NATIONAL ATOMIC ENERGY AGENCY

NATIONAL ASSESSMENT REPORT OF POLAND

TOPICAL PEER REVIEW 2017

AGEING MANAGEMENT

Polish 1th TPR national report prepared in accordance with the Nuclear Safety Directive 2014/87/EURATOM

(blank page)

TABLE OF CONTENTS

Executive summary

Information included in this report describes selected by ENSREG issues related to ageing management of the only one nuclear facility in Poland that meets ENSREG threshold for peer review - MARIA Research Reactor. Description of licensee approach to ageing management has been included in this report, however some chapters which do not apply to MARIA Research Reactor design were skipped (ie. Reactor Pressure Vessel, Calandria/Pressure Tubes and Pre-Stressed concrete pressure vessel). In each chapter national nuclear regulatory body (PAA) added it’s comments related to licensee description and information on inspections of ageing of selected components.

Identified main issues related to ageing of licensee is that they conduct reactor operation with ageing control programme but not ageing management programme. The main difference between this two programs is that actually licensee performs only control of ageing parameters without using feedback from this control to perform proper ageing management process.

Nevertheless, from year 2015 licensee makes effort simultaneously with development of new Safety Classification (and using feedback from this document), to develop the Ageing Management Programme. New deadline approved by regulatory body for developing the Ageing Management Programme is set for the end of 2017 year. Due to that reason licensee added to this report some information about content and requirements that will be used during development process of new document.

The report is written by both license and regulatory body representatives in accordance with requirements included in document ‘Report Topical Peer Review 2017 Ageing Management Technical Specification for the National Assessment Reports’.

Preamble

This report is based on the European Union Nuclear Safety Directive 2014/87/EURATOM amending the previous directive on this issue 2009/71/EURATOM. The reason for the changes was to regulate safety issues at EU level. Following the events at the Fukushima Daiichi Nuclear Power Plant, the European Council has mandated national regulatory bodies to carry outthe Stress Tests. A significant conclusion from the Stress Tests, beyond the possible areas where the approach to nuclear safety could be improved, has shown the key role of mechanisms for closer cooperation and coordination among all parties responsible for nuclear safety. Peer evaluation has been proven to be a good way to build trust in order to collect and exchange experiences and to ensure the common application of high standards of nuclear safety. It is also important to underline reviews of the existing legal framework for the safety of nuclear facilities carried out by the European Commission. These events were the basis for updating the Nuclear Safety Directive.

The previous Directive 2009/71/EURATOM imposed on the Member States the obligation to carry out periodic self-assessment of the national frameworks and of the competent regulatory bodies which precedes the conduct of international reviews. The new Directive also imposes an obligation to carry out a thematic evaluation every 6 years starting from 2017. The first step is to make a report on the assessment of the previously selected nuclear safety issue of relevant nuclear facilities. ENSREG also includes a research reactors with a thermal capacity of more than 1 MW, which entails carrying out a topical peer review for Poland, focused on the Maria Research Reactor (30 MW) and delivering a report, verified by the nuclear regulatory authority - National Atomic Energy Agency (PAA), to the European Commission by the end of 2017.

The goals of National Assessment Report are to:

- describe the general ageing management program,
- evaluate applications to identify strengths and weaknesses,
- identify actions to address areas that need improvement,
- prepare the report at an appropriate level of detail allowing for substantive peer evaluation.

The topic chosen by ENSREG is the management of ageing processes. Ageing is defined as the process of gradual changes in the characteristics of SSC occurring during its operation. These changes may lead to degradation of materials and equipment, and may result in a reduction in the safety margin of the reactor. In extreme cases, it may be necessary to shut down the reactor. The concept of ageing is related to many factors occurring during the operation of the reactor. They can be divided into two groups: physical factors and non-physical factors. The report addresses only the issue related to ageing caused by physical factors.

With respect to the ageing management program (AMP), these SSC that are relevant for nuclear safety are taken into account. AMP's task is to prevent, detect and mitigate the various types of ageing mechanism that may affect the safety functions. In the National Assessment Report, general ageing management programs are taken into account as well as more detailed descriptions of the selected elements of the SSC.

Member States through ENSREG have decided that issues related to electric cables, concealed pipelines, reactor pressure vessels and concrete structures will be included in the National Assessment Report. This selection allows to draw conclusions of the overall of ageing management functions in practice. The selected elements of the SSC serve as a basis for evaluating the AMP implementation.

The scope of nuclear facilities, to be assessed in the National Assessment Report, is as follows:

- Nuclear power plants,
- Research reactors with a thermal power of 1 MWt or more.

Research reactors of smaller capacities may be considered by Member States on a voluntary basis. The report will evaluate reactors that will be operated on 31 December 2017 or has been under construction on 31 December 2016. Reactors that are completely decommissioned or are authorized by the appropriate regulator for decommissioning stage do not fall within the scope of the National Assessment Report.

Taking into account range of the facilities described above, the National Assessment Report of Poland will cover one nuclear facility - MARIA Research Reactor - which meets the threshold of research reactors with a thermal power of 1MWt or more. The further content of the report will be related only to the assessment of this one facility.

Taking into account the above issues, the report will consist of the following chapters:

1. General Information
2. Overall ageing management programme requirements and implementation
3. Electrical cables,
4. Concealed pipelines,
5. Reactor pressure vessels,
6. Calandria / pressure tubes (CANDU),
7. Concrete containment structures,
8. Pre-stressed concrete pressure vessels (AGR),
9. Overall assessment and conclusions,
10. References.

It is important to emphasize that actually chapters 6 and 8 are not not applicable to the Maria Research Reactor design. . Due to the transparency and comparability of reports from different countries, these chapters have been retained in the report, although they are left just with short explanation.

1.GENERAL INFORMATION

1.1. Nuclear installations identification

The MARIA research reactor is located in Otwock - Świerk and operated by National Centre for Nuclear Research (NCBJ). MARIA is a high flux, light water cooled reactor of a pool type, water and beryllium moderated. Fuel elements in the shape of concentric tubes are placed in pressurized channels of a Field's pipe type. Fuel channels embedded in beryllium blocks, along with a graphite reflector are the core that is submerged in the reactor pool. The reactor pool and fuel channels have separate cooling systems. The reactor uses low-enriched (<20% 235U) nuclear fuel in the form of U3Si2 or UO2 dispersed in aluminium, covered with aluminium cladding. The MARIA reactor is equipped with vertical channels for irradiation of target materials, a pneumatic transfer system for short irradiations, and horizontal neutron beam channels.

MARIA reached its first criticality in December 1974. The reactor was operated until 1985, then it was temporarily shut down for the purpose of upgrading technological systems. The modernization included the following: enlargement of beryllium matrix, inspection of graphite blocks, upgrading of ventilation and temperature measurement systems. The second stage of modernization was conducted between 1996 and 2002, during regular maintenance and it consisted of: replacement of heat exchangers, replacement of instrumentation and control (I&C) systems, upgrading dosimetry system, modernization of fuel element integrity monitoring system. New coolant pumps in fuel channels circuit and residual heat removal pumps were installed in 2014. In September of the same year, the reactor was fully converted to low enriched fuel (LEU). As for 2017, there is no end of operation date for the MARIA reactor scheduled.

The major areas of the MARIA reactor application are the following: production of radioisotopes, testing of fuel elements and structural materials for nuclear engineering, neutron radiography, neutron activation analysis, neutron transmutation doping, education and basic research with neutron beam application.

The main characteristics of the MARIA reactor are as follows:

- Nominal power 30 MWt
- Moderator H2O, beryllium
- Reflector Graphite
- Fuel assemblies:
    - Material U3Si2/UO2 dispersed in aluminium
    - Enrichment <20% 235U
    - Cladding Aluminium
    - Shape Concentric tubes
    - Active length 1000 mm

The MARIA reactor facility was designed as a complex of several buildings, functionally connected (Fig.1). The reactor core is housed in building B. The building consists of three main structural elements of the concrete safety housing: the upper dome, the cylinder, and the bottom plate (slab). During normal operation the building

B concrete structure is designed to withstand the vacuum operating pressure up to 200 Pa (200 mm H2O). During emergency situations the concrete safety housing (confinement) is designed to withstand the overpressure 0.01 MPa. Maintaining the vacuum pressure in the reactor hall, enables limitation and control of releases of the radioactive gases.

