ISI). Therefore, managing passive systems and structures degradation is likely to be the key to planning a good life management programme not only for the plant initial design life but also for its economic viability in longer term operation.
— Systematic assessment of maintenance (SAM): This form of assessment is less onerous than LA and can be profitably applied to critical systems with emphasis on active components that are generally designed to be replaceable, when needed, in order to preserve the system functions. SAM makes use of Failure Mode & Effect Analysis (FMEA). Input information for the analysis is extracted from internal and external feedback and R&D findings. Results are processed utilizing streamlined Reliability Centred Maintenance (RCM) techniques, as modified for nuclear plant applications
— Condition assessment (CA): this is a technical procedure, typically performed for small active categories of components. The CA process assigns components to commodity groups such as

instrument categories, valve types, small rotating or reciprocating machinery, etc. where they are evaluated by group. The methodology entails a general review of plant data in order to establish current condition and to evaluate ageing degradation. Recommendations out of a condition assessment should normally provide the technical basis for on-going ageing management of the component category or commodity and may identify the need for further or more rigorous assessment.

PLiM provides the opportunity to optimally deploy online condition-based monitoring and including the use of remote monitoring and guide maintenance planning and practices which altogether allow for a better prognosis capability and more reliable predictions on the remaining useful life (RUL) of SSCs. This proactive approach, known as prognostic-based health management (PHM) or integrated system health management (ISHM) requires online tools to detect degradation early in the SSC lifecycle (diagnostics), and estimate RUL for the degraded component or system (prognostics). One such methodology developed by KHNP in Korea for life time assessments is shown in Figure 5 through which a remaining life assessment or prognosis is conducted in 6 steps [6].

Age-related degradation mechanisms (ARDMs) are identified and the aging effects of the ARDMs are reviewed by evaluating the design and material data of structures and components with the operational and environmental conditions of the systems. Qualitative life evaluations such as aging management reviews (AMR) as well as quantitative life evaluation as time limited aging analyses (TLAA) are carried out during aging assessments. The life of a system is defined as the shortest life of a structure or component within the system boundary.

In conducting an aging evaluation of the screened structures and components (SCs), aging mechanisms of the sub-components should be recognized. Guidelines to this effect can be found in:

— the methodology outlined in ASME Section III Appendix W that describes aging mechanism in operating nuclear plants,
— American Concrete Institutes codes for structures,
— IAEA TECDOCs.

In order to ensure the aging mechanisms selected for the sub-components are correct, the following information should be acquired:

— A good understanding of aging mechanisms in nuclear power plants
— Susceptible operating environments that can cause aging degradation.
— Functions, materials, design, fabrications, and operating conditions of the sub-components
— Review of operating experience feedback and technical documents and reports issued of degradation experience worldwide

Aging mechanisms are a function of the design characteristics and operating environments and can be extracted from the design documentation. Material characteristics and properties are normally taken from the certified material test report (CMTR) and from the final safety analysis report (FSAR). Operating evironments can be working fluid, temperature, pressure, humidity, water chemistry, radioactive fluence, etc. For LTO applications, time limited aging assessments (TLAAs) are required. They 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. The guiding principle is always that all system intended functions should not be compromised by aging effects during plant operation. In a qualitative aging management review (AMR) all the plant data records since first reactor criticality are assembled and systematically assessed for function and integrity for the period under review. TLAAs, on the other hand, are quantitative assessments. Usually they apply to relatively small components but they are intense activities in terms of engineering effort and scientific investigations.

Generic aging lessons learned (GALL) report of NUREG-1801 have been published to facilitate ageing assessments and are normally taken into account in PLiM analysis, particularly in LTO submissions where most recommendations resulting from detailed life assessments of a PLiM project are applied to the plant aging management programs (AMPs). These recommendations essentially aim at preserving the SSC intended functions during the extended operation period (LTO). They may be implemented in various ways such as overall plant refurbishments, component replacements, design modifications, addition of performance monitoring functions, upgradings, etc. Before implementation,

all AMPs and derived maintenance activities should be compared with those of other similar national and off-shore nuclear power plants, if available.

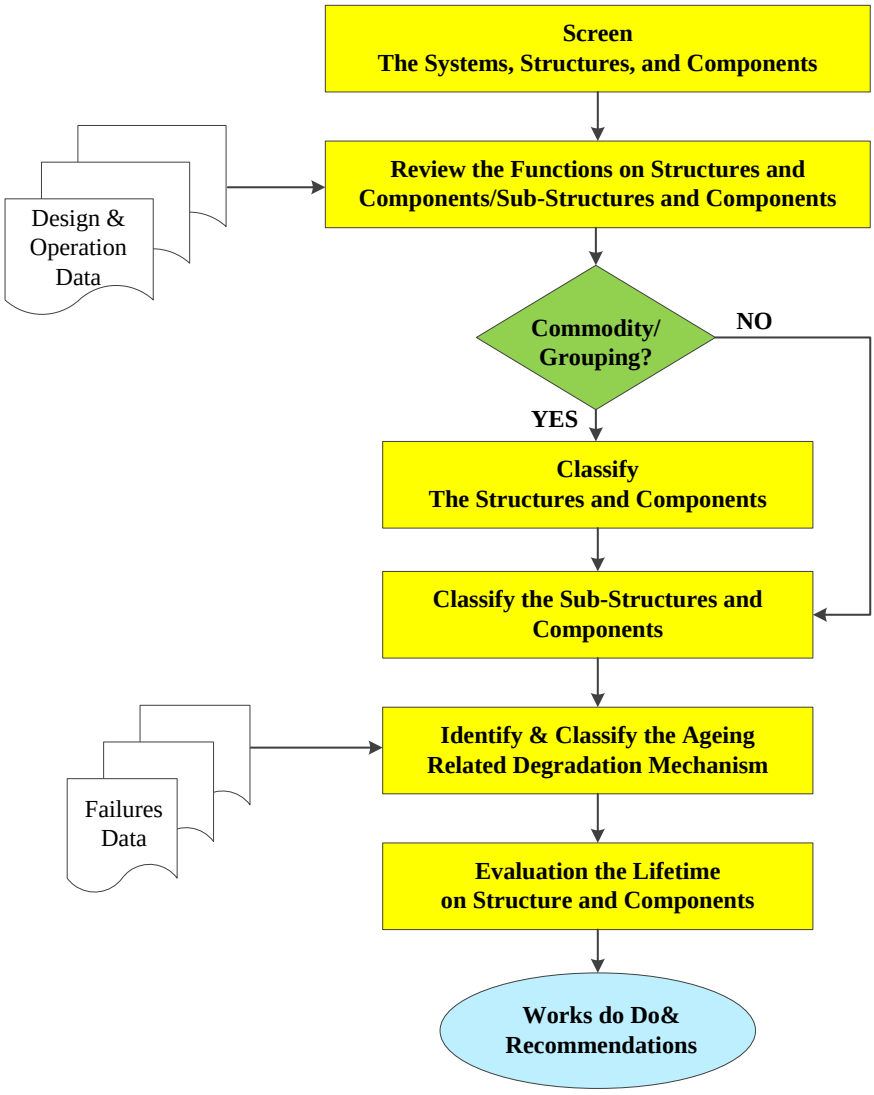


FIG.5. Remaining Life Evaluation

2.3. LESSONS LEARNED FROM THE FUKUSHIMA DAI-ICHI NUCLEAR POWER PLANT ACCIDENT ON AGEING MANAGEMENT

2.3.1. Background

Concern has been expressed that ageing of the Fukushima Dai-ichi nuclear power plant facilities might have played a role on the accident or might have increased the effects of the accident. In the "Report of the Japanese Government to the IAEA Ministerial Conference on Nuclear Safety" of June 2011, 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 method and results of the ageing technological assessment conducted in the past, a theoretical assessment of safety-critical equipment of the Fukushima-Dai-ichi units was conservatively conducted to at least preliminarily understand whether there could have been an impact of the seismic ground motion on the safety-related equipment as a result of the earthquake. The assessment of the

equipment ageing was conservatively considered for an extended period, from the time of commissioning to a full 60-year operation. [8]

2.3.2. Results of the assessment

Assessment of each ageing event

In the assessment of the upper grid plate damage, low-cycle fatigue cracking and irradiation-induced stress corrosion cracking have been examined, since they were the two ageing-related phenomena that may have contributed to the damage to the upper grid plate resulting from the Fukushima-Daiichi accident. The assessment was carried out using the seismic ground motion of the earthquake and the results of the ageing technological assessment conducted in the past. It was found that the impact of the seismic ground motion was sufficiently small to preclude any contribution from these ageing phenomena to the damage resulting from the accident. The impact on the safety margin to the acceptable limit was small and the acceptable limit itself was originally selected with some additional margin to the actual expected damage limit. Therefore, the comprehensive review showed that it was difficult to consider any impact of the ageing phenomena on the upper grid plate following the accident.

Similarly, in the assessment on the neutron irradiation degradation of the RPVs using the seismic ground motion of this earthquake and the results of the ageing technological assessment conducted in the past, the impact on the safety margin to the acceptable limit was found to be small and in turn the acceptable limit was established with some safety margin to the actual damage limit. Therefore, it was again difficult to conclude that ageing has had an impact on the accident such as the loss of safety functions following the seismic ground motion induced by this earthquake.

Seismic impact assessment on major safety-related facilities

During the seismic safety assessment, the surface corrosion of the basic volute of the reactor shut-down cooling system pump was considered 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 this earthquake and the results of the ageing technological assessment conducted in the past. It was found that the impact on the safety margin to the acceptable limit was small.

Regarding the low-cycle fatigue cracking of the shroud support, the low-cycle fatigue cracking of the main steam system piping, and the low-cycle fatigue cracking of the nuclear reactor recirculation piping, the assessment was also conducted using the seismic ground motion of this earthquake and the results of the ageing technological assessment conducted in the past. The results showed that it was difficult to attribute to these ageing phenomena a contribution to the damage from the accident such as the loss of function of such components, because the impact on the safety margin to the acceptable limit was sufficiently small and the acceptable limit itself had still some additional margin to the actual damage limit.

2.3.3. Summary of the impact of ageing on the Fukushima 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. Also, it is difficult to consider that the ageing events contributed to the occurrence and aggravation of the Fukushima Dai-ichi NPP accident during the period between the accident occurrence and the time when the accident developed beyond the NPP design basis. However, at this point, it is difficult to conduct an on-site inspection of the equipment to confirm if ageing had an impact on the safety equipment. This is only a theoretical assessment of the impact of ageing by analysis, utilizing the technological assessment on ageing conducted in the past. Therefore, as new findings become available in the future, through on-site confirmation, etc., an additional review will be conducted on the impact of ageing.

3. PROACTIVE AGEING MANAGEMENT IN OPERATING NPPS

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

— Recognizing common weaknesses in conventional ageing management of NPP, especially the inability to trace root causes, to develop lessons-learned and to prioritize the use of resources to mitigate the effects of ageing in SSCs important to safety and reliability;
— Application of a proactive and systematic ageing management process and by pursuing a continuing improvement of the ageing management programme.

It has been common practice in the past 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 and corrective measures are taken only in reaction to component failure(s). The corrective or reactive management of ageing has led to surprises with safety consequences, as exemplified in the steam generator tube rupture events of Palo Verde NPP and the failure of the control rod driving mechanism (CRDM) penetration nozzle at the Davis-Besse NPP.

Since then, preventive maintenance methods have evolved to become more proactive and ageing management programmes are now better suited to prevent such mishaps with the use of plant condition monitoring systems capable of anticipating failures. In addition, proactive SSC ageing management programmes take into account not just current and anticipated events, but the whole life cycle of a plant and includes systematic reviews of world-wide events and operating experiences at each stage of the lifetime. It starts, whenever possible, at the design stage and continues through the component manufacture/fabrication phase, through plant construction, system installation, commissioning, operations (including long term operation and extended shutdowns) and extends all the way to the decommissioning period.

A proactive ageing management programme inevitably extends to both the safety and economic aspects. It extends also to all associated external activities such as engineering, procurement, fabrication, transportation, preservation in storage until installation etc. It involves continuous learning and a continuous search for improvement of methods and processes from the experience accumulated during operation of the plant itself and from that of other nuclear power plants. This will help avoid recurrent issues and prevent the plant from lagging behind in operation techniques and in achieving plant performance targets.

Proactive ageing management implies accountability. The operating organization takes on the responsibility to address ageing concerns specific to the plant by issuing operating instructions and procedures. Similarly it assumes the obligation of identifying generic ageing issues known nationally and internationally from operations feedback, from vendor recommendations, from manufacturer operating manuals and from plant designers. Each known ageing issue should be addressed with an adequate ageing management programme. [2] A good example of a proactive ageing management culture in a plant would be its underpinning practice of familiarizing its staff with proactive ageing management principles at all levels and enforcing a protocol such as the Plan-Do-Check-Act (PDCA) developed by the IAEA. Figure 6 shows the relationship between proactive aging management and advanced maintenance technology.

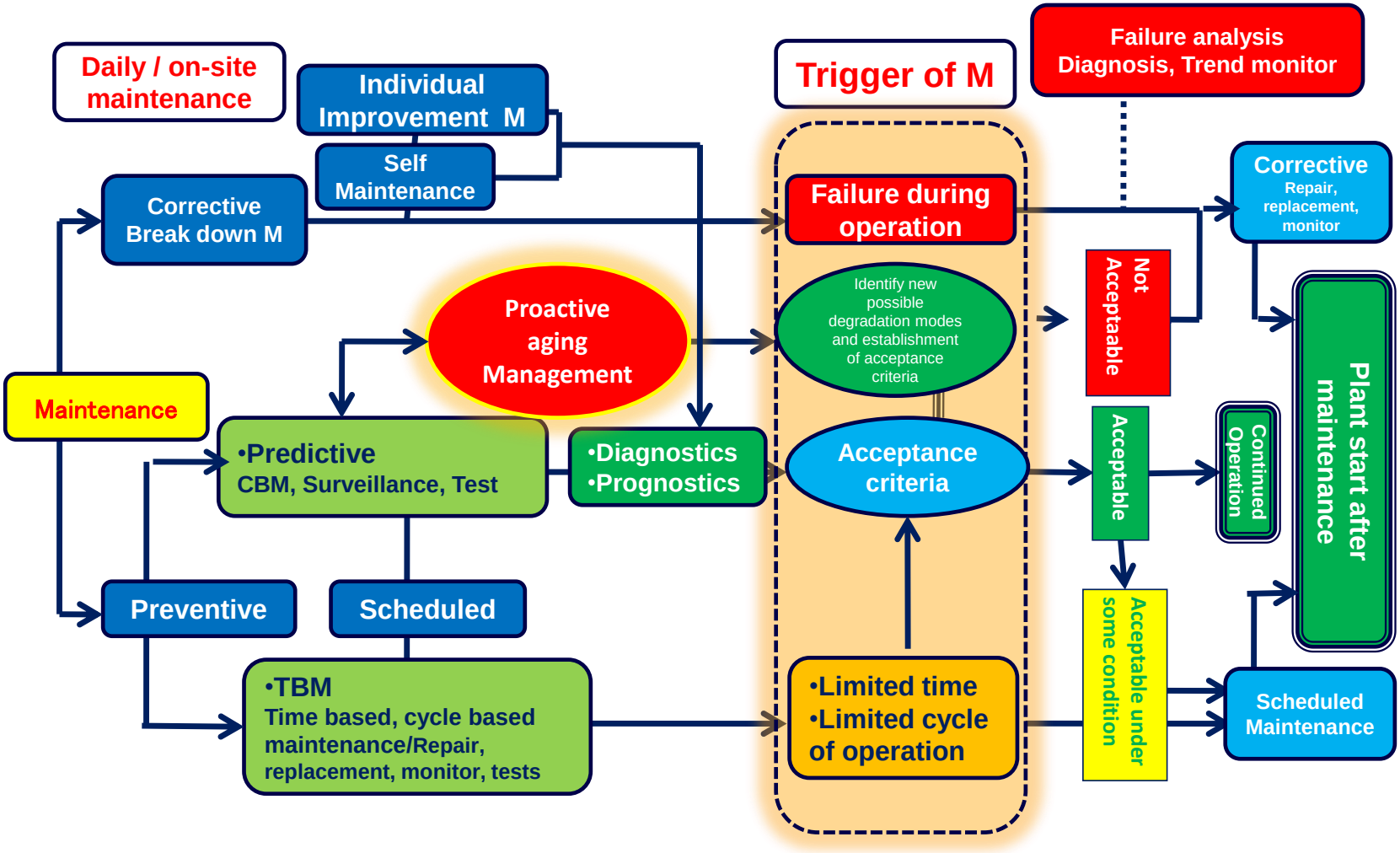


FIG.6. Proactive aging management and advanced maintenance technology

3.1. AGEING CONSIDERATIONS IN DESIGN

Conceptual design

During the conceptual design phase of any nuclear power technology, ageing considerations of all SSCs must be taken into account in the high level design target or design objective documents, in the safety design guides and in the design specifications of the SSCs. In the design requirement documents and in the basic assumptions and input data to the safety, thermo-hydraulic and stress analysis, as well as in the safety analysis report (SAR) reference should be made to the appropriate articles on ageing in the applicable codes and standards.

Particular attention should be paid to assure appropriate technical margins. The margins consist of design margin, operational margin and safety margin to prevent component failure. The operator should ensure and the regulator should verify that margins adequately include the ageing effects on SSCs, particularly on those important to safety. Margins should include the operational margin considering effects of wear mechanisms and other age related degradation to ensure their safety and commercial functions are ensured for the duration of the SSC mission time. If the mission time extends to the plant service life, as is the case for passive non-replaceable components, the design should also involve analysis of the duration of the entire plant service life. This of course includes all normal and transient operating conditions, testing, maintenance, and outages. Anticipated design basis accidents following all postulated initiating events are to be considered part of the operating life and hence part of the design margin calculations.

The appropriate design document should address ageing management and include as a minimum the following topics [1]:

— A recommended strategy for ageing management and prerequisites for its implementation;
— A list of all safety significant SSCs of the plant that could be affected by ageing;
— Proposals for appropriate materials monitoring and sampling programmes where ageing may affect the capability of critical SSAs to perform their function throughout the lifetime of the plant;
— Appropriate consideration of operating experience with respect to ageing;
— Ageing management recommendations for critical SSCs (concrete structures, mechanical & electrical, I&C components, cables, etc.) and measures to monitor and mitigate their degradation;
— Environmental 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 severe accident mitigating features, including portable SSCs may have been introduced in the context of increasing robustness and/or flexibility of the plant in mitigating the consequences of severe accidents. They may or may not be aligned and energized and some of these may be in storage during normal operation. Nevertheless they should all be included in the ageing management programme and maintenance plan.

Environmental qualification (EQ) of SSCs

The EQ process is illustrated in figure 7. The design vendor and installation documentation normally provide important information needed before an EQ programme can be established for specific component and system applications. An ageing management programme must include activities prescribed in the Design documentation and in the EQ programme and should particularly be concerned with preserving the EQ of SSCs requiring it, throughout their service period.

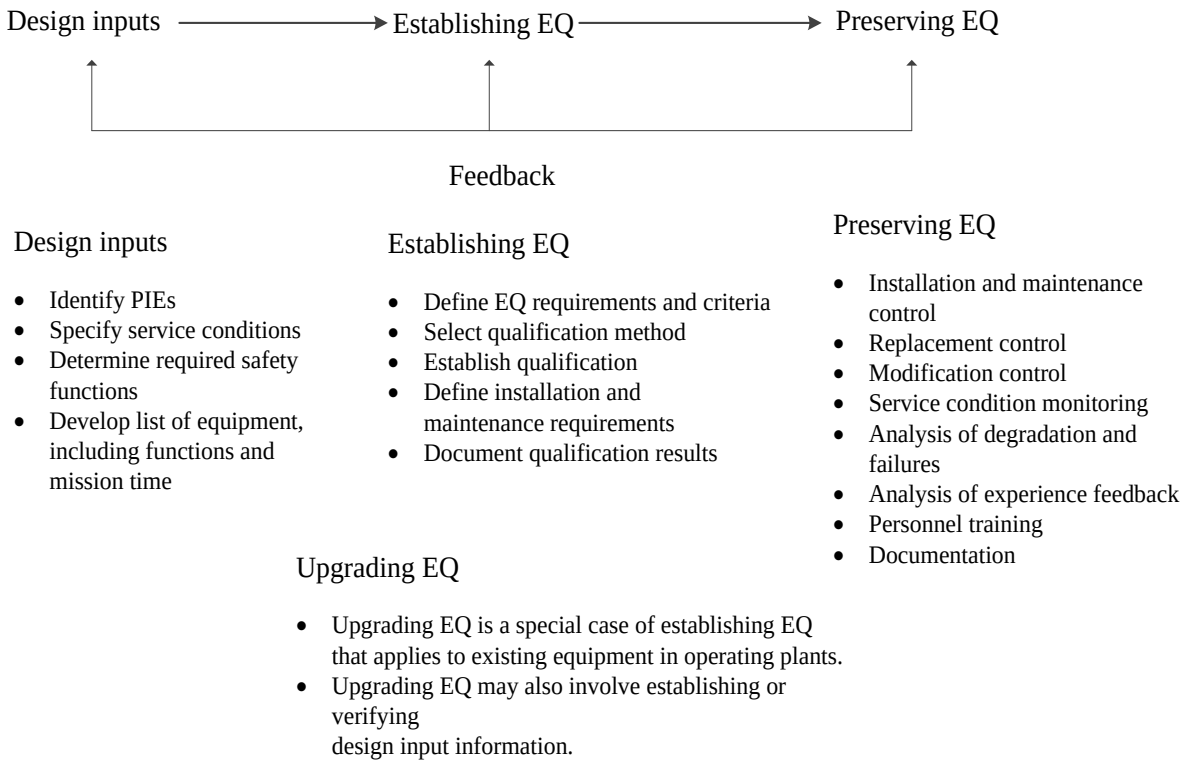


FIG.7. The equipment qualification process; PIE: postulated initiating event.

Contractual and bid-evaluation phases

In the pre-contractual phase in the bid specification the plant owner or operator must clearly communicate ageing criteria and requirements to the plant supplier, design organizations and manufacturers in order to receive a design that complies with all criteria and requirements for reliability, performance and even long term operation (if necessary).

During the bid evaluations, ageing considerations and equipment qualification (EQ) are important issues and complementary to each other in designing SSCs. A proper EQ programme provides effective means for ageing management. Ageing should be one of the items being evaluated for each of the technologies under scrutiny.

During the contractual phase, following the purchase of a generic design, the detail design phase and the plant deployment are negotiated and agreed upon. It is important that all ageing considerations and assumptions made by the designer on how the plant should be operated, including operating margins are crucial and should be handed over by the technology vendor to the operator.

While ageing considerations are given at the stage of the conceptual design, they must not be underestimated or worse be compromised during the site specific detail design phase. Detail design is normally governed by a division of responsibilities in the contract.

The procurement phase

Ageing consideration 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 EQ tests are in compliance with their own EQ programmes. In this respect, they should guard against non-conformances, interfacing discontinuities or material incompatibilities.

3.2. AGEING CONSIDERATION DURING FABRICATION & CONSTRUCTION.

During fabrication and construction installation non-conformances or quasi non-conformances may occur. Consequently, residual stresses may be induced that Design did not assume. For example sometimes during construction a welder may apply a certain force to bring two parts (e.g. two pipe spools) closer together. When the force is released residual stresses are induced in the weld, thus increasing the risk of cracks. Although force application in pipe welds is not forbidden, it is important that the plant operator be informed of the practice so that appropriate measures may be taken to appropriately manage 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 multi pass welding, inducing unacceptable or accumulated heat input, geometrical discontinuities and other relevant details that are not properly taken into account or modelled in stress analysis; unaccounted for damage during transportation etc. [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]

The owner/operator should ensure as a minimum that:

— Current knowledge on the factors affecting ageing is available to SSC manufacturers and is properly taken into account in the fabrication and construction of SSCs;
— Current knowledge about relevant degradation mechanisms and mitigation measures are taken into account during the manufacturing and installation of SSCs;
— The appropriate amount of reference data (baseline) is collected at the factory and documented;
— Surveillance specimens for ageing monitoring programmes are correctly specified, made available and installed in accordance with design specifications.

3.3. AGEING CONSIDERATIONS DURING COMMISSIONING

During commissioning, the functional capability of SSCs is checked and baseline data are collected to record conditions at start up for later reference in ageing management during operation. Also 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 of 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 premature ageing cases 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 clean up may be undertaken during commissioning, residual aggressive agents may remain in the system and cause early corrosion and cracks. Similarly, if abnormal temperatures occur during commissioning, values and length of time that high temperature manifest themselves should be notified and discussed with Engineering so that corrective or protective action may be taken to prevent systematic re-occurrence and accelerated ageing in adjacent SSC’s such as concrete structures, cables etc. Transients and extreme conditions during hydrostatic testing and hot conditioning tests should be recorded and engineering 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 commercial operations.

All parameters that can influence ageing degradation should be identified and recorded during commissioning. The information should be made available to operations in order that it could 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.

The regulatory body, as part of its review and inspection programme, should ensure that critical service conditions (as used in equipment qualification) are in compliance with the design requirements and analysis assumptions.

3.4. AGEING MANAGEMENT IN OPERATION

Natural Nuclear Plant Operator will be focusing on SSCs of safety significance. However, they also will include in their AM programmes also those SSCs that may not directly be of safety significance, but whose failure would have a significant financial impact, for example those leading to long down time and/or substantial repair or replacement cost penalties.

A systematic approach to managing ageing such as PLiM techniques if applied during plant operation can help optimize mitigation of the effects of ageing. Application of PLiM since the early stages of plant operation will automatically apply a proactive approach to ageing management because PLiM is based on understanding degradation mechanisms, on prevention and mitigation, rather than supposedly minimizing expenses with a reactive approach by responding only to SSC failures, which instead will inevitably result in increased cost and performance reductions.

A proactive approach to ageing management during operation will also look at continuing education in ageing management and PLiM techniques, at increasing the awareness, familiarization, the motivation and sense of ownership of operations, maintenance and engineering support staff. In order to ensure the correct application of proactive ageing management principles, appropriate procedures, tools, materials and qualified staff should be at hand, appropriate storage of spare parts and consumables should be available to minimize degradation and control shelf life. The use of multidisciplinary teams should be foreseen for dealing with complex ageing management issues. The use of feedback of operating experience should be required, and the use of on-line and ad hoc monitoring and of NDE should be encouraged.

Figure 8 shows the typical elements necessary in the development of an ageing management strategy during operation of an NPP.

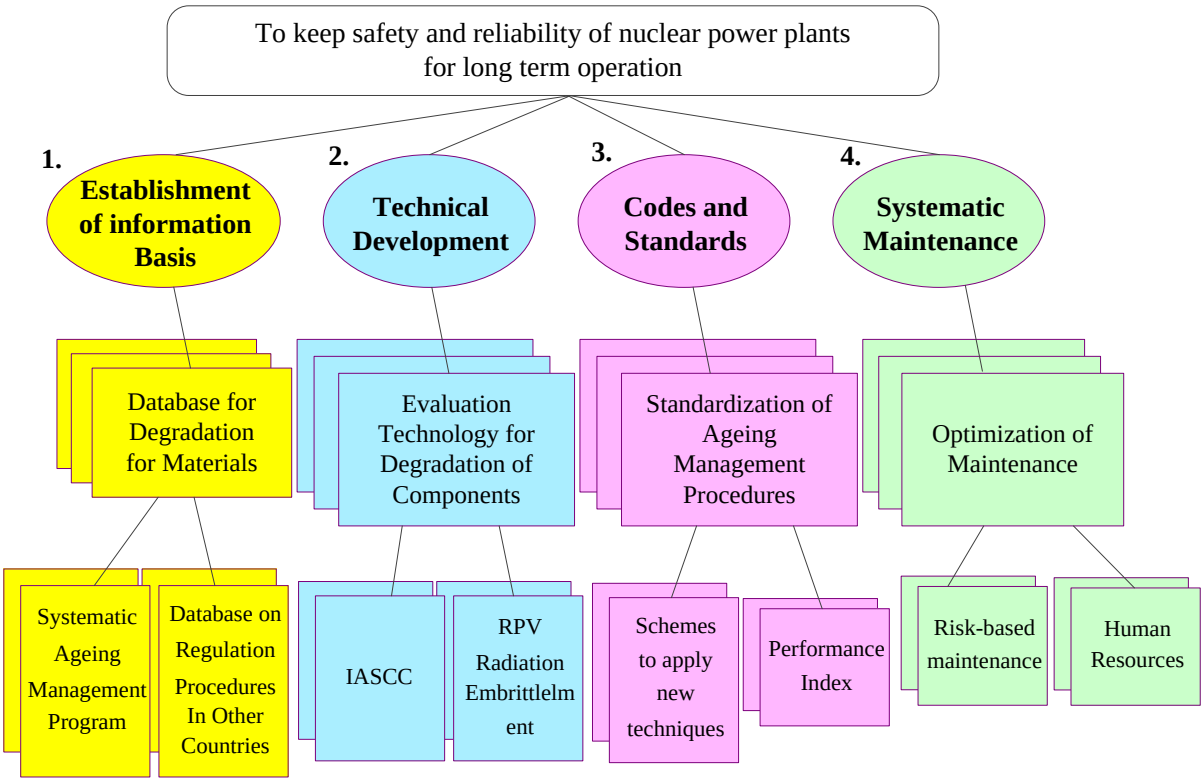


FIG. 8. Strategy Map for Ageing Management

This diagram was first presented by the Atomic Energy Society of Japan in 2009 and updated yearly until 2011. It constitutes a strategy map for ageing management in that it points at the four major principles that should govern the implementation of an ageing management programme in NPPs:

- establishment of a material degradation database,
- development of a technical understanding and an evaluation of component degradation in the plant, the application of codes and standards and
- setting up of a systematic approach to maintenance which leads to maintenance optimization
- use of tools such as risk-based guidance in the maintenance planning process.

Maintenance during operation should always be mindful of ageing and should be conducted pro-actively. The best way to achieve this is to integrate maintenance programmes with ageing management activities and if the plant has a Plant Life Management programme, both maintenance and ageing management should be integrated into the PLiM programme. Figure 9 shows the interconnectivity of maintenance and ageing management. Figure 9 presented OECD-NEA project on Stress Corrosion Cracking and Cable ageing management in Tokyo, Japan dated May 25-26, 2010.

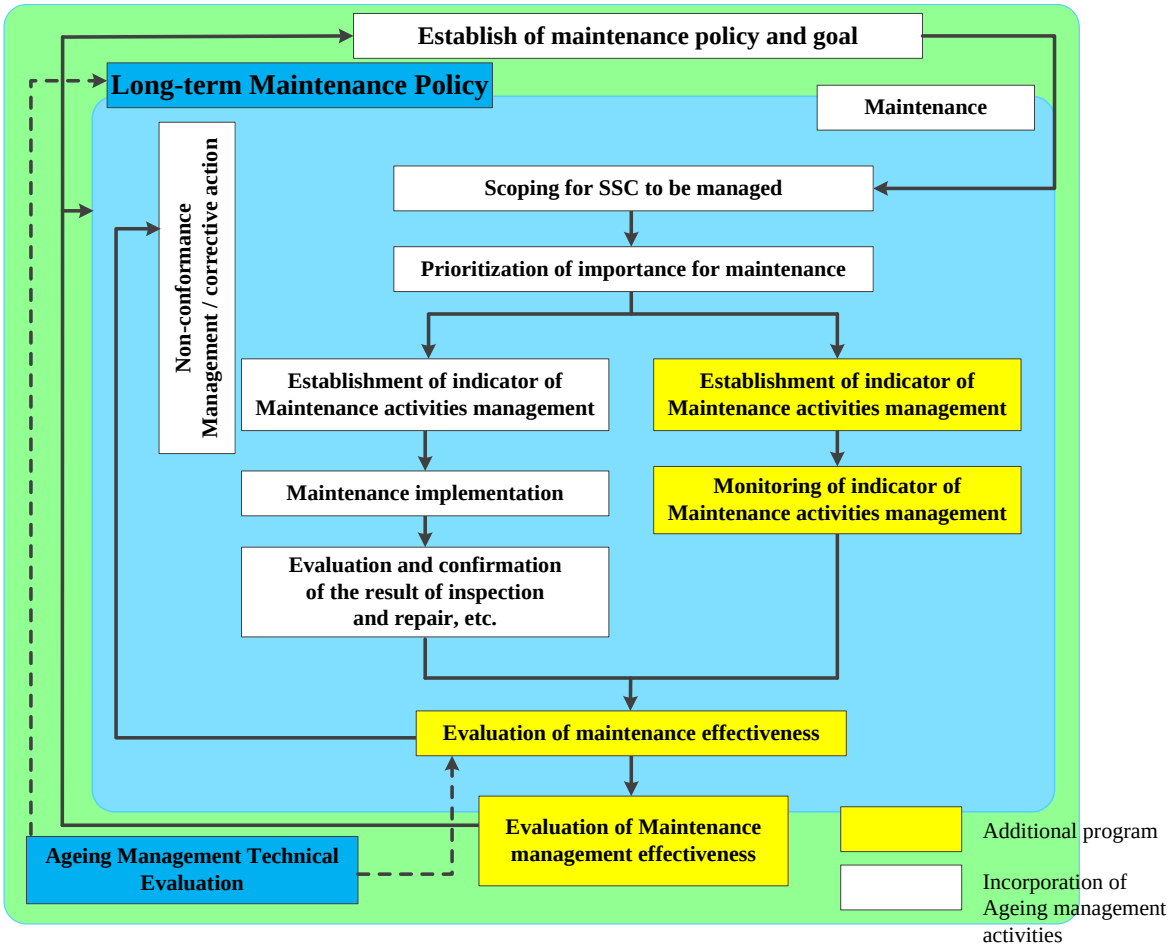


FIG. 9. Integration of a routine maintenance programme and ageing management activities.

The white boxes indicate the interfacing activities of the large maintenance envelope and ageing management. This interfacing gives rise to additional programmes indicated by the yellow boxes which become part of the overall maintenance policies and goals and of the routine maintenance activity chart.

The IAEA developed PLAN, DO, CHECK and ACT (PDCA) cycle to help implement a systematic ageing management process. Fig. 10 represents the key elements of an ageing management programme utilizing the PDCA cycle for the reactor pressure vessel in a PWR. Similarly, Figure 10-11 represent the key elements of an ageing management programme utilizing a systematic ageing management process for the reactor pressure vessel internals in a pressurized light water cooled reactor.

The box in the middle is the first preparation step. It entails the determination of all the ageing mechanisms of a structure or component. If one of the mechanisms is new or not known, then R&D is required to develop an understanding. For the well-known mechanisms only recognizing them and listing them or extracting them from a PLiM data base will be sufficient. Other information on the SSCs themselves such as material properties, fabrication methods, stressors, operation experience, R&D findings is also crucial to complement the understanding of the degradation mechanisms.

The peripheral boxes in the PDCA cycle describe the four steps of a systematic implementation programme: The PLAN box at the top is the first of these steps and involves the planning of activities aiming at optimizing ageing management of a SSC and by so doing at improving its effectiveness. The second peripheral box to the right is the DO box that contains a collection of all activities aimed at minimizing the expected degradation during the operational phase of the SSC including the careful operation of the SSC within its technical specification and according to the applicable procedures, the use of the best chemistry and environmental control possible and the careful recording of transients and historical events. The box at the bottom is the CHECK box that requires periodic inspection, monitoring and condition assessment of the functional capabilities of critical SSCs.

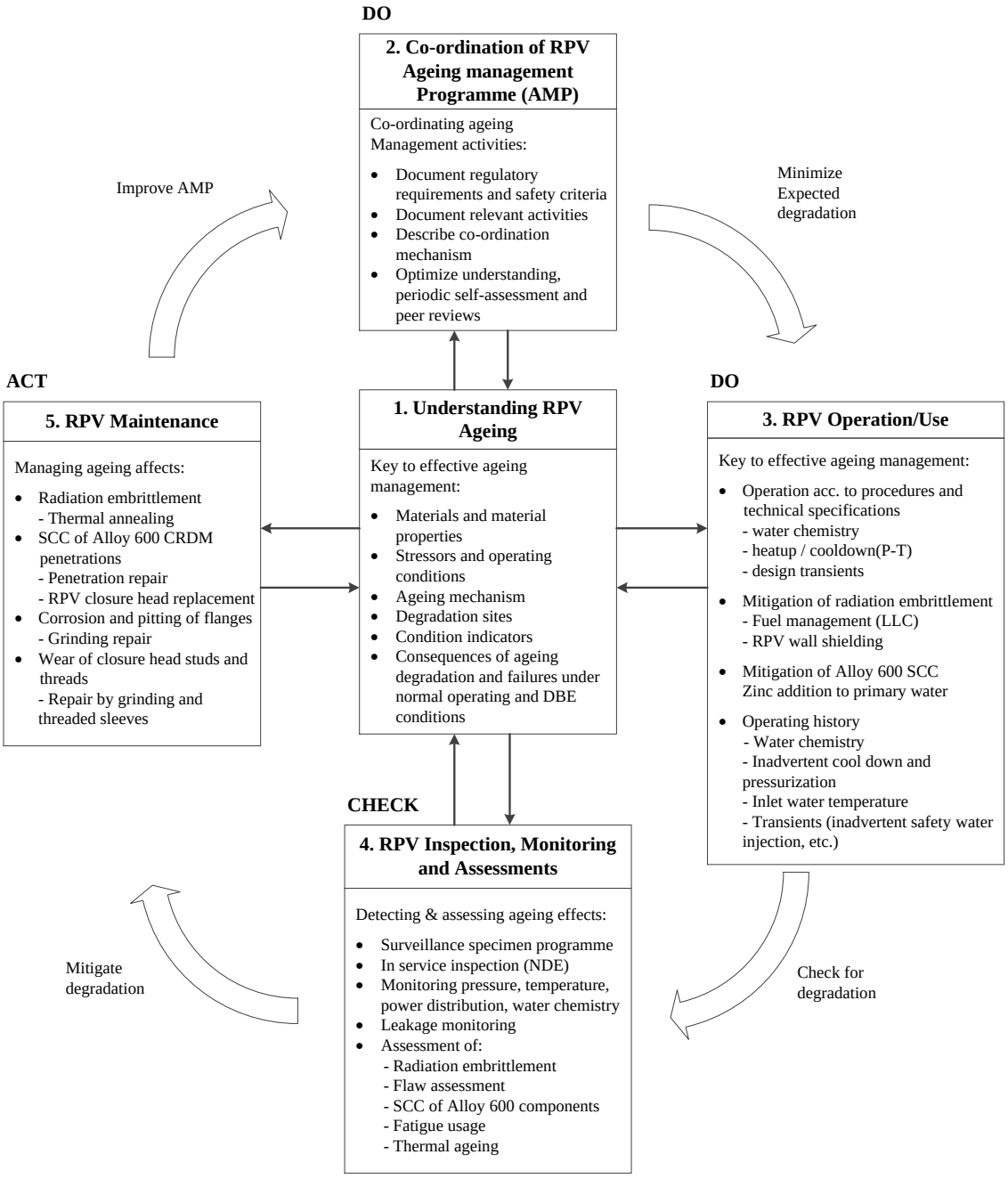


FIG.10. Key elements of a PWR pressure vessel ageing management programme

This can be done by calibrating instruments, testing or inspecting SSCs to detect ageing symptoms such as leaks, vibration and by following up by means of record keeping and continuing updates. When conditions grant it a fitness for service investigation may be required. The fourth peripheral box is the ACT box that deals with mitigating degradation. This may involve preventive and corrective maintenance action, spare parts inventory management and record keeping of all maintenance history.

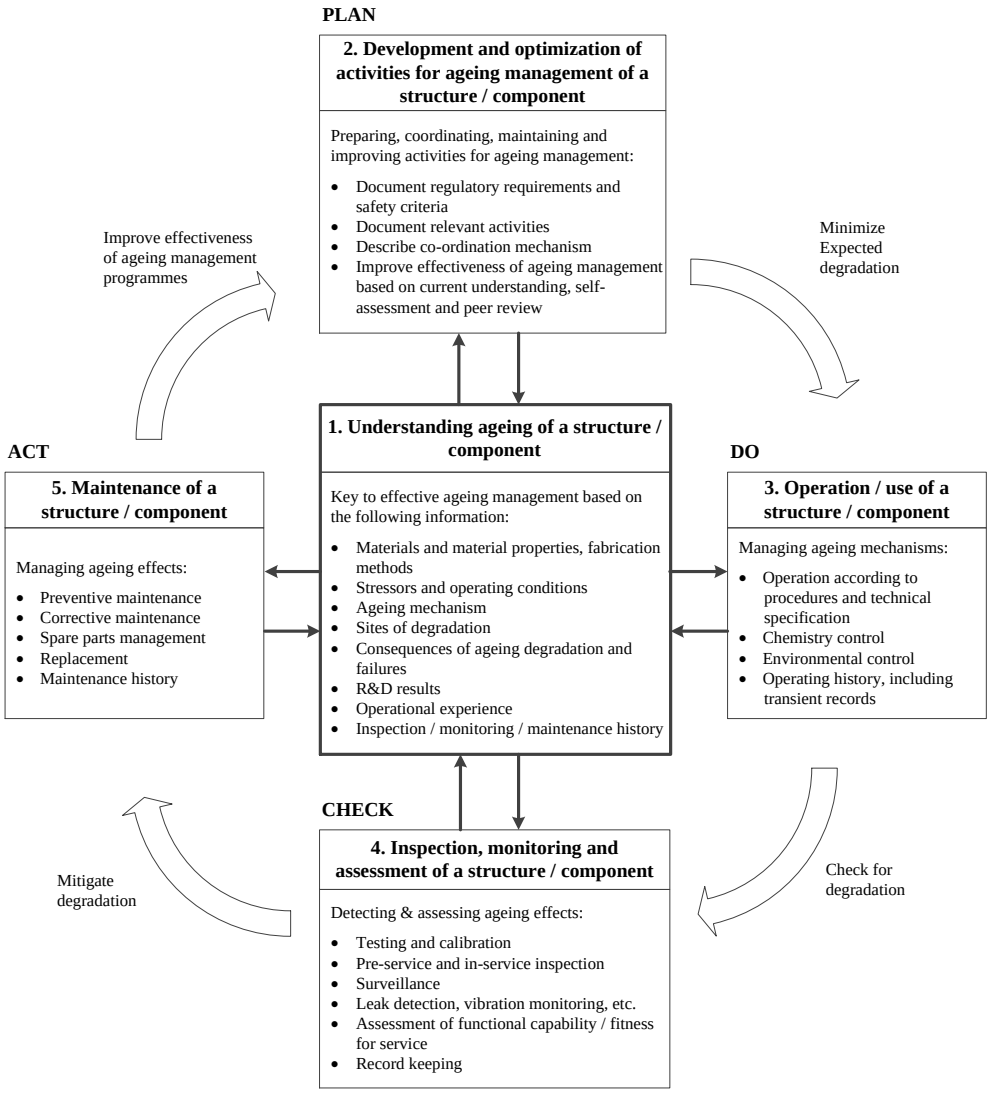


FIG.11. Systematic approach to managing ageing of a reactor pressure vessel internal structure for a PWR

At the end of the cycle, the experience gained going systematically through the four peripheral boxes closes the peripheral loop with the corrective action planning box at the top, and an analysis of the loop with a view to gain feedback and further improve the ageing management programme effectiveness.

The operating organization should also identify and address commonly encountered issues in ageing management programmes of NPPs:

— Premature ageing of SSCs caused by more severe service conditions than assumed by design or by errors in design, fabrication, installation, commissioning, operation and maintenance,
— Scattered or out of control ageing by lack of coordination or unforeseen ageing phenomena or by an exclusively reactive ageing management programme
— Lack of a technology watch function causing lack of awareness of operating experience feedback and research results;
— Unexpected stress loading to structures or components due to external events (e.g. earthquakes, floods, etc.).

The cooperation and the beneficial relationship among all stakeholders in SSC ageing management in a typical nuclear power plant in Japan is shown as an example in the diagram of figure 12 obtained from the Japanese regulator.

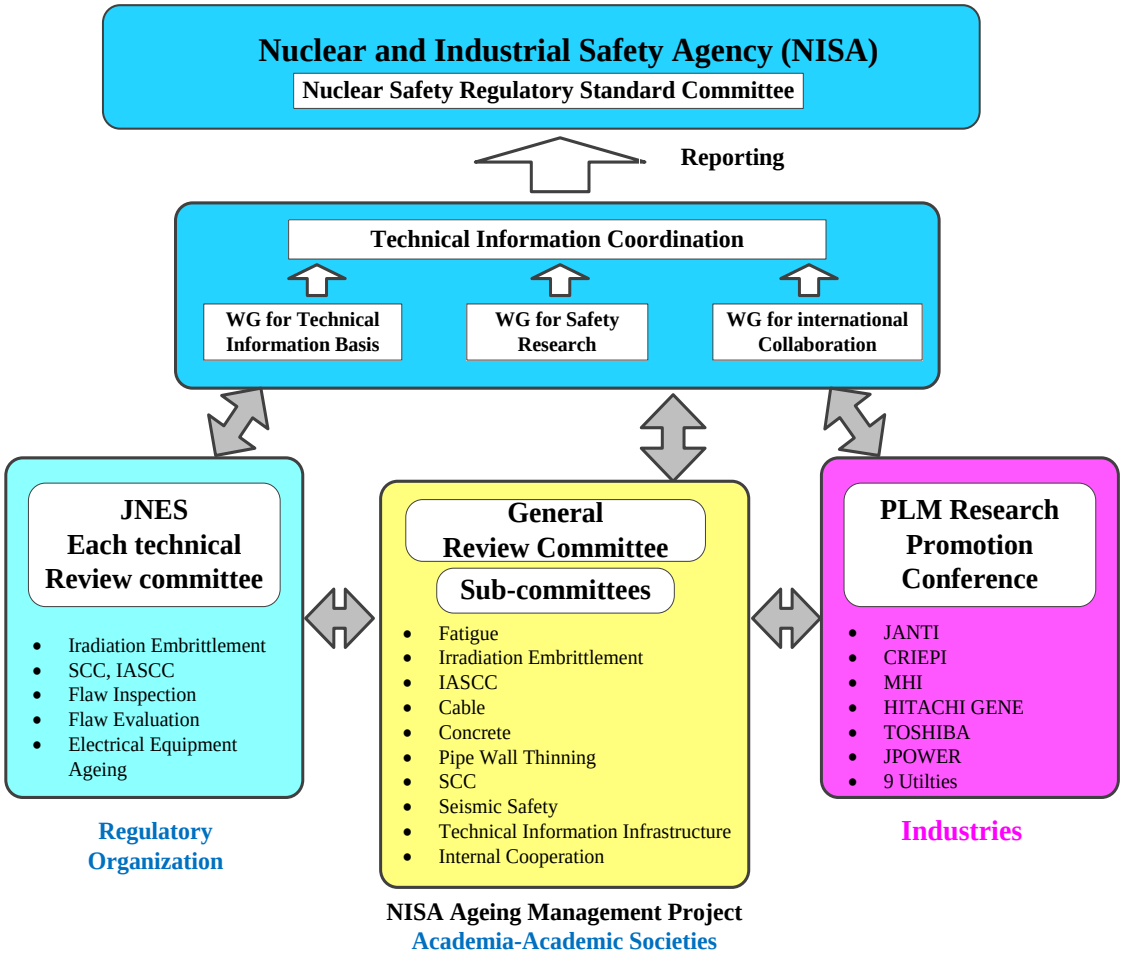


FIG.12. Intervention and relationship among all stakeholders of a NPP

As shown in the top box of figure 12, a multi-representative Nuclear Safety Regulatory Standard Committee assumes a coordination role for all stakeholders involved in ageing management of a typical nuclear power plant in Japan. It interfaces with the regulatory organization (JNES), academia and academic societies in Japan and with the industrial groups providing services to the plant ageing management and to the maintenance groups. Coordination is executed through working groups operating in each of the three larger stakeholder groups.

