ROMANIA

National Assessment Report for the EU Topical Peer Review on Ageing Management for Nuclear Installations

December 2017

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EXECUTIVE SUMMARY

This report has been prepared by the National Commission for Nuclear Activities Control (CNCAN), the nuclear regulatory authority of Romania, following the ENSREG specification developed by WENRA for the European Union Topical Peer Review (TPR) on Ageing Management for Nuclear Installations.

In preparation of the National Assessment Report (NAR), CNCAN has reviewed the reports prepared by the licensees, the National Company Nuclearelectrica (SNN) and the Institute for Nuclear Research (RATEN – ICN) and has conducted specific inspection activities at the Cernavoda Nuclear Power Plant (NPP) and the TRIGA Research Reactor (RR).

The report presents the regulatory framework for AMP, the overall AMPs developed by the licensees, as well as the specific applications of the AMPs in accordance with the TPR specification.

Based on the regulatory reviews and inspections performed so far, CNCAN is satisfied with the adequacy of the licensees'AMPs and with their overall implementation. No major issues have been identified.

01. GENERAL INFORMATION

01.1 Nuclear installations identification

In accordance with the ENSREG specification, developed by WENRA, the scope of the Romanian national assessment report (NAR) for the EU Topical Peer Review (TPR) on ageing management for nuclear installations covers the following installations:

- Cernavoda NPP Units 1 and 2, in operation
- TRIGA research reactor, in operation.

Table 1.1 provides a summary of basic data on these installations.

Table 1.1: Nuclear installations addressed in the Romanian NAR

Name	Licensee	Type of reactor	Power output	Year of first operation	Scheduled shutdown
Cernavoda NPP Unit 1	National Company Nuclearelectrica (SNN)	Pressurized heavy-water reactor (PHWR) – CANDU 6	Design net capacity: 650 MW(e)	1996	N/A
Cernavoda NPP Unit 2	National Company Nuclearelectrica (SNN)	Pressurized heavy-water reactor (PHWR) – CANDU 6	Design net capacity: 650 MW(e)	2007	N/A
TRIGA research reactor	Institute for Nuclear Research (RATEN - ICN)	Dual core pool type TRIGA reactor	TRIGA SSR (Steady State Reactor) - 14 MW	1979	N/A

Cernavoda NPP, the only nuclear power plant in Romania, has two units in operation, pressurised heavy water reactors of CANDU 6 design (CANadian Deuterium Uranium), each with a design gross output of 706.5 MWe. Unit 1 and Unit 2 started commercial operation on the 2nd of December 1996 and on the 1st of November 2007, respectively. The plant was initially intended to have 5 units. The construction of the other three units on the site was stopped at different stages, and these units are currently under preservation. All units are pressurised heavy water reactors (PHWR), CANDU 6 type. The licence holder for Cernavoda NPP is the National Company Nuclearelectrica (SNN - Societatea Nationala Nuclearelectrica S.A.)

Romania also has only one research reactor (RR) in operation. It is a dual core pool type TRIGA reactor, which has achieved the first criticality on the 18th of November 1979. The research reactor is primarily used for materials testing. The Institute for Nuclear Research (RATEN - ICN) in Pitesti is the license holder for this research reactor. The reactor is composed of the following cores which are contained in the same pool:

- TRIGA SSR (Steady State Reactor) - 14 MW reactor; the conversion of the TRIGA-SSR Reactor started in 1992, from HEU fuel (Highly Enriched Uranium) to LEU fuel (Low Enriched Uranium) and was completed in 2006; the modernization of the reactor safety systems and of the control room has been completed in 2011 to support the life extension of the facility;

- TRIGA ACPR (Annular Core Pulse Reactor); the ACPR reactor, with LEU fuel, can be operated for a maximum pulse of 20.000 MW; it has a single large central irradiation channel for fuel and structural materials irradiations under pulsed modes.

01.2 Process to develop the national assessment report

The present report has been prepared by the National Commission for Nuclear Activities Control (CNCAN), the nuclear regulatory authority, following the ENSREG specification developed by WENRA.

CNCAN has requested the licensees, the National Company Nuclearelectrica (SNN) and the Institute for Nuclear Research (RATEN - ICN), to prepare reports covering the contents in the specification for the NAR and conducted reviews and inspections to verify the information provided in the licensees' reports. Following these reviews and inspections, CNCAN prepared the NAR report.

02 OVERALL AGEING MANAGEMENT PROGRAM REQUIREMENTS AND THEIR IMPLEMENTATION

02.1 National regulatory framework

In January 2016, CNCAN issued a specific regulation on ageing management for nuclear installations – Nuclear safety requirements on ageing management for nuclear installations (NSN-17). Before this, general requirements on ageing management for nuclear installations had been formulated in the license conditions, as well as in the context of the regulation NSN-10 on periodic safety review (PSR), issued in 2006, and of the regulation NMC-10, issued in 2003, on management systems for the operation of nuclear installations, as well as in license conditions. A regulatory guide (GSN-01) was issued in 2015, establishing the list of codes, standards and national prescriptions to be respected by the NPP in all stages of plant operation, including the implementation of the ageing management program.

The requirements in the regulation are based on the relevant IAEA safety standards (NS-G-2.12) and WENRA Safety Reference Levels (Issue I).

Examples of requirements directly based on the WENRA SRLs on Ageing Management:

- The licensee shall establish an ageing management program (AMP);
- The licensee shall assess structures, systems and components important to safety taking into account relevant ageing and wear-out mechanisms and potential age related degradations in order to ensure the capability of the plant to perform the necessary safety functions throughout its planned life, under design basis conditions;
- The licensee shall provide monitoring, testing, sampling and inspection activities to assess ageing effects to identify unexpected behaviour or degradation during service;
- The Periodic Safety Reviews shall be used to confirm whether ageing and wear-out mechanisms have been correctly taken into account and to detect unexpected issues;
- In its AMP, the licensee shall take account of environmental conditions, process conditions, duty cycles, maintenance schedules, service life, testing schedules and replacement strategy;
- The AMP shall be reviewed and updated as a minimum with the PSR, in order to incorporate new information as it becomes available, to address new issues as they arise, to use more sophisticated tools and methods as they become accessible and to assess the performance of maintenance practices considered over the life of the plant;
- Ageing management of the reactor pressure vessel (or its equivalent) and its welds shall take all relevant factors including embrittlement, thermal ageing, and fatigue into account to compare their performance with prediction, throughout plant life;
- Surveillance of major structures and components shall be carried out to timely detect the inception of ageing effects and to allow for preventive and remedial actions.

Other requirements established in the regulation on ageing management for nuclear installations:

- The licensee shall identify the systems, structures and components (SSCs) important for the safety of the nuclear installation, which cannot be replaced and the ageing of which limits the duration of the operational lifetime of the installation;
- The lifetime of the non-replaceable safety-related SSCs shall be evaluated taking

into consideration all the relevant ageing and wearing mechanisms known, the results of the in-service inspections, the internal and external operating experience, the results of the relevant research and the newest analytical instruments available. This evaluation has to be performed every 10 years since the start of operation and every time new information becomes available that indicates the need for the revision of the assumptions at the basis of the evaluation;

- The licensee shall identify the nuclear safety analyses that use assumptions related to the lifetime of the SSCs and their ageing, including analyses that have assumptions related to a certain planned / estimated operational lifetime of the nuclear installation as a whole. Such analyses include, for example, fatigue analyses, analyses at the basis of the environmental qualification of the SSCs, analyses of the behavior of primary heat transport system components in operation. These analyses have to be revised and revalidated periodically, as the need arises and at least every 10 years;
- The licensee shall describe the ageing management program in a stand-alone document;
- The document describing the AMP shall include at least the following information:
    - Philosophy, strategy, objectives, organization and resources allocated for the AMP, together with the way in which the AMP integrates the activities relevant for SSCs ageing management performed in the frameworks of other programs and processes;
    - The methodology for identification and the criteria for the inclusion of the SSCs in the AMP;
    - The nuclear safety analyses that use assumptions related to the lifetime of SSCs (TLAAs) and the records of the ageing of the respective SSCs;
    - The analyses and documentation regarding the ageing related degradation that has the potential to affect the nuclear safety functions of the SSCs, including the analysis of the variation / the trending of the failure rates of SSCs due to ageing;
    - The main ageing mechanisms of the SSCs and their effects;
    - The availability of the data necessary for evaluating the ageing related degradation of the SSCs;
    - The effectiveness of the operation and maintenance programs for the ageing management of the components that can be replaced;
    - The acceptance criteria and the safety margins imposed for the SSCs in accordance with the design requirements;
    - The measures established for monitoring the evolution of the ageing mechanisms and for their mitigation, in particular for the SSCs important for nuclear safety that cannot be replaced and the ageing of which limits the operational lifetime of the nuclear installation;

- Information on the physical state of the SSCs, including the current safety margins, as well as any other characteristics that may limit their service lifetime;
- The history of the inspection, surveillance, testing and maintenance of SSCs;
- The preventive activities implemented or planned to reduce at a minimum and to control the ageing related degradation of SSCs;
- The corrective actions for preventing ageing related failures of the SSCs and the establishment of the acceptance criteria in relation with which the need for corrective actions is evaluated;
- The use of research and experience exchange programs at national and international level, from which the licensee obtains relevant information for the optimization of the AMP;
- The use of internal and external operating experience, relevant for the ageing management of SSCs, including of the lessons learned at international level in the field of ageing management;
- The measures established for resolving aspects related to SSCs obsolescence;
- The list of codes, standards and guides used by the licensee for establishing the requirements and procedures of the AMP, in accordance with good practices recognized at international level in this field.

The licensees are complying with the regulation of NSN-17 and have submitted to CNCAN the relevant documentation of the AMP to demonstrate compliance with the regulatory requirements.

02.2 International standards

The regulation NSN-17 - Nuclear safety requirements on ageing management for nuclear installations incorporates all the relevant WENRA safety reference levels.

Regarding the use of international standards, the regulation NSN-17 - Nuclear safety requirements on ageing management for nuclear installations imposes on the licensees that they include in the documentation of their AMP the list of codes, standards and guides used in establishing the requirements and procedures for the AMP, in accordance with the good practices recognized in this field at international level.

The regulation NSN-17 also requires that the licensees include in the documentation of their AMP the use of research and experience exchange programs at international level, through which the licensees obtain information relevant to the optimization of their AMP. Moreover, the regulation NSN-17 also requires that the licensees include in the documentation of their AMP the use of internal and external operating experience, relevant for the management of ageing of the SSCs, including lessons learned at international level in the field of ageing management. This requirement actually aimed at imposing the use of IAEA’s International Generic Ageing Lessons Learned (IGALL) program, among other sources of lessons learned in this field.

The international standards and guides used for the ageing management of Cernavoda NPP mainly consist of the mandatory codes and standard requirements (e.g. ASME, CSA, ASTM, ANSI, IEEE, IEC, ISO) and best practice guides recommended by EPRI, WANO, INPO, COG, etc.

Cernavoda NPP carried out the first PSR for Unit 1 between October 2008 and June 2012. This comprehensive and systematic evaluation of nuclear safety was based on the practical experience of conducting such reviews within the European Union, and was in compliance with both national regulations (NSN-10 regulation on PSR, issued in 2006) and the IAEA guidelines of that period (safety guide NS-G-2.10), that included the safety factor "Ageing Management". At that time, the applicable codes and standards used in establishing the AMP requirements for Cernavoda NPP were included in the list of standards used for the performance of the PSR.

In accordance with the requirements of the NSN-17 regulation, Cernavoda NPP maintains the list of codes, standards, regulations, prescriptions and guides used in establishing the requirements and procedures for the AMP.

Examples of international standards and good practice documents used by Cernavoda NPP for their AMP include the following:

- Safety Requirements: SSR-2/1 Safety of Nuclear Power Plants: Design and SSR-2/2 Safety of Nuclear Power Plants: Commissioning and Operation;
- Safety Guides: NS-G-2.12 - Ageing Management for Nuclear Power Plants, NS-G-2.6 - Maintenance, Surveillance and In-service Inspection of NPPs and SSG-25 Periodic Safety Review for Nuclear Power Plants;
- Safety Reports: SRS 62 Proactive Management of Aging Management for Nuclear Power Plants, SRS 57 Safe Long Term Operation in Nuclear Power Plants, SRS 82 Ageing Management for NPPs- International Generic Ageing Lessons Learned (IGALL);
- IAEA-TR 338 Methodology for the Management of Aging of Nuclear Plant Components Important to Safety;
- TECDOC 1503 NPP life management processes: Guidelines and practices for heavy water reactors, TECDOC 1037 Assessment and management of ageing of major NPP components important to safety – CANDU pressure tubes, TECDOC-1998 Assessment and management of ageing for Concrete Containment buildings.

In 2008-2009, when Cernavoda NPP documented the ageing degradation mechanisms and elaborated PLiM manuals for the components selected in scope of the PLiM Program, only SRS 15 Implementation and Review of Nuclear Power Plant Aging Management Program and TECDOC 1503 were in effect as general guideline reference at international level. At that time, the most important guidance in defining the AMP and lessons learned from operating experience were the North-American model and US-GALL. As a consequence, Cernavoda NPP developed the AMP mainly following the INPO AP-913 model for Equipment Reliability, Preventive Maintenance Basis templates and Guidelines developed by EPRI and American / Canadian Code requirements.

Nevertheless, attention was paid to comply also with the IAEA standards and guidelines, when the selection of PLiM Components was performed and documented in the technical report. As such, the specific evaluation against IAEA guidelines, demonstratesd full compliance with IAEA guidelines defined in TECDOC 1503.

Since 2016, Cernavoda NPP has nominated representatives in IGALL discipline Working Groups and they participated to the annual meetings and documents' review, as requested. Romania has also representatives in the IGALL at Steering Committee level from SNN corporate and CNCAN.

As regards the TRIGA Research Reactor, the licensee has used the IAEA Specific Safety Guide No. SSG-10 - Ageing Management for Research Reactors for the development of its AMP, as well as the relevant documents referenced in the bibliography of SSG-10 (e.g. TECDOC-792 - Management of Research Reactor Ageing, NS-G-4.2 - Maintenance, Periodic Testing and Inspection of Research Reactors).

02.3A Description of the overall ageing management Program for Cernavoda NPP

02.3.1A Scope of the overall AMP for Cernavoda NPP

a) Assignment of responsibilities within the licensee's organisation to ensure an overall AMP is developed and implemented

Starting with 2008, after Unit 2 started commercial operation, Cernavoda NPP adopted a new policy to ensure the reliability of critical SSCs at competitive costs developing technical programs (common for both units) according to INPO AP-913 Equipment Reliability philosophy.

To attain this objective, a new process was developed - "Maintaining Equipment Reliability" process (MERP), as the most important pillar of the overall Aging Management strategy for Critical SSC.

Important modifications have been implemented to address the new organizational and procedural needs, personnel qualification requirements and changes to duty areas, in order to integrate the new processes and reinforce the old ones and to sustain a nuclear power plant with 2 units in operation. The leading role to implement the MERP process was attributed to a new group - Component Engineering Department, which had the main task to develop technical programs, consisting in a set of preventive activities, inspections, testing, surveillance and condition / life assessments to maintain SSCs reliability over the plant life.

For this purpose, Cernavoda NPP adopted the organizational structure of the Technical Unit presented in Fig.2.1.

The Process Systems Department (DSP) has the overall responsibility and accountability for the safe and reliable long term operation of the assigned systems.

The Design Engineering and Technical Support Department (DPST) has the overall responsibility and accountability to maintain Design Basis and Configuration Control.

The Component Engineering Department (DIC) has the mission to ensure equipment reliability over plant design life by developing Inspection, Preventive Maintenance and Life Cycle Management (PLiM) Programs and to continuously improve their performance, as per the current codes, standards and good practice guidelines.

At station level, besides the MERP, other processes have been defined to ensure the safe operation, reliability and economic competitiveness of the plant in accordance with licensing basis.

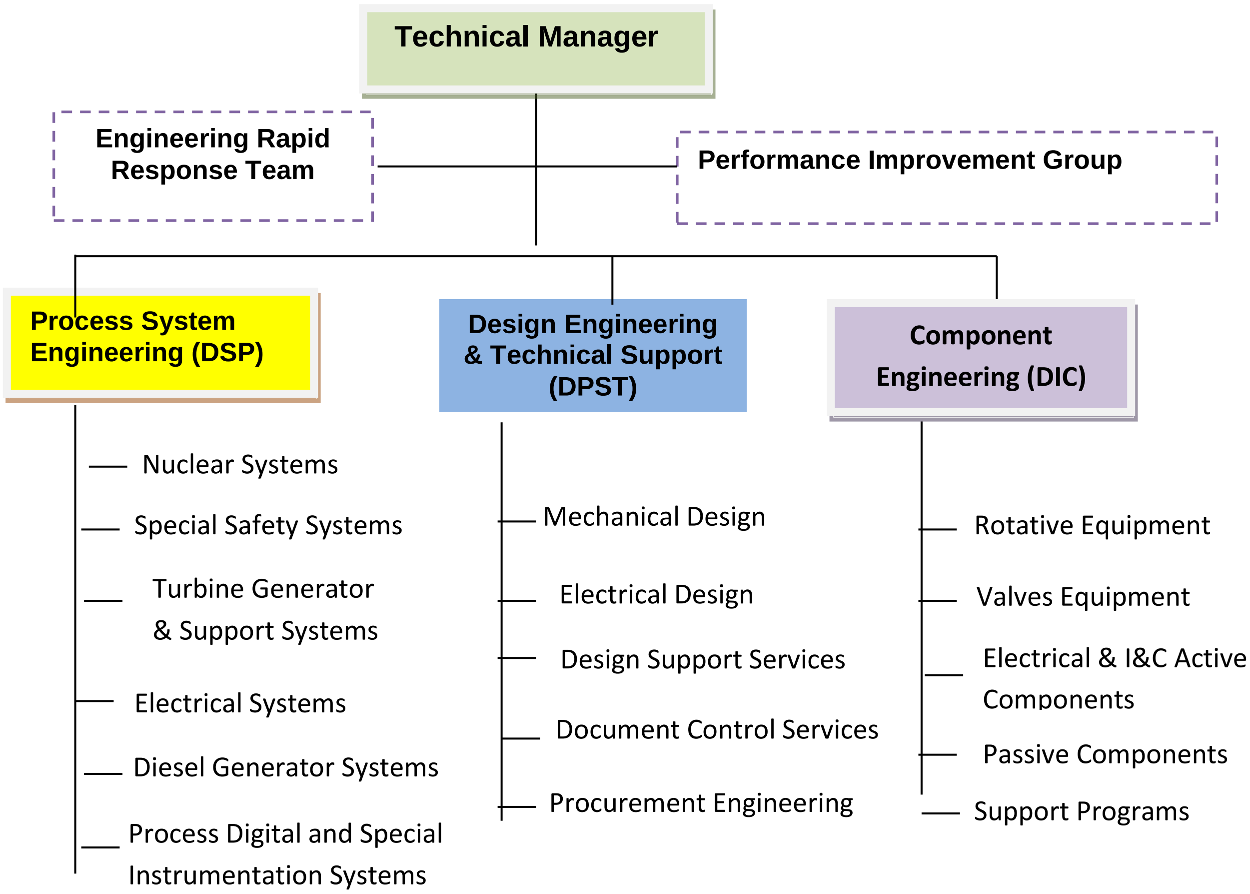


Fig. 2.1

In order to meet the above requirements, a set of principles have been defined, which underpin the activities related to the MERP process:

- The organization resources are focused on critical SSCs to ensure safe and reliable plant operation.
- Ensuring and maintaining the reliability of critical SSCs is coordinated by suitably organized technical staff: System Engineering (DSP) and Component Engineering (DIC). Other departments / services provide specialist support, as appropriate.
- Control over design configuration and design bases is provided by DPST personnel who support Responsible System/Component Engineers in design margin analysis.
- The organization ensures a process for continuous improvement and alignment to the latest international standards, guides and best practices applicable for Equipment Reliability (ER).

Based on MERP philosophy and principles, various programs have been developed, in accordance with IAEA, INPO, EPRI Guides, at system or component level for timely identification and mitigation of ageing degradation mechanisms and effects, as follows:

- Preventive Maintenance (PM) programs for active components (templates of predefined tasks for generic groups of components);

- Plant Life Management (PLiM) programs mainly for passive components, for which PM strategies are not sufficient to control degradation and require "Management" of complex ageing issues;
- Support Programs (e. g**.** Equipment Qualification, Shelf Life, Periodic Inspection, Predictive Maintenace, Proactive Obsolescence, Testing) to ensure oversight coordination & systematic approach.

Besides defining PM/PLiM and support programs, other activities conducted at station level ensure the efficient control of Ageing Effects as per the "PLAN-DO-CHECK-ACT" loop:

1. Ensuring the resources for implementation in terms of spare parts availability, or inspection tools and expert services required for implementation
2. Planing of programs activities, depending on regulatory commitments, mandatory testing intervals, maintenance windows, and last, but not least: availability of spareparts and services
3. Implementing the programs for Equipment Reliability providing feed-back to technical responsibles about inspection, testing, surveillance results
4. SSC performance monitoring, implemented through specific Health Monitoring programs developed at System and at Component level to detect ageing degradation trend and margin left
5. Taking corrective actions**,** in case of SSC degraded performance, to mitigate ageing, avoid failures and improve SSC s condition
6. Periodic assessment of program efficiency and compliance based on internal /external OPEX, R&SD new findings or against international codes & standards.

These activities are sustained by several processes, which involve almost all the plant departments:

1. Acquisition and Materials Control
2. Work Control and Planning
3. Operations and Maintenance
4. Technical Unit, with support from Safety & Licensing, Chemical Service, Fuel Handling personnel
5. Performance Monitoring and Improvement Group
6. Development and Monitoring of Management Systems.

However, the most significant activities for ageing management control and mitigation of ageing effects are performed by Technical Unit personnel with support from Safety & Licensing, Reactor Physics, Chemical Service, in developing, assessing and continuously improving the AMP. They assist RSEs (Responsible System Engineers) / RCEs (Responsible Component Engineers) with their specific expertise in evaluation of the operational or design margins as per Current Licensing Basis (CLB).

The overall AMP strategy, indicating the responsibilities within Cernavoda NPP organization to develop and implement specific AMP activities, as recommended in the IAEA NS-G-2.12 is presented in Fig. 2.2.

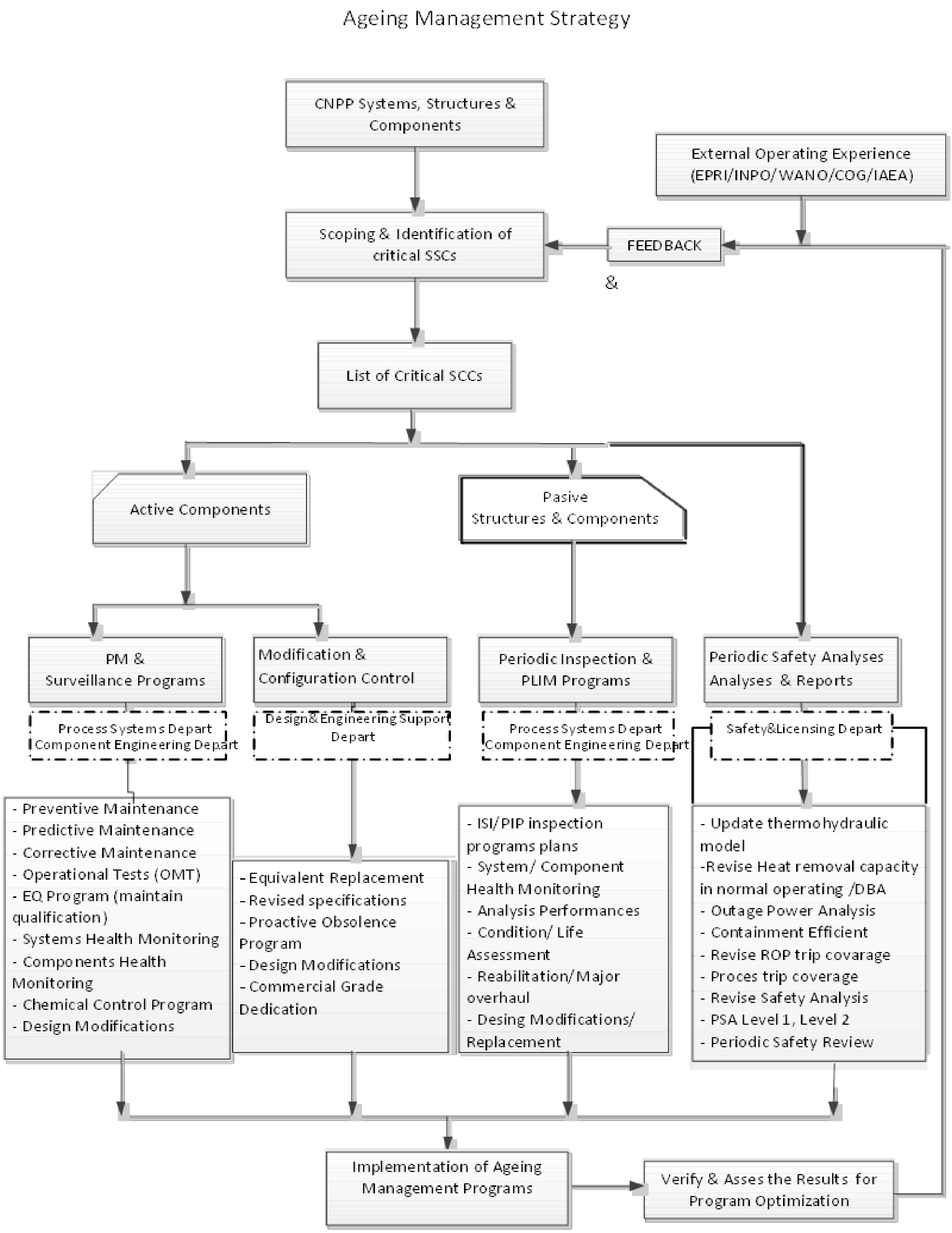

Ageing Management Strategy
CNPP Systems, Structures & Components
External Operating Experience (EPRI/INPO/WANO/COG/IAEA)
Scoping & Identification of critical SSCs
FEEDBACK
&
List of Critical SCCs
Active Components
Pasive Structures & Components
PM & Surveillance Programs
Modification & Configuration Control
Periodic Inspection & PLIM Programs
Periodic Safety Analyses Analyses & Reports
Process Systems Depart Component Engineering Depart
Design& Engineering Support Depart
Process Systems Depart Component Engineering Depart
Safety&Licensing Depart
- Preventive Maintenance
- Predictive Maintenance
- Corrective Maintenance
- Operational Tests (OMT)
- EQ Program (maintain qualification)
- Systems Health Monitoring
- Components Health Monitoring
- Chemical Control Program
- Design Modifications
- Equivalent Replacement
- Revised specifications
- Proactive Obsolence Program
- Design Modifications
- Commercial Grade Dedication
- ISI/PIP inspection programs plans
- System/ Component Health Monitoring
- Analysis Performances
- Condition/ Life Assessment
- Reabilitation/Major overhaul
- Desing Modifications/ Replacement
- Update thermohydraulic model
-Revise Heat removal capacity in normal operating /DBA
- Outage Power Analysis
- Containment Efficient
- Revise ROP trip covarage
- Proces trip coverage
- Revise Safety Analysis
- PSA Level 1, Level 2
- Periodic Safety Review
Implementation of Ageing Management Programs
Verify & Asses the Results for Program Optimization

Fig 2.2

The licensee’s organizational arrangements and processes ensure that the AMP activities for Cernavoda NPP are budgeted, planned, implemented, assessed and improved, in accordance with the current IAEA standards. Cernavoda NPP has in place all types of activities recomended by IAEA guidelines, to ensure systematic control of ageing: PM, Water chemistry, In-service Inspection, In-service Testing, Environmental Qualification (EQ), etc. Arrangements are in place with external organizations to benefit from international R&D findings, international operating experience and lessons learned, benchmark the best methodologies and approaches, in order to continuously improve AMP efficiency.

b) Methods used for identifying SSCs within the scope of overall AMP

Cernavoda NPP has a policy associated with the critical SSCs classification in various categories of importance, which represents a graded approach to establish, as the first step, the critical systems and, in a second phase, the critical components, based on the impact on plant's capacity to operate safely and reliably. The process also establishes that the priority to address eventual SSCs problems depends on the impact of SSCs failure on maintaining critical safety functions, or on their potential to affect the critical functions.

The selection of the critical systems was performed by RSE based on Functional Failure and Effect Analysis, taking into account the effect of their malfunction on the plants’ capability to operate safely, reliably, and efficiently from the nuclear safety, environmental, radiological and production perspectives.

Using this approach, critical systems were identified. The list of critical systems selected in the scope of AM strategy was documented in an information report which represents the main reference in prioritizing plant programs and projects.

Secondly, the selection of critical components was performed by RSE and RCE, with expert panel support, according to the following general considerations:

- Determination of component criticality is consequence-based - the higher the consequence associated with the failure of the component, the higher is its criticality.
- Both active and passive functions of a component are considered.
- Consideration must be given to the role of a component in a system (e.g. in a critical system, not all components are critical and each must be examined separately and in a non-critical system, there cannot be a critical component; otherwise such a system would have been classified as critical system).

Based on the above considerations and safety impact, the critical components to be addressed through PM Programs are classified in:

- SPV (Single Point of Vulnerability)
- SC1 (Safety Critical):
- OC2 (Other Critical)
- PC3 (Production Critical)
- Non-critical components: fulfilment of legal requirements for non-critical SSC.
- Run-to-Maintenance (RTM) part of non-critical systems, for which periodic testing may still be performed, if required.

Details on SSCs screening for criticality and grouping for PM programs strategy are detailed below:

SPV (Single Point of Vulnerability): If the component failure could cause reactor or turbine/generator automatic trip or impose manual shutdown.

SC1 (Safety Critical): If failure of the component could cause:

- Safety or Safety Related System Impairment;
- Inability to mitigate accidents or implement emergency procedures;
- Indication of moderate or high risk significance for safety, based on PSA models.

OC2 (Other Critical): If failure of the equipment could cause:

- Failure to comply with legal, regulatory requirements (other than the regulations issued by CNCAN) for SSCs which concur to safe production of electricity.
- An unacceptable increase in environmental or radiological safety hazards, as resulted from PSA.

PC3 (Production Critical) If failure of the component could cause:

- Unplanned power reduction ≥5% FP (full power);
- Operator and/or maintenance workaround required to maintain the unit at full power;
- The standby component to be put in service to avoid unplanned power reduction or unit shutdown.

Non-critical components: If not already classified as "critical", but any of the following criteria is satisfied:

- Component failure causes operator and/or maintenance burden;
- Component replacement/repair will cost ≥10000 EUR;
- The component is obsolete and design modification will be required for replacement;
- There is a long lead time for parts or replacement component delivery which will not allow timely repair.
- The component ensures fulfillment of legal requirements for non-critical SSCs.

Run-to-Maintenance (RTM) mainly equipment/components part of non-critical systems, for which:

- Failure of the component does not degrade a critical function;
- PM does not represent a cost effective solution to ensure its operability;
- Unavailability can be tolerated for extended periods of time without any consequence or concern;
- Periodic testing may still be performed, if required.

First, the active critical components were grouped into generic categories, such as: valves, pumps, compressors, motors, driers, medium voltage cells, instrumentation transmitters, convertors, cards, etc.).

After that, the critical components of one generic type were grouped into families, based on the same model, manufacturer, operating conditions, etc.). For a family, a template of PM tasks and frequencies was established, consisting of: inspection, testing, replacement activities or activities needed to meet other regulatory commitments.

Basically, it was an expert panel decision between EPRI predefined PM Basis templates, manufacturer recommendations, actual operating conditions and internal operating experience, to decide on how aggressive the PM program should be.

Detailed guidance on how to define PM Programs activities and frequencies has been defined in a specific procedure (Guide for PM Programs documentation), taking into account: component design function, materials, manufacturer recommendations, normal and abnormal working conditions, duty cycles, environmental stressors and industry best practice guidance models for PM program tasks.

For a family, a leader component was selected, which in most cases met the worst working conditions.

Initially, the PM Programs were defined very conservatively, as the PM tasks were defined such as to ensure the reliability target for the leader, usually the SPV, which was the "worst case scenario".

Starting with 2014, based on the feedback received from implementation, especially the "as-found" condition, reported during component major revisions /overhauls, the PM matrixes were differentiated for every component, in order to eliminate the conservatism induced by the family, which was set to meet the requirements for the leader component.

Next step, after finalizing the SSCs segregation according to the new criticality classes in 2014, is to revise PM Programs tasks depending on the new component criticality described above (SPV, SC1, OC2, PC3).

According to the new criticality approach (the PM Programs should have different requirements in terms of tasks and frequencies, depending on component criticality, going down from very robust PM Programs for SPV, less for SC1 and lesser and lesser for OC2 and PC3 components, until fulfillment of only legal requirements for non-critical ones.

The list of PM Programs developed for generic types of critical components is presented below.

<u>Mechanical - PM:</u>

- Air operated valves (AOV)
- Manual valves (MV)
- Check valves (NV)
- Motorized operated valves (MOV)
- Pumps
- Compressors
- Dryers
- Heating, ventilation and air conditioning (HVAC)
- Safety / relief valve (RV)

<u>Instrumentation and Control (I&C) - PM:</u>

- Nuovo Pignone panels

- Marconi Logic panels

- Electronic controllers (EC)

- Switches

- Transmitters

Electrical - PM:

- Electric Motors – High voltage

- Power center

- Motor control center (MCC)

- Cells medium voltage (CMV)

- Electric Motors – Low voltage

Special equipment - PM:

- Emergency Diesel Generators

- Standby Diesel Generators (SDG)

- Containment gamma detector

- Reactivity mechanisms

- Distributed Control System (DCS)

- Digital Control Computers (DCC)

- Reactor Building

- Chillers

- Switchgear Equipment – Main Breaker

- Power transformers

- Cells High Voltage (CIV)

- Airlocks (including pneumatic / hydraulic actuated components)

- Class I and II Equipment Batteries

Special equipment – PliM components – detailed in the following.

According to the MERP philosophy, Cernavoda NPP developed PLiM Programs are developed for plant assets with major impact on safety, production or environment, which require proactive identification of aging mechanisms, that cannot be addressed by normal plant PM strategies.

Guidance on how to manage and document the ageing degradation mechanisms and the corresponding mitigating activities for ageing effects and define PLiM Programs activities are detailed in a specific procedure for PLiM Program Manual elaboration.

In order to select the components requiring a PLiM Program, after documenting the ageing degradation mechanisms and the corresponding mitigating activities for ageing effects, a supplementary evaluation was performed in respect with the following criteria:

1. components that, in case of major degradation, need a long outage to repair, exceeding the extent of a normal Planned Outage;

2. require an action plan to address the reliability problems > 1.000.000 EUR;
3. high complexity components, with a limited life, which might become “obsolete”, cannot be replaced, or which have associated TLAAs (Time-Limited Ageing Analyses) in the Current Licensing Basis;
4. components which, in case of improper operation / maintenance, have serious consequences on nuclear safety, power production, public or environment, requiring a “Life Assessment” study.

If any of the above criteria is met, then the component is considered “PLiM component”, for which a PLiM Program should be developed. Where not very clear, the components were included conservatively into PLiM Program.

The list of PLiM components is provided below:

- Power transformers
- Fuel channels
- Steam generators
- Feeders
- Heat exchangers
- Piping surveillance:
    - Flow-Accelerated Corrosion (FAC)
    - Supports
    - Snubbers
    - Buried piping
    - Expansion joints
    - Small bore piping
    - Tubing
- Turbogenerator
- DCC
- Reactor Building
- SDG
- Cables.

As the list of PLiM component contains mainly unique components, for which the ageing degradation mechanisms may not be completely known, or for which other new degradation mechanisms (not taken into account in the design) might appear, a close contact with the industry and R&D support is strongly recommended.

The selection of Cernavoda NPP PLiM components was finalized in 2009. The approach of Cernavoda NPP was validated afterwards, when the list of PLiM components was compared with other CANDU 6 plants (Point Lepreau and Wolsung 1), which seemed to have developed a similar selection of PLiM components. In 2009, CNPP approach was compared also with IAEA TECDOC 1503 and was considered to be in line with the recommendations in effect at that time.

