EU TOPICAL PEER REVIEW

AGEING MANAGEMENT ASSESSMENT OF NPP

NATIONAL REPORT

REPUBLIC OF BULGARIA

Nuclear Regulatory Agency

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1. OVERVIEW

1.1. Identification of the Nuclear Facilities

Kozloduy NPP Plc is a public limited company. Its Registered Office address is: Kozloduy 3321, Kozloduy Municipality.

A part of the Kozloduy NPP Plc business activity is to use nuclear energy for generation of electricity. This activity is based on the current operating licences of the units as well as the licence for electricity and heat generation issued by the Bulgarian Nuclear Regulatory Agency (NRA) and the Energy and Water Regulatory Commission (EWRC) respectively.

The National Assessment Report on Ageing Management is focused on the units in operation of Kozloduy NPP, which are units 6 and 6. The Kozloduy NPP Units 5 and 6 consist of WWER-1000 reactors, Model B-320, enveloped in a leak-tight reinforced concrete protective structures (containments), and turbine generators (steam turbines К 1000/60 1500 2 and electricity generators ТВВ 1000 4У3).

Units 5 and 6 were commissioned in 1987 and 1991 respectively and their design lives expire in 2017 for Unit 5 and 2021 for Unit 6. Upon completion of a review and analyses as well as implementation of corrective actions included in the Plant Lifetime Extension (PLEX) Programme, in November 2017, the Operating Licence of Unit 5 was renewed for another 10 years. Analogous activities are being implemented on Unit 6 too. The Units' key parameters are presented in Table 1.

Table 1. Key parameters of the Kozloduy NPP Units 5 and 6

Parameter	Unit 5	Unit 6
Licensee	Kozloduy NPP Plc	Kozloduy NPP Plc
Reactor type	WWER-1000, Model B-320	WWER-1000, Model B-320
Installed capacity	1000 MWe	1000 MWe
Year of commissioning	1987	1991

1.2. Process of preparation of the National Assessment Report

This National Report was prepared by the NRA based on a self-assessment conducted by the Licensee and was approved through a Council of Ministers Decree. The Report was prepared in compliance with the Technical Specification drawn up by WENRA and approved by ENSREG in late December 2016. The structure of the Report corresponds to Annex 1 of the Specification. The information presented concerns the nuclear facilities specified in Clause 1.1 and covers the regulatory requirements and activity as well as the Licensee activity in respect of Ageing Management.

2. OVERALL AGEING MANAGEMENT PROGRAMME REQUIREMENTS

2.1. National Regulatory Framework

The state control over the safe use of nuclear energy and ionizing radiation, and radioactive waste and nuclear fuel safe management is performed by the Bulgarian Nuclear Regulatory Agency (NRA) which is an independent specialized executive body.

The requirements of the national legislation as regards NPP ageing management are specified in the following documents:

- Act on the Safe Use of Nuclear Energy;
- Regulation on Ensuring the Safety of Nuclear Power Plants;
- Regulation on the Procedure for Issuing Licenses and Permits for Safe Use of Nuclear Energy

Additional instructions regarding the implementation of the requirements contained in the regulatory documents and relevant to the ageing management process are contained in the following regulatory guides:

- Guide on Implementing Periodic Safety Review of Nuclear Power Plants;
- Guide on Safe Operation of Nuclear Power Plants

The Act on the Safe Use of Nuclear Energy(ASUNE) is the basic law in the field of nuclear energy use, which regulates the public relationships as regards the state regulation of safe use of nuclear energy and ionizing radiation, and safe management of radioactive waste and spent nuclear fuel. The Act on the Safe Use of Nuclear Energy defines the national framework for state regulation of the activities in the field of nuclear energy use. The law determines an authorization regime for the regulation of the activities.

The Regulation on Ensuring the Safety of Nuclear Power Plants defines the main criteria and rules for nuclear safety and for radiation protection, as well as the organizational arrangements and technical requirements to ensure safety during siting, design, construction, commissioning and operation. Section 3 from Chapter 8 of the Regulation requires that during commissioning and operation should be developed and implemented programmes for maintenance, testing, surveillance and inspections, targeted at ensuring compliance of operability, reliability and functionality of structures, systems and components (SSCs) that are important to safety, with the design requirements throughout the whole operating lifetime of the nuclear power plant. These programmes take into consideration the operational limits and conditions and are revised taking into account the operating experience. The predictive, preventive and corrective maintenance programmes comprise activities to control the degradation to prevent failures and to restore operability and reliability of SSCs important to safety. The maintenance programmes take into account the ageing management programme results and include replacement of obsolete SSCs or SSCs whose lifetime has expired, as well as the re-qualification of the safety related SSCs and the use of new maintenance technologies.

Art. 228 of the Regulation requires that the operating organization should develop, implement, evaluate and improve the ageing management programme in order to identify all the safety related SSCs ageing mechanisms, to define their potential consequences, as well as the measures needed to maintain their operability and reliability. All SSCs important to safety shall be assessed against the envisaged safety functions during their design life, taking into account the ageing and wearing mechanisms, as well as possible degradation related to the time of their operation and the load cycles. The environmental conditions, parameters of technological processes, maintenance and testing schedules, and the safety related SSCs replacement strategy shall be considered as factors during ageing management. Control, testing, sampling and inspections shall be implemented in order to assess the ageing effects and to detect unexpected behaviour of degradation at an early stage.

The Regulation on the Procedure for Issuing Licences and Permits for the Safe Use of Nuclear Energy defines the procedure for issuing licenses and permits for performing activities regulated by ASUNE. Art. 23 therein defines the required documents which should be submitted in relation to the license renewal. One of these documents is an Updated Safety Analysis Report for the nuclear facility. In its essence, this report is a periodic safety review of the nuclear facility. Art. 83 therein defines the necessary minimum number of safety factors (16) which should be taken into account when performing the periodic safety review; among these factors is the "ageing management" factor.

The Guide on Implementing Periodic Safety Review of Nuclear Power Plants provides detailed instructions on the process of conducting the periodic safety review. The "ageing management" factor is also considered in detail. The purpose of the ageing management review is to determine whether the ageing aspects having an impact on the SSCs important to safety, are properly managed, whether there is an effective ageing management programme, and whether all the required safety functions for the design lifetime are ensured. The review comprises the ageing management programme, and the following aspects in particular: timely detection and mitigation of the ageing mechanisms and effects; completeness of the programme; efficiency of operating and maintenance policies and/or ageing management procedures for replaceable components; assessment and documenting of ageing related potential degradation, which could impact the safety related SSCs safety functions; ageing effects management for SSCs necessary for safety during decommissioning; performance efficiency indicators; accountability. The review assesses the following technical aspects: ageing management methodology; licensee's understanding of the dominant ageing mechanisms and phenomena, including information on the actual safety margins; acceptance criteria and required safety margins of SSCs important to safety; methods for ageing control and ageing effects mitigation; etc.

2.2. International standards

The ageing management approach adopted by Kozloduy NPP Plc is directly related to the established organization for lifetime management, while constantly updating the programmes and practices related to the process. The up-to-date requirements of the relevant regulatory framework of the Republic of Bulgaria, the IAEA safety standards, and the applicable documents of the supplier are introduced for the maintenance, non-destructive control, inspections, surveillance and equipment qualification activities. The goal of the current operating practices is to adequately comply with the national regulatory documents, observing the relevant international safety standards, to ensure the plant safety in the process of long-term operation.

The basic internationally adopted standards, guides and other documents, used in the development of an Ageing Management System are, as follows:

- IAEA, Ageing Management for Nuclear Power Plants, Safety Standards Series Safety Guide No. NS-G-2.12, IAEA, Vienna (2009);
- IAEA, Safe Long Term Operation of Nuclear Power Plants, Safety Reports Series No. 57, IAEA, Vienna (2008);
- IAEA, Methodology for the Management of Ageing of Nuclear Power Plants Components Important to Safety, Technical Reports Series No. 338, IAEA, Vienna (1992);
- IAEA, Data Collection and Record Keeping for the Management of Nuclear Power Plants Ageing, Safety Reports Series No. 50-P-3, IAEA, Vienna (1991);
- IAEA, Implementation and Review of a Nuclear Power Plant Ageing Management Programme, Safety Reports Series No. 15, IAEA, IAEA, Vienna (1999);
- IAEA, Equipment Qualification in Operational Nuclear Power Plants: Upgrading, Preserving and Reviewing, Safety Report Series No. 3, IAEA, Vienna (1998);
- IAEA-EBP-SALTO‘ Final Report of the Programme on Safety Aspects of Long Term Operation of Water Moderated Reactors, Recommendations on the Scope and Content of Programmes for Safe Long Term Operation, Vienna, (2007);
- IAEA, Plant Life Management for Long Term Operation of Light Water Reactors, Technical Reports Series No. 448, IAEA, Vienna (2006);
- IAEA, Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL), Safety Report Series No. 82, IAEA, Vienna (2015);

IAEA, Maintenance, Surveillance and In-service Inspection of Nuclear Power Plants, Safety Standards Series Safety Guide No. NS-G-2.6, IAEA, Vienna (2002);

- IAEA, Periodic Safety Review of Nuclear Power Plants, Safety Standards Series No. SSG-25, IAEA, Vienna (2013);
- IAEA, A system for the Feedback of Experience from Events in Nuclear Installations, Safety Guide No. NS-G-2.11, IAEA, Vienna (2006);
- WENRA Report, Safety Reference Levels for Existing Reactors, 2014

2.3. Description of the Overall Ageing Management Programme

The Programme ensures integration of the controlled documents in force from Kozloduy NPP Plc Management System, that are related to the implementation of lifetime and ageing management activities. The interrelations between the programmes, activities and responsibilities regarding the control, monitoring and minimization of the degradation effects for SSCs covered by the Programme, are defined in detail.

An Equipment Ageing Management Plan was developed and put into effect in 2003. The Plan defines the required actions for the implementation of the recommendations resulting from conducted reviews and assessments, as well as the deadlines and the financial resources ensuring the lifetime, functionality and reliability of the equipment.

The preparation for the development of Units 5 and 6 Lifetime Management Programme (the PROGRAMME) started simultaneously with the implementation of the Plan. The PROGRAMME was put into effect in 2007; it defines the ageing management activities.

The goal of the PROGRAMME is to assure that the major ageing mechanisms and effects thereof shall be identified and reduced to ensure safe and reliable operation. The implementation of the PROGRAMME is one of the prerequisites for obtaining a license for operation of each unit, in compliance with the requirements of the Regulation on the Procedure for Issuing Licences and Permits for the Safe Use of Nuclear Energy.

The following main ageing management activities are defined in the PROGRAMME:

- screening of structures, systems and components (SSCs) at the NPP, which have a significant impact on safety;
- defining of dominant ageing mechanisms of the screened SSCs and developing of effective practical methods for monitoring and slowdown of the ageing process;
- using effective methods for design, manufacturing, storage, installing, control and operation.

The PROGRAMME features a systematic approach that comprises activities, related to the degradation check (non-destructive control, surveillance, diagnostics, assessments, etc.), correcting unacceptable degradations (repair, replacement, etc.), minimizing degradation effects (operation in compliance with the operating procedures and technical requirements, water chemistry, etc.), and improving ageing management programmes effectiveness. All the ageing management activities are implemented according to approved procedures and programmes.

As a result of the conducted periodic safety reviews in 2009 and 2016, the internal programmes in place for preventive maintenance, non-destructive control and inspections, surveillance and equipment qualification have been updated, and they are covered by the PROGRAMME.

2.3.1. PROGRAMME scope

2.3.1.1. Allocation of responsibilities within the licensee organization to ensure the development and the implementation of the PROGRAMME:

A special structural unit has been set up within Kozloduy NPP Plc structure. Its main functions are related to the development and maintaining of the PROGRAMME. The allocation of the functions and the tasks related to the implementation of the ageing management activities is specified in the rules of organization and operation of the respective structural units and subunits.

The main ageing management activities are specified in the PROGRAMME.

The specific responsibilities for the implementation of each activity, specified in the PROGRAMME, have been allocated in the existing programmes for preventive maintenance, non-destructive control and inspections, surveillance and equipment qualification, water chemistry control.

2.3.1.2. Methods used for identification of SSCs within the PROGRAMME scope:

The requirements to the process of screening of SSCs covered by the PROGRAMME have been developed on the basis of the IAEA recommendation to the NPP ageing management adjusted to the ageing management objectives at Kozloduy NPP, in accordance with the following documents:

- IAEA, Ageing Management for Nuclear Power Plants, Safety Standards Series Safety Guide No. NS-G-2.12, IAEA,Vienna (2009)
- IAEA, Methodology for the Management of Ageing of Nuclear Power Plant Components Important to Safety, Technical Reports Series No.338, IAEA, Vienna (1992);

An approach has been adopted to define the SSCs within the PROGRAMME scope, according to which the PROGRAMME comprises the respective SSCs related to safety assurance. SSCs safety classification is considered during the screening.

The PROGRAMME scope covers the following main groups of SSCs:

- SSCs important to safety that ensure the fulfilment of the safety functions in all anticipated operating events and accident conditions;

- SSCs for normal operation, whose failure could prevent the fulfilment of safety functions by SSCs important to safety;
- Other SSCs for which safety analyses have been performed, and the latter have been included in the Safety Analysis Report:
  - SSCs that are necessary to manage or mitigate the consequences of: transients, accompanied by anticipated transients without scram (ATWS); station blackout (SBO) and pressurized thermal shock (PTS).
  - SSCs required to deal with internal floods (FP);
  - SSCs that have environmental qualification (EQ) for operation at anticipated operational conditions and accident conditions.
- SSCs that perform functions to ensure plant availability and reliable electricity generation.

When screening the civil structures only those that could limit the operational lifetime of the power plant were selected. Structures that could be easily replaced, or whose characteristics could be easily restored, are not covered by the ageing management programme. They are subject to other programmes.

2.3.1.3. Methods for SSCs grouping in the screening process

For the purposes of the ageing management and the review made within the comprehensive assessment, related to the units lifetime extension, the SSCs have been allocated in the following functional groups:

- mechanical equipment;
- electrical and I&C equipment;
- civil structures.

These SSCs have been assessed for their real condition and the residual lifetime. Each group is subdivided into sub-groups. The components from the first two groups have been allocated in subgroups (9 subgroups for mechanical equipment and 3 for electrical and I&C equipment). As for the civil structures, they are allocated in 2 subgroups.

2.3.1.4. Methodology and requirements for assessment of existing maintenance practices and development of the ageing management programmes according to the identified significant ageing mechanisms;

The methodological requirements for assessment of the existing maintenance practices are specified in procedures determining the performance of the following activities:

Preventive maintenance

The preventive maintenance procedure for structures and components from the technological systems specifies:

- the activities/processes, responsibilities and deadlines for planning of preventive maintenance;
- modifications, preparation, ensuring and documenting of the implementation;
- the analysis of the maintenance, repairs, inspections and control data;
- the activities, responsibilities and deadlines for discussing, approving and implementation of the recommendations (remedial actions) from the trends analyses;
- the process of the assessment of the effectiveness of the remedial actions implementation.

Predictive maintenance

The predictive maintenance procedure defines the activities, responsibilities, planning, implementation and analysis of the results. Predictive maintenance is performed mainly according to recommendations resulting from the SSCs condition diagnostics control.

Implementation of emergency (corrective) maintenance

The activities, responsibilities and deadlines for maintenance implementation are defined in an emergency (corrective) maintenance procedure.

The preventive maintenance activities take into consideration the ageing effects and mechanisms and are targeted at minimization of the material degradation.

With the purpose of control and mitigation of the SSCs ageing effects, ageing management programmes are developed and implemented within the general PROGRAMME for the SSCs, as follows:

- reactor pressure vessels;
- steam generators;
- primary circuit pipelines;
- pressurizer;
- reactor coolant pumps;
- diesel generator station (DGS);
- accumulator batteries;
- unit transformers and house load transformers;
- reactor internals (RI);
- main heat-exchange equipment;
- buried pipework from the service water system (QF);
- hydraulic shock absorbers;
- cables and cable facilities;
- civil structures of Kozloduy NPP;
- units 5 and 6 containments;

- hydraulic engineering facilities for service water supply (HEF);

The ageing management programmes are improved based on the analyses of the maintenance and the prescribed remedial actions in case of acceptance criteria breach, as well as the consideration of the operating experience and the scientific research in that field.

2.3.1.5. Quality assurance of the PROGRAMME:

2.3.1.5.1. Gathering and storage of data and information on the trends in maintenance history and operational data;

An Integrated Management Systems is in place, which has been established in compliance with:

- IAEA, The Management System for Facilities and Activities, Safety Requirements, GS-R-3, IAEA (2006), replaced by IAEA, Safety of Nuclear Power Plants: Leadership and Management for Safety, General Safety Requirements No. GSR Part 2, IAEA, Vienna (2016);
- Quality management systems. BDS EN ISO 9001:2008/2015 requirements;
- Environmental management system. Requirements and instructions for implementation, ISO 14001:2004/2015;
- Health and safety at work management systems, OHSAS 18001:2007.

The integrated system is described in Management System Manual. The system was built on a process principle taking into account the specifics of Kozloduy NPP and the processes in place.

Document management at Kozloduy NPP is structured in an auxiliary process *Documents and records management*, covering the activities for the management of the controlled documents and records throughout their lifetime.

The documents and records management activities are specified in the procedures for the management of documents, records and archive documents, databases operation procedures, etc.

The following documents are being collected and archived since Kozloduy NPP commissioning:

- designer and manufacturer’s documentation;
- regulatory documents, on whose basis the plant design was produced and implemented;
- operational records;
- reports and other technical documents and correspondence on performed installation, start-up and adjustment, and maintenance works;
- documents on implemented technical solutions, which introduced design modifications;

- documents with results from performed non-destructive in-service control of basic metal, built-up surfaces and welded joints;
- documents from other activities providing impartial information for performing periodic assessments and documenting of potential degradation mechanisms, which could have an impact on the SSCs safety functions.

Besides, records are documented and stored regarding:

- process parameters (characteristics) of the major and auxiliary equipment and systems;
- major equipment worked-out hours;
- technical and economic indicators;
- manifested SSCs defects and failures;
- documents from events analyses;
- documents proving planned and implemented preventive maintenance of SSCs over the years, in-service non-destructive control, geodesic control and measurements, planned and implemented schedules of technical surveillance (performed by the state and the plant);
- engineering chemistry working programmes;
- programmes and certificates of functional and complex tests of the operation and the engineering support teams, and
- other documents providing information for assessment and documenting of potential degradation mechanisms, which could have an impact on the SSCs safety functions.

Documents that are filed in the central archive offices:

- all documents describing the way of implementation of the project management, functional operation and engineering support activities;
- the documents containing the design description (Safety Analysis Report, Probabilistic Safety Analysis, and Technical Specifications);
- manufacturer's documentation of the materials used in the production of the supplied equipment;
- reporting documents on maintenance, equipment and facilities modification (certificates, records, etc.).

Kozloduy NPP uses also electronic information systems and databases through which the process of data collection and records storage is implemented.

The data collection and records storage system that is in place provides diverse opportunities for:

- trending of the SSCs characteristics degradation;

- predicting the functioning of each component and its operating life;
- identification and assessments of degradation;
- failures and non-compliances caused by ageing.

2.3.1.5.2. Indicators used for effectiveness assessment.

Kozloduy NPP Plc Integrated Management System contains separate subprocesses "SSCs lifetime management" and "Qualification management of SSCs important to safety".

The requirements, main principles, responsibilities and obligations regarding Kozloduy NPP effectiveness assessment are defined in the Quality Assurance Rules. System of self-assessment indicators for effective management of Kozloduy NPP Plc is applied.

The programme's effectiveness assessment is performed using the Self-assessment Indicators System. A part of this System are the summarized, functional and specific indicators reflecting short-term and long-term tasks and the activities being performed. Part Effective Operation of the Self-assessment Indicators System defines the following functional indicators related to the lifetime management:

- Effectiveness of maintenance;
- Major equipment lifetime management;
- Frequency of performance deviations of major equipment;
- Operability of safety systems;

Functional indicator "Major equipment lifetime management" consists of 5 specific indicators, as follows:

- Design modes impacting the major equipment lifetime;
- Safety systems SSCs runtime;
- Safety systems SSCs number of cycles;
- Chemistry Performance Indicator;
- Failures of qualified equipment under the Safe Shutdown Equipment List (SSEL).

2.3.2. Ageing assessment

<u>2.3.2.1. Main standards and guides, as well as key design, manufacturing and operational documents used to prepare the PROGRAMME</u>

The principal regulatory documents, standards and guidelines as described in items 2.1, 2.2 and 2.3 herewith have been used for the development of the PROGRAMME.

The requirements for ensuring reliable equipment operation provided in the manufacturer's documents and the operational documentation have been taken into account at the ageing management review. The operational history, failure history, water chemistry, maintenance, as well as the information regarding start-up trials and commissioning of equipment are taken into account.

2.3.2.2. Main elements used in the NPP programmes for ageing assessment

For the purposes of effective ageing management, in compliance with the pointed IAEA guideline documents, each ageing management programme should possess the following nine attributes (elements):

1. Scope of the AMP based on understanding ageing;
2. Preventive actions to minimize and control ageing degradation;
3. Detection of ageing effects;
4. Monitoring and trending of ageing effects;
5. Mitigating ageing effects;
6. Acceptance criteria (admissibility);
7. Remedial actions;
8. Operating experience feedback and feedback of research and development results;
9. Quality management and administrative control;

The analysis of the compliance of Kozloduy NPP programmes that are in place in the areas of: maintenance, equipment qualification, in-service control, surveillance and monitoring and water chemistry control regarding the reactor pressure vessel, electric cables, buried pipelines and civil structures, with the above attributes, is presented in sections 3-7 of this National report.

2.3.2.3. Processes/procedures to determine the ageing mechanisms and the possible consequences thereof

During the effective ageing management of SSCs important to safety, it is necessary to determine the ageing mechanisms and the critical sections of components taking into account the type of materials, the condition and the impact of the stressors.

The following approach is applied to determine the ageing mechanisms for SSCs covered by the PROGRAMME:

- Gathering input data
- Information analysis
- Determining potential and dominant ageing mechanisms

The information that is used as input data to determine the SSCs ageing mechanisms is derived from the following sources:

- design documentation, equipment certificates, etc.;
- technical specifications, manufacturer's operating instructions for the equipment and the organization of the production, process diagrams and technical descriptions, and other kind of documentation;
- documentation regarding design modifications, replacements (designs, solutions, technical solutions);

- reporting documentation on maintenance (incl. in-service control, tests, verification and trials);
- reporting documentation on water chemistry control of media and corrosion condition of elements;
- reports on deviations in the equipment operation;
- available databases on reliability, defects, maintenance/repairs, incl. for the whole period of operation;
- documentation regarding the report on spent lifetime, load cycles of the equipment;
- reporting documentation on previously performed assessments and analyses

Based on the received information on the actual condition of the SSCs, an analysis is performed of the design documentation and the history of their operation, the in-service control results, tests and trials, performed in compliance with the requirements of the current technical documentation.

The potential ageing mechanisms are determined for each SSC according to preliminary assigned lists. Based on the analysis of the received information and on expert evaluations, the dominant (among the detected) ageing mechanisms are determined, that have been manifested for the specific facilities, as well as the critical locations.

2.3.2.4. Identification of acceptance criteria related to ageing

The acceptance criteria are determined for each component or equipment of the same type (commodity group), and are related to the degradation mechanisms in connection with the performance of the design function by the respective SSC.

The acceptance criteria are expressed as limit value for the monitored ageing parameter, including the required safety margin.

The acceptance criteria are determined in compliance with the design calculations results, regulatory requirements or results from calculations and analyses performed based on the gathered operating experience.

The specific acceptance criteria for each of the SSCs considered in the report are specified in the respective parts of this report.

2.3.2.5. Using the scientific and research programmes

Kozloduy NPP hires various scientific and research institutes and organizations during the implementation of the ageing management activities and defining of lifetime of SSCs covered by the PROGRAMME. Relevant information is contained in items 3, 4, 5 and 7 herewith.

2.3.2.6. Using internal and external operating experience

The Regulation on Ensuring Safety of Nuclear Power Plants requires development and systematic application of a programme to collect, analyse, and document the internal and external operating experience.

The operating experience feedback process at Kozloduy NPP is described and specified in hierarchically organized documents. The underlying document that regulates the process is *Safety rules. Operating experience feedback system.*

The procedure for obtaining, evaluation and distribution of the operating experience information is defined in *Procedure for use and distribution of operating experience.* The procedure defines the activities and the responsibilities regarding the review and the evaluation of internal operating experience, as well as the distribution and the implementation of the operating experience of foreign NPPs and industrial organizations, amongst Kozloduy NPP staff. The procedure covers the activities for sharing operating experience with the members of the World Association of Nuclear Operators (WANO) and the International Atomic Energy Agency (IAEA). It also covers the process of applicability assessment and use of information from other sources - publications, topical workshops, reports from business trips abroad, periodic reports and analyses, work practices, proposals by the personnel, etc. The procedure defines the work of the permanent Committee for review and evaluation of operating experience, which gathers at least once per month.

Gathering, evaluation, classification and the records on internal events and failures is implemented in accordance with:

- Methodology for analysis of events and operating experience;
- Procedure for reporting and analysis of events at Kozloduy NPP Plc - specifies the procedure and the responsibilities for reporting, analysis of events, and assignment and monitoring of the implementation of remedial actions;
- Procedure for reporting and analysis of low-level events and near misses - specifies the activities related to the implementation of systematic approach for identifying and reducing latent weaknesses (low-level events and near misses) before they lead to serious deviations, events, or incidents, as an effective method for increasing safety and for saving considerable financial resources;

The Operating Experience Council - which is a consultative body with the Production Director, makes the final approval of the corrective actions resulting from external operating experience. The Council's work is specified in the Rules for the activity of the Operating Experience Council.