3.4.1. Refurbishments, Modernizations and Power Uprating

In the event of reactor power uprating, large plant refurbishments, important modifications or large equipment replacements, the operating organization should identify and justify possible associated changes in process conditions (e.g. flow pattern, velocity, vibration) that could cause accelerated or premature ageing or even failure of some components. Also if a component must be replaced because of obsolescence or for other reasons with similar but not identical specifications without a thorough assessment being made or without the operator being informed of such changes, material compatibility may arise, even though form and functions remain the same. In addition, different specs may produce accelerated ageing issues from which safety consequences may derive.

A good example of this is the case of the guide tube pins in the reactor 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 CANDU type reactors, typically made of ASME SA106 carbon steel. Thinning occurred in the early feeders even fabricated within the same manufacturing tolerances

allowed for the chromium content of this material. Operating experience and R&D evidence determined that feeder pipe bends in the reactor face region are much more susceptible to stress corrosion if fabricated with chromium content within the lower percentile (< 0.2%) of the Chromium band, than feeder pipes fabricated with the Chromium content within the upper percentile (> 0.2 %) of the band. This evidence induced designers in refurbishment and new build projects to more closely specify the chromium content for feeder pipes.

If a new ageing mechanism is discovered (e.g. through feedback of operating experience or research), operating organizations should prepare contingency plans or modify their maintenance plans to deal with the condition.

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 ageing degradation can often be reduced by optimizing operating parameters. In the case of primary system components, proper recording of transients allows the fatigue usage factor to be improved if actual operating transients and stressors are used instead of those assumed by the original design organization. Also optimizing the fuel pattern in the core to reduce fluence on the vessel in PWRs and reduce the embrittlement rate is a good example of optimization of the operating parameters to mitigate the effects of ageing. [9]

3.4.2. Long Term Operation:

PSR is the standard approach for those Member States that do not follow the license renewal application process. The plan of approach shown in Figure 13 illustrates a systematic phased approach developed by the Swedish nuclear safety authority during the preparation phase of the Swedish operators' applications for longer term operation beyond the plant lifetime as postulated in the original design. This methodology has proven successful in the preparation of the periodic safety review submission for longer term operation.

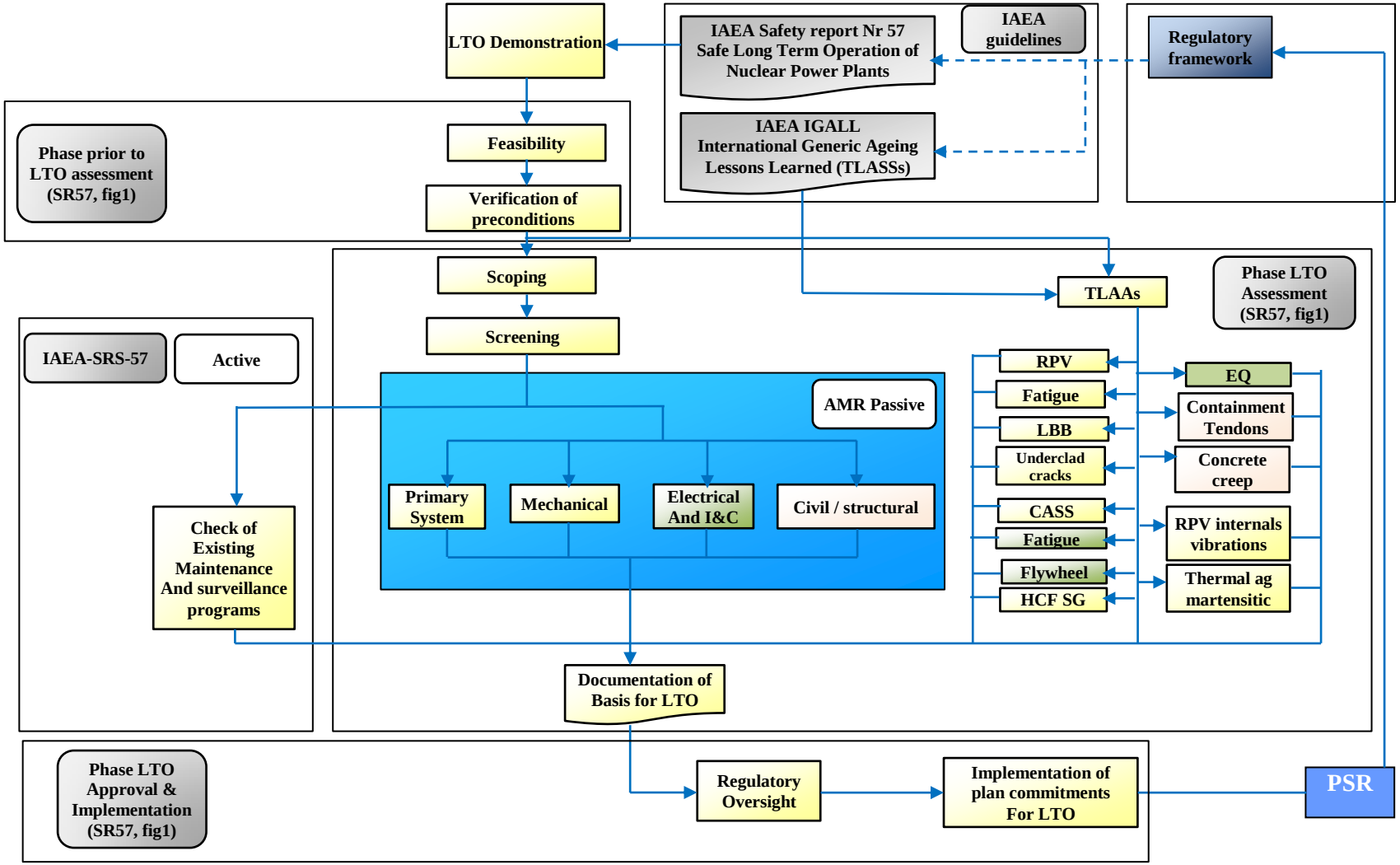


FIG.13. Plan of approach for LTO projects (Swedish Nuclear Safety Authority)

The first phase at the top is the preparatory phase prior to the application for Longer Term Operation. The second block represents the LTO assessment phase and the third block is the implementation of the actual application commitments. A good ageing management programme conducted from day one in the service life of a plant will guarantee that the plant will be in its best possible condition and will require the minimum possible changes at the time of its PSR submission for longer term operation which is usually more demanding than regular PSRs. In line with IAEA's safety reports series No. 57 [10], the diagram shows that Sweden's nuclear regulatory framework integrates the contents of IAEA's recommendations and equally relies on the periodic safety review process.

An equivalent model within the context of a licensing framework that requires a safety review submission for an operating period beyond the period first postulated in the original design is exemplified in figure 14 provided by the Qinshan phase 1 nuclear power development in Qinshan province, China.

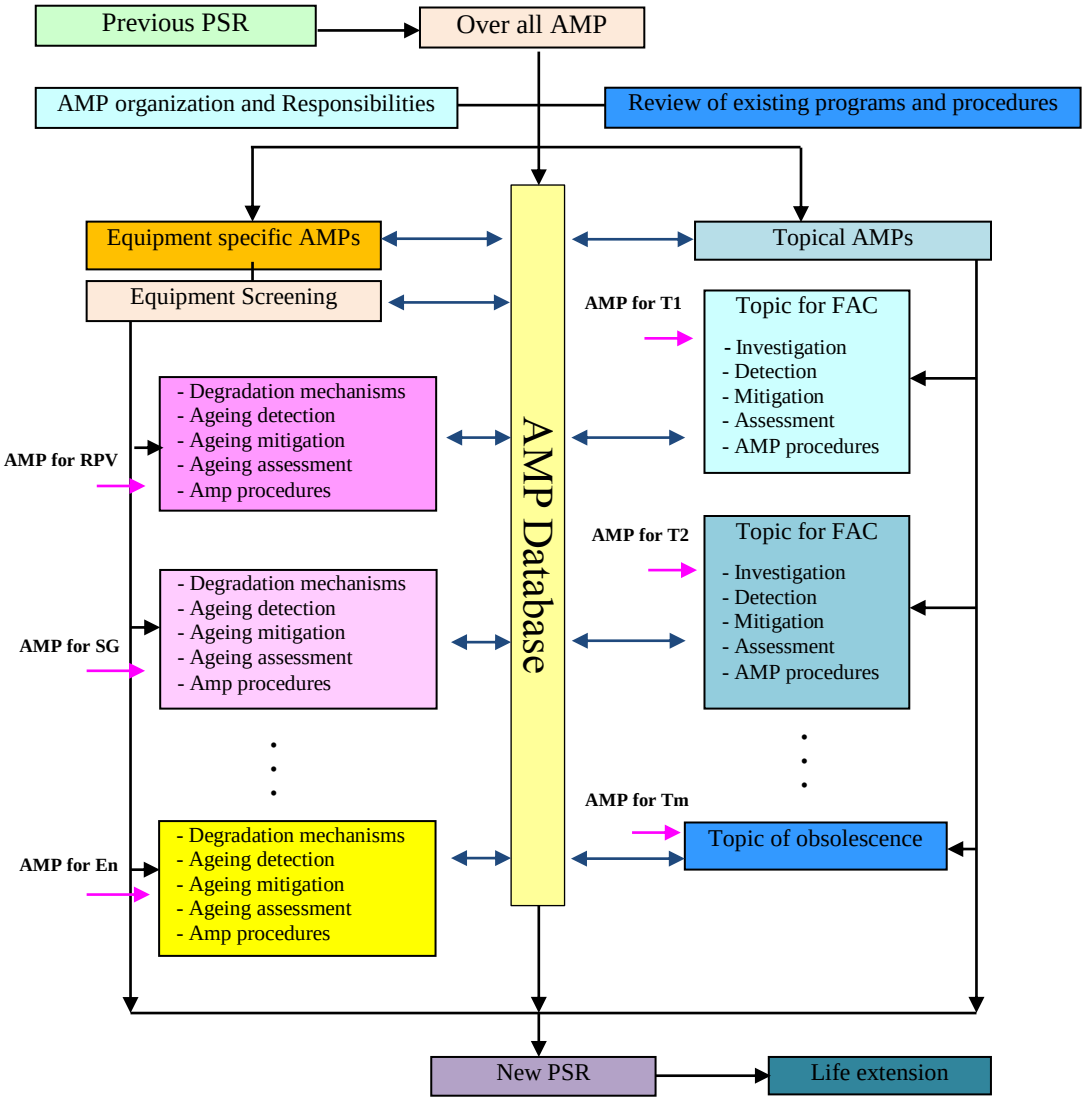


FIG. 14. Systematic Ageing Management Model for Qinshan-1, China

When a design upgrade is deemed necessary to enhance safety or performance in parallel to an LTO permit submission, the process to assess the project is shown in figure 15. 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 judgment supported by probabilistic insights (if available). The

list of possible changes is consolidated and a short list of LTO issues are set aside to be included in the Periodic Safety Review report and submission to the regulatory body for an LTO permit. A cost benefit analysis is conducted on all consolidated issues to orient the proposed improvements in a feasibility context which includes an estimate of the safety benefits. All inputs are then collected and an integrated assessment is conducted. The process leads eventually to an improvement plan which includes both the short-listed LTO drivers and the work plan containing the approved design upgrades.

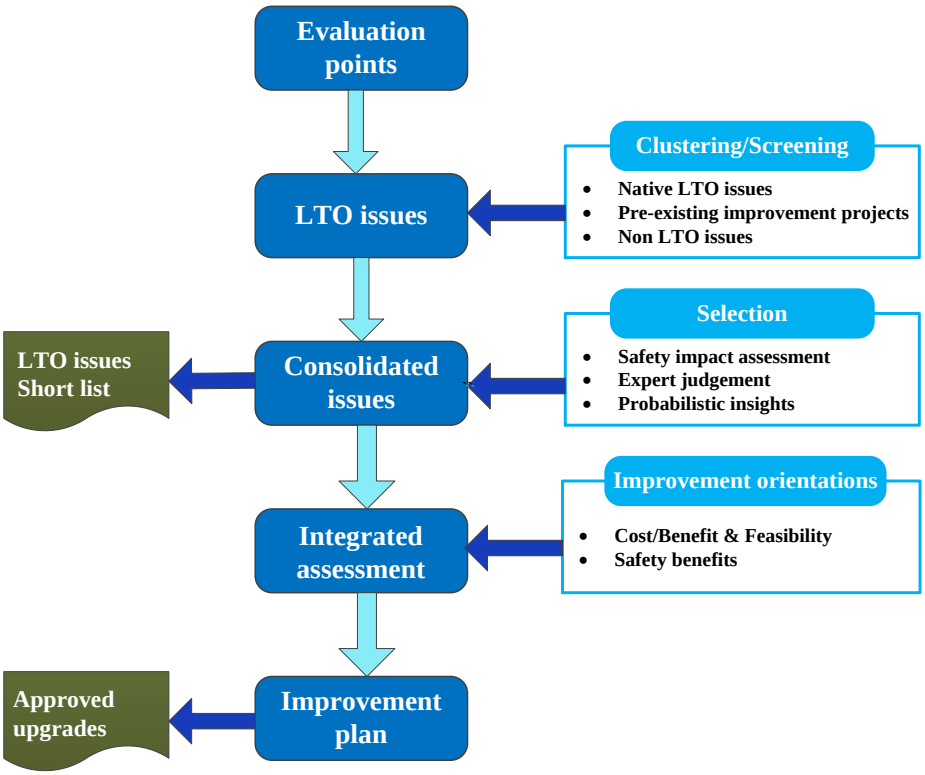


*FIG.*15. Design Upgrade Assessment Process

3.5. AGEING MANAGEMENT PRE-REQUISITES FOR DECOMMISSIONING

Appropriate arrangements, including ageing management, upgrades, replacements, etc. should be made before decommissioning to ensure that SSCs and tools required to implement decommissioning (e.g. containment system, cooling equipment, lifting equipment and condition monitoring equipment) remain available and functional.

Once a licensee decides to decommission a plant, an application is usually required. As part of this application an environmental assessment is included to show how the environmental footprint left by the decommissioned plant will be acceptable and meet the environmental laws of the land for industrial operations and the international commitments of the country in which the decommissioning is taking place.

In terms of safety of the decommissioning operations, the licensee is bound to maintain operational and preserve the functionality of all those facilities and tools essential to carry out safely all decommissioning activities and ensure the protection of the decommissioning staff, of the public and of the environment.

Decommissioning activities can occur in various stages imply various stages that can be increasingly financially onerous. To endure these obligations will be met, regulators usually require, a licensee to submit a decommissioning plan and provide financial guarantees to execute the decommissioning plan of the facility, as part and as early as the initial construction and operating license submission(s).

Initially the licensee may put the plant in a safe storage (safstor) condition until sufficient radioactive decay has occurred in the pant and residual radioactivity is more manageable. All physical

preparation steps can be completed during this period before decontamination operations are started. Generally, most parts of a nuclear power plant do not become radioactive and those that do, generate only low level radiation. Decontamination however may involve the construction of large decon systems and of a storage tank facility to contain the by-products of decontamination activities. Usually low and medium level radiation safe storage and safe burial space may also be required. However most of the metal recovered from dismantling SSCs can be recycled. Even before dismantling occurs however, defueling and removal of the reactor coolant and moderator are necessary steps. Facilities and equipment necessary to implement these steps (including drums and storage vessels and their support systems) must be maintained in a functional operating state throughout the life cycle of the plant including decommissioning. Approximately 99 % of the radioactivity in a NPP about to be decommissioned resides in the fuel and is removed from the plant when the reactor and the spent fuel bays are defueled. Certainly surface contamination will be present in the systems containing the primary coolant and 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. For example, Electricité de France (EdF) has decided to adopt partial dismantling with postponement of final demolition for 50 years.

In countries without decommissioning experience, the main issue is the lack of clearly defined regulatory requirements for the decommissioning of nuclear facilities and sufficiently funds may not have been set aside. Another issue may be a weak waste management policy including transportations and routing, long term storage and final disposal facilities. Fuel disposal and the management of spent fuel in general may also be an unresolved issue and should be addressed before decommissioning of any NPP in those countries may start. [11] [12] [13]

3.6. EFFECTIVE AGEING MANAGEMENT PROGRAMMES

The attributes of an effective ageing management programme are summarized in Table 1 below taken from reference [1].

— The first item in an effective ageing programme is normally the establishment of the steps necessary to survey and understand the extent of ageing issues in the plant covering both the known phenomena and the newly discovered issues through inspection, monitoring and testing. This includes the identification of all relevant SSC ageing mechanisms, the deployment of appropriate condition indicators, an understanding of the stressors, the mapping of the degradation sites and the establishment of limiting conditions. Once the facts are known, then predictive models of SSC ageing can be developed.
— The second item is the adoption of barriers against ageing to slow down the consequences to a manageable if not optimized operation. Selection of the barriers and of devices or administrative processes to monitor their effectiveness and the establishment of a limiting condition or of an acceptable criterion on the progress and speed of degradation. Barriers usually include maintenance interventions, specifically controlled operating conditions or enhancements in the treatment of the system flow or of the external environment.
— The third item is the establishment of a programme of inspection and monitoring aimed at detecting new hot spot before they become incipient degradation issues leading to possible failure.
— The fourth item consists in the deployment of monitoring and trending devices to collect data and facilitate SSC prognosis
— The fifth item involves mitigation of degraded SSC approaching their acceptance limit. This usually involves maintenance activities including repair & replacements.
— The sixth element found in good ageing management programmes deals with acceptance criteria, including the update of such criteria for repaired and replaced items wherever necessary.
— The seventh item is a process or norms for the enactment and approval of corrective actions when SSCs fail the acceptance criteria
— The eight element to look for in a good ageing management programme is a set of instructions on how to deal with operating experience feedback and feedback from research and development, a process for analysing and implementing operating experience and R&D feedback, and the recording of the evidence to that effect
— The ninth element to look for is quality management of the ageing management programme itself

and its implementation Administrative controls that document the implementation of the ageing and of its implementation. This should include a verification process to ensure the adequacy of all preventive and mitigating actions and the appropriateness of the planning, the implementation and the record keeping of all AM activities.

TABLE 1 EFFECTIVE AGEING MANAGEMENT PROGRAMMES

1. Scope of the ageing management programme based on understanding ageing	• Structures (including structural elements) and components subject to ageing management • Understanding on 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

Administrative control of the development and execution of an ageing management programme should not be neglected. It is an important activity aimed at maintaining quality control over the process.

Figure 16 shows a four step procedural control of the ageing management programme at the PAKS nuclear power plant in Hungary. Administrative control extends to the scoping, development, execution and feedback steps.

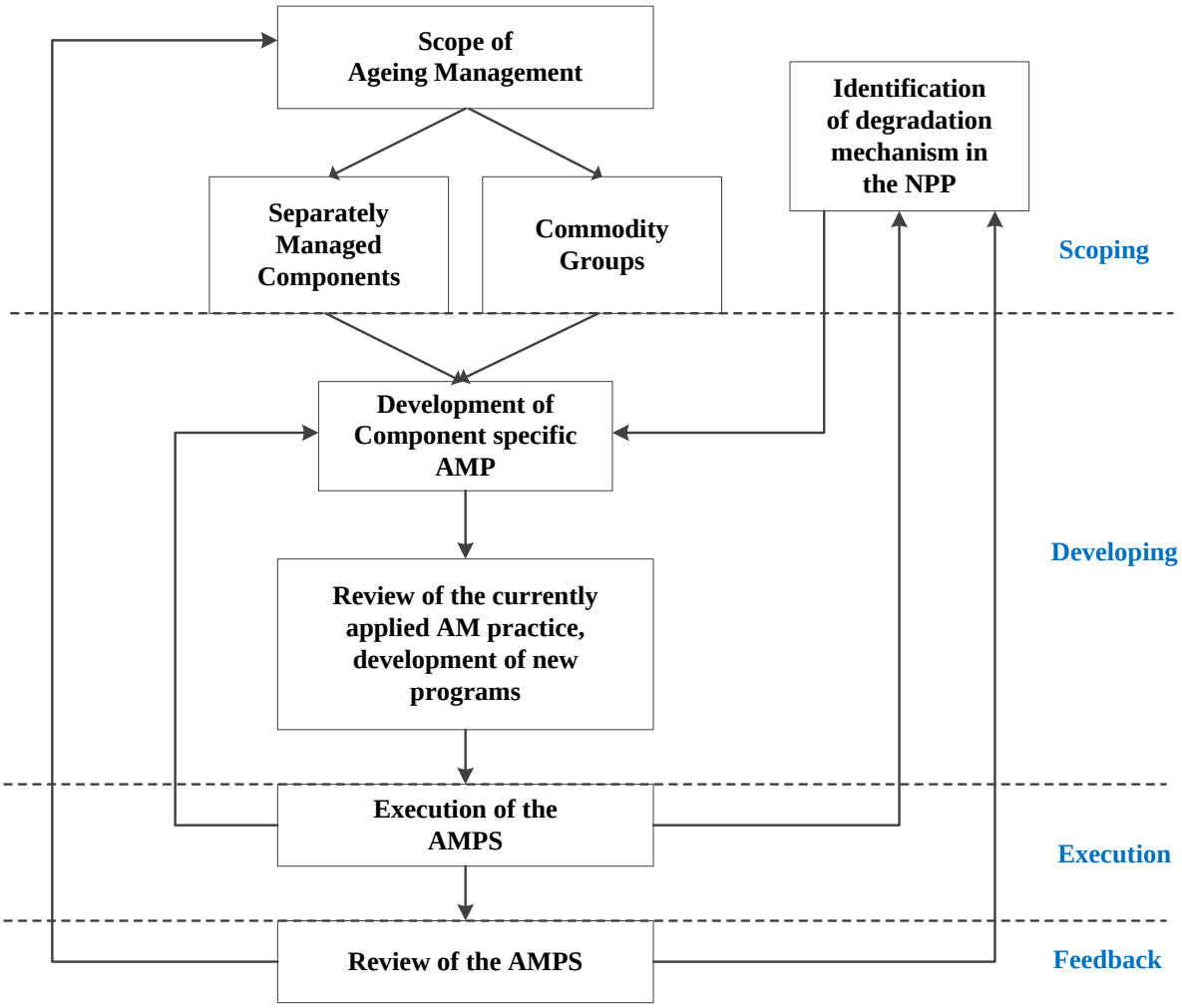


FIG.16. Administrative control of the AMP at the NPP in PAKS, Hungary

4. AGEING MECHANISMS

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

— Those affecting the internal microstructure or chemical composition of the material and thereby changing its intrinsic properties (thermal ageing, creep, irradiation damage, etc.).

— Those imposing macro geometrical damage on the component either through metal loss (corrosion, wear) or through cracking or distortion (stress-corrosion, deformation, cracking).

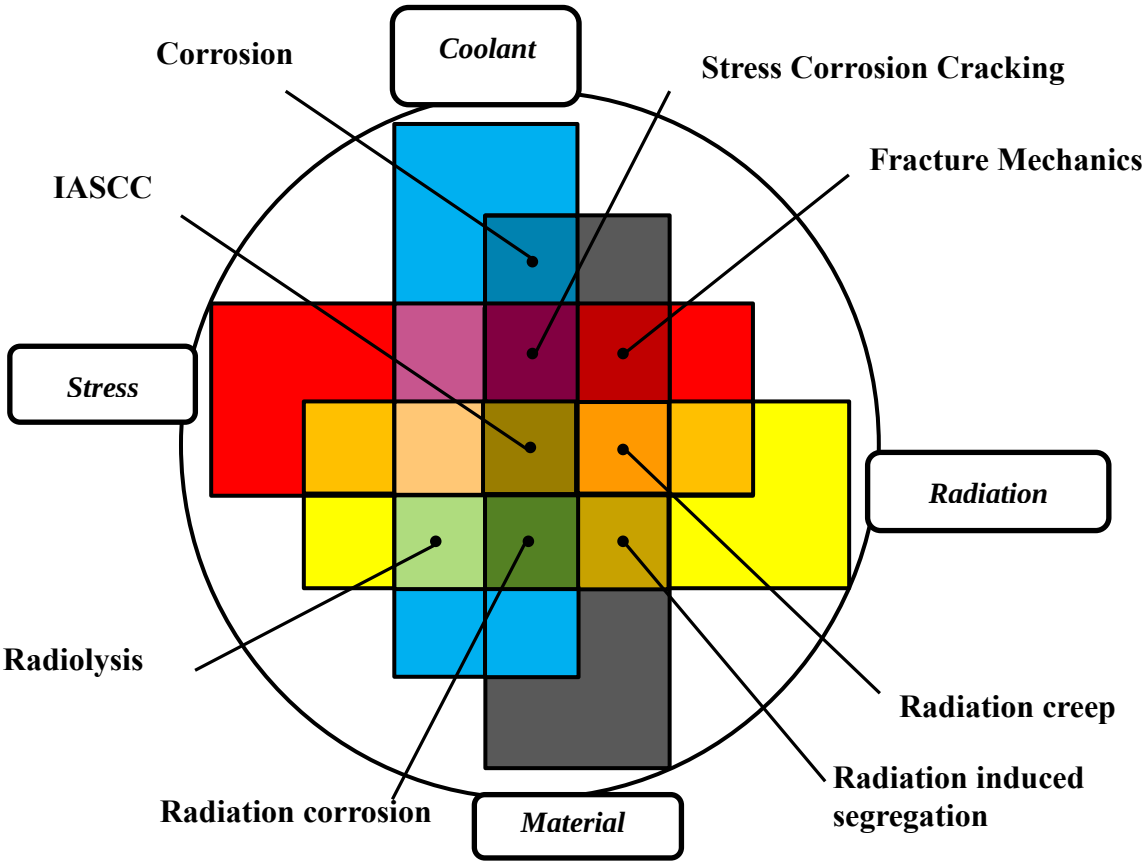


FIG. 17 the concurrent action of all stressors present on the material.

From the point of view of the stressors figure 17 illustrates the concurrent action of typical stressors present on material exposed to the environment of a water cooled nuclear reactor.

Under the most demanding operating conditions near the core of water cooled nuclear reactors, metallic materials represented by the grey rectangle are subject to the concurrent influence of the cooling medium, of stress fields and of high radiation levels. The mixed coloured squares superimposed over the grey rectangle denoting the SSC material represent different combinations of stressors on the material. The square located at the centre of the diagram indicates a condition known as irradiation-assisted stress corrosion cracking (IASCC) of austenitic stainless steels, a condition to which are typically subjected light water reactor vessel internals.

The light green square in the lower left region represents the conditions under which the coolant itself, whether it be light or heavy water, exposed to ionizing radiation undergoes high-energy radiolysis of H_20 / D2O molecules into H^+ / D+ and OH^- radicals. These radicals are themselves chemically reactive, and in turn recombine to produce a series of highly reactive compounds such as superoxide (HO_2) and peroxide (H_20_2), which enhance oxidative damage to metallic surfaces. The squares to the right of the light green square represent the effect of radiolysis on the corrosion of stainless steel. The combination of stressors becomes particularly aggressive in the presence of oxygen, a condition conducive to inter-granular stress corrosion cracking.

The effect of radiation and water radiolysis on activity transport in nuclear power systems 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, 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 due to radiation exposure (namely its strength, ductility, elastic moduli) that may lead to swelling, elongation, metal fatigue and even to creep-related ruptures. Swelling is exacerbated by the effects of temperature, cold bending, impact and stress levels.

Stress corrosion cracking (SCC) is the condition whereby cracks form typically in austenitic steels and aluminium alloys subject to a corrosive environment such as that induced by the presence of chlorides, alkali, nitrates, ammonia that can lead to sudden failures, especially at elevated temperatures. The metal surface may appear shiny and unaffected but the cross section can harbour cracks which often go undetected but progress rapidly producing devastating and unexpected failures. When cracks are detected through in-service inspection (ISI) the analytical methods called fracture mechanics can be used to make predictions. Fracture mechanics applies the theories of elasticity and plasticity to the microscopic crystallographic defects characterized by ISI and is able to predict mechanical failures with acceptable accuracy.

Degradation mechanisms include metallurgical phenomena such as irradiation embrittlement, fatigue, corrosion, interaction and combinations thereof. Table 2 illustrates the interaction of such phenomena. It should be noted however that the table shows only the main degradation mechanisms for different NPP components.

TABLE 2. MAIN DEGRADATION MECHANISMS IN PWR COMPONENTS – AGEING RAINBOW CHART

Material	**Phenomena**	**Sub-components**	**Mechanism**	**S**	**K**	**C**
C&LAS base metal	Irradiation effects	Nozzle, intermediate, & lower shell courses; inlet & outlet nozzles	Emb.	2.9	2.1	2.9
C&LAS welds and clad	Irradiation effects	Nozzle course welds, inlet & outlet nozzle welds	Emb.	2.8	2.1	2.8
Ni-alloy base (A600)	SCC	Bottom mounted nozzles & repair pieces (B&W)	IG/	2.3	2.8	2.8
Ni-alloy W&C (A82/182)	SCC	DM welds for: inlet & outlet nozzles, core flood nozzle (B&W), safety injection nozzles (W), bottom mounted nozzles, repair pieces	IG/	2.6	2.6	2.8

Source: Fictitious data, for illustration purposes only

Figure 18 below correlates the stressors present in the material to the initiation of ageing mechanisms which in turn cause the observable physical consequences.

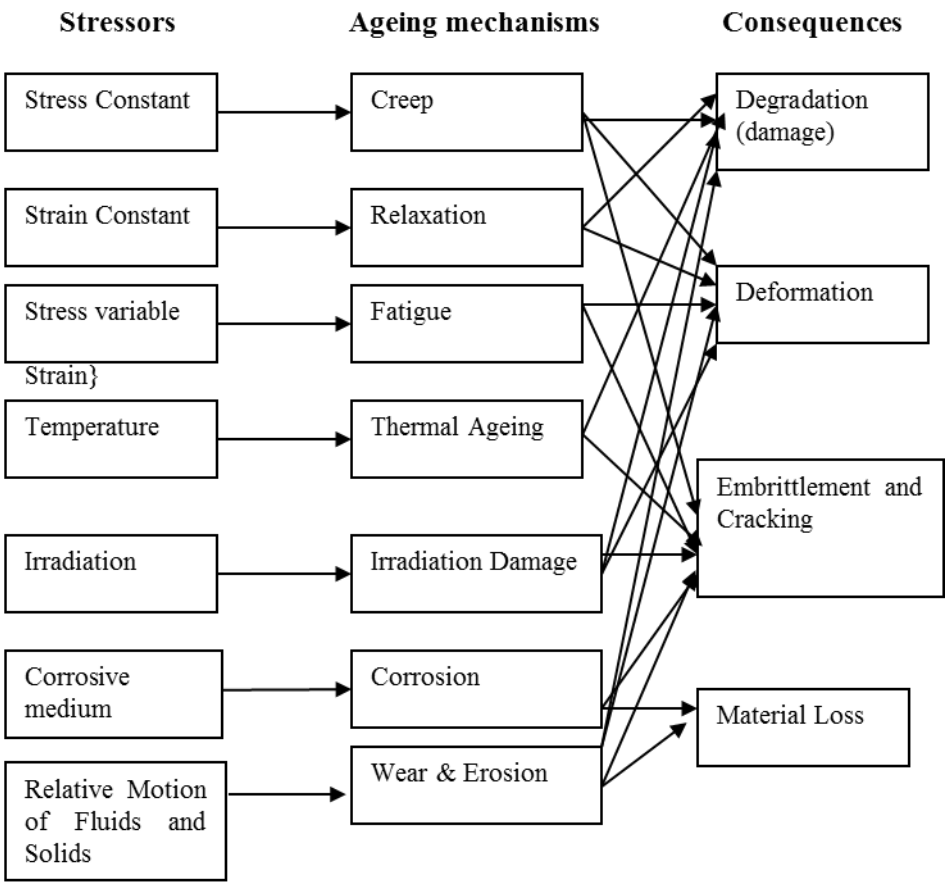


FIG.18. Ageing factors, ageing mechanisms and consequences

Field experience and experimental work have shown that several austenitic stainless steels, such as types 304, 316 L, 316 CW and 347, are susceptible to irradiation-assisted stress corrosion cracking (IASCC) an ageing degradation mechanism affecting mainly the reactor internals. In practice, it may be difficult to decide simply through field observations unsupported by R&D, whether cracking is caused by IASCC or by other types of SCCs.

4.1. RADIATION DAMAGE

Radiation-induced microstructural changes significantly degrade material properties. Radiation damage in RPV steels is usually correlated with neutron fluence on a given material. Two different threshold energies are used for characterization of irradiation conditions - neutrons with energies larger than 1 MeV for PWR and BWR RPVs of western design and with energies larger than 0.5 MeV for WWER RPVs. Unfortunately, ratio between both neutron fluences is not constant for all irradiation conditions, thus any comparison of data must be carefully analyzed. Radiation damage materials for reactor internals is best correlated to displacement damage to quantitatively characterize it. A physical unit called displacements per atom or (dpa) has been used. It is a damage-based or consequence-based radiation exposure 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 to correlate radiation to physical property alterations

Reactor pressure vessels of both PWR and BWR plants are fabricated from low-alloy steels, either of Mn-Ni-Cr-Mo or Cr-Mo-V in western designs and of Ni-Cr-Mo-V material in WWER designs. The most damaging mechanism is radiation embrittlement (together with radiation hardening) that practically determines the RPV lifetime. Radiation embrittlement is characterized by the material transition temperature shift which causes a change in the fracture type from brittle-to-ductile as we move to higher transition temperatures and depends on the material type, the alloy composition and the presence of detrimental elements (like phosphorus and copper) and neutron fluence.

Reactor vessel internals of both BWR and PWR plants are mainly fabricated from austenitic stainless steels. Unlike the fuel elements, which are removed after a few years of service, the internals are intended to remain for the full life of the plant and consequently 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; significantly affecting its susceptibility to stress corrosion in both reactor types. Other effects of radiation damage are swelling, creep, IASCC etc. [14] [15]

Neutron irradiation is the cause of several degradation processes. Figure 19 shows the major degradation processes affecting austenitic steels under neutron irradiation and their effects on the material physical and mechanical properties.

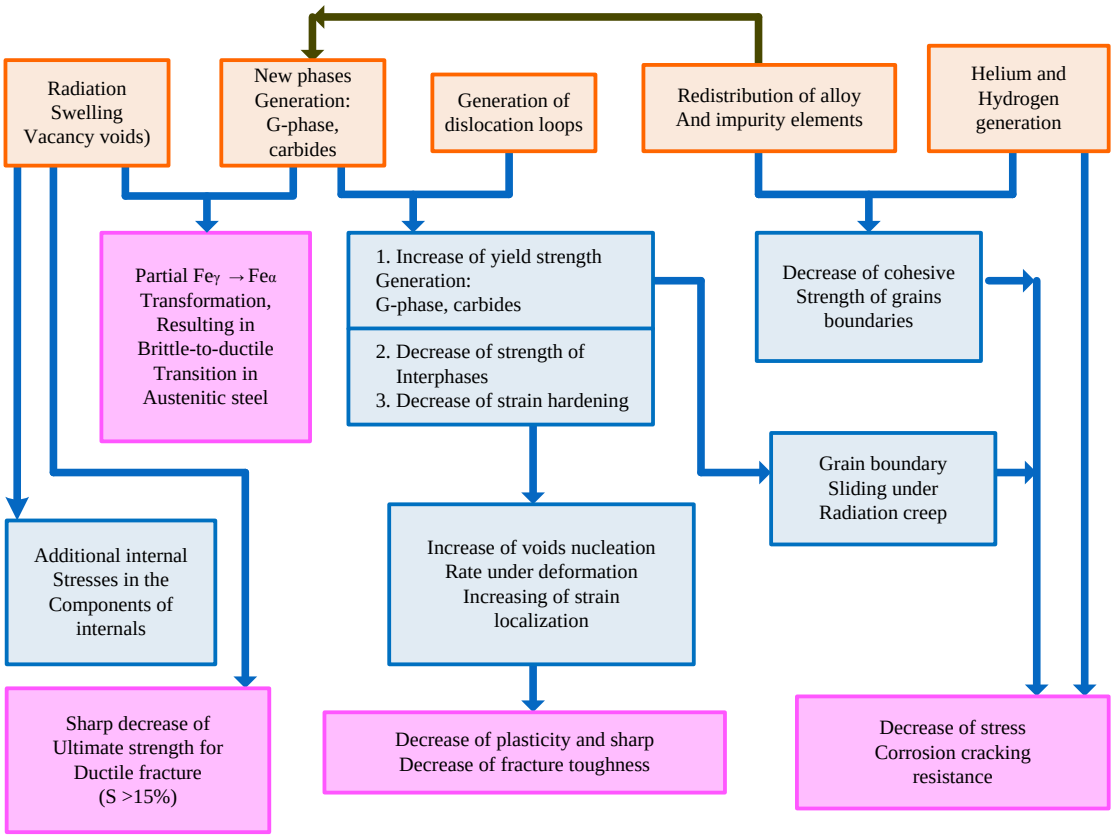


FIG. 19. Effects of neutron irradiation in austenitic steels

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

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

Microstructure

— High irradiation induced dislocation
— loop density;
— Cavities (bubbles and voids).

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$^{1/2}$).

Chemical composition:

— Radiation induced segregation (RIS) at grain boundaries (mainly, Cr, Mo and Fe depletion and Ni and Si enrichment).

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 50s when fatigue failures caused multiple crashes of De-Havilland's Comet, the world's first jet airliner. The characterization of irregular alternating loads and reversed/repeated stress cycles by means of the conversion of irregular time series into a sequence of 'cycles" of stress vs time curves allowed the development of high cycling and low cycling methodologies for fatigue lifetime evaluations and the development of usage factor criteria.

Under repeated cyclic loading, if sufficient localized micro-structural damage has been accumulated, crack initiation can occur in the most highly stressed locations. For metallic materials, a key factor in recognizing fatigue damage is the existence and the shifting of dislocations.

Fatigue damage can be divided into two recognizable stages: In the first stage, fatigue causes local accumulation of irreversible plastic deformation with micro crack initiations. In a second stage, fatigue causes cyclic crack growth that continues until the crack reaches its 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 deformation). When a very high number of cycles (above 10^9) is involved, fatigue is referred to as "vibration fatigue".

Fatigue can be caused by alternating mechanical load as well as by alternating thermal loading (thermal transients). When cyclic thermal loading is involved, fatigue is sometimes referred to as thermal fatigue. Fatigue is one of the phenomena more often considered in time-limited ageing analysis. [16] [17]

4.2.1. Mechanical fatigue

Mechanical fatigue occurs preferentially at those locations of a structure or component, affected by imperfections caused by technological manipulation (e.g. impact of fabrication) or by the presence of geometrical discontinuities. These locations may show signs of local deterioration of mechanical properties, structural defects, discontinuities and local stress concentration. Key parameters for the evaluation of fatigue damage in the first stage are:

— local excursions of stress tensor components in time, usually converted into a sequence of sinusoidal cycles,
— fatigue and tensile properties of the components or materials under consideration,
— environmental characteristics and residual stresses from fabrication and other pre-existing conditions.

In terms of diagnosis there are physical signs of the presence of fatigue damage that are recognizable and in terms of prognosis or predictions regarding the evolution of fatigue damage, there are mathematically sophisticated methods, often codified in national codes and standards (ASME, KTA, RCC, KEPIC, JSME, etc.) which take into account the necessary level of conservatism to reflect the uncertainty of the input data (number of cycles, stress level).

Particularly in the second stage of fatigue in a component, crack growth occurs, and therefore it is necessary to know the shape and dimensions of the cracks, the mechanical parameters characterizing the crack growth and the fracture toughness of the material. Also in this area, there are well developed computational methods to help with the diagnosis and the prediction of the consequences on the component in time (crack growth and criticality).

Normal operation is the condition in which fatigue mechanisms more frequently manifest themselves in a variety of SSCs. Monitoring fatigue damage in the first stage allows in most cases the use of computational post-mortem evaluations. It is necessary to have access to all material properties and loading history consisting of a sequence of operational stress regimes, or conditions.

If a structure or component is designed to a known load spectrum characterized by measurable representative parameters (temperature, pressure, flow rate), monitoring of such parameters is simple. Instrumentation to that effect should be available and the SSC operating history should be recorded. In particular any deviations from the design assumptions should be recorded and safeguarded in the ageing management documentation and in the supporting plant life management databases (if available). In some cases, deviations beyond a certain threshold should be reported to the regulatory body. In cases where it is not possible to codify operating spectra with known parameters and where the actual operating regime is random (e.g. vibration cases, acoustic resonances), it is 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 is possible to continuously measure the deformations etc. An alternative may be direct online monitoring of the manifestations of these mechanism. The same requirements apply to the monitoring of fatigue damage regardless of its development stage.

Most of the fatigue tests are performed in air while SSCs are usually in contact with coolant - high pressure and high temperature water with low (in PWR) or high (in BWR) oxygen content. Thus, design fatigue curves should be adjusted to take into account the effects of the water environment.

4.2.2. Thermal fatigue

Thermal fatigue is a major stressor in surge lines, spray lines and associated nozzles and in mixing tee lines that are subject to thermal stratification, thermal shock, turbulent flow penetration, and thermal cycling during normal operation and even worse during reactor power manoeuvring and plant start-up/shutdown transients. The original fatigue analysis of these components in most older operating reactors includes only design basis transients, but not phenomena such as thermal stratification present in the surge and spray lines, and thermal cycling present in mixing tees; these phenomena were discovered more recently and their effects are only considered in newer designs and for component and piping replacements in older reactors.

Geometrical discontinuities such as connections between thin piping runs and thick valves generally contain inclusions and fabrication defects and the weld geometry induces inherent stress concentration whose effects are similar to those of cracks. Such discontinuities often involve dissimilar materials for example welds between stainless steel piping and low-alloy steel vessels, as in piping to steam generator nozzle welds, or piping to pressurizer nozzle welds. The degradation mechanism in such cases is known as carbon migration. It involves decomposition of carbides, followed by diffusion of carbon from the base steel into the butter material. The driving force in such migration is the carbon content gradient between the ferritic steel base metal and the stainless steel weld metal.

4.2.3. Vibration fatigue

Vibratory fatigue is another type of mechanical fatigue that occurs primarily in small-diameter piping. They have been a significant source of problems in NPPs. The failures have occurred mainly at weldolets and socket welds in small-diameter pipe lines. The failures result from high-cycle mechanical fatigue combined with low-amplitude cyclic stress. There are three main contributing factors:

— Mechanical excitation mechanisms generating fatigue: An example of this type of excitation is the pressure pulsations in centrifugal pumps occurring at frequencies, multiple of the vane passing frequency. They happen in the range of acoustic frequencies. They 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.
— Cavitation: This 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 vibration of piping 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 multi-stage orifices have revealed high frequency, broadband vibrations with accelerations in the hundreds of g's and velocities at some locations as high as hundreds of inches per second. In addition to severe vibrations, the collapse of the cavities on a solid surface removes material by mechanical erosion, damaging piping and other components. One of main differences between cavitation and mechanical excitation is the frequency content (broadband) of the excitation.

— Flashing is a third excitation mechanism that occurs when the temperature of water is higher than its saturation temperature at a given pressure and the water flashes into steam. This results in broadband pressure pulsations causing vibration of the piping downstream of the flashing component and steam hammer phenomena. This is the reason why valves controlling oversaturated flow into a vessel kept at lower pressure are installed as close as possible to the entry nozzle so that the flashing may occur in the larger volume of the vessel and not in piping between the valve and the vessel. In a somewhat opposite phenomenon to flashing, the collapse of steam bubbles may also cause water hammer which may synergistically interact with vibratory fatigue and cause failures.

4.2.4. Fatigue by geometrical discontinuities

Geometric discontinuities caused by a localized thickness change in a cross section of such welds intensify the stresses in a very localized area. The stress amplitude decays rapidly to nominal values as we move away from the discontinuity. These discontinuities are intrinsic characteristics of weldolets and socket weld connections prone to fatigue. In addition to geometric discontinuity, imperfections and crack-like weld discontinuities in the weld metal or in the heat-affected zone also induce stress concentration which drastically reduce the weld-joint fatigue strength.

Inadequate piping supports can also inducing 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 natural frequency. If an excitation source exists in the natural frequency range of the piping system (e.g. the vane passing frequency of a centrifugal pump in the system) resonance may occur and high frequency fatigue damage may result.

It is to be noted that the excitation mechanisms described above generate high-frequency vibratory loads and acoustic resonance that are generally not considered in older NPP stress analysis reports which deal almost exclusively with low frequency alternating loads (such as seismic). Acoustic modelling has only been introduced relatively recently. Vibrations of this nature however should be quantified and adequately monitored during plant operation. They should be fed back to Engineering or R&D for adequate resolution or acceptable mitigation.

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 that 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 structure or 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 (uniform corrosion, local corrosion attack, selective corrosion),
— Corrosion combined with mechanical loads (stress corrosion cracking, hydrogen cracking, irradiation assisted corrosion, corrosion cracking due to deformation, corrosion fatigue) and

— Flow assisted corrosion (erosion-corrosion, flow accelerated corrosion).

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 loss of the material that involves a wide area without any appreciable localized attack. Degradation begins with the formation of a surface layer and continues 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 resulting in a reduction of the material thickness. Its rate depends on the oxygen content, on the temperature, on the flow rate of the medium, etc. General corrosion does not change the material microstructure underneath the surface layer.

Local corrosion has different characteristics. It usually appear in crevices as localized pitting or small holes through the protective surface layer caused by the local concentration of certain chemical agents, as oxygen, chlorides, ammonia etc. Pitting for example often occurs on the steam generator tubes at their place of attachment.

Selective corrosion appear in certain areas such as the area adjacent to the grain boundary, or it may affect only certain specific elements of the alloy. A typical example of selective corrosion is inter-granular oxidation in some austenitic steels and some other nickel and iron based alloys. Inter-granular 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 inter-granular 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 or pools, tanks etc. or in condensation phenomena and other sources of humidity in the presence of oxygen forming salty acidic or caustic solutions particularly giving rise when in contact with steel to galvanic and stray currents that displace metallic ions.

Galvanic corrosion is accelerated corrosion occurring when a metal is electrically coupled with a more noble metal with different surface potential, when they both have a common electrical path and they are both sharing the same electrolyte. The flow of electrons is due to the difference of potential between the two metals moving from the more active anode (less noble) to the less active, the 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 near vent line bellows, flanged hatches in the wet well in BWRs, penetrations attachment to the containment in supports, ducts, grounding wire etc.