The MARIA reactor is powered by two transformer stations: OPT-11 and OPT-12 6/0,4kV. They are low voltage switching stations (nn) 0,4kV and they provide power supply to the reactor's main (RG-I, RG-II) and auxiliary (RP-I, RP-II) switching stations. In the MARIA reactor, all power supply cables are low voltage cables.

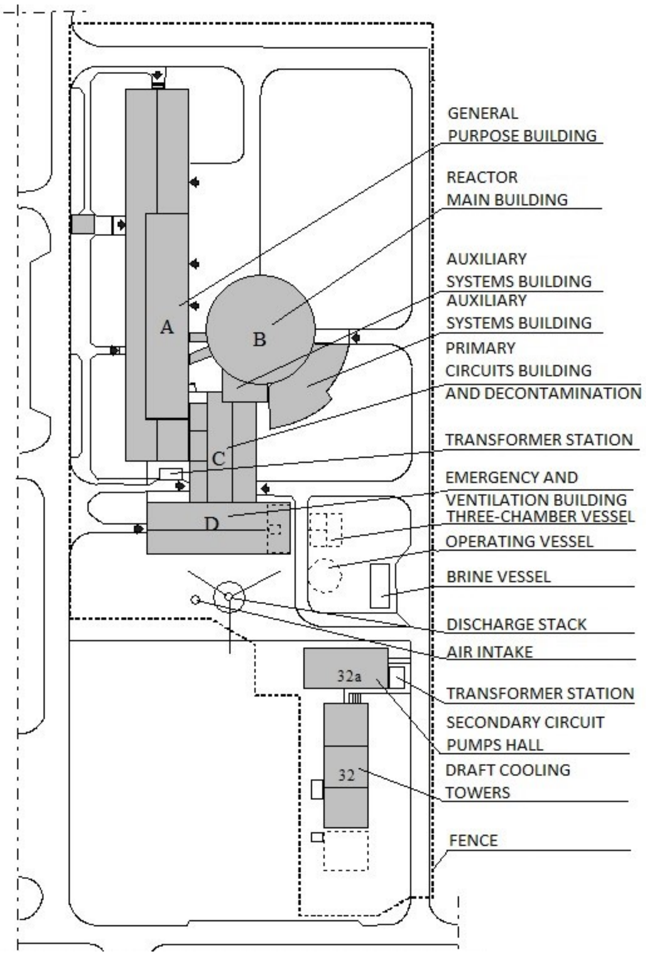

Fig.1. Site plan of the MARIA reactor facilities.

1.2. Process to develop the national assessment report

Poland has only one nuclear facility which can be placed under the ENSREG guidelines - the MARIA research reactor located in Otwock. Therefore, the National Assessment Report was prepared in co-operation of the PAA and the operator of the Maria research reactor. In May 2017, the PAA asked the NCBJ to prepare the report in accordance with the methodology and specifications described in the WENRA Technical Specifications, as adopted by ENSREG. Within the project the tasks were divided between PAA and NCBJ.

The role of the PAA was to, oversight and assess the materials prepared by the NCBJ and present the results and conclusions resulting from the carried out work.

The final version is to be published on the PAA website in December 2017 and accessible through ENSREG website.

2. Overall ageing management programme requirements and implementation

2.1. National regulatory framework

General requirements of nuclear safety of nuclear facilities are governed by the Law of 29 November 2000 called "Atomic Law".

Atomic Law and its subsequent specific regulations contain provisions that regulate requirements for:

1. radiological protection (personnel, society and patients);
2. nuclear safety and radiological protection, including:

- safety of nuclear facilities,
- nuclear material and sources of ionizing radiation,
- radioactive waste and spent nuclear fuel,
- transport of nuclear materials and radioactive sources and spent nuclear fuel and radioactive waste,
- assessment of radiation levels and emergency measures;

3. physical protection (nuclear facilities and nuclear materials);
4. non-proliferation of nuclear materials and technologies;
5. civil liability for nuclear damage.

In the Polish legal framework specific Regulations of the Council of Ministers has been adopted. Some requirements relevant to ageing management are described below.

Regulation of the Council of Ministers of 31 August 2012 on nuclear safety and radiological protection requirements which must be fulfilled by a nuclear facility design

Within the framework of the Regulation of the Council of Ministers on Nuclear Safety and Radiation Protection Requirements, the requirement to indicate in the design these SSC of a nuclear facility important for the nuclear safety and radiological protection whose proper functioning (which can be degraded by ageing mechanism, in particular as a result of environmental conditions, in particular such as: vibration, temperature, pressure, impact of liquid stream or splinters, electromagnetic interference, irradiation, flooding, humidity and any possible combination of these factors occurring at the time when the operation of these SSC will be necessary) are guaranteed by appropriate qualification tests.

The designs of SSC that are essential for nuclear safety and radiological protection have to include in the project appropriate safety solutions which takes into account exploitation mechanisms of these SSC and their potential for technological degradation associated with aging, to ensure the ability of SSC of a nuclear facility to perform safety features throughout the facility's intended use period. Also the effects of their ageing under normal operating conditions, maintenance, repair and upgrading operations, as well as the state of the facility during and after the occurrence of postulated initiating events have to be taken into account.

Regulation of the Council of Ministers of 31 August 2012 on the scope and method for the performance of safety analyses prior to the submission of an application requesting the issue of a license for the construction of a nuclear facility and the scope of the preliminary safety report for a nuclear facility

Under the Regulation of the Council of Ministers on the scope and method of carrying out safety analyses conducted prior to the application for a permit for the construction of a nuclear facility and the scope of the initial safety report for a nuclear facility, the requirement to analyse whether long-term ageing mechanisms of the nuclear facility have been identified in the nuclear project which can reduce its reliability and to ensure that they are monitored and appropriate measures will be taken.

Probabilistic analysis of the safety of a nuclear facility should take into account all internal and external events that may occur in all normal operation modes of the facility and lead to the release of radioactive material from any source at the nuclear facility. The analysis should identify all sequences of failures and errors that may increase the risk. For this purpose, ageing mechanism of SSC of a nuclear facility also have to be taken into account.

Regulation of the Council of Ministers of 11 February 2013 on requirements for the commissioning and operation of nuclear facilities

The regulation of the Council of Ministers on the requirements for commissioning and operation of nuclear facilities take into account the management of ageing processes within the framework of the program of maintenance and repairs, research, supervision and control of SSC of a nuclear facility important for nuclear safety. Incorporation of ageing management and appropriate measures ensure reliable performance of required safety functions by SSC throughout the lifetime of a nuclear facility, taking into account in particular long-term degradation processes occurring at the effect of operation and environmental conditions.

Under this program, the frequency of activities included in the document must also be specified which takes into account the estimated possibility of degradation of SSC of a nuclear facility during operation and the characteristics of their aging.

Regulation of the Council of Ministers of 11 February 2013 on nuclear safety and radiological protection requirements for the stage of decommissioning of nuclear facilities and the content of a report on decommissioning of a nuclear facility

In accordance with the scope of this Council of Ministers Regulation, periodic safety inspections shall be carried out at the frequency specified in a specific permit, which shall not be less than one in 10 years. Under the aforementioned, the ageing of SSC of a nuclear facility essential for nuclear safety and radiological protection must be taken into account.

2.2. International standards

The reference documents of the International Atomic Energy Agency have been used in assessment of the Ageing Management:

- Management of Research Reactor Ageing - IAEA, TECDOC-792,
- Ageing Management for Research Reactors - IAEA, Specific Safety Guide No. SSG-10,
- Maintenance, Periodic Testing and Inspection of Research Reactors - Safety Guide No. NS-G-4.2.,
- Report WENRA Safety Reference Levels for Existing Reactors, Issue I:Ageing Management.