Kozloduy NPP performs periodic review of the efficiency of the external experience utilization process based on preliminary set objectives and criteria. The main goal of the use of external operating experience is to prevent recurrence of safety-significant events, which have already been registered and analysed by other nuclear power plants.

The review of the process of utilization of external operating experience is performed once a year by the Committee on review and evaluation of operating experience, preparing a report in compliance with the requirements of an existing Procedure for utilization and dissemination of operating experience. The report is subject to review by the Operating Experience Council.

2.3.3. Monitoring, testing, sampling and inspection

Programmes comprising monitoring, testing, sampling and inspection activities are approved and implemented at Kozloduy NPP.

2.3.3.1. Programme for monitoring of the condition indicators, parameters and trending

Diagnostic systems.

A procedure for SSCs monitoring, analysis and assessment is implemented within the predictive maintenance system. It defines the principles, scope, responsibilities and the reporting on the activities regarding the monitored facilities, used methods for technical condition diagnostics, way of gathering and storing of information, methods for analysis, conclusions and recommendations, as well as the control over the quantity and quality of the performed analyses; the requirement that the results from the SSCs diagnostic through the regular systems, fixed and mobile devices and equipment, should be documented, systematized and analysed.

The monitoring of the condition and the working conditions of the SSCs covers the following engineered features:

- Computer information system OVATION visualized the information, provided by an unified information system. This includes current information on the unit equipment condition, the database of the unified information system, access to archived data and information, provided by the applied programmes. The main applied programmes are:
    - generator monitoring - monitoring and determining of maximum and minimum temperatures of the generator stator, rotor and coolers. Calculating the hydrogen and nitrogen leakage degree and the hydrogen absolute humidity degree;
    - motors monitoring - provides the user with information on the condition of the selected operational characteristics of the particular motor, for example the number of motor start-ups and accumulated operation time of the motor. For selected motors, it accumulates hot start-ups, cold start-ups, start-ups under load, idle run start-ups and start-up attempts;

- protections and interlocks control programme - automatically monitors the operational characteristics, in compliance with the process activation conditions of the discrete devices and components in the plant protection and control system equipment. Indicates to the operator the monitored components of the plant equipment, which are in an alarm condition, if the component is foreseen to be in abnormal condition;
- turbine monitoring programme - it monitors the turbine heating and thermal conditions. The heating of the main steam line, heating of turbine stop valves and the temperature differences in the turbo-generator body are monitored during start-up and operation of the turbo-generator;
- flow and level adjustment programme - it calculates the adjustment coefficients for the determined flows and levels, and after that applies these coefficients to the flow and level measurements;
- warm-up/cool-down programme - it performs calculations on primary circuit heat removal change rate, and compares these rates with the limit values;
- back-up measurements programme - it calculates the mean value from a certain group of signals, determines the best mean value from the back-up sensors, ignoring the sensor with incorrect indications;
- load cycles automatic reporting programme - it monitors the work of a set of switches and valves. Operational characteristics of the components are calculated, such as number of switched on, open, total number of cycles and total operation time;
- unit mode programme - it monitors preliminary defined groups of selected parameters of the plant to determine the mode and sub-mode of unit operation.

- Loose Parts Monitoring System (KŰS). Computer information system to monitor and localise loose parts and/or loose mechanical joints on important elements of the primary circuit equipment - reactor, main coolant pipelines, main coolant pumps, steam generators. The shocks or impacts from loose parts on the internal surface and the primary circuit facilities cause single sound events. These sounds are reflected as structural noises on the primary circuit components and are registered by accelerometers which are attached to the primary circuit external surface.
- Leak Before Break System(FLUS). Computer information system for early detection and localization of leaks on important elements from the primary circuit equipment - reactor (upper unit), main coolant pipelines, pressurizer, specific areas in the containment. The system measures the level of humidity over a local area through a system of sensors and lines which take the air flow to them.

- Fatigue Monitoring System (FAMOS). Computer information system for constant monitoring of the temperature regime of preliminary determined primary circuit equipment pipelines. The system gathers and stores the measured process values on-line, periodically recording data.
- Monitoring system of steam and gas mixture (coolant level) occurrence in the reactor pressure vessel - ensures monitoring of the following parameters:
  - presence or absence of liquid phase coolant discretely in three levels (level indicator monitoring spots) along the reactor height;
  - temperature at the fuel assembly outlet and in the upper part below the reactor head;
  - temperatures in reactor accident condition (up to 1200°C).
- Reactor protection system against cold overpressurization - its purpose is to limit the consequence from inadvertent activation of pressure increasing devices through switching off pressure increase systems components, and through closing feeding lines, respectively. The system was designed to limit the excessive pressure increase as a result of water feeding and/or non-removal of decay heat generated in the primary circuit during unit start-up and shut-down and at abnormal operations.
- Internal containment pre-stress control system of Kozloduy NPP Units 5 and 6 - designed to visualize on-line, to record, document and process data on the tendons pre-stress by measuring the relevant forces in six separate wires in one tendon, with the purpose of obtaining primary information (in real time) on the containment.
- System for technical diagnosis of primary circuit main coolant pumps (ГЦН-195М). The automated system is designed to determine the current technical condition of the reactor plants main coolant pumps through processing of the data from the heat-exchange and vibration control of the pumps.
- Information system for primary and secondary circuit water chemistry parameters monitoring – it is designed to ensure automatic measurement, recording of the coolant physical and chemical indicators and their visualization on the operative personnel work stations.
- Periodic monitoring of the technical condition of electric motors 6 and 0.4 kV under load. The monitoring is performed by measuring the electric parameters, start-up and work characteristics in asynchronous motors 6 kV and 0.4 kV in operation and under load, by the means of instrument, using clamp meters and voltage measurement probes.
- Aggregates vibration condition monitoring system (INTELLINOVA). The system is designed for constant on-line monitoring of the bearings condition and the vibrations

intensity of almost 100 secondary circuit aggregates, ventilation systems in the containment and primary circuit emergency cooling pumps.

- Vibration monitoring of more important SSCs from Reactor Building, Turbine Hall and Auxiliary Building 3, and the main steam pipelines of Units 5 and 6, which belong to the safety systems, systems important to safety and normal operation systems. The assessment of the vibration condition is performed at nominal operation modes. The system monitors vibration displacement, vibration velocity and vibration acceleration, which allows for a high quality assessment of the technical conditions of the machines and diagnostics of defects and the degree of their impact in the process of operation.
- Thermo-visual testing. The equipment pre-outage diagnostics arrangements include thermo-visual testing of the facilities. The scope of this testing covers the main electric equipment, such as unit transformers, houseload transformers, backup transformers, turbine generators and exciter, surge arresters, cable heads, busducts 6 kV and 0.4 kV, generator breaker, electric motors, transformers 6/0.4 kV, switchboards 0.4 kV, etc. The testing is performed in accordance with the general list of monitored equipment, as well as specific lists that are produced before unit outage.
- Noise monitoring of reactor internals. The noise diagnostics applications are used through the existing neutron flux density measurement chambers, which secure additional useful information on the reactor internals vibrations. The noise sources in the neutron flux signal can be separated and subject to correlation analysis in order to determine the causes. Experimental data obtained at different unit thermal power levels and at different number of effective days are considered.

The listed diagnostic systems monitor the degree of impact of ageing with the aid of periodic calculations of accumulated damages, as well as by using the increments of the partial values of the accumulated damages or calculations on the basis of records of measurement of operational parameters.

The systems monitor the defined parameters whose values serve as input data to calculate the accumulated fatigue damages and perform periodic calculations of the accumulated fatigue damages in the points for assessment. This includes both the parameters measured through the process measurements systems, and the data from the operational transients' history.

Based on results from performed assessments, already recorded, the respective software programmes forecast damages in the observation points, or provide information for performing expert assessments.

2.3.3.2. Inspection programmes

During plant operation, monitoring is carried out to identify possible deterioration of the status of the SSCs in order to assess if it is acceptable to continue to operate safely or there is a need of taking up recovery measures. Special attention is paid to the primary and secondary circuits.

Established were baseline databases (manufacturer's passport data) plus additional ones, obtained during installation and receiving inspection of equipment. These data provide the basis for comparison with the data from follow-up inspections. The scope of the in-service inspection includes systems and components selected in accordance with their importance for safety:

- Parts of components, subject to the impact of the coolant pressure;
- Components of the primary circuit system or associated components that are important to provide for the safe shutdown of the reactor and for the nuclear fuel cooling in accordance with the normal operation conditions and during postulated accidents;
- Other components, such as a main steam pipeline or feedwater pipelines, the displacement or failure of which may endanger the system.

The systems and components operating under pressure are subjected to:

- Leak rate and strength tests of the systems;
- Reactor coolant pressure boundary leak test before of the reactor after outage, when the first circuit leak-tight integrity has been compromised;

The periodical monitoring during operation of units 5 and 6 ensures that any deterioration in the condition of the most loaded component will be identified before a failure occurs.

In-service Inspection Work Programmes

The work programmes for in-service inspection of the equipment and the pipelines of the first and the second circuits are an integral part of the documents for outages preparation. The terms for preparation of the work programmes are stipulated in the preventive maintenance procedures for the components of the process systems. The In-service Inspection of SSCs is planned on the grounds of the Procedure for In-service Inspection of Base Metal, Welded on Surfaces and Welded Joints of Equipment and Piping of WWER-1000 units at Kozloduy NPP.

The programs are prepared and agreed by the responsible structural units of Kozloduy NPP and submitted for agreement to the NRA before the beginning of the specific outage under the licensing conditions for operation of Units 5 and 6.

Based on previous years' results and the accumulated operating experience, the scope of inspection in the work programs could be increased compared to the one stated in the Procedure for In-service Inspection of Base Metal, Welded on Surfaces and Welded Joints of Equipment and Piping of WWER-1000 units at Kozloduy NPP, on the grounds of a decision of a Technical council and /or upon request by the NRA.

The scope, frequency, regulatory and technical documentation and the method of inspection for each individual facility and each assembly are defined in the in-service inspection procedures. Conducting in-service inspection under the different methods is carried out according to approved and standardised (or validated) inspection procedures and methodologies. These documents contain sections describing how to identify imperfections in the metal, classify them against a standard and assess their acceptance according to the limits. The follow-up action, after identified and reported imperfections, is to take decision at a Technical council session to replace (repair) the assembly, or to increase the scope of inspection. The activities themselves are stipulated in the In-service inspection procedure.

The inspection frequency is determined in accordance with the requirements of the following documents: Rules for the Arrangement and Safe Operation of Equipment and Pipelines of Nuclear Power Plants. Moscow, Energoatomizdat (ПНАЭ Г-7-008-89) and Russian Federal Norms and Rules in the Areas of Usage of Nuclear Energy, Fundamentals of Ensuring Safety of Nuclear Power Plants (НП-001-15). The inspection frequency vary depending on the active degradation mechanisms. Frequency of inspection of components with registered discontinuities shall be determined by a decision of a Technical council in accordance with the In-service inspection procedure.

Trends are determined by performing a comparative analysis of the condition of the metal of equipment and pipelines, based on the results obtained under current and prior periodic inspections. Comparative analyses and conclusions on the discontinuities development are attached to the implementation reports of the inspection work programmes.

The acceptance criteria for the parameters monitored during the inspection are defined in the regulatory documents referenced in the In-service inspection procedure.

When registering unacceptable discontinuities (defects) in accordance with the evaluation limits established in the regulatory documents, additional inspection is conducted to determine the parameters of the defect, and analysis of the causes that led to the emergence (increase) of the defect is performed. The Technical Council takes the decisions on removal of the established causes (replacement of equipment, change of operating mode, maintenance, etc.) and determines the possibility for further operation of the equipment and piping, based on:

- the documents reflecting the results of the inspection of the condition of the metal, in the area of the discovered discontinuities, provided by the organization, performing inspection;
- report for a defective assembly analysis;
- other available documents - quality certificates, strength assessments, corrosion survey assessment, etc.

Inspected sections of equipment with registered discontinuities, allowed to operation without maintenance, are inspected during the next outage in a 100% scope with the same methods that they have been discovered with. The subsequent scope and periodicity of the inspection shall be determined by a decision of a Technical council in accordance with the documents in force at Kozloduy NPP.

The Inspection is performed by the accredited body Inspection Centre type C - Diagnostics and Control Centre.

A system for qualification of non-destructive testing is set up and functions at Kozloduy NPP, in accordance with the requirements of the Methodology for Qualification of In-service Inspection Systems of WWER Nuclear Power Plants (IAEA-EBR-WWER-11, Methodology for Qualification of ISI Systems for WWER Nuclear Power Plants, March 1998).

After an outage, the reports on the ISI performed are prepared and submitted to NRA.

2.3.3.3. Surveillance Programmes

The purpose of the surveillance programme for the SSCs important for safety, is to maintain and improve the availability of the equipment, to verify the compliance with the operating limits and conditions, and to detect and remedy any abnormal operation mode prior to any safety significant consequences occur. The abnormal operating conditions include not only defects in SSCs and software support, procedural errors and human errors of the personnel, but also emerging trends against the operating limits, the analysis of which may indicate a deviation of the NPP from the design limits and conditions. The programme has to confirm the availability of a sufficient safety margin to ensure safety in the event of impact of anticipated operational occurrences, errors and deviations. Special attention is paid to:

- The integrity of the barriers between radioactive materials and the environment;
- Safety systems availability;
- Availability of devices, the failure of which may adversely affect safety.

Surveillance measures taken to confirm the integrity and assess the residual life of the primary circuit are:

- Leakage measurement;
- Inspection and strength test of primary circuit;
- Registration of transients and comparing them with the ones analysed in the Safety Analysis Report (SAR);
- Testing the operability and leak-tightness of isolating devices, forming the primary circuit boundary;
- Surveillance of the Early Leak Detection System;

- Surveillance of the chemical composition of the coolant in the primary and secondary circuits;
- Surveillance of the radiation surveillance specimens.

Surveillance measures taken up to confirm the containment integrity are as follows:

- Containment leak tests;
- Penetrations leak tests;
- Structural integrity inspection;
- Control of the conditions inside the containment.

The frequency and the scope of the separate SSC surveillance are determined by:

- The level of importance of the SSC to safety;
- The recommendations of the manufacturer;
- The expected mechanism of the failures;
- The experience with the intensity of the failures;
- The degree of automation of surveillance.

The stipulated objectives of the Surveillance programme of Units 5 and 6 are:

- Periodic functional tests;
- Metrological verification of the instrumentation devices operability. Calibration of sensors and devices;
- Tests after maintenance or overhaul;
- Surveillance programmes after implementation of modifications important to safety;
- Control (monitoring);
- Inspections and checks of the condition of technical surveillance facilities

2.3.3.4. Measures for identification of unexpected degradation

Methods for establishing unexpected degradation include analysis of the results of the implementation of the monitoring, inspection and surveillance programmes and the operating experience.

2.3.4. Preventive and corrective actions

At Kozloduy NPP is established a system of preventive and corrective actions regulated in the following documents:

- Procedure for Organisation and Control During the Installation of Equipment and Pipelines;
- Procedure for Corrective (Emergency) Maintenance;
- Procedure for Performing Compliance Checks and Quality Control when Carrying out Activities Related to the Maintenance of Structures, Systems and Components;

- Procedure for Preventive Maintenance and Repair of Structures and Components of Process Systems.
- Procedure for Predictive Maintenance and Repair.

Kozloduy NPP applies a set of maintenance procedures (repair procedures) for each process system. The procedures describe the systems, their functions, associated technical specifications and a list of equipment and components, a list of all legal and binding requirements.

Preventive and corrective maintenance is performed in accordance with the long-term, annual and monthly schedules. Reports are prepared on the activities carried out, containing conclusions, findings and recommendations for corrective actions.

2.4. Review and updating of the PROGRAMME

The Programme, as part of the Kozloduy NPP Plc Management System, is subject to updating after reviews of the system. The main activities relevant to the review are:

- Observation and measurement;
- Self-assessment;
- Independent assessment:

A profound independent review of the ageing management programme had been completed within the Units 5 and 6 Lifetime Extension Project initiated by Kozloduy NPP Plc, where an Ageing Management Review (AMR) and residual life assessment of the equipment and facilities of Units 5 and 6 was performed.

The Ageing Management Review and residual life assessment were carried out in compliance with the Ageing Management Review and Assessment of the Residual Life of the Equipment and Facilities at Units 5 and 6 Methodology. According to the methodology the assessments are based on the following information:

- Information on the history and operating conditions;
- Existing lifetime characteristics;
- Operating conditions;
- Information on the water chemistry;
- Design solutions modifications;
- Maintenance and repair strategy, maintenance documentation, unscheduled repairs, provision of resources for maintenance and repairs;
- Information from the in-service inspection, diagnostic systems;
- Test reports;
- Abnormal operating modes, failures and damage of the equipment;
- Reports from the Ageing Management Review and residual lifetime justifications.

As a result from the comprehensive assessment performedwere determined the following:

- the technical condition of SSCs with the identification of prevailing and potential ageing mechanisms;
- SSCs with residual lifetime the operation of which can be extended for a given time period within LTO of the plant;
- SSCs whose technical condition and lifetime characteristics can be recovered or maintained through maintenance and repair activities throughout the plant LTO;
- SSCs that pursuant to preliminary (expert) assessment have sufficient residual lifetime for LTO of the units, to which end it is necessary to undergo additional assessments, evaluation and justification of residual lifetime following special methodologies and/or programmes involving specialised organisations;
- SSCs the lifetime of which will expire when the unit operational lifetime is reached and whose replacement is appropriate during the preparation of the unit for LTO;
- efficiency (technical and economic) of the current system for maintenance and repair including monitoring during operation allowing in particular to identify the operability of the SSCs in the LTO period of Units 5 and 6;
- capability of maintaining the operable condition of SSCs for the LTO period of Units 5 and 6, adequacy of the performed routine measures to provide reliability of components and prompt identification of their transition in boundary condition;

As an independent expert review of the PROGRAMME, in the context of activities for lifetime extension of units 5 and 6, Kozloduy NPP, in agreement with the NRA, have initiated conducting of IAEA SALTO peer review mission. In 2016 a Pre-SALTO Peer Review Mission of Unit 5 was conducted by a team of IAEA experts in the following fields:

A: Organisation and functions, current licensing basis, configuration/modifications management;

B: Determining the scope and selection, and plant programmes related to the extended operational life;

C. Ageing management review, review of ageing management programmes and revalidation of time limited ageing analyses for mechanical components;

D: Ageing management review, review of ageing management programmes and revalidation of time limited ageing analyses for I&C components;

E: Ageing management review, review of ageing management programmes and revalidation of time limited ageing analyses for civil structures;

2.4.1. Audits and inspections of the license holder

Conducting audits and inspections is part of the independent assessment of the activities at Kozloduy NPP.

Internal audits are carried out on a planned basis and according to a documented system, where the findings and conclusions of the audits are based on objective evidence. Conclusions from the audits include assessment of the compliance with pre-defined audit criteria, discrepancies detected, established good practices. Corrective measures with deadlines and responsibilities are defined for each discrepancy.

The activities included in the PROGRAMME are covered by the processes that are part of the Kozloduy NPP Management system, which are subject to a periodic audit every 5 years.

In the management structure of the plant there are specialised units with control functions with a certain degree of independence, which carry out independent inspection control over the compliance with the requirements of the international and national regulatory documents and standards in the field of nuclear safety, radiation protection, quality assurance, environment, fire safety, industrial safety, technical supervision and metrological support.

Corrective/preventive measures are defined for the removal of the identified discrepancies.

The information on the implementation of the measures identified during audits and inspections is registered with the Kozloduy NPP Management System.

2.4.2. Assessment of the Internal and External Operating Experience

Requirements, basic principles, responsibilities and obligations in the use of operating experience at Kozloduy NPP are listed in the procedure: Safety rules, Operating Experience Feedback System. A structural unit was established - Operational Experience and Self-assessment Indicators, and there are also Operational Committee to review and assess the feasibility of operating experience (screening after the initial review) and Operating Experience Council. The feedback system consists of two major programmes:

- Programme for utilization of the internal (own) operating experience;
- Programme for utilization of the external (industry) operating experience;

Sources of internal operating experience are operational events that occurred at Kozloduy NPP, including low level events and near misses. The main stages in the Programme for internal operating experience feedback include:

- Reporting and recording of an event in the information system;
- Event investigation;
- Analysis of the causes - definition of direct, indirect and root causes (for level 1 and level 2 events);

- Analysis of trends in low-level events and near misses and establishing common causes (program and organizational);
- Determining the appropriate corrective actions to prevent recurrence of similar events (for 1st and 2nd level events) and reducing the frequency of such events (for 3rd level events);
- Implementation and reporting on the corrective measures;
- Evaluation of the efficiency of implemented corrective measures;
- Periodic review of the effectiveness of the programme, including independent external assessments (NRA, IAEA, WANO).

Sources of external operating experience are operational events published in the information networks of WANO and IRS-IAEA, materials from seminars and conferences, and also the established good international practices. The main stages in the Program for External Operating Experience Feedback include:

- Initial examination (screening) of the applicability of the information published in relevant international information networks (WANO, IRS-IAEA);
- Review of the feasibility of the selected information by the Committee for review and evaluation of operating experience (Screening Committee) and identification of relevant corrective measures;
- Approval of the correctives measure by the Operating Experience Council;
- Implementation and reporting on the corrective measures;
- Evaluation of the efficiency of corrective measures and the programme for utilization of external operating experience.

2.4.3. Assessment of design modifications that could have an impact on the PROGRAMME;

Modifications of SSC important to safety are carried out after issuing a permit by the NRA in compliance with the ASUNE and the Regulation on the Procedure for Issuing Licences and Permits for Safe Use of Nuclear Energy. The Regulation defines the documents that should be submitted by the applicant for review and evaluation. Follow-up control of the modification made is accomplished by conducting inspections by NRA.

Modifications of the plant design are carried out under the Procedure for implementing design modifications at Units 5 and 6, which stipulates the activities on:

- analysis and evaluation of the design modifications proposals;
- drawing-up the necessary documents;
- sequence of the design modifications implementation stages;

- implementation and control of the activities related to the management of the process of implementing design changes (both permanent and temporary);
- carrying out tests after the modification;
- amendments to existing documentation.

The approved and implemented design changes are also reflected in all ageing management related documents in the field of operation, supervision, maintenance, and equipment qualification. Such changes are also reflected in the PROGRAMME by updating its scope and the ageing management activities.

2.4.4. Evaluating and measuring the ageing management efficiency;

The efficiency of the programmes for testing, maintenance, repairs, inspection and water chemistry control of the structures, systems and components (SSC) is evaluated by means of:

- quantitative and qualitative indicators, stipulated in the Rules for Applying the Self-assessment Indicators System for Effective Management of Kozloduy NPP Plc.
- periodic protocols, reports and other types of documents of the Kozloduy NPP's structural units;
- trend analyses of the values of the specific and functional indicators,
- the results obtained from the maintenance activities during the operation of the units at power;
- the results obtained from the maintenance and repair activities and modifications during an outage.

2.4.5. Time Limited Ageing Analysis – TLAAs

During the realisation of the whole Programme for Preparation of Unit 5 for Operational Lifetime Extension, the implementation of a group of activities has been initiated for further analyses and justifications of the residual life of the SSCs. Developed was a set of safety analyses – PTS analyses, cyclic strength calculations, radiation embrittlement and thermal fatigue, and seismic calculations that serve for residual life assessment purposes.

The PROGRAMME is updated on the grounds of the analyses made and the additional measures for ensuring the life of SSCs for LTO are identified.

2.4.6. It incorporates the most advanced technological achievements and results of research and development activity;

The exchange of information between the Kozloduy NPP, equipment manufacturers and research institutes is a continuous process and it aids to:

- perform an assessment of the operational life, the effects of certain combinations of certain work conditions and operating conditions, rate of progression and the presence of unforeseen essential ageing mechanisms;
- implement cost-effective solutions for maintenance the qualification of SSC;
- better understanding of the ageing effects on the equipment and the reasons for ageing.

As an example, more than 5000 modifications to the initial design of Units 5 and 6 have been currently implemented. A great part of these modifications are based on state-of-the-art technological achievements and research and development results.

2.4.7. Reporting of the amendments to the current licensing or regulatory framework in the PROGRAMME;

Periodic reviews of the PROGRAMME are made within and under the terms of the current Units 5 and 6 licenses from 2003 until now. Three revisions of the PROGRAMME have been developed, and a number of new requirements have been expanded and included, resulting from amendments to the NRA regulations and guidelines issued during this period.

A new revision of the PROGRAMME is being prepared, taking into account the requirements of the Regulation for Ensuring the Safety of Nuclear Power Plants issued at the end of 2016.

2.4.8. Identifying the need of additional research and development activity

In some cases, the experience from operation or research results raises the question of the timeliness of previous assessments, such as assessments of the life extension and qualification of installed equipment. After evaluating this information, Kozloduy NPP assigns performance of additional research and development activities.

2.4.9. The strategy for periodic review of the PROGRAMME, including potential correlation with periodic safety reviews;

Periodic review is performed:

- once every 5 years according to the documents management process of Kozloduy NPP;
- each year as part of the review of the Management System;
- in case of necessity, arising from the implementation of the activities under item 2.4. of this Report.

A compulsory review of the PROGRAMME has also to be carried out in the framework of the periodic safety review, conducted every 10 years, as part of the units re-licensing process. The scope and manner of conducting a periodic safety review is defined in the Regulatory Guide “Conduct of Periodic Safety Review of Nuclear Power Plants” and the Methodology for

Conducting Periodic Safety Review of Units 5 and 6 of Kozloduy NPP, which is agreed with the NRA. The review includes determining the status of the following safety factors related to ageing management:

Factor 1: The design of the NPP at the time of commissioning;

Factor 2: Actual state of the structures, systems and components (SSC);

Factor 3: Equipment qualification;

Factor 4: Equipment ageing.