In conclusion, the selection of SSCs within the scope of overall AMP for Cernavoda NPP is in line with IAEA documents requiring to ensure the fulfilment of the critical safety functions, the critical support functions and is also taking into account PSA results for normal, abnormal and accident conditions. Determination of component criticality/importance is consequence-based (the higher the consequence associated with the component failure, the higher its criticality/importance). Both active and passive functions of a component are considered.

c) Grouping methods of SSCs in the screening process

To establish a cost-effective process to address all the potential threats to plant safe and reliable long term operation, Cernavoda NPP has developed 2 types of programs for the critical components selected in scope of MERPs:

- Preventive Maintenance (PM) programs for generic active components (predefined tasks repeated periodically for families of components, based on EPRI PM Basis templates for different types of components),
- Plant Life Management (PLiM) programs for complex, unique components, mainly passive, for which PM programs are not considered sufficient to control degradation, as confronted with complex ageing issues.

For major passive components, specific to CANDU design, (Fuel Channels, Steam Generators, feeders) the PLiM program consists of periodic inspections and surveillance activities based on the CAN CSA N 285.4 standard requirements.

Supplementary prescriptions apply to Pressure Tubes, according to CSA N285.8 - Technical Requirements for In-Service Evaluation of Zirconium Alloy Pressure Tube in CANDU NPP and to feeders as per ASME, ASTM codes.

For the Reactor Building, the PLiM program consists of periodic inspections of metallic embedded components in accordance with the requirements of the standard CSA-N285.5 - Periodic Inspection of CANDU Nuclear Power Plant Containment Components.

Other requirements for containment materials and testing are mandatory in accordance with the standards CAN3-N287.5-M81 - Testing and Examination Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants, CAN3-N287.2-1977 - Material Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants, or CAN3-N287.3 - M82/1990 Design Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants.

The passive components not specific to CANDU were grouped in a Piping Surveillance Program**,** defined for: FAC, supports and snubbers, buried piping, expansion joints, Small Bore piping, Tubing.

The inspection program for FAC, supports and snubbers is defined based on ASME Sect. XI Nuclear In-service Service requirements.

Other technical programs for Piping surveillance subprograms (Buried piping, Expansion Joints, Small Bore piping) were developed in accordance with EPRI guidance.

Due to the large number of components in the scope of FAC, Buried piping, the inspections are planned in accordance with the risk ranking methodology, calculated with a software application; depending on the impact in case of failure, operating conditions and thinning/failure rate (CheckWorks, BPWorks, SnubWorks).

For PLiM active components (Turbogenerator, Main Transformers, DCC, Diesel Generators) the program consist of PM requirements for periodic preventive tasks, predictive tests or planned replacements, overhauls, as well as design modifications to resolve obsolescence or SPV mitigation, in accordance with IEEE standards, EPRI Guides and manufacturer recommendations.

In conclusion, for critical active components, the grouping method was based on generic type and families, for which PM templates were determined based on international experience. For passive components, PLiM Programs were defined in accordance with Canadian standards and to American codes, as applicable. For active PLiM components, IEEE standard, EPRI guidance or Manufacturer recommendations were used in defining PLiM activities and frequencies. For the PLiM Programs with a large number of components in scope, risk ranking tools are used to prioritize inspections depending on failure impact, in order to ensure plant safe and reliable operation.

d) Methodology and requirements for evaluation of the existing maintenance practices and developing of ageing programs appropriate for the identified significant ageing mechanism

As already mentioned, a PLiM Program is developed for unique, complex components and require important resources to be allocated, therefore a thorough, comprehensive methodology was established to determine its necessity.

For every PLiM component, the ageing degradation mechanisms and the activities to mitigate ageing effect are documented in the PLiM Manual, elaborated according to the methodology briefly described below:

1. Provision of a short technical description of the PLiM component
2. Identification of the significant ageing degradation mechanisms: first, all the mechanisms and modes of potential degradation were identified and documented in a global list based on the design basis documentation and operating experience, then a list of plant specific degradation mechanisms and modes has been established by eliminating from the global list those not applicable to Cernavoda NPP due to design and operating conditions differences; then, an analysis of the potential consequences was performed and the mechanisms and modes of degradation with no impact on the function of the components have been eliminated from the plant specific list; finally, the areas / components affected by the relevant mechanisms and modes of degradation have been identified and documented.
3. Establishment of the activities required to mitigate the ageing degradation mechanisms: an evaluation of the relevance and importance of the degradation mechanisms was performed, taking account of their evolution and monitoring, as well as of their consequences and impact; preventive and mitigating solutions were evaluated, established and implemented.
4. Determining the long term plan to attain PLiM Component design life (Life Cycle Management Plan - LCMP). The plan is a combination of the following activities:
    - Testing, inspection, preventive maintenance,
    - Condition monitoring,
    - Actions to mitigate and control the ageing,

- Continuous contact with relevant international industry experience.

Depending on component type passive/active, LCMP is using testing and inspection activities to detect ageing effects and monitor the condition for passive components, or by performing PM tasks and monitoring failure rate for active ones.

Different inspection requirements are defined for passive PLiM components or passive functions of active components (e.g. pressure boundary integrity). For instance, for nuclear major CANDU components it is performed in accordance with Periodic Inspection Program (PIP) detailed in station procedures based on the applicable standards and approved by CNCAN.

For active components selected in PLiM Program, the most common ageing problems is related to the obsolescence of electronical subcomponents, as is the case for DCC, Turbogenerator, Diesel Generators.

For passive components, the PIP ensures the monitoring of the evolution of ageing effects (loss of material /thinning, dimensional/ structural modifications, weld cracking, fatigue. etc.) and assessment of the component condition against certain acceptance criteria, usually indicated in the standards and codes. The acceptance criteria are normally set such as to ensure fitness for service until the next scheduled inspection. The inspection results and condition assessment that the component is fit to sustain the fulfillment of component design functions until the next scheduled inspection are reported to CNCAN.

For active components, specific condition monitoring parameters are set, referring to the failure rate, which are trended and reported if adverse trend is detected.

The condition of all PLiM components is monitored against specific performance indicators (PIs) established in correlation with the acceptance criteria, having administrative limits (set with sufficient margin to allow mitigating actions to be implemented). The individual PLiM component performance is reported once per year as part of Component /System Health Monitoring Process.

New actions to mitigate and control the ageing could be necessary based on the feed-back from implementation. This will impose the revision of the actual PLiM Programs, to ensure a high level of confidence in PLiM components ability to perform according to design requirements, in all normal, abnormal or accident conditions.

Depending on the case, PLiM Programs revision could be triggered by:

- adverse trend in PIs evolution, or different trend than expected,
- relevant operating experience identifying new ageing mechanisms, not previously taken into account,
- revision of codes and standards requirements in latest editions,
- Life Assessment results,
- Periodic self-assessment of PLiM Programs efficiency, to ensure compliance with international best practice in ageing management.

At present, Life Assessment studies have been performed for the majority of PLiM Components. They have been contracted to be developed in a standard format which corresponds to the list of degradation mechanisms, locations, contributors, materials and operating conditions developed in the PLiM manuals. This ensures conditions for an easy and fast revision, upon validation of the current PLiM Programs or recommendations for improvement of the PLiM Manuals, as the Life Assessment studies have been contracted with subject matter experts, usually the designer or manufacturer of the most important assets.

Cernavoda NPP has continuous contact with relevant international industry experience. For more complex ageing issues, arrangements are in place with CANDU Owners Group (COG) to benefit from international operating experience lessons learned and from the latest Research & Development (R &D) results. Cernavoda NPP is participating to COG R&D Program since 2006 and it collaborates also with local design and research institutes. In 2016 Cernavoda NPP has joined also the IGALL Program, in order to maintain a close collaboration with other NPPs, more experienced in the implementation of AMPs.

e) Quality assurance of the overall AMP in particular

According to the new functional groups and duty areas created to support MERP implementation, resources have been ensured for PLiM Program such as:

- a PLiM Coordinator to ensure Single Point of Contact duties and overall monitoring and reporting requirements;
- the supervision role is detained by a Program Engineer responsible for each PLiM component.

The PLiM Coordinator ensures the following coordination duties:

1. Documents components selection to be included PLiM program scope and the overall implementation strategy.
2. Documents the long term Plant Life Cycle Plan (PLCP).
3. Monitors the adherence to the implementation schedule of PLiM activities and planned budget execution, for the periodic update of PLCP.
4. Documents the integrated AMP and ensures its periodic revision to support the alignment to the latest good international practices. Ensures also the implementation of the lessons learned from internal or international operating experience, or on the latest findings from R&D studies.
5. Ensure the transmittal of PLiM program documents to CNCAN for regulatory review.

In addition, all RSE/RCE are responsible identification of future major problems which might challenge the plant SSCs. Newly identified major problems are included in the next System Monitoring or Component Health Monitoring report and the validated technical problems with high impact will be then solved by: design modifications, capital investment project, supplementary analysis to refine the initial design assumptions, or participation in integrated COG Joint Projects or R&D studies, for generic CANDU complex issues. The process to include new major problems in the PLiM Program ensures that PLiM is a living program, in continuous improvement. It also provides a formal confirmation process to ensure that preventive actions are adequate and appropriate and that all corrective actions have been completed and are effective.

Collection and storage of data and trending of information on maintenance history and operational data is performed for all SSCs covered by the AMP, in accordance with station specific procedures that are part of Cernavoda NPP Integrated Management System. As already mentioned, the condition of all PLiM components is monitored against specific performance indicators (PIs) established in correlation with the acceptance criteria.

Administrative controls are in place that document the implementation of the AMP and ensure that actions taken, in form of the strategic documents being approved at Station Manager level or at corporate level.

The overall effectiveness of the AMP is verified periodically through self-assessments, using all the relevant information (e.g. from inspection, testing, surveillance, maintenance, changes in regulations and standards, internal and external operating experience, new R&D results, etc.)

02.3.2A Ageing assessment for Cernavoda NPP

a) How key standards and guidance, as well as key design, manufacturing and operations documents are used to prepare the overall ageing management program

Based on the individual AMPs and LCMPs developed for each PLiM component according to the methodology previously described, an overall Plant Life Cycle Plan (PLCP) was developed. The PLCP represents the sum of individual level plans developed for each PLiM component with main milestones and corresponding budget.

PLCP contains all the inspection, testing in accordance with codes and standards requirements, PM activities, as well as technical assessments and studies (COG R&D included) necessary to attain or extend component end of life.

Since 2016, when the specific regulation for ageing management (NSN-17) was issued by CNCAN, a new requirement was identified to document the overall AMP program in a stand-alone document. The requirement on the content of the document describing the obverall AMP has already been presented in Section 02.1 National regulatory framework.

The overall AMP for Cernavoda NPP is described in an integrated manner, consolidating all the available information from different sources in a single document at station level:

- Reference documents and station procedures,
- List of critical systems, components or PLiM components,
- List of codes and standards
- PM or PLiM Manuals, inspection reports,
- System/component health monitoring plans and reports,
- Other technical strategies (PLiM implementation Strategy, Safety Analysis Strategic Plan),
- Final Safety Analysis Report, chapter 15 (for deterministic Safety Analysis).

Other history data, such as: PM feedback, internal/external operating experience and R&D reports are also available in different databases.

However, the intention of the overall AMP document is to describe the overall process defined for ageing management, outlining the attributes of a comprehensive and systematic control of ageing and giving direction on where the specific detailed information can be found.

b) Key elements used in plant programs to assess ageing

Cernavoda NPP has developed a performance monitoring process for critical SSCs in a graded manner:

- since 2004, Health Monitoring at System Level,
- since 2011, Component Health Monitoring,
- since 2013, Program Health Monitoring.

The System/Component Health monitoring program was developed in accordance with System/Component Health monitoring plans |(SHMP/CHMP) containing the following data:

1. System /Component performance objectives/goals (failure rate or ageing trend),
2. FFA (Functional Failure Analysis) including: Functional role; Failure Mode and Causes; Degradation Mechanism and Indicators; Failure Location; Failure Cause,
3. Routines, outage walk-downs, inspection plans (in the form of a detailed review sheets),
4. Performance indicators with acceptance criteria.

A continuous assessment of system/component health and performance is ensured through RSE/RCE permanent review, analysis and trending of data collected from station logs, Work Management System, abnormal condition reports, operating experience, walk-downs. The purpose of this data is to assess current health and to predict future health of a component, anticipating major defects/degradations or generic deficiencies that can be mitigated by timely implementation of appropriate preventive measures, to avoid unwanted consequences.

A Program Health Monitoring (PHM) process was also developed to support the timely identification of relevant ageing and wear-out mechanisms and potential age related degradations. PHM Reports are prepared for the coordination of support Programs (Environmental Qualification, Predictive Maintenance, Shelf Life, Fluid Leak Management) and were defined to provide a measurable indication for the overall supervision and standardized documentation and implementation, in line with continuously changing requirements and approaches at international level.

c) Processes/procedures for the identification of ageing mechanisms and their possible consequences

The processes/procedures for the identification of ageing mechanisms and their possible consequences have been presented in Section 02.3.1A Scope of the overall AMP for Cernavoda NPP - d) Methodology and requirements for evaluation of the existing maintenance practices and developing of ageing programs appropriate for the identified significant ageing mechanism.

d) Establishment of acceptance criteria for ageing

Performance indicators (PIs) have been established in health monitoring plans in the form of measurable parameters that indicate the degree to which a component is able to perform its intended design function(s).

Mandatory PIs - provide a general overview of component health process. The indicators (direct, indirect) are mandatory to establish consistent performance for Components Programs.

Direct PIs are those indicators that are directly quantifiable from the operation of equipment (e.g. component functional failure - valve stroking times; pump vibration, pressure, temperature, etc).

Indirect PIs are historical, programmatic information for a component that can provide a qualitative assessment health (e.g. Deferral of PM activities, Corrective Maintenance Backlog, number of TD-MODS, etc).

Specific and additional PIs are indicators of component reliability and/or condition. Depending on component type active or passive, different performance objectives are defined in Component Health Monitoring Plans (CHMPs).

Specific PIs are established at the component type level in scope to reflect component condition and to predict component health. These PIs are established considering the degradation mechanisms and methods to detect the degradation.

Specific PIs are used for:

- evaluation of long term trends,
- determination of whether acceptance criteria are being met or whether there is deterioration over time,
- determination of whether there is any risk or threat for safe and reliable continued operation of the system/plant.

For the PIs acceptance criteria and immediate action limit are defined, as applicable.

For active components specific PIs are used to identify the generic deficiencies and will include trends:

- from Predictive Maintenance Feedback (Vibration Monitoring, Infrared Thermography Measurements, Oil Analysis, Viper Results)
- from Preventive Maintenance (e.g. electrical parameter for motors, ripple for power supplies, current consumption for I&C equipment)

An immediate action limit is established for each indicator/sub-indicator and is determined based on the components inability to perform its function, from operation or integrity point of view, because of condition degradation (e.g.: minimum allowable thickness, chemical parameter values).

When the monitored parameters exceed the immediate action limit during the reporting period, an abnormal condition report will be issued and the impact on the component integrity or capability to accomplish the operational role is evaluated for both short/medium term and the long term. Based on the analysis results the required corrective actions are identified.

Any SSC anomalies, deficiencies, degradations or other indication of failure or falling performance in respect to the acceptance criteria, are addressed and documented, as required:

- Raise work requests;
- Issue Abnormal Condition Reports;
- Perform Technical Operability Evaluation (TOE).
- Raise ECR (Engineering Change Request).
- Other appropriate actions that address the issues.

Any SSC degradation, failure and urgent operational or maintenance issues are analyzed by and dispositioned according to the impact: TOE for issues with nuclear safety impact, or Operational Decision Making (ODM) for those with production impact.

RCE is taking the leading role in ensuring that Component Plan Recommendations (CPRs) from the approved CHMRs are implemented at the deadlines established. Where applicable, the RCE completes actions or works with other departments staff to complete actions as per station processes. Approved CPRs are ranked / graded and managed as per specific procedures.

The RCE follows up CPRs implementation as scheduled, to establish the effectiveness of these actions and give a feed-back to PLiM Coordinator, in charge with the administrative duties related to monthly CHM Reporting and overall PLiM Program Coordination monitoring and reporting via PHMR.

e) Use of R&D programs

The R&D Program includes an element of looking forward not addressed by engaging single suppliers or joining a joint project to address a specific issue. The intent of R&D is to do work that allows more accurate prediction of future behaviour and hence the capability to develop mitigating strategies before safety issues and/or economic losses occur.

Participation in R&D Programs has certain benefits, as compared to individually financing the work on specific problems:

- Access to an integrated approach for problem solving;
- Enhancing the capabilities of Cernavoda NPP staff by access to Technical Committees and Working Groups (as a form of maintaining and continuously improving technical competences);
- Use of predictive capabilities for asset management.

This is a key element of the overall AMP, as R&D deliverables can be used as inputs to maintenance strategies.

Therefore, since 2006 Cernavoda NPP is participating to COG R&D Program and collaborates also with local design and research institutes (Nuclear Research Institute – ICN Pitesti, Center for Technological Nuclear Engineering – CITON).

At plant level, Cernavoda NPP participation in COG R&D sub-Programs: Fuel Channels (FC), Safety and Licensing (S&L), Chemistry, Materials and Components (CM&C), Health, Safety & Environment (HS&E) and Industry Standard Toolset (IST) ensured access to state-of-the-art reports elaborated by subject matter experts and research professionals, which were used in developing PLiM Manuals, or improving the computer based predictive tools, used in assessing cumulative effects of PHT System Ageing.

Results of R&D studies are used also in defining Health monitoring Plans, or “fine tunning” of the Specific PIs acceptance criteria, or in setting appropriate limits for additional indicators.

f) Use of internal and external operating experience

Using lessons learned from the external operating experience information within current activities of any nature and the implementation of the actions resulted from the operating experience feedback process is a day-to-day practice in the entire Cernavoda NPP organization.

In the last couple of years it is part of daily duties for the coordinating personnel (Superintendents, Specialist Engineers) when attending the Technical Unit Meeting to assess the applicability of a historical event proposed for evaluation each day. For each applicable event, an external Abnormal Condition Report is initiated, which is then processed in accordance with specific station procedure requirements.

One level down, the Single Point of Contact for operating experience (a dedicated person in every department) is responsible for:

- Transmitting questions and answers to the relevant questions to the COG/WANO contact officer in order to be posted on COG/WANO websites;
- Initiation of external abnormal condition reports based on external operating experience, when the information is considered applicable.

Periodically System/ Component Health Monitoring Reports are presented in front of HMR (Health Monitoring Review) Board for approval of major technical problems identified at system/component level, as recommended in the final section of SHMR/ CHMR according to the specific departmental procedure.

Every RSE/RCE, in his /her SHMR/CHMR, places a special focus on:

- Identifying Persistent Component Issues and developing long-term recommendations and activities;
- Highlighting of the problem with medium/long term impact (e.g: obsolescence, spare parts availability);
- Reviewing all Long Term Actions (assessing status of these recommendations and initiating appropriate actions if resolutions deadlines are not being met);
- Reviewing internal and external operating experience for component related issues and sharing / disseminating the relevant operating experience.

Any relevant and significant internal operating experience that would apply to component performance and issues is assessed and reported periodically. The assessment takes into account the extent of condition and extent of the problem in order to prevent future similar failures. The recommendations / actions to be taken to prevent the occurrence are formulated accordingly.

External operating experience (compiled by COG, INPO, WANO, IAEA, regulatory authorities from other countries, such as the Canadian Nuclear Safety Commission, US Nuclear Regulatory Commission), identified during the reporting period, are evaluated for relevant events that would apply to component performance. Any recommendations to prevent its occurrence take into account the type of the problem (specific or generic issues).

02.3.3A Monitoring, testing, sampling and inspection activities for Cernavoda NPP

The main objectives of Cernavoda NPP Maintenance Surveillance and In-Service Inspection Program (MS&I) is to maintain long term availability of the SSCs, such to ensure safe and reliable operation of the plant. This program is consistent with the IAEA Safety Guide, NS-G-2.6. The main document governing this program is also MER Process (MERP), according to Fig. 2.3.

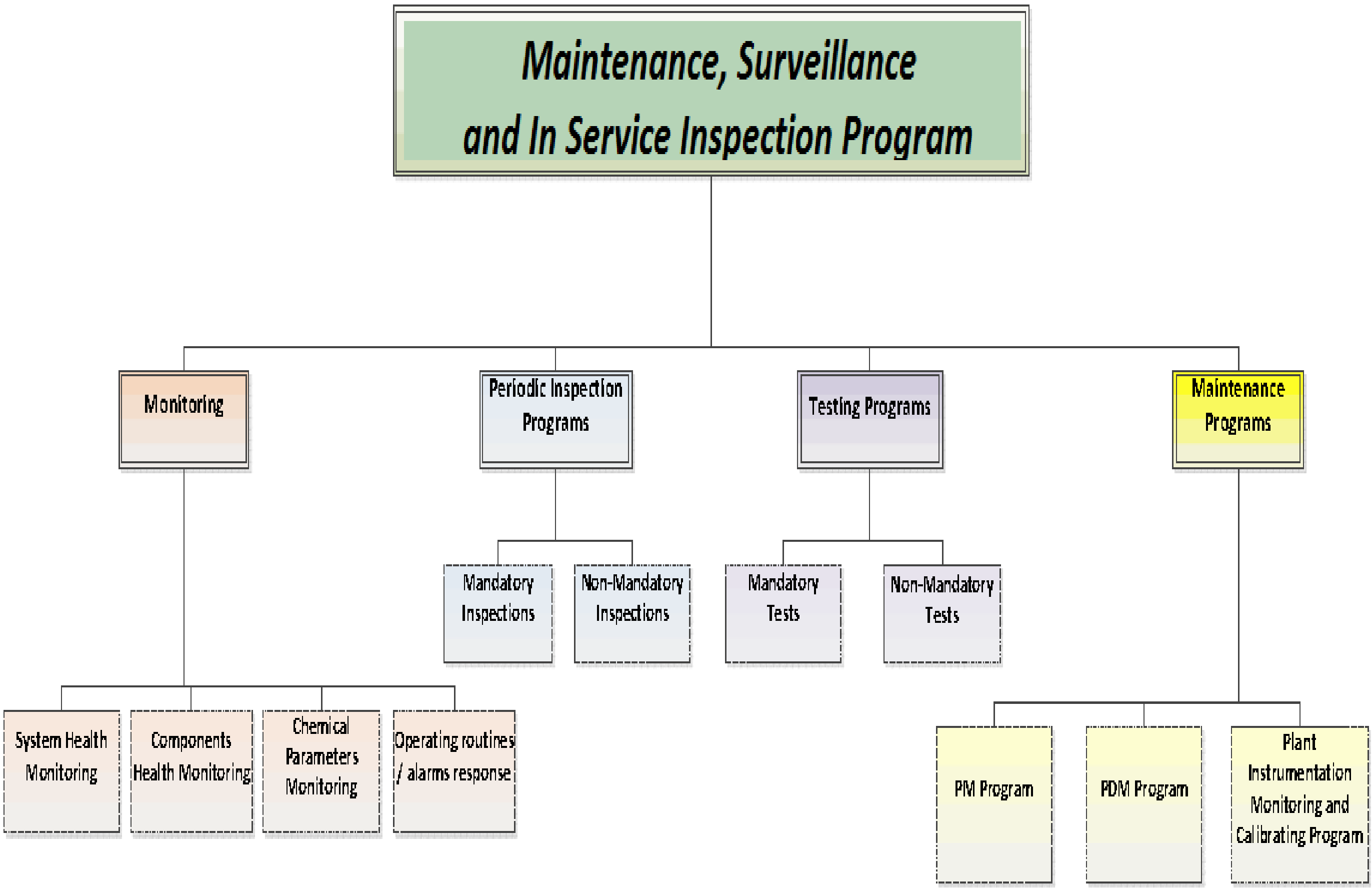


Fig. 2.3

The MS&I program was developed to cover four main directions:

A. Monitoring

A.1 System Surveillance

The central part of the system surveillance program is the procedure "Performing System Health Monitoring", which details surveillance program requirements and activities for critical systems, the objective aiming to continuously confirm and maintain the plant within the safe operating envelope. System performance is assessed by analyzing data collected from different plant sources (including field walkdowns) which is then trended, analyzed and reported as part of the System Health Reports. The system health reports are presented to the Station System Health Review Board, where they are analyzed from the point of view of health degradation causes, actions needed and effectiveness of implemented actions.

A.2 Component Surveillance

The Component Health Monitoring (CHM) Process is a structured method which provides performance monitoring of critical and essential components, in order to identify trend of degradation and to initiate the necessary corrective/ preventive actions. For each type of component, specific performance indicators are defined in accordance with design features, operating and maintenance conditions, possible stressors and are monitored according to the Component Health Monitoring procedures.

The Responsible Component Engineer (RCE) performs all the duties and responsibilities related to the assigned components, acting as the "Component Owner". The RCE represents the "driving force" for component problems, investigation and resolution.

The monitoring results are included in Component Health Monitoring reports and these are subject to the board evaluation similar to Systems Health Reports.

A.3 Chemical Surveillance

The group responsible with the monitoring of all chemical aspects within the plant is the Chemistry Group. The responsibilities, specific actions and monitoring means, are detailed in approved station instructions and operating manuals. Chemistry monitoring results are included in System/Component Health reports.

A.4 Operating routines and alarm response

Operator routines are complementary actions to System/Component Surveillance Programs. Specific documents detail the actions and responsibilities in preparation and conducting operating routines. Completed routines are sent to RSE / RCE for analysis and trending.

Abnormal conditions are prompted as alarms for operators and specific response alarms are developed enabling operator to the actions to restore the systems / components in a normal and / or safe condition.

B. Inspection Programs

The inspection program is detailed in the Periodic Inspection Program (PIP) procedure and consists of:

The Mandatory Inspection Program, required in the operating license issued by CNCAN, covering SSCs important to nuclear safety and performed according to Canadian Standards CAN CSA 285.4 and respectively CAN CSA 285.5, which are part of the licensing basis for the plant. According to these standards, periodic inspections are performed on fuel channels, feeders, steam generators, nuclear piping, pumps, vessels, valves, flanges, PHT & auxiliary systems supports and respectively on Containment structure. Inaugural inspections were performed on all required equipment to establish a "baseline" of initial conditions and measurements.

The Non-Mandatory inspection program is performed according to station internal procedures, prepared using the best industry practices. The following programs are developed and used as inputs for the SSC Surveillance:

- Heat Exchangers Inspection Program;
- Pressure Vessels Inspection Program;

- Piping Inspection Program;
- Main Steam Lines Inspection Program;
- Supports and Hangers Inspection Program;

Many of these inspections include activities performed by ‘third party certification organizations’, where Cernavoda NPP does not have the required skills, the expertise, the tools or the know-how to perform a credible Fitness-For-Service assessments or Life Assessments, based on inspection results.

C. Testing Program

The testing program is covered by Operating Manual Test procedures and consists of mandatory tests planned and executed in accordance with a station procedure approved by CNCAN and non-mandatory tests governed by specific procedures for tests with the plant at full power and for outage tests, respectively.

Mandatory tests are developed for safety-related SSCs to demonstrate to the nuclear regulatory authority that specific commitments made by Cernavoda NPP are being met and to support reliability claims made in the licensing submissions.

Non-Mandatory Tests are developed based on management requirements, technical and engineering rationale, to ensure a good reliability of the systems used for production.

The results from the Mandatory and Non-Mandatory Tests are used as inputs for the Systems and Components Health Monitoring Reports.

D. Maintenance Programs

As part of the Surveillance Program, several specific maintenance programs are developed, according to IAEA Safety Guide NS-G-2.6, to detect and mitigate degradation of a functioning SSC.

The maintenance Programs are:

- Preventive Maintenance Program;
- Predictive Maintenance Program;
- Plant Instrumentation and Calibration Program.

The objectives and responsibilities for these programs are detailed in the specific procedures. Inputs from these programs are used in different technical Programs, implemented to monitor the performance of the critical SSCs.

02.3.4A Preventive and remedial actions for Cernavoda NPP

The programmatic approach for preventive and remedial actions represents the main objective of mandatory inspections, as defined in the Periodic Inspection Program (PIP), i.e. to determine that new, unaccepted degradation of the inspection zones did not appear or**,** if identified, there is a high probability to keep it at an acceptable level for the entire period of planned operation of the plant.

PIP is a preventive type of Program, due to:

I. The methodology and criteria used for the selection of the inspection zones. A graded approach is used to determine PIP inspection zones, as follows:

1. First, systems selection:

   a) systems containing fluids used for nuclear fuel heat removal and transport;

   b) systems with an important role in safe shutdown of the reactor and/or ensuring safe fuel cooling in normal operation, in shutdown state, but also in case of a process system failure;

   c) systems which, if fail, have impact on the above safety functions.

2. Secondly, selection of the components to be inspected, based on the criteria:

   a) the degradation/failure could lead to:

      - reduced capability to perform in accordance with the design,
      - or has significant impact on a safety function;
      - major failures of a process system.

   b) systems/ components for which the operating experience indicated a high probability for major degradation;

   c) include a sufficient number of components to ensure that general degradation (such as erosion- corrosion) is timely detected;

3. Finally, the zones selected for PIP inspection, which should include, as a minimum:

   a) zones with significant indication accepted, after detection in previous inspections;

   b) zones susceptible to be exposed to a significant process of erosion-corrosion;

   c) zones with severe operating conditions which might determine high stress, in general cycling loads;

   d) components which should be replaced instead of being repaired.

II. The criteria used to establish inspection frequency. Inspection frequency is based on code CAN CSA N285.4 –Ed. '94, where the following rules apply:

a) Inaugural Inspection **-** to be performed after the pressure test, before placing in service the SSC.

b) Periodic Inspection

- first in the period of 5 years, starting one year after the first power generation.
- subsequent periodic inspections shall be performed in intervals, such as to not exceed 10 years.

In case of supplementary mandatory inspections for Fuel Channels, Steam Generators, Feeders, the frequency, selection of inspection zones, as well as remedial actions, or extension of initial scope, in case that indications are identified, the prescriptions for all these cases all are contained in the standard CAN/CSA N285.4.

Similar requirements apply to the Reactor Building for which periodic inspections for concrete containment are dictated by the standards CSA N285.5 "Periodic Inspection of CANDU Nuclear Power Plant Containment Components" focusing on metallic components, and CSA N287.7 “Periodic Inspection of Concrete Containment” for the concrete.

Codes and standards requirements are continuously revisited and updated based on the international experience, new R&D results, or the identification of new degradation mechanisms not previously taken into account. All these activities require state-of-the-art tools in terms of sophisticated technologies combined with highest professional skills and knowledge.

Cernavoda NPP understood the importance of new R&D results, therefore participated since 2006 to all 5 COG R&D sub-Programs: Fuel Channels, Safety & Licensing, Chemistry, Materials and Components, Health Physiscs and Environment and Industry Toolset, using COG R&D studies in elaboration of the PLiM Program Manuals and of the Life Cycle Plans for critical components.

02.3B Description of the overall ageing management Program for TRIGA RR

02.3.1B Scope of the overall AMP for TRIGA RR

The SSC Ageing Management Program (AMP) at the TRIGA-14MW reactor operated by the Institute for Nuclear Research Pitesti, Romania aims at identifying and implementing effective management actions and practices (techniques / methods / standards) to ensure early detection and mitigation, counterbalancing the effects of aging with consequences in nuclear safety.

The systematic aging management program at the TRIGA Research Reactor is based on the gradual application of:

- Provisions of the CNCAN regulations NMC 10, NSN-10 and NSN-17;
- Provisions of Safety Guide No. SSG-10 Specific Safety Guide for Ageing Management for Research Reactors as well as of the documents referenced in the bibliography of SSG-10.

The AMP is developed and interfaced with other programs of existing activities and processes that are part of the integrated management system developed by the Institute and presented in the Manual for Integrated Management System.

For example, existing programs as part of the integrated management system are as follows:

- the preventive maintenance program which is one of the main means of the AMP;
- the in-service inspection, the periodic testing and monitoring program provides data on SSC status and operating indicators;
- the data and reporting management program provides information on the performance of all systems covered by the aging management plan;
- the equipment qualification program that provides data and information on the maximum useful life of a piece of equipment or components that are part of an SSC;
- the Chemical Surveillance Program that monitors issues related to water chemistry and long-term corrosion of SSCs;
- the TRIGA 14MW reactor operating program follows and complies with the procedures for continuous operation;
- feedback of operational experience, significant event analysis and research programs;

- spare parts and spare parts insurance program to ensure when needed availability;

The Integrated Management System of ICN Pitesti incorporates all conventional systems and processes into a common framework in which it is possible to programmatically carry out all the activities related to the implementation of the activities, their control and their reporting.

The objectives of the AMP at TRIGA Reactor at ICN-Pitesti are:

- Ensuring nuclear safety in reactor operation;
- Ensuring the availability of the reactor for research and economic activities;
- Reduction of operating costs;
- Application of the management program and related procedures;
- Training and retraining the operating and maintenance staff.

The Integrated Management System of the Institute also includes provisions for the management of the operating organization of nuclear installations, including the functions of the system for controlling the planning and applying the activities at the TRIGA Research Reactor.

The Institute's integrated management system is periodically audited by CNCAN and includes management responsibilities, resource management, process implementation, measurement, evaluation and annual management performance analysis of performance indicators.

Applying the aging management requires that the following requirements are met:

a) Planning and prioritizing maintenance works;

b) Meeting the requirements of the regulatory authority, rules, codes, standards;

c) Overseeing the fulfillment of licensing requirements, operational limits and conditions and of the requirements derived from the safety analysis report;

d) Ensuring the availability of resources for spare parts, materials, tools and equipment;

e) Performing inspections, verifications, tests and current analysis of their results;

f) Ensuring the qualified personnel, knowledgeable of the installations and equipment with the necessary skills for maintenance, inspections, check-ups, reporting and analysis,

g) Developing, updating and applying procedures in a quality supervision system including procedures for finding and solving nonconformities;

h) Identifying and using good practice information from other reactor system projects, new products and the experience of other operating organizations;

i) Synthetizing the appropriate documentation and procedures for performing inspections, verifications, tests and measurements;

j) Analyzing the causes of failure, degradation, ageing of SSC and establishing corrective, preventive actions.

Method for Selecting Components and Equipment Structures Systems for their inclusion in the aging management program of TRIGA-ICN-Pitesti Research Reactor:

All reactor structures, systems, components and equipment are subject to the natural aging phenomenon in one form or another, but the susceptibility to degradation is different depending on the operation conditions, operation under stable regimes and under the conditions of postulated initiating events. A systematic approach to aging concerns the assessment of those structures, systems, components and equipment that have safety functions relevant to the safe operation of the reactor and which are clearly susceptible to aging degradation.

A systematic approach based on nuclear safety criteria has been applied to select the SSC (screening) that have been included in the aging management plan. According to the Specific Safety Guide No. 10, this approach involves three phases, three levels of selection.

The flowchart in Fig. 2.4 reflects the systematic approach by applying the three levels of selection.