2.3. Description of the overall ageing management programme

In 2012, the Ageing Control Programme (ACP) [1] included in a System Procedure, was implemented in the MARIA reactor. The scope of ACP contains monitoring and inspections of ageing processes with the application of procedures included in the MARIA reactor technological instructions. No ageing management is provided in ACP. It will be supplemented in a new Ageing Management Programme (AMP) being implemented in the MARIA reactor, and also expanded with the missing elements. The Ageing Management Programme is supposed to replace ACP.

2.3.1. Scope of the overall AMP

Assignment of responsibilities for Ageing Management Programme (AMP) development and implementation

In the current MARIA reactor Ageing Control Programme, included in the System Procedure, there are noparticular persons designated to be responsible for its overall implementation. According to the technological instructions, the managers of divisions are responsible for implementation of the technical condition check of the reactor equipment and systems. Furthermore, the responsibility for appointing the Committee conducting periodic assessment of the condition of the facility was assigned to the Head of the Maria Reactor Operation Division (EJ2). The members of the Committee are the section managers of the Operation Division, as well as the Quality Assurance Specialist of the MARIA reactor. The appointed Committee prepares the assessment of the reactor facility, recommends its further operation, and suggests changes or updates of the ACP.

Systems, Structures and Components (SSC) of the reactor, within the scope of ACP, have been divided into groups, and the implementation of Ageing Control Programme of the SSC groups has been assigned to the relevant sections of the Operation Division (EJ2). According to ACP, the managers of the sections are responsible for implementation of the instructions. The MARIA reactor Operator Section (DOM) performs inspections of SSC classified to groups including the following: the reactor core, the reactor pool, and the storage pool. The MARIA reactor Mechanical Section (DMM) performs inspections of SSC classified to groups including: the reactor building (R2-B), ventilation system, and general infrastructure of the building. The Mechanical Section (DMM) is also responsible for procedures related to controlling mechanical part of the reactor cooling circuits.

SSC assigned to the power supply systems and power supply components of the control circuits are inspected by the MARIA reactor Electrical Section (DEM). SSC included in the control system unit and the cooling circuit of the reactor, are inspected by the Instrumentation Section (DAM) of the MARIA reactor. SSC assigned to the dosimetry system are inspected by the Dosimetry Lab of the MARIA reactor.

Methods used for identifying SSC within the scope of overall AMP

The selection of SSC within the scope of ACP has been based on the operating experience of the MARIA reactor. In particular, components that affect nuclear safety of the facility, as well as these particularly susceptible to ageing processes, fall within the scope of ACP. The ACP contains elements included in the Safety Classification of the MARIA reactor SSC [2] that perform safety functions during normal operation and anticipated operational occurrences, and are subject to ageing. The development and adoption of the new MARIA reactor AMP based on the SSG-10 methodology [3] is planned until the end of the current year. The scope of AMP will be updated periodically along with the SSC changes taking place in the MARIA reactor.

Grouping methods of SSC in the screening process

In order to effective manage of ageing inspection process, according to SSG-10, selected SSC were grouped with regard to the similarities in the components structure (groups: electrical cables, blocks situated in the reactor matrix, matters related to cooling circuit of fuel channels, measurement instrumentation), and also due to operation under similar conditions (high neutron radiation, pressure and flow, temperature and humidity). The division facilitates the assignment of responsibilities for particular screening to the relevant sections of the MARIA Reactor Operation Division.

SSC groups included in the ageing control programme, as follows:

- Reactor core,
- Reactor pool and storage pool,
- Control systems,
- Reactor building (R2-B),
- Reactor cooling circuits,
- Electric power supply system,
- Ventilation system,
- Dosimetry system,
- Infrastructure - assessment of building technical conditions and building installations.

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

Currently applied ACP is not an ageing management programme. The methodologies and requirements used to assess present maintenance procedures and to develop ageing management programmes have not been included in ACP.

This methodology is described in The Quality Assurance Programme of The MARIA Reactor Facility (PZJ-MARIA-15) [4].

The MARIA reactor applies operation and maintenance manuals within the scope of maintenance procedures. Due to lack of accelerated degradation of SSC, so far there has not occurred any need to evaluate the documents mentioned above.

Quality assurance of the overall AMP

The quality assurance system in ACP of the MARIA reactor is based on the current PZJ-MARIA-15 [4].

The quality assurance system is defined in this PZJ-MARIA-15 as a set of documents of a hierarchical structure, conventionally divided into three levels. Lower quality assurance level documents determine requirements resulting from the documents of a higher level. PZJ-MARIA-15 is of the highest importance (1st level documents).

The MARIA reactor Ageing Control Programme in the form of Instruction 03-ZR-15 is a second level document - a system procedure. The programme contains instructions and procedures that identify the ways in which actions can be taken to assess the effects of ageing, being the third level documents.

According to PZJ-MARIA-15, the third level documents are detailed documents - technological and control instructions, as well as operation and maintenance manuals. These documents refer to the activities performed during the operation of the reactor, the scope of responsibilities of the individual employees, the organizational chart of the maintenance teams, the actions to be taken during screening to the ageing control inspections, the tests and other activities that are performed at the frequency specified in those documents.

Data collection and storage

Any data obtained during implementation of the MARIA reactor ACP are stored either in paper or electronic form or in both forms, namely:

- post-inspection protocols – paper and in an electronic form,
- results of measurements – electronic form,
- results of monitoring – electronic form.

Document evaluation

The Nuclear Facilities Operation Department performs the evaluation of documents in accordance with the preparation, approval and archiving procedures of the MARIA reactor operating documents. ACP is also subject to this procedure.

According to this document, every procedure and instruction is subject to continuous evaluation during the established verification periods (once every five years for system procedures and once every three years for operational procedures). This document also requires revision of procedures and instructions as a result of any changes in the reactor technological processes, technical standards and conditions, quality assurance programmes or as a result of post-inspection requests (internal and external audits).

2.3.2. Ageing assessment

Key standards and guidance, as well as key design, manufacturing and operations documents are used to prepare the overall AMP

Preparation of the current ACP has been based on the guidelines and recommendations published by the International Atomic Energy Agency, in particular:

- Specific Safety Guide No. SSG-10,
- Management of Research Reactor Ageing TECDOC-792 [5].

Key elements used in plant programmes to assess ageing

The guidelines included in the SSG-10 have been used to develop the criteria for grouping SSC elements in the review process.

Data obtained from TECDOC-792 have been used to define ageing and to determine physical and non-physical factors affecting the ageing of the MARIA reactor SSC.

Included in ACP, the review schedule of the technical condition of the reactor equipment and systems has been based on a chapter relating to ageing control activities presented in Management of Research Reactor Ageing.

The MARIA reactor Safety Analysis Report [6] was used to analyse Systems, Structures and Components (SSC), in particular to analyse the MARIA reactor unique elements of SSC.

Individual tests that fall within the scope of ACP, as well as the other procedures not included in ACP used to ageing assessment, have been based on Polish and international standards, operation and maintenance manuals, and on operating experience presented in detailed technological instructions. Individual SSC are susceptible to various ageing processes to varying degrees. The ageing control programme particularly refers to the ageing processes mentioned below.

Tab. 1.

Operating/ambient conditions	Ageing processes	Consequences
Radiation	Change of Properties	Dislocations in graphite grid Beryllium brittleness Decomposition
Temperature	Change of Properties	Durability decrease Concrete dehydration Curing of polymers
Flow/pressure	Erosion	Change of durability Leakage
Water chemistry	Corrosion	Change of durability Leakage
Vibrations	Change of quality	Change of mechanical properties Decomposition

For all the SSC included within the scope of ACP, there are physical parameters defined as the criteria used to ageing assessment and/or having a significant impact on the ageing process. The key parameters are the following:

- fluence of gamma and neutron radiation,
- contamination,
- temperature,
- dimensions,
- dielectric resistance,
- corrosive accretion and erosive losses.

