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PLANT LIFE MANAGEMENT MODELS FOR LONG TERM OPERATION OF NUCLEAR POWER PLANTS

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FOREWORD

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

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

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

As of the end of 2014, there were 438 nuclear power plants operating around the world. Of these plants, 225 had been in service for over 30 years. When these plants reach the end of their operating licence, they will undergo a periodic safety review and an ageing assessment of their essential structures, systems and components to validate or renew their licence to operate beyond the originally intended service period.

Three different plant life management models have been used to qualify plants to operate beyond their operating licence. The models are based on the licence renewal application concept, the periodic safety review process, or a combination of both.

In the light of the lessons learned from the 2011 accident at the Fukushima Daiichi nuclear power plant, regulators and operators around the world reviewed the safety of their nuclear plants and their accident response capabilities to better prepare for beyond design basis accidents. Any design changes and accident mitigation measures introduced by the safety review, such as measures adding operating flexibility under severe accident conditions or increased robustness of a plant in response to beyond design basis conditions, were to be included in nuclear power plant ageing management programmes and addressed in long term operation (LTO) applications to ensure plant functionality at all times, including operation beyond the plant design life.

In this report, the IAEA has collected samples of licensing practices for LTO from Member States. The various plant life management models used to obtain LTO authorizations are described here and comparisons are drawn against the standard periodic safety review model. Lessons learned and warnings about possible complications and pitfalls are also described to minimize the risk of licensing for LTO applications.

The IAEA expresses its appreciation for the generous contributions of several Member States. The IAEA is particularly grateful to the members of the working group for their contribution to the report.

The IAEA officers responsible for this publication were F. Nuzzo and K.S. Kang of the Division of Nuclear Power.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

Many Member States have given high priority to licensing their nuclear power plants (NPPs) to operate for terms longer than the time frame originally anticipated (e.g. 30 or 40 years). As of December 2014, out of 438 NPPs operating in Member States, approximately 80% had been in service for over 20 years. The task of managing plant ageing is assigned in most Member States to an engineering discipline called plant life management (PLiM), which applies a systematic analysis methodology to the ageing of structures, systems and components (SSCs). Specifically, PLiM can be defined in one sentence as the integration of ageing and economic planning for the purpose of maintaining a high level of safety and optimizing plant performance by dealing successfully with extended life ageing issues, maintenance prioritization, periodic safety reviews (PSRs), education and training. This discipline is particularly useful in helping plant owners make an informed decision on continuing to operate their plants longer than their originally assumed design life.

From the licensing standpoint, there are three conceptual approaches that licensees use to obtain an authorization to operate their NPP unit beyond its design service life. One approach is based on the licence renewal application (LRA) concept, the second on the PSR concept and the third on a combined approach. The United States of America practices the LRA concept, while most European States and Japan use PSRs to obtain the authorization to continue operation of a plant beyond the original design life, also called long term operation (LTO). In some Member States (e.g. Hungary, the Republic of Korea and Spain), these two different concepts and related regulatory approaches have been combined, encompassing elements of both approaches to better meet local requirements.

Licence renewal applications in the United States of America and in States following the US model are based on the assumption that ageing management of active components and systems is adequately addressed by the maintenance rule or similar regulatory processes. The LRA prerequisites are:

— Integrated plant assessment to evaluate the ageing management of passive, long lived SSCs, to ensure that they can support continued safe plant operation beyond the 40 year term of the original licence;
— Assessment of SSCs with time-limited ageing to justify the additional years of operation;
— Environmental impact assessment for the additional service life.

If a licensee follows the maintenance rule and other US operating and licensing practices, it is likely that elements of the US LRA process may be incorporated into the LTO authorization process.

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 ten 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's 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.

Taking into account the diversity of approaches to LTO authorizations, the IAEA has collected, in this report, technical and licensing information on LTO authorizations from a range of Member States that disclosed their PLiM model and their experiences in NPP LTO. This report also contains detailed comparisons between PSR and LRA practices for the benefit of all Member States, including newcomers to nuclear power generation.

1.2. TERMINOLOGY

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

— *Plant life management (PLiM)*. This is defined as the integration of ageing and economic planning to optimize NPP investments in favour of safety, commercial profitability and competitiveness, while providing a reliable supply of electrical power. There are other definitions of PLiM. The Electric Power Research Institute in the United States of America produced a glossary of common ageing and PLiM terms. This glossary is being internationalized by the Organisation for Economic Co-operation and Development's (OECD's) Nuclear Energy Agency (OECD/NEA) and will probably become the basic communication tool in this field. When PLiM is properly applied, it can:
  - 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;
  - Provide NPP utilities/owners with the optimum preconditions for achieving LTO.

— *Long term operation (LTO).* The LTO of an NPP may be defined as operation beyond an established time frame set forth by, for example, licence term, design, standards, licence or regulations, which have been justified by safety assessment with consideration given to life limiting processes and features of SSC. Long term operation is conditioned by regulatory requirements and subject to regulatory authorization, and usually also to public or political acceptance. In practice, LTO is only possible when an appropriate safety assessment has been performed, and the results have been found to be favourable concerning safety of the NPP involved.

1.3. OBJECTIVE

This report addresses the various PLiM models for LTO to help Member States build the most appropriate model for their particular case, as follows:

— Describing and discussing the three approaches to PLiM for LTO used by Member States;
— Comparing differences and highlighting equivalencies between PSRs and LRAs;
— Highlighting the problems and drawing long term conclusions and recommendations from Member State LTO experience.

A PLiM model for LTO should not be considered only as an asset management tool applicable from the outside, but as a tool driven by the owner/operator's attitude to keep the plant safe and to fulfil the owner's business goals.

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

1.4. SCOPE

Guidance in this publication is based on experience gathered from the worldwide nuclear power industry, as well as input from experts on SSC ageing, LTO, LRA, power uprates, regulatory aspects, PSRs and safety and economic issues.

Section 2 contains the various Member State approaches to LTO, particularly the role that ageing and PLiM played in the achievement of an NPP design life and the definition of the key parameters governing the viability of operation beyond its nominal design life. The approach used to achieve LTO closely follows the type of LTO licensing rules that are adopted by Member State authorities.

Section 3 deals with lessons learned from operating experience (OE), with the resulting safety and operating improvements and with other changes in the context of the applicable LTO process, which may involve any or all of the following categories: licensing changes, design basis updates, plant configuration alignments, operating procedures, accident management protocols and processes, organizational changes and changes to supporting assets. In Section 3.1, each participating Member State presents its approach to LTO and compares it to the international standard based on the PSR framework, but only if such a comparison was ever made in the country. In Section 3.2, the participating Member States describe their experiences and lessons learned from the conduct of their LTO processes. In Section 3.3, the same Member States describe how they handled design and licensing changes in their LTO process. Design changes may result from various sources, such as new regulatory requirements, modernization, obsolescence, expansion of environmental qualification programmes, large scale refurbishments, operational feedback and those dictated by their PSRs. In Section 3.4, the experience of Member States in dealing with beyond design basis issues is detailed.

Section 4 contains descriptions of areas requiring particular attention, followed by recommendations for implementing typical LTO related modifications and for operating the plant beyond the end of its design life, after the LTO changes have been implemented, with particular emphasis on aspects relevant to ageing management during the LTO period.

The Appendix contains special applications and developments related to LTO in specific Member States that may be of interest to others.

1.5. USERS

This guide is intended to support NPP owners/operators planning an extension of their plant operation beyond its original design life. Such a programme requires the cooperation of vendors, manufacturers, prime contractors, consultants and regulators to generate the documentation required to demonstrate the safety, economy and environmental acceptability of the planned LTO.

This report is also intended for newcomer countries interested in procuring, from the beginning, the tools necessary to implement ageing management in their future plant with LTO in mind. The following are foreseen as users of this guide:

— Utilities;
— Regulatory bodies;
— Architect–engineers/prime contractors;
— Consultants;
— Subcontractors.

2. MEMBER STATE APPROACHES TO PLANT LIFE MANAGEMENT FOR LONG TERM OPERATION

This section covers the technical steps required to justify LTO beyond the nominal design life of a nuclear power generating unit. The process may be part of an LRA or reflect specific requirements of the last PSR cycle before the originally assumed design life of the plant. The LTO process implies a detailed screening of SSCs to select the candidate SSCs for an in-depth ageing evaluation. The SSCs selected are subject to a comprehensive safety margin assessment for the extended operating period. The safety related and/or economically important SSCs also become determining factors in a feasibility study to economically justify the LTO project. This section also includes a description of the documentation required by the regulatory body.

Although each Member State and each reactor technology may have its particular needs and LTO justification methods, these various approaches could be grouped into three main categories:

— The PSR method, which is typically used in Member States with unlimited or continuing licences. European States and Japan use the PSR methodology. The process is almost seamless in that a PSR follows the same rules as an LTO safety review.

— An approach based on a limited term licence and a licence renewal concept. The United States of America uses the LRA concept based on a limited licence term, of normally 40 years, that regulates the current licensing basis (CLB) and a certain number of requirements regarding ageing and regulatory action items, a CLB compliance check, a historical check of maintenance and performance monitoring data and a final safety analysis report (FSAR) update.
— A combination of the previous two approaches, as is the case in the Republic of Korea, where PSR is used for regular safety checks every ten years within the framework of a limited licence term of 30 or 40 years. For the last PSR term leading to LTO, the Republic of Korea has developed an intensified safety review process focused on ageing management and critical component life assessments as prerequisites to obtaining an operating licence extension beyond the original term of a nuclear power unit licence.

2.1. CANADA'S APPROACH TO PLiM FOR LTO

2.1.1. Organizational structure

In Canada, PLiM methodologies and tools have been developed and applied to the State's Canadian deuterium–uranium plants. The organization requires that obtaining a long term operating licence in Canada be based on the logic of a plant refurbishment outage, during which all aged SSCs undergo a thorough assessment and upgrades involving component replacements, system overhauls, system boundary reclassifications and updates to more recent code versions.

The model used to establish a PLiM programme in a CANDU plant involves the training of internal specialists, selected from among the operational staff, on the use of PLiM tools and methodologies. These specialists are responsible for updating the PLiM or ageing management databases and the human interface. They coordinate the various stakeholders, including individual system engineers, maintenance groups and cost engineering groups. The gathering of system data remains the responsibility of the system engineer, and the various skills required by the PLiM programme are drawn from operations, maintenance and from station design and engineering support groups. PLiM tools can provide many of the inputs to the economic assessments that are necessary before deciding on LTO licence applications, including reactor core re-tubing when necessary.

2.1.2. Licensing requirements

The existing plants have been licensed on the basis of requirements established during the design process. To demonstrate continued compliance with safety margins, design parameters and regulatory requirements, the day to day operating envelope is maintained within the bounds of the assumptions of the plant safety analysis. This envelope includes special safety system set point limits and system availability, the acceptable range of process parameters, allowable equipment configurations and operating states.

The key element in a Canadian heavy water reactor LTO licence application is fuel channel replacement (FCR), since fuel channels have known degradation mechanisms that limit their service life. The usual approach to this special outage involves performing both the FCR work as well as any other necessary refurbishment work. The FCR outage is an opportunity to consider rehabilitation of other heavy water reactor systems or components to ensure that extended service life is achieved without the need for other extended outages.

It is, however, important to keep the planning for this type of work subordinate to the work actually required for LTO, to ensure that the FCR outage duration is not affected by non-essential maintenance work.

Regulatory requirements for LTO include safety analysis and probabilistic safety assessment (PSA) updates to address ageing degradation and the resolution of generic action items that may also require configuration updates.

The Canadian Nuclear Safety Commission (CNSC) has established a formal process called the integrated safety review (ISR) for LTO licence submissions. The ISR is a comprehensive assessment of plant condition and operational history. It is conducted at the time the licensee considers life extension of its NPP. The main objectives of the ISR are to determine the extent to which the licensing basis will remain valid over the proposed extended operating life. This includes assessing the adequacy of the arrangements that are in place to maintain plant safety for LTO, the improvements to be implemented to resolve any issues that have been identified, and the extent to which the plant conforms to modern standards and practices.

The basic requirements of an ISR are essentially similar to the PSR process recommended by IAEA Safety Standards Series No. SSG-25, Periodic Safety Review for Nuclear Power Plants [1]. The ISR, however, contains a few important additions to account for the specific circumstances of a CANDU plant at the time of an LTO application.

As in the PSR process, the ISR involves an assessment of the current state of the plant and of the plant performance, taking into account all relevant OE in Canada and around the world, new knowledge from R&D activities, and advances in technology. This process codifies the establishment of appropriate modifications to SSCs, procedures and emergency management arrangements, to enhance safety to a level approaching that of modern NPPs.

2.1.3. Plant condition assessment

2.1.3.1. Scoping and screening

The regulator requires proof of an efficient SSC AMP at the time of an LTO application. The operator applies PLiM techniques to assess the conditions of the plant in specific ways, particularly at the time of an extended LTO decision. The method includes a screening step in which a set of critical SSCs (replaceable/irreplaceable), and groups of less critical components, are divided into the following three categories:

— Life assessments, which are performed for critical irreplaceable (mostly passive) structures and components;
— Systematic assessments of maintenance, which are performed for critical active systems using the failure modes and effects analysis and simplified reliability centred maintenance techniques;
— Condition assessments, which are performed for other SSCs and for groups and categories of large numbers of less critical components.

The tools used in the performance of these tasks are:

— An information database to allow for quick resolution of ageing issues and gaps in knowledge;
— Predictive tools to forecast trends;
— Monitoring tools to track progress and ensure that changes happen as predicted;
— Integrated models to assess the impact of ageing phenomena on safety and operating margins;
— Feedback into the knowledge database for future reference, and to help isolate possible common degradation drivers in the plant.

The database tool allows a structured and systematic analysis of critical components following the seven principles of the reliability centred maintenance method.

Once the screening is complete, a function failure analysis and a failure modes and effects analysis are conducted. The analysis is supported by database templates that contain input information regarding the components, their failure modes, and suggested maintenance tasks and recommended frequencies.

The general principles in assessing analysis results are the same before or during LTO. In the failure modes and effects analysis step, the failure modes with higher probability of occurring and/or greater consequences to business or safety are identified.

The PLiM analysts seek to predict, prevent, eliminate or reduce the consequences of each possible failure. For this to happen, proper procedures, spare parts and resources need to be in place to act in the shortest possible time and prevent undesired consequences.

With these concepts in mind, the PLiM analyst then sets action and task prioritization based on the probability and consequence of each failure mode. During this phase, ‘run to failure’ cases are eliminated from the assessments. These involve less critical SSCs for which failure modes cannot be predicted, prevented or eliminated.

Whenever run to failure is an unacceptable maintenance strategy, the PLiM analyst establishes when and where predictive and preventive maintenance should be applied. PLiM provides an SSC health prognosis with predictive recommendations to ensure design life attainment or longer life. Typically, such recommendations may include:

— Additional data collection and record keeping;
— Enhancements to maintenance and inspection programmes (such as improved detection);
— Enhancements to operating conditions (such as fluid chemistry specifications or monitoring);
— Modifications or early replacements;
— Further R&D activities to improve techniques to better assess the impacts of ageing;
— A recommended schedule for the assessment update.

2.1.4. Evaluation of SSCs

2.1.4.1. Critical (mostly passive) irreplaceable components

The first and most important category of station assets subjected to PLiM is the group that includes all critical SSCs. They are normally considered irreplaceable because if they fail, they can limit the plant design life. They are generally designed to be long lived and passive in nature. Typical examples in CANDU plants are: the containment structure, the reactor structures, the calandria and the end shields. These SSCs are the subject of rigorous studies, testing and knowledge sharing. These evaluations are usually referred to as 'life assessment', which in Canada follows the Institute of Nuclear Power Operations (INPO) rules of systematic life assessment, INPO AP-913.

The main goals of a life assessment for long lived, mostly passive, critical components are to:

— Maintain safety and performance targets;
— Determine degradation mechanisms;
— Identify remedial action options;
— Provide input for technical/economic evaluations;
— Achieve design life;
— Keep the option of LTO open and ensure positive outcomes of technical/economic evaluations.

The life assessment study starts with a component and system health monitoring programme to determine the current physical condition of the selected SSCs. This may include thermohydraulic studies, performance indicators, estimates of operating and safety margins, among other things. The evaluation may involve R&D if the ageing mechanisms are not known. For example, stress-strain state analysis and supporting experiments to assess the condition of steam generator joints, or accelerated tests to support a creep life assessment of pressure tubes and calandria tubes. Once sufficient information has been accumulated, the responsible system engineer can evaluate the degradation of these components and subcomponents. This is also called the 'diagnostic phase'. It is followed by a recovery investigation task in which various intervention options are examined to re-establish configuration integrity. An intervention strategy suitable for the component, or its category or class, is then developed, eventually leading to an optimization step. The results are then used to perform an asset management review and a cost–benefit analysis. The cycle ends with a PLiM study report which, if approved, produces an input to the system maintenance data store for planning and execution of the recovery strategy. Life assessment is a time consuming, complex and costly process and is usually reserved for life limiting critical SSCs.

2.1.4.2. Critical but replaceable SSCs

These SSCs are generally 'active' in nature in the sense that they normally include energized (or active) constituents (i.e. primary system motors, resistors, sensors, power operated isolation components and emergency safety systems, among other things). Some are replaceable only during extended outages (large steam generators and pressurizers) and are done through plant refurbishment programmes. They deserve and should undergo an assessment, although less rigorous than a life assessment, but still systematic in nature. One of the assessment methods used for this category of SSCs in CANDU reactors is the systematic assessment of maintenance method. A systematic assessment of maintenance is conducted in parallel with existing maintenance, surveillance and inspection programmes. It is used to optimize maintenance and surveillance inspection and to support life management plans. The systematic assessment of maintenance uses streamlined but advanced techniques such as:

— Simplified failure modes and effects analysis (deterministic);
— Simplified reliability centred maintenance (probabilistic).

Typical critical but replaceable components selected for systematic assessment of maintenance in a CANDU unit are:

— Fuel channels;
— Steam generators;
— Reactor structure;
— Containment structure and major civil structures;
— Nuclear piping and major conventional piping;
— Turbine generator and major auxiliary systems;
— Large pressure vessels, pumps and heat exchangers;
— Large motors, motor terminations and cable systems;
— Containment penetrations;
— Component cooling water pump house/intake/outfall.

The steam generators normally belong to this category and their refurbishment or replacement may also be planned for the extended refurbishment outage. In addition, some subcomponent replacements may be necessary to address flow accelerated corrosion issues.

2.1.4.3. Less critical SSCs

These are the most numerous components in the plant. They are typically subject to a simple condition assessment. The first step is to gather such components in commodity groups (i.e. instruments and valves). Second, the PLiM analyst performs a general review of operational data in order to assess the current condition. The analyst evaluates ageing degradation of strategically selected components and provides a prognosis on the attainment of design life. Recommendations may typically include an ongoing ageing management plan and further assessments.

2.1.4.4. Non-safety items

For other categories and isolated components, recommendations can be issued to enhance their AMPs. Components such as the station control computers can be included in the refurbishment outage. These can become obsolete when spare parts are no longer available, resulting in the potential for their declining reliability to climb exponentially. Similarly, the programmable digital comparators used during reactor shutdown are normally items replaced during an LTO outage.

The windings of the main generator have a life normally insufficient to span the proposed LTO period and their refurbishment may also be included. Upgrades tied to modern human factor engineering requirements for human–machine interfacing components found in control centres are usually also included in the refurbishment of older reactor units, and many enhancements in the control room may be identified and planned for implementation during an LTO outage.

2.1.4.5. Balance of plant components

Balance of plant components are much easier to inspect and replace than those in the nuclear steam plant; however, a number of additional investigations should be undertaken to assess ageing of balance of plant components that are critical for plant performance during LTO.

2.1.4.6. Environmental assessment

In accordance with the Canadian Environmental Assessment Act, LTO projects may be required to undergo an environmental reassessment. In such cases, proof that the LTO will not have significant adverse environmental effects is required prior to any licensing action being initiated.

As the appointed authority for the conduct of the environmental assessment process, the Canadian Nuclear Safety Commission (CNSC) determines whether an environmental assessment is required and ensures that the process is carried out appropriately.

2.1.5. LTO feasibility study

A key part of an LTO programme is to utilize the PLiM assessments and the ageing management strategy to enhance current PLiM programmes for extended operation, such as planning the optimized surveillance, maintenance and operations programmes to achieve the utility's targets for safety, reliability and production capacity during its extended life.

With the PLiM work, the state of the SSCs should have already been determined. The information is subjected to R&D if understanding of the degradation mechanisms is necessary. The operational history of the SSC, and its design and fabrication, should be reviewed focusing on age sensitive characteristics. With these elements in mind, it should be possible to diagnose ageing stressors and ageing mechanisms during all modes of operation, to assess maintenance in terms of effectiveness in ageing management and to prepare a life prognosis. In parallel, recommendations for improvements to optimize LTO can be obtained. There may be a need for further R&D. When all elements are available, a business case for LTO can be assembled that compares the cost of refurbishing the NPP with the cost of alternatives within the overall business objectives and the feasibility parameters.

2.1.5.1. Verification and conclusions

Recommendations are subsequently screened by a risk informed valuation process and all final decisions are retained in the station database for implementation.

Experience has shown that a plant condition assessment conducted in this way is a highly effective and auditable process for scoping the refurbishment work of a CANDU plant.

2.1.6. Documents package structure

The information package structure of a typical application to the CNSC for the release of an operating licence for LTO is described below.

2.1.6.1. The licence application for LTO

The licensing process for LTO begins with the licence holder formally advising the CNSC of the intention, and submitting the LTO project description. The application includes:

- Definition of project scope and objectives;
- Status of current plant design and operation;
- Project elements and structures (such as permanent and temporary structures, special infrastructure and required construction equipment, among other things);
- Expected project activities (such as operational phases, timing and scheduling of each phase, functional grouping of preparatory and refurbishment activities, configuration changes);
- Site information (such as social impact, environmental features and land use);
- Transportation and population protection, training requirements, waste treatment and disposal;
- Anticipated milestones.

2.1.6.2. ISR basis document

Prior to performing the actual ISR, the licence holder prepares an ISR basis document, which includes the plan, the scope and methodology for the conduct of the ISR. The ISR basis document should specify:

- The proposed extended operation period;
- The safety factors to be addressed as listed in SSG-25 [1];
- An assessment of any unresolved shortcomings related to IAEA safety standards and other appropriate modern international standards and practices;
- The criteria for the selection of applicable standards, specifications and practices that will be used in the review of each safety factor for all expected modes of operation (i.e. normal operation, maintenance, refuelling, shutdown and startup activities) to determine whether there is any potential for increased or unacceptable levels of risk during LTO;
- An assessment of the outstanding CNSC generic action items and station specific action items, and a description of the method to be used to arrive at an acceptable resolution;
- An assessment of plant non-compliance with modern licensing requirements and the methodology to be used to resolve the non-compliances or to justify any deviations;
- An assessment of the plant configuration and performance in light of the CNSC regulatory documents that would apply to a new build;
- In the case of multiple units, the physical status of each unit considered separately, as well as the impact of dependencies on common services;
- An assessment of the adequacy of the management arrangements to implement LTO and to operate during the LTO period.

In terms of plant performance, the licence holder should also identify and address any gaps between current and desired plant state and performance, documenting the significance of any gaps and prioritizing corrective actions and improvements. The commission reviews the ISR basis document for acceptance.

2.1.6.3. Conformity review

A conformity review report is required to confirm that the NPP meets, and will continue to meet, the current plant specific licensing and design basis. This review is based on a systematic point by point comparison of the plant condition against modern standards and practices to assess the level of safety compared with that of modern NPPs. The shortcomings are identified and their safety significance assessed. Any modifications judged necessary to improve the level of safety need to be listed and a global assessment of plant safety for LTO is described. The licensing basis is used in the conformity review and also serves as the baseline in the comparison against modern standards and practices.

2.1.6.4. ISR safety factor report

The ISR safety factor report is submitted to the CNSC for review and approval. This document is concerned primarily with requirements stemming from the plant safety analysis review. It contains the review results, including proposed corrective actions and safety improvements for specific topics. The ISR safety factor report also contains a summary of the results of conformity reviews and comparisons against modern standards and practices.

2.1.6.5. Global assessment

The licence holder incorporates the results of the environmental assessment and the ISR safety factor reports in a global assessment report, which includes an integrated implementation plan. The global assessment report presents significant ISR results, including plant strengths, the integrated implementation plan for corrective actions and safety improvements, and an overall risk judgment on the acceptability of continued plant operation. Interactions between safety factors, individual shortcomings, corrective actions and safety improvements, including compensatory measures, should be considered in assessing the overall plant safety and the acceptability of continued operation. The global assessment should also show the extent to which the safety requirements of the defence-in-depth concept are fulfilled.

2.1.6.6. Integrated implementation plan

In view of each of the ISR safety factors, an integrated assessment of the acceptability of plant operation during the licence application period is evaluated. All corrective actions and safety improvements for each of the shortcomings are described.

The integrated implementation plan indicates the schedule for implementing the safety improvements and provides justification for deferral of the work if the safety improvements cannot be completed during the nearest outage. Finally, all station documentation, such as the safety analysis report, operating and maintenance procedures, training materials, environmental qualification and pressure boundary records, among others, are updated to reflect the outcomes of the ISR.

2.2. THE CZECH REPUBLIC'S APPROACH TO PLiM FOR LTO

2.2.1. Organizational structure

2.2.1.1. PLiM organizational structure for LTO

A project team to prepare for the LTO of the Dukovany NPP was launched to operate for an additional ten years starting from 2015. A major task of the PLiM working sub-team in this project team was the development and implementation of a new PLiM programme as described in an IAEA publication [2].

For the day to day production activities, which include periodic assessments of the AMP recommendations (e.g. AMP of the reactor pressure vessel (RPV), low cycle fatigue AMP, flow accelerated corrosion AMP, cable systems AMP, motor operated valve and solenoid operated valve AMP), a PLiM and LTO department was established, consisting of eight specialists serving all six Czech units. The department work scope was to analyse and integrate all important technical and economic data concerning the state of the irreplaceable, long lived and economically important SSCs. The outcome of this work is a comprehensive assessment of the SSCs to assist system engineers with their final recommendation regarding the future of each of the essential SSCs. System engineers were members of the PLiM working sub-team and participated in the implementation of a new integrated PLiM programme.

2.2.1.2. Organizational structure created specifically to prepare the documents needed for LTO

A project team of about 40 people was formed for the preparation of the Dukovany NPP LTO implementation plan, which lasted from 2005 to 2008. The financial and human resources were planned in accordance with the approved yearly schedules. It is important to note that representatives of the external suppliers were incorporated into the project team structure.

The make-up of the project team was reviewed periodically and experts were integrated as required to provide the skills needed in the development of specific parts of the documentation. In January 2009, the project team was modified for the actual LTO implementation stage in accordance with the project implementation plan.

The general project team structure remained unchanged, as shown in Table 1.

2.2.2. Licensing requirements

The State Office for Nuclear Safety (SONS), responsible for the supervision and the administration of nuclear energy, issues operation permits for nuclear power units for periods of ten years. The necessary preconditions are derived from legislative requirements, as shown in Fig. 1.

TABLE 1. PROJECT TEAM STRUCTURE

Project sponsor	The Power Division Director, reporting to the board of directors, provides the required human and financial resources
Steering committee	Provides the highest level of work control, assigning goals and priorities, carrying out work verification and approving tasks led by the NPP Director
Project manager	Manages the project, responsible for communication with the regulatory authority
Working team leader	Leads the development of project tasks
Legislative part working sub-team	Responsible for the preparation of documents required for renewal of the operational permit
Technical part working sub-team	Responsible for tasks in areas important for LTO — PSR, FSAR, configuration management, PSA, design basis, environmental qualification, in-service inspection (ISI), severe accident management, IAEA safety issues, among other things
PLiM working sub-team	Divided into groups (mechanical, electrical, instrumentation and control, cables, civil), responsible for ageing management review, PLiM implementation, SSCs assessment

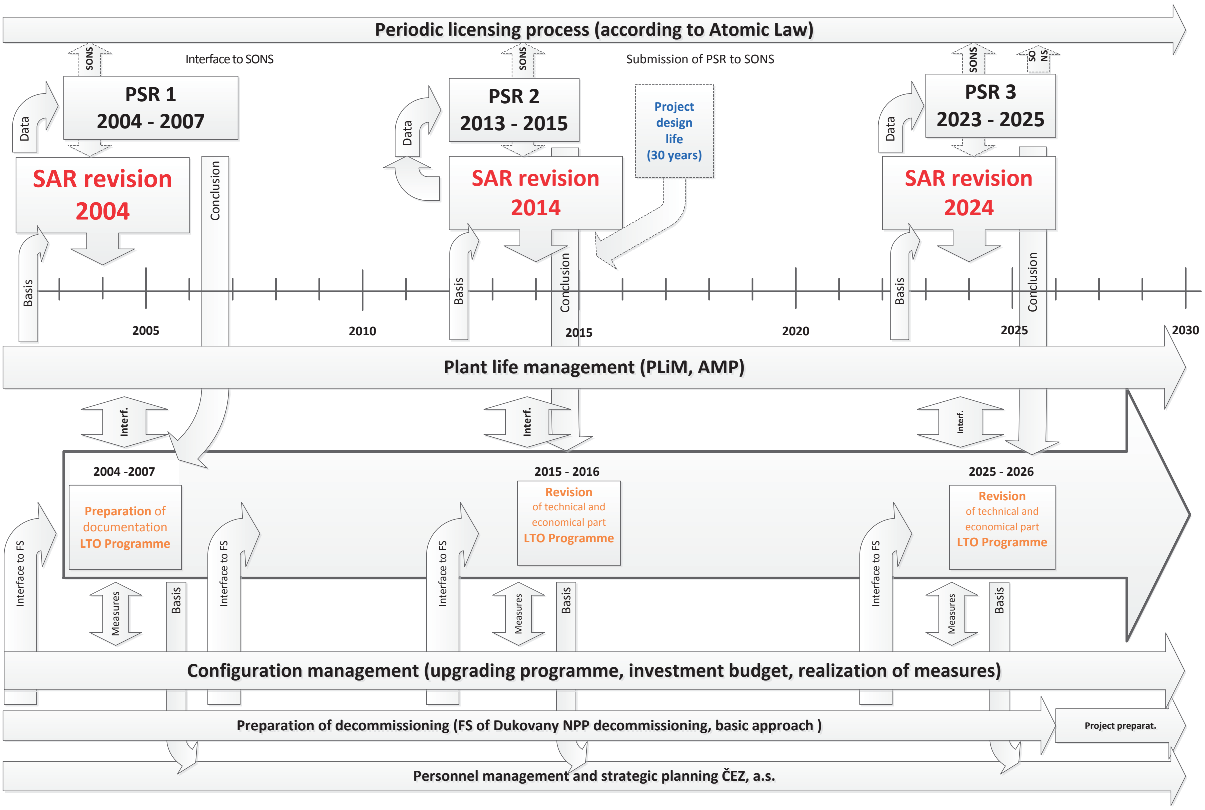


FIG. 1. Licensing procedure in the Czech Republic.

2.2.2.1. Final safety analysis report

The first precondition to obtain the permission of SONS to operate an NPP in the Czech Republic is the update of the FSAR after 30 years of operation. The report needs to be prepared based on the synopsis for FSAR revisions in the quality assurance programme. The scope and content of the FSAR are based on United States

Nuclear Regulatory Commission (NRC) regulation RG 1.70 (Rev. 3) with the appropriate adjustments to account for technology differences of the WWER-440/213 reactors at Dukovany and for differences in the layout and construction solutions. It also includes agreements issued of negotiations with SONS. From this point of view, RG 1.70 (Rev. 3) can be seen as a general directive in terms of content, structure and depth of the topics contained therein. A successful submission of the safety report after 30 years of operation is the first precondition to obtaining a renewal of the licence to operate the plant for extended LTO.

2.2.2.2. SONS rulings

SONS ruling R-SÚJB 24237/2005 on the permit application to continue operating Dukovany Unit 1, valid until 31 December 2015, contains the following requirements, which the operator must meet:

- To complete the refurbishment of the instrumentation and control (I&C) systems within the scope of modules M1 and M2 according to the conditions found in SONS regulation No. 12040/3.2/2001.
- To assess and analyse all non-conformance, if any before the installation of the I&C changes in modules M1 and M2. The operator will also assess reliability on an annual basis in accordance with ČSN IEC 50 (191) and ČSN IEC 60605-4, and inform SONS of the results.
- To include the safety documentation related to I&C modernization and the selected type of fuel, complete with safety analyses based on a verified set of input data by means of codes validated via operational measurements within the unit, and available data from suitable experimental devices. The analysis should demonstrate that the new configuration resolves all findings from the analyses of Section 15 of the FSAR Rev. 2 for Unit 3. This work needs to be completed before the refurbished I&C hardware and software are commissioned and put into operation.
- To update level 1 and level 2 PSAs, the so-called 'living PSA', in connection with SSC changes. Furthermore, the applicant periodically needs to update the level 1 and level 2 PSAs in five year intervals. The applicant also needs to submit these analyses to SONS.
- To carry out probabilistic reliability assessments during operation and inform SONS of the results on a quarterly basis.
- To submit to SONS the assessment of operational and safety indicators. The scope, dates and form are found in SONS letter Reg. No. 26020/2005.
- To develop and annually update the accident management plan, which needs to include the handling of severe accidents and the training of staff, and to forward it to SONS.
- To submit to SONS, a PSR proposal with the content and scope of the review before it is conducted and, once completed, to provide to SONS the report and recommendations.
- To develop and submit to SONS its LTO strategy proposal. The strategy will be based on the relevant IAEA publications and on internationally accepted practice.

Similar requirements were issued for the remaining units of the Dukovany NPP. One additional requirement was issued at a later date.

In connection with the long term strategy, the applicant needs to submit to SONS a comprehensive programme for the LTO of the Dukovany units containing a feasibility study to be annually updated in order to document progress on the LTO relevant programmes and the resolution of safety, ageing and human resources issues.

2.2.2.3. Periodic safety review

Another SONS requirement is the submission of a PSR report to be prepared in accordance with the IAEA's methodology. The PSR report is divided into 14 sections containing the analysis of each of the safety factors. In the case of a PSR for an LTO application, the sections carrying the most weight are Section 2 on the actual condition of the SSCs and Section 4 on ageing.

The objective of the PSR is to determine whether ageing in an NPP is being effectively managed so that all required safety functions are maintained and the AMP is adequate, even for long term plant operation. The following safety factors are to be considered:

— A documented method and criteria for identifying SSCs covered by the AMP;
— A list of SSCs covered by the AMP and a record containing the SSC ageing data;
— Evaluation and documentation of potential ageing degradation mechanisms that may affect the safety functions of the SSCs;
— The extent of the current understanding of all dominant SSC ageing mechanisms;
— The availability of data necessary for a proper assessment of ageing degradation, including baseline data and operating and maintenance history;
— The effectiveness of operational and maintenance programmes in managing the ageing of replaceable components;
— Acceptance criteria and required safety margins for SSCs;
— The effectiveness of the surveillance, inspection and maintenance programme in the management of the ageing of replaceable components;
— Awareness of the physical condition of SSCs, including actual safety margins and conditions that would limit service life;
— Programme policy, organization and resources.

Ageing management of SSCs important to safety requires that degradation be controlled within defined limits. Effective control of ageing degradation is achieved by means of a systematic ageing management process consisting of the following ageing management tasks:

— Operating guidelines aimed at minimizing degradation rates;
— Inspections and monitoring consistent with the applicable requirements aimed at achieving the timely detection and characterization of any degradation;
— Assessment of observed degradation in accordance with appropriate guidelines to assess integrity and functional capability;
— Maintenance (repair or replacement of degraded parts) to prevent or remedy unacceptable degradation.

2.2.3. Scoping and screening method

The process of selecting SSCs for PLiM is shown in Fig. 2. It covers the scope described in Ref. [2] for LTO, plus economically significant non-safety SSCs and replaceable safety related SSCs.

With regard to PLiM, a graded approach was implemented. All power plant equipment was divided into three categories in terms of safety, technological and economic importance in accordance with the strategic vision of further operation. In each category, PLiM is governed by specific procedures, as described in the following.

(a) Category 1: PLiM programme

Scoping criteria:

— SSCs (non-replaceable) critical for LTO;
— Passive mechanical SSCs, active mechanical components fulfilling a passive function;
— Economically important SSCs.

Principles: Managing degradation mechanisms through the management of causes (such as temperature, pressure, flow, regimes and chemistry), monitoring of ageing impacts (such as diagnostics, measurement and testing) and their mitigation.

Category 1 scope:

— Primary circuit pressure boundary;
— Containment pressure boundary;
— Cost sensitive SSCs: safety cabling, cooling towers, turbines, generators and transformers.

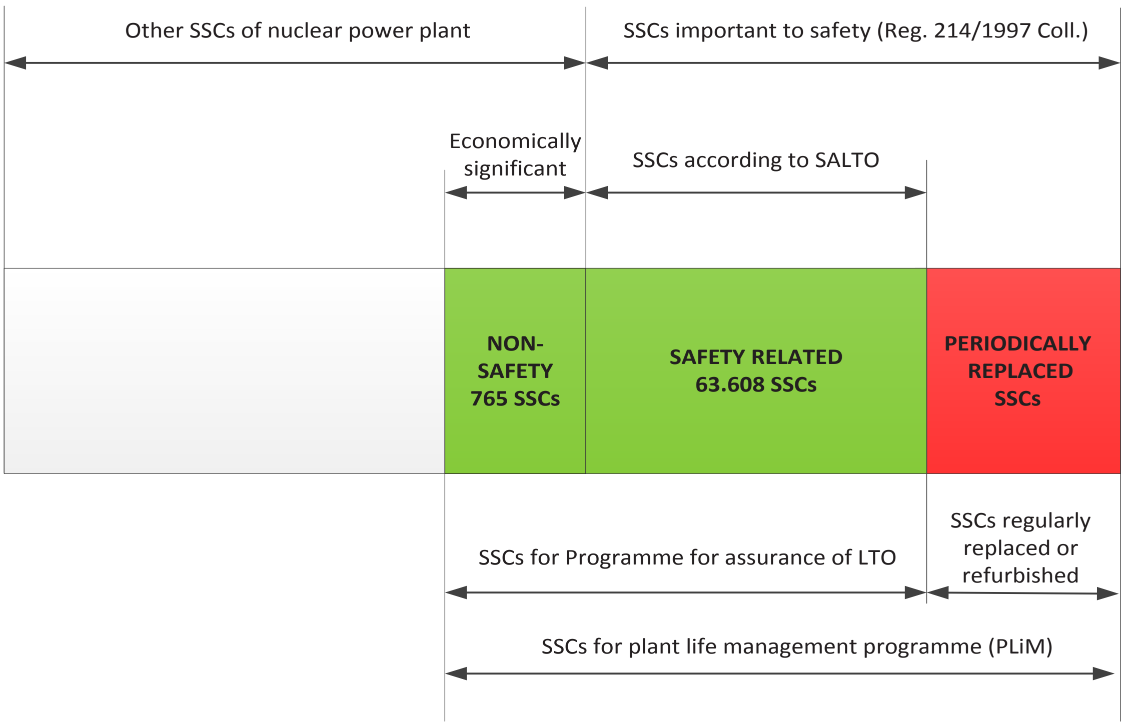


FIG. 2. The scoping and screening process.

(b) Category 2: PLiM based on preventive maintenance

Scoping criteria:

— SSCs important for the fulfilment of operational safety targets;
— SSCs important for nuclear and technical safety.

Principles: Monitoring of ageing impacts. Ageing impacts are monitored through diagnostics, measurements, testing and ageing mitigation by means of preventive maintenance.

Category 2 scope:

— The remaining safety relevant SSCs;
— SSCs with direct influence on availability;
— SSCs with high maintenance costs.

(c) Category 3: PLiM based on corrective maintenance

Scoping criteria:

— The scoping exercise entails the selection of SSCs whose failure does not impact safety, direct unavailability, or important economic impacts.

Principles: Corrective maintenance methods.

The PLiM screening process is used for the preparation of the LTO application, as described in Ref. [2]. For those SSCs determined to be within the scope of the LTO evaluation, it is necessary to identify which of them are subject to an assessment aiming at demonstrating that the effects of ageing degradation will be managed for the planned period of LTO so that all relevant licence requirements are upheld. It will include primarily:

— An assessment of the current physical condition of the SSCs and the determination of significant ageing effects;
— A demonstration that significant effects of ageing are properly managed;
— A determination of the extent of the time-limited ageing analysis (TLAA) review programme;
— A TLAA evaluation.

2.2.4. Evaluation of SSCs

An ageing management review for LTO purposes is conducted as follows:

— Preparation of the strategy and of detailed methodologies for mechanical, electrical, I&C and civil structures and components;
— Selection and classification of SSCs and collection of input data on the SSCs [3];
— Definition of potential degradation mechanisms and ageing impacts on SSCs;
— Assessment of existing information useful for the understanding of SSC ageing;
— Development of ageing management matrixes to ensure that all degradation mechanisms and ageing impacts are covered by acceptable AMPs;
— Definition of gaps in the understanding of SSC ageing, including degradation mechanisms and ageing impacts;
— Formal assessment of conformity of implemented AMPs with IAEA descriptions of acceptable AMPs [2];
— Evaluation of results and determination of corrective measures.

In addition to the ageing management review, done primarily for LTO purposes, there is an ongoing PLiM programme to manage all SSCs during normal operation. The following parameters are typically assessed after each campaign for each SSC or group of SSCs in the PLiM scope:

— Physical ageing of SSCs;
— TLAAs;
— Conceptual ageing;
— Technical ageing;
— Maintainability (i.e. spare parts and suppliers);
— Reliability (failure rates);
— Maintenance costs.

The results of this assessment are:

— Determination of residual lifetime of the selected SSCs;
— Definition of necessary diagnostics, monitoring, surveillance and testing;
— Proposals for replacements, refurbishments and exchanges;
— Proposals for R&D.

2.2.5. Feasibility study

International practice recommends that the decision process to proceed with an LTO should take into account technical and economic analyses to determine under which conditions an LTO is warranted. The practice is summarized in an IAEA publication [4]. The main elements influencing the cost of an LTO, according to the IAEA methodology, are called ‘cost drivers’, and are listed below:

— Safety upgrades to meet regulatory requirements;
— Other non-safety and conventional system upgrades;

— Management programmes and processes;
— Environmental impact assessment;
— Maintaining expertise;
— Public acceptance;
— Radioactive waste and spent fuel management;
— Decommissioning;
— Licensing process;
— Operation and maintenance review;
— Operational spare parts and consumable materials;
— Fuel cycle improvements;
— Overall risk assessment.

The technical and economic feasibility study of the LTO of the Dukovany NPP contains an analysis of the ageing effects on SSCs, a proposal of possible design changes for the mitigation of these impacts and of their cost aimed to determine the total possible NPP operating time extension and the most economical strategy to achieve its implementation.

The economic assessment focuses on the increased cost items accumulated during plant operation over the original design lifetime of the SSCs. This implies that the technical and economic feasibility study may limit itself, without serious impacts on its usefulness, to the equipment with long lifetime (corresponding to the original design lifetime of the NPP), the replacement or extensive refurbishment of such SSCs would be significant.

2.2.5.1. Technical assessment

In December 2005, NRI Řež prepared a methodology for the technical and economic assessment for the safe LTO of Czech NPPs. In accordance with this methodology, the list of SSCs selected for evaluation was consolidated into 248 commodity groups. This number was further optimized. Subsequently, the best estimate of the operating time extension of such equipment was carried out considering ageing, obsolescence, availability of spare parts, maintenance technologies and new safety and economic requirements.

For individual commodity groups, the end of their service life was determined taking into account current or planned operating methods. Making use of PLiM techniques, the main degradation mechanisms affecting individual SSCs were determined to facilitate the estimate.

Facultative technical measures aimed at ensuring the achievement of the target operating time (40, 50 and 60 years of operation scenarios) were developed and proposed, which included replacements, repairs, condition monitoring and adjustment of operating conditions. The proposal included previously planned modernizations and reconstructions, and specific analysis of all possible functional failures and degradation during the LTO, such as:

— Loss of equipment functions that may occur during LTO;
— Equipment failures leading to loss of function;
— Effects of ageing causing failures;
— Operational conditions causing failures;
— Degradation mechanisms causing ageing effects;
— Operational conditions, equipment features, material characteristics, design or construction features and production technologies causing the degradation mechanisms and their effects on SSCs.

From the total list of optional technical measures, all possible variants and combinations were reviewed and a short list of the most suitable solutions was prepared based on the impact and feasibility of each measure. The final selection’s objective was to determine the most suitable measures for each operating scenario (40, 50 and 60 years of operation).

2.2.5.2. Economic assessment

The first part of the assessment report describes the methodology used to develop the LTO cost optimization model, which takes into account financial, technical and reliability aspects in accordance with the requirements of the Dukovany NPP LTO programme, by the production division of ČEZ, a.s. This part of the economic assessment also includes a documentation list and interface requirements with the technical assessment.

The proposed methodology is based on Ref. [4], adjusted to the current control documents of the Generation Division of ČEZ, a.s., and is compared with experience feedback from similar evaluations carried out in foreign NPPs.

The second part of the economic assessment contains calculations and an economic analysis that takes into consideration the outputs of the technical assessment. It also contains the analyses of risks and other necessary parameters as described in the methodology part. The final output is the determination of the optimum duration of the LTO.

The economic assessment also includes a publication review (i.e. a comparison of the selected modifications with the features of a classic power plant), and it defines the boundary conditions and discusses the validity of the improvement selection. The technical and economic feasibility study of the Dukovany NPP LTO programme was completed in September 2007.

2.2.6. Documents package structure

The documentation required for the Dukovany NPP LTO programme consists of the following parts:

- Part 1: A thesis on the Dukovany NPP LTO;
- Part 2: A technical and economic feasibility study of the Dukovany NPP LTO;
- Part 3: Safety aspects of the Dukovany NPP LTO (certificate of acceptability of LTO);
- Part 4: A long term investment plan;
- Part 5: A summary report;
- Part 6: Attachments.

The documentation to support the PLiM requirements is shown in Fig. 3.

2.3. FRANCE'S APPROACH TO PLiM FOR LTO

2.3.1. Organizational structure

In preparation for the ten year outages, an AMP is in force at Électricité de France (EDF) in order to justify that all SSCs concerned with an ageing mechanism remain within the applicable design and safety criteria. The AMP procedure is carried out in three main steps, in coordination with French regulations and with IAEA Safety Standards Series No. NS-G-2.12, Ageing Management for Nuclear Power Plants [5], as follows:

(1) Selection of safety related SSCs affected by an ageing mechanism;
(2) Review of all SSCs 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;
(3) Detailed ageing management reports required for some sensitive components (such as the reactor pressure vessel, reactor internals, civil structures, I&C or electrical cables).

Each NPP should provide plant ageing management reports to the relevant nuclear safety authority at least 12 months before the ten year outage. Relying on a thorough ageing analysis of the safety components of the plant, the report is supposed to justify the ability of the plant to operate for ten more years.

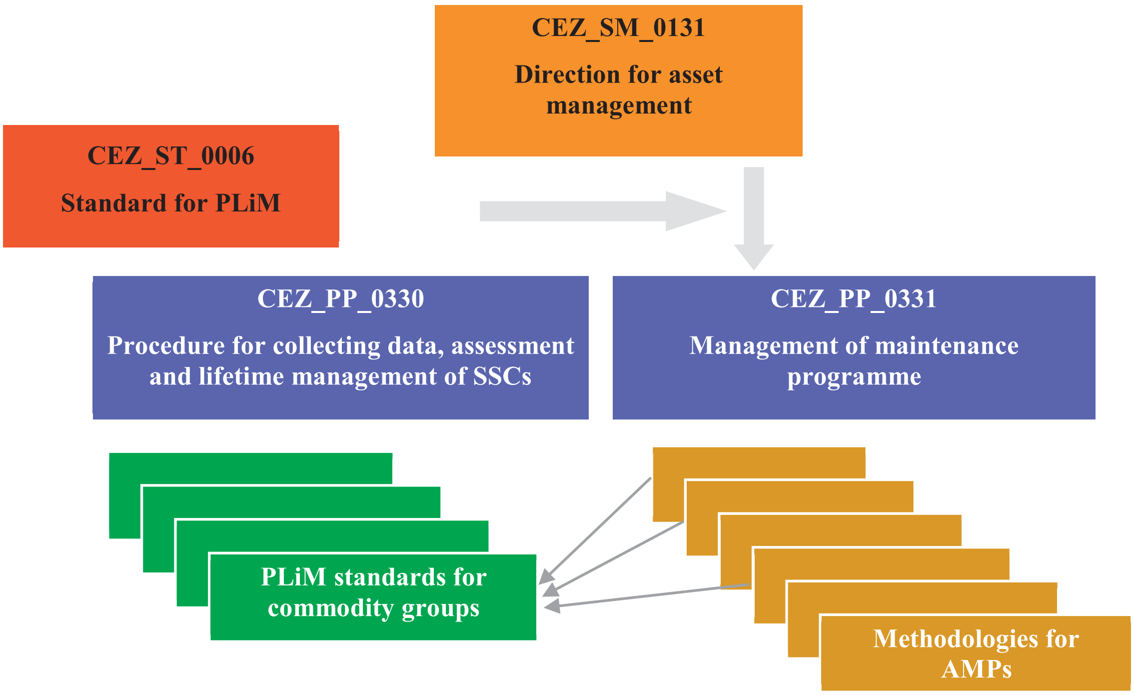


FIG. 3. Documentation to support PLiM requirements for the Dukovany NPP.

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 that the licensee has an effective understanding of ageing mechanisms and an efficient method of integration of operating feedback.

The LTO strategy is illustrated in Fig. 4 and includes:

— A diagnosis of the state of the plant based on ageing analysis and operating feedback;
— A prognosis of 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) that includes an exceptional maintenance programme periodically revised by the EDF's executive committee for LTO.

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

— If the estimated end of life is beyond 60 years, the strategy is updated periodically by the executive committee. A component replacement feasibility file (to cope with obsolescence and the availability of spare components) is prepared at the same time as the repair/replacement process in order to plan strategic modifications, if necessary, to cope adequately with an unexpected demand for replacements. Moreover, an R&D programme on issues such as material research, evolution of methods, non-destructive testing evaluation and configuration modifications is established in support of the strategy.
— If the estimated end of life is between 40 and 60 years, 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 plan it correctly (this occurs between the third or fourth ten year outage), a decision making tool is used and specific methods are implemented to test various schedules and assess the consequences on safety and operating conditions.
— If the estimated end of life is less than 40 years, the replacement/refurbishment of components is decided by the executive committee as part of the routine asset management programme aiming at an extension of operations to up to 60 years.