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

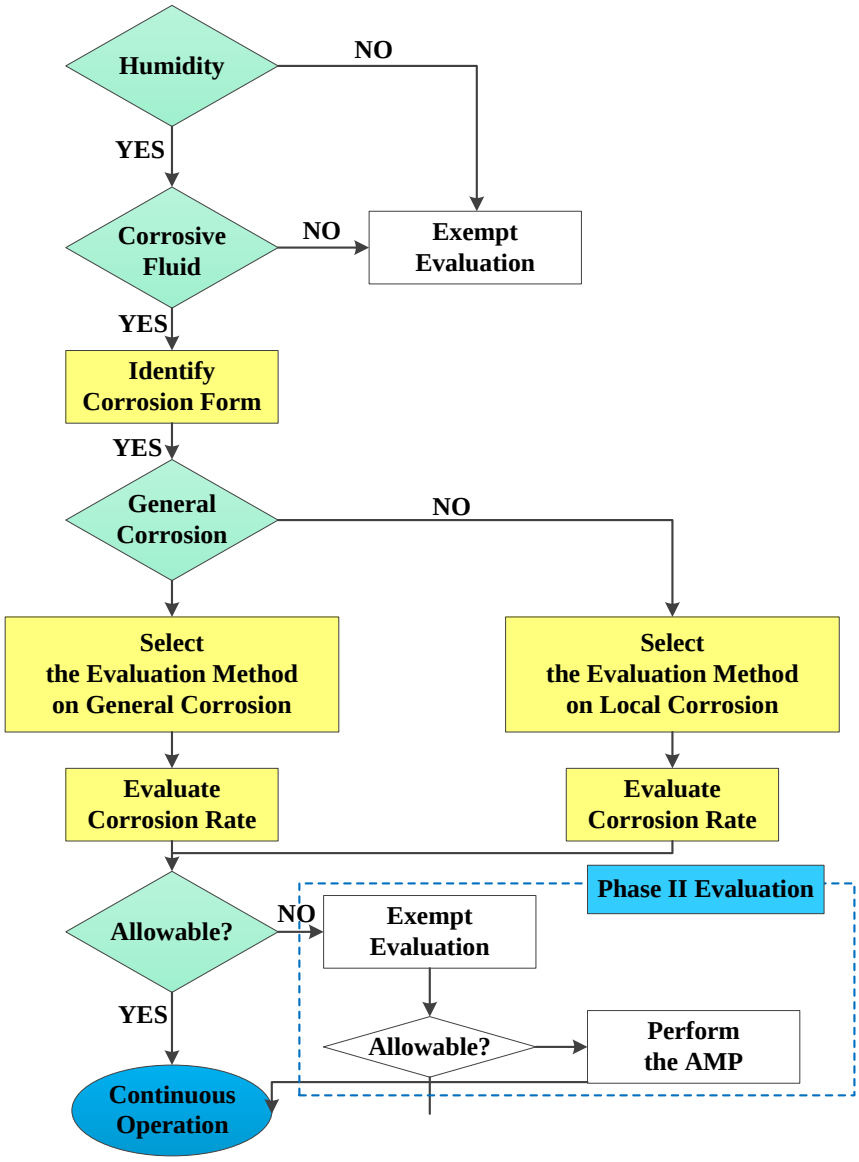


FIG.20. Evaluation of general and localized corrosion by KHNP (Korea)

Figure 20 shows the process used by KHNP in Korea to evaluate the extent of general and localized corrosion in SSCs operated in humid environments and in those containing corrosive fluids. In any case, the operator should identify the form of corrosion that can occur. If only localized corrosion is found, the corrosion rate is evaluated in a different way than in general corrosion cases. Aging evaluation for corrosion is based on an estimate of the corrosion rate. If the estimated of the corrosion rate is not acceptable, a second tier investigation (PLiM phase II project) is conducted on the component which will this time involve direct thickness measurements. NDE may be performed for the measurements. If the remaining thickness is found to be unacceptable in accordance with code requirements, then mitigation of the condition should be planned via AMPs.

4.4. STRESS CORROSION CRACKING (SCC)

Stress corrosion cracking is a complex phenomenon driven by the synergistic interaction of mechanical, electrochemical and metallurgical factors. Stainless steels are particularly susceptible to stress corrosion cracking in the presence of chlorides and in neutral pH demineralized water containing oxygen such as the recirculating coolant in a BWR

Both BWR and PWR components can suffer from SCC, which may have trans-granular (through the grains - TGSCC) or inter-granular (along the grain boundaries - IGSCC) morphology.

Sometimes the modes are mixed or the mode switches from one to the other. IGSCC and TGSCC can occur in the same alloy, depending on the environment, the microstructure, or the stress/strain state. SCC usually propagates perpendicularly to the principal tensile stress. Cracks can also vary in the degree of branching.

All SCC has a brittle-like appearance, since cracks 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 as a result of irradiation damage. Many alloys are susceptible to SCC in at least one environment.

However, SCC does not occur in all environments. On the other hand, an environment that induces SCC in one alloy above a certain temperature does not necessarily induces SCC in another alloy. Nickel-chromium-iron alloys can exhibit inter-granular stress corrosion cracking (SCC) at temperatures greater than 250 °C when exposed to high purity water. Among the nickel alloys, alloy 82 is the only commonly used nickel base weld metal considered to be practically resistant to SCC since no failures have occurred in service to date. Other nickel alloys, such as alloy 600, need to be evaluated for SCC resistance on a case by case basis.

SCC is usually divided into an initiation and a propagation phase. SCC initiates from sites of localised 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 sub-phases, a ‘slow’ propagation phase and a ‘fast’ propagation phase. The latter is usually characterized by crack tip stress intensities, KI, exceeding a characteristic apparent threshold value in pre-cracked type specimens known as KISCC in fracture mechanics. Perhaps the most interesting fact concerning SCC is that three preconditions are necessary and must be present simultaneously for SCC to take hold. The elimination of any one of these pre-conditions or the reduction of one of the three below some threshold level can, in principle, prevent SCC. The three necessary preconditions for SCC to occur are shown in figure 21.

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

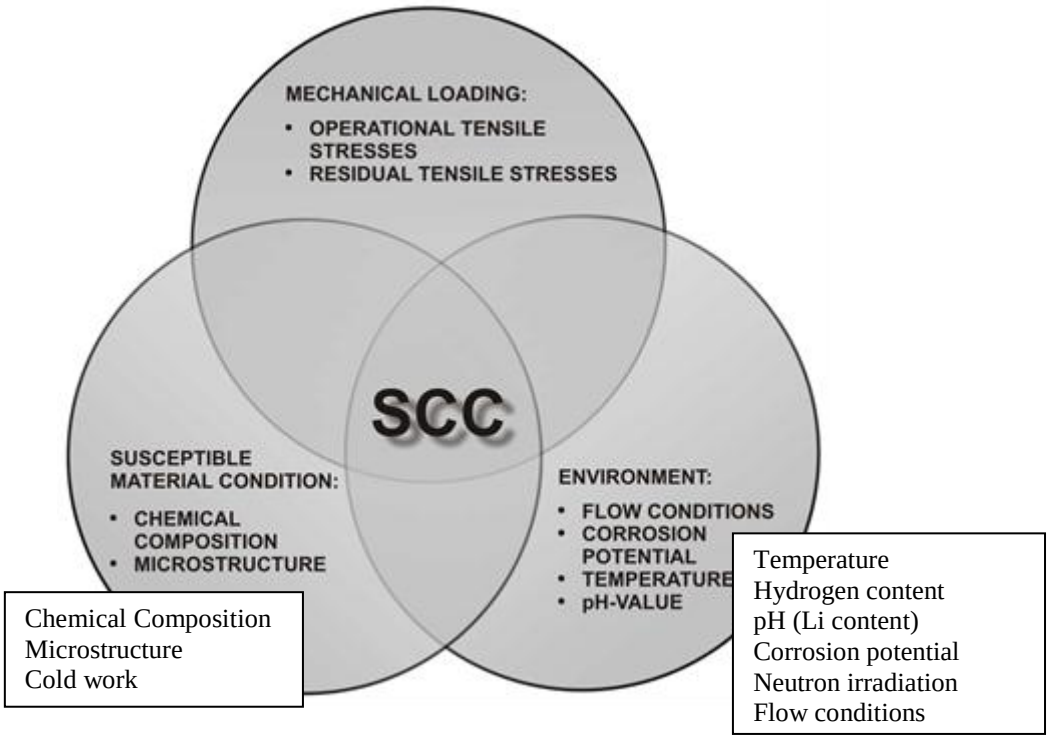


FIG. *21. Factors of Stress Corrosion Cracking*

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

IGSCC and IASCC have been significant for some components in BWRs made of austenitic stainless steel or nickel based alloys. Examples of such components are the recirculation piping, core internals and some parts of the RPV such as the in-core monitor (ICM) housing and the control rod drive (CRD) stub tubes.

4.4.1. Transgranular Stress Corrosion Cracking

A SCC issue common to both types of light water reactors is trans-granular stress corrosion cracking (TGSCC) of austenitic stainless steels, which is primarily due to chloride contamination although other halide anions such as fluoride can also cause TGSCC. It is generally a problem that initiates on the outside surfaces of austenitic stainless steel components mainly due to 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 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 predominantly in the past and has the advantage of having large concentrations of soluble silicate which have a favourable buffering action in the presence of chloride contamination of external surfaces of austenitic stainless steel. The allowable limits for surface chloride contamination in combination with the soluble silicate content of thermal insulation are encapsulated by the Karne's diagram shown in figure 22 taken from Ref. [18] in the process of being updated - draft Reg. Guide DG-1312 [19]

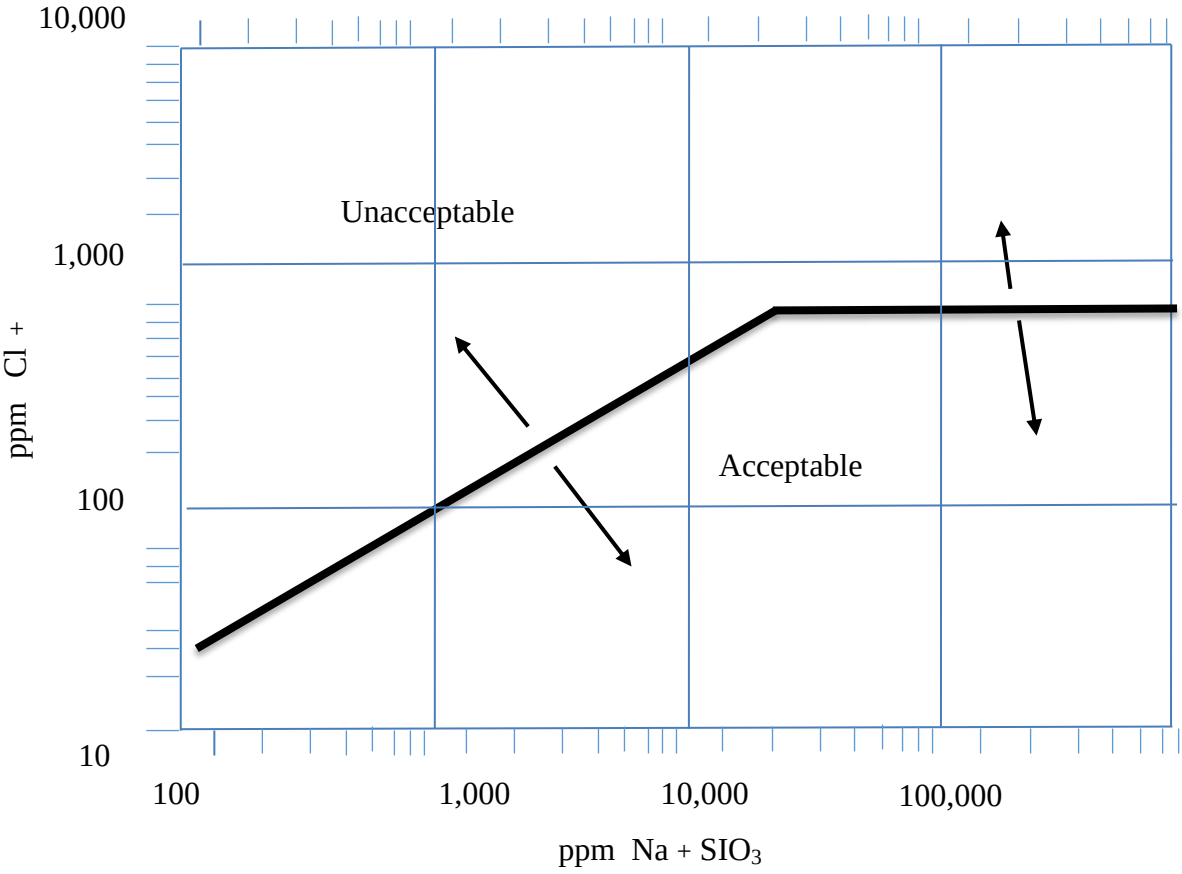


FIG. 22. Karnes ASME (USNRC Reg. Guide 1.36) showing the safe and unsafe areas for chloride and soluble silicate in insulation material.

In newer plants, mineral wool insulation is tending to replace fibreglass in many countries because of concerns about clogging of reactor building sump pump filters during major loss of coolant accidents caused by debris from glass fibre shredding. Mineral wool insulation is less prone to clogging such filters but has the disadvantage that it has much less soluble silicate and is therefore unfortunately much less tolerant of surface chloride contamination with obvious consequences for 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 in dead legs of chloride contamination and oxygen. One location that has been rather frequently affected by TGSCC in PWRs is the canopy seals that assure the pressure boundary of threaded connections in the control rod drive housings that are located above the reactor pressure vessel upper head. Leaks from canopy seals have caused serious boric acid corrosion of the upper head low alloy steel.

The origin of the problem of cracking of canopy seals is air bubbles that are trapped during refuelling when the reactor pressure vessel is open to air followed by inadequate de-oxygenation procedures to remove air bubbles from such locations with a very complicated pathway to the reactor vessel itself. Procedures during plant start-up for eliminating these air pockets vary between operators but the one acknowledged reliable method involves 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 (PWSCC)

Primary water stress corrosion cracking (PWSCC) refers to SCC in the primary water environment of PWRs. It indicates the degradation that certain alloys undergo in contact with the reactor primary coolant. More susceptible to this type of degradation among the materials used in the primary and interconnected systems in PWRs is Inconel 600 182 and 82. There is no general consensus on the mechanism responsible for it. Two dominant hypotheses have been proposed:

— hydrogen embrittlement (HE), a mechanism involving the diffusion of hydrogen and its concentration at the crack initiation tip.
— dissolution/oxidation at the crack tip with de-alloying of the Cr and surface diffusion

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

Another particularity is that PWSCC requires the simultaneous presence of high tensile stresses, a conducive environment (i.e. high temperature water) and a susceptible microstructure.

PWSCC of alloy 600 was reported for the first time in the steam generator tubes at Obrigheim in the Federal Republic of Germany in 1971. Similar degradation was observed worldwide until such time as the tubes were replaced with thermally treated alloy 690.

Significant damaged areas were the tight row 1, 2 and 3 U-bends, roll transitions, and some 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. This weld was made with Inconel 600 alloy 182. It was the last weld made during SG fabrication. It was done manually and was not followed by any thermal treatment. The stubs and divider plates in most steam generators were made from forged plates of alloy 600 whose final thermal treatment occurred at the end of their forging sequence.

Damage was reported to Inconel 600 material also outside of steam generators. In the US during the 1989 leakage was reported in 20 pressurizer heater and instrument penetrations at San Onofre, in the pressurizer thermal sleeves and in instrument nozzles at Calvert Cliffs

PWSCC has never been observed in alloy 800 SG tubing or in replaced steam generators with alloy 690 tubing.

In contrast to the IGSCC problems experienced in 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. [20]

4.4.3. Irradiation Assisted Stress Corrosion Cracking (IASCC)

Irradiation assisted stress corrosion cracking is extensively studied in research and test reactors and evaluated in actual plants. Figure 23 shows a scheme developed by the Tokyo Electric Power Company (TEPCO) used to evaluate IASCC.

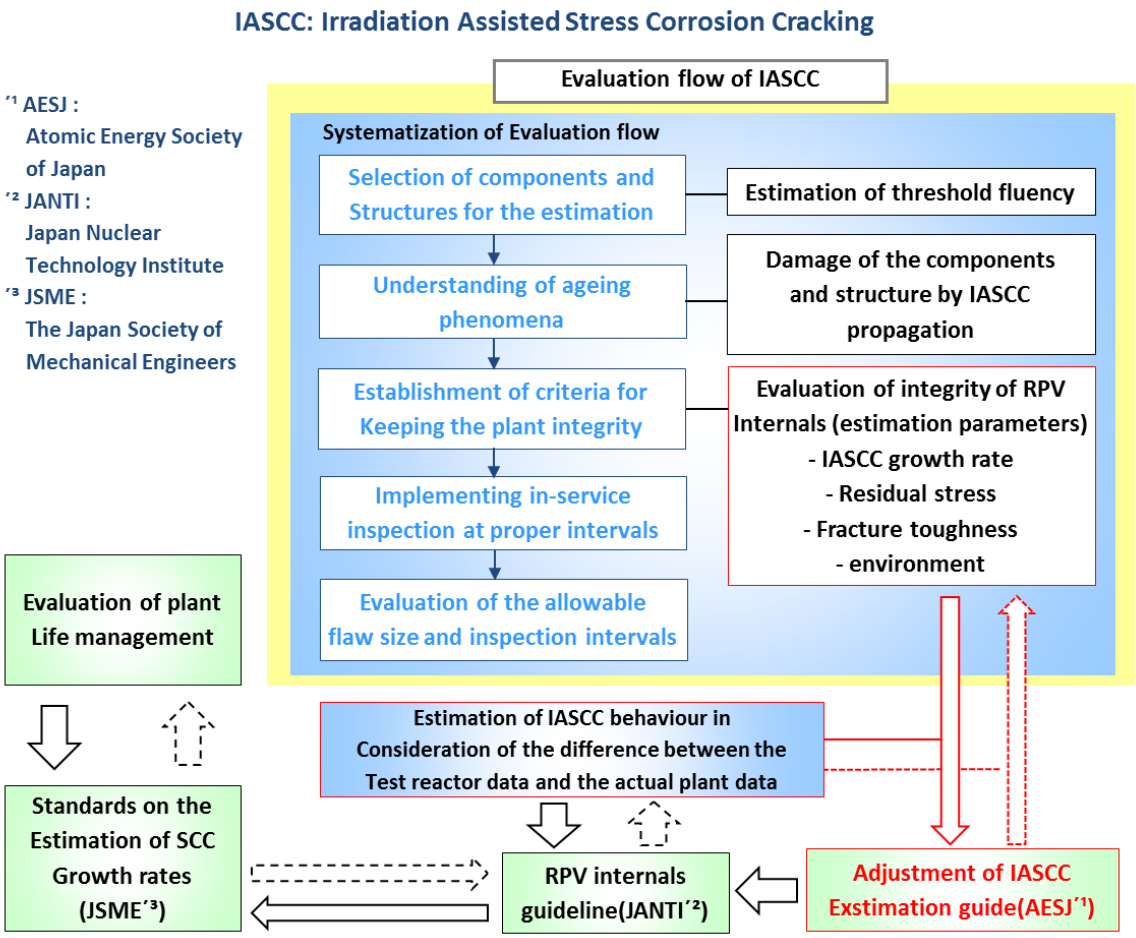


FIG.23: IASCC evaluation scheme for RPV internals developed by TEPCO, Japan

This systematic approach to an IASCC assessment begins with a selection of highly irradiated RPV internals in Boiling Water Reactors. Components are selected if subjected to an appropriately established threshold fluency or above it. Understanding of the phenomena is normally acquired through an analysis of the information collected and the characterization of surveillance coupons exposed in power reactors. Another source of information is from the results of R&D projects making use of research reactors. This 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 allows an estimate of the remaining life of the reactor internals in the actual plant by studying the difference between the test reactor coupon data and the actual in-reactor conditions. Finally guidelines have been developed to directly evaluate the RPV internals in the whole reactor fleet.

The susceptibility of IASCC attack under various neutron fluences and irradiation doses are shown in figure 24.

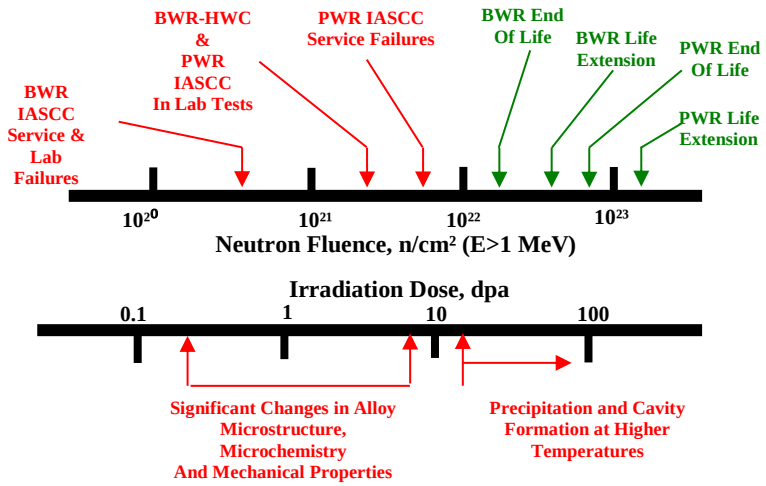


FIG. 24. IASCC susceptibility in relation to neutron fluence and irradiation dose

For BWRs two threshold values for the onset of irradiation assisted stress corrosion cracking (IASCC) have been reported depending on the stress level of the component. For components with high tensile stresses the threshold is ~5 × 10^{20} n/cm^2 (E>1MeV); with lower tensile stresses the threshold is ~2 × 10^{21} n/cm^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 get a better understanding of and to establish a database for IASCC of austenitic stainless steels. In 2007, two relationships between and applied stress were published that were based on experimental results in simulated PWR primary water using irradiated samples removed from several operating plants. Although there are some differences in the threshold stress reported in both papers, the threshold stress for IASCC initiation clearly decreases with increasing fluence. Highly irradiated 316CWss at a neutron fluence of more than ~30 dpa showed IASCC susceptibility above relatively low stresses between ~0.4 σ_y to 0.6 σ_y (where σ_y is the as-irradiated yield stress of typically ~1000 MPa at such high fluence). [20]

4.5. FLOW ASSISTED CORROSION

Flow Accelerated Corrosion (FAC) or Erosion Corrosion (EC) is a chemical corrosion mechanism affecting metals such as carbon steels, which owe their corrosion resistance to the formation of oxide layers. If a moving fluid is in contact with such a metal as in a piping system, the oxide layer protecting the metal surface may be dissolved. As a result a continuing process sets itself in: the constant creation and dissolution of the surface oxide layer which gives rise to metal wall-thinning. This of course if allowed to continue indefinitely may ultimately cause perforation of the pipe wall. The corrosion rate is proportional to the solubility of the oxide layer, to the material, the content of Cr, Mo and Cu, the temperature, the flow velocity, the surface roughness and the pressure/flow profile of the fluid. In addition, FAC is enhanced by the impingement of liquid droplets and by cavitation phenomena. Flow accelerated corrosion manifests itself in a generalized fashion. The attack is widespread and thinning occurs throughout the affected region, rather than in localized as in the case of pitting or cracking. FAC has been experienced worldwide and can occur in all types of nuclear and fossil power plants. Although the FAC mechanism has been studied for a long time these last 40 years, it remains one of the major safety issues in terms of the potential for unexpected failures, injury and even loss of life. FAC is also a concern for plant availability.

Flow accelerated corrosion (FAC) has caused deterioration to internal parts in steam generators such as tube supports and moisture separators and in BOP components, particularly piping in several plants in Japan, France and in the US. EPRI has issued guidance documents dealing with FAC, notably:

— Flow accelerated Corrosion in Power Plants [21]
— PWR secondary water chemistry guidelines [22]

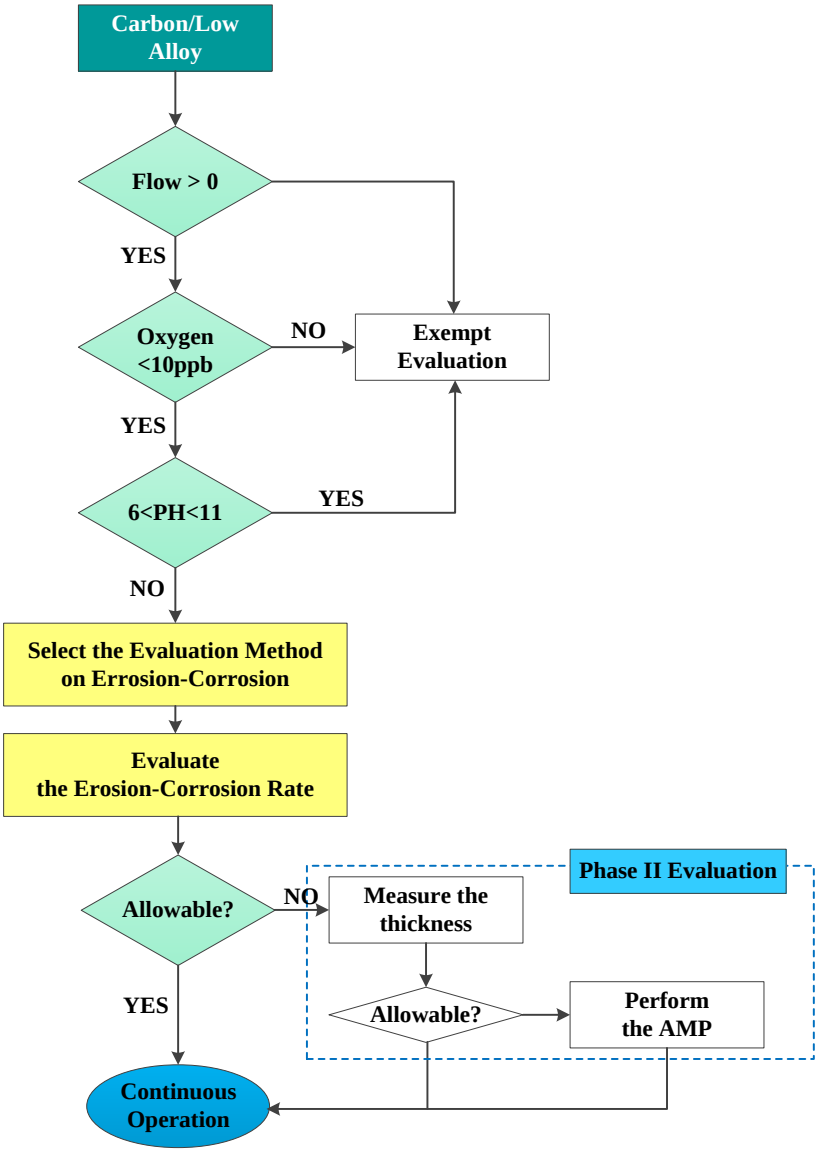


FIG. 24. FAC Erosion-Corrosion evaluation

A successful FAC programme should focus on reducing the number of inspections, the FAC wear rate by improving materials, the water chemistry; and the design.

An example of an effective erosion-corrosion (or FAC) evaluation process is shown in Figure 24 from Korea's Hydro and Nuclear Power corporation (KHNP), a subsidiary of the Korea Electric Power Corporation (KEPCO). If flow velocity and oxygen content in the water are beyond or below certain values as shown in the Figure, the component is either exempt from evaluation or the erosion-corrosion rate is evaluated and, if the operating conditions warrant it, an ageing management programme for continuous operation is applied. The general approach to a technical evaluation of FAC / EC in the context of an ageing management programme is well described in EPRI's document [21].

4.6. THERMAL AGEING

Thermal ageing is a normally non-reversible degradation mechanism heavily dependent on temperature, the material microstructure and time. The material may lose ductility and become brittle because of microstructural changes in the form of precipitates separating from the metal matrix. In the case of the RPV steel with appreciable copper impurity, most precipitates are copper-rich. The

precipitates block dislocation movement thereby causing hardening and embrittlement. Copper in RPV steel is initially trapped in a super-saturated state. With time at normal PWR operating temperatures (~290°C), it may be ejected to form stable precipitates as the alloy strives toward a more thermodynamically stable state, and this occurs even if there are no radiation related acceleration of thermal ageing in the metal microstructure. Thermal ageing depends solely on time and temperature of exposure, together with the material type and its chemical composition.

Thermal ageing has no hardening effects in RPV steels, as only changes in the transition temperature can be observed, without any measurable changes in yield strength. As this damage 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 been studied in several countries (USA, France, Germany, Japan). Literature searches suggest that no embrittlement occurs in typical RPV steels at these temperatures for operating times as great as 100,000 h.

The 15Kh2MFA and 15Kh2MFAA type steels used to fabricate the 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 10 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), as well as boat samples taken from operating plants.

The 15Kh2NMFA type steel used to fabricate the WWER-1000 pressure vessels is slightly susceptible (a shift in the ductile-brittle transition temperature of 10 to 20°C) to thermal ageing at operating temperatures, due to the high nickel and low vanadium content of this material.

Cast austenitic stainless steel (CASS) is a very ductile material, resistant to stress corrosion cracking. It is vastly used in safety related components. CASS materials of complex geometries are however difficult to inspect. The challenge is to identify the most effective technologies. Multi-angle inspections have been used to improve the signal to noise ratio. If the molybdenum content is high for a static component with greater than 14% ferrite content, fracture analysis may be conducted. The input parameters are determined such as the J-integral versus crack growth resistance (J-R) curves for polymeric materials with both elastic and plastic contributions. These can be obtained directly from load vs. displacement records in literature without installing an on-line crack monitoring system. Once the input parameters are obtained, the crack 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 ferrite content greater than 20% fracture analysis can be conducted the same way as for static components. If the ferrite content is below 20 %, the component can be exempt from evaluation. Figure 25 shows the thermal aging evaluation approach for cast austenitic stainless steels developed by KHNP, Korea. The process is well documented in IAEA's IGALL report [23]

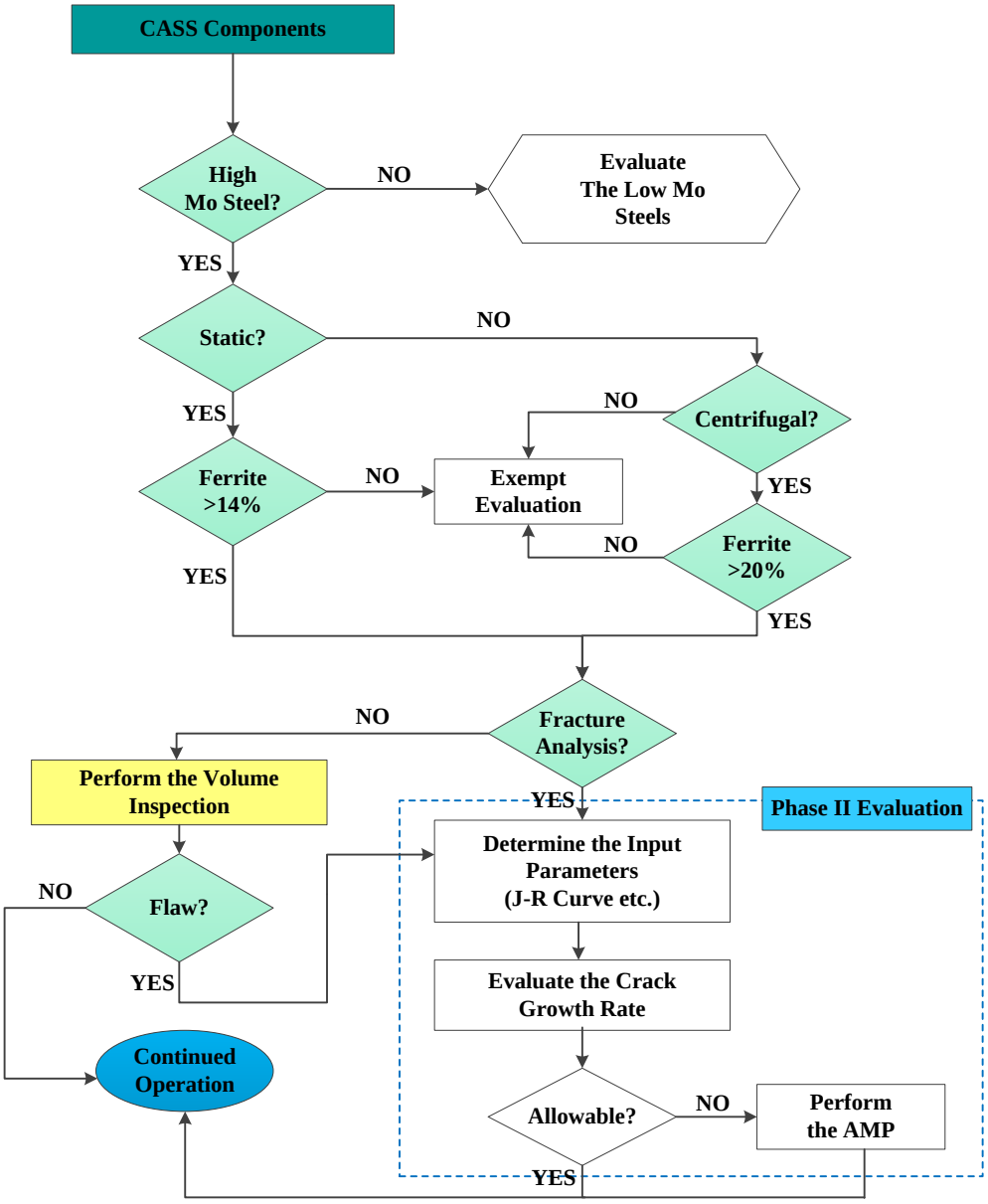


FIG.25. Embrittlement evaluation of cast austenitic stainless steel (CASS) KHNP, Korea

5. DEGRADATION AND MITIGATION OF STRUCTURES, SYSTEMS AND COMPONENTS

Ageing management programmes usually include periodic in-service inspection (ISI), periodic inspection programmes (PIP) for components selected for preventive maintenance (PM), Water chemistry control programmes, and specific structures and components intended to undergo life cycle management (LCM) etc. These programmes include health monitoring of critical structures, system and components during normal operation and ultimately in service extended beyond the original design life. Figure 26 shows how all elements of an ageing management programme come together in a simplified diagram.

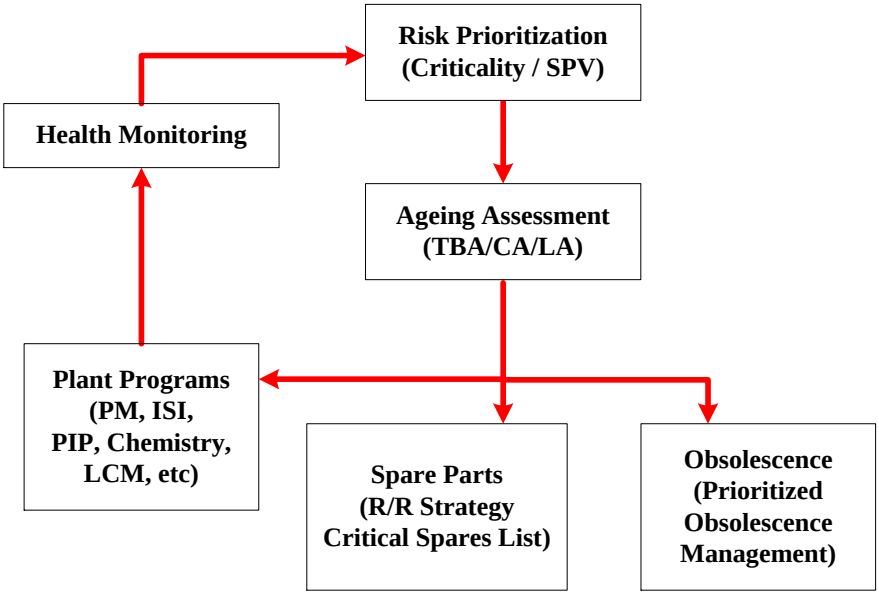


FIG. 26. Elements of a typical ageing and obsolescence programme (simplified diagram)

When Plant Life Management techniques are adopted, they are capable of optimizing maintenance intervention and mitigation activities and establish their prioritization on the basis of risk and SSC criticality. Prioritization can also be established by means of single point of vulnerability (SPV) analyses. Depending on their categorization, SSCs will be assigned to a particular ageing assessment programme (time based assessment, condition assessment or life assessment). As a result of these assessments, a critical spare part list can be prepared, spare part inventories can be defined in support of repair/replace (R/R) strategies. Obsolescence is also part of any ageing assessment programme and its mitigation should also be prioritized.

Having defined a prioritized critical SSC list, the AMPs for each item are defined with input from operating experience, locally and internationally and from R&D feedback.

5.1. DEGRADATION OF STEEL CONTAINMENT

General corrosion of exposed metal surfaces begins with the formation of a thin layer that continues as long as the reactants can penetrate the layer. It is an electrochemical process that requires an electrolyte as in metals in contact with the reactor coolant or exposed to humid environments such as air containing suspended salts, acid or caustic agents, and its natural constituents oxygen and carbon dioxide. These may be acquired by the air from system leaks, flanges, the suppression chamber, the fuel pool, the refuelling operation, condensation, leaks during maintenance, groundwater intakes etc. The general corrosion mechanisms affecting metal structures inside containment as described in chapter 3 are:

— atmospheric corrosion

— aqueous corrosion: Flaws in aqueous environments can be initiation sites for crevice corrosion
— galvanic corrosion
— stray-current corrosion

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

— crevice corrosion
— pitting corrosion
— biologically and microbiologically-induced corrosion

Mechanically-assisted or accelerated degradation involves corrosion coincident with mechanical wear and/or metal fatigue. In terms of consequences, overload conditions can deform, bulge, bend, buckle, stretch the steel and produce detrimental effects on the containment capacity and leakage tightness. Sources of this kind of degradation can be erosion, fretting wear, surface flaws, arc strikes, and overload during in-service inspections, repair work or missiles generated by equipment failure or pipe whip events. Mechanical loads of this kind however are not expected to be frequent events affecting containment steels. Possible exceptions are found in the BWR design:

— Lubrite contact surfaces for example those between the BWR Mark I metal containment torus support column base plates and the base mat due to relative motion, thermal transients and pressure testing
— Steam impingement during safety/relief valve discharge into the suppression pool. 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.

Stress Corrosion cracking and hydrogen induced cracking are phenomena not quite categorized as general corrosion, but in BWR containments they can attack large portions and hence be called general corrosion. They are caused by exposure of metal surfaces to an aggressive environment. Hydrogen becomes aggressive in the presence of high temperatures and/or high pressures but this is not be a serious problem for BWR containments since H_2 is not expected, either at high pressure or at high temperature.

Stress corrosion cracking effects BWR containment steels to some extent. It can occur at relatively low tensile stresses, often below the material yield strength. It may be caused by applied loads, residual stresses. It may appear in the heat affected zones after welding and in the presence of corrosive agents. Stress corrosion cracking is however rare in carbon, low-alloy and stainless steels of the BWR containment liners, but they can occur when those surfaces are exposed to solutions containing sulphates, hydroxides, chlorides, ammonia, fluorides, carbonates and decontamination fluids. The most probable degradation mechanism for BWR containment steels remains corrosion associated with mechanical wear or abrasion. However, stress corrosion cracking may more easily affect stainless steel bellows, electrical and piping penetrations, and the wet portions of the ABB Atom design.

Another observed degradation phenomenon in BWR containments may be fatigue. In all fatigue cracking, cyclic stresses are the main cause. When the initial crack reaches a critical size and the remaining un-cracked 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 (start-up/shutdowns, power manoeuvring, etc.)
— Piping loading at penetrations
— Pressure testing and containment leak rate test
— Safety relief valve discharge testing (including steam condensation loads)

BWR containments are designed with sufficient fatigue margins, however local fatigue-related issues may arise. Fatigue may affect for example locations with geometrical discontinuities, locations with material discontinuities as in dissimilar metal welds with materials of different thermal expansion coefficients, locations where installation misalignments occurred. Examples of locations where stress concentrations occur in BWR containments are drywell locations with discontinuities in the region connecting the cylindrical and spherical portions of the Mark I design. Another typical location with stress concentrations is the embedded portion of the drywell base. Precisely, at the concrete-metal interface where crevices can form and moisture containing corrosive substances can enter the crevice initiating corrosion assisted fatigue.

A possible stress concentration location in BWR containments may be misalignments resulting from weld shrinkage or steel plate offsets occurred during construction. Other stress concentration sites can be the containment penetration bellows, typically subject to low-cycle fatigue and stress corrosion cracking during normal operation by virtue of their service requirements. Bellows are made of thin plies that are susceptible to indentation incurred during fabrication, and operation related stress concentration can reduce fatigue life. In addition, transgranular SCC of bellows has occurred from high residual tensile stresses in coincidence with exposure to chlorides, sulphides, fluorides that may have accumulated during fabrication, installation or operation.

Another fatigue source may be the repeated condensation of steam bubbles producing pressure oscillations on the suppression pool liner and on the spot welds of the anchor-bolts. Non-metallic materials are also subject to deterioration, leading to their destruction if not appropriately monitored and tested through ageing management programmes. Penetrations and airlock doors using non-metallic seal material are typical of such a category. Even low radiation doses and low temperatures may cause their embrittlement and failure. Leak tests should be conducted on doors and hatches using such materials at regular intervals and every time they are opened.

5.2. RPV AGEING DEGRADATION IN PWRS

Degradation and ageing are terms used to describe the deterioration of components, but there is a distinction:

— The term “degradation” denotes either a fast or gradual deterioration of characteristics of an SSC that could impair its ability to function within acceptance criteria.
— The term “ageing” refers to a general process in which the SSC characteristics gradually deteriorate with time or use.

When ageing processes are known, they can be monitored through an appropriate AMP and PLiM programme and potentially mitigated. Previously unknown or unexpected degradations, if undetected or underestimated, on the other hand, can lead to accidents

5.2.1. Irradiation embrittlement in the RPV beltline region

In terms of irradiation-related deterioration, of primary concern is the core beltline, typically located in the intermediate and lower shells directly surrounding the height of the fuel element assemblies plus an additional volume both below and above the active core. The low alloy steels making up the beltline are subject to irradiation hardening (increase in yield strength and tensile strength) and embrittlement (shift in transition temperature or decrease in upper shelf energy) that can lead to loss of toughness (elongation, reduction of area) as well as of fracture toughness.

RPV steel toughness is strongly affected by the presence of copper, nickel and phosphorus. Because operating vessels fabricated before 1972 contain relatively high levels of impurity copper and phosphorous, irradiation damage becomes a major consideration for their continued operation. Sensitivity of welds to irradiation degradation and the degree of embrittlement is determined as a function of the chemical composition and the level of neutron exposure. The embrittlement of high copper and high nickel welds plays a key role in the assessment of the significance of pressurized thermal shock (PTS).

Main parameters governing the sensitivity of materials to radiation damage are:

— The flux spectrum that may influence the degree of radiation embrittlement caused in ferritic steels;
— The microstructural characteristics, such as grain size and metallurgical phases that can influence the severity of radiation damage.
— Alloy composition which has a strong effect on radiation sensitivity, especially with elements such as nickel that increases the SSC sensitivity to radiation embrittlement with increasing fluence
— The presence of impurities such as copper and phosphorus that increases sensitivity to radiation embrittlement
— Lower flux levels, lower irradiation temperatures and increased manganese (Mn) levels that may also add to the deterioration.

— A decrease in radiation embrittlement at temperatures above 310°C that has been attributed to the dynamic in-situ "annealing" of the material.

The synergisms and complex nature of the alloys examined makes a complete interpretation of the mechanisms difficult. Prediction of radiation embrittlement is usually based on normative trend curves for transition temperature shifts. These curves/formulae are not generic for all RPVs even if manufactured according to the same material standards. They are mostly manufacturer specific, i.e. they are different for RPVs manufactured in different countries. Weldments however are always critical in the beltline region.

Predictions are then verified by testing of the surveillance specimens that are inserted in special capsules into the RPVs and periodically withdrawn for testing. Surveillance programmes were originally designed for design life of 40 years. Thus, some corrective measures (additional capsules, prolongation of withdrawal scheme etc.) must be taken to assure data collection and continued testing remain possible for RPV lifetime extensions. As a general rule, materials with a tensile strength at room temperature above 700 MPa cannot be used for pressure boundary applications.

5.2.2. Under clad cracking in RPVs:

Cracks confined to the cylindrical portion of the RPV appeared in some first generation vessels after the final stress relief heat treatment of the RPV fabricated from A508 Class 2 forging steel or the equivalent European grades. Post-weld stress relief heat treatment at elevated temperature resulted in de-cohesion of the grain boundaries and small cracking occurred. Under clad reheat cracks are approximately 2 to 3 mm in depth and can be detected during pre-service NDE using special high angle beam transducers.

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

5.2.3. Primary water stress corrosion cracking (PWSCC) in RPVs

PWSCC requires the simultaneous presence of high tensile stresses, a corrosive environment (i.e. high temperature water) and a susceptible microstructure. Alloy 600 is most susceptible to this type of corrosion. PWSCC has caused cracking and leakage in RPV head penetrations and in dissimilar metal butt welds (DMW). There is no general consensus on the degradation mechanism of Alloy 600 182 and 82 in contact with primary water. Two dominant hypotheses have been proposed:

— hydrogen embrittlement (HE) mechanism involving the diffusion of hydrogen and its concentration at the crack initiation tip.
— dissolution/oxidation at the crack tip with de-alloying of the Cr and surface diffusion

It was found that Alloys 690/152/52 are more resistant to PWSCC than Alloys 600/182/82 possibly due to higher Cr and the protective influence of the carbides $Cr_{23}C_{6.}$

PWSCC attacks particularly control rod drive mechanism nozzles and other vessel head penetration nozzles welded to the upper reactor vessel head. This degradation is of safety concern since it could infringe on the reactor coolant system pressure boundary. A control rod drive mechanism (CRDM) break may also cause a control rod ejection accident. See figure 27.

The CRDM penetrations in the pressure vessel head of western RPVs are made of stainless steel. They are usually fabricated from forged or rolled bars. The ASME specification used is 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.

In the Siemens-KWU vessel heads, CRDM nozzles are made of ferritic steel, manufactured as co-extruded pipes and cladded with stainless steel.

Historically, vessel head penetration cracks were first found in France in 1991 at Bugey 3. A year later five 900 MW and four 1300 MW vessel heads also showed cracked CRDM penetrations. They occurred at the earliest after 30,000 operating hours. Due to similarities in materials and fabrication of

all their vessel heads, EdF decided to replace them all. By 2009, fifty 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. Later inspections 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 6 inch (15.24 cm) thick low alloy carbon steel RPV head material with stainless steel cladding. The cavity was about 6.6 inches (16.8 cm) long and up to 5 inches (12.7 cm) wide, extending through the carbon steel all the way down to the 0.25 inch (6.4 mm) internal surface stainless steel, type 308 cladding of the RPV head. The root cause of the corrosion was the result of boric acid interaction with the carbon steel of the RPV head. The source of the boric acid was a primary water leak via a through-wall crack in a CRDM nozzle due to PWSCC.