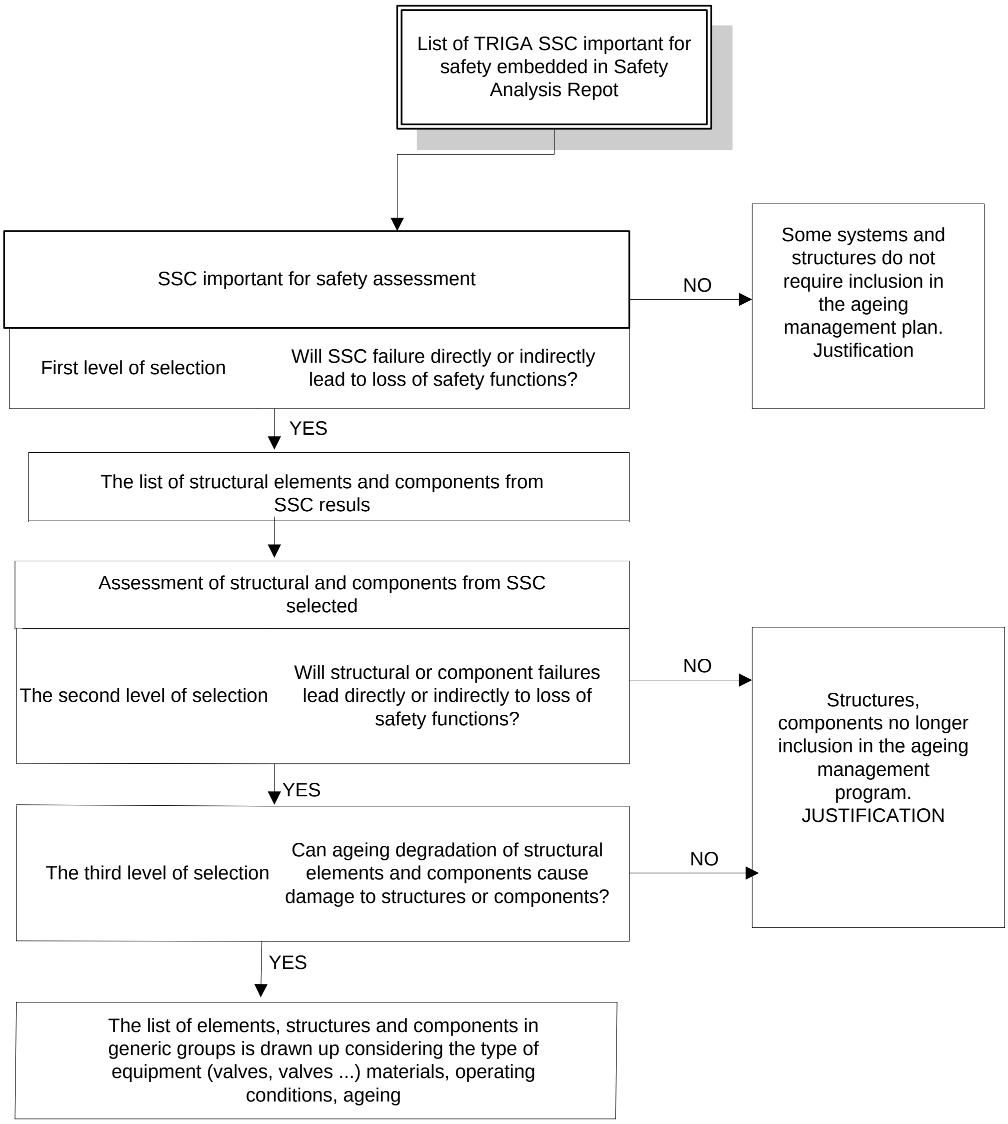


Fig. 2.4

02.3.2B Ageing assessment for TRIGA RR

Further, the SSC list prepared according to the method described above was analyzed from the point of view of aging and its consequences on normal operation and on accident situations in relation to design basis and postulated initiating events.

The AMP addresses the identification and application of effective measures and practices to ensure early detection and remediation of the effects of aging on nuclear safety related SSCs defined below.

SSCs that make up the "TRIGA Reactor" nuclear installation are defined according to IAEA nuclear safety standards and are classified according to their function and importance for nuclear safety on three levels 1, 2, 3 starting from the general category of all systems in the nuclear installation including the nuclear reactor (core) with all its systems.

Level 1. The first level of classification of systems in the general category concerns:

a) SSCs Important for safety;

b) SSCs that do not have safety functions or are not important for safety;

Level 2. The second level of classification refers to the content of Category 1 (a) "SSCs Important for safety " which contains:

c) "SSCs support for nuclear safety" means SSCs that are important for nuclear safety but which are not part of a safety system and

d) "Nuclear Safety SSCs" (Safety Systems).

Each of the selected systems, including their components susceptible to failure due to aging, has been associated with one or more of the safety functions they have to perform, for example:

F1 Reactivity control and criticality;

F2 Reactor safe shut down;

F3 Ensuring negative reactivity of the core, core geometry;

F4 Evacuation of residual heat;

F5 Prevention of release of radioactive products from the actireactor core and from the reactor hall;

F6 Ensuring the shielding function.

Safety functions are analyzed against the mechanisms of degradation / aging of the elements that support them under the conditions of the postulated external and internal initiating events.

The detection of the consequences of the failure mechanisms is determined through inspection procedures, periodic tests and check-ups. The monitoring is ensured by collecting inspection data, periodic checking and processing, based on the initial values established by the project and verified during the commissioning period, reference values.

The evaluation of trends in the aging processes is carried out by annual / multiannual analysis of operating parameters and the effects of degradation mechanisms.

02.3.3B Monitoring, testing, sampling and inspection activities for TRIGA RR

The limits and technical conditions in the license for the TRIGA reactor provide for requirements for the periodic verification of systems and components with safety functions. For this, a regular type of surveillance and periodic inspection program was developed and approved, as shown in Table 2.1 below.

The program refers to the following requirements:

- Identification of the scheduled test /verification, inspections;
- Specification of the applicable procedure;
- The person responsible according to the attributions and competencies for performing the tests / works;
- Annual testing / verification planning;
- The procedures include requirements on acceptability limits and comparison with test results.

Table 2.1. - The program of surveillance and periodic inspection tests for the TRIGA SSR-14 MW reactor

No.	Test / Inspection Name	Document applied	Testing / Inspection Officer	Execution period (month)											
				I	F	M	A	M	I	I	A	S	O	N	D
1.	Determination of the compensation capacity of the control bars	EO-TH-184T	Responsible for operation					x ✓						x	
2.	Determining the drop time of the control bars	EO-TH-185T	Responsible for operation			x ✓		✓	x			x			x
3.	Visual Inspection of Control Bars	EO-TH-165T	Responsible for exploitation		x ✓										
4.	Instrumentation check (before each start)	EO-TH-174T	Senior reactor operator												
5.	Fuel Bending Expansion and Folding Inspection (after 1500MWdays)	EO-TH-468T	Responsible for operation											x	
6.	Calibration of fuel temperature measurement channels	EO-TH-174T	AMC team responsible		x ✓			x		✓	x			x	
7.	Verification of threshold values of the instrumentation for radiation monitoring	EO-TH-383T	Responsible for personel dosimetry			x ✓			x ✓			x ✓			x
8.	Determination of closure time of pool isolation valves	EO-TH-214T	Responsible for operation		x ✓				x ✓				x		x

9.	Verification of safety, signaling and alarm functions of the reactor-related systems	EO-TH-625T	AMC team responsible											**x**	
10.	Determination of delay time for shutting down main pumps	EO-TH-589T	AMC team responsible				**x** ✓							**x**	
11.	Verification of reactor behavior when power supply failure	EO-TH-410T	Responsible for operation							✓					**x**

- LEGEND: **x** scheduled
- ✓ completed
- NOTE: The performance data is to be found in the „Registry for Surveillance and Periodic Inspection Surveys" according to surveillance requirements in a specific procedure

02.3.4B Preventive and remedial actions for TRIGA RR

The maintenance procedures specific to selected systems determine the measures to compensate for or limit degradation. Compensation measures generally concern the replacement of concerned / degraded components as a result of aging wherever possible in maintenance procedures. Degradation limitation measures for non-replaceable components may concern additional actions to analyze and prevent the consequences of the failure or to limit the availability of the reactor for extensive remediation work.

02.4A Review and update of the overall AMP for Cernavoda NPP

The first PSR for Cernavoda NPP started in 2005 for Unit 1, being a licensing condition. CNCAN required that the PSR Program to be developed using the IAEA NS-G-2.10, Safety Guide "Periodic Safety Review of Nuclear Power Plants". Later, in May 2006, based on the IAEA guide, Romania issued its own national regulation, NSN-10 "Norm for the Periodic Safety Review of Nuclear Power Plants". The PSR for Cernavoda Unit 1 covers 14 safety factors, recommended by IAEA in NS-G-2.10 and required by the CNCAN specific regulation NSN-10.

As recommended by IAEA, the Cernavoda PSR has been split into three phases:

- Phase 1 - Project Set-up and Planning – started in 2005 and was completed in 2007, after CNCAN approval of the PSR Policy, Scope and Program and Quality Assurance Plan.
- Phase 2 – Main PSR Activities –performed between 2009 and 2012.
- Phase 3 – Corrective Action Plan & Implementation. The Unit 1 Periodic Safety Review findings and observations were assessed by Cernavoda NPP, including the impact of these recommendations on Unit 2, and appropriate actions have been set and implemented.

Cernavoda Unit 1 PSR considered all safety related aspects in an integrated manner. To aid this process, the Cernavoda Unit 1 PSR Nuclear Safety Assessment Principles have been developed and used for plant evaluation. It was a review that provided an opportunity to give confidence that the plant is safe, as judged by modern standards and from operational experience.

The Safety Factor "Ageing" SF4, primarily concerned with the condition of the SSCs in the future, as managed by the overall Aging Management Program (AMP), was one major review topics of the Unit 1 PSR project. The review in this area indicated that:"Ageing management is captured within a reliability management system. This is based on international experience and a series of supporting documents that provide a management framework and the other major elements of a best practice program".

After undertaking a systematic review of the current plant design, ageing mechanisms and their management against internationally accepted safety standards, codes and practices, the PSR for Cernavoda Unit 1 has concluded in the discipline report for Ageing, Degradation and Obsolescence: "Overall, the Cernavoda Unit 1 ageing management program is considered compliant with industry best practice. In addition, there are no high or medium nuclear safety significant PSR findings resulted in this discipline report". There were only some clarifications on ageing items and recommmendations, considered opportunities for improvement, related mainly to personnel training in addressing ageing management issues, integrating ageing issues in developing operating procedures and defining and implementing a critical spare parts policy.

In 2013, CNCAN considered the PSR results during the Unit 1 and Unit 2 license renewal process, and granted licenses for 10 years to Unit 1 and 7 years to Unit 2 (so that Unit 2 completes its own full-scope PSR).

As required by NSN-10, a PSR is performed after each 10 years of unit commercial operation. Unit 2 is in operation since November 2007. Therefore, in the next couple of years Cernavoda NPP is preparing for conducting the second PSR evaluation of Unit 1 and the first PSR for Unit 2. Both PSR evaluations will be based on NSN-10 and IAEA Safety Guide SSG-25.

In 2016, the action plan for completing alignment to the NSN-17 regulation requirements on ageing management, Cernavoda NPP considered a good opportunity to perform an in-house evaluation / self-assessment of the current AM Programs against the latest IAEA documents issued on ageing management, such as revised / improved guidance and IGALL documents.
The first in-house evaluation of the AMP effectiveness was a major task at plant level, completed at the end of 2017, intended to assess and reinforce the current AMPs, based on the following new information which became available over the time:

- Life Assessment recommendations for LCM Plans after receiving the Life Assessment studies for the most important PLiM components (Fuel Channels, Feeders, Steam Generators, Heat Exchangers, Reactor Building);
- Component and System performance, as reported in the last couple of years, since SHMR/CHMR has become a mature process;
- The latest IAEA documents issued on ageing management topics, which were adopted also by CNCAN.

The Final Safety Analysis Report (FSAR), Chapter 15 - Deterministic Safety Analyses includes a list of nuclear safety analyses that use assumptions about the lifetime of the SSC and their ageing.

Among these are: Primary Heat Transport (PHT) ageing analysis and determination of Unit 1 End of Life using NUCIRC computer code, Accident Analysis (Loss of Coolant Accidents, Main Steam Line Break) that determine the environmental qualification requirements of SSC with safety functions, PHT thermal-hydraulic analysis in various configurations and according to updated models, to reproduce as close as possible the actual installation parameters:

The ageing phenomena that affect the components of the PHT circuit are taken into consideration by adjusting the ageing parameters in the NUCIRC model. The main objective is to reproduce the plant parameters in aged conditions. The adjusted model is then transferred to the CATHENA code. In this way, the aged model is used to evaluate the plant response for design basis accident scenarios. The effect of pressure tube creep is also included in the models used for physics simulations.

In order to ensure the capacity of the nuclear installation to perform its nuclear safety functions, compliance with the design bases and safety margins, Cernavoda NPP ensures the periodical update of the nuclear safety analyses according to the strategic deterministic nuclear safety analysis program, which is also imposed through license conditions and regulations and approved by CNCAN. The program is in line with the requirements of the IAEA SSG-2 Safety Guide and CNCAN regulations.

These analyses use codes and techniques specific to deterministic analyses and up-to-date models of the installation (the reactor core / the PHT circuit / the secondary circuit), starting from the as-built configuration and taking into account the effects of aging observed up to date.

Code requirements are continuously revisited and updated based on the international experience, new R&D results, or the identification of new degradation mechanisms not previously taken into account.

As previously mentioned, Cernavoda NPP participates in and makes use of various R&D activities, including 5 COG R&D sub-Programs: Fuel Channels, Safety & Licensing, Chemistry, Materials and Components, Health Physiscs and Environment and Industry Toolset.

The modifications in the current licensing or regulatory framework (including those related to ageing management program) are controlled through the plant process that documents the management of the current licensing bases (CLB). This process ensures that all license conditions and other regulatory specific requirements are identified, analyzed from safety and licensing point of view, and actions are taken as per plant procedure to implement those requirements. All the actions are tracked for their status until implementation.

02.4B Review and update of the overall AMP for TRIGA RR

The AMP of the TRIGA Mark II Piteşti Research Reactor was first issued in 2013, taking into account the state of the installation and the technology existing at that time. The program took into consideration: the mechanisms of degradation, the characteristics of the site, susceptibility of the SSCs to ageing-related degradation, the program of operation of the reactor and its operating experience.

The overall AMP for TRIGA RR is reviewed and updated in accordance with the provisions of the IAEA Safety Guide SSG-10, taking into account the latest CNCAN requirements (NSN-17), IAEA standards and guidance and the operating experience. Periodic self-assessments and audits are performed, in accordance with the integrated management system requirements and the resulting findings are used to improve the AMP relevant activities, as applicable. The results of the monitoring, testing, sampling and inspection activities are used to confirm the effectiveness of the AMP and to identify and implement corrective actions where necessary.

When the operating license for the reactor was renewed in 2017, a reassessment of the ageing management program was performed by the licensee and by CNCAN. The reassessment and the revision of the program took into consideration the history of operation, the evolution of the physical state of the SSCs, any new mechanisms of degradation and works of repairs and refurbishment performed by the licensee in the period 2013-2017, and also the modification of the procedures and safety documentation (FSAR). The review also took into account the implementation of the new regulatory requirements in the regulation NSN – 17.

02.5 Licensees' experience of application of the overall AMP

A. Cernavoda NPP

The international approach has changed in the last years from the north-american pragmatic approach of Maintaining Equipment Reliability process (MERP) based on INPO AP-913 philosophy to a more comprehensive concept: "Ageing Management", based on IAEA documents. IAEA has enhanced in the last years the specific AM Standards, Guides, Safety Reports , TECDOCs developing reference documents in IGALL program to support MS to validate and enhance their AM programs. The new AMP philosophy requiring a systematic and integrated control of ageing was embraced by Romanian and Canadian Regulatory bodies, as it takes into account the cumulative ageing effects on overall systems and the risk for safety performance.

Cernavoda NPP assessed the conformity with the requirements of the CNCAN regulation NSN-17 on ageing management, identifying the following strengths:

a) The processes and the organizational arrangements set in place since 2008 support all the activities required for the overall AMP. Where there are no internal resources available, arrangements were made with external organizations to support with the required tools, manpower or specific expertise for technical Programs implementation;

b) Long term agreements are in place with subject matter experts to enable the implementation of specialized PLiM services (Life Assessments or Condition Assessments included), technical support services with OEM (Original Equipment Manufacturer). This will ensure continuity in the evaluation of inspection results and an increased credibility with CNCAN;

c) Periodic evaluation of ageing effects in safety analysis and reassessment of safety margins is performed for each unit according to a specific elaboration timetable strategy, correlated with the latest licensing requirements following the lessons learned from the Fukushima accident;

d) The AMP is continuously improved, based on information exchange, joining external organizations (EPRI, COG, INPO, WANO, IAEA), participation in training and benchmarking missions to keep alignment to the latest industry initiatives.

The self-assessment of the current Cernavoda NPP overall AMP concluded that MERP philosophy is simple, but comprehensive, so that no significant changes have been needed to the organization, content and/or structure of the overall AMP based on the new NSN-17 regulation issued by CNCAN in 2016.

Cernavoda NPP identified also 3 opportunities for improvement:

1. Revision of Equipment Reliability Process and PLiM procedures to integrate all the activities at interface (such as: Chemical control, Margin management, proactive Obsolescence management) and better sustain the international approach for an integrated AM Program;

2. Documenting the strategies to keep alignment of Periodic Inspection Program with recent Code standard editions;

3. Analysis of the actual PM/PLiM programs conformity with IAEA best practice model, verifying the 9 generic attributes of an effective AMP against IGALL documents. It is considered to be a good exercise to identify eventual gaps and to revise the current CNPP Programs, prior to the next PSR evaluation, scheduled to start in 2019.

B. TRIGA RR

Based on its experience with the implementation of the AMP for the TRIGA research reactor, the licensee considers it has a good practice in the utilization of the AMP interfaced with other programs of the Integrated Management System and identified the following areas for improvement:

- Increased utilization of the internal operational experience and the experience of other comparable research reactors;
- Procurement of specific equipment, tools and instruments dedicated to periodic testing and inspection for identification of effects of ageing;
- Training of maintenance staff in the area of ageing identification, data collection and processing and designing of solution for reduction of cumulative effects of ageing;
- Since the ageing of staff is faster than the ageing of the installation, a continuous effort should be developed in the area of the human factors to ensure training and succession of generation for the management the installation lifecycle;
- The development of the acceptance criteria (quantitative) and the values of measured accepted parameters to prevent the qualitative evaluation and improper diagnostic for each category of SSC. The evaluation of degradation and consequences are specific to each SSC selected in the AMP.

The licensee implemented the AMP and the necessary compensatory measures concerning reparation, refurbishment, modernization of equipment and systems preventing the ageing effects from impacting on nuclear safety of the research reactor.

02.6 Regulatory oversight process

CNCAN implements a regulatory oversight process that consists of review, assessment and inspections.

The review and assessment performed by CNCAN as part of the licensing process and as part of the continuous regulatory oversight focuses primarily on:

- Operating licence renewal documents, including updates to the FSARs;
- New or updated safety analyses performed by the licensee;

- Resulting of periodic safety reviews (PSR or other more frequent routine reviews);
- Station safety performance;
- Significant events reported by the licensee;
- Temporary configuration changes;
- Plant modifications;
- Results of the periodic inspection program and of the mandatory tests for safety-related SSCs;
- Implementation of the preventive and corrective maintenance activities for safey-related SSCs.

The review and assessment activities aim at verifying compliance with the following:

- Regulatory requirements, safety principles and design criteria;
- Defence in depth concept achievement;
- Systems separation philosophy;
- Special safety systems design requirements;
- Design codes, standards and safety guides that are part of the current licensing basis.

The review and assessment activities are performed with the objectives of determining whether the applicable safety objectives and requirements for each aspect or topic have been met, whether the safety analyses cover both normal and fault conditions and whether the safety submissions provided are sufficiently complete, detailed and accurate.

The objective of the inspections performed by CNCAN is to monitor compliance with the legal, regulatory and licensing requirements, and to take enforcement action in the event of non-compliance. The inspections for Cernavoda NPP and TRIGA RR are planned in a systematic manner by the staff from CNCAN headquarters (and by the resident inspectors for Cernavoda NPP), with the aim of ensuring a proactive identification of the deficiencies and deviations from good practices that could result in non-compliances.

The inspection planning is periodically reviewed and modified as new information on the facility or organisation is obtained. The inspections are normally focused on those areas that would pose a significant risk, or for which a poor performance has been recorded. However, if an assessment finds good performance in an area, the results may be used to reduce the frequency and depth of the future inspections.

The inspections performed by CNCAN include:

- scheduled inspections;
- unscheduled and/or unannounced inspections, some of these being reactive inspections, in response to incidents;
- routines and daily observation activities performed by the resident inspectors for Cernavoda NPP.

Examples of inspection activities and tasks performed by CNCAN inspectors are given below:

- review of nuclear installations’ operation reports;

- review of progress on outstanding safety issues;
- review of the past safety performance of the nuclear installations;
- review of the status of committed safety improvements;
- management system audits;
- review of the implementation of temporary and permanent modifications to ensure they are consistent with the licensing basis for the installations;
- system inspections;
- observation of operating and maintenance practices and work.

Resident inspectors in the Cernavoda NPP Surveillance Unit have a very important role in the daily observation and assessment of the activities on site. Examples of activities performed by the resident inspectors are given below:

- verification of the implementation of the dispositions and recommendations resulted from previous inspections;
- independent preliminary investigation of events significant for safety;
- inspections in the field for observing and gathering information on the general progress of plant activities;
- detailed system inspections, for observing the performance of maintenance activities and the status of related documentation;
- daily verification of the various records and reports related to the operation of the plant;
- evaluation of the practices in different areas of activity to observe adherence to procedures, with focus on preventive maintenance activities, testing of the special safety systems, personnel training, quality assurance, radiation protection aspects;
- surveillance of the performance of activities during the planned outages with regard to configuration of the safety related systems, radiation protection of the personnel, work involving contractors, elaboration and review of the safety documentation (procedures, work plans, modification proposals, etc.);
- witnessing the performance of tests or other activities performed on safety related systems, usually according to an inspection plan that includes Witness Points (WP) and Hold Points (HP) (this approach is used mainly for monitoring the activities during planned outages).

A series of routine inspections is used by the Cernavoda NPP Surveillance Unit to monitor the physical state of the systems and the operating parameters, that cover all safety relevant areas of the plant. The areas covered by the routine inspections are:

- Reactor Building;
- Service Building;
- Turbine Building;
- High Pressure Emergency Core Cooling Building;
- Emergency Water System Building;
- Secondary Control Area;
- Standby Diesel Generators Building;
- Spent Fuel Bay;
- Pump House;

- Chillers Building;
- Fire Response Command Area.

During planned outages, inspections are performed also in the areas not accessible during operation at power.

Besides the routines, the resident inspectors perform daily visits to the control room, for verifying the main operating parameters and the different aspects related to work planning and control of temporary modifications. The resident inspectors participate also as observers in the daily planning meetings of the plant management. Daily reports are elaborated by the Cernavoda NPP Surveillance Unit and forwarded to the CNCAN headquarters for information on the plant status and for ensuring awareness of any inspection findings.

Following the issuance of the specific regulation on ageing management (NSN-17) in 2016, CNCAN performed reviews of the documentation submitted by the licensees to demonstrate that their overall AMPs meet the latest regulatory requirements and that all the necessary elements are in place. CNCAN also performed specific inspections to verify the implementation of the overall AMP, as well as specific ageing management activities for selected safety-related SSCs, at Cernavoda NPP and TRIGA RR.

Previously, ageing management related regulatory reviews and inspections had been performed in the context of the regular review and inspection of the implementation of preventive maintenance and periodic in-service inspection programs for the nuclear installations, both for Cernavoda NPP and the TRIGA RR, as well as in the context of the first PSR for Unit 1 of Cernavoda NPP.

With the issuance of the new NSN-17 regulation, the regulatory review and inspection of licensees’ AMPs has become more systematic.

02.7 Regulator’s assessment of the overall ageing management program and conclusions

In addition to its regular oversight activities aimed at verifying compliance with the new regulation NSN-17 on ageing management (based on the relevant WENRA reference levels), CNCAN performed reviews and inspections to verify the self-assessment reports prepared by the licensees in accordance with the WENRA / ENSREG specifications for the TPR.

Based on the regulatory reviews and inspections performed so far, CNCAN is satisfied with the adequacy of the licensees’AMPs and with their overall implementation. No major issues have been identified. The main findings of the licensees’ self-assessments have been presented in section 02.5 Licensees’ experience of application of the overall AMP.

Starting with 2017, ageing management related topics are included more systematically in the regulatory inspection plans and the regulatory oversight activities for monitoring implementation of the licensees’ AMPs and their overall effectiveness will continue.

03A ELECTRICAL CABLES - Cernavoda NPP

03.1A Description of ageing management programs for electrical cables

There is an ever-increasing growing interest in the world in terms of aging-related cable status, especially in nuclear power plants. The unanticipated or premature aging of the wiring system may result in the unavailability of essential or safety equipment, may affect the safety and health of staff and the population, or may lead to significant production losses.

From the analysis of the international experience in the field, it resulted that the electric cable faults can have an impact on the nuclear safety or impose large losses of production.

Thus, it was concluded that the performance of the electrical cables directly affects the performance of the systems supplied, with an impact on the economic efficiency of the nuclear power plant and / or safety. For this reason, the need to implement a Cables Program as part of the PLiM programs at Cernavoda NPP was imposed in order to monitor the behavior of the cable system.

The program tracks:

- identification of aging mechanisms;
- tracking the ageing process of the wiring system in the plant and its evolution, to prevent the total failure of the constituent components and implicitly the consequences of their degradation;
- reducing the degradation of the cable system through planned repairs or replacement work.

The program aims to estimate the ageing, the evolution of this phenomenon, where it occurs, and also to identify the appropriate actions to avoid unavailability of the serviced equipment. Evaluating the components of the cable system that are exposed to premature ageing and identifying the timing of the planned replacement or repair of damaged ones will minimize production losses due to unplanned stops or damage to the safety features of the plant.

To achieve the above mentioned objectives, it was necessary to develop technical programs for the electrical cable system, whose complexity is proportional to the impact of their failure and which will include:

1. Selection and classification of cables according to the impact of their failure on the reliable and safe operation of the plant;

2. Collecting and organizing design, manufacturing, assembly, commissioning, operating parameters, and test results, tests, inspections, and maintenance and repair history until the current date;

3. Identification and analysis of degradation mechanisms, failure modes and identification of performance status parameters / indicators that can effectively track their evolution;

4. Development and implementation of inspection, testing and preventive maintenance activities;

5. Periodic assessment and reporting of health.

The "Cables" program addresses specific activities to track the behavior of the power and command cables system in critical equipment depending on the importance and impact on the power plant and the cost associated with the tracking activities in operation (inspections tests / verifications, repairs, replacements).

Considering the large number of cables for critical equipment and SPV and the complexity of the NPP wiring system (redundancy, sewerage, construction types and different features, different functions, operating parameters), the cable program is developed as a mixed program PM and PLiM).

The selection of programs applicable to the cable system families was made according to the following criteria (listed in order of importance):

- Possibility of replacement (redundancy, reactor building penetrations)
- Impact of failure (circuit breaks with safety or major impact in power generation, long-term outages, high cost of repair / replacement)
- Working environment (characterized by temperature, radiation, humidity, hot spots)
- Operating experience (number of events due to failure of cable system components)
- Other considerations.

03.1.1 Scope of ageing management for electrical cables

Following the evaluation of critical components lists at Unit 1 and Unit 2, approximately 4900 cables and related components (connecting elements, support and protection systems) were identified as important to ensure the operation of these equipment. Out of these, 3600 belong to critical nuclear safety equipment and the rest belong to equipment critical from the point of view of production.

<u>Environmental qualification</u>

At Unit 1, the cables with environmental qualification requirements have the following qualification level:

- The Phillips power and instrumentation cables, according to the test report, the lifetime of the cables is 30 years at a temperature of 90°C. Also, according to the test report, the respective cables were tested at a total integrated dose (normal dose + accident dose) of 100 Mrads;
- The Boston Ins. (used for Ion Chambers), according to the test report, the lifetime is 40 years at a temperature of 74°C. According to the same test report, the respective cables were tested at a total integrated dose (dose in normal operation + accident dose) of 200 Mrads;
- The BICC fabrication cables, used for SDS1 (Shutdown System #1) and VFD (Vertical Flux Detectors), according to the test report, were tested at a total integrated dose (normal dose + accident dose) of 200 Mrads. The operating temperature for these cables is 150°C;
- The Canada Wire fabrication cables, used for SDS 2 (Shutdown System #2) and HFD (Horizontal Flux Detectors), according to test report, were tested at a total integrated dose (normal dose + accident dose) of 300 Mrads. The operating temperature for these cables is 150°C.

At Unit 2, the cables with environmental qualification requirements are Nexans power and instrumentation cables, which, according to the IEEE 383 Type Test Report, have a 40-year life span at a temperature of 90°C and were tested at a total integrated dose (normal dose + accident dose) of 99.8 Mrads.

Unit 1 has been in operation for 21 years and Unit 2 has been in operation for 10 years and the temperatures and the level of radiation mentioned above have not been achieved in the normal functioning of the nuclear power plant.

The cable system was grouped by families according to the following criteria:

- the cables have the same voltage level (LV, MV),
- the cables have the same type of insulating material,
- the cables serve equipment installed in the nuclear side (also affected by radiation) or in the classical part of the plant.

According to the technical literature (EPRI, COG, IAEA) criteria have been defined to differentiate the cables for which a PM program is sufficient, from the cables for which a PLiM program is required. Some of these are set out below:

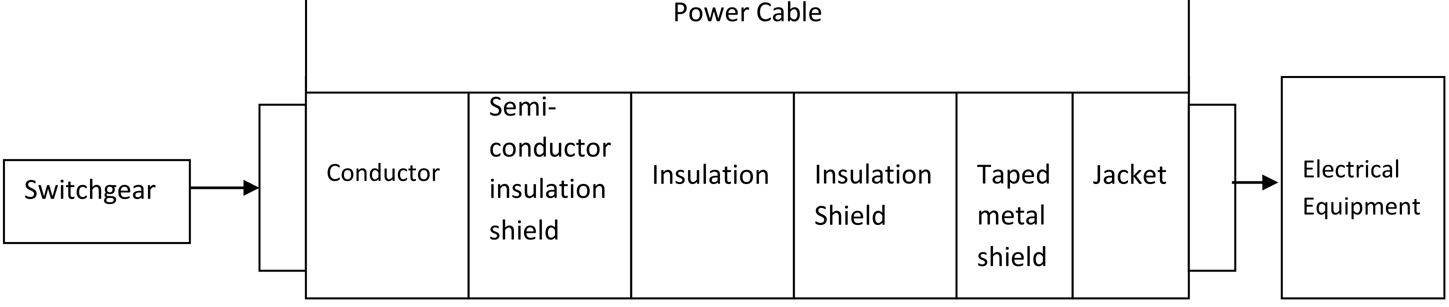
- "a cable program should begin with identifying harsh environments, assessing the performances of installed cable types (operating experience), and conducting initial inspections to determine future actions" (EPRI 1003317);
- "the program will include cables with nuclear safety function and those with significant impact on production; initial inspections may be limited to cables in hard areas and not necessarily to follow a complete list of cables; the purpose of the program is to determine if cables are significantly damaged in areas with the toughest environments" (EPRI 1003317);
- "the process of drawing up a list of cables will be limited to identifying the types of insulation for the cables in the plant and the areas with harsh environments (temperature, radiation, humidity, chemistry)" (EPRI 1013475).

<u>Constructive wiring systems and aging considerations</u>

A cable system in a nuclear power plant is not limited to the cables in the field and includes the following components:

- the cable itself,
- connectors (connectors, terminal heads, slippers, joints),
- protective systems and cable supports (cable beds, protective pipes).

With regard to cables, the following are specified: A simple cable contains two parts - the metal conductor and non-metallic insulation covering the conductor. A more complex cable may contain several wires with sections depending on the voltage level, the insulation filler, the sheath, a screen, individual wires and a common sheath of the cable.



The metal wire, reinforcement and screen are not normally rated for ageing. Insulation and sheath are the benchmarks in aging management, and insulation degradation is the most important factor under consideration. Insulation and sheath materials used in electrical cables are a combination of base polymeric structures and filler additives, resulting in a material that has good mechanical properties, electrical and flame propagation resistance.

With regard to the connection elements and the protective system and cable support, the following are specified: According to the technical literature, it results that the connection elements and the protective system / cable support are not evaluated from the point of view of ageing, most defects occurring due to the incorrect design, manufacturing and installation of the components / equipment.

Types of insulating materials for electrical cables at Cernavoda NPP

The types of insulating materials most used at Cernavoda NPP are:
At Cernavoda Unit 1:

- PVC (polyvinyl chloride),
- XLPE (Reticulated Polyethylene)
- EPR (ethylene-propylene rubber)
- CSPE (Hypalon)
- TFE (teflon insulation - TEFZEL)
- PE (polyethylene).

At Cernavoda Unit 2, according to the technical specifications and procurement documents, most of the cable insulation are made of new generation materials with properties superior to the cable insulation purchased for U1:

- XLPE (Reticulated Polyethylene),
- EVA (ethylene-vinyl acetate),
- EPR (ethylene-propylene rubber),
- HI-TEMP PE (High Temperature Polyethylene)
- EFTE (TEFZEL200 + TEFZEL280).

a) Methods and criteria used for selecting electrical cables within the scope of ageing management

Knowing and analyzing degradation mechanisms and defective contributing factors are the basic information for determining behavior in operation, assessing the condition of cables, and identifying actions to reduce degradation processes and their associated effects. The most significant degradation mechanisms affecting the cable system are listed in "Aging Management Guideline for Commercial Nuclear Power Plants - Electrical Cable Terminations, Table 4-18", SAND 96-0344, "Cable System Aging Management", EPRI 2002, Report 1003317 and "Medium Voltage Underground (MVU) Cable White Paper" NEI 06-05 (April 2006).

Generic degradation mechanisms

The list of the most relevant degradation mechanisms, breakdown modes, probable causes and the main contributors to the degradation rate applicable to cables are shown in Table 3.1. The table highlights the methods and activities needed to identify and prevent degradation.

Table 3.1

Failure Location	Degradation Mechanism	Degradation Influence	Discovery Prevention Opportunity
Cable conductor	Corrosion	Water ingress, chemistry	None
	Mechanical stress (bending)	Too small a bending radius	
Cable insulation	Electrical trees	Manufacturing defect (voids and protrusions or contaminants in insulation)	Partial discharge
	Material degradation	Normal life (that is, remaining dry)	Power factor or dissipation factor (tan delta)
	Mechanical damage leading to tracking	Improper transportation, handling or installation	None
	Mechanical stress	Improper pulling technique	None
	Thermal damage	High temperature operation (high load factor, bad heat transfer, high ambient temperature)	Power factor or dissipation factor (tan delta)
	Water trees caused by water ingress through damaged jacket	Damaged by foreign material or protrusions in the duct during pulling. Embrittlement of jacket due to high temperature operation	Power factor or dissipation factor (tan delta)
	Water trees (that is, those that lead to electrical trees, mainly water trees that exceed 30% of the insulation thickness)	Water ingress in the presence of voltage, especially in the presence of manufacturing defects (voids, contaminants, ionic impurities)	Power factor or dissipation factor (tan delta)
Cable insulation shield	Electrical discharge	Gaps between insulation and insulation shield- possibly installation damage	Partial discharge
Cable jacket	Damaged	Foreign material or protrusions in the duct	Inspection
	Embrittlement, leads to loss of jacket integrity	Thermal aging due to high temperature operation. Circulating currents in neutral.	Inspection
Cable taped metal shield	Corrosion	Water ingress chemistry	Loss of continuity or tracking
Terminations	Degraded insulation	Installation error allowing ingress of moisture	Partial discharge
	Overheating from high resistance connections	Improperly installed, corrosion	Partial discharge or thermography
	Partial discharge and tracking	Installation error	Partial discharge
	Surface tracking	Moisture, especially at higher voltages	Partial discharge or inspection
	Tracking	Manufacturing defect	Partial discharge

Mechanisms of degradation specific to Cernavoda NPP

Degradation mechanisms, their effects on the parts of the cable system at Cernavoda NPP, have been determined by the main causes that can lead to the degradation of the cable system (heat, radiation, handling, humidity). These are presented in the Table 3.2.

Table 3.2

Area / Subassembly	**Mechanism of failure**	**Causes and Symptoms**	**Impact**		**Possibility of replacement**		**Tracking mode**
			At the equipment level	**At system level**	**Yes**	**No**	
Cable / Insulation and jacket	**Thermal and thermal degradation** of organic parts caused by ambient heat and ohmic heat (Joule - Hertz effect)	Heat, Oxygen	Brittleness, cracking, melting, discoloration	Low insulation resistance, electrical failure, increased vulnerability of defect in hard working environments	Yes	No (for cables passing through reactor building penetrations)	Visual Tactile indenter E-AT-B Electrical tests
	Chemical decomposition of organic parts, radiation-induced oxidation	Radiation, Oxygen	Brittleness, cracks, discoloration, dilation	Low insulation resistance, electrical failure	Yes		Visual indenter E-AT-B Electrical tests Chemical tests (OIT, Spectroscopy)
	Wear due to work in the area, human trafficking, or faulty cable system support practices	External mechanical effort	Cutting, cracking, chopping	Interrupt loops / circuits Low insulation resistance, electrical failure	Yes		Visual Electrical tests
	Arc-over	Fault by manufacturer (holes) Electric arc	Brittleness, cracks Electric trees	Short circuits	Yes		Visual Electrical tests
	Structural degradation due to water / moisture in	Moisture	Water trees	Short circuits	Yes		Chemical tests (OIT, Spectroscopy)

Area / Subassembly	Mechanism of failure	Causes and Symptoms	Impact		Possibility of replacement		Tracking mode
			At the equipment level	At system level	Yes	No	
	protective channels and pipes						
Connection element **(Connector, slipper, contact surface)**	**Electrochemical stress** (moisture, oxygen etc)	Corrosion and oxidation of metals	Corrosion and oxidation of external contact surfaces	Increasing contact resistance and local warming, loss of circuit continuity	Yes		Visual Electrical tests
	Fatigue and deformation of metals	Vibrations, Stretching Efforts	Weakness of shoe and conductor contact, shoe rupture	Interrupt loops / circuits.	Yes		Visual Electrical tests
Racks system and cable protection pipes	**Vibrations** Construction activities Inadequate maintenance practices	Missing: supports, anchorages, sealing caps (fireproof)	Mechanical degradation of cable insulation leading to short circuits and overload	Interrupt loops / circuits.	Yes		Visual
	Vibrations Weak parts of the cable rack system Improper reinstallation after maintenance or project modifications	The systems of vertical supports damaged or broken parts	Mechanical damage to cable insulation due to the concentration of vertical loads, leading to cable break at the top of the cable rack Degrading the seismicity characteristics.	Interrupt loops / circuits.	Yes		Visual Tactile

b) Processes/procedures for the identification of ageing mechanisms related to cables

The inspection / testing activities performed on the cable system are aimed at maintaining its integrity in order to be able to fulfill its function according to the design requirements.