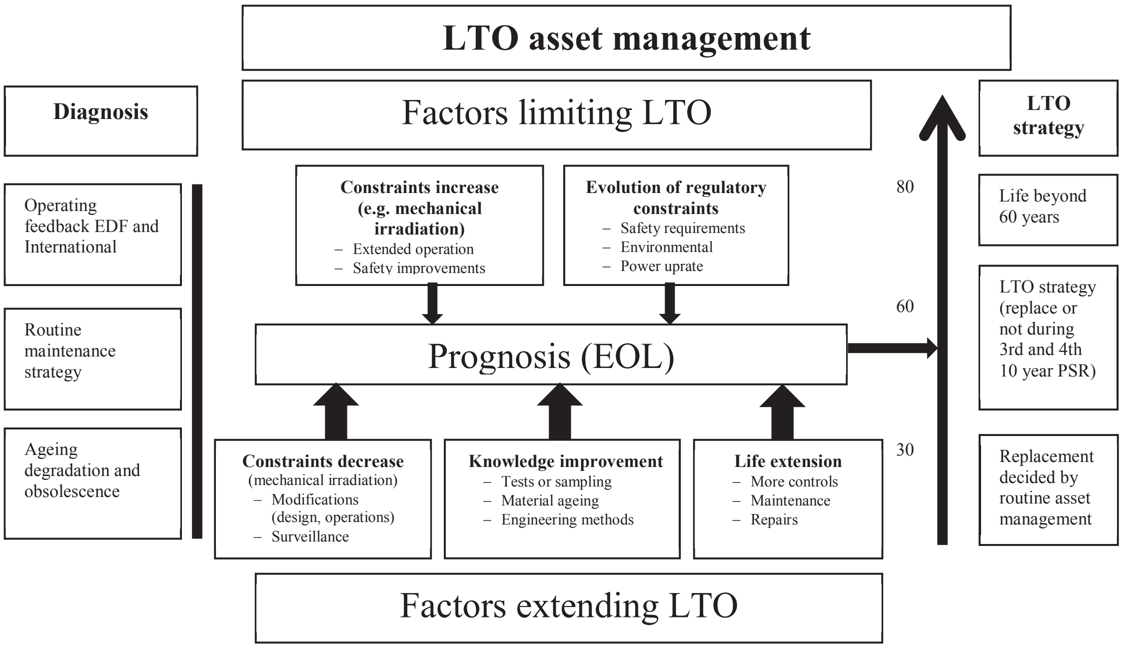


FIG. 4. The LTO decision making process for major NPP components.

In terms of PLiM, a distinction has to be made between replaceable components such as pumps and steam generators and non-replaceable components such as the RPV and the containment building. Appropriate strategies need to be set-up in both cases.

2.3.2. Licensing requirements

The French Nuclear Safety Authority, formerly the Directorate for the Safety of Nuclear Installations, does not issue a licence 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 12–16 months, depending on the series and the fuel cycle retained). An agreement was reached between EDF and the safety authorities to minimize the need for long outages to implement modifications at the ten year safety review milestone, during which time a complete check of the unit is performed as prescribed by the 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 according to the context and the implementation plan of the modification batch proposed (the same for all of the units in the series), the series is allowed to operate for ten more years (except if a specific problem in one unit becomes a common case and a generic action item).

Currently, the oldest 900 MW units of the CP0 series (24 years old) and of the CP1 series (21 years old), 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 it will not consider a life extension request before the third ten year outage, and not for more than ten years at a time.

2.3.3. Scoping and screening method

The scoping process is driven by the PLiM programme and its ageing management techniques and is focused primarily on the selection for ageing analysis of SSCs considered passive and long lived. During the scoping of the SSCs, history and ageing data are collected to facilitate the economic analysis consistent with the long term business objectives of the owner. The most obvious selections include safety related non-replaceable SSCs, safety related replaceable SSCs, as well as conventional systems in need of upgrades or refurbishments. These are aimed at improving efficiency and, in some cases, at increasing plant output and reliability or to optimize operational costs. Such items could include:

- Civil structure upgrades;
- Turbine generator overhaul with or without power upgrades;
- Condenser tube replacement;
- Power transformers;
- Switchgear;
- Nuclear and non-nuclear piping, especially cooling water piping;
- Cables for both power and control applications;
- Communication equipment.

Some changes may be imposed by the regulator, such as environmental impact improvements, fire protection and heating and ventilation air conditioning upgrades. The selected SSCs are then analysed in detail to determine replacements/upgrades required to meet the new expected service life.

2.3.4. Evaluation of SSCs

The plant life monitoring programmes in France are designed to predict ageing, provide mitigating solutions for critical non-replaceable components, increase performance of active components and optimize the capacity factors of the plants. The programme has two overall high level objectives: to ensure that all components reach their design lives in safe and good working order; and to leave the option open for LTO beyond design life. The programme has led to the strengthening of the irradiation resistance monitoring programme, customized to the individual component conditions. Containments, in particular, are being monitored on-line during their entire operating lifetime. Mitigation solutions are being implemented to match any degradation incurred.

Replacement components present a variety of challenges involving a number of corresponding mitigation solutions. Vessel head and steam generator replacements have been implemented wherever necessary, and other special maintenance strategies have been provided by PLiM studies, which drive the maintenance, monitoring and surveillance programmes.

In economic and market analysis, a plant is rarely evaluated in isolation, especially where the grid and the nuclear power generators need to be flexible and operationally interdependent. This requires a complete power system analysis that involves long term modelling of the entire power supply system and the examination of competing scenarios, including alternative energy sources and the alternative strategies of LTO of existing plants versus new builds.

The first step in evaluating the lifecycle of a critical SSC is to look at its declared design life, its design basis and its design assumptions, and compare these to its operating history and to the R&D connected to the component and its materials. The critical components include:

- RPV head;
- Large diameter piping in the primary circuit;
- Steam generators;
- Primary pumps;
- Pressurizer;
- Control mechanisms;
- Vessel internals;
- Containment building;
- Turbine and generator;
- Electrical cables;
- Cooling towers, if applicable.

The RPV head is the most difficult and complex item to replace. The main ageing mechanism is neutron induced embrittlement. This ageing mechanism is well tolerated, except under thermal shock conditions. The issue is brittle failure. Normally, this vulnerability should be taken into account at design time by specifying steels bound with elements that are naturally more resistant to embrittlement, namely copper and phosphorus. These elements will lower the steel’s nil ductility transition reference temperature. Depending on how low we can have this reference temperature at reactor startup conditions, we can also determine how ageing will affect it and predict what will be its value at the end of the component design life (after 40 years). It is highly desirable for the reference temperature to be below 100°C by a substantial margin.

Steam generator replacements are carried out programmatically in France. The operator and utility EDF maintains a number of spare steam generators in store that are compatible with the replacement programme, and maintains a standing order with the vendors for a continuing supply ahead of the immediate need. This frees the utility from the risk of late deliveries as a result of such a long lead manufacturing schedule, allowing it to develop a no-risk replacement schedule for the entire fleet of reactors in France.

The main primary circuit pipes elbow casting material CF8-CF8M has proved susceptible to a decrease in operating toughness. Theoretical studies and a strict on-site monitoring programme have helped resolve this problem. The results of such studies show that the cold leg elbows will survive for at least their 40 year design life.

Containment is obviously an entity that cannot be replaced. Its ageing is, therefore, closely monitored. Two types of containments are found in French reactors:

— The single wall prestressed concrete containments with metal liners, featured in the 900 MW(e) series, in which the strength is given by the concrete and the leaktightness by the metal liner.
— The double wall containment found in the new N4 and 1300 MW(e) units, where the prestressed inner wall acts as the containment barrier, and the reinforced outer wall is designed to resist external hazards. The interspace between walls is kept at a negative pressure and fitted with a leak detection system.

The double wall solution of the more modern units is recognized as an additional safety factor to mitigate radiological releases during accident scenarios.

R&D programmes for containment walls are established to provide a clearer understanding of the most debilitating degradation mechanisms and their dynamics, including erosion–corrosion, fatigue, wear, thermal and irradiation ageing. In addition, cooperation programmes with State operating plants older than the French plants are providing good insights into the ageing of equivalent components. The licence renewal awarded to plants that have reached their original design lives in the United States of America, and the dramatic performance improvements achieved by those plants, confirms the importance of using all PLiM tools and methodologies to control ageing and to maximize the operating life of all French plants in a safe and environmentally friendly manner.

2.3.5. Feasibility study

The technical, economic and political evolution of the entire energy sector in France will dictate the feasibility of keeping nuclear units in operation beyond their originally intended service life or, conversely, to decommission them. The government policy framework is particularly crucial. If the climate is that of support, it gives investors an economic incentive and reduces regulatory risk connected to LTO projects. LTO in parallel with new builds can fill the capacity gap and allow the decommissioning of fossil plants and of nuclear power units that are no longer rentable. The replacement of nuclear units and new builds are planned together. With fossil fuel prices increasing and the low availability factors of renewables, the deployment of new generation reactors in France becomes a reality, both as base load and as load following units.

With the introduction of large reactor units such as the European pressurized reactor, the grid to which they are connected needs to meet certain characteristics of robustness and stability, since a large amount of electricity suddenly connecting or disconnecting requires a robust tie-in point and a stable grid. Key parameters in forecasting future electricity demands are the evolution in time of the projected peak loads and the risk of capacity drops affecting the reliability of supply. With a capacity margin of 20%, the flexibility of power generation is manageable if the total energy mix is taken into account, especially the three major generator types: coal, gas and nuclear.

If the share of intermittent renewable sources is increased beyond 15% of the energy mix, 20% capacity margins may not be enough and the question of global reliability of the supply becomes more critical. The summer peak average temperature and the winter peak average cold spell play a fundamental role in determining the minimum capacity necessary and the projected demand growth rate for the study period under consideration.

Economic analysis will play a major role in deciding the LTO of an existing nuclear power facility. EDF is dedicated to optimizing investment planning for nuclear assets in order to support decisions for LTO. Investments are intended as long term asset management activities, such as preventive maintenance tasks, modernization and enhancement and logistic purchases, such as the strategically planned purchase of spare parts. The three decisive points to investigate in investment planning are:

(a) The selection and planning of an optimal investment strategy;
(b) The measure of profitability of a portfolio of investments;
(c) The risk of a portfolio of investments.

For new builds, the share of capital in the lifetime levelled cost of nuclear electricity is about 65% without counting the higher risk premium of new construction. Such a high incidence on the overall cost of a new build usually plays in favour of the number of plants striving for LTO.

With respect to aged plants under an LTO programme, their existing flood defences may have to be maintained, repaired or improved. For plants near the ocean, of special concern is a possible design water level rise due to climate change. According to the Intergovernmental Panel on Climate Change, sea level increases at the end of the century could be anywhere from 1 m to as much as 1.7 m, depending on the site. This increase in the average sea level may require a more comprehensive and robust coastal protection plan. Considerations of this nature will have to be taken into account in any LTO or new build feasibility study.

2.3.6. Document package structure

In France, a periodic licensing validation process is practiced and is regulated by a PSR process, therefore the documentation required for an operating period beyond the plant's original design life is not very different than the documentation contained in a regular PSR report that is augmented by additional analysis, particularly a special SSC ageing analysis. Any additional calculation required will include:

— Life assessment studies and related reviews;
— Technical and economic assessments of any proposed upgrades;
— Feasibility studies to select the best option for LTO, including technical and economic evaluation;
— Licensing requirements, such as safety analysis FSAR updates, environmental impact reassessments and reviews;
— Special studies deriving from the records of questions and answers;
— Any permanent changes and actions agreed upon with the regulator.

2.4. HUNGARY'S APPROACH TO PLiM FOR LTO

2.4.1. Organizational structure

2.4.1.1. Organization of PLiM for LTO

There is a composite organization at the Paks NPP extending to all PLiM related areas. The areas listed below, which are part of the cycle and configuration management programmes, are covered on a daily basis by the responsible sections or departments in the plant. All tasks within each area are governed by operating instructions, procedures and quality assurance programmes, including:

— Design.
— Fabrication.
— Procurement (including the rating).
— Erection (construction, assembly and installation).
— Commissioning.

— Operation covering:
    - Operational tests;
    - Monitoring and surveillance;
    - In-service inspection;
    - In-service testing;
    - Condition monitoring;
    - Preventive maintenance;
    - Corrective maintenance;
    - Maintenance effectiveness monitoring (Hungarian maintenance rule);
    - Spare parts management;
    - Configuration management;
    - Ageing management;
    - Environmental qualification.

— Replacement, reconstruction.
— Education.
— Asset management and economy planning.
— Control of plant safety.

2.4.1.2. Organizational structure for developing the operating licence extension application

A formal project team (consisting of eight employees) was set-up for the preparation of the LTO programme and the LRA report. 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 Figs 5 and 6. The project staff is responsible for several tasks, which include:

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

An in-house forum of experts helps with achieving the project technical tasks and 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 Fig. 6.

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 technical support organization 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 Safety Aspects of Long Term Operation (SALTO) of NPPs mission conducted by the IAEA.

2.4.2. Licensing requirements

In Hungary, a comprehensive regulatory system has been developed specifically to oversee the safety of the LTO of the WWER-440/213 reactors at Paks. Compliance with the current licensing basis is controlled 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 should demonstrate compliance with the current

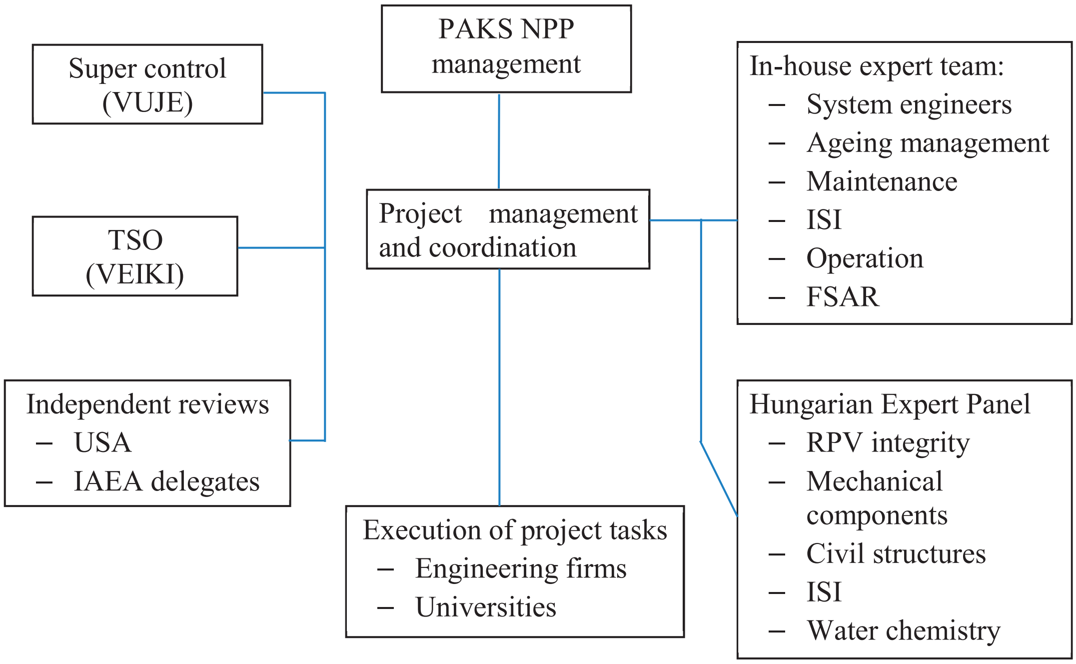


FIG. 5. Project organization for developing the LTO programme.

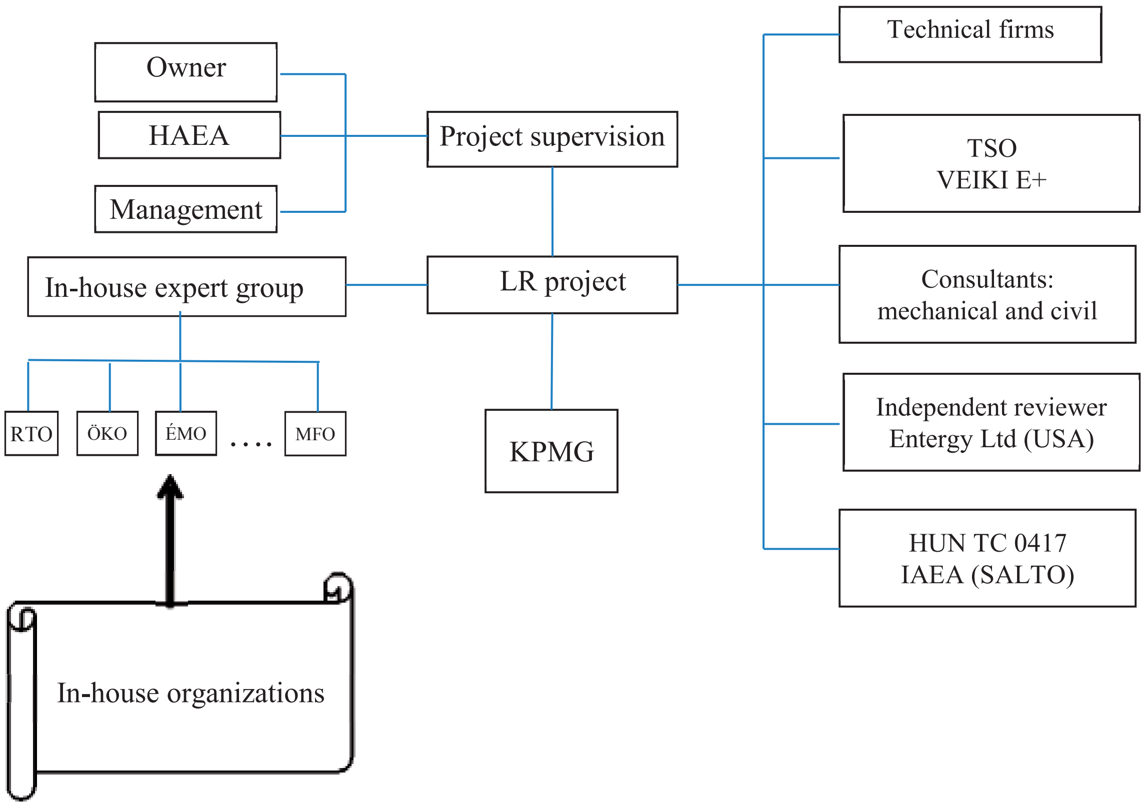


FIG. 6. Project organization for developing the application report for licence renewal.

licensing basis. To that end, it also includes the definition of design basis. The FSAR content is similar to that required by NRC RG 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 ten years primarily to assess the overall ageing of the plant SSCs on a time scale broader than 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 publications [1, 2].

Licence renewal is the formal process used in Hungary to apply for an operating licence extension beyond the original design term. This process is similar to the one governed by the 10 CFR 54 licence renewal regulation in the United States of America, with some notable deviations.

The licensee has to prepare and submit the LTO programme to the regulator no later than four 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 licence 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 LRA has to be submitted one year before the licence expiration date. The application needs to 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.

2.4.3. Scoping and screening method

The licence renewal scope extends to all safety classified 1–3 SSCs. Of the non-safety-related SSCs, those whose failure may jeopardize the safety functions of the safety related SSCs also need to be included in the licence renewal scope. In turn, of this total scope, only the passive and long lived SSCs remain in the integrated plant assessment scope during the licence renewal process (e.g. review of the applicable AMPs). The remaining SSCs can be screened out of the ageing management review.

The scope of the FSAR, PSR and maintenance rule is shown in Fig. 7. According to Hungarian Guideline 4.14, Activities to be Implemented by the Operator to Support the License Application for Operation Beyond Design Lifetime, management of the passive SSC safety functions is based on the AMPs, and that of the active SSC safety function is based on the maintenance rule (see Fig. 8). Additionally, the environmental qualification department manages I&C components subject to a harsh environment.

2.4.4. Evaluation of SSCs

The LTO programme focuses on the ageing of long lived passive SSCs, while the performance of active systems and components is controlled via the maintenance rule, as in the United States of America. TLAAs belong to the LTO programme core tasks, which include the environmental qualification as well.

2.4.4.1. Reconstitution or extension of the TLAAs

The licensee has to identify and justify the list of potential TLAAs. A minimum list of TLAA tasks are provided in Hungarian Guideline 4.14. The list was based on the document, Industrial Guidelines for Implementing the Requirements of 10 CFR 54 — The License Renewal Rule (NEI 95-10 (Rev. 6)), and on the feasibility study findings. The final list is based on the review of all domestic and international TLAA related documentation, such as the original design documentation, the FSAR, US and other international practice.

The TLAA analysis can be used to achieve the following goals:

— Demonstrate that the original calculation can be extrapolated to the extended operational term.
— Replace the conservatism used in the original TLAA with less conservative approaches. In this case, particular attention should be paid to the required safety margins, which should not be decreased.
— Reconstitute the TLAA by applying the most relevant Hungarian guidelines and the US Regulatory Guides 1.38 and 3.25 [6, 7].
— Demonstrate how degradation limiting measures, if adopted during the extended operational life, will slow down the ageing processes.

A ten year reserve was assumed in the extension or reconstitution of the TLAAs.

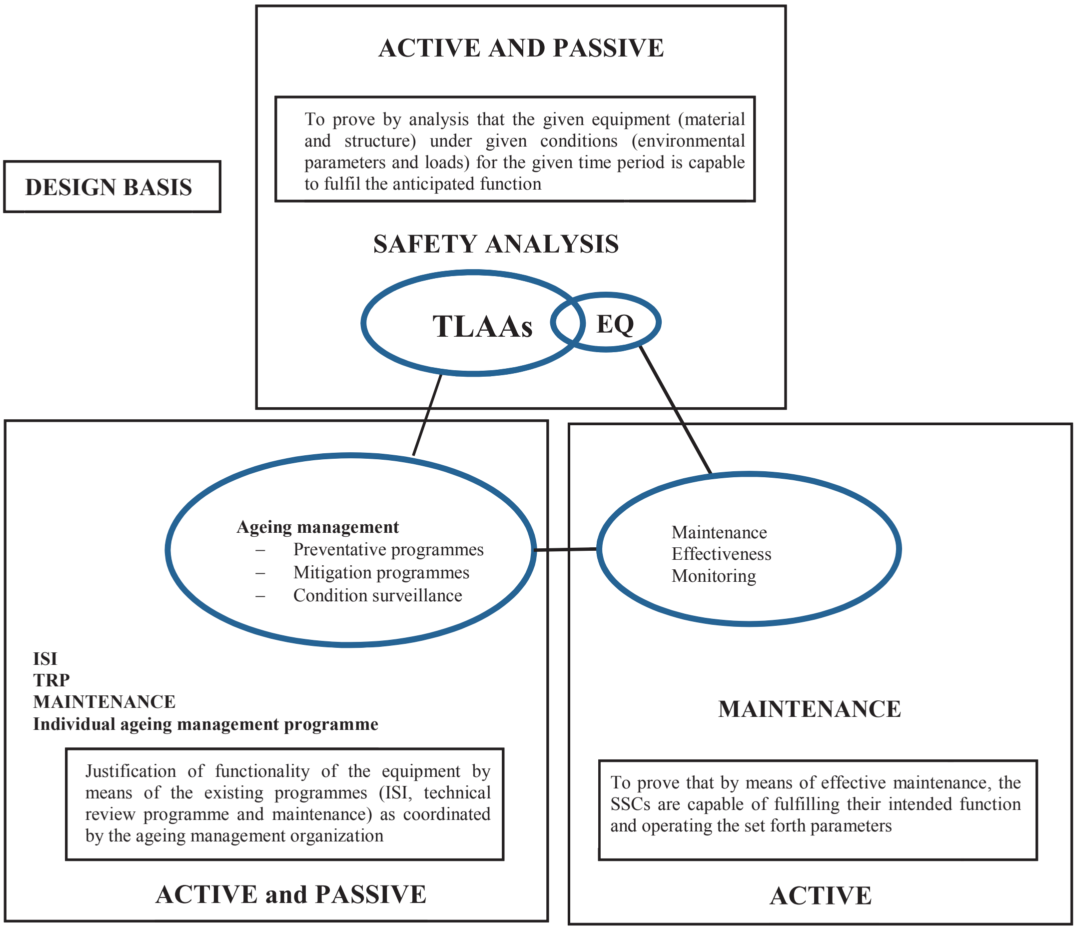


FIG. 7. Scoping of licence renewal, FSAR, PSR and maintenance rule.

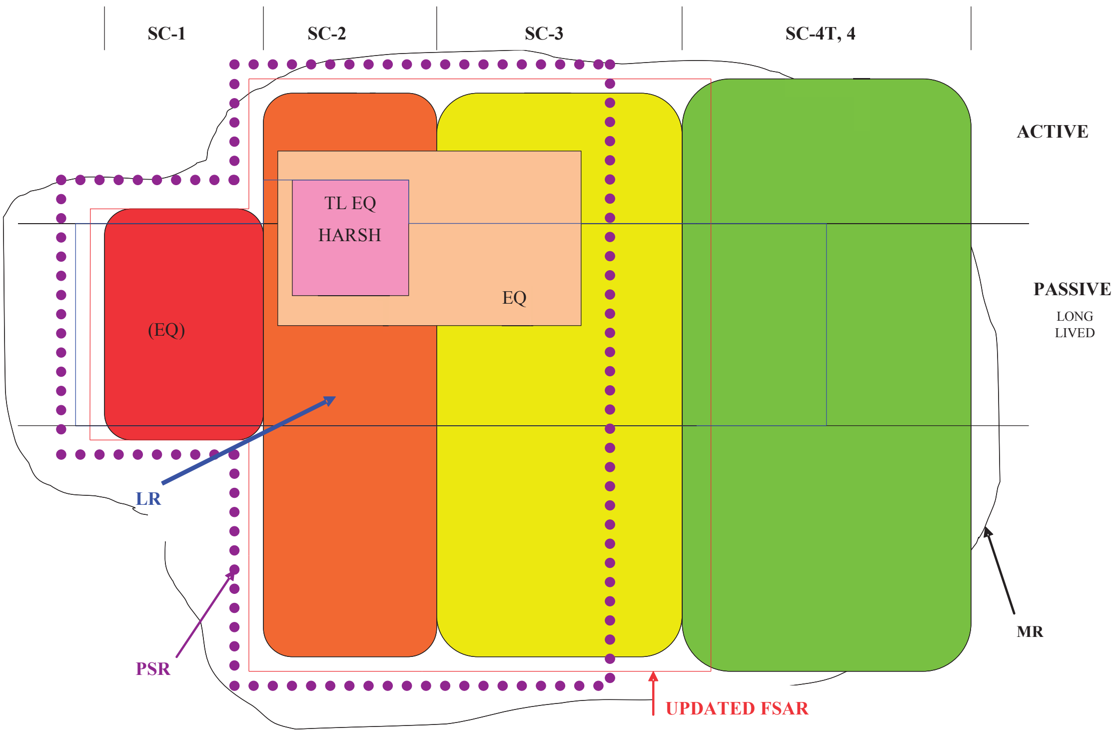


FIG. 8. Managing the safety functions of the SSCs.

2.4.4.2. Review of the AMPs

The main goal of reviewing the AMPs is to demonstrate that degradation processes are detected by the current AMPs in a timely manner, allowing the implementation of the necessary mitigating measures. This review is based on the attributes defined by Hungarian Guideline 4.12, Ageing Management during Operation of Nuclear Power Plants. These attributes comply with the Generic Aging Lessons Learned (GALL) Report (NUREG-1801) [8] and NS-G-2.12 [5]. The review includes:

— Determination of the degradation mechanism and component affected;
— Mitigation and preventive measures;
— Parameters to be monitored;
— Detection of ageing effects;
— Monitoring, trending and condition assessment;
— Acceptance criteria;
— Corrective actions;
— Feedback, efficiency and improvement of the AMP;
— Administrative control, quality assurance, coordination and documentation;
— Feedback from operation and component condition.

2.4.4.3. Managing the active safety function of the SSCs applying the maintenance rule

The SSC active safety function assessments follow the Hungarian maintenance rule, which requires the management of:

— Safety class 1–2 components at the component level (item by item);
— Safety class 3–4 components at the system or unit level.

The maintenance rule also requires the establishment of the maintenance targets, which can be set by either deterministic or probabilistic methods, and are assessed and monitored within the maintenance rule framework.

2.4.5. Feasibility study

A feasibility study is conducted to facilitate the owner's decision to embark on an LTO programme. In this study, the following aspects are evaluated:

— Assessment of the overall plant safety (with inputs from the FSAR and PSR).
— Assessment of the plant technical condition and operating practice to ensure the plant configuration and state are within expectations (based on feedback from the AMP, maintenance and ISI programmes).
— Evaluation of the changes and measures needed for LTO (safety upgrades, replacements and reconstructions).
— In the business evaluation, a comparison is made to predict the cost, prices, profit and cash flows of other options, for example, the assessment of different extension terms (in Hungary's case, this is 30, 40 and 50 years), construction of a new NPP, or a new conventional power generating station (see the business model in Fig. 9).
— Other conditions (regulation and public acceptance).

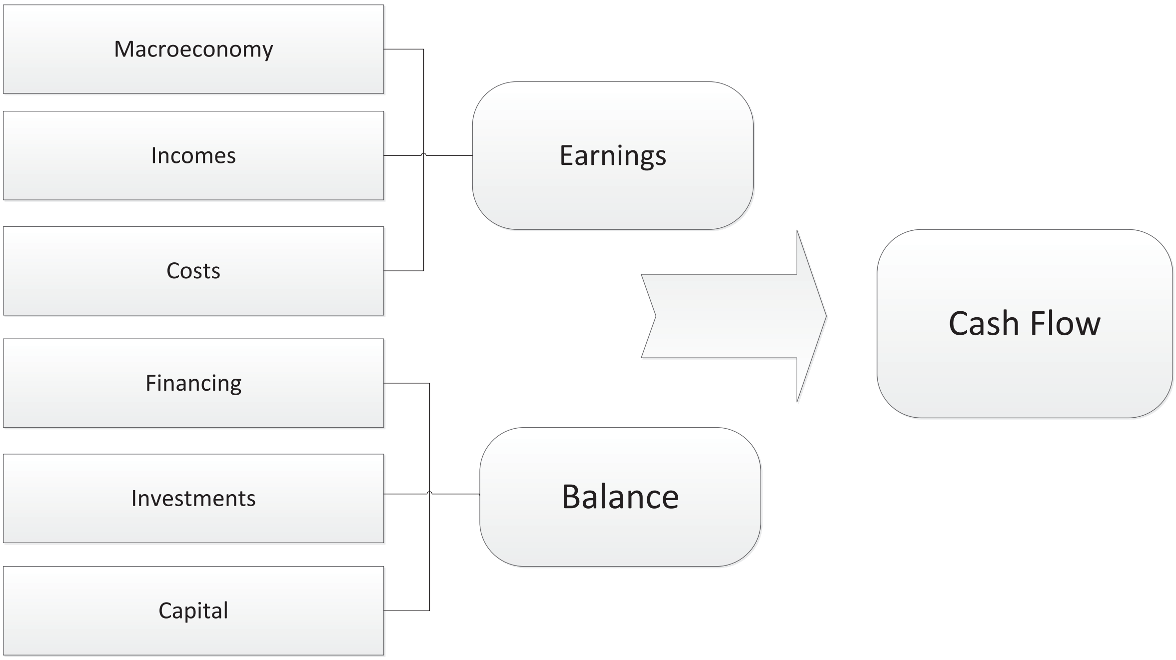


FIG. 9. Business model for a new NPP.

2.4.6. Documents package structure

The formal content of the LTO programme and the LRA report includes:

- General information about the plant and the licensee;
- Determination of the scope for lifetime extension (licence renewal; condition assessment of safety class 1 and 2);
- Integrated plant assessment (SSC screening and list of passive long lived components; review of AMPs);
- Revalidation of TLAA (including environmental qualification);
- Modification of FSAR (supplement);
- Modification to the technical operational procedure (supplement);
- Revision of other operational documents (supplement).

2.5. INDIA'S APPROACH TO PLiM FOR LTO

This section covers a general overview of PLiM at Indian pressurized heavy water reactors (PHWRs).

2.5.1. Organizational structure

A multitier system was adopted for PSRs managed through various safety committees at the plant level in operating organizations and at the regulatory body level. A number of NPPs, particularly the older ones (Tarapur Atomic Power Station, Rajasthan Atomic Power Station (RAPS), Madras Atomic Power Station (MAPS), Narora Atomic Power Station (NAPS) and the Kakrapar Atomic Power Station (KAPS)), have undergone PSR, based on which a number of retrofits and upgrades have been implemented. Key elements considered in a PSR are:

- Actual physical condition of the NPPs;
- Safety analysis (both deterministic and probabilistic);
- Equipment qualification;
- Management of ageing;

— Safety performance;
— Use of experience from other NPPs and research findings;
— Operating procedures;
— Organization and administration;
— Human factors;
— Emergency planning;
— Environmental impact.

2.5.2. Licensing requirements

The Nuclear Power Corporation of India Ltd (NPCIL) conducts PLiM exercises at its plants as per requirements specified in NPCIL instruction HQI-7005 and in the Atomic Energy Regulatory Board (AERB) — the Indian regulatory body — Safety Guide SG-O-14. These guidelines cover the requirements that are based essentially on information made available from national and international OE and feedback systems, and from the IAEA [9–12]. Within the operating licence, the AERB grants initial authorization for a specified period, which may range from five to nine years, and renewal of authorization for further specified periods following a safety assessment. In particular, NPCIL HQI-7005 covers the various key elements of a typical life management programme:

— Screening and prioritization of SSCs based on safety importance;
— Ageing assessment methodologies to assess the effect of degradation on SSCs based on degradation mechanisms;
— Assessment of service life;
— Appropriate inspection and mitigation techniques (repair, refurbishment and replacement).

In renewal of authorizations, a comprehensive safety review of NPPs is required, focusing on the cumulative effects of plant ageing and irradiation damage on the results of in-service inspections, system modifications, operational feedback and performance of safety systems, among other things. Renewal of the Authorisation for Operation of Nuclear Power Plants (AERB Safety Guide AERB/SG/O-12), covers these requirements. The regulatory approach in India for LTO requires the following steps:

— Re-authorization with a limited period of five years;
— Renewal of authorization towards the end of current authorization;
— Application for renewal of authorization (ARA);
— The renewal authorization based on the safety assessments of ARA and PSR, alternatively at five year intervals;
— PSR.

The process of safety review for renewal of authorization is carried out periodically. For this purpose, a standard categorization of SSCs into four categories was made. In a PSR, the health of all systems is reported based on the ageing management applicable to each SSC:

— Category 1: Major critical SSCs limiting plant life;
— Category 2: Critical SSCs;
— Category 3: Important SSCs;
— Category 4: Other SSCs.

The regulatory system in India does not specify any time limit on the service life of NPPs. The plants can continue operation as long as they satisfy the regulatory requirements with respect to reliability and safety margins and they satisfy the safety case.

Currently, the combined approach with ARA and PSR is in force for LTO. Every NPP is required to undergo a PSR once every ten years in accordance with AERB Safety Guide AERB/SG/O-12. The scope and depth of a PSR carried out in India is at par with the recommendations in the IAEA publication on PSR, SSG-25 [1].

During a PSR, a comparison of the NPP features with regard to the current standards is also conducted. A PSR normally establishes also the implementation schedule of recommended improvements. The safety significance of the PSR findings, the updated PSA of the plant, and technoeconomic considerations governed the final implementation plan. It is to be noted, however, that older plants up to KAPS 1 had used zircaloy-2 material for pressure tubes in their original design. Given the reduced design life of such material, a systematic pressure tube replacement programme has been implemented together with major retrofits. RAPS 2, MAPS 1 and MAPS 2 have already undergone this exercise.

In the MAPS units, apart from the normal safety upgrades, steam generator replacements, feeder length replacement at the outlet ends and activities suggested by the PLiM teams have also been completed. In addition, special safety upgrades have also been completed in RAPS 1. At present, NAPS 1 has initiated pressure tube replacement. Feeder length replacement is being considered for NAPS units. It is foreseen that from the NAPS reactors onwards there will be no need for as many retrofits as have been implemented in older units since materials have been replaced to meet modern practices and critical components present no negative trends and no signs of degradation as did the older units.

2.5.3. Scoping and screening method

The guiding document on ageing management of Indian PHWRs (NPCIL HQI-7005) identifies the following areas for the scoping and screening method:

— Ageing management for coolant channels and feeders;
— Ageing management of mechanical equipment as D_2O heat exchangers and steam generators;
— Ageing management of reactor components and reactivity mechanisms;
— Ageing management of civil structures;
— Ageing management of secondary cycle equipment and piping;
— Ageing management of I&C, including issue of obsolescence;
— Ageing management of electrical items including power transformers.

SSCs in Indian reactors are categorized as follows:

(a) Category 1. Major critical SSCs limiting plant life: The major critical components are those for which integrity and functional capabilities have to be ensured during plant operation and shutdown conditions. These have the highest safety significance. These components are non-replaceable and control plant life, and include the calandria end shield components, calandria tubes, in-core components for reactivity mechanisms and moderator system piping (inside the calandria vault).
(b) Category 2. Critical SSCs: These components are required for plant operation and plant shutdown. They have high safety significance and are recognized as major critical components. Usually, they are difficult to replace due to radiation exposure, long shutdown periods and high cost, and include the primary heat transport system piping and equipment, pressure tubes, steam generators, primary coolant pumps, primary heat transport feeders, emergency core cooling system (ECCS) piping and equipment, shutdown cooling, moderator cooling heat exchanges and pumps.
(c) Category 3. Essential SSCs: For these components, preventive maintenance, ISI and condition monitoring and assessment are possible to mitigate ageing. They can be replaced and are replaced in a planned manner during the NPP’s operating life, and include the end shield cooling system equipment, the calandria vault cooling system components, the primary heat transport feed pumps, the turbine generator system, process water piping and equipment, feedwater piping and equipment, heat exchangers, diesel generators, uninterruptible power supplies and batteries.
(d) Category 4. Other SSCs: These are safety related support systems managed by planned preventive maintenance, ISI and conditioning monitoring and assessment. They are routinely replaceable. This category includes components, such as air compressors and instrument air systems, heavy water recovery dryers, main exhaust fans, transformers and power and control cables.

There are 13 PHWR units in operation in India and another five under construction. Design details, including material specifications and quality control techniques, have seen improvements from plant to plant. Also, control of the local environment, and operating practices and chemistry control have been upgraded in time. NPCIL has developed plant specific life management programmes to effectively monitor the condition of SSCs and take corrective action in order to maintain safety margins. To effectively manage the ageing of SSCs, the plants have a programme that provides timely detection and mitigation of ageing degradation in order to ensure that the required integrity and functional capability of SSCs are maintained throughout the service life, and beyond, for LTO.

2.5.4. Evaluation of SSCs

The coolant channels in PHWRs are called pressure tubes. They carry nuclear fuel and form the primary coolant pressure boundary. Zircaloy-2 pressure tubes with loose fit garter springs were used originally in RAPS, MAPS, NAPS and KAPS 1. From KAPS 2 onwards, Zr-2.5%Nb alloy pressure tubes with tight fit garter springs have been introduced in more recent units. Zircaloy-2 pressure tubes are susceptible to degradation and require replacement after about 10 to 12 full power years. An elaborate life management programme has been developed for coolant channels over the years. Design modifications and improvements in manufacturing procedures have enhanced the service life of pressure tubes.

A systematic coolant channel replacement programme has been completed for RAPS 2, MAPS 1 and 2, NAPS 1 and 2 and KAPS 1. The pressure tubes in these units have been replaced with Zr-2.5%Nb alloy pressure tubes with tight fit garter springs. New pressure tubes are expected to have a significantly longer life cycle. The coolant channel condition is closely monitored by an expert group from AERB.

Feeder pipes (made of carbon steel conforming to American Society of Mechanical Engineers (ASME) specifications) carry the reactor coolant from the main primary coolant headers to the pressure tubes. They are susceptible to thinning due to flow accelerated erosion–corrosion. Large scale inspections were conducted in RAPS 2 during an en masse coolant channel replacement outage, and in the MAPS units, based on international experience. Inspection findings confirmed the thinning of outlet feeders near elbows. The ISI programme was enhanced, and detailed assessments of safety margins and residual life were carried out. A systematic feeder replacement has been completed for RAPS 2, MAPS 1, NAPS 1 and 2 and KAPS 1. New feeders are of ASTM A-333 grade-6 material with 0.2% chromium, which is more resistant to flow accelerated corrosion. A flow accelerated corrosion management programme for secondary cycle piping is in place in all the stations. Detailed ISI programmes exist for steam generators and D_2O heat exchangers. An equipment qualification programme is in place

Various safety upgrades have been incorporated in older generation stations like RAPS and MAPS. Station specific problems and other issues of obsolescence have been addressed at NAPS and KAPS 1. Other major upgrades covered the following:

- Replacement of control and power motor generator sets with static inverters of the supplementary control room;
- Main control room computer systems and controllers;
- Modernization of the fire alarm systems.

NPCIL has shown its commitment to ensure that safety systems are in place in Indian NPPs to handle a Fukushima Daiichi type event. Task forces were formed in NPCIL to revisit the safety status of all the Indian NPPs with specific reference to extreme natural events. Reviews have confirmed the robustness of the Indian PHWR design.

To further improve defence in depth, some recommendations were made for which detailed engineering has been carried out and regulatory approvals obtained. Recommendations include, among other things, the provision of additional tie-in points to important safety systems, additional instrumentation for the measurement of important parameters, provisions to ensure sufficient water inventory on site and additional measures to mitigate beyond design basis accident situations. A regulatory body/committee has also carried out a review of the safety status of Indian NPPs.

2.5.5. Feasibility study

Before embarking on an LTO application, a feasibility study is undertaken to justify the exercise. In most cases, the project is economically viable unless the plant was subjected to a damaging event. The NPP owner or operator will decide based on the findings of this feasibility study and on their corporate goals. The feasibility study covers technical and economic aspects and the licensing implications, such as an evaluation of the cost of satisfying all prerequisites including data collection, SSC life assessments, condition assessments, any upgrades and collateral costs, a budget estimate and all economic and financial implications.

2.5.6. Document package structure

The ARA is submitted three months prior to completion of the five year licence limit and covers the aspects important to plant operation.

In addition, a PSR submission covers the review of safety factors important to plant safety. The PSR is a more comprehensive review and is intended to further ensure a high level of safety throughout the service life of the plant.

2.6. CHINA'S APPROACH TO PLiM FOR LTO

2.6.1. Organizational structure

The first NPP in China, Qinshan-I, was put into operation in 1991. Since then, 15 units have been built and put into operation of various reactor types, namely pressurized water reactors (PWRs), water cooled moderated power reactors (WWERs) and CANDU reactors, for a total capacity of 12 550 MW(e). PLiM LTO in Chinese NPPs was developed in the late 1990s. The 15 NPPs in operation to date are owned and operated by two groups: the China National Nuclear Corporation (CNNC) and the China General Nuclear Power Group (CGN). Although slight differences may exist in various NPPs, the organizational structures to carry out PLiM activities are generally similar. Normally, a general manager or deputy general manager may take on the overall PLiM responsibility and a specialized group within a specific department, such as the technical support department or the equipment management department, takes charge of the AMPs.

In China, it is the technical support organizations that provide a wide range of PLiM support to NPPs. The main technical support organizations include SNERDI (Shanghai Nuclear Engineering Research and Design Institute), RINPO (Research Institute of Nuclear Power Operation), SNPI (Suzhou Nuclear Power Institute), NPIC (Nuclear Power Institute of China), CNPEC (China Nuclear Power Engineering Company) and SNPSC (State Nuclear Power Plant Service Company).

2.6.2. Licensing requirements

As shown in Fig. 10, for every NPP in China, the regulator, the National Nuclear Safety Administration (NNSA), requires that a safety review be completed as one of the preconditions for proceeding to the next step, as is done when passing from the design stage to the construction stage or from the construction stage to operations. During operation, PSRs are required every ten years. The conclusions and recommendations of previous PSRs or FASRs are taken as input to AMP activities and the outcomes of AMPs are reviewed in preparation for the next PSR. Figure 11 shows a typical model of a systematic AMP in relation to two successive PSRs. The NPP operator is required to update the FSAR to reflect safety related modifications to essential SSCs during operation. To date, no Chinese NPP unit has reached the end of its design life. The NNSA has not yet decided on the licensing requirements for NPP units to continue operation beyond their design life. However, the design life of the first Chinese NPP, Qinshan-I, is 30 years; only 20 years have passed and ten more years of service remain. The LTO strategy will be decided in the near future.

2.6.3. Scoping and screening method

The scoping and screening of SSCs for a PLiM study vary from plant to plant, however, the principles described in IAEA publication Methodoloy for the Management of Ageing of Nuclear Power Plant Components Important to Safety [9] are used and referred to in combination with economic considerations, which may vary from plant to plant. Figure 12 shows the typical screening steps for components under an AMP.

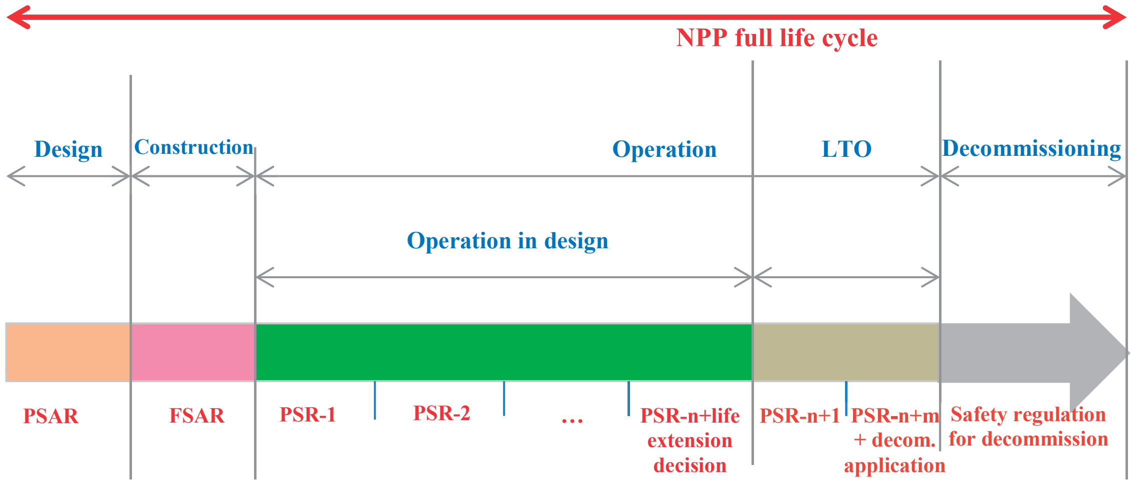


FIG. 10. Licensing steps for Chinese NPPs.

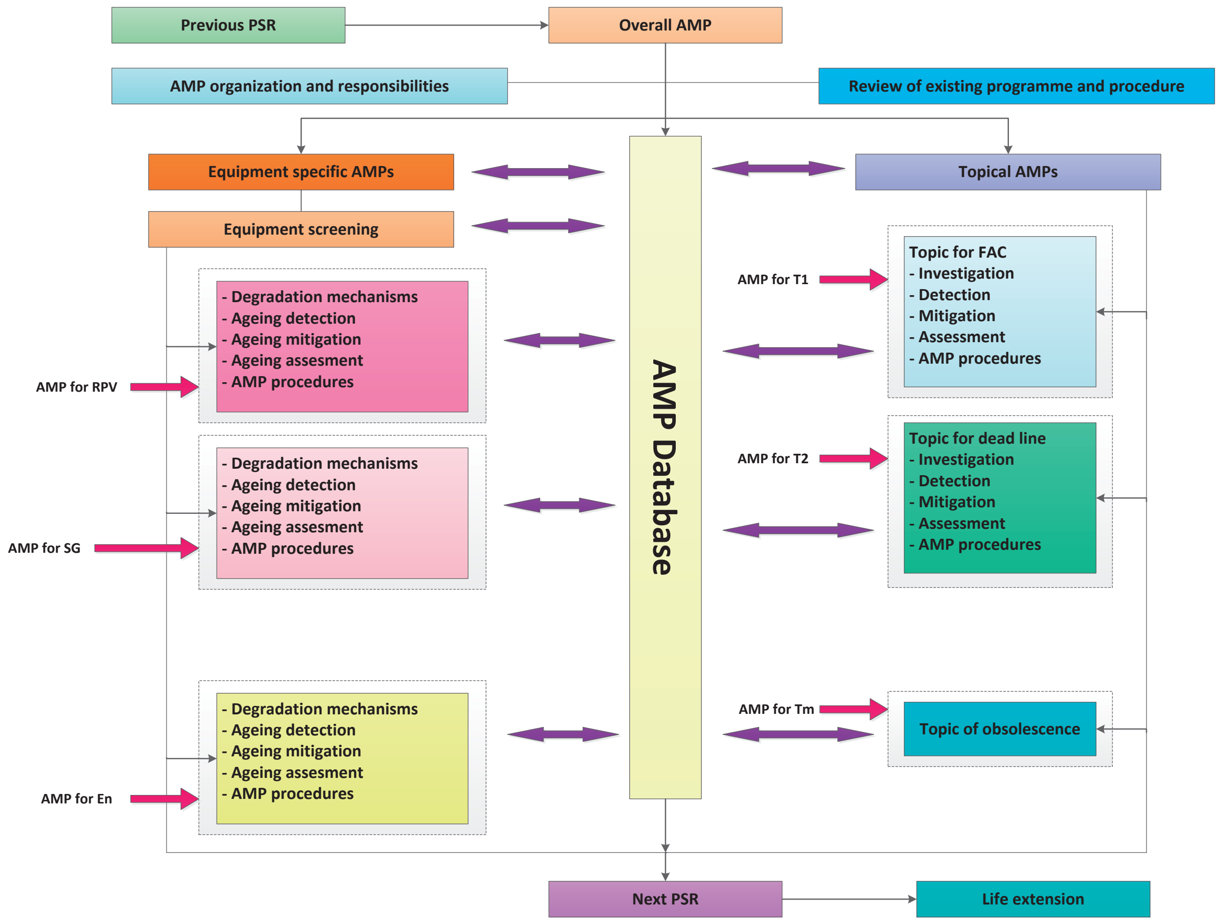


FIG. 11. Systematic AMP model and its relation to the PSR in Chinese NPP.