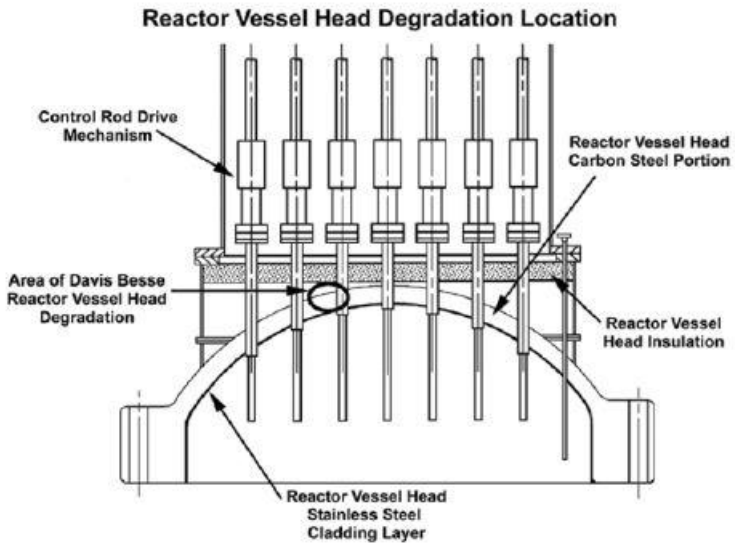


FIG.27. RPV Head Degradation Location

In the Russian VVER reactor pressure vessels, nickel based alloys have not been used. The WWER 440, as well as the WWER-1000 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 (for control rods, instrumentation, etc.) are made of austenitic materials of the same type as the cladding itself, i.e. the first layer/bead on the ferritic material 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 is not a severe matter of concern for the ageing management of the WWER RPVs

5.2.4. Bottom Mounted Instrumentation (BMI) penetration leaks in RPVs

In April 2003, small boron deposits around two of the 58 BMI penetrations (penetrations 1 and 46) were identified in 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 the Takahama PWR Unit 1 plant (figure 28). Laser and water jet peening have been applied as mitigation measures at the Takahama unit and at other Japanese PWRs.

FIG.28. Corrosion near the BMI penetration nozzle at the Takahama Unit 1 plant

5.2.5. Thermal Ageing in RPVs

High temperatures produce in RPV materials a time dependent degradation mechanism. Thermal ageing does not appear to be generic but depends on the heat treatment, chemical composition and service time at temperature. The material may lose ductility and become brittle because of impurities precipitating and affecting the microstructure. The role of phosphorus in the overall embrittlement of western-type RPV materials has been a 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 however is also dependent on the type of microstructure (or state) 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 10 years of exposure at about 300°C indicate that the shift in the Charpy ductile to brittle transition temperature is small. RPV steels with phosphorus content well above about 0.02 wt% may be susceptible to temper embrittlement during fabrication. However, the western RPV materials normally contain less than 0.020 wt % phosphorus. Therefore, it is unlikely that any western RPVs will exhibit temper embrittlement. If a 450°C thermal annealing of an irradiated RPV is required for recovery of the fracture toughness, the possibility of temper embrittlement should be evaluated.

Transition temperature shifts due to use and ageing: The brittle to ductile transition temperature (critical temperature) of pressure vessel materials is time or use dependent, since many damaging mechanisms can affect it primarily the temperature shift due to radiation embrittlement. A shift also be caused by a low-cycle fatigue damage, corrosion and pitting.

5.2.6. Fatigue in RPVs

In nuclear power system, designs components are mainly subject to low cycle fatigue (<1,000 cycles) and the chief sources of cyclic stresses are temperature and pressure transients and to some extent vibration due to abnormal operation with phenomena such as cavitation, acoustic resonance, wear and tear if undetected, unmitigated or unresolved. In western RPVs the closure studs have the highest usage factor of any other subcomponent. It is in the order of 0.66 for the 40-year design life of the vessel. 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 western RPVs.

The service life of the WWER-440 closure studs is limited to 15 years of operation 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 one for the total RPV service time. Additionally they are subject to periodic inspections.

5.2.7. Corrosion in RPVs

Corrosion is the result of the interaction of the metal with its environment. Three major types have been identified:

— Corrosion without mechanical loading (uniform / local attack, intergranular attack): Material loss occurs. Metal ions dissolve in liquid electrolyte (anodic dissolution) and hydrogen is produced.
— Corrosion with mechanical loading (stress corrosion cracking, corrosion fatigue) The combined action of a corrosive environment and mechanical loading can cause cracking even when no material degradation would occur under either the chemical or the mechanical conditions alone
— Flow assisted corrosion attack (erosion-corrosion, flow induced corrosion, 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 PWRS

Another type of vessel that is usually subjected to a life assessment for the development of a customized ageing management programme is the pressurizer in pressurized water (PWRs) and heavy water reactors (PHWRs).

5.3.1. Pressurizer Relief Line cracks

During the 13th periodic inspection of Tsuruga Unit 2 in September 2003, a crack was found in the pressurizer relief line nozzle stub weld. The cracks were in the weld metal 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) and the welding metal materials were changed to nickel based alloy 690

The life assessment process for SSCs may vary in accordance with the regulatory practices of each country, with the type of plant and the SSC. The life assessment procedure applied to pressurizers in Korea are shown in figure 29.

In identifying subcomponents within pressurizers of a specific type or class of plant, that should be included in their ageing evaluation, KHNP makes use of the technical information in design guides and design manuals. Next, the operating principles and the functions of each of the relevant subcomponents are identified by reviewing the design specification, the technical specification, the bid specification, the manufacturing drawings and the final safety analysis report (FSAR). In order to identify and classify the degradation mechanisms, the operating and maintenance history is reviewed. Finally, the appropriate technical codes and standards are used to evaluate the lifetime of the sub-structures and components.

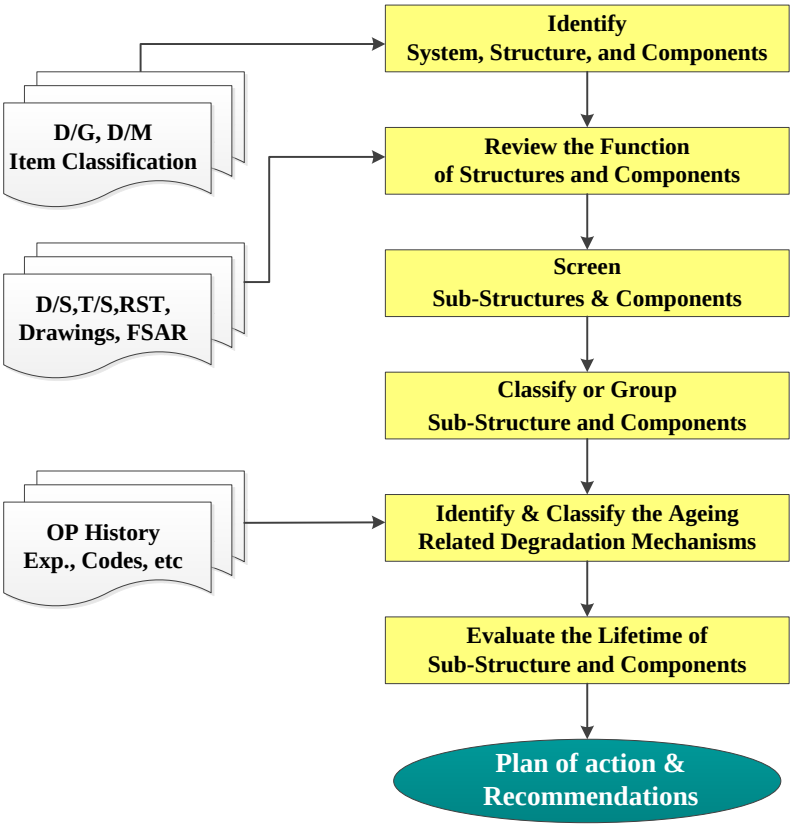


FIG.29. Life Assessment of a Pressurizer by KHNP, Korea

5.4. AGEING MONITORING AND MITIGATION METHODS IN PWRS

5.4.1. Mitigation of Radiation Embrittlement

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

Fuel Management methods of flux reduction:

— Low Leakage Core (LLC) flux reduction can be achieved by implementing a low neutron leakage core (LLC). A LLC is a core that utilizes spent fuel elements on the periphery of the core which reflect neutrons back into the core or absorb them rather than allowing them to bombard the RPV wall.

— Dummy Fuel Elements: A more drastic reduction of neutron flux can be achieved by inserting shielding dummy fuel elements (made from stainless steel) into the periphery of an active core, for example into the corners of the WWER active core hexagons. The use of dummy elements usually results in a significantly different neutron balance in the core. The radial gradient is increased and thus the power distribution is disturbed in such a way 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.

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

Thermal annealing (a maintenance intervention acting on the RPV microstructure):

Once a RPV is degraded by radiation embrittlement (e.g. significant increase in Charpy ductile-brittle transition temperature or reduction of fracture toughness), thermal annealing of the RPV is the

only way to recover the RPV material toughness properties. Thermal annealing is a method by which the RPV (with all internals removed) is heated up to some temperature by use of an external heat source (electrical heaters, hot air), held for a given period and slowly cooled. In the US Codes and Standards that refer to annealing are: 10 CFR 50.61 and 10 CFR 50 Appendices G and H; 10 CFR 50.66 (the thermal annealing rule) addressing the critical engineering and metallurgical aspects of thermal annealing and requiring a thermal annealing report and Regulatory Guide 1.162 describing the format and content of the required report.

In Russia, it was found that an annealing regime with a temperature above 475°C for a hold period of at least 100 hours results in acceptable mechanical property recovery (about 90 %) and a residual embrittlement which does not depend on neutron fluence. Such annealing treatment was implemented in 14 RPVs of the WWER-440 type units.

5.4.2. Mitigation of SCC in CRDM penetrations

The maintenance methods developed to correct the Alloy 600 reactor vessel head penetration cracking problems include:

— stress improvement methods;
— grinding and re-welding 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 more protective and stable oxide coating, which delays initiation of PWSCC;
— surface treatments, such as special grinding techniques, nickel plating, etc.;
— peening with shot or other methods which replaces high tensile residual stresses on the surface with compressive stresses. (Shot peening however 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 the treated surfaces from the PWR coolant attack, stop existing cracks from propagating and fill small cracks.

5.4.3. Vessel Head repairs and replacement:

Repairs of the nozzle penetrations are possible by grinding out the stress corrosion crack and filling the resulting cavity with a suitable weld metal or by inserting a thin liner (tube) of thermally treated Alloy 690 TT 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. [24] There are three types of examinations used during in-service inspection: visual, surface and volumetric. The three types of in-service inspections are carried out always in reference to the pre-service inspection (PSI) that is required for all RPVs. Inspection plans are usually prepared, in which are decided the first and the subsequent in-service inspection intervals.

All shell, head, shell-to-flange, head-to-flange and nozzle-to-vessel welds and 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 (at an interval of 3 to 10 years). Successive inspections also require 100% volumetric examination of all these welds. The nozzle inside surface are all subjected to a volumetric examination during each of the four inspections. Of the external surfaces

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-safe end 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.

All PWR plants in the US have closure bolts in compliance with ASME Section III and are inspected according to ASME Section XI. Regulatory Guide 1.65 provides guidance on vessel closure bolting materials and inspections. Regulatory Guide 1.150 provides guidance on ultrasonic test procedures [25] 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 intervals for the RPV are 4 years (for conventional vessels it is 5 years); however, 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 with acoustic emission monitoring, NDE during outages, a material surveillance programme, loose-parts (noise) monitoring during operation, leak detection during operation and fatigue monitoring. The Code specifies a complete programme including both the utility and regulatory agency inspections. Areas of the RPV that must be inspected include the beltline region of the shell, all the welds, the top and bottom heads, the nozzles, the penetrations, the control rod drive housings, the studs, 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 internal tests must be performed at a RTNDT temperature of + 30°C

For WWER plants the RPV in-service inspection 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), surveillance specimen evaluation and hydraulic testing. Parts and sections of the reactor to be inspected, locations, volume and periodicity are procedurally specified. Mandatory are the examinations of the RPV base and weld metal in the zones with stress concentrations or high neutron flux, 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.

5.4.5. Surveillance Programmes

In the USA fracture toughness requirements are governed by Appendix G of 10 CFR Part 50, which contains requirements for prevention of fracture of the ferritic materials in the primary coolant pressure boundary of US NPPs, with emphasis on the RPV. With the publication of Appendix G of 10 CFR Part 50, "Fracture Toughness Requirements", the USNRC also published Appendix H in the same code, a set of rules for the reactor vessel material surveillance programmes.

In Germany, radiation embrittlement can be neglected when the neutron fluences are lower than 10^{21} n/m^2 (E > 1 MeV). Since the maximum allowed RPV fast neutron fluence in Germany is limited to 1.1 x 10^{23}n/m^2 (E > 1 MeV) and KTA 3203 is valid for this fluence or lower values only, the number of radiation 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 respectively). KTA 3203 allows higher lead factors (>3) on the radiation capsules.

In France the material surveillance programme is specified in RSE-M. It is similar to the US programme. Some changes in the French surveillance programmes (re-arrangement of the capsules, new material in the capsules, laboratory tests, etc.) 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 on whether to plan and proceed with LTO at about the 20 year mark of plant operation.

In Russia the requirements for the WWER material surveillance programmes are given in the following documents:

— Rules for Design and Safe Operation of Components in NPPs, [26]
— Tests and Research Reactors and Power Plant Metallurgy, Moscow - 1973 [27]

— Rules for Design and Safe Operation of Components and Piping in NPPs. PNAE G-7-008-89, Energoatomizdat, Moscow - 1990 [28]

They were applied only to the WWER-440/V-213 and WWER-1000 NPPs. 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 WWER-440/V-213 are supplied with supplementary programmes that contains also specimens from austenitic cladding and IAEA reference steel JRQ. In addition to the specimens for monitoring the radiation damage in the beltline materials, the WWER surveillance programmes includes specimens for monitoring the thermal ageing damage in the pressure vessel materials. Two complete sets of 90 surveillance specimens are located well above the active core (receiving virtually no appreciable neutron flux), in front of the upper (outlet) nozzle ring. These sets are usually removed and tested after 5 and 10 (or 20) years of operation.

5.5. RPV, INTERNALS AND PIPING AGEING DEGRADATION IN BWRS

In order include in ageing management programmes materials typically subjected to ageing degradation phenomena, they are mapped in connection to three main factors, material type/environment/stressors as shown in figure 30, where two or three major degradation mechanisms are shown as acting simultaneously.

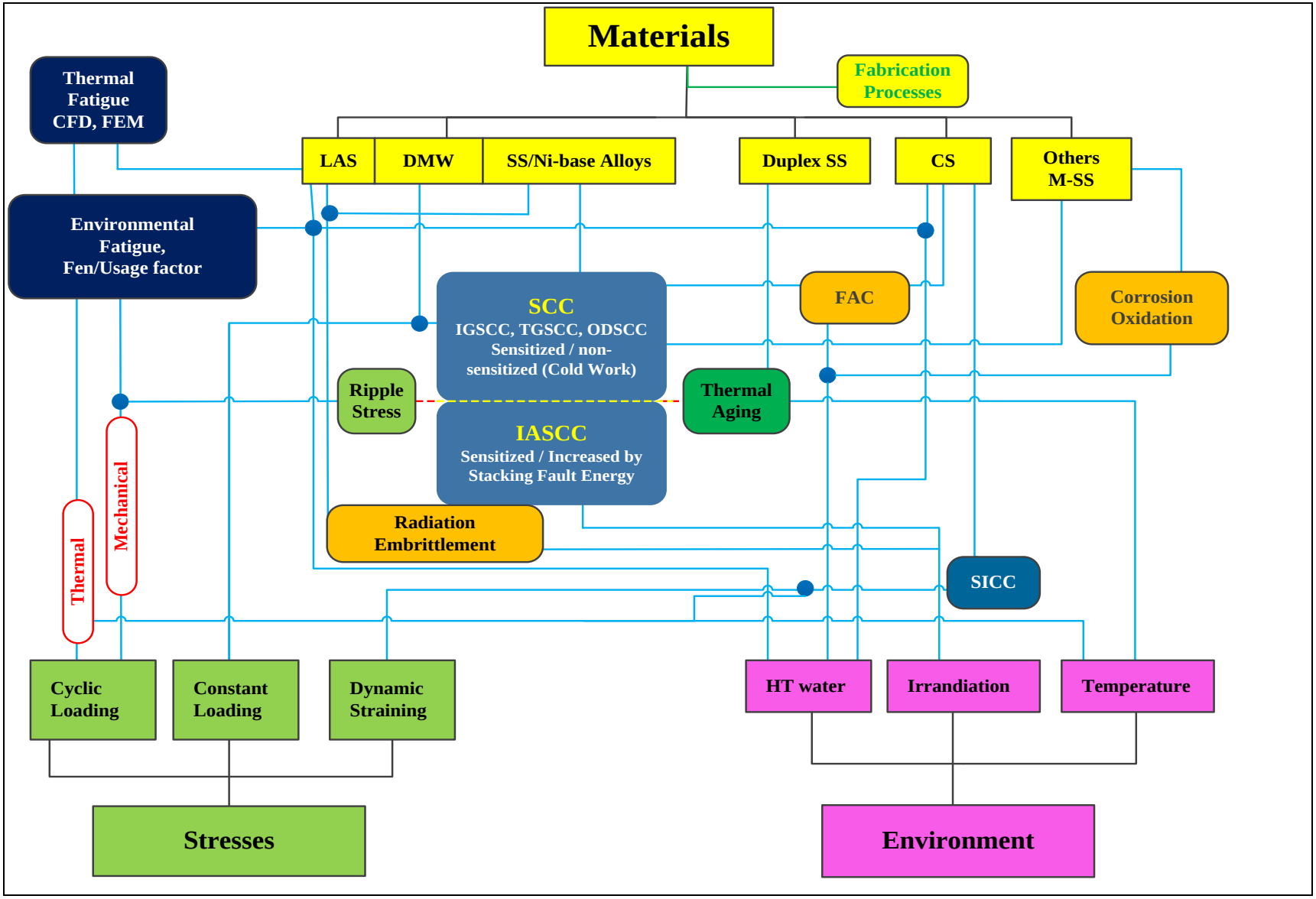


FIG.30. Typical Material Ageing Degradation resulting from a combination of Material types, Environments and Stressors

The yellow boxes in the centre of the diagram represent the screening and examination of the various materials found in BWRs such as low alloy steels (LAS), dissimilar metal weld (DMW) material, Nickel-based stainless steel alloys, duplex stainless steels with mixed microstructure of austenite and ferrite and higher strength, better resistance to pitting, stress corrosion cracking than austenitic stainless steels, carbon steels, martensitic stainless steels and others

In the blue boxes on the left hand side are shown the stressors causing fatigue and the computational codes or calculations required, namely Computational fluid dynamic analysis (CFD) used to determine the constant and cyclic loading functions and time histories and Finite element Analysis (FEM) used in studying 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 time and of the evolutions of the dissolved oxygen (DO) content in time, in accordance with the ASME Section III Pressure Vessel Code for pressure vessels in nuclear power applications. The CUFs are then duly augmented by the Environmental Effect Correction factor (FEN) as a multiplier of the usage factor. For piping calculations are similar but somewhat simpler. Linear models can be used instead of three dimensional or axisymmetric finite element models, except for particular case such as t/3t or 3/5t discontinuities, (where t denotes the material thickness), special piping inserts such as thermal sleeves, special flanged connections, flued heads found in line anchors or floating piping, metal expansion joints, bellows, etc.

In pink are shown the stressors on the materials, namely the two phase flow water/steam and metal interaction, the irradiation and the temperature distribution. In the central part of the diagram are shown the degradation mechanisms resulting from the superimposition of the main factors contributing to the degradation effects, and the degradation type to be expected for example:

— In Carbon steels acting coincidentally are residual stresses from fabrication processes, if applicable, (e.g. cold bending), hydrogen treated water and metal interaction producing flow assisted corrosion, and dynamic straining producing strain induced corrosion cracking (SICC).
— In duplex stainless steels, residual stresses inherited from the fabrication processes (if any) may seemingly be the only main contributor to thermal ageing degradation
— In martensitic and other types of stainless steels contributing to ageing degradation are residual stresses inherited from the fabrication processes, water / coolant and metal interaction producing various types of SSC, namely inter-granular stress corrosion cracking (IGSCC), trans-granular stress corrosion cracking (TGSCC) and for tubular components outside diameter stress corrosion cracking (ODSCC) either sensitized or non-sensitized type, and if in the presence of radiation fields, irradiation-assisted stress corrosion cracking (IASSC) possibly increased by the dislocation of crystals in the metal due to stacking fault energy, if present, caused by layer interruptions in the stacking of the micro structure, as well as other types of corrosion / oxidation.
— In Nickel-based alloys, contributing to ageing degradation are residual stresses inherited from the fabrication processes, radiation, water to metal interaction and constant loading, if present, are the main contributors to ageing degradation which can take the form of radiation embrittlement, and if in contact with the reactor coolant SSC as IGSCC, TGSCC, and for tubular components ODSCC either sensitized or non-sensitized and if in the presence of radiation fields, IASSC possibly increased by crystal dislocation due to stacking fault energy, if present, caused by layer interruptions in the stacking of the metal micro structure.
— In dissimilar weld materials, contributing to ageing degradation are residual stresses inherited from the fabrication processes, constant loading, radiation, if present, causing radiation embrittlement, and if in contact with the reactor coolant also SSC as IGSCC, TGSCC, ODSCC either sensitized or non-sensitized and if in the presence of radiation fields IASSC, possibly increased by crystal dislocation due to stacking fault energy, if present, caused by layer interruptions in the stacking of the metal micro structure.
— In low alloy steels, contributing to ageing degradation are residual stresses inherited from the fabrication processes, if applicable, hydrogen treated water and metal interaction, ripple stresses, radiation, if applicable, causing radiation embrittlement, SSC as IGSCC, TGSCC, and for tubular components ODSCC either sensitized or non-sensitized and if in the presence of radiation fields, IASSC possibly increased by crystal dislocation due to stacking fault energy, if present, caused by layer interruptions in the stacking of the metal micro structure.

For more information, the following reports can be consulted on materials degradation and inspections in BWR RPV, RPV internals, nozzles and piping: [29] [30] [31] [32] [33] [34]

5.5.1. Radiation Embrittlement in RPV and Overlay Clad

Irradiation embrittlement in ferritic materials in BWRs is also one of the key ageing management issues and is addressed by performing a surveillance programme, even though the estimated maximum dose at the end of a BWR vessel's operational life time is lower than that of PWR vessels, due to the larger vessel diameters in BWRs, hence the greater distance between the reactor core and the RPV wall. Fracture toughness and non-destructive radiographic testing (RT-NDT) evaluation provides a quantitative assessment of RPV integrity.

Another concern in BWRs is the weld overlay susceptible of irradiation-related embrittlement. This cladding is typically made of 308 or 309 austenitic stainless steels with some amount of 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 were fabricated using knowledge gained from surveillance programmes and more modern methods such as the use of large ring forgings to reduce the number of welds in the core beltline. IAEA's technical document on BWR Pressure Vessels [30] discusses in detail the effects of neutron irradiation on the mechanical properties of the SS cladded steels and welds of the RPVs of light water cooled and moderated reactors (PWRs, BWRs and WWERs)

5.5.2. IASCC, IGSCC, TGSCC and ODSCC in Core Shroud, Piping and Spent Fuel Pool

BWR water chemistry 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 than in PWR 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 have been experienced in BWRs as early as the 1960's. The most common mitigation processes used individually or in combination to substantially reduce SCC in BWRs are:

— Design changes: Changing the design to implement internal jet pumps for water recirculation where there are no longer any recirculation pipes.
— Material changes: using more IGSCC resistant materials, for example changing the recirculation pipes from 304 stainless steel to a lower carbon content stainless steel as type 316L. 304 steels also showed SCC susceptibility when they are 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 type 308L stainless steel or alloy 82 or noble metal cladding have proven successful as isolation techniques. Also it should be mentioned that hardened surfaces tend to increase susceptibility to SCC as experienced in operation and predicted by the crack tip strain model theory.
— Environment Improvements: Changing water chemistry to reduce the ECP is achieved via an drastic reduction of the oxidants by injecting hydrogen into the feedwater (0.3-2.5 ppm depending on the core loading and the reactor fuel cycle as a function of burnup). This process is known as hydrogen water chemistry (HWC). One of the drawbacks to using hydrogen injection into the feedwater is an increase in the radiation levels in the main steam line caused by the activation of nitrogen as N^{16}. In order to reduce the dose increase in the system, platinum and rhodium are injected into the reactor water (Noble Chemistry Injection or NCI) either on line or during an outage to favour the catalytic recombination of the hydrogen and the added bonus of further reducing ECP. Another method to reduce radiation fields in the system is to clad (welding method) or to coat (plasma spray method) the reactor inner surface with noble metal.

Other interventions to reduce SCC is changing surface finishes, where necessary, to minimize the surface hardening layer and surface roughness by polishing with better methods such as buffing. Also to change tensile residual stresses in hardened surfaces to compressive stresses may be achieved by surface peening (shot peening, water jet peening, laser peening etc.) It is important to note here that

peening remains effective only as long as a compressive stresses are maintained. Otherwise, peening produces a hardened layer which may promote SCC. Hence it is important to ensure that compressive stresses remain such throughout the life time of the plant.

SCC has also been experienced in the Spent Fuel Pool. The fuel structural material consists mainly of Zircaloy and stainless steel Zircaloy. It has proven to be insensitive to any kind of corrosion phenomena (uniform corrosion, stress corrosion cracking (SCC), electro corrosion) in the temperature range <=60°C. Above this temperature residual stresses in welds play a key role in promoting corrosion. Spent Fuel Pools have received attention recently after the Fukushima accident, where it has been assumed that a shortage of cooling water may have produced spent fuel criticality. Consequently, the reliability of the cooling water system and its maintenance have been the subject of concerns since then. Any leaks of coolant as well as the coolant level in the Spent Fuel Pool must be constantly monitored. [35]

5.5.3. SCC in Dissimilar Metal Welds

The reactor pressure vessel base metal in BWRs is normally made of low alloy steel (LAS) internally protected against corrosion by means of a stainless steel cladding. The most common type of cladding is the weld overlay cladding application. This process creates a dissimilar metal weld (DMW) situation at the interface between the stainless steel clad and the LAS 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 DMW region, propagate in the weld and reach the fusion boundary. The safety concern is whether the SCC originated and propagated can penetrate into the LAS base metal. In BWR environment, laboratory data showed the crack penetration even under a constant load crossing the fusion boundary and propagating in LAS where a material's chemical composition such as Sulphur content is important in growth rate.

The first reactor to produce electricity in Japan was a 45 MW prototype boiling water reactor: the Japan Power Demonstration Reactor (JPDR) which reached criticality in 1963 and provided a large amount of information for later commercial reactors. In 1969, the reactor core was remodelled and converted to a 90 MW BWR and continued to run until 1976. It also later provided the test bed for reactor decommissioning. After 13,000 hrs of operation, stress corrosion cracking was found in the stainless steel weld overlay cladding of the JPDR reactor pressure vessel in the surface exposed to the reactor coolant. The corrosion cracks penetrated to the interface between the stainless steel weld overlay clad and the RPV base metal. Detailed analysis revealed that the crack terminated at the fusion boundary, forming a large pit in the low alloy steel (LAS) material, as it tended to relieve the stress /strain concentration at that location. To prevent similar conditions to repeat themselves in the 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 increase resistance to SCC even when a crack reaches the fusion boundary of the weld overlay cladding
— Use of Low Sulphur RPV steels

In most situations in commercial power plants, SCC stops at the fusion boundary where the risk of crack penetration into the LAS base metal is much lower. However, another possibility of crack propagation at the LAS/Ni based weld metal interface should be acknowledged where an SCC susceptible microstructure might form along the DMW fusion boundary. In such cases, a crack may very well propagate along the fusion line to cause a circumferential crack which may cause a sudden rupture of the fusion boundary, when the residual stress field is favourable for such a crack propagation type.

5.5.4. Environmental Fatigue in Nozzle and Piping

Environmental effects on metal fatigue, being time dependent degradation phenomena, have been a concern since significant environmental effects resulted in lab-simulated BWR environments in the mid 1970's conducted in Japan to analyse the JPDR experience discussed above. Intensive

research revealed that the crack terminated at the LAS/Clad interface in the JPDR case, but also revealed the possibility that cracks continue to penetrate into the LAS base material under specific conditions. Since then, extensive efforts have been focused on the environmental effects, resulting in both crack propagation and initiation. ASME Sec. III provides a design curve for fatigue calculations in pressure boundary components. Environmental effects are always present with varying intensities in all materials used in BWR vessels (CS, Stainless Steels, Nickel-based alloys and their weld metals). They affect the S-N curve shown in figure 31, representing fatigue limits in terms of alternating stress amplitude (S) versus number of cycles to failure (N), as demonstrated in extensive laboratory data obtained especially in Japan and in the USA.

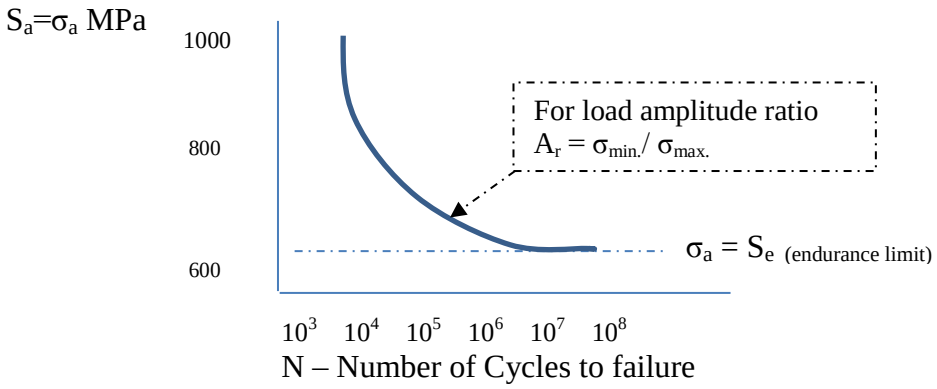


FIG.31. - S-N fatigue limit curve for a specific loading amplitude

Environmental effects can be quantitatively expressed in terms of an environmental Factor F_{en} defined as:

$$F_{en}=da/dN|_{air} / da/dN|_{environment}$$

Original data are available in references [36]. Environmental effects are more evident at lower frequencies or lower strain rates in fatigue, but it is not always simple to exactly define the endurance limit stress, below which fatigue life is infinite, in other words the strain rate or the frequency at which the environmental effect is no longer a factor. However for such materials a working endurance limit may in any case be specified at some large number of cycles. It was also found that the most prominent environmental effects in the S-N curve (F_{en}) is observed in carbon steels. A project attempting to understand the conditions under which fatigue cracks initiate and propagate has been recently included in the Japanese NRA programme. Also it is important to understand how environmental effects on thermal fatigue cause the reduction of fatigue life. In general, it has been found that thermal fatigue damage is limited only to the surface in correspondence to the thermal stress distribution. However, if there are tensile and/or bending stresses in a piping system, small thermal fatigue cracks may grow and cause failure in pipes.

5.5.5. Flow Assisted Corrosion in Carbon Steel Piping

FAC is an ageing degradation phenomenon acting particularly in carbon steel components with high flow velocity. For FAC prone components, periodic in-service inspection (ISI) is required to make sure sufficient pipe thickness above the minimum requirement remains available for the expected service period. Two ISI methods are used, one is fixed points monitoring at each ISI to evaluate the FAC trend and the other is a wide area inspection covering all possible FAC zones in a component, within a certain period of plant operation. Key factor in 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. FAC models and data base have been developed to help with quantitative prediction of FAC rate.

5.5.6. SCC and Fatigue in Pump and Valve

Several SCC/Fatigue in Pumps and Valves are reported in BWR plants. One important incidence is cracking found in valve stems made of high strength martensitic stainless steel. Also pump casings and valve bodies made of cast stainless steels (duplex stainless steels) have been subject to cracking and a concern has been raised in the context of proactive ageing management about whether SCC/Fatigue susceptibility is increased by thermal ageing or not.

5.5.7. Thermal Ageing Embrittlement of Duplex Stainless Steels

Thermal ageing embrittlement may become a material degradation issue in cast stainless steels and austenitic stainless steel weld metal. Both of them contain the delta ferrite phase of austenitic material and thermal ageing takes place mostly in the delta ferrite phase, as the so called spinodal decomposition, which is a kind of kinetic phase decomposition in a nano-scale pattern, forming a very highly Cr-enriched zone. This results in a marked increase in hardness and embrittlement and the component displays a fractural behaviour.

Evidence of this kind of embrittlement is the frequently observed increase in the ductile-brittle transition temperature (DBTT) in Charpy impact tests. Thermal ageing degradation can take place only below 450 °C, above which the ageing produces only a σ phase embrittlement, which is of no concern in LWRs. However, under the BWR normal operating temperature of 288 °C, this ageing phenomenon can be of concern in the long run. Thermal embrittlement degradation can and should be monitored during periodic in-service inspections by measuring the metal hardness on site and determining the degree of embrittlement and its trend. Some 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 the thermo-electromotive force measurements and the calorimetric measurement. More information in section 5.8

5.6. AGEING DEGRADATION IN CANDU TYPE REACTORS

5.6.1. CANDU Reactor assemblies: Calandria, shield tank and End shields

In CANDU reactors ageing 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: the shielding of the reactor faces and the radial shielding of the cylindrically shaped calandria.

— The end shield structures containing a recirculating light water cooling system acting as a heat sink at both reactor ends together with the fuel channel shield plugs provide biological protection at the reactor faces, with the reactor shutdown. The end shield structures are made of two tube sheets supporting the fuel channel end fittings and steel balls filling the interspace between the tube sheets. The total thickness of each of the shielding structures is about one meter. Since the fuel channels penetrate the end shields to allow on-line refuelling, biological shielding from the fuel channels is provided by shielding plugs located between the CANDU fuel and the fuel channel closure plug, as shown in figure 32. End shields and shield plugs together complete the shielding of the 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 achieved by a reactor vault, a steel lined concrete construction, filled with light water, which also serves as calandria support. In CANDU 6 the reactor vault combines thermal shielding and biological radial shielding.

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

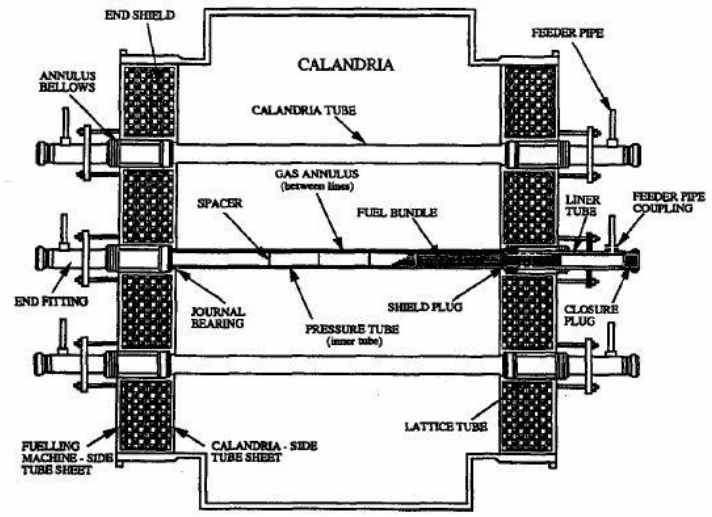


FIG. 32. CANDU reactor assemblies

The end fittings encased in the end shields support the zirconium alloy pressure tubes to which they are connected through rolled joints.

5.6.2. Pressure Tubes – Ageing deformations

Pressure tubes are exposed to temperatures up to 313°C and internal pressures of up to 11 MPa, neutron fluxes of up to 3.7 x 1013 n/cm2/s and fluences of up to 3 x 1022 n/cm2 (in 30 years of operation at 80% capacity). These conditions cause changes in dimensions and material properties through irradiation damage and microstructural evolution.

Pressure tube deformation is a leading ageing factor in operating Heavy Water Reactors (HWRs) In CANDU type reactors, horizontal pressure tubes undergo three types of deformations: diametrical creep leading to flow by-pass of the fuel bundles and the related penalty to critical heat flux, longitudinal creep that may lead to interference of feeder pipes with the fuelling machine, and sagging that may lead to interference with in-core components and potentially also contact between the pressure tube and calandria tube shown in figure 33.

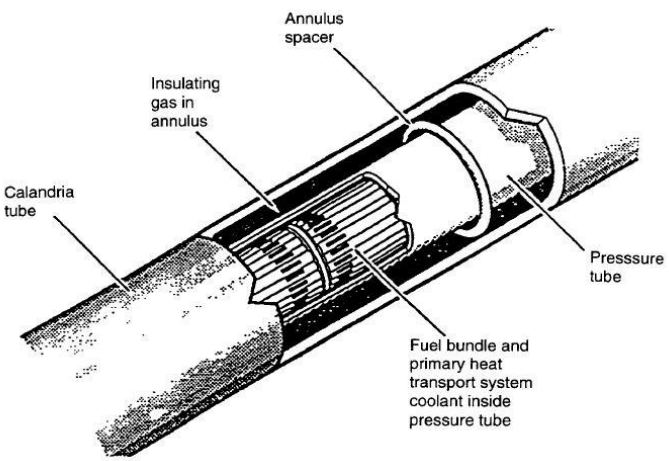


FIG. 33. CANDU reactor: Fuel Channel schematic

Early CANDU reactors did not have deformation allowances for 30 years of operation. The channels associated with these reactors have required an ongoing programme of axial repositioning

and it is expected that many will be replaced before they have operated for 30 years. The channel design in later reactors includes modifications such as longer fuel channel bearings, smaller diameter coils for the annulus spacers and additional spacers associated with each pressure tube, to provide for deformation allowances that can accommodate a 30 year fuel channel design life.

Tube lamination flaws: A few top-of-ingot tubes in early CANDU units have had a significant tensile stress concentration which had escaped detection during manufacture. They contained sharp lamination flaws caused by deep shrinkage cavities. The tube manufacturing process has been changed to minimize the size of the shrinkage cavities in ingots and the ingot tops are now cropped and removed. For tubes now entering service, final inspections employing enhanced techniques ensure that such laminations are detected and the tube rejected.

The presence of hydrides has been associated with crack initiation and propagation in zirconium alloy pressure tubes. Therefore it is imperative that the Hydrogen content in new zirconium alloy tubes and the Deuterium absorbed by the tubes during service be minimized.

Because all crack propagation, and most crack initiation for CANDU pressure tubes has been associated with delayed hydride cracking (DHC), the tube design and its specification have been modified to almost eliminate the probability of DHC occurrence and increase the probability that leak before break (LBB) instead of tube rupture occurs if DHC still happens at operating temperature. Although the overall performance of pressure tubes has been good, some in early CANDU reactors have leaked and two pressure tubes have ruptured, one while at operating conditions, due to delayed hydride cracking (DHC).

5.6.3. Pressure tube geometry monitoring:

The small wall thickness reduction that occurs during reactor operation is of no concern as it is more than compensated for by irradiation strengthening. Diametrical expansion of the pressure tube results in flow bypass that 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 became a concern, fuel management and design changes can be used to mitigate fuel cooling by-pass. 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 multi-tasking tool. Other than diameter and wall thickness,the CIGAR tool is also capable of flaw detection, pressure tube sag measurements, spacer location, inside surface profilometry, video visual inspection, gap measurement between the pressure and calandria tubes 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 many positions along the 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 downwards 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 is achieved through the scanning tool for elongation measurement (SCEM). 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 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 provides lead time for planning corrective action

Spacer movements 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 contact. A Spacer Location and Repositioning (SLAR) System has been developed by the CANDU Owners Group. SLAR has a fast-scan blister detection system that uses 6 line focused

ultrasonic transducers to inspect the outside surface of the bottom 60 degrees 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.

Deuterium and Hydrogen ingress is a deterioration mechanism in pressure tubes that is monitored periodically. The Periodic Inspection Programme specified in CSA Standard N285.4 requires that six pressure tubes be scraped about 10 years after first power generation of the lead unit of a multi-unit power generating station. A sampling tool is available that scrapes thin slivers of metal from operating tubes, leaving a smooth, rounded groove whose depth is much smaller than the tube's corrosion and wear allowance. These samples are analysed for H and D using a specially developed vacuum extraction technique.

Fuel channels in CANDU reactors are equipped with a leak detection system: Dry CO_2 gas flows through the annulus between each pressure tube and its calandria tube. Its moisture content is constantly monitored during reactor operation using very sensitive leak detection equipment. This almost immediately identifies if any pressure tube leakage starts to occur. If a leak were to occur, the channel is inspected and replaced if necessary.

5.6.4. Flow-Accelerated Corrosion (FAC) in feeders

Feeders carry the reactor coolant to and from individual fuel channels. They are part of the pressure boundary and any leak would constitute a small LOCA. A case of flow accelerated corrosion occurred in primary system feeder pipes of early designs made of A106B material (see figure 34). Considerable feeder thinning was observed for the first time at the Point Lepreau reactor in 1995, especially on the outlet elbows close to the exit of the pressure tube. The degradation mechanism identified was determined to be Flow Assisted Corrosion (FAC). This prompted an inspection programme to assess the extent of the problem.

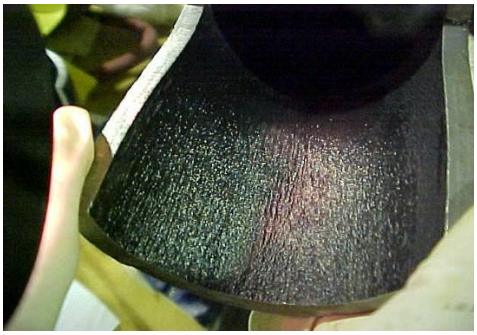


FIG. 34. Flow accelerated corrosion occurred in primary system feeder pipes

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. Advancement in wall thickness measurement techniques were made such as the METAR Crawler developed by Hydro Québec that scans feeder bends and uses 14 Eddy Currents probes and motors to push the bracelet to measure wall thickness 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 water wedge housing.

It was recognized that flow accelerated corrosion (FAC) of the outlet feeder piping was exacerbated by low Cr contents of the steel and by local hydraulic behaviour. In addition, the coolant becoming significantly unsaturated in Fe as it traverses the non-ferrous zirconium alloy pressure tubes and is heated to ~300°C causes it to dissolve more Fe when it reaches the feeder pipes. This, combined with turbulence and high mass transfer as the coolant enters the feeder bends, results in wall-thinning rates at the bends high enough to reduce feeder life. Based on available solubility data for Fe and Fe_3O_4 some reduction in rate can be achieved by tightening the coolant pH from the previously controlled range of 10.2 to 10.8 to the reduced range of ~10.2 to 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 Cr contents of the feeder steel ≥0.3 wt % reduces feeder thinning rates by 50%,. Typically the Cr contents of early CANDU

reactor feeder pipes were very low, <0.02 wt % for instance, which is a consequence of the low Cobalt content originally imposed in order to reduce corrosion product activation.

More recent reactors, and all new CANDU-6 reactors, specify higher Cr in the steel, while remaining within the A106 specification. Given that the feeder pipe FAC mechanism is well understood and predictable, this degradation mechanism can be readily managed for existing and extended operation. [37]

5.7. STEAM GENERATORS IN PWRS AND PHWRS

5.7.1. Steam Generator tubing materials and their degradation mechanisms

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) as well. The ensuing tube damage resulted in substantial economic loss for utilities in the 1980s and premature replacements of SGs in the USA and elsewhere. In the early 80s, the alloy X-750 GT support pins also began to suffer from PWSCC and many had to be replaced. ODSCC of mill annealed alloy 600 steam generator tubing has also continued to the present day and led to many SGs replacements. ODSCC of thermally treated alloy 600 SG tubing has also been observed.

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, on the tube support design and on the tube installation. Initially, in Western countries the SG tubing was made of nickel based Alloy 600. The first German steam generators designed by Siemens developed leaks caused by stress corrosion cracking (SCC) after only two refuelling cycles. They were replaced by SGs with Titanium stabilized Alloy-800NG tubing. Since the early 1970s most German and Canadian SGs have been tubed with Alloy 800 NG (I 800NG) material, and have given good long term performance. Today AREVA NP and Babcock & Wilcox Canada continue to supply steam generators with Alloy-800NG tubing.

Other suppliers followed a different route. In France, Japan, South Korea and in the USA steam generator manufacturers use thermally treated Alloy 690 (Alloy 690TT) tubes. This alloy has more chromium (29.5%) than alloy 600 (15.5%) and proportionally less nickel. It is very resistant to primary water stress corrosion cracking and improved corrosion resistance in secondary side environments. Newly fabricated Alloy 690TT tubes are subjected to thermal treatment at 705°C for 15 hours to relieve fabrication stresses and improve their microstructure. No cracking on Alloy 690TT SG tubes has been reported in operating power plants since their introduction in 1989. [38]

In CANDU practice, the SG tube material used in the 1960's 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 also difficult to inspect with standard eddy current tools. In later units Babcock & Wilcox Canada Ltd. manufactured CANDU steam generators with titanium stabilized nuclear grade Alloy 800NG tubes and a manufacturing method, which precludes any random heat addition to the tubing.

The Russian WWER-440 and WWER-1000 steam generator tubing is made of 08Kh18N10T material which is also a titanium-stabilized austenitic stainless steel.

5.7.2. Degradation mechanisms in Steam Generators

In general degradation in steam generators that may be safety concerns involves mainly SG tubes and divider plates. SG tubes constitute a delicate primary to secondary side boundary, given their reduced thickness. When tubes crack and leak, a loss of coolant event occurs and the secondary side which has no nuclear safety barriers and hence is open to the public at large may become contaminated. It is important to note that there has been practically no damage in the free span sections of SG tubing, where the tubes are in contact with turbulent water. Degradation has occurred in bends and terminals of SG tubing, affecting both the primary and the secondary sides. Problems with SG

tubes have often forced costly unscheduled and extended outages with high personnel radiation exposure and loss of power. Consequently, many SG replacements became necessary.

To partially offset this vulnerability, steam generators are designed with a 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 practiced by inserting a short section of tube (the sleeve) into the tube to cover the degraded zone. The sleeve is then welded to the tube.

A lot of R&D effort was undertaken in the world to improve the SG design and materials as well as the secondary side water chemistry. Although progress has been made, SG degradation remains a continuing challenge.

Understanding degradation mechanisms has been the most important factor in finding solutions. It was found that SG tubing corrosion depended on the interaction of three factors:

— SG and balance of plant system design
— SG and balance of plants materials
— Water chemistry

Flow Assisted Corrosion (at the tube supports): Significant thinning of the tube support plate (TSP) ligaments with drilled hole carbon steel TSPs 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 eggcrates in a different design as the ABB/CE steam generators at San Onofre 2 and 3 and Maine Yankee. FAC was also diagnosed as the main degradation mechanism in moisture separators, feedwater headers, nozzles and water boxes, blowdown headers etc.

For WWER Steam Generators the prevalent degradation mechanism is outer surface stress corrosion cracking, that occurs due the ingress of impurities, mainly iron and copper oxides, through the feedwater flow caused by the degradation of the main condensate system equipment. In fact 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-related problems and associated steam generator problems.