The following describes all the tests currently used in the industry to determine defects, ageing and duration of remaining life of the cables. Over the last 20 years, specialized external organizations have developed numerous reports on cable aging and cable status monitoring practices in nuclear power plants. The national standard uses the PE 116 (and related technical datasheets), which presents the methods and tests used to determine the condition of the cables after certain operating periods. Below are the most effective test methods for determining the condition of the cables, depending on their voltage level.

These methods can be globally divided as follows:

- Destructive methods / Non-destructive methods;
- Methods for determination of cable status (aging, remanence) / Methods for determining cable defects (local, global).

c) Grouping criteria for ageing management purposes

A) Control and signal cables (I&C)

Control cables (including thermocouple extension conductors) are low voltage cables (U <1kV, most of which have a rated voltage of 300V). They are used for analog and digital signals from different types of transducers. Thermal resistance detector (RTD), pressure transducers (PT) and thermocouples contain shielded twisted pair cables. Radiation detectors and neutronic flux monitoring circuits often use shielded coaxial or triaxial cables.

Control cables are low-voltage cables with low amperage used in instrumentation and control circuits for auxiliary components (switches, valves, relays and contactors). There are multi-conductor cables that are screened only when installed near high voltage systems.

To determine the state / degree of degradation of I&C cables, the most effective methods are:

- Polymer Aging Monitor (INDENTER) - a tool that measures the compression coefficient for Neoprene, Hypalon, and PVC sheathing cables to identify the strength;
- Different chemical tests, including swelling and gel factor (XLPE and PVC);
- Induction time of oxidation and oxidation induction temperature (XLPE);
- Density (XLPE and EPR);
- Elongation at break;
- Measurement of insulation resistance and ohmic resistance;
- Spectroscopy & Reflectrometry;
- Models of predictive methods of cable ageing (accelerated ageing tests, irradiation, Arrhenius method, EQ qualifications, etc.).

B) Low voltage cables (LV)

Low voltage cables (U <1kV) are used to power auxiliary components, such as motors, MCCs, heaters, and small transformers. These cables may be monofilament or multifilament and are generally not screened.

To determine the state / degree of degradation of LV cables, the most efficient methods are:

- Polymer Aging Monitor (INDENTER) - a tool that measures the compression coefficient for Neoprene, Hypalon, and PVC sheathing cables to identify the strength;
- Different chemical tests, including swelling and gel factor (XLPE and PVC);
- Induction time of oxidation and oxidation induction temperature (XLPE);
- Density (XLPE and EPR);
- Elongation at break;
- LIRA (Line Impedance Resonance Analysis);
- Measurement of insulation resistance, ohmic resistance and polarization index;
- Tanning measurement;
- Spectroscopy & Reflectrometry;
- Models of predictive methods of cable ageing (accelerated ageing tests, irradiation, Arrhenius method, EQ qualifications, etc.)

C) Medium voltage cables (MV)

Medium voltage cables (1 kV <U <60 kV) are used to power high power electrical equipment in the plant.

To determine the state / degree of degradation of MT cables, the most effective methods are:

- An alternative voltage test;
- Measurement of insulation resistance, ohmic resistance and polarization index (relevant only for insulated paper cables);
- Increased voltage tests (no longer practiced - is destructive);
- Measurement of partial discharge at industrial frequency
- Low Frequency Alternative Test (f <1Hz);
- Tanning measurement;
- Measuring partial discharges;
- Elongation at break;
- LIRA (Line Impedance Resonance Analysis);
- Different chemical tests;
- Models of predictive methods of cable ageing (accelerated ageing tests, irradiation, Arrhenius method, EQ qualifications, etc.)

It should be noted that at this time there is no unique test / monitoring technique to be used to determine the state of all types of cables and insulation. Techniques used in industry to determine the lifetime of cables are chosen depending on: the voltage level, the type of insulation, the color of the insulating material, the length of the cable, the types of strain interacting with the cable, the temperature and the humidity of the environment.

Given the above elements, one can conclude that:

Connection elements are tracked by:

- Electrical tests, visual and tactical inspection (PM activities)
- Repairs / Replacements (PM activities)

The protective system and cable support is tracked by:

- Visual and tactical inspection (PM activities)
- Repairs / Replacements (PM activities)

Cable with its components:

(i) The conductor is tracked by:
  - Electrical tests (PM activity)

(ii) The insulation is tracked by:
  - Electrical tests, visual and tactical inspection (PM activities)
  - indenter
  - Elongation at break (E-at-B)
  - Chemical tests
  - Spectroscopy

(iii) The screen is tracked by:
  - Electrical tests (PM activity)

(iv) The shield is tracked by:
  - No inspection activities required

(v) The jacket is followed by:
  - Electrical tests, visual and tactical inspection (PM activities)
  - indenter
  - Elongation at break (E-at-B)
  - Chemical tests
  - Spectroscopy.

03.1.2A Ageing assessment of electrical cables

The collection of data on cables and their organization in an easily readable electronic format is necessary to facilitate their analysis for the purpose of producing periodic health reports, lifelong learning and monitoring, inspection, testing and repair plans.

The primary sources of technical data used in the program are as follows:

- controlled documents used in the design, manufacture, installation, commissioning and storage of cables;

- Operating Manuals, Operating Logs, Operating Manual Tests, verification bulletins, etc.;
- inspection and repair procedures, work orders, work plans, bulletins and inspection reports and non-destructive examinations, work reports for maintenance, repairs, replacements or alterations.

The main objectives of the collection and organization activity are the following:

- collecting design documents (including project changes during the operating period), specifications, applicable standards, regulatory requirements, certificates, reports, operating and maintenance manuals, etc.;
- identification of the technical characteristics of the equipment, the materials used in the manufacture, their properties and the operating parameters stipulated in the design;
- recording of actual operating parameters under normal operating conditions, transient regimes, overloads, planned tests and stops, relevant for the assessment of the aging process;
- recording the results of non-destructive tests, inspections and examinations, before and after interventions / repairs;
- registration of maintenance and repair / component replacements, or modifications.

The cable system elements included in the program are evaluated and monitored through current preventive maintenance activities, visual / tactile inspections and electrical tests. The activities specified above correlate with existing preventive maintenance activities for the serviced equipment.

Their activities and frequencies will be adjusted taking into account the evolution of the degradation mechanisms as well as the specific conditions imposed by each equipment such as the possibility of isolation, the importance category, the history of operation, inspections and repairs and the analysis of the results of the inspections and tests.

a) Ageing mechanisms requiring management and identification of their significance

Degradation mechanisms, their effects on the parts of the cable system at Cernavoda NPP, have been determined by the main causes that can lead to the degradation of the cable system (heat, radiation, handling, humidity). These are presented in the Table 3.3.

Table 3.3

Area / Subassembly	Mechanism of failure	Causes and Symptoms	Impact		Possibility of replacement		Tracking mode
			At the equipment level	At system level	Yes	No	
Cable / Insulation and jacket	**Thermal and thermal degradation** of organic parts caused by ambient heat and ohmic heat (Joule - Hertz effect)	Heat, Oxygen	Brittleness, cracking, melting, discoloration	Low insulation resistance, electrical failure, increased vulnerability of defect in hard working environments	Yes	No (for cables passing through reactor building penetrations)	Visual Tactile indenter E-AT-B Electrical tests
	Chemical decomposition of organic parts, radiation-induced oxidation	Radiation, Oxygen	Brittleness, cracks, discoloration, dilation	Low insulation resistance, electrical failure	Yes		Visual indenter E-AT-B Electrical tests Chemical tests (OIT, Spectroscopy)
	Wear due to work in the area, human trafficking, or faulty cable system support practices	External mechanical effort	Cutting, cracking, chopping	Interrupt loops / circuits Low insulation resistance, electrical failure	Yes		Visual Electrical tests
	Arc-over	Fault by manufacturer (holes) Electric arc	Brittleness, cracks Electric trees	Short circuits	Yes		Visual Electrical tests
	Structural degradation due to water / moisture in protective channels and pipes	Moisture	Water trees	Short circuits	Yes		Chemical tests (OIT, Spectroscopy)

Area / Subassembly	Mechanism of failure	Causes and Symptoms	Impact		Possibility of replacement		Tracking mode
			At the equipment level	At system level	Yes	No	
Connection element **(Connector, slipper, contact surface)**	**Electrochemical stress** (moisture, oxygen etc)	Corrosion and oxidation of metals	Corrosion and oxidation of external contact surfaces	Increasing contact resistance and local warming, loss of circuit continuity	Yes		Visual Electrical tests
	Fatigue and deformation of metals	Vibrations, Stretching Efforts	Weakness of shoe and conductor contact, shoe rupture	Interrupt loops / circuits.	Yes		Visual Electrical tests
Racks system and cable protection pipes	**Vibrations** Construction activities Inadequate maintenance practices	Missing: supports, anchorages, sealing caps (fireproof)	Mechanical degradation of cable insulation leading to short circuits and overload	Interrupt loops / circuits.	Yes		Visual
	Vibrations Weak parts of the cable rack system Improper reinstallation after maintenance or project modifications	The systems of vertical supports damaged or broken parts	Mechanical damage to cable insulation due to the concentration of vertical loads, leading to cable break at the top of the cable rack Degrading the seismicity characteristics.	Interrupt loops / circuits.	Yes		Visual Tactile

b) Establishment of acceptance criteria related to ageing mechanisms

Knowing and analyzing degradation mechanisms and defect contributing factors is the basic information for determining behavior in operation, assessing cable status, and identifying actions to reduce degradation processes and their associated effects.

The inspection / testing activities carried out on the cable system are aimed at maintaining its integrity in order to be able to fulfill its function according to the design requirements.

The following describes all the tests currently used in the industry to determine defects, aging and duration of the remaining life of the cables. Over the last 20 years, specialized external organizations have developed numerous reports on cable aging and cable status monitoring practices in nuclear power plants. The national standard uses the PE 116 (and related technical datasheets), which presents the methods and tests used to determine the condition of the cables after certain operating periods. Below are the most effective test methods for determining the condition of the cables, depending on their voltage level.

These methods can be globally divided as follows:

- Destructive methods / Non-destructive methods;
- Methods for determination of cable status (aging, remanence) / Methods for determining cable defects (local, global).

The methods of tracking the status of the cable system are divided as follows:

A) Preventive Maintenance Methods (Electrical Tests, Visual and Tactile Inspection) that can be performed with internal resources and addresses the following components:

- cable;
- connection elements;
- protective systems and cable supports.

B) Methods for determination of defects and remaining lifetime (Chemical tests, Spectroscopy, Elongation at break, Complex electrical tests) that can be performed with external resources and addresses:

- cable insulation;
- cable jacket.

The most significant degradation mechanisms affecting the cable system are listed in "Aging Management Guideline for Commercial Nuclear Power Plants - Electrical Cable and Terminations, Table 4-18", SAND 96-0344, "Cable System Aging Management", EPRI 2002, Report 1003317 and "Medium Voltage Underground (MVU) Cable White Paper" NEI 06-05 (April 2006).

Other standards used by Cernavoda NPP in the AMP for cables include the following:

- PE 116, Norm of tests and measurements for electrical equipment and installation (national standard);
- IEEE Std 400-2001, “IEEE Guide for Field Testing and Evaluation of the Insulation of Shielded Power Cable Systems”
- Norm on the design, execution and operation of the electrical installations associated with buildings I7 - 2011 (national standard).

Operating experience, both internal and external, is screened for identifying events and lessons learned that are relevant and applicable for the electrical cables of Cernavoda NPP. For example, the EPRI Report #3002007991 – "Plant Engineering: Low-Voltage Cable Susceptibility to Wet Aging" was analysed and no actions were found necessary because at Cernavoda NPP there are no cables cables that are subject to submersion and exposure to wet environments can occur only accidentally for a limited time. The station internal procedures and the PM program offer the necessary means for checking periodically and detecting the deterioration of cable insulation (e.g. visual inspections, insulation resistance measurement, color change).

The technical support for the development and research required for the AMP program for electrical cables will be provided by participating in the COG R&D program, which can provide answers to the development of a unique cable aging monitoring technique that applies to all types of insulation and cables. Until now, COG reports have been issued regarding the condition of low voltage cables and the development of non-destructive methods for cable testing. The continuation of this research will be useful for the development of the cable program at Cernavoda NPP.

03.1.3A Monitoring, testing, sampling and inspection activities for electrical cables

The knowledge and analysis of the degradation mechanisms are the basis for determining the causes of defects, being the starting point in the preparation of the lifetime studies or the assessment reports of the state of the structures and components as well as of the action plans to reduce the process ageing of cables and its effects.

The analysis of degradation mechanisms and fault modes is done for each cable family and aims at technical documentation of the following aspects:

- the predominant degradation mechanisms for each cable type
- possible failure modes and their effects
- status parameters that highlight degradation
- the means of determining (measuring) them and the frequency of their collection.

The main degradation mechanisms and defective cable modes proposed for the PLiM program are described below:

Thermal degradation: This mechanism can be defined as a thermo-oxidative degradation of the organic parts caused by the heat of the environment and the heat emitted by the electric current passing through a cable (Joule - Hertz effect). It is the most important mechanism that leads to the long-term aging of the cable system. As long as the cables are installed in the vicinity of hot equipment, in enclosed environments with high temperature or the cable is under-dimensioned, the risk of thermal degradation and defects increases after a shorter operating period than the one projected. To control the appearance and development of this defect, visual inspections, complex electrical tests, use of Indenter, and destructive checks of the E-at-B type are required.

Chemical decomposition by irradiation: In the areas where the cables are affected by radiation, the chemical decomposition of the organic parts of the cable (insulation, mantle) occurs, resulting in cracks that are the main defect precursors. Radiation will also cause hardening of the most important insulation materials. Radiation, as well as temperature, causes an increase in the cross-sectional (chain section) of the long chain formed by polymeric molecules in rubber materials. In the end, the material will suffer splitting, through which long chains will break into smaller segments. From here, the material will have a low tensile strength and will crack at lower bending forces. Inside the reactor building, radiation levels will generally cause fewer defects than thermal ageing, except where there are direct radiation outbursts through the walls of biological screens installed around process lines. In order to control the appearance and development of this defect, visual inspections, spectroscopy, comparison of real values with those obtained from accelerated ageing, Indenter use and E-at-B and OIT destructive checks are required.

Structural degradation: This mechanism is carried out in high humidity conditions and leads to the appearance of water in the cable insulation / jacket and then to short circuit. If bending causes the insulation to break, contaminants and moisture, which will crack the cracks, can cause defects. If the cable does not bend after the aging, then the material will remain as an insulating tube. Another condition where humidity would cause degradation of insulating materials is an accidental condition in which fluid and steam are released at very high temperatures. This problem is being investigated at this time for EPR insulation covered with Hypalon slides. After a severe aging, the Hypalon sheath can crack in an accident involving steam. Then the crack will propagate into EPR, exposing the conductor to external factors. In order to control the appearance and development of this defect, visual inspections, spectroscopy, chemical tests and destructive checks of type E-at-B are required.

Arc-over: It is the phenomenon of electrical discharge between the metal parts of a cable in the gas layer that appears near the insulation, due to defects in the manufacturer or installation. It occurs at medium and high voltage cables and the effects are devastating. In order to control the appearance and development of this defect, electrical tests, spectroscopy, chemical tests are required.

The types of representative cables that are included in the PLiM program will be subjected to tests to determine ageing and remaining life:

- Tests to be performed with internal resources (inspections, insulation resistance, ohmic resistance, Indenter);
- Tests to be performed by a specialized company (accelerated aging + Indenter, chemical tests, complex electrical tests, elongation at break, OIT, FTIR, determination of plastic content).

Given the above elements, one can conclude that:

Connection elements are tracked by:

- Electrical tests, visual and tactical inspection (PM activities)
- Repairs / Replacements (PM activities)

The protective system and cable support is tracked by:

- Visual and tactical inspection (PM activities)
- Repairs / Replacements (PM activities)

Cable with its components:

I. conductor is tracked by:
   - Electrical tests (PM activity)

II. insulation is tracked by:
  - Electrical tests, visual and tactical inspection (PM activities)
  - indenter
  - Elongation at break (E-at-B)
  - Chemical tests
  - Spectroscopy

III. Screen is tracked by:
  - Electrical tests (PM activity)

IV. The shield is tracked by:
  - No inspection activities required

V. The jacket is followed by:
  - Electrical tests, visual and tactical inspection (PM activities)
  - indenter
  - Elongation at break (E-at-B)
  - Chemical tests
  - Spectroscopy.

03.1.4 Preventive and remedial actions for electrical cables

The list of the most relevant degradation mechanisms, breakdown modes, probable causes and the main contributors to the degradation rate applicable to cables are shown in the following table (Table 3.4). At the same time, the same table highlights the methods and activities needed to identify and prevent degradation.

Component	Degradation mechanism	Factors that influence degradation	Possibilities of identification and prevention	Preventive maintenance strategy
Neutral cable	Corrosion	Water, Chemicals	Electrical tests Visual inspections	Periodic Inspection (Electrical Tests, Visual Inspections)
Cable lead	Corrosion (Al multifilar only)	Water, Chemicals	Electrical tests Visual inspections Laboratory tests	Replacement
	Mechanical stress (bending)	Incorrect bend radius	Visual inspections	Recovery / Re-route cable route
Jacket / Cable Insulation **(general)**	Chemical decomposition (discolouration, shrinking)	Existence of acid droplets Get rid of chemicals	Visual inspections	cleaning Spectroscopy Regular (visual) inspection to avoid recurrence
	Miscellaneous defects (plastic sucking, soaking, greasy surface, contaminants)	Faulty manufacturer Installation defects	Electrical tests Visual inspections Laboratory tests	Periodic Inspection (Electrical Tests, Visual Inspections) Accelerated aging tests Chemical tests Spectroscopy Elongation to Break Replace the cable in the next stop of the unit

Component	Degradation mechanism	Factors that influence degradation	Possibilities of identification and prevention	Preventive maintenance strategy
Jacket / Cable Insulation **(general)**	Structural defects insulating material *Air bubbles*	Humidity under the shell Faulty manufacturing	Visual inspections Electrical tests Chemical tests	Assessing the degree of degradation Inspection at next stop Replacement
	Swelling	External local heat source	Visual inspections Identify and remove the heat source Screening the heat source	Evaluarea gradului de degradare (teste de imbatranire accelerata)
	Fragility	Heat in excess (amperage, local source) Excessive radiation	indenter Electrical tests Visual inspections Identifying the source Re-analyzing circuit loads (current charging capacity) Screening the source	Periodic inspection (if not very fragile) Replacement (if fragile)
	Cracks	Aggravated aging Mechanical degradation Radiation light (ultraviolet)	Electrical tests Visual inspections (The crack depth will be determined)	Periodic Inspection (Electrical Tests, Visual Inspections) to determine the degree of change of the depth

Component	Degradation mechanism	Factors that influence degradation	Possibilities of identification and prevention	Preventive maintenance strategy
Cable insulation	Electric trees	Fault by manufacturer (holes)	Indenter Electrical tests Chemical tests	Assessing the degree of degradation (accelerated aging tests)
	Electric trees	Electric arc	Indenter Electrical tests	Assessing the degree of degradation Inspection at next stop Replacement
	Water trees	Fault by manufacturer (holes)	Indenter Electrical tests Chemical tests	Assessing the degree of degradation (accelerated aging tests)
Cable insulation	Mechanical stress	Inappropriate drawing methods	Visual inspections Laboratory tests	Periodic inspection / replacement Accelerated aging tests Spectroscopy Elongation to Break
	Mechanical degradation (leading to contamination)	Inappropriate transport, handling or installation	Visual inspections Laboratory tests	Periodic inspection / replacement Spectroscopy Elongation to Break
	Material degradation	Storage / Installation	Visual inspections	Periodic inspection / replacement Spectroscopy Elongation to Break
	Degradation due to animals	Rodent animals	Visual inspections	Removal Causing Factors
Cable insulation (TR-XLP, EPR-DR only,	Transient thermal	High temperature operation, High load factor	Visual inspections Electrical tests	Periodic inspection / replacement Spectroscopy

Component	Degradation mechanism	Factors that influence degradation	Possibilities of identification and prevention	Preventive maintenance strategy
Rubber Butilic)			Re-analyze circuit loads Screenings	Decreasing the amperage value
Insulation screen	Strength and fragility (lead to loss of conductivity)	Contamination with oil, resins, preservatives	Visual inspections	Cleaning Spectroscopy Regular (visual) inspection to avoid recurrence
	Strength and fragility (lead to loss of conductivity)	Thermal aging (depends on load factor, ambient temperature, UV radiation)	Indenter Electrical tests Visual inspections Screening the source	Inspectie periodica
	Loss of adhesion	Thermal immersion (depends on load factor and ambient temperature)	Electrical tests Visual inspections	Periodic inspection
Jacket the cable	Failure	Digging, Thick elements in the cable duct	Analyzing design documents	Manual digging
	Degradation due to animals	Rodent animals	Visual inspections	Removal Causing Factors
	Strength and fragility (lead to loss of conductivity)	Thermal aging (depends on load factor, ambient temperature, UV radiation)	Indenter Visual inspections Laboratory tests	Periodic inspection / replacement Accelerated aging tests Spectroscopy Elongation to Break
	Strength and fragility (lead to loss of conductivity)	Conductor overload, Circulation current through	Indenter Visual inspections	Periodic inspection / replacement Accelerated aging tests

Component	Degradation mechanism	Factors that influence degradation	Possibilities of identification and prevention	Preventive maintenance strategy
		null	Laboratory tests	Spectroscopy Elongation to Break
Joints	Partial discharge and arc-over	Installation Errors	Visual inspections Good installation practices	Periodic Inspection / Replacement
	Arc-over	Human errors (Exposure of cables to excessive humidity) Faults by manufacturer	Visual inspections	Periodic Inspection / Replacement
	Overheating	Inappropriate crippling practices	Visual inspections Electrical tests	Periodic Inspection / Replacement
	Corrosion	Water, Chemicals	Visual inspections	Periodic Inspection / Replacement Cleaning
Terminal heads	Partial discharge and arc-over	Installation Errors	Visual inspections Good installation practices	Periodic Inspection / Replacement
	Arc-over	Faults by manufacturer	Visual inspections Electrical tests	Periodic Inspection / Replacement
	Arc-over	Dust and moisture	Visual inspections Electrical tests	Periodic Inspection / Replacement
	Arc-over	UV radiation	Visual inspections	Periodic Inspection / Replacement
	Overheating	Inappropriate crippling practices	Visual inspections Electrical tests	Periodic Inspection / Replacement
	Corrosion	Water, Chemicals	Visual inspections	Periodic Inspection / Replacement

Component	Degradation mechanism	Factors that influence degradation	Possibilities of identification and prevention	Preventive maintenance strategy
				Cleaning
Connector / Contact Area	Electrochemical stress (moisture, oxygen etc)	Corrosion and oxidation of metals	Visual inspections Electrical tests	Periodic Inspection / Replacement
Connection element / Slipper	Fatigue and deformation of metals	Vibrations, Stretching Efforts	Visual inspections Electrical tests	Periodic Inspection / Replacement
Channel and pipe protection system for cables	Vibrations Construction activities Inadequate maintenance practices	Missing: supports, anchorages, sealing caps (fireproof)	Visual inspections	Periodic visual inspections

Table 3.4

03.2A Licensee's experience of the application of AMPs for electrical cables

The Program Manual for the AMP for electrical cables at Cernavoda NPP was issued in 2009, based on reference documents issued by EPRI, COG, IEEE, IAEA and on plant-specific information. It presents the following information:

- generic and specific degradation mechanisms;
- the analysis of the impact of specific degradation mechanisms;
- maintenance, testing and surveillance activities, together with the principles for their selection for various types of cables and insulation materials;
- description of the AMP for the electrical cables;
- selection and classification of the components of the cable system included in the AMP;
- the list of cable families grouped function of the voltage, type of insulating material and the systems supplied;
- the timeline of major activities belonging to the AMP for electrical cables, for 2009 – 2019.

The current in-service monitoring activities provide physical status monitoring, based on the as-found files of visual inspections (mainly performed during planned outages), as well as monitoring of the values and evolution of the operating parameters obtained in the electrical tests.

Health status reports provide the necessary technical information for an efficient management of safe operation and maintenance of selected cables. These reports contain sections with the analysis of the results of the monitoring of the functional parameters, the results of the inspections, the operating experience information, the evolution of the component condition. The Health Report will have a separate section mentioning the actions needed to correct indicators that have a worsening trend. Reducing the influence of these contributors will be achieved by identifying them in the inspections and removing the causes.

No significant age-related issues have been encountered so far, as the design and environmental qualification of the safety related electrical cables has ample margins when compared to the actual operating conditions. Nevertheless, for the cables included in the PLiM program it is necessary to continuously obtain information on their state and condition, so that measures can be taken before the occurrence of defects leading to unavailability of the serviced equipment, with a major impact in the operation of the plant.

A global Life Assessment (LA) study for the electrical cables is being contracted and will be available by the end of 2020, which will include also an overall assessment of the effectiveness of the AMP for electrical cables.

03.3A Regulator's assessment and conclusions on ageing management of electrical cables for Cernavoda NPP

CNCAN has reviewed the Program Manual for the AMP for electrical cables of Cernavoda NPP and found it comprehensive and in line with good practices at international level, as outlined in EPRI and IAEA guidance documents. However, since the program manual was first issued in 2009, CNCAN recommended that it is updated to take account of the latest relevant standards, results of R&D in the field of ageing management for NPP electrical cables and operating experience accumulated in the period 2009 – 2017.

CNCAN is performing regulatory oversight of the AMP for electrical cables using the information / records coming from operation and maintenance logs, inspections performed during the planned outages, system health reports, abnormal event reports, etc.

Based on the experience so far, CNCAN concluded that the in-service inspections and tests have been effective in preventing unexpected age-related degradations in safety-related electrical cables. More in-depth regulatory reviews will continue when the LA study for electrical cables will be finalized, while the planned inspections programs will include ageing management of electrical cables on a regular basis.

03B ELECTRICAL CABLES - TRIGA RR

03.1.B Ageing management for electrical cables.

Ageing management for electrical cables is a part of the overall AMP presented in Chapter 2. The AMP for electrical cables interfaces with other programs, as presented in Chapter 2, with some specific technical requirements applicable to electrical materials-cables.

03.1.1B Scope of ageing management for electrical cables

The scope of the AMP for electrical cables is:

- To identify and to evaluate the degradation which may lead to failure and malfunction of systems to which the cables belong;
- To predict future performances;
- To decide on the type and time of preventive action;
- To optimize where possible the operating condition;
- To apply the practices forecasted for reduction of ageing degradation;
- To identify the trends and new emerging ageing effects before they jeopardize the safety and service life.

03.1.2B Ageing assessment of electrical cables

Electrical cables important to safety of the reactor

Cables important to the safety of the reactor are those which belong to systems which perform the safety functions described in the previous chapter. Those cables are classified following the safety class of the system which incorporate the concerned cable.

There are two categories of cables:

- Cables which belong to the reactor control system and safety system;
- Cables which belong to the engineering safety features and support system of those;

Methods and criteria used for selecting electrical cables with the scope of ageing management

The selection of cables is the result of engineering design of systems to which the cables belong, for this scope only standard environmental qualified cables are selected.

The criteria concern:

- Requested reliability of cable for periodic replacement or for life of system;
- Decision of the type and time of preventive maintenance;
- Environmental qualification EQ;
- Fail-safe of system and safe shutdown of reactor;
- Radiation resistant;
- Fire resistant in certain condition;
- Ability to be installed and replaced when necessary.

Process/procedures for the identification of the ageing mechanism related to cables

- Visual inspection;
- High voltage testing;
- Temperature during operation of some cables;
- Thermography of cable, junctions and terminals in both ends;
- Insulation resistivity;
- Care cable electrical resistivity;
- Appearance, support – connection panels;
- Vibration.

For each cable subject to periodic inspection, testing, condition monitoring by dedicated procedures, the data are recorded, processed and used for ageing effects trending and identification of prevalent ageing mechanism in spite of ageing the design basis remain valid.

Grouping criteria for ageing management purposes

The TRIGA 14MW research reactor safety systems do not contain high voltage cables above 3kV subject of hostile environment.

Medium voltage cables buried or in trench are those outside of the reactor building and sustain the normal operation of reactor, operation of primary and secondary coolant system. The failure of those cables affects only the availability of reactor.

Neutron flux and temperature measurement cables are used in the reactor systems which are installed in the hostile environment of direct radiation of reactor. The logic diagram of reactor protection showns that in case of failure the safety function for reactor safety are not affected.

03.1.3B Monitoring, testing, sampling and inspection activities for electrical cables

Monitoring of cables failure is ensured by design of electrical system following the applicable standards and regulation in the real time of operation.

Ageing of cables followed by decrease in electrical parameters is assessed by periodic testing and inspection activity focused on condition of operation of cables, load and environment. Specific procedures are developed and applied for the cables testing depending on the environment.

03.1.4.B Preventive and remedial actions for electrical cables

Preventive actions are determined by design specification concerning environmental qualification related to the reactor environment, position inside the reactor building, temperature of operation, radiation exposure and integrated radioactive dose compatible with cable environmental qualification.

Remedial actions concern the entire length of cable replacement with a new cable with the condition of root causes of failure identification and prevention of repetition.

03.2B Licensee’s experience of the application of AMP for electrical cables

The experience consists of the application of maintenance programs for the systems and equipment to which the cables belong. Regular testing of cables and data recording and processing reveal the trends of ageing effect on cables. Unplanned maintenance of failed system reveal that the failure was produced by a defective cable. Cause of failure analysis lead to further improvement corrective actions to prevent the repetition.

03.3B Regulator’s assessment and conclusions on ageing management of electrical cables for TRIGA RR

The regulatory oversight of the AMP for electrical cables consists of planned inspections as well as reviews of the information / records coming from operation and maintenance logs, inspections performed by the licensee, system health reports, abnormal event reports, etc.

Based on the experience so far, CNCAN concluded that the in-service inspections and tests have been effective in preventing unexpected ageing-related degradation in safety-related electrical cables.

04A CONCEALED PIPEWORK – Cernavoda NPP

04.1A Description of ageing management programs for concealed pipework

At Cernavoda NPP, the ageing management program for concealed piping (subsequently named Buried Piping Program or BP Program) was initiated in 2011, following an Area for Improvement (AFI) received after a WANO/INPO peer review of the plant in November 2010.

The BP Program at Cernavoda NPP was initiated in accordance with INPO AP-913 – Equipment Reliability Process main frame and was developed by using the following reference documents:

- EPRI TR 1016456, „Recommendations for an Effective Program to Control the Degradation of Buried and Underground Piping and Tanks", December 2010;
- NEI 09-14, „Guideline for the Management of Buried Piping Integrity", Nuclear Energy Institute, January 2010;
- COG-08-4066, „Guideline For Implementing a Buried Piping Program at CANDU Stations", April 2009.

Other documents produced by the industry have also been considered (e.g. NUREG/CR-6876, BNL-NUREG-74000-2005, „Risk Informed Assessment of Degraded Buried Piping Systems in Nuclear Power Plants", June 2005), as well as internal and external operating experience. This approach makes the Cernavoda NPP AMP for BP compliant with industry best practice.

The Cernavoda BP program was implemented in 3 major stages:

Stage 1 - Defining the program requirements, which was completed between 2011-2012 and included:

- preparation of the program strategy in order to define program requirements, approach, and resources needed for program implementation;
- allocation of internal resources (Program Engineer) and necessary budget;
- basic training of allocated BP Program Engineer.

Stage 2 - Perform initial evaluation of BP actual condition, which was completed between 2012-2014 and has included the following activities:

- collect required data (system drawings and documents related to underground piping lines and tanks, materials of construction, operating conditions, previous inspection data, cathodic protection measurement results, etc);
- define BP program initial scope by ranking each buried piping system based on the "Risk Ranking Number Value";
- perform initial risk-ranking process for each buried piping systems from scope;
- establish initial inspection scope based on the allocated risk;
- perform initial inspections on high-risk system buried piping;
- perform initial condition assessment of high-risk buried piping;
- perform soil analysis (defining soil sampling scope, perform soil sampling, analyze and report the soil samples results).

Stage 3 - Development of the BP Program started in 2015, by contracting a 4-years integrated services with an external company. This stage is under implementation and includes the following activities:

- perform electronic plant mapping of BP in both Cernavoda NPP units;
- perform a detailed "Risk Ranking" (at piping segment level) using BPWORKS™ software elaborated by EPRI;
- issue multi-annual plan for BP inspections in Unit 1 & Unit 2;
- perform BP Inspections (direct and indirect);
- perform "Fitness for Service" assessments based on both inspection results and engineering evaluations, in order to make run/repair decisions and to identify mitigation strategies;
- prepare an Asset Management (Long Term) Plan for managing the structural and leakage integrity of buried piping. Key elements of the Asset Management Plan shall include:
    - Inspection plans;
    - Planned maintenance activities;
    - Plans for repair;
    - Anticipated buried piping replacements.

  The Asset Management Plan will be a living document that will be periodically reviewed as more plant data becomes available through physical assessments and other means and also as industry knowledge and technology evolve.
- repair affected buried piping as required.
- use feedback to revise risk-ranking and AMP as required.

The performance of all underground plant components (piping and tanks) selected in the BP Program is continuously monitored via Component Health Monitoring (CHM) process, actions for improvement are recommended and the progress in their implementation is annually reported to high level plant management. The resolutions of all technical problems identified in Component Health Monitoring Reports (CHMR) are ranked and graded, based on the safety impact and are planned accordingly afterwards.

04.1.1A Scope of ageing management for concealed pipework

In accordance with the approved related Program Manual and the EPRI recommendations mentioned above, at Cernavoda NPP this program applies to all underground piping contained within the critical, non-critical and safety related systems of Unit 1 and Unit 2 and includes:

- Piping buried in soil;
- Piping encased in concrete;
- Piping located in trenches.
- Piping located inside valve pits or access pits.

Cernavoda NPP BP program includes in its scope also the following major commodities:

1. All existing underground tanks, which contain fuel oil and supply the Standby Diesel Generators (SDG) and the Emergency Power Supply (EPS) generators located, for both Unit 1 and Unit 2 of the plant.
2. The Cathodic Protection systems (galvanic type) installed on the EPS & SDG piping and tanks, for both Unit 1 and Unit 2 of the plant.

The scope of Cernavoda NPP BP Program does not apply to other types of underground commodities such as conduits, concrete pits, cables and intake structures, which are monitored by separate programs.

A full list of the piping systems included in the scope of Cernavoda BP Program based on their criticality is presented in the following Table 4.1.