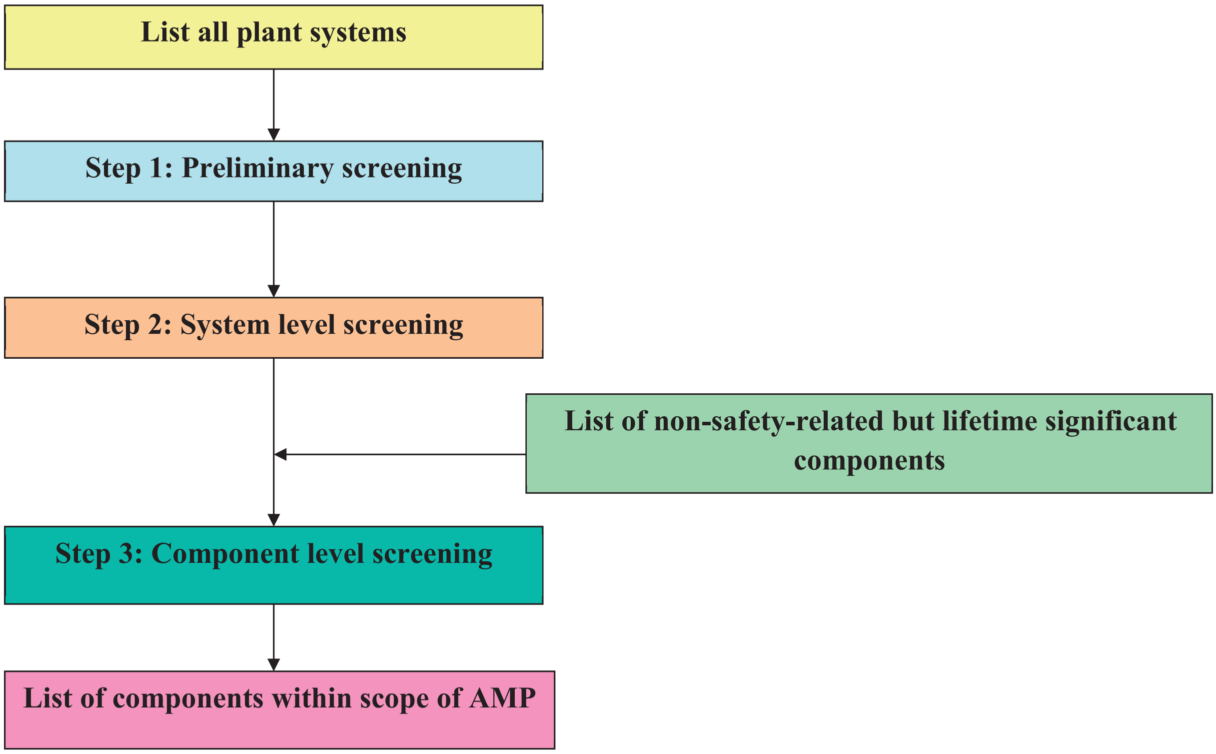


FIG. 12. Typical screening steps for AMP components.

2.6.4. Evaluation of SSCs

During a PSR, ageing is reviewed in terms of 14 safety factors at the overall plant management level, as well as at the SSC level. The following elements are taken into account:

— Programme policy, organization and resources;
— A documented method and criteria for identifying SSCs covered by the AMP;
— A list of SSCs covered by the AMP, and a record that provides information in support of the ageing management methodology;
— Evaluation and documentation of potential ageing degradation that may affect the safety functions of SSCs;
— The extent of understanding the dominant ageing mechanisms of SSCs;
— The availability of data for assessing ageing degradation, including baseline, operation and maintenance history;
— The effectiveness of operation and maintenance programmes in managing the ageing of replaceable components;
— The programme for timely detection and mitigation of ageing mechanisms and ageing effects;
— Acceptance criteria and required safety margins for SSCs;
— Awareness of the physical condition of SSCs, including actual safety margins and any features that would limit service life.

For some key components in the primary loop, such as the RPV, steam generator, primary piping and cables, among other things, TLAA may be carried out to assess their actual ageing status. For example, a mid-term ageing assessment study was carried out for RPVs, steam generators and the surge line of Qinshan-1 during the period 2006–2008, and an on-line fatigue monitoring system was established on the surge line for this purpose.

2.6.5. Feasibility study

A feasibility study for LTO should be carried out by utilities from the points of view of both safety and economy. All PLiM activities related to major changes, such as power uprates, RPV head replacements and renovation of safety related digital I&C systems, among other things, usually start off with a comprehensive feasibility study. If the feasibility study report is related to safety SSCs, it is submitted to the NNSA for review. Only if the safety review conclusions are favourable to the proposed design change can the change be implemented under the supervision of the same nuclear safety regulatory authority.

2.6.6. Documents package structure

In China, the PSR process for an NPP begins with the submission of a PSR plan to the NNSA for review before the PSR is actually carried out. The documents package includes two levels of information, the management level and the detailed technical level. The management level documents include the PSR plan, the quality assurance programme, the overall implementation programme, as well as the implementation procedures for each individual safety factor. The technical level documents include the general PSR report, the individual PSR reports for specific safety factors, and the related supporting materials. In some cases, topical reports that are not related to safety factors are also prepared.

China has no experience in preparing documentation for the purpose of LTO or plant life extension. However, it can be assumed that the documentation system would be similar to that of a PSR with additional considerations related to ageing over and above a standard licence renewal practice.

2.7. THE REPUBLIC OF KOREA’S APPROACH TO PLiM FOR LTO

2.7.1. Organizational structure

With the introduction of PSR in the year 2000, the regulator aimed at confirming that NPPs in the Republic of Korea maintain an appropriate safety level, that the ageing phenomena are managed properly, and that well trained engineers in the design, operation and maintenance of NPPs have been involved in implementing the PSRs.

Korea Hydro and Nuclear Power Company has established an enterprise wide, dedicated PSR management team coordinating continued operation, but it relies on various organizations, including contractors, to conduct needed engineering activities. Implementing continued operation is achieved by a cooperative work system, as shown in Fig. 13, among the organizations related to the project. The head office sets up a plan and secures the staffing and funds to implement the LTO project. The central research institute plays a pivotal role in preparing the data package for licensing application with support from contractors. The plant site takes charge of the field work, including plant innovation, licensing and public acceptance activities.

2.7.2. Licensing requirements

As in many other countries, the licensing period is not fixed in the Republic of Korea. However, if the operator of a nuclear power reactor wishes to continue operation after the expiration of the plant design life, an approval from the Nuclear Safety and Security Commission (NSSC) is required.

The licensing requirements are founded on two principles:

(a) The current licensing basis should be maintained during the period of continued operation to ensure that the safety level is no less than the level before expiration of the design lifetime;
(b) The acceptance standards, taking recent safety research results and OE into account, should be met in order to ensure that the highest safety level is maintained.

The Nuclear Safety Act in the Republic of Korea establishes the review subjects concerning a PSR implementation for commercial reactors and related facilities. Enforcement decrees state the time, methods, standards, processing period, details and criteria for a PSR. NSSC Notice No. 2012-25 is a guideline on the application of technical standards for the assessment of continued operation of nuclear power reactor facilities beyond their design life.

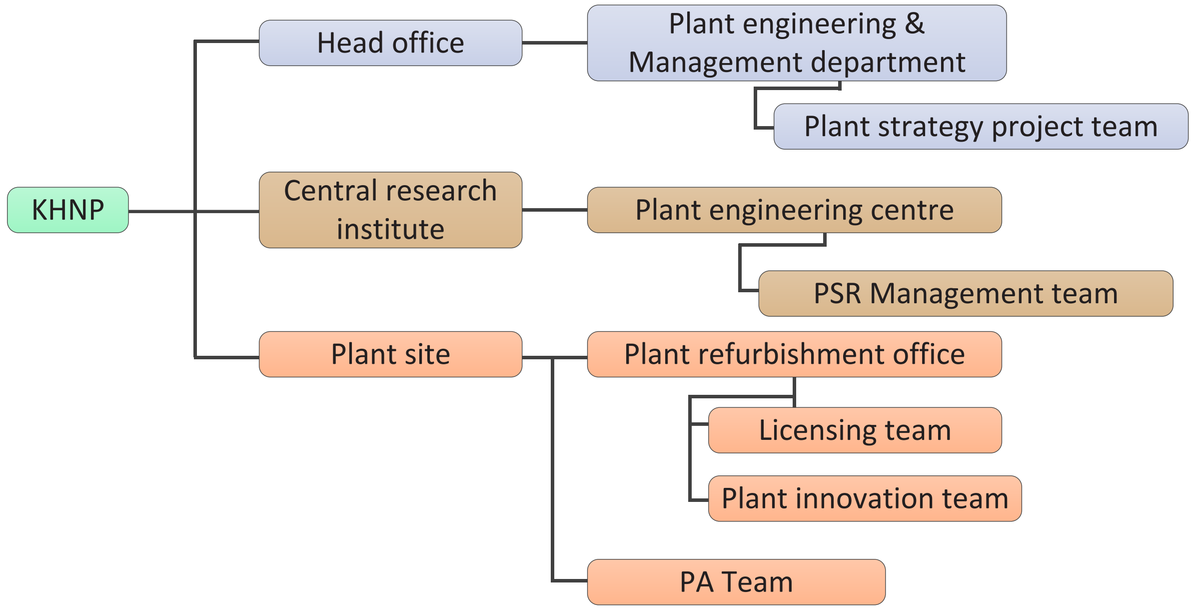


FIG. 13. Organizational structure for the continued operation project.

Based on the Enforcement Decree of the Nuclear Safety Act, to operate a nuclear reactor beyond its design life, a licensee is required to submit an assessment report from two to five years in advance of the expiration date of the facility design life. Safety reviews by the regulatory body usually take 18 months. The NSSC will approve continued operation for a maximum of ten years at a time. There is no limit to the number of operating cycles, as long as the plant can continue to run safely. When an NPP reaches the end of its design life, the licensee has the option to terminate or continue operation. Figure 14 shows the overall licensing process of a permanent shutdown or of continued operation.

2.7.3. Scoping and screening method

The licensee determines the SSCs to include in the scope of a continued operation application and submits to the regulator a list of SSCs subject to an AMP authority. In principle, the scoping process is focused on the selection of the SSCs that are screened as passive and long lived. The scoping and screening process is illustrated in Fig. 15. The methodology for ageing management is in accordance with the safety review guidelines developed based on the NRC report, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, Final Report (NUREG-1800) [13].

The selection criteria for SSCs subject to the ageing management review are determined based on multiple standards and regulations, including the enforcement regulation on PSR, US regulations on licence renewal (10 CFR 54.4), and the definitions of quality class used in plants in the Republic of Korea. The selection criterion is based on quality classes ‘Q’ and ‘A’ and characterizes safety related SSCs. To these are added the items defined in 10 CFR 54.4. The selection criteria can be summed up as follows:

- Safety related SSCs (Quality Class ‘Q’).
- Non-safety-related SSCs, whose failure could prevent a safety related function (Quality Class ‘A’).
- Other SSCs affected by retroactive regulations. This last criterion is applied only to specific SSCs, the others apply to all.

2.7.4. Evaluation of SSCs

The actual physical condition and the degree of ageing degradation of the SSCs are evaluated with reference to the design and manufacturing data, taking into account the test, operation and maintenance data in accordance with the applicable codes and standards.

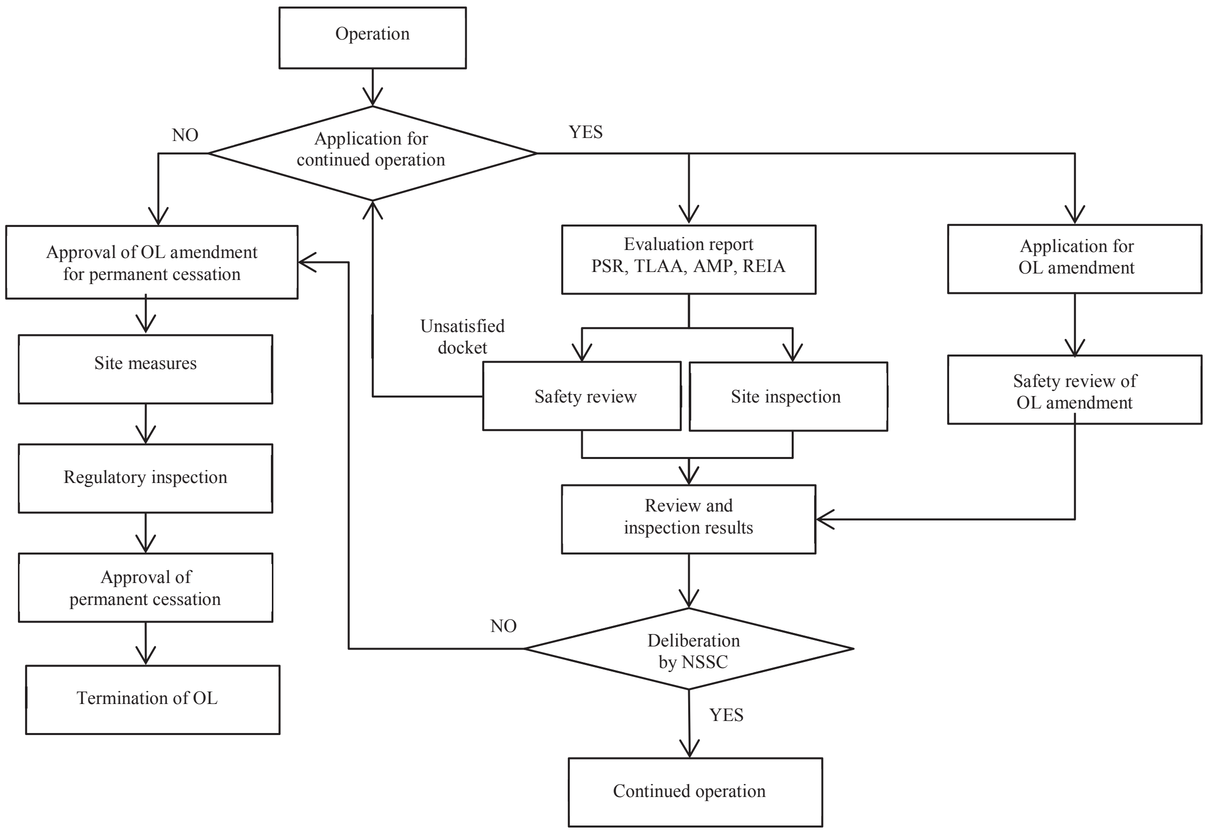


FIG. 14. Licensing process of permanent shutdown or continued operation.

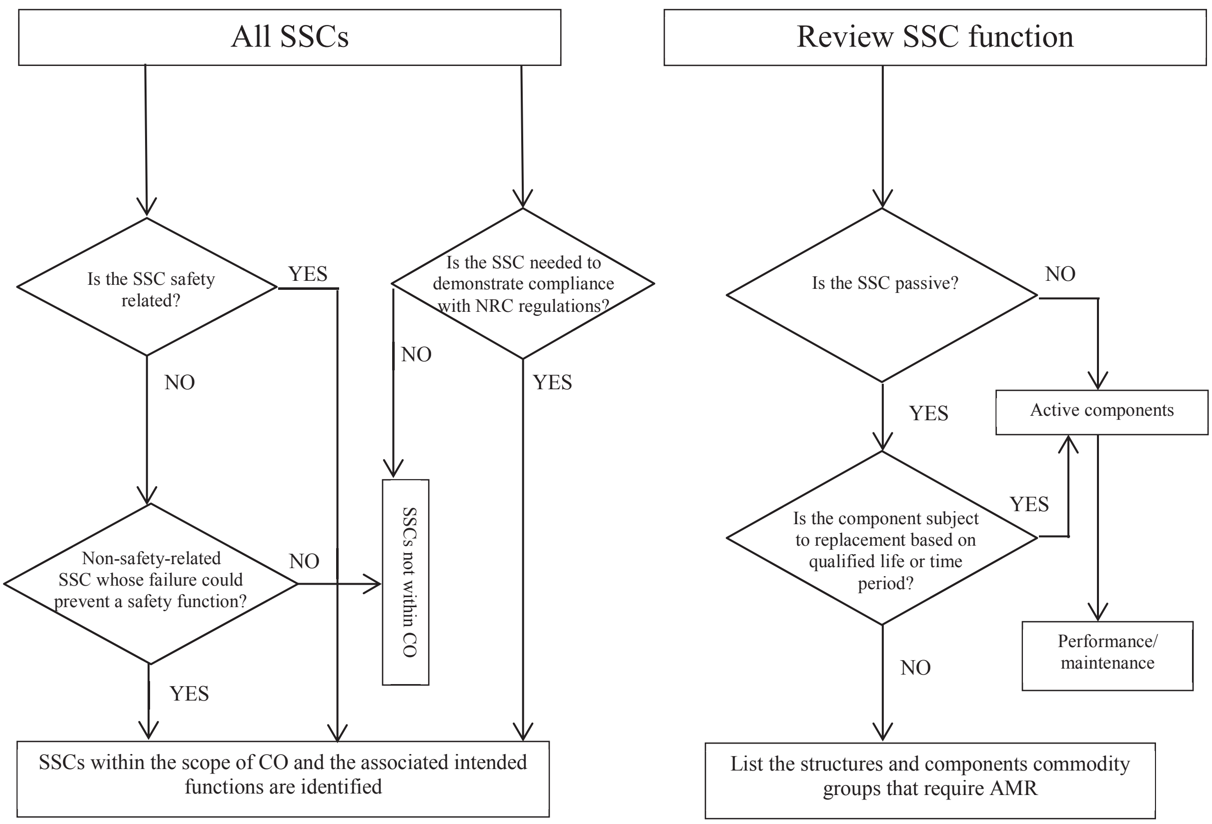


FIG. 15. Scoping and screening method.

In general, systems or components are evaluated in compliance with relevant regulatory requirements, applicable codes, standards, national laws and regulations. In addition, domestic and foreign OE and research findings are taken into account. For detailed evaluations, the actual physical condition of the SSCs is confirmed through analysis of the data collected, taking into account the original design and manufacturing specifications, any design changes, test results, the inspection and maintenance history, and any walkdown reports.

After confirming whether available records correctly apply to the facility condition assessment, ageing phenomena are analysed by system and by component. They are evaluated in terms of their capability to maintain their intended function during the continued operation period. Based on the results of this evaluation, the AMPs adopted in the facility are assessed and an audit covering safety, quality and schedule is conducted on the specific implementation of the safety improvements.

A comprehensive ageing evaluation is conducted for passive long lived components, while for active components, evaluations focus on performance, on the maintenance records and on the AMPs. Any non-conformance is noted and recommendations are issued in the final condition assessment report.

2.7.4.1. Physical condition

An assessment on whether the records currently available show the correct condition of the SSCs under review is conducted by validating their configuration and functions and by examining actual design versus expected design intent, functional requirements, operating characteristics, operating history (including test, inspection and maintenance management programmes) and the data collection and management system.

2.7.4.2. Selection of components subject to ageing management

Components of the plants are classified by safety related class (Quality Class Q) and non-safety-related SSCs whose failure could affect the safety related function (Quality Class A), and reliability class (Quality Class S) in accordance with the quality class criteria of Korea Hydro and Nuclear Power Company. This ageing review is conducted for Quality Classes Q and A. Among the components with Reliability Class S, referring to the method suggested in NEI 95-10 (Rev. 6) Appendix F, those that could have impact upon the Quality Class Q components are additionally selected.

As for the evaluation of groups of components, such as pressure vessels, valves, pumps and heat exchangers, groups are formed based on material characteristics and operational features. Among components from the groups, those operated under adverse conditions are selected. In addition, long lived, passive subcomponents performing their intended function are selected for the review.

2.7.4.3. Ageing mechanism identification

This analysis is performed to select the ageing mechanism that can affect the components subject to the ageing management review. The following methods are used to derive the ageing mechanism that could affect the components subject to the review among the ageing mechanisms that can occur in NPPs, referring to the 17 ageing mechanisms suggested in the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Appendix W:

- — Identify each ageing mechanism;
- — Identify occurrence frequency and condition of each ageing mechanism;
- — Determine to which of the ageing mechanisms under consideration a component is susceptible;
- — Determine whether domestic and foreign OE cases relevant to the component and any known research findings are reflected in the analysis.

2.7.4.4. Ageing management review

The ageing evaluation determines whether the subcomponents can maintain their integrity throughout the LTO period, resisting the impact of all applicable ageing mechanisms, according to the following procedures:

— Examination of the ageing mechanisms that could occur in each subcomponent;
— Analysis of the design and operational condition of the subcomponents;
— Evaluation of the impact of ageing of each subcomponent.

2.7.4.5. Ageing management programme

The ageing management review is conducted to confirm whether the AMPs applied to the equipment subject to the PSR are systematic and comprehensive enough to guarantee safe plant operation. The review comprises the following steps:

— Reviewing the existing AMP;
— Examining whether the programme is properly organized;
— Examining the details of the newly improved AMP.

The existing AMP refers to that originally in force in the operating NPPs. It includes programmes such as ISI and water chemistry control. When these programmes are found to be unsatisfactory in managing equipment degradation, an improved version or a totally new more systematic and comprehensive ageing management plan is established, as described in NUREG-1801 [8]. AMPs can be classified into the following four categories:

— Prevention: preventing ageing (e.g. coating and painting);
— Mitigation: mitigating the impacts of ageing (e.g. water chemistry control, boric acid corrosion and fuel chemistry);
— Condition monitoring: detecting the impact of ageing (e.g. visual inspection and measuring thickness);
— Performance monitoring: equipment performance tests (e.g. thermal performance of heat exchangers).

Among the AMPs, there are those lacking measures for prevention or mitigation. For AMPs with mitigation measures, such measures are included in the description. The ageing evaluation process is illustrated in Fig. 16.

2.7.4.6. Time-limited ageing analysis

TLAAs for continued operation of the plants are identified in accordance with the requirements of NSSC Notice No. 2012-25 in reference to NUREG-1800 [13]. The licensee lists the items subject to TLAA and demonstrates that the analyses remain valid for continued operation to the end of the period of continued operation, or that the effects of ageing on the intended function will be adequately managed for the period of continued operation. Items subject to TLAAs are selected based on the six selection criteria suggested in NUREG-1800 [13]:

(a) SSCs that are long lived and passive;
(b) SSCs that display effects of ageing during inspections;
(c) SSCs on which time-limited assumptions were defined by the operating term;
(d) SSCs determined to be relevant by the licensee from a safety perspective;
(e) SSCs whose capability to perform their intended functions may be in doubt;
(f) SSCs that are contained or incorporated by reference in the current licensing basis.

All exemptions based on TLAAs need to be justified. The technical justification of each exemption proposal for the period of continued operation needs to be accurately verified. Typical TLAA items are suggested in Tables 2 and 3.

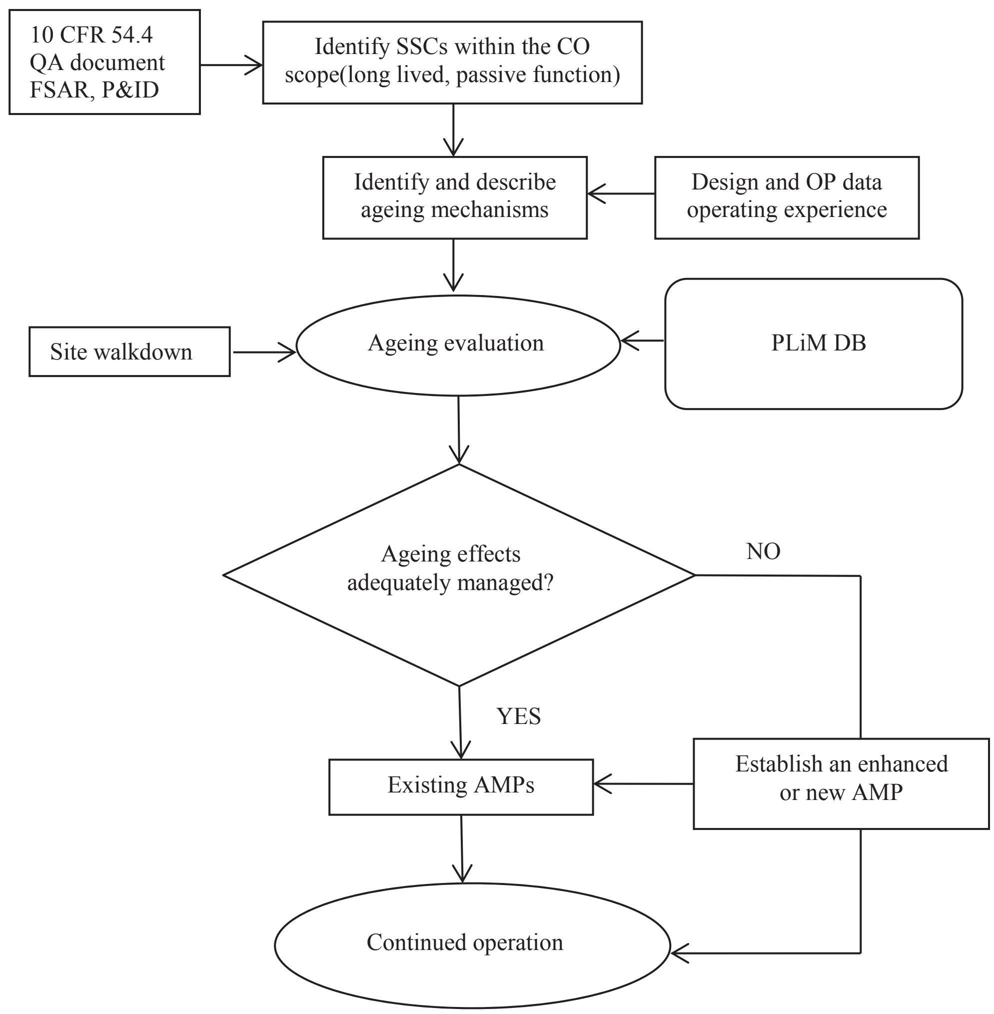


FIG. 16. The ageing evaluation process.

TABLE 2. TYPICAL TLAAs FOR PWRs

TLAA items	Reference regulations and technical standards
Reactor vessel irradiation embrittlement analysis	10 CFR 50 Appendix G
Metal fatigue analysis	10 CFR 54.21
Environmental qualification of electric equipment	10 CFR 50.49
Concrete containment tendon prestress analysis	10 CFR 54.21
Containment liner plate, metal containment and penetrations fatigue analysis	10 CFR 54.21
Other plant specific TLAAs	10 CFR 54.21

TABLE 3. TYPICAL TLAAs FOR PHWRs

TLAA items	Reference regulations and technical standards
Life assessment of reactor assembly and fuel channels	CNSC Reg. Guide G-360 and 10 CFR 54.21
Metal fatigue analysis	10 CFR 54.21
Environmental qualification of electrical equipment	10 CFR 50.49
Concrete containment tendon prestress analysis	10 CFR 54.21
Fatigue analysis of containment penetrations	10 CFR 54.21
Other plant specific TLAAs	10 CFR 54.21

2.7.5. Feasibility study

The first step when considering continued operation is to perform a feasibility study to verify if the goals of current operation can be maintained or even improved, and to objectively show the viability of the LTO programme. Continued operation is, in most cases, technically feasible, but its implementation, based on the LTO goals, may not always be feasible from an economic standpoint. In order to compile the technical, economic and licensing requirements, the feasibility study team reviews all requirements, collects all data, reviews all current regulatory requirements, conducts an evaluation of key SSCs that could tilt the balance and determine the feasibility of an LTO programme, and conducts a preliminary economic assessment.

2.7.6. Documents package structure

The licensee who wants to operate an NPP beyond its original design life submits a PSR report evaluating the required 11 safety factors, and the following two additional reports:

— A life prognosis of the SSCs under consideration for the period of continued operation;
— An assessment of the radiological and environmental impact on the territory, considering the plant age condition and all design changes since the start of operation.

The documentation for an LTO permit application should provide sufficient information to clearly justify the safety of the plant’s continued operation beyond its original design term.

The life evaluation report for the major components should contain, in sufficient detail, technical information about the various types of ageing effects that might be encountered during the continued operation term and how the licensee will manage those effects, as follows:

— Scoping and screening report of the SSCs subject to ageing management review;
— AMPs;
— TLAAs, including the continued operational term;
— Operating experience feedback and important safety research results.

AMPs should contain the programme scope, preventive actions, monitoring and inspection parameters, detection of ageing effects, monitoring and trending, acceptance criteria, corrective actions, administrative controls and OE. In addition to these elements, special attention should be given to OE and safety research results as requested by the safety review guidelines of the Korea Institute of Nuclear Safety. AMPs should indicate all measures to be taken prior to continued operation, and after the start of continued operation, depending on the design characteristics and OE.

The radiological environmental impact assessment (REIA) is to be carried out in accordance with the technical standards in effect at the time the REIA was most recently conducted at the same site. Providing that the NPP has implemented the REIA and submitted it to NSSC in a previous operating cycle, the evaluation can be limited only to the parts not previously evaluated for earlier operating cycles. Evaluations should cover:

— The continued operation plan;
— The environmental status of the territory;
— A global condition assessment of the NPP;
— The environmental effects of continued operation on the territory;
— The effects of postulated accidents;
— The environmental monitoring programme.

In addition, if there are configuration changes identified through the safety review of LTO, FSAR and technical specifications, supplements should be submitted in parallel with the continued operational application. Alternatively, the licensee could submit the FSAR and technical specification supplements independently after the continued operation application.

FSAR supplements should contain a summary description of the FSAR parts affected by the continued operation application. The supplement should also contain details about the programmes and activities for managing the effects of ageing and the evaluation of TLAAs during the continued operational term. Changes or additions to the technical specification, if any, should also be provided as technical specification supplements with the permit application for configuration changes. If full justification for the changes or additions is addressed in the technical specification supplement and attached to the continued operation application, it should not be repeated in the application documentation.

2.8. THE RUSSIAN FEDERATION’S APPROACH TO PLiM FOR LTO

2.8.1. Organizational structure

In the Russian Federation, the approach strategy to PLiM for LTO began to develop at the end of the 20th century, when the economic feasibility of the life extension of Russian NPPs beyond the original 30 year design life became apparent. The first NPP to reach its design life in 2001 was Novovoronezh (Units 3 and 4 with WWER-440s). Table 4 provides a summary reporting on the life extension granted to Russian reactors.

TABLE 4. SUMMARY OF THE LIFE EXTENSION OF RUSSIAN REACTORS

Reactor type	Life extension duration
WWER-440 (1, 2 Kola NPP, and 3, 4 NVNPP)	Extended to 15 years. Total: 45 years. Currently considering to further extend operating life for another 15 years
WWER-1000	Life extended for 30 years. Total: 60 years
High-power channel-type reactor (RBMK)	Life extended for 15 years. Total: 45 years
BN-fast neutron reactor (Beloyarsk NPP)	Life extended for 15 years. Total: 45 years
EGP-6 small RBMK (Bilibino NPP)	Life extended for 15 years. Total: 45 years

In 1998, a ministerial order was issued by the Russian Ministry of Nuclear Industry about conducting a feasibility study to demonstrate the viability of the life extension of first generation Russian NPPs. To this end, a new technical organization was created based at Rosenergoatom, also known as the Concern for Production of Electric and Thermal Energy at Nuclear Power Plants. The main task of this structure was to develop an LTO strategy, taking into account the operating features of the Russian NPPs. Within Rosenergoatom, a special PLiM programme for LTO was created.

The scope concerning the plant life extension (PLEX) of Russian NPP units included the following tasks:

- Analysis and substantiation of the social and economic advantages of PLEX projects;
- Safety improvements of the NPP units (modernization and reconstruction);
- Assessment of the residual lifetime of major SSCs;
- Environmental impact study (nuclear and radiation), fire protection and nuclear safety during the additional operating period of the NPP units.

The PLEX project's main task was to demonstrate that the investment in the LTO project would produce economic and social benefits in parallel with acceptable safety improvements. This is within the competence of the Government of the Russian Federation, according to the federal law of the Russian Federation on the use of atomic energy. Issues regarding the safety of Russian NPP units are within the competence of the supervisory (oversight) body specially authorized by the Federal Environmental, Industrial and Nuclear Supervision Service (Rostechnadzor). Rostechnadzor has developed and approved industrial norms and procedures to ensure the appropriate level of technical and ecological safety of NPP units, even beyond their design lives.

Rosenergoatom is the operating company responsible for the generation of electrical and thermal energy at all NPPs in the Russian Federation. Although lifetime extension studies are executed by the individual NPP, since all Russian NPPs are affiliated branches of the operating company, all plants need to follow the standard norms and rules mandatory for the entire Russian nuclear industry.

A specialized department of modernization and lifetime extension exists in each NPP. This department provides common coordination of PLEX work and routine progress control of the LTO implementation plan. A special commission has been set-up for the assessment of the state of the plant SSCs (condition assessment) and for the estimate of their residual life. The commission includes representatives of the operating company, the main designer, the reactor construction company, as well as representatives of the manufacturers, the head of the material science organization and other specialized companies. As a rule, for the implementation of a PLEX programme, the NPP hires, on a contract basis, specialized external companies accredited with Rosenergoatom. All contractors are required to apply for Rostechnadzor work permits, giving them the authorization to execute the prescribed work. The selection of all specialized companies for PLEX work execution is always carried out on a competitive commercial basis.

Rostechnadzor is responsible for the independent selection of expert organizations and for the appraisal and quality assessment of the executed PLEX work. It is guided by industrial norms and rules when it executes a comprehensive review of the PLEX documents justifying the NPP unit life extension. The NPP provides responses to all questions and observations provided by Rostechnadzor. Only after all questions have been answered and PLEX work has been executed in its full scope and in full compliance with the prescribed norms will Rostechnadzor approve the scope of work, giving to Rosenergoatom the licence to operate the NPP units beyond their design life.

Overall, the legal basis of the PLEX norms is comprehensive and provides ample guidance for NPP unit life extension beyond its 30 year design term. It covers the documentation review, the confirmation of the residual service life of its SSCs, the economic analysis, the confirmation of NPP safety and the issue of the new licence for operation. The concerted justification of all the essential aspects of safety, maintenance and ageing analysis leads to LTO licence approval. Thus, the NPP units operating under an LTO licence become an instrument for the advancement of nuclear power and the benchmark of actual progress in NPP efficiency, safety and social acceptability.

2.8.2. Licensing requirements

The Gosatomnadzor publication Basic Requirements for Power Unit Lifetime Extention of Nuclear Power Plant[s] (NP-017-2000) sets the design service life of Russian NPP units of the first generation at 30 years. Beyond this term, further operation can be continued only after approval of the special decision accepted on the basis of safety investigations and economic assessments.

Taking into account the positive OE of the first generation of NPPs and the strong economic drivers in favour of LTO, it was decided to extend the LTO validation of these plants for an additional 15 years of operation (+100 000 hours). To this end, specific regulatory guidelines have been issued. They are a step by step procedure to help prepare the documentation and execute the activities necessary to obtain a plant life extension licence. The regulatory guidelines consist of two document types:

— A first tier document defining the PLiM aspects, its processes and activities;
— A second tier document containing concrete (specific) guidelines for the SSC (vessels, pipes and buildings, among other things) ageing prediction calculations supported, when necessary, by experimental substantiation.

These regulatory documents are developed within the framework of the NPP safe operating principles, formulated in the Russian industrial norm Basic Principles of Nuclear Power Plant Safety (OPB-88/97); Rules for Arrangement and Safe Operation of Equipment and Piping of Nuclear Power Installations (PNAEG-7-008-98); PNAEG-7-002-86 [14]. These documents regulate NPP safety at all stages of plant life.

The provisions for life extension of Russian NPP units beyond their 30 year design term are prescribed in item 5.1.14 of NP-001-97, General Regulations on Ensuring Safety of Nuclear Power Plants. Depending on the SSC residual lifetime assessment and the results of the safety review, the operating company can decide to extend its NPP unit service life. In this case, a new operating licence for the unit must be obtained.

As they have been developed precisely for a long term operating licence application, the regulatory documents also constitute the basic norms for a PLiM programme. The structure of the normative documents is presented in Fig. 17.

The main document governing a PLEX implementation programme at Russian NPPs is NP-017-2000, which carries the weight of federal norms and rules. According to NP-017-2000, before an NPP unit reaches its design service life, the operating company must perform a comprehensive safety review of its ageing conditions, based on which it will decide whether to prolong its service life, or decommission it. The methods used for safety assessments should include a degree of conservatism commensurate with the uncertainty of the input data. Any software used should be certified according to the prescribed requirements.

The length of time an NPP unit can operate beyond its design service life needs to be established in accordance with technical and economic factors, taking into consideration the following specific issues:

— Possibility of ensuring and holding safety margins during the NPP unit LTO period;
— Existence of sufficient residual service life of non-replaceable equipment;
— Availability of temporary storage to contain the additional amount of spent fuel produced, or the possibility to transport it outside the NPP site;
— Safety assurance for the handling of additional radioactive waste formed during the NPP LTO period;
— Safety assurance of the NPP unit during its decommissioning at the end of its extended operating period.

The operating company sends to Rostechnadzor the final results of its PLEX study in order to obtain the operating licence beyond the NPP's design life (during the additional period of operation). The list and contents of the documents substantiating safety of the NPP unit during the additional period of operation are prescribed by Regulatory Document RD-04-02-2006, Requirements for the Documents Package List and Documents Contents Substantiating Safety of an NPP Unit during the Additional Period of Operation. The main stages of the PLEX work are presented in Fig. 18.

To apply for an LTO licence in the Russian Federation, a deterministic approach was selected. The essence of this approach consists of the assessment of the state of the SSCs (condition assessment) on the basis of:

— Review of the original 30 year operating licence period;
— Condition assessment of the SSCs at the end of the 30 year operating period, based on the ageing calculation estimate, supported by experimental confirmation;
— Prognosis regarding the operating period extension beyond design life of the unit.

Federal normative documents of the top level
NP-001-97 (OPB 88/97), PNAE G-7-008-89, PNAE G-7-002-86, NP-017-2000

Standards of the operating company (Rosenergoatom)
STO 1.1.1.01.006.0327-2008, Lifetime extension of an NPP unit
(determines the procedure, order and terms of PLEX works execution of an NPP unit)
STO 1.1.1.01.007.0281-2010, Lifetime management of NPP units' elements
(determines the methodology and procedure of lifetime management of NPP units' elements)

Guidelines of the operating company (Rosenergoatom)
There are more than 50 guidelines which determine requirements on the scope and order of works execution concerning assessment of technical condition and substantiation of residual lifetime of various NPP units' elements.

— RD EO 1.1.2.09.0774-2009, Procedure for assessment of technical condition and substantiation of residual lifetime of pipelines, vessels and pumps of NPP units;
— RD EO 0538-2004, Procedure for lifetime substantiation of containment of NPPs with WWER-1000 units;
— RD EO 0447-03, Procedure for assessment of technical condition and substantiation of residual lifetime of NPPs' concrete structures, important for safety;
— RD EO 0146-99, Procedure for assessment of technical condition and substantiation of residual lifetime of electric cables of NPPs' systems, important for safety and instrumentation and control systems;
— RD EO 0190-00, Procedure for assessment of technical condition and substantiation of residual lifetime of armature of NPP units' operating systems;
— RD EO 0195-00, Procedure for assessment of technical condition and substantiation of residual lifetime of NPPs' diesel generators;
— RD EO 1.1.2.22.0283-2008, RD EO 0141-98, RD EO 0234-00, RD EO 0277-2006, RD EO 0294-01. RD EO 0362-02. RD EO 0410-02.

FIG. 17. Structure of normative documents regulating PLEX work.

The approach to LTO is based on the PSR concept. Such an approach is justified by the fact that the NPP service life is mostly determined by the actual condition of the SSC materials. If a LTO licence is granted for a 15 year term, the service life can be extended further from 15 to 30 years. A good PLiM programme includes technical and economic estimates and sets the preconditions for:

— Maintenance level needed to maintain the required safety level and the desired capacity factor;
— Optimization of maintenance by implementing correctly prioritized repairs;
— Ageing detection, life prognosis and consequent ageing rate control.

The life management programme and the method used needs to meet the requirements of:

— National laws (mandating quality in activities involving radiation) and defining criteria for the equipment classification in safety levels;
— IAEA guidelines for the plant life management of NPP equipment, PSR and LTO [1].

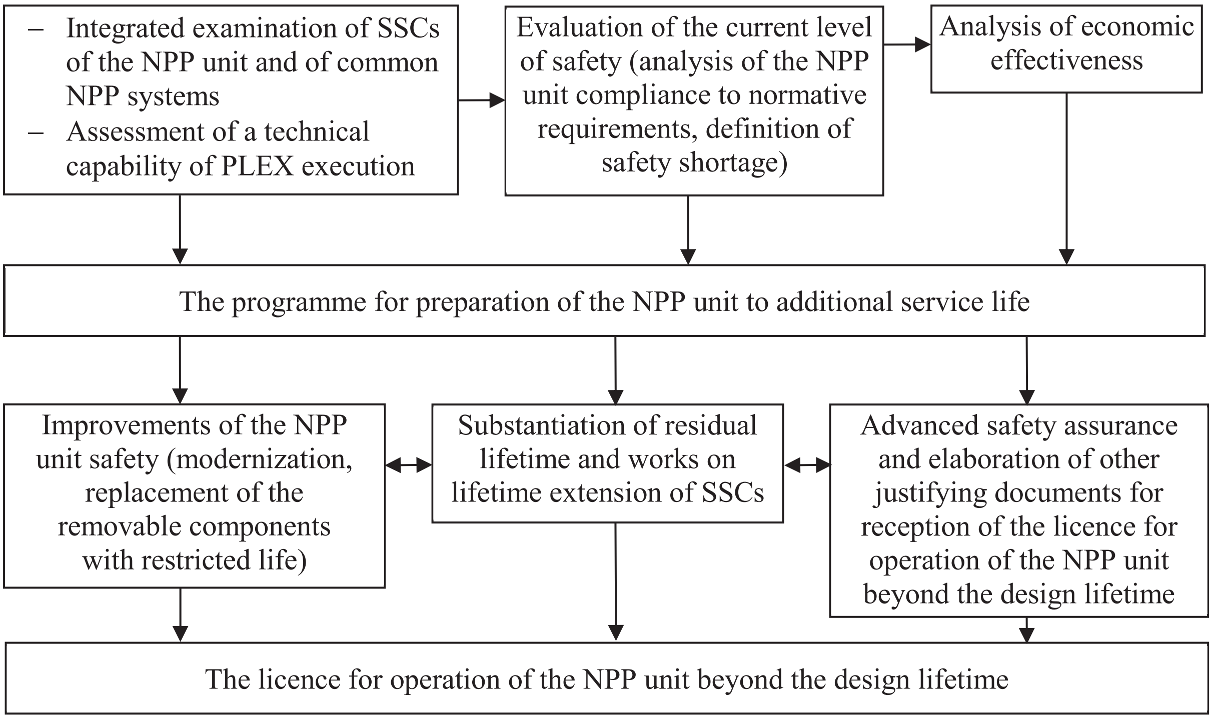


FIG. 18. Main stages of a PLEX project.

2.8.3. Scoping and screening method

The inspection procedure of the safety function integrity of passive, active and other SSCs is based on the following general requirements:

— Prediction of equipment ageing;
— Monitoring of the consequences of ageing on equipment performance;
— Compensation of the consequences of ageing;
— Maintenance surveillance and inspection programme improvement taking into account the new knowledge;
— Estimation of programme efficiency.

Lifetime management of SSCs in NPP units is carried out along the following three tracks.

(a) Maintenance during operation

Lifetime management of SSCs are governed by design, operational or normative documents recommending a programme of periodic inspections, testing, preventive maintenance or even an extensive overhaul carried out within the framework of the maintenance programme. The prioritization of such elements and their periodicity is defined in the maintenance or replacement schedule.

For SSCs being managed through PLiM, the documents which confirm the possibility of an operating period up to the next periodic inspection are the work order on a repair service (recovery and extensive refurbishment), or the work order of an ISI confirming that the component will be able to perform its functions.

Components are replaced when they reach their ultimate inadmissible state, following periodic inspection confirming their technical state as beyond repair or from testing results confirming that recovery (repair) or even a heavy overhaul of those components is admittedly technically or economically inadvisable.

(b) Replacement (modernization)

A decision on the replacement of NPP components is imperative when the component loses its functionality and its recovery cannot be economically achieved.

(c) Life extension (refurbishment)

A condition assessment through inspection, testing and residual life estimates should be done for non-recoverable and irreplaceable SSCs, or for SSCs whose design life is close to expiration or whose operational or normative constraints dictate it. Well before the end of the design or the added service life of NPP units, a review or a refinement of its non-replaceable SSC condition assessment and a prognosis or a projection of their expected performance and the level of maintenance, surveillance and inspection recommended during the desired extended period, is required.

To prepare for an LTO or second LTO application, a list of SSCs for which life extension work is required should be compiled in addition to a separate list for each NPP unit and a special sub-list for systems shared or in common with other NPP units.

The SSC list should contain components such as:

— Equipment and pipelines of safety classes 1, 2 and 3 according to NP-001-97 and covered by the requirements of PNAE G-7-002-86 [14], for which allowable fatigue cycles and other lifetime performances will be exhausted before the end of their design life or the end of their additional service life period.
— Equipment and pipelines of safety classes 1, 2 and 3 according to NP-001-97, for which lifetime performances are not recoverable, supported or controlled by the operating maintenance system and accepted at the NPP. These are components for which methods and means for inspection, condition assessment, residual lifetime estimates and repair and recovery methods do not exist at the NPP.
— Turbines, shutoff valves of the reheat system, cross-over pipelines within the limits of the turbine and steam extraction pipelines (if shutoff valves exist) from the turbine up to the shutoff valve.
— Buildings and construction important to safety.
— Climbing cranes registered with Rostechnadzor.
— Metal structures and covers sealing internal space of water graphite reactors, including the jacketed graphite stack and its corresponding elements.
— Internal components of WWER type units.
— Other components for which lifetime management was judged to be technically feasible and economically convenient.

Not included in the special list are elements for which lifetime management is carried out by operating maintenance, namely:

— Controllable, repairable and recoverable elements (including elements located inside the pumps, vessels or pipelines);
— Elements subjected to replacement or modernization;
— Heat transfer and mechanical equipment and pipelines of safety class 4.

2.8.4. Evaluation of SSCs

An example of the approach used is given in the following for a condition assessment of NPP components made of construction steels (heat transfer, mechanical equipment and connecting pipelines), with an estimate of residual lifetime and related documentation. The approach is standard in the Russian Federation for NPP components of safety classes 1, 2 and 3, and follows the requirements and recommendations of industrial norms, rules and standards, as well as guidelines and specific procedures of the operating company.

In the first stage, comprehensive collection, review and analysis of design and construction data and documentation are conducted for each component, as follows:

- — Integrated examination of the NPP unit and of its operating history;
- — Maintenance documentation;
- — NPP unit passport data (technical specifications);
- — Stress analysis data (if available);
- — Layouts;
- — Assembly drawings;
- — Results of on-site inspections and technical examinations;
- — Repair documentation;
- — Data on operating modes and conditions, running hours, number of cycles of heat-up/cool down (start/stop), number of hydro tests, etc.

In the second stage, a preliminary stress analysis is executed according to PNAE G-7-002-86, Norms for Stress Analysis of Equipment and Pipelines of Nuclear Power Plants, and RD EO 0330-01, Guidelines for Stress Analysis of Equipment and Pipelines of RBMK and WWER Reactors in the Operating Configuration. According to RD EO 0330-01, cyclical fatigue effects by the working fluid are taken into account.

All relevant information concerning construction materials, geometry and design features, among other things, obtained from the design documentation and passports, is used as input data for the stress analysis. Actual operational modes and historical conditions, all recorded failures and deviations of actual operating parameters from design, are also considered in the stress analysis.

The model of expected load combinations for the extended lifetime is developed for the purpose of calculating the residual lifetime of an element. The average loading spectra of the last ten years of operation is assumed to be the model of future annual loadings. Other loadings can also be considered in the operating modes and conditions of the extended operation period.

Based on the stress analysis results, the most loaded areas are those in which the metal may be potentially more heavily subjected to ageing degradation and to loss of physical–mechanical properties as a result of ageing stressors such as thermal deformation and low cycle fatigue, among other things. In the most loaded zones it is also possible to expect the formation of operational flaws. Therefore, special emphasis is put on the zones highlighted by the stress analysis for follow-up, testing and non-destructive inspections during the LTO period. This occurs particularly in welded joint areas. These required inspections and tests are then incorporated into the PLEX recommendations and from there into NPP planned activities.

In the third stage, a collection of input data for each of the selected components is systematically compiled from the results of past ISIs, from the examination of data on repairs and replacement activities and from the definition of areas of potential high stress, as revealed by the stress analysis.

If a condition assessment is to be carried out for one component group of the same type, one or several samples from the group can be selected for a detailed comprehensive inspection. In this case, the components for special inspection are selected based on the time they have been in operation, on the number of stress cycles (heat-up/cool down and hydro tests) and on the operating conditions, especially those more favourable to the initiation of operational flaws. Another selection criterion is that of components that may have been subjected to repairs by welding.

Once the selection of the samples has been completed, if their inspection yields unsatisfactory results in one or more of the inspection criteria applied (e.g. if the metal condition does not comply with industrial standards), then an increase of the inspection scope may be recommended and justified for the component group. Conversely, if the inspection of the typical component representative batch is satisfactory, there should be no increase of the inspection scope.

The Russian technical documentation provides guidelines to assess the ageing mechanisms and their effects on all SSCs through the use of comparative key parameters and quality criteria to uniquely characterize the SSC condition. In addition, the documentation provides guidance on the most appropriate methods of non-destructive inspection to typify the SSC conditions at the end of their design life.

The detailed selection of the inspection zones (welded joints, heat affected zone and base metal) within the equipment and the pipelines, and the selection of the most suitable inspection methods, is based on considerations related to their installation features, to their actual operating history, and to statistical data on the characterization of flaws related to operational modes and environmental conditions.

The selection of inspection zones in metal components hinges on observed tendencies (or lack thereof) for degradation in the main parameters characterizing the metal condition (e.g. mechanical properties, wall thickness and presence of defects). To account for uncertainties during the selection of inspection zones, conservatism is applied, and therefore the worst zones are slated for the most comprehensive examinations. These are typically zones in which the maximum degradation is expected to occur, as indicated by the stress analysis and by experience feedback associated with the SSC type, as opposed to zones for which there are no objective causes for concern.

Detailed inspection requirements and the scope of special examinations for the selected zones are recorded by the PLEX project staff in a document entitled, Programme of Inspection, Condition Assessment and Residual Life Estimate of Essential Components.

Metal inspection, as a rule, is conducted using the following methods:

— Visual control;
— Dye penetrant inspection;
— Ultrasonic testing;
— Ultrasonic wall thickness measurement;
— Mechanical properties testing and measurement;
— Ferrite phase measurement.

The specification of specially designed or customized methods depends on the component specific features.

In the fourth stage, the metallic component integrated inspection programme is prepared by combining the outcomes of the individual inspection programmes for each selected component. Non-destructive inspection is executed in compliance with the integrated programme during a scheduled outage of the NPP unit. In order to conduct high quality inspections, the most advanced methods and inspection tools are used, which help to obtain a more precise assessment of the metal conditions, and include:

— The precision measurement of mechanical properties, using the kinetic indentation method, and the projection of possible degradation during the extended period of operation, taking into account all dominating ageing mechanisms.
— Ultrasonic testing of weld joint integrity, using phased array techniques (the latest development in non-destructive methods of welded joints inspections) allows detection of the flaw type and the precise measurement of its specific geometric dimensions.
— The on-line continuous measurement of metal wall thickness reduction using the combined electromagnetic-acoustic method for selected carbon steel piping elements particularly subject to erosion–corrosion (bends, straight pipe spans located downstream of pressure regulating and flow throttling fittings). These tools feature three dimensional plotting capability for flaws and wear signs, and the ability to detect and localize areas exhibiting maximum metal wear.