2.4.10. Incorporation of new or unexpected circumstances in the PROGRAMME

The PROGRAM is an open document and, in case of new or unexpected circumstances, it is subject to updating.

An example of this is the update of the PROGRAMME after the approval of the National Action Plan following the Fukushima accident.

2.4.11. Using the results from monitoring, testing, sampling and inspection activities to review the PROGRAMME;

On the grounds of the monitoring, testing, sampling and inspection activities completed (work reports, installation reports, single test and multiple test reports, non-destructive examination reports, reports for implemented modifications and other documents) a review of the PROGRAMME is performed with regards to the adequacy and periodicity of the monitoring. Long-term monitoring schedules and procedures, and the PROGRAMME respectively are updated if necessary.

2.4.12. Periodic evaluation and measurement of the ageing management efficiency

Evaluation and measurement of ageing management efficiency are described in section 2.4.4. of this Report. The frequency of efficiency evaluation and measurement is in accordance with the effective management indicators system - once every 3 months.

2.5. License Holder's experience in implementing the PROGRAMME

On the basis of the evaluated operating experience and in order to optimize the ageing management activities, a structural change was made in 2005 at Kozloduy NPP, as a result of which a structural unit was established, the basic functions of which relate to coordinating the activities under the PROGRAMME, reviewing and improving the PROGRAMME as well as organizing activities related to the development of new programmes.

As a result of the evaluation of the experience in implementing the PROGRAMME, in accordance with international best practices, a programme for management of equipment qualification and lifetime management programmes were developed and implemented in 2009 to manage the lifetime of the individual components within the scope of the PROGRAMME.

Following a WANO Significant Operating Experience (SOER) announcement in 2011, the scope of the PROGRAMME was expanded and a lifetime management programme for Unit Transformers and Houseload Transformers was further developed.

As a result of the analysis of the experience and practices adopted at the plant related to ageing management, in the period 2012-2015, during the comprehensive assessment for the purposes of extending the lifetime of the units, an in-depth review of ageing was carried out and as a consequence areas of improvement were identified, for which new, specific programmes were developed covering:

- buried pipelines from the service water system;
- cables important to safety in addition to the qualified cables scope;
- hydraulic shock absorbers from the systems important to safety;

Based on the results of the comprehensive assessment, much of the existing ageing management programmes in the plant have also been updated.

Within the periodic safety review of Unit 5 in 2016, a self-assessment of the PROGRAMME was carried out which showed a high degree of compliance with the current requirements in the area of ageing management, including also those related to WENRA Safety Reference Levels for Existing Reactors, 2014.

2.6. Regulatory Supervision Process

Implementation of the activities related to ageing management falls within the scope of the regulatory control exercised by the NRA.

The PROGRAMME is one of the documents necessary for the issuance of operating licenses for Units 5 and 6 and is subject to an ongoing process of regulatory control for compliance with the conditions set in the licenses. Regulatory evaluation of the PROGRAMME is carried out within the scope of the procedures for issuing the operating licenses of the units.

Under the conditions of the operating licenses of Units 5 and 6, Kozloduy NPP is obliged to carry out a comprehensive assessment of the actual condition of the equipment and facilities in order to prepare the relevant unit for lifetime extension (LTO).

The comprehensive assessment of the actual condition of the equipment and facilities of the units in order to determine their residual life was performed in a scope and terms preliminarily agreed with the NRA. The lifetime extension programmes were developed on the grounds of the results of the assessment and were agreed by the NRA four years before the expiry of the respective operating license. By means of topical inspections carried out by NRA inspectors, the implementation of the programmes for preparation for lifetime extension of the respective unit was verified. Periodically, the NRA was submitted six-month and annual reports on the status of implementation of the measures from the programmes as well as the final completion reports. As a result of the inspections carried out and the analysis of the reports received, it was found that the programme has been fulfilled in accordance with agreed deadlines and scopes.

In 2017 the operating license of unit 5 was renewed. During the licensing process an evaluation and analysis of the PROGRAMME was carried out, it was updated during the periodic safety review conducted in 2016. This periodic safety review has been carried out in accordance with the NRA requirements for re-evaluation of the site's external hazards, the concept of continuous improvement of safety, including implementation of measures to manage severe accidents in the context of the Fukushima NPP accident. Within the scope of the periodic safety review in 2016, a review and evaluation of the ageing management activities and an effective ageing management program were carried out to confirm the possibility of long-term operation of the units.

2.7. Regulator's assessment of the general ageing management program and conclusions

The PROGRAMME presents the overall process of lifetime an ageing management, clearly defines the tasks, the distribution of responsibilities and the interconnections between the organizational structural units with respect to SSCs that are important for safety. The specific programmes developed are in line with IAEA regulatory standards, safety standards, provide a framework for coordinating all operation, maintenance, monitoring and oversight programmes and activities that are important for safety.

The scope of the PROGRAMME includes not only SSCs that are important for safety but also other SSCs that the License Holder has deemed necessary to be included in the programme in terms of extending the NPP lifetime. For all SSCs, ageing processes and degradation mechanisms are defined. Ageing processes and degradation mechanisms are subject to continuous monitoring and analysis. Trends are assessed and measures are taken to slow down the ageing process, to restore the SSC's characteristics by means of preventive or corrective maintenance, SSC's maintenance and repair improvement, and other measures.

The PROGRAMME has been updated during the periodic safety reviews in 2008 and 2016. The Licensee Holder has gained considerable experience in managing ageing processes, monitors and analyses the information from international experience regarding ageing of SSCs and applies the information relevant and applicable to the units.

As a result of the updates of the PROGRAMME, a number of improvements have been made, related to the outages planning and implementation and the identification and fulfilment of specific measures for the ageing management and limiting the SSCs degradation mechanisms. An example of such a measure is the realisation of a low-flux core refuelling pattern in order to reduce the neutron flux influence of on the reactor pressure vessel (vessel's irradiation embrittlement).

The efficiency of the PROGRAMME is confirmed by the results of the SSC's comprehensive assessment, as part of the implementation of the lifetime extension project.

The control of the implementation of the ageing management activities is carried out by the NRA in the process of issuing licenses and permits and the follow-up control of their implementation, also by means of inspections.

In the current long-term inspection programme that defines the main areas of control, ageing management is included in the areas of Maintenance, Technical Support, Management System, and SSCs important to safety.

The implementation of the PROGRAMME is also checked during the inspections of the readiness of the units for startup and operation after annual outage and refuelling. Inspected are the implementation of the scope of maintenance works and the quality of their execution, the modifications of the SSC and the conditions of the issued permits, as well as the control of the corrosion of process equipment of the unit and verification of the obtained results.

When conducting these inspections, particular attention is paid to the results of the in-service inspection of the metal and its compliance with the requirements of the Procedure for In-service inspection of Equipment and Pipelines, which contains the criteria laid down in the regulatory documents and standards of the manufacturing country. As a good practice are considered the additional in-service inspection programmes for equipment beyond the scope of the above procedure. The additional scope of monitoring is determined on the basis of data obtained from internal or external operating experience, assessment of the operating / maintenance personnel and the monitoring carried out for the purpose of the lifetime extension.

For assessing the SSCs maintenance and repair system, as part of the PROGRAMME, inspections are carried out covering:

- Supervision during carrying out corrective maintenance of major equipment;
- Registering deviations and defects from the normal state of the SSC's of systems important for safety and performing independent supervision of the maintenance activities;
- Organization and control of compliance with spare parts and materials storage requirements.

The main results of the inspections performed show compliance of the developed documents for carrying out activities with the requirements of the regulatory documents and the standards used in the design of the equipment. Correspondence was also established between the activities performed and the requirements of the internal documents (programmes, procedures, etc.) developed by the License Holder.

Inspection activities also cover the indicators for assessing the efficiency of the ageing management process. It was found that the system of indicators works effectively and provides possibility for corrective feedback and assessment of the status of important areas of the plant's operation. The achievement of the objectives set is carried out in accordance with clearly defined criteria. In compliance with the conditions of the operating licenses of the units, the NRA is periodically submitted information, including the indicators related to the ageing management.

Through the Operational Control Department at the Kozloduy NPP, the NRA performs oversight on the operation of the units, including the periodic tests of the SSCs, the outages and the removal of defects and failures of SSCs that are important for safety.

Changes resulting in modification of SSCs important to safety are authorised and controlled by modification authorisation permits issued by the NRA. A Regulation defines the documents that should be submitted for review and evaluation. The documents submitted in connection with the issuance of the permit are reviewed and evaluated by the NRA, taking into account the ageing management related requirements.

In conclusion, the NRA considers that THE PROGRAMME has been developed in compliance with the regulatory requirements and the applicable IAEA safety standards. The reviews and analyses completed have confirmed its efficiency.

3. ELECTRICAL CABLES

3.1. Description of ageing management programmes for electrical cables

3.1.1. Scope of ageing management for electrical cables

Cable ageing management is a set of activities, such as identifying the effects of ageing and degradation mechanisms, implementation of operational measures, procedures, methodologies, etc., taking into account the state-of-the-art achievements of science and technology. Activities are focused on the basic remaining features of the cables, with a defined lifetime, initial and marginal values of the status indicators, and resistance to external factors in design basis accidents. The intensity of cable ageing and the duration of their operation are determined by the cable's construction, the type of insulation material, the types of stabilizers (antioxidants, plasticizers, etc.), the duration of the impact, the amount and intensity of the degradation factors.

The process of cable ageing management is implemented by:

- defining dominant ageing mechanisms and the effects of ageing on the basis of scientific research results;
- determining the resistance of the cables to external factors of impact in design basis accidents and severe operating conditions, based on the results of type-certificated tests;
- implementing measures to reduce the intensity of external factors of impact;

- performance of diagnostics (control) of the technical condition and periodic reassessment of the service life of representative cable samples;
- performance of diagnostics (control) by means of assessment of cable connections in terminals, conductors, surfaces of bolts and busbars for all cabinet terminal blocks, I&C and electrical equipment, automatic and packed switches and connectors, power cable sockets, contact cable surfaces, bolt connections and busbar surfaces of power lines;
- preventive replacement of cables when replacing the respective SSCs and timely replacement of cables that have reached end of their service lifetime.

For the purpose of electrical cables ageing management a Program for Life Time Management of the Cables and Cabling at Units 5 and 6 of the Kozloduy Nuclear Power Plant was developed and put in place. The scope of the Program includes cables and cabling of Units 5 and 6, including hermetic cable penetrations.

Before all, these are cables and hermetic cable penetrations from the technological systems needed to ensure the performance of the basic safety functions, such as:

- a safe reactor shutdown and maintaining it in a safe condition during and after the accident;
- residual heat removal from the reactor core after reactor shutdown during and after the accident;
- reduction of radioactive materials release and to ensure that any release would remain within permissible limits during and after an accident.

An up-to-date list of main and ancillary systems is provided to ensure that the above mentioned functions are fulfilled. The list also includes:

- power supply cables for qualified equipment in the containment
- cables of protection, alarm and control systems;
- power supply cables for qualified equipment out of the containment

Secondly, the cables of technological systems that are important for the production of electric power. Those are cables with a medium voltage of 6 kV, in the composition of long cable lines, difficult to access and the replacement of which is related to the performance of a large volume of activities of substantial material value.

The Kozloduy NPP operates groups of cables, power cables 6 kV;

- power cables up to 1 kV;
- control cables for I&C purposes for control, protection, alarm, technological control;

The criteria for grouping the cables are: voltage and environmental qualification.

The main impact on the operability of the cables and the hermetic cable penetrations during their service life comes from the environmental conditions: temperature, humidity, radiation dose rate, possible impact of degradation factors in case of design basis accidents. The environmental conditions as listed, according to their influence on the development and intensity of ageing processes of the cabling, can be divided into adverse and non-adverse:

- adverse operating conditions - above all these are locations where temperature, pressure and level of ionizing radiation might change, or presence of a high temperature steam-gaseous mixture – premises where a loss of coolant accident (LOCA) and high energy line break (HELB) might occur: In addition, as a rule, there are locations at the Units where the level of operational degradation factors is higher than the average for the Unit, the so called "hotspots" of operation. Most often these are locations with higher temperature, the source of which is the operating equipment. Although "hotspots" are not very numerous, their contribution to ageing of components can be determining;
- non-adverse (normal) operating conditions (MILD, NOC), are characterized by an ambient temperatures within the range of 15°C to 50°C, normal pressure, low level of ionizing radiation. Such conditions are to be found in most premises outside the containment.

Cables operating in adverse conditions are subject to a qualification program as specified in Quality Procedure "Equipment Qualification Management at the Kozloduy NPP Units 5 and 6".

The activities regarding cabling lifetime management at Units 5 and 6 are performed in accordance with established programs, procedures, maintenance and technical service instructions, testing procedures, etc. This documentation describes the specific maintenance methodology, diagnostics, testing, as well as the acceptance criteria.

3.1.2. Ageing assessment of electrical cables

At Units 5 and 6 of Kozloduy NPP there are control and power cables installed with polyethylene (PE) insulation, cross-linked polyethylene (XLPE) polyvinyl chloride (PVC), ethylene vinyl acetate (EVA), ethylene propylene rubber (EPR) , a silicone halogen-free compound (SiHF). Insulated polymeric cables of EPR, EVA, SiHF were brought into use under the modernization of systems and equipment that took place at Units 5 and 6 in recent years. Polymers of the SiHF type are also applied in the insulation of airtight cable penetrations.

Physical ageing of electrical cables is primarily related to the change in the properties of the insulating polymeric materials they are made of. These changes cause deterioration in performance or a decrease in the functional reliability of cable lines.

The rate of change in the physical properties of the construction materials is determined by the intensity and type of degradation factors taking effect during operation, such as increased temperature; increased or decreased humidity; radiation exposure; vibrations; chemical impacts due to various substances, etc.

The classification of the potential ageing mechanisms of cable insulating materials, considering the types of cables at Units 5 and 6 as well as the presence of external factors during operation, is as follows:

- for cables with polyethylene insulation coatings, mainly thermal-oxidation destruction, which leads to a decrease in density, crack formation and embrittlement, is observed;
- for polyethylene crosslinked (XLPE), ethylene vinyl acetate (EVA), ethylene-propylene (EPR), silicon halogen-free compound (SiHF) insulation materials, thermal radiation oxidation destruction and electric field destruction (for power cables) resulting in decrease of density, fissure formation, embrittlement and dendrite development in power cables;
- for polyvinyl chloride (PVC) insulated PVC cables, plasticizing agent elution and destruction of molecular chains is observed that results in loss of mass, increased density, fissure formation and embrittlement.

Cables that were LOCA and HELB qualified are re-qualified through the application of the Laboratory Cable Assessment Testing Methodology.

The methodology aims to systematize the methodological, technical and organizational measures and procedures through which cable qualification is applied in order to confirm the ability of the cables to perform the functions assigned to them both in normal operation and as well as under conditions of a design basis accident for a specified period of operation.

In accordance with the objectives as set for cable qualification, the Methodology defines:

- requirements for qualifying cables;
- sequence of conducting cable qualification tests;
- requirements regarding measurement of representative cable samples using laboratory methods when conducting qualification tests;
- evaluation criteria regarding the results of the laboratory assessment of cables during qualification tests performed;
- evaluation of the cable service life (cable qualification);
- documentation requirements and quality assurance in the execution of the works.

During the elaboration of the Methodology the requirements of the following documents were taken into account:

- IAEA Nuclear Energy Series No.NP-T-3.6 Assessing and Managing Cable Ageing in Nuclear Power Plants, Vienna 2012;
- IEC 60780:1998 ”Nuclear power plants. Electrical equipment of the safety system. Qualification”;

- СТП 0.03.050-2009 “Qualification of plant equipment and components. General requirements”;
- СТП 0.03.083-2009 ”Environmental equipment qualification. General requirements”;
- СТП 0.03.079-2009 ”Quality assurance and control system for equipment qualification.”;

In the management of cable ageing of technological systems, important for the electric power production at Kozloduy NPP, non-destructive methods of cable insulation diagnostics are used, by measuring the electrical parameters that directly or indirectly characterize the technical condition of the insulation. In order to systematize the ways and the sequence for testing the condition of power cables for rated voltage of 6 kV and assessing the degree of ageing of the insulation and the risk of defects, a Method for 6 kV rated voltage power cables condition assessment was developed. The following measurements are included in the methodology:

Insulation resistance measurement.

Insulation resistance measurement (Rin) appears to be a sensitive and inexpensive method for controlling the homogeneity of insulation. When measuring R_{in} it is necessary to take into account the polarization and absorption phenomena. For this reason, not only R_{in} absolute significance, but also the absorption coefficient and the polarization index, are criteria for assessing the condition of the insulation.

When determining the absorption coefficient $Ka = R_{60s} / R_{15s}$, insulation resistance values, measured over 15 seconds for R_{15s} and 60 seconds for R_{60s}, are used. When determining the polarization index $PI = R_{10min} / R_{1min}$, insulation resistance values, measured over 1 minute R_{1min} and 10 minutes R_{10min}, are used.

The value of the "Ka" absorption coefficient and the polarization index "PI" are used to assess the condition of the polyethylene insulation cables. For normal values are considered Ka> 1.6 and PI> 4.0.

"Power and Control Cable Testing Procedure" was developed according to the general requirements and criteria for determining the electrical resistance of the cable products insulation, as regulated in the following documents:

- Standard BDS 1986-1982 - Cables, conductors and cords. Insulation electrical resistance measuring method;
- Testing Standards for Electrical Machines and Equipment", Sofia, 1995,

Measuring insulation capacity (C) and dielectric loss angle tangent (tgδ)

The measurement is performed at various frequencies. In these tests, the increase in tgδ is assessed as the applied voltage increases. This feature determines the presence of gas inclusions in cable insulation. Measurement of C capacitance and dielectric loss tgδ, analogously to R_{in}, is compared for each phase.

Measurement of partial discharge characteristics (PD)

One of the progressive diagnostic methods in which we measure:

- distribution of partial discharges along the length of the cable;
- quantity of partial discharges at the locations of defects;
- voltage of occurrence and extinction voltage of partial discharges.

On the basis of the results of the measured PD characteristics the distance to the location of the defect is defined. This method gives good results for polyethylene insulation cables.

Criteria for estimating measurement of partial discharges results are determined according to:

- Standard BDS EN 60270-2003 - High voltage test methods. Measurement of partial discharges;
- Standard IEEE 400.2-2013 – IEEE Guide for Field Testing of Shielded Power Cable Systems Using Very Low Frequency (VLF) (less than 1 Hz);
- Standard IEEE 400.3-2006 – IEEE Guide for Partial Discharge Testing of Shielded Power Cable Systems in a Field Environment;

Ageing mechanisms of cable accessories and cable trays and cable penetrations are also identified. Cable structures, including cable trays, grills, ladders, cable conduits, cable glands, are subject to ageing due to general corrosion, loss of physical properties, hardening, and cracking due to overload, bolt oxidation, and deterioration of the ground circuit. Cable penetrations are with reference ageing mechanisms that include loss of insulation resistance, loss of tightness. Cable connections such as cable joints, cable connectors, bolts, breakers, disconnectors and terminals are subject to thermal and radiation ageing, insulation resistance reduction, contact surface oxidation, all effectively leading to possible electrical damage, measurement accuracy reduction, safety features degradation and reduced reliability.

Information on research and development programs and operational experience is to be found in paragraph 3.2.

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

Tracking and determination of the ageing trend of cables subject to qualification is done by testing and assessing the test results of cable samples in order to determine their residual service life. Tests are carried out in specialized laboratories, which have a LOCA chamber for simulating accident and post-accident environmental conditions. Prior to this, cable samples are subjected to heat and radiation impacts, thus taking into account the specific qualification period.

According to the Methodology for Laboratory Assessment of Cables, initial tests are provided, such as:

- conducting a visual inspection and photographing cable samples in their original state;
- verification of the integrity of the conductive cores and screens;
- insulation resistance measurement;
- mechanical tests on the samples;
- higher voltage tests.

Accelerated thermal and radiation ageing in thermal chambers and exposure to Cobalt 60 isotope is applied. Intermediate functional tests include external visual inspection and photographic specimens of cables after ageing; verification of the integrity of the conductive cores and screens; measurement of insulation resistance and testing in modelled accident and post-accident conditions (LOCA-test, HELB) in LOCA chambers. The final tests include external visual inspections and photographing of cable samples after the LOCA test, checking the integrity of the conductive leads, measuring the insulation resistance, performing higher voltage tests, and measuring the tear resistance and relative elongation limits.

Power cables operated under favourable (normal) operating conditions (MILD, NOC) of 6 kV and 0,4 kV at Kozloduy NPP are subject to periodic monitoring, which is performed during the planned Unit's outage according to the schedule for preventive maintenance and service.

A basic method providing preventive actions for monitored cable connections in terminals, conductors, bolts and busbar surfaces is the infra-red thermal imaging camera control. It is performed on an annual schedule in accordance with an approved procedure, as well as before shutdown and after start-up of the unit after a planned outage or in-service corrective action for all cabinet terminals, system control panels and electrical equipment panels, automatic and circuit breakers and disconnectors, cable heads of power cables, contact cable surfaces, bolted connections and busbars.

Cable lines laid in tunnels, ducts and collectors in the buildings of the Plant which are not subjected to corrosion, mechanical damage (closed trays), have no coupling sleeves and open end sleeves with an obsolete structure are tested at least once in every 3 years. Testing of power supply cables is carried out during the overhaul of the equipment

Measurements and control of the technical condition of the cables are carried out in accordance with the Procedure on Power and Control Cables Testing in the scope, as follows:

- Measurement of absorption coefficient ($K_{abs} = R_{60sec}/R_{15sec}$).
- Measurement of the polarization index ($PI = R_{10min} / R_{1min}$).
- Measuring leakage current.
- Measuring cable capacity.

The Procedure on Power and Control Cables Testing also specifies the criteria for the acceptability of measurement results.

Verification of the hermetic and the technical condition of the ELOX electrical penetrations operated at above atmospheric pressure, with nitrogen is carried out by personnel in accordance with the corresponding procedures and in the appropriate volume and periodicity as follows:

- Within the unit's outage an external visual inspection of the condition of the anti-corrosion coating, the coating layers in the area of the welds, the presence of the protective sleeves on both sides of the penetrations and the presence of blind flanges of the spare locations, the insulation at the connection points of the cable core with the penetration and visual control of the cable pressure indicator gauge is performed;
- measuring the insulation resistance of the cable lines along with the penetration as related to the enclosure, as an acceptance criterion of $\geq 0.5 M\Omega$ for type BT hermetic cable penetrations and $\geq 5 M\Omega$ for airtight cable penetrations of the MT type;
- once a month, while the unit is operated on power, air tightness of "ELOX" electrical penetrations is checked by the readings of the indicator mounted on each hermetic cable penetration (taking into account the temperature changes in the room), as well as during planned annual outage and before putting into operation, with acceptance criteria as defined in the methodology.

3.1.4. Preventive and remedial actions for electrical cables

Kozloduy NPP Plc has introduced a system of preventive and corrective actions the main purpose of which is to effectively maintain the operability of the SSC. In this respect, long-term maintenance and repair schedules for the SSCs, tailored to the requirements of the manufacturers, have been developed and implemented. As part of this system is also the task to maintain operability of the cables and hermetic cable penetrations. Preventive and corrective actions on cables and hermetic cable penetrations are also taken in case of disturbance of environmental conditions and impaired design characteristics of cables and hermetic cable penetrations, established during periodic tests and walkdowns.

The maintenance of the technical condition of the cabling is carried out in accordance with the following programs that are in place:

3.1.4.1. Equipment Surveillance Programme

The Surveillance Programme objectives are:

- verification that the design conditions, under which plant safety has been justified, are maintained during operation;

- verification that safety level is in compliance with the requirements and provides for sufficient margins during anticipated operational events, personnel errors and equipment failures;
- maintenance and improvement of equipment preparedness, confirmation of respective operational limits and conditions;
- detection and elimination of any violation from normal operation, before the occurrence of significant implications to safety.

In accordance with the requirements of the Program, the methods of surveillance of the equipment being applied are:

Oversight

Oversight provides direct information regarding Units' equipment condition. It is carried out by the operating staff through recording the parameters, plant walkdowns and their records, inspection of the condition of the stand-by equipment (thermal insulation inspection, visual inspection, inspection of the condition of the valves according to check-list, etc.). Defects or deviations from normal parameters are recorded in a database, and the necessary actions are then taken by the personnel responsible for the defective equipment.

Periodic functional tests

The purpose of periodic functional tests is to verify the operability of systems in standby mode (safety systems and separate systems for normal operation performing safety functions). The main procedures in compliance with which the safety systems parameters and operability are checked are the procedures for testing the safety systems of Units 5 and 6. Another principal document directly related to the oversight is the Instruction Regulating the Type, Procedures and Conditions for Performing Testing of the Systems and Equipment of Units 5 and 6 upon Shutdown for Scheduled Outage, During Planned Outage and Start-up after Planned Outage.

Metrological verification of measuring instruments performance. Calibration of sensors and instruments

The metrological control of the measuring instruments and the measuring systems are carried out according to the organization and the order as regulated in the Instruction for Metrological Control of Measuring Instruments at Kozloduy NPP

Post technical service or maintenance testing

After a planned annual outage, pre-planned complex functional tests of separate pieces of equipment and systems are carried out, and appropriate documentation (functional test certificates and system preparedness certificates) is produced. Checking the entire measuring channel, including cables, is performed. The purpose is before putting into service to confirm that the maintenance objectives have been met and that the requirements regarding the element (the system) are complied with.