The parameters mentioned above are strictly defined in respective technological instructions.

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

The physical parameters mentioned above are controlled by monitoring, inspections and tests specified in the review schedule of the technical condition of the reactor equipment and systems, included in the ACP. The individual instructions specify actions that lead to the detection of ageing mechanisms and determine the critical values of physical parameters used in control of ageing mechanisms

Establishment of acceptance criteria for ageing

The process of establishing criteria has not been described in the current ACP. However, the acceptance criteria for ageing included in the MARIA reactor technological instructions are based on internal operating experience, technical standards, experience of other reactors, and research and development projects.

Use of R&D programmes

The process of using research and development projects was not described in the current ACP during its preparation. The individual technological instructions of the MARIA reactor used for ageing assessment, were based mainly on available standards and operating experience. In the absence of the above mentioned, the results of internal and external R&D projects were used to develop, verify or improve individual instructions.

Use of internal and external operating experience

The process of using internal and external operating experience was not described in the current ACP during its preparation. Nevertheless, the experience gained during operation of the MARIA reactor was implemented in individual technological instructions used in the programme. Likewise, external operating experience is analysed, among others, to evaluate the instructions. In the absence of internal data required to develop the programmes, the experience of other nuclear facilities is used.

2.3.3. Monitoring, testing, sampling and inspection activities

Programmes for monitoring condition indicators and parameters, and trending

The current ACP process does not include monitoring programmes dedicated to the management of ageing processes, however, to monitor ageing processes, operational monitoring is being used. According to the MARIA reactor Operation and Maintenance Instruction [7], the operating parameters of the reactor equipment are examined. Among them, there are parameters monitored to determine the condition of the following equipment and reactor systems:

- temperature of the main and residual heat removal pump bearings and their motors in the channel cooling circuit,
- vibrations of the residual heat removal pump bearings of the channel cooling circuit,
- the bearing temperature of the reactor building fans (Building B).

During the operation, the parameters recorded every two hours in the Reactor Operation Chart enable to track the trends in the devices exhaustion. Ageing of the elements is determined by the observed increase in temperature and vibration on the monitored bearings. This suggests excessive bearing wear and the necessity of replacement or maintenance.

In addition, the temperature and vibration values of the fuel channel circuit pumps are recorded digitally.

Inspection programmes

To assess the technical condition and the ageing processes effects, the ACP applies periodic tests, inspections and controls of the devices and their components. The actions are carried out in accordance with operating instructions.

The following tests are conducted in the reactor:

- inspection of the safety and control rods condition;
- inspection of block apertures for safety and control rods;
- inspection of the control rods channels;
- inspection of the control rods diameter;
- measurement of force required to pull the control rod in a channel;
- water tightness test of the following: reactor pool liner, storage pool, and the water sluice;
- water tightness test of the heat exchangers of the primary cooling circuits;
- air tightness test of delay gas tank;
- water tightness test of emergency core cooling system valve;
- air tightness test of the reactor building;
- inspections of the beryllium and graphite blocks condition;
- inspection of the technological ventilation.

Surveillance programmes

The current ACP process does not include surveillance programms. However, the surveillance of following components is performed:

- beryllium and graphite blocks
- fuel element cladding

Beryllium blocks have a parameter that limits their maximum operating time in the reactor core – maximum fluence of fast neutrons absorbed by a beryllium block. The fluence is controlled using the numerical codes: Monte Carlo type codeand diffusion code.

The tightness of cladding is also monitored by fuel element integrity monitoring system for fuel elements (WNEP) installed in the reactor. The detection system traces the slow degradation of the fuel element cladding during long-term operation in the reactor (radiation damage, corrosion, erosion, etc.). It also enables to predict the leakage degree gained by the operating element, resulting in the necessity of removing such an element from the reactor core.

Provisions for identifying unexpected degradation

The guidelines for identifying all expected and unexpected degradation in the MARIA reactor are not included in the ACP, however, they are provided by the tests and inspections of the MARIA reactor operational control programme.

2.3.4. Preventive and remedial actions

Preventive and remedial actions are not included in the current ACP of the MARIA reactor. The assessment of the condition of the facility is conducted by a committee appointed by the Head of the Reactor Operation Division. The committee prepares an assessment of the reactor facility, recommends its further operation and may submit requirements for changes or improvements.

Despite the absence of a systematic approach to preventive and remedial actions, a number of instructions have been implemented: operational, chemical, electrical, mechanical, dosimetry guidelines, and instrumentation manuals. Indications of preventive, remedial and maintenance actions of selected SSC are included in the instructions.

2.4. Review and update of the overall AMP

Within the scope of the being currently prepared AMP, the means of the following processes implementation will be described (some of them are implemented according to other documents, to which ACP is subject):

Implementation of internal audits findings

Principles of planning and conducting internal audits are not included in the ACP, however, they are currently included in the Integrated Management System of the National Centre for Nuclear Research [8]. Both PZJ-MARIA-15 and ACP are subject to the Integrated Management System procedures. Internal audits are conducted by the MARIA Reactor Quality Assurance Specialist (SZJ-RM). The Quality Assurance Specialist is obliged to submit the conclusions from these reviews and inspections in writing to the Director of the Nuclear Facilities Operations Department. The Director makes further decisions related to their implementation.

Evaluation of plant specific and others' operating experiences

Collection of information on operating experience from reactor users will be held once a year, every time regarding the past calendar year. Moreover, information will be collected from people supervising the operation of the reactor.

On the basis of collected internal information, as well as other external information, an assessment of their impact on the ageing process of the reactor will be performed.

Evaluation of plant modifications that might influence the overall ageing management programme

Modifications introduced in the facility, that may influence AMP, will be evaluated during every implementation.

These actions are currently being implemented in accordance with the Procedure for the Development, Approval and Archiving of Operating Documents of the MARIA Reactor, ACP is also subject to this procedure.

Evaluation of the effectiveness of ageing management

In the Maria reactor, evaluation of documents including assessment of their effectiveness, is conducted in accordance with the MARIA reactor operating documents development, approval and archiving procedures. ACP is also subject to this procedure and the evaluation is held every five years.

Evaluation of ageing analyses that are time limited

The MARIA Reactor Quality Assurance Specialist is obliged to monitor the topicality of the ageing analysis contained in the ACP. In case of expiry of their period of validity, the MARIA Reactor Quality Assurance Specialist is entitled to apply to the Director of Nuclear Facilities Operations Department (DEJ) for their updating.

How current "state of art", including R&D results, is taken into account

Analysis of current knowledge and determination of whether new methods of research should be implemented, replacing those previously adopted in AMP, will take place once every two years by the persons responsible for the study. Once the analysis has been done for every possible range of research, a discussion will be held every year. The discussions should result in requests for R&D activities that can be performed to improve the research methods available.

Consideration within the overall ageing management programme of modifications in the current licensing or regulatory framework

A review of regulatory and licensing changes will be conducted once every two years, each time regarding the past two years. In exceptional cases of sudden regulatory changes, this review will be performed on an ad hoc basis, following the adoption of the relevant legislation, during the "vacatio legis".

Identification of need for further R&D

Based on the current state of knowledge and the results of internal audits of AMP, the Quality Assurance Specialist of the MARIA reactor will identify and report the need to apply new R&D programmes to the DEJ Director.