As a rule, the methods and tools used in these special inspections are not usually specified in normal periodic inspection programmes for metallic components. The condition assessment of metallic components is performed in full compliance with the requirements of the Russian industrial guidelines.

In the fifth stage, a document on the condition assessment and residual life of essential components is prepared for each of the selected components. It contains the component's technical and operational history, the stress analysis results, the governing degradation mechanisms, any ageing reports, the results of in-service and special expert inspections, and the projections of the metal condition evolution to the end of the component service life. The acts, reports and protocols of the condition assessment and the inspection scope, defined in the inspection programme, are appended to this document and summarized in its conclusions.

Certified finite element software is normally used when highly detailed stress analysis is required. This method allows more precise material behaviour projections for the extended lifetime. The finite element analysis executed at this stage incorporates the measured values of the input parameters obtained from the component condition assessment (wall thickness, mechanical properties and flaws, among other things) at the end of its design life. The model also includes the degradation forecasts of such parameters during the planned additional service life.

The definition of the governing degradation mechanisms is important to support any prediction of changes to the metal properties and to the fundamental ageing parameters of the SSC during its extended life term. A typical prediction of mechanical property degradation is made taking into consideration the actual measurements of mechanical properties at the end of the component design life, and the performance data resulting from investigations on ageing mechanisms and their effect on the class of components of the same type in actual or simulated LTO conditions.

In the conclusion of the document on condition assessment and residual life of essential components, there are also recommendations with regard to the implementation of compensating life management measures to account for the more challenging conditions of the extended service period and any other inspections or testing that need to be strictly scheduled during the LTO in order to guarantee the safe operation of the selected essential components beyond their design life in the context of the safe LTO of the NPP unit as a whole.

In the sixth stage, taking into consideration the main results and the recommendations of the PLEX programme as described in phase five, a final decision document on the terms and conditions for further operation of the unit is detailed for each of the essential components.

In the seventh stage, a programme for lifetime management of NPP unit components during the extended service life is prepared based on the assessment of the state and residual life of each component. This document includes:

— The required periodic inspection of the mechanical properties and ageing parameters of the essential components during the additional extended service life, which enables the evaluation of the actual degradation trends of such parameters, and allows the operator intervention to prevent the component from reaching its ultimate inadmissible state. This action is based on the known dominant degradation mechanisms and their expected effects on component ageing.
— The implementation of measures for the optimization of operating conditions and the mitigation of ageing mechanisms on the essential components.
— Operating practices in support of component reliability in accordance with the requirements of the Russian normative and technical requirements in order to guarantee the necessary level of safety during the LTO period.
— A list of components due for replacement as they approach their ultimate acceptable operating state.

In the eighth stage, the full technical documentation package containing the complete condition assessments and the life prognosis throughout the LTO period of each of the components selected in the PLEX programme is sent to Rostechnadzor for comprehensive expert review and approval. If the review is judged satisfactory, it is followed by the issue of a licence for the NPP unit extended operating period.

2.8.5. Feasibility study

A feasibility study for an LTO investigation is a set of specific studies aimed at assessing the technical capability, the safety performance and economic justification for an extension of an NPP unit service time beyond its original design life. A feasibility study is usually the first stage of a PLEX programme. This stage, as a rule, begins eight to ten years before the end of the design service life and includes the following main activities:

— An integrated examination of the state of the NPP unit.
— A safety assessment of the NPP unit.
— The scope and list of activities required to prepare the NPP unit for extended service, including activities such as component replacement, modernizations and technical justification for a life extension.
— Development of an investment model for the PLEX project.
— Components marked for replacement in case of technical or economic limitations preventing their lifetime extension. Component replacement is evaluated in view of the following factors:
    - Existence in the Russian Federation, or abroad, of a proven replacement technology for the component types;
    - Possibility to dispose (e.g. burial) of large scale contaminated components;

- Radiation burden on NPP personnel during component replacement;
- Duration of the NPP unit outage connected with component replacement and corresponding financial losses;
- Requirements for normative documentation;
- Cost of the replacement components.

In terms of the economic analysis of a PLEX project, all possible scenarios should be considered such as various durations of the additional service life. Depending on the reactor type, durations will vary. For RBMK units, for example, it has been decided that the considered additional service life durations will be limited to 10, 15 and 20 years and for WWER reactors to 15, 20, 25 and 30 years.

According to the feasibility study results, and taking into consideration the investment model for PLEX, the operating company approves the preparation of the NPP unit for the additional service life. Such a decision needs to be approved five years prior to the expiration of the NPP unit design life.

After the authorization to proceed with the NPP unit life extension, the technical and administrative documentation, defining the sequence and terms of the PLEX activities, as well as the sources and amounts of financing, is prepared.

2.8.6. Document package structure

The NPP unit licence for LTO can be obtained by the operating company through the submission, to Rostechnadzor, of an application package reporting on the nuclear safety, the environmental impact, the radiation protection and the industrial safety during the extended period of operation beyond the original design term.

The documents include the following:

— The advanced safety assurance report;
— The unit preparation programme for lifetime extension programme and assurance of its implementation;
— The integrated evaluation report of the state of the NPP unit;
— The unit technical condition assessment and SSC residual life estimates, which include the justification, limitations and conditions associated with operation of the SSCs beyond their design life;
— The PLiM programme for the NPP unit during the LTO period;
— Operating regulations of the NPP unit during the LTO;
— The passport of the reactor facility of the NPP unit;
— Instruction on accident prevention at the NPP unit;
— Guidelines on the management of beyond design basis accidents;
— Staff protection measures during accidents at the NPP unit;
— Quality assurance programme for the NPP unit operation;
— Information on selection, training, qualification and admission to independent work of the NPP unit personnel;
— Instruction on assurance of nuclear safety during storage, handling and loading of nuclear fuel;
— Instruction on the registration and control of nuclear fuel and materials at the NPP unit;
— Notice on the maintenance of the NPP unit physical protection features;
— Copy of the interdepartmental commission certificate on the organization of the NPP unit protection;
— Notice on the registration and control system of radioactive water and radioactive waste.

The documents listed above are enclosed as attachments to the application for a new LTO licence for the NPP unit. They should be sent to Rostechnadzor at least one year before the expiration of the unit original design life.

2.9. SPAIN’S APPROACH TO PLiM FOR LTO

There are eight nuclear units in Spain located in six different sites. All were designed for 40 years of operation. One example is Santa Maria de Garoña (SMG), which reached LTO in 2011. This plant has completed the process and the application required to operate beyond 40 years. The other plants in Spain have not yet applied for LTO, as they will not reach 40 years of operation until the 2020s. The SMG plant is taken as an example of a fully completed LTO application process.

2.9.1. Organizational structure

In Spain, the organizational structure for PLiM varies widely from plant to plant. Some have a dedicated organization, others do not. The main task in an LTO application is the preparation of an integrated ageing management assessment plan. At SMG, this was developed by a team specifically created for this purpose. A group of 12 specialists from Spanish engineering companies worked for 2.5 years, together with a group of 10 staff from the plant. Experts in the US licence renewal process were also used during the project to assess the work done. Figure 19 shows the structure of the team that prepared the integrated ageing management assessment plan. The red colour represents the plant staff, and the blue colour the contractors.

After submitting the application for LTO, a new position was created in the plant to manage and coordinate the AMP. In addition, 10 programme owners were nominated to manage the 43 AMPs.

2.9.2. Licensing requirements

In Spain, there is no legal or administrative limitation on the operating life of NPPs. However, the operating permits are warranted through an ongoing assessment and PSRs. The Ministry of Industry issues the new operational permits based on the technical evaluation made by the Nuclear Safety Council (Consejo de Seguridad Nuclear, CSN), in a safety evaluation report.

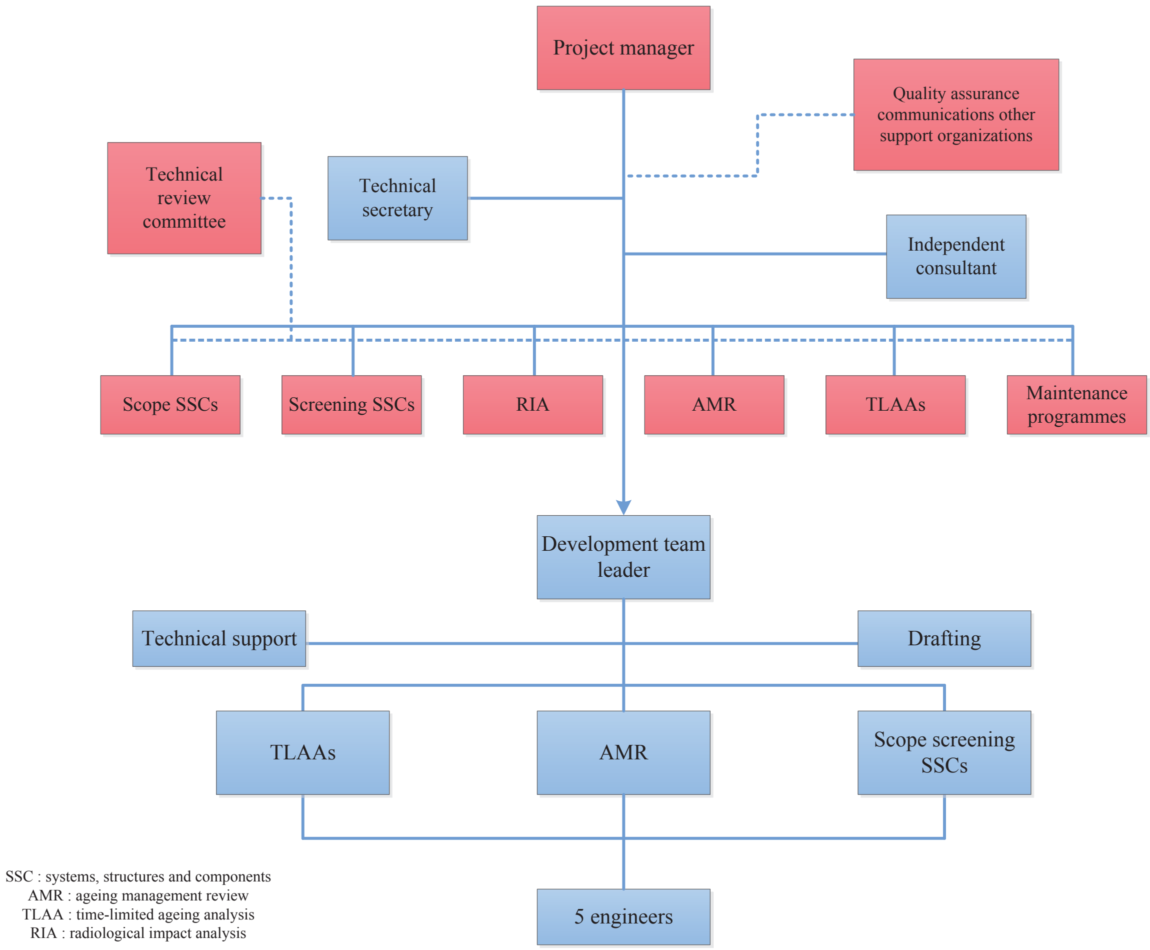


FIG. 19. Licence renewal 2009–2019: The ageing management review team.

The PSR is the basis for the renewal of the operating licences and is required at each ten year interval. CSN Safety Guide 1.10 establishes the content of a PSR. In addition, every ten years a new regulatory applicability study is performed as required by a complementary technical instruction. This study contains a comparison of the plant against modern rules and standards not included in the licensing basis. When the plant design life is reached, the following is also required:

— A radiological impact report associated with LTO;
— Proposal of revision of the radioactive waste management plan;
— An integrated ageing management assessment plan;
— An FSAR supplement, including the analysis supporting LTO;
— A technical specifications revision, including changes required to support LTO.

The importance of the new regulatory study increases when LTO is involved. The selection of the rules and standards to be applied is made by the regulator. No cost–benefit considerations are made at this time. The process is shown in Fig. 20 and usually involves plant modifications. In the case of the SMG plant, this task has required almost the full dedication of the mechanical, electrical and I&C engineering support teams for a period of four years.

2.9.3. Scoping and screening method

The Spanish regulation requires managing the ageing of passive SSCs. Safety Instruction IS-22 requires that ageing management for passive long lived equipment meets 10 CFR 54. Active equipment is managed by the maintenance rule defined in Safety Instruction IS-15. The maintenance rule has been implemented in Spanish plants since 1999 and follows the requirements of 10 CFR 50.65.

Safety Instruction IS-22 also contains the requirement of an integrated ageing management assessment plan including TLAA at the time of an LTO application. The reference is 10 CFR 54 and NRC documents NUREG-1800 [13], NUREG-1801 [8] and the US industry guide NEI 95-10 (Rev. 6). This plan is accompanied by a proposal to supplement the FSAR, including the analysis that supports LTO and a proposal to revise the technical specifications to include the changes required for LTO. This documentation must be issued three years before the beginning of the operational permit entering LTO.

The scoping of passive components follows those defined in 10 CFR 54.4(a) and includes:

— Safety related SSCs;
— Non-safety-related SSCs whose failure could prevent the execution of a safety function;
— SSCs relied upon in safety analyses or plant evaluations to perform a function that demonstrates compliance with regulations for fire protection, environmental qualification, pressurized thermal shock, anticipated transients without scram or station blackout.

Life management of other equipment not subjected to the previous regulation is treated with different approaches depending on the plant. In some plants, life management activities are distributed throughout the whole organization, including: maintenance, operation, engineering and investment committees. Other plants have a dedicated organization for life management. The scope of life management programmes is typically based on reliability and economics.

2.9.4. Evaluation of SSCs

Passive SSCs are managed by 43 AMPs, while active SSCs are evaluated through the maintenance rule. The responsibility of each AMP rests with its owner, mostly selected among the maintenance and engineering staff. In turn, all AMP owners report to the AMP coordinator.

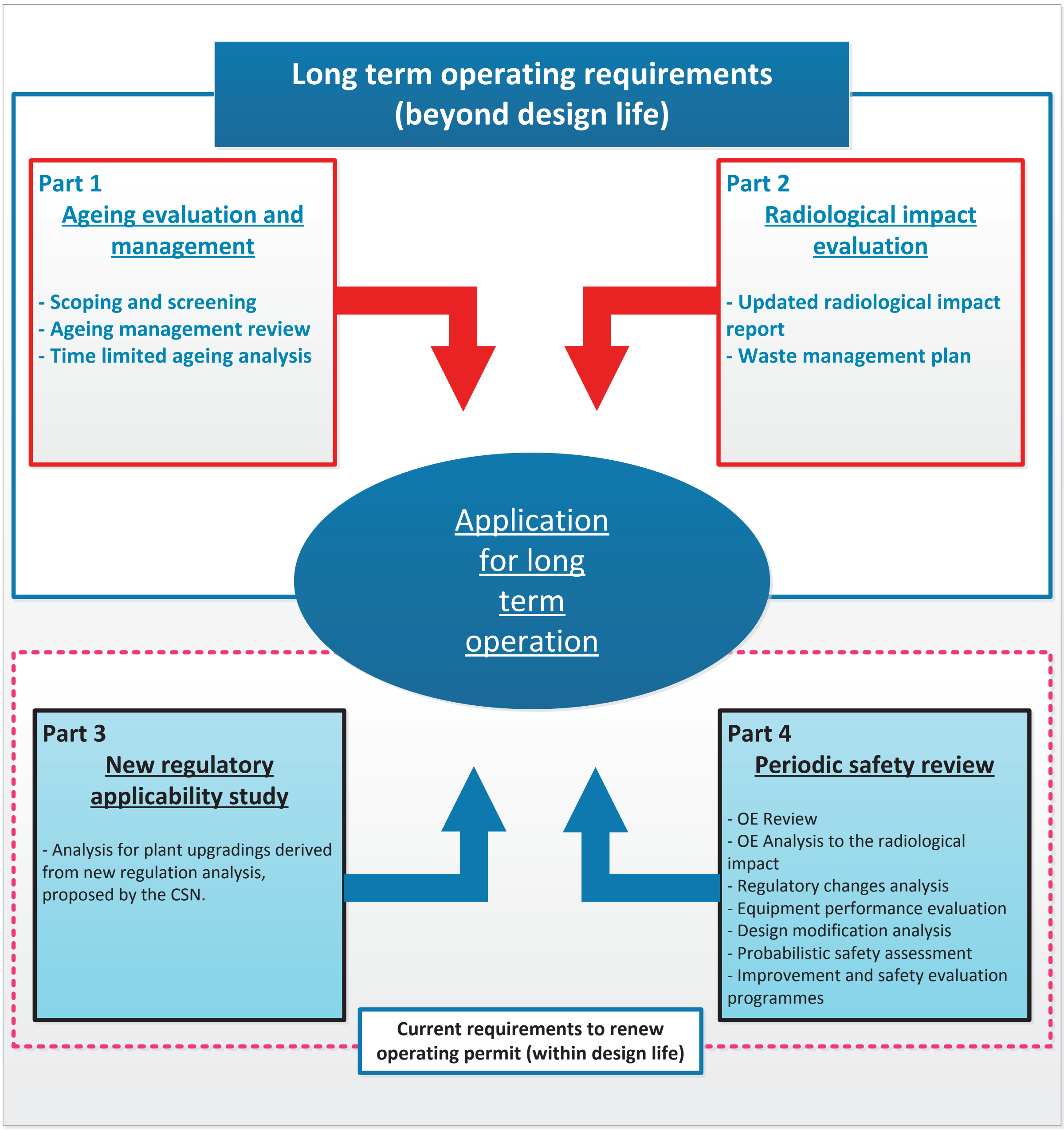


FIG. 20. LTO authorization requirements.

AMPs are defined in the integrated ageing management assessment plan; however, these are also living programmes continuously updated and improved through various assessments such as:

— A yearly ageing management report, which includes the corrective actions and activities necessary to control the ageing mechanisms. These reports are sent to the regulator.
— Health reports of the AMPs.
— Ad hoc assessment of scope changes due to physical modifications or changes to the licensing basis.

In addition, when an LTO application is submitted, the integrated ageing management assessment plan, which includes TLAA, is included. An evaluation of the plant design is performed in conformity with the new regulatory applicability study that includes a comparison of the plant configuration to the regulatory study findings beyond the licensing basis and a proposal of technical solutions to address deviations found in the comparison. As a result, two large modifications have been committed at the SMG plant to receive the approval for LTO. The first relates to NRC Regulatory Guide 1.75, Criteria for Independence of Electrical Safety Systems, in which a method to achieve independence of safety related equipment as per the Institute of Electrical and Electronic Engineers 384-1992 is suggested. As a result, SMG has committed to install new cables and trays for division A, separate equipment and cables inside the electrical panels and segregate non-safety-related equipment connected to 1E sources. This project requires complex work in the control room. It includes the installation of over 1000 I&C and 600 power cables. The second modification relates to ASME. In this respect, SMG has committed to install a new standby gas treatment system designed to the latest code of nuclear air and gas treatment. This requires a new emergency ventilation building, ducts and filters, among other things.

2.9.5. Feasibility study

Nuclenor, which owns and operates SMG, began its life management activities in the 1980s following a pilot project commissioned to US utilities in cooperation with the US Department of Energy and the Electric Power Research Institute regarding the feasibility of LTO for the Monticello NPP.

SMG was not the only Spanish plant interested in a PLiM implementation plan. In the 1990s, Spanish plants together developed a methodology for life management called the Spanish Electricity Industry Association (UNESA) methodology. Nuclenor adopted this methodology for its life management plan, and in 1987, it conducted its own feasibility study for LTO at SMG. As a result, the company made a strategic decision to establish an advanced PLiM programme in order to reach the end of the plant design life in 2011 in the best possible condition, and to stand the best chance of obtaining an LTO licence and continuing operation beyond 2011. To this end, a comprehensive life management plan was created aiming at reaching this objective, which became the central focus of Nuclenor's strategic plan.

To meet the company's strategic objective, many modernization activities were performed, of which the most important were:

— Replacement of the clean-up piping and related heat exchangers in 1996;
— New safety related cables in 1996;
— Installation of a safe shutdown remote panel in 1996;
— Repair of the core shroud in 1997;
— Extension of the fuel pool storage capacity in 1998;
— New ECCS suction pump filters in 1999;
— New condensate water filtering system in 1999;
— Replacement of the low pressure turbine rotors and casings during 1992–2001;
— Retubing of the main condenser during 1999–2001;
— Replacement of the reactor control rod indication and control system in 1999;
— New core spray tube inside the vessel in 2001;
— Replacement of the neutronic flow instrumentation in 2003;
— New control room habitability system in 2005;
— Addition of a new full scope simulator in 2005;
— Replacement of the DC batteries in 2007;
— Change of the buried safety piping to an above ground layout in 2009;
— Replacement of the low pressure cooling injection motors in 2011;
— Replacement of the main transformer in 2011.

Nuclenor applied for a new operating permit for the period 2009–2019, which meant entering into a full-fledged LTO. To do that, a project was set-up.

2.9.6. Document package structure

The documents submitted to the regulator in the SMG application for LTO were:

- An integrated ageing management assessment plan, including TLAA;
- An FSAR supplement proposal, including the analysis that supports LTO;
- A technical specifications revision proposal, which included the needed changes for LTO;
- A PSR report;
- A radiological impact report associated with the LTO;
- A radioactive waste management plan revision proposal;
- A new regulatory applicability study that included a comparison of the plant configuration with regulatory requirements not included in the licensing basis and a proposal of technical solutions for the deviations found in the comparison.

In addition, a yearly ageing management report was prepared during the LTO period to include the activities necessary to manage ageing mechanisms as required by the regulator.

SMG applied for an operating permit for the period 2009–2019. The documentation package was sent in June 2006 and a review was conducted in June 2008. In June 2009, a safety evaluation report in favour of an operating permit of ten years until 2019 was issued by the CSN. However, due to changes in the State's energy policy, the Ministry of Industry reduced the permit to four years, ending in 2013. By the end of 2011, general elections led to a change of government. As a result, it is expected that SMG may not be politically shut down, but may obtain a ten year operating permit until 2019, as CSN had decided in 2009.

2.10. THE UNITED STATES OF AMERICA'S APPROACH TO PLiM FOR LTO

2.10.1. Organizational structure

The organizational structure for PLiM in the United States of America 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 and operations) for implementation. Other plants may not have a dedicated PLiM organization, but rely on various organizations (e.g. design engineering, system engineering and maintenance) to conduct needed PLiM studies for LTO on a case by case (e.g. by component groups, such as piping, cables and transformers) or by individual components (turbine generator or steam generator).

The organizational structure for preparing an LRA varies from utility to utility, but a typical structure is based on creating a project team to address each major discipline of the needed studies. The team members may be from utility organizations, contracted consultants who specialize in preparing LRAs, or a combination of the two groups, which is most common. A typical organizational structure is shown in Fig. 21.

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.

2.10.2. Licensing requirements

Based on the US Atomic Energy Act of 1954, the NRC issues initial licences for commercial power reactors to operate for up to 40 years and allows these licences to be renewed for an additional 20 years with each renewal application. A 40 year licence term was selected on the basis of economic and antitrust considerations, not technical limitations. There is no limit on the number of licence renewals as long as the plant can continue to be run safely and in accordance with environmental requirements. The decision whether to seek licence renewal rests entirely with NPP owners, and typically is based on the plant's economic situation and whether the plant can continue to meet NRC requirements for continued safe operation.

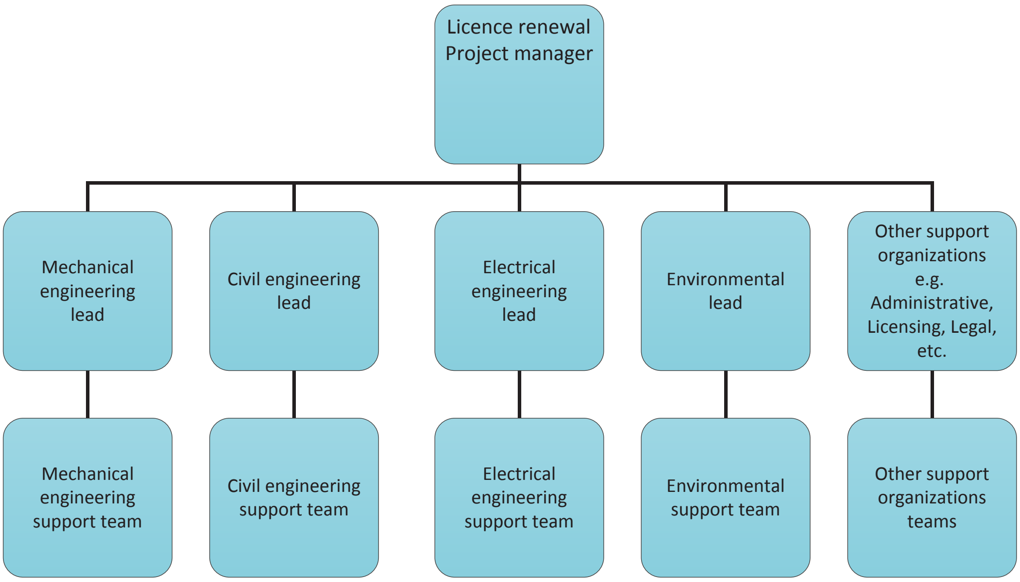


FIG. 21. Typical licence renewal project team organization chart.

An LRA may be submitted to the NRC after 20 years of operation and must be submitted prior to 35 years of operation (i.e. more than 5 years before the original 40 year licence expires). This 15 year window of opportunity was set by the NRC to allow a reasonable time for utilities to plan for long term power generation needs (i.e. most utilities require at least ten years to plan for alternative power supplies if licence renewal is unsuccessful), and the NRC mandated at least five years to perform a review of the LRA. If a utility submits an LRA more than 5 years before the original licence expires (called 'timely renewal application'), the NRC will allow continued operation beyond the 40 year term, if its review takes more than 5 years. 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 licence renewal process proceeds along two tracks — one for review of safety issues (10 CFR 54) and another for environmental issues (10 CFR 51). An applicant must provide the NRC with an evaluation that addresses the technical aspects of plant ageing and describes the ways those effects will be managed. It must also prepare an evaluation of the potential impact on the environment, assuming the plant operates for another 20 years. The NRC reviews the application and verifies evaluations through inspections and audits. The NRC review process is shown in Fig. 22.

The NRC provides regulatory guides and NUREG documents to define the regulatory process for an LRA. In addition, the Nuclear Energy Institute (NEI) has developed an industry guidance document, NEI 95-10 (Rev. 6), on how to prepare an LRA, that is endorsed by the NRC. The NRC regulations, guidance documents and background information are available to the public on the NRC web site.

The licence renewal applicant needs to conduct an integrated plant assessment, address needed changes to TLAA and conduct an environmental impact review. The IPA involves an ageing management review of SSCs within the scope of licence renewal, which are passive and long lived. The licence renewal rule credits the maintenance rule (10 CFR 50.65) for ageing management of active components so that an additional review of active components for licence renewal is not required.

2.10.3. Scoping and screening method

Scoping and screening for PLiM for LTO involves the entire plant and goes well beyond the NRC process for licence renewal for LTO. The PLiM process varies widely from plant to plant and is usually based on economic and technical evaluations done in accordance with life cycle management guidelines from the Electric Power Research Institute that are outside the scope of the regulatory review for LTO. Much of the work done for PLiM for LTO can be used for a licence renewal application, the NRC process for licence renewal is usually a subset of the PLiM activities (if a plant has a formal PLiM project).

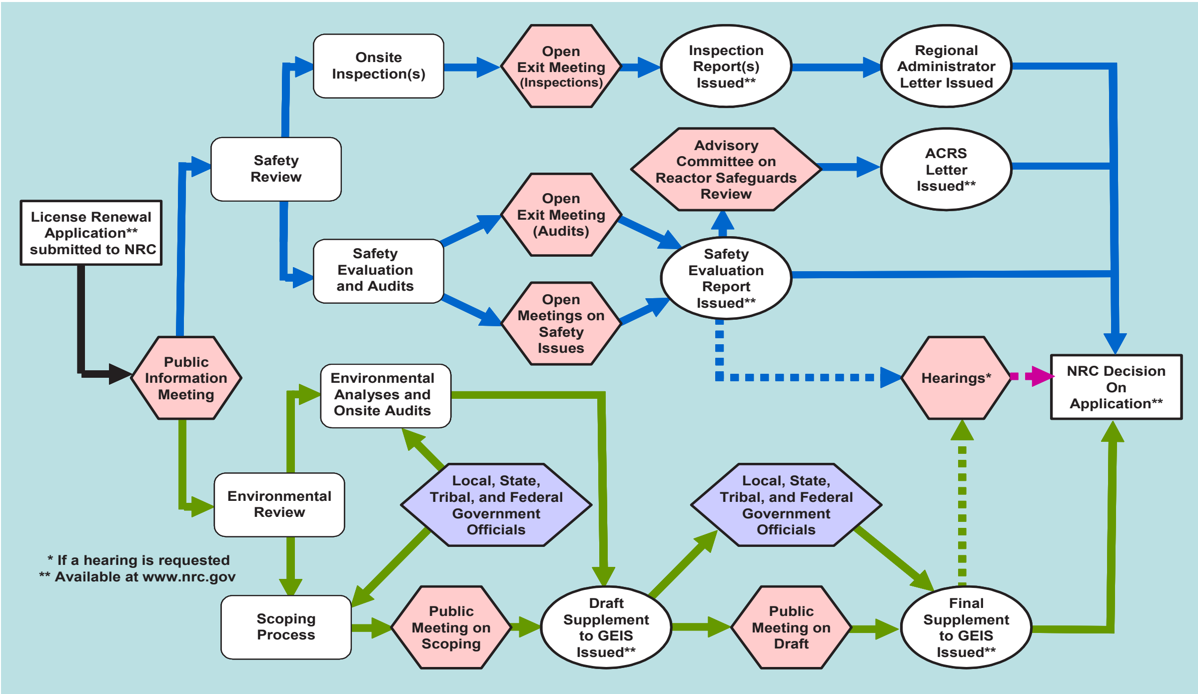


FIG. 22. NRC LRA review process (reproduced from NUREG-1850, Frequently Asked Questions on License Renewal of Nuclear Power Reactors).

Scoping for an LRA is very well defined in NRC regulations and in industry guidelines (e.g. NEI 95-10 (Rev. 6)). The major categories of scoping and screening include:

— Safety related SSCs that meet the criteria of 10 CFR 54.4(a)(1);
— Non-safety-related SSCs whose failure could prevent accomplishment of a safety function that meets the criteria of 10 CFR 54.4(a)(2);
— Other SSCs relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with NRC regulations for fire protection, environmental qualification, pressurized thermal shock, anticipated transients without scram or station blackout, that meet the criteria of 10 CFR 54.4(a)(3).

Although the licence renewal scope includes both active and passive SSCs, the screening process for licence renewal eliminates active or short lived SSCs from further ageing evaluation, since these components are already adequately covered by the maintenance rule (10 CFR 50.65) for ageing management or do not require ageing management since they are replaced prior to LTO (i.e. short lived SSCs).

2.10.4. Evaluation of SSCs

The evaluation of SSCs for PLiM also varies widely from plant to plant and may be either limited to major critical components or may be a comprehensive study of all plant SSCs. The major elements of a typical PLiM evaluation for LTO includes categorization of the SSCs based on their importance to safe and economical operation, potential for ageing effects to impact the intended function of the SSCs and identification of appropriate ageing management activities to ensure maintenance of the intended functions. 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 and operations) for implementation. Other plants may not have a dedicated PLiM organization, but rely on various organizations (e.g. design engineering, system engineering, maintenance and contracted consultants) to conduct needed PLiM studies for LTO on a case by case basis (e.g. by component groups or by individual components).

The evaluation of SSCs for an LRA is based on following the guidance in NEI 95-10 (Rev. 6). The evaluation includes an integrated plant assessment and a review of TLAA for SSCs within the scope of licence renewal. The integrated plant assessment consists of identifying the material and environment combinations for in-scope components and structures, and identifying the intended functions, applicable ageing effects that could result in a loss of intended function and the AMP needed to maintain the intended function.

The integrated plant assessment also includes a review of OE to ensure all applicable ageing effects have been identified and to ensure any existing AMPs credited with licence renewal are effective. If existing programmes are not determined to be effective, then new or modified AMPs need to be identified and implemented for LTO.

The evaluation of TLAAs involves identifying the plant specific TLAAs as defined in 10 CFR 54.3. Once the TLAAs (e.g. environmental qualification, fatigue analysis and neutron embrittlement analysis) are identified, they must be assessed to demonstrate that the analyses remain valid for the extended period of operation, that the analyses have been projected to the end of the period of extended operation, or that the effects of ageing will be adequately managed for the period of extended operation.

The results of the integrated plant assessment and TLAA evaluations are summarized in the LRA, in accordance with guidance in NEI 95-10 (Rev. 6) to support the regulatory process for licence renewal.

2.10.5. Feasibility study

A feasibility study to support a decision to seek LTO is primarily an economic analysis. However, it may also include political, environmental and other analyses based on local, state or federal circumstances. If the plant is well maintained and operating safely, economically and efficiently, then the economic analysis may be simplified. In most cases, US NPPs seeking licence renewal are the lowest cost, most reliable sources of electricity production compared with other large scale sources (e.g. fossil power plants). In addition, most NPPs are continually well maintained, so much so that major component upgrades or replacements are not needed to support LTO.

However, for NPPs that do face major component upgrades or replacements to allow LTO, the economic analysis will require more detail and effort. Some of the major inputs to the economic analysis include market price for electricity, regulated or competitive sale of electricity, ongoing operation and maintenance, cost of operation and expected cost of capital improvements (refurbishment or replacement of SSCs) needed for LTO, among other things. If the economic analysis or feasibility study supports LTO, then the plant owner can make a well informed decision regarding the option of licence renewal.

2.10.6. Document package structure

The document package structure for the LRA is defined by NUREG-1800 [13] and NEI 95-10 (Rev. 6), which is endorsed by NRC regulatory guidance. The structure for the LRA is as follows:

1. ADMINISTRATIVE INFORMATION

2. SCOPING AND SCREENING METHODOLOGY FOR INDENTIFYING STRUCTURES AND COMPONENTS SUBJECT TO AGEING MANAGEMENT REVIEW AND IMPLEMENTATION RESULTS
   - 2.1. Scoping and Screening Methodology
   - 2.2. Plant Level Scoping Results
   - 2.3. Scoping and Screening Results: Mechanical Systems
     - 2.3.1. Reactor Coolant System
     - 2.3.2. Engineered Safety Features
     - 2.3.3. Auxiliary Systems
     - 2.3.4. Steam and Power Conversion System
   - 2.4. Scoping and Screening Results: Structures
   - 2.5. Scoping and Screening Results: Electrical and Instrumentation and Controls Systems

3. AGEING MANAGEMENT REVIEW RESULTS
   3.1. Ageing Management of Reactor Vessel, Internals and Reactor Coolant System
   3.2. Ageing Management of Engineered Safety Features
   3.3. Ageing Management of Auxiliary Systems
   3.4. Ageing Management of Steam and Power Conversion System
   3.5. Ageing Management of Containments, Structures and Component Supports
   3.6. Ageing Management of Electrical and Instrumentation and Control

4. TIME-LIMITED AGEING ANALYSES
   4.1. Identification of TLAAs
   4.2. Reactor Vessel Neutron Embrittlement Analysis
   4.3. Metal Fatigue Analysis
   4.4. Environmental Qualification of Electrical Equipment
   4.5. Concrete Containment Tendon Prestress Analysis
   4.6. Containment Liner Plate, Metal Containments and Penetrations Fatigue Analysis
   4.7. Other Plant Specific TLAAs

APPENDICES
   A: FINAL SAFETY ANALYSIS REPORT SUPPLEMENT
   B: AGEING MANAGEMENT PROGRAMMES AND ACTIVITIES
   C: (OPTIONAL)
   D: TECHNICAL SPECIFICATION CHANGES
   E: ENVIRONMENTAL INFORMATION

The documents needed to support the studies and evaluations that are summarized in the LRA typically include 30–50 separate technical reports and calculations. The number of supporting documents is dependent on the quality of the nuclear plant configuration and current licensing basis documents. In some cases, it may be necessary to reconstitute missing, out of date, or difficult to retrieve current licensing basis documents as part of the LRA project. A typical list of supporting documents for an LRA is as follows:

— Scoping report;
— Mechanical system ageing management review reports (by system);
— Civil and structural ageing management review reports (by structures);
— Electrical and I&C ageing management review report;
— OE review report;
— Environmental review report;
— AMP review reports;
— TLAA reports.

3. IMPLEMENTATION OF PLiM FOR LTO

3.1. COMPARISON BETWEEN DIFFERENT APPROACHES TO LTO AND THE PSR FRAMEWORK

3.1.1. LTO in Canada compared with the PSR framework

A comparison between the CNSC Regulatory Document RD-360 [15], Life Extension of Nuclear Power Plants, from 2008, and SSG-25 [1] from 2013, shows many similarities and differences. The RD-360 could be applied to LTOs, refurbishments, restarts after a prolonged layup period and licence renewals. The Canadian ISR process is illustrated in Fig. 23. The general objectives of the Canadian RD-360 and PSR are the same and include:

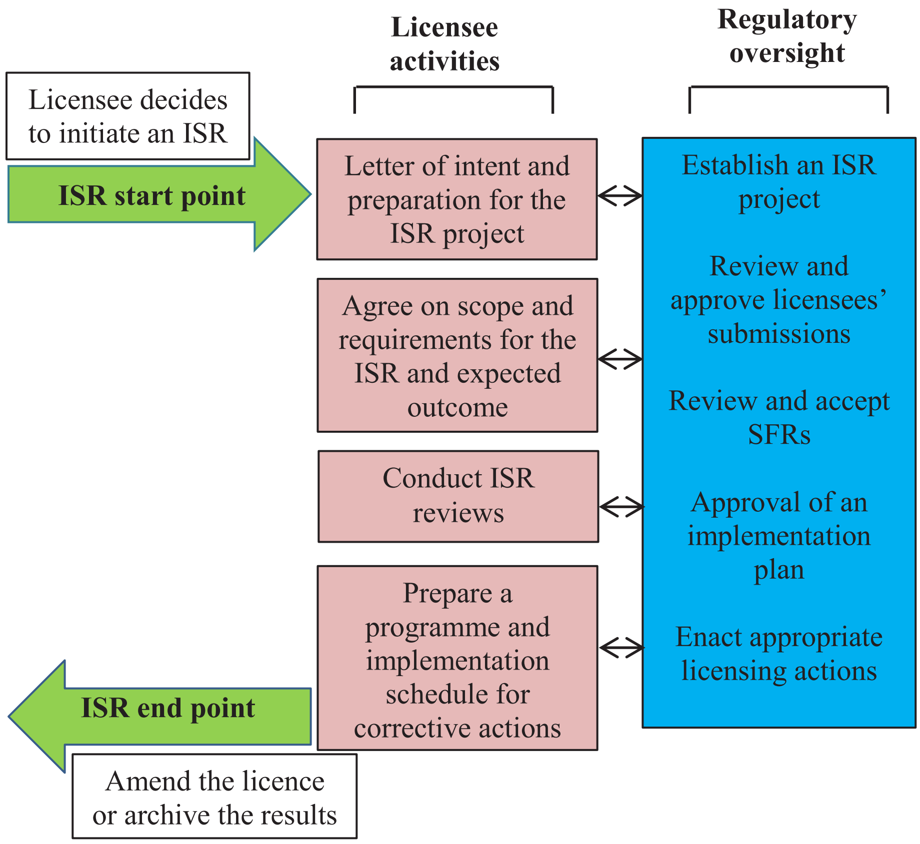


FIG. 23. The Canadian ISR process.

— Assessing the extent of conformity of the candidate plant to modern standards and practices;
— Assessing the extent of the continued validity of both design basis and licensing basis;
— Assessing the adequacy of current arrangements to maintain safety of the facility;
— Identifying improvements to be implemented to resolve identified safety issues and to increase robustness.

In the Canadian ISR process shown in Fig. 23, the licensee submits an ISR document with its LTO application. Contractors could be used to develop the ISR, or parts of it, but the owners remain responsible and should carefully coordinate and integrate their work within the overall plan. The ISR process is designed so that the documentation is self-contained and presented as a complete package.

The ISR requirements are in compliance with those contained in SSG-25 [1] in addition to those deriving from the CNSC's national mandate and regulatory goals. Table 5 shows the differences.

All differences between the Canadian LTO application and the PSR practice stem from typical national requirements, other than specific lessons learned from the refurbishment of Canadian reactors, and in part from the communication protocol between regulator and licensee in Canada. An important lesson learned in the refurbishment of Canadian reactors is that LTO planning should take place sufficiently in advance to minimize any accumulated negative impact from ISR findings at the time of its submission for regulatory approval. The CNSC requirement for an ISR methodology document may have been inspired by this lesson learned. In the ISR, the licensee explicitly declares the LTO project goals and objectives, the methods and the tools that will be used and the assumptions that will be made. This document constitutes a clear road map and a valuable reference tool for both the implementation and the oversight functions. Such a methodology document is not explicitly requested in SSG-25 [1]. Within the ISR process, the check on safety factors and margins is defined both conceptually and in detail, and is agreed upon between the regulator and the licensee before the ISR package is submitted. Finally, RD-360 [15] explicitly requires a final global assessment of the proposed LTO configuration to check the correct integration of all changes with the declared intent.

TABLE 5. COMPARISON BETWEEN RD-360 IN CANADA AND THE IAEA PSR PROCESS

RD-360 [15]	SSG-25 [1]
The ISR is performed one time for refurbishing a facility or extending its life (e.g. by 25–30 years)	PSR is performed every ten years
Would likely require an environmental assessment	Does not require an environmental assessment
Focus is on the condition of SSCs and fitness for service for the proposed period (25–30 years)	Focus is on the condition of SSCs and fitness for service for the next ten years, until the next PSR
Performed outside the licensing cycle, the ISR results are transferred to the licensing process	The PSR and its results are part of the licensing process
The CNSC expects 17 safety factors to be covered (added security, safeguards and quality management)	Calls for 14 safety factors
The resulting integrated improvement plan includes the results of both the ISR and environmental assessment and factors such as station specific action items and generic action items, among other things	The resulting integrated improvement programme includes the results of the PSR
Detailed process, but no details on safety factors objectives, description and elements	Detailed process and safety factors, objectives, description and elements

3.1.2. LTO in the Czech Republic compared with the PSR framework

A modernization programme called MORAVA has been implemented at the Dukovany NPP since 1998. Under this programme, favourable conditions for operation beyond the original design lifetime have been established. The Dukovany NPP started preparation of its LTO programme in 2004. It consisted of the following topics:

— Theses;
— Feasibility study of the Dukovany NPP LTO;
— Assessment of the safety aspects of the Dukovany NPP LTO;
— Long term modernization plan;
— Summary report;
— Appendices.

A risk analysis was conducted separately. On the basis of these documents, the Dukovany NPP LTO strategy was developed.

In January 2009, the LTO programme and the Dukovany NPP LTO strategy were approved by the České Energetické Závody (ČEZ) Board of Directors. Operation for 60 years is considered in the programme (until 2045 for Unit 1). The first step in the LTO programme implementation, the LTO preparation project (for safe operation beyond 2015), was approved by the ČEZ Board of Directors at the same time The project contains safety improvements (resulting from quality assurance, PSR and PSA, among other things), calculations and analyses necessary for operation until 2025, but also allows further operational extensions until 2045. A report entitled The Safety Case for LTO of Dukovany NPP Unit 1 was prepared for Unit 1 operational permission renewal. It will serve as the basic document demonstrating readiness for the LTO period.

According to Czech law, a licence to operate an NPP is unlimited in the Czech Republic, but it is necessary to obtain operational permit renewals every ten years for each unit independently, as shown in Fig. 24. One of the SONS preconditions for NPP operational permits is a successful PSR outcome and demonstration that the analysis results and the action plan to implement any necessary remedial measures is sound (see also Fig. 1). The same procedure will be also used for Units 2–4.

A comparison has been made between the Czech Republic licensing practice for LTO and the PSR process by listing the features of the Czech LTO project (2009–2015) and SSG-25 [1], as shown in Table 6.

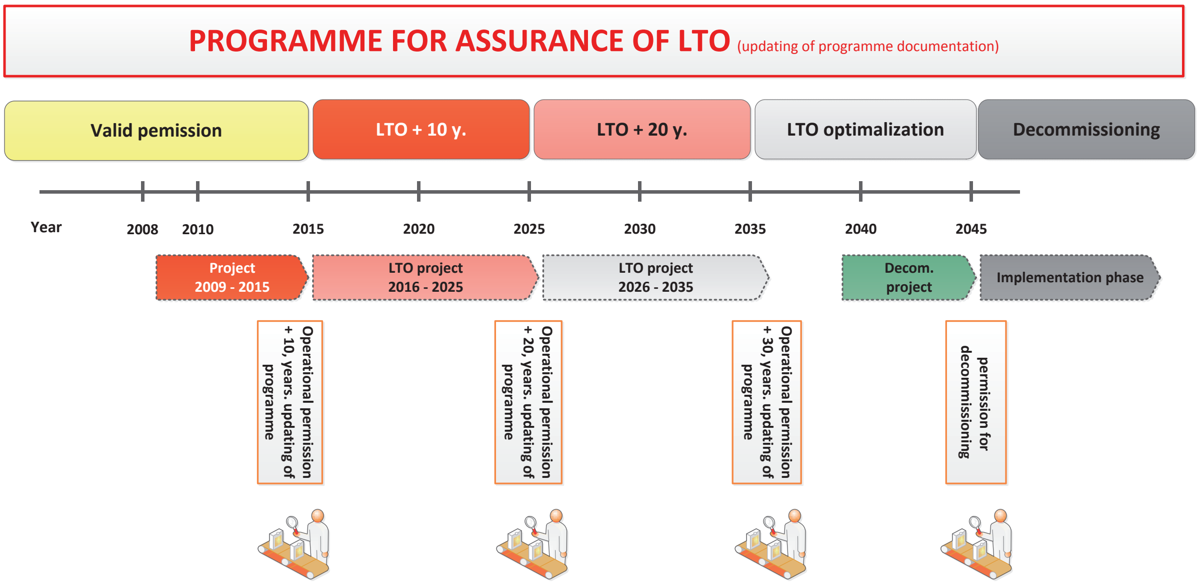


FIG. 24. Operational permit renewal diagram for Dukovany Unit 1 in the Czech Republic.

TABLE 6. COMPARISON BETWEEN THE CZECH LTO FEATURES AND THE IAEA PSR REQUIREMENTS

LTO programme and LTO project	SSG-25 [1]
LTO programme is a living document LTO project is reviewed at each of the ten year PSR cycles	PSR is performed every ten years
Does not require an environmental assessment	Does not require an environmental assessment
Focus is on the condition of SSCs and fitness for service for the proposed LTO period (up to 60 years)	Focus is on SSCs conditions and fitness for the service for next ten years until the next PSR
Documents the safety case (results of the LTO project) and is a fundamental part of the operational permit renewal process (PSR is part of the safety case)	PSR and its results are part of the operational permit renewal process
Evaluates the correctness of the PSR implementation process	14 PSR safety factors are assessed
The LTO project contains all remedial measures coming from the PSR, IAEA safety issues and SALTO issues, among other things	Remedial measures are defined and progress is monitored annually
Progress is reported annually to the regulatory committee (including PSR)	Progress is reported annually to the regulatory committee as a part of the LTO report

3.1.3. LTO in France compared with the PSR framework

The LTO of the French NPP fleet beyond 40 years (the initial design basis for mechanical structures) is one of the major objectives for EDF. As most of the EDF NPPs were built and connected to the grid in the 1980s, a lifetime limited to 40 years would lead to an important investment programme over the next 20 years for new plants (nuclear units or others, such as gas, coal or renewable) starting as early as 2017 to compensate for the shutdown of the oldest NPPs. Operating nuclear units for 10 or 20 additional years will allow EDF to smooth the commissioning flow for the new build programme (2020–2050), which represents an industrial and economic advantage.

Unlike US regulations, no limited licensing lifetime has been defined, even though a design basis of 40 years has been initially taken into account to justify the structural integrity of the major mechanical components. As a consequence, PSR as well as in-depth inspections of systems, structures and equipment, have to be performed every ten years for every unit in order to check compliance with all the applicable nuclear safety requirements (nuclear safety references). These nuclear safety references are up to date references including operating feedback experience (national and international), taking into account the best international practices and the most recent nuclear safety standards, such as the reference levels for existing plants from the Western European Nuclear Regulators Association. Therefore, PSRs are based on a continual nuclear safety improvement process.

3.1.4. LTO in Hungary compared with the PSR framework

As in many European States, PSRs are an important element of Hungarian regulations. Although PSR is not a licensing tool, limitations and conditions for the operational licence, including safety upgrades, may be defined as a result of a PSR. Although the scope of the PSR and the content of the PSR are very close to those recommended in SSG-25 [1], the main focus of the PSR in Hungary is ageing, and the assessment of changes and tendencies in support of LTO.

The Paks NPP's first PSR was carried out between 1995 and 1996 for Units 1 and 2, and between 1997 and 1999 for Units 3 and 4. It was the first systematic assessment programme for overall plant safety. The review provided a basis for the renewal of the permanent operating licence at that time. The critical findings of the first PSR were the urgent need for a systematic design basis reconstitution programme, the initiation of a systematic ageing management and equipment qualification programme, and a comprehensive list of safety upgrades. The PSR provided a substantial contribution to the safety level of the Paks NPP.

The second PSR was performed between 2006 and 2008 for all four units, simultaneously. The review assessed overall plant safety in the context of LTO, such as ageing management, the level of R&D support, knowledge of science and technology, the development of safety analysis methods, new evidence related to hazards and their impact on plant safety and the evaluation and feedback of operational and other experiences, among other things.

In addition, PSR provided crucial technical information applicable to the development of the LTO programme. Simultaneous execution of the PSR and of the project in support of the LRA for LTO provided positive, synergetic effects, given the commonality of goals and overlapping of scopes between PSR and licence renewal work. The benefits drawn from this kind of synergy compensated for the heavier workload and doubling of effort carried out in the plant.

The Paks NPP considers LTO a strategic programme that involves and develops all areas of activity in the operation of a plant, for example, human resource management and knowledge management, among other things.

3.1.5. LTO in India compared with the PSR framework

There are 17 units currently operating in India: 2 BWRs and 15 PHWRs. In addition, there are six units under construction: three PHWRs (220 MW(e) each); two WWERs (1000 MW(e) each); and one Prototype Fast Breeder Reactor (500 MW(e)). The total installed capacity includes 4120 MW(e) in operation and 3120 MW(e) under construction. Safety reviews are performed in India during the course of the NPP's normal service life. Inspections extend to periodic performance reports, event reports, safety related engineering modifications, planned maintenance outages, radiological safety performance, radioactive waste disposal history and practice evolution, and the outcome of regulatory inspections. Figure 25 shows the safety review process applied in India for operating NPP units. The regulatory body in India, AERB, appoints a safety review committee for operating plants (SARCOP) made up of expert groups and a unit specific review committee. The expert groups include reactor physics specialists, reactor chemistry specialists, instrumentation and control experts, coolant channel safety experts, equipment qualification experts and ageing management experts.