Other degradation factors can be at play in SGs. Among them are:

— 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 (maintenance tools and substances for example),
— impurities in the conditioning chemicals

5.8. REACTOR COOLANT AND INTERCONNECTED SYSTEM PIPING

The reactor coolant and interconnected system designs vary with the technology, the vendor 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 steel. This material displays good higher strength than austenitic steels and resistance to pitting corrosion and stress-corrosion cracking also higher than that of austenitic steels. 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 however are subject to phase precipitation of many compounds at temperature above 300 °C especially in welds if not perfectly executed or in the presence of cold bending and other mechanical stressors in the fabrication processes. A well-known phase separation mechanism with Cr precipitation is that of the spinodal decomposition consisting of Cr migration forming chromium rich and chromium poor segregated ferrite phases, causing the material to become brittle. As aging proceeds beyond 3000 hours of operation at temperatures above 300 °C, a secondary precipitation occurs in the Chromium-rich alpha phase with separation of ferrite and carbide. In PWRs operating temperatures are above 300 in certain areas of the system reaching a maximum of 325 °C. Studies have indicated that Duplex steels generate an electron flux in the presence of a heat flux and a

temperature gradient. This allowed the development of instrumentation capable of measuring the thermal-electric phenomenon (the Seebeck effect) and correlate it to ageing and temperature allowing diagnostics on the state of the unmixing of the ferritic phase and prognostics on the ageing progression and on the remaining life of DSS materials. Other techniques to evaluate ageing effects in DSS have also been developed, based on measurements of the lattice strain of the alpha ferrite. Predictions for the integrity of the DSS components of LWRs are particularly important in the justifications for the extended service life of these reactors up to and beyond 60 years [39] [40] [41]

5.8.1. Thermal Stratification

The stratification of layers of coolant at different temperatures and temperature gradients in piping components, particularly in legs where during normal operation there is no flow, as in branch pipes connected to the main coolant piping, may result in thermal fatigue degradation. The mechanisms that lead to fatigue in nominally stagnant piping systems near the reactor coolant such as the pressurizer surge line is turbulence induced from the reactor coolant system (RCS) line. Hot RCS water is transported into the stagnant and hence colder water. The stratification of the cold and hot fluids results in thermal stresses.

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 institutions such as EPRI and the European Union have funded research to quantify failure margins and predict behaviours in the field. Benchmark studies were conducted using the CFD code FLUENT against experiments, road maps and procedures are available but a consistent methodology has not been reliably mapped so far. Monitoring degradation and using non-destructive inspection is necessary if these cases cannot be designed out.

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, elbows, etc. in the piping at or downstream of the tee. Assessment of the high-cycling fatigue incidence on the piping in order to quantify the failure margins and make reliable life predictions is difficult since conventional thermocouples are useless given the high turbulence and arbitrary nature of the phenomenon. Computation fluid dynamics (CDF) modelling to determine the loadings and advanced fatigue and fracture mechanics analysis guided by experimental evidence is needed to develop a predictive procedure.

5.8.4. Fouling

In general terms, fouling is the accumulation of foreign material on surfaces, such as the inside of pipes, steam generators or heat exchangers. In service-water pipes and in component cooling heat exchangers particularly in once through raw water systems the fouling material is carried by the liquid. It can be organic material such as algae or bacteria, in which case it is referred to as bio-fouling, since it is mainly of the biological sort. Bio-fouling occurs in water systems when organic matter is carried from outside the system, for example from an open body of water through an intake canal or from an open cooling tower.

Bio-fouling is further subdivided into two types: macro-fouling and micro-fouling depending on the foreign material unit size and on the type of wall adherence and 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 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 organic material will accumulate inside system components until fouling plugs pipes, heat exchanger tubes and other tubular components.

Having less scaling or corrosion products in a pipe, gives the fouling material less hold on the inner surface of a pipe. Fouling in steam generators, heat exchangers and steam turbine condensers must 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, must be checked periodically for fouling. Temperature measurements at the inlet and outlet streams can be used to infer the acceptability of the heat exchanger performance for the remainder of the maintenance cycle. This is because the NPP operator is usually required to monitor and demonstrate that the accumulation of fouling in the component cooling heat exchanger is not going to affect the availability of the safety loads for the remainder of the maintenance cycle, which normally coincides with the fuelling cycle.

5.8.5. Water-hammer and steam-hammer

Water hammer is a pressure surge that can arise in any pumping system that undergoes an abrupt change in its flow-rate or change in direction and usually results either from pump starts and stops, or from the opening and closing of valves. Alternatively it can be caused by water column separation and collapse. These abrupt changes can cause all or part of the flowing water column to undergo a 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, another valve downstream, several elbows in between, etc. 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 resulting from the conversion of velocity or momentum energy into pressure energy. Since liquids have a low compressibility, there is little attenuation and the resulting pressure energy tends to be high.

Water hammer may affect piping networks during transients in both conventional and nuclear power plants. Water/steam hammer are usually created by water slugs and steam bubbles entrapped and collapsing in piping containing sub-cooled liquid. Ideally the piping systems should be designed to absorb pressure transients or attenuate them. Water hammer in PWR steam generator sparger feed lines for example has been a problem when void formed during operational transients. Water Hammer has also been experienced in cooling water networks after a system drain. A cooling water system typically reaches all heat exchangers and other heat loads throughout the plant. If void forms at higher elevations and pumps are re-started after maintenance activities, air, vapours and other compressible gas bubbles that gathered at high elevation suddenly collapse and a consequent water/steam hammer event occurs.

5.8.6. High-energy pressure breakdown orifices

High energy multistage orifices are found in minimum recirculation flow lines for costly multi-stage centrifugal pumps with a large number of stages even > 10, and capable of generating high discharge pressures, For this type of pumps, such as the primary coolant circuit charging or make up pumps, manufacturers generally recommend a minimum return flow of 15% or even greater, to protect the pumps when their flow control valves are 100% closed due to operating configuration or even inadvertently due to operating or testing errors, especially during commissioning. Protection is normally provided by a smaller pump by-pass line with a high-energy multi-stage pressure break down orifice, since these devices cannot be inadvertently closed.

Other applications of such high energy pressure break-down orifices in water cooled reactors may be found in the by-pass line to the control valves on the main discharge line of the charging pumps. During the first fill-up of the reactor coolant circuit and interconnected systems these orifices may be subject to extremely high differential pressures between the charging pump discharge and the atmospheric pressure of the system being filled up. Typical issues may include inter-stage cavitation and cavitation in the line downstream of the orifice assembly. Erosion of the walls may ensue. If the

orifice line has a nominal size much smaller than the main line and is connected to the main line by means of weldolets or equal, any vibration may become a problem for the small line.

Another even more destructive issue may be the onset of acoustic resonance. If the orifice emits an acoustic frequency that is too close to the system natural frequency, a pipe break may occur during operation.

High energy multi-stage pressure breakdown orifices can also become a problem as they age if erosion and corrosion affect their geometry and their configuration in the system. Monitoring of high-energy orifices is highly advisable. This can take the form of velocity transducers and of flow or pressure differential measurements. In the beginning these could be kept constantly on-line and when enough confidence is gained tests could be done periodically.

Pump and valve cavitation

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

In a centrifugal pump, to characterize the risk of void creation and cavitation, it is customary to define a parameter called the Net Positive Suction Head (NPSH), the difference between the total head on the suction side of the pump - close to the impeller, and the liquid vapor pressure at the operating temperature. There are two classical ways to look at the NPSH: one is from the system viewpoint, referred to as the NPSHa (Net Positive Suction Head available), the other from the viewpoint of the pump called NPSHr (Net Positive Suction Head required).

The NPSHa of the system at the pump impeller is the sum of the static head of the fluid column including any external pressure above the water column, of the velocity head of the fluid passing through and of the vapour head of the fluid at the operating temperature, which can be expressed as the vapour pressure divided by the specific weight of the fluid and of friction losses.,. The NPSHa is related to the system characteristics and can be changed by changing the system design.

The NPSHr depends on the pump design. Each pump has its own minimum NPSH required. In a centrifugal pump as the impeller curvature accelerates the fluid beyond its normal operating range the fluid can vaporise and cavitate. Cavitation causes damage downstream of the vaporisation point in the pressure recovery section. Somewhere in that section, along the flow path, velocity decreases, as pressure increases and re-crosses the vapour pressure line. At this point the vapour bubble (also called void) collapses,

With the NPSH quantities at hand, the cavitation prevention rule becomes very simple: The NPSHa of the system should always be larger than the NPSHr of the pump with an adequate margin to ensure that cavitation will not occur. The original system design should have been delivered cavitation free, but also any design changes that may be undertaken during construction, commissioning or operation of the system should again be verified for cavitation prevention. Moreover, if inspection or monitoring during operation detects vibration, or other abnormal symptoms, ISI should verify that the system configuration has not changed due to ageing, fouling, clogging or if the operators have not changed the system operating parameters. In principle any change to the parameters defining NPSH may cause a crossing of the cavitation threshold for the pump, which will cause cavitation and induce vibrations and operating abnormalities.

The cavitation mechanism has been categorized into two major types:

— Inertial or transient type where a void is formed and then rapidly collapses which produces the shock wave. The collapsing void usually causes vibration with possible contact between the impeller and static parts such as wear plates or wear rings. Premature failure of seals and bearings can also be expected. Cavitation usually damages the impeller as well causing pieces of metal to break off from the blades and collide with other blades and other components in a chain reaction fashion.
— Non-inertial type where the void does not collapse, but moves and oscillates changing its size. This is caused by the superposition of an energy input as for example of an acoustic field.

As for pumps, cavitation in a valve occurs, 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 way to avoid cavitation in valves is to size them for the proper fluid velocities, typically for velocities less than 15 feet per second. 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 handling for the most part hot and re-heated condensate mixed with preheated make-up streams and taking suction from de-aerators. The NPSHa can also be improved by locating components such as pumps and control valves in the colder return lines of a system or on the colder side of heat exchangers. If in addition seals are damaged and air is entrained into the suction side of the valve or a pump for that matter, air bubbles also form and add to the vibration and to the wear damage to a valve or a pump. Unlike cavitation air entrainment can be quickly eliminated by identifying the source of air ingress and reconstituting the system integrity. This may be counter-intuitive but air ingress can sometimes offer some relief to cavitation if the air bubbles are large enough to cushion the impact of void implosions. Too much air ingress on the other hand can cause priming problems and reduce pump capacity.

Monitoring for signs of cavitation can help saving pumps and control valves. The most evident waning sign of cavitation are vibration and noise. This can be easily achieved by simply using the naked human ear of trained personnel or by using suitable noise level meters. The noise resembles that of small stones going through the system. Usually it increases with time when in addition to void collapsing, material is chipped away. This increases the wall roughness causing higher friction factors between fluid and wall. More sophisticated vibration measuring equipment can also be employed to detect cavitation. These portable devices can connect to the pump bearing housings to detect movement (displacement).

Several measures can be applied on the system design to eliminate cavitation. One way is to increase the gap between the static pressure at the maximum velocity point and the vapour pressure at the operating temperature as for example, if at all possible, by increasing the system operating pressure. This can be done for example by breaking down the pressure drop by using multi-stage control valves, just as is done with orifices in series in a multi-stage orifice assembly and in the case of pumps by increasing the number of blades or by changing their curvature or by increasing the number of stages or by changing the pump altogether for example search for a larger pump or for a pump that runs at a lower speed, which would normally produce a lower NPSHr.

Another way of eliminating cavitation can be achieved by lowering the pump closer to the fluid suction nozzle. All conditions that increase friction losses and therefore reduce the NPSH available should also be removed. For example if

— the suction pipe diameter is too small forcing high acceleration on the suction side of the pump
— the suction pipe is too long or if it has too many changes of direction
the pump runs too fast.

5.8.7.1. Acoustic Vibration

Some boiling water reactors (BWRs) have experienced significant acoustic resonance in their reactor and steam systems. In some cases, acoustic resonance has resulted in damage to plant components in reactor pressure vessel (RPV) and steam lines.

In general all NPP designs have at one point experienced the excitation of acoustic standing waves in closed side branches, e.g., safety relief valves (SRVs), due to vortex shedding generated by 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, the acoustic waves generated in the side branches may cause vibration problems locally and may lead to complications such as valve-seat wear. Therefore, the structural components are subjected to high-cycle fatigue loads, which over time may severely impact their functionality and safety.

As a result, nuclear power plant operators evaluate potential adverse flow effects during initial plant design and startup of new reactors, and following major design changes such as when implementing power uprates of operating reactors. Turbulent steam flow in RPVs can excite large-scale, low-frequency acoustic modes. Steam flow over safety valves (SVs), safety relief valves (SRVs), and other branch lines can couple with high-frequency acoustic mode of branch steam column. Pressure fluctuations with low and high frequencies in RPV and steam lines can impact steam dryer. Acoustic resonance has caused severe vibration of steam line components.

In June 2002, the steam dryer cover plate of a Boiling Water Reactor (BWR) failed after 90 days of power uprate operation (117% original power) at the Quad Cities Nuclear Power Station, Unit 2 (QC-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% original power). The 6x9 inch plate of the steam dryer outer bank was lost in the reactor coolant and steam system; Damage also was found to several steam line components. The Unit 2 steam dryer was instrumented with pressure sensors, strain gages, and accelerometers. The steam lines were instrumented with strain gages to calculate the pressure load on the Unit 1 steam dryer based on the Unit 2 benchmark. The highest acoustic pressure load was found to be 0.65 psi2/Hz (168 dB) on the steam dryer during the Quad Cities unit 2 BWR power uprate operation [42]

In late 2005, the safety relief valves (SRVs) experienced short circuiting at Quad Cities, 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 SRVs. 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 (SV) and SRV branch line length had not been reduced sufficiently to cause acoustic resonance to occur at steam velocity beyond the operating conditions. To address the issue, the owner-operator installed a T-connection and dead leg pipe in the SV and SRV branch lines to effectively lengthen the branch line and cause acoustic resonance to occur at lower steam velocity with reduced pressure fluctuations. Upon reactor restart, measurements indicated that the sources of acoustic resonance had been eliminated.

NRC Regulatory Guide 1.20 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. [43], 44], [45]

5.9. EMERGENCY FEEDWATER SYSTEM

In PWRs the Emergency Feedwater System also called Auxiliary Feedwater System is used to supply makeup water to the steam generators when operated as heat sinks to remove decay heat, when the reactor is shutdown. The system is equipped with a number of throttling valves to adjust the flow to the steam generators during refuelling and reactor start-up and to isolate the pumps and other components during maintenance. This system suffered from piping wall thinning in the US and low flow operation and flow accelerated corrosion were determined to be the root cause. Through-the-wall erosion was found at the inlet end of the cross-over A106-B pipe and at the tee. Probable cause was postulated to be the high crossover piping velocity of 11m/s at 245°C/4.74 MPa and pH = 9.4 on morpholine water treatment. It is believed that much of the damage may have occurred during early operation at reduced pH. Under high pH conditions, erosion is greatly reduced. As a corrective action the EWS/RHC headers and their thermal sleeves were replaced by more erosion resistant components (SA-335, 1-1/4 Cr material).

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

5.10. BURIED PIPING

Underground or buried piping and tanks are used in both safety and non-safety applications in a nuclear power plant. They may contain liquids or gases.

Buried piping or tanks are below grade and in direct contact with the soil. In contrast to buried piping, underground piping and tanks even though below grade as well, are not kept in direct soil contact; for example piping contained in a concrete duct.

Recent regulatory action and public concern in some jurisdictions has been associated with leaks of radioactive or environmentally deleterious material from buried piping systems at NPPs. Although releases are typically within prescribed limits, such releases can impact negatively on a NPP's reputation as an environmental steward. Additionally, leaks from piping can cause economic loss to plants as they may require a unit outage or other compensatory measures until the affected system are repaired.

There are differences between underground piping in an NPP and those in conventional industries such as oil and gas, or municipal water systems. Pipelines used in conventional industry are typically straight, long, and electrically isolated, with much simpler infrastructure as shown on figure 35. In contrast, at NPPs underground pipes are made of a variety of materials. They are often shorter in length, come in a far greater range of diameters with overcrowded infrastructures. In addition they are electrically interconnected and grounded for safety reasons. Due to the distinct design complexities of underground or buried piping in nuclear power plants, in order to control ageing mechanisms special considerations must be given to the piping material conditions and coatings [46] as well as to the necessity of scheduling periodic inspections and of providing suitable inspection techniques.

FIG.35. Sample Guided Wave Collar.

Due to the expense of digging up buried piping for inspections, often orders of magnitude greater than the direct inspection cost, NPPs often take a risk informed approach to selecting inspection locations, and utilize available commercial software tools for assistance in selecting such locations. There is an increasing trend to leaving diagnostic equipment, such as guided wave collars, installed underground with test leads going to the surface to minimize repeat costs of digging over the operating life of a NPP.

5.11. SECONDARY SIDE PIPING IN PWRS - FLOW-ACCELERATED CORROSION

Flow Accelerated Corrosion (FAC), also called erosion-corrosion (EC) is a chemical corrosion mechanism affecting metals such as carbon steels, which owe their corrosion resistance to the formation of oxide layers. If a moving fluid is in contact with such a metal as in a piping system, the oxide layer protecting the metal surface may be displaced and dissolved in the fluid. It is the constant repetition of two processes, namely the continuing creation and dissolution of the surface oxide layer that gives rise to the metal wall-thinning, which may ultimately cause perforations and in the case of pipes or even pipe ruptures. The corrosion rate is proportional to the solubility of the oxide layer, to the Cr, Mo and Cu content of the material, to, the operating temperature, the flow velocity, the surface roughness, the flow velocity and the pressure profile of the fluid. In addition, FAC may be enhanced by the impingement of liquid droplets and by cavitation phenomena.

Carbon steel components in the secondary side, such as feedwater heaters, condensers, valve bodies, and piping have experienced severe wall thinning in NPPs. Nuclear regulatory bodies of

various countries require utilities to provide wall-thinning management programmes for carbon steel components.

The Mihama incident August 2004 in Japan occurred in a 22 in. carbon steel straight run of condensate pipe, one foot downstream of a flow orifice. See figure 36.

FIG. 36 (Mihama (Japan)) A 22 inch condensate line downstream of an orifice ruptured catastrophically

Another similar event occurred in the Surry plant in 1986, where 4 workers were killed by a piping rupture in a condensate line. See figure 37.

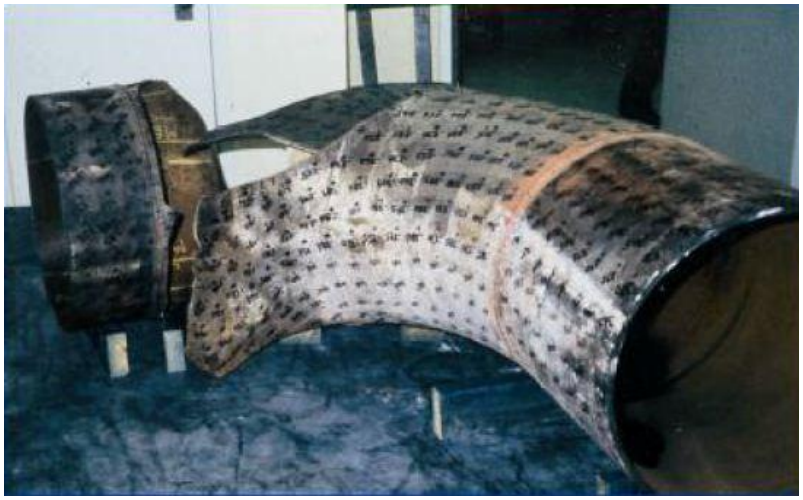


FIG.37 (Surry Unit 2 (USA)) An 18 inch elbow in a condensate line ruptured catastrophically

The two accidents are quite similar. In the period between 1986-2002 in the Russian Federation (RF) and in the Ukraine 45 cases of operational failures in NPPs were the result of FAC in pipelines:

— WWER-440 NPPs - 19 failures,
— WWER-100 NPPs - 17 failures,
— RBMK NPPs - 7 failures.

Many operating plants conducted reviews of their high energy piping systems. At the Loviisa NPP comprising two VVER 440 units dating back to the late '70s and early '80s, the FAC programme review was conducted after the Surry accident in 1986. to incorporate the immediate feedback from that events. Since the review was limited to feedback from the Surrey accident, flow control orifices were not included in the review programme and a guillotine break of the feed water system piping

occurred in 1990 in Unit 1. A second guillotine break of the feed water system piping occurred in 1993 in Unit 2. In order to reduce EC of the feed water system piping, the secondary water chemistry was changed from neutral to alkaline. Also, a change from carbon steel to low alloy or stainless steel piping was carried out. Changing piping materials and water chemistry, coupled with on-line monitoring has improved resistance to FAC. In order to improve the inspection programme, it was recognized that specialized computer software was also going to be necessary.

Quality standards today recommend that all plants should have a flow-accelerated corrosion (FAC) programme covering all high-energy secondary side systems. Inaugural inspection and periodic inspections are mandatory in CSA N285.4 and N285.5.

EPRI first began research into FAC in response to the Surrey incident. 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 plant life management of mechanical components in power plants. As in the case of CHECKWORKS the degradation analysis functions include lifetime prediction in respect to flow accelerated corrosion (FAC), droplet impingement corrosion, cavitation corrosion, etc.

In Japan, utilities studied the wall thinning phenomenon using the available large amount of measurement data on feedwater and steam systems. One of those utilities, Kansai Electric reviewed data taken at about 30,000 components in its nine PWR plants. Upon completing a statistical evaluation of the data produced by Kansai Electric and other utilities in similar reviews, the utilities established guidelines in 1990, and issued the “Guideline for Management of Pipe Wall Thinning in PWR Secondary Systems” document, stipulating the inspection method, the inspection targets, their frequency, the acceptance criteria, corrective actions (e.g. pipe replacement) and so on. Wall thickness is normally measured by means of using ultrasonic test (UT) equipment. Pipes found to have a remaining service life of two years or less, require corrective action

5.12. CABLES AND I&C SYSTEMS

Cable ageing and the need for condition monitoring (CM) is an important aspect of plant life management. Cables, especially their insulation and jacket material are vulnerable to ageing degradation during normal operation and means must be established to ensure that cable ageing does not lead to unsafe operation. Cable fires, moisture intrusion, loss of functionality, and sensitivity to noise/interference are examples of cable issues giving rise to current and future concerns about long-term performance of cables in nuclear facilities.

Cables in nuclear power plants can typically be grouped into the following functional types:

— Instrumentation and control cables (coaxial, triaxial, twisted, shielded)
— Low and medium voltage power cables (less than 1kV)
— Specialty cables (such as articulating cables in moving equipment close to the reactor or stationary cable still subjected to high radiation levels)
— General services cables (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 low voltage power cable are (see figure 38):

— Conductors, insulation, shielding, jacket.

Other components which may be present in a cable include:

— Filler or bedding materials, which occupy the gaps between insulated conductors in multi-conductor or multi-core cables, which improves the mechanical stability of the cable
— Tape wraps, which may provide additional electrical, mechanical, fire protection, or identification of conductor grouping
— Armouring layers, below the outer jacket, which are sometimes used for mechanical protection

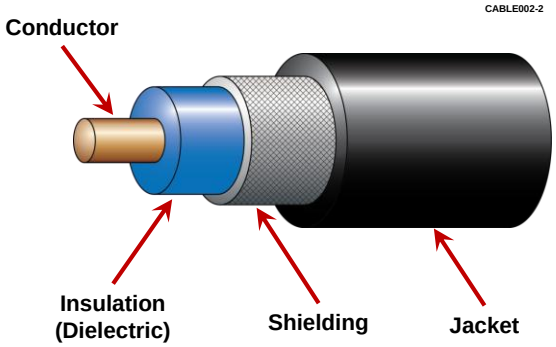


FIG.37. Schematic of a typical cable

Instrumentation cables (including thermocouple extension wires) normally carry low voltage and low current either analogue or digital signals used by instruments. Resistance temperature detectors (RTDs), pressure transducers, and thermocouples are normally composed 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 standards writing organizations such as the International Electro-technical Commission or IEC to understand the effect of ageing on cables and to establish effective and reliable means for cable qualification and condition monitoring.

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

— Reduction in the insulation resistance (IR) in elevated temperatures (above 70° C) and ageing.
— Gassing and disposition of these gaseous products on surrounding surfaces
— Formation of conductive salts on the surface of the insulated conductors

In order to guard against the adverse consequences of cable ageing and degradation, a cable ageing management programme 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 measure the health of cables.

5.12.1. Cable testing

The nuclear power industry has recognized that there are limitations in cable qualification testing as pre-ageing and the use of models such as the Arrhenius law for assessing qualified life. As such, periodic cable testing in nuclear power plants is recognized as vital for troubleshooting and identifying 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.

A wide range of cable testing methods have been developed for cable ageing management. Available cable-testing techniques have been used to test all parts of a cable (insulation, jacket, and conductor) as well as connections, penetrations, splices, and terminations. In the broadest terms, there are 4 types of testing techniques (visual/tactile, electrical, mechanical, chemical) for cable condition monitoring, management, and troubleshooting as shown in Table 3

TABLE 3 CABLE CONDITION TESTING TECHNIQUES

No	Technique	Monitoring Type	Description	Destructive/ Intrusive	Features	Remarks/ Applicable Materials
1	Walkdowns	Visual and Tactile	Capable of identifying cable conditions, by appropriate manipulation and observation. This procedure should be considered for more 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.
2	Thermal Imaging/ Thermograph	Visual and Tactile	A measurement used during walkdowns for identifying environmental hot-spots and unanticipated operating conditions.	No/ No	The best supplement the plant walkdowns 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.	The technique has significant limitations for use as a diagnostic tool since internal hot-spots are not detectable and it cannot be used for locating external hotspots during the plant full power operation / All materials.
3	EAB	Mechanical	The benchmark property of the cable by tensile testing.	Yes/ Yes	Although this method is impractical as a routine CM method, it generates optimal data for cable condition assessment and is particularly useful where cable samples have been placed in a sample deposit, specifically for CM.	50% absolute EAB normally defines the end-of-life condition based on a conservative estimate of aged cable surviving a DBA / All materials.
4	Indenter Modulus (IM)	Mechanical	A measurement carried out by clamping he cable jacket or insulation while the probe penetrates the surface of the test	No/ Mainly No	This technique only provides information on the cable condition at the location being tested and could be used on operational	A probe with known dimensions should be driven through the surface of a sample at a

No	Technique	Monitoring Type	Description	Destructive/ Intrusive	Features	Remarks/ Applicable Materials
			material a few hundred microns. The IM 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.		cables.	carefully controlled speed. / PVC, CSPE, EPR, EPDM.
5	Recovery Time	Mechanical	A 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	The recovery time measurement gets the higher sensitivity and stronger correlation with elongation than that of the IM.	A parameter that has proven very useful to assess the degradation of cables. / All materials.
6	Frequency domain reflectometry (FDR)	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 as long as the cable under test is shorter than the FDR 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.

No	Technique	Monitoring Type	Description	Destructive/ Intrusive	Features	Remarks/ Applicable Materials
7	Time domain reflectometry (TDR)	Electrical	A technique based on the transmission line theory just as FDR. It involves sending a DC 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	The technique allows cable diagnostic by providing information about the conductor and any connector or connection in the circuit, and to a lesser extent, about the cable insulation material.	It allows diagnostics about a device at the end of the cable, such as a RTD or thermocouple. The test depends largely on comparisons with a baseline TDR. / All materials.
8	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, of the insulating material, and of the end device.	No/ Disconnection Needed	The measurement 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.
9	Insulation resistance (IR)	Electrical	Measurements are made using an IR instrument at specific voltages to validate the cable insulating material characteristics.	No/ Disconnection needed	The measurement for detecting expected IR 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. Furthermore, the current in the cable is limited to avoid cable damage. /

No	Technique	Monitoring Type	Description	Destructive/ Intrusive	Features	Remarks/ Applicable Materials
						All materials.
10	Tan delta	Electrical	Monitoring integrity of the cable insulation by measuring the tangent angle (loss) between the resistive current and the capacitive current under AC voltage.	No/ Disconnection needed	This measurement is used to detect resistance current increases due to impurities or inclusions in the insulation.	Methods include using a fixed frequency such as 60Hz and stepping the AC voltage up to 1.2 times the rated cable voltage or applying a low voltage with a variable frequency up to 20kHz. / All materials.
11	Density	Chemical	The measurement of density of polymer samples using the Archimedes principle that 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 the equilibrium is reached).	No/ Micro-sampling required	The measurements results are known to correlate with EAB.	The higher the level of ageing, the greater the concentration of oxidation products, the higher the density. / XLPE, other thermoplastics and some elastomers EPR.

No	Technique	Monitoring Type	Description	Destructive/ Intrusive	Features	Remarks/ Applicable Materials
12	Ultrasonic Velocity (UV)	Chemical	The 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	This technique is designed to measure the properties of the cable jacket over a small volume between the transducer probes. This technique is still under development.	The measurements obtained can be strongly dependent on the cable construction and on the specific formulation of the jacket material. Extensive baseline data may be required. / PVC, PE, EPR.
13	OIT/OITP	Chemical	Measures the loss of anti-oxidants 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 oxydizing environment.	No/ Micro-sampling required	The method has been standardized for use in CM and has been shown to correlate well with degradation of certain polymers (e.g, OIT for polyethylene and ethylene propylene rubbers and also CSPE and PCP).	Degradation products from halogenated materials can damage expensive calorimeter cells, and continued multiple testing on this kind of material is impractical, unless specific non-corrodible cells are used. /EPR, PE, XLPE, PVC.
14	TGA	Chemical	Measures changes in weight relative to changes in temperature.	No/ Micro-sampling required	The measurement is particularly useful in diagnostics of cable insulation ageing in that it allows mapping of the degradation temperatures. The sample preparation is similar to that in OIT/OITP testing. Usually carried out on samples that involve corrosive degradation products as	The reference temperature is determined by 1 of 2 methods: Determining the temperature at which 5% mass loss occurs or the temperature corresponding to the point at which the

No	Technique	Monitoring Type	Description	Destructive/ Intrusive	Features	Remarks/ Applicable Materials
					the sample chambers are chemically far more robust than those used in DSC.	maximum rate of weight loss occurs. /CSPE, PCP, PTFE, PVC, EPR.
15	Fourier Transform Infrared (FTIR) Spectroscopy / Near infrared reflectance (NIR)	Chemical	Polymers change their light absorption characteristics due to the formation of new chemical bonds as they degrade. The measurement of 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 infra-red analysis in reflectance mode is carried out based on detecting the development of ageing induced infra-red 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.	/ The measurement is not applicable to polymers that contain heavily absorbent materials such as carbon black, e.g CSPE and PCP./FTIR:PVC,XLPE, EPR,PE NIR: EPR, PE.

5.12.2. 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 change with ageing should be determined and used to define acceptance criteria. For example cable elongation, which is the measurement of how far the material will stretch before breaking and which diminishes with ageing, can be used to set an acceptance criteria, such as a certain amount of minimum elongation expressed in percentage in relation to the years of continuous operation. An acceptance criterion of 50% of the ultimate elongation has been used for some materials. Elongation is an example of a property that can be easily measured and correlated to the insulation resistance to ageing stressors and mechanical loading such as vibration, bending or other movements. Elongation criteria effectively provide a quantifiable margin from unacceptable events, such as cable cracking on bending.

The IAEA published a report [47] in 2012 containing information on the state-of-the-art in cable qualification, condition monitoring, and ageing management. Both the Institute of Electrical and Electronics Engineers (IEEE) and the International Electrotechnical Commission (IEC), the international standards and conformity assessment body for all fields of electrotechnology have developed many standards which provide guidance on different methodologies for cable testing and diagnostics. Standard IEC 62465 is on cable testing and standard IEC 62582 addresses specific test methods for cable condition monitoring. IEC 62465 covers cable conductors while IEC 62582 addresses cable insulation and jacket material.

IEEE provides standards for qualification of equipment and cables such as IEEE 323 and IEEE 383 used for Class 1E safety functions. Any I&C cable installed in a nuclear facility for the purpose of supporting a safety function must be qualified to perform not only during normal service conditions but also during and after a design basis event (DBE).

5.13. CONCRETE AND NON-METALLIC AMP(S)

5.13.1. Containment Designs

The containment system is primarily designed to contain any radioactive material that may be released from the primary system in case of an 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 in western NPPs 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 up to 25 mm thick, joined by welding, and anchored to the concrete wall by studs, structural steel, or other steel products.

Containments usually have a large, thick base mat that provides seismic capability. Base mats also support the structures, and may help contain molten material during a severe accident.

The technical definition of NPP 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 must also include an evaluation of these penetrations.

Nuclear power reactors are usually designed, fabricated, erected, constructed, tested, and inspected according to the requirements of the ASME Boiler and Pressure Vessel Code

Containment leakage rates are determined in the safety analysis report (SAR) and in the plant Technical Specifications or equivalent. Leakage rates are expressed in terms of a percentage of the total containment atmosphere mass (or weight) leaked over a 24 hour period.

5.13.2. 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. A comprehensive understanding of

concrete structures ageing, of the degradation mechanisms, and of their impact on the structure 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 where repairs would be costly, the monitoring of the environment or of potential stressors that could lead to degradation are mandated.

5.13.3. Ageing mechanisms for concrete structures

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

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

— Salt Crystallization is produced by repeated wetting and drying cycles.
— Freezing and thawing attack in concrete in saturated conditions
— Abrasion/Erosion/Cavitation: Progressive loss of material at concrete surfaces
— Thermal Exposure/Thermal Cycling: Beyond 350°C there can be a rapid decrease in strength.
— Irradiation: Direct neutron and gamma rays
— Fatigue and vibration
— Chemical attack including efflorescence, leaching, sulphate attack, ettringite formation

The relationship between the root causes of concrete degradation and types of cracks before and after hardening is shown in figure 39. Primary causes are on the left (a) and the relationships between some of the main causes and degradation types are reported in the table on the right (b)

Primary causes are grouped in table (b) into two categories: cracks due to intrinsic (rheological) concrete properties and cracks caused by external loading. [48] [49]

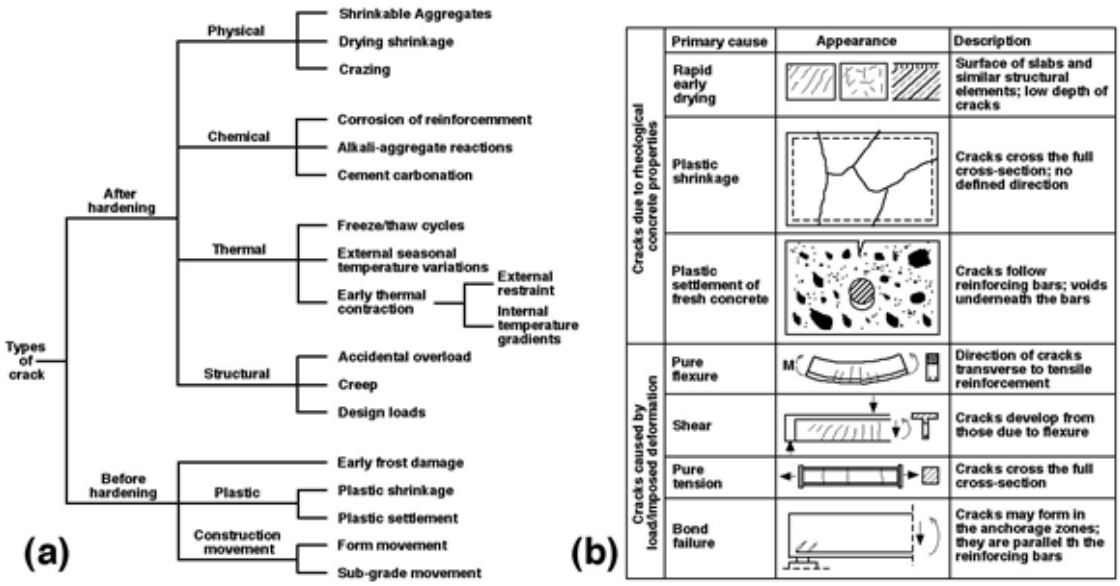


FIG.39 Relationship between primary causes and types of cracks in concrete

It is interesting to note that degradation may not be always due to ageing. Cracks due to the rheological properties of reinforced concrete are exclusively due to the hardening process or to the intrinsic discontinuities of the aggregates and the presence of reinforcing steel bars. Cracks caused by the material properties are of varying nature. Some may be confined to the surface region, others may follow the reinforcing bar pattern, and others are influenced by the aggregates.

5.13.4. Inspection and monitoring of concrete and metallic structures

Concrete is a durable material and its performance in NPPs has been good. However, experience shows that accelerated degradation of concrete may occur with poor quality construction, aggressive environments, excessive operational loads, extreme accident conditions. An impaired concrete structure increases 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 plant life. The scope of an ageing management programme for concrete structures should cover potential ageing mechanisms, age-related degradation, inspection, monitoring, as well as condition assessments and remedial actions not just for the concrete for interconnected structures such as mild steel reinforcing, pre-stressing systems where applicable, penetrations, liners (metallic and non-metallic), anchors, 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 over a wider area. [50] [51]

The methods used to assess concrete conditions vary in relation to the concrete property to be assessed:

— Air Permeability to check air content;
— Audio Methods to check cracking;
— Break-Off Methods to check compressive strength;
— Carbonation Depth (D) 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, voids;
— Infrared Thermography to assess delamination Instrumentation to assess, cracking, creep, 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, Frost Damage, Honeycomb;
— Probe Penetration to assess compressive strength
— Pull-out Testing to assess compressive stress and pull-out strength;
— Radar to assess voids, concrete cover, embedded parts;
— Radiation/nuclear to assess Cracking , Creep, Delamination, Density, Embedded Parts , Honeycomb, soundness, void;
— Rebound Hammer to verify compressive stress, uniformity;
— Stress Wave Transmission to check voids, cracking, delamination;
— Tomography to check cracking, delamination, voids;
— Ultrasonic Pulse Velocity to verify compressive strength, cracking , delamination , honeycomb, modulus of elasticity, voids;
— Visual Inspection to assess bleeding channels, chemical composition, Corrosive Environment, Cracking, Delamination, honeycomb, structural performance, aggregate quality, uniformity, voids

In pre-stressed systems, primary ageing mechanisms are concrete shrinkage and creep, and 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 re-tensioning. For grouted pre-stressing systems, cementitious grout provides excellent corrosion protection for the tendons when used properly. Additional information on inspection of post-tensioning systems is available in Subsection IWL of Section XI of the ASME Boiler and Pressure Vessel Code. Tests on non-grouted tendons include:

— Pre-stressing force with time to be determined at regular intervals
— Mechanical tests on tendon materials
— Tests on corrosion protection
— Visual inspection

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

— Liners and liner coatings: Containment leak tightness is verified periodically through leakage rate testing. Ultrasonic, magnetic particle, liquid penetrant or electromagnetic can be used. 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 in-service inspection programmes
— Structural anchorage
— Foundations, Piles, and Underground Structures

Other structures in a nuclear power plant 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.

In conclusion, concrete ageing has been identified as one of the most important issues in relation to plant life management and life extension. More information on how concrete structures may be managed for long-life operation and what must be considered to ensure that they are reliable throughout their service life can be found in IAEA's technical document on concrete containment [50]

5.14. METALLIC STRUCTURES

Accessible metallic structures and components inspections and assessments usually 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 COMPONENTS AND PROTECTIVE MEDIA PROPERTIES OR CHARACTERISTICS

Evaluation Method / Property or Characteristic	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 (P)				X									
Coating Bond Performance (P)		X											X
Broken Wires (T)													X
Coating Thickness (P)	X												
Weld Cracks (L)							X		X				X
Coating Distress (P)													X
Elongation (T)											X		
Free Water Quantity (P)				X									
Leakage (L)								X					X
pH Value (P)				X									

Evaluation Method / Property or Characteristic	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
Prestressing Force Loss (T)						X							
Reinforcement Corrosion (R)			X		X					X			X
Structural |Degradation (L, T)													X
Surface Cracks (L, P)							X		X				X
Ultimate Strength (T)											X		
Wall Thickness (L)										X		X	
Yield Strength (T)											X		

Metallic or non-metallic liners are attached on the inside surface of many single-wall containments to provide a barrier against gas leakage. Metallic liners consist of up to 25 mm thick carbon steel plate, joined by welding, and anchored to the concrete wall by studs, structural steel shapes, or other steel products. PWR containments and dry-well portions of BWR containments are typically lined with carbon steel plate. Liners of wet-wells of BWR containments, and of the fuel pool structures in most light-water reactors (LWRs) consist instead of stainless steel plates. The liner is not considered to contribute to the strength of the structure it protects.

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

Pipe penetrations may be either rigidly attached to containment or a bellows may be provided as pressure boundary, with the pipe being allowed to move relative to the containment wall.

— 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 (flued) head or a flat plate welded to the sleeve and to the penetrating pipe spool.
— Flexible containment piping penetration are provided with bellows (figure. 40). They are typically constructed of stainless steel and are primarily used in steel containments, with only limited use in reinforced and pre-stressed concrete containments. The bellows minimize the loading imposed on the containment caused by differential movements between the steel containment and the pipe. [52]

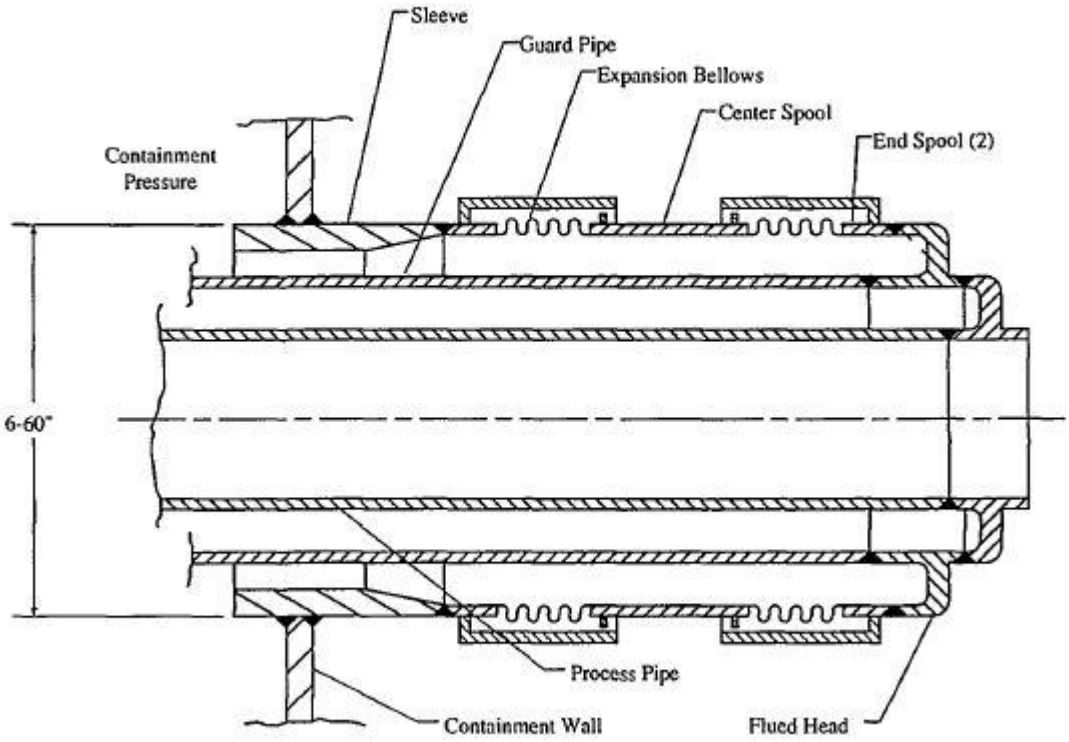


FIG.40. Typical Containment Piping Penetration Bellows

A Spent Fuel Pool (SFP) is usually a multi-storey reinforced concrete structure supported by bearing walls and/or by a base mat. They are used to store used fuel assemblies, fuel bundles, or control and/or adjuster rods for a period of time until they are sufficiently cool to be transferred to dry cask or other storage. See figure 41 for cross section pf SFP building. 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.

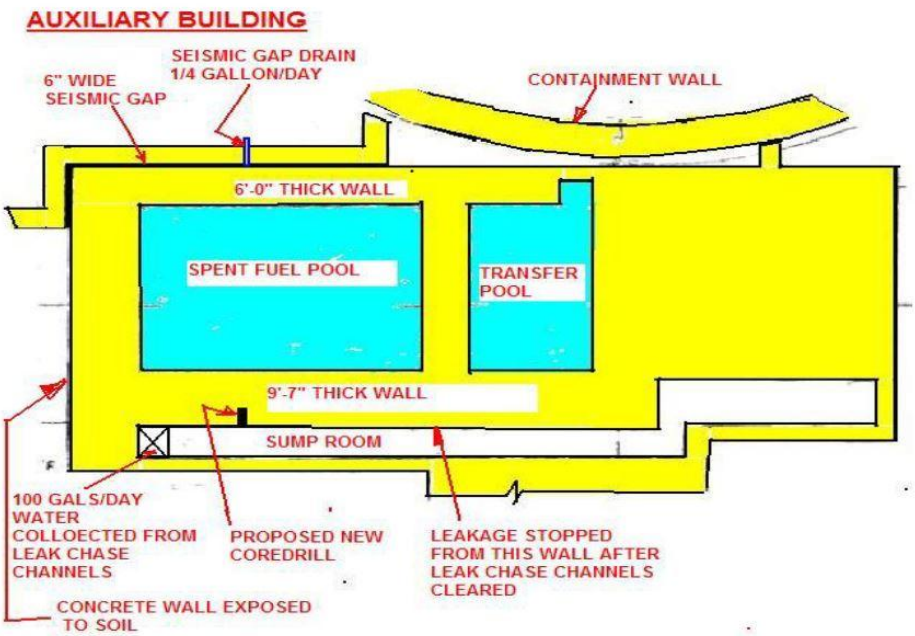


FIG.41. Typical cross section schematic of a Spent Fuel Pool

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 I structures and are designed to withstand natural phenomena (e.g. earthquakes and tornadoes). See figure 42.

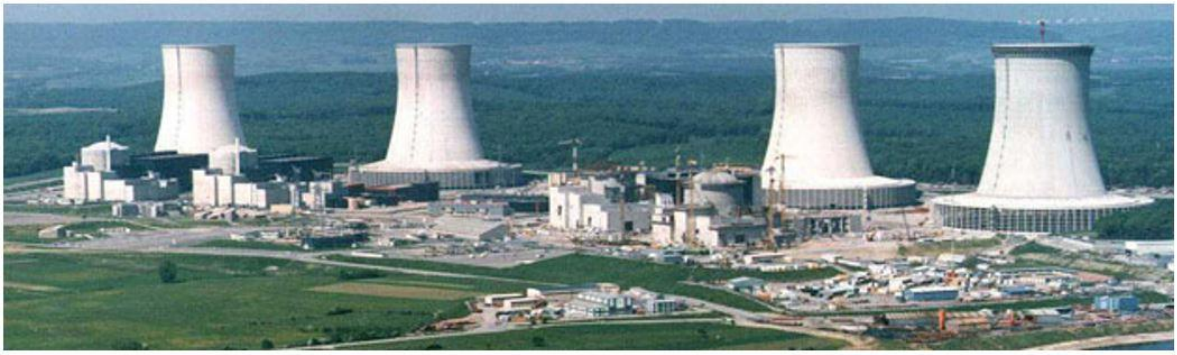


FIG.42. Cooling towers in nuclear power plants

During operation, the concrete inside a mechanical draft cooling tower (MDCT) shell is subjected to temperatures of about 40°C and to 100% relative humidity. “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 structure. Water intake structures are 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.