Furthermore, the forthcoming AMP will also address the following:

> **Strategy for periodic review of the overall AMP including potential interface with periodic safety reviews**
>
> The AMP update will be performed once every four years, in each case regarding the past four years, and will take into account the conclusions drawn during the periodic reviews made in the past four years.
>
> To that end, a separate, brief report will be prepared, describing the scope of changes and their causes. In the case of changes to the AMP structure, a reference table will be enclosed to facilitate navigation of modified AMP extracts.
>
> In the event of the impact of the AMP update on the content of the Safety Analysis Report (SAR), amendments will be made to it in the form of an Annex, agreed with the National Atomic Energy Agency (PAA).
>
> **Incorporation of unexpected or new issues into the AMP**
>
> In the case of unexpected necessity to apply urgent changes to the AMP, their incorporation will be performed as soon as possible. Amendments agreed with the PAA will be applied to the AMP, as well as to the SAR, if necessary.
>
> **Use of results from monitoring, testing, sampling and inspection activities to review the overall AMP**
>
> Once a year, regarding the past calendar year, an impact assessment of the results of the facility condition monitoring, its periodic structural testing, and the AMP tests, will be performed.
>
> **Periodic evaluation and measurement of the effectiveness of ageing management**

In the MARIA reactor, documents evaluation including assessment of their effectiveness, is conducted in accordance with the MARIA reactor Development, Approval, and Archiving Procedure of the MARIA reactor operating documents. ACP is also subject to this procedure, that is held every five years.

2.5. Licensee's experience of application of the overall AMP

The current ACP being applied in the MARIA reactor, is not a programme for managing ageing processes in the strict sense of the term. The scope of the ACP includes monitoring and control of ageing processes, however, it does not contain guidelines for the management of ageing programmes.

ACP guidelines do not provide any directives for assessing the quality of ageing control processes, what negatively affects the supervision of its correct implementation. Quality control of all the ACP processes and the forthcoming AMP, should be included in the system process for ageing management.

The operating instructions of the ACP are the technological instructions of the MARIA reactor also used for current operation. In addition, some of the procedures used to monitor, evaluate and prevent ageing have not been included in the ACP, despite their active deployment. The absence of system approach to managing ageing processes results in the dispersion of documents used to control ageing and responsibility for individual programmes. During the implementation of the new AMP, a thorough analysis of the documents of ageing management procedures used in the MARIA reactor, will be conducted. In the MARIA reactor, planned deployment of a new AMP is to replace the ACP, and also will be completed with missing elements.

2.6. Regulatory oversight process

Regulatory assessment of the overall AMP and its modification.

Polish regulatory framework is constructed by prescriptive and non-prescriptive way of regulation and as it was mentioned in previous chapter it doesn't have precise requirements about AMP. Nonetheless ageing program as part of quality assurance program was taken in consideration during the last process of renewal licence in 2015 year. PAA assessed documentation related with ageing management and gave additional licence condition to NCBJ in order to make additional review of the reactor SSC from ageing point of view it's impact on reactor operation. Subsequently, PAA required NCBJ to update the reactor MARIA ageing control program. This process was completed by NCBJ in timely manner. Nonetheless revised ACP document submitted to PAA President was inadequate in comparison to IAEA standards. Therefore, after the assessment, the PAA has decided to set up new the deadline for preparation of the AMP by the end of 2017 year.

Regulatory inspection of implementation of the overall AMP

The ACP is subject to regular PAA inspection program and any findings resulting from inspections are to be included into further correction actions.

2.7. Regulator's assessment of the overall ageing management programme and conclusions

The regulator assessment of ageing management processes described in this chapter

Information's presented by NCBJ are based on ageing control program from year 2015. This causes that many requirements and recommendations from WENRA reference levels or IAEA standards weren't achieved. PAA identified some areas which need to be improved and potential good practices which are positive signal for future.

Weaknesses:

- Responsibility for ACP is *divided* for managers of the sections what causes that final response is scattered. There is no single person responsible for development and implementation of AMP.
- Methodology used for identifying SSC has been based on the operating experience of the MARIA reactor. It seems that this approach cannot be sufficient and should be, as it was written by NCBJ, replaced with a new version.
- Methodology and requirements for evaluation of existing maintenance practices and developing of ageing programmes, appropriate for the identified significant ageing mechanisms does not exist as a result of lack of management module in current ACP.
- Lack of indicators in process of effectiveness assessment and trending information relay on maintenance history. Apart from this, NCBJ hasn't documented process of establishing criteria, R&D, internal and external operating experience, monitoring condition indicators and parameters, surveillance programs and others. Current activity related with ageing management like process of acceptance criteria for ageing is based on many sources like internal operating experience, technical standards, experience of other reactors and R&D projects, but it was not described in ACP and in many examples also not sufficiently documented. These gaps causes that program is not manageable and operator cannot use potential which it has, to implement preventive program and remedial action.

Strengths:

- Grouping methods of SSC in the screening process by using similar components is suggested by IAEA standards and this is a good step which can improve efficiently screening process.
- Existing in NCBJ quality assurance cover ACP and could be used for AMP .
- Identification of key elements of ageing mechanisms and their possible consequences are sufficient for basic level of ACP but this also should be developed adequately to scope of new program.

Experience from inspections and assessment as part of regulatory oversight

Official ACP document has a lot of gaps in comparison to international standards in that area. Nonetheless it is still basic document which is used during process of inspection conducted by PAA. Regulatory body also takes in consideration others documents, not mentioned in ACP, used by NCBJ in processes related with ageing control which comply with the industrial standards. PAA identified most of gaps in ACP during the process of assessment in 2015 year. NCBJ is aware of their weakness and is going to systematize whole activity which has been conducted with close relation with ageing management. These topics should be covered in new AMP.

3. Electrical cables

3.1. Descriptions of ageing management programmes for electrical cables

Electrical cables have not been included within the scope of the current ACP. Nevertheless, there have been introduced guidelines for testing of power cables, batteries, uninterruptible power supply units, main switchboard, and auxiliary switchgears. Electrical cables are under the supervision because they fulfill the safety functions included in the Safety Classification of System, Structures and Components, and are located in rooms and ducts under adverse conditions (moisture, radiation, external conditions, etc.). There is a new Ageing Management Programme being prepared in the MARIA reactor.

The programme concerning electrical cables will include the following:

- Prevention of cable ageing.
- Assessment of cables ageing.
- Monitoring, testing and inspections of electrical cables.
- Preventive and remedial actions for electrical cables.

3.1.1. Scope of ageing management for electrical cables

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

According to the law [9], constructions should be periodically inspected during their use. At least once every 5 years, the inspection should consist of checking the technical condition and suitability of the facility. Electrical installations including as follows: connection efficiency, fittings and insulation resistance, should also be covered by the inspection. According to the Act mentioned above, all electrical cables in the MARIA reactor are subject to periodic inspection.

Processes/procedures for the identification of ageing mechanisms relating to cables

In order to identify the ageing mechanisms of electrical cables in the MARIA reactor, an analysis of Polish standards has been conducted [10], as well as the International Atomic Energy Agency [11,12] publications and American regulator's [13]. Based on the foregoing documents, mechanisms described in the chapter 3.1.2., have been defined.

Grouping criteria for ageing management purposes

Beginning with 2018, the following cable grouping regarding working conditions will be applied:

Group 1 - Measurement cables exposed to high neutron and gamma radiation.
Group 2 - Power cables exposed to at least 1 adverse environmental factor, resulting in accelerated degradation of a conductor or insulation parameters.
Group 3 - Power cables exposed to standard environmental factors causing degradation of a conductor or insulation parameters.

Tab. 2.

Cable Group	A building or a Process Room of the Reactor	Number of Adverse Environmental Factors	Adverse Environmental Factors
1,2	Reactor Hall (Building B)	2	neutron radiation, gamma radiation, humidity
2	Pump House (Building C)	2	humidity, gamma radiation
2	Building 32a	2	humidity, corrosive environment
2	Decontamination Hall (Building C)	1	gamma radiation
3	Building D	0	-
3	Building A	0	-

The harshest conditions exist in the pump house, where the primary circulation circuits main pumps and the residual heat removal pumps are located. They are responsible for the correct cooling of the reactor core, both during normal operation and in emergency situations. There are two exposure factors existing simultaneously, namely - high humidity and high gamma radiation.

In the building 32a, water is prepared for the reactor cooling circuits. In order to ensure the proper operation of the water preparation equipment (demineralization station, ion exchange units), brine and hydrochloric acid are applied. They cause corrosion of the contacts and cable ends in the auxiliary switchgears, also located in the building 32a. Another exposure factor for cables is increased air humidity, which is caused by the vicinity of a water-cooling tower of the secondary coolant circuit.