The AERB collects the inputs and performs a safety review of operating plants. This provides continuity of action, coordinates all experts and supports the law to enforce the regulations and the licensing requirements and any additional safety requirements. The AERB is responsible for overseeing the follow-up and implementation of all action items issued by a safety review. Finally, it is responsible for verifications and inspections.

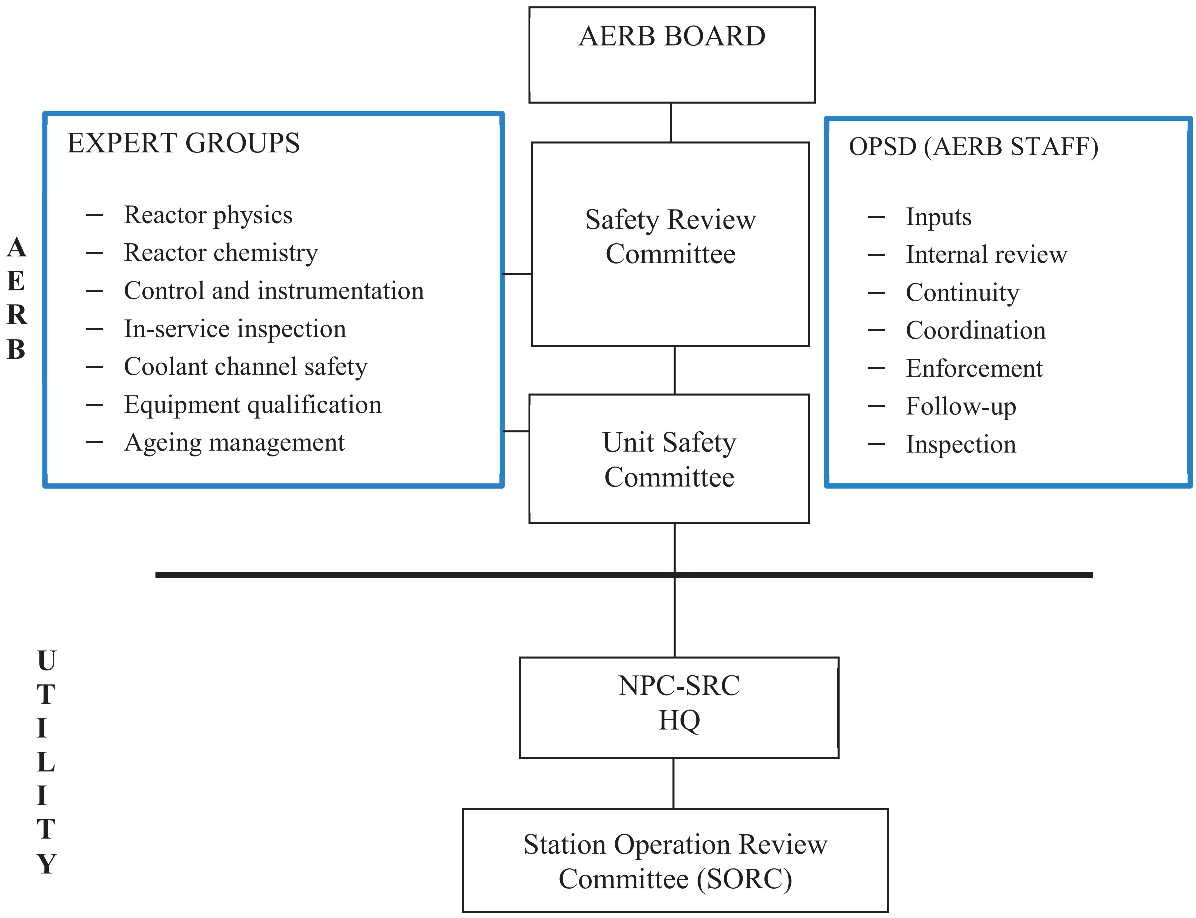


FIG. 25. The safety review process for operating NPPs in India. OPSD: Operating Plants Safety Division; NPC: Nuclear Power Committee; SRC: Safety Review Committee.

The PSR process was not established in India from the beginning. It underwent an evolution beginning in 1993 when an authorization for operation process was instituted with a validity of only five years. At that time, a safety assessment report on the operating unit was performed in order to obtain a renewal of the authorization for operation. The document produced was called the Safety Assessment Report for the Renewal of Authorization (SARRA). This process was applied to the Tarapur Atomic Power Station, RAPS 1 and 2 and NAPS units. In retrospect, it is obvious that, at that time, issues such as ageing management, equipment qualification and comparisons with current standards were not adequately addressed.

AERB Safety Guide AERB/SG/O-12, Renewal of Authorisation for Operation of Nuclear Power Plants was published in 2000. It established the necessary principle of periodicity of safety inspections. For all intents and purposes, this meant the establishment of a PSR process that was to be conducted every ten years for NPPs of standard designs. For new designs, the first PSR is to be performed after five years of operation.

Today, in India, a PSR is a prerequisite for reauthorization of a licence to operate an NPP. Table 7 illustrates the renewal authorization, the PSR submission requirements and the submission timeline.

3.1.6. LTO in China compared with the PSR framework

The first NPP unit in China, Qinshan-1, with a 30 year design life, has operated since 1991. Although neither the owner or operator nor the NNSA have decided whether an LTO programme should be carried out for Qinshan-1 after it reaches the end of its design life, feasibility studies are being conducted in parallel by both sides. The design life for the second NPP unit in China, Daya Bay 1, which started operation in the early 1990s, is 40 years. The LTO for Qinshan-1, which started in 2010, is the only LTO activity in China. Thus, the LTO related activities for Quinshan-1 represent a pilot project for the Chinese LTO process development.

TABLE 7. COMPARISON BETWEEN ARA AND PSR

	ARA	PSR
Submission	Three months before authorization expiry	Six months before expiry of authorization
Factors	Safety performance Equipment reliability OE feedback ISI status and major work completed Public concerns	Actual physical condition Safety analysis Equipment qualification Ageing management Safety performance OE feedback Procedures Organization and administration Human factors Emergency planning Environmental impact

From the NNSA side, the study will focus on the development of a suitable licence renewal procedure and a safety assessment methodology. As a result, a series of regulatory requirements and guidelines are being developed as a template for LTO applications. The PSR policy has been adopted in China as a means to support the licence of operational NPPs, and it can be expected that the LTO review procedure may utilize as much as possible of the PSR conclusions in combination with some elements of the licence renewal procedures.

On the utility side, the study will focus on both safety and economy. From the safety point of view, the integrated plant assessment and current licensing basis processes will be carefully studied. The TLAA will certainly be carried out for critical SSCs with significant ageing mechanisms, also taking into account affects related to ageing. From the economic viewpoint, heavy component replacement evaluations and power uprate studies could also be performed.

3.1.7. LTO in the Republic of Korea compared with the PSR framework

All operating NPPs in the Republic of Korea are required to conduct a PSR every ten years during the plant’s design life. Even if the PSR concludes that the plant is in compliance with the safety standards in effect, that does not mean that the plant can continue to operate beyond its design life because the regulations associated with the PSR do not apply to continued operation beyond design. It is unlikely that a routine PSR could identify safety shortcomings and significant increases in risk or reduction of safety by using the licensing basis applicable to the plant design life. Most issues identified through a PSR are categorized as recommendations rather than compulsory corrective actions. In order to identify safety improvement items, the more recent safety standards and practices can be used. The licensee is then required to submit an implementation plan for all improvements considered feasible. The regulator should monitor the execution of the implementation plan even in the event that no immediate compulsory corrective action is needed.

Since the most important factor in a PSR is ageing assessment, the PSR process is only regarded as supporting material in a more extensive application for continued operation beyond the plant design span. However, since the PSR is a key regulatory instrument for maintaining the safety of plant operation, an application for continued operation beyond the plant design term also remains stipulated under the legal framework of the PSR process, and rule making is used to supplement the PSR framework with specific LTO requirements. Continued operation can be seen, therefore, as a PSR extension in that two more rigorous safety assessments are added: an AMP that includes TLAA, and an assessment of the radiological impact on the environment for the long operating term.

In summary, the safety review of an LTO application is still conducted under the regular PSR framework, with additional safety requirements more specific to LTO safety and to the environmental impact of the plant on the territory. Figure 26 illustrates how PSR is extended to LTO.

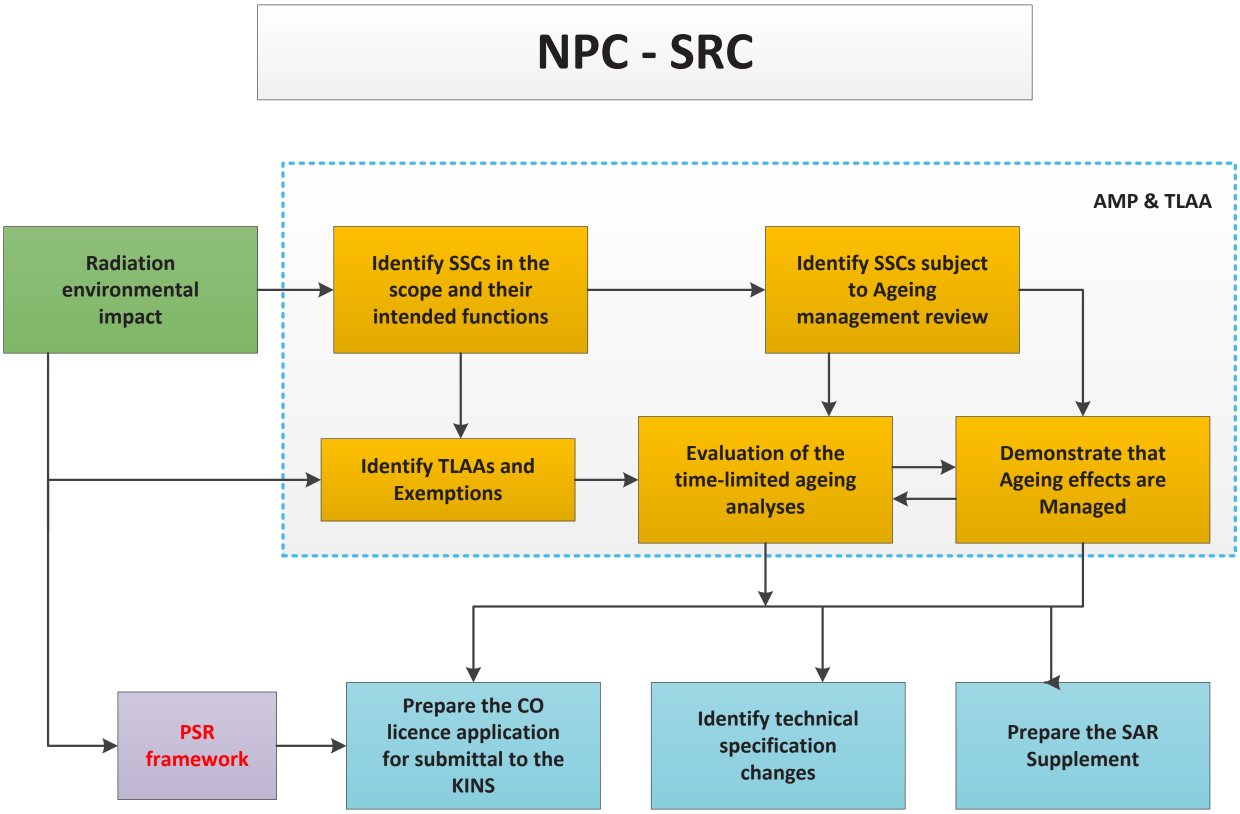


FIG. 26. Extended PSR for continuous operation. NPC: Nuclear Power Committee; SRC: Safety Review Committee; CO: continuous operation; KINS: Korea Institute of Nuclear Safety.

3.1.8. LTO in the Russian Federation compared with the PSR framework

Structures, systems and components in Russian NPPs are subjected to a variety of mechanical, physical and chemical conditions during operation. The proven NPP practices of monitoring, maintaining and operating are the primary means used for managing ageing.

Rosenergoatom is the operating company managing the generation of electrical and thermal energy at all NPPs in the Russian Federation. All Russian NPPs are affiliated branches of the operating company. However, NPPs are directly responsible for their own operation and for the work they perform on life extension of Russian NPPs; all NPPs are required to follow the unified norms and rules of the Russian nuclear industry.

A specialized department of modernization and lifetime extension exists in the organizational structure of each NPP. These departments fulfil a coordination role in the PLEX work and routinely control their execution. A special commission oversees the assessment of the condition and residual lifetime of SSCs performed by the modernization and lifetime extension departments.

The main document governing PLEX work at Russian NPPs is NP-017-2000, Basic Requirements for Power Unit Lifetime Extension of Nuclear Power Plant[s], which is part of the federal norms and rules. This document requires that before an NPP unit reaches its design service life, the operating company is to perform a comprehensive safety and economic study, based on which it may decide whether to prolong its service life or decommission it. The methods used in the safety assessments should be conservative to compensate for uncertainties in the input data. In addition, the software used should be certified. The design service life of Russian NPP units of the first generation is set at 30 years. Operation beyond this term is granted in accordance with the following technical and economic factors:

- Demonstration of the capability to uphold safety margins throughout the proposed LTO period beyond the unit's design service term;
- Existence of sufficient residual life for non-replaceable essential equipment;

- Availability of sufficient temporary storage space for the additional amount of spent fuel or existence of a plan meeting all requirements for its safe transport outside the NPP site;
- Existence of a safety insurance plan for the handling of radioactive waste produced during the additional period of operation;
- Existence of a safety insurance plan and environmental impact statement for unit decommissioning.

The operating organization sends the final results of the PLEX study to Rostechnadzor, with the application for an operating licence of the NPP unit for an LTO period beyond its design life. The list and contents of the necessary documents to be submitted are prescribed in Rostechnadzor's regulatory document, RD-04-02-2006, Document List and Contents in Support of the Safe Operation of an NPP Unit Beyond its Design Service Life.

To demonstrate the safety of LTO in the Russian Federation, a deterministic approach was selected. The essence of this approach consists of the following elements:

- An analysis of the past operation during the original 30 year design term;
- A comprehensive assessment of the condition at the end of the 30 year operating period based on a theoretical estimate supported by experimental confirmation;
- An ageing prognosis and safety margin estimate for the operating period beyond design.

Substantiation of LTO in the Russian Federation is based on the PSR approach, which allows determination of the actual SSC conditions. At this time, the service life extension for Russian reactors varies from 15 to 30 years.

Experience with life extension showed the usefulness of a number of improvements to existing LTO procedures. Currently, an extension period of 30 years with a one time complex SSC survey (examination) at the end of the design life is proving to be very long. It is advisable to aim for shorter periods of time, such as ten years, as is generally practiced in Europe.

The reduction of the first LTO period will allow the operator to provide more objective information about the general ageing of the main equipment, the current condition of the equipment, along with the results of the regular periodic monitoring. This will also allow the introduction of more effective modern survey and new monitoring methods, techniques and hardware and the creation of more objective databases, and an increased safety level through additional independent controls (inspections) of the most problematic areas. It will also allow more effective and customized recommendations regarding modernization on the basis of a deeper understanding of the ageing mechanisms and the capability to meet new regulatory and safety requirements.

In addition to life management programmes and the technical and economic estimates supporting design life extension (30 years), a number of specific preconditions are imposed in the Russian Federation. They include:

- Demonstration that the unit will maintain its required safety margins at the expected capacity factor during its LTO period;
- An optimization plan of operating conditions and an associated list of repairs, design changes and their implementation schedule;
- The capability to detect and control the consequences of ageing during the operation period beyond the unit's design service life.

In the Russian Federation, the inspection requirements of the safety functions and safety margins of the essential passive and active SSCs include the following capabilities during the LTO period:

- Accurate prognosis of the ageing of essential equipment during the LTO period;
- Monitoring of the consequences of ageing relative to the service loads on the equipment;
- Compensatory measures to mitigate and optimize the consequences of ageing, where necessary;
- Improvement of the programme taking into account new available knowledge and R&D indications;
- Estimate of the programme efficiency.

3.1.9. LTO in Spain compared with the PSR framework

An ageing management review was carried out for the LTO application in 2009 for SMG. It was performed as a dedicated project by a specific team. As a result, new activities were generated and distributed throughout the plant organization. To maintain the quality of the AMP, two positions were created: the programme owner and the ageing management coordinator.

A general procedure that applies to the whole company was prepared to detail their job descriptions, their responsibilities and the assessment activities pertaining to ageing management. Periodic assessments are carried out to keep the AMP up to date. Every two years, programme owners evaluate their programmes and the AMP coordinator reviews their evaluations.

3.1.10. LTO in the United States of America compared with the PSR framework

The practice of assessing the safety of operating NPPs through the use of PSRs is well established among Member States. The objective of a PSR, as stated in SSG-25, is to ensure a high level of safety throughout the plant's operating life by systematically assessing the cumulative effects of plant ageing, plant modifications, OE, technical development and siting aspects.

In the United States of America, there is agreement with the IAEA premise that vigilant oversight and ongoing reviews are essential to ensuring safety throughout the life of the plants. Historically, it has engaged the international community through the development of the process to conduct PSRs to ensure the objective of maintaining safety throughout the entire operating life of a plant. However, in the United States of America, this objective is accomplished through the comprehensive set of NRC regulations, inspections and safety review programmes, rather than by using the PSR approach.

The US regulatory structure was well established when the PSR approach was being developed. During the formulation of the licence renewal rule in the early 1990s, the NRC specifically considered the concept of performing a comprehensive review of a plant to bring it closer to the current standards (a goal of the PSR approach). The NRC did not adopt this approach in part because it believed that the robust and mature NRC programmes, including the on-site resident inspector programme, generic issue identification and systematic evaluation process, afforded adequate protection to the public. The NRC has reviewed the PSR and has maintained that the safety functions of PSRs are achieved by the US system. The Appendix presents an overview of the US regulatory structure, salient features of the US regulations consistent with the PSR approach and a comparison between the safety factors in the PSR Safety Guide and US activities.

Following issuance of the initial operating licence and during the period of extended operation, the NRC continues providing oversight of plant operations to verify that they are being conducted in accordance with NRC regulations. The oversight includes daily monitoring by the on-site resident inspectors and periodic regional inspections, OE evaluations, generic issue resolution, biennial updates of the licensing basis and imposition of new requirements.

Current NRC policies and programmes have five main strengths that make them comparable to the PSR process. First, the NRC regulatory process emphasizes ongoing technical evaluation and oversight of plant operations. Because the design basis evolves during the entire licence period, a continuing oversight process ensures facility safety throughout the life of the plant. Annually, the NRC devotes significant resources to the oversight process at each plant. Through the use of resident inspectors, who provide daily inspections, and regional specialists, each plant receives 6000 to 10 000 hours of inspection. Focused in-depth inspection teams are routinely scheduled to evaluate the safety of licensees' designs and operations. For example, the NRC spent approximately 20 000 staff hours conducting component design basis inspections at 24 facilities in 2009. The purpose of component design basis inspections is to verify the initial design and subsequent modifications, and provide monitoring of the capability of the selected components and operator actions to perform their design bases functions. Additionally, over 1200 hours are spent evaluating licensing tasks at each plant. This level of effort gives the NRC the confidence that its oversight process produces a level of safety comparable to that afforded by the PSR process.

Second, the NRC regulatory oversight programme is comprehensive. It encompasses a wide spectrum of programmatic activities ranging from initial licensing and inspection to cross-cutting safety culture issues that incorporate all of the safety factors evaluated in the PSR process. To assess whether there were any significant gaps between the PSR process and the US regulatory oversight process, NRC staff from several divisions of the

Office of Nuclear Reactor Regulation and other programme offices studied the IAEA PSR safety factors to verify that the US programme elements accomplish the associated functions. Table 8 includes a complete comparison between the PSR safety factors and global assessment to US programme elements. As indicated in Table 8 there are no significant gaps between the IAEA PSR safety factors and the US programme.

TABLE 8. CROSS-SECTION OF SELECTED PSR SAFETY FACTORS IN US PROGRAMMES

IAEA SSG-25 [1] safety factors	US programme elements
Actual condition of SSCs	In-depth daily inspections by resident inspectors Focused routine inspections by specialists (e.g. maintenance rule, corrective action programme) Reactor oversight process performance indicators
Equipment qualification	Inspections tied to environmental qualification rules (10 CFR 50.49) Component design basis inspections Permanent plant modifications Licence event reports
Ageing	AMPs (10 CFR 54) Licence renewal inspections (passive components) Maintenance rule and other reactor oversight process inspections (active components)
Deterministic safety analysis	Evaluation of changes to the design and licensing basis Changes to the FSAR (10 CFR 50.59) Daily inspections that compare everyday operation to design bases
Probabilistic safety assessment	PSAs used in selecting inspection samples Plant specific PSAs for internal and some external events PSAs can be used in lieu of deterministic assessments (RG 1.174) Maintenance rule (10 CFR 50.65) PSAs not required to be updated in the USA

Third, the NRC reviewed several international PSR related documents to confirm that the outcomes from performing PSRs and conducting the NRC regulatory programmes are similar. Because the actual PSRs submitted to the regulatory authorities were not readily available to the NRC for review, the agency reviewed a number of international regulators' PSR evaluations and a PSR summary report submitted by an international plant. A high level comparison suggested all findings and other recommendations reviewed are in areas that have received similar regulatory attention through the ongoing NRC regulatory process.

Fourth, the US commercial nuclear utilities regularly assess the safety performance of their NPPs. Following the 1979 Three Mile Island accident, the US nuclear power industry formed INPO to promote safe and reliable operation of NPPs. INPO conducts biennial independent assessments at all member stations using a multidisciplinary team of INPO employees and independent industry peers with supervisory or technical expertise in the areas they are assessing. INPO assesses the plants in the following areas:

— Operations;
— Maintenance;
— Work management;
— Configuration management;
— Design engineering;
— Equipment reliability;
— Radiological protection;
— Chemistry;
— Training;

— Organizational effectiveness;
— Safety culture.

During the assessments, the evaluation team observes operations, analyses processes and observes personnel. The assessments are preceded by a three week preparation process in which the team members review critical data from plant operation (e.g. corrective action information, plant performance data and self-assessments) collected since the last assessment. The evaluation team uses detailed performance objectives and criteria for each area being assessed. The team briefs the plant senior management on the output of the assessment, which consists of area performance summaries and areas for improvement. Staff of the NRC routinely review these reports as an independent check to ensure that NRC processes are capturing similar performance insights.

Finally, owners groups and equipment vendors have long played the role of providing unified industry approaches to generic nuclear regulatory and technical issues, and coordinating interactions with the NRC. The Boiling Water Reactor Owners Group and the Pressurized Water Reactor Owners Group were formed to share industry OE. At the request of the owners groups, the NRC may comment and review the owners groups' topic reports. The NRC regularly meets with these owners groups to stay abreast of the existing and emerging plant safety issues of mutual interest.

In summary, the objectives of the PSR process are well served by the current US regulatory process. The NRC agrees with three main goals of the PSR process:

(a) To confirm that the plant is as safe as originally intended;
(b) To determine whether there are any SSCs that could limit the life of the plant in the foreseeable future;
(c) To compare the plant against modern safety standards and identify where improvements would be beneficial at justifiable cost.

The NRC manages ageing through insights gained from inspections of active components (e.g. maintenance and corrective action inspections and follow-up generic actions) and passive components through the formal licence renewal process, which establishes comprehensive ageing programmes for passive long lived components (e.g. reactor vessels, cabling and buried piping). These processes are informed by a robust OE programme that screens both domestic and international experience for insights that can be used to improve plant performance.

The US system also requires that plants upgrade to more modern safety standards on an ongoing basis through new regulations and orders that impose new requirements. Similar to what is accomplished through the PSR process, the NRC evaluates these changes for safety benefit before requiring implementation. Although the US system does not require its licensees to summarize performance with, for example, a recurring ten year submission to the regulator, the US Government believes that its day to day focus on inspection and assessment ensures that these improvements are evaluated year to year. For example, the NRC recently reviewed several international PSR evaluations. Issues identified and documented in these ten year reviews appear to be very similar to those identified, documented and evaluated annually in the inspection, licensing and generic actions under the US process.

The NRC regards the process of standing back and performing a holistic in-depth evaluation of each plant at a regular interval to be beneficial. It follows this practice on a shorter interval, evaluating OE, considering upgrades and performing assessments annually. It then uses its formal licence renewal process to further evaluate extending the licence expiration date for LTO.

3.2. IMPLICATIONS OF OPERATING EXPERIENCE AND LESSONS LEARNED

3.2.1. Operating experience and lessons learned in Canada

The decision to continue operation in Canada depends ultimately on economic factors. This decision is made for each unit independently, even if the unit is part of a multiunit station. An effective PLiM implementation programme plays a pivotal role in LTO decisions because it is capable of integrating ageing management and economic planning. The use of an economic model allows PLiM to evaluate implementation alternatives, such as optimizing the staffing curve to shorten the LTO outage, helping to decide on capital upgrades if it can be shown they can increase capacity factors or the maximum continuous rating. Plant life management can also help select the optimum conditions and the best time for an extended LTO outage.

Ultimately, it is important to find the point where the operating and capital costs of changes are no longer financially viable, taking into account risks and uncertainties (such as the costs of resolving licensing issues, major refurbishments and future electricity prices). This can be done via an economic sensitivity analysis.

Before large unit refurbishments and modernizations are conducted, three changes should take place in the management of the plant, for ageing in particular, if the refurbishment programme is to be efficiently run and to be ultimately successful:

(a) Condition based decision making implemented using advanced maintenance information and monitoring techniques, specifically health monitoring information.
(b) Control systems in place whereby timely flow of corrective action information goes to key decision makers, for example providing an electronic portal to the maintenance review team.
(c) Effective use of age related information. Decisions are optimized and linked to the work management system.

Running metrics on these parameters measures how well PLiM is being implemented and allows fine tuning and continuing improvements leading to refurbishment. Ideally, a PLiM team should be involved in LTO support to, among other things, perform a systematic screening of all SSCs through a request for relevant inputs from inaugural inspections and historical performance records from chemistry control programmes and from technology watch programmes, which provide obsolescence projections and warnings. The analysis of this data allows trustworthy and defensible prognoses on the SSCs. Technical results are then consolidated into an economic model before results are finalized and recommendations on the plant ageing mitigation programmes are issued (i.e. recommendations on improvements to the current maintenance, surveillance and inspection or on targeted changes to the configuration). Finally, benchmarking should also be used with similar programmes in other areas. During implementation of unit refurbishment and modernization, the PLiM team should continue to be available to support the engineering and planning groups in the preparation of individual SSC life cycle plans.

3.2.1.1. Refurbishment of CANDU reactors for long term operation

Design changes to support LTO may be carried out either with the reactor being fuelled, or defuelled, depending on whether or not interventions on the core are included. When fuel is present, the reactor is maintained in a guaranteed shutdown state. Refurbishment is then executed using an ‘outage model’. The work on SSCs is organized based on functional outage groups (FOGs).

A FOG can be defined as a safe perimeter of isolated SSCs selected to safely allow field work. They may contain groups of systems or components that are de-energized and secured within a certain space-time window. The isolated state is safeguarded and the heat sinks are controlled by the main control room shift supervisor and the operators. All SSCs requiring replacement or refurbishment are assigned to a FOG. Operations issues work permits on each system design change package to the field organization responsible for the schedule and the work implemented within the FOG envelope.

For a refurbishment programme done with the reactor fuelled, a sample organization chart for a refurbishment outage is presented in Fig. 27. Under the outage manager, there are six functional groups in charge of the six major categories of activities necessary to maintain the plant in a safe shutdown state, while controlling the implementation of engineering change packages. The outage coordinator manages the schedule taking into account feedback from the field, organizes break plans and resolves schedule adherence issues. The operations coordinator reviews work permit requests, liaises with the shift manager and posts the maintenance testing logs. The heat sink manager coordinates (with operations) the reconfiguration of heat sinks and the swapping of heat sinks where necessary to allow the implementation of design changes or maintenance work on systems. The system window coordinator coordinates with the various project managers responsible for the work on the FOG envelope, ensuring that work is at all times compatible with the FOG window permit. The system window coordinator also receives feedback from the outage coordinator and liaises with the schedule manager and the project managers regarding break plans or changes in the overall schedule, when and where required.

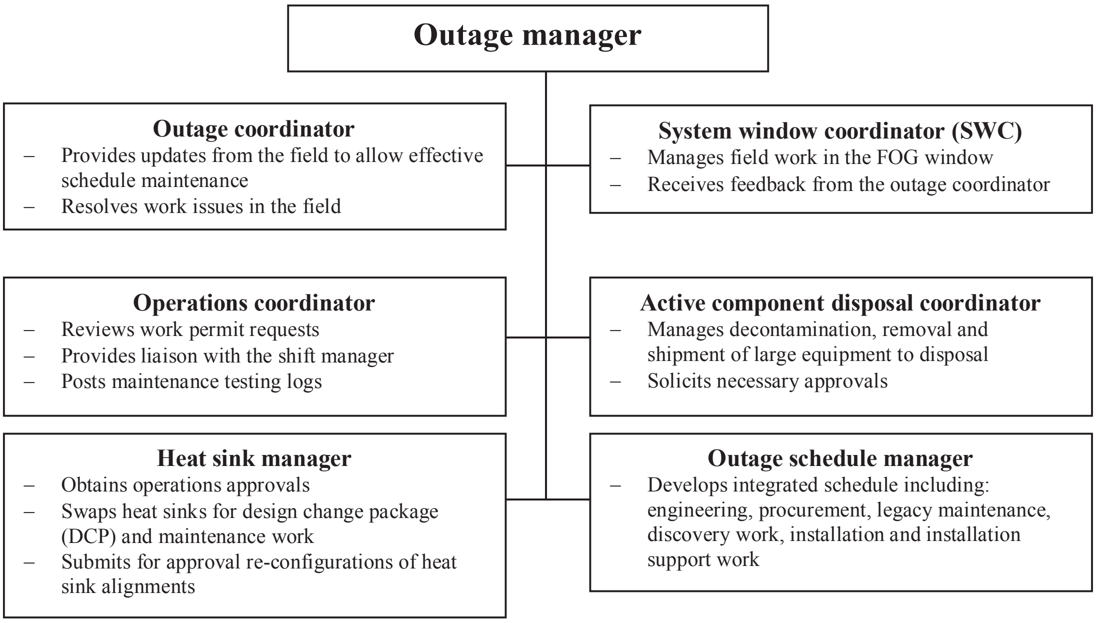


FIG. 27. Organization during the execution of a CANDU refurbishment outage.

The active component disposal coordinator oversees the removal, decontamination and shipment of large equipment and solicits all necessary approvals.

The outage schedule manager is responsible for the development and maintenance of the integrated outage schedule involving field engineering activities, procurement, legacy maintenance work, demolition and temporary modifications, unplanned discovery work and break plans, cranes and other infrastructure, security, radiation confinement, waste management work and installation activities.

The infrastructure necessary to support the project can at times be underestimated or simply forgotten. This infrastructure includes:

- Facilities to support the substantial increase in the number of employees, workers and contractors, such as enhanced security systems, increased radiation monitoring equipment, increased number of qualified radiation escorts and radiation trainers for external contractors;
- Strengthened laundry facilities;
- More overalls, plastics, double plastics with respirators;
- Increased numbers of lockers and sanitation;
- Cafeterias;
- Additional temporary lighting in the reactor building and confinement areas;
- Additional parking and office space;
- Preparation of large equipment lay down space and access routes;
- Preparation of temporary modifications and demolition work,
- Preparation of new cranes to allow new access;
- New storage and equipment movement routes;
- Coordination of scaffolds;
- Sufficient time allocated for dismantling within the implementation schedule.

A good engineering change control process is also essential. It should be regulated by a tested procedure, capable of integrating engineering, procurement and installation. Dealing with non-conformance issues should also be a prioritized management controlled activity. It should include a good station condition record system coupled with a prompt resolution process capable of driving root cause analysis and corrective actions.

In the realm of engineering, a substantial amount of hidden work is usually underestimated in project budgets. This work may, however, impact the project schedule if not well defined and caught in time. In Canada, this work is sometime defined as ‘gap engineering’, and indicates the substantial amount of detailed definition work that is normally not provided by the engineering agency or architect–engineering firm developing the design change packages. Gap engineering includes activities such as the development of on-line wiring drawings, shop drawings and the inevitable amount of field changes and field engineering work deriving from the walkdowns of work orders and work packages or the resolution of non-conformances with project managers, field assessors and foremen.

Preventing stock code number duplications of materials in stores or on order has proven to be a challenge. Further challenges include:

- The lack of standardization of commodity items, such as civil and architectural hardware (i.e. specialty concrete anchors);
- The lack of a robust standardization of installation procedures, including shimming, torque values, gasket standards to avoid rework and, more importantly, non-conformance and loss of traceability.

3.2.2. Operating experience and lessons learned in the Czech Republic

This section summarizes the conditions of critical components and refers to the main modifications adopted in support of LTO. Based on the RPV surveillance programme, the Dukovany RPV can achieve at least a predicted safe service life of 60 years, until 2045, and even longer. The only exception is the RPV for Unit 1, which has a service life prediction that is limited to 2040. This difference can be resolved by annealing the pressure vessel, or by less expensive measures (e.g. heating of the reactor make-up tanks). Experimental projects are under way to explore the option of extending the lifetime of the vessel's internal parts. The critical spot is the bearing cylinder, which may yield a negative evaluation and thus its replacement may be recommended. For the remaining internal parts of the reactor vessel, a positive evaluation is anticipated.

In 2025, it will be necessary to change the control rod drives. As far as other essential equipment of the primary circle — steam generators, main circular pumps, and pressurizers — all significant ageing effects are known and monitored by adequate programmes. Methodologies for mitigation of these effects are being applied. Significant investments beyond the scope of the current programme are not envisaged.

Equipment in the secondary side has been, or will be, extensively refurbished in the near future to address degradation and issues related to design margin usage. In 2012, the low pressure and high pressure turbine rotors were replaced with machines featuring double the guaranteed lifetimes. In addition, measures to reduce erosion–corrosion rates in the main condensers and in other components of the entire secondary circle were installed.

The I&C systems were extensively updated. This work started in 2009 and will be completed in 2016. It is assumed that the lifetime of this installation should extend to 2025–2030. For longer service life extensions to 2035 and 2045, it would be necessary to perform further I&C renovations.

The turbine generator reconstruction and that of the 110 kV distribution points were also completed. As a result of the degradation of the insulating materials, it will be necessary to carry out a gradual reconstruction or replacement of other electrical appliances. With the exception of those in the hermetic zone, the issue of cabling seems to be one of the most pressing problems in LTO. The challenges related to the collection of sufficient data on the cabling condition inhibit the determination of their residual service life, thereby making it necessary to take appropriate measures that will allow the determination of the residual lifetime of the cabling outside of the hermetic zone.

As far as structures and structural components, a complete reconstruction of most of the roof claddings of buildings is to be expected, as well as a gradual replacement of the internal and external claddings of the cooling towers. Repairs are planned for the steel liner in the spent fuel pools to protect the storing of fuel and shafts, and the reconstruction of some underground piping and cable channels is planned.

Among the programme's improvements, it is particularly important to mention the introduction of a new PLiM programme and partial AMPs, such as that for valves, or the extension of the AMP for cables to cabling outside the hermetic zone. It is also necessary to carry out modification of the RPV surveillance programme to address the specific needs of the LTO period.

Risks and savings opportunities in the implementation of modifications were assessed, as were other sources of risks.

During the evaluation of individual cost categories, potential risks related to LTO were identified. Remedial measures were proposed and costs for their implementation were evaluated. A conservative approach to evaluation was adopted and the proposed measures were submitted for economic calculation.

The PLiM programme resolved the monitoring and condition assessments of the SSCs. The PLiM programme also recommended other activities for the NPP Dukovany LTO to be included in further revisions of the LTO programme documentation.

3.2.2.1. Other sources of risks

The risk study output also deals with other sources of risk from the assessment of cost categories for which remedial measures were currently not defined. They may be important sources of risks, such as:

— Changes to regulatory body requirements;
— Extension of planned outages;
— Problems with staffing;
— Business risks (internal and external);
— Timing of initiation of the proposed measure and changes;
— Incorrect estimate of time and cost;
— Political risks.

The recommendations and conclusions did not include fundamental obstacles that could endanger the NPP LTO. A method to minimize the impacts of all identified spheres of risks has been proposed in order to plan the work and to implement the LTO programme.

3.2.3. Operating experience and lessons learned in France

The most critical component in the service life of French NPPs is the RPV. During its service life, the RPV steels are exposed to neutron irradiation, which causes microstructural changes and a degradation of mechanical properties. As the age of NPPs increases and life extensions and service lives spanning up to 80 years are on the agenda, some existing open issues regarding the understanding and prediction of RPV irradiation embrittlement need to be clarified. This is especially important for materials with higher contents of copper, phosphorus or nickel. Irradiation embrittlement effects resulting from high neutron fluences need to be adequately considered in RPV surveillance and safety assessments. Currently, high fluence data for original RPV materials are used in national programmes. Unfortunately, the surveillance database for long irradiation times (>20 years) and low neutron fluxes is rare. Consequently, the treatment of such long term irradiation effects is often affected by large uncertainties requiring the generation of new data and of their assessments.

In this context, the availability of microstructural data is essential for the understanding of the underlying mechanisms. The further qualification and validation of appropriate safety concepts, such as the master curve, is important because in the absence of long term irradiation data, best estimate assessment tools with reasonable conservatism become essential to justify LTO beyond the originally assumed plant design life. The availability of well developed and validated prediction tools for irradiation embrittlement would be advantageous for the implementation of any dedicated safety assessment and decision making on the economic service life of a nuclear power unit.

Project LONGLIFE (treatment of long term irradiation embrittlement effects in RPV safety assessments) was initiated to improve the understanding of the LTO irradiation effects and to determine the most appropriate application of the embrittlement surveillance procedures. The scope of work of the LONGLIFE project comprises the analysis of LTO boundary conditions, microstructural investigations and supplementary mechanical tests on RPV steels from decommissioned plants, with recommendations for RPV material assessments and irradiation embrittlement surveillance as well as LTO specific training requirements.

In the most recent series of 1450 MW(e) N4 units, the predicted end of life transition temperature RT_{NDT} (reference temperature for nil ductility transition in RPV belt line materials) was 42°C. Since predictions may be different from the actual ageing in the field, surveillance, monitoring and recording of historical operational data becomes an important factor in ageing management of components, particularly of the RPV.

If, at the time of design and fabrication, materials were specified compatibly with thermal treatments, then physical methods to control embrittlement may be applied, such as thermal conditioning to restore the RPV materials approximately to their initial state.

In France, ageing management of pressure vessels is mainly done using a low fluence fuel management programme. In addition, removable surveillance specimens, normally held among the vessel internals in strategic positions where neutron fluxes are higher than the vessel wall ever experiences, are periodically retracted and analysed to determine or correct trends in order to more accurately control and predict the end of life transition temperature. At the moment, predictions allow planning for life extensions of the vessel from 40 to 60 years with a sufficient margin.

Of concern are also some areas of the pressure vessels that contain Inconel 600 material susceptible to stress corrosion cracking. The French systematic replacement programme of its 54 vessel heads, activities that did not influence outage duration and that remained limited in terms of cost and dosimetry, have eradicated the problem entirely.

Among the RPV internals, a number of baffle bolts and baffles in high flux zones of the 900 MW(e) units developed cracks due to loss of ductility. New materials for the internals have been developed, and where necessary, replacement has been the preferred solution.

Operating experience with steam generators has shown that the control of tube leaks is a major undertaking in the life management of these components, in France and abroad. Inspections and analysis showed that root causes of tube leaks are stress corrosion cracking, support plate tolerances, clogging and tube surface fouling. Significant blockage of the quatrefoil broached interspace between the tube and the support plates has been observed, resulting in an increase of local fluid velocities. This is the root cause of vibration induced fatigue and, hence, of tube impact loads and ruptures. Interspace clogging causes pressure drop increases in the overall secondary side flow and reductions in the circulation ratio. Continuous on-power monitoring using the wide range level instrumentation can be correlated to the overall pressure drop. On-line backflow level monitoring was implemented for the entire fleet in France. Alarm thresholds were also established to help decide when to take compensatory measures. During outages, televisual inspections of the tube support plates and eddy current testing of the tubes is universally practiced in France.

Another ageing phenomenon is the fouling of exposed surfaces of the secondary side, resulting from deposits of impurities and iron oxide contained in the feedwater due to specific operating conditions, such as the need to keep a low pH environment in the secondary circuit, the presence of additive residues (aggressive complex ammonia compounds), raw water infiltrations (e.g. condenser leaks into the feedwater stream), the presence of erosion–corrosion products in suspension and an imperfect chemistry control programme. External tube surface fouling inevitably decreases the overall steam generator heat exchange capability. With increasing secondary side fouling, the amount of steam produced, and the overall steam pressure, decrease. Comparing the steam pressure drop to predefined thresholds and using this input in mass balance calculations throughout the lifetime of the steam generator (with due consideration to the blowdown history, lancing operations, chemical cleaning sessions and other interventions) allows fairly accurately plotting of the steam generator secondary side fouling history.

The lifecycle of currently operating steam generators is strongly linked to the percentage of tube rupturing, and, therefore, of plugged tubes. The number of plugged tubes has become the prominent factor in steam generator replacement planning because this greatly affects heat exchange capability. Ageing mechanisms have adversely affected the alloy 600 tubes used in the first generation steam generators in France and in other parts of the world. This experience has induced steam generator tube manufacturers to switch to the use of alloy 690 (more resistant to rupture), without having to appreciably tax the steam generator heat transfer coefficient, and its size.

For steam generators with Inconel 600 tubes in France, a ceiling of 15 effective full power years has been considered an average life span. Replacement steam generators with alloy 690 tubes will allow longer life spans. Ideally, the aim is to ensure that the new material will not require a second steam generator replacement.

3.2.4. Operating experience and lessons learned in Hungary

The plant AMPs may be used in LTO, provided that they meet the evaluation criteria of plant programmes for LTO and licence renewal. A regulatory guideline provides the requirements for the acceptance and optimization of plant programmes to ensure that the required plant conditions and the intended safety functions and performance are maintained. The programmes have to be reviewed and the adequacy of each has to be demonstrated along with its completeness and compatibility with the other LTO programmes.

3.2.4.1. Review of the ageing management programmes

The review and qualification of the AMPs need to meet the following criteria:

— Determination of the degradation mechanism and affected areas;
— Mitigation and preventive measures;
— Parameters to be monitored;
— Detection of ageing effects;
— Monitoring, trending and condition assessment;
— Acceptance criteria;
— Corrective actions;
— Feedback, efficiency and improvement of the AMP;
— Administrative control, quality assurance, coordination and documentation;
— Feedback from operation and condition of the component.

The AMP review is one of the most essential parts of the LTO and licence renewal programme. The review conclusions are documented. Some existing programmes may be qualified without modifications, whereas others may require filling the gap of new developments. All these can be completed within the framework of updating the LTO programme.

3.2.4.2. Review and modification of the ISI programmes

Hungarian nuclear safety regulations allow for the possibility of adopting requirements of national (if available) or international codes. They must in any case be recognized codes, such as the ASME, as regulators do not specify codes and standards covering the design and commissioning of an NPP or the ISI to be performed during operation. Therefore, internationally recognized codes should be adopted before the start of the ISI programmes. Once the code is adopted, verification can be conducted on the safety performance of pressure retaining components. As a result of ISIs, component repair and replacement may have to be carried out if inadequate ISI results are obtained.

Adoption usually also includes tailoring of the selected code requirements to the specific circumstances at the NPP under consideration. Adopting a new code may replace a former, outdated code, thereby ensuring compliance with one that is up to date and internationally recognized. This change, however, cannot be implemented on a routine basis.

The Appendix contains more information on the Hungarian experience with respect to altering the applicable codes and standards during operations, and the extra work necessary to accommodate the necessary changes before applying for an LTO licence.

3.2.5. Operating experience and lessons learned in India

The Indian regulatory body, AERB, implemented PSR in India for the first time in 2003. Certain weaknesses were observed, namely:

— The NPP reports did not adequately address some of the review elements envisaged in the AERB guide.
— Ageing management and equipment qualification were found to not be adequately addressed, and the operator was asked to develop comprehensive programmes to resolve these issues.
— Internal reviews conducted by the utility were found to be inadequate in order to draw the appropriate conclusions and inferences, and as a result the NPP was asked to supplement the PSR report.

- Other site specific issues included failures in channel monitoring resistive temperature detectors in the Narora Atomic Power Station, the necessity of reducing tritium concentration in accessible areas of the reactor building, the necessity to relocate ECCS pressure and level transmitters from a high enthalpy to a lower enthalpy area in NAPS and KAPS NPPs, the repeated failures of adjuster rod drives in KAPS, the lack of baseline data on the primary coolant system feeders in KAPS and the necessity of seismic upgrades for the secondary side piping and equipment in MAPS.

Similarly, the strengths observed during the conduct of the PSR included:

- The programmes for maintenance, in-service inspection and chemistry control that have been in place since the beginning of plant operation;
- Constant improvements recorded regarding operational performance;
- A reduction in the number of events and a decrease in the trend of collective doses;
- Well established OE feedback systems for all NPPs in India, and close compliance with regulatory recommendations;
- Safety analysis was wide ranging, that is, it encompassed safety concerns connected to postulated initiating events that had previously not been analysed;
- No concerns regarding any of the life limiting ageing related degradation of non-replaceable components;
- Radioactive emissions and discharges were well below the limits set in the technical specification.

A number of action items were issued in connection with the periodic inspection programme. Among the most important were:

- An expert group reviewed the existing safety analysis report in light of the latest AERB requirements (AERB/SG/D-5) for newly designed plants. Recommendations included a revision to the safety analysis with models reflecting the latest plant modifications and with additional postulated initiating events, using the latest computer codes with accepted validation and verification certificates. The analysis report was also to be drafted following a specified format.
- The establishment of comprehensive AMPs that went beyond good maintenance, surveillance and inspection programmes. The programme was to be established in accordance with AERB Safety Guide AERB/NPP/SG/O-14, Life Management of Nuclear Power Plants, which requires the identification of SSCs important to safety using AERB Safety Guide AERB/SG/D-1, Safety Classification and Seismic Categorisation for Structures, Systems and Components of Pressurised Water Reactors. Ageing management was to also include on-line monitoring and ageing mitigation programmes.
- A systematic and comprehensive equipment environmental qualification programme to include a detailed master list of equipment and components based on their safety functions and their design characteristics as described in AERB/SG/D-1. The qualification is not only to be demonstrated by comparing the service conditions with the conditions assumed in the design and operating manuals, but the programme should also contain provisions to preserve the qualification in time at least for the authorization period. Finally, the owner/operators are to produce guidelines and testing sequences for loss of coolant and other accident conditions.
- An update of the technical specifications to establish less stringent and less conservative requirements, and the postponement of surveillance requirements because of the continued operation of the units. This entailed a revision to the surveillance frequency based on the failure data with special consideration given to requirements, the inclusion of design and procedural changes, if and when necessary.
- A further reduction of the collective dose. Efforts are required to obtain a systematic reduction, which may entail an increased use of remote tools and an upgrade of the maintenance procedures with dose reduction as the main goal.
- The shortfall regarding compliance with ISI requirements. This was attributed to overly conservative ISI practices, such as requiring repeated inspection for identical equipment. A more targeted inspection programme was recommended with a particular focus on areas with known flow accelerated corrosion such as feeders and the secondary side pipeline. A revision to the current ISI manual was recommended to align it with current international practices.

In conclusion, it was recognized that the PSR experience was an exhaustive exercise. The resources, effort and time spent to carry out a PSR were considerable. In that respect, there is a need for further optimization. The very first PSR reports did not meet the regulatory body expectations, but subsequently the PSR brought into focus issues such as the lack of systematic equipment qualification and AMPs. The PSR helped not only to ascertain the current safety of NPPs by checking and testing the safety margins of essential systems, structures and equipment, but also allowed the identification of all the differences between the safety standards in the plant and the current safety standards for new plants.

The AERB safety guide is being revised based on the experience accumulated, the revisions to IAEA Safety Standards Series No. NS-G-2.10 (this has been superseded by SSG-25) and international experience.

3.2.6. Operating experience and lessons learned in China

All 15 NPP units in operation in China are owned by two groups: CNNC and CGN. Technical organizations and vendors from various industrial groups provide technical support to ensure the safe operation of the NPPs.

On 18 April 2007, the China Nuclear Energy Association (CNEA) was established as a national non-profit and non-governmental organization with the mandate to function as a bridge between CNEA members, government agencies and foreign vendors. CNEA's mission is to implement national policies on nuclear energy development, promote independent industrial innovation and technical advances in nuclear power applications and promote improvements in safety, reliability and economics of nuclear energy applications. Under CNEA, there is a committee responsible for the organization of peer reviews and the collection and dissemination of experience feedback to support the continued safe operation of NPPs. The committee members are selected from NPP operations, NPP design organizations, as well as from technical support organizations. All NPPs are requested to periodically send their event reports, outage plans and performance indicators, to the committee. In turn, the committee publishes a quarterly comprehensive operational performance summary report that includes general information on NPP operation, outage activities, important maintenance activities on main systems and heavy components, technical innovation introduced into important SSCs, data on waste disposal and environmental monitoring and performance indicators from the World Association of Nuclear Operators and operational and internal events. The committee also prepares topical annual reports, including the following:

- Plant Events and Experience Feedback: Annual Report for Chinese NPPs;
- Key Performance Indicators: Annual Report for Chinese NPPs.

The committee's main responsibility is to plan and conduct peer reviews on various topics. Peer review team members are usually experts selected from NPPs other than the one under review. Through these activities, operational experience and lessons learned are efficiently and transparently shared among similar operating organizations.

As for PLiM programmes, many activities including R&D, standardization, guidelines and procedure preparation, AMP practices and SSC innovations, among other things, are carried out by utilities supported by their technical support organizations. Ageing management activities in Chinese NPPs are generally split into two categories: overall AMP and component specific AMPs (also known as topical AMPs). The overall AMP category includes:

- Plant AMP policies;
- Organizational structure and allocation of responsibilities;
- Relationship between AMPs and existing operation and maintenance programmes and procedures;
- Requirements for data collection, record keeping and plant database systems for ageing management applications;
- Screening of SSCs in the scope of an AMP;
- Methodology to identify significant ageing mechanisms for SSCs covered by an AMP;
- List of component specific AMPs;
- List of topical AMPs.