Table 4.1 - List of system (BSI) included in the Buried Piping Program scope, for U1 & U2

ID	BSI	System Name	Piping Material	Fluid	Safety related	Critical system	Remarks
1.	34320	Emergency Core Cooling	CS	H_2O	SSS	C	-
2.	34410	Spent Fuel Cooling/Purification	SS	H_2O + D_2O	SRS	C	
3.	34610	Emergency Water Supply	CS	H_2O, steam	SSS	C	-
4.	43220	Condensate Make–up & Storage	SS	H_2O	SRS	C	Piping in trench
5.	52320	Fuel Oil System (for Standby Diesel Generators)	CS	fuel oil	SRS	C	-
6.	71210	Circulating Water Supply	CS	H_2O	-	C	Partial concrete embedded piping
7.	71310	Raw Service Water	CS	H_2O	SRS	C	Partial concrete embedded piping
8.	71400	Fire Protection	CS, HDPE	H_2O	SRS	C	-
9.	71500	Potable Water Supply	CS, HDPE	H_2O	-	-	Galvanized steel
10.	71610	Raw Water (from pila dubla to WTP)	CS	H_2O	-	C	
11.	71690	RSW Backup Cooling	CS	H_2O	SRS	C	in U1 only
12.	71740	Active Drainage (from S/B to condenser discharge)	CS	H_2O	SRS	-	Partial concrete embedded piping
13.	71750	Sewage	CS	H_2O	-	-	-

ID	BSI	System Name	Piping Material	Fluid	Safety related	Critical system	Remarks
14.	72210 / 52900	Main Fuel Oil Distribution (for Emergency Power Supply - EPS)	CS	fuel oil	SRS	C	-
15.	73020	Hot Water (from estacade to Adm. Building – Pump Station)	CS	H_2O	-	-	-
16.	79210	Liquid Handling	CS	low rad. waste water	SRS	-	Partial concrete embedded piping

Notes: H_2O notation includes: river water / service water / demi water.
Legend: Safety related: C - Critical System ; SRS - Safety-Related System; SSS – Special Safety System
Piping material: CS – Carbon Steel; SS – Stainless Steel; HDPE – High Density Polyethilene

a) Methods and criteria used for selecting buried piping within the scope of the Cernavoda NPP ageing management program

To establish the scope of the BP ageing management program, Cernavoda NPP has used the risk-ranking methodology, which is based on the consequence and likelihood of pipe failure (see Figure 4.1 below).

The selection of the BP ageing management program scope was performed in two steps:

Step 1: identification and risk-ranking of the plant systems (BSI) containing buried or underground piping.

Step 2: Identification and risk-ranking of the buried & underground piping lines segments

For step 1 (initial risk-ranking at system level), the evaluation methodology involved:

- Collecting required data (design drawings and documents related to buried assets, piping materials, operating parameters, etc);
- Performing initial risk-ranking analysis for each BP system by assessing /justifying the consequence of failure, likelihood of failure and its operating experience and then calculating the product of each factor to determine the “Risk Ranking Number Value” (or Risk Score);
- Use the “Risk Score” to define BP initial scope and to list each system evaluation in descending value of “Risk Ranking Number Value” (or Risk Score)”.

The result of Cernavoda NPP initial risk-ranking (at system level) was documented in a specific Information Report. A summary of the results is presented in the Figure 4.1:

Figure 4.1 – Initial Risk-Ranking of Buried Piping Systems at Cernavoda NPP

No.	System BSI	System Name	Pre-Assessment (ordered by allocated score)			
			Impact Ranking Factor (IMP)	Likelihood of Failure Factor (LFF)	Operating Experience Factor (OEF)	Risk Priority Number
	34320	Emergency Core Cooling	50	3	3	450
	34610	Emergency Water Supply	50	3	3	450
	52320	Fuel Oil System (for Standby Diesel Generators)	45	3	3	405
	72210 / 52900	Main Fuel Oil Distribution (for Emergency Power Supply - EPS)	45	3	3	405
	71310	Raw Service Water	45	3	2	270
	71690	RSW Backup Cooling	45	3	2	270
	73020	Hot Water (from estacade to Adm. Building-Pump Station)	30	3	3	270
	71210	Circulating Water Supply	40	3	2	240
	43220	Condensate Make–up and Storage	45	2	2	180
	71750	Sewage	20	3	3	180
	79210	Liquid Handling	45	3	1	135
	71610	Raw Water (to Water Treatment Plant)	40	3	1	120
	34510	Resin Transfer	45	2	1	90
	71740	Active Drainage (from S/B to condenser discharge)	45	2	1	90
	71400	Fire Protection	45	0.5	3	67.5
	71500	Potable Water Supply	20	0.5	1	10

Risk Score Levels

Priority	Very High	High	Medium	Low
	≥ 360	**≥ 200, < 360**	**≥100, <200**	**< 100**

For step 2 (final risk-ranking, for pipe segments) the Cernavoda BP program scope was determined at a more detailed level by using BPWORKS software, an electronic application that was developed by the Electric Power Research Institute (EPRI).

The BPWORKS software application is made of two interconnected modules:

<u>Data Management Module</u> – which assists users to develop and manage a database for the buried & underground piping systems, by collecting critical information such as: soil environment, piping material, coating & lining, cathodic protecion and other significant parameters. This information can be modified and updated as necessary.

<u>Risk Ranking Module</u> – which calculates the Likelihood of degradation and the Consequence of failure of the piping systems using the collected data and a set of algorithms.

Using the two mentioned modules, BPWORKS software combines the likelihood of degradation with the consequences of failure to assess risk in a manner similar to ASME Section XI Code Cases N-560, by summing the consequence of failure and the likelihood of failure.

These results are then binned into a 12 categories matrix, as shown in the Figure 4.2 - Risk Ranking Matrix below:

Figure 4.2: Risk Ranking Matrix

	No Consequence	**Low Consequence**	**Medium Consequence**	**High Consequence**
High Likelihood				
Medium Likelihood				
Low Likelihood				

<u>High Risk segments</u> (shown in red), therefore, are those which are determined to have High Consequence and High Likelihood, High Consequence and Medium Likelihood, or Medium Consequence and High Likelihood.

<u>Medium Risk segments</u> (shown in yellow) are those that are determined to be High Consequence and Low Likelihood, Medium Consequence and Medium Likelihood, or Low Consequence and High Likelihood.

<u>Low Risk segments</u> (shown in green) are the remaining risk categories, with No Consequnce or Low Consequence.

In summary, BPWORKS establishes the BP segments which have similar charachteristics and determines the relative risk ranking of these segments, in order to prioritize their inspections. Additionally, this data can be used to schedule mitigation activities, piping replacements and repairs and develop strategies for an Asset Management Plan.

b) Processes/procedures for the identification of ageing mechanisms related to Cernavoda NPP buried piping program

The internal procedures used by Cernavoda NPP for the identification of ageing mechanisms for buried piping are based on the following reference documents:

- „Aging Identification and Assessment Checklist - Mechanical”, EPRI report, August 2004;
- „Balance of Plant Corrosion – The Underground Piping and Tank Reference Guide” Revision 1. EPRI, Palo Alto, CA: November 2013. 300200682;
- BPIG Position Paper No. 1, “Guidance for the Development of Buried and Underground Piping and Tanks Asset Management Plans,” October 2013;
- BPIG Position Paper No. 3, “Underground Piping and Tanks Integrity Program Excavation/Indirect Inspection/Direct Examination Checklist,” December 2013;
- Cathodic Protection Application and Maintenance Guide Volume 1 and Volume 2. EPRI, Palo Alto, CA: July 2013. 3002000596;
- Generic Aging Lessons Learned (GALL) Report – Final Report (NUREG-1801, Revision 2), XI.M41 Buried and Underground Piping and Tanks, December 2010;
- “Guideline for the Management of Underground Piping and Tank Integrity,” Nuclear Energy Institute, NEI 09-14, Revision 3, April 2013;
- “INPO Chemistry Department – Evaluator How-To Underground Piping,” Institute of Nuclear Power Operations, Revision 6, March 2011;
- Inspection Methodologies for Buried Pipe and Tanks. EPRI, Palo Alto, CA: August 2010.1021561;
- Nondestructive Evaluation: Buried Pipe NDE Reference Guide – Revision 2. EPRI, Palo Alto, CA: December 2012. 1025220;
- Nondestructive Evaluation: Buried Pipe NDE Reference Guide – Revision 2 – Addendum EPRI, Palo Alto, CA: November 2013. 3002000447;
- Nondestructive Evaluation: Inspection Methods for Tanks and Containment Liners. EPRI, Palo Alto, CA: December 2012. 1025215;
- NRC Inspection Manual, Temporary Instruction 2515/182, Review of the Implementation of the Industry Initiative to Control Degradation of Underground Piping and Tanks, August 2013.

c) Grouping criteria for ageing management purposes

The detailed “Risk Ranking” process (at piping segment level) using BPWORKS™ software involved the following activities:

- Collect piping segment specific data;
- Input data to the BPWORKS™ database module and define each piping segment;
- Use BPWORKS risk-ranking module to perform risk ranking for each piping segment;
- Define program exceptions;
- Use risk ranking to define BP program detailed inspection scope;

- Define inspection scope and methodology (i.e. direct and/or indirect examinations techniques).

The models used by BPWORKS software to determine the Likelihood of Degradation and the Consequences of Failure are presented below:

- ***BPWORKS Modeling of Likelihood*** (rows in the Risk Ranking Matrix)

BPWORKS software models three basic types of failure mechanisms, which are are defined as follows:

Occlusion: Loss of flow area caused by turbercles (corrosion nodules) formed by general corrosion acting alone, or with the participation of microbes, bi-valves (for example, zebra mussels), sedimentation, or debris.

Leakage: the escape of the contained fluid due to a hole or a crack.

Break: A break may consist of a burst (caused by over-pressure, either by a steady pressure rise or by a sudden pressure transient / water hammer, with or without corrosion), a guillotine break (due to soil movement) or a brittle fracture (due to overload of a pipe made of brittle material such as cast iron or a material susceptible to selective leaching, like cast iron, brass alloys).

The Likelihood of Leak, Break and Occlusion are then determined from a summation of the susceptibilities from each of the contributors (e.g., pitting, stress corrosion cracking, etc) and interpreted as follows:

Total susceptibility points < 20 means Likelihood = Low

Total susceptibility points between 20- 40 means Likelihood = Medium

Total susceptibility points > 40 means Likelihood = High

The various parameters that contribute to the Likelihood of Leakage, Break, or Occlusion are described in the BPWORKS Software Users Manual.

- ***BPWORKS Modeling of Consequence*** (columns in the Risk Ranking Matrix)

BPWORKS software assigns points to various parameters to determine consequence of failure. All parameters are equally important to consequence. To differentiate this, BPWORKS assigns weighting to the more important parameters.

The BPWORKS Software Users Manual shows the various weighting factors used. The strongest weights are assigned based on:

- Cost of repair (Very high – 7, High – 4, Medium – 2, Low – 0),
- Failure can cause collateral damage (High – 7, Medium – 3, Low – 1, No – 0),
- Failure effect safe shutdown or core damage frequency (Yes – 10, No – 0),
- Failure effect worker safety ? (Yes – 5, No – 0), and
- Failure results in lost generation (>7 days – 10, 3-7 days – 6, 1-2 days – 5, <1 day – 4, Power Reduction – 3, No – 0).

A point system is used to determine the Consequences of failure for both ID and OD degradation. The point system for the Consequences of failure is the same for OD initiated degradation and ID initiated degradation. The points associated with Consequences of leaks and break are different from the point system used for consequences of occlusion.

The Consequences of leak, break or occlusion are determined from a summation of the points and are interpreted as follows:

Total points < 2 means Consequences = Very Low

Total points between 2- 5 means Consequences = Low

Total points between 5- 10 means Consequences = Medium

Total points > 10 means Consequences = High

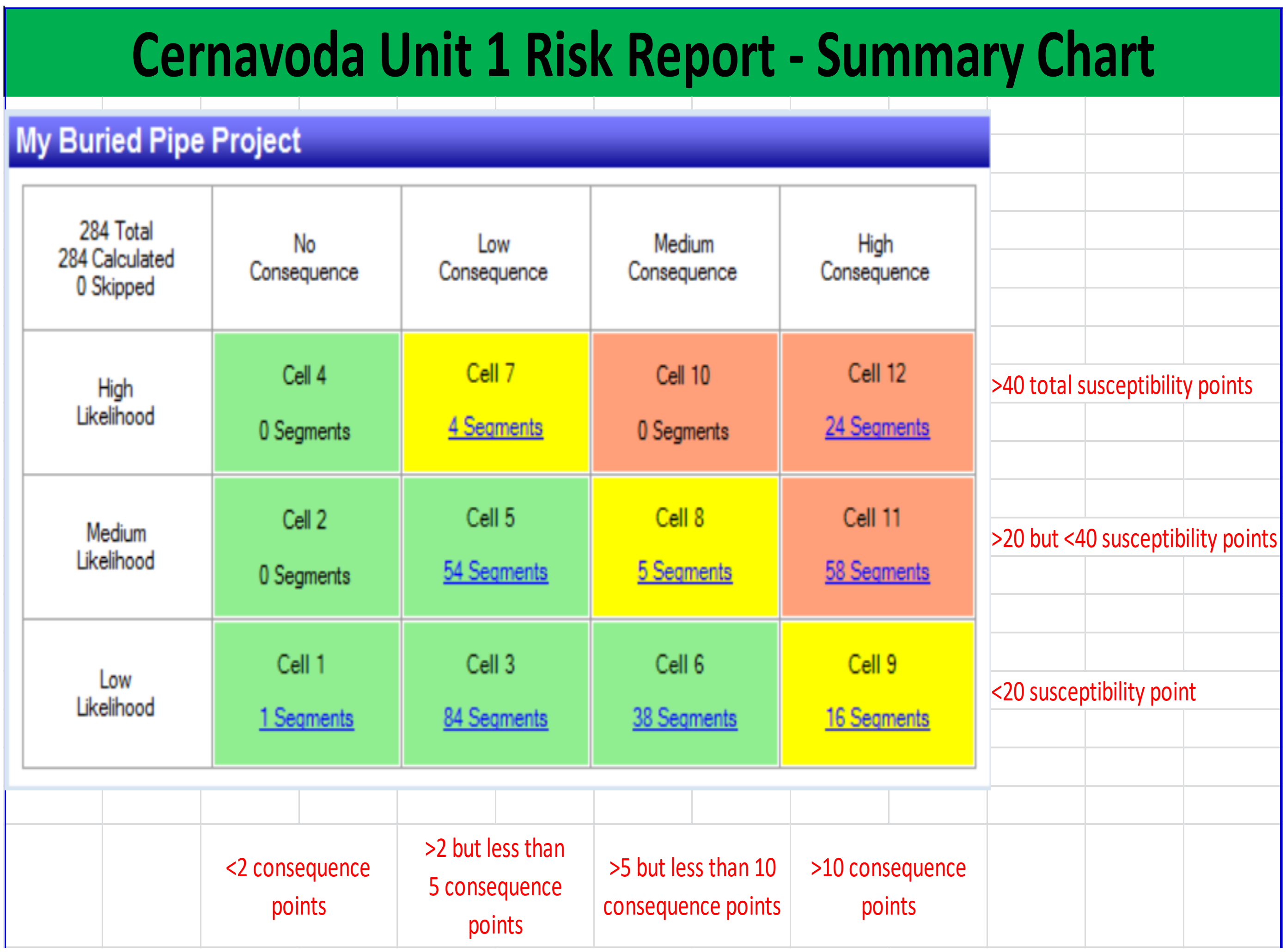


Figure 4.3: Risk Ranking Summary Chart – Unit 1

Fields which are used in the Risk Ranking algorithms are indicated in the BPWORKS Data Management Module by an asterisk (*) next to the name. If data is not entered into that field, the

Risk Ranking Module will be populated by default values which are generally conservative. This means that the better the data provided the more accurate the risk assessment will be. The default values used by BPWORKS are available in the Risk Ranking Module as well as in the BPWORKS Software Users Manual.

Following data entry into BPWORKS and the risk-ranking process, the buried piping segments in Unit 1 and Unit 2 of Cernavoda NPP have been grouped based on the ID and OD risk ranking results, as presented in the below Figures 4.3 and 4.4. The purpose of these Summary Chart images is to give an understanding of the distribution of results (number of segments) in the Risk Ranking Matrix format presented earlier.

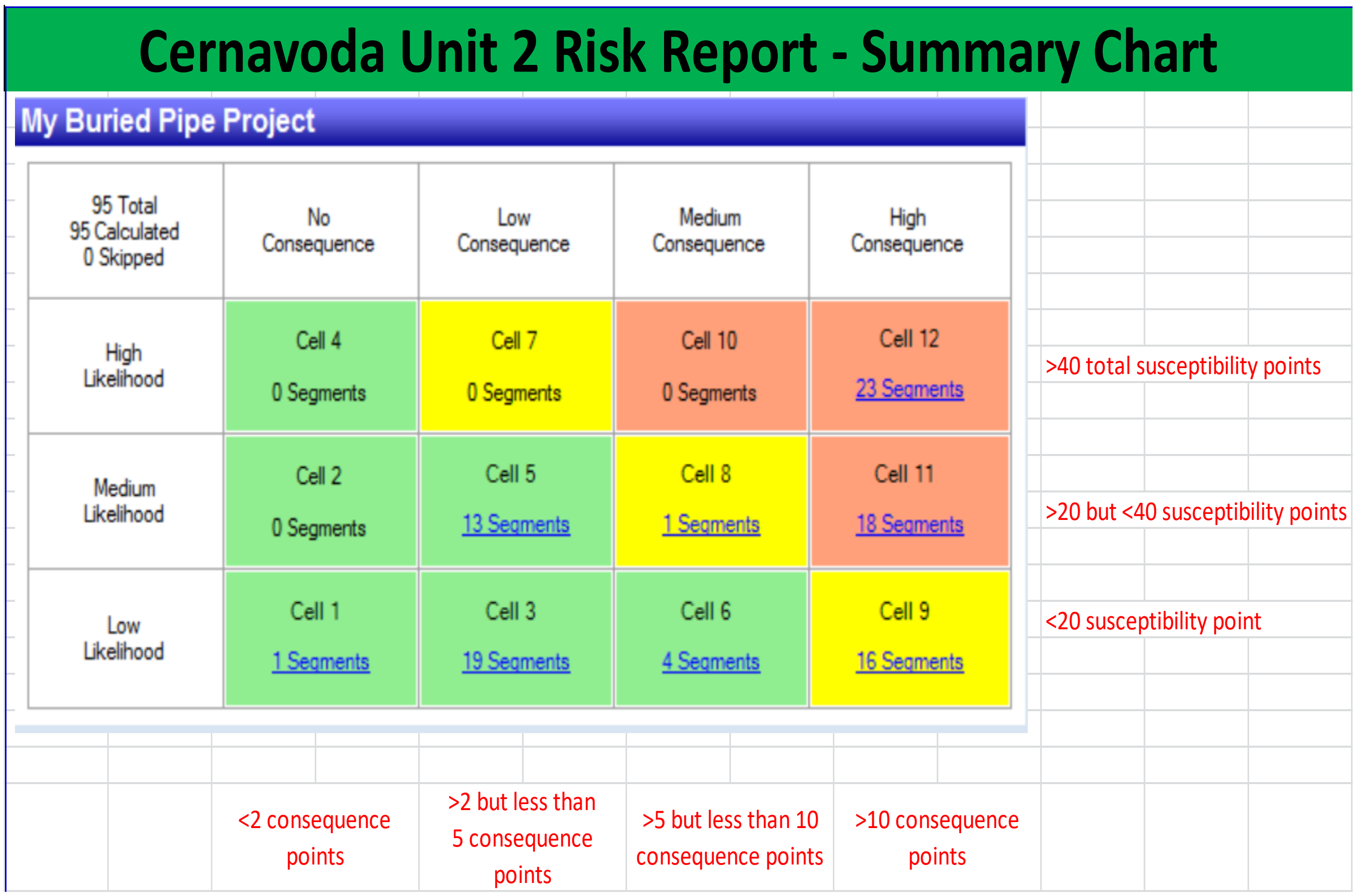


Figure 4.4: Risk Ranking Summary Chart – Unit 2

Following the risk-ranking process, there were 3 groups of underground piping identified at Unit 1 & Unit 2 of Cernavoda NPP with the highest risk of degradation.

These are presented as follows.

- **Group 1 - Piping systems that transfer fuel oil for emergency power generation;**

System name	System BSI code	Pipe diameter	Pipe material (per ASTM or STAS)	Fluid	Filling material
Fuel Oil System (for SDG - Standby Diesel Generators)	52320	1” 1 ½ “ 2”	Carbon steel A53 Gr.B	Fuel oil	Engineering fill
Main Fuel Oil Distribution (for EPS - Emergency Power Supply)	52920	1” 1 ½ “ 2”	Carbon steel A53 Gr.B	Fuel oil	Engineering fill

- **Group 2 - Piping systems that provide cooling to SSCs important to safety.**

System name	System BSI code	Pipe dia	Pipe material	Fluid	Filling material
Emergency Core Cooling (ECC)	34320	16” 2“	Carbon steel A106 Gr.B	Demi water	Engineering fill
Emergency Water Supply (EWS)	34610	10”	Carbon steel A 53 Gr.B	Raw (river) water	Engineering fill
Condenser Cooling Water (CCW)	71210	144 “ 114 “ 16”	Carbon steel OL 42.2K	Raw (river) water	Partially concrete embedded
Raw Service Water (RSW)	71310	60” 48” 14”	Carbon steel OL 42.2K	Raw (river) water	Partially concrete embedded
RSW Backup Cooling (BCWS)	71690	16”	Carbon steel A106 Gr.B	Raw (river) water	Partially concrete embedded

- **Group 3 - Piping systems that contain radioactive effluents;**

System name	System BSI code	Pipe dia	Pipe material	Fluid	Filling material
Liquid Handling system	79210	3 inches	Carbon steel A53 Gr.B	Low rad. waste water	Partially concrete embedded

04.1.2A Ageing assessment of concealed pipework

a) Ageing mechanisms requiring management and identification of their significance

The BP Program Manual, which is the technical basis of the BP Program at Cernavoda NPP, has taken into consideration all the degradation (ageing) mechanisms known in the industry. Then, based on the specific piping design data and operating conditions in Cernavoda NPP, the applicable degradation mechanisms that could potentially affect the function of an underground component (pipe/tank) together with related impact have been identified and considered for the application of a dedicated BP program.

These degradation mechanisms considered in the Program Manual for the underground piping and tanks at Cernavoda NPP nuclear plant are as follows:

- General Corrosion;
- Pitting;
- Under-deposit Corrosion;
- Crevice Corrosion;
- Galvanic Corrosion;
- MIC (Microbiologically Influenced Corrosion);
- External Corrosion;
- Chemically Influenced Corrosion;
- Occlusion.

For Group 1 of SDG and EPS fuel oil piping systems, which are made of carbon steel, the ID (inner diameter) corrosion is much less probable than the OD (outer diameter) corrosion, since the fluid is not a factor for degradation. For this group of components, the major degradation mechanisms of the external piping surface are general corrosion or pitting, as a result of coating degradation.

For Group 2 of cooling water piping systems, which are made of carbon steel and are partially concrete embedded, the outside concrete envelope acts as a passivation environment, protecting the pipes metallic external surface, and therefore the OD corrosion is unlikely to appear. The only piping areas which are exposed to OD corrosion are the ones located inside the accessible valve pits, where pipe is exposed to atmosphere. For this group of components, the main factor of degradation identified is the ID corrosion, as a result of degradation mechanisms specific to the raw (river) water: under-deposit corrosion, occlusion, MIC (Microbiologically-Influenced Corrosion).

For Group 3, which includes the Liquid Handling piping, the only portion of the 3” carbon steel piping which is exposed to OD corrosion is the one located inside a valve pit. The rest of the Liquid Handling underground pipe line is concrete embedded and therefore not exposed to OD degradation. For this group of components, the main factor of degradation identified is the ID corrosion, as a result of degradation mechanisms caused by the low rad waste water (mainly general corrosion).

b) Establishment of acceptance criteria related to ageing mechanisms

After inspection results are obtained, a Fitness-For-Service (FFS) assessment is made to calculate and demonstrate the adequacy of degraded pipe for continuing operation and the margin for failure, based on the identified degradation mechanisms and the design basis of the system.

For the FFS evaluation of buried and underground piping an estimated corrosion rate is used in order to predict the remaining pipe life and/or the time until the next inspection.

To prepare the Cernavoda NPP Buried Piping AMP, the following key technical documents have been used:

Guidance reports:

- COG-08-4066, Guideline For Implementing a Buried Piping Program at CANDU Stations, April 2009;
- EPRI TR 1016456, Recommendations for an Effective Program to Control the Degradation of Buried and Underground Piping and Tanks, December 2010;
- NEI 09-14, Guideline for the Management of Buried Piping Integrity, Nuclear Energy Institute, January 2010;
- INPO Chemistry Department, Evaluator How-To Underground Piping, Sept. 2010;
- Generic Aging Lessons Learned (GALL) Report – Final Report (NUREG-1801, Revision 2), XI.M41 Buried and Underground Piping and Tanks, December 2010.

Standards:

- ASME sect. XI In-Service Inspections;
- NACE SP0169-2013 Control of External Corrosion on Underground or Submerged Metallic Piping Systems – to determine the acceptance criteria for Cathodic Protection.

Design documentation:

- Plant system Design Manuals (DM) – to determine buried piping operational parameters and flow type (continuous, intermitent, stagnant);
- Isometric Drawings – to determine buried piping location and burial depth;
- Plant Layout – to determine buried piping location in respect to U1 & U2 buildings and to identify the valve pits location;
- Bill of Material – to determine buried piping material, pipe dimension and its nominal thickness.

Operating experience:

Both internal and external operating experience is screened and used to improve the AMP for BP at Cernavoda NPP. Examples of external operating experience reports used for the improvement of BP AMP related activities are provided below:

- COG OPEX record no. 51075 - Nondestructive Evaluation: Buried Pipe Structural Health Monitoring. This event was evaluated and found applicable for Cernavoda NPP. Actions have been taken to improve BP examination methods;

- EPRI 3002002949 - Recommendations for Managing Cathodic Protection. This operating experience report was evaluated and found applicable for Cernavoda NPP. A gap analysis was performed to identify the opportunities for improvement for the Cathodic Protection activities at Cernavoda NPP;
- RCA 2015- 02341E for COG OPEX - EPRI report no. 3002004395 - Nondestructive Evaluation: Buried Pipe NDE Reference Guide - Revision 3. This operating experience report was evaluated and found applicable for Cernavoda NPP. Two actions have been initiated (for EPRI report analysis and to identify improvements for the NDE examination techniques);
- COG OPEX - EPRI report 3002005294 - Soil Sampling and Testing Methods to Evaluate the Corosivity of the Environment for Buried Piping and Tanks at Nuclear Power Plants. This operating experience report was evaluated and found applicable for Cernavoda NPP, but no actions needed to be taken (soil sampling and analysis was already performed at Cernavoda NPP at the time the report was issued).
- INPO Event Report 17-2 (Level 4) - Information on Nuclear Industry Issues Related to Cathodic Protection systems. This operating experience report was evaluated and found applicable for Cernavoda NPP. As a result, an improvement action was initiated to elaborate an Information Report in order to present the results, conclusions and recommendations of the external evaluation completed in 2016 for the Cathodic Protection systems at Cernavoda NPP.

<u>R&D programs</u>:

The Buried and Underground Piping Program at Cernavoda NPP was developed in accordance with COG R&D study: COG-08-4066, Guideline for Implementing a Buried Piping Program at CANDU Stations, issued in 2009. Since 2006, Cernavoda NPP is participating to COG R&D sub-programme for Chemistry, Materials and Components, and BP Programme Engineer is participating to the the specific COG BP Working Group.

04.1.3A Monitoring, testing, sampling and inspection activities for the concealed pipework

<u>Sampling</u>

To evaluate the corrosivity of the environment for the plant buried piping and tanks, soil sampling and analysis were performed in 2013 in both units, at locations close to these underground components.

Sampling was done twice per every location, in winter and summer time, in order to determine the differences of soil properties between the “dry” and “wet” seasons. For each soil sample, physical composition and a set of chemical parameters have been determined, including pH, rezistivity, moisture content, chlorides, sulphites, etc.

This activity was performed by a specialized, external company and has involved:

- Execution of geotechnical drills in 22 relevant locations in Unit 1 & Unit 2;
- Sampling the soil probes from the drilling locations;

- Analyzing each of the soil samples, to determine physical composition and the set of chemical parameters;
- Documenting the soil analysis results for each unit.

Results of the soil analysis have been used in the risk-ranking assessment, by input the required technical data into the BPWORKS software.

Soil sampling and analysis activities are planned to be done during every specific buried piping inspection which involves digging the soil and the results are planned to be introduced in the BPWORKS, to update the related risk-ranking for that specific pipe segment/location.

<u>Inspections</u>

a) Initial inspections

In accordance with the BP Program strategy, piping identified with high risk during preliminary risk assessment (at system level) have been inspected to determine their as-found condition. Inspections were executed for Unit 1 piping, because this unit is the first one placed into operation in 1996, therefore its piping is more susceptible to degradation than Unit 2 piping (which was placed in operation in 2007).

For inspection purposes, direct examinations have been used, justified by the fact that this is the most conservative examination method.

These direct inspections involved the following generic activities:

- Establish the appropriate location to perform inspection, based on pipe accessibility;
- Provide access to the buried piping inspection location (external or internal);
- Local removal of external coatings (for buried pipes only);
- Perform direct inspections (UT thickness measurements);
- Condition assessment;
- Establishing compensatory measures.

All mentioned activities were performed using internal, qualified plant resources.

A summary of initial underground piping inspections and the associated condition assessment results for Cernavoda NPP is presented in Table 4.2.

The Unit 1 fuel oil EPS buried tanks (2 pcs, 25 cbm each) and the fuel oil SDG buried tanks (4 pcs, 250 cbm each) were also inspected to determine their as-found condition. For each tank, the inspection activities consisted in:

- emptying the reservoir by removing the existing fuel oil
- chemical cleaning and ventilation of tank internal
- UT scan of the tank bottom and approx. 500 mm of the tank circumferential plate

The examination results showed a very good condition of EPS and SDG buried tanks, with only a few spotted areas with wall thinning that have been repair by welding.

b) Planned inspections

Based on the risk-ranking results obtained in 2016 with BPWORKS software for the buried piping segments in Unit 1 and Unit 2, multi-annual inspection plans have been developed, in order to establish the inspection scope, inspection type (direct/indirect) and the appropriate inspection time for the next 3 years of existing contract (2017-2019). These planned inspections include piping segments from all three groups, as they were previously defined in this report.

Monitoring

Monitoring is performed for the Cathodic Protection (CP) systems associated with plant EPS/SDG, by specific activities consisting in CP potential and current measurements, followed by the interpretation of results. These monitoring activities are executed twice a year (6 months frequency), during winter and summer periods.

The applicable acceptance criterion used at Cernavoda NPP is "- 850 mV ON potential", as per NACE SP0169 standard. To be noted that all CP systems in the plant are galvanic type (with sacrificial anodes) and there is no current interrupter or "decoupler" provided by design, to allow measuring the "instant OFF" potential, as per the practice in the industry. The CP measurement results have been recently evaluated by an external, NACE CP3 qualified expert. The CP evaluation report concluded that cathodic protection systems related to the EPS and SDG Fuel oil piping are working, but improvements are necessary, in order to revise the existing Maintenance Procedures used for periodic measurements. On the long term, the NACE expert recommended changing the sacrificial type Cathodic Protection existing at Cernavoda NPP with impressed type Cathodic protection. The required actions shall be included in the BP Asset Management Plan.

By design, at Cernavoda NPP there are no coupons installed which could allow the on-line monitoring and trending of the corrosion process for the plant buried piping. The necessity to install these coupons shall be assessed after planned inspections are completed and recommendations shall be included in the BP Asset Management Plan.

Table 4.2 – Summary of initial (as-found) inspections and associated condition assessment results for Cernavoda NPP underground piping

1. System name	System BSI code	Pipe dia	Pipe material (per ASTM or STAS)	Fluid	Filling material	Main activities performed	Condition Assessment results (for inspected area)
Group 1 - Piping systems that transfer fuel oil for emergency power generation							
Fuel Oil System (for SDG - Standby Diesel Generators)	52320	1” 1 ½” 2”	Carbon steel A53 Gr.B	Fuel oil	Engineering fill	Digging for access to pipe Coating inspection + local removal Direct inspections (UTT) Indirect inspections (CP measurements) – every 6 moths	Coal-tar external coating adherent to pipes. No pipes indication at UT thickness. CP not working properly (depleted Mg anodes)
Main Fuel Oil Distribution (for EPS - Emergency Power Supply)	52920	1” 1 ½” 2”	Carbon steel A53 Gr.B	Fuel oil	Engineering fill	Direct inspections (UTT) Coating inspection Indirect inspections (CP measurements) – every 6 months	Coal-tar external coating adherent to pipes. No pipes indication at UT thickness. CP not working properly (degraded electrical connections, partially depleted Mg anodes)
Group 2 - Piping systems that provide cooling to SSCs important to safety							
Emergency Core Cooling (ECC)	34320	16” 2”	Carbon steel A106 Gr.B	Demi water	Engineering fill	Digging for access to pipes Coating inspection + local removal Direct inspections (UTT) – full scan Local coating renewal	Coal-tar external coating adherent to pipes. No pipes indication at UT thickness.
Emergency Water Supply (EWS)	34610	10”	Carbon steel A 53 Gr.B	Raw (river) water	Engineering fill	Digging for access to pipes Coating inspection + local removal Direct inspections (UTT) – full scan	Coal-tar external coating adherent to pipes. No pipes indication at UT thickness.

						Local coating renewal	
Condenser Cooling Water (CCW)	71210	144“ 114” 16”	Carbon steel OL 42.2K	Raw (river) water	Partially concrete embedded	Periodic internal mechanical cleaning of the large bore pipes (during Outages) Direct inspections (UTT) – spot checks	No degradation identified. UT spot checks found 144” pipe thickness close to nominal value.
Raw Service Water (RSW)	71310	60” 48” 14”	Carbon steel OL 42.2K	Raw (river) water	Partially concrete embedded	Periodic internal mechanical cleaning of the large bore pipes (during Outages) Periodic chemical treatment (with biocide) Direct inspections (UTT) – spot checks	UT spot checks found 144” pipe thickness close to nominal value. No relevant degradation identified.
RSW Backup Cooling (BCWS)	71690	16”	Carbon steel A106 Gr.B	Raw (river) water	Partially concrete embedded	Direct inspections (UTT) – full scan of pipe accessible inside valve pits (3 locations)	Indication found for local wall thinning (pitting).
Group 3 - Piping systems that contain radioactive effluents;							
Liquid Handling system	79210	3”	Carbon steel A53 Gr.B	Low rad. waste water	Concrete embedded	Direct inspections (UTT) of pipe accessible inside valve pits	General corrosion of external surface for the pipe accessible area not embedded in concrete (pipe located inside CV1 valve pit).

04.1.4A Preventive and remedial actions for concealed pipework

For metallic buried and underground piping, Cernavoda NPP applies a preventative maintenance strategy, consisting in:

- periodic cathodic protection verifications;
- visual examination (in accesible areas);
- periodic UT thickness measurements (in accesiblle areas).

If the risk of failure is identified to be unacceptably high, immediate measures are taken through Contingency Plans, to repair the affected area and to mitigate the causes of degradation. Up to the present moment, no unacceptably degradation has been identified on the inspected underground piping and tank in the plant.

At Cernavoda NPP, the run-or-repair decision of the underground piping is based on the Level 1 uniform loss assessment, which is the most conservative approach. The corroded pipe is presumed to be corroded uniformly down to the minimum measured thickness value ($t_{min\ meas}$) and an estimated corrosion rate (mm/year) is calculated, based on the pipe service life (in years). The remaining life (RL) is then calculated considering the measured minimum measured thickness ($t_{min\ meas}$), the minimum calculated thickness per ASME Code ($t_{nim\ calc}$) and the estimated corrosion rate (CR). The acceptance criteria used for buried piping inspection evaluation is that the measured wall thickness should exceed the calculated minimum thickness ($t_{min\ calc}$), as resulted from the construction code ASME B31.1 requirements. However, the Construction Code ASME B31.1 presently used for thickness evaluation does not contain rules for evaluating of buried piping. A specific standard is currently under development by ASME and shall be considered after its official approval.

For Group 1 of inspected EPS/SDG fuel oil piping, it was determined that buried piping is in appropriate condition, but the associated cathodic protection systems do not work as expected for Unit 1 EPS and SDG systems.