Surveillance programs after implementation of modifications that are important for safety

Following the implementation of the design modifications, specific functional test programs are developed. The programs have two objectives - to verify the conformity of the modification with the design requirements and to determine, on the basis of accumulated experience, the optimal scope and periodicity of the surveillance.

<u>3.1.4.2.</u> <u>Technical servicing and maintenance</u>

Technical servicing and maintenance include a set of activities to restore the operability, reliability, service lifetime of equipment or parts of it and are carried out:

- according to a schedule;
- in case of deterioration of the technical condition;
- preventive survey of cable connections in terminal blocks, conductors, bolts and busbar surfaces;
- in case of failures and disruptions.

Preventive maintenance involves removing the equipment from operation, regardless of its condition, with a periodicity defined in the technical documentation and depending on the maintenance category the scope of work is defined related to dismantling, control and fault detection of the equipment. Preventive maintenance is regulated in the Procedure on Preventive Maintenance of Structures and Components of Technological Systems

Corrective maintenance is carried out in the event of deterioration in the technical condition or failure of the equipment. Corrective maintenance is carried out in accordance with the Procedure on Execution of Corrective (Emergency) Maintenance.

<u>3.1.4.3.</u> <u>Analysis of failures</u>

Failure analysis helps to identify trends in equipment degradation and the way potential failures might occur. Analyses of the failures are performed according to the Methodology for failure analysis of equipment. The results of the analysis help assessing the need to review previous qualification conclusions or requirements related to installation, maintenance and replacement of equipment. The results of the trend analysis are used for recommendations as follows:

- for replacement of equipment of a type or a component thereof;
- to change the parameters of the environment in the premises or the operating conditions;;
- to improve the operational and maintenance procedures
- to modify the configuration of the systems.

3.1.4.4. Control of environmental conditions in the premises

The Kozloduy NPP monitors the parameters of the environment in the premises through the information systems sensors. The data obtained from temperature, pressure, humidity and radiation measurements are used to determine the so-called "Hotspots". After analysing the results of the measurements, activities are then developed to improve the environment (improvement of the ventilation, provision of additional heat insulation).

Monitoring the conditions specifies unforeseen workloads leading to premature ageing of the equipment, which may lead to shorter service life.

Monitoring the conditions refers to the actions performed to assess functional operability, operational availability, equipment service life, and contributes to:

- the assessment of ageing effects and the residual service life of equipment;
- identification of ageing mechanisms that could be inadequately addressed in the primary qualification;
- identification of emerging defects.

The Procedure for Performing of Predictive Maintenance and Repairs describes the activities, organization and responsibilities of preparing and executing predictive maintenance and repairs on the SSC in order to prevent the occurrence of defects and / or failures and to reduce the number of corrective maintenance activities.

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

Ageing Management of cabling is carried out throughout the entire lifetime of the Plant. Over the past few years, the Plant on its own and via external services, carried out activities related to the management and ensuring the residual lifetime of cables and airtight cable penetrations, namely:

- tracking and determining the ageing trend of cables subject to qualification (tests and evaluation of cable samples test results);
- timely maintenance and repair works;
- Periodic monitoring of the technical condition of the cables (diagnostics) and their lifetime extension (re-qualification);
- replacement (for unsatisfactory performance or modernization).

In the period 1992-1995, during the operation of Units 5 and 6, defects (insulation breakdowns) were found on the cable lines 6 and 0.4 kV, that provide power for the equipment in containment and in the 6 kV cable lines between Units 5 and 6 and the common plant facilities. Due to the extreme importance for the normal operation and safety of the Units, decision was made for the 6 kV cables in containment powering the main circulation pumps to be replaced while for the rest of 6 and 0.4 kV cables, supplying safety systems equipment and equipment of systems important to safety an assessment of the state of their insulation is envisaged.

In 1996, under Units 5 and 6 Modernization Program, a measure "Introduction of methods and means for determining the residual service life of cables" was launched. The implementation of the measure also involved the Scientific Research Sector at the Sofia Technical University with the task of "Determining the state of insulation of power and control cables of Kozloduy NPP Unit 5". In accordance with BSS 2581-86 "Power cables for stationary laying with polyethylene and chemically crosslinked polyethylene insulation" a program was developed and tests for cables from the containment (power control cables low voltage) were made in laboratory conditions. As a result of the survey, the following findings and recommendations were made:

- in most of the surveyed cable lines a deteriorated state of the insulation in the hermetic penetrations was observed. Replacement of the cable penetrations to be planned, with a priority to be given to the replacement of the 6 kV hermetic penetrations, providing power to the main coolant pumps;
- in certain cable lines cases of poor insulation were found localized in the cable heads at the supply sections 6 and 0.4 kV and in the cable heads of the electric motors. It was recommended to plan the gradual replacement of cable heads in the future using state-of-the-art technology and a new type of insulating material;
- in order to determine the residual service life of cables, periodically (in 3-5 years), identical tests of the tested cable lines to be carried out as to give account how the following factors affect the ageing of the insulation: current load, increased temperature, humidity and radiation.

In line with the recommendations made, the following activities were carried out:

- the step-by-step replacement of power and control cable hermetic penetrations of 6 and 0.4 kV was performed. The replacement started in 1999 with 6 kV penetrations for the main coolant pumps. The total replacement of the hermetic cable penetrations was completed in 2006. At present Kozloduy NPP uses hermetic cable penetrations of the "Elox" type, which are qualified in accordance with IEEE 317-83 standards and meet the technical conditions of "Проходки герметичные типа “ЭЛОКС” для АЭС”, ТУ 5057;
- gradual replacement of cable heads and cable lines in safety-critical equipment was carried out. The replacement took place in the period 2000 ÷ 2005;
- a periodic survey of the isolation status of cable lines 6 and 0.4 kV was introduced to determine their residual life.

In 1999 and 2005, the following surveys were planned and implemented:

- "Survey on the state of insulation of cable conductive lines and asynchronous electric motors 6 kV at Units 5 and 6";
- Determination of Residual Service Life of Power and Control Cables at Units 5 and 6"

As a result of the surveys conducted the following conclusions were made:

- no changes in the characteristics of the electrical insulation of the cable lines for the period 1997 ÷ 2005 were observed;
- all tested 6 kV power lines have a normal state of electrical insulation as well as the installed cable heads and couplers.

The test results show that, after replacement of the airtight cable penetrations, the overall insulation condition of the cable lines 6 and 0.4 kV is good and their residual life is sufficient to ensure the normal operation of Units 5 and 6.

On the basis of the Methodology for testing the state of power cables for rated voltage of 6 kV, a survey on the state of the 6 kV power cables was carried out in 2009. The purpose of the survey is to summarize the data obtained from measurements of the electrical resistance of insulation, partial discharges and tests with increased DC voltage of 6 kV power cables between Units 5 and 6 and the common plant facilities and establish the state of insulation of the 6 kV cable line.

The results of the survey are presented in a report with the following conclusion: "The technical condition class corresponds to power cables with normal operating environment. No ageing of cable insulation of 6 kV cables is observed."

For the purpose of tracking and identifying the trends of changes in the state of insulation, a second survey of cables 6 and 0,4 kV was carried out in 2015 ÷ 2016.

The main purpose of the survey on the state of Units 5 and 6 cabling and the common plant facilities is to measure a set of parameters that determine the condition of the cable insulation at the given moment of time. This set of parameters includes measuring: the insulation resistance value of the cable lines; absorption coefficient and polarization index, tangent delta tgδ, and partial discharge measurements to identify problem areas and local accumulations of partial discharges in the cable insulation. The combination of these measured parameters provide information about the presence of moisture and sections with initial form of structural changes in cable insulation and the accumulation of partial discharges along the cable length or at local points, such as cable joints and cable end joints - cable heads.

In 2008-2010, a re-qualification of electrical and electronic equipment, including cables and airtight cable penetrations, was carried out for the design lifetime in view of beyond design basis impacts - LOCA or HELB.

In connection with the implementation of the Plant Lifetime Extension Project a thorough Ageing Management Review to assess residual lifetime of the equipment and facilities of Kozloduy NPP Units 5 and 6 was performed. The list of SSC to be reviewed also includes an assessment of the residual life of cabling in the composition of the electrical equipment and the equipment of the systems for control of safety systems and the systems that are important to safety.

At the first stage of the Ageing Management Review, the possibility of further operation of the cables being operated under "MILD" environmental conditions was confirmed with the current maintenance and technical service system at Kozloduy NPP.

Upon completion of the cabling review in premises where LOCA and HELB design basis accidents might occur, it was recommended to carry out an additional survey and re-qualification of the cables for the given conditions, taking into account the Plant lifetime extension. During the period 2015-2016, cables, operating under severe conditions LOCA and HELB were re-qualified.

According to a recommendation from the Review, a re-qualification of 6 kV power cables from the safety systems was carried out, with regard to the results of the measurements of the absorption coefficient and the polarization index.

The activities, related to cables ageing and hermetic cable penetrations management are enlisted in the Long Term Schedule for Periodic Monitoring of Cables 6 and 0.4 kV. The purpose is to monitor ageing trends in cable insulation and to identify appropriate corrective measures. The continuous development and improvement of the cable ageing management system and airtight cable penetrations is an obligatory factor that ensures the safe operation of the Plant and Kozloduy NPP and the staff is conscious of that.

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

The approach demonstrated by the licensee as regards ageing management of the cabling is justified by being correctly directed and helps achieve good results in the process of maintaining and enhancing the level of safety at Kozloduy NPP.

An important contribution to the improvement of the level of safety has the implementation of the latest achievements in science and technology, cooperation with leading scientific organizations (OJSC “All-Russian Research Institute for Nuclear Power Plants Operation” - Moscow, Technical University of Sofia, Engineering and Technical Center "Equipment Qualification and Service Life Evaluation"- Kharkov), as well as own efforts.

The documents developed (instructions, programs, procedures, etc.) are updated in a timely and adequate manner and reflect the requirements of the national regulatory framework, the applicable standards and those of the IAEA related to the ageing management process of the cabling components. Testing programs have been developed, studies and tests are being carried out to confirm the qualification status of electrical equipment and cables. The necessary methodologies and analyses were developed in order to provide the necessary resources for the project and for identifying the qualification status of the cables. Prior to the expiry of the cable qualification deadlines, preventive and corrective actions, including re-qualification or replacement of the necessary components, are being undertaken. In the course of the implementation of the Project for Plant Lifetime Extension of Kozloduy NPP, a survey regarding the residual life of cabling was performed. At locations where work conditions can be worsened during an accident, cable qualification activities have also been carried out. As a result of the survey and qualification, cable replacement activities are foreseen to be performed or are already performed for the cases that have not met the requirements for the additional service life and / or the environmental conditions in the premises where they are located.

The NRA inspections are part of the activities that cover the overall control over the ageing processes of the nuclear facilities cabling. Within the framework of the inspections to determine the Units' start-up readiness after an outage, the planning, organization and reporting of the performed maintenance works, the compliance with the required documentation, the status and the operability of the technological systems, including their pertaining cables and cable penetrations is checked. The implementation of the measures regarding cabling planned to be completed during the outage and relating to the Plant lifetime extension is checked. The following activities can be mentioned as an example:

- Qualification of 6 kV safety systems cables for extended service life according to measurements results of the absorption coefficient and the polarization index. (During the planned 2017 outage at Unit 6, 7 cables were measured). A 10 year long-term schedule was developed;
- Reinforcement of cable ducts between the reactor building and diesel generator stations by modernizing penetrations making new expansion joints;
- Replacement of stands, sample impulse lines of I&C sensors, junction boxes, cables and cable trays of equipment installed in HELB conditions.
- Replacement of cables for temperature control of the reactor coolant circuits and the pressurizer involved in measuring channels of the control and protection safety systems from the junction boxes to hermetic penetrations.

The replacement of cables and related equipment, as well as cable penetrations, in case they are in safety-critical systems, is subject to authorization. The Nuclear Regulatory Agency reviews and assesses the documents submitted with the application and issues permits for implementation of the modifications/changes.

The ageing management of cables and cabling at the Kozloduy NPP is carried out in accordance with the documentation as developed, implementing good practices by the licensee, thus ensuring that safety at Kozloduy NPP shall be achieved and maintained at the level required.

4. CONCEALED PIPELINES

4.1. Description of the concealed pipeline ageing management programmes

4.1.1. Scope of concealed pipeline ageing management

A lifetime management programme for Component Cooling Water System [CCWS] buried pipelines is developed to implement the process of concealed pipelines ageing management. The lifetime management programme covers the underground trunk pipelines which provide service water for cooling of important to safety structures, systems and components.

The underground trunk pipelines lifetime management programme is based on the knowledge and understanding of the main material degradation mechanisms in result of ageing factors. The understanding of underground trunk pipelines ageing is developed on the grounds of basic documentation, operational and maintenance history and the international experience in this area.

The scope of the Component Cooling Water System [CCWS] Buried Pipelines Lifetime Management Programme covers the following:

- main ageing mechanisms of buried pipelines;
- integration of activities related to underground trunk pipelines lifetime;
- design basis, applicable codes and standards, and requirements enforced by the regulatory body;
- data on underground trunk pipelines structure modernisation;
- data on the abnormal underground trunk pipelines operation;
- data on the thermo-mechanical loadings of underground trunk pipeline components;
- data on operational medium parameters (water chemistry parameters, data on corrosion status control, etc.);
- operational and maintenance history, including surveillance and in-service inspection results;
- information on implementation of measures for underground trunk pipelines operation reliability enhancement;
- worldwide experience in the assessment of the state of the underground trunk pipeline metal.

The buried (underground) trunk pipelines of the component cooling water system are related to systems that are important to safety. These are part of the component cooling water system designed to cool down essential loads in all operating modes and emergency conditions. The pipelines are divided in three independent systems which operate in closed circuits and water is cooled in the spray pools. They are located in the area from the reactor Building to the Spray Pools, from the Spray Pools to the intake structures in Diesel Generator Stations, and from Diesel Generator Stations to the Reactor Building. The pipeline diagram with numbering is shown on Fig. 1.

The pipeline system of the component cooling water system comprises gravity, discharge and recirculation branches.

The gravity pipelines supply water from the spray pools to the intake structures of the pumps located in the diesel generator stations and have a constant diameter of Dn 1200 throughout their entire length.

The discharge pipelines, supplying water from the Reactor Building to the Spray Pools, have different diameters - in their first parts (branches) the nominal/external diameter is Dn 600, in the medium branches the diameter is Dn 800, and in the end parts the provisional diameter is Dn 1000.

The recirculation pipelines of service water pumps (located in the Diesel Generator Stations) have permanent diameter of Dn 250 throughout their entire length.

Discharge and gravity pipelines are from spirally welded tubes of low-alloyed steel 10Г2САФ under the BDS 4880-79 “Low-alloyed structural steel”, and the recirculation ones – from tube units of carbon structural steel ст20 under ГОСТ 1050-74 “Carbon structural steel of commercial quality. Technical conditions”. The geometric dimensions of the main components (provisional diameter, wall thickness) are the following: Ø273x8, Ø630x6, Ø820x6, Ø1020x7, Ø1220x8. The total length of the pipelines is around 5726 m. Formed components are made of carbon steel ВСт3сп4 under BDS 2592-71 “Carbon structural steel of commercial quality”. The total number of the formed components (elbows, three-way pipes, connectors, etc.) amounts to around 150 pieces.

The pipelines have reinforced outer anti corrosion coating in the form of a ultra-reinforced three-layer tape of the type “СИЛ–Б” according to ОН 0265129-79 0 - the pipelines are laid in trenches with bottoms of compacted sand and then backfilled with the excavated material. Some parts of the pipelines are beneath concrete pavements or their route passes through concrete shafts and penetrations.

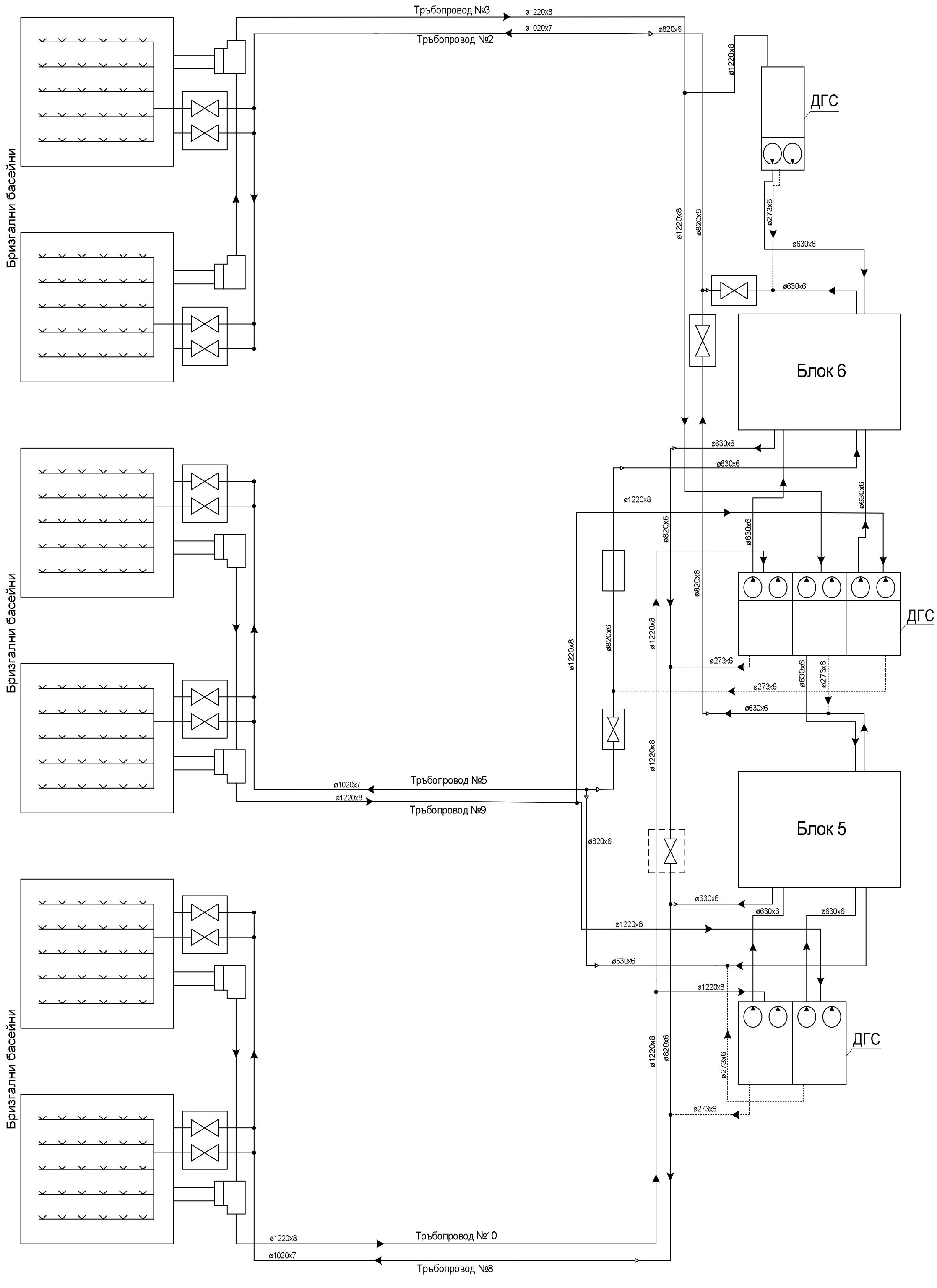


Fig. 1 Diagram of Kozloduy NPP Unit 5 and 6 Component Cooling Water System [CCWS] pipelines.

Operating pressure: max 0.46 MPa in the discharge pipelines, max 0.2 MPa in the gravity pipelines, max 0.6 MPa in the recirculation pipelines.

Water working temperature: 12÷45°C in the discharge pipelines, 5÷33°C in the gravity pipelines, 35°C in the recirculation pipelines.

Calculated discharge of cooling water: 3000 m^3/h;

Working medium: service water (pH 25ºC – 7.8÷9.0 ed).

Most of the pipelines have sections which are buried in the ground, as well as sections penetrating concrete, and sections lain in covered trenches and shafts with restricted access.

4.1.1.1. Methods and criteria for selection of concealed pipelines within the frameworks of ageing management

The selection of critical components of the underground trunk pipelines is defined in the Lifetime Management Programme for Kozloduy NPP Units 5 and 6 Component Cooling Water System [CCWS] Buried Pipelines which addresses ageing management and is based on the results from the implementation of operational, maintenance, surveillance, water chemistry and in-service inspection programmes, including:

- requirements for state monitoring;
- scope and frequency of examinations;
- identifying defective sections by non-destructive testing methods (NDT);
- maintenance of water chemistry within the established operating limits;
- maintenance of operating modes within the established operating limits;
- requirements for leak monitoring;
- grouping of system components according to their degradation level, requirements for repair or replacement and analysing by considering a number of acceptance criteria.

The critical sections of the pipelines have been identified on the basis of:

- Results of input data analysis.
- Availability of pipeline sections in which evolution of degradation ageing mechanism is expected, for instance:
    - sections where fluid direction changes (elbows, connectors, etc.);
    - sections in which the pipeline diameter changes;
    - sections that have three-way pipes, branch jointing;
    - sections at the place of welded connections crossing;
    - congested areas due to inappropriate routing where pipelines cross with other underground communication lines;
    - stagnant sections or drainages.
- Observations of degradation effects and occurring ageing mechanisms during operation; repaired sections (sections with impaired insulation, places of bad quality of the pipeline trenches backfilling).

4.1.1.2. Processes/instructions for detection of ageing mechanisms related to concealed pipelines

The activities associated with identification of concealed pipeline metal ageing mechanisms cover the following:

- Analysis of defects and failures found during the period of operation;
- Non-destructive testing of pipelines:
    - Visual inspection and ultrasonic measuring of the pipeline wall thickness at the accessible sections (existing shafts for valves and observations) without excavation works;
    - visual inspection and ultrasonic measuring of pipeline sections with excavation works and removal of the outer anti-corrosion protection during repairs because of found leaks;
    - visual inspection of pipeline internal surfaces with remote control devices through the existing technological manholes.
    - visual inspection and ultrasonic thickness measuring of pipeline sections during installation of isolation valves;
    - test drilling for visual inspection and ultrasonic thickness measuring to confirm the results from the conducted magnetometric diagnostics.
- Survey of dismantled pipeline sections with defects registered with destructive and non-destructive testing methods.
- Water treatment analysis for circulation cooling systems with spray pools, including corrosion speed measuring.

4.1.1.3. Criteria for grouping* of pipeline sections for the purposes of ageing management:

Pipelines can be separated in the following sections on the basis of their routes and expected degradation mechanisms:

- buried pipelines: the pipelines that have direct contact with soil or concrete (for instance a wall penetration);
- underground pipelines: the pipelines are under the level, but they are placed in tunnels or shafts - thus they have contact with the air and the access to these places for inspection is restricted.

** The criteria for grouping are selected according to Table 5 of Annex 3 and the Ageing Management Programme Type 125 of the "Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL), Safety Reports Series No 82, IAEA" (further called – IGALL)*

4.1.2. Assessment of concealed pipeline ageing

The programme for underground pipeline lifetime management is developed in compliance with IAEA methodology, using the following key standards and guidelines:

- Ageing Management for Nuclear Power Plants, Safety Standards Series, Safety Guide No NS-G-2.12, IAEA (2009);
- Safe Long Term Operation of Nuclear Power Plants, Safety Standards Series No 57, IAEA (2008);
- Implementation and review of a nuclear power plant ageing management programme, Safety Reports Series 15, IAEA (1999);
- Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL), Safety Reports Series No 82, IAEA (2015);
- Ageing Management Program (AMP) 125 Buried and underground piping and tanks (Annex to IGALL).
- Requirements for Nuclear Power Plant Equipment and Pipelines Lifetime Management. Fundamentals. НП-096-15 (2015).

The following key projects and their process and operational documents were used for the development of the programme:

- Plant Life Extension of the Kozloduy NPP Plc Units 5 and 6 Project;
- Programme for investigation, assessment of the technical condition and justification of the residual life of the underground trunk pipelines between the spray pools, Reactor Building and diesel generator stations of Kozloduy NPP Units 5 and 6.
- Results of the technical condition and quantitative assessment of the residual life of underground trunk pipelines between the spray pools, Reactor Building and diesel generator stations of Kozloduy NPP Units 5 and 6;
- Analysis. Water treatment for circulation cooling systems with spray pools;
- Instruction: Monthly monitoring of service water buildings and facilities - service water for essential loads (civil engineering and hydrotechnical part);
- Procedure for operating control of service water pipelines - essential loads, group A of Units 5 and 6 Component Cooling Water System [CCWS];
- Assessment of Kozloduy NPP Units 5 and 6 buried service water discharge pipelines with magnetic methods;
- Work programmes on non-destructive testing (NDT) of Kozloduy NPP Units 5 and 6 underground trunk pipelines;
- Operating procedure for component cooling water system for essential loads;

- Diagnostic procedure on contactless magnitometric testing of pipelines technical conditions, 2-008-2002;
- Operating procedure on service water corrective treatment for consumers important to safety in the circulation cooling systems with spray pools;
- Inspection procedure. In-service inspection of base metal, overwelding and welded joints ot the equipment and pipelines of Kozloduy NPP with WWER-1000 reactors.

Item 4.1.2.1 and item 4.1.2.2 of this Report specify how the above-mentioned documents were applied.

4.1.2.1. Ageing mechanisms requiring management and identification of their significance

Critical components of the buried/underground pipelines were selected for ageing assessment. The critical components were identified by analysing the data obtained from: the repair works conducted for removing leaking sections, non-destructive metal testing (NDT) and pipeline corrosion state assessment.