AMPs cover a variety of components, including RPV, reactor vessel internals, steam generator, pressurizer, reactor coolant pump, primary piping including the surge line, cables and containment, among other things. For a PHWR, special CANDU components (such as the calandria, the pressure tubes and feeder pipes) are in the list to establish the respective AMPs. A typical component specific AMP may include the following:

- Component description, including design documents, manufacturing information, commissioning information and operation and maintenance records, such as design changes during operations, FASR and PSR regarding ageing issues for the component;
- Regulatory requirements, codes, standards and guidelines for ageing management and analysis;
- Ageing mechanism analysis indicating the most significant mechanisms and their characterization;
- Organizational structure and responsibility allocation for ageing management for the component;
- Measurements for timely detection of ageing degradation;
- Measurements to mitigate significant ageing degradation;
- Methodology for ageing analysis and its main conclusions;
- Experience feedback and a database system.

The topical AMPs focus on general issues and are not specifically for one component, such as AMP for deadline ageing and AMP for obsolescence.

3.2.7. Operating experience and lessons learned in the Republic of Korea

It may not be strictly required that operational experience feedback be considered during the safety reviews of the NPP design life term, however, some experience feedback and research items, be they domestically or internationally produced, may be considered important enough to include in the AMP. By reflecting on OE, a licensee can improve the ageing management quality of its SSCs. In the Republic of Korea, important OE to be considered in the ageing management of the plants is listed in NSSC Notice No. 2012-25, Evaluation Scope and the Applicable Guidance on the Technical Basis of CO (continuous operation).

Tables 9 and 10 illustrate the OE feedback items and research findings that need to be considered by licensees. In addition, when new OE and research results are published during the LTO of a plant, they should either be reflected in the existing AMP, or alternatively a new AMP should be established so that a higher SSC safety level can be maintained for the period of continued operation beyond the original NPP design term.

TABLE 9. OPERATING EXPERIENCE AND RESEARCH FINDINGS

TLAA items	Reference regulations and technical standards
Evaluation for fire protection	10 CFR 50.48
Seismic qualification of equipment	Regulatory Guide 1.100
Pressurized thermal shock of reactor vessel	10 CFR 50.61
Anticipated transient without scram	10 CFR 50.62
Active components management plan	ASME Operation and Maintenance of Nuclear Power Plants
Evaluation for thermal stratification of piping	NRC Bulletin 88-11
Safety assessment for ignition of combustible gas	IAEA NS-G-1.10
Evaluation of capability for coping with station blackout	10 CFR 50.63

TABLE 10. OPERATING EXPERIENCE AND RESEARCH FINDINGS TO BE REFLECTED FOR PHWRs

TLAA items	Reference regulations and technical standards
Evaluation of fire protection	CAN/CSA-N293, Fire Protection for CANDU Nuclear Power Plants
Seismic qualification of equipment	CAN3-N289.1, General Requirements for Seismic Design and Qualification of CANDU Nuclear Power Plants
Active components management plan	ASME Operation and Maintenance of Nuclear Power Plants Korea Electric Power Industry Code Maintenance and Operation of Nuclear Power Plants CAN-N290.1, General Requirements for Safety Systems of Nuclear Power Plants CAN/CSA-N290.5, Requirements for Electrical Power and Instrument Air Systems of CANDU Nuclear Power Plants
Evaluation for thermal stratification of piping	NRC Bulletin 88-11
Safety assessment for ignition of combustible gas	IAEA NS-G-1.10

The management programme for active components important to safety is one of the lessons learned in the process of implementing LTO. The following items are considered during the programme review:

— An effective record of the active components subject to evaluation, which includes a condition assessment of the components.
— SSCs selected from among the safety related components or non-safety-related components whose failure could prevent safety related components from fulfilling their safety functions.
— The function and performance of components are maintained.
— The appropriate corrective action programme is established.
— The appropriate maintenance programme is established.
— The risk of severe accidents is minimized, and the licensees provide a safety assessment of the consequences of hydrogen ignition. The accident sequences are to be selected from PSA and the quantity of produced combustible gas is to be estimated using realistic analysis methods.

The concentration of combustible gas needs to ensure at all times that the flame acceleration threshold is not exceeded or that the detonation transient does not occur. Alternatively, another reasonable or practical method should be provided to ensure containment integrity is maintained.

3.2.8. Operating experience and lessons learned in the Russian Federation

The first lifetime extension project for the second generation WWER-440 plants started in the Russian Federation over 12 years ago for Units 3 and 4 of the NVNPP (Novovoronezh Nuclear Power Plant). The task was called project B-179. Since then, significant experience was accumulated in the lifetime extension of second generation nuclear power units of all types in operation in the Russian Federation (WWER-440, WWER-1000, RBMK-1000, EGP-6 and BN-600). This experience gave the possibility to develop, and later significantly improve, the norms on lifetime extension. Furthermore, high efficiency modern instrumentation, tools and methodologies have been applied, providing precise condition assessments of SSCs and the most complex and reliable information about the NPP as a whole, pinpointing the exact specificity of its condition and of its ageing mechanisms.

Currently in the Russian Federation, first generation WWER reactors in the NVNPP, comprising Unit 1 WWER-210 (in service between 1964 and 1984) and Unit 2 WWER-365 (in service between 1969 and 1990) have been taken out of service after reaching their design life and are being prepared for complete decommissioning. All second generation WWER reactors, including WWER-440 (projects B-179, B-230 and B-213), WWER-1000 (projects B-187, B-302 and B-320), and the RBMK-1000, EGP-6, BN-600 reactors, are now in the process of completing their lifetime extension process or have just about completed it, even before reaching their design lifetime.

The design and construction of the first NPPs in the Russian Federation were based on regulatory documents, norms and standards in use between 1960 and 1970. These regulations were produced for conventional power stations and later approved as NPP standards with a few additions for the design of the first series of nuclear power units. Generally, these adjustments addressed radioactive shielding, biological protection from nuclear reaction and material irradiation. The first NPP safety requirements, as such, were formulated in 1973. The design of nuclear power units focused mainly on high quality materials and equipment, attention to maintenance and operation (particularly a periodic condition assessment of all metallic equipment and pipelines) and high personnel qualification standards. Significant improvements to these requirements were recently introduced during the life extension evaluation projects of the second generation of Russian NPPs, where high safety and reliability margins of the primary safety related components were adopted. Equipment and systems modernization projects have been carried out to increase overall NPP reliability and safety.

Nuclear power units in the Russian Federation today operate in strict compliance with: the requirements of current standards, rules, guidelines and improved operating procedures and working instructions; the timely implementation of maintenance and repair activities; the periodic condition assessment of safety related SSCs, and; the continued education, application and management of safety culture principles. Compliance with these ensures an acceptable safety level of Russian NPPs at every stage of their life cycle.

The successful life extension of the second generation of Russian NPPs, and the accumulated positive experience of power units operating beyond their design lifetime, illustrates that the Russian life extension assessment standards are correct and adequate, and that the additional measures adopted are sufficient to ensure safety and reliability of all NPPs operating under the Russian LTO norms.

Operating experience with different NPP types in the world, supported by R&D to fill knowledge gaps, has significantly increased the industry's technical capabilities and knowledge base. This also allows for a substantial improvement of the old standards and safety guidelines supplemented by new design standards and requirements aimed at improving radiation safety and reliability in new designs and in operating nuclear power units.

The scope of work to justify an NPP life extension in the Russian Federation will depend on the fatigue and stress calculations for all important components based on the updated Russian norm PNAE G-7-002-86 [14]. Certain components may not meet the new requirements. In such cases, the scope of work will include modernization and reconstruction of such components. Usually, the assessment leads to a modernization and refurbishment of pipelines, of hanger support systems, replacement of flange couplings or the addition of supports to meet higher earthquake loads, among other things. Some refurbishments and modernizations require a long time and are usually implemented in several stages during major outages for several years before the end of the design life, and in some cases, some work is carried out even afterward.

All NPP safety normatives change over time and these changes can be very significant. As a result, older NPPs may not meet a number of new requirements towards the end of their lifetime. If new normative rules, and requirements are not met, a list of deviations is submitted to the regulatory body. Deviations are categorized according to their safety importance and a corrective action plan is proposed aiming at minimizing the impact that these deviations may have on safety.

During the life extension process of Russian power units, operators found that there were components important to safety for which there was no documentation because it was partially or totally lost. This is not acceptable in light of the safety requirements for life extension. For these components, design basis reconstitution projects have been launched. In some cases, design data is obtained using advanced technologies (e.g. laser scanning for components with complex geometry). The geometry of piping runs is recovered by creating new 3-D isometric diagrams either conventionally or by means of optical scanning. The chemical composition of steel is measured and the steel grade is obtained by testing. The reconstituted equipment and piping specifications are then drafted, certified and registered with the regulatory authorities.

Even when specifications, drawings and other documentation is available, it is always to be analysed with respect to the current normative. Old equipment specifications and pipeline certificates may not fulfil all current requirements, both structurally and in terms of content, since their issue precedes, by decades, the publication of the latest safety normative. Usually, deviations of this kind and the reconstitution of the design basis documentation are allowed by the established procedures, and the new technical specifications should not be a hindrance to the operability of the affected component.

However, according to the new normative requirements for lifetime extension, the design basis reconstitution work for such components, and improvements to the related operations documentation (e.g. operating instructions, safety engineering studies or operation inspection project), normally lead to improvements in the operating and safety culture and reduce the possibility of human error, thereby benefitting safety in the operating power unit.

During the service life of the power unit, configuration improvements are carried out to modernize and refurbish areas for continued enhancement of reactor reliability and safety. Within the context of lifetime extensions, modernization and refurbishment work is of the utmost importance because it is an instrument that allows the power unit to meet the new nuclear radiation protection and nuclear safety requirements. It is clear that a full alignment of the older power units, built between 1960 and 1970, with the current NPP safety requirements may not be technically or economically possible, but a significant reduction of the safety deficit is certainly an achievable goal within the framework of a lifetime project.

For example, safety improvements performed on the pilot WWER-440 projects, namely the modernization and refurbishment of Units 3 and 4 of the NVNPP plant, which were the first units undergoing a life extension of 15 years, included important additions and design changes such as:

— The creation of two independent safety system channels with internal active components;
— The extension of the design accident safety requirement from a 32 nominal diameter ($D_{nom}32$) break of the primary circuit pipeline to a 100 nominal diameter ($D_{nom}100$) break;
— Reduction of the risk of primary circuit pipeline breaks for $D_{nom}200$ and $D_{nom}500$ using the leak before break (LBB) concept and adopting a three channel integral leak monitoring system;
— Reduction of the radiation exposure of the staff, of the public and of the environment in case of beyond design accidents (loss of coolant accident $D_{nom}200$ and more) achieved through technical and organizational means;
— Resolution of the third and fourth category deviations from safety normative requirements (according to the IAEA’s classification);
— A substantial reduction of the active area failure rate from 1.08×10^{-3}/a to 3.44×10^{-5}/a and 5.12×10^{-5}/a for Units 3 and 4, respectively, as demonstrated through a first level PSA.

Although the recommended probability value of 1×10^{-5} per reactor per year of cumulative severe beyond design basis accidents as per OPB-88/97 was not achieved, it is nonetheless evident that modernization work performed within the context of the life extension projects represents a big step in the right direction. Given the positive LTO experience during the prolonged service life of pilot Units 3 and 4 in the NVNPP, work for a second life extension of Unit 4 is currently being planned for the first time in the history of the Russian nuclear industry.

It is important to note that the requirements for a second life extension have become even more stringent. Safety requirements for the design accident spectrum of the main circulation pipeline now include the rupture of a 500 mm pipe and increased confinement tightness. This goal will be reached through the following safety system modernization steps:

— An improved ECCS capable of supplying cooling water to the reactor active area in case of rupture of a primary circuit pipeline of a nominal diameter above 100 mm:
  - Implementation of a passive ECCS (provision of additional hydro-accumulators for the ECCS);
  - Implementation of an active low pressure ECCS (long term emergency make-up via low pressure pumps injecting cooling water into the reactor through the primary circuit).
— Modernization of the entire reactor confinement area to meet the indicated accident requirements, without exceeding the stated radiological impact criteria.

The plan also includes complex material science investigations of the main equipment and pipeline conditions, taking into account the long term ageing process. These goals are taken for granted in modern project power units, but considering that all this work needs to be performed on original main circuit equipment, without significant modifications to the building, they present unique challenges and will require new approaches and technical solutions, which will go beyond the scope of the current life extension norms.

Activities in the work scope are aimed at defect elimination, reduction of the ageing mechanisms and recovery of the reliability and capacity factor in order to reach the durability targets for the NPP unit main components. Refurbishment and modernization, in the context of lifetime extension, usually consists of replacing the equipment and the secondary coolant circuit pipelines with stainless steel (mitigation of flow accelerated corrosion effects), and replacement of copper parts in components handling condensate, worn equipment, cables, driving gears of the reactor control rods, automatic control system components and hardware and software of the central unit control room, among other things. In some cases, extensive compensatory actions need to be taken to meet the required life targets of NPP components.

Developing and installing on-line monitoring systems in the most problematic areas is one example of such actions. On-line monitoring systems allow on-time detection of operation related defects and the development of optimized maintenance tasks (see the Appendix). Because of the implementation of such systems, there has been a steady reduction in operational violations (according to the International Nuclear and Radiological Event Scale) in Russian NPPs during the last several years.

Organizations involved in NPP operation share experiences in the Russian Federation through monthly operators meetings and NPP management meetings, and also in quarterly and annual meetings of NPP profiled sections and departments. An investigation commission analyses emergency situations and breakdowns as they occur and conducts a technical search of similar components and systems breakdowns. During the year, practical research and technology conferences on different topics are held with the involvement of key institutes and high profile organizations, Russian NPP and foreign specialists, including representatives of NPPs and high profile state institutes. These actions allow the Russian Federation to follow global trends in NPP life management, life extension and ageing control, among other things, to adopt and to implement during production lessons learned worldwide, and also to share valuable experiences and feedback on relevant issues facing nuclear power operators in the world.

3.2.9. Operating experience and lessons learned in Spain

Ageing management is seen as a process that needs periodic assessments and updates. One of the main sources for improvement in ageing management is feedback from OE, which is therefore used to continually improve the AMPs that are the main tools to ensure SSC ageing is properly managed in the plant.

Most of the changes that arise in the programmes are connected with maintenance activities. Some examples are:

- New guidelines to reflect the condition of equipment accessible during maintenance activities;
- New guidelines to inspect cables and determine their condition;
- Training and guidelines for walkdowns;
- Improvements to the identification of structural components.

The implementation of these AMPs has shown growing issues that could be expected in the LTO of other plants. Some examples are:

- *Flow accelerated corrosion.* This becomes a challenge in LTO as the replacement of large equipment is required for plants where carbon steel is issued. A flow accelerated corrosion programme is a classic ageing management intervention typically implemented well before reaching LTO to avoid its impact becoming too costly, since the need for inspections, repairs and replacements increases with time.
- *Cable ageing.* This is another challenging issue in operating plants. Cable replacement is a complex operation simply because the scope of the programme involves thousands of cables. A substantial effort is required and many new procedures for inspection walkdowns and testing had to be created. This programme also required much planning to define which areas require inspection and testing, and to minimize loss of availability and high doses related to the sheer number of inspections and tests required. Environmental qualification has led to the replacement, before LTO, of large equipment, such as four emergency electrical motors. Similar situations could happen in other plants as the qualification of some expensive equipment normally ends at the beginning of the LTO period.

The condition of buried piping has increasingly become an issue mainly due to the difficulties related to the use of standard inspection techniques. Development is ongoing to deliver new and more effective inspection techniques.

All the OE accumulated in the plant with regard to ageing has been reviewed by the AMP owners and by the coordinator. In addition, the NRC's Generic Aging Lessons Learned (GALL) Report (currently NUREG-1801 [8]) is periodically reviewed to include OE from US plants. These reviews are analysed, since the GALL Report is one of the main references in the definition of AMPs.

3.2.10. Operating experience and lessons learned in the United States of America

One of the key elements for continuous improvement of ageing management is the use of OE and lessons learned during the implementation of AMPs. The document NEI 95-10 (Rev. 6), which is the basis for licence renewal evaluations for LTO in the United States of America and many other States, has identified OE as one of the ten key elements of every AMP for LTO. The OE element is described as:

> "Operating experience of the aging management activity, including past corrective actions resulting in program enhancements or additional programs or activities, should provide objective evidence to ensure that the effects of aging will be adequately managed so that the intended functions of the structure or component will be maintained during the period of extended operation."

Although this ageing management activity element is described in terms of justifying the adequacy of the existing ageing management activity for LTO, it is also intended that OE reviews continue for the life of the ageing management activity to demonstrate programme effectiveness. For example, in NUREG-1800 [13], Section A.1.2.3.10, the NRC states that:

> "Consideration of future plant-specific and industry operating experience relating to aging management programs should be discussed. Reviews of operating experience by the applicant in the future may identify areas where aging management programs should be enhanced or new programs developed. An applicant should commit to a future review of plant-specific and industry operating experience to confirm the effectiveness of its aging management programs or indicate a need to develop new aging management programs. This information should provide objective evidence to support the conclusion that the effects of aging will be managed adequately…"

In order to ensure that ageing management activities and programmes are effective and are continuously improved during the extended period of operation or LTO, the OE programme needs to address ageing issues. In the United States of America, nuclear plant operators have a comprehensive OE programme that is based on INPO 10-006, Revision 1, Operating Experience (OE) Program and Construction Experience (CE) Program Descriptions, and INPO 97-011, Guidelines for the Use of Operating Experience. These OE programmes monitor sources of plant and industry information, such as:

— NRC licensee event reports;
— NRC generic communications (bulletins, generic letters, regulatory issue summaries and information notices);
— INPO event report documents;
— Nuclear steam supply system information;
— Owners group reports;
— Vendor bulletins;
— 10 CFR 21 reports.

These sources of OE information, and others, are screened by site personnel to determine whether they may have implications for the specific nuclear plant site subject to the review. Further documentation in the corrective action programme, which is part of the quality assurance programme, is required when the initial OE evaluation identifies a condition adverse to quality, non-conformance; potential inoperability of any structures, systems or components; degraded equipment or equipment not performing as expected or per design. Degraded equipment includes equipment degraded due to the effects of ageing. The results of an OE review can include enhancement of existing ageing management activities or development of new AMPs. Examples of new AMPs created due to OE reviews in the past include the flow accelerated corrosion programme, the buried piping inspection programme and the medium voltage cable inspection programme, among other things.

Another element of OE activities is the updating of industry and regulatory ageing management guidelines to incorporate lessons learned. One example is the ongoing revision of the NUREG-1801 [8] GALL Report. The GALL Report contains the NRC's generic evaluation of existing AMPs that have been determined to be adequate for LTO, and programmes that should be augmented in order to be adequate for LTO. The report originated in 2001 and was revised in 2005 and 2010. The revision process was necessary to ensure lessons learned based on OE during the years between revisions were incorporated. The NRC plans to continue to monitor the effectiveness of AMPs listed in NUREG-1801 [8] and incorporate lessons learned based on OE when needed.

In summary, safe and reliable LTO is dependent on activities, such as OE programmes, corrective action programmes and self-assessments of ageing management activities to ensure lessons learned are captured and continuous improvements are made when weaknesses are identified. These activities are crucial to the success of LTO.

3.3. HANDLING OF DESIGN AND LICENSING CHANGES

3.3.1. Handling of design and licensing changes in Canada

Design changes required at the time of an LTO application may stem from various sources, such as new regulatory requirements, modernization programmes, obsolescence, the introduction or expansion of environmental qualification programmes, motorized valve and driver upgrade programmes, changes dictated by operation feedback or lessons learned from the Fukushima Daiichi accident (which has prompted various degrees of intervention in plants around the world).

The original design life of a plant is driven by the design life of its major components. However, service life, the time during which the plant can operate safely and reliably, may exceed the design life when conditions warrant it. This is because the actual operation in terms of, for example, fatigue cycles and ageing factors such as erosion–corrosion, among other things, for most components may be considerably lower than their design targets. When possible, selected components are replaced to re-establish margins and full operability. In CANDU reactors, the core components that require replacement in order to extend plant life are primarily the fuel channels at about 30 years of operation. Removal and replacement of fuel channel components in a CANDU unit is usually an economically viable option and has been a pivotal milestone in the life management of CANDU units. Extensive analysis and studies of fuel channels have been undertaken over the years and lessons learned summarized in a number of publications [15–18]. Bulk lower feeder replacement is another significant activity that may be required in older plants for LTO. Replacing feeders during the large scale fuel channel replacement (LSFCR) outage may actually reduce the duration of the LSFCR by improving access to the fuel channels.

Other important components that may need refurbishment or even replacement for LTO are the steam generators. The latest design of steam generators includes improved materials, and these components may not have to be replaced. In addition, all plants in Canada have implemented steam generator life cycle management, optimizing inspection, monitoring materials and applying ageing mitigation to achieve design life and beyond.

In terms of changes stemming from safety reviews following the Fukushima Daiichi accident, the CNSC has requested a comprehensive safety review of all NPPs in the country, and of all research and isotope reactors and nuclear laboratory facilities with particular focus on their defence in depth philosophy and features. The safety review included:

— External hazards, such as seismic, flooding, fire and extreme weather events;
— Measures for prevention and mitigation of the consequences of severe accidents;
— Emergency preparedness and emergency response plans;
— Severe accident management procedures;
— Design basis and accident analysis;
— Multiunit plant accident propagation prevention and mitigation capability;
— Overall nuclear emergency management system including the State's emergency management framework and related processes.

Licensees have examined events (more severe than those that have historically been regarded as credible) and their impact on NPPs. Certain design enhancements for severe accident management, such as changes to improve containment performance to more adequately prevent unfiltered releases of radioactive products, or to enhance control capabilities of hydrogen releases and to improve the survivability of equipment and instrumentation, were evaluated and implemented wherever practicable. Finally, changes were also proposed in an effort to enhance safety to a level approaching that of modern NPPs including also, modifications to procedures, processes and emergency management arrangements, wherever needed.

3.3.2. Handling of design and licensing changes in the Czech Republic

3.3.2.1. Reconstitution of the design basis

Knowledge of the current NPP design basis is a fundamental starting point for any LTO programme. The design changes and assessments of SSC ageing effects on their performance and functions need to be supported by easily accessible design basis information to prevent undesirable infringements on NPP safety.

The safety design bases reconstitution project started in 2003. The central data gathering process was completed in 2006. In 2007, the reconstitution of the design basis parameters started. Both the design basis functions and parameters were assembled. The reconstitution was prioritized according to the needs and requirements of the recommended modifications.

3.3.2.2. Configuration management

Configuration management is a process ensuring that the installation, operation, maintenance and testing of equipment occurs in accordance with its design and interface requirements as defined in the design documentation. Configuration management's key role is to ensure that accurate information is available during the lifetime of the plant, and that it remains consistent with the plant's physical state and with its operating parameters as it ages. It also ensures that any approved design changes maintain the correct correspondence between design requirements, the physical state of the plant and the controlled documentation during the plant's lifetime.

This is achieved by periodically identifying, verifying and managing information (including the management of electronic documentation) related to the physical configuration of the plant, and ensuring that at all times the configuration meets the design requirements.

The Dukovany NPP LTO programme is based on the assumption that all design changes are governed by the system configuration management, and that they are systematically assessed using a standardized approach.

3.3.2.3. Legislative requirements of SONS (regulatory requirements)

The State Office for Nuclear Safety, responsible for the supervision, administration and use of nuclear energy in the Czech Republic, issues operation permits to its NPP licence holders for ten year cycles. The precondition is that the licensee meets a set of legislative requirements as follows:

(a) Final safety analysis report

The first precondition for obtaining an operating permit for an individual NPP unit (Reg. Nos 6773/3.1/01, 55714/2006, 4866/98/3.2/30 and 11167/2002/3.1) is the submission of an updated FSAR after the first 20 years of continuous operation. The actual report (as submitted) was prepared based on the synopsis contained in the quality assurance programme for the preparation of an FSAR after the first 20 years of continuous operation (PLNB J63, item 125), which was approved by SONS (Reg. No. 13251/2001). The synopsis, content and the scope of the FSAR revision are based on NRC RG 1.70 (Rev. 3), with due adjustments to account for the specific technological and configuration differences in the Dukovany NPP, which comprises WWER 440/213 reactors. All adjustments are negotiated and documented in formal agreements with SONS.

NRC RG 1.70 (Rev. 3) is essentially used as a directive for contents, structure and depth of the FSAR revision. Similarly, after 30 years of operation, a new revision of the safety report must be submitted to the regulatory body for approval as the first precondition for an operating permit renewal.

(b) SONS rulings

SONS R-SÚJB 24237/2005 conditional ruling for Unit 1, valid until 31 December 2015, contains the following requirements that must be met before the onset of the LTO:

(a) To submit to SONS the following documentation:
   (i) An FSAR revision, reflecting the changes that occurred during the previous years of operation and including a summary of the residual lifetime assessment of components and systems essential to safety;
   (ii) An updated list of all added or modified SSCs essential to nuclear and technical safety.
(b) To complete the instrumentation and control systems upgrades within the scope of modules M1 and M2 based on the conditions contained in SONS ruling Reg. No. 12040/3.2/2001. Up until the installation date of the I&C upgrades described in modules M1 and M2, the licensee will assess and analyse non-conformances in these systems (T1) and their reliability in accordance with ČSN IEC 50 (191) and ČSN IEC 60605-4 and it will inform SONS of the results (T2).
(c) To submit the safety documentation associated with the I&C upgrades and with the supply of the selected type of fuel. This will contain safety analyses based on the verified set of input data. The analyses must be performed using codes validated by data obtained from operational measurements within the unit and/or by data collected in suitable R&D experimental programmes. Results must agree with the evaluation of analyses in Chapter 15 of the FSAR (Rev. 2) for Unit 3. This effort must be completed and approved before the I&C upgrades are commissioned and put into operation.
(d) To update the Level 1 and Level 2 living PSA in relation to the SSC design changes, to update all PSA documentation in five year intervals and to inform SONS of the analysis results.
(e) To conduct a PSA of the plant in operation and inform SONS of results on a quarterly basis.
(f) To submit to SONS the dataset used for the assessment of operational and safety indicators. The scope, dates and form are provided by SONS Regulatory Letter No. 26020/2005.
(g) To further develop an accidents control programme that includes the handling of severe accidents and inform SONS of the test results on an annual basis.
(h) To carry out PSR by first submitting the PSR proposal containing the contents and scope of the assessment to SONS, and then provide SONS with the results.
(i) To submit its LTO strategy proposal. The strategy will be based essentially on IAEA documents and on internationally accepted practice.

Similar rulings will be issued also for the remaining NPP units.

(c) Periodic safety review

The third legislative requirement is the PSR submission prepared in accordance with the methodology in SSG-25 [1]. The review is divided into 14 sections, which contain the analysis of the safety factors. From an LTO viewpoint, the important sections address actual condition of the essential SSCs and ageing.

A PSR main objective is to determine whether ageing in an NPP is being effectively managed so that the required safety functions are maintained, and whether an effective AMP is in place for future plant operation. The review focuses on the following safety factors:

— Programme policy, organization and resources.
— A documented method and criteria for identifying SSCs covered by the AMP.
— A list of SSCs covered by the AMP and records that provide information in support of the management of ageing.
— Evaluation and documentation of potential ageing degradation that may affect the safety function of SSCs.
— The extent of the understanding of the dominant SSC ageing mechanisms.
— The availability of data for assessing ageing degradation, including baseline data, operating and maintenance history.
— The effectiveness of operational and maintenance programmes in managing ageing of replaceable components.
— Acceptance criteria and required safety margins for SSCs.

— The programme for timely detection and maintenance to properly manage the ageing of replaceable components.
— Awareness of the physical condition of SSCs, including actual safety margins and features that would limit service life.
— Managing the ageing of SSCs important to safety, which requires the age related degradation of the SSCs to be controlled within defined limits. Effective control of ageing degradation is achieved by means of systematic ageing management processes consisting of the following ageing management tasks, based on the understanding of SSC ageing:
  - Operation within operating limits with the aim of minimizing the rate of degradation;
  - Inspections and monitoring consistent with the applicable requirements with the aim of timely detection and characterization of any degradation;
  - Assessment of observed degradation in accordance with appropriate guidelines to assess integrity and functional capability;
  - Maintenance (repair or replacements of parts) to prevent or remedy unacceptable degradation.

(d) SONS criteria for long term operation

SONS imposes various requirements on the operator of an NPP applying for licence renewal. These requirements follow IAEA safety regulations in the area of ageing management. They take into account the conclusions of the IAEA's SALTO reviews, and the implications of a comparison of the requirements of the Czech and US legislation for LTO of NPPs, particularly 10 CFR 54, NUREG-1800 [13] and the GALL Report [8]. The criteria deal with the physical impact of ageing on SSCs with long design lifetime. The obsolescence issue, for example, must be resolved in a PSR review to support an LTO application. The criteria stipulate rules for the following activities:

— Selection of SSCs;
— Assessment of the acceptability of LTO of selected SSCs (assessment of appropriate ageing management);
— Assessment of TLAAs;
— Assessment of current ageing programmes.

The analysis of LTO requirements in the PSR and in the FSAR, and the requirements obtained from SALTO, often overlap. Generally, it is possible to state that the criteria for LTO from extrabudgetary SALTO reviews are more modern (they follow a tested logical procedure for documentation), and therefore it is possible to consider a replacement or modification of the original criteria for LTO in the PSR. At any rate, it is possible to demonstrate that the original safety factors in section 4 of the PSR are encompassed by the SALTO procedure and listed criteria.

3.3.3. Handling of design and licensing changes in France

In France, the approach to obtaining an LTO operating licence is primarily based on the PSR system, which offers the opportunity to both examine, in depth, the installation conditions and to check that it actually complies with all the applicable regulatory requirements and provisions (conformity check). Another objective of a PSR in an LTO context is to improve the safety level of the operating unit, aligning it as much as possible with the most recent requirements applicable to newer installations with higher safety objectives, taking into account the latest developments in national and international expertise and OE (safety reassessment).

Periodic safety reviews cover all the risks or drawbacks the installation may present in terms of safety, public health and environmental protection, including radioactive waste and releases. Within this PSR framework, the decision to authorize LTO is likely to require addressing issues covering a timeframe longer than ten years, although, when the period of extended operation is entered, PSRs continue to be conducted every ten years.

The focus is on three major issues:

(a) Monitoring the reactor vessel integrity and strength;
(b) Monitoring steam generator maintenance and replacement;
(c) Monitoring and maintaining a high level of containment integrity.

In the future, reactors in operation will run alongside the more recent third generation models, designed to a significantly higher level of safety. This raises the question of prolonging operation of older reactors beyond their 40 year design life, when a safer and more current technology is available.

In light of the Fukushima Daiichi accident, safety requirements associated with LTO have become even more stringent than they already were. In particular, when comparing the LTO candidate unit safety level to that of the European pressurized reactor type reactors or their equivalent, taking into account lessons learned from the Fukushima Daiichi accident, these added requirements have produced proposals of design changes for LTO, triggering significant improvements to reactor safety.

The French Nuclear Safety Authority considers that any life extension could only be contemplated if it is associated with:

— A proactive and ambitious upgrade programme to improve the installation's safety by an order of magnitude, an amount far greater than the continuing improvements resulting from the PSRs. R&D work in France and elsewhere has already suggested improvements that would provide significant reductions in radioactive releases in case of severe accidents.
— Demonstration of a strict compliance of the reactors with the applicable regulations.
— An adequate management of ageing and obsolescence programme. As far as these efforts are concerned, the French Nuclear Safety Authority expects far reaching proposals from the licensee.

3.3.4. Handling of design and licensing changes in Hungary

Regulatory and licensing matters related to the siting, construction, commissioning, operation and decommissioning of NPPs are within the remits of several different authorities in Hungary, as describing in the following:

— Nuclear regulation, licensing and regulatory oversight are within the responsibilities of the Hungarian Atomic Energy Authority (HAEA), as stated in the Atomic Energy Act CXVI, 1996;
— Environmental protection aspects of the use of nuclear energy are within the realm of the National Inspectorate for the Environment, Nature and Water, as stipulated by the Environmental Protection Act LIII, 1995;
— Production and commercial aspects of power generation are within the responsibilities of the Hungarian Energy Office, on the basis of the Electric Energy Act LXXXVI, 2007;
— Health physics aspects are under the responsibility of the National Public Health and Medical Officer Service, on the basis of a government directive.

In addition, several government and ministerial directives have included regulations related to NPP operation. The nuclear safety regulation is the most significant of these for the handling of design and licensing changes.

The Atomic Energy Act was passed in 1996. It was prepared in accordance with the most advanced international standards and principles and it complied with the requirements and recommendations of the European Union, the OECD/NEA and the IAEA. The Act came into force in July 1997. It reflects, primarily, the importance of the independence of the regulator, and it transfers the regulatory responsibility of nuclear facilities to the HAEA.

The HAEA Nuclear Safety Directorate has defined a pyramid of legal documents (listed below) that are applicable to nuclear safety; complying with the first three levels is mandatory:

— The Atomic Energy Act;
— Government directives;
— Nuclear Safety Regulations, in a recent edition in ten volumes;
— HAEA regulatory guidelines;
— Internal documents of the HAEA and the licensee.

The Atomic Energy Act and subsequent government directives (Government Decree 118/2011 on the nuclear safety requirements of nuclear facilities and related regulatory activities, is currently in force), form a legislative system covering all the basic issues that need to be addressed with any modification to the configuration or to the licence.

The mandatory safety requirements for operating NPPs are presented in the first four volumes of the Nuclear Safety Regulations, issued as annexes to Government Directive 118/2011. These cover:

— Vol. 1, Regulatory Procedures for NPPs;
— Vol. 2, Quality Assurance;
— Vol. 3, General Requirements for the Design of NPPs;
— Vol. 4, Operational Safety Requirements for NPPs.

The safety requirements are in accordance with those issued by the IAEA. Most of the guidelines issued by the HAEA are based on international standards, in particular IAEA publications and on the World Association of Nuclear Operators recommendations.

The Atomic Energy Act requires the regulations to be updated in line with developments in technology and science, operational experience and safety research. All key international conventions were ratified and included in the national legislation.

Any planned changes or component replacements that impact the current configuration (licence) have to be performed according to the legislation system previously described.

The safety reviews, for example, the PSR or any potential abnormal event during operations, and the related lessons learned, could be the source of configuration modifications. In these cases, the most important regulation is Government Directive 118/2011. After completing configuration changes of this kind, their description, justifications analysis results and other implications on safety and reliability have to be input into the FSAR, which is updated annually.

3.3.5. Handling of design and licensing changes in India

In preparing the PSR for LTO authorization, all the agreed upon design changes and lessons learned from the Fukushima Daiichi accident were taken into consideration and particular emphasis was put on ageing and plant life management related findings.

Major findings from the owners and utilities assessment of existing NPPs in India were that they:

— Meet regulatory requirements for external events;
— Have sufficient provisions to ensure core cooling in absence of off-site and on-site power supplies;
— Have sufficient on-site water storage for decay heat removal;
— Implement procedures to handle external events, loss of ultimate heat sink and station blackout.

Gaps were identified for review and design change. The following list includes ways to address these gaps:

— Incorporate an automatic reactor trip in all NPPs for seismic events;
— Strengthen the provisions for beyond design basis accidents;
— Increase on-site water supplies;
— Procure and install additional diesel generator sets for charging batteries and running small capacity pumps;
— Incorporate provisions for containment safety and fuel pool safety;
— Formalize and implement severe accident management guidelines.

The magnitude of postulated design basis natural events and the related requirements for siting and design of NPPs, as specified in AERB safety regulations, are appropriate and sufficiently conservative. However, in the light of the Fukushima Daiichi accident, it is prudent to further enhance this conservatism and also to postulate beyond design basis natural events.

Additional measures incorporated following the Fukushima Daiichi accident include:

— Tie-in points outside the reactor building to connect to an external water supply to inject water into the core from outside the reactor building;
— Guidelines made available to all NPPs on the use, quantification and incorporation of safety margins with respect to earthquakes and external flooding;

— Early tsunami warning system to be made available at all coastal NPPs;
— An automatic reactor trip on seismic events at all NPPs (this already exists at two nuclear generating stations);
— The capability of all NPPs to cope with extended station blackout conditions;
— Guidelines on how to handle beyond design basis accidents and completion of a comprehensive severe accident management guidelines document.

In parallel, the implications from a comparison of the plant conditions with that of modern safety requirements should be analysed and, in turn, the unit safety justified to the licensing authority.

The regulatory body in India completed special inspections of all NPPs in order to assess their capability to deal with natural events and station blackouts.

3.3.6. Handling of design and licensing changes in China

In China, two nuclear safety requirements published by NNSA in 2004 are regarded as fundamental safety regulations:

— HAF 102, Design Safety Requirement for Nuclear Power Plants;
— HAF 103, Operation Safety Requirements for Nuclear Power Plants.

General safety requirements, including design and licensing changes, are stipulated in the two regulations. They are the equivalent of IAEA requirements Safety of Nuclear Power Plants: Design, SSR-2/1 and Safety of Nuclear Power Plants: Commissioning and Operation, SSR-2/2, respectively.

Modifications for NPPs are regulated by the following procedures:

— Procedure for SSC modification;
— Procedure for operational limits and conditions modification;
— Procedure for instruction and control system modification;
— Procedure for organization modification.

Any proposed modifications that affect the configuration on which the basis of the operating licence was issued needs to be submitted to NNSA for approval prior to implementation. Any modifications involving plant configuration and the operational limits and conditions need to conform to the safe design requirement stipulated in HAF 102 and, in particular, the modifications should preserve the plant capability to perform all safety functions without degradation. After the modification, all related documents used for operation are to be updated in a timely manner to reflect all changes. Any organizational modifications, if related to safe operation, are to be submitted to NNSA for approval.

During the conduct of a PSR, weaknesses in safety may be identified. In this case, the operators need to make improvement commitments to the NNSA. These commitments may cause changes of the design and licensing bases. In this case, the FSAR needs to be expeditiously updated and approved by NNSA to reflect the changes.

Changes of operational limits and conditions may have an impact on ageing management. The NPP AMP team should evaluate such effects. If necessary, an assessment or a qualification, such as a TLAA or an environmental qualification reconfirmation, should be carried out, reviewed and approved for specific SSCs.

3.3.7. Handling of design and licensing changes in the Republic of Korea

Various design and licensing changes may be implemented throughout the operating life of NPPs to improve safety and reliability of SSCs. They may involve adding, deleting or modifying the SSCs, and changes may also affect the plant configuration management. All documentation related to design and licensing changes should be updated to address not just the change itself but also what impact the changes will have on the current operation and maintenance practices.

For the LTO period, in preparing the licensing application, most of the design and licensing driven changes implemented during the NPP design life are reviewed in terms of whether they may impact ageing management for the period of continued operation. Any impact on ageing management during the period of continued operation is recorded and incorporated into the AMP as part of the LTO application. In order to maintain a current AMP, design and licensing changes are included as part of the ongoing review process.

3.3.8. Handling of design and licensing changes in the Russian Federation

Design changes in NPP units that aim at eliminating or minimizing the effects of ageing mechanisms are usually the result of complex analyses and of R&D conclusions. Design changes can also be the result of feedback, both positive and negative, from OE in similar units. Design changes are implemented taking into account the recommendations from international organizations (including the IAEA), and lessons learned from events and accidents around the world. These usually require preventive and compensatory measures in order to remove weaknesses and avoid analogous situations in the future.

For example, as a result of experience accumulated from the operating steam generators PGV-1000 and PGV-1000M in WWER-1000 units, some abnormal hydrodynamic regimes deviating from the design were observed in the heat exchange section, and increased damage to the steam generator pipe nozzles was observed. High stresses (including residual stresses) in the piping joints were noted. As a result, the steam generator design was altered, and these changes either reduced or completely eradicated the degradation phenomena, thereby increasing steam generator reliability and serviceability. The following design improvements were implemented:

- Installation of deflection plates in the upper part of the steam generators in order to eliminate local steam breakout (see Fig. 28);
- Provided coverage to some sectors of the down comer circulation channels in the heat exchanging tube bundle region to eliminate the steam–water mixture breakthrough and blow out;
- Increased by 8% the number of perforations in the perforated plate in order to improve hydrodynamics (see Fig. 29);
- Changed the position of the flow rate sensors on the level equalization volumes in order to increase data stability and reliability and to correct the readings;

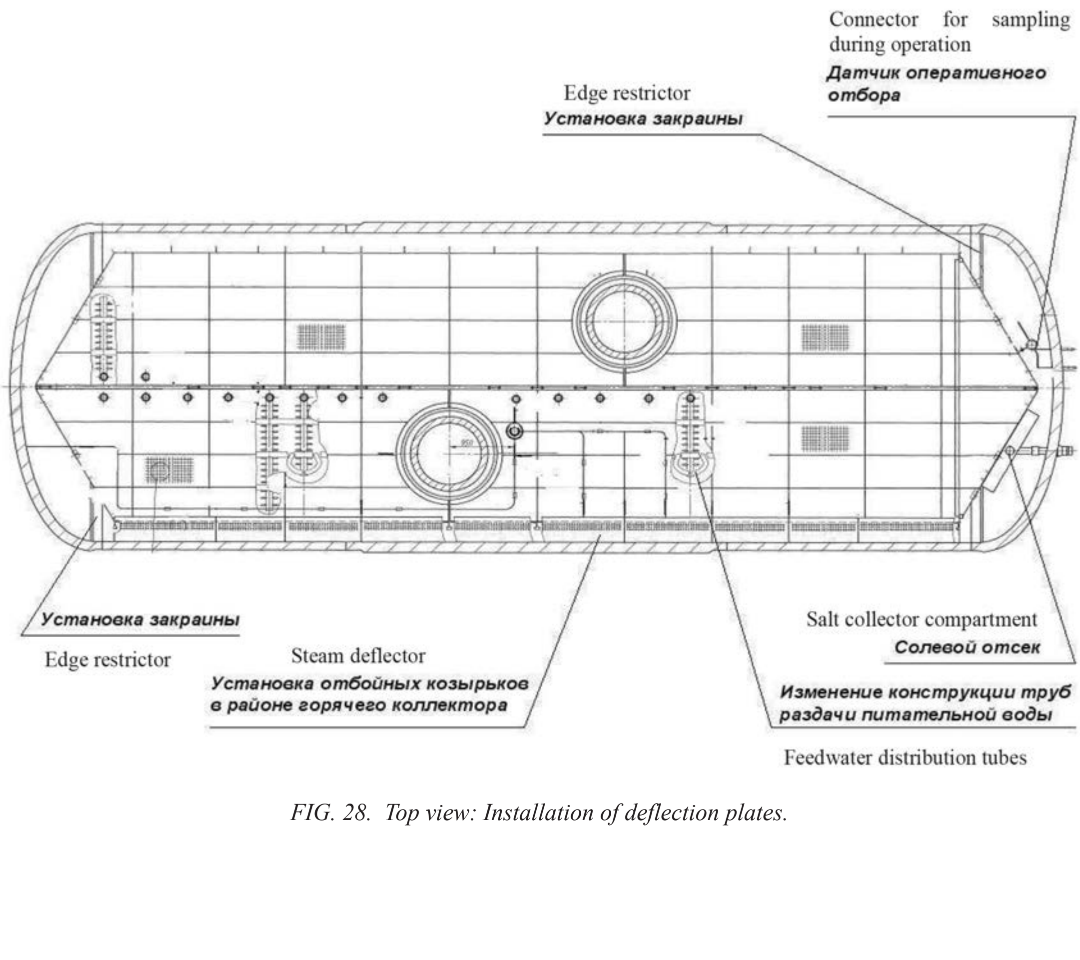


FIG. 28. Top view: Installation of deflection plates.

— Reorganized the brine collecting space and improved the feedwater and blowdown systems (including the location of the sample points) in order to decrease the impurity level (salt concentration) in the maximum thermal stress regions;
— Installed a chemical reagent distribution system in the steam generator to implement chemical cleaning during cool down to remove sediments from the exchange surface and to decrease the accumulation of corrosion deposits (see Fig. 30);
— Installed support sleeves and cover plates, each made of two parts, to decrease thermal stresses in the welded joints;
— Installed cleaning and inspection nozzles in the lower steam generator part and in the heat exchange tube region. Each nozzle will be sealed with expanded graphite sealing (see Fig. 30);
— Cut out window openings in spaces situated below the heat exchanger tubes;
— Sealed all primary and secondary flanges (including manholes) with expanded graphite sealing (removed all nickel based sealing) and modified the sealing surfaces (see Figs 29 and 30).

With the above modifications, the steam generators of model PGV-1000 (1000M) reliably supply steam with the required characteristics and meet the specified design parameters.

An example of a WWER-1000 design change to improve emergency response is the implementation of a boric acid solution addition in the ECCS hydro-accumulators heated to 55–60°C. This system was not in the original design. It was designed and installed in operating units to decrease the thermal shock to the pressure vessel base metal during the passive response of the ECCS to design basis and beyond design basis accidents.

Changes driven by licensing rules and implemented in operating units have one main goal in common: the improvement of unit operation reliability and safety based on modern safety requirements. An example of these changes for units designed between 1960 and 1970 is the adoption of the LBB concept for large diameter pipelines of the main circulation circuit. This is considered an improvement factor of great significance for the licensing terms of older NPP units. Implementation of the LBB concept provides operators with time to take appropriate measures in order to prevent loss of coolant accidents and to protect the third physical barrier from the release of ionizing radiation and radioactive materials to the environment. The successful use of the LBB concept allows for the avoidance of massive design changes to protect against pipe whip, jet impingement, and other dynamic effects following large loss of coolant accidents with guillotine failure of the high energy primary reactor coolant piping.

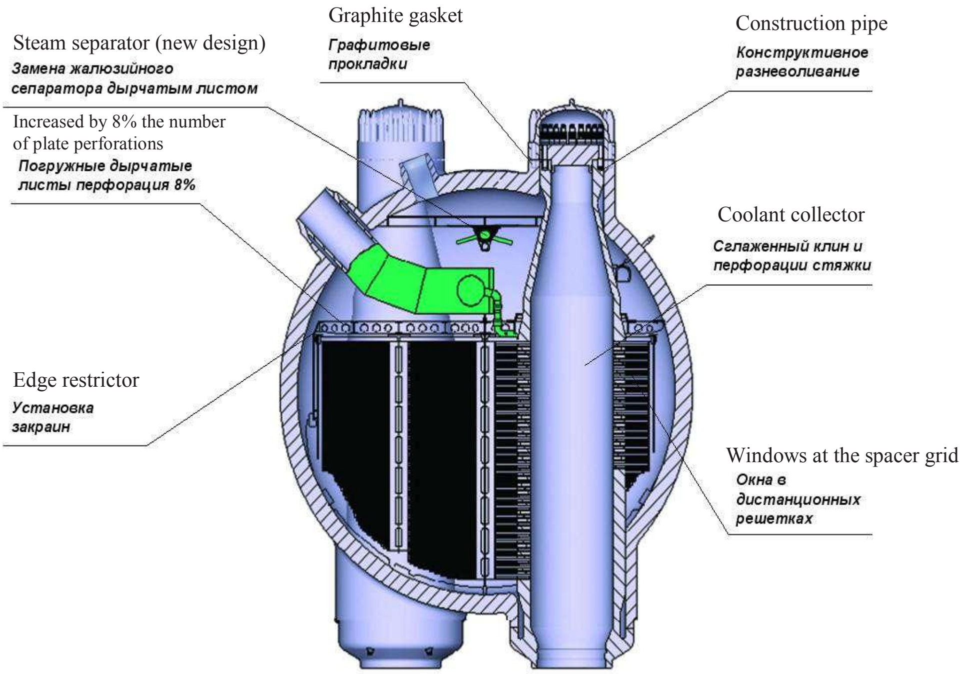


Рис.2

FIG. 29. Perforation increase and feedwater pipe modifications.

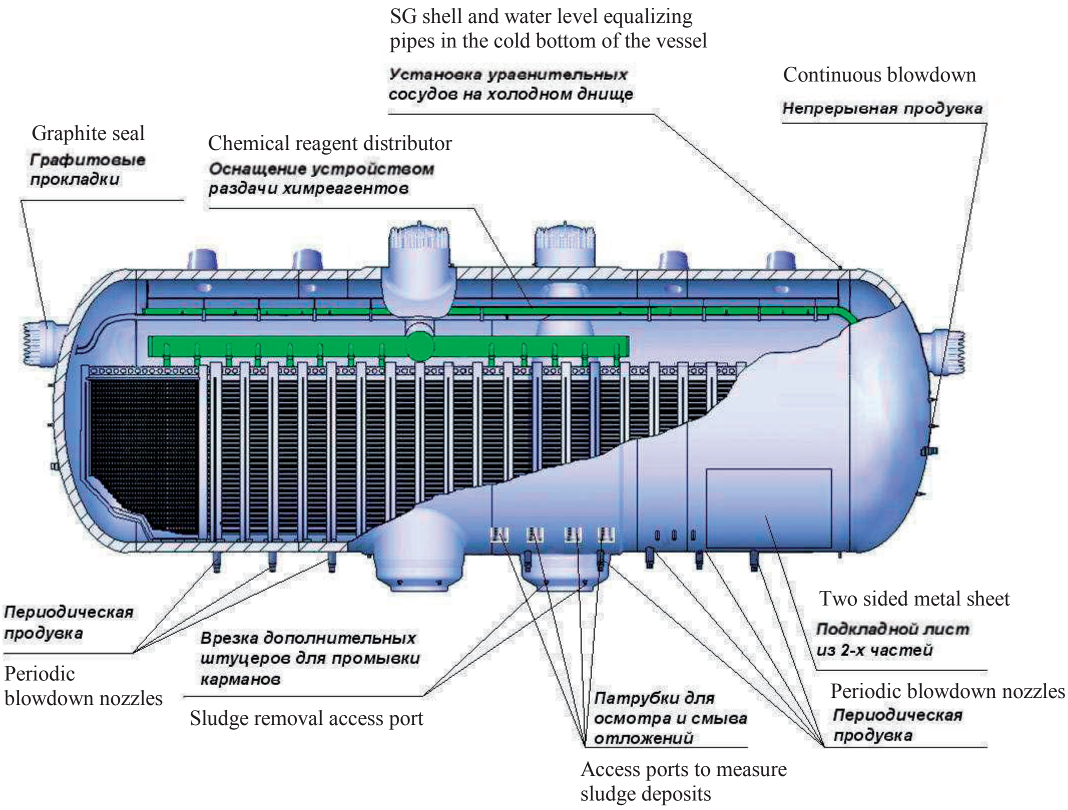


FIG. 30. Modifications to the steam generator chemical reagent distribution system and expanded graphite sealing.

Retrofitting nuclear power units with supersensitive leak detection instrumentation on the primary coolant allows the early detection of leaks long before the leak point turns into a crack and reaches its critical length. This technique allows time to safely shut the reactor down, make a pipeline repair or replace the leaking section, and by doing so, excludes having to design for sudden pipe ruptures. For welded joints (weld metal) in the most loaded pipelines, experimental ultrasonic inspection tools with increased sensitivity levels are being developed using phased array technology. This will greatly enhance the implementation of the LBB concept and the justification for finding the likelihood of sudden gross guillotine failures in high energy large bore reactor coolant pipes.