Remedial actions have been implemented as a plant project and consisted in:

- full replacement of the cathodic protection system components (anodes, wires, connection boxes) for EPS in Unit 1.
- replacement of affected connection boxes related to cathodic protection, for SDG in Unit 1

For Group 2 of inspected water cooling system piping, no major degradation was identified. Small leaks caused by pitting appeared in 2014 on a 16” Backup Cooling pipe located inside a valve pit and were promptly repaired by welding a metallic patch. To estimate the remaining life, the pipes are periodically monitored (by annual UT thickness) and the measured pipe wall thickness is compared with the calculated minimum wall thickness (as per construction code ASME B31.1 requirements for Class 3 non-nuclear piping).

For Group 3 of inspected low rad waste piping, general corrosion was identified on the 3” pipe external surface during inspection (inside the valve pit CV1, which is the only accessible area). No leaks were identified or expected. To ensure long time operation, all accessible pipe inside the CV1 pit was planned for replacement during the next Unit 1 outage, in 2018.

Description of the preventive and remedial actions to be taken for the buried piping in Unit 1 & Unit 2 of Cernavoda NPP shall be provided by the Asset Management Plan, as per the industry approach. As per existing contract, this document shall be elaborated by the Services

Provider in 2019, after all the planned inspections for underground pipes will be completed and the results will be assessed.

04.2A Licensee's experience of the application of AMPs for concealed pipework

The inspections and condition assessments which have been performed up to present demonstrated that the Unit 1 buried and underground pipes identified with the highest risk of degradation have a good condition of their external surface, mainly due to a very thick and adherent coal tar enamel coating applied at the time of their installation.

However, implementation of Cernavoda NPP concealed piping program is still in progress, with actions taken for more as-found inspections and condition assessment of these pipes from Unit 1 and Unit 2. Therefore, a final conclusion cannot be taken at this moment regarding the way that ageing mechanisms have acted over time on these pipes.

The BP Asset Management Plan (AMP), which shall be developed after completion of all underground piping and tanks planned inspections, will conclude about the plant concealed piping condition and the required repairs or replacement activities and will recommend possible changes in the way that BP program is implemented at Cernavoda NPP.

4.3A Regulator's assessment and conclusions on ageing management of concealed pipework for Cernavoda NPP

CNCAN performed a review of the Program Manual for the AMP of the BP of Cernavoda NPP and found is comprehensive and in line with the international good practice documents and guidance available on this topic, issued by EPRI, COG, INPO, NEI, ASME, etc. CNCAN also reviewed the operating experience, the design modifications/improvements implemented and inspection records for Cernavoda NPP BP. There were no significant ageing-related issues identified that could affect the safety-related BP.

Based on the experience so far, CNCAN concluded that the AMP implemented by Cernavoda NPP has been effective in preventing unexpected age-related degradations in safety-related concealed piping. More detailed regulatory reviews will continue when the LCMP (Life Cycle Management Plan) for BP will be issued.

04B CONCEALED PIPEWORK – TRIGA RR

04.1B Description of ageing management programs for concealed pipework

04.1.1B The scope of the program

The scope of the program is to ensure the monitoring, prediction and timely detection and mitigation of degradation of in concrete embedded pipes part of the primary heat transport system to ensure in all circumstances a minimum amount of water in the reactor pool to cover the core and to ensure the dissipation of residual heat.

04.1.2B Ageing assessment of concealed pipework

The embedded pipes of TRIGA14MW are made from aluminum alloy T6061, insulated with 3 layers of tar pitch coal to prevent the direct contact with concrete, with 800mm diameter for primary flow and 300mm for anti-siphon pipes. For this reason, those pipes are grouped in a single category of safety components.

The pipes are passive components embedded for life in steel bars reinforced concrete of the shielding of reactor, containing as well a 300m^3 pool for this reason the materials of embedded pipes and pool operate at quasiconstant ambient temperature with a difference of 7^0C, filled with high quality demineralized water in continuous circulation through the purification system. The tightness of embedded pipes is continuously surveyed by the level of pool water.

The inner surface of the embedded pipes was inspected with a special borescope 20m long. An air leack test was performed with no indications of defects on the inner surface of pipes or air leaks. At the same time, during reactor shutdown and reactor operation, these pipes contain low radioactive water contaminated with corrosion activated products.

A simple emergency cooling system is submerged in the pool, mounted on the pool bottom, to ensure the residual heat removal from the reactor core in the pool water.

The TRIGA ICN reactor is provided with pipelines embedded in the concrete structure of the reactor pool, which are part of the primary cooling circuit of the reactor. The secondary circuit is provided with pipes buried in the ground. There are no pipes located in covered channels.

The primary circuit of the reactor filled with demineralised water is a closed circuit that communicates with the atmosphere in the reactor hall through the surface of the pool water in the reactor.

The closed primary circuit located in the reactor building is made of stainless steel and an aluminum embedded part. Primary circuit components, pipes, valves, heat exchanger, delay tank and pumps were designed for a future reactor power of 28MW considering doubling the power in the future. This was the approach when developing the TRIGA SSR 14MW reactor design.

To prevent the consequences of external events (earthquake) followed by pipe breaks and loss of pool water where the cooling of the reactor core is no longer possible, the primary circuit design stipulates the inclusion of aluminum inlet and outlet ducts in the massive protection of concrete so that mechanical stress is evenly distributed and discharged in the concrete.

At the entrance and exit of the stainless steel ϕ800 pipes in the concrete protection a pair of ϕ 800mm valves is provided, of which one valve is manually operated and the second valve is pneumatically activated automatically when the pool water level drops below the limit of alarm, compared to the operating level. The pneumatic valves are actuated also by continous measurement of difference between the inlet flow and the output flow caused by a leakage of a pipe; should that happen, the pool water level would be reduced by pumping. Additionally, the primary pump from the primay circuit are shut down by the same signal for differential level.

To control a minimum cooling water level in the pool, one meter above the top of the reactor core, at the top of the each of the ϕ 800mm pipes embedded, there were provided two pipes of ϕ 300mm embedded also in concrete, equipped at the entrance in the pool with 2 floating valves that meet the requirements for redundancy and reliability being gravitationally triggered when the level of water in the pool drops (the decrease of water level 2.5 m below the standard level) for breaking the siphon.

The diagram of the piping embedded in the primary cooling and antisiphon circuit is presented in Fig. 4.5.

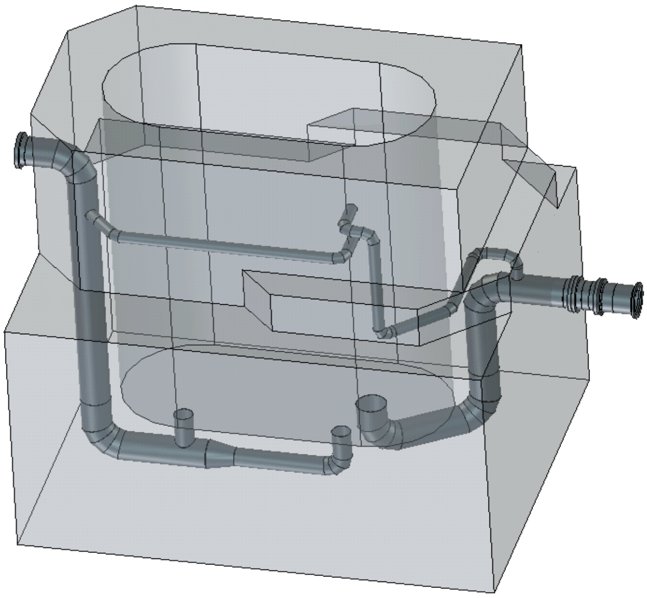


Figure 4.5

04.1.3B Monitoring, testing, sampling and inspection activities for the concealed pipework

The tightness of embedded pipes is continuously surveyed by monitoring the level of pool water. The inner surface of the embedded pipes was inspected with a special borescope 20m long. An air leack test was performed with no indications of defects on the inner surface of pipes or air leaks.

04.1.4B Preventive and remedial actions for concealed pipework

The ageing management for embedded pipes does not currently include activities for reparation, refurbishment or replacement as for many other pool type research reactors, where reparation of embedded pipes was possible with a special design for very special conditions.

04.2B Licensee's experience of the application of AMPs for concealed pipework

The ageing management for concealed pipework of TRIGA14MW reactor is a part of Ageing Management Program of the reactor and consists mainly of monitoring and inspection activities relevant for the integrity of the primary heat transport system.

04.3B Regulator's assessment and conclusions on ageing management of concealed pipework for TRIGA RR

Regulatory oversight focuses on the licensee's monitoring and inspection activities for ensuring the integrity of the reactor PHT system. No major performance issues have been identified in this area.

05 REACTOR PRESSURE VESSELS

This chapter is not applicable to Cernavoda NPP.

Although this chapter is not applicable to the TRIGA RR either, due to the fact that the reactor pool is not a pressure vessel, some information is provided on the ageing management of the reactor pool.

Ageing management of the TRIGA RR REACTOR POOL

05.1. Methods and criteria used for selecting components subject to AMP

The criteria which lead to the selection of the pool liner and reinforced concrete structure as components to be included in the scope of ageing management are that the pool and the associated components ensure the performance of the following safety functions:

- Contain the reactor core;
- Sustain the core cooling in all circumstances;
- Provide radiation shielding in all circumstances;
- May retain and delay the releases from the fuel clad failure.

The pool liner is a thin aluminum alloy T6061 welded structure, provided with external layers of tar pitch coal and embedded in a passive structure of reinforced concrete.

The external layers of tar pitch coal felt prevent the contact with concrete structure, ensure the thermal expansion of components and distribute all mechanical efforts to concrete structure.

05.2. The ageing mechanisms

The ageing mechanisms considered for the pool liner concern:

- The corrosion of aluminum alloy surface by water contained in pool;
- The embritlment of aluminum plates due to neutron irradiation;
- The mechanical damage by dropping heavy object inside the pool;
- The natural ageing of aluminum alloy in time.

To prevent corrosion of aluminum surface, the water quality is essential; the water parameters are subject of Operating Limits and Conditions. To avoid the stagnant water and continuous control of water chemistry the purification system containing mechanical filters and ion exchanger column, made of stainless steel are maintained in continuous operation.

05.3. The main structure and components that constitute the pressure boundary

The pressure boundary term for a pool type reactor may have the meaning of limits of controlled volume contained in reactor pool, pipes (embedded and conventional) installed in the lower level rooms of reactor, supported by structures which discharge the load to concrete structural elements.

The heat exchangers contain a controlled water volume, pumps, valves and associated pipes are also filled with water at atmospheric pressure, plus the water column pressure which reach the pool bottom.

The reactor pool limit is the pool liner described above where the water pressure, at the top of the open pool is the atmospheric pressure and at the bottom is of 9.4m water column (0.94 bar). At the entry in the core the pressure is some 7m water column (0.7bar). Water going through the reactor core reaches to the delay tank installed at -12m level in the reactor building structure. The delay tank volume is 110 cubic meters and allows the decay of N^{16}. The outlet of delay tank goes to a connecting tube to the inlet of the four primary pumps. With the exception of embedded pipes described above and pool liner which is Aluminum alloy made, all other components are manufactured from stainless steel AISI304.

The ageing of the primary circuit components using demineralized water at the maximum operating temperature 45^0C and design temperature 60^0C is very low.

Inner inspection of the surface and the welding denoted the absence of corrosion traces and deformation.

05.4. Description of reactor pool and the reactor core

The reactor pool is inside made of reinforced concrete block with 12.4*8.8*12m dimensions with a volume of concrete of $1267m^3$. The pool space is lined with welded aluminum plates and external reinforcement. The liner has several penetrations for connection with embedded pipes of primary cooling circuit and anti-siphon circuit. In the lower part there are 4 penetrations for radial and tangential beams tubes for neutron scattering experiments, having 200mm diameter. An underwater transfer channel 5.5m deep and 1m large provide connection with the Post Irradiation Examination Laboratory and with the pool for irradiated/spent fuel intermediate storage. The transfer channel is provided with a vertical closure port with compressed air rubber gasket.

The reactor pool liner was installed inside of previous casted pool shielding having a 1m space around the pool liner to allow the acces for final He leack test of welding and connections as well for X ray radiography and spot check as a final examination before embedment of the liner shielding concrete.

The next operations were the application of external anticorrosive layers from tar pitch coal felt and progressive filling of space with concrete, embedment. At the end of embedment operation, the initial inspection and testing activities were performed. Before filling with demineralized water, the late test for iron contamination traces of inside surface was performed using the feroxil procedure.

Two reactor cores are installed on the bottom of the previously described pool, consisting of the 14MW steady-state reactor and the Annular Core Pulsing Reactor (ACPR) following the standard design of TRIGA. The reactor core is located at opposite ends of the reactor pool as shown in Figure 5.1.

The steady-state 14MW reactor core is installed in a square grid array 11x12 grid positions. The active core is made up of 29 fuel rod clusters, 8 control rods, several in core experimental positions/vertical channels and is surrounded by a region of beryllium reflector blocks. In time the core configuration was modified in order to accommodate the irradiation facilities, in core capsule and loop for experimental fuel samples irradiation.

The fuel cluster is made of 25 fuel rods, each with a 0.342 inch outer diameter and 30inch long.

The Incoloy 800 cladding contains erbium-uranium-zirconium-hydride, fuel moderator meat. The fuel pins are supported and spaced within the cluster by aluminum bottom plate support and three Inconel spacers. The pin clusters and spacer assemblies are installed inside a 3.5 inch square tube, aluminum shroud with a lower aluminum fitting to be inserted in the reactor grid plate and a top fitting for fuel assembly handling.

The eight control rods are square annular assemblies containing sintered B_4C pellets in 20 rods each. The control rods are installed in the reactor core inside of square guide tubes to ensure safe insertion by gravity. The control rods are driven by standard TRIGA rack and pinion control rod drivers.

The reactor grid plate made from cast aluminum is assembled with an aluminum structure in the bottom of the pool connected with outlet pipe for primary cooling circuit. The cooling water circulation is downward to delay tank.

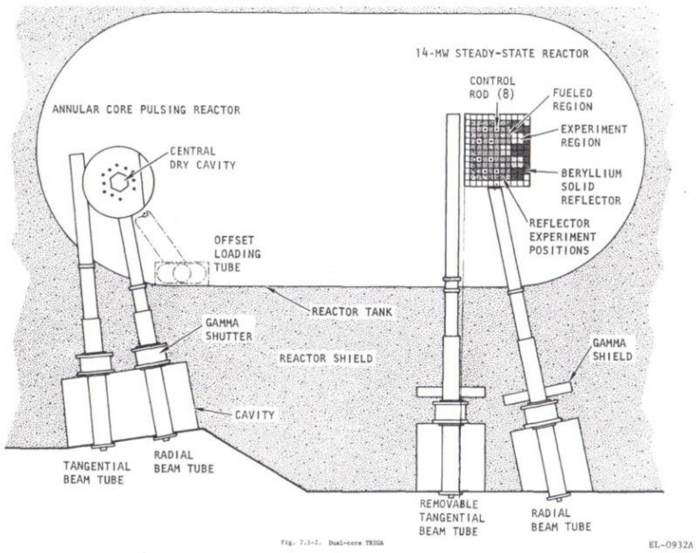


Figure 5.1. TRIGA dual-core

05.5. Reactor pool liner inspection

The quality of the pool liner inner surface was subject of visual inspection using the borescope and video camera. The surface of the liner walls and the bottom of the liner were inspected using the mentioned devices. Video images of the aluminum surface has not revealed indications of corrosion effects or damages of the liner.

The reactor core structure was inspected using the video camera, without any structural defects. During the reactor core structure inspection, the junction between the liner and the primary circuit pipes (welded junctions) was inspected, with no indications of defects on the seam weld.

A special device for controling of the integrity of the aluminum liner using US method was designed and now is under manufacturing. The inspections have started at the end of 2017.

06 CALANDRIA/PRESSURE TUBES (CANDU)

06.1 Description of ageing management programs for calandria tubes/pressure tubes

a) Description of Calandria Tubes / Pressure Tubes

The fuel channel used at Cernavoda NPP (illustrated in Fig. 6.1) consists of a Zr 2.5%Nb pressure tube (PT) centred in a Zr-2 (zircalloy) calandria tube (CT). The PT is roll-expanded into stainless steel end fittings at each end. CANDU6 reactors have 380 fuel channels and each fuel channel holds 12 fuel bundles. The reactor core length is approximately 6000 mm, and the PT is approximately 6150mm long (design dimension). Reactor coolant flows through adjacent fuel channels in opposite directions.

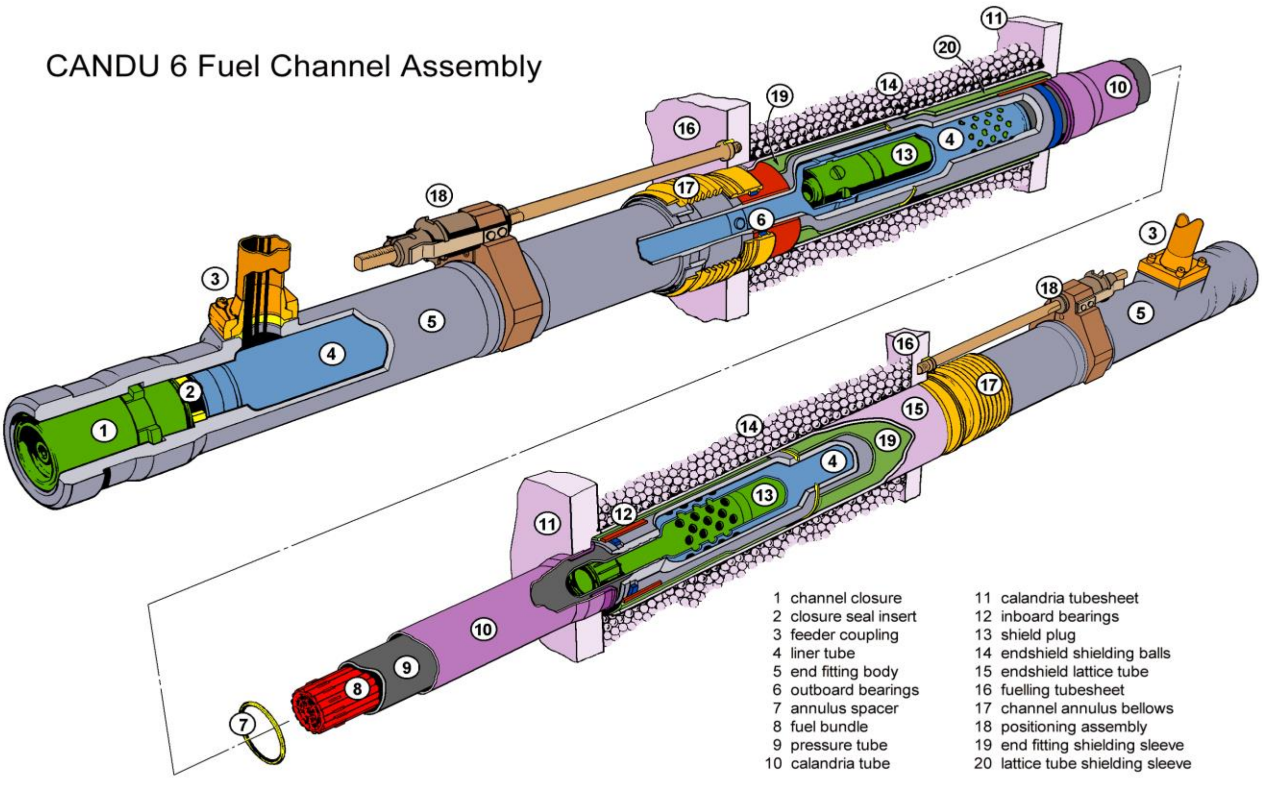


Fig. 6.1 – CANDU 6 Fuel Channel Assembly

Each pressure tube is thermally insulated from the low-temperature moderator by the annulus gas between the pressure tube (PT) and the calandria tube (CT). Fixed annulus spacers, positioned along the length of the PT, maintain an annular gap and prevent contact between the two tubes (details are presented in Fig. 6.1).

Each end fitting holds a liner tube, a fuel support plug (shield plug) and a channel closure plug. The end fittings provide connections to various interfacing components such as PHT feeder pipes, fuelling machines, positioning assemblies and channel annulus bellows. The outboard end of each end fitting is sealed by a channel closure when not connected to a fuelling machine (details are presented in Fig. 6.2).

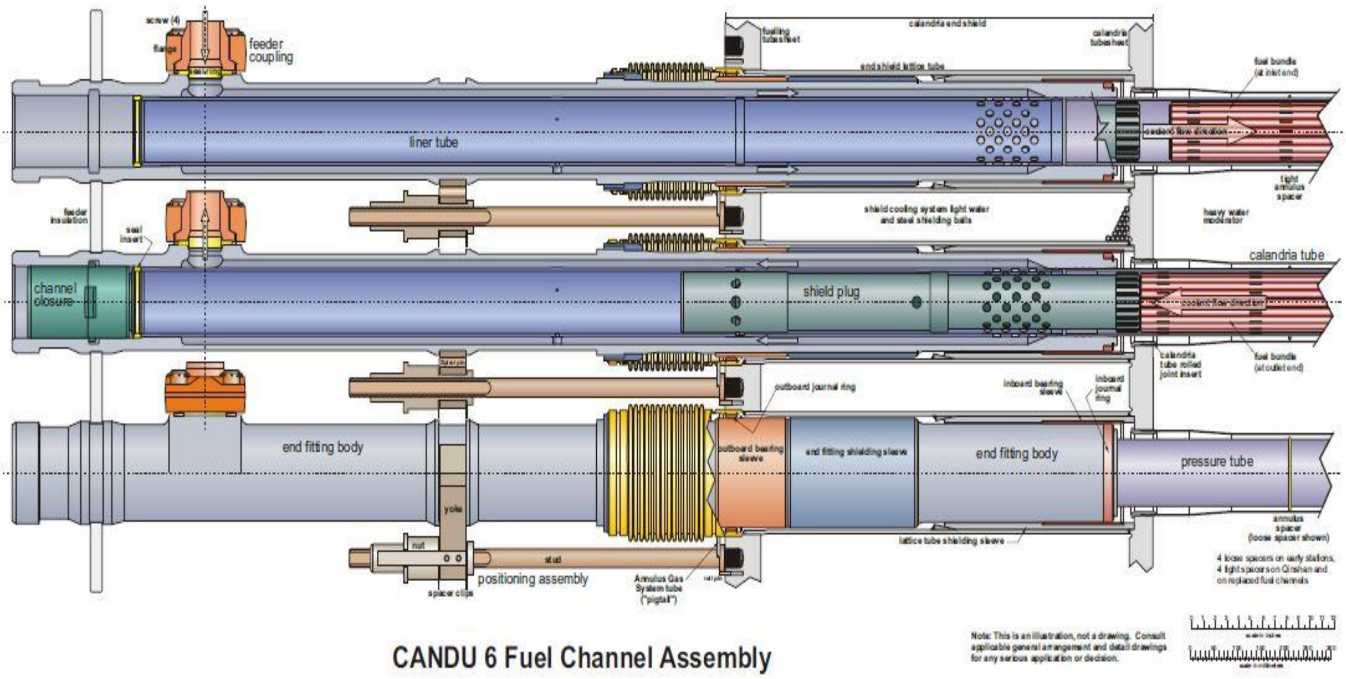


Fig. 6.2 – CANDU 6 Fuel Channel Assembly – details of the end fitting

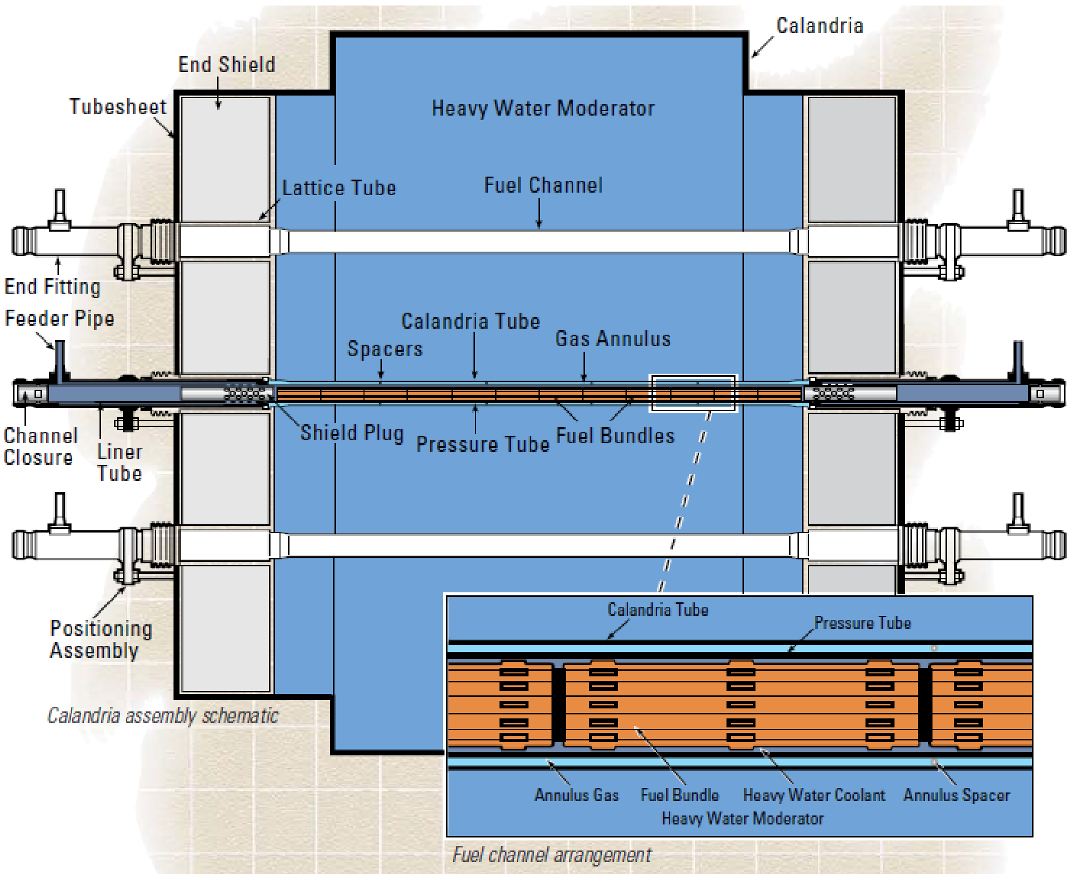


Fig. 6.3 – Calandria Vessel Assembly Schematic

The fuel channel is ‘fixed’ to the end shield at one end via a positioning assembly and is ‘free’ at the opposite end, thus allowing movements in the free direction due to thermal expansion, thermal and irradiation creep and well as irradiation growth. Each end fitting is supported in its end shield lattice tube on two sliding journal bearings. The positioning assembly device is installed at both ends to enable switching the fixed and free ends at about mid-life of the reactor. At the original installation, the fuel channel is ‘fixed’ at the reactor ‘C’ end, and ‘free’ at the opposite end. This arrangement allows the total axial creep elongation to be shared between the two ends of the channel.

The fuel channels are fuelled during normal reactor operation at power by two remotely-controlled fuelling machines which attach to the outboard ends of each channel during fuel changing to form leak-tight connections. One machine inserts new fuel into one end of the channel, and the other machine receives spent fuel from the opposite end of the channel. The outboard end face of each end fitting allows a sealed pressure boundary connection to be made with the fuelling machine during fuel insertion and removal.

The annulus spacer is a wire coil with 15 coils per inch and 4.826 mm in diameter, assembled around a circular girdle wire to form a torus. The wire at each end of the coil is formed into a hook. With the hooks joined together, the coil is held snugly around the pressure tube. The ends of the girdle wire overlap by about 180°. Each fuel channel assembly has four spacers, spaced about 1.02 m (40 in) apart and located symmetrically with respect to the calandria centreline. Besides maintaining a PT-to-CT gap while allowing radial expansion of the PT, the spacers also provide a rolling contact between the PT and CT to accommodate PT elongations and contractions.

The annular gap between the PT and CT is filled with a dry, continuously recirculating inert gas (CO_2 and a small amount of O_2), and is sealed at both ends by bellows. The annulus gas system permits the detection of leaking primary coolant from the pressure tubes or moderator leaking from the calandria tubes in the event that such leakage develops. The relatively dry gas also helps minimize corrosion of the calandria tube inside diameter, the pressure tube outside diameter and the annulus spacers. The gaseous atmosphere acts as an insulator, reducing the heat transfer from the primary coolant in the pressure tube to the moderator in the calandria.

b) Cernavoda NPP Pressure Tube (PT) Material

The PTs are made of Zr-2.5%Nb. The PTs made for Cernavoda Unit1 have numbers with a prefix M. The exception to this rule are 3 pressure tubes - P09, R18 and U17 - with the prefix H from Darlington material, and 5 pressure tubes – P16, R13, R14, R16 and T07 - with the prefix N from Cernavoda Unit 2 material. These H and N-series tubes were manufactured to the same technical specifications as the M-series tubes and are manufactured using the identical process for M-series manufacturing. All of the H and N-series tubes in Cernavoda Unit1 were extruded in 1982 except for the tube in lattice site R18 which was extruded in 1980. All pressure tubes in Cernavoda Unit 2 have numbers with prefix N. As such, the PT behaviour is identical between the two units.

c) Rolled joints

Like on all recent CANDU reactors, the fuel channels for the Cernavoda CANDU Units 1 and 2 use the Zero Clearance Rolled Joint design. This type of rolled joint makes use of an initial interference fit between the PT outside diameter and the end fitting inside diameter. The elimination of an initial pressure tube-to-end fitting radial clearance by use of an interference fit has been found to result in low residual stresses in the PT at the rolled joint.

Low residual stresses are essential to reduce the possibility of DHC (Delayed Hydride Cracking) in the PT.

d) Pressure Tube Orientation

PTs in both Cernavoda 1 and 2 were installed with their back end at the outlet side of the fuel channel. The back end is the end of the pressure tube that was extruded last.

e) Cernavoda Calandria Tube Material

CTs installed in Cernavoda Units 1 and 2 are made of Zr-2 (zircalloy) material. The ingots were made by Wah Chang, and the tubes were manufactured by Bristol Aerospace. The tubes have been annealed and stressed relieved.

f) Design Envelope for Pressure Tubes

The stress analysis report provides the operating envelope for dimensional changes due to PT creep and growth. As long as this operating enveloped is respected, the pressure tubes meet ASME code requirements.

Table 6.1 - Design Envelope for Pressure Tubes

Item	Value, mm
Maximum value of internal diameter	108.72
Minimum value of wall thickness	3.66
Minimum value of available bearing travel for creep on the fuel channel A-face	85.7
Minimum value of available bearing travel for creep on the fuel channel C-face	80.7

06.1.1 Scope of ageing management for calandria tubes / pressure tubes

Methods and criteria for selecting components in scope of the AMP

The fuel channels operational conditions (temperatures, pressures, high neutrons flux) cause dimensional, material properties and structural changes of the PTs by deterioration due to irradiation. The PTs are also subjected to corrosion caused by flowing of PHT heavy water coolant (slightly alkaline) inside the PTs, and to embrittlement phenomena caused by the fields of neutrons produced from fission reaction and from ingress of deuterium resulted following radiolysis reaction produced by heavy water flowing in the reactor core. Based on CANDU units operating experience and following technical documentation analysis, it was observed that there are several processes that could affect the fuel channels operation behaviour.

Part of the potential degradation mechanisms for fuel channels have been removed based on the results obtained from researches and tests, as follows:

- The hydrogen presence into the PT material has been reduced by using Zr-2.5%Nb alloy and manufacturing technology;
- In the high stress areas of the PT material, part of the contributing factors to the local hydrides accumulation process has been eliminated by:
    a) improving of the manufacturing process;

b) using of advanced techniques for the PT final inspection (ultrasonic);

- The improved manufacturing process and the detailed final inspections of the PT have conducted to removal of any pressure tubes defects appeared during manufacture process;
- Changes in the rolled joints design and the rolling procedures revision have conducted to reduction in, or even elimination of the tensile stress in these areas.

The degradation mechanisms with major impact on Cernavoda NPP fuel channels are those which can conduct to dimensional and structural changes that can influence the operational normal parameters of the system, or can result in FC unavailability, with major consequences on plant operation.

The main degradation mechanisms that affect the operational behaviour of the zirconium alloy pressure tubes are the following:

i) DHC (Delayed Hydride Cracking): cracking caused by the hydrides appearance (hydrogen and deuterium presence), and by the PT defects / flaws (from manufacture or appeared during operation).

DHC conducts to pressure tube cracking, appearance of heavy water leaks and necessity of reactor shutdown for PT replacement. It is expected that a “leak before break” (LBB) event, which could be detected by an increase of the humidity into the annulus gas system, is associated with a pressure tube failure by DHC. However, the PT rupture could occur if the heavy water leakage is not detected before the crack grows to an unstable length.

Currently, the deuterium ingress into the pressure tubes in CANDU reactors is monitored by removal of material samples from operating pressure tubes.

Data collected from the deuterium concentration measurements is necessary not only to give a fast warning if tubes have a significantly higher ingress than anticipated, but also to evaluate the fitness for service of any pressure tube flaws that may appear during operation.

Also, it was observed that the deuterium ingress rate is higher in the rolled joint region of the pressure tube compared to the others areas (the body of the pressure tube), because of additional H/D ingress galvanic reactions in the crevice between the pressure tube and the end fitting, that results in additional H/D ingress into the pressure tube ends.

Other important aspect that has to be monitored is the fuel channel spacers position that ensures a minimum clearance between the PT and the CT. Following tests and operating experience, it resulted that a PT could sag until it contacts the surrounding CT, because of the garter springs removed from their designed locations, and as results, a significant temperature gradient will occur through the thin wall of the PT. In these conditions, at the contact location brittle hydride “blisters” can develop and accelerate a crack extending on the PT length, such that “leak before break” may not occur.

Based on operating experience it was observed that the “tight-fit” garter springs used at Cernavoda NPP stay on the installation position, but this has to be confirmed during the periodic inspections.

ii) Irradiation enhanced deformation: elongation, sag, diametral expansion, wall thinning

a) Axial creep

The PTs from CANDU reactors elongate during operation due to the operational temperature and pressure of the heat transport system coolant (heavy water), and the fast neutrons flux from the reactor core.

The elongation is an important parameter taken into consideration in the design of the fuel channel components (bearings, bellows, positioning assemblies), and also in the design of the interfacing systems and components (feeders, fuelling machine).

Based on the studies performed until now it has been observed that the fuel channel axial creep is a linear function in time, with a nominal linear rate of approximately 5 mm per year. Such, it has been predicted that the fuel channels elongation in 30 years is approximately 153 mm, each channel being allowed to elongate an average of 76.2 mm ± 10% at each end.

Measurements accumulated in time from different operational plants have shown that there is a considerable variation between the elongation rates of the different pressure tubes (the elongation rate is directly proportional with the fast neutron flux and stress in pressure tube). Such, the monitoring of these rates for all fuel channels is required. This monitoring is necessary to establish the adequate period for the fuel channels re-adjustment (by fixing the positioning assembly at the channel free end and freeing the channel at the fixed end), and as data used to verify any possible contacts between feeders, even if the elongation changes of the pressure tubes are taken into consideration in the reactors design.

The fuel channels elongation monitoring can be used for an early determination of any corrective actions that have to be taken into consideration if the fuel channel elongation with a high rate of elongation becomes a problem that could limit the operational life period of the fuel channels. Even if such kind of situations occur, there are solutions that could be taken into consideration, such as defuelling the fastest elongation pressure tubes, and then adequately limiting the flow in these empty channels.

b) Sag

The PT sag occurs because of the irradiation during reactor operation. To avoid the contact between the PT and surrounding CT, four spacers are designed and installed between these tubes to ensure that the PT will not contact the CT for the fuel channel designed life. During operation, the PTs sag on the portions between spacers.

In addition to the PT sagging because of the irradiation from the reactor core, the entire fuel channel assembly may sag. Although this sag is not an issue regarding refuelling during operation (by passing the fuel bundles through the channel), in the Calandria vessel there are horizontal pipes that are part of the liquid poison injection system that adds poison into moderator system, such that some Calandria tubes may sag to contact these reactivity mechanisms. Although this situation is not expected to occur during the operating lifetime of the plant, monitoring of the gap between the CT and these reactivity mechanisms is required, to ensure that a contact does not occur sooner than expected. In case of occurence of such kind of situation, remedial actions can be initiated, which would likely involve moving of the mechanisms down to increase the gap between them and CT.

c) Diametric expansion

Diametric creep results in external diameter increasing and wall thinning of the PT and is considered a major aspect in design of the garter springs.