Electrical installations in the decontamination hall can be subjected to temporary radiation, which is related to the location of the hall above the pump station, as well as the expedition of radioactive materials, the storage of radioactive waste, cleaning of heat exchangers, etc.

The reactor core and spent fuel storage pool are located in the building B. Some cables and wires located in the hall B, concerning both measuring instrumentation and auxiliary power supply, are therefore exposed to increased levels of neutron radiation and gamma radiation, and also adverse environmental conditions in the form of increased humidity.

3.1.2. Ageing assessment of electrical cables

Ageing mechanisms requiring management and identification of their significance

In the MARIA reactor, there are two types of adverse factors, which influence ageing of the electrical components, in particular the conductors:

- External - adverse factors that occur independently of the operation of the wires, when the devices are either on or out of operation.

- Internal - adverse factors resulting from the operation of electrical systems connected to the wires and their peripherals.

Tab. 3.

-	Adverse Factor	Damage Mechanism	Damage Effect
External adverse factors	moisture	disintegration of the cable insulation sheathing; corrosion of unshielded cable connectors; delamination of insulated wires.	**Electrical** breakdown, loss of electrical continuity
	neutron and gamma radiation	decrease in the effectiveness of antioxidants effects in organic insulating materials, resulting in disintegration of the cable insulation	**Electrical** breakdown
	chemicals	corrosion of unshielded cable connectors	Loss of electrical continuity
Internal adverse factors	heating of components resulting from electrical or mechanical load	continuous operation of electrical wires at high load and at high ambient temperatures can lead to a decrease in conductivity and insulation values.	**Electrical** breakdown, loss of electrical continuity
	physical stresses caused by vibrations	loosening of connectors, increase of temperature in connector area	**Electrical** breakdown, loss of electrical continuity
	wear of electrical components, e.g. contacts caused by operation	repeated maintenance activities may cause fatigue of the contact material at the mounting site, which may result in mechanical damage	Loss of electrical continuity

Establishment of acceptance criteria related to ageing mechanisms

Criteria to be met by cables and protective devices have been established on the basis of the national standard PN-HD 60364-6.

3.1.3. Monitoring, testing, sampling and inspection activities for electrical cables

Description of activities

In accordance with the standard PN-HD 60364-6, the periodic tests of electrical installation include, inter alia, the following measurements:

- Electrical continuity test.
- Measurement of insulation resistance of electrical installation.
- Inspection of the effectiveness of the indirect contact protection by means of automatic power shutdown.
- Measurement of short-circuit impedance.

The electrical continuity test is performed for protective conductors. Insulation resistance measurements are made between the active conductors and the neutral-protective conductor (PEN) with an ohmmeter of a large measuring range. The short-circuit impedance measurement is required to verify the protection effectiveness by means of automatic power shutdown, and is performed by means of a short-circuit impedance meter. It is planned to implement an additional technique for identifying ageing processes in the new AMP, namely - TDR time domain reflectometry.

Frequencies

In the MARIA reactor facilities, electrical measurements for all groups of cables are performed at least once every 5 years, according to the Act of 2002 - Construction Law, article 62. Beginning with 2018, the frequency of measurements will be increased, concerning individual groups as follows:

- Group 1 - once a year,
- Group 2 - at least once every 3 years,
- Group 3 - at least once every 5 years.

Acceptance criteria

Criteria for electrical installation approval for further operation are the following:

- breaker switching off time is less than 0.4s,
- touch voltage at short circuit ≤ 50V (for alternating voltage),
- for a nominal voltage circuit up to 500V, the required insulation resistance is at least 1MΩ,
- continuity of protective conductors.

On the assumption, cables or safety devices do not meet the foregoing criteria, the installation is not suitable for further operation.

3.1.4. Preventive and remedial actions for electrical cables

The following preventive actions are being performed:

- inspection connections on terminals in RG-I and RG-II switchgears;
- inspection connections at the pump and fan motors terminals installed in the reactor process rooms.

At current flow, loose connections lead to a temperature increase in this area, which may damage the insulation of the conductor. Inspection of switchgear connectors is conducted twice a year, in accordance with the instructions for operation of RG and RP electrical switchgear units supplying the MARIA reactor facilities. Inspections of pump and fan motors terminals are held once a year.

The current ACP process does not determine remedial actions programms. However, according to the technological instructions, both not included and within the scope of ACP, provide guidelines for repairs, modernizations and refurbishment.

Despite the lack of guidelines for repairs of electrical cables, the following methods are used in MARIA Research Reactor:

- flawed cable replacement,
- flawed cable coupling,
- cable ending replacement,
- tightening of terminals.

3.2. Licensee's experience of the application of AMPs for electrical cables

So far, there have never occurred any necessity to replace the cables due to the failure to comply with the requirements of PN-HD 60364-6 standard, in the MARIA reactor.

Based on the results of the electrical measurements performed so far, the MARIA reactor is unable to determine the ageing process trends of the electrical cable components. These measurements have only been tagged as "acceptable", what precludes the possibility to track the measurement history.

3.3. Regulator's assessment and conclusions on ageing management of electrical cables

The regulator assessment of ageing management processes described in this chapter

Despite the fact that electrical cables have not been included within the scope of the current ACP, the NCBJ carries on testing of electrical cables. The NCBJ accurately analysed adverse conditions in the MARIA reactor to which electrical cables are exposed. In the AMP electrical cables will be grouped, giving the highest priority to measurements of cables important for safety which were exposed to high neutron and gamma radiation.

Detailed identification of the ageing mechanisms and their effects will allow to take measures to prevent damages of electrical cables. However, the programme for monitoring and trending has not been developed. It has a negative impact on the effective prevention of cable ageing.

The NCBJ is going to perform preventive and remedial actions for electrical cables in AMP based on current inspection program. As a result of this, the likelihood of sudden and severe cable damage will be significantly reduced.

Experience from inspections and assessment as part of its regulatory oversight

During the inspections the PAA focuses mainly on the operation of electrical equipment and electrical cables that are in current ACP. Previous inspections showed no non-conformities in control of ageing of electrical cables. Inspections of electrical cables described in the ACP are performed timely and without any technical problems.

Strengths and weaknesses identified by the regulator

The frequency of testing of some electrical cables in the MARIA reactor will increase from 2018 year in comparison to current dates. The most vulnerable cables will be checked every year. The licensee will also implement the Time Domain Reflectometry (TDR) method in order to identify the ageing process more effectively. As a result, the probability of detecting of cable damage will increase.

The weakness of the programme is that the ageing process trends of electrical cables are not being determined. Consequently, it is difficult to detect the ageing process and replace important cables before they will be having negative impact on safety.

Conclusions

The NCBJ is going to prepare a new AMP which will cover all current and planned activities related to ageing management of electrical cables. Licensee is going to group cables taking to an account established criteria. This will ensure that the relevant priority will be given to cables that fulfill safety functions. The NCBJ should also determine trending process for electrical cables and components.

4. Concealed pipework

The reactor design makes the number of concealed pipework relatively low. The sole systems, of which parts are buried in soil, encased in concrete or in covered trenches are determined below:

- secondary coolant circuit,
- impulse pipework.

The current ageing programme does not include the components mentioned above. They have not been contained in the AMP due to the lack of influence of the selected systems on nuclear safety of the facility, and only on the availability of the reactor.

The secondary coolant circuit pipework is designed to transfer the heat transferred from primary coolant system (building C) through cooling tower (building 32) to the environment.

The system consists of main circulating pumps, residual heat circulating pumps, water purification system and forced draft wet cooling towers. The pumping station is located in building no. 32, the pipeline connecting cooling tower and primary circuit heat exchangers is situated along buildings C and D on their east side. The pipework is insulated, encased in tar and partially buried in soil. During normal operation there are no radioactive substances in secondary coolant circuit.[1]

Although during normal operation there are no radioactive substances in secondary coolant circuit, the heat exchanger tube puncturing may occur realising the radioactive waste to environment. At the same time it is possible to isolate the faulty heat exchanger and therefore the secondary coolant circuit pipework was assigned to third safety class. Additionally low residual power enables the energy dissipation using only the primary circuit and the following auxiliary systems: demineralized water system and the radioactive effluents system. Due to the foregoing, the secondary coolant circuit pipework has not been included within the scope of ageing control programme.