Concrete anchors or screws are required for heavy machinery, structural members, piping, ductwork, cable trays, towers, and many other types of structures and components in NPPs. Anchorage requirements may relate to load capacity, resistance to vibration, preload retention, temperature range, corrosion resistance, post-installation or pre-installation, ease of installation, ease of inspection and stiffness as required in the German nuclear safety standard KTA 2201 series on seismic designs complemented by the design and construction guidelines of the Deutsches Institut für Bau Technik (DIBt). In the U.S. the requirements on concrete anchors for NPPs are those of the US NRC regulatory guide for the design of anchors and structural supports [53] complemented by the code requirements of the American Concrete Institute [54].

Embedded anchors are preferred wherever possible. Where embedded anchors have not been provided during the pouring of concrete (for example in case of modifications, or in case of repositioning of an anchor point following installation in a misplaced position) concrete perforation type anchor systems can be used.

Concrete anchors therefore can be subdivided into two primary types: mechanical and bonded. Mechanical anchors are those using an embedded part emerging from the concrete surface. In these cases, embedding depth, plate size and torque applied are critical. Bonded anchorage systems uses drilled holes in the concrete in which are inserted anchor bolts that can be grouted (i.e. bonded to the concrete using cementitious grout or an adhesive that chemically bonds the anchor bolt to the concrete)

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

Waterstops are static seals used between joints in concrete structures to prevent passage of water or other fluids. Waterstops are embedded in the concrete or run across and/or along joints. Waterstops

are used in containment, spent fuel pools, wastewater treatment plants, secondary hazardous fluid containment structures, tunnels, ponds, water reservoirs etc.

Joints 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.

6. ASSESSMENT FOR AGEING MANAGEMENT

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 visual inspections for corrosion damage, leakage-rate and post tensioning testing, where applicable. Visual inspection, other than naked eye, may include remote devices such as still or video cameras, bore scopes, periscopes to detect corrosion products, flaking, discoloration, blistering, peeling, abrasion and underwater techniques such as dislodging, ultrasonic mapping, coating adhesion tests, dry-film thickness determinations. 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, where the detection of corrosion and its assessment has to rely on robotics and advanced techniques. Containment leakage rate testing is done by pressurizing containment and then monitoring changes in pressure, temperature and vapour pressure. If the pressurisation is done with a mixture of air and gas, then halogen gas detectors can be used to enhance detection. Containment testing can only be done with the 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 they should be included in periodic inspection programmes to be evaluated. However it is to be noted that these are replaceable items. In case of uncertainty in the assessments, they can always be replaced.

Stainless steel bellows in some penetrations are subject to fatigue and stress corrosion cracking and they can be tricky to inspect.

Protective coatings may also be difficult to inspect and diagnose. Their ageing evaluation can be done by measuring their characteristics (e.g. thickness and bond to substrate. The economic life of a coating is determined by comparing maintenance cost with recoating costs (including personnel dose).

6.1.1. Non-destructive testing methods

Non-destructive examination both surface and volumetric methods are widely used. Among them we count:

— Visual examination: Three types of visual examinations are classified in the ASME Boiler and Pressure Vessel code: VT-1 to detect discontinuities and imperfections on the surface such as cracks and corrosion regardless of the complexity of the surface; VT-2 to detect leakage from pressure retaining components; VT-3 to determine the general mechanical and structural conditions of components and their supports. They are heavily dependent on the skills and experience of the examiner and without dismantling they are limited to accessible areas.

— Liquid penetrant: To detect cracks, porosity, laminations and capillary discontinuities in solid components. The method makes use of the capillary effect by which the liquid is absorbed by the surface defect

— Eddy Current: for detection of cracks, porosity, inclusions of electrically conductive materials. It is based on the principle that alternating currents through a coil close to a conductor induces a current in the conductor and creates a magnetic field opposite 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.

— Half-cell potential measurements have been used with great success to detect corrosion of reinforcement steel in 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.

— Magnetic particles used to detect surface and shallow subsurface discontinuities in ferromagnetic materials. A magnetic field is induced into the material; its surface is dusted with iron particles either dry or in coloured or fluorescent liquid suspension. The lines of the magnetic flux are locally disrupted in correspondence to a flaw

— Ultrasonic testing: uses high frequency sound waves to detect surface and subsurface flaws (volumetric testing capability). Echoes are scattered or reflected back to a receiver from internal imperfections. 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 provides the flaw indication. The method requires interpretation by operators familiar with patterns, signal magnitude, timing and probe positioning to determine flaw size, distance and reflectivity. The measurement is volumetric and good for up to 6 m for axial penetration. It is very 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, signals are reflected from the front and back surfaces and the difference in arrival times of the two signals can indicate the thickness. Ultrasonic can also be used to detect and monitor corrosion. Ultrasonic imaging techniques can also be used to detect and localize thickness reductions in the metallic containment of BWRs.
— Electromagnetic acoustic transducers (EMATs) are a form of ultrasound examination technique in which high energy ultrasound shear waves that can travel a good distance, are generated directly in the metallic component being examined rather than in the transducer and it is non-dispersive. They can be generated even through surface coatings of thicknesses of up to 1.5 mm.
— Magnetostrictive sensors are devices that launch guided waves and detect electromagnetically elastic waves in ferromagnetic materials. The guided wave technique is more sensitive than shear waves used in the EMAT system. These waves can follow the contour of a structure, accommodate thickness discontinuities, travel considerable distances and be picked up at a remote location. They are therefore more suitable for examining piping. This method can be used to characterize corrosion damage and to inspect structures whose surfaces are not accessible as in the case of painted surfaces.
— Radiographic techniques use penetrating Gamma or X radiation and flaw detection is based on the differential absorption of radiation. Radiation is made to pass through the material and is captured on film on the opposite side of the source. Defects are indicated in 2D as density changes (i.e. various shades of grey) on the film, which remains as a permanent record. Radiography can detect both surface and internal defects. Gamma 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. Cracks are harder 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, accessibility to the specimen to test may be a problem, access from both sides is mandatory, heavy biological shielding is usually necessary and even total area evacuation may be required.
— Acoustic emission inspection uses small amplitude stress waves that are emitted by kinetic energy as the material is locally strained beyond its elastic limit. Piezo-electric transducers located on the material surface will 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 continuous monitor of critical SSCs for first detection of crack propagation. The system is extremely sensitive, fast, obstructive. Applications include stress-corrosion cracking, hydrogen cracking, and gas evolution. Conducting the test and interpreting results requires considerable experience since background noise often interferes with signals.
— Thermographic inspection is used for the detection of property variations at interfaces of layered materials. The material must be thermally conductive. 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 difference 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 temperature difference the better the flaw detection capability.
— Electrochemical corrosion monitoring: Corrosion rates are measured directly. DC current is directed from a counter electrode to the working electrode and the change in potential is measured between the two electrodes. Alternatively AC impedance changes can be used and converted into corrosion rate information.

Figure 43 correlates in a visually compact manner the various primary causes of ageing to the most appropriate detection method, tool or instrument available for inspection, monitoring or testing.

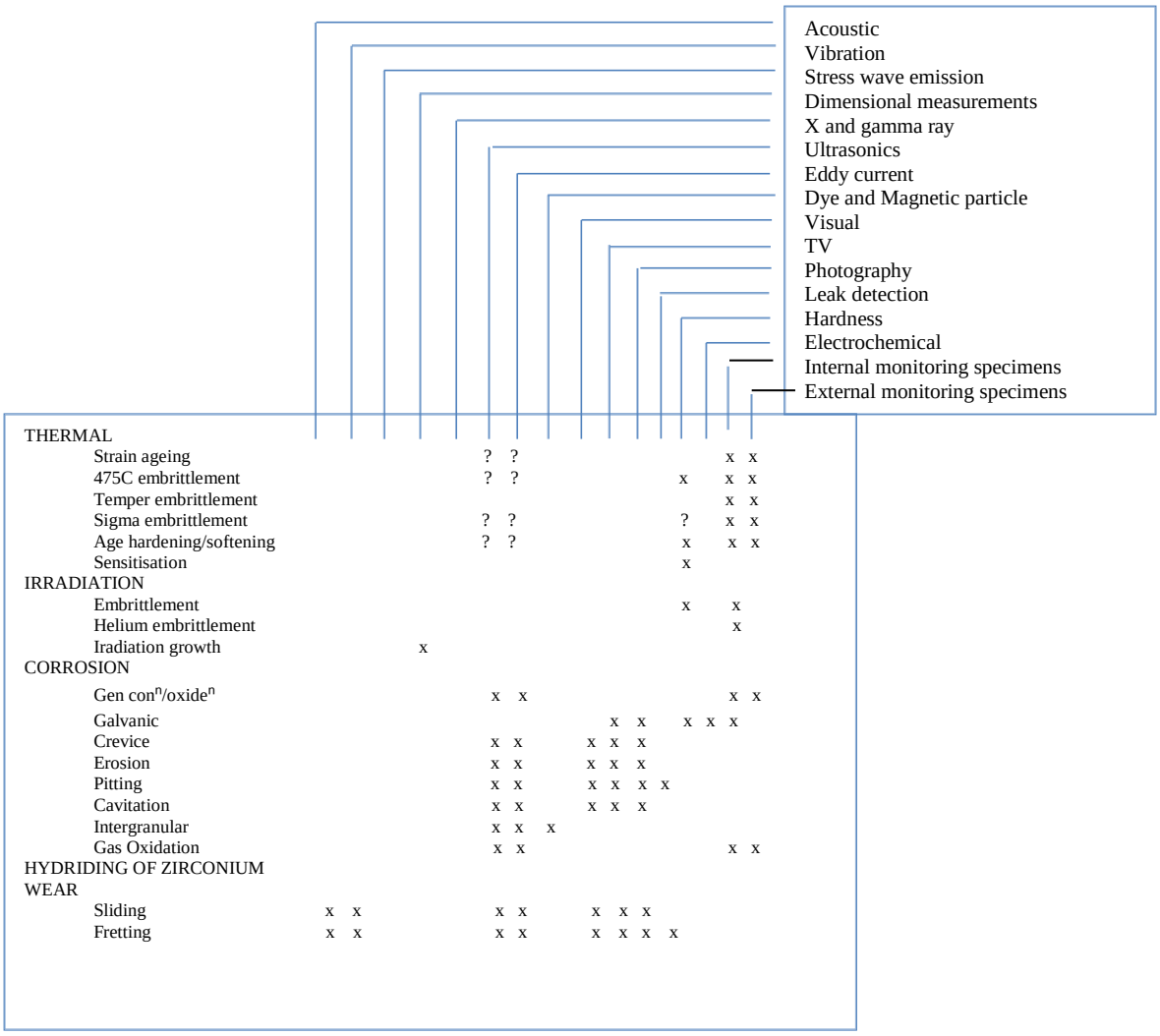


FIG.43. Inspection, monitoring and testing tools used in ageing diagnostics

6.1.2. Destructive testing methods

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 specimen are inserted into the RPV, periodically withdrawn and destructively tested; these tests serve for periodical evaluation of RPV material embrittlement and are a necessary part of PSRs. In some special cases, as for instance a Russian rule still requires [28] one weld from the primary piping in each plant is to be cut out after 100,000 operating hours, to derive specimens for testing to determine potential changes in the mechanical properties of the material that has seen all stressors during operation. Of particular interest are the tensile properties; These tests are now mostly omitted, as other non-destructive methods are available today;

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

Not all tests require a complete destruction of the specimen. The degree of interference with the component integrity can be used to divide them further into:

— Semi-destructive tests: A test is called semi-destructive when it has practically negligible effects

on the component integrity, its shape and size. Examples of tests of this nature can be the hardness tests, the instrumented hardness test (ABIT - Automated Ball Indentation Testing), replicas for metallographic examination etc.

— Destructive tests: In this case, some material must be removed from the component surface by mechanical or electro-erosive machining. It is then used as test sample. The removed material can be either practically negligible (like taking scraps for chemical analysis or for neutron dosimetry measurements) or substantial in size (samples for metallographic tests, mechanical tests, tensile tests, sub-size 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 without the material removed (effect of sampling on stresses and potential notch effect on fatigue) is necessary to evidence that the component safe operation will not be affected. In all such cases, previous authorization/approval from the regulatory body must be obtained.

6.1.3. Leak-Before-Break (LBB)

Leak-Before-Break (LBB) is a criterion that allows the relaxation of design requirements to mitigate postulated double-ended guillotine breaks (DEGBs) 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 more 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 otherwise not been economically feasible.

The LBB concept is based on the fact that sudden DEGBs of primary circuit piping are extremely unlikely and that tests have shown that the failure mode has a high probability to be gradual. The most likely failure mode starts with a through-the-wall crack producing a leak that does not challenge immediately the pipe capability to withstand its design loads. The leak can then be detected by appropriate inspection and by using either existing or newly added monitoring techniques.

The LBB concept has been interpreted by some as a pro-active and preventive approach to DEGB 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 developed, that were capable of predicting the evolution of postulated DEGBs under all operational loads.

This newly proven capability allowed operators to use the time interval between the first leak and the critical crack length to control, inspect and monitor on line their critical piping systems. These defensive measures, together with conservatively selected critical crack parameter thresholds, allowed the development of administrative procedures for the prevention of catastrophic failures, without having to introduce massive design changes with the risks that these involve.

If a leak grows large enough to start producing void in the pipe, a power pulse may follow. To avoid such an occurrence, mitigating action should be taken before voiding starts; this must be long before a crack has grown to a critical crack length. The proportion of leaks that would propagate quickly to DEGBs was estimated for CANDU reactors and gave failure rates of 7.7E-10/ry for 18 inch pipes and 8.8E-10/ry for 22 inch headers. Failure rates for smaller pipes were much higher. If LBB were credited, all pipes greater than 7 inches in diameter would have “extremely low” failure rates. If LBB were not credited, only pipes greater than 18 inches. would have extremely low failure rates. [55] Other concepts somewhat similar to LBB, notably the Break Preclusion (BP) concept and the Low Break Probability (LBP) concept have also been used.

6.2. 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 ageing degradation before failure of the structure or component occurs. Sampling checks of equipment should be applied to detect precursors of ageing degradation when applicable. In the evaluation of existing monitoring methods to identify effective and practical monitoring methods and technology, the following should be addressed:

— Functional parameters and condition indicators for detecting, monitoring and trending ageing

degradation of the structure or component;

— An assessment of the capability and practicability of existing monitoring techniques to measure these parameters and indicators with sufficient sensitivity, reliability and accuracy;
— Data evaluation techniques for recognizing instances of significant degradation, failure rates and their tendencies, for predicting future integrity and functional capability of the structure or component.

The results of the review relating to the monitoring of ageing should be documented.

6.3. REACTOR PRESSURE VESSEL NDE/ISI

RPV integrity is of critical importance for stable and reliable operation of nuclear power plants. Therefore, during the plant service life, periodic in-service inspections must 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 such as ASME B&PV Code (US), RSE-M (FRANCE), PNAE-G-7-010-89 (Russian Federation). In addition to this, certain European countries follow their own national standards that are based on the European Network for Inspection and Qualification (ENIQ) methodology [56].

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 (common approach). 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 figure 44.

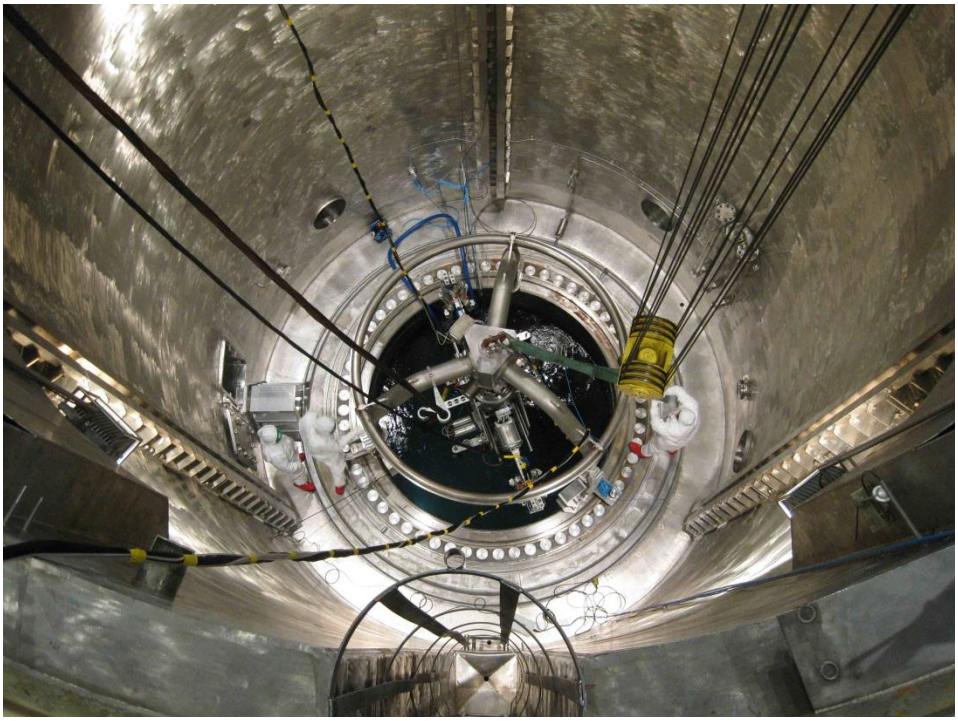


FIG. 44. RPV inspection manipulator - example

Scope of the RPV examination differs for the different plant types:

— PWR – circumferential and longitudinal welded joints of PWR, inlet/outlet/safety injection nozzle welds, nozzle inner radiuses, nozzle-to shell welds, baffle-to-former plate bolts
— VVER-440 – circumferential welded joints, inlet/outlet nozzle welds and nozzle inner radiuses,

vessel material in the active region of the RPV (zone exposed to highest neutron flux)

— VVER-440 - circumferential welded joints, nozzle welds and nozzle inner radiuses, vessel material in the active region and during a special outage 100% of the RPV surface is examined using the ultrasonic and eddy current methods.

— BWR – inspection of nozzles (ID and OD) and certain welds and RPV supports

The RPV inspection is an activity on the critical path of each power plant outage and must be performed with maximum reliability. RPV inspection robots are developed in such way that maximum efficiency and inspection speeds are achieved, reducing the total outage time. New technologies, such as highly automatized robots, associated software packages, ultrasonic phased array technology, and high-resolution visual inspection cameras are applied to provide enhanced inspection capabilities.

However, quality and technical requirements of an RPV inspection must never be compromised. A common requirement in all types of power plants is the qualification process. There is a number of qualification approaches (ENIQ, ASME, IAEA) which differ in technical and formal aspects, but all approaches ensure that the inspection system and the technique used can meet the code requirements and this even if new technologies must be developed to reach the goal.

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

— Core barrel support lugs

— Bottom mounted instrumentation

— RPV threaded flange holes, nuts and bolts

BMI are structures for which NDE examination is required. This has been emphasized in the last ten years following the ageing damage incurred in plants that use BMIs. Applicable orders and code cases require application of ultrasonic and eddy current inspections. Ultrasonic testing is used to inspect the BMI tube inner surface and eddy current testing is used to inspect the tube J-weld. See figure 45-46.

FIG.45. Examples of BMI inspection equipment

Recent operating experience from VVER power plants revealed issues with the integrity of the RPV sealing components like bolts, nuts and threaded holes. The threaded holes of the RPV flange and studs are examined by ultrasonic and eddy current NDE methods.

a) b)

FIG. 46 – RPV studs and holes inspection manipulators; a) VVER RPV studs inspection manipulator, b) VVER RPV holes inspection manipulator

6.4. STEAM GENERATOR NDE/ISI

Steam generator in-service inspections, using various NDE techniques, are designed to provide vital information about the SG conditions. The scope, the frequency and the type of NDE techniques used in SG inspections are defined by various regulations and standards such as the ASME B&PV Code (US), RSE-M (FRANCE), GOCT (Russian Federation). In addition, certain European countries follow their own national standards that are based on the ENIQ methodologies.

The main SG inspection activity is the eddy current ID examination of the SG tubing. Inspections are performed with remotely controlled inspection robots that are installed inside the SG primary channel heads while the eddy current probe is positioned on the desired tube. Years of R&D resulted in the development of robots of compact design with computer-controlled operation that do not require human intervention. Figure 47 shows examples of SG inspection robots for VVER and PWR/CANDU.

a)

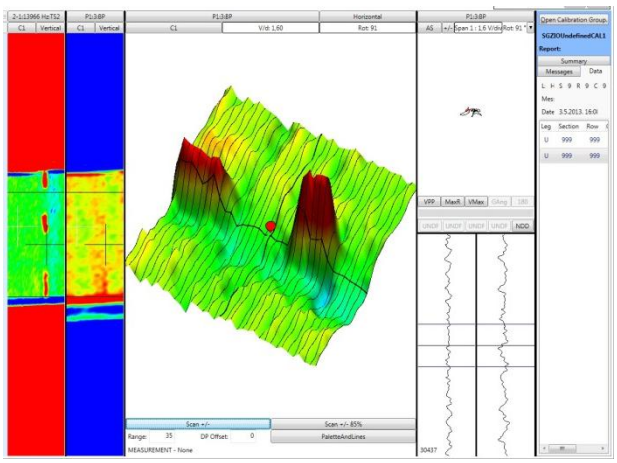
b)

FIG. 47 – SG inspection manipulators; a) VVER inspection manipulator, b) PWR inspection manipulator

The differences between the SG tubing inspections of VVER and PWR/CANDU plants are mainly due to the different steam generator configurations. The CANDU (as also the PWRs SGs) have a vertical tube bundle configuration that uses robots attached to the tube sheet that can easily move from tube to tube across the tube sheet. Such designs offer the possibility of multiple robots per SG channel head, hence distributing the work more efficiently.

R&D in eddy current testing also resulted in inspection methods that offer more details about the condition and degradation of the tubing. The latest technology involve a probe array that offers more information about the tube, enabling the examination of the entire tube circumference and allowing the determination of size (length, depth) and morphology of the detected degradation (the source of defect initiation and orientation). Such technology provides us with valuable information for further condition assessments, operational assessments and ultimately a more efficient ageing management. See figure 48.

a)



b)

FIG. 48. SG eddy current array probe; a) example of probe, b) example of EC array probe data

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

Beside the standard eddy current data on tube degradation, new and improved hardware and software allows the collection, processing and evaluation of data on other subcomponents such as:

— Tube support plate,
— Anti-vibration bar gap measurements
— Independent tube ID validation
— Deposit mapping.

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

Each SG inspection system and technique must 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 the EPRI Guidelines, while others apply their own qualification procedures.

6.5. REACTOR PRESSURE VESSEL HEAD NDE/ISI

Inspection of RPV Heads and sub-components encompasses ultrasonic, eddy current and visual examination. Remotely controlled robot systems automatically position the inspection devices on the desired RPVH penetration.

In the VVER fleet, RPVH inspections are performed automatically only in certain plants. In PWR power plants, after the well-known events involving the RPVHs in 2001 and 2003, inspection became mandatory and resulted in a number of RPVH replacements. RPVH inspections are normally performed from under the RPV head, where the inspection robot is deployed. See figures 49 - 51.

FIG.49. RPVH inspection probes manipulator

Inspections of PWR RPV Heads include:

— Bare material visual examination – top of the RPV Head is visually inspected a crawler robot fitted with a camera
— RPVH penetrations – The penetration welds through the RPV head are inspected from the inside surface of the penetration - Ultrasonic examination is used for the entire material volume, while eddy current 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 must be performed on the J-welds and the outside surface of the penetration tubes. Inspection is done using eddy current array probes with large number of coils (16 or more eddy current coils)
— CRDM joint weld – The bimetal weld is examined by ultrasonic and eddy current similarly to the RPVH penetration tube inspection.

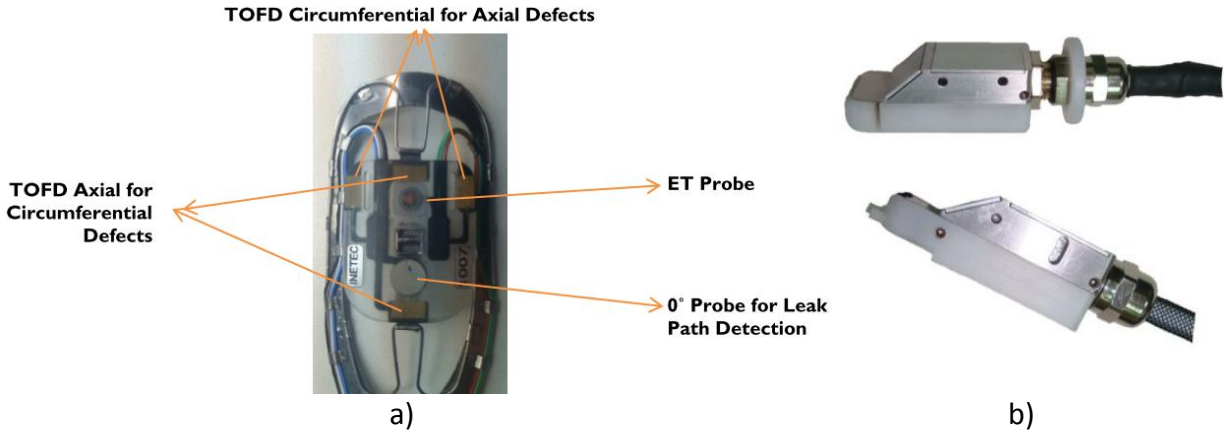


FIG.50. RPVH inspection probes: a) Ultrasonic probe for RPVH penetration inspection, b) eddy current array probes for J-weld examinations

Inspection methods applied in RPVH examinations include time-of-flight-diffraction (TOFD) ultrasonic inspection technique that provides best results for smaller material thickness. Inspection probes for RPVH penetration examination combine both ultrasonic TOFD crystals and eddy current

coils. J-weld Eddy current inspections are done with array probes that utilize multiple coils enabling coverage of large inspection volume.

Each inspection system and technique must 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 RPV Heads, 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 corrective action is presented in Figure 51.

FIG.51. PVH repair solution

7. REGULATORY FRAMEWORK

7.1. INTRODUCTION

The ageing management framework in most countries is designed to provide the licensees of nuclear power plants with a common approach to demonstrate the adequacy of the plant critical SSCs and the operator's continuing capability to operate the plant safely and reliably throughout the licensing period. In general each member state sets its own ageing management requirements or adopts internationally recognized requirements, but they all include common principles and essential safety features. All regulations in terms of ageing management aim at confirming throughout the plant life that its 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 license under all operating conditions, including accident conditions, even as its structures, systems and components continue to age. To this end, at regular intervals, the plant safety analysis is updated and the SSCs important to safety are evaluated with regard to the effects of ageing. In addition to this periodic review at regular intervals, every time a major change is introduced in the plant configuration, the plant safety analysis models 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 control functions including set points and control parameters and its capability to mitigate the effects of internal and external events including extreme conditions, The safety review also includes a review of the plant operating and emergency procedures, of its maintenance, and inspection programmes, of human performance, of the staff knowledge management and the adequacy of the processes and procedures to monitor and manage ageing mechanisms.

The safety analysis reviews incorporate deterministic and probabilistic methods which to a large extent are complementary. The probabilistic method namely the probabilistic safety analysis (PSA) 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 the deterministic methods provide insights into the plant response to the various operational conditions and the integrity of the SSCs critical to safety.

7.2. VARIOUS NATIONAL APPROACHES

From the licensing standpoint, there are three conceptual approaches used to obtain an authorization to operate nuclear power units even beyond their design life. They are all based on plant configuration control and a good control and awareness of SSC ageing. Three type of conceptual approaches to licensing have been developed:

— the license renewal application (LRA) process,
— the periodic safety review (PSR) approach
— a combined approach of the former two.

The U.S. practice the first concept, while most European countries and Japan use PSR to obtain their license and the authorization to continue operation beyond the original design life of a plant (also called long term operation or LTO). In some Member States (e.g., Spain, Hungary, and the Republic of Korea), these two different concepts and related regulatory approaches have been combined encompassing elements of both LRA and PSR to better meet local conditions and licensing requirements. The various national approaches to license renewal are described in IAEA's technical document 1736 [23]. License renewal applications in the US and in countries following the US model is based on the pre-requisite that ageing management of active components and systems is adequately addressed by the maintenance rule (MR) or similar regulatory processes. The LRA pre-requisites are:

— An integrated plant assessment evaluating the AMPs of passive, long-lived SSCs. This is a prerequisite ensuring that the AMPs can support continued safe plant operation beyond the 40-year term of the original license. This usually requires:
— An assessment of SSCs with time-limited ageing to justify the additional years of operation;
— An Environmental impact assessment for the additional service life.

In general, if a licensee of a certain country follows the U.S. maintenance rule and other US operating and licensing practices, it is likely that elements of the US LRA process may be incorporated into their LTO authorization.

7.3. ORGANIZATIONAL STRUCTURE

7.3.1. Current Practice

The organization of an ageing management programme (in terms of the main elements making up the programme) is shown in figure 52. The central column shows the sequence of activities; to the left and to the right are shown the supporting data. On the left we have the data coming from the ageing knowledge base, the technology and engineering tools used to implement the programme. To the right of the main activity sequence, we have direct input data coming from the same plant specific operation records. Below are shown the acceptance criteria and the decision making processes.

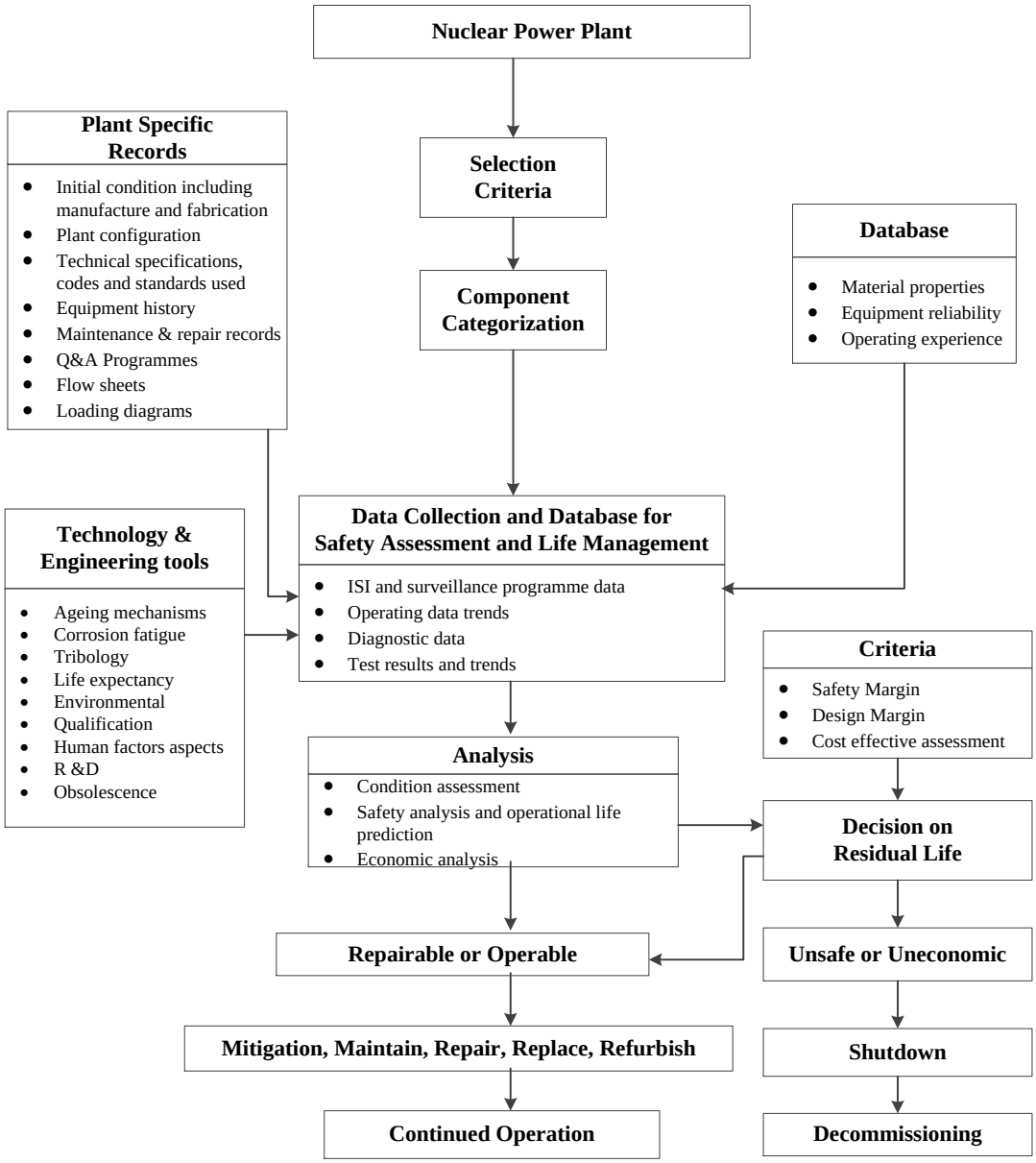


FIG. 52: Main elements of an ageing management programme

An example of an ageing management programme for a nuclear power plant is shown in figure 53 for Ringhals, a Swedish nuclear power plant with one BWR and three PWRs south of Gothenburg.

The division of responsibilities is based on a traditional grouping by discipline handling SSC ageing and below them the sub-programmes developed for the ageing management programme.

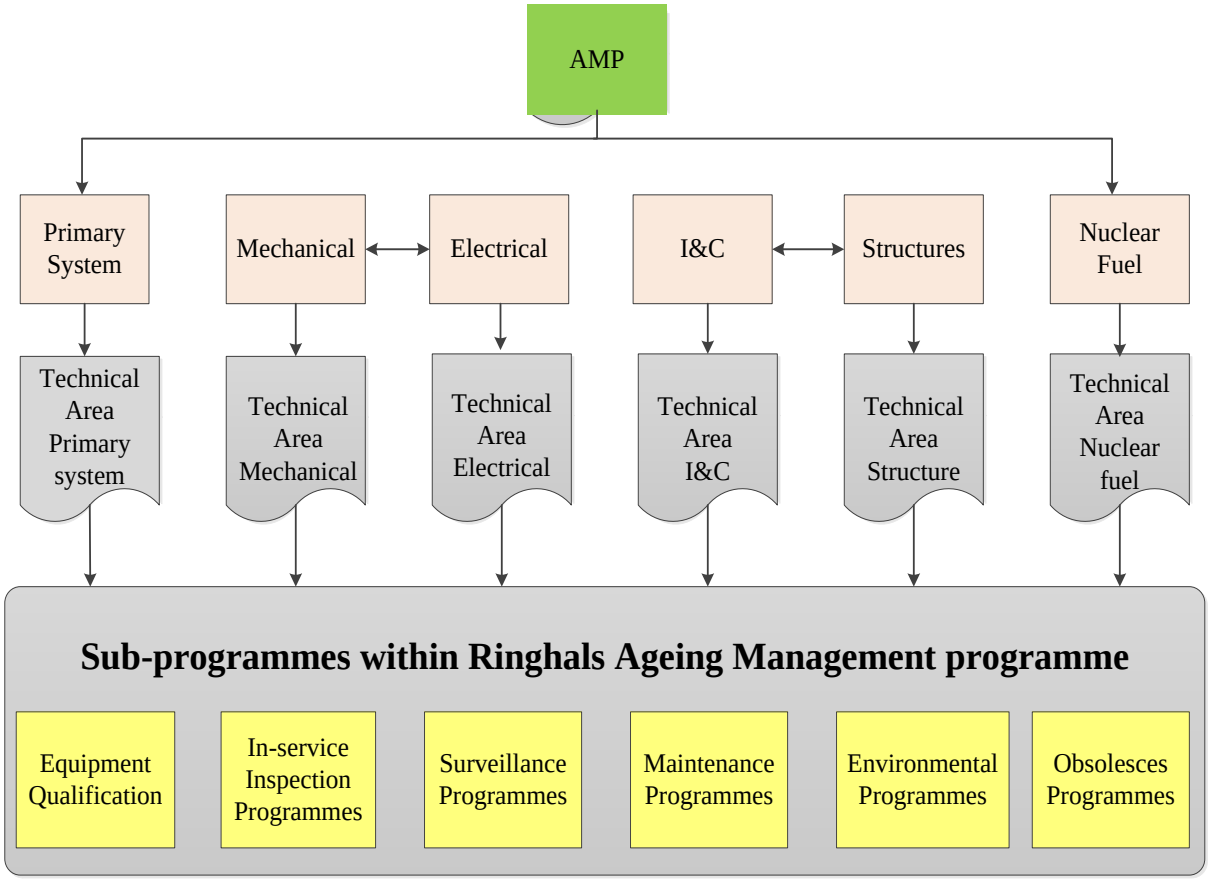


FIG.53: Ringhals, Sweden – Ageing Management Programme Structure

7.4. LICENSING FRAMEWORK IN THE UNITED STATES

In the United States, the Nuclear Regulatory Commission (NRC) issues initial licenses for commercial power reactors to operate for up to 40 years and allows these licenses to be renewed for an additional 20 years with each renewal application. A maximum license term of 40 years was selected on the basis of economic and antitrust considerations, not of technical limitations. There is no limit on the number of license renewals as long as the plant can continue to be run safely and in accordance with environmental requirements. The decision whether to seek license renewal rests entirely on nuclear power plant owners, and is typically based on economic factors and safety considerations mainly whether the plant can continue to meet NRC requirements for a prolonged operation period.

A license renewal application may be submitted to the NRC as early as after 20 years of operation and not beyond 35 years of operation. This 15 year window of opportunity was set by the NRC to allow a reasonable time for utilities to develop their long term business plans and to allow the NRC staff at least five 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 more than five years, due to the adjudicatory process, but the plant will be allowed to continue operation. The license renewal process proceeds along two tracks: one for review of safety issues (10 CFR Part 54) and the other for the review of environmental issues (10 CFR Part 51).

NRC's code of federal regulations 10 CFR 54.21, describes the license renewals. It requires license renewal applications to include:

— an integrated plant assessment (IPA),

— an evaluation of the changes to the current licensing basis (CLB),
— an evaluation of time-limited ageing analyses (TLAAs), and
— a final safety analysis report (FSAR) supplement.

The NRC has issued regulatory guides and NUREG documents defining the regulatory process for an LRA. In addition, the Nuclear Energy Institute (NEI) in support of its members has developed an industry guidance document, NEI 95-10 Rev. 3: "Industry Guideline for Implementing the Requirements of 10 CFR Part 54" on how to best prepare a license renewal application, which has been endorsed by the NRC in its Regulatory Guide 1.188: "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses". Regulations, guidance documents, and background information are available to the public on the NRC website at www.nrc.gov. The Standard review plan for a licence renewal (SRP-LR) sections are keyed to the RG-1.188, Standard Format document; the sections are numbered according to the section numbers in that document.

In the license renewal application the applicant provides the information needed to assure the NRC that any changes to the safety and performance envelope and the effects of ageing will meet the license renewal requirements during the entire extended operating period. In order to meet this goal, an applicant should have analysed the plant to ensure that actions have been or will be taken to:
— manage the effects of ageing during the period of extended operation (this determination should be
— based on the functionality of structures and components that are within the scope of license
— renewal and that require an ageing management review),
— perform Time limited ageing analysis (TLAA)s on all components requiring it (for details on
— TLAAs refer to section 8.3

Most of the programmes to manage ageing that are credited for license renewal are existing programmes. In a staff paper (SECY 99-148), "Credit for Existing Programmes for License Renewal," dated June 3, 1999, the staff described options and provided a recommendation for crediting existing programmes to improve the efficiency of the license renewal process. In a staff requirements memorandum (SRM) dated 27 August 1999, the NRC approved the staff recommendation and directed the staff to focus the review guidance in the SRP-LR on areas NUREG-1800 2 April 2001 where existing programmes should be augmented for license renewal. Under the terms of the SRM, the SRP-LR would reference a "Generic Ageing Lessons Learned" (GALL) report, which evaluates existing programmes generically, to document
— the conditions under which existing programmes are considered adequate to manage identified ageing effects without change and
— the conditions under which existing programmes should be augmented for this purpose.

The GALL report (NUREG-1801) 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 acceptable with respect to such a matter when the application references the GALL report and the evaluation of the matter in the GALL report applies to the plant. They ensure that the material presented in the GALL report is applicable to the specific plant involved and that the applicants have identified specific programs as described and evaluated in the GALL report if they rely on the report for license renewal. The results of the GALL effort are presented in a table format in the GALL report, Volume 2. The table column headings are:
— Item,
— Structure and/or Component,
— Material,
— Environment,
— Ageing Effect/Mechanism,
— Ageing Management Programme (AMP), and
— Further Evaluation.

In terms of how the NRC review is conducted is spelled out in the commission's standard review plan for license renewal applications of nuclear power plants NUREG-1800

The staff's evaluation of the adequacy of each generic programme in managing certain ageing effects for particular structures and components is based on the review of the following program attributes (or elements):
— Scope of the program: The scope of the program should include the specific structures and components subject to an ageing management review.

— Preventive actions: Preventive actions should mitigate or prevent the applicable ageing effects.
— Parameters monitored or inspected: Parameters monitored or inspected should be linked to the effects of ageing on the intended functions of the particular structure and component.
— Detection of ageing effects Detection of ageing effects should occur before there is a loss of any structure and component intended function. This includes aspects such as method or technique (i.e., visual, volumetric, surface inspection), frequency, sample size, data collection and timing of new/one-time inspections to ensure timely detection of ageing effects.
— Monitoring and trending Monitoring and trending should provide for prediction of the extent of the effects of ageing and timely corrective or mitigating actions.
— Acceptance criteria Acceptance criteria, against which the need for corrective action will be evaluated, should ensure that the particular structure and component intended functions are maintained under all current licensing basis (CLB) design conditions during the period of extended operation.
— Corrective actions Corrective actions, including root cause determination and prevention of recurrence, should be timely.
— Confirmation process: The confirmation process should ensure that preventive actions are adequate and appropriate corrective actions have been completed and are effective.
— Administrative controls Administrative controls should provide a formal review and approval process.
— Operating experience: Operating experience involving the ageing management program, including past corrective actions resulting in program enhancements or additional programs, should provide objective evidence to support a determination that the effects of ageing will be adequately managed so that the structure and component intended functions will be maintained during the period of extended operation.

The SRP-LR is divided into four major chapters:

— Administrative Information;
— Scoping and Screening Methodology for Identifying Structures and Components Subject to Ageing Management Review, and Implementation Results;
— Ageing Management Review Results; and
— 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 SRPLR section, as appropriate for the application. In some cases, the major portion of the review of a plant program or activity may be done on a generic basis (with the owners' group of that plant type) rather than in the context of reviews of particular applications from utilities. In other cases, a plant program or activity may be sufficiently similar to that of a previous plant that a complete review of the program or activity is not needed. For these and similar reasons, reviewers need not carry out in detail all of the review steps listed in each SRP-LR section in the review of every application. The subsections are:

— Areas of Review: This subsection 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 systems, structures, components, analyses, data, or other information that are reviewed as part of the license renewal application. It also contains a discussion of the information needed or the review expected from other branches to permit the primary review branch to complete its review.
— Acceptance Criteria: This subsection contains a statement of the purpose of the review, an identification of applicable NRC requirements, and the technical basis for determining the acceptability of programs 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, the technical bases for some sections of the SRP-LR can be provided in branch technical positions or appendixes as they are developed and can be included in the SRP-LR.
— Review Procedures: This subsection discusses the way 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 subsection 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 (SER), in which the reviewers publish the results of their review. The SER also contains a description of the review, including which aspects of the review were selected or emphasized; which matters were modified by the applicant, required additional information, will be resolved in the future, or remain unresolved; where the applicant's program deviates from the criteria provided in the SRP-LR; and the bases for any deviations from the SRP-LR or exemptions from the regulations.

— Implementation: This subsection discusses the NRC staff's plans for using the SRP-LR section.
— References: This subsection lists the references used in the review process.

The SRP-LR incorporates the staff experience in the review of the initial license renewal applications. It may be considered a part of a continuing regulatory framework development activity that documents current methods of review and provides a basis for orderly modifications of the review process in the future. The SRP-LR will be revised and updated periodically, as needed, to incorporate experience gained during future reviews, to clarify the content or correct errors, to reflect changes in relevant regulations, and to incorporate modifications approved by the NRR Director. A revision number and publication date are printed in a lower corner of each page of each SRP-LR section. Because individual sections will be revised as needed, the revision numbers and dates will not be the same for all sections. The table of contents indicates the revision numbers of the most current sections. Comments and suggestions for improvement should be sent to the Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. Notices of errors or omissions should be sent to the same address.

Under the United States License Renewal Rule (also adopted by many other regulators with western type systems) when applying for a license renewal for a period extending beyond the first license, an SSC evaluation called "Time-Limited Ageing Analysis" (TLAA) is required when certain criteria are met within the scope of the license renewal to continue to operate the plant beyond the first license expiry date. These analyses must include:

— The effects of ageing on the SSC;
— The 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 SSC to perform its intended functions
— All calculation are contained or incorporated by reference in the current licensing basis.

A screening of SSC involving TLAAs must be conducted. The TLAAs then must be evaluated by one of these three methods:

— Demonstrate that the analyses remain valid for the period of extended operation;
— TLAAs have been projected to the end of the license renewal period,
— Ageing can and will be managed adequately for the period of extended operation.

An example of TLAAs submitted by Excelon to the USNRC by one of three methods listed above is shown in table 5.

TABLE 5. EXAMPLES OF TLAA TABLE 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 (USE) 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 End-of-Life Re-flood Thermal Shock Analysis	Validation of the Analysis for the period of Extended Operation

7.4.1. Organizational structure in the Unites States

The organization of PLiM in the U.S. 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, operations) for implementation. Other plants may not have a dedicated PLiM organization, but relies on various organizations (e.g., design engineering, system engineering, maintenance) to conduct needed PLiM studies for LTO on a case-by-case e.g., by component groups (piping, cables, transformers) or by individual components (turbine-generator, steam generator).

The organization structure for preparing a 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 organization structure is shown in figure 54.

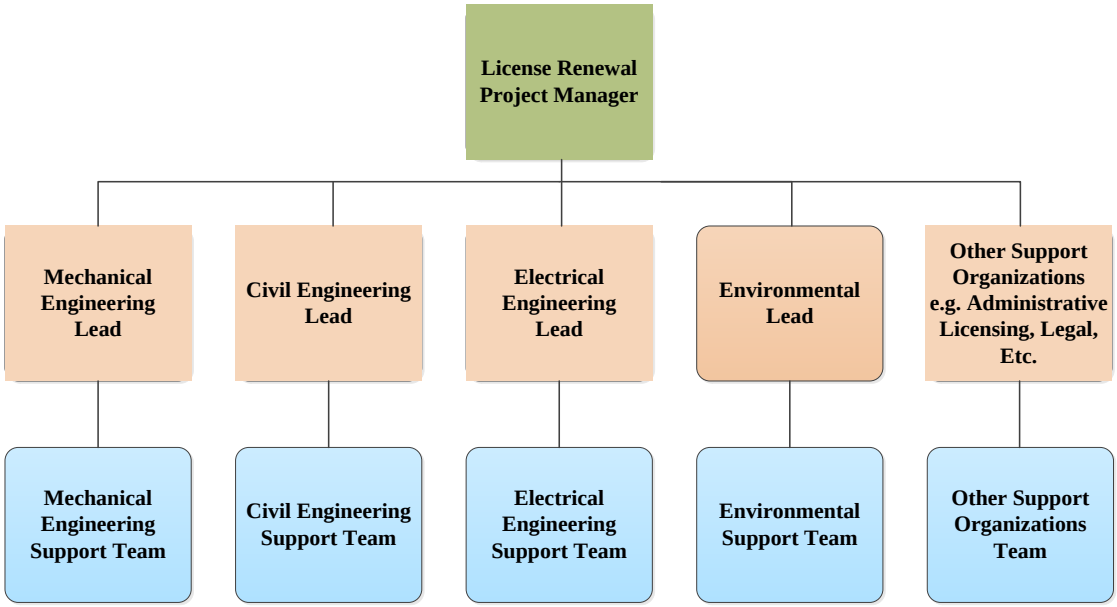


FIG. 54. Typical License Renewal Project Team Organization Chart.