The diametric expansion of the PT allows an increasing amount of the primary coolant to flow around fuel bundles, which could lead to a slightly reducing of the channel power at constant flow. Although this issue can be compensated by an overall increase in flow, and the redistribution of the flow from the lower power channels to the higher power channels, it will eventually results in an unacceptable fuel cooling.

By design, an increase of the diametric expansion of the PT is allowed, this being limited to 5% of the initial PT diameter. This limitation is provided to avoid a creep rupture of the PT, and to ensure that the garter springs are not squeezed between the PT and CT.

Speciality studies showed that the increasing rate of the PT diameter is approximately 0.1 mm per year. The maximum diametric creep of the PT for 30 years is estimated to be approximately 4.1% and occurs in an area which is approximately 75% of the length from the inlet end.

iii) Changes in pressure tubes material properties

Based on tests performed and data collected from the operating reactors, it has been observed that the presence of the PT in radiation fields increases tube hardness, yield and tensile strength, and reduces ductility and fracture toughness. The consequences of such changes are that the PTs become more susceptible to fracture, that means the limits associated with demonstrating their LBB (“leak before break”) behaviour are decreased, and hence, the possibility of tube rupture is increased.

In order to maintain the PTs in operation, assessments are required to demonstrate that these tubes continue to show high confidence and assurance that DHC will not occur, and also that if DHC occurs the result will be detection of a leak that could allow the reactor to safely shutdown before this crack becomes unstable and to lead to a break.

To ensure that LBB requirement is satisfied, there must be a high level of confidence that the time necessary for a crack to grow to an unstable length is less than the time available to initiate adequately actions after leak detection.

From the assessment of the showed data results that the PTs from the fuel channels are subjected to some degradation mechanisms, which could conduct to:

- Cracking or even singular rupture of a PT;
- Reaching of the admissible limits provided by design for the entire PT population (internal diameter, length or deuterium/hydrogen content into tube material).

Taking into consideration the impact of the above mentioned issues, it results that a life management program for fuel channels fitness for service is required.

Processes for the identification of ageing mechanisms

CANDU fuel channels are subject to a number of degradations mechanisms. The fuel channel degradation mechanisms can be divided into a few sub-categories.

1. Degradation mechanisms related to dimensional changes resulting from thermal and irradiation creep and irradiation growth:
    a) Axial elongation
    b) Diametric expansion
    c) Wall thinning

d) Sag of pressure tube

e) Sag of calandria tube

2. Degradation mechanisms related to dimensional changes:
   a) Nip-Up
   b) Calandria tube to LISS (Liquid Injection Shutdown System) nozzle contact
   c) Pressure tube to calandria tube contact and blister formation
3. Other degradation mechanisms:
   a) Material loss from corrosion of the inside surface of a pressure tube and oxidation of the outside surface
   b) Deuterium uptake from corrosion and ingress at the rolled joints
   c) Changes in mechanical properties due to irradiation
   d) Service induced damage (i.e., fuelling scratches, crevice corrosion marks, debris fret marks, etc.) which could be the starting point for DHC or fatigue.

The causal factors and contributors to the degradation mechanisms, as well as the methods used to mitigate them are compiled in table 6.2.

Table 6.2.

Component	Degradation mechanism	Cause	Contributors	Monitoring or mitigation methods
Pressure Tube	DHC (Delayed Hydride Cracking)	Local Hydride Accumulation (Large tensile stress concentrations)	Pressure Tube manufacturing process Commisioning Plant operating conditions	Improved manufacturing process Pressure tube inspections during commisioning Avoid transient conditions
Pressure Tube	DHC (Delayed Hydride Cracking)	Manufacturing flaw	Pressure Tube manufacturing process	Improved manufacturing process Improved inspections during manufacturing
Pressure Tube	DHC (Delayed Hydride Cracking)	Fueling scratches	Refueling	Monitor flaw indications during periodical inspections
Pressure Tube	DHC (Delayed Hydride Cracking)	Pressure tube fretting flaw from contact with fuel bundles	Particularities of the refueling system. Fretting with the fuel bundle’s skates	Modifications applied to the fuel handling system.
Pressure Tube	DHC (Delayed Hydride Cracking)	Corrosion Cracking	Concentration of LiOH in crevices between the fuel bundle and the pressure tube	Monitor flaw indications during periodical inspections

Component	Degradation mechanism	Cause	Contributors	Monitoring or mitigation methods
Pressure Tube	DHC (Delayed Hydride Cracking)	Foreign material fretting	Impurities in the Heat Transport System	Foreign Material Exclusion (FME) procedures during all stages of construction, commisioning and operation. Monitor flaw indications
Pressure Tube	DHC (Delayed Hydride Cracking)	Flow of D_2O cooling agent through the pressure tubes	Improper chemistry control	Monitor deuterium ingression Maintain chemistry control Develop empiric models for deuterium concentration
Pressure Tube	Iradiation-induced deformation	Sag of the pressure tube or fuel channel assembly	Reactor operating parameters Shifting of the spacers between pressure tubes and calandria tubes	Monitoring of pressure tube sag and elongation through periodic inspections Monitoring of spacers position through periodic inspection Monitor the distance between fuel channels and Liquid Injection Shutdown System (LISS) lines Shift LISS lines or horizontal flux detector in order to increase the distance to the calandria tube
Pressure Tube	Iradiation-induced deformation	Elongation	Reactor operating parameters	Monitor pressure tube elongation during refuelling Reconfiguration of fuel channels fixed side.
Pressure Tube	Iradiation-induced deformation	Diametral Expansion	Reactor operating parameters reactorului	Monitor through periodic inspections
Pressure Tube	Iradiation-induced deformation	Pressure tube wall thinning	Reactor operating parameters	Monitor through periodic inspections
Pressure Tube	Changes in material properties	Iradiation	Reactor operating parameters	Monitor through extracted material samples
Spacers	Chamges in material properties	Iradiation	Reactor operating parameters	Monitoring and condition assessment

Component	Degradation mechanism	Cause	Contributors	Monitoring or mitigation methods
Rolled Joint	DHC (Delayed Hydride Cracking)	High rate of deuterium ingression	Galvanic corrosion in the rolled joint	Addition of O_2 in the Annulus Gas System Monitor and screen for any flaws through periodic inspections
Rolled Joint	DHC (Delayed Hydride Cracking)	High remanent stress	Improper rolled joint execution during installation	Update procedures governing rolled joint execution Ensure proper positioning of rolling equipment
End Fitting	Scratches and surface oxidation	Fuel Handling maneuvers or unrelated work-tasks in the area	Halogen contamination	Control of all halogens in the Reactor Building
Positioning Assembly	Mechanical failure	Stuck assembly pin		Repairs or replacement
Calandria Tube	Iradiation-induced deformation Changes in material properties	Iradiation	Reactor operating parameters	Replacement

The PTs installed in Cernavoda NPP were manufactured using tight controls on the process. It has already been observed that better control of the manufacturing process has led to less variance in dimensional changes in later generation CANDU6 reactors when compared to the first generation CANDU6 reactors (Gentilly 2, Point-Lepreau, Wolsong 1 and Embalse). The reduction in variance should also apply to deuterium ingress, but it must be kept in mind that deuterium ingress is affected by other factors such as water chemistry.

Following industry practice, Cernavoda NPP Fuel Channels (FC) PLiM program focuses on the pressure tube degradation mechanisms defined in Section 12 of the standard CSA N285.4-09. These degradation mechanisms must be monitored and managed in order to demonstrate the structural integrity of the PHTS during the entire life of the reactor. CTs are, strictly speaking, not fuel channel components, but are often treated as such because of their interaction with the PTs.

Assessments of the other fuel channel components such as the end fittings, bellows and CTs are traditionally not included in a PLiM program, as there are no known active degradation mechanisms affecting the structural integrity of these components. These components are therefore not considered as life-limiting to the operation of the reactor.

06.1.2 Ageing assessment of calandria tubes/pressure tubes

Fuel channel inspections according to CAN/CSA-N285.4-94 requirements using Ultrasound (US) & Eddy Current (EC) detectors were performed in 1999, 2003, 2008, 2010, 2013, 2014.

These inspections were performed with Atomic Energy of Canada Limited (AECL) using AFCIS (Advanced Fuel Channel Inspection System) - the latest technology developed for fuel channel non-destructive examination.

A 10 years Framework Agreement was signed in 2012 with the CANDU original designer (Candu Energy Inc. which took over AECL) and the Life Assessment (LA) study was completed in 2015, and it includes Unit 1 inspection results obtained from 1999 until 2014.

Any external or internal operating experience is subjected to an assessment in order to determine if there are any potentially new ageing mechanism concerned, and to assess the applicability of the respective operating experience reports for Cernavoda NPP.

Additionally, the licensee for Cernavoda NPP is a full member in the Candu Owners Group Fuel Channels Working Group and has participated in the Candu Owners Group Fuel Channels Research and Development program between 2006 and 2015. All assessments were executed using the latest information, models and procedures developed by the industry and provided through the Working Group, R&D program and service providers that participate in them.

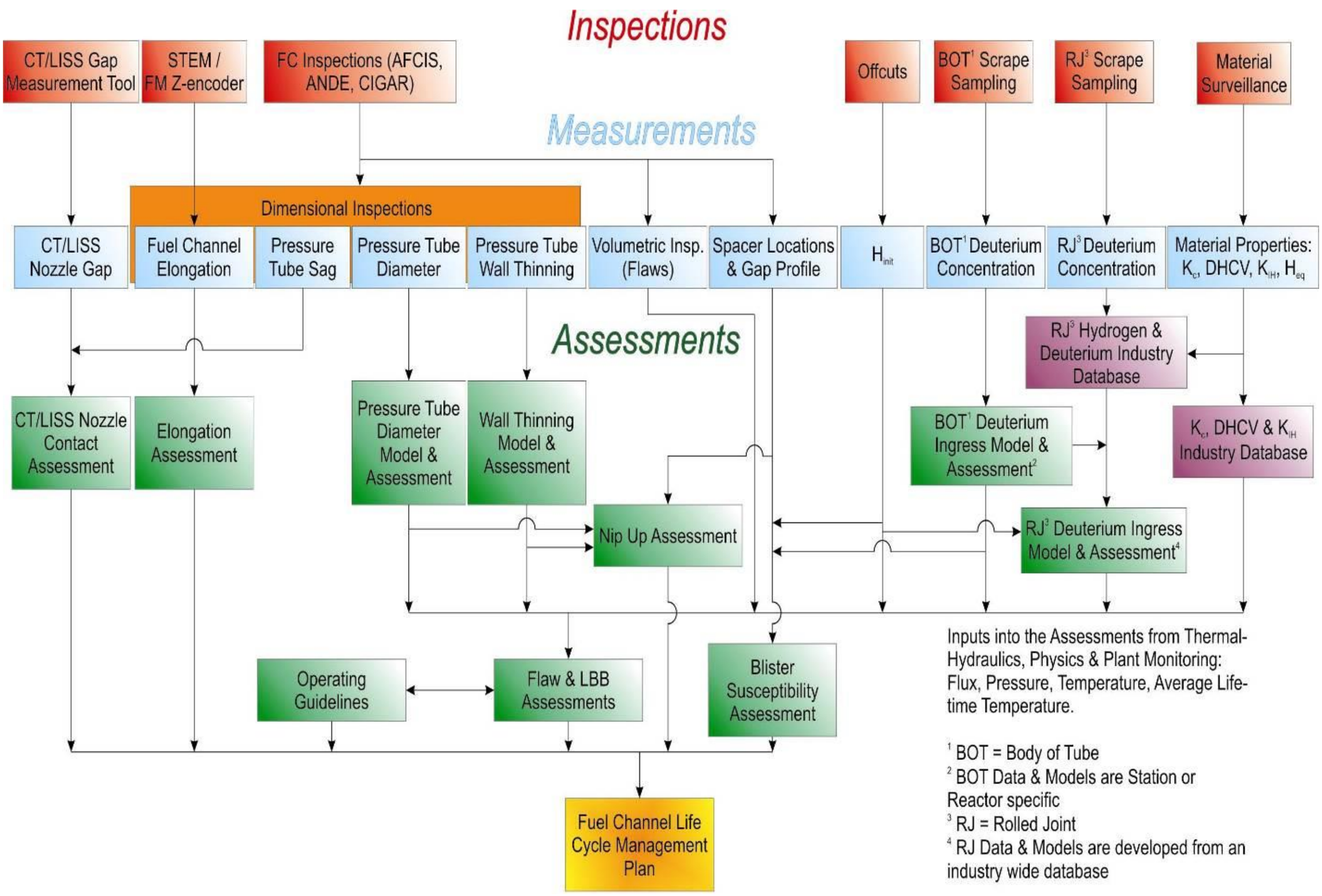


Fig. 6.4

Previously, inspections and assessments were performed in accordance with the standard CSA N285.4-94 and the Fitness-for-Service Guidelines (FFSG). Industry is currently migrating to CSA N285.4-09 and CSA N285.8-10.

Cernavoda NPP is currently using N285.4-94 and is planning to switch to N285.4-09 in 2018. With regards to FFS evaluations (flaws), Cernavoda NPP is already using N285.8-10 (the latest standard at the time of the last Fuel Channel Inspection campaign in 2014), and will switch to the latest version (2015 edition or later) for the Fuel Channel Inspections in 2019 and 2020, which is consistent with the industry practice of using the latest available standard.

Acceptance criteria are derived from the IAEA-TECDOC-1037 - Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: CANDU Pressure Tubes, as well as from the applicable standards.

6.1.3 Monitoring, testing, sampling and inspection activities for calandria tubes/pressure tubes

Periodic Inspection Program (PIP)

The PT periodic inspection must be performed on the full length of the selected PTs including the rolled joint regions and consists of two parts: the volumetric inspection and the dimensional inspection. The requirements of the volumetric inspection are to scan the PT to detect flaws on the pressure tube inner and outer surfaces, as well as internal flaws (laminar flaws). The dimensional inspection includes measurements of internal diameter, wall thickness, sag, elongation, PT to CT gap, and spacer locations.

According to CSA N285.4-94 Clauses 12.3.2.2 and 12.3.2.3, baseline inspection sample size must be a minimum of 12 tubes of the lead unit (C1) and a minimum of 10 tubes for the second unit (C2). The periodic inspection sample size must be a minimum of 5 pressure tubes for the lead unit (C1) and a minimum of 4 pressure tubes for the second unit (C2) as per Clauses 12.4.2.2 and 12.4.2.3 respectively.

According to CSA N285.4-94 Clause 12.8.3.1, the power plant lead unit is defined as the unit having pressure tubes with the most EFPH of operation of all units. However, any unit within 7000 EFPH of the lead unit may be substituted.

According to CSA N285.4-94 Clause 12.3.1, the baseline inspection shall be performed with a 2-year period commencing after 7000 EFPH of operation. According to Clauses 12.4.3.1 and 12.4.3.2, the first PIP inspection in the power plant lead unit (C1) shall be performed within a 3-year period commencing 4 years after the generation of the first net power. Subsequent inspections shall be planned for time intervals that do not exceed 6 years or 1/5 of the component design life, whichever is less. These inspections shall be performed over the last half of the inspection interval. For the subsequent units in the plant (C2), the first PIP inspection shall be made within a 3-year period commencing 4 years after the generation of the first net power. Subsequent PIP inspections shall be planned for time intervals that do not exceed 10 years or 1/3 or the component design life, whichever is less. These inspections shall be performed over the last half of the inspection interval.

According to CSA N285.4-09 Clauses 12.2.2.2 and 12.2.2.4, inspection sample size must be a minimum of 15 tubes for the baseline inspection and minimum of 10 tubes for all subsequent periodic inspections. A minimum of 5 of the 10 tubes selected for each inspection interval shall be a repeat of a periodic inspection from the previous inspection interval

(Clause 12.2.2.4 (b)). The inspection intervals are specified in Clause 12.2.3.1 of CSA N285.4-09.

The periodic inspections must be completed within the specified inspection interval. Results are reported to the regulatory authority within 90 days after completion of the periodic inspection program (CSA N285.4-94 Clause 11.3.2 and CSA N285.4-09 Clause 11.3.2.2).

According to CSA N285.4-94, the baseline hydrogen measurement for Unit 2 should be performed 9-10 years after the net power date. As such, the scrapping (sample prelevation) campaign is scheduled during the next Fuel Channel Periodic Inspection Campaign (2019), since the same equipment is needed for both. The sample size is scoped at 10 PT, which complies with both the 1994 edition of the standard and the 2009 edition.

Material Property Testing

Requirements

According to Clause 12.8.2.4 from the 1994 edition of CSA N285.4, material surveillance shall be performed on 1 PT within a 2-year period commencing 12 years after first net power. Subsequently, material surveillance shall be performed on 1 PT at intervals not exceeding 3 years.

The material property testing of pressure tubes removed from operating reactors must meet the following requirements:

1. The material property testing must include measurements of:
    a) Fracture toughness (K_c)
    b) Delayed hydride crack growth rate (velocity) (DHCV)
    c) Threshold isothermal stress intensity factor (K_{IH}) for onset of delayed hydride crack (DHC) initiation from a crack
2. There must be a sufficient number of measurements to establish the variation in material properties along the length of the PT.
3. A sufficient number of measurements must be taken to establish the [H_{eq}] profile along the length of the PT, including the rolled-joint regions.

Schedule

According to CSA N285.4-94 Clause 12.8.2.1, the CANDU lead unit is defined as the unit having pressure tubes with the greatest fast neutron fluence. However, any unit in the plant with pressure tubes having fast neutron fluence within 5% of the CANDU lead unit may be substituted. Note that the CANDU lead unit definition is different from the power plant lead unit used for FCI and [H_{eq}] measurements.

Until the unit’s shutdown for refurbishment, Wolsong 2 was the CANDU lead unit in regards to Cernavoda’s material properties program. Before Wolsong 2, the CANDU lead units were Bruce Power’s Unit 3 (Bruce 3) and Point Lepreau.

Using the information from surveillance pressure tubes extracted from the CANDU lead units, SNN was able to deferr the extraction of a pressure tube while operating under CSA N285.4-94.

According to CSA N285.4-09 Clauses 12.4.2.2.2 and 12.4.3.1, in a multi-unit station, one tube must be removed from the lead unit (C1) for material property testing within each interval. The power plant lead unit is defined as the unit having pressure tubes with the most

EFPH (Effective Full Power Hours) of operation of all units in the power plant. The concept of CANDU lead unit is no longer available for dispositioning of pressure tube extraction. Once Cernavoda NPP switches to a later revision of the standard, a tube extraction or a Integrated Material Surveillance Program will be necessary to comply to Clauses 12.4.2.2.2. and 12.4.3.1.

Integrated Material Surveillance Program (IMSP)

According to CSA N285.4-09 Clause 12.4.2.2.3, the sample size may be reduced when justified by an integrated material surveillance program that satisfies the requirements of Clause 12.4.1.2 and is accepted by the regulatory authority.

The material testing for the surveillance program is managed through a COG Joint Project under the direction of the COG Fuel Channel Working Group. Currently, the organizations participating in the COG Joint Project are: Ontario Power Generation, Bruce Power, Hydro-Québec, New Brunswick Power, SNN, Atomic Energy of Canada Limited and the CANDU Owners Group.

Recently, OPG and Bruce Power have implemented an IMSP in order to use 2 tubes recently removed from Embalse as part of their own surveillance program and avoid pulling 1 tube each in their own reactors.

SNN / Cernavoda NPP has initiated the process to join the IMSP with OPG and Bruce Power, and a report was prepared that demonstrate that the reactors in SNN are sufficiently similar to satisfy the requirements of an IMSP, and can use the results from the inspection of the 2 pressure tubes removed from Embalse.

The results of examination of Embalse pressure tubes Q05 and Q07 will be used as substitution for the intervals corresponding to the approximately equivalent amount of EFPH at the time of the removal of the PT in Embalse (i.e. up to 202000 EFPH).

In-Service Monitoring

Some of the factors that influence the degradation speed of fuel channel components are:

- The number of Effective Full Power Hours (affects pressure tube elongation, diametral expansion and material property changes);
- Heavy water chemistry might affect deuterium concentration (althought experimental studies show that the effect is minor or non-existent) and corrosion processes;
- Temperature transients in the PHT can lead to evolution of flaws.

As such, these parameters are monitored at all times in order to evaluate the pressure tube and fuel channel condition.

Aditionally, the following parameters are monitored on-line, after each refuelling of a fuel channel:

- Fuelling Handling Machine Z-index, providing accurate data on the pressure tube hot elongation;
- Pressure-drop measurements for each channel, which provides some indication on the evolution of the internal diameter.

Calandria Tube – Liquid Injection Shutdown System Gap Measurements

In all CANDU reactors, fuel channels and liquid injection nozzles (LINs) sag over time due to axial creep and growth under the effect of loading by their weight. The fuel channels sag more than the LINs, and as a result, the calandria tube (CT), may close the gap and come into contact with the LIN below it.

In order to investigate the possibility of CT/LIN contact, CT/LIN gap reduction assessments were prepared utilizing data from fuel channel and LIN measurements. The gaps between CTs and LINs were measured during an outage of Unit 1 in June of 2016 at 155190 EFPH. The present assessment used the same methodology followed in other CT/LIN gap assessments performed for other CANDU units.

The purpose of this assessment is to predict the contact times between CTs and LINs, and to ensure that no CT/LIN contact is predicted prior to the recommended next inspection interval to satisfy the acceptance criteria based on CSA N285.4.

The main result and conclusion of this assessment is that the predicted contact times between CTs and LINs for Cernavoda Unit 1 show the earliest contact time of 264 kEFPH at location Q11/LIN-5. Since Unit 1 will enter refurbishment before 247,000 EFPH, contact will not occur during the operational life of Unit 1.

Hydrogen Equivalent Concentration Determination

A sufficient number of measurements must be taken to establish the [Heq] profile along the length of the selected PTs. Measurements need not be taken in the rolled joint region. Determination of the rolled joint [Heq] profile may be done by calculation and using data from PT surveillance.

According to CSA N285.4-94 Clause 12.8.3, [Heq] only needs to be measured in the power plant lead unit (C1). There is no requirement to measure [Heq] in C2. However, with the adoption of the 2009 edition of the standard, [Heq] measurements for C2 will be performed during the planned outage in 2019.

Initial Hydrogen [Hinit]

Chemical analysis of pressure tube archive samples was performed to determine the initial hydrogen concentration of the Cernavoda ingots in Units 1 and 2. Normally, the measurements were taken at the top and the bottom of the ingot. For Cernavoda 1, as of June 2008, [Hinit] values were measured on the off-cuts of 25 pressure tubes and ranged from 3.93 to 10.98 ppm.

Body of the Tube (BOT) Deuterium Ingress

Scrape sampling is a process in which small curls of material referred to as micro-samples or scrapes are removed from the inside surface of a pressure tube and analysed for deuterium. SNN performed scrape sampling in C1 on 6 tubes in 2008 and on 2 tubes in 2014.

6.1.4 Preventive and remedial actions for calandria tubes/pressure tubes

Preventive and remedial actions for calandria and pressure tubes are described in table 6.2. Preventive actions are comprised mostly of careful monitoring of the component condition, either on-line or through periodic inspections, as well as efforts to ensure stable chemistry control and operating parameters.

For most defects that cannot be dispositioned according to CAN CSA N285.8, or which compromise safe operation the remedial action is Single Fuel Channel Replacement (SFCR), operation that consists of replacing either an entire fuel channel assembly (including calandria tube) or a partial replacement of pressure tube, annulus spacers and end-fitting.

De-fuelling of fuel channels at risk of developing flaws is also considered, if such a flaw is postulated. Some minor mechanical defects can be repaired or mitigated, after an assessment on a case-by-case basis.

6.2 Licensee's experience of the application of AMPs for calandria tubes/pressure tubes

Fuel Channel inspections have been performed at Cernavoda NPP periodically, in accordance with the applicable standards, using Ultrasound (US) & Eddy Current (EC) detectors. The inspections were performed using AFCIS (Advanced Fuel Channel Inspection System) - the latest technology developed by Atomic Energy of Canada Limited (AECL) for fuel channel non-destructive examination.

A 10 years Framework Agreement was negotiated with Candu Energy Inc. to perform the mandatory inspections, in accordance with the regulations and standards, for the period 2013-2022 and the Life Assessment (LA) studies (2015, 2020). The first LA study confirmed the actual good condition of Unit1 fuel channels.

The AMP for Fuel Channels in CANDU/PHWR involves a comprehensive set of activities including in-service or periodic inspections, maintenance, surveillance, post-irradiation examination, engineering assessments for fitness-for-service and residual life estimation, research and development in the fields of design, manufacture, operation, in-service inspection and life extension.

However, applying this comprehensive knowledge base to an existing power plant like Cernavoda NPP requires careful assessments of the recommendations and guidelines included in the AMP. The small variability between different units and operators means that a blind implementation of an AMP program can compound the risk of a specific issue being ignored from an over-reliance on the acceptance criteria provided by the AMP.

At Cernavoda NPP each requirement and suggested action was assessed in order to determine if it is applicable for the Cernavoda NPP Units, both from a technical and regulatory perspective. As a result of AMP, the periodic inspection program (PIP) was streamlined in order to both cover all susceptive equipments and components and ensure that vital resources are not wasted on characterising degradation mechanisms that are unlikely to be encountered. However, those unlikely events are still screened through measures that ensure that, if encountered, they can be mitigated before they can develop into major flaws or present a safety risk. As an example, CT-LISS measurements confirmed the previous evaluation that a contact is unlikely to develop at Cernavoda Unit 1, and will allow an accurate assessment of the same condition for Cernavoda Unit 2. Fuel Channel Annulus Spacers is another field where resources are focused on specific issues, adapted to Cernavoda NPP's particularities (priority being on spacer integrity, and not spacer movement which while screened for is not postulated to occur during the life of either Unit 1 or 2).

All monitoring guidelines were implemented in Cernavoda NPP Fuel Channel Program and the long term strategy follows the development of AMP in order to align to an accurate snapshot of Fuel Channel condition, as well as future issues that can develop. The process is an iterative one, involving regular updates to both the AMP and the Fuel Channel program, in order to implement the latest development in the industry, as well as research and development through national research institutes, COG and various other entities.

6.3 Regulator's assessment and conclusions on ageing management of calandri-a tubes/pressure tubes

CNCAN regulatory oversight programs for Cernavoda NPP have always included, as one of the most important areas, the review and inspection of licensee's activities for monitoring and maintaining the integrity of the PHT system in accordance with the mandatory regulations and standards that are part of the licensing basis for the plant.

As part of the regulatory oversight of the AMP for fuel channels, CNCAN reviewed the results of the fuel channels inspections (after each inspection campaign) and the LA study and Life Cycle Management Plan for Unit 1 fuel channels.

Also, as part of the regulatory oversight of the updated Final Safety Analysis Report, CNCAN is verifying periodically how the licensee updates the safety analyses to take account of the effects of ageing.

Based on the reviews performed so far, CNCAN concluded that the Cernavoda NPP AMP for fuel channels is comprehensive and in line with the good practices at international level, taking account of the latest available R&D results and operating experience.

No unexpected degradation of the fuel channels due to ageing has been identified so far in Cernavoda NPP. The regulatory reviews will continue for verifying the effectiveness of the AMP for fuel channels on the longer term.

07A CONCRETE CONTAINMENT STRUCTURES – Cernavoda NPP

07.1A Description of ageing management programs for concrete structures

07.1.1A Scope of ageing management for concrete structures

The Containment Structure (the exterior shell of Reactor Building (RB)) housing the nuclear reactor and process equipment) was designed as a fully pre-stressed reinforced concrete structure, and was isolated from the Internal Structure except at the base slab.

The Containment Structure essentially consists of a vertical cylinder (perimeter wall) with a ring beam on top, capped with a hemispherical dome, and a concrete base slab, all of them made of pre-stressed reinforced concrete.

The foundation is a pre-stressed reinforced concrete slab, 1.68m thick, approximately 49.08m in diameter. The base slab is pre-stressed with tree layers of horizontal pre-stressing tendons, on top side and bottom side. The layers of pre-stressing tendons form a 120° angle in between, each layer having 21 tendons. On the base slab rest the cylinder (perimeter wall) and internal structures.

The perimeter wall (cylinder) has the internal diameter of 41.45m, is 1.07m thick, 42.29m height and is pre-stressed with 124 vertical tendons and 147 circular tendons.

The ring beam is 1.905m thick and 4.267m in height, pre-stressed with 8 layers of tendons, is located on the top of perimeter wall. The ring beam makes the connection between perimeter wall and upper and lower dome.

The upper hemispherical dome has the interior radius of 41.45m and is 0.62m thick; it is pre-stressed with three layers of tendons form 120° angle in between, each layer having 47 tendons.

The lower hemispherical dome has the interior radius of 41.45m and is 0.38m thick; it is made of reinforced concrete, located 2.44m below the upper dome; it is not part of containment boundary.

The space contained between the upper dome and the lower dome serves as an emergency dousing water tank.

A sliding membrane was provided between the sub-base and base slab to allow radial movement of the base slab during post-tensioning and due to shrinkage. Also, shear keys was provided between sub-base and base slab to prevent any relative movement other than radial. A structural reinforced concrete hinge was provided at the base of the perimeter wall where it meets the base slab to reduce the moment in the wall end and in the base slab.

The Containment Structure has the following structural roles:

- To house the reactor and the auxiliary systems and protect them from the environmental conditions and severe events occurring outside the reactor building.
- To provide radiological shielding in order to limit the radiation exposure of the personnel and public during normal operation and accident conditions.
- To withstand the pressure generated under the containment envelope during a loss of coolant accident and to ensure that the leakage rate of the containment envelope it is not greater than the allowable limit.

In order to keep the level of radiation under allowable limits for the protection of the personnel and public during normal operation and accident conditions, the containment structure was completely lined on the inside by means of an epoxy liner, except at selected locations where fiberglass epoxy liners are used.

The containment liner and penetrations have no structural role, but do fulfill a number of system roles:

- To form an identifiable, inspectable and testable membrane over the inside faces of the containment structure to prevent the uncontrolled release to the environment of fission products.
- To allow fluids, ventilation and electrical systems to penetrate the containment pressure boundary.
- To provide a plant access equipment and personnel airlocks, and a fuel transfer tube to allow equipment, personnel and fuel to pass through the reactor building shell.
- To prevent ingress of fuel storage pond water from the spent fuel bay.
- To provide features to facilitate periodic leak rate testing at full containment test pressure of the full containment liner and local leak rate testing of penetrations.

In order to obtain a more flexible coat which would improve leak tightness, over the above mentioned liner a NORMAC polyurethane liner was applied. All joints are prepared and sealed with Thiokol sealant and then either of the above mentioned liners applied over as required. Electrical and mechanical services pass through this liner via specially-designed containment penetrations.

Access for personnel and material was achieved through the Personnel Access Airlock and the Equipment Access Airlock. Fuel movements between the Reactor Building and the Spent Fuel Bay take place through a special penetration system.

RB Containment Structure Description

The RB containment is a pre-stressed concrete structure consisting of the base slab, the perimeter wall and the dome. Bonded pre-stressing system was used as principle reinforcement for concrete containment structure. Permanent protection of tendons against corrosion was provided by cement grout.

Base Slab, Sub-Base and Gallery

The base slab is a 1.68m thick pre-stressed concrete flat plate supporting the perimeter wall and internal structure. The base slab has post-tensioning layers. Each of these layers consist of three groups of parallel tendons offset 60° from each other. There are 21 cables in each direction generally spaced at 2.025m. The total number of tendons in the base slab is 126.

The base slab is supported on a sub-base slab that is made of reinforced concrete. A sliding membrane is provided under the base slab to facilitate radial deformation during pre-stressing of the base slab. A central shear key and a set of radial shear keys are provided in the base slab for transferring the horizontal and torsional forces due to environmental and accidental loads.

The sub-base has a circular channel (gallery) allowing access to the perimeter wall pre-stressing cable heads.

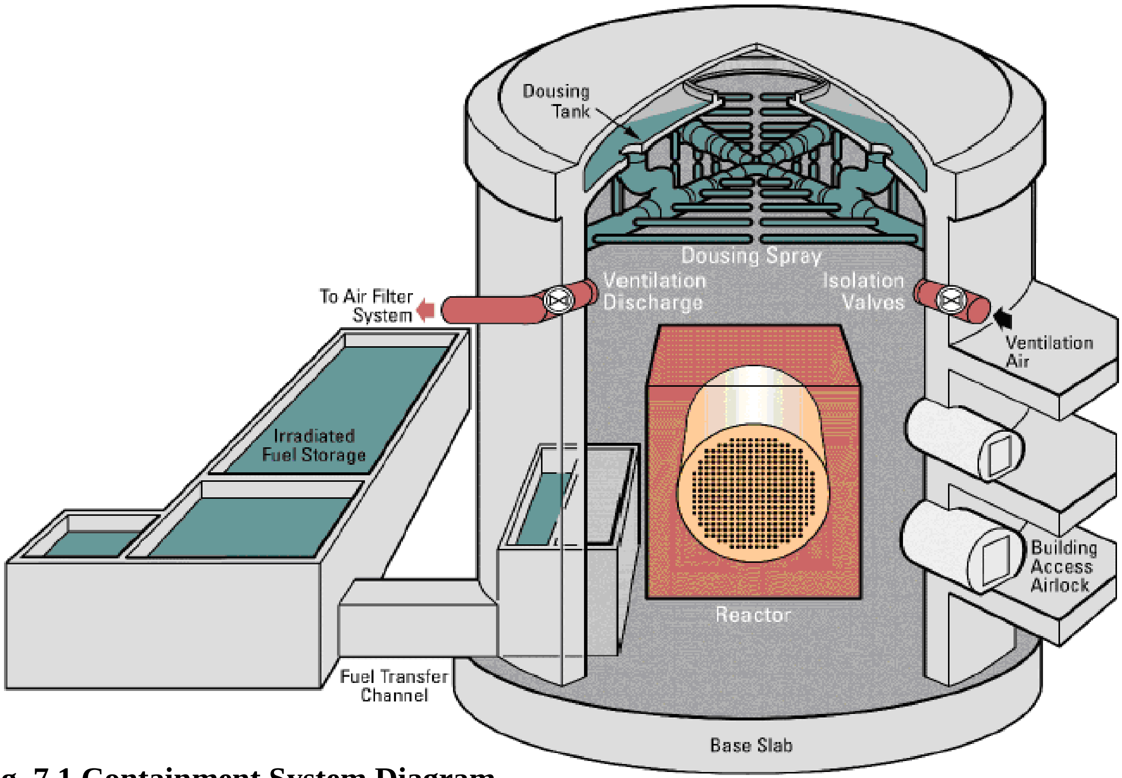


Fig. 7.1 Containment System Diagram

Perimeter Wall and Buttresses

The perimeter wall is a pre-stressed concrete cylinder, 1.07m thick, 41.46m high with internal radius of 20.73m. There are vertical and horizontal pre-stressing steel in the perimeter wall. The horizontal pre-stressing cables are laid out in rings consisting of 2 semicircles located 20cm from the outer surface. There are a total of 147 horizontal cables spaced at about 60cm. The vertical cables are located in the middle of the wall spaced at about 1 m. There are a total of 124 vertical cables. There are 4 buttresses at 90° from each other to house perimeter wall horizontal pre-stressing cable heads. Each buttress is 1.9m thick.

There are two permanent openings in the perimeter wall, for equipment and personnel airlocks. Two large temporary openings (7.92m² and 10.36m²) were used for access of equipment during the construction period. They were subsequently closed up during second stage concrete pour, which is connected to the perimeter wall by a system of horizontal and vertical pre-stressing cables and reinforcing steel connections.

There is a structural hinge provided at the bottom of the perimeter wall to relieve the bending stresses. The wall thickness is decreased at the joint between the perimeter wall and the base slab. The compressive force from the self-weight and pre-stressing, in combination with the sealant material and liner is supposed to minimize the leakage at the location of the hinge.

Ring Beam

The 1.90m wide ring beam constitutes a transition element between the upper dome and the perimeter wall. It houses the vertical perimeter wall pre-stressing cable heads, those of the dome pre-stressing and those of its own pre-stressing. In addition, the ring beam also provides support to the inner dome, which forms the bottom of the dousing tank.

The ring beam consists of two parts: the first concrete pour phase supports the inner dome and the perimeter wall, while the second pour phase supports the upper dome. The first pour phase of the ring beam is 1.36 m in height while the second phase is 2.91 m high. All the pre-stressing cables for the upper dome and the vertical cables of the perimeter wall are anchored in the second pour phase.

Each phase of the ring beam has 8 semicircular cables forming 4 rings.