The types of potential ageing mechanisms that could evolve in the base metal and welded joints of buried and underground pipelines are presented in Table 2.

Table 2. Identification of concealed pipeline ageing mechanisms

Selected pipeline group	Ageing mechanism	Description	Ageing effects
Buried Underground	General corrosion	Known also as homogeneous corrosion which spreads on the metal surface with almost the same corrosion rate.	Loss of material (thinning of walls)
Buried Underground	Pitting (local) corrosion	Localized corrosion of the metal surface limited to a spot (pitting) or a small area in the form of a cavity (pit).	Change of dimensions (thinning of walls)
Buried	Corrosion cracking as a result of deformation	Corrosion where the presence of local dynamic deformation is an essential reason for formation of cracks but there is no cyclic loading or the cyclic loading is limited to very few rare events.	Crack growth
Buried Underground	Erosive-corrosive wear	Gradual removal of metal corrosion products as a result of the mechanical interaction between the surface and the circulating fluid	Change of dimensions (thinning of walls)

All defects discovered during the operation up to year 2016 are in the base metal (there is only one discovered in a welded joint) of the straight sections of buried pipelines and the main reason for that is the presence of building waste in the backfill which impaired the outer corrosion protection. No defects were found in the sections with formed details which are considered as potentially critical sections because of the possibility of erosive-corrosive wear of pipelines. No cracks related to stress corrosion were found.

Irrespective of the fact that cathode protection is not provided by the design, no defects were found in the crossing sections of pipelines with existing cable ducts. Therefore, the influence of laying cables in cable ducts on the other underground communications is limited.

According to the analysis of potential ageing mechanism evolution of buried and underground pipelines of the Component Cooling Water System [CCWS], it can be concluded that general and pitting corrosion are the main ageing mechanisms.

4.1.2.2. Identification of acceptance criteria related to ageing mechanisms.

The main acceptance criteria related to ageing mechanisms of Component Cooling Water System [CCWS] underground pipelines are the following:

The condition of insulation and its capability to perform its functions - integrity, good adhesion with the pipeline metal.

The state of insulation is examined by means of:

- periodic visual inspections for leaks along the route of buried pipelines, monitoring of the condition of penetrations and the visible parts of tubes in the shafts along the pipeline routes, the readings of instrumentation and control systems according to the Operating procedure on the component cooling water system for essential loads;
- conduct of contactless magnetometric diagnostics of pipelines without digging according to the Diagnostic procedure on magnitometric testing of pipelines technical conditions, 02-008-2002, developed in compliance with Russian state standards, and Methodology for non-contact magnetometric diagnostics of operatble heating lines, Moscow, 2009";
- visual inspection of critical sections of pipelines according to the results of the contactless magnetometrical diagnostics, maintenance and repair.

Corrosion rate.

The rate of corrosion on the internal side of the pipelines is monitored and controlled by maintaining appropriate chemistry of the water running in them, application of water treatment programmes according to the operating procedure for service water treatment system for essential loads in circulation cooling systems with spray pools.

Dimensions of defects registered with the help of non-destructive testing (NDT) methods are specified:

- for pipelines commissioned by the year 1991 – the assessment of the quality of welded connections is according to the requirements of "Regulations on the inspection of welded joints and overwelding of assemblies and structures of nuclear power plants, and research nuclear reactors and installations," ПК 1415-72, Moscow, Metallurgy;

- for the pipelines commissioned after 1991 (sections of pipelines that have been replaced or repaired after the year 1991) – the assessment of the quality of welded joints is according to the requirements of PNAE G -7-010-89, Equipment and pipelines in nuclear power plants. Welded joints and overlays. Inspection rules, Gosatomenergonadzor of the USSR, Moscow, 1991*;
- the quality assessment of base metal is performed according to the codes (standards), described in the "Inspection procedure. In-service inspection of base metal, overwelding and welded joints of the equipment and pipelines of Kozloduy NPP with WWER-1000 reacors".

**Comments: After the new revision of the procedure became effective: "Inspection procedure. In-service inspection of base metal, overwelding and welded joints of the equipment and pipelines of Kozloduy NPP with WWER-1000 reactors", , the quality assessment of base metal and welded joints of pipelines is to be performed according to a new normative document - НП-089-15. Federal norms and rules in the area of usage in nuclear ebergy. Regulations on the structure and safe operation of the equipment and pipelines of nuclear power plants, Rostechnadzor, Russia, 2015.*

Assessment of pipeline wall thinning.

Ultrasonic thickness measuring according to the following documents:

- Work programme on ultrasonic thickness measuring of underground pipelines of the service water system.
- Programme for assessment of dismantled sections of the underground pipelines of the components cooling water system;
- Assessment reports of the Component Cooling Water System [CCWS] underground pipelines according to the results of the contactless magnetometric diagnostics.
- Programme for assessment, evaluation of technical conditions and justification of the residual life of underground trunk pipelines between the Spray Pools, Reactor Building and diesel generator stations under Kozloduy NPP Units 5 and 6 Plant Life Extension Project, completed within the period 2014÷2016.
- Procedure for field inspection of service water pipelines - essential loads of the Component Cooling Water System [CCWS];

The acceptance criteria for pipeline wall thickness are according to the codes specified in i.4.2 of "ПНАЭ Г 7-002-86 Rules and codes in nuclear power industry. Procedure for evaluation of equipment and pipeline strebgth properties, Moscow, Energoatomizdat, 1989".

The revised design and quantitative assessment of component cooling water system underground pipeline state, performed under the Project for Kozloduy NPP EAD Units 5 and 6 Life Extension within the period of 2014÷2016, are the following:

- analyses of static and dynamic strength and stability under seismic impact and by considering the real (measured) characteristics of the pipeline metal, as well as with the predicted thickness, for the end of operational life;
- inspection for ovalization and static stability of the section (diameter);
- analyses of overloaded sections with finite elements method.

Most of the verification calculations were performed with the software CESAR II which allows precise modelling and assessment of underground pipelines according to the codes of ASME BPVC SECTION – III, Division 1 Subsection ND, Class 3 Components, Rules for Construction of Nuclear Facility Components, 2007, for Level D.

Upon analysing the results and assessing their reliability and applicability, and according to the Conclusions of the technical condition and quantitative assessment of the residual life of the underground trunk pipelines between the spray pools QF10,20,30W01,02, Reactor Building and Kozloduy NPP Units 5 and 6 Diesel Generator Stations, was found that the mean corrosion thinning of Component Cooling Water System [CCWS] underground pipelines (after operation of more than 30 years) is currently 0,6 mm for all pipelines. The extrapolation to the end of the extended operational life shows that the corrosion thinning of all pipelines will be 1,2 mm.

The underground pipelines of the Component Cooling Water System [CCWS] have sufficient capacity and towards the end of their extended operational life, the minimum wall thickness will exceed 4 mm which meets the standards given in i.4.2 of ПНАЭ Г-7-002-86, Regulations for strength calculation of the equipment and pipelines of nuclear power plants.

The non-linear analyses, conducted according to the finite elements method for the overloaded sections, confirm that stress concentrations for all of the analysed sections are in very limited areas and fade away quickly. Developing non-linear deformations are limited and localized mainly in the areas of the welded seams. Developing deformations are of the order of 1%. This value can be assessed as acceptable because the formed (curved) components (which are overstressed and develop non-linear deformations) are made of ВСт3сп steel which has a clearly expressed yield site before secondary strengthening begins and after that it may eventually break.

4.1.3. Monitoring, testing, sampling and inspection of concealed pipelines

Monitoring of buried/underground pipelines condition cover the implementation of the following activities:

- Periodic visual inspections for leaks along the routes of buried/underground pipelines according to the Operating Procedure on component cooling water system for essential loads.
- Condition monitoring of penetrations and the visible parts of tubes in the shafts and manholes along the pipeline routes, the readings of instrumentation and control

systems according to the Operating procedure on the component cooling water system for essential loads.

- Monitoring of vertical deformation settlement of concrete supports of pipelines Dn 1000 from Spray Pools to the water intake shafts and diesel generator stations with the help of the installed reference marks according to the Instruction for Monthly Monitoring of Component Cooling Water System [CCWS] Buildings and Facilities for Essential Loads (civil-engineering and hydro technical part).
- Monitoring for occurrence of cracks in the walls when Dn 600 pipelines penetrate them and the Dn 400 pipelines from the Reactor Building to the spray pools and Dn 1000 pipelines from the spray pools to the water intake shafts and diesel generator stations in accordance with the Monthly Monitoring Instruction for buildings and facilities of the Component Cooling Water System [CCWS] for essential loads (civil-engineering and hydro technical part).
- Corrosion rate measuring under water treatment programmes according to the operating procedure on water treatment of essential load component cooling system in circulation cooling systems with spray pools.

Within the period 1992÷1999 the programme of the company "Nalko" was used for water treatment of circulation cooling systems. According to this programme, corrosion rate was measured every week with a portable device.

Since the year 2000 an on-line corrosion rate measuring is introduced – by the year 2007 the SCA-1 Corrater device was used under the programme of the company GE Betz, then a device of the company GE Water & Process Technologies is being used. The device is installed on the discharge pipelines on each of the three systems. The device triggers a signal on every 10 minutes to the automated computer system located in the carriages over the water intake shafts of the three systems. The stand is adjusted to measure the corrosion rate of carbon steel.

The corrosion rate on-line monitoring results are verified every year by placing the so-called "corrosion coupons" in a special stand for a period of three months which are then sent to a laboratory in Belgium to analyse the corrosion state.

4.1.3.1. Contactless magnetometric diagnostics

The analyses of the reasons for corrosion formation on the outer surface of the buried pipelines identify that the problem areas are on the sections which backfill is of bad quality. The presence of waste from building materials in the backfill impairs the insulation integrity and accelerates corrosion formation at these places.

Upon investigations into the methods that could be used for detection of places with impaired insulation without digging of the pipelines, a decision was made to perform contactless magnetometric diagnostics based on the measurements of earth magnetic field distortion caused by the change of tube metal permeability in the stress concentration areas (corrosion, pipeline insulation damage, signals due to the influence of the electric cables or metal items fallen during pipeline laying, etc.)

Examinations were performed by hired certified staff in compliance with Procedure for non-contact magnetometric diagnostics of pipeline technical conditions, РД 102-008-2002, developed in compliance with the Russian State Standards and Methodology for non-contact magnetometric diagnostics of operable heating lines СО ОАО "МТК", Moscow, 2009".

The results of the examination carried out in 2014 identified 6 sections with anomalies and increased concentration of stress - then, following the decision of the Specialized Technical Council, these sections were dug out and tested with non-destructive methods.

The examination carried out in 2017 did not identify any anomalies of the heaviest (first) category (with maximum concentration of stress and critical distribution of magnetic fields). It is recommended to dig out and assess (non-destructive testing) the sections with anomalies of category 2 that tend to reach category 1, of a total length of 127,2 m, which is 2,37% of the entire assessed length.

The anomalies in signals due to the influence of the electric cables are described in the reporting documentation. These sections of the buried pipelines are to be monitored because impaired outer insulation can affect corrosion rate.

4.1.3.2. Non-destructive in-service metal control

The non-destructive testing of underground pipelines of the component cooling water system is conducted at the accessible places (shafts and manholes), after digging out of sections when leaks were identified along the pipeline route, and when deviations are registered at conducting contactless magnetometric diagnostics. The following methods are applied: visual inspections, penetration testing (using the results of the visual inspection) and ultrasonic thickness measuring, and where possible - visual inspection of pipeline internal surface with remote control devices.

The Methods, non-destructive testing periodicity, regulatory requirements for testing and assessment of buried pipeline test results are regulated in the Procedure for field inspection of underground pipelines of the component cooling water system.

4.1.3.3. Specialised assessment of underground section

During the implementation of the plant lifetime extension (PLEX) project, a Programme for assessment of a compromised (replaced) section of buried Dn 1000 pipeline was implemented.

The following non-destructive testing activities were performed on the dismantled section of a Dn 1000 discharge pipeline and the following intolerable deficiencies were identified:

- ultrasonic thickness measuring on the surface of the scraped spots of the defective section without removing the depositions on the internal side of the pipeline;

- ultrasonic thickness measuring on the internal surface of the scraped spots upon removal of deposition, and on the outer side of the areas against the scraped ones;
- on the internal surface of the scraped spots of the area against defective sections;
- hardness inspection of a 50x50 mm section, cut from the defective section, and of the base metal of this section of the pipeline;
- spectral analysis (to specify the chemical composition);
- metallographic analysis of metal structure;
- analysis of metal mechanical properties; tensile tests, bending and impact strength;
- ultrasonic thickness measuring and thickness measuring of an item with dimensions of 50x50 mm with an incremental measuring device after polishing;
- ultrasonic thickness measuring of the outer surface of defective section of the item after sand blasting.

The following conclusions were drawn on the basis of the results of the specialized assessment of a dismantled section with dimensions of Ø1020x7mm:

- welded joints - there are no deformations.
- the results of the ultrasonic thickness measuring of the pipeline are as following:
    - the thickness of the pipeline is of wide margin, i.e. it sufficiently exceeds the calculated minimum value with technological additives (according to ПНАЭ Г-7-002-86);
    - the deposition which is built-up on the internal side of the pipeline does not substantially affect the results;
        - according to the metallographic analysis results, corrosion is insignificant;
        - cracks - none are registered.
        - the chemical analysis of the steel of the defective section demonstrates that it complies with the requirements for steel, 10Г2САФ according to БДС (BDS) 4880-79, according to the design.
        - the mechanical testing is within the allowances for spirally seamed pipe of Ø1020x7 mm under the BDS (БДС) 14479-78.
        - the analysis of the available corrosion on the outer side of the pipelines proved that corrosion appeared mainly in the areas which anti corrosion coating was damaged during laying or burying of pipes. It is evident from the visual inspection that in the areas with good insulation, the latter has good adhesion to the base metal and successfully performs its protective functions. It is evident from the visual inspections that regarding external corrosion spreading, the pipelines are in operable state and can remain operable during the time of the extended lifetime of system operation.

- the visual inspections of the internal surface of the pipeline identified depositions due to the lack of water treatment of the circulation cooling water in the spray pools with chemical reagents during the first years of operation, and also due to the availability of mechanical pollution according to the data presented in the "Water treatment analysis for circulation cooling systems with spray pools". The conducted analysis of depositions proved that the first, the most external layer, were depositions that could be easily removed. The second layer of depositions which was observed underneath, was comparatively difficult for removal. The third layer of depositions which is closest to the base metal is most difficult for removal - under it the base metal is not significantly damaged by the corrosion. it is evident from the chemical analysis of the depositions that limestone types of depositions prevail in the first two layers, and in the third, the closest one to the base metal, metal oxides are prevailing which serve as protection of the base metal.

4.1.4. Preventive and remedial actions for concealed pipelines

The following preventive and remedial actions are foreseen to guarantee the lifetime of pipelines during their extended period of operation:

- revision of two penetrations according to the Results of the technical condition and quantitative assessment of the residual life of underground trunk pipelines between the spray pools, Reactor Building and diesel generator stations under the lifetime extension project;
- conduct of continuous water corrective treatment for consumers important to safety in the circulation cooling systems with spray pools;
- enhancement of the monitoring system and registration of buried/underground pipeline condition in documents;
- building of inspection wells and piesometers;
- digging out and survey (non-destructive testing) of the sections with anomalies of category 2, tending to reach category 1 (the highest category with maximum stress concentration and critical distribution of magnetic fields), according to the recommendations of the contactless magnetometric diagnostics, conducted in 2017.
- conduct of periodic walkdown inspections of sections of the buried pipelines where the contactless magnetometric diagnostics registered anomalies of lower category (2), and the anomalies related to the signals due to the influence of the electric cables.

4.2. Licensee's experience in the application of AMPs for concealed pipelines

Kozloduy NPP experience in the application of the Ageing Management Programme for buried/underground pipelines demonstrates that:

- no other mechanisms of ageing of the base metal and welded connections of buried/underground pipelines components have been identified, apart from the ones presented in Table 2, i.4.1.2 of this Report. According to the results of Component Cooling Water System [CCWS] pipeline assessment with the help of non-destructive and destructive testing methods, the main degradation mechanism is due to general and pitting corrosion;
- the corrosion on the outer surface of the pipelines is spreading in the areas which anti corrosion coating was damaged during laying or burying of pipes. In the areas with good insulation, the latter has good adhesion to the base metal and successfully performs its protective functions;
- it was observed during the survey of the internal surface of the sections of the buried pipelines that depositions were evenly distributed and consisted of three layers according to the chemical analysis (limestone types of depositions prevail in the first two layers, and in the third which is the closest to the base metal, metal oxides prevail), and that depositions do not significantly affect corrosion processes;
- the application of contactless magnetometric diagnostics to the pipelines provides additional information about the state of the buried pipelines without having to dig them out. The reliability of the method will be identified upon digging out and assessing the sections with registered anomalies and by the contactless magnetometric diagnostics conducted in 2017.

4.3. Regulator's assessment and conclusions on ageing management of concealed pipelines

The developed Lifetime Management Programme for Kozloduy NPP Units 5 and 6 Component Cooling Water System [CCWS] Buried Pipelines addresses ageing management and considers the results from the implementation of operational, maintenance, surveillance, water chemistry and in-service inspection programmes.

The in-service activities for the buried/underground pipelines are performed according to work programmes, covering visual inspections, television image control, ultrasonic testing and thickness measuring of the underground pipelines in the accessible sections/in the shafts and assessment of the pipeline sections after digging out. Measuring of the thickness of these sections provides general information about the state of the metal of the entire service water system pipelines;

An Operating procedure for Units 5 and 6 Component Cooling Water System [CCWS] pipelines is developed. The procedure regulates the method, periodicity and scope of buried pipelines non-destructive testing, regulations for implementation of monitoring and assessment of test results.

The activities under corrosion level monitoring of underground pipelines are performed under the implementation of water treatment programmes;

Periodic visual inspections are conducted and pipeline routes are monitored according to Instruction for monthly monitoring of the buildings and facilities of the Component Cooling Water System [CCWS] for essential loads (civil engineering and hydro technical part), and the Operating procedure for Component Cooling Water System [CCWS] for essential loads;

Apart from the conventional methods for non-destructive testing (NDT) (visual inspection, television remote inspection and ultrasonic thickness measuring), Kozloduy NPP applies a new method for contactless magnetometric diagnostics which provides additional information about the state of the service water system buried pipelines;

Currently applied assessment and non-destructive testing (NDT) methods confirm the availability of the dominant ageing mechanisms described in the Programme for buried/underground CCWS pipeline lifetime management;

Within the framework of Kozloduy NPP EAD Units 5 and 6 Lifetime Extension Project, revised design and quantitative assessment of component cooling water system pipeline condition were completed on the basis of the data from the specialized assessment and non-destructive testing (NDT) and metal ageing mechanisms were considered.

It is evident from these assessments that the CCWS pipelines can be operated within the licensed period provided the prescribed corrective measures are implemented and monitoring and operating procedures are observed.

The activities for buried pipeline ageing management are also monitored under the scope of conducted inspections. The scope of reviews covers condition, organisation and activity monitoring during operation and maintenance of the service water supply system. It is evident from the results that the activities related to the pipeline ageing management are effectively monitored. The necessary technical and organisational measures for establishment and mitigation of ageing effects, and for reassessment of pipeline life characteristics are being implemented.

The licensing condition to submit a report on the results of equipment, including buried pipelines, residual lifetime assessment within one month after the unit restart upon the completion of an outage, is complied with.

The activities performed under Kozloduy NPP Units 5 and 6 CCWS buried/underground pipeline ageing management comply with the requirements of the Regulation on Ensuring the Safety of Nuclear Power Plants, the IAEA Safety Standards and with other internationally accepted documents.

5. PRESSURE VESSELS

5.1. Description of ageing management programmes for RPVs

A RPV ageing management programme is developed to coordinate the process of ageing management for reactor pressure vessels. The programme describes the set of activities performed by the plant in order to assess, prevent, identify, control and mitigate the consequences of the specific ageing effect on the structures, systems and components or group of components of reactor pressure vessels.

The objectives of the Programmes are as follows:

- present the set of organizational, technical and methodological actions focused on ensuring or reassessment of lifetime characteristics of the item established in the regulatory and technical requirements or design documents;
- provide for a clear allocation of responsibilities among the plant organizational structures for performance of the activities ensuring lifetime of reactor pressure vessels;
- presents the deadlines and tools for ensuring lifetime of reactor pressure vessels at Units 5&6 at Kozloduy NPP;
- provide for keeping and developing documentation for lifetime management of reactor pressure vessels as well as its lifetime extension.

The programme is based on the knowledge and the understanding of the degradation processes of material properties, resulting from ageing factors, inherent to the nuclear power plant in operation. Mechanical degradation leads to decrease in lifetime, which is one of the equipment reliability indicators.

5.1.1. Scope of ageing management for RPVs

The scope of the programme covers the reactor pressure vessel and the upper head. Critical locations subject to surveillance are: seal arrangement, external and internal surfaces of the reactor pressure vessel and the reactor head, base metal of components and welds.

The reactor pressure vessel structure incorporates its main components: flange, cylindrical rings, elliptic bottom and nozzles.

The diagram of the reactor pressure vessel and the upper head is given in Appendix 1.

5.1.1.1. Methods and criteria for selecting components within the scope of ageing management programme for reactor pressure vessels

The requirements to the process of selection of components included in the scope of the Programme are developed on the basis of the IAEA recommendation to plant ageing management, in accordance with the documents described in Section 2.3.1.2.

A safety-based approach is adopted, according to which the scope is defined based on whether the relevant components are related to ensuring safety of nuclear facilities. The safety classification is taken into account in the process of selection.

The selection of components is based on:

- design basis data, including applicable codes and standards, and requirements imposed by the regulatory body;
- data from operating and maintenance history, including surveillance and oversight;
- data from inspections;
- data from international experience in studies on the reactor pressure vessel metal condition.

5.1.1.2. Processes and procedures for selection of ageing mechanisms for the different materials and components of reactor pressure vessel

The processes and procedures for selection of ageing mechanisms for the different materials and components of reactor pressure vessel are described in Section 2.3.2.3 of this Report.

Dominating degradation mechanisms of equipment metal are identified, based on the results of the analysis of technical documentation, design specifics, manufacturing technology and operating condition, as well as the results from in-service inspections of the RPV. , , these degradation mechanisms determine its technical condition and residual lifetime.

The degradation mechanisms are identified in the process of operation based on the results of the in-service inspection of base metal and welds through studying the structure and properties of materials.

Table 3 contains the main potential degradation mechanisms of base metal, welds and cladding of the reactor pressure vessel, which can influence its operability.

Table 3. Potential degradation mechanisms of the reactor pressure vessel

Equipment description	Equipment qualification	Potential degradation mechanism
Reactor pressure vessel	Welds Base metal	Radiation embrittlement Thermal ageing Fatigue
	Flange surface Stud holes	Local corrosion Stress corrosion
	Cladding	Intercrystalline corrosion Radiation embrittlement

Equipment description	Equipment qualification	Potential degradation mechanism
		Fatigue

For the identification of ageing mechanisms of the different materials and components of reactor pressure vessels, the following parameters of their technical condition are used:

Characteristics of mechanical properties of base metal, metal of welds and cladding, including embrittlement are determined through:

- hardness measurement
- specimen testing
- comparison of results of the previous metal in-service inspection
- analysis of the data from technical documentation taking into consideration the regulatory values.

The assessment criteria are in compliance with the requirements of the PNAE G-7-002-86 Strength Calculation Limits of Equipment and Pipelines in Nuclear Power Plants, 1989.

Flaws (defects), their location and geometric dimensions (length, depth, location, spatial orientation) are defined through:

- analysis of the data from technical documentation
- methods of non-destructive examination of reactor pressure vessel.

Assessment criteria:

- The requirements of the Russian code Equipment and Pipelines in Nuclear Power Plants. Welds and Cladding. Inspection Rules, PNAE G-7-010-89, are applied for assessment of quality of welds and corrosion protection weld overlay
- The assessment of base metal quality is performed according to codes (standards) described in the inspection procedure.

Since November 2017, the assessment has been performed according to НП-089-15 Federal Norms and Rules in the Area of Usage of Nuclear Energy. Rules for Arrangement and Safe Operation of Equipment and Pipelines in Nuclear Power Plants, Rostechnadzor, Russia, 2015.

Monitoring that the values of reduced stresses in the components of the reactor pressure vessels are kept within of the design limits.

The assessment criteria for not exceeding the relevant allowable stress values are in accordance with the requirements of the Russian Code PNAEG-7-002-86 Norms for Strength Calculations of Equipment and Pipelines in Nuclear Power Plants, Moscow, „Energoatomizdat“, 1989” and
РД ЭО 1.1.2.05.0330-2012 Guide for Strength Calculation of Equipment and Pipelines of RBMK, WWER and EGP Reactor Installation in Operation Including Long Term Operation. OAO Concern Rosenergoatom SA, 2012”;

The number of loading cycles during normal operation, abnormal operation, design basis accidents, both for the time of operation and their predicted number for the extended life time is defined through:

- analysis of the data from technical documentation
- expert assessment of the number of loading cycles for the period of extended operation

Assessment criteria - not exceeding the number of loading cycles by the end of the extended time of operation

The actual cumulative and predicted fatigue damage for the extended operation time are calculation parameters.

Assessment criteria - not exceeding the allowable value of the cumulative fatigue damage by the time of expiry of the extended life time.

Geometric dimensions of the reactor pressure vessel components are defined with the analyses of the results from reference measurements.

Assessment criteria - compliance with the requirements of the design documentation and passport data.