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

7.5. THE PERIODIC SAFETY REVIEW (PSR) APPROACH TO LICENSING

In countries where the safety performance of NPPs 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 original design life of a nuclear power generation unit. The fundamental requirement is for the licensee to demonstrate a good understanding of the plant condition and of its capability to operate safely for the duration of the PSR cycle. If the new operating period reaches or crosses the end of the plant design life, the main focus of the LTO authorization process becomes that of determining whether the ageing of critical SSCs is being effectively managed so that all required safety functions can be maintained through the LTO period. In other words, the regulator focuses on the effectiveness and on the capability of the ageing management programme (AMP) to adequately cover the LTO period. Regulators may also use PSR as a tool to identify and resolve safety issues in NPPs. The PSR process is shown in figure 55.

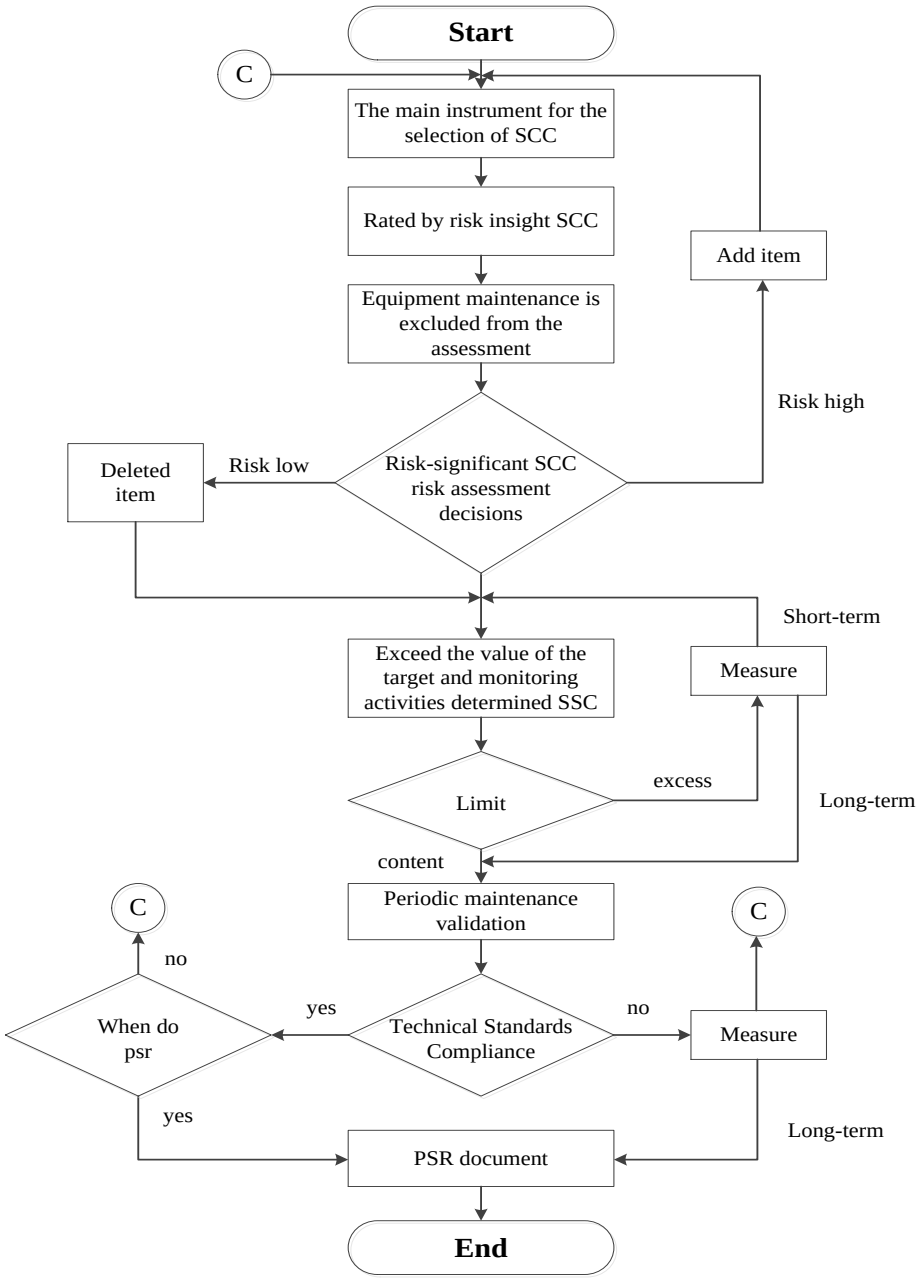


FIG.55. Periodic Safety Review Report Preparation Process

As in the case of the license renewal process, the selection of SSCs for the LTO period safety review is a very important process. It determines which plant SSCs will be included in the review scope. The result provides the utility with list of short and long lived SSCs. Short lived SSCs are subject to relevant plant programmes (maintenance, EQ, etc.). Long lived SSCs are subject to relevant ageing management programmes (AMPs) and time limited ageing analyses (TLAAs). Some national regulators also require that all SSCs credited in a safety analysis with an essential function that mitigates certain types of events must be included in the scope. Such events may include:

— Fires and floods;
— Extreme weather conditions;
— Earthquakes;
— Pressurized thermal shock;
— Anticipated transient without scram;
— Station blackout.

Taking into account the above list, the scoping exercise follows the decision making process depicted in figure 56.

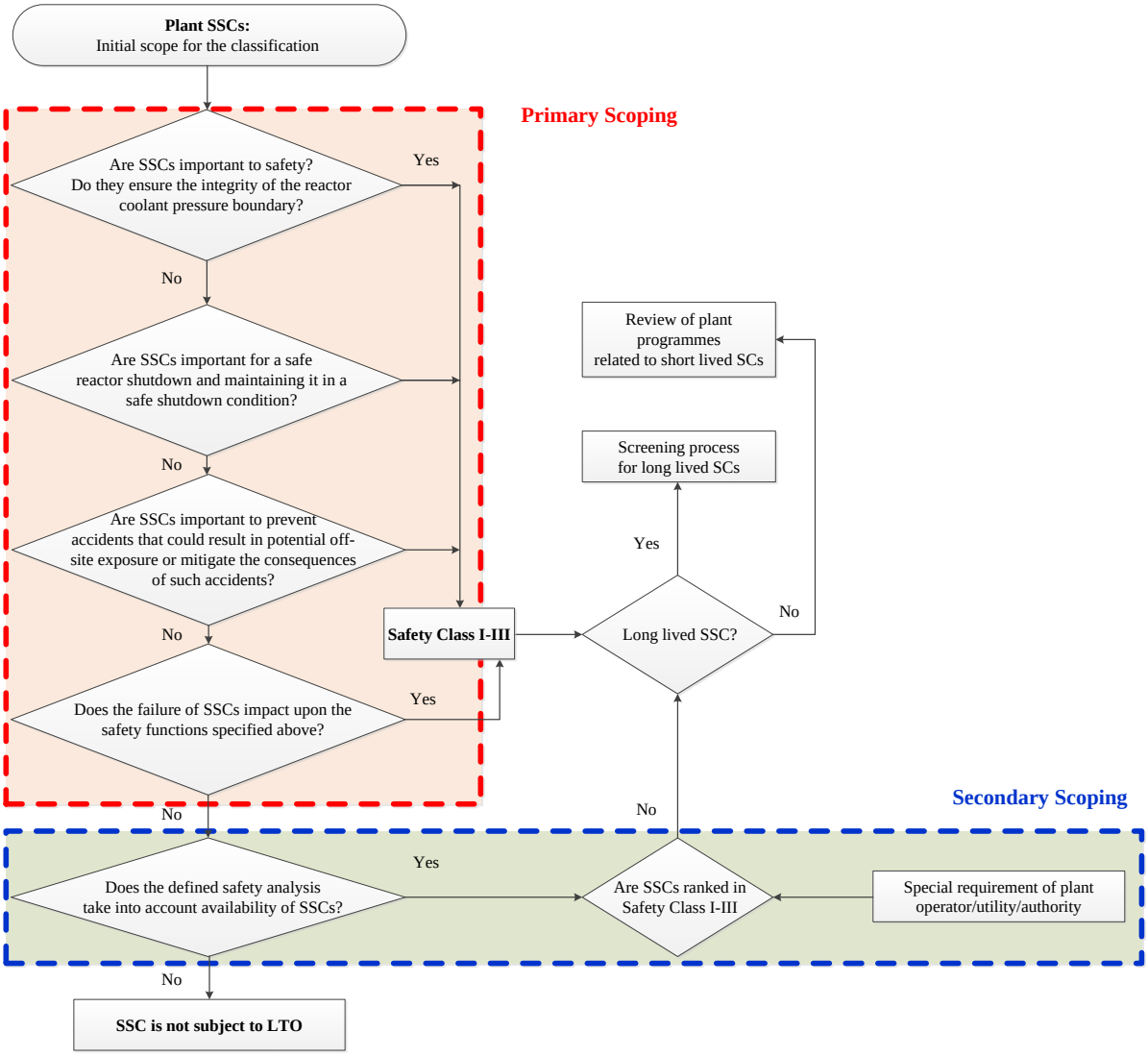


FIG. 56. Scope setting process ("scoping").

Depending on the licensing framework in force, an evaluation of TLAAs, as prescribed in the US licence renewal approach, may be required by regulators also within the PSR approach, particularly for PSR reports submitted for long term operation permits. A prerequisite for a PSR submission is that the following plant programmes have been well established from the beginning of plant operation:

— Maintenance;
— Equipment qualification (EQ);
— In-service inspection (ISI);
— Surveillance and monitoring;
— Monitoring of chemical regimes;
— Operational diagnostics;
— Configuration management.

Other existing relevant programmes at the plant are:

— Resolution of safety issues;
— Probabilistic safety assessment;
— Periodic safety review;
— Regular update of safety analysis report.

As shown in figure 57 long lived SCs enter a screening process. The purpose of screening is to identify SCs 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 SCs for whose AMPs to be developed;
— A list of TLAAs whose validity must be reviewed;
— A list of SCs with no ageing effects identified (no TLAA needed).

Short lived SCs are managed by plant programmes such as maintenance and equipment qualification. Short lived SCs are subject of regular replacement.

Time limited ageing analyses (TLAAs) are plant specific analyses for the time period under consideration. The TLAA time period may cover a limited number of loading cycles or a more limited duration of a specifically defined event. If the originally design life of a system or component is at the end of its originally intended duration in service, then its TLAA must be reviewed and revalidated with respect to the longer service period. It is necessary to verify whether the conditions do not generate new TLAAs. If a number of new TLAAs emerge as a result of the conditions considered, new analyses must be performed to demonstrate that the ageing effects on required SC functions will be adequately managed throughout the entire extended service period. 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 case of unfavourable results, corrective action(s) must be taken to ensure that the effects of ageing on the intended functions will be adequately managed throughout the entire period.

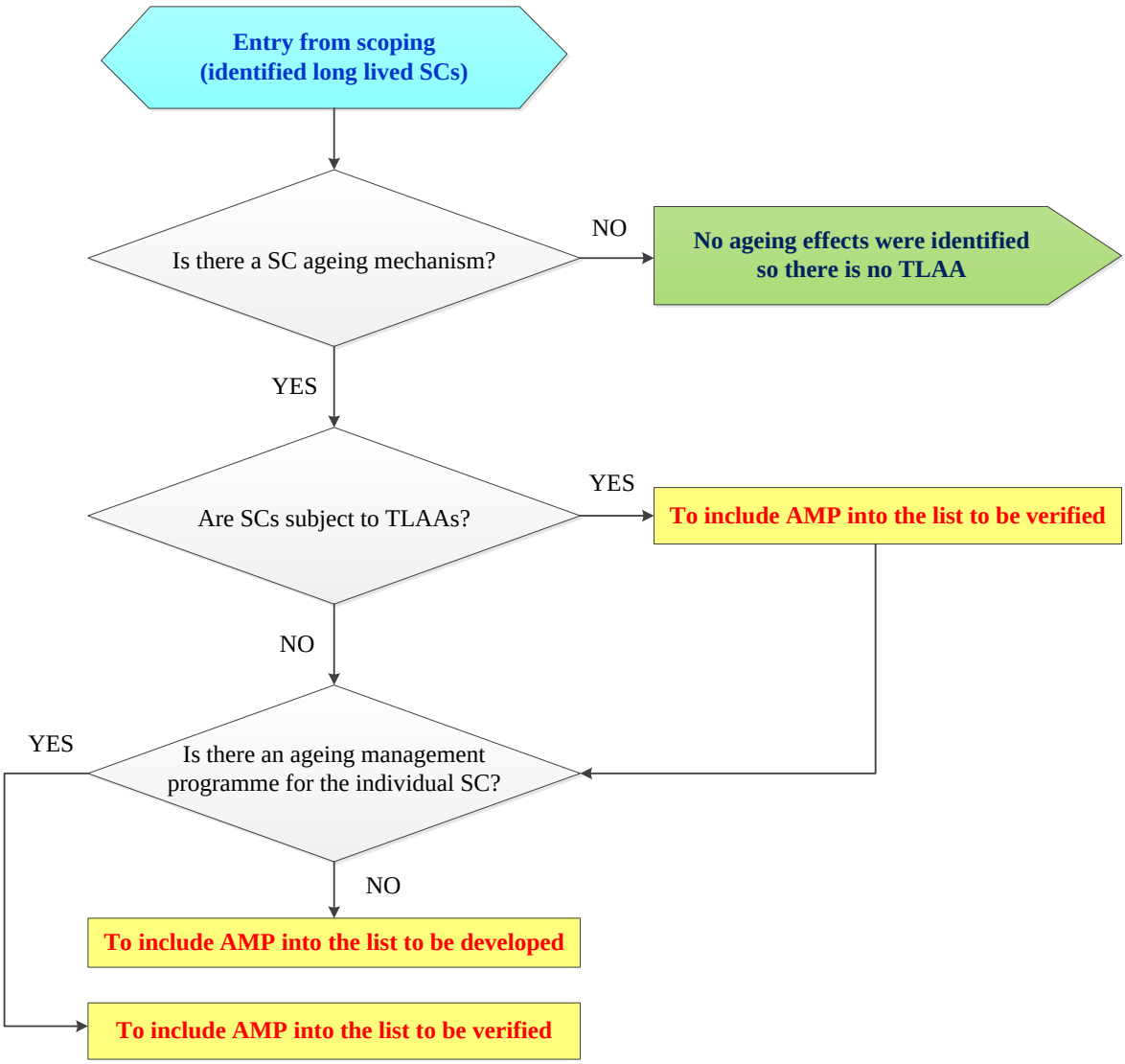


FIG. 57. Screening process.

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

— Crack growth analyses (flaws in SCs which were found by ISI);
— Analysis of the reactor pressure vessel (RPV) resistance against brittle/fast fracture (pressurized thermal shock analyses)
— Determination of RPV pressure-temperature dependence curves (p-T curves for normal conditions, interference modes, pressure tests, auxiliary modes);
— Low cycle thermal and mechanical fatigue of the main primary mechanical SCs;
— Thermal stratification of pressurizer surge line;
— Thermal stratification of steam generator feed water inlet nozzles;
— Consequences of postulated high energy piping ruptures (with respect to fatigue)
— Leak before break analyses;
— 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 SCs 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 to a possible damage of primary and SCs active devices damage
— Environmental impact on electrical and I&C equipment / SCs (Safety Classes I–III)
— Impact of ageing effects on confinement penetrations;
— Ageing of confinement structures (welds, transition welds);
— Ageing of cranes which may have impact to 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 statics of structures and the important functions of buildings.

Requirements regarding the TLAA documentation submissions are:

— Terms of reference;
— Computational model(s), tests, 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 SC ageing;
— Conclusions, recommendations, the proposed corrective actions.

7.6. LICENSING REQUIREMENT IN FRANCE

In France the Regulatory Authority is “L'Autorité de sûreté nucléaire (ASN)”, formerly “Direction de la Sûreté des Installations Nucléaires“(DSIN). Its technical support organization (TSO) is the “Institut de Radioprotection et de Sûreté Nucléaire (IRSN)” the Institute for Radiological Protection and Nuclear Safety. ASN does not issue a license for a specified period of time. The original design life of the units is 40 years. The Safety Authorities give an authorization to restart each unit after reloading at the end of each cycle (roughly every twelve to sixteen months, depending on the reactor design series and of the fuel cycle).

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

All modifications are defined taking into account the results of a PSR, performed not unit by unit, but at the same time for the entire series of plants of the same design. This occurs before the ten-year outage of the first unit of the series. According to the results of the PSR, and to the context and the implementation plan of the modification batch proposed (the same on 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, are implicitly 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 they will not consider a life extension request before the 3rd PSR cycle, and not for more than 10 years at a time.

7.6.1. Organizational structure

In preparation for the 10-year outages, an AMP is in force at EdF in order to justify that all SSCs concerned by an ageing mechanism remain within the applicable design and safety criteria. The AMP procedure is carried out in 3 main steps, in agreement with French regulations and with IAEA Safety guide NS-G-2.12 on Ageing Management for Nuclear plants:

— Selection of Safety related SSCs affected by an ageing mechanism;
— Review of all the SSC subjected to degradation mechanism selected by the experts using ageing analysis sheets in which maintenance adaptability, difficulty of repair and replacement as well as risk of obsolescence are taken into account;
— Detailed Ageing Management Reports (DAAR) required for some “sensitive” components (such as the reactor pressure vessel, reactor internals, civil structures, I&C or electrical cables).

Each NPP 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 safety components of the plant, the report must justify the ability of the plant to operate for 10 more years.

A detailed, systematic procedure is now 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 of operating feedback. The LTO strategy is illustrated in figure 58. It includes:

— 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 the factors facilitating lifetime extension;
— A Strategy (Asset Management actions) including an exceptional maintenance programme periodically revised by the EDF's executive committee for LTO.

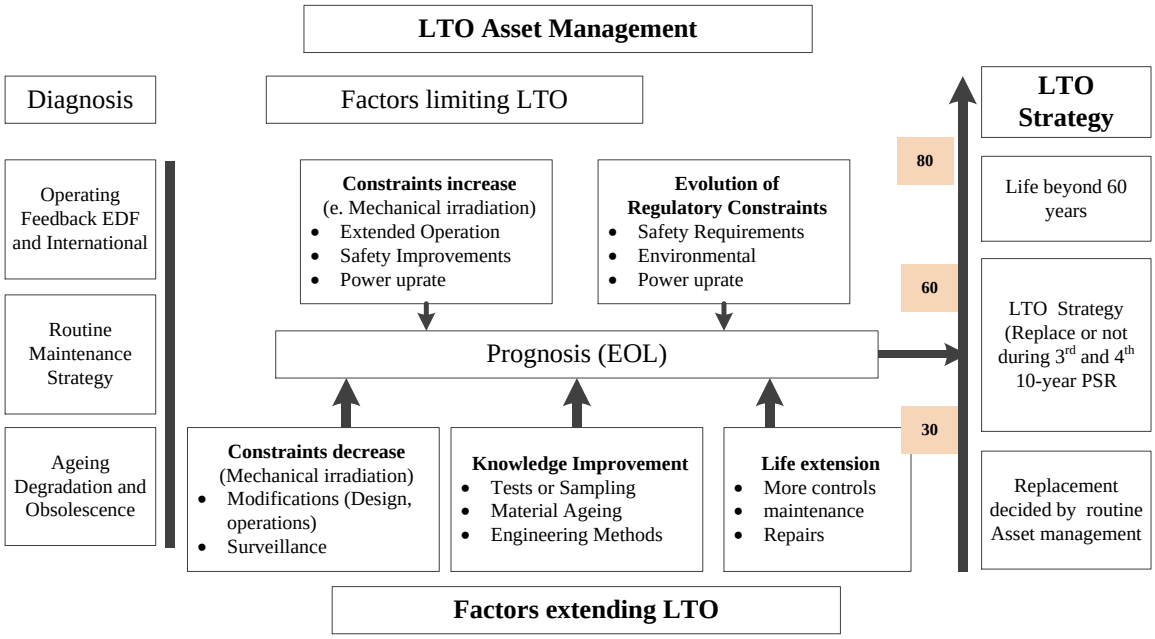


FIG. 58. LTO decision making process for major NPP components.

The strategy is selected according to the estimated end of life (EOL):

— If the estimated EOL is beyond 60 years, the strategy is periodically updated by the Executive Committee. At the same time, a Component Replacement Feasibility File is prepared to cope with obsolescence and the availability of spare components, with the repair /replacement process in order to plan strategic modifications if necessary in order to adequately cope with an unexpected demand for replacements. Moreover, a R&D programme on issues such as material research, evolution of methods, NDT evaluation, configuration modifications etc. is established in support of the strategy.
— If the estimated EOL is between 40 and 60 Y, the LTO strategy includes an Exceptional 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 planning it correctly (this occurs between the 3rd or 4th 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 EOL is less than 40 Y, 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 to up to 60 years.

In terms of plant life management, a distinction has to be made between replaceable components such as pumps and steam generators and non-replaceable components such as the Reactor Pressure Vessel (RPV) and the containment building. Appropriate strategies must be set up in both cases.

7.7. LICENSING REQUIREMENT IN HUNGARY

In Hungary, a comprehensive regulatory system has been specifically developed to oversee the safety of LTO of the four WWER-440/213 reactors at Paks. Compliance with the current Licensing Basis (CLB) is periodically verified via an annual updating of the FSAR and routine regulatory inspections and approvals. The annually updated FSAR has to be in accordance with the actual plant configuration, and demonstrate compliance with the CLB. To that end, it also includes the definition of design basis. The FSAR content is similar to that required by U.S. NRC Regulatory Guide 1.70 (Rev. 3).

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 time scale broader than the routine daily or even yearly checks. The broader time scale 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 specific safety guide on periodic safety reviews [57]. License renewal is the formal process used in Hungary to apply for an operating license extension beyond the original design term. This process is similar to the one governed by the 10 CFR 54 LR rule in the U.S., with some notable deviations.

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

A formal LR application has to be submitted one year before the licence expiration date. The application must include an assessment of the three 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. The contents of the LR application report are described in section 2.2.6.

7.7.1. Organizational structure

Organization of PLiM for LTO

There is an organization based on functions, extending to all PLiM related disciplines at the Paks NPP. The disciplines listed below, which are part of the cycle and/or configuration management programmes, are covered by the responsible sections or departments in the plant. All tasks within each discipline are governed by operating instructions, procedures, and QA programmes:

— Design;
— Fabrication;
— Procurement (including the rating);
— Erection (construction, assembly, installation);
— Commissioning;
— Operation covering:
— operational tests;
— monitoring & surveillance;
— in-service inspection;
— in-service testing;
— condition monitoring;
— preventive maintenance;
— corrective maintenance;
— maintenance effectiveness monitoring (Hungarian MR);
— spare parts management;

— configuration management;
— ageing management;
— environmental qualification (EQ).
— Replacement, reconstruction;
— Education;
— Asset management and economy planning;
— Control of the plant safety.

Organizational structure for developing the operating license extension application

A formal project team (of 8 employees) was set up for the preparation of the LTO programme and the license renewal application. A project manager and technical deputy direct the execution of the project tasks, prepared for implementation by the project staff.

The project team follows a project plan, approved by management, which includes technical tasks and the budget details. The project team works within the Technical Support Department. The relationship between the project team and other in-house teams can be seen in figures 59-60. Project staff is responsible for several tasks, among which:

— the implementation of the project technical tasks;
— coordination between the internal and external experts and the organizations involved;
— the recording and filing of the analyses and of other project-related documentation;
— the preparation of regular progress reports for the project management and for the nuclear authority;
— the organization of technical meetings;
— the adherence to budget and deadlines.

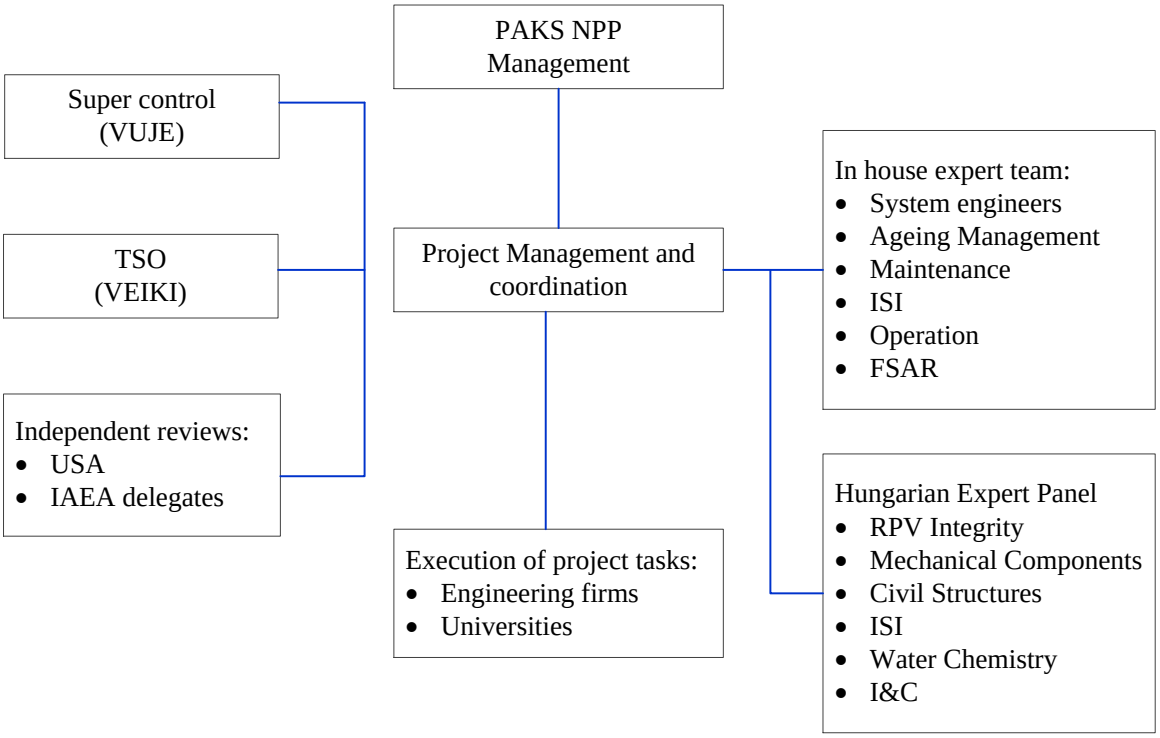


FIG. 59. Project Organization for developing the LTO programme.

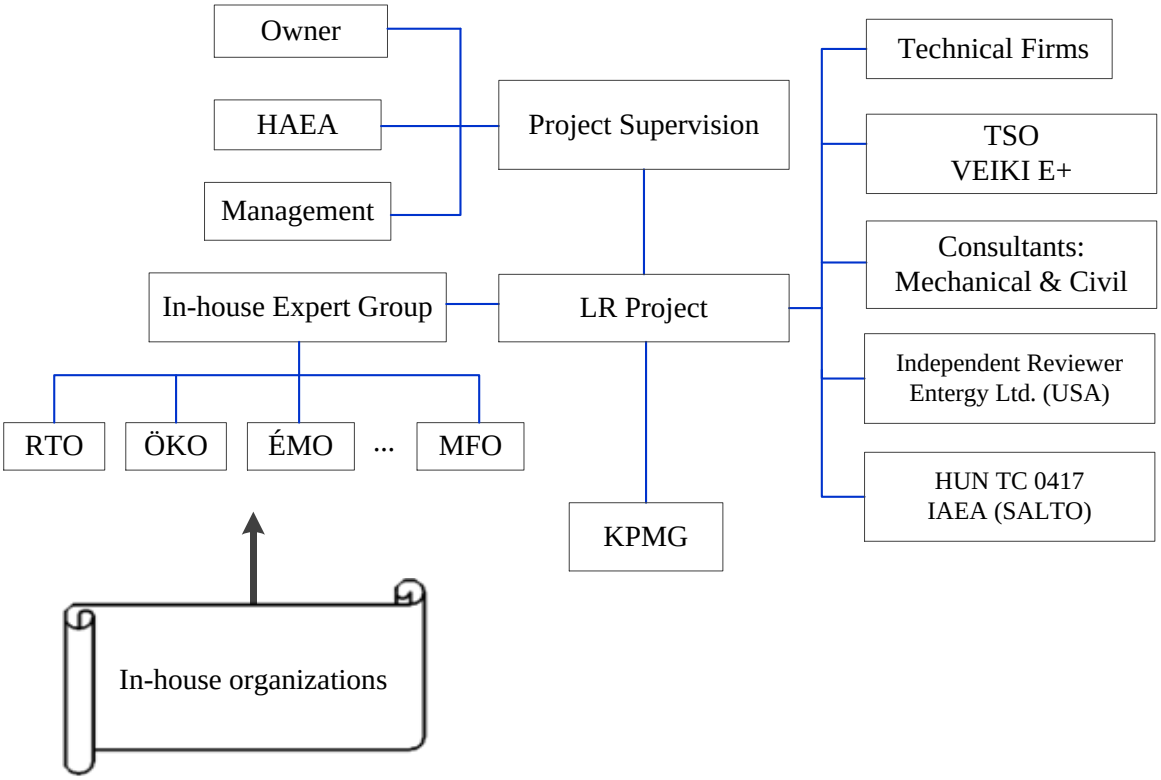


FIG. 60. Project Organization for developing the application report for LR.

An in-house forum of experts helps with achieving 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 forum also helps with the in-house verification process and the approval phases. These activities and relationships are represented in figure 60.

Management regularly reviews progress against the project plan. A weekly meeting chaired by the Head of the Technical Support Department is held to discuss progress and future activities and a monthly meeting is chaired by the General Manager of the plant where issues that may have arisen are reconciled, gaps are filled, and issues resolved.

The work of the project is supported by the TSO Company and by dedicated Hungarian engineering firms and universities. The formal application for Licence Renewal was reviewed and commented upon independently by expert firms (such as Entergy Ltd.) and by a formal SALTO mission conducted by the IAEA.

7.8. INTERNATIONAL COMMON APPROACH

The governmental and legal framework have the purpose of ensuring the effective regulatory control of facilities and activities in the country. Other responsibilities and functions are also included, such as liaison within the global safety regime and liaison for providing the necessary support services for the purposes of safety (including radiation protection), emergency preparedness and response, nuclear security and the State control and accounting of, nuclear material The need for requirements governing an nuclear effective regulatory body and the definition of a regulatory framework. The regulatory body develops strategies and promulgates regulations in implementation of laws and policies. One of such regulations with the regulatory framework for safety, are the rules for setting up organizations to deal with operations of a nuclear power plant, including SSC ageing management. Figure 61 shows the processes used by member states to ensure their organizations, their authorization procedures regarding operation and safety policies comply with the regulatory framework criteria.

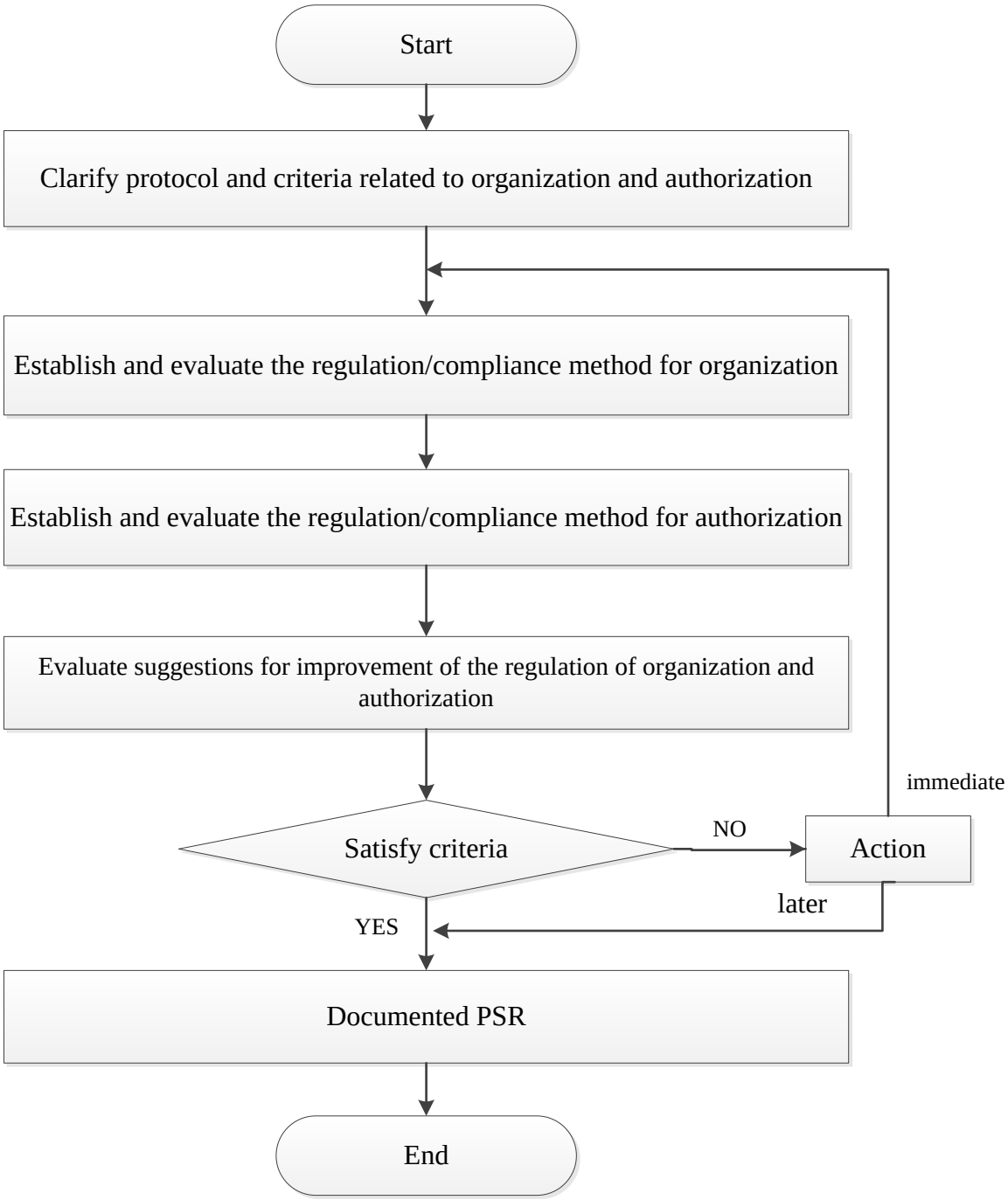


FIG.61 Compliance with the regulatory framework

7.9. CODES AND STANDARDS

The international standards related to NPP ageing management are:

— The Ageing Management for Nuclear Power Plants, Safety Guide NS-G-2.12 from the International Atomic Energy Agency (IAEA)
— The Safe Long Term Operation of Nuclear Power Plants, IAEA Safety Report Series No. 57,
— The safety aspects of nuclear power plant ageing IAEA-TECDOC-540.
— The Glossary of Nuclear Power Plant Ageing from the Organisation for Economic Cooperation and Development (OECD) Nuclear Energy Agency

In the context of ageing management an important milestone is reached at the time of a periodic safety report submission or of a license renewal application.

In the United States and in some of the countries operating US-designed plants, the license renewal is governed by the Code of Federal Regulations (10 CFR Part 54). [16] Operators are required to provide an integrated plant assessment (IPA), an evaluation of time-limited aging analyses (TLAAs), a supplement to the plant's final safety analysis report (FSAR), any necessary changes to the plant's technical specifications (along with related justifications) in accordance with USNRC's Reg. Guide 1.188. [58]

A systematic tool used by the US NRC is "The Generic Aging Lessons Learned (GALL) Report" [59], which contains the US NRC staff's evaluation of the existing plant programs. The information in the GALL Report has been incorporated into NUREG-1800 "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," [60] as directed by the Commission, to improve the efficiency of the license renewal process.

At the time of the original construction licence, an operator is required to submit a report on the potential environmental impact of the proposed station and associated facilities. This report is regulated by Regulatory Guide 4.2, [61] and Regulatory Guide 4.2S1 - Supplement 1 [62]
Renewal of a nuclear power plant operating license requires an update to the original environmental assessment through the preparation of an environmental impact statement (EIS), regulated by NUREG-1437 Volumes 1, 2, 3 and supplements. [63] [64] [65] These Regulatory guides are periodically reviewed and supplements are issued as the US NRC gains experience from plant sites that have undergone environmental reviews in the context of their licence renewal applications.

7.10. RECOMMENDATIONS

This section offers recommendations on ageing management in connection with the long term operation of a nuclear power plant. To facilitate LTO of a nuclear power plant, the operating organization should demonstrate, and the regulatory body should oversee, that the safety of the nuclear power plant is acceptable when compared with current safety standards. The most important issue with respect to safe operation and in view of preparing the plant for LTO is the understanding of the ageing mechanisms of the plant SCCs and the management of the impact of ageing on plant safety. The operator first preoccupation should be to ensure at all times the integrity of the plant defence in depth barriers and of the safety system performance. 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 methodology. The tools usually required to reach this understanding are:

— Analysis of operating experience (Review of event reports)
— Ranking of parts & components most subject to failures
— Classification of Failure severity (Incipient, degraded, catastrophic)
— Identification of failure mechanisms
— Early detection methods through plant surveillance testing and maintenance programmes

The in-depth review of ageing management should ensure that plant programmes and practices that will be used to support the management of ageing effects during long term operation are reviewed and are consistent with the generic attributes of an effective ageing management programme such as that given in Table 1 "effective ageing management programmes".

The review process should involve the following main steps:

— An appropriate screening method to ensure that structures and components important to safety will be evaluated for long term operation;
— Demonstration that the effects of ageing will continue to be identified and managed for each structure or component during the planned period of long term operation;
— Revalidation [1] of safety analyses that were developed using time limited assumptions, to

[1] Revalidation of safety analyses with time limited assumptions is an assessment of degradations under normal service conditions applied to safety analyses done at the time of first design on the basis of a specified length of plant operation. It would include fatigue calculations, pressurized thermal shock analysis and equipment qualification of electrical and instrumentation and control cables [1] [2]

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 throughout the planned period of long term operation.

In order to consider time limited ageing phenomena, it is necessary to evaluate for each of the risk significant components, the increase of the likelihood of their unavailability with time as described in IAEA's TECDOC 540 – Safety aspects of NPP ageing [66].

A time limited analysis method should provide the 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 must 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, the understanding of the time dependant risk factors is normally required in periodic safety review or licensing renewal application to the regulatory body. 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, on-line and otherwise, to help achieve the early detection of material degradation symptoms. The data obtained from surveillance or inspection in various plants world-wide In-Plant Reliability Data System (IPRDS) support the recent ASME code emphasis on pump and valve testing to improve the reliability of pumps/ valves and diesel generators.

Data reporting system should include key information needed to evaluate ageing concerns such as materials involved, environmental conditions contributing to failure, an assessment of the capability to detect component degradation in the incipient state before failure. In this research is essential to achieve a correct understanding of the causes of instrument drifts and prevent piping and other component failures.

In addition, the following two technical areas that are deemed to be especially significant or closely related to ageing management should be considered in an ageing management programme.

— Equipment qualification: equipment qualification (EQ) is a pre-requisite for long term operation of NPPs and qualification of equipment important to safety in NPPs ensures its capability to perform designated safety functions on demand under postulated service conditions including harsh accident environment resulting from postulated design basis accidents.
— Periodic safety review : the objective of the review of management of ageing in a periodic safety review is to determine whether ageing in a nuclear power plant is being effectively managed so that required safety functions are maintained, and whether an effective ageing management programme is in place for future plant operation

8. INNOVATION TECHNIQUES AND R&D

The selection of R&D tasks, the development of new knowledge and the collection and incorporation of innovative techniques and new information coming back from research are all activities that usually imply large investments and hence require a systematic approach.

8.1. R&D AND MITIGATION TECHNOLOGIES

Ageing management studies in support of operating nuclear power plants have been conducted in many countries as plants approach their design life and safety reviews for longer term operation permits are being performed. Many insights have been gained and knowledge gaps filled for components such as steam generators. Wear assessment programmes have been developed and results have been incorporated into thermal hydraulic codes, flow-induced vibration and wear assessment codes. R&D was conducted by EPRI and others to develop material reliability programmes to check material integrity and manage material ageing. The susceptibility of nickel-alloy (Alloy 600) to PWSCC has been a generic issue worldwide, R&D developed a good understanding of the degradation mechanism of alloy 600, based on which NPP 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 periodic safety review, R&D may be needed to acquire supplementary information or to generate new understanding of a certain degradation and derive the correct mechanism before designing and implementing any mitigation or corrective actions. The R&D results must lead to solutions which will eventually be incorporated together with their justification into the PSR safety documentation. Once R&D results are available a process to turn the understanding into engineered solutions and operational practice is needed. One such process is shown figure 62.

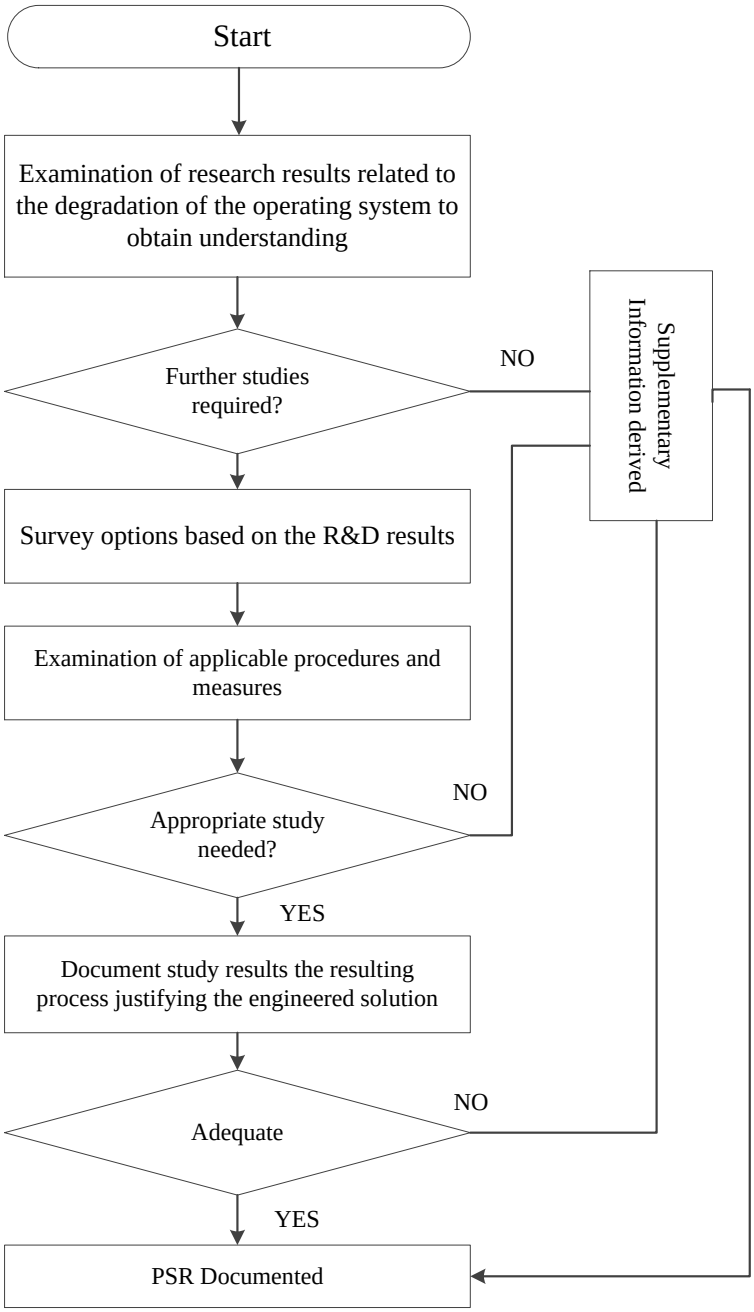


FIG.62. Procedure to incorporate R&D finding into the PSR documentation

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 supplementary 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, all possible options must be surveyed, examined, the best option capable of adequately mitigating or correcting the consequences of the degradation must be selected. Practical measures must be engineered, justified and their implementation appropriately planned. The study results and the engineered solution with its justification including calculation and verification should be reported in the PSR documentation. If the engineered solution requires supplementary vendor information, the latter must be obtained before submitting the PSR documentation to the regulatory body.

8.2. JAPAN

Figure 63 shows the validation of the diagnostic and prognostic process for concrete structures in use in Japanese nuclear power plants.

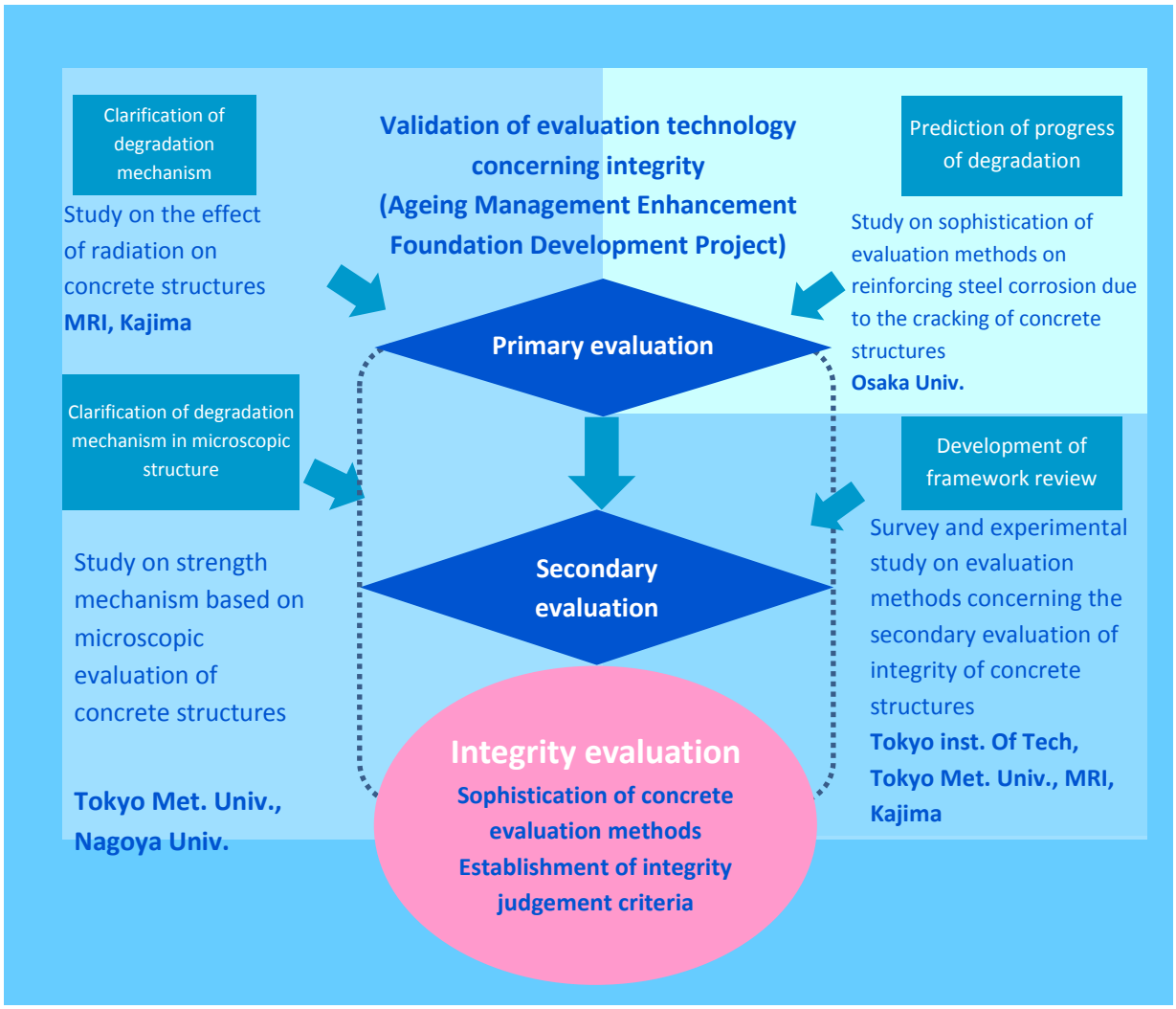


FIG.63. Validation of ageing management validation technology Tokyo Met. University

The primary integrity evaluation of the concrete structure is conducted by gathering information on the SSC condition (left hand side of the primary integrity evaluation diamond). 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 (shown on the right hand side) an adequacy assessment (degree of sophistication) of the evaluation methods, particularly when it comes to identifying the degree of corrosion of the reinforcing steel due to cracking in the concrete structure.