The WNEP (Fuel Elements Leaks Detection) system monitors and detects leaks in fuel elements. The online gamma and delayed neutron radiation activity measurement in water samples enables fast and precise detection of fuel element piping degradation. The burnup of the fuel and the activity of the water sample is compared to the manufacturer's leakage-burnup plot. Any increase in the activity is displayed in the MARIA control room WNEP panel.

The part of WNEP piping is encased in biological shield of the MARIA research reactor. [1] Basalt concrete thickness of the storage pool shield at the height of the WNEP room is 1.4 m. The pipes are set in the serial arrangement, the vertical and horizontal pitches are 20 mm (Fig. 3).

Despite the fact that the components belong to the first safety class, this system has not been included in the ageing control programme. Loss of tightness concerning WNEP pipes with a diameter of 16 mm is estimated as low risk circumstance (local leakage from primary circuit and increase in dose rate of ionizing radiation in aerosols in the WNEP room). A small distance (4 mm) between these pipes significantly impedes the ability of non-destructive testing to investigate parts of all pipes embedded in the concrete shield (Fig. 3). Due to the difficulty of testing and easy implementation of remedial actions, consisting in cutting and clogging of the damaged tube and replacing it with a reserve tube, in relation to the obtained results, it has been decided not to include the these pipes in the ageing programme.

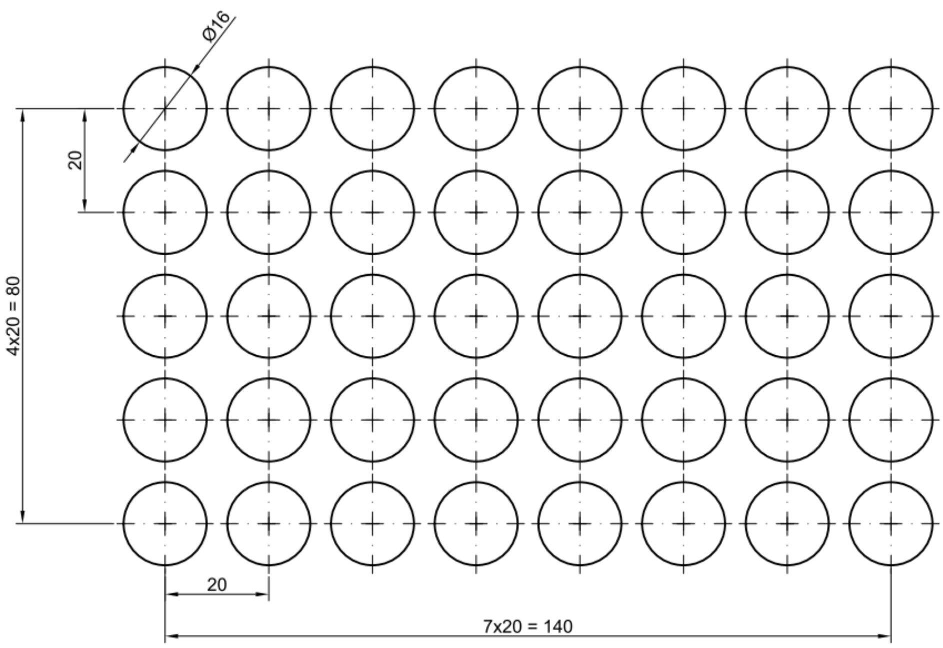

Fig.3. WNEP pipes arrangement.

Regulator's assessment and conclusions of ageing management of concealed pipework

The regulator assessment of ageing management processes described in this chapter

PAA agrees with NCBJ argumentation regarding skipping of concealed pipework from ACP and NAR. Both, secondary cooling circuit and WNEP pipes, do not have any serious impact of nuclear safety which justified them embrace by AMP.

Experience from inspections and assessment as part of its regulatory oversight

Safety assessment and inspections which have been made by PAA, have found no additional concealed pipework which should be covered by ACP. This approach is consistent with a graded approach attitude and have no impact for nuclear safety.

5. Reactor pressure vessels

The MARIA research reactor is a light-water reactor of a pool type, and it does not have a reactor pressure vessel.

PAA agrees with NCBJ argumentation regarding skipping of reactor pressure vessels chapter from national report.

6. Calandria/pressure tubes (CANDU)

The MARIA research reactor as a light-water pool type reactor does not have any calandria and pressure tubes specific for the CANDU type reactor.

PAA agrees with NCBJ argumentation regarding skipping of calandria/pressure tubes chapter from national report.

7. Concrete containment structures

No dedicated ACP for concrete containment structures has been implemented in the MARIA reactor. An analysis of the capabilities of the containment control condition, including its concrete part and the influence of the ageing process on its fulfillment of the safety function, is currently being conducted.

7.1. Description of ageing management programmes for concreted structures

7.1.1. Scope of ageing management for concrete structures

Methods and criteria used for selecting concrete structures within the scope of ageing management

The reactor MARIA containment is fulfilling the safety function, it enables limitation and control of the radioactive discharges into atmosphere and therefore it was integrated within the ACP scope. Only the analysis of the condition of the MARIA containment building in terms of its tightness has been established in the ACP.

Processes/procedures for the identification of ageing mechanisms for the different materials and components of the concrete structures

The ACP does not include the procedures determining the ageing mechanisms for materials and components of containment structures, including concrete containment structures. The inspection procedures of building are conducted in accordance with the provisions of the Act – Construction Law. [9]

7.1.2. Ageing assessment of concrete structures

Ageing mechanisms requiring management and identification of their significance

ACP does not analyse the ageing mechanisms and adverse factors leading to accelerated degradation of concrete containment structures. Only the effect – the decline in air tightness of reactor building – is analysed. [14]

Establishment of acceptance criteria related to ageing mechanisms

Included in the MARIA reactor ageing control programme, acceptance criteria for air tightness of the buildings are derived from calculations of physical quantities related to gas transformation.

7.1.3. Monitoring, testing, sampling and inspection activities for concrete structures

Description of activities

Periodic inspections of the technical condition of the reactor facility are performed. The scope of inspections covers the control of the technical building condition, building and installation elements exposed to adverse environment and damaging effects, occurring during the operation of the facility. The inspection is based on analysing the technical documentation of the facility, and the external inspection of the elements, where outcrops or other invasive methods of testing may be performed.

Within the scope of the current inspection procedures, the air tightness of the reactor building is controlled, which results from the requirements for safe operation of the facility. According to the MARIA reactor license issued by the President of PAA [15], maintaining the vacuum pressure in the reactor hall at a level of -5 mm H2O, enables controlled release of radioactive gases generated during the operation of the MARIA reactor.

Measurement of the reactor building air tightness is implemented in the technological sealing conditions of rooms and the whole building. The pressure increase provided by the fans being then shut down, results in gradual overpressure drop in the reactor hall, which is measured to determine the room tightness.

Frequencies

The inspections of building are conducted in accordance with the provisions of the Act – Construction Law:

- Periodic inspections, once a year, consisting of checking the technical conditions,
- Periodic inspections, once every five years, consisting of checking the technical conditions and usefulness.

Acceptance criteria

The acceptance criteria for air tightness test result from the design assumptions for the MARIA reactor building, for which the total building volume of the building B net is 17700 m3. The project assumes that at 5 mm H2O vacuum pressure, air penetration into the reactor hall through concrete walls and leaks does not exceed 1000 m3/h.

Criteria for classifying the technical condition of building components should meet the requirements of standards, approvals, certificates and technical conditions contained in the Act - Construction Law.