Critical brittle temperature of reactor pressure vessel

Assessment criteria - not exceeding the maximum allowable brittle temperature of reactor pressure vessel metal (defined by the calculation of strength of material compared to brittle fracture) by the end of the extended operation/life time.

Table 4 describes the dominating parameters for the condition of base metal, welds and cladding of reactor pressure vessel components.

Table 4. Dominating parameters for the condition of metal of reactor pressure vessel components

Monitored effect of metal degradation	Dominating parameters of the metal condition
change in the mechanical properties	temporary strength, yield strength relative elongation, relative narrowing
embrittlement	critical brittle temperature T_{K},, plane strain fracture toughness K_{IC}
crack growth	geometric dimensions of cracks (length, depth, opening, spatial orientation
pitting (pitting corrosion)	area of damage, defects per unit of area, depth of defects
structure change	phase distribution, phase modification growing of microflaws (micro cracks, pores, condition of the grain limits

5.1.2. Ageing assessment of RPVs

5.1.2.1. Ageing mechanisms requiring management and identification of their significance

The main monitored degradation mechanisms ranked according their significance are as follows:

- fatigue
- radiation embrittlement
- thermal ageing
- local corrosion
- stress corrosion
- intercrystalline corrosion.

5.1.2.2. Establishment of acceptance criteria related to ageing mechanisms

The assessment of degradation of critical items of the reactor pressure vessel is performed though comparison of the design requirements and the information obtained from:

- flaws and failures occurred during operation
- deviations in the operating modes of reactor installation
- results of the applied techniques for non-destructive examination and expert in-service inspection of rector pressure vessel
- conclusions of the existing analyses and assessments.

When the actual values for the dominating parameters of the metal condition of reactor pressure vessel comply with the values set in the design documentation and current regulations, and actual loading parameters and their number complies with the design ones (but does not exceed them), it is accepted that the criteria will be satisfied.

5.1.2.3. Key standards and guidance used to prepare the programme for reactor pressure vessels

The methodology for development of the programme and its structure complies with the IAEA requirements.

The basic standards and guidelines used for development of the Ageing Management Programme are given in Section 2.2.

For development of the Programme, the Russian regulation Fundamentals of Ensuring Safety of Nuclear Power Plants, ОПБ-88/97 (PNAE G -01-011-97, NP-001-97), Gosatomnadzor, Russia, 1997, was also used.

In the developed Life Management Programme for reactor pressure vessels of Units 5&6 at Kozloduy NPP, data for the design parameters, data for operating and maintenance history, including supervision and oversight, data from inspection, data from the international operating experience in studying the condition of metal of reactor pressure vessel and welds of similar designs are included.

The programme is developed to provide for the full integration with the existing plant programmes. At the same time the data in the programme are considered when developing new operating procedures and programmes for maintenance, modernizations, testing. The programme for reactor pressure vessels is related to a number of documents of the Kozloduy NPP plc including:

- Technical specifications for safe operation of Unit 5(6) of Kozloduy NPP with WWER-1000 (B320) reactor;
- Long-term schedule for maintenance of main and auxiliary equipment at Unit 5&6; съкращението липсва в списъка със съкращения
- Units 5&6 Life Management Programme
- Reactor installation operating procedure
- Procedures for maintenance and repair
- Inspection procedures
- Procedure for testing of the safety systems at Unit 5(6)
- Procedure for periodic technical reviews of facilities with increased risk, which are of significance to nuclear safety
- Procedure for preparation of reports for neutron-physics characteristics of the WWER -1000 reactor

5.1.2.4. Compliance of the Programme for reactor pressure vessels and the operating procedures

The ageing management of reactor pressure vessels includes adherence throughout the entire period of operation to all the restrictions such as not exceeding the allowable values of mechanical or physical loads and providing their operability according to Technical Specifications of Units 5 and 6. The Technical Specifications include the design margins and conditions for safe operation, rules and main principle for safe operation and the general sequence of safety related operations.

Exceeding the allowable operating parameters such as temperature, pressure, rate of changing the operating parameter, water chemistry could lead to acceleration of ageing processes and premature degradation of mechanical properties.

The operator's practices have an influence on the operating parameters of the reactor installation and operator's actions have a significant role when implementing the programme for decreasing the degradation of the metal of the reactor pressure vessels through control of the allowable limits.

5.1.2.5. Compliance of the Programme for reactor pressure vessels and the maintenance and repair procedures

There is an established and reinforced practice for performance of planned annual outages at Units 5&6 at Kozloduy NPP Plc. A long-term schedule for maintenance of main and auxiliary equipment at Unit 5&6 equipment is prepared and updated annually. The long-term schedule covers and contains the distribution (per years) of the type preventive activities (types of maintenance and planned modifications) of the main equipment at Units 5&6, and its Section 1.1. is related to reactors. The long-term schedule defines the priorities and coordinates the preventive maintenance activities. This document is kept updated and reports such as completed work reports, installation reports, single test and multiple test reports, non-destructive examination reports, reports for implemented modifications, and other current documents at Kozloduy NPP plc are used as feedback.

5.1.2.6. Compliance of the Programme for reactor pressure vessels and the in-service inspection procedures

Non-destructive examination of the components of the reactor pressure vessels in operation is performed in a planned manner, during the planned annual outages, and inspection results are used for the assessment of the current condition of base metal and welds of the reactor pressure vessels, as the main objective of the non-destructive examination is the detection, localisation and sizing, and monitoring of the flaws in the reactor pressure vessels.

5.1.2.7. Key projects used for preparation of the Reactor Pressure Vessel Programmes

Kozloduy NPP takes part into the following international projects which are related to the ageing management of reactor pressure vessels:

- Joint activities for improvement of the experimental and regulatory and methodological justification for ensuring the lifetime extension of the reactor pressure vessels of WWER-1000 reactors were carried out with Russian engineering companies.
- IAEA project - preparation of regulatory (technical) documents for ageing management of plant equipment (IGALL International Guide Ageing Learned Lessons).

The objective of the IGALL project is to support the management of nuclear power plants and regulators to sustain the required level of safety during plant operation taking into consideration equipment degradation mechanisms as well as prepare new guidelines for criteria and practices of the separate countries applicable during plant lifetime extension.

5.1.2.8. Research and Development Programmes

The purpose of the programme for studying the specimens is to validate the conservatism of the relation of the material property change of reactor pressure vessel under the impact of the operating factors and recording of the inspection results. This programme allows for receiving experimental information about the radiation and thermal embrittlement and is a basis to clarify the regulatory relations designed for assessment of the radiation and thermal fracture of the materials of the reactor pressure vessel. The study and testing of the specimens is performed by the Bulgarian Academy of Science (BAS). Additionally, BAS provides also scientific and technical assistance for preparation of the ageing management programme for reactor pressure vessels of Units 5&6 at Kozloduy NPP.

5.1.2.9. Internal and international operating experience

The information from other plants with WWER-1000 reactors includes analysis and application, if appropriate, of the results of the external and internal operating experience for ageing management of the RPV components.

The activities and responsibilities for the review and assessment of the internal and external operating experience are described in details in Chapter 2 of this Report. Disclosure to personnel and application of the operating experience from other plants is performed according to the procedure for the application and disclosure of the operating experience.

The information obtained by international scientific experience and participation of the personnel in workshops, conferences and training is used for comparison, analysis and determination of preventive and corrective measures, as well as for prompt update of Programme for reactor pressure vessels.

- Internal operating experience

In the frame of lifetime extension project calculations of the temperature fileds and stresses, static and cyclic yield strength, and brittle fracture strength of the reactor pressure vessel of Unit 5 were performed, taking into consideration the predicted fluence of fast neutrons on reactor pressure vessel for the period of extended operation.

The calculation analyses are performed for combinations of the following loads and modes:

- normal operation modes
- abnormal operation modes
- design basis accidents
- design basis earthquake
- safe shutdown earthquake

As a result of performed calculations, new temperature regimes of the metal of reactor pressure vessel for hydraulic tests and number of loading cycles of the equipment of reactor installation were defined. The ageing management programme for reactor pressure vessels was revised in 2016.

The same activities are under implementation for Unit 6 at Kozloduy NPP in accordance with the lifetime extension project.

- External operating experience

Following the information for defects, detected in the base metal of the reactor pressure vessels in the Belgian Nuclear Power Plant Doel 3 and Tianz 2 in 2012, and in accordance with WENRA's recommendations to the nuclear safety authorities, the materials and the structural integrity of reactor pressure vessels of PWR/WWER reactors fabricated prior to 1985 were verified. In the course of this verification, the available data, including the NDE methodologies and methods, were analysed and the quality of base metal of the reactor pressure vessels was reassessed. The results of the reassessment show that:

- no indications were detected in the base metal of the reactor pressure vessels at Units 5&6 with the non-destructive examinations in the scope indicated in the in-service inspection procedure, as well as with the additional ultrasonic inspection of the base metal of the rings;
- due to the fact that the reactors of Units 5&6 at Kozloduy NPP are of a newer generation, and are manufactured according to a different technology, the occurrence of defects of hydrogen flocken type of base metal of reactor pressure vessels is impossible.
- The qualification of the systems for inspection of reactor pressure vessels at Kozloduy NPP is performed by an independent qualification body in compliance with the adopted international norms by the IAEA and ENIQ.

Based on the above findings, conclusion is made that there is no need to increase the scope of inspections beyond the frames of the existing in-service inspection programmes.

5.1.3. Monitoring, testing, sampling and inspection activities

The monitoring and inspection of the condition of reactor pressure vessels is provided through:

5.1.3.1. Water chemistry Control

Kozloduy NPP maintains optimal water chemistry in order to slow down the process of corrosion of construction materials, to reduce the deposits on the heat exchanging surfaces, to increase the lifetime of equipment, reduce the rate of build-up of radioactive products and decrease the dose exposure to the personnel.

The water chemistry control is performed according to the requirements of the operating procedures and requirements of the regulating and operating documentation.

Based on the procedures and approved scope of maintenance activities, a specific programme for corrosion inspection of the equipment is prepared at the beginning of every planned annual outage of the units.

According to the requirements of the Procedure for water chemistry, the results of chemistry control are assesses and analysed on a daily basis, monthly basis, and reports for the condition of water chemistry of primary and secondary circuit at Units 5&6 are developed accordingly. Monthly and annual reports as well as fuel cycle reports for the maintained water chemistry of primary and secondary circuit at Units 5&6 are prepared. In the reports is included analysis of the water chemistry data, causes for deviation and /or disturbances, if any, and performed corrective actions.

5.1.3.2. Monitoring of the cumulative nuclear fluence

The monitoring of the accumulated neutron flux is performed in accordance with the Procedure for preparation of reports for neutron-physics characteristics of the reactor.

Radiation loading of the reactor pressure vessels is defined through combination of measurements and calculations. The neutron flux for given areas of reactor pressure vessel and locations of the specimens is defined through calculation of the absolute values. The calculation results are compared to the measured results of the specimens and ex-vessels neutron detectors located and exposed on the outer side of the reactor pressure vessel. The Russian methodology Methodology for Neutron Control of the External Surface of Pressure Vessel of WWER-type Reactor , RB-018-01, 2001, is used.

5.1.3.3. Monitoring of the process of change in the mechanical characteristics

Monitoring the process of the change in the mechanical characteristics of the base metal and welds of the reactor pressure vessel is performed according to the programme for removal and testing of specimens.

The standard programme for specimens provides for receiving experimental information about the radiation embrittlement of the material of reactor pressure vessel and is a basis for clarification of the regulatory relations, for assessment of radiation damage of materials.

Data obtained during testing of specimens provides for correct information about the properties of materials of reactor pressure vessel both at the initial condition and in the process of operation. The results are reliable and representative, which allows definite determination of the required parameters, for example critical brittle temperature of the metal of reactor pressure vessel.

In order to define the actual changes in the mechanical properties of the reactor pressure vessel metal (yield strength, temporary strength, relative elongation, and relative narrowing) and characteristics of resistance against brittle fracture (critical temperature of ductile to brittle transition of the metal), specimens are periodically removed from the reactor and tested in specialized laboratories.

The list of the characteristics defined through specimens, their installation location and method of fixing are shown in the design documentation of the specimens. The number of specimens for every type of testing, survey during exposure with indicators of the neutron flux and temperature, deadline for their testing and accountancy of the testing results are also established there.

5.1.3.4. Non-destructive examination of reactor pressure vessel

The non-destructive examination of the reactor pressure vessels is carried out according to the Long-term schedule for preventative maintenance and repair, and work programmes for in-service inspection of the primary equipment and pipelines, which are developed for the related planned annual outage. The work programmes are developed in compliance with the approved Procedure for in-service inspection. The procedure provides for the methods, regulations, scope and frequency of the non-destructive examination of the components of the reactor pressure vessels.

The following non-destructive examination methods are applied to the reactor pressure vessels: visual inspection, TV remote inspection, dye penetrant inspection, manual and automatic ultrasonic inspection, eddy current inspection.

The non-destructive examination according to the Procedure for in-service inspection is performed from outside and inside the reactor pressure vessels. The frequency of outer-surface non-destructive examinations (visual inspection, dye penetrant inspection, manual and automatic ultrasonic inspection) is maximum 30 000 hours (4 years) in operation, and inside-vessel (manual and automatic ultrasonic inspection, eddy current inspection) - maximum 60 000 hours (8year) in operation. The TV remote inspection of the inner surface of the reactor pressure vessels is performed within 30 000 hours in operation. Non-destructive examination (visual inspection, dye penetrant inspection, manual and automatic ultrasonic inspection) of the flange of reactor pressure vessel is carried after every reactor opening.

The scope of automated ex-vessel non-destructive examination includes the following critical items:

- ultrasonic inspection and TV remote inspection of welded joints No.1÷4*
- ultrasonic inspection of the ring (600mm) opposite the reactor support structures including the welds of the supports of the reactor pressure vessel
- the TV remote inspection of the base metal of the ring opposite the reactor core
- ultrasonic inspection and TV remote inspection of the base metal of the bottom

Ultrasonic inspection is performed for the entire thickness of the component including the boundary with the cladding.

**COMMENT: Due to the presence of the support ring the scanning (ultrasonic inspection) of weld joint No. 4 is one-sided. Visual inspection, dye penetrant inspection, manual and automatic ultrasonic inspection) are performed on welds No 5- 7.*

In 2002, a continuous ex-vessel scanning of the cylinder surface of the reactor pressure vessel of Unit 6 was performed. Starting from the weld of the bottom, the lower ring, and to level 1000mm under the support ring, i.e. ultrasonic testing of 100% of the base metal of the lower ring (between welded joints No.2 and No.3) and about 80% of the base metal of the upper ring (between welded joints No. 3 and No. 4) was performed.

The scope of automated non-destructive examination of the reactor pressure vessel from inside includes the following critical items:

- ultrasonic inspection of welds No 1÷7 including the cladding;
- ultrasonic inspection, eddy current inspection and TV remote inspection of the cladding of the ring (1000 mm) opposite the reactor core;
- ultrasonic inspection and TV remote inspection of the cladded surfaces of the nozzles Dn 850 including the welds of the reactor pressure vessel to main coolant line Dn 850
- ultrasonic inspection, eddy current inspection and TV remote inspection of the cladding of the radius transitions of nozzles Dn 850.
- The TV remote inspection of the entire cladding of the reactor pressure vessel including the welds of the protective shells of the nozzles of the Emergency Core Cooling System, the supports to the reactor pressure vessel, dowels of the supports, contact surfaces of the dowels and pins.

The ultrasonic inspection of the cladding includes: weld overlay, area of melting of the base metal/welded joint in depth of up to 25mm. The eddy current inspection is performed for surface and subsurface defects with depth up to 4 mm.

The non-destructive examination of metal is carried out in accordance with the procedures for visual inspection, TV remote inspection, dye penetrant inspection, manual and automatic ultrasonic inspection and eddy current inspection, and specialized methods developed for automated ultrasonic inspection, eddy current inspection and TV remote inspection for the reactor pressure vessel.

Programme for qualification of the non-destructive examination systems (procedures, equipment and personnel) of safety important components is implemented. The outside and inside Reactor Pressure Vessel ultrasonic inspection and inspection of the Dn850 nozzles to the reactor pressure vessel are qualified. The qualification is based on the IAEA methodology (IAEA-EBR-WWER-11 Methodology for Qualification of ISI Systems for WWER Nuclear Power Plants. 1998) and European network for inspection and qualification ENIQ (EUR 17299 EN. European Methodology for Qualification of Non-Destructive Testing, (Third Issue) 2007). The Certificates for the applied inspection procedures are recertified periodically, if there are changes in important parameters. The personnel performing the inspection are recertified in 5 years period according to the Quality System adopted by the Qualification Centre.

The assessment criteria of the non-destructive examination results are according to the regulations listed in Section 5.1.1. of this Report.

For the registered indications, a special monitoring regime has been introduced. After every inspection, a comparative analysis of the indications is carried out. The analysis is included in a separate appendix to the Completion Report of the work programme for non-destructive examination. Currently, no changes of the monitored indications are recorded.

As part of the measures for lifetime expansion of Units 5&6, calculation of the allowable size of the recorded indication in the welds of the reactor pressure vessels is performed. The calculations are performed in the frames of the activities under the PLEX project of Units 5&6 at Kozloduy NPP, during which was proven that the indications would not exceed the allowable limits.

5.1.3.5. Monitoring of the cyclic yield strength

The cyclic yield strength of the reactor pressure vessel is monitored on the basis of the requirements considering the relation of the allowable loading cycles and actual loading cycles (including vibration loading) for the entire period of operation.

The number of allowable loading cycles including the reactor pressure vessel in different operating modes with normal operation, abnormal operation and design basis accidents is established in the regulatory, design and operating documentation. The actual number of loading cycles as one of the leading parameters for the technical condition of the reactor pressure vessel condition and allowable number of loading cycles defines the residual lifetime of the reactor pressure vessel under cyclic loading.

The input for the calculation of the values for the cumulative fatigue fracture of the components of reactor pressure vessel is the transient modes and loads causing cyclic stress, number of their recurrences and characteristics required for the calculation of tension, as well as data for vibration loading of the components of the reactor pressure vessel for the period of operation of reactor installation.

The number of operation and load modes, data for their recurrence are defined from the analyses of the operation of the reactor installation or through expertise. As a result, the conditions of the operating cyclic loading in the most loaded areas of the components of the reactor pressure vessel are identified – tension amplitudes and number of cycles of different type for the period of operation.

5.1.4. Preventive and Corrective Actions for reactor pressure vessel

The system of preventive and corrective actions established in the plant is described in Section 2.3.4 of this Report.

5.1.4.1. Preventive Actions for reactor pressure vessel

The preventive actions performed in the process of reactor pressure vessel ageing management at Kozloduy NPP plc are developed in the frames of the reactor pressure vessel programme. These activities are presented in a short in Table 5,

Table 5. Preventive actions performed in the process of reactor pressure vessel ageing management

No.	Actions for reactor pressure vessel	Reference ageing mechanism	Guiding documents	Final document	Implementer
1	Maintenance activity	Fatigue Local corrosion, Radiation embrittlement	Procedure for maintenance and repair of primary circuit main equipment, WWER-1000 reactor	Reports for performed maintenance and repair	Organizational Structure at Kozloduy NPP
2	Non-destructive examination NDE	Material fatigue, local corrosion, stress corrosion	1. In-service inspection programme 2. In-service inspection procedures	Reports, NDE Records	Organizational Structure at Kozloduy NPP
3	Testing of specimens	Thermal embrittlement, Radiation embrittlement	Time schedule for removal of specimens at Kozloduy NPP (according to Ageing management programme for reactor pressure vessels	Contract reports	Engineering companies
4	Loading of reactor core applying partial low leakage pattern	Radiation embrittlement	Administrative procedure Preparation of reports for neutron-physics characteristics of the WWER -1000 reactor	Neutron-physics characteristic Report	Organizational Structure at Kozloduy NPP
5	Monitoring of corrosion condition	Local corrosion, stress corrosion	Work programme for monitoring of corrosion status of primary and secondary circuits equipment for WWER 1000	Corrosion condition record	Organizational structure at Kozloduy NPP
6	Monitoring of the cumulated nuclear fluence	Radiation embrittlement	Time schedule for long-term activities for reactor pressure vessels, Units 5&6 (according to the programme for reactor pressure vessels)	Fuel cycle reports	Organizational structures of Kozloduy NPP , Engineering companies

No.	Actions for reactor pressure vessel	Reference ageing mechanism	Guiding documents	Final document	Implementer
7	Technical assessment		Procedure for safe maintenance of vessels and pipelines in Reactor Building of Units 5&6 and Auxiliary Building -3	Technical assessment reports	Organizational structure at Kozloduy NPP
8	Maintaining optimal water chemistry	Local corrosion, stress corrosion	1. Operating procedure Water chemistry of primary and secondary circuits and clean-up systems 2. Maintenance procedure Procedure for monitoring the corrosion condition of primary and secondary circuits equipment at WWER 1000 plants	Forms and records	Organizational structure at Kozloduy NPP
9	Hydraulic testing of strength	Local corrosion, stress corrosion Fatigue	Set of operating procedures for Units 5&6 start-up	Completion reports	Organizational Structure at Kozloduy NPP
10	Monitoring during operation	Fatigue	Operating procedure for the use of nuclear application software for the Ovation Control Information System, Operating procedure for RPV Large-scale temperature monitoring system .	Forms, records and reports for diagnostics systems	Organizational structures at Kozloduy NPP

5.1.4.2. Corrective measures

In the event that a structure or component of the reactor pressure vessel does not satisfy the acceptance criteria, corrective actions will be taken, and they are described in the Programme for reactor pressure vessels with the required level of completeness. The corrective actions including the identification of the root cause and prevention of recurrence are prompt and are performed in compliance with the applicable norms and standards approved by the regulator, BNRA.

The identified flaws (defects) are analysed and resolved through qualified techniques for maintenance and repair, new maintenance technologies and materials are applied, and corrective actions are taken in order to suppress ageing mechanisms and mitigate the consequences of their effects and slow down the process or reverse the trend. The programme for reactor pressure vessels defines the following corrective actions:

- assignment and performance of additional analyses (including strength analyses, PTS, neutron fluence) by competent engineering companies and scientific institutes, and if required, in compliance with the results of analyses, engineering or component replacement

- change in the operating modes, practices and functional tests
- modification (change) of water chemistry
- additional assessment of the condition of metal of reactor pressure vessels through methods of non-destructive examination or expert methods for in-service inspection for the purpose of operating diagnostics of the cause for occurrence and distribution of flaws in the critical areas of the flaws in critical areas of equipment. Organization and performance of supervisory (oversight) non-destructive examination. Update of the programme for specimens at Kozloduy NPP plc.
- Location of the addition monitoring sensors whose indications together with the data received from the stationary sensors (pressure, temperature, flow rate, etc.) provides for obtaining the maximum complete information for the condition of the reactor pressure vessel in order to develop the additional measures for protection of the reactor pressure vessel against adverse factors.

One of the main corrective actions at Kozloduy NPP plc is Maintenance strategy for the reactor pressure vessels at Kozloduy NPP plc. For development of maintenance strategy for the reactor pressure vessels, the following stages are included and performed:

- *Diagnostics* – the identified flaws or indications are subject to analysis according to the flow chart given in Figure 2.

Stressor → **Ageing mechanism** → **Ageing effect** → **Key condition parameter**

Figure 2. Flow chart of the process of diagnostics of an item with flaws

- *Prediction* – assessments of the technical capabilities for performing of maintenance and its cost efficiency.
- *Time schedule* – forthcoming maintenance activities are ranked according to their priorities.
- *Selection of maintenance method* – selection of method depends on the nature of damages and flaws, applicability and cost.
- *Preparatory maintenance activities*– preparatory maintenance activities are related to the provision of the qualified and trained personnel, use of qualified techniques for maintenance and repairs, development and application of new maintenance technologies and materials.
- *Adequate application of maintenance measures*

5.2. Licensee's experience of the application of AMPs for RPVs

5.2.1. Operating Experience

The change in the properties of the reactor pressure vessels of WWER-1000 reactors during operation is monitored through periodic testing of specimens. For this purpose the manufacturer developed and Kozloduy NPP applied the so called Typical/Standard Programme for Monitoring of the Reactor Pressure Vessels. The programme defines the types and number of specimens, their material, scope of mechanical testing and reference testing parameters, as well as the frequency of testing.

The operating experience acquired during the performance of the standard programme for monitoring the WWER-1000 reactor pressure vessels, as well as the new scientific studies and developments in the area of study of the occurrence and progress of degradation processes in the materials of reactor pressure vessel show that that the standard programme of the specimens for reactor pressure vessels also in other nuclear power plants with WWER-1000 reactors can be modernized.

5.2.2. Applied changes in the Programme as a result of operating experience

Determination of the fluence of fast neutrons in the reactor pressure vessel is fundamental for the assessment of the change in the properties of material, which provides for the integrity of reactor pressure vessel in all modes of operation and is the base for assessment of the residual lifetime of the installation. Fast neutrons induce processes in the metal of reactor pressure vessels that result in increasing the strength and decreasing the plasticity and ductility of metal. For the assessment of mechanical properties of the metal of the reactor pressure vessel and welds, the neutron flow with energy beyond 0.5 MeV (fluence), which is integral by time, is used. For the assessment of the accuracy of the calculated neutron fluence in the reactor pressure vessel, data from calculated and measured activities of the specimens and detectors in the standard assemblies are used.

The following was performed:

- experiments to identify the behaviour of the metal of factory weld with high nickel content (1.7%) upon exposure to beyond design neutron fluence
- PTS analysis and determination of the allowable critical brittle temperature of reactor pressure vessels of Units 5&6
- assessment of the thermal ageing effect of the material of reactor pressure vessels . Thermal specimens from Units5&6 were removed and tested.