The third concrete pour phase consisted of a 0.20m wide and 4.27m high concrete ring that is required to provide a protective cover for the anchor heads.

Upper Dome

The upper dome is a pre-stressed concrete spherical element with the internal radius of 41.45m and a thickness of 0.61m. The upper dome pre-stressing was made up by three groups of cables arranged so that they form spherical equilateral rings with the anchor heads located in the ring beam. Each layer has 47 tendons. The total number of tendons is 141.

Containment Liners and Joint Sealants

All the internal exposed surfaces of the containment/internal structure are covered / protected with epoxy liner systems that have various roles: enhancing leak-tightness, accommodating maintenance and ease of decontamination. The functions of the liners are as follows:

- The containment quality liner applied to the internal surfaces of the RB containment shall continue to function effectively for the lifetime of the plant and to ensure that the plant can meet the regulatory requirements for the allowable leakage rate and protect the public from excessive radiation exposure in case of postulated accidents. In addition, it has to meet the maintenance/decontamination requirements.
- The characteristics of the fiberglass reinforced epoxy liner must be similar to the containment liner. In addition, it enhances water tightness of the water retaining structures, i.e., Dousing Tank and Fuel Transfer Structure.

The characteristics of the decontamination liner applied to the RB internal structure have to satisfy decontamination/maintenance requirements. Its significance consists in allowing maintenance and easy decontamination of all the RB internal surfaces. No leak tightness enhancement is required.

A poly-sulphide compound (Thiokol) is used as a joint sealant in all joints to ensure leak tightness. At the hinge area (the wall to base slab connection), Thiokol is applied both at the inner and outer surfaces.

Containment Penetrations and Airlocks

The equipment airlock and the personnel airlock are accessible through the Service Building (SB). The four main steam pipes penetrate the RB containment structure perimeter wall. There are numerous other mechanical and electrical systems, which penetrate the containment.

07.1.2A Ageing assessment of concrete structures

Life Assessment Study

In 2015, a 10 years Framework Agreement was awarded to CEI (CANDU Energy Inc., which took over AECL), which includes the performance of Life Assessment (LA) studies for Units 1 and 2 containment structures. The LA study for Unit 1 has been finalized in 2017. In this LA, the aging mechanisms that can affect the performance of the RB structures were identified and the risks associated with the effects of aging degradations were considered. The LA study for Unit 2 will be finalized in 2018.

The methodology used to perform the Life Assessment (LA) study included a detailed review of design, manufacturing, construction, commissioning, operations, inspection, and maintenance data in order to assess the effect of aging degradation on the RB structures and their components, establish their current condition, and provide a prognosis for future performance (with associated recommendations). Consideration was given to applicable EPRI, IAEA, COG, CSA reference documents.

Ageing mechanisms requiring management and identification of their significance

Aging Related Degradation Mechanisms (ARDMs) that may adversely affect the properties of concrete structures under assessment are discussed in this section. As a structure ages, changes in material properties arise from continuing microstructural changes and environmental influences. As much as possible, these changes are accommodated in the design; however, some of the changes in material properties withtime cannot be accommodated in the design.

Mechanisms that may affect performance of nuclear concrete structures are presented in Fig. 7.2 and are briefly discussed in the following.

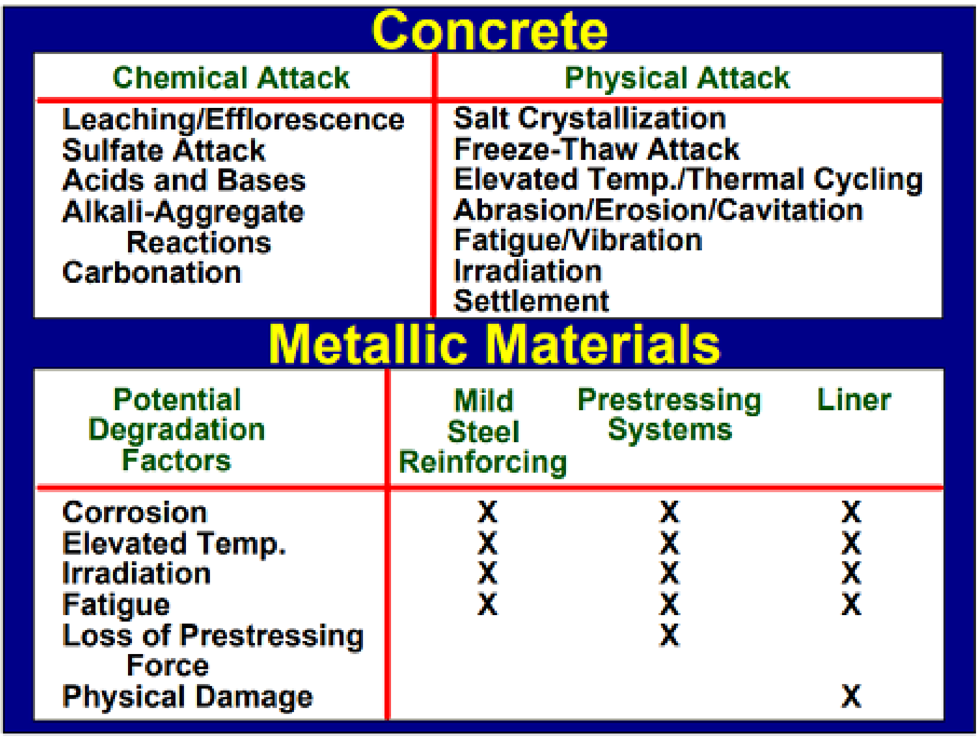


Fig. 7.2 - Mechanisms affecting the long term performance of containment concrete buildings (from IAEA-TECDOC-1503)

Sub-Base and Pre-stressing Gallery

The following ARDMs are considered for the sub-base and pre-stressing gallery:

- For reinforcing steel – corrosion
- For concrete - chemical attack including leaching, sulphate attack, acid attack, ASR (Alkali Silica Reaction) and salt crystallization.

The ground water elevation measured in the vicinity of the RB is about 9.16m below El. 100. The RB basement floor is at El. 93.90m and the pre-stressing gallery floor is at El. 89.18m. Thus, only the bottom part of the pre-stressing gallery seems to be occasionally exposed to the groundwater. The presence of moisture facilitates a variety of degradation mechanisms. A medium ranking (in terms of importance) was given to the leaching ARDM.

Sulphate attack, acid attack, ASR, salt crystallization and corrosion of reinforcement had not been detected at Cernavoda Unit 1, however they are considered applicable based on the material and environment. Sulphate and acid attack may occur locally; therefore it is not considered likely to affect the target life of the facility. Given that fact, sulphate and acid attack were ranked low.

Although the mortar bar test method (ASTM C227-71) was specified for the aggregates, this method was later identified to have some shortcomings and is not applicable for testing of all types of aggregates. Therefore, it is prudent to consider that if access to the moisture exists, the ASR may be initiated with time and cause expansion and cracking of concrete. ASR and corrosion of reinforcement have a tendency to become a widespread phenomenon in the high humidity environment and could potentially lead to a significant degradation. Thus, ASR and corrosion of reinforcement were ranked medium.

Base Slab

The following ARDMs are considered for the base slab:

- For concrete - elevated temperature exposure, creep and shrinkage;
- For pre-stressing system – loss of pre-stressing forces, corrosion; and
- For liner and joint sealant material – weathering, wear, elevated temperature exposure, irradiation.

Elevated temperatures inside the RB are associated with the relaxation of steel and volumetric changes of concrete including creep and shrinkage that may cause the pre-stressing force losses to be higher than those accounted for in design phase, therefore this ARDM is given a high ranking.

Shrinkage of concrete is usually considered an early age phenomenon and is, therefore, not typically considered an ARDM. However, massive RB structures might take a longer time to undergo shrinkage considering the typically very dry environment of the R/B.

Creep is referred to the time-dependent increase in strain and deformation under permanent stress. Creep and shrinkage are given medium ranking.

ARDMs related to the pre-stressing system were detected at other reactor buildings around the world and they might be applicable to Cernavoda Unit 1. There is a monitoring program in place for the pre-stressing system at Cernavoda 1.

Aging degradation of the concrete and joint sealant resulting in cracking, increased permeability, and volume changes and may also contribute to the loss of pre-stressing forces and corrosion of the pre-stressing steel. Since ARDMs associated with the pre-stressing system may potentially affect the life attainment, particularly considering long term operation, a high ranking is given. Material degradation of the pre-stressing tendons such as corrosion and stress corrosion cracking is also possible. Considering that consequences of corrosion for grouted tendons are not as severe as for un-grouted ones, these mechanisms are associated with medium ranking.

There is a program in place to detect and mitigate the degradation due to weathering. It is important that the maintenance is performed promptly to ensure that aggressive substances are not allowed to penetrate and deteriorate structures. Thus, a medium ranking is given. Joint sealant materials, liners, and waterstops are exposed to the elevated temperature of the RB.

Delamination appears to be a less important issue for the base slab liner (epoxy) as compared to perimeter walls that use an elastic liner (Normac). A low ranking is assigned.

Corrosion of embedded parts and rebar is considered as low ranking since it is a local phenomenon.

Perimeter Wall and Buttresses

The ARDMs considered for the perimeter wall and buttresses are similar to those considered for the base slab.

Gamma radiation values measured close to the perimeter wall result in a cumulative dose slightly below the threshold. Radiation exposure may cause deterioration of the internal surface of the concrete and facilitate ASR. It is unlikely that irradiation alone will significantly affect design life of the structure considering that the compressive strength of the concrete is likely more than 10% higher than the design strength. Thus, a low ranking is given.

Delamination of the elastic liner (Normac) is typically detected in association with the leak rate tests, with a medium ranking applied.

Upper Dome and Ring Beam

Similarly to the perimeter wall and the buttresses, exposure to elevated temperatures, shrinkage and creep, as well as ARDMs related to the pre-stressing steel, liner and joint sealant material are considered applicable to the upper dome and the ring beam and are given the same ranking. In addition to these, ARDMs i.e., leaching, ASR, freeze/thaw cycles, and corrosion associated with the dousing water and rainwater that may be collecting at some locations on the dome are considered applicable.

Fuel Transfer Structure (FTS)

The following ARDMs are considered for the fuel transfer structure:

- For concrete - chemical attack including leaching, acid attack and ASR, erosion and cavitation, elevated temperature exposure;
- For reinforcing steel, embedded parts – corrosion;
- For liner and joint sealant material – weathering, wear, delamination elevated temperature exposure, irradiation.

All of the above have been assigned a medium ranking.

Dousing Tank

The following ARDMs are considered for the dousing tank:

- For concrete - chemical attack including leaching and ASR, erosion and cavitation – medium ranking;
- For reinforcing steel and EPs – corrosion - ranked medium except for corrosion that is given a low ranking considering the pH of the water is 9;
- For liner and joint sealant material – weathering, wear, delamination elevated temperature exposure, irradiation – medium ranking.

Calandria Vault Structure (CVS)

The following ARDMs are considered for the CVS:

- For concrete – leaching, ASR, erosion and cavitation, acid attack, effects of elevated temperature exposure, fatigue;
- For reinforcing steel, steel liner and embedded parts – corrosion, fatigue
- For joint sealant and epoxy liner – exposure to elevated temperatures and radiation, weathering.

All of the above have been assigned medium and low rankings.

Effects of ARDMs

Effects of ARDMs considered applicable to RB structures were divided into three groups:

- Effects of ARDMs associated with exposure to moisture;
- Effects of ARDMs associated with exposure to elevated temperature/temperature gradient/temperature fluctuations and
- Effects of ARDMs associated with exposure to radiation.

Standards

The main standards and reference documents used in the AMP for the concrete containment structures of Cernavoda NPP include:

- CSA N285.5 - Periodic inspection of CANDU nuclear power plant containment components;
- CSA N287.1 - General requirements for concrete containment structures for nuclear power plants;
- CSA N287.2 - Material requirements for concrete containment structures for CANDU nuclear power plants;
- CSA N287.3 - Design requirements for concrete containment structures for nuclear power plants;
- CSA N287.4 - Construction, fabrication, and installation requirements for concrete containment structures for CANDU nuclear power plants;
- CSA N287.5 - Examination and testing requirements for concrete containment structures for nuclear power plants;
- CSA N287.6 - Pre-operational proof and leakage rate testing requirements for concrete containment structures for nuclear power plants;

- CSA N287.7 - In-service examination and testing requirements for concrete containment structures for CANDU nuclear power plants;
- IAEA NP-T-3.5, Aging Management of Concrete Structures in Nuclear Power Plants, 2016;
- COG-08-4042. Ageing Management Program for CANDU Concrete Containment Structures. May 2009;
- COG-10-4042. Concrete Stress and Strain Data Collection and Analysis for Ageing Management of Nuclear Containment Buildings. March 2011;
- COG-13-4036. State of the Art Report – Evaluating Integrity of Concrete Containment Structures with Grouted Post-Tensioning Systems. January 2014;
- EPRI#3002000462. Non-Destructive Evaluation: NDE for Tanks and Containment Liners. November 2013;
- EPRI#3002003027. Strain Development in Post-Tensioned Containment Concrete. December 2015;
- EPRI#3002005389. Tools for Early Detection of ASR in Concrete Structures. September 2015;
- ACI 349.3 Evaluation of Existing Nuclear Safety-Related Concrete Structures. 2002;
- PE 432.1 / 2002 – Norm on the monitoring of the behaviour in time of the constructions (national standard).

The establishment of acceptance criteria related to ageing mechanisms is done based on these standards and guidance documents.

Use of operating experience

A review of available operating experience has been performed as part of the AMP for the containment, as well as part of the LA study, using the relevant IAEA, COG, EPRI, NEA/OECD reports as sources of information.

Significant aging events related to concrete structures of NPPs have occurred and have been evaluated recently, e.g. delamination of concrete, ASR, larger than anticipated loss of the pre-stressing force, spent fuel pool leakage, corrosion of steel in water intake structures, etc.. These events led to a better understanding of the degradation mechanisms and necessitated new aging management actions.

In particular, the IAEA document NP-T-3.5 includes state of the art information regarding ageing management of concrete structures in NPPs. It contains background material on ageing related degradation mechanisms that is necessary to ensure understanding of ageing. It also includes new and practical information on current practices and techniques for inspection, monitoring and mitigation of ageing degradation.

Lessons learned from the operating experience are continuously used for the optimization of the AMP related activities for the containment structures of Cernavoda NPP.

Use of R&D

The information on latest relevant research and development activities was reviewed and used for the AMP and for the LA study. These included latest EPRI and COG reports related to the performance of various materials under conditions expected inside the RB structure, non-destructive evaluation and evaluation of the containment post-tensioning system.

For the most part, R&D documents reviewed are related to four major areas:

1. Activities necessary for effective aging management of nuclear structures;
2. Methods for evaluating integrity of containment prestressed using bonded posttensioning system;
3. Qualification of non-metallic repair materials to be used for repair of nuclear structures;
4. Non-destructive techniques for inspection of nuclear structures.

The connection with the nuclear industry operating experience and R&D is achieved through the collaboration of Cernavoda NPP through COG. Continuous information received from COG staff provides access to all debates on topical issues where Cernavoda NPP representatives directly intervene. In order to clarify the lesser known aspects of the influence of some factors on the degradation mechanisms, a joint research development program is being developed through COG. Cernavoda NPP has joined this program, thus providing access to the necessary information to improve the program based on the results of the research. Cernavoda NPP proposed to COG a series of themes related to the aging aspects of structures and finishes at CANDU plants or on Normac's film repair methods. Among the research reports developed by COG with impact on the structure and finishes of the Reactor Building are: “Assessment of Potential Testing Techniques for Difficult-to-Access CANDU Containment Concrete”, “Effective Repair Materials in Concrete Containment Curing Times of Repair Materials for Vacuum Building Outages”, “Long-Term Performance of Concrete and Repair Materials”.

07.1.3A Monitoring, testing, sampling and inspection activities for concrete structures

Since 2001, inspections are performed inside the RB in compliance with national regulations and standards, every outage. Monitoring of building behavior is achieved through direct visual examination. Some of the RB structures (e.g. pre-stressing gallery, CVS) are not included in the scope of in-service inspections.

Additional monitoring related to aging management of RB structures includes:

- Monitoring ground water quality (chemistry);
- Monitoring of settlement;
- Monitoring of RB deformations;
- Monitoring of RB pre-stressing system (Test Beam);
- Inspection of passive protection barriers against fire.

Surveillance walkdowns are performed in the RB by the maintenance staff periodically (annualy at a minimum). Also, inspections of the passive protection barriers against fire are performed.

Leak rate tests (LRTs) are performed periodically, in accordance with the mandatory regulations and standards included in the licensing basis.

The RB containment structure is monitored using a system of strain and temperature measurements. The system includes Vibrating Wire Strain Gauges (VWSG) embedded in the base slab, perimeter wall and dome of the RB.

The gauges are connected by cables embedded in concrete to a local junction box. Embedded in concrete conduit contains multi-conductor cables that connect the junction boxes to a switching panel via a central main junction box. A total number of the gauges is 118.

Test beams were cast at the time of construction of Cernavoda Unit1. The intent of the program was to evaluate the effectiveness of the cement grout as a corrosion protection medium and to evaluate the long term volume changes of concrete (creep and shrinkage) and pre-stressing steel (relaxation).

A number of 3 beams were to be cast with un-bonded tendons protected with petroleum and the pre-stressing force was to be measured by Dynamometers installed on these beams. All beams are to be constructed with 91(7mm) wires cables. The tests are specified to be performed at the time of Proof Pressure Test and subsequently 1, 2, 5, 10, 15, 20 and 25 years after.

The specified tests include:

- visual examination,
- stress relaxation measurements on the three specimens cast with un-bonded pre-stressing tendons,
- flexural test,
- destructive examination, 7 beams were cast to be destructively tested at 1, 2, 5,10,15,20 and 25 years.

A monitoring program of the vertical deformation measurements is in place. The measurements are collected and evaluated using Leica DNA03 digital level and 0.93m and 1.85m invar staffs. The measurements are collected semi-annually. The vertical displacements are shown as negative (signifying settlement) or positive (signifying swelling).

07.1.4A Preventive and remedial actions for concrete structures

Preventative Maintenance to Maintain RB Air Leak Tightness

Cernavoda NPP procedures specify the preventive maintenance of the RB containment structure to maintain containment leak tightness. These documents make reference to the standard CSA N287.7-08. It lists the following activities to be performed that should be implemented either during the LRT (leak rate test) or immediately after LRT:

1. Applying additional layers of the Normac liner on the perimeter wall, outside walls of Cernavoda U1 R-001 (FTS), around Main Steam Line penetrations, and temporary openings in case the leaks or deterioration of the liner is discovered.

2. Perform repairs of the Thiokol around embedded parts, cables, anchors, joints and R-001.

3. Perform leak searches outside the RB in accordance with a specific maintenance procedure. Where significant leakage is found during LRT, it is to be repaired using a two-component injection resin. Flexible material should be injected in the joints to allow for movement. Experienced personnel should be involved in the repairs.

4. Perform inspections in R-001 after LRT. If necessary, an injection of two-component resin and repairs of the Thiokol and the Normac liner are to be performed in accordance with the work plan after draining and decontamination are complete.

07.2A Licensee's experience of the application of AMPs for concrete structures

The technical basis and program manual was developed with external support (from AECL - Atomic Energy of Canada Limited) in 2009.

Although Cernavoda NPP is a station with a relatively short period of operation, the Reactor Building and civil structures are older, as they were built in early 1980s.

The components selected in scope of the program are: Containment, Dousing Tank, Reactor Vault/ Reactor Building, Fuel transfer channel.

There were no major events related to ageing of civil structures, the internal operating experience indicates that the main ageing issues related to buildings and structures at Cernavoda U1 up to date have been with the elastomeric components.

Inspection and test results showed that RB foundation settling is less than expected in Units 1 and 2. Also, the pre-stressed concrete structure is in good condition. However, the degradation mechanisms identified in the international technical literature for concrete are expected to apply to Cernavoda NPP Unit 1. Therefore, some improvements are necessary for monitoring and testing of civil structures.

In 2015, a 10 years Framework Agreement was awarded to CEI (CANDU Energy Inc., which took over AECL), which includes the performance of Life Assessment (LA) studies for Units 1 and 2 containment structures. The LA study for Unit 1 has been finalized in 2017. In this LA, the aging mechanisms that can affect the performance of the RB structures were identified and the risks associated with the effects of aging degradations were considered. Based on the recommendations of the LA study, the licensee is drafting an action plan, to be issued in 2018, for reinforcing the actual RB PLiM program.

The LA study for Unit 2 will be finalized in 2018.

07.3A Regulator's assessment and conclusions on ageing management of concrete structures for Cernavoda NPP

The regulatory oversight of the ageing management of the concrete containment structures of the Cernavoda NPP units has traditionally focused on the performance and results of the leak rate tests, which offer an overall view of the state of the containment building. The LRTs performed so far for Units 1 and 2 have not revealed any unexpected degradations in the containment performance. However, taking into account the more comprehensive approach involved by a proactive AMP, as required by the regulation NSN-17 issued in 2016, further reviews and inspections will focus on more detailed aspects of the AMP for concrete containment structures.

As part of the regulatory oversight of the AMP, CNCAN reviewed the Program Manual for the AMP of the containment structures and the RB LA study for Unit 1 and found them to be comprehensive and adequate in terms of evaluations performed and recommendations issued. Further reviews will be performed in 2018, when the licensee issues the action plan for the improvement of the AMP for Unit 1 RB, based on the recommendations of the LA study.

07B CONCRETE CONTAINMENT STRUCTURES – TRIGA RR

07.1B Description of ageing management programs for concrete structures

7.1.1B Scope of ageing management for concrete structures

The ageing assessment of concrete structures concern the selected structure which performs a safety function, constitutive materials selected for design and construction technologies of selected structures, considered the operation conditions in correlation with ageing mechanism.

The main concrete structures selected for the evaluation are:

- The reactor building which contains the structural elements with safety functions are reactor pool shielding, delay tank shielding, shielding of equipments of water purification system and shielding of liquid radioactive waste, tanks;
- The confinement and delay/control of releases in normal operation and accident conditions are performed by tight reactor hall;
- The releases of radioactive gases and aerosols are controlled by a high (60m tall) concrete chimney (stack);
- Cooling towers are referred for normal operation; residual heat after reactor shutdown is dissipated in the reactor pool water.

7.1.2B Ageing assessment of concrete structures

The nominal degradation mechanism of structural concrete depends on materials, construction technology and operational environment/stress.

a) Reactor pool shielding made from reinforced concrete, service conditions are:
    - ➢ Constant loads – static in normal operation;
    - ➢ Variable loads – variable/dynamic design (Design Basis Earthquake).

- Environment condition:
    - ➢ Constant or slow variation of temperature starting from ground temperature i.e. 12^0C till 37^0C maximum water temperature in pool;
    - ➢ Low temperature, freezing is prevented by design, below the ground level;
    - ➢ Dry and clean air atmosphere around pool shielding prevents the chemical attack from alkali, acid/base, elevated temperature/cycling concrete degradation;
    - ➢ Fatigue vibration are not considered as being degradation mechanism;
    - ➢ Irradiation is not significant because the distance from the reactor core to the structural concrete is 1.5m of water and 1m filling concrete outside of the pool liner to the reinforced concrete, leads to low level of integrated dose.

None of above presented degradation mechanisms has the potential to prevent the pool shielding to fulfill the role in normal operation. From the technical literature the doses from gamma irradiation were 10^9 rad degradation of mechanical property and this was not significant. The dynamic behavior of reactor shielding in case of earthquake design basis will not produce the deterioration of the pool.

b) Delay tank shielding; accommodate the delay tank with a volume of ~100m^3 water.

This shielding is part of the building structure forming a large room at 12m level. The conditions of operation of this concrete structure are approximately similar to those of the pool shielding, the same environmental conditions. The actual condition of those walls and ceiling are similar with the reactor pool shielding.

c) The reactor hall starts from level “+0” until +14m. The construction following a special design is made from two profile U shape poured in place, the other structure elements of reactor hall are fabricated beforehand and assembled on site forming a structure of pillars and beams filled with bricks. The internal surfaces are finished and painted to ensure some internal lower pressure relative to outside atmosphere. This building is continuously ventilated and the temperature of the construction remains between 10^0C - 22^0C. The external load changes of wind and snow do not produce visible degradations. The reparation is feasible with conventional means.

d) The reactor chimney (stack) is a tall concrete construction with a hight of 60m on a large and heavy basement. The exhausted air from the reactor building is released through the chimney in the atmosphere. The chimney (stack) is subject to annual climatic variation with summer maximum +40^0C and winter smallest minimum -20^0C. Wind pressure generates mechanical efforts in this structure compatible with design provision.

7.1.3B Monitoring, testing, sampling and inspection activities for con-crete structures; preventive and remedial actions for concrete structures

The periodic evaluation of reactor structure and building was performed by experts from design authority and qualified entities.

The last evaluation report concerning the chimney (stack) was performed in September 2017 by a licensed expert. The conclusion regards the conformity of the actual structure in relation with the design provision, the recommendations concerns the reparation in the lower part, base, of protective concrete layers and to remake the external protection paint.

The permanent responsibility concerning the inspection of the reactor reinforced concrete structure rests with the licensee. The supervision activity concerns the annual walk-down of a team composed by operating personnel and a licensed construction engineer. The report reveals the status of inspected areas and items, conclusions and corrective actions if any.

7.2 Licensee’s experience of the application of AMPs for concrete structures

The ageing assessment of the concrete structures is part of the overall AMP for TRIGA RR and consists primarily of period inspections, with maintenance and repair activities performed as necessary. No unexpected ageing-related degradations have been observed in safety-related concrete structures.

7.3 Regulator’s assessment and conclusions on ageing management of concrete structures

CNCAN performes periodic inspections at the TRIGA RR in order to verify the licensee’s activities for maintaining the integrity and functional performance of safety related SSCs. So far there have been no safety concerns identified with the AMP of the concrete structures of the TRIGA RR.

08 PRE-STRESSED CONCRETE PRESSURE VESSELS (AGR)

This chapter is not applicable to the nuclear installations in Romania.

09 OVERALL ASSESSMENT AND GENERAL CONCLUSIONS

The regulatory framework for AMP has been completed in 2016 with the issuance of the new regulation NSN-17 - Nuclear safety requirements on ageing management for nuclear installations. The regulation is based on the relevant WENRA reference levels and introduces requirements that are aimed at driving the licensees to implement systematic and proactive AMPs, going beyond the traditional approach limited to maintenance, in-service inspections, surveillance and testing.

The licensees are complying with the regulation of NSN-17 and have submitted to CNCAN the relevant documentation of the AMP to demonstrate compliance with the new regulatory requirements. Several improvement opportunities have resulted from the licensees' self-assesments and the regulatory reviews concerning the implementation of effective AMPs and these will be tracked by CNCAN through its regular oversight processes.

Both Cernavoda NPP and the TRIGA RR use relevant international standards and good practice / guidance documents, as well as R&D results and operating experience feedback for the development, implementation and optimization of their AMPs.

In addition to its regular oversight activities aimed at verifying compliance with the new regulation NSN-17 on ageing management, CNCAN performed reviews and inspections to verify the self-assessment reports prepared by the licensees in accordance with the WENRA / ENSREG specifications for the TPR.

Based on the regulatory reviews and inspections performed so far, CNCAN is satisfied with the adequacy of the licensees'AMPs and with their overall implementation. No major issues have been identified.

While no unexpected ageing-related degradations have occurred so far in the safety related SSCs of the nuclear installations of Cernavoda NPP and TRIGA RR, the effectiveness of the licensees' AMP can only be proven in the long-term. Constant examination is required to ensure that the operational limits and conditions continue to be met, including the preservation of adequate safety margins.

Starting with 2017, ageing management related topics are included more systematically in the regulatory inspection plans and the regulatory oversight activities for monitoring implementation of the licensees' AMPs will continue.

10 REFERENCES

1. NSN-17, Nuclear safety requirements on ageing management for nuclear installations, regulation issued by CNCAN in 2016;
2. Report, Topical Peer Review 2017, Ageing Management Technical Specification for the National Assessment Reports, RHWG Report to WENRA, 21 December 2016;
3. Ageing Management for Nuclear Power Plants, Safety Guide No. NS-G-2.12, IAEA Safety Standards Series, 2009;
4. Nuclear Power Plant Life Management Processes: Guidelines and Practices for Heavy Water Reactors IAEA TECDOC Series No. 1503, IAEA 2006;
5. Maintenance, Surveillance and In-service Inspection in Nuclear Power Plants, Safety Guide IAEA Safety Standards Series No. NS-G-2.6, 2002;
6. Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: CANDU Pressure Tubes, IAEA TECDOC Series No. 1037, IAEA 1998;
7. Ageing Management for Research Reactors, Specific Safety Guide, IAEA Safety Standards Series No. SSG-10, 2010;
8. Aging Management Guideline for Commercial Nuclear Power Plants - Electrical Cable and Terminations, SAND 96-0344, 1996;
9. Cable System Aging Management, EPRI 2002, Report 1003317
10. Medium Voltage Underground (MVU) Cable White Paper" NEI 06-05, April 2006;
11. IEEE Std 400-2001, IEEE Guide for Field Testing and Evaluation of the Insulation of Shielded Power Cable Systems;
12. Aging Identification and Assessment Checklist - Mechanical, EPRI report, August 2004;
13. Balance of Plant Corrosion – The Underground Piping and Tank Reference Guide, Revision 1. EPRI, Palo Alto, CA: November 2013. 300200682;
14. Cathodic Protection Application and Maintenance Guide Volume 1 and Volume 2. EPRI, Palo Alto, CA: July 2013. 3002000596;
15. Generic Aging Lessons Learned (GALL) Report – Final Report (NUREG-1801, Revision 2), XI.M41 Buried and Underground Piping and Tanks, December 2010;
16. Guideline for the Management of Underground Piping and Tank Integrity, Nuclear Energy Institute, NEI 09-14, Revision 3, April 2013;
17. INPO Chemistry Department – Evaluator How-To Underground Piping, Institute of Nuclear Power Operations, Revision 6, March 2011;
18. Inspection Methodologies for Buried Pipe and Tanks. EPRI, Palo Alto, CA: August 2010.1021561;
19. Nondestructive Evaluation: Buried Pipe NDE Reference Guide – Revision 2. EPRI, Palo Alto, CA: December 2012. 1025220;
20. Nondestructive Evaluation: Buried Pipe NDE Reference Guide – Revision 2 – Addendum EPRI, Palo Alto, CA: November 2013. 3002000447;
21. Nondestructive Evaluation: Inspection Methods for Tanks and Containment Liners. EPRI, Palo Alto, CA: December 2012. 1025215;

22. NRC Inspection Manual, Temporary Instruction 2515/182, Review of the Implementation of the Industry Initiative to Control Degradation of Underground Piping and Tanks, August 2013;
23. CAN/CSA N285.4 94 “Periodic inspection of CANDU nuclear power plant components”, December 1994;
24. CAN/CSA N285.4 05 “Periodic inspection of CANDU nuclear power plant components”, June 2005;
25. CAN/CSA N285.4 09 “Periodic inspection of CANDU nuclear power plant components.” June 2009;
26. CAN/CSA N285.4 14 “Periodic inspection of CANDU nuclear power plant components.” May 2014;
27. CAN/CSA N285.8-05 “Technical requirements for in service evaluation of zirconium alloy pressure tubes in CANDU reactors” , June 2005;
28. CAN/CSA N285.8-10 “Technical requirements for in service evaluation of zirconium alloy pressure tubes in CANDU reactors” , July 2010, Including Update No. 1 (2011) and Update No. 2 (2013);
29. CAN/CSA N285.8-15 “Technical requirements for in service evaluation of zirconium alloy pressure tubes in CANDU reactors.” Published in 2016;
30. CSA N285.5 - Periodic inspection of CANDU nuclear power plant containment components;
31. CSA N287.1 - General requirements for concrete containment structures for nuclear power plants;
32. CSA N287.2 - Material requirements for concrete containment structures for CANDU nuclear power plants;
33. CSA N287.3 - Design requirements for concrete containment structures for nuclear power plants;
34. CSA N287.4 - Construction, fabrication, and installation requirements for concrete containment structures for CANDU nuclear power plants;
35. CSA N287.5 - Examination and testing requirements for concrete containment structures for nuclear power plants;
36. CSA N287.6 - Pre-operational proof and leakage rate testing requirements for concrete containment structures for nuclear power plants;
37. CSA N287.7 - In-service examination and testing requirements for concrete containment structures for CANDU nuclear power plants;
38. IAEA NP-T-3.5, Aging Management of Concrete Structures in Nuclear Power Plants, 2016;
39. EPRI#3002000462. Non-Destructive Evaluation: NDE for Tanks and Containment Liners. November 2013;
40. EPRI#3002003027. Strain Development in Post-Tensioned Containment Concrete. December 2015;
41. EPRI#3002005389. Tools for Early Detection of ASR in Concrete Structures. September 2015;
42. ACI 349.3 Evaluation of Existing Nuclear Safety-Related Concrete Structures. 2002;
43. PE 432.1 / 2002 – Norm on the monitoring of the behaviour in time of the constructions (national standard).

LIST OF ACRONYMS

AECL - Atomic Energy of Canada Limited

AFCIS - Advanced Fuel Channel Inspection System

AMP - Ageing Management Program

ANS – American National Standard

ANSI – American National Standards Institute

ARDM - Aging Related Degradation Mechanism

ASME – American Society of Mechanical Engineers

ASR – Alkali Silica Reaction

BP – Buried Piping

CANDU - Canadian Deuterium Uranium Reactor

CEI – CANDU Energy Inc.

CIV - Cells High Voltage

CLB – Current Licensing Basis

CNCAN - National Commission for Nuclear Activities Control

COG – CANDU Owners Group

CPRs - Component Plan Recommendations

CSA – Canadian Standards Association

CT – Calandria Tube

CVS – Calandria Vault Structure

DCC - Digital Control Computers

DCS - Distributed Control System

DHC - Delayed Hydride Cracking

ECR - Engineering Change Request

EFPH - Effective Full Power Hours

EPRI – Electric Power Research Institute

EQ – Environmental Qualification

ENSREG – European Nuclear Safety Regulators Group

FAC - Flow-Accelerated Corrosion

FC – Fuel Channel

FFA - Functional Failure Analysis

FSAR - Final Safety Analysis Report

FTS – Fuel Transfer Structure

FP – Full power

IAEA - International Atomic Energy Agency

IEEE – Institute of Electrical and Electronics Engineers

IEC – International Electrotechnical Commission

IGALL - IAEA's International Generic Ageing Lessons Learned program

IMSP - Integrated Material Surveillance Program

INPO – Institute of Nuclear Power Operations

ISCIR - State Inspectorate for Boilers, Pressure Vessels and Hoisting Installations

ISO – International Organization for Standardization

LCMP - Life Cycle Management Plan

LRT – Leak Rate Test

MERP - Maintaining Equipment Reliability Process

MS&I - Maintenance Surveillance and In-Service Inspection Program

NMC - Norms on Quality Management

NPP - Nuclear Power Plant

ODM - Operational Decision Making

OEM - Original Equipment Manufacturer

PHM - Program Health Monitoring

PIs – Performance Indicators

PIP - Periodic Inspection Program

PLCP - Plant Life Cycle Plan

PLIM - Plant Life Management Program

PM – Preventive Maintenance

PHT – Primary Heat Transport (System)

PHWR - Pressurised Heavy Water Reactor

PSA - Probabilistic Safety Assessment

PSR - Periodic Safety Review

PT – Pressure Tube

QMS - Quality Management System

RATEN - Technologies for Nuclear Energy State Owned Company

RATEN - CITON - Centre of Technology and Engineering for Nuclear Projects

RATEN - ICN – Institute for Nuclear Research

RB – Reactor Building

RCE - Responsible Component Engineer

RSE - Responsible System Engineer

RTM - Run-to-Maintenance

SDG – Standby Diesel Generators

SHMP/CHMP - System/Component Health monitoring plan

SHMRB - System Health Monitoring Review Board

SNN - National Company “NUCLEARELECTRICA”

SPOC - Single Point of Contact

SPV - Single Point of Vulnerability

SSCs – Systems, Structures and Components

TLAA - Time-Limited Ageing Analysis

TOE - Technical Operability Evaluation

TPR – Topical Peer Review

TRIGA - Training, Research, Isotopes, General Atomics

VWSG - Vibrating Wire Strain Gauges

WANO - World Association of Nuclear Operators

WENRA - Western European Nuclear Regulators Association