Following the Fukushima Daiichi accident, great attention has been paid to requalifying safety related systems and components, in particular, for earthquake scenarios. Seismic qualification requirements are contained in the Russian regulatory documents for NPP life extension. Currently, all NPP sites are being re-evaluated to lower probability and higher intensity earthquakes producing higher ground motions and related effects. In most cases, such revisions suggest an increase in the licensable plant design basis criteria. For example, for the Novovoronezh NPP, the region MP3 was increased from 4.9 to 7 points in the Medvedev Sponheuer Karnik macroseismic intensity scale MSK-64 scale and, as a result, revisions to the NPP component seismic qualification are being performed. For safety related components that were never seismically qualified in the older units, design changes are also being performed. These changes usually include installation of additional supports and mechanical and hydraulic dynamic snubbers, among other things. These extra measures provide increased reliability for safety and safety related systems so as to meet the new more severe design basis events.

3.3.9. Handling of design and licensing changes in Spain

Changes to the design and licence basis are analysed to ensure that AMPs and TLAA both include and justify them during LTO.

Analysis of the impact of design changes on ageing management may be done as part of the configuration control process, or be managed, for example, by the ageing management coordinator. This is the case for the SMG plant.

Changes to the licence are reviewed at least every two years. Those related to ageing management or LTO are evaluated through the scoping criteria. For those changes that meet the selection criteria in 10 CFR 54.4, an ageing management review is performed and new scope added to the AMPs.

With respect to active equipment, the maintenance rule scope is periodically updated as changes in safety functions are analysed. These changes are considered in the review of the licence scope.

Modifications are reviewed by the ageing management coordinator to determine whether they have an impact on the AMP. If they do, the new scope is defined for the AMP or TLAA, where a thorough review of the modifications before implementation can improve the solution. For example, a design review and a walkdown can prevent installation of new safety related equipment in non-safety areas and protection can be requested for the safety equipment. If this is not possible, equipment in these areas should be added to the ageing management scope, as their failure could affect the newly added safety related equipment in the area as stipulated by 10 CFR 54.4(a)(2).

Changes in safety classification are treated as changes to the licensing basis. As a result of the Fukushima Daiichi accident and related stress tests, some changes to the licensing basis may be introduced. For example, if new requirements for station blackout are introduced, then new components may be incorporated and some of these may meet the US selection criteria in 10 CFR 54. Therefore, once the analyses have been completed, a review of the conclusions and recommendations will be undertaken to capture, if required, a new scope for the AMP.

3.3.10. Handling of design and licensing changes in the United States of America

Since NPPs are subject to ongoing design and licensing basis changes throughout the operating term, the AMPs and activities also need to be modified and updated as part of the change process. When SSCs are added, deleted or modified as part of the change process, the impact of the change on the ageing management activities also need to be addressed. The modification process should include consideration of the operating term, maintenance and inspection requirements during the operating term and ageing management activities for the SSC. Once a plant has decided to operate beyond the original design life, the modification process should incorporate the new operating term (e.g. 60 years) into the basic assumptions for subsequent modifications. This will ensure that design and licensing basis changes include appropriate consideration of ageing management activities for the remaining operating term.

In the case of adding new SSCs to an operating plant, there may be a need to create new, or modify existing, AMPs as part of the modification process. This is best handled during the development of the modification package so that when the new SSCs are installed, the new or modified AMPs are also put into place. The consideration of LTO during the design and licensing basis change process is critical to the success of LTO.

In the case of changing the licensing basis for an operating plant, there may also be a need to create new, or modify existing, AMPs as part of the change process. For example, if the function of an existing SSC is changed (e.g. non-safety function changed to safety function), but the SSC is not modified (e.g. no physical equipment change, just document changes), a mechanism needs to be in place to ensure the appropriate new or modified AMPs are addressed as part of the licensing basis change. Without adequate change management controls, the licensing basis change could result in inadequate ageing management for LTO. One method to ensure controls are in place is to require that licensing basis changes be implemented using the design change process discussed above. However, if licence basis changes are handled outside the design change process, procedures need to be in place to include the impacts of AMP due to the licensing basis change.

In summary, safe and reliable LTO is dependent on an adequate design and licensing basis change process that considers the need for new or modified AMPs. These change process controls are essential to the success of LTO.

3.4. BEYOND DESIGN BASIS ISSUES

3.4.1. Beyond design basis in Canada

In response to the Fukushima Daiichi accident, the Canadian Government required:

- Ongoing environmental monitoring on Canadian territory from coast to coast.
- Deployment of experts to the IAEA.
- Issuance of a directive pursuant to the regulatory requirements to all major nuclear facilities to review initial lessons learned.
- Re-examination of the safety cases with a focus on:
  - External hazards and measures to prevent or mitigate severe accidents;
  - Emergency preparedness;
  - Implementation of immediate short term and long term measures.

At the regulatory level, the CNSC carried out reviews of the safety status of the plants in Canada and abroad. The CNSC site staff carried out focused inspection on seismic, fire, flooding, backup power, hydrogen igniters and passive recombiners, in addition to ongoing inspections against external hazards. The CNSC conducted inspections of spent fuel bays, their components and equipment, heat sinks and alarms, as well as the availability of on-site and off-site resources.

Another area of review has been a thorough verification of the defence in depth strategy and of the measures to minimize the frequency of abnormal operation and failures, the control of abnormal operation, the detection of failures, the capability to limit the progression of an accident to within the design basis, the capability to control severe plant conditions (severe accident management guidelines), and to mitigate radiological consequences (emergency management) [19–23].

The industry, on the other hand, established a working group under the CANDU Owners Group to exchange information and define response strategies. The working group was tasked with verifying station capability to mitigate conditions during beyond design basis events, including station blackout conditions, internal and external flooding events, other events concurrent with a seismic event and environmental monitoring and reporting. The working group, which included the CANDU designer Atomic Energy of Canada Limited and the engineering group CANDU Energy, issued its conclusions and recommendations regarding the application of the lessons learned from these reviews and a recommended plan to incorporate design and administrative improvement.

For the station blackout, CANDU features an automatic reactor shutdown system, efficient passive natural circulation convective cooling, and two independent and diverse sources of back-up power. In terms of external hazards, all sites have been assessed for a range of concurrent hazards, including flooding, severe weather and seismic activity.

All sites in Canada are located in zones of low seismic activity. Every site has its own specific seismic hazard assessment and plant designs have been re-evaluated with respect to seismic events, and confirmed to be seismically robust. In terms of the fuel pools, they are seismically qualified, double-walled pools. The spent fuel is removed routinely into dry storage to minimize bay inventory, and on-line fuelling minimizes heat load in the fuel bay. All plants possess several diverse means of adding water to fuel bays. It was concluded that the reactors have been designed to withstand external hazards.

In terms of emergency management, both the on-site and off-site response have been found to be adequate. On-site, the individual stations have a multilevel response, which ensures adequate resources and communication with the province and municipality, and adequate severe accident management guidelines. Off-site, mechanisms are in place to ensure good collaboration between municipal, provincial and federal emergency management organizations, and efficient decision making processes to guarantee adequate actions are taken to mitigate impacts.

However, a number of follow-up actions to strengthen nuclear safety against extreme events, their combinations, and the response capability of the plants, were recommended, and their implementation has been translated into short and long term action items for the individual plants.

The implications of the changes to increase the robustness of CANDU plants, and their response capabilities to extreme events on the PLiM programmes and LTO, inevitably need to be made part of each NPP life management programme. Any design modifications or additions of new mitigation equipment or systems, regardless of whether they are active and in-service or inactive and kept in storage for emergency usage only, need to be adequately maintained and ready for service at all times. They may be of a non-safety grade and some may even be of a mobile nature. They nevertheless need to be submitted for review to the PLiM specialist or person responsible for maintenance, surveillance and inspection in the plants, so that the equipment may be incorporated into the maintenance inspection and surveillance programme.

Systematic scrutiny is applied at the time of licence renewal, particularly for an LTO application to the plant life management and maintenance, surveillance and inspection programmes, to verify that severe accident equipment, and system and component storage facilities, have been properly included.

3.4.2. Beyond design basis issues in the Czech Republic

The AMP must fulfil the Czech Atomic Law requirements (Act 18/1997 Coll., §4) for the preparation and execution of activities aimed at avoiding the initiation of nuclear accidents or (if already initiated) mitigating them and, hence, reducing public exposure to ionizing radiation, upholding nuclear safety, enacting radiation and physical protection, and providing emergency preparedness consistent with state of the art atomic science and technology (Law 18/1997 Coll., §17).

Accident management programme targets include:

— Preventing the initiation of emergency regimes and facilitating accident management;
— Avoiding severe accidents;
— Mitigating severe accident consequences;
— Achieving long time stable status.

The programme fulfils defence in depth requirement levels 2, 3 and 4, and to some extent level 5. Accident management procedures aimed at minimizing the risk of population exposure are continuously updated, tested and enforced. They contain operational procedures, equipment fitness for service, modifications and organizational issues, among other things.

The most important national and international guidance documents incorporated into Czech practice are the Parliament Act 18/1997, SONS decrees 106 and 195, and IAEA publications (see Refs [4, 19–23]).

The AMP's main objective is to continuously ensure an acceptable level of nuclear safety during all phases of each abnormal and emergency event. The system of prevention, management and implementation of protection from nuclear accidents combines organizational, administrative and technical measures.

It presents updated abnormal, emergency and severe accident operational procedures. The selected strategies need to reflect the current level of science and technology and need to be consistent with FSAR and PSA results. Implemented modifications are to be compatible with the applicability of operational procedures. It was necessary to implement the following modifications to enhance the capability to prevent or mitigate the consequences of severe accidents:

— External cooling of the outer RPV surface;
— Hydrogen removal;
— Continuous level measurement in steam generators;
— Continuous pressure measurement in steam generators;
— Hydro-accumulation or isolation valves modification;
— Pressurizer relief valve modification;
— Reactor cooling circuit de-aeration at operating pressure;
— Water level measurement in the space under the reactor bottom, among other things.

Development of severe accident operational procedures for non-operational regimes will remain a primary task until 2013. They will be based on PSA2 results for non-operational regimes and will be turned into symptomatically oriented procedures.

3.4.3. Beyond design basis issues in France

Following the accident on 11 March 2011 at Fukushima Daiichi NPP, the French Nuclear Safety Authority issued a decision (on 5 May 2011) detailing the specifications of beyond design basis assessments, consistent with the work of European safety authorities (Western European Nuclear Regulators Association and European Nuclear Safety Regulators Group). The main topics for evaluation, for reactors and spent fuel storage pools, were:

— External flood and flood related effects (such as hail, wind and lightning);
— Beyond design basis earthquakes;
— Loss of heat sink and electrical power supply at the site level;
— Operational management of accident situations, including serious accidents with core meltdown (reactors and spent fuel pools).

Complementary safety assessments were conducted. They have not revealed any critical failures in the expected operational nature of the systems and organizations. The few failures or threats identified on the systems are now corrected, or will be, during the first outage.

As a result of these investigations, EDF developed an action plan, structured in three phases, including the following very short term (summer 2011) actions:

— A complete review of crisis organization for managing serious accidents and a comprehensive review of all backup systems, their support systems and the fire-fighting systems needed to manage a serious accident;
— A review of the management plan for sustained partial/total power failure or fire;
— A complete review of the flood management plan concerning internal or external flood risk prevention (maintenance programme, volumetric protection and administrative procedures);
— The verification of the equipment necessary to protect from floods;
— A walkdown of the seismically qualified equipment required to manage a fire or flood at each site.

Some of the main additional short term actions suggested include:

— Installation of one emergency back-up diesel generator per reactor, capable of withstanding beyond design basis earthquakes, flooding and tornados;
— Duplication of auxiliary feedwater storage tanks on 1300 MW(e) and 1450 MW(e) units;
— Reinforcement of the spent fuel pool make-up system;
— Installation of a last resort make-up water supply.

The complementary safety assessments have also shown that, in order to enhance plant resistance to beyond design basis hazards, the volumetric protection levels need to be increased at several sites as well as the earthquake resistance of some equipment needed to face a station black out.

The medium and long term actions to be taken from 2012 onward include a medium and long term phase consisting of:

— Assessing the impact of the post-Fukushima Daiichi action plan on ongoing projects;
— Re-evaluating the need for changes in other safety requirements, particularly regarding the protection of plants against external hazards.

Modifications are to be implemented before the subsequent periodic safety review. Priority actions and modifications are expected to be completed before the end of 2018. They include the creation of a nuclear rapid response force, the creation of ad hoc electrical and mechanical connection points for mobile equipment, the installation of a last resort make-up water supply and the addition of small diesel generators.

3.4.4. Beyond design basis issues in Hungary

3.4.4.1. General issues

Lessons learned from the Fukushima Daiichi accident will probably have a limited impact on ageing management, but external events characterization may very well be affected. Whatever lessons are learned from the accident at the Fukushima Daiichi NPP, they will most likely affect the design basis of NPPs.

For LTO, the hazards from external events may need to be reconsidered. What was designed for protection against external events in the original design, and any special systems that may be added as a result of lessons learned from the Fukushima Daiichi accident, will have a specific AMP for the LTO.

3.4.4.2. Severe accident management equipment, stocked equipment

Severe accident management systems are normally classified as non-safety. Those pre-dating the Fukushima Daiichi accident, and any additional ones added as a result of lessons learned and stress tests, among other things, are usually not meant to be used during normal operation. This may create a maintenance and availability issue. When called to operate, they may not be in top working condition and, hence, fail for lack of maintenance. Since severe accident management systems are normally not safety graded, maintenance programmes tend to disregard them and they may not have an adequate AMP. Following the Fukushima Daiichi related stress tests performed on the Paks NPP, as was done in all other nuclear plants in the world, the operator decided to increase the level of protection against severe accidents (including increasing the reliability of the essential SSCs) to ensure the long term availability of electric and water supply during extreme events so as to better manage that class of event. In some cases, new components and mobile equipment may also be procured. Any newly installed severe accident systems or components, and any spare parts designated for service under extreme conditions and kept in storage, will have to be included in the station AMPs. Some equipment may have to be stored in a secure location away from the NPP, and as is the case with all stored equipment, it will not be in service during normal operation. This does not mean that such equipment should never be inspected and tested; severe accident mitigating equipment needs to be ready for service at all times. Consequently, the inspection, testing and maintenance programmes should include such equipment, their protective structures (if any) and access ways, and the plant ageing management model should recommend their inclusion in the station AMPs.

3.4.4.3. Classification

The only function of these special SSCs is that of acting as a last resort to mitigate severe accident conditions when the plant safety systems give way under extreme circumstances, before the last defence in depth feature of the plant is breached and large releases into the environment occur. Examples of severe accident mitigating systems and components could be the isolation of wall penetrations, door and window isolations and the availability of mobile diesel generators, extra battery packs, extra reserve water inventories and mobile pumping capability, among other things. Since they are not part of the defence in depth line-up, no nuclear safety classification and no redundancies are applied to such equipment. However, they may need to be classified according to their exposure to severe external conditions and this may be a complex administrative task.

3.4.4.4. Probabilistic safety assessments

PSAs may have an effect on the scoping and definition of special equipment, for example, seismic PSAs and severe external condition PSAs may be used to identify extra equipment that may have a positive impact on the overall core damage and release frequency probability figures, without necessarily having to upgrade such equipment to a safety category and safety related quality. However, existing PSAs may not need to be permanently extended to include such extra protective features.

3.4.4.5. Time-limited ageing analysis and design review

Among the TLAAs analyses required in the United States of America (e.g. NEI 95-10 (Rev. 6)), only those components with time-limited features are included and covered by the PLiM/LTO process. This is not the case in Hungary, where all TLAAs targeted in the design review, not only those categorized as having time-limited features, may be included in the PLiM/LTO process. At Paks, the LTO project also considers tasks with design aspects, and establishes a minimum set of TLAAs to be included in the PLiM studies. Therefore, although essentially based on the pre-Fukushima Daiichi, US practice for TLAA analysis, the LTO application in Hungary would nevertheless require an additional justification that the original design basis will be upheld during the extended LTO period. This means that a full design review scope would be a precondition for an LTO application, and would have to take into account the most recent requirements of the applicable standards and any inputs from all external event hazard analyses as well as a PLiM model that includes the extra equipment dictated by the lessons learned from the Fukushima Daiichi accident.

3.4.4.6. Site management of severe accidents

Equipment and stored goods used for the management of the consequences of severe accidents will need to be included in the revision of the severe accident management procedures to incorporate all lessons learned from the Fukushima Daiichi accident, and the recommendations from the stress tests on the plant.

3.4.5. Beyond design basis issues in India

Following the Fukushima Daiichi accident, the AERB concluded that the design, operating practices and regulations in India have inherent strengths to deal with natural events and their consequences. For the older BWRs, Tarapur Atomic Power Station 1 and 2, improvements to ensure continuous reactor cooling under prolonged station blackout with concurrent loss of both on-site and off-site power are being considered, as are measures for inerting the containment with nitrogen to prevent hydrogen gas explosions. A containment filtered venting feature is also being evaluated. A large number of safety upgrades have been implemented over the years based on the outcome of the safety review. These upgrades have substantially enhanced the NPP capability to withstand natural events.

Since the suboceanic faults threatening the Indian coast lie between 800 and 1300 km from the coastline, the simultaneous occurrence of strong earthquakes and tsunamis is not foreseen. Nevertheless, design changes are being envisaged to ensure that the basic safety functions of the Indian NPPs are not impaired even when faced with beyond design basis accident conditions. In Indian NPPs, submergence of the fuel in the pool is assured for one week under station blackout, and under beyond design basis conditions. Reliable back-up provisions to add water to the primary heat transport system, and operability of the fire water system are to be enhanced to ensure functionality, even during a flooding event. The class III power supply system in Tarapur Atomic Power Station 1 and 2 is to be upgraded to meet the revised maximum flood levels, and design changes to the MAPS plant are being developed to account for the upwardly revised flood level from a study of the Nicobar–Sumatra suboceanic fault.

Design provisions to mitigate a core meltdown will require mobile equipment additions as well as permanent design changes entailing detail engineering, procurement of material and component, and construction, commissioning and licensing updates.

3.4.6. Beyond design basis issues in China

In China, SSCs covered by the AMP for LTO are screened based on the principles spelled out in an IAEA publication (see Ref. [9]). Only key SSCs are selected and included in the scope of the AMP for LTO. The vast majority of the SSCs beyond those included in the scope of the AMP are managed by the general maintenance plan. Although lessons learned from the Fukushima Daiichi accident may have a limited impact on the AMP for LTO, there are some challenges originating from the analysis of events beyond design basis, especially external events, that will dictate some specific ageing programmes not previously included in the AMP.

After the Fukushima Daiichi accident, all NPPs in China carried out a comprehensive safety assessment taking into consideration extreme external events. The plants were assessed for their capability to resist:

— Extreme floods;
— Extreme seismic events;
— Fires;
— The overlapping of natural disasters (and the mitigation measures required);
— Station blackout (and the availability of additional power supply when emergency power is lost).

Specific measures for the prevention and mitigation of severe accidents were investigated and the feasibility of their deployment was assessed.

The above assessment may have no direct effects on LTO, but those components that should remain available to fulfil their mitigation functions under severe accident conditions should be qualified to beyond design basis criteria. It will be necessary to establish regulatory requirements and technical criteria, including protection from extreme external events, to ensure the equipment survivability and its effectiveness in the presence of widespread destruction, as determined by the analysis in the regions where the plants are located.

3.4.7. Beyond design basis issues in the Republic of Korea

Shortly after the Fukushima Daiichi accident, all nuclear facilities in the Republic of Korea were inspected to ensure the countermeasures against severe accidents were adequate and in full agreement with the safety inspection principles. In particular, a close examination of Kori Unit 1, the oldest reactor in operation in the country, was undertaken to confirm its safety during the period of continued operation beyond its design life.

The inspection team consisted of 56 experts in the areas of earthquake/tsunami, electric power/fire/reactor cooling, severe accident, emergency planning and LTO.

This review included:

— Confirming unit safety during a design basis earthquake and tsunami;
— Securing reactor cooling capability in case of station blackout due to severe flooding;
— Securing containment integrity, assuming loss of reactor cooling capability;
— Adequacy of the emergency response, assuming large release of radioactive materials;
— Close examination of the ageing of the Kori Unit 1 AMP.

Based on special inspections, investigations and research in the country, it could be confirmed that NPPs in the Republic of Korea will remain safe even during a massive earthquake and tsunami. Nevertheless, dozens of short and mid-term severe accident coping features have been recommended to better mitigate beyond design basis accidents.

However, these severe accident mitigation features remain outside the scope of PSRs for continued operation beyond the plant design life, since they include non-safety-related equipment. Components dedicated to severe accident mitigation do not go through ageing related degradation mechanisms because they are in a standby state throughout the NPP operating life. However, they are to be incorporated into the plant life management programmes and are to be periodically inspected and tested to ensure their availability and good performance in the unlikely event they may be required to operate. Therefore, it can be said that the Fukushima Daiichi accident did not directly impact the current regulatory rules regarding LTO in the Republic of Korea. The SSCs required to safely shutdown a nuclear reactor in case of a design basis accident remain within the scope of an intensified PSR for continued operation beyond the NPP original design life.

3.4.8. Beyond design basis issues in the Russian Federation

The primary reactor cooling system forms the third defence in depth physical barrier on the radioactive material propagation path to the environment. During normal and emergency conditions, the effectiveness of its function as a barrier depends on a wide range of internal processes including automatic and operator actions as well as external events impacting the plant. Within the bounds of NPP life extension, a Level 1 PSA is carried out to quantify the risk of a nuclear accident. The output of a Level 1 PSA is a severe accident frequency value corresponding to the probability that the reactor containment may be breached.

A set of all the relevant initiating events and their combinations is built into the NPP design basis, therefore all safety analyses of the consequences of these events on the plant are performed in response to the licensing requirements. In a PSA, several types of analyses are performed, including probabilistic reliability analysis of all normal operating regimes, of design basis accidents and also of beyond design basis accidents caused by each type of initiating event, including (fire, flood and earthquake). Initiating events are events that either directly or indirectly cause damage to the reactor core, for example events that cause reactor core damage due to failure of the reactor safety system functions or to non-compliance with the safety and licensing requirements under design basis accidents, or events that cause severe accidents (beyond design basis), that produce catastrophic damage to the reactor core.

Reactor equipment reliability is analysed using probabilistic methods of fracture mechanics, which are widely used for calculations of failure probabilities in NPP pipelines and other pressure boundary devices. Probabilistic methods allow designers and operators to evaluate the probability of cracks occurring and growing to their critical values during normal operation and emergency situations. Taking into account the defect detection probability, these methods can evaluate the probability of occurrence of all events connected with leaks and ruptures of the primary circuit. The method includes:

- Constructability analysis of the SSCs, definition of the static properties of all materials, analysis of the reliability and functional readiness of the equipment. Equipment categorization is performed according to the materials used, construction method and loading during operations (on the vessel, the cover, the studs and the welded joints).
- Analysis of the equipment loading regimes: normal operation regimes, abnormal regimes and accident regimes are grouped together, and the corresponding probabilities are calculated for each group.
- Equipment and component stress-strain analysis for all loading regimes, which provides information about the conditions that initiate and grow cracks and defects in the base metal and in the welded joints.
- A flaw parameter analysis for the base metal and the welded joints is also carried out. Flaws are detected using non-destructive inspection methods and characterized by means of parameters describing the size, shape and location within the metal matrix. However, very small cracks may not be detected by such methods. A model of the equipment is then developed, containing its actual fault pattern and probabilistic small crack distribution. This model is used for further fracture mechanics analysis to establish the fault/defect growing mode, the growth speed of the crack, and their interactions.
- Fatigue–corrosion analysis of cracks is conducted to predict the crack growth mode, its critical size, and the time when the component fracture occurs. The critical size of cracks is determined by means of fracture mechanics criteria. The analysis is performed using probabilistic strength methods, that is, the probability of catastrophic failure is calculated as the combined probability of the manufacturing cracks already existing at the time of the first installation that reach their critical size, and that of defects created during operations by the applied loads. The records of non-destructive periodic inspection with repairs of inadmissible defects are also taken into account. Reliability is determined by calculating the probability of detecting defects of various dimensions.

In each NPP unit, instructions for NPP accident management, and procedures for beyond design basis accident control and mitigation, are licensing requirements. They need to be available and approved for use. The NPP operator also needs to conduct periodic staff retraining in the educational/training centre with simulation of different accidents and different pathways of accidents, each with a controlled staff action. The construction and layout features of the facility, the functional capabilities of its safety systems, the training of its operating staff, the set of approved organizational accident response procedures and the available mitigating measures, all need to be aimed at ensuring that the probability of large scale destruction (beyond design accidents causing reactor vessel destruction, reactor cover tear-off, primary circuit steam generator collector destruction), would result in a core damage frequency below 1×10^{-7}, as required by regulation OPB-88/97.

3.4.9. Beyond design basis issues in Spain

Following the accident on 11 March 2011 at the Fukushima Daiichi NPP, the CSN submitted a complementary technical instruction, requiring Spanish NPPs to perform stress tests consistent with the work of European safety authorities (the Western European Nuclear Regulators Association and the European Nuclear Safety Regulators Group). The main topics for evaluation were: earthquakes, external floods beyond design basis, prolonged station blackout combined with loss of heat sink, emergency organization and resources, severe accident management and the management of degraded condition in the spent fuel pools.

As a result of the stress test analysis, commitments for changes and modifications that go beyond the plant design basis were made. The commitments are grouped into three categories: short term (2012); medium term (2014); and long term (2014).

Other activities related to beyond design basis have been the analysis and the commitments made to improve management of large fires, and improved explosion prevention capability. These activities, and the follow-up commitments, were in response to a CSN complementary technical instruction.

In Spain, ageing management for LTO is defined in the integrated ageing management assessment plan. The scope of this plan is based on the requirements of 10 CFR 54.4. Typically, equipment that does not meet 10 CFR 54 is not included in AMPs; however, the vast majority of such equipment is covered by maintenance plans.

Equipment introduced as a result of recommendations issued by the stress tests may not meet scoping criteria for ageing management. They should, however, be tested and maintained, even if they are in storage, to guarantee (until the end of the LTO period) their functionality when required to operate under severe accident conditions. Even though stress test results may have no direct relation to LTO, new complementary technical instructions can originate a new licensing basis. If that happens, the AMPs should be updated to reflect the changes.

At SMG, in addition to the integrated plan, a PSR and a design review were conducted in order to apply for an operating period beyond the plant design term. Long term operation was issued by the CSN, and new requirements were added for the release of a new operation permit for LTO. Some of them led to modifications that went beyond the previous design basis. Some examples are:

- Seismic qualification of the fire protection system;
- Installation of a new standby gas treatment system designed to the latest code for nuclear air and gas treatment;
- New cables and trays for division A to fulfil NRC Regulatory Guide 1.75, Criteria for Interdependence of Electrical Safety Systems.

3.4.10. Beyond design basis issues in the United States of America

The accident at the Fukushima Daiichi NPP highlighted the need for NPPs to be prepared for beyond design basis accidents. The equipment needed to respond to such beyond design basis accidents may include some non-safety-related equipment, which is outside the scope of the traditional PSR, LRA, maintenance rule, or other ageing management evaluation processes. Such non-safety-related equipment should also be subject to ageing management in order to ensure proper LTO functioning during the period of extended operation.

For example, non-safety-related SSCs (e.g. portable pumps, piping spool pieces, hoses and electrical cables) that are stored for long periods of time in warehouses and storage buildings, for example, may require periodic maintenance or inspection (e.g. lubrication changes, periodic rotation of active pumps, periodic performance testing) to ensure proper function during LTO. If these ageing management activities are not controlled, evaluated and updated based on OE and lessons learned, they may not be implemented sufficiently to support successful LTO. This may require some additional administrative controls and evaluation activities including scoping, screening, ageing management reviews and the identification of AMPs necessary to maintain the intended function of the beyond design basis SSCs.

4. TECHNICAL ISSUES IN APPLYING PLANT LIFE MANAGEMENT FOR LONG TERM OPERATION

4.1. TECHNICAL TASKS

In preparation for an LTO permit application, an ageing management review is usually conducted to establish the current state of critical SSCs and their usage factor for fatigue assessments, in order to determine their fitness for prolonged service to the end of the LTO permit duration.

The PLiM analyst needs to look into the past history of the SSCs, making use of all available records, including those generated by on-line monitoring and diagnosis systems, wherever available. On-line monitoring systems, if selected and set-up for the purpose, can provide a precise record of any deviations from the SSC technical specification by recording changes in parameters and variables, such as peak values, Fourier spectra, vibration residues, critical speeds and chemistry values, among other things. Monitoring can also provide information on the ageing assessment of SSCs, such as pressure boundary leaktightness, number and entity of pressure and thermal transients and functional anomalies in components, such as inlet valves, primary coolant pumps and steam generator internals. Monitoring systems can also provide information on unaccounted stressors and interference with the functionality of systems and components, including cases such as the inadvertent introduction of loose parts. Data, in the most advanced on-line monitoring systems, are post-processed, and recommendations are automatically issued to help operators optimize the planning of maintenance activities and, in special cases, design upgrades and system improvements can be suggested. On-line monitoring systems allow analysts to follow and trend the equipment behaviour and provide meaningful data for an LTO feasibility analysis.

The information is compiled and a documentation update package, which usually includes FSAR and technical specification updates, is prepared and submitted to the regulatory body for approval.

4.1.1. Scoping and screening of SSCs for LTO

All safety related and non-safety-related SSCs, whose failure could impact safety related SSCs, should be included in an ageing assessment in an LTO application. SSCs performing active functions are normally covered by the maintenance practice in the individual Member States (e.g. the maintenance rule in the United States of America).

The ageing and condition assessment of safety related and generally passive SSCs, whose failure may have more serious safety and economic consequences, are usually subject to extensive studies to define maintenance activities or update and upgrade interventions. In condition assessments for LTO applications of slow ageing, generally passive and irreplaceable SSCs and special refurbishment programmes, detailed studies may be conducted that require specific testing and R&D support.

4.1.2. Missing reference data

In the case of missing design information or lack of traceability (origin of the data unknown) in the TLAA management or in AMPs, the remedial steps reported below may be followed:

- The missing information (design parameters) should be measured and collected at the site, deduced by calculation or obtained through the cooperation of NPP operators of similar plants.
- An analysis of any assumptions made may have to be considered. If the assumption is based on engineering judgment, sensitivity analyses should be conducted to account for potential uncertainties.

4.1.3. Considerations in ageing management

Monitoring systems should be in place to reliably follow the SSC ageing process caused by known degradation mechanisms, including:

— Irradiation embrittlement (particularly reactor internals);
— Creep;
— Corrosion (water chemistry control);
— Wear;
— Flow accelerated corrosion and environmentally assisted corrosion, wall thinning of housings and piping (on-line detectors);
— Elasto-plastic thermal deformation phenomena.

Other parameters that need to be monitored include vibration and thermal stratification (thermocouples and dilatometers). Degradation detection and mechanism analysis of the main safety related components are described in more detail in Ref. [16].

Special attention should be paid to stress corrosion cracking of steam generator tubes from the safety and economic point of view for LTO. Critical components with perhaps difficult access, such as certain RPV base metal sections, reactor supports and reactor structures and other components not included in normal ISI programmes, such as buried pipes and underground pipes, should also be included in ageing evaluation for LTO following best international practice and taking into account operations feedback, possible special design requirements and assumptions.

4.1.3.1. Environmental fatigue of key components

In the original design of second generation reactors, environmentally modified fatigue curves were not considered. The trend today is to take into account environmental effects in fatigue calculation as more and more R&D results become available in this area.

4.1.3.2. Buried and underground piping

Regulators may be concerned about possible radioactive fluid leaks from buried pipe contaminating the surrounding soil and underground water. Buried and underground piping are usually inaccessible or partially inaccessible. Detection of leaks and condition assessment of such piping is difficult. Buried pipes typically contain cooling water, service water, fire water, oil, diesel oil, gas or hazardous fluids. They can be attacked both internally and externally. Attack could be internal through the electro-chemical corrosive activity of the transported fluids, and external through aggression from the environment and through soil pressure in locations where the piping runs below access routes (e.g. rails and heavy trucks). Soil pressure on buried piping from periodic surface loads (e.g. trucks and railway passing), may change the ground characteristics, bend the pipe and deform (ovalize) its cross-section.

Tools used for underground and buried piping leak monitoring include detection of small current variations in the electromagnetic field, microphones to detect water sounds and other sonic devices, robot inspections, ground probing drilling, excavations and probabilistic calculations, among other things.

Leaks from buried piping may reach under groundwater bodies and introduce some levels of radiation into the groundwater.

4.1.4. Containment

Concrete structures play a large role in safety related systems such as containment and other economically significant NPP structures. Ageing management programmes and methodologies have been developed for NPPs and are generally well established. They are similar to those of other plant programmes such as those for electrical or mechanical components. Degradation mechanisms have been investigated and concrete inspection, monitoring and repair techniques have been developed and applied in operating NPPs and continue to be refined. Some specific issues that need particular attention are:

— Increasing requirements on concrete structures beyond what was originally envisaged due to plant life extensions, security and evolving safety requirements;
— As structures age, environmental stressors increase and may potentially impact their functionality and durability;
— The need for a routine re-evaluation of OE and research results, and consequently a periodic adjustment, as required, to the concrete AMP;
— The need for strict quality control during the construction phase of new NPPs to ensure concrete quality for long term reliability;
— Provisions for monitoring (e.g. using embedded instruments) and inspecting (i.e. accessibility considerations) concrete structures in new builds;
— Incorporating long term durability requirements in the requirements and design of new structures.

The most commonly observed form of concrete degradation is cracking. Factors of primary concern are corrosion of steel reinforcement due to carbonation or the presence of chloride ions, excessive loss of prestressing force, excessive containment leakage and leaching due to percolation of fluids. The loss of prestressing in prestressed reinforced concrete containments is likely to come from delayed strains (creep and shrinkage) in certain areas if these have been underestimated.

Containments can be monitored by a comprehensive set of special strain gauges installed in the concrete wall, and also by other redundant measuring methods, including equipment installed on a permanent basis around the nuclear island. Various monitoring methods are used to detect local deformation, possible losses of prestressing values and overall displacements occurring in the containment. For example:

— Local deformation in the containment is detected by strain gauges installed in the base mat, the gusset plate, the barrel and in the dome. Plumb lines are used for overall displacements of the structure and variations in containment diameter. Invar wires are used for variations in the height of the cylindrical wall and optical levelling for reactor building settling and slope.
— Variations in tension are monitored on a few vertical prestressing cables grouted with grease and fitted with dynamometers. The first plant unit on every site includes four cables fitted with instruments.
— Thermal variations in the structure are monitored using thermocouples installed in the same areas as the extensometers.

In most cases, plant operators adopt a quarterly metering cycle on all detectors, outside special periods, such as prestressing status verification time or containment pressure testing time. A certain number of units have adopted a monthly metering cycle. All measuring devices are checked, either in person or using automatic data acquisition devices, to ensure they are capable of transmitting all measurements to a central database via a computer. Experts then conduct analyses aimed at isolating irreversible structural deformations, such as shrinkage, creep, relaxation of prestressing cables and reversible distortions of thermal or pressure related origin.

Continuous monitoring is the ultimate tool of ageing management. It is best implemented at the very start of an NPP project and should end when final shutdown takes place. It can give an accurate picture throughout the life of the structure, including the phenomena affecting the prestressing cables, hence the residual compression in the structure, and the overall and relative settlement of the reactor building.

Pneumatic containment testing is usually prescribed in commercial nuclear power regulations. There are tests to measure the containment system’s overall integrated leakage rate under simulated loss of coolant accident peak pressure. Other tests may be required to measure local leakage rates and containment isolation valve leakage rates. For each type of test, regulators may define various intervals and when each type of test can be conducted. The ultimate goal is to assess the capacity of containment buildings to prevent the release of radioactivity during an accident and to meet the mechanical design criteria during operation.

Intervals may be prescriptive or they may vary based on monitoring performance. Apart from the inaugural tests, repeat containment leak tests may be conducted every ten or more years. Some plants today are pursuing intervals greater than ten years for leakage tests. A balance should be struck with intervals and test intensities to avoid overstressing or fatiguing the containment structures and interconnected systems and components. Specific monitoring during the test is usually conducted to evaluate the containment leaktightness and to check its performance against the design criteria and new licensing requirements. With LTO applications, regardless of when tests are conducted, licensees are required to substantiate containment integrity for the LTO period.

4.1.5. Time-limited ageing analysis

The following important tasks may be selected to verify that TLAAs are valid for the LTO period:

- Reactor vessel embrittlement analysis (irradiation and thermal);
- Pressure temperature curve derivation for the primary coolant;
- Known crack stability analysis;
- Metal fatigue analysis (including thermal stratification and RPV internals);
- Amendment of high energy line break analysis;
- Material property degradation analysis;
- Environmental qualification of electric equipment (electrical cables and I&C components);
- Concrete containment tendons prestress analysis;
- Containment liner plate, metal containment and penetrations fatigue analysis;
- Fatigue analysis of safety cranes;
- Fatigue analysis of flywheel of main circulation pump.

There may be other plant specific TLAAs to be added to the list.

High energy line break analysis as defined in the applicable codes is reviewed for LTO applications. The analysis of high energy line breaks evolved in time to encompass dynamic effects. There are two typical consequences of high energy line breaks:

- Dynamic effects, such as pipe whip, jet impingement on safety related targets and water hammer effects, such as sudden breaks of the pipe or unintended pressurization of confinement rooms or the reactor building (feedwater pipe breaks, penetration failures and main steam line breaks);
- Environmental effects in the form of flooding and spray wetting from fluid discharge from the break, possibly producing harsh ambient temperatures and humidity.

Challenged by these effects, plant systems and components need to be designed to operate and help bring the plant to a safe shutdown, as required under all operating conditions.

Protective structures may be designed using energy methods or dynamic analysis techniques, including finite element analysis. For LTO analysis, a regulator may require focus on the ECCS, water hammer analysis and break size, shape, opening area and opening time.

4.2. REGULATORY PROCESSES

State specific regulation and guidance for the submission of LTO applications should be in place before embarking on an LTO project.

In States following the licence renewal method, an application for life extension can typically not be submitted before a certain number of years (e.g. 20 years) of operation. A minimum number of operational years of experience are necessary to allow a licensee to collect enough information on the physical conditions of the SSCs and build enough knowledge on ageing mechanisms in order to confidently plan future life management activities and maintain appropriate safety margins and economic viability for the whole LTO period. In addition to the licence renewal submission work, regulators may impose updates and design changes to the plant based on lessons learned from OE feedback and new R&D findings.

States that have no limited licensing term, adopt a periodic safety review process, which is conducted every ten years. Each PSR reviews the plant configuration in light of OE, new R&D findings and lessons learned.

Some States have adopted a combination of the two methods previously described by issuing a licence covering the whole nominal design life (30–40 years), but conditional to the submission of a periodic safety review every ten years of the nominal design life. At the end of the design life, the licensee submits an application to justify LTO. This application should allow enough time for regulatory review (one to three years).

From a licensing and regulatory viewpoint, the three methods are equivalent in the end, because major safety and regulatory concerns are addressed regarding ageing management capability, including TLAA on SSCs to prove the operability of all safety critical items for the longer term. In addition, the conformity of the plant to modern standards is assessed, taking into account that modern standard requirements are:

— Not included in the current licensing basis;
— Affected by the new long term operating conditions;
— Identified as being important for safety on the basis of systematic and justified methods (i.e. cost–benefit analysis in terms of safety enhancement).

4.3. RESOURCES ASSOCIATED WITH PLiM FOR LTO

A licensee applying for an LTO permit will most likely use the services of a dedicated internal organization or specialized external contractors. In any case, the licensee should maintain full control of the project and of the interfaces. An independent review of the LRA by an external organization will align the results with international practice.

Organizational aspects of an LTO project are crucial. It is a complex undertaking and may require expert advice. Preparation for the project includes the extra housing, restoration and sanitary facilities to host the increased number of tradespeople and personnel entering the plant. Correspondingly, there may be the need for the following:

— Extra parking;
— Security devices;
— Entry points and turnstiles;
— Temporary lighting in the plant;
— Washrooms;
— Medical support;
— Laundry facilities to wash contaminated coveralls and browns and to treat wastewaters;
— Transportation and distribution;
— Cranes;
— Locker space for the increased number of workers;
— Decontamination equipment;
— Tools and machinery;
— Waste treatment;
— Off-site storage space;
— Adequate (new) equipment and bulk material storage.

Human resources will increase in an LTO outage, requiring extra training facilities to qualify new radiation workers in order to allow them safe entry into the controlled zones, and to work on pressure boundary components. Additional qualified resources will be needed to carry out emergency drills. Qualified radiation protection escorts will be needed in sufficient numbers to accompany external people and consultants. Systematic training will have to be conducted to qualify contractors' personnel and augmented internal staff.

To maintain full control over the various contractors and their interfaces, and to fulfil the LTO licensing commitments taken, an independent organization should be created from the ranks of the licensee's project management and internal technical support organization teams to oversee the transition from the initial licensing requirements to the LTO requirements (which may be very different) and to ensure the additional commitments are correctly implemented during LTO. The primary responsibility of such a team will be to ensure that the licensing commitments taken at the time of application submission are not changed or misinterpreted by the implementing organization, especially when there is a large time interval between the LTO application submission and its implementation. This requires continuity and control on the part of the licensee.

Good timing in preparing the necessary infrastructure is crucial, and its planning should be comprehensive and completed in advance of the LTO outage to allow for timely implementation.

The conclusions reached in the LTO licence are not cast in stone for the duration of the licence. The licensee should be prepared to upgrade the plant configuration if new requirements and new generic action items, or internally generated design changes, become necessary. A typical example may be the occurrence of an event such as the Fukushima Daiichi accident. Lessons learned from that accident will have to be analysed, and the licensing documentation updated in operating plants, including those with an LTO licence. The implementation of lessons learned may induce configuration changes and safety improvements to the plant. Usually, major system modifications may require the licensee to submit a design modification information package and analysis reports to the licensing authority, as well as an FSAR and technical specification updates for approval. If the licensing envelope does not change, then a reconciliation statement may be sufficient to the licensing authorities.

In other jurisdictions, during the LTO, a periodic review of AMP is conducted in different depths and degrees, depending on the circumstances. These reviews include OE feedback analysis and R&D findings evaluations. Any design modifications with safety impact are analysed in light of the current licence.

Should the original design basis of a plant be found to be incomplete, or if it has not been properly preserved prior to the operator applying for an LTO, the design documentation will have to be reconstituted.

4.4. ADDITIONAL ASPECTS

4.4.1. A State’s energy strategy

One consideration is capacity replacement, the building of new plants, versus keeping capacity in operation through LTO. Developments in technology, trends in demographics, changes in the economy and advancements of the renewable energy sector mean that an electricity production plan needs to be continually updated in order for a State to meet the needs of an ever evolving economy and shifting electricity demands, while at the same time being able to provide affordable electricity. The growth forecast will determine the new capacity required, which includes the capacity to be replaced, the capacity to be added and the implementation schedule. It is important that, as the State moves forward, spare capacity always be available to allow for some flexibility. The challenge is in choosing the right combination of generation sources and the necessary level of investment to be able to modernize the infrastructure to meet future needs.

4.4.2. Design updates

A licensee applying for an LTO licence or permit is to ensure that the original design is updated to meet the regulations that may affect plant safety that are valid at the time the LTO application is submitted. For example, new external hazard characterizations, such as:

(a) Seismic flood and increased probability and intensity of forest fires in the dry seasons.
(b) The inclusion of new loads and effects not present or not considered in the original design, such as the greater awareness of the consequences of the following:
   — Thermal stratification phenomenon;
   — New fire protection requirements;
   — New station blackout durations;
   — New seismic updates;
   — Possible presence of new dams or flood causing agents;
   — Presence of new industries, new railroads, infrastructures;
   — Tree growth in the area of the plant.

The licensee is to ensure that all changes deriving from these aspects have been implemented before submitting an LTO application.

4.4.3. Long term operation limitations

The original period of licensed operation for reactors is based mainly on economic considerations rather than the limitations of nuclear technology or the actual materials of construction. Long term operation of an NPP may be conditioned by life limiting processes and features of SSCs. It is, therefore, important that the licensee identifies all those limitations, both technical and economic, to LTO so that an informed decision can be made to extend the life of a plant. The decision to pursue LTO is usually based on an evaluation that covers strategic elements, such as:

- The need for electric power and issues concerning supply diversity;
- Regulatory requirements, including new requirements;
- A technical assessment of the physical condition of the plant, including the need for enhancements or modifications;
- The impact of changes to programmes and procedures necessary for continued safe operation,
- The results of an environmental impact and the consequent impact on the site (land and water usage) during the LTO;
- The impact of the necessary updates to the emergency procedures on the infrastructure and personnel inside and outside the plant perimeter;
- The cost of the engineering, contractual, legal and licensing documentation updates including the FSAR and operational documentations;
- A comprehensive economic assessment.

A decision to proceed with LTO is made only if the results of the activities indicated above demonstrate that the plant can be operated safely for the planned period of LTO and that it will still be economically and strategically viable to do so.

If risk informed ISI programmes are to be used for the planned period of LTO, there should be a comprehensive set of regulatory requirements for the implementation of risk informed ISI. The methodology, equipment and personnel should be qualified in their capacity in accordance with national standards, regulatory requirements and IAEA recommendations [10, 17] (for further information, see also Ref. [10]), and the qualification process should include requirements that provide a quantitative measure of the effectiveness of the ISI process.

5. SUMMARY

The LTO process varies based on the type of licence applied for in each of the Member States. For States with licences limited to a certain number of years, a licence renewal process that includes ageing reports, modernization upgrades and new licensing requirements, is required. The decision whether to seek licence renewal or life extension is normally based on the results of the plant economic analysis, which needs to include all safety and licensing requirements for continued operation.

For States with unlimited operating licences, the approach to LTO is the regular periodic safety review with extra requirements specific to ageing and safety margins. In the PSR, Member States will check whether SSC ageing is being effectively managed and whether an effective AMP is in place for the period for which the plant is intended to continue operating beyond its original design life. Certain States with specific technology and requirements may use a hybrid process with some elements of a licence renewal within the structure of a PSR process.

The Fukushima Daiichi accident highlighted the need for NPPs to be prepared for prolonged beyond design basis accidents, and has led to extraordinary safety reviews of NPPs and their safety margins in many countries. These reviews have generated safety improvements, new licensing requirements and the addition of equipment capable of increasing operator flexibility for the mitigation of the consequences of severe accidents.

When nuclear power plants reach the end of their nominal design life, they undergo a safety review and an ageing assessment of their essential SSCs for the purpose of validating or renewing their licence to operate for terms beyond the originally intended service period. Three different PLiM for LTO models were introduced to qualify these nuclear power plants to operate beyond their nominal service life. A model followed in the United States of America, and in some of the Member States, is based on the LRA concept, in which the regulator issues operating licences for up to 40 years. Beyond the 40 year limit, licences can be renewed for an additional period to a maximum of 20 years for each renewal application. Another licensing model based on the PSR process is used primarily in Europe. In this model, the NPP operating licence is virtually unlimited on the condition that the NPP undergoes standardized PSRs, usually every ten years, to confirm that it continues to meet its licensing terms and environmental conditions. The other model is a combination of elements of these two models, and is normally used in regions where the PSR model is prevalent or in countries where the PSR model needs to be complemented by additional specific local requirements.

Appendix

SPECIAL APPLICATIONS AND DEVELOPMENTS

A.1. HUNGARY

There are two basic goals set with the adoption of a new code: improving safety performance and improving cost effectiveness of plant operation. The changes to the applicable codes and standards in the Paks NPP in Hungary included the following:

- Engineering support of operation and maintenance, based on the new code which made state of the art implementation of inspection, testing and maintenance possible.
- New safety and structural analyses, which allowed for comparisons of the results against the most recent regulatory requirements in the interest of safety and performance.
- Indirect impact of safety improvements on facilitating national and international acceptance of the NPP lifetime extension plans.
- Compliance with the new code requirements, which provided the opportunity for an extension of the original ISI cycle time. This intended extension of the inspection interval has a major potential for enhancing the cost effectiveness of future operation and maintenance.

Existing procedures need to be replaced by new in-service inspection programmes. The in-service inspection programmes put great emphasis on the ageing management side of inspections. They are, therefore, best suited to effectively contribute to the objectives of the LTO programme. The selected codes and standards may also cause modifications to non-destructive examination (NDE) programmes, and revisions of the NDE framework programme and acceptance criteria documentation. This revision ensures uniformity of the engineering inspections and of NDE, and eliminates deficiencies found in the former NDE programmes.

The generic and specific requirements for the content and structure of procedures, calibration, execution, evaluation and documentation of examinations, are included in the revised NDE procedures.

The use of qualified procedures is specified for in-service NDE activities. This means the implementation of inspection qualification is not only due to the adoption of the new code, but it is also required for compliance with the current regulatory requirements. The NPP had to establish the appropriate qualification infrastructure, and work systematically on ensuring that the qualification activity meets the regulatory requirements.

Some examples of qualified NDE applications in which the component or system condition assessment and qualification involved flaw detection and flaw sizing, are listed below:

- Steam generator primary collector and dissimilar weld;
- Steam generator heat exchanger tube;
- Steam generator bolting;
- RPV welds, beltline and nozzle inner radius;
- RPV cladding;
- RPV welds;
- RPV bolting;
- Main coolant pipe circumferential and longitudinal welds;
- Pressurizer nozzle dissimilar weld;
- Main cooling pump stud.

As a result of the work described, the content of the ISI/NDE framework programmes achieve good agreement with the selected code requirements. To facilitate their application, a guideline is available that contains instructions on how to translate the results of examinations between the old and new systems.

A.1.1. Modification of the maintenance programme

AMPs developed and updated for all passive components contain all the information related to ageing and the activities that have to be implemented. A general approach may be to formally introduce and perform these programmes for ageing management for only safety class 1 components. For other safety related components they will serve only as a basis for the activity and all the relevant information will be input into the maintenance procedures. During continuous operation and maintenance activities on these systems, the information is fed back periodically to the AMPs. Any evaluation and modifications required for the long term operating LRA are carried out in the context of the LTO programme.

The content of the AMPs includes:

— Identification of the scope and of the ageing mechanisms;
— Preventive measures;
— Parameters to be monitored;
— Detection of ageing effects;
— Monitoring and trend analysis;
— Acceptance criteria;
— Corrective actions;
— Feedback of the programme results and efficiency;
— Administrative controls and quality assurance;
— Feedback of operational experience.

A review of the content of the AMP is needed to establish what is new and what needs to be incorporated into the maintenance procedures, then all the maintenance procedures affected are supplemented.