The purpose of performed tests is to identify the level of change in the properties of the metal of reactor pressure vessels at Unit 5&6 as a result of thermal effect by the time of removal of the specimens from the reactors.

The received new testing results together with the data base from the previous testing of specimens from these units provide for more precise prediction of the change in the properties of metal required for justification of the design and beyond design operation of the units.

Based on the obtained results, a change in the standard programme for monitoring of the reactor pressure vessels and change in the Time schedule for removal and testing of specimens, which is an Appendix to the Ageing management programme for reactor pressure vessels, was made.

5.2.3. Assessment of the reactor pressure vessels for the needs of extended operation

In the frames of the large-scale programmes for assessment and justification of the plant extended operation, a number of measurements, studies and monitoring, as well as analysis of the obtained results were carried out.

Radiation embrittlement

The cumulative fluence in the reactor pressure vessel is monitored through measurements, analytical calculation and periodic study of the irradiation specimens (mechanical, thermal and ionizing). In the frames of the PLEX project, the fluence values were calculated. The calculated fluence considering the new fuel type and the operation at 104% power will reach approximately 40% of the design values by the end of the extended operation period. This is due to the use of low leakage fuel loading pattern, which lead to significant decrease in the neutron fluence on the reactor pressure vessel wall.

Thermal ageing of reactor pressure vessel

In the process of operation the reactor pressure vessel is subject to the effects of increased temperatures (290 – 322°C), which leads to development of thermal ageing and change in the mechanical properties and characteristics of metal. This effect was considered by the strength calculation of the reactor pressure vessel.

Intercrystal corrosion, local corrosion

The inner surface of the reactor pressure vessel and flange surface are cladded with a two and three layers of austenite cladding.

According to the results of in-service inspection, no defects causing local (pitting) corrosion of the inner surface of the reactor pressure vessel and flange surface were identified. The corrosion defects are not mechanism limiting the operating lifetime of reactor pressure vessel, provided that the primary water chemistry is kept within limits.

Fatigue

The dominating ageing mechanism is cyclic fatigue. The strength and integrity are defined mainly by brittle fracture strength, cyclic strength and cracking susceptibility of the base metal, metal of welds and corrosion protection weld overlay.

Calculations considering the change in the properties of materials and actual loading cycles during operation were considered.

The analysis of the operating regimes shows that there is a high strength margin in terms of cumulative cyclic fracture.

Stress corrosion

This ageing mechanism is typical for the areas of sealing grooves and threaded surfaces of the stud holes. These areas are monitored periodically, they are subject to maintenance and are not limiting to the operating lifetime of the reactor pressure vessel.

The actual measured values and characteristics are compared to the design ones. The data show that the ageing rates are lower than the predicted ones.

The achieved results show that the established and applied ageing management programme is adequate. The programme objectives in terms of coordination of organizational and technical measures for ensuring reliability and safety of reactor pressure vessels are achieved.

5.3 Regulator's assessment and conclusions on ageing management of RPVs

The ageing management programme for RPVs of Units 5&6 at Kozloduy NPP is developed in compliance with the regulations and IAEA safety standards. The scope of the programme includes the reactor pressure vessel and upper head. The sealing flange, internal and external surface of the reactor pressures vessel, base metal and welds are considered as areas of special interest.

For preparation of the ageing management programme for RPVs, design and construction documentation, data from operation, maintenance and monitoring of the facilities as well as data from study of the condition of metal of reactor pressure vessels were used.

The programme is updated and completed after the performed large-scope assessment in the frames of the plant lifetime extension programme.

The life management programme for reactor pressure vessels is part of the licensing documentation and is subject to review and assessment by the regulator.

As part of regulatory oversight of the ageing management processes, review of the in-service inspection programmes for reactor pressure vessel submitted to the BNRA prior to every planned annual outage is also performed.

The oversight of the ageing management processes is also provided through regulatory inspections performed mainly during the planned annual outage and prior to unit start-up. The scope of these reviews should also include the review of the scope and results of the applied in-service inspection to the reactor pressure vessels, as well as maintained water chemistry and results of corrosion assessment.

Based on the performed regulatory oversight it can be confirmed that the ageing management programme for reactor pressure vessels and performed in-service inspection are effective and in compliance with the existing regulatory requirements.

6. CALANDRIA/PRESSURE TUBES (CANDU)

Irrelevant

Kozloduy NPP operates WWER-1000 reactor types of Russian design, which is a PWR type according to the westerns terminology.

7. CONCRETE CONTAINMENT STRUCTURES

7.1. Description of ageing management programmes for concrete structures

7.1.1. Scope of ageing management for concrete structures

<u>7.1.1.1.</u> <u>Description of the types of structures, basic functions, and main structural elements</u>

The reactor building structure (of Units 5 and 6) represents a spatial structural system that consist of three main parts - foundation block, containment, and auxiliary facilities. The three main parts are united by a reinforced concrete plate, 2.40 m thick. The containments of KNPP units 5 and 6 are special pre-stressed reinforced concrete protective structures comprising as follows: a cylinder, a dome and a support ring that joins them. The auxiliary facilities denote a structure, 66 x 66 m, made of reinforced concrete walls and plates that surround the containment. The foundation block consists of a foundation plate and two interim plates, while the walls execute a stiffening effect.

The reactor is placed within the containment, the boundary of which is the containment structure. Figure 3 shows a cross-sectional view of the reactor building.

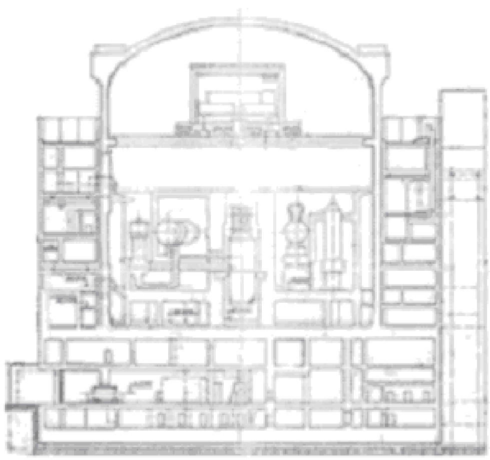


Figure 3. Cross-sectional view of the reactor building

The Safety Analysis Report (SAR) presents the design documentation, analyses and calculations of the civil structures of Kozloduy NPP reactor building. The containment enclosure system is documented in the SAR as being a part of the localisation (confining) safety systems. The structures of the containment enclosure system meet the following requirements:

- in terms of being part of the localisation systems, they perform the functions assigned (retaining of leaktight integrity and strength), and confine the active fission products under all design modes of operation;
- function as barrier and support structures both under operating conditions and during transients;
- provide biological shielding both under operating conditions and in case of design basis accidents.

The containment structure strength calculations took into account the impacts of natural phenomena, man-made events, combinations of impacts caused by rupture of the largest reactor coolant pipeline and a design basis earthquake (DBE). The safety margins have been determined for a beyond design basis earthquake, or the combination of a beyond design basis earthquake with flooding, as well as the way to maintain the containment structure integrity during severe accidents.

Figure 4 shows a general layout diagram of the containment area.

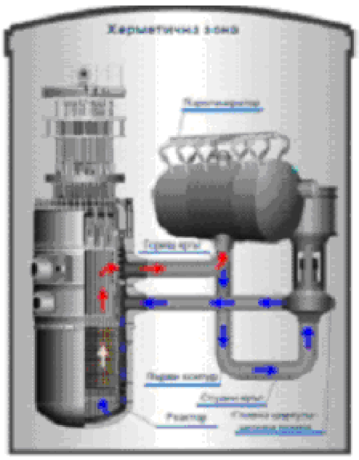


Figure 4. General layout diagram of the containment area

The containment structure also includes the emergency makeup tank room. The room has a lining of one layer of sheet steel plus a second layer of stainless sheet metal; leaks between these two layers are monitored by a dedicated system.

The inner wall of the containment structure has a steel lining (steel class ВСт3сп5), which ensures leak tightness. Outer wall of the containment cylinder part has been covered by a cement mix layer (sprayed concrete) . The concrete surface is protected by an epoxy anti-corrosion coating.

A cladding of 0.8 mm thick profiled steel sheets (LT) is installed above the level of the outdoor part of the containment to ensure additional protection. On the horizontal part of the support ring, a protective cement mix layer has been applied plus two hydro insulation strata. The following elements form the dome protective coating: light concrete to shape the draining sloping surfaces; lute, to level the surface; four layers of insulation material and concrete to protect the hydro insulation. In addition, in 2014 the dome received treatments with materials that prevent moisture entry into the reinforced concrete.

To ensure the required strength of the containment structure, the latter is equipped with a special system for prestressing composed of prestressed tendons (cables) in the cylinder wall and the dome, while the ends of them all are anchored in the containment support ring. The purpose of this system is to create a minimum uniform level of containment prestressing, in order to ensure fulfilment of its basic functions. The prestressing cables are installed in polyethylene tubes (so called channel-forming elements), grouted in the concrete cylinder walls and in the containment dome. The channels have not been injected, which allows, if necessary, performing of inspections, replacements or further retensioning of the prestressed cables. The prestressing system is the key factor determining the structure's prestressed state both during operating or accident conditions. Each prestressed cable (tendon) consists of steel high-strength strands (ropes), while each strand is composed of one central straight wire with several cold drawn wires coiled around it. The strands are of untwisting type, and their relaxation is minimal. A prestressing cables' anchoring device consists of an anchoring cylinder, triple wedges, an anchoring screw and a nut.

A dedicated system performs on-line monitoring of the prestressing force in the prestressing cables (of the containment cylinder and dome). The system relies on standard, electrical resistance strain gauges grouped, as per the design, in custom-made anchoring measuring devices called "measuring test-anchors". In accordance with the adopted concept, measurement is performed on a projected number of cables located within the cylindrical part of the containment structure and in the dome, while the sensors are installed at the strands of each cable.

The sensors have cables to connect them to the measurement station situated inside the reactor building. The system also has a mobile measurement station. The data obtained from all the test anchors are recorded, processed, summarised in a systematic manner and interpreted on a computer using a specially adapted software. As a result, the average force in the tension cable is obtained.

7.1.1.2. Methods and criteria for selection of components for ageing management

Based on the criteria described in section 2.3.1.2 of this Report, the following components belonging to the containment structure, the auxiliary building and the foundation block of the reactor building have been identified as the subject of periodic inspections and control:

- the concrete cylinder and dome parts;

- the steel lining;
- tension cables (tendons);
- anchoring devices of the tension cables;
- protective concrete coating of the dome;
- dome drainage system;
- containment penetrations;
- the inner and outer reinforced concrete walls and plates of the auxiliary facility and the foundation block of the reactor building.

7.1.1.3. Processes/procedures for identifying the ageing mechanisms of different materials and components of the concrete structures

The ageing mechanisms, potential and dominant ones, for the concrete structures of the airtight structures, foundation block and auxiliary facility have been identified on the basis of long-term periodic observations and monitoring of their condition. The planning, organisation and implementation of condition monitoring activities for the airtight structures, foundation block and auxiliary facility, together with the interpretation of results have been regulated in the Instruction for Condition Monitoring of Containment (Airtight) Structures, and Instruction for Construction Sites Lifetime Management.

Periodic as well as extraordinary visual inspections have been performed of the visible parts of the containment structures, foundation block and auxiliary facility, namely:

- the visible external part of the pressurised structures, cylinder part, ring and dome;
- bottom support blocks;
- visible internal parts of the pressurised structures;
- anchorage elements of the tension tendons of the prestressing systems;
- steel truss supports of the ventilation stack;
- the inner and outer reinforced concrete walls and plates of the auxiliary facility and the foundation block of the reactor building.

Periodic visual inspections are undertaken by a designated committee:

- quarterly visual inspections covering the protective lids and the tension cables' anchoring;
- visual inspections within the framework of technical visual assessments of the civil structures, focused on the condition of the inside and outside sheet iron lining, as well as the concrete of the gastight structure, foundation block and auxiliary facility (technical visual assessments are scheduled within a year period).

Based on the data of the condition monitoring of the containment structures, foundation block and auxiliary facility of the reactor building, evaluation is made of the current state of the following:

- the shielding concrete and the external sheet iron (LT) lining of the containment cylinder part;
- the protective steel lids for troughs' draining and tension cables anchoring;
- the inner sheet iron lining of the containment structure (ensuring airtightness);
- the outer and inner reinforced concrete walls, plates of the auxiliary facility, and the foundation block of the reactor building.

Special reports are issued to document the results from condition monitoring of the containment structures, foundation block and auxiliary facility of the reactor building, as well as the corrective actions identified. Those reports contain:

- justifications for monitoring performance:
- participants in the committee that performed the visual inspection;
- owner and operating organisational unit;
- data about the technical documentation reviewed and geodetic control;
- description of any defects found, based on preliminary filled-in evaluation sheets and with references thereto;
- any other data on the inspected sites, found during the review of the technical documentation available or during the visual inspection (e.g. data of completed or scheduled significant maintenance jobs, surveys and results thereof, interpretations of the results from other types of monitoring, data of changes occurring in the function or status of the site, etc.)
- a conclusion on the general condition of the containment structures, foundation block and auxiliary facility;
- suggestions for follow-up activities.

It is mandatory to attach to the reports all the filled-in sheets recording any defects found.

The results from condition monitoring of the containment structures of the system for automatic control of pre-stressed and deformed state are then documented in a Certificate of Systems' Testing and Monitoring, which contains as follows:

- degree of containment structure prestressing as compared against the condition following the replacement of the prestressing system in 2005;
- degree of containment structure prestressing as compared against the design condition during previous monitoring performed by the automatic control of pre-stressed and deformed state;

- degree of containment structure prestressing as at a specific date, as compared against the design condition shown by the automatic control of pre-stressed and deformed state;
- the average force of the tension cables.

The Certificate is issued quarterly, prior and following leak tightness tests of the containment structure, and also in case of incidents with the prestressing system, and/or earthquakes.

The data resulting from the containment structures monitoring through the computer (automated) monitoring system for cable prestressing are documented in "Analysis of data from the containment cables prestressing computer monitoring system regarding the ten monitored tension cables of the prestressing system", and also in "Certificate of inspection of the containment cables prestressing computer monitoring system". Upon reaching of the operating limits, a Programme is drafted for cables retensioning, scheduled to be implemented during the subsequent annual outage.

7.1.2. Ageing assessment of concrete structures

7.1.2.1. Ageing mechanisms requiring management and identification of their significance

On considering the potential ageing mechanisms of the civil structures of the leakproof structures, foundation block and auxiliary facility of the reactor building, the following ageing mechanisms have been identified/selected that will require management and evaluation of their significance:

- protective concrete coating of the dome - ageing and gradual decomposition of the joints' hydroinsulation material, ageing of the main concrete resulting from atmospheric impacts, formation of small, local damages in the concrete;
- dome drainage system - retaining rainwater in the anchoring troughs due to impaired leaktightness of the drainage system and corrosion of the ring-like, steel drainage channel along the periphery of the ring at the respective elevation, as well as all possible ageing factors resulting thereof;
- the anchor troughs of the cylindrical cables with protective steel lids and the anchoring of the tension cables - corrosion of the steel linings along the walls and bottom of anchor troughs, discontinuities on the corrosion-proof coating and corrosion on the steel lids and anchor plates;
- the containment structure cylindrical part above the open part - penetration of rainwater behind the protective LT-sheet iron panels and wetting of the concrete surface, due to displaced LT-sheet iron panels and all possible ageing factors ensuing thereof;

- the tension cables in the area of goosenecks - surface damaged coating of the strands with conservation lubricant;
- the visible part of the containment structure steel lining - cracking, blistering or peeling of the anti-corrosion coating;
- base metal of the containment structure steel lining - presence of areas with visible local warpings of the steel lining resulting from its distortion;
- base metal of the containment structure steel lining - occurrence of deformations due to external loads that cause deviations from the nominal shape of the cylindrical part;
- stress relaxation in the tension cables (due to material creeping) and loss of tensioning force;
- tension cables - decrease of tension (relaxation) of the tension cables;
- base metal of the containment structure steel lining - degradation of the physicomechanical properties of the material under the impact of operating environment factors (such as high temperature values, ionising radiation, etc.);
- welding joints of the containment structure steel lining - occurrence of defects in the welding joints and/or presence of surface blemishes;
- steel lining and other metal parts of the containment structure - corrosion of the structural steel;
- concrete and metal parts of the containment structure, foundation block and auxiliary facilities of the reactor building - high temperature and the temperature cycles;
- concrete and metal parts of the containment structure, foundation block and auxiliary facilities of the reactor building - material fatigue;
- concrete and metal parts of the containment structure, foundation block and auxiliary facilities of the reactor building - uneven relative subsidence of individual sections of the structure;
- concrete and steel of the containment structure, foundation block and auxiliary facilities of the reactor building - radiation and exposure;
- steel of the containment structure - embrittlement;
- concrete and steel of the containment structure, foundation block and auxiliary facilities of the reactor building - corrosion of concrete and steel due to boric acid, chlorides, etc.;

The concrete of the containment structure, foundation block and auxiliary facilities of the reactor building are subject to the types of monitoring/inspection listed below:

- concrete toughness as a result of increased local humidity and penetration of different salts, mainly chloride ions;

- occurrence of disintegration processes inside the concrete resulting from the aggressive impact of sulphate aggressive media (delayed ettringite formation, DEF, in the concrete);
- physicomechanical characteristics of concrete as a result of alkali-silica reaction (ASR) occurring in concrete;
- the half-cell potential between concrete and the reinforcement surface; impaired contact between concrete and the bearing reinforcement bars, and development of corrosion of the reinforcement bars;
- the concrete integrity and in-depth structure as a result of crack formation, concrete disintegration;
- dissolution and leaching of calcium hydroxide from concrete;
- development of chemical aggressiveness (including of chlorides, sulphates, etc.)
- alkali-aggregate reaction of the concrete additive materials;
- cyclic freeze/thaw;
- steel corrosion in rebar elements;
- concrete creep;
- concrete shrinkage;
- abrasion (erosion) of concrete and cavern formations;
- carbonisation (depassivation) and neutralisation of concrete.

7.1.2.2. Establishment of acceptance criteria related to ageing mechanisms

According to the Instructions for Life Management of Kozloduy NPP Construction Sites, criteria have been introduced for assessment/categorisation of any defects found (current condition) in important elements and assemblies (Table 6) on the basis of visual observations mainly.

Table 6. Criteria for assessment/categorisation of any defects found (current condition) in important elements and assemblies

Category	Definition/Description
Category A	**Significant defects** or severe impairments of main SSCs of primary importance to safety, with immediate possibility of disintegration, causing abrupt (within a short time period) failure or disintegration. This category includes defects that considerably decrease the bearing capability of the civil structure and present a danger to its reliability. **Defect recovery actions:** Immediately perform maintenance, recovery works and/or other emergency actions.
Category B	**Deviations or impairments of SSCs of moderate significance** (average importance) to safety, with a possibility of local (partial) degradation that will cause limited and gradual (in the long term) failure. This category includes deviations, the existence or further development of which may pose danger to the reliability and normal operation of the building or facility. **Defect recovery actions:** repairs, recovery works, and/or detailed

Category	Definition/Description
	assessment/monitoring - review (discussion) and prioritised implementation in the short term.
Category C	**Negligible** deviations or impairments of secondary SSCs not related to safety, with limited potential of local (partial) disintegration within exceptionally (too) long time period. This category includes deviations the existence or further development of which does not pose any danger to the reliability and normal operation of the building or facility. Nevertheless, they can endanger the normal operation of equipment, the life or health of the operating personnel, or disrupt the civil site architectural features or hygiene. **Defect recovery actions:** Repairs are performed locally or long-term monitoring (tracking) of the impairment. Immediate repair actions are undertaken only if the deviation poses risks to the personnel life or health.

In connection with the identified ageing mechanisms, the following acceptance criteria have been approved, as described in Instruction for Condition Monitoring of the Protective Shells of Kozloduy NPP Units 5&6, or in the Instruction for Life Management of Kozloduy NPP Construction Sites, together with the respective specific activities:

- Measurement of the average force in the tension cables (tendons) is performed in conformity with the provisions of document "US NRC Regulatory guide 1.35 "In-service inspection of ungrouted tendons in prestressed containments". Acceptance criteria have been established for the cylindrical, regular and non-regular tendons, as well as for the dome tendons regardless of their position.
- The measured depth of neutralisation/carbonisation that may not exceed the thickness of the reinforcement protective concrete coating functions as an acceptance criterion for measuring the depth of the protective concrete coating and the degree of concrete neutralisation/carbonisation (as per BSS CR 12793:2003 "Measurement of the Carbonisation Depth of Hardened Concrete");
- A criterion for acceptance in the compressive strength studies is that the established average value for compressive strength should be equal to or greater than the concrete design stamp compared against the standard traits of concrete during the construction period;
- During the concrete resistivity measurements (as per AASHTO Designation: T XXX-08 „Standard Method of Test for Surface Resistivity Indication of Concrete's Ability to Resist Chloride Ion Penetration") – the measured values for concrete should not be less than the established limit (permitted) values;
- The geodetic measurement of any ground shifting (subsidence) below the civil structures foundations follows the prescriptions of the new European standard - BSS EN 1997-1:2005/NA:2015 "Eurocode 7: Geotechnical Design. Part 1: General

Rules". On the grounds of this, the geodetic measurements of any deformations of the civil structures of units 5&6 reactor buildings (containment buildings) conform to the following critical (maximum permitted) displacement values: maximum, absolute and relative deformations - not to exceed the permissible limits;

- The half-cell potential measurements (as per ASTM C876-91/1999, "Standard Test Method for Half-Cell Potentials of Uncoated Reinforcing Steel in Concrete") use as an acceptance criterion the rule that the measured half-cell potential should exceed the limit (allowable) value;
- The established degree of delayed ettringite formation of concrete remaining below the permissible limit value is the acceptance criterion concerning the assessment of the degree of concrete delayed ettringite formation development;
- The established degree of alkali-silica reaction of concrete remaining below the permissible limit value is the acceptance criterion for assessment of the degree of concrete alkali-silica reaction.

Regarding all the rest (not mentioned above) of ageing mechanisms, the acceptance criteria are derived on the basis of expert evaluations using the above Table 6.

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

The following activities for monitoring, testing, sampling and inspection for concrete structures are performed:

- visual inspection of protective concrete coating of the containment structure dome and the auxiliary facilities; the periodicity of this inspection is 1 year;
- visual inspection of the dome drainage system; the periodicity of this inspection is 1 year;
- visual inspection of the anchor troughs of the cylindrical tendons with the protective steel lids and the anchoring of the tension tendons; this inspection has quarterly periodicity;
- visual inspection of the cylindrical part of the containment structure above the open part; the periodicity of this inspection is 1 year;
- visual inspection of the tension cables in the gooseneck areas; this inspection has quarterly periodicity;
- visual inspection of the visible part of the containment structure steel lining; the periodicity of this inspection is 1 year;
- *in situ* and laboratory survey to measure and assess the level of carbonisation/neutralisation of concrete - this type of test should take place once every ten years;

- instrumentation testing of the concrete resistivity/conductivity - this type of test should take place once every ten years;
- conduct of “lift-off” test involving not less than 10% of the tension tendons of the containment shell – this type of test should take place once every ten years;
- implement geodetic diameter measurement inside the containment structure, at the standard test points - this type of test should take place at least once annually;
- implement a geodetic measurement of absolute and relative deformations (slump) of the foundation beneath the civil structure of the reactor building and the containment structure, at the standard test points - this type of test should take place at least once annually;
- measuring the half-cell potential of reinforcement steel - performed during specialised surveys and/or upon identification of the need thereof;
- annual leak tests are performed during the outages and in conformity with the Containment Structure Leak Test Procedure.

In the course of specialised surveys (and/or upon identifying the necessity thereof), specialise Kozloduy NPP departments or external organisations perform the following:

- non-destructive testing (NDT) of the weld joints of the containment structure steel lining including visual inspection with dye-penetrant liquids (dye-penetrant testing);
- NDT of the containment structure steel lining through ultrasonic thickness measurement (UT);
- NDT of the containment structure steel lining through metal hardness measurement;
- visual inspection for any deformations (swelling) of the containment structure steel lining;
- instrumentation testing to take a picture of the rebars in typical cross sections and elements of the bearing reinforcement concrete structure of the buildings, determining the thickness of the protective concrete coating, and identifying the width and depth of any cracks in the concrete;
- instrumentation testing to identify the potential compressive strength of concrete through wrenching;
- instrumentation testing to identify the potential compressive strength of concrete through measuring the elastic rebound;
- *in situ* and laboratory testing to determine the degree of progress of alkali-silica reaction (ASR) – Internal Swelling Reaction (ISR);
- *in situ* and laboratory testing to determine the concrete volumetric density and compressive strength;

- *in situ* and laboratory testing to determine the degree of progress of concrete ettringite formation (DEF).

7.1.4. Preventive and remedial actions for concrete structures

The main criteria, procedures, measures and activities that need to be taken upon identifying the need of preventive and corrective actions are described in the Instruction for Life Management of Kozloduy NPP Construction Sites. The preventive and monitoring measures are classified as per their priority, resulting from the expert assessment issued by the Hydro Engineering Facilities and Civil Structures workshop, as follows:

- Top priority (urgent action).

Undoubtedly, top priority is attributed to maintenance action aimed at recovery of defects on elements whose condition is “unsatisfactory”;

- Second priority (necessary action).

These are actions addressing all defects of elements that are in “satisfactory” or “good” condition and are outside the scope of the top priority.

The second priority actions include the necessary repair and preventive actions the implementation of which is not urgent, or is associated with specific conditions (such as recovery of a certain defect, detailed design or dismantling of process equipment). These actions focus on recovery of defects that do not have the preconditions to develop further and, therefore, may be undertaken with some delay, in conformity with the opportunities for funding and organising of the maintenance activities.