7.1.4. Preventive and remedial actions for concrete structures

Neither preventive nor remedial actions have been included in the current ACP for the MARIA reactor containment concrete structures. During the long-term operation of the building B, there were no indications to take such actions. The foregoing programme is primarily based on continuous monitoring of conditions.

In case, however, remedial actions for concrete structures prove to be necessary, repairs on the basis of expertise or existing concrete repair standards of PN-EN 1504 [16] will be made.

7.2. Licensee's experience of the application of AMPs for concrete structures

So far, during the MARIA reactor operation there have not been any unacceptable changes in the tightness of the containment.

7.3. Regulator's assessment and conclusions on ageing management of concrete structures

Regulator assessment of ageing management processes described in this chapter

Concrete structures were not taken into account in current safety classification of SSC and due to this fact NCBJ is not certain if this should be included in ageing management. Nevertheless NCBJ in accordance with the ACP performs air leak tightness test in the reactor concrete building and periodic tests of all building structures according with Construction Law.

First test from mentioned above, confirms whether reactor concrete building has enough air tightness. This test covers tightness of concrete and all related components like gaskets and seals.

Second periodic inspection is typically dedicated to civil structures without specific requirements for nuclear buildings and additionally is even not mentioned in the ACP.

Experience from inspections and assessment as part of its regulatory oversight

As it was mentioned above concert structures do not belong to Safety Classification of SSC and also they are not mentioned in inspection procedures and inspection programs. The inspections and assessments of this issue are being conducted by PAA during regular inspections of ACP.

PAA expects from NCBJ to take into consideration concrete structures and their impact to safety during current process of Safety Classification. Results of Safety Classification should have impact on decision whether any concrete structure should be included in new AMP or not.

Strengths and weaknesses identified by Regulator

NCBJ has not put enough effort to analyse ageing impact on the concrete structures important for nuclear safety and this is the most important weakness. Without knowledge about influence of exact concrete structures on nuclear safety all actions related to ageing management are useless.

Conclusions

NCBJ is aware of their gaps in the ACP in relation to concert structures. Preparation of the new AMP NCBJ should reflect results of review of safety classification.

8. Pre-stressed concrete pressure vessels (AGR)

The MARIA research reactor as a light-water pool type reactor does not have any pre-stressed concrete pressure vessels specific for AGR reactors.

9. Overall assessment and general conclusions

PAA would like to summarize the national assessment report for 1st topical peer review. In spite of only partial compliance of ACP scope with IAEA standards and WENRA reference levels, NCBJ carries on process of ageing control focusing on this parts of SSC which are the most important for nuclear safety. This process is mainly based on operating experience which has been collected through the whole life time of operation. PAA has divided assessment process on two parts, first - general approach for ageing processes based on ACP document and the second one conducted activities related with ageing processes which are covered or not by ACP.

Regarding ACP, this document need to be reviewed and updated regularly. Lack of assigned responsibilities to apply of overall AMP caused that process is no coherent and each part of process related from owners of process is proceeds in different manners and different conditions. Absence of tools in ACP like: analysing of inspection results, trending of parameters and maintenance history, cause that actions taken by licensee are not based on data evaluation but in progress observation coming from inspections, monitoring and surveillance. The basic for preventive maintenance and remedial actions is to relay on evaluation, which also doesn't exist in current ACP. Collection and storage of data from operations without analyses of them have no use from the point of view of ageing management.

Another example of gap which causes that ACP is inefficient, is lack of screening and SSC identification methods. NCBJ's identification process is based on the operating experience which is very difficult to prove and gave opportunity to skip some weak chain link. Screening and identification of SSC methods should consist of interrelated both: safety functions with consequences of their failures and ageing degradation mechanisms with their consequences.

The strength of this part is that the NCBJ is aware of their weaknesses, areas of improvements and is willing to carry on necessarily actions to improve situation. They have experience both from ACP and maintenance program, so they can use it as the basic for establishing new AMP which will be covering all necessary issues related to ageing management.

Regarding the second part of report strictly connected with technical areas PAA has assessed that NCBJ maintains SSC on the proper safety level. Ageing management process, despite that not every single SSC is covered by ACP, is maintained on the basic level relaying on their experience and industrial standards. However NCBJ doesn't take in consideration all aspects of ageing mechanisms which exist what causes that their inspections are not fully efficient.

Due to relatively simplified construction of reactor Maria, not comparable to typical NPP design like PWR, BWR or CANDU, two chapters related to reactor pressure vessel and calandria/pressure tubes have been skipped. Subsequent chapter about concealed paperwork was also skipped due to lack of evidence which justified necessity to carry out specific ageing related activities.

Finally, despite of many gaps and weaknesses in ACP and taking into consideration all efforts which has made by NCBJ to implement new AMP in the near future and to keep SSC in good conditions, give PAA reasonable evidence that process of ageing is under control and on the sufficient level. However it doesn't mean that PAA resigned from the previous established by NCBJ program for new AMP and still claims that process is not sufficiently covered in existing documents and must be updated on the basic of self-assessment findings which has been made during peer review process and IAEA guides.

10. References

[1] P. Nowakowski, Program kontroli starzenia reaktora MARIA, Instrukcja 03-ZR-15

[2] Klasyfikacja bezpieczeństwa Systemów oraz Elementów Konstrukcji i Wyposażenia reaktora badawczego MARIA, Elżbieta Borek-Kruszewska et al

[3] Ageing Management for Research Reactors. Specific Safety Guide No. SSG-10, IAEA, 2010

[4] E. Borek-Kruszewska, Program Zapewnienia Jakości dla Obiektu Reaktora MARIA, PZJ MARIA 15

[5] Management of research reactor ageing, IAEA-TECDOC-792, IAEA, 1995

[6] K. Pytel et al, Eksploatacyjny Raport Bezpieczeństwa Reaktora MARIA, ERBM-2015

[7] J. Idzikowski, A. Hryczuk, Instrukcja obsługi i ruchu reaktora MARIA, 05 ZR 15

[8] J. Ośko, Księga Zintegrowanego Systemu Zarządzania NCBJ

[9] Ustawa z dnia 7 lipca 1994 r. Prawo budowlane, Dz.U. 2017 poz. 1332

[10] Norma PN-HD 60364 Instalacje elektryczne w obiektach budowlanych

[11] Management of life cycle and ageing at nuclear power plants: Improved I&C maintenance, IAEA-TECDOC-1402, IAEA, Vienna, 2004

[12] Assessment and management of ageing of major nuclear power plant components important to safety: In-containment instrumentation and control cables Volume I, IAEA-TECDOC-1188, International Atomic Energy Agency, Vienna, 2000

[13] Condition-monitoring techniques for electric cables used in nuclear power plants, Regulatory Guide 1.218, U.S. Nuclear Regulatory Commission, Washington, 2012

[14] Pytel K., Nowakowski P., 2015. Pomiary szczelności budynku B reaktora MARIA, Procedura 60-MT-15.

[15] Zezwolenie nr 1/2015/Maria z dnia 31 marca 2015 r., Prezes Państwowej Agencji Atomistyki

[16] PN-EN 1504 Wyroby i systemy do ochrony i naprawy konstrukcji betonowych

Annex 1 Abbreviations used in this report

Abbreviation	Meaning
ACP	Ageing Control Programme
AMP	Ageing Management Programme
DAM	The MARIA Reactor Instrumentation Section
DEJ	Nuclear Facilities Operations Department
DEM	The MARIA Reactor Electrical Section
DMM	The MARIA Reactor Mechanical Section
DOM	The MARIA Reactor Operator Section
DTR	Operation and Maintenance Manuals
EJ2	The MARIA Reactor Operation Division
NAR	The National Assessment Report
PAA	National Atomic Energy Agency (nuclear regulatory authority)
PZJ	Quality Assurance Programme
R&D	Research and Development
R2-B	The MARIA Reactor Building
RG	Main Switching Station
RP	Auxiliary Switching Station
SAR	Safety Analysis Report
SSC	Systems, Structures and Components
SZJ- RM	Quality Assurance Specialist of the MARIA Reactor
WNEP	Fuel Elements Leaks Detection