Some examples for what needs to be incorporated into the maintenance procedures are listed below:

— A diagram of the component showing the detailed location of the critical degradation mechanisms;
— A data sheet with information related to ageing mechanisms, including identification of the critical location in the diagram and the location of the critical ageing mechanism;
— Critical ageing mechanisms that can be assumed, whether a given location is affected by that mechanism or not; any necessary additional control activities, such as surface checking and wall thickness measurement, among other things;
— Supplement with the specified driving torque of screws;
— Documentation of the issues with measurements and controls and any new recommendations arising from the modification.

A.1.2. Review and modification of other operational programmes

During the development and review of the AMPs, some operative plant programmes may not meet the expectations of the revised ageing management system.

These are operative programmes used during the ageing management of specific components. They may have not received due emphasis in the original plant programmes. An example of one of the most important programmes assigned to mechanical components is the Hungarian practice shown in Fig. 31.

Some specific categories of the Hungarian AMPs are addressed in more detail below.

A.1.3. RPV surveillance programme

To monitor irradiation damage of the reactor structural materials, RPVs were originally equipped with specific surveillance specimen sets. To eliminate the weaknesses of the original surveillance programme (i.e. the extremely high lead factor, inaccuracy of neutron fluence monitor positions and the inadequate temperature monitor) a supplementary surveillance programme was designed and launched earlier on. Archive (reconstituted Charpy V-notch) and reference materials were used. Neutron fluence monitors were placed in controlled positions and a new type of temperature indicator was selected.

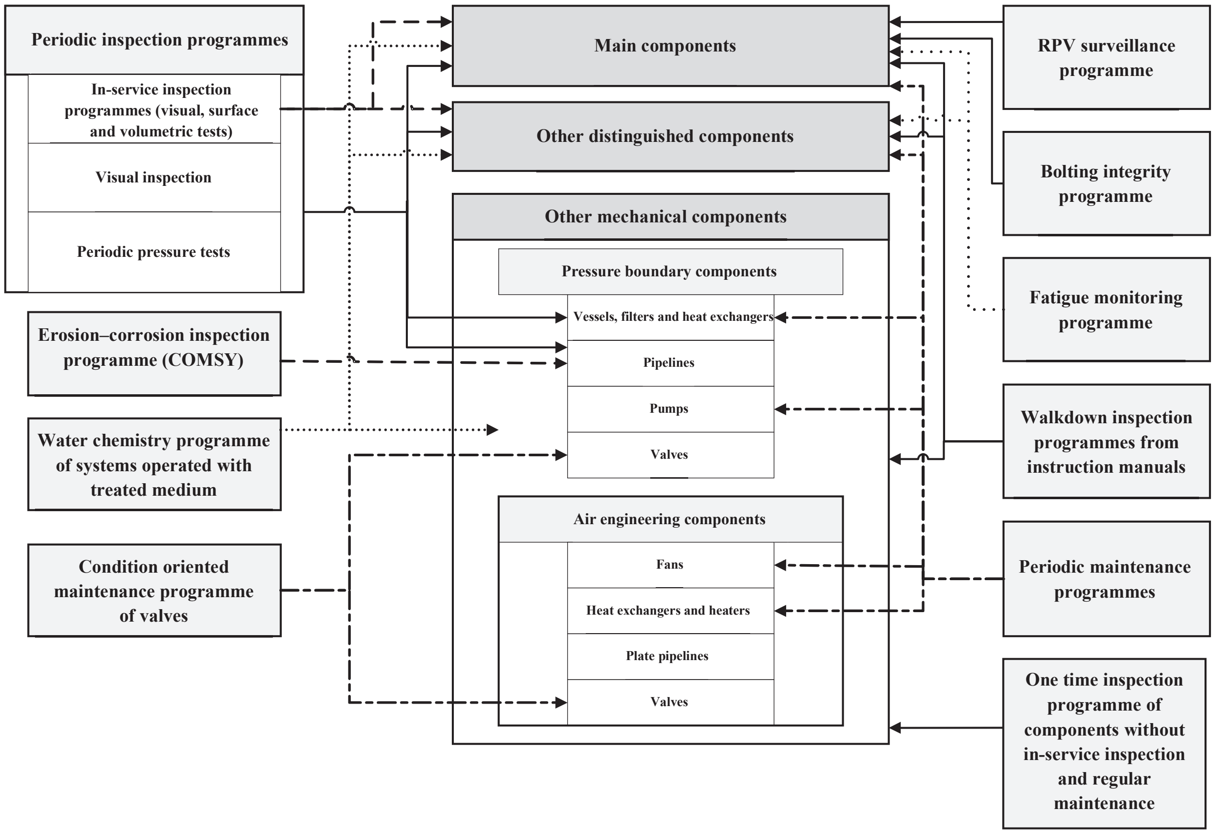


FIG. 31. The operative AMPs of mechanical components.

As this supplementary surveillance programme was designed for 30 years of operation, the extension of plant lifetime required further RPV monitoring. Consequently, an updated supplementary surveillance programme has been prepared. Its main objective is to meet the requirements of ASTM E 185, Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, that is, to be able to monitor the effects of neutron irradiation of the RPV throughout its operating lifetime, and for the lead factor to approach the range of between one and three.

A.1.4. Walkdown inspection programmes

One of the important methods for the management of ageing mechanisms of systems, structures and components is the regular on-site inspection of the actual condition of operating equipment. The documented walkdown inspection of components can be used to detect leakages, unusual vibrations and deformations associated with several degradation mechanisms.

On-site inspections of SSCs operating in serviceable and limitedly serviceable rooms are based on the tasks and responsibilities defined in work descriptions. They are performed by the personnel on duty from the appropriate operative team (i.e. engineering, electric, I&C or dosimetry), in accordance with the various system manuals.

Certain systems and components of the primary circuit operate in rooms that are non-serviceable during the operation of the plant. The on-site inspection activity in these rooms is ‘high radiation hazardous’ and so it can be performed only on the basis of a high radiation hazardous work programme or after the shutdown of the reactor. The inspection activity should be based on route plans, checklists and data sheets. Among others, its scope includes:

— Identification of the source of any extraordinary noise or vibration;
— Adequacy of the installation and intactness of thermal insulations;
— Checking the integrity of equipment and pipelines containing operating fluid;
— Adequacy of the controlled drainage of identified and controllable leakages and of temporary protection of surrounding equipment.

The ageing management review concluded that the documented walkdown inspection programmes can be used to prevent and recognize boron acid induced corrosion degradation of equipment if the relevant inspection guideline specifies in detail the environments of equipment to be checked, where leakage of the primary coolant can be assumed.

A.1.5. Water chemistry programme

Chemistry programmes have a close relationship with ageing and ageing management. Corrosion problems that required the review and modification of chemistry programmes may appear during the entire lifetime of the units.

Based on operational experience, the water chemistry of the primary circuit should provide compatibility of the construction and structural materials of the primary components with the chemical composition of the coolant. The water service of the primary circuit has significant impact on the general and local corrosion of surfaces, the transport of corrosion products, the extent and type of radioactive contamination, as well as the decontamination technologies used.

In line with the compatibility principle, additional improvement measures to be implemented during the service lifetime extension programme were specified as follows:

— Reduction of the control range for boron equivalent, potassium ion and dissolved hydrogen concentrations in the primary coolant during power operation to further decrease the general corrosion of primary circuit surfaces;
— Reduction of the potassium equivalent concentration in the last 500 service hours before shutdown for refuelling to further decrease the transport of primary circuit corrosion products.

In the secondary circuit, the stress corrosion cracking of heat exchanger tubes in the steam generator is the most relevant degradation mechanism. The only action that effectively prevents this degradation mechanism is the establishment of compatibility between the structural materials used and the water chemistry of the fluid in them. The concentration of impurities is kept as low as is technologically feasible through the implementation of the following actions:

— Decreasing the secondary circuit corrosion products deposited on the surface of the heat exchanger tubes by increasing the pH value of the feedwater;
— Decreasing the secondary circuit corrosion products deposited on the surface of the heat exchanger tubes by replacing the carbon steel elements with corrosion resistant steel;
— Elimination of copper from the secondary circuit, including the replacement of copper tube main condensers with corrosion resistant steel tube ones;
— Increasing the ratio of regenerated lateral precipitations (in order to decrease the ion flow in the steam generators);
— Conservation of surfaces in shutdown state, reduction of the size of water drops by replacing ammonium hydroxide in the feedwater with multifunctional polyamine.

A.1.6. Erosion–corrosion monitoring programme

The COMSY code, developed by AREVA NP GmbH, is used in Hungary for the analysis of erosion–corrosion degradation of piping systems based on theoretical and experimental models. The results of the programme have been verified and corrected through a large number of model experiments. When using the programme, the systems to be monitored are divided into segments (e.g. feedwater), sections (e.g. the pipeline from the feedwater collector to the steam generator) and components (e.g. straight pipes or valves). Calculation results from the programme are as follows:

— In addition to erosion–corrosion phenomena, a forecast of the potential for cavitation and erosion;
— Forecast of the minimum lifetime of the components;
— Designation of degradation locations to be monitored;

— Evaluation of monitoring results;
— Evaluation of lifetime usage;
— Updated lifetime forecast;

The following systems have been reviewed regarding erosion–corrosion:

— Feedwater system;
— Fresh steam system;
— Bleeding system;
— Main condensate system;
— Auxiliary condensate system;
— In-house service steam system;
— Turbine gland steam system.

The ageing management review established and demonstrated that the condition oriented ageing and plant life management system can be used to prevent, detect and monitor the erosion–corrosion degradation of the affected components. The modifications proposed were to measure and incorporate the actual chromium content of the monitored component materials in the assessment.

A.2. RUSSIAN FEDERATION

A.2.1. Example of an intelligent system of condition monitoring in the context of PLiM and PLEX

Plant life extension beyond the originally anticipated life is one of the main concerns in the nuclear industry of the Russian Federation, as well as abroad. Along with consideration of the accumulated experience gained during LTO of different NPP unit types, in the context of PLEX activities, the set of technical measures designed to ensure safe and reliable operation of NPPs beyond their originally assumed service life should be used to justify LTO. Nuclear power plant LTO stipulates the necessity of changing the traditional approaches applied to conducting an assessment of the degradation rate, also referred to as ageing, in zones mostly subjected to operational loading of equipment important to safety. Considerable work has been done and experience accumulated to justify operation beyond the originally assumed service life of Russian NPPs. In parallel, new approaches have been developed such as the design of new diagnostics tools to help assess operational damage. The approach is based on the application of conventional NDE methods in conjunction with on-line monitoring to determine the consequences of thermal and mechanical loading, such as the actual deformation and loss of structural and functional performance of essential and potentially life limiting equipment. Numerical calculations of the accumulated metal damage detected in the monitored zones can then be conducted more precisely. The block diagram of a life cycle of NPP equipment subject to thermal and mechanical loading is presented in Fig. 32. It includes all main sequences in the service life of a safety related component, from its design phase all the way to its decommissioning.

A detailed analysis of the main stages of the NPP equipment lifecycle complemented by the accumulated experience of LTO of NPP units shows that traditional approaches used for condition assessments of thermomechanical equipment in NPPs are insufficient from the standpoint of solving the complex safety related SSC ageing projections during the NPP LTO period. Operation related damage, as a rule, presents the following typical characteristics:

— Operational damage presenting regular and yet unstoppable escalation.
— Operation related cracks growing very fast in a short period of time. This is a direct threat to NPP safety and challenges the main safety requirements set by the industrial normative.
— Annual ISI cannot guarantee that through-the-wall cracks will not be detected during the fuelling campaign of an NPP unit.
— Exact causes of crack formation and growth are not clarified.
— Corrective measures taken do not apparently lead to successful results.

— Repairs to the damaged zones having negative effects on the life characteristics of the component being repaired and potentially causing its premature replacement.

The solution selected to help resolve operation related damage involves the development and application of an intelligent diagnostic system capable of monitoring operational damage, including crack formation and the actual loading history of the main SSCs, such as extensive cyclic expansion.

This is a new approach to monitoring operational damage and justifying SSC survivability for the duration of an operating term longer than the originally assumed NPP design life.

The current state of the art in stress analysis makes it possible to predict the behaviour of almost any defect, but to really make use of all of its potential, it is necessary to collect much more information than is possible through the most advanced non-destructive testing inspection methods and means. The difference between the proposed methodology and conventional approaches in diagnostics is the fact that in the classical method, inspection results are used as input data to the stress analysis, while with the new methodology, the problem is reversed.

In the proposed new approach, first a detailed finite element model of the monitored equipment is created, then preliminary calculations are carried out using classical means. In other words, the main working parameters obtained from the regular sensors installed on the equipment of an operating NPP unit make up the operational load combinations and time history and the stress intensity profiles and the code evaluations are calculated from these. These preliminary calculations allow an optimization of the selection of the most appropriate locations for the

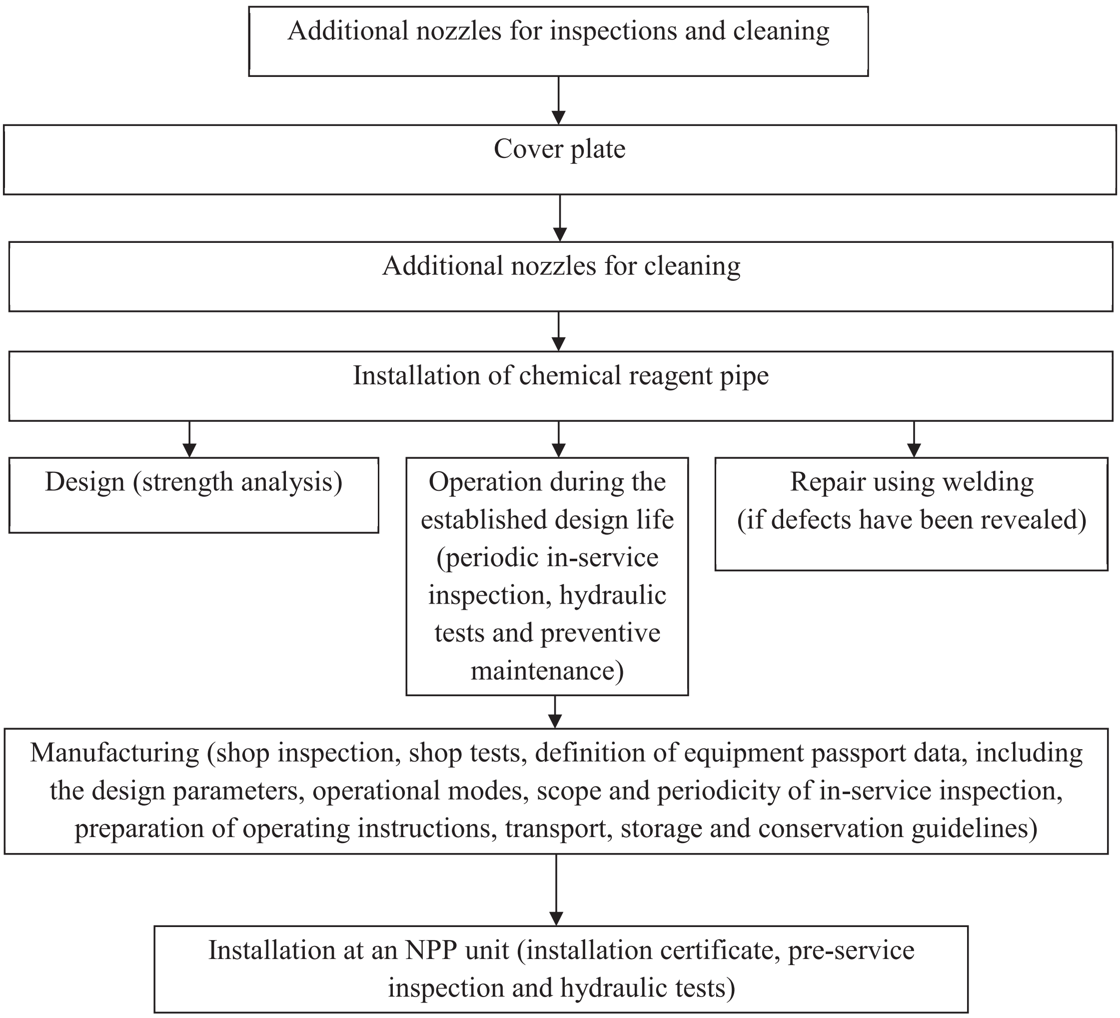


FIG. 32. Life cycle of the NPP thermomechanical equipment.

installation of the new advanced control sensors, and help in the selection of the most appropriate type of sensors in order to obtain the most precise results. The type of control sensors currently used are high temperature strain gauges, temperature probes, as well as pressure, acceleration and displacement sensors. Moreover, in the precise zone of potential incipient damage, special high temperature resistant sensors, intended for the monitoring of defect kinetics, or of incipient cracks, can be installed. For this purpose, two types of redundant, diverse and independent sensors are normally used: acoustic emission sensors and ultrasonic sensors.

All sensors can operate on-line for several years. The optimal frequency of data records is selected. All recorded data, after prompt processing, is transferred as input to the finite element model for strength calculations of the monitored zone. As a rule, once a year a comprehensible classical non-destructive inspection of the mechanical properties (using the kinetic indentation method), of the metal integrity (employing the phased array techniques), and of the residual stresses (utilizing the magnetic method) are carried out. The data collected allow for a precise condition assessment of the structural material in the monitored zone, and serve as verification of the results based on the advanced diagnostic system as a whole and of the calculation module in particular.

The software for strength calculations should be customized, since it should work on-line automatically with the specific sensors and the computer interface in use. Such an approach allows prompt verification of the stress model using live records as they are received from the control sensors. The simultaneous use of the thermomechanical loading sensors and defect monitoring sensors, in conjunction with the finite element stress model, allows maintenance not only to foresee the most unfavourable scenario(s) resulting in SSC damage, but also to provide prompt analysis and data processing to recommend the most effective compensating measures (in order to reduce negative operational impacts), and an effective plan of action for the appropriate modernization or refurbishment effort, which is after all the most effective action plan to successfully justify an NPP operation permit extension beyond its design service life.

A block diagram of the work implementation links, with respect to monitoring of operational damageability and justification of survivability of NPP equipment during LTO, is presented in Fig. 33, and takes into consideration the philosophy previously described.

The application of the new approach for element diagnostics gives the NPP the necessary toolkit, which allows prompt and, above all, effective resolution of the following main tasks:

— Improvement of safety of NPP unit operation;
— Determination of the cause and effect links of degradation and defect propagation in the critical zone, and clarification of dominating mechanisms and of the factors responsible for the degradation;
— Development of the effective compensating measures intended to eliminate, or at least considerably mitigate, the main degradation factors;
— Development of new acceptance criteria for operation dependent cracks (based on length, height, equivalent square, orientation and location along the perimeter, among other things) in order to justifiably reduce conservatism in rejections in accordance with modern industrial norms for quality assessment;
— Reduction in the number of unnecessary over preventative repairs of defects based on their continuous on-line monitoring during operation.

A.2.2. Example of an application of NPP on-line monitoring and procedural stages

The on-line monitoring concept previously described has recently been fully applied on operating NPP equipment. On-line monitoring of operational degradation and the effects of thermomechanical loading have been successfully applied in the Russian Federation. Since the proposed approach is universal and can be easily adapted to the monitoring of almost any critical equipment or pipeline, it is advisable to concentrate on the general features of the new approach.

In the first stage, the primary finite element model of a closed circuit was created (see Fig. 34). The system selected was of particular interest due to the necessity of solving several tasks. The model of the closed system components was finely detailed, including pressure vessels, interconnected pipelines, supports, hangers and anchors, among other things, since they will all have an effect on the stress deformed state of the circuit components.

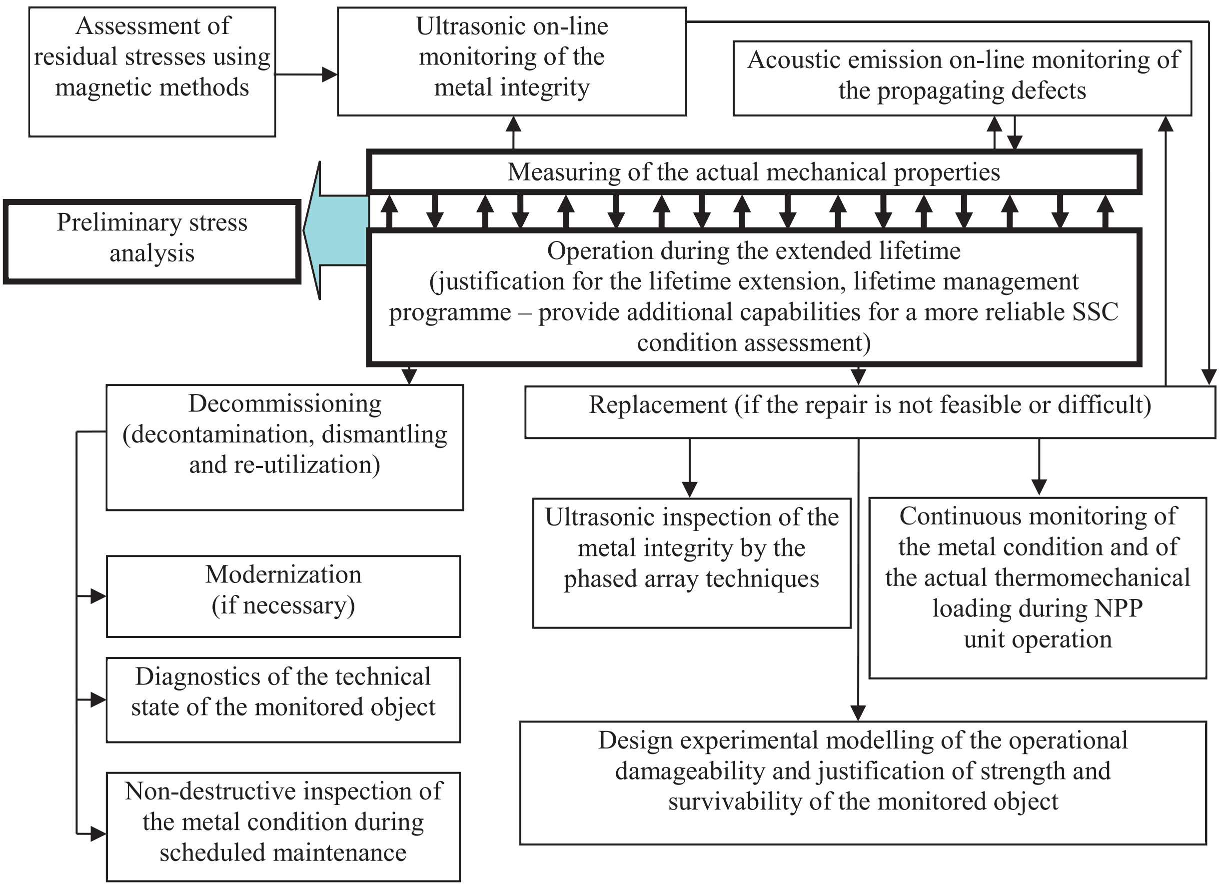


FIG. 33. The work implementation block diagram.

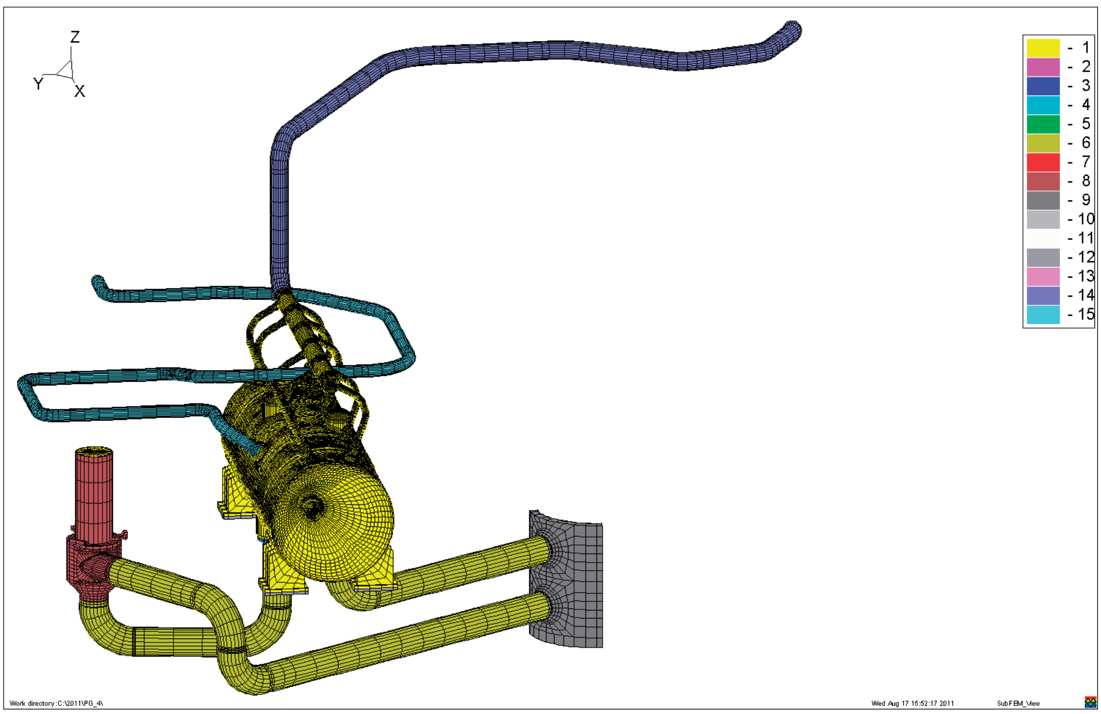


FIG. 34. Primary finite element model.

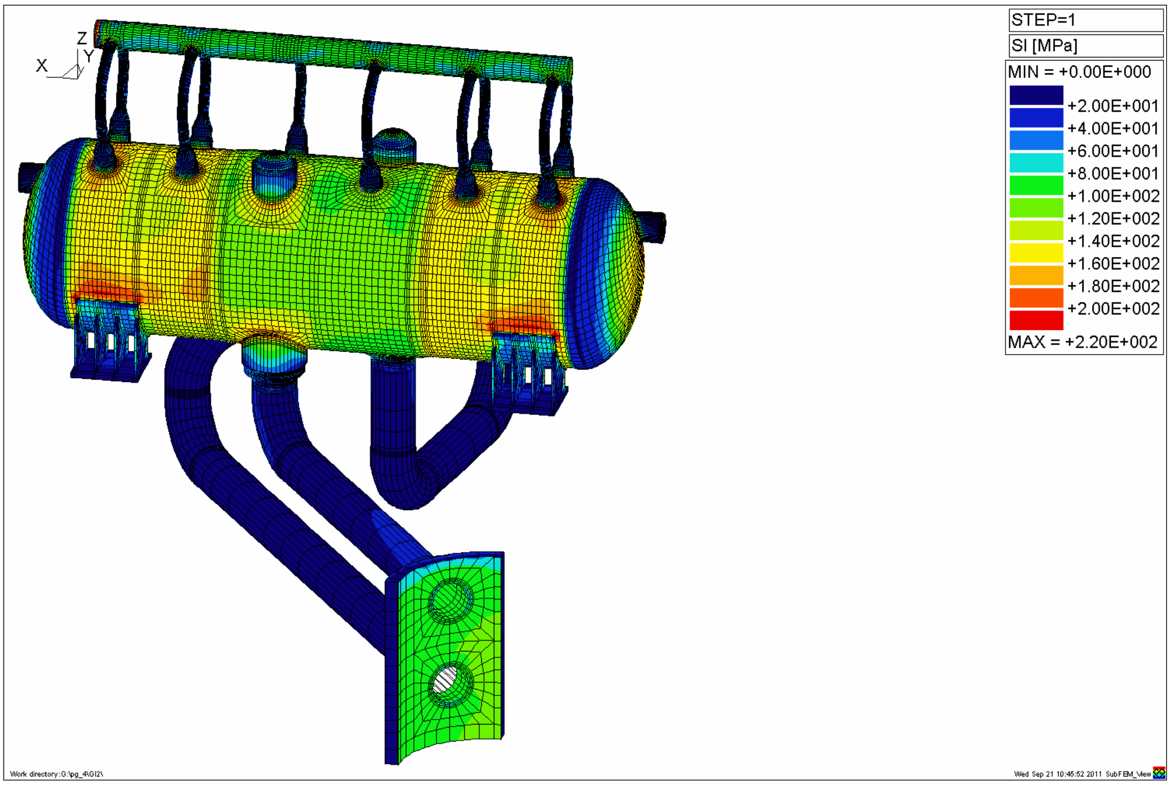


FIG. 35. Preliminary strength analysis considering the design operational modes.

In the second stage, preliminary strength calculations were performed using the finite element model. The results of calculations (see Fig. 35) were used to determine the zones subject to maximum stress levels, and hence rupture in relation to a possible rapid propagation of operation related degradation. It is important to note that preliminary results of the analysis are considered together with the results of the periodic ISI data and operation history. Such an approach allows the selection of zones, where defects are most likely to appear, on which to apply the advanced on-line monitoring instruments. To achieve the most effective monitoring system, it is necessary to design the system in such a way that data from the additional control sensors (strain gauges, thermocouples, displacement sensors and ultrasonic and acoustic sensors, among other things) in conjunction with data obtained from regular sensors (pressure, thermocouples and flow rates, among other things), would allow the maximum amount of comprehensive data collection on the state of the monitored object. Then for one or several selected zones, the required types of control sensors are to be determined, including their number and technical performance requirements. The exact location selection and the design of the attachment of the selected control sensors to the objects being monitored can then be finalized. The optimization of the sensor type and location scheme is to be closely followed at installation time.

In the third stage, the architecture of the monitoring system of operational degradation and the consequence of thermomechanical loading is developed. One of the main requirements of the monitoring system is the acquisition of reliable information on the actual stress deformed state, and on the defects of the inspected object for each of the different operational modes of the NPP unit. Moreover, enough monitoring data should be collected and conveyed to the finite element model as input in order to allow the correct combined analytical experimental justification of the monitored object’s condition, strength, accumulated degradation (remaining cycles and residual lifetime) and LTO survivability.

Considering the operational features of the inspected object, the following requirements have been taken into account during the development of the monitoring system:

— The system is to operate continuously during at least one fuel campaign with no possibility of personnel access for any service work in view of the fact that the system in question may be located inside the sealed containment area.

— Maximum reliability, service lifetime survivability and a close adherence to the temperature transients as per design should be guaranteed (temperature and humidity).
— It is necessary to provide remote system control, effective handling of the considerable scope of collected data and its storage and transfer to the end user for prompt analysis.
— The measuring components (sensors, lubricant for acoustic coupling and accessories, among other things) need to be optimally selected, taking into account that the outer surface temperature of the inspected metal can reach up to 300°C.

Taking into account these requirements, and the intended functions of the monitoring system, the following subsystems have been arranged within the system as a whole:

— On-line ultrasonic monitoring subsystem with high temperature sensors intended for the assessment of the propagation speed of the defect’s growth in all different operational modes;
— On-line acoustic emission monitoring subsystem with high temperature acoustic emission sensors scanning specific sectors of the monitored zone intended to detect the moment of the crack formation and its growth velocity under all different operational modes;
— On-line monitoring subsystem of thermal deformation and displacement detection with control sensors of temperatures, deformations and displacements, intended to collect reliable data on actual thermalmechanical loading of the monitored zone.

The architecture of the on-line monitoring system is shown in Figs 36 and 37.

In the fourth stage, the hardware design and fabrication of the measuring components of the monitoring system was carried out.

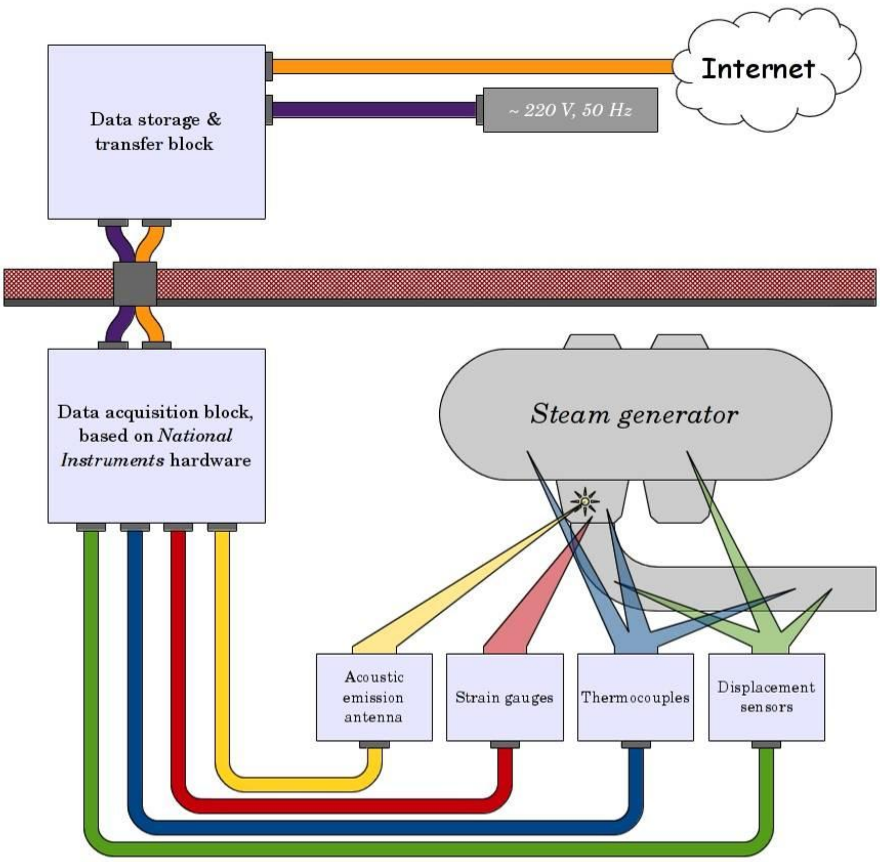


FIG. 36. Integral structure of the monitoring system.

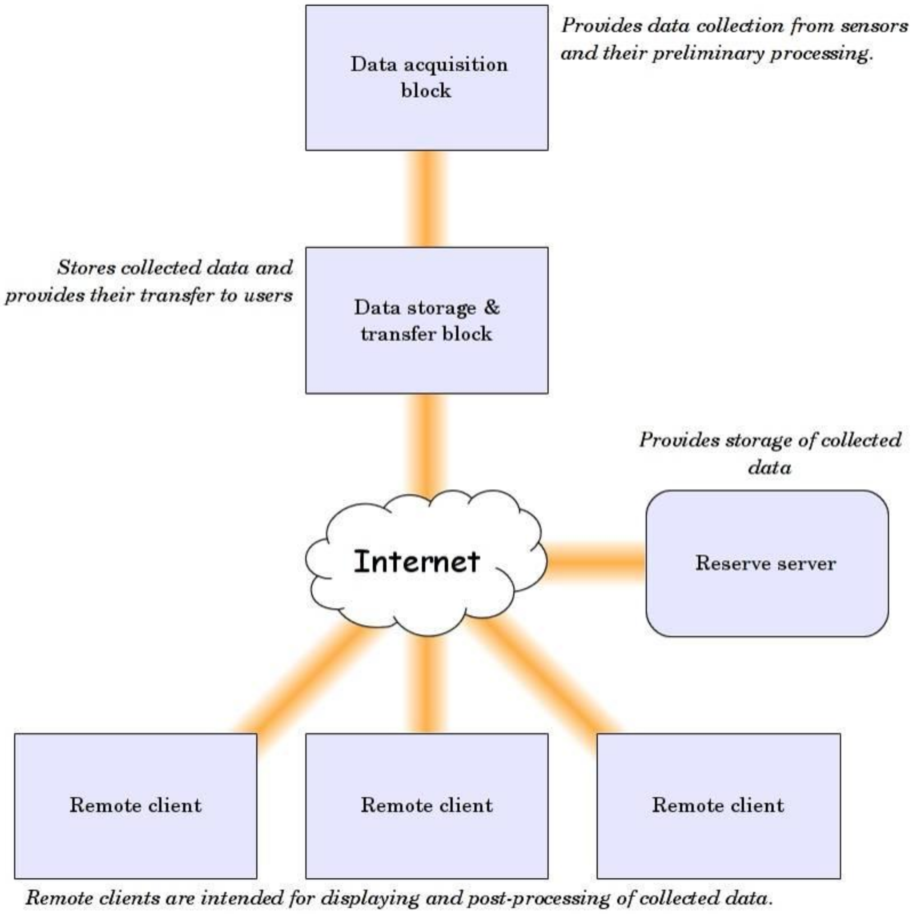


FIG. 37. Base scheme of the data exchange links.

Taking into account the multifunctional tasks of the system and the interface requirements of separate modules (subsystems) the decision was made to build the monitoring system based on multichannel microprocessor measuring modules on the national instruments chassis, integrated into the common measuring processing complex (see Fig. 38).

For the purpose of implementing the specific tasks, special software was designed to test the system operability, data acquisition, storage and transfer ability, its capability to perform preliminary analysis of the collected data, and other requirements.

Taking into account the outline and basic design features of the object to be monitored, and the accepted scheme of the control sensor's arrangement, special accessories were designed and manufactured to allow the fastening of the system elements on the monitored object (see Fig. 39).

In the fifth stage, the mounting, adjusting and testing of all monitoring system modules were completed. Trial operation of the system in a continuous mode was performed for one month. Tests were carried out in a laboratory using the full scale test bench, which precisely models the monitored zone. The test bench included the real segment monitored zone, which was separated from the dismantled equipment (see Fig. 40).

In the sixth stage, the mounting, adjusting and testing of the monitoring system in the NPP unit were completed (see Figs 41 and 42).

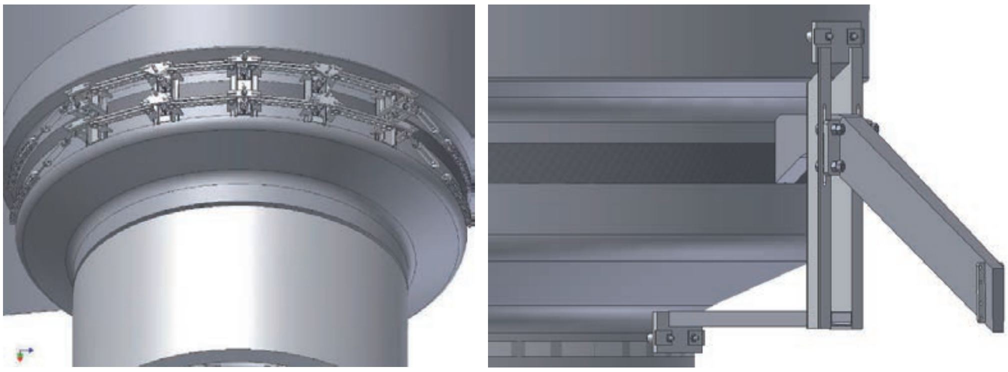
In a seventh stage, the adjustment and calibration of the finite element calculation module were carried out. Special software developed for the analytical experimental analysis of the stressed state and survivability of the monitored zone becomes the central core of the monitoring system. This software works in parallel with the acquisition of diagnostics data through the measuring channels of the monitoring system. The finite element model of the monitored object contains control nozzles at the control sensors location. In the control nozzles, the calculation results need to be compared to the measured temperature, displacement and deformation data. The distinctive feature of the model is its ability to use the data collected from the continuous monitoring of the object

(a)

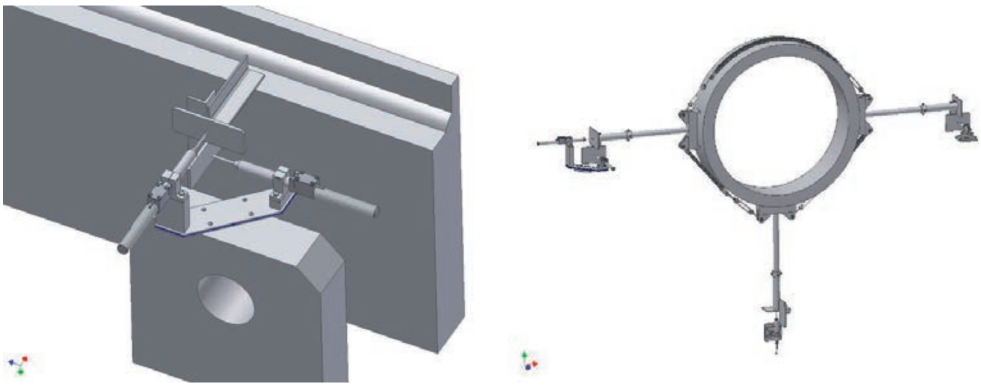
(b)

(c)

FIG. 38. Hardware of the monitoring system. (a) and (b) Data acquisition and processing block; (c) Data storage and transfer block.



(a) Fastening of the AE sensors (b) Fastening of the US sensor



(c) Control of equipment displacements (d) Control of pipeline displacements

FIG. 39. General view of the special accessories for fastening.

FIG. 40. Laboratory testing of the monitoring system on the full-scale test bench.

FIG. 41. Mounting of the acoustic emission, ultrasonic and displacement sensors.

FIG. 42. Mounting of the strain gauges and thermocouples.

(fields of temperatures, deformations, displacements and pressure) both as actual input data for calculations and as testing data for the finite element module calibration.

Calibration of the finite element module was performed by means of a comparison of the calculation data at the control nozzles and of the experimental data on deformations, displacements and temperatures in different operational modes and at different loading levels of the monitored object. Data on pressure and temperatures registered by the regular NPP unit sensors were used in the finite element module as the actual loading parameters of the object. If a satisfactory match of the calculation results and the experimental data is not obtained, it is likely that improvements in the calculating core are to be implemented, including, among other things:

— Revision of the software;
— Refinement of the monitored object's geometry;
— Changing of the finite element mesh in specific zones;
— Correction of the boundary condition.

As a rule, only after several iterations can a satisfactory agreement of calculations and experimental data be achieved.

After adjustment and calibration of the finite element calculation module and the subsequent comparative analysis of experimental and calculated data with good agreement is achieved (see Fig. 43), a conclusion is provided on the efficiency of the developed model and on the adequacy of modelling of the object stressed state in different operational modes.

In an eighth stage, the continuous recording of the monitored data in parallel with data transfer to the interfacing calculation module, can take place. The monitoring system provides the possibility to make on-line assessments of the kinetics of crack initiation and of its propagation. Correspondingly, the analysis allows the building of a picture of the stressed states in the monitoring zone. This can be done at any time using the currently live loading parameters as input data. The method allows for assessment of the actual accumulated deterioration, taking into consideration the recorded operational load cycles. It also gives the life forecast, justifying the monitored component longevity and its LTO survivability, if this is the case. Moreover, all events recorded by the monitoring system are analysed with consideration given to the operational modes, which occurred during the specified time period, and their characteristics and technological features. It provides a stable feedback on the metal behaviour with regard to defects and their growth as a function of actual relevant factors and parameters, and it provides an understanding of the cause and effect relationship responsible for the defect initiation, its growth and speed of growth.

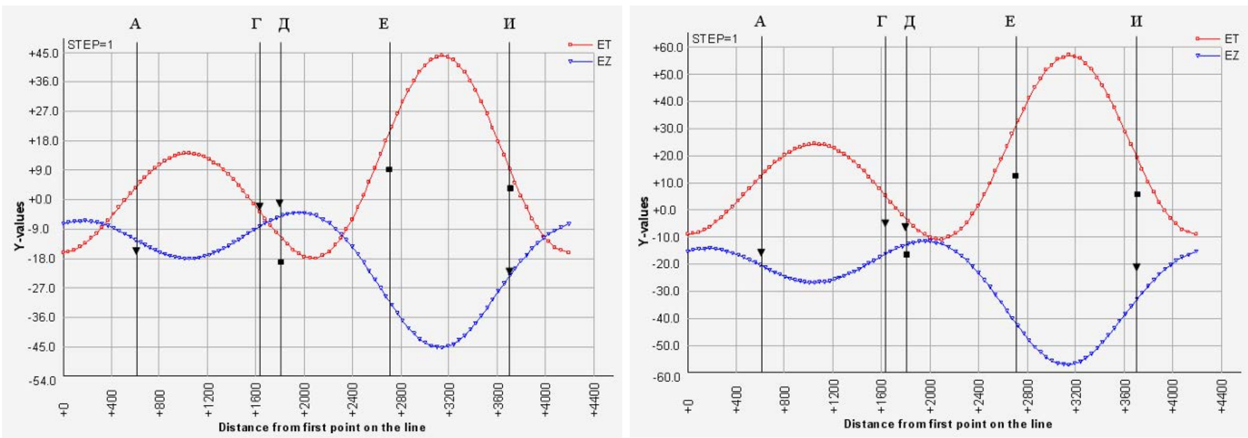


FIG. 43. Comparison of calculated (black triangles and squares) and experimental (red and blue curves) data on deformations in the monitored zone at different loading parameters.

For any specified time period, the database (knowledge base) on the behaviour of the monitored equipment under different operational modes is collected and stored. Correspondingly, a comprehensive analysis of the stressed state and defect behaviour is also performed with the new approach described above. The data obtained allows for the development of effective compensatory measures intended to mitigate the main damaging factors and to improve the longevity of the monitored object.

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ABBREVIATIONS

AERB	Atomic Energy Regulatory Board
AMP	ageing management programme
ARA	application for renewal of authorization
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
CANDU	Canada deuterium–uranium (Canadian pressurized heavy water reactor)
ČEZ	České Energetické Závody
CFR	Code of Federal Regulations
CGN	China General Nuclear Power Group
CNNC	China National Nuclear Corporation
CNPEC	China Nuclear Power Engineering Company
CNSC	Canadian Nuclear Safety Commission
CSA	Canadian Standards Association
CSN	Nuclear Safety Council (Consejo de Seguridad Nuclear)
ECCS	emergency core cooling system
EDF	Électricité de France
FCR	fuel channel replacement
FOG	functional outage group
FSAR	final safety analysis report
GALL	generic ageing lessons learned
HAEA	Hungarian Atomic Energy Authority
HAF	Chinese regulatory document: NPP safety requirements and quality assurance
INPO	Institute of Nuclear Power Operations
ISI	in-service inspection
ISR	Integrated Safety Review
I&C	instrumentation and control
KAPS	Kakrapar Atomic Power Station
LBB	leak before break
LRA	licence renewal application
LSFCR	large scale fuel channel replacement
LTO	long term operation
MAPS	Madras Atomic Power Station
NAPS	Narora Atomic Power Station
NDE	non-destructive examination
NEA	OECD Nuclear Energy Agency
NEI	Nuclear Energy Institute
NPCIL	Nuclear Power Corporation of India Ltd
NPP	nuclear power plant
NNSA	National Nuclear Safety Administration
NRC	United States Nuclear Regulatory Commission
NSSC	Nuclear Safety and Security Commission
NVNPP	Novovoronezh Nuclear Power Plant
OE	operating experience
OECD	Organisation for Economic Co-operation and Development
PHWR	pressurized heavy water reactor
PLEX	plant life extension
PLiM	plant life management
PSA	probabilistic safety assessment
PSR	periodic safety review
PWR	pressurized water reactor

RAPS	Rajasthan Atomic Power Station
RBMK	high-power channel-type reactor
REIA	radiological environmental impact assessment
RPV	reactor pressure vessel
SALTO	Safety Aspects of Long Term Operation
SMG	Santa Maria de Garoña
SONS	State Office for Nuclear Safety (Czech Republic)
SSCs	structures, systems and components
TLAA	time-limited ageing analysis
WWER	water cooled water moderated power reactor

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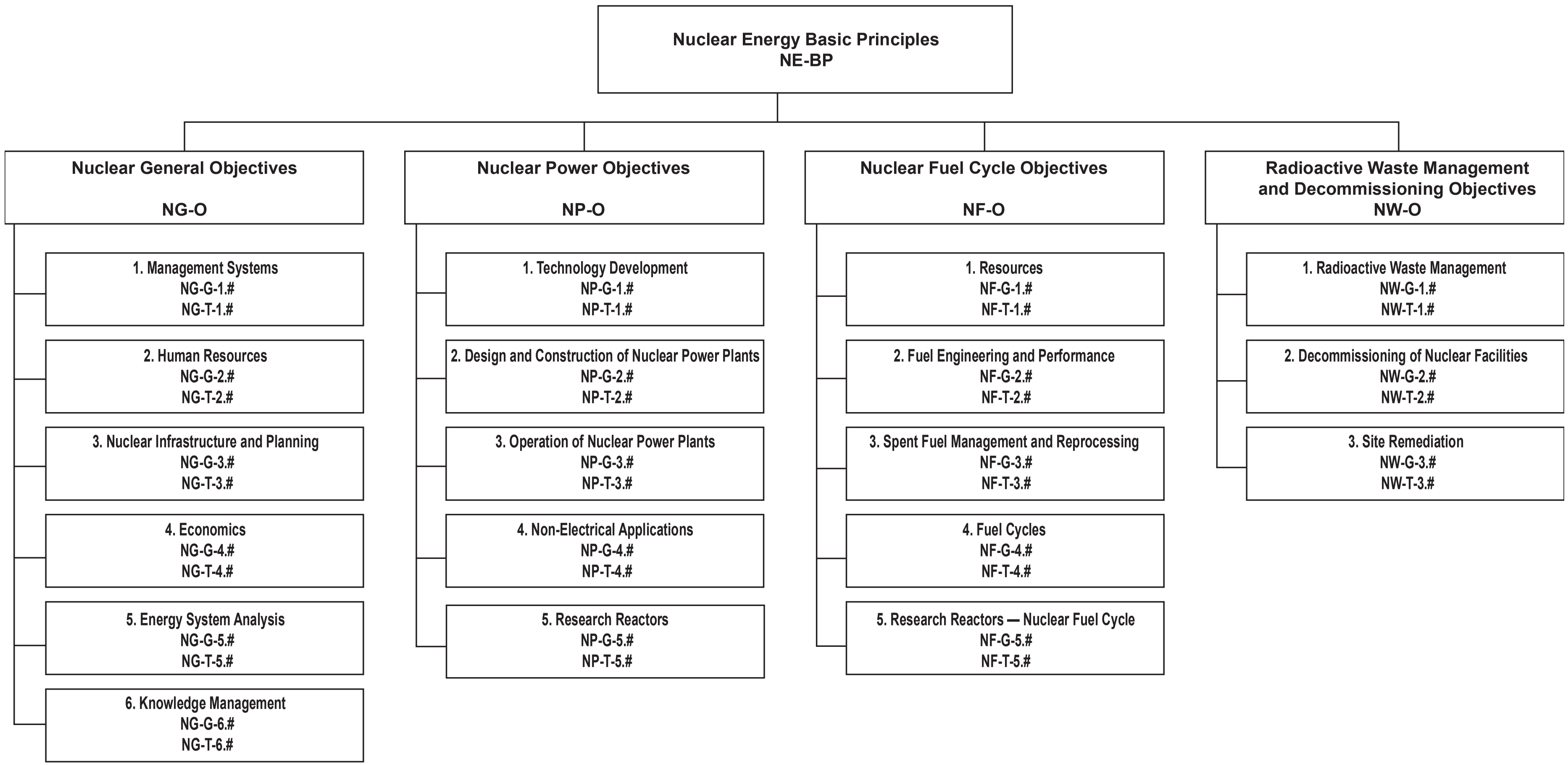

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