A secondary integrity evaluation is conducted by investigating degradation of the structure at the microscopic level (right hand side of the secondary integrity evaluation diamond). This investigation allows the evaluation of the remaining strength of the concrete structure. Shown on the right hand side of the secondary evaluation diamond is the development of the process used in the microscopic evaluation. This is done by surveying and conducting experimental studies on the evaluation methods used in the secondary integrity evaluation of the concrete structures.

8.3. China

A recent research programme undertaken in China in the area of RPV ageing management is shown in figure 64.

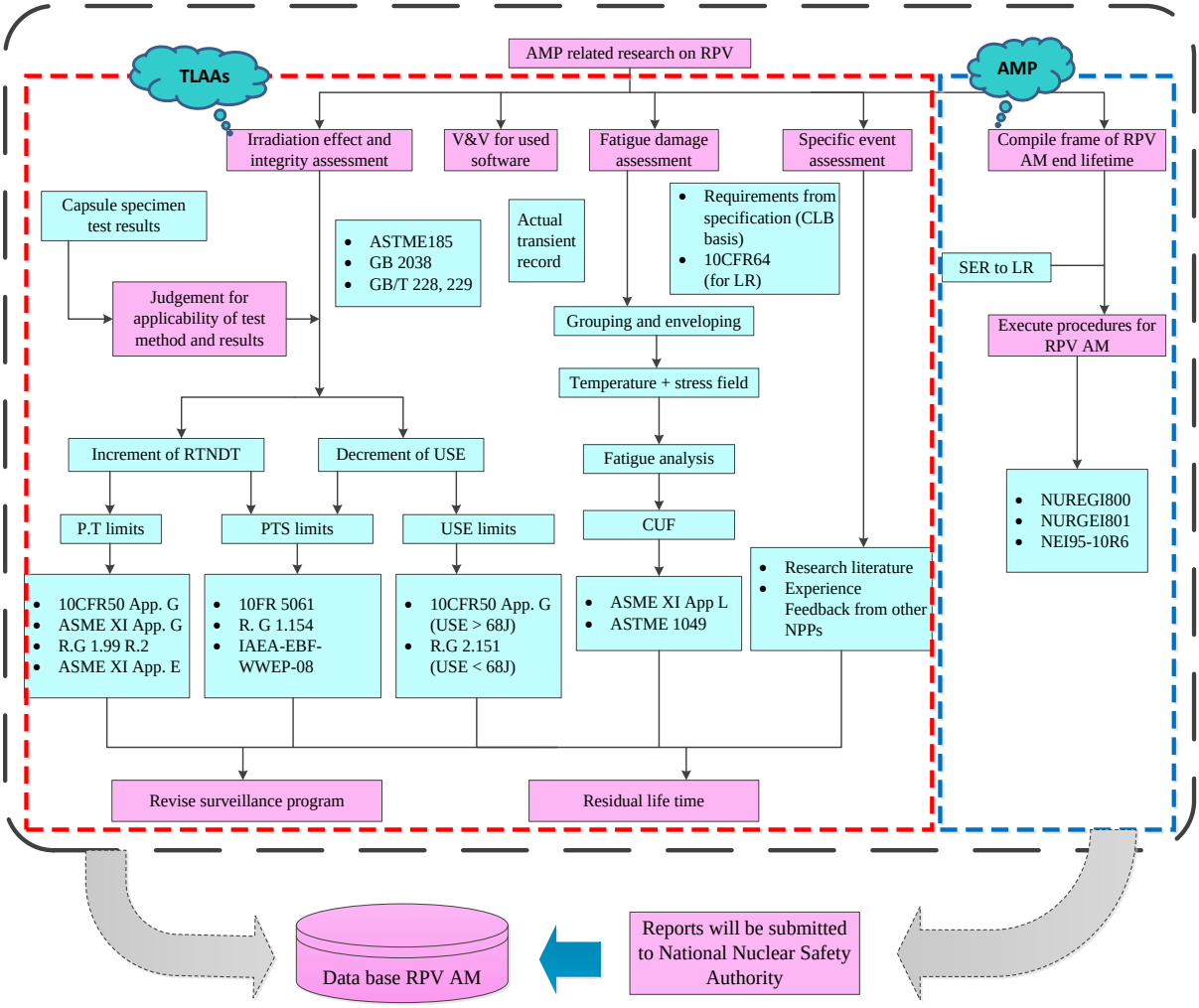


FIG. 64. RPV Ageing Management Programme (AMP) related research

In figure 65 is shown the RPV ageing management documentation system as it was applied to the Qinshan phase 1 plant.

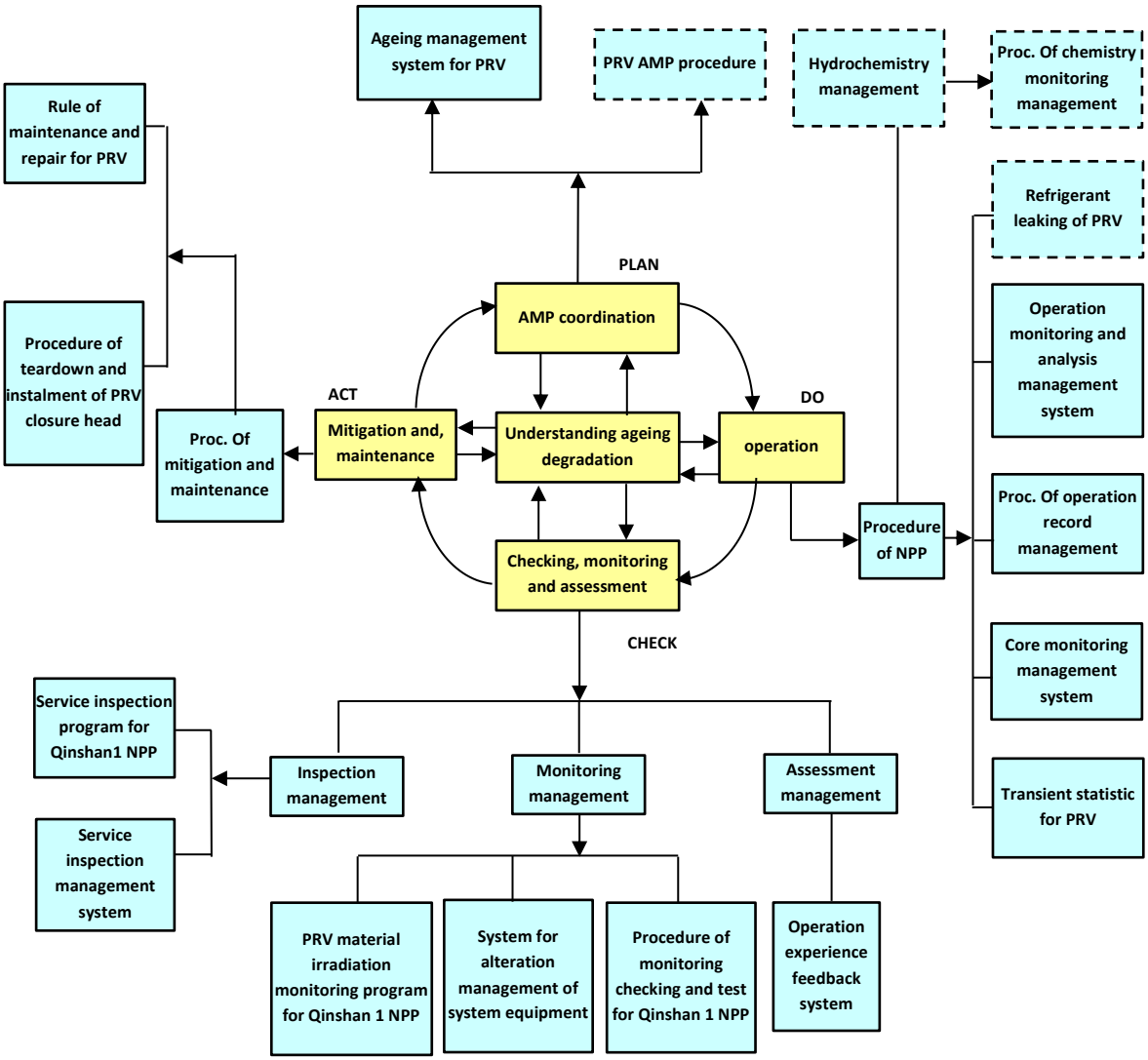


FIG.65. RPV ageing Management documentation system for Qinshan phase 1

At the centre in yellow directing everything else in blue around is the PLAN, DO, CHECK, ACT cycle, a systematic approach to ageing management. Beginning with the planning phase, the AMP coordination group after receiving feedback from the field and from OPEX and after building a sufficient understanding of all RPV degradation mechanisms develops the RPV ageing management plan and the ageing management procedures. For the "DO" step, Operations, represented by the yellow box to the right, applies the NPP 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 66 shows the ageing management activities and the milestone achievements since 2009 in the Chinese plants of Daya Bay and Ling Ao.

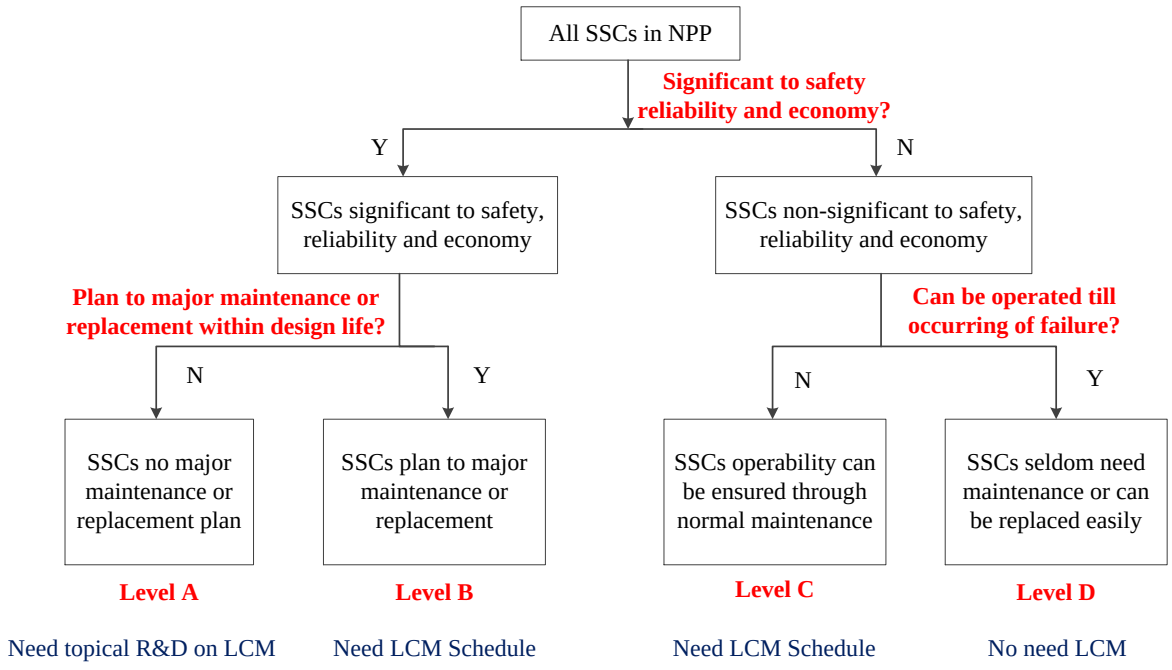


FIG.66. Screening, life cycle management and preventive maintenance activities in Daya Bay and Ling AO NPPs in China

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.

Those SSCs non-significant to safety, reliability and production can either be run to failure if they seldom require maintenance and they can be replaced easily without shutting down, or if their operability can be ensured through regular maintenance, a simple a periodic maintenance schedule is all they require.

On the other hand, the SSCs significant to safety, reliability and electricity production are the object of a major ageing management programme or replacement within the design life. Of these, the SSCs without a clear ageing management or replacement plan require topical R&D and the development of a Life Cycle Management (LCM) programme. Those critical SSCs with a major AMP simply need a schedule for their Life Cycle Management (LCM) programme.

Figure 67 represents the generic process, common to all NPPs in China, for the preparation, execution, assessment, corrective actions and reporting in a typical periodic safety review (PSR) in Chinese NPPs. As a starting point, the NPP Owner/Operator as the licensee submits the PSR plan inclusive of the scope, the specific requirements and deliverables to the regulator for approval. The activities can be grouped as those requested by the regulatory body and those resulting from the owner's own safety improvement, the plant diagnostics and the corrective action plan. The periodic safety review ends with the regulatory approval of the PSR corrective action plan.

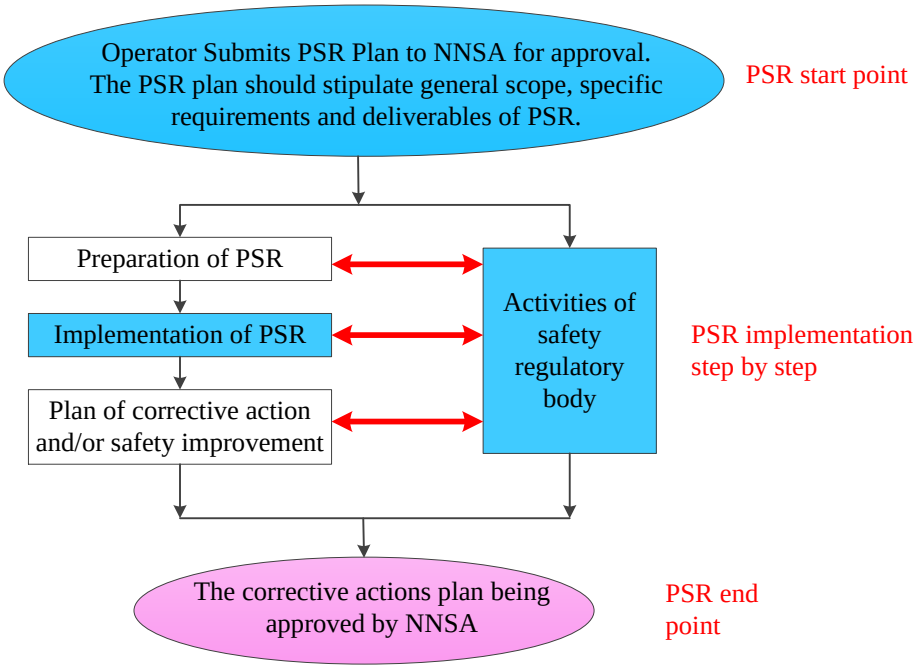
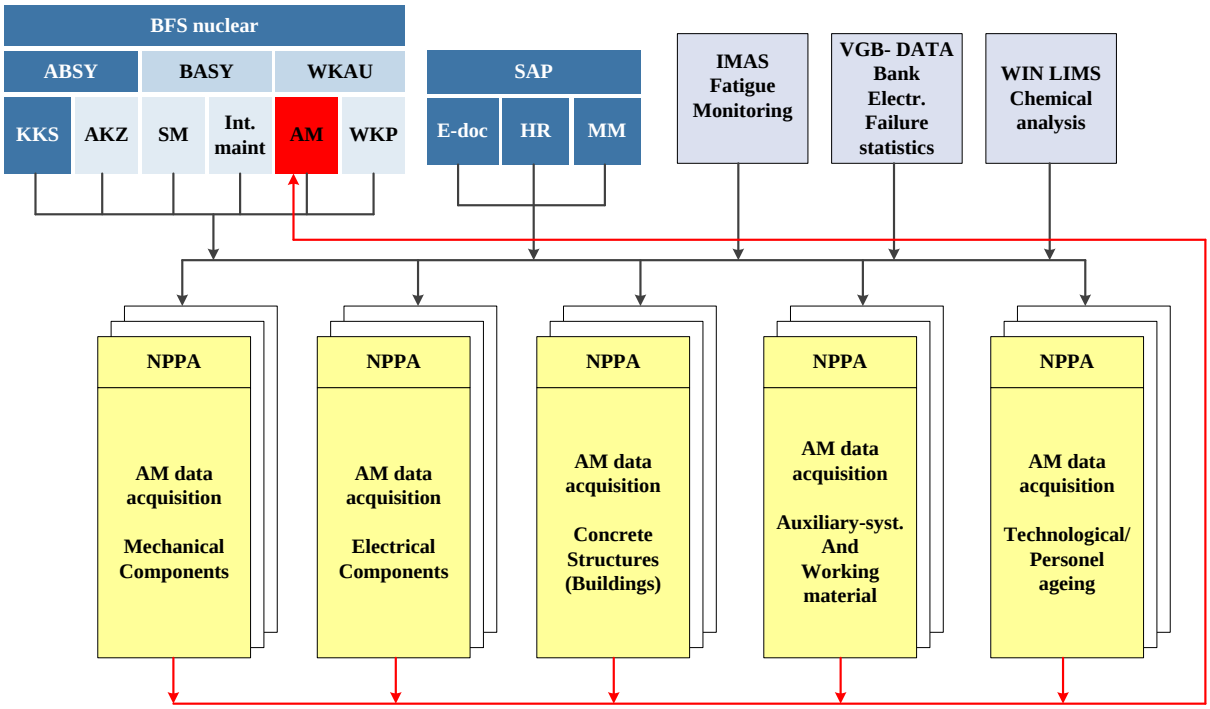


FIG. 67. Generic flowchart for Periodic Safety Reviews in Chinese NPPs

8.4. GERMANY

Figure 68 shows the data acquisition system used in German nuclear power plants. [67]



BFS: Betriebsprüfungssystem = Nuclear Operation Management System
ABSY: Anlagenbeschreibungsystem = Component Specification Database
BASY: Betriebsauftrags- und Störmeldesystem = Normal and Upset Condition Reporting System
AKZ: Anlagenkennzeichnungssystem = Equipment tagging system
KKS: Kraftwerk-Kennzeichensystem = Unique Component Designation and Identification System
MM: Material Management
SM: Störmangelmeldungen = Alarm Annunciation and Recording System
WKAU: Wiederkehrende Prüfungen Ausgabe = Comprising the Ageing Management Programming Database
WKP: Wiederkehrende Prüfungen = Periodic Testing
HR: Human Resources
WIN LIMS: Windows labour-Information-Management System

FIG.68 Data acquisition and management system in German Plants

The top boxes from left to right show the various information systems:

a. The first on the top left (BFS) Betriebsprüfungssystem is the nuclear operation management system which comprises:

— The Anlagenbeschreibungsystem (ABSY) = the component specification system, inclusive of the Anlagenkennzeichnungssystem (AKZ) = the equipment tagging system and of the Kraftwerk-Kennzeichensystem, (KKS), the unique designation and identification system

— the Betriebsauftrags- und Störmeldesystem (BASY), the normal and upset condition reporting system, inclusive of the Störmangelmeldungen (SM) the alarm annunciation and recording system

— the Wiederkehrende Prüfungen Ausgabe (WKAU) inclusive of the Wiederkehrende Prüfungen (WKP) (periodic testing programme) and of the ageing management (AM) programme

b. The SAP data management system which is the enterprise resource and information system, comprising:

— the electronic documentation management system,

— the human resource training & qualification database,

— the material management module

c. The fatigue monitoring system (IMAS)

d. The electrical failure statistical data-bank

e. The chemical analysis system.

In the yellow boxes below the first row of blue and white boxes are shown the data acquisition systems for the various disciplines, the auxiliary systems, the system ageing and the personnel training and qualification data system.

The output from all data acquisition systems goes to the Ageing Management database and to the PLiM database (if one is in force in a specific plant)

Figure 69 represents an Integrated Maintenance Management programme as applied in Germany. The concept follows the PLAN, DO, CHECK, ACT cycle but the diagram focuses on a concept designed to guarantee the remaining life time of a critical SSC. Starting from the left, the yellow box represents the pre-operational steps that should be taken pro-actively 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 causes 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 is made up of a pro-active part shown as shaded and crossed out with monitoring devices and measures to control degradation with in parallel condition verification steps, involving analytical evaluation of operating experience feedback, stress analysis, fracture mechanics etc. It is shaded because it is built-into the system as time-based and frequency-based elements. The other side of the operation period is represented by the dark yellow box to the right which consists 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 making use of in-service inspections, leak detection and exploratory investigations by maintenance.

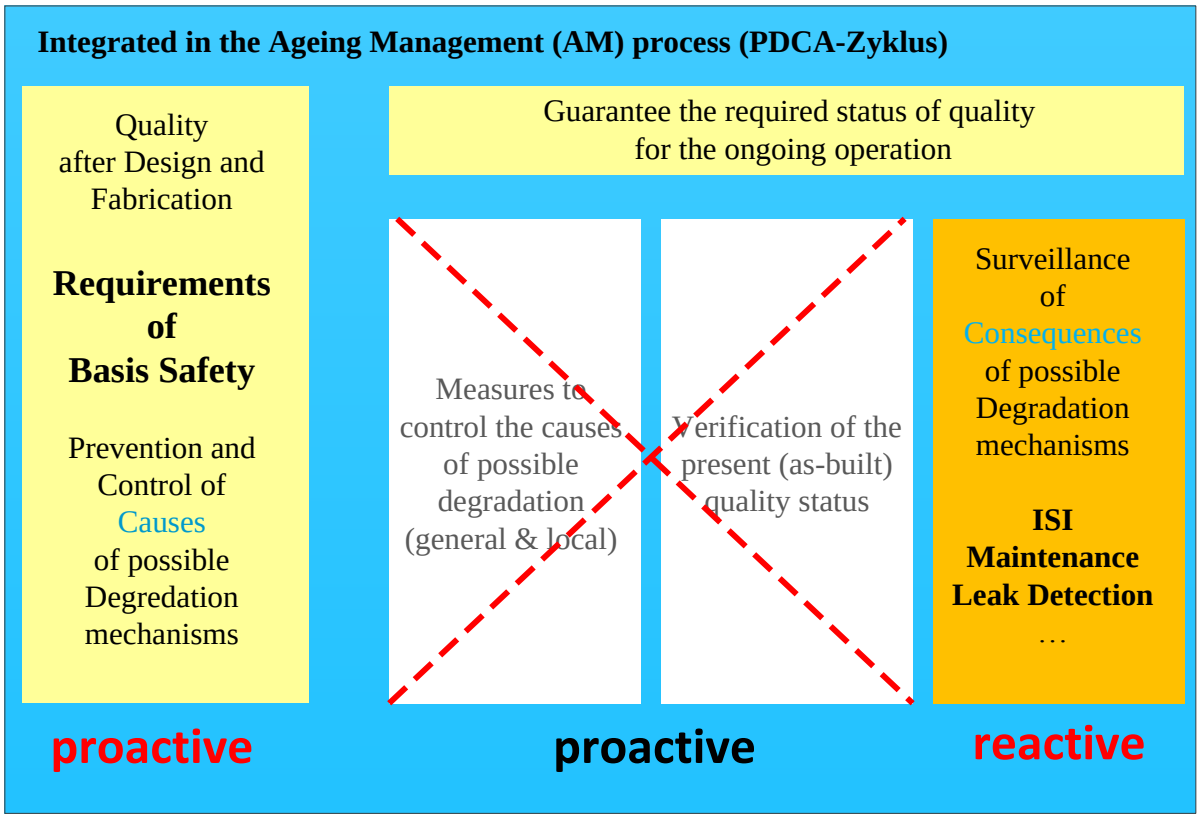


FIG. 69. Integrated Ageing Management Process as applied in Germany.

8.5. INTERNATIONAL COLLABORATION ON AGEING MANAGEMENT

Standardization of Ageing Management Programmes is achieved through International collaboration on ageing management and through the cooperation of International agencies operating in the nuclear power field. They include OECD/NEA, the IAEA and lately also organizations such as WANO, EPRI and is achieved through working groups and international meetings. Figure TTMM is a representation of the various aspects handled by the international agencies and relevant international working groups and meetings. The OECD/NEA released a Long Term Operation called the green booklet for the colour of its cover. This is a general guide for regulatory bodies regarding application for operation beyond the nominal design life of a plant. Cooperation initiatives include the Piping Failure Data Exchange (OPDE) Project. The OPDE Project includes events of interest with regard to piping failures. It covers the main safety systems piping and those non-safety piping systems whose failure 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 Stress Corrosion Cracking and Cable Ageing Project (SCAP) 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 the commendable practices 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 (CODAP) [38] is an umbrella programme that combines the follow-up of both the OECD Pipe Failure Data Exchange Project (OPDE) and the stress corrosion cracking (SCC) and Cable Ageing Project (SCAP). The IAEA has sponsored an International Generic Ageing Lessons Learned (IGALL) for Nuclear Power Plants through an extra-budgetary programme established in September 2010 [23]. Its first phase was completed in 2013 and its second phase in 2014-2015. The IGALL Programme has developed and maintains documents and a database to provide a technical basis and practical guidance on managing ageing of mechanical, electrical and instrumentation and control components and civil

structures of nuclear power plants important to safety in order to support the application of the IAEA guidance on: Specific Safety Requirements on Design [3], Commissioning and Operation [4]; Safety Guide on Ageing Management [1]; Safety Guide on Periodic Safety Review [57]; and Safety Report on Safe Long Term Operation [10].

The IGALL database contains:

— A generic sample of ageing management review tables;
— A collection of proven ageing management programmes;
— A collection of typical time limited ageing analyses

International cooperation meetings fostering collaboration on NPP ageing management include the following programmes:

— PARENT on cyclotron-produced radionuclides for medical diagnostics and therapies
— RCOP, a cooperative project among China, Japan and the Republic of Korea) aimed at the establishment and broader use of information networks.
— IGRDM, the International Group for Radiation Damage meeting
— Zorita is a cooperative research project on ex-plant materials from José Cabrera (Zorita) NPP (ZIRP) with the Spanish Consejo de Seguridad Nuclear (CSN, SPN). a site under decommissioning.
— IFRAM is the International Forum for Ageing Management, a global cooperation initiative in identifying and addressing technical issues in the ageing of nuclear power plants (NPPs)

8.6. INTERNATIONAL ACTIVITIES ON AGEING MANAGEMENT

The IAEA has long been involved in activities aimed at supporting member states with their NPP ageing management programmes both in terms of safety standards, guides and services for ageing programmes and programme implementation guidelines

8.6.1. AM initiatives of IAEA's activities

8.6.1.1. Safety standards, guides and services for ageing programmes

The higher level documents on ageing management are the IAEA safety standards and safety guides on ageing management NS-SSR-2/1 and /2 [3[4]] which contain the requirements related to NPP Ageing management such as

— Requirement 14: The operating organization shall ensure that an effective ageing management programme is implemented to ensure that required safety functions of systems, structures and components 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 time-frame established in the licence conditions, design limits, safety standards and/or regulations

Another IAEA guiding document still part of the IAEA safety standards is: Safety Guide No. NG-G-2.12 Ageing management for nuclear power plants [69] whose objective is that of providing a set of guidelines and recommendations for managing ageing of Systems Structures and Components (SSCs) important to safety in nuclear power plants. It covers SSCs mainly focusing on physical ageing but it also includes the management of obsolescence.

Other IAEA activities related to ageing management are Peer Review Missions that are services to the Member States in support of their Long Term Operation programmes. These missions are conducted by international expert teams and led by IAEA representatives. They are regulated by the IAEA-EBP-SALTO guidelines: "Safety aspects of Long Term Operation (SALTO) of Water Moderated Reactors" [70]. This document contains recommendations on the scope and content of a typical programme for safe long term operation of water moderated reactors. It deals with the framework necessary to embark in long term operation (LTO) or operation beyond the service time assumed in the original design covering in detail mechanical components and materials, electrical components and cabling, instrumentation and control equipment and also with structures and structural elements.

The IAEA has issued a safety report series No. 57 Safe Long Term Operation of Nuclear Power Plants [10] which 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 ageing degradation, Safety analysis revalidation using time limited assumptions, documentation required and regulatory oversight. This document can be used as preventive guidance before NPP Operators begin the development of Long Term Operation Programmes for their NPPs.

8.6.1.2. International Generic Ageing Lessons Learned (IGALL)

Another IAEA ageing management initiative relates to the development of the International Generic Ageing Lessons Learned (IGALL) database. [23] This activity resulted in the preparation of a state of the art IGALL report containing an itemized table of recommended ageing management programmes particularly aimed at facilitating the safe long term operation of either a new or an operating NPP. The IGALL database was developed by over 100 key international experts on ageing management issues from 13 Member States and the European Commission represented by regulators, utilities, vendors, manufacturers, and the originators of codes and standards. The IGALL report was based on the results of the application of systematic approach to ageing and maintenance techniques for a variety of NPP structures and components as described in safety guide NS-G-2.12 [69]. It contains consolidated international information on SSC degradation mechanisms, 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 ageing management programmes. It is intended to function as a guide for the implementation of AMPs for NPPs. It serves the main nuclear power technologies in use by the member states (PWR, BWR, VVER, CANDU, PHWR). The intent is for IGALL to be a live document to be updated at least every 5 years.

8.6.1.3. AM programme initiatives of IAEA's Coordinated Research Programmes (CRP)

Beside safety standards and safety related requirements and documents related to NPP ageing and ageing management, the IAEA has also issued a series of implementation engineering documents and application models describing how the safety level requirements can be met in practice. A list of these documents published to date is reported in the bibliography of this handbook.

IAEA's Nuclear Power Engineering division has sponsored a number of Coordinated Research Programmes (CRP), and international workshops. They are an important IAEA mechanism 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 availability of the developed results through electronic media and to production of relevant technical document, or guideline, containing detailed description of the subject of the research.

Examples of CRPs are:

— Qualification, condition monitoring, and management of ageing of low voltage cables in nuclear power plants. 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 NPPs: 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 (SPD) techniques for the monitoring of SSC functions and integrity in nuclear power plants. [71]

— Review and benchmark of calculation methods of piping wall thinning due to erosion-corrosion in NPPs. Flow-accelerated corrosion is considered as wall thinning of steel piping due to turbulent and fast flowing water or wet steam that wears away the protective film (oxide layer) and leads to continued corrosion of the underlying metal. Wall thinning in piping and vessels because of FAC has caused sudden ruptures in high and moderate energy systems, resulting in plant transients and affecting safety/non safety related equipment by leaking steam and water and risking personnel safety.

8.6.1.4. IAEA Coordinated Research Projects on RPV integrity

The IAEA has sponsored a series of Coordinated Research Projects (CRPs) that have led to a focus on reactor pressure vessel (RPV) structural integrity application of measured best irradiation fracture parameters using relatively small test specimens.

Engineering judgement - approach used in time limited ageing analyses as well as in the design of all the plant structures and components take into consideration safety margins in safety coefficients (safety factors) which in the outcome bring considerable degree of conservatism. Real experience from testing materials, from structural analyses shows that the results obtained in such way in numerous cases do not reflect the real rate of damage, real condition of a given structure or component. A typical example may be fatigue analyses which with great help of tools (FEM, computational codes, worst material properties taken from standards) try to provide the plant with the results of damage based on real operating experience (transients associated with the changes of temperature and pressure and some other physical quantities) at infinitely small theoretical mathematical points. Structural analyses of the affected (concerned) materials, carried out at places that should manifest the usage factor value up to 1 show more-less just exactly that difference in safety factors.

This can be proven by independent analyses carried out by several research institutions, laboratories. In that light it is always extremely helpful and beneficial to have different comparative round-robin exercises, benchmark comparative studies, which are carried out, for example through the mechanism of the IAEA coordinated research activities (CRPs). The impact of different evaluation methodologies for testing of the same identical material has been studied many times over the past twenty years in the nuclear industry. It quickly has been recognized and understood that it is very important to have harmonized methodologies, codes and standards that are universally used for the investigation of the similar/same/identical material structures and components, for structures that have been designed by the same designer - technology supplier and/or produced by the same manufacturer.

The first project (or CRP Phase 1), "Irradiation Embrittlement of Reactor Pressure Vessel Steels," focused on standardization of methods for measuring embrittlement in terms of both mechanical properties and the neutron irradiation environment. The main results were published in 1975 in IAEA Technical Report Series (TRS) No.163 [72].

CRP Phase 2, "Analysis of the Behaviour of Advanced Reactor Pressure Vessel Steels under Neutron Irradiation," involved testing and evaluation of so-called advanced RPV steels that had reduced residual compositional elements (copper and phosphorus). The results were summarized in IAEA Technical Report Series (TRS) No. 265 [73].

CRP Phase 3 "Optimising Reactor Pressure Vessel Surveillance Programmes and Their Analyses" addressed the direct measurement of fracture toughness using irradiated surveillance specimens. A key achievement was the acquisition a series of RPV steels for radiation embrittlement research. The JRQ reference material was documented in IAEA TECDOC-1230 [74].

The main emphasis during CRP Phase 4 was the experimental verification of the Master Curve approach for surveillance size specimens. [75] Application included a large test matrix using the JRQ steel and other national steels including VVER materials. No differences in laboratories were identified, and results 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 [76]. The large CRP group consisted of 20 testing laboratories representing 15 Member countries. This CRP had two main objectives:

— Develop a large database of fracture toughness data using the Master Curve methodology for both precracked Charpy size and one-inch thick (25.4 mm) compact tension (1T-CT) specimens to assess possible specimen bias effects and any effects of the range of temperatures used to

determine T_0, either using the single temperature or multi-temperature assessment methods and

— Develop international guidelines for measuring and applying Master Curve fracture toughness results for RPV integrity assessment. The results were documented in IAEA Technical Report Series (TRS) No. 429 [16] and IAEA TECDOC-1435 [76].

CRP Phase 6 "Effects of Nickel on Irradiation Embrittlement of Light Water RPV Steels" comprised procurement of materials, determination of mechanical properties, irradiation and testing of specimens and microstructural characterization. The results clearly show the significantly higher radiation sensitivity of the high nickel weld (1.7 wt%) compared with the lower nickel base metal (1.2wt%), as documented in IAEA TECDOC-1441[77].

CRP phase 7 was focused on VVER-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 VVER-440 RPVs that provides better predictive capabilities based upon chemical content and neutron exposure. This new correlations were developed in a framework that better simulates the known embrittlement mechanisms for these steels, and was published in IAEA TECDOC-1442 [78]. The CRP was accomplished through the completion of four tasks:

— Collection of VVER-440 surveillance and other relevant data and input into the IAEA International Database on RPV Materials (IDRPVM),
— Analysis of radiation embrittlement data of VVER-440 RPV materials using the IDRPVM database,
— Evaluation of predictive formulae depending on material chemical composition, neutron flux and fluence
— Guidelines for prediction of radiation embrittlement of operating reactor pressure vessels of VVER-440 including methodology for evaluation of surveillance data of a specific operating unit.

CRP Phase 8 is an ongoing extension of CRP-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-8 include:

— Better quantification of fracture toughness issues relative to testing surveillance specimens for application to RPV integrity assessment,
— Development of approaches for addressing MC technical issues in integrity evaluation of operating RP Since 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 the coordinated research project CRP Phase 9 was to perform benchmark deterministic calculations of a typical Pressurised Thermal Shock (PTS) regime with the aim of comparing 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 used for RPV integrity assessment. This is the case not only between and VVER 440-230 and PWRs reactor types, but also within each group. These differences are based, in principle, on different codes and rules used for design, manufacturing and materials used for the various types of reactors on one side, and on the different level of implementation of recent developments in fracture mechanics on the other side.

Benchmark calculations were performed to improve the user qualification and to reduce the user effect on the results of the analysis. This addressed generic PWR and VVER reactor vessels types, as well as sensitivity analyses to check several points. The complementary sensitivity analyses showed that the following factors significantly influenced the assessment: flaw size, shape, location and orientation, thermo-hydraulic assumptions, material toughness. Applying national codes and procedures to the benchmark cases produced significantly different results in terms of allowable material toughness. This is mainly related to the safety factors used and approaches to postulated defects, postulated transients, representation of material toughness.

8.6.2. AM related initiatives of other international organizations

International initiatives regarding NPP ageing management have also been undertaken by other international organizations such as IFRAM, OECD-NEA, EC-JRC etc.

An International Forum for Reactor Ageing Management (IFRAM) has been formed to facilitate the sharing of information among participants world-wide on advances in materials ageing management (including diagnostics and prognostics) for NPPs.

Individual activities related to diagnostics and prognostics are underway at the European JRC (and in member nations of the EU), South Korea, China, Japan, India, Canada and the USA, where the Electric Power Research Institute (EPRI), the US Department of Energy and the US Nuclear Regulatory Commission are all engaged.

The US federally-funded activities, under DOE-NE, are organized in the Light Water Reactor Sustainability Programme (LWRS), with a programme office at the Idaho National Laboratory. Some people in the community addressing the technical challenges related to reactor life extensions, which involves universities and national laboratories, are considering how to integrate advanced diagnostics into new build and next-generation NPPs

8.6.2.1. NUGENIA

NUGENIA has been established as a natural progression of the NULIFE mission (it was envisaged to create NULIFE Association / Institute) and as a fruit of close collaboration with the SNETP and SARNET as shown in figure 70. The legal entity of an association was chosen after a careful review of all European possibilities. The Association being an independent legal entity is open to all nuclear stakeholders endorsing the Statutes. All members of SNETP are welcome as are international and non-European organisations.

NUGENIA is the result of the integration process of 3 groups active in the nuclear energy research field of Generation II & III technologies:

— The first of the three SNETP pillars: Technology Working Group Gen II & III
— The NULIFE Network of Excellence on nuclear plant life management
— The SARNET Network of Excellence on severe accidents

Each of these groups shall bring its expertise, strength and experience together in NUGENIA. The first group has an SNETP mandate of action ending in June 2012 that NUGENIA aims to receive at that time. The two Networks of Excellence have a common goal of reaching their sustainability beyond the current contracts with the European Commission, ending respectively in June 2012 for NULIFE and in March 2013 for SARNET.

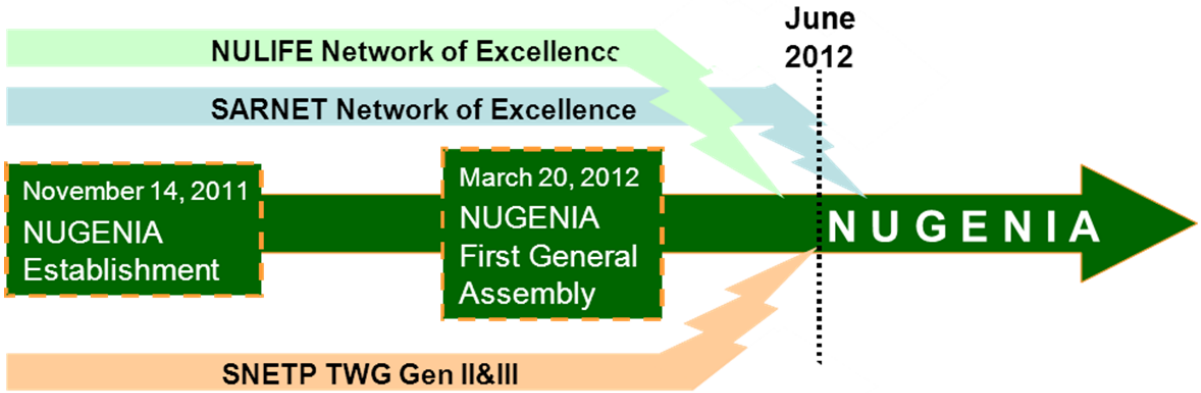


FIG.70 - Establishment of NUGENIA

As a product demonstrating the success of the 3 networks (SNETP TWG Gen II&III, NULIFE and SARNET), NUGENIA is set up to be the starting point of a more ambitious and united community to advance the safe, reliable and efficient operation of nuclear power plants. NUGENIA shall provide, in a transparent and visible way, a scientific and technical basis by initiating and supporting international R&D projects and programmes. NUGENIA will contribute to innovation and facilitate implementation and dissemination of R&D results. NUGENIA scope of activities covers 8 main technical areas:

— Plant safety and risk assessment
— Severe accidents
— Improved Reactor Operation
— Integrity assessment of Systems, Structures and Components
— Fuel Development, Waste and Spent Fuel Management and Decommissioning
— Innovative LWR design & technology
— Harmonisation
— In-service Inspection and Non Destructive Examination[79]

Definitions taken from the IAEA Safety Glossary 2007 Edition

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 throughout 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.7

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.

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.*

legal person

Any organization, corporation, partnership, firm, association, trust, estate, public or private institution, group, political or administrative entity or other person designated in accordance with national legislation who or which has responsibility and authority for any action having implications for *protection and safety*.

- Contrasted in legal texts with natural person, meaning an individual.
- See also *applicant, licence* and *registration.*

licence

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***).

! In IAEA usage, a *licence* is a particular type of *authorization*, normally representing the primary *authorization* for the *operation* of a whole *facility* or *activity*. The conditions attached to the *licence* may require that further, more specific, *authorization* or *approval* be obtained by the *licensee* before carrying out particular *activities*.

licensee

See *licence*.

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).

licensing process

See *licence*.

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.

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 *ageing management.*

life management (or lifetime management)

See *ageing management*.

long term operation [2]

Long term operation means operation beyond an established timeframe set forth by, for example, licence term, design, standards, licence and/or regulations, which has been justified by safety assessment, with consideration given to the life limiting processes and features of *structures, systems and components (SSCs)*. If an operating organization decides to pursue *long term operation*, justification is supported by the results of *periodic safety reviews*, including review of management of ageing, and overseen by the *regulatory body*.

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.

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[2] The definition of **long term operation** is taken from Section 2.17 of the IAEA Safety Guide on **ageing management** for nuclear power plants [20].

- 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*.

nuclear installation

1. A *nuclear fuel* fabrication plant, *research reactor* (including subcritical and *critical assemblies),* nuclear power plant, *spent fuel storage facility,* enrichment plant or *reprocessing facility.*
   - This is essentially any *authorized facilities* that are part of the *nuclear fuel cycle* except *facilities* for the mining or processing of uranium or thorium ores and *radioactive waste management facilities.*

2. [For each Contracting Party, any land based civil nuclear power plant under its jurisdiction, including such *storage,* handling and *treatment facilities* for *radioactive materials* as are on the same site and are directly related to the operation of the nuclear power plant. Such a plant ceases to be a *nuclear installation* when all *nuclear fuel elements* have been removed permanently from the reactor core and have been stored safely in accordance with approved *procedures,* and a *decommissioning* programme has been agreed to by the *regulatory body.*]

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.

- 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 equipment

Equipment certified as having satisfied *equipment qualification* requirements for the conditions relevant to its *safety function(s)*.

qualified life

See *life*.

safety case

A collection of arguments and evidence in support of the *safety* of a *facility or activity*.

- This will normally include the findings of a *safety assessment* and a statement of confidence in these findings.
- For a *repository*, the *safety case* may relate to a given stage of development. In such cases, the *safety case* should acknowledge the existence of any unresolved issues and should provide guidance for work to resolve these issues in future development stages.

safety function

A specific purpose that must be accomplished for *safety*.

- Reference [21] 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.*

- 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.*

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.

surveillance testing

Periodic testing to verify that *structures, systems and components* continue to function or are capable of performing their functions when called upon to do so.

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ABBREVIATIONS

AE	Architect Engineer
AMP	Ageing Management Programme
AMR	Ageing Management Review
ASME	American Society of Mechanical Engineering
ASN	Autorité de Sûreté Nucléaire (French Nuclear Safety Authority)
ASTM	American Society for Testing and Materials
ATWS	Anticipated Transient Without Scram (in LWRs)
BDBA	Beyond Design Basis Accident
BOP	Balance of Plant
BWR	Boiling Water Reactor
CANDU	Canada Deuterium (Canadian pressurized heavy water reactor)
CLB	Current Licensing Basis
CM	Configuration Management / Condition Monitoring
CNSC	Canadian Nuclear Safety Board
CSA	Canadian Standard Association
DM	Dissimilar metal
DB	Design Basis
DBA	Design Basis Accident
DHC	Delayed Hydride Cracking
DSA	Deterministic Safety Analysis
ECCS	Emergency Core Cooling System
ECT	Eddy-Current Testing
ÉDF	Électricité de France
EQ	Environmental Qualification
FAC	Flow-Assisted Corrosion
GALL	Generic Ageing Lessons Learned
GS	Garter Springs in Fuel Channel type reactors
ISI	In-Service Inspection
LA	Life Assessment
LBB	Leak Before Break
LR	License Renewal
LRA	License Renewal Application
LTO	Long Term Operation
MSI	Maintenance Surveillance Inspection
NDT	Non-Destructive Test
NEI	Nuclear Energy Institute
OM	Operation and Maintenance
PLEx	Plant Life Extention
PLiM	Plant Life Management
PSA	Probabilistic Safety Analysis
PSAR	Preliminary Safety Analysis Report
PSR	Periodic Safety Review
PTS	Pressurized Thermal Shock
PWR	Pressurized Water Reactor
RCM	Reliability-Centred Maintenance
RCS	Reactor Coolant System
RI-ISI	Risk-informed In-Service Inspection
ROP	Reactor Oversight Process (USNRC)
RPV	Reactor Pressure Vessel
RTNDT	Reference Temperature for Nil Ductility Transition (in RPV Belt line materials)
RTD	Resistive Temperature Detector
SAM	Severe Accident Management
SAMG	Severe Accident Management Guidelines

SIT	Structure Integrity Test
SSC	Structures Systems and Components
SWC	System Window Coordinator (CANDU – Field terminology)
TLAA	Time Limited Ageing Assessment
TSO	Technical Support Organization
VVER	Vodo-Vodyanoi Energetichesky Reactor (Water-Water Power Reactor)
WANO	World Association of Nuclear Operators

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