- Third priority (recommended actions).

The third priority actions include those ones that are not absolutely necessary, but may, however, exercise a positive impact on the containment structure. These actions focus on improving the status of the civil structure and rectifying of negligible (mild) faults.

The Instruction for Life Management of Kozloduy NPP Construction Sites also lists prescriptions for preventive / corrective actions regarding concrete structures, such as:

- while performing of instrumentation monitoring of concrete resistivity / conductivity:
  - In case the measured concrete resistivity exceeds the limit value specified in the Instruction - no preventative/corrective actions are needed;
  - In case the measured concrete resistivity remains below the limit value specified in the Instruction - it is required that an option be considered and discussed for increasing the frequency of monitoring performed in the respective areas in order to be prepared (if applicable) to proceed with developing of a special technology and implementing of sanitary and rehabilitation activities in the areas in question;

- while performing tests to determine the degree of progress of concrete ettringite formation (DEF):
    - In the event of identifying concrete ettringite formation within the regulated progress limits - no preventive/corrective actions need to be undertaken;
    - In case the measured degree of concrete ettringite formation exceeds the limit value specified in the Instruction - it is required that an option be considered and discussed for increasing the frequency of monitoring performed in the respective areas in order to be prepared (if applicable) to proceed with developing of a special technology and implementing of sanitary and rehabilitation activities in the areas in question;
- while performing tests to determine the degree of progress of concrete alkali-silica reaction (ASR):
    - In the event of identifying concrete alkali-silica reaction within the regulated progress limits, on the surface tested - no preventive/corrective actions need to be undertaken;
    - In case the measured degree of concrete alkali-silica reaction exceeds the limit value specified in the Instruction - it is required that an option be considered and discussed for increasing the frequency of monitoring performed in the respective areas in order to be prepared (if applicable) to proceed with developing of a special technology and implementing of sanitary and rehabilitation activities in the areas in question;
- while performing instrumentation monitoring to measure the half-cell potential of reinforcement steel:
    - In case the measured half-cell potential exceeds or is equal to the specified maximum limit value - no preventive/corrective actions need to be undertaken;
    - In case the measured half-cell potential falls within a given interval (between the minimum and the maximum limit values) - it is required to consider and discuss an option for increasing the frequency (at least twice higher) of the monitoring performed in the areas in question;
    - In case the measured half-cell potential stays below the minimum limit value - it is necessitated to consider and discuss an option for implementing other (more accurate) methods of instrumentation monitoring in the areas in question, in order to confirm (or not) the negative results; On the grounds of this, there has to be a preparedness to proceed (if applicable) with

developing of a dedicated technology and performing of renovation and rehabilitation activities in the areas in question;

- while performing an *in situ* and a laboratory test to measure and assess the thickness of concrete coating of the concrete structure - if the measured concrete coating thickness stays below the specified limit value, it is necessitated to consider and discuss options for more intensive measuring in the areas in question;
- while performing an *in situ* and a laboratory test to measure and assess the concrete carbonisation/neutralisation degree - if the measured concrete carbonisation depth exceeds the specified limit value, it is necessitated to consider and discuss options for more intensive measuring in the areas in question and/or develop a dedicated technology and perform renovation and rehabilitation activities in the areas in question;
- while implementing lift-off tests:
  - In case the measured tension forces in the tension cable remain below the: specified force regarding the cylindrical regular tendons; the specified force for cylindrical non regular tendons; specified force for dome tendons - it is necessary to perform retensioning of the respective tendon until reaching the controllable tensioning force;
  - In case the measured forces exceed the specified limit values - no preventative/corrective actions are needed. The tensioning forces losses are not linear and they are calculated for each specific moment. The predictable subsequent losses due to concrete shrinkage or creep, or relaxation or reinforcement uses are calculated as the function of change with time logarithm.

When the average tensioning force within a cable reaches its operating limit, as per the Operating Manual (Technical Specifcations), retensioning of the respective cable is undertaken.

Upon failure of a cable, a programme must be drafted for this cable recovery and, if possible, the programme is scheduled to be implemented during the annual outage closest in line. The materials necessary to make the cables, such as strands, anchoring devices as well as all the machinery (pressure gauges, cranes, jacks, pump stations, etc.) are available on stock at Kozloduy NPP. In accordance with the Safety Analysis Report, and the Instruction for Condition Monitoring of the Kozloduy NPP Units 5&6 Containment Structures, the so-called “operating limit” has been specified for monitoring of failed tensioning cables at units 5 and 6. Upon reaching of this operating limit (as a result of bending of an anchoring device, or breaking of strands), the failed elements are recovered, the activities for which are performed during the next outage of the respective unit.

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

As a result of the periodic visual observations and inspections of units 5&6 in 2008 and 2009, the necessity has been identified of repairing the steel structures at upper elevations of the units 5&6 reactor buildings. The respective repair activities were performed during the outages in 2008, 2009 and 2010. The repairs performed at the upper levels of units 5&6 included as follows: repair and replacement of steel metal structures; metal stairs; support blocks of the dome cables; the ventilation stack supports; sandblasting and application of anticorrosion coating; dismantling and installation of sheet iron LT55; dismantling and installation of galvanised sheet iron; procurement and installation of removable, hot-dip galvanised grids, with mesh size 33.3x33.3, and fixing elements; final painting of the metal structures.

The 2012 ageing management review of SSCs at KNPP units 5&6 identified the need of a timely renovation and maintenance in good technical condition of the hydro insulation of the dome protective concrete, and the drainings of units 5&6 reactor buildings. The necessary repair activities of the units 5&6 hydro insulation of the dome protective concrete and reactor buildings' draining were implemented in 2013.

In 2015 and 2016, a specialised assessment and analysis were accomplished at Kozloduy NPP to confirm the residual life of the protective containment structure and the prestressing system of units 5&6, in connection with Stage 2 of the Plant Life Extension Project for Units 5&6. The assessment included and analysis of the control of the actual tension forces in the tension tendons of the containment structures of units 5&6, performed in 2015 by a team from the Metal Science Institute at the Bulgarian Academy of Science. The lift-off test was undertaken to measure the actual tension forces. During this activity, the requirements were followed of the US regulatory documents: NRC RG 1.35– “In-service inspection of ungrouted tendons in pre-stressed concrete containments”, NRC RG 1.35.1– “Determining pre-stressing forces for in-service inspection of pre-stressed concrete containments”. Lift-off tests were performed, and during the measurements, the forces in the cables' ends at the anchoring devices were monitored. The forces were monitored in a total of 28 tension tendons – 22 in the cylindrical part and 6 in the dome of unit 5. As regards unit 6, the forces were monitored in 29 tendons – 23 in the cylinder part, and 6 in the dome. As per NRC RG 1.35.1, the top and bottom limits were defined for the tension forces within the tendons and, on the grounds of the data obtained in the current and in previous tests, the relations were defined for future time periods. A negligible difference of 0.3% was found between the average tension forces in the measured tendons and the data recorded by the prestressing computer monitoring system.

The results from the specialised survey of unit 5 show that in some of the monitored areas of the reinforcement steel elements the measured concrete resistivity was lower than the limit specified, which points to the existence of a potential probability of corrosion processes developing in the reinforcement steel of these areas. The recommendations led to corrections made to the Instruction for Life Management of Kozloduy NPP Civil Sites.

The specialised survey of the feasibility of lifetime extension of units 5&6 provided an evaluation of the current condition of the containment civil structure, the auxiliary facilities and foundation block, based on the data from the assessment conducted to identify the actual condition of the pre-stressing system of the containment structures, the real traits of concrete, the actual strength traits of the steel in the containment structures lining, the results from the additional visual inspections and analysis of the operating and maintenance documentation.

The actual technical condition was determined through the conduct of control static and dynamic analyses inclusive of combinations of the impact of design basis accident and the seismic impact of RLE. During the implementation of the activities for assessment of the current condition of the containment structures civil structures, the auxiliary facility and the foundation slab, and the analysis and interpretation of results, due consideration was given to all the conclusions from the comprehensive assessment completed at Stage 1 of the Project.

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

In the course of implementing the Project for Lifetime Extension of Unit 5, a regulatory review was performed of the ageing evaluation of reinforcement steel structures and of the reporting documents from the actions completed. Among the more important of these were:

- confirmation of the preliminary residual life of the containment structure and auxiliary facilities;
- confirmation of the residual life of the pre-stressing system;
- defining the actual pre-stressing forces of the tendons.

In the course of the review process, the Bulgarian Nuclear Regulatory Agency assigned external, independent expert assessment to assess the performance of individual measures, inclusive of ones concerning civil structures (the containment structure, the auxiliary buildings, ventilation stack, and spray pools). The external assessments performed confirmed the conclusions made by the licence-holder through the assessment of SSCs, the tests and analyses conducted. The results from these activities demonstrate the possibility of safe operation of the civil structures for the lifetime extension period of operation.

The Bulgarian Nuclear Regulatory Agency undertakes periodic reviews in connection with the ageing management of reinforced concrete structures. The results from these reviews are positive, indicating that the civil structures in question are in a stable condition and meet the design requirements. They also prove that the activities required for periodic observations, assessment and maintenance of the structures in question have been conducted as necessary.

On these grounds it can be confirmed that the practices currently applied at Kozloduy NPP for ageing management of concrete structures (including programmes, instructions, schedules, etc.) have been effective and conform to the existing regulatory requirements and IAEA standards.

8. PRE-STRESSED CONCRETE PRESSURE VESSELS (AGR)

Not applicable.

Kozloduy NPP operates Russian design reactors of the WWER-1000 type, which correspond to the PWR reactors, as per the western terminology.

9. OVERALL ASSESSMENT AND GENERAL CONCLUSIONS

Current report provides information on the existing national regulatory frame in the area of ageing management. The main regulatory documents have been described, the interrelations among them, the provisions they contain as regards the activities associated with the ageing management process of SSCs within the scope of this Report. The Act on the Safe Use of Nuclear Energy and the regulations of its application define the national framework for state regulation of the activities in the field of nuclear energy use.

At the end of 2016, the fully revised Regulation on Ensuring the Safety of Nuclear Power Plants was approved. New conceptual requirements have been introduced on the safety of nuclear power plants. Due consideration has been given to the WENRA Safety Goals for New NPP Designs, and the updated (after the Fukushima NPP accident) Reference Levels for Safety Harmonisation of NPPs in Operation in terms of their ageing management (Issue I: Ageing Management), as well as the latest IAEA safety standards in this field. Due consideration has also been given to the latest revisions of IAEA standards regarding the subject of this Report, such as Periodic Safety Review for Nuclear Power Plants, No. SSG-25, IAEA, 2013; Safety Classification of Structures, Systems and Components in Nuclear Power Plants, No. SSG-30, IAEA, 2014, etc.
Additional instructions on applying of the provisions of the Regulation have been identified in the Regulatory Guide on Implementing Periodic Safety Review of Nuclear Power Plants and in the Regulatory Guide on Safe Operation of Nuclear Power Plants.

Bulgarian Nuclear Regulatory Agency fulfils its control functions regarding the SSCs ageing management through the following activities: review and evaluation of documents enclosed to the applications for issuance of operating licences; review of the periodically submitted information in terms of licensing conditions implementation; review, analysis and assessment of any modifications of SSCs and issuance of permits for their practical realisation; undertaking of scheduled as well as extraordinary inspections in compliance with a long-term programme and the annual plan for inspections.

Kozloduy NPP implements a general Programme of Units 5&6 Lifetime Management (the PROGRAMME). It serves to introduce an integrated approach to observations, identification, documenting and analysing of the ageing of SSCs, in view of identifying all the ageing mechanisms, potential consequences of the processes and possible actions for restoration of their operability. The PROGRAMME defines the activities for maintenance and repairs, non-destructive testing, modifications, trials and qualification of SSCs. Based on the criteria identified in the PROGRAMME, SSCs have been selected whose ageing is being monitored and assessed. Individual activities and specific programmes related to the ageing management are evaluated in terms of their effectiveness. To ensure control and mitigation of ageing effects, within the framework of the general PROGRAMME, individual programmes are applied for ageing management of irreplaceable or difficult to replace SSCs that are reviewed in the respective sections of this Report.

The PROGRAMME, being a part of the Kozloduy NPP Management System, is a subject to review and updating. Taking into account experience and best international practices, the PROGRAMME has been improved. The most significant modifications are related with improvements of existing programs and deriving of new and specific programmes for ageing management of the following SSCs:

- qualified equipment;
- buried pipework of the Service Water System;
- cables important to safety, in addition to the qualified cables;
- the hydraulic shock absorbers of the systems important to safety.

Taking into account the information presented in this Report, both regarding the licence-holder activities related to the ageing management process, and the regulatory control performed, the following major conclusions can be drawn:

- Generally, the existing PROGRAMME together with the specific plant programmes for ageing management comply with the regulatory requirements and provide due consideration to the IAEA safety standards.
- The interrelations among the programmes, activities and responsibilities regarding the control, monitoring and minimisation of the degradation effects for SSCs covered by the PROGRAMME, are defined in detail.
- The PROGRAMME implementation will accomplish its main objective - that is to ensure that the basic ageing mechanisms and effects thereof will be identified and reduced to guarantee safe and reliable plant operation.

As a result of the self-assessment provided in this Report it is possible to identify the following potential good practices and areas for improvement:

- The process of preparing additional specific programmes for SSCs in-service inspection, which expands the scope of the existing Instruction for In-Service Inspection of Equipment and Pipelines. This Instruction is subject to revision in view of extending the scope, methods and frequency of in-service inspection, taking into account the data from plant own operating experience, international practices and the activities for plant operating life extension.
- The use of an unconventional method such as the contactless magnitometric diagnostics for diagnostic monitoring of buried. The implementation of such a technique provides additional information about the zones where stresses are concentrated. The presence of similar stresses may be interpreted as a sign of corrosion, damaged insulation of the buried pipelines or nearby electric cables or metal objects that fell there during the process of placement of the pipelines.
- The application of a low-flux scheme for core refuelling may be regarded as an activity related to the ageing management process of the reactor pressure vessel and having a positive effect on its lifetime. This considerably reduces the negative impact of the neutron flux on the metal of the reactor pressure vessel (RPV).
- The activities initiated and included in the programme for study and analysis of RPV surveillance specimens, concerning the staged upgrading of the surveillance specimens assemblies and placing of new ones. It will ensure a possibility for study and analysis of the RPV material in view of its long-term operation;
- The regulatory frame introduced is generally in line with the approaches adopted by the IAEA and WENRA regarding the ageing management processes. Respective provisions have been included in the applicable regulatory documents. To facilitate the licence-holder, and in view of more comprehensive understanding and implementation of the regulatory provisions, due consideration should be given to the possibility of preparing a regulatory guide for ageing management. The objective of such a guide would be to provide more detailed instructions and directions for the practical application of the existing and modified regulatory documents;
- Bearing in mind the comprehensive nature of the activities related to ageing management, and the large scale of the plant life extension project, it is required to review the processes on the grounds of which a long-term regulatory inspection programme is developed. The purpose should be to optimise the programme scope in terms of types of SSCs and periodicity of inspections, which will enhance the effectiveness of the inspections performed.

In the context of activities for lifetime extension of units 5 and 6, Kozloduy NPP has initiated the conduct of IAEA SALTO review mission. In 2016, a pre-SALTO mission was conducted on unit 5. It should be noted that the pre-SALTO review mission took place prior to the completion of all the activities related to the lifetime extension of Unit 5. The mission scope also included a review of the programmes and activities associated with ageing management of SSCs, which is the subject of this Report. The pre-SALTO review mission on unit 6 is due in 2018, while the main SALTO review is scheduled for 2020.

The main results from the past pre-SALTO mission confirmed that the approach of Kozloduy NPP and the activities under way for plant lifetime extension and ageing management generally comply with the safety standards of IAEA and the approved international practices. The vibration condition monitoring of rotating mechanisms, including those not related to safety, received a positive review. Such monitoring provides the opportunity for a comprehensive analysis of the deviations, enabling the prediction of the potential time of failure, evaluation of the level of reliability and planning of the maintenance activities.

Five areas of the standard scope of the mission have been reviewed; six recommendations and five suggestions have been made, concerning mainly the improvement of the coordination among the individual programmes and updating the scope of some SSCs covered by specific ageing management programmes.

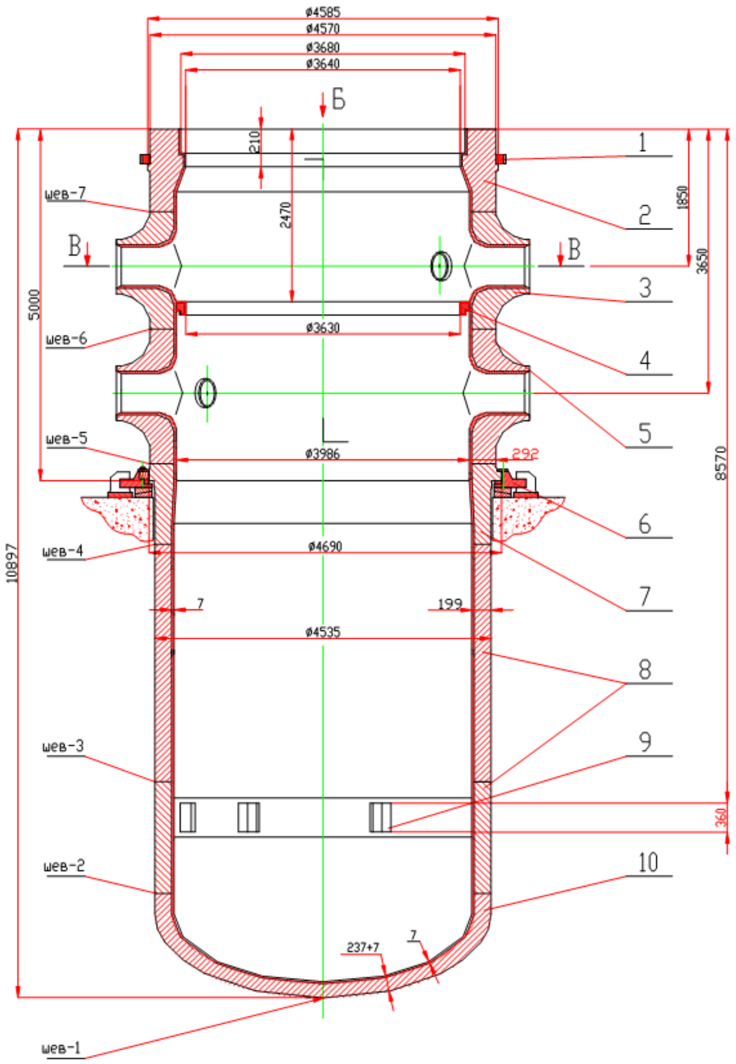
A summary of the final report of the review mission is available on the IAEA site.

The information provided in the current Report shows that over the past eight years the ageing management activities have been significantly influenced by the implementation of the project for lifetime extension of the units in operation. The results from the lifetime evaluation performed on critical plant equipment, which has a determining influence on the operating lifetime of the units, have confirmed the effectiveness of the ageing management approach applied by Kozloduy NPP.

LIST OF ABBREVIATIONS

DEF	Delayed Ettrigite Formation
HELB	High Energy Line Break
IAEA	International Atomic Energy Agency
IGALL	International Generic Ageing Learned Lessons .
INSAG	International Safety Advisory Group
ISR	Internal Swelling Reactions (in concrete)
LOCA	Loss of coolant accident
MILD	Mild (normal) operating conditions
Pre-SALTO Mission	Preliminary Review Mission of Safety Aspects of Long Term Operation
SALTO Mission	Review Mission of Safety Aspects of Long Term Operation
SSSL	Safety Shutdown System List
TLAA	Time Limited Ageing Analysis
WANO	World Association of Nuclear Operators
WENRA	Western European Nuclear Regulators Association
NPP	Nuclear power plant
BNRA	Bulgarian Nuclear Regulatory Agency
BAS	Bulgarian Academy of Sciences
BDS	Bulgarian National Standard
WWER	Water-cooled Water-moderated Energy Reactor
Plc	Public limited company
ASUNE	Act on the Safe Use of Nuclear Energy
I&C Systems	Instrumentation & Control Systems
SSCs	Structures, Systems, and Components
D&C Centre	Inspection Body of type C – Diagnostics and Control Centre
The PROGRAMME	Ageing Management Programme

Diagram of WWER – 1000/B320 reactor pressure vessel



1. Thrust ring; 2. Vessel flange; 3. Nozzle ring – upper;

4. Coolant flow deflector; 5. Nozzle ring – lower; 6. Vessel support ledge;

7. Vessel support ring; 8. Vessel rings; 9. Vessel clevis inserts; 10. Core barrel bottom

Dimensions:

-	Height	10,897 mm
-	External flange diameter	4,570 mm.
-	External ring diameter	4,535 mm
-	External diameter at the cross section of nozzles DN850	5,260 mm
-	Weight	329 t.

Chemically treated and demineralised water is used as reactor coolant and moderator. It contains boric acid which concentration vary between 16 and 0 g/kg in the process of operation. The coolant is forced (by the Reactor Coolant Pumps) into the reactor vessel through four inlet nozzles (Position 5), then passes between vessel and barrel, reaches the perforated barrel bottom and through the openings of the assembly supports it enters the core flowing upwards.

Flowing between the fuel assemblies, the coolant heats up due to the heat released by nuclear fission and exits the core through the perforated lower plate of the protective tube unit (PTU) flowing around the tubes. Further, through the perforated cylinder of the PTU and core barrel, the coolant enters circular space between the core barrel and reactor pressure vessel and then flows out through the four outlet nozzles (Position 3).

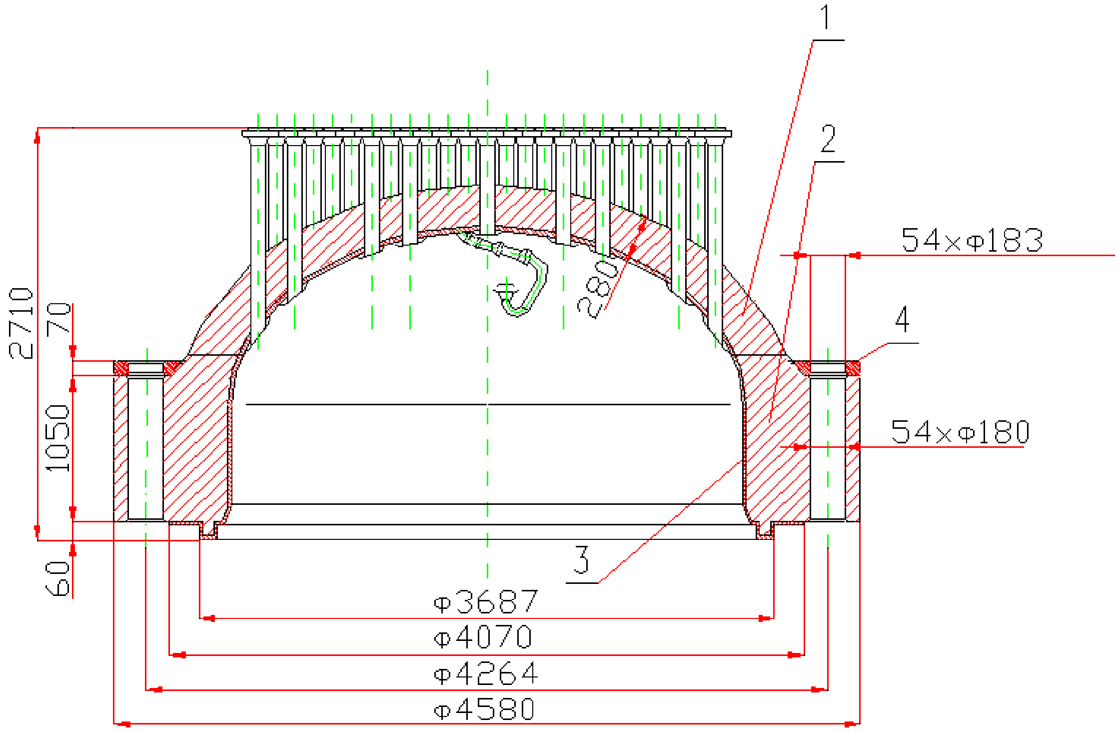
RPV basic design parameters:

- Nominal pressure, MPa	15.7
- Hydrostatic test pressure,	MPa 24.5
- Design nominal pressure, MPa (kgf/cm^2)	17.6 (180)
- Design temperature, °C	350
- Maximum heatup rate, °C/h	20
- Maximum cooldown rate, °C/h	30
- Design maximum fluence of fast neutrons which energy is > 0.5 MeV, neutron/cm^2	5.7.10^{19}

RPV strength is determined based on the material content of the base metal , cladding, and welds , and for the nominal temperature and pressure ranges.

According to equipment specifications, the RPV base metal is 15ХНМФА steel, core ring is 15Х2НМФА-А, and the cladding is 7 mm thick 08Х19Н10Г2Б steel.

Diagram of WWER – 1000/B320 reactor vessel head



1. Truncated ellipsoid; 2. Vessel head flange; 3. Corrosion-protective overlay; 4. Circular section

The vessel head is part of the integrated head assembly (IHA) and represents a forged and welded structure consisting of an ellipsoid and a flange machine-welded together.

The entire flange is forged of 15X2HMФA steel. On the flange inner surface as well as on the contact surface of the lower butt face of the flange, a two-layer stainless corrosion-protective weld overlay is applied.

The elliptic head dome is forged of a flat workpiece cladded in advance with a two-layer corrosion-protective weld overlay. This flat workpiece for the head dome is manufactured of 15X2HMФA steel and subjected to complete thermal treatment: annealing.

The dome and flange overlays are machine-welded under flux using welding band Св-07X25H13 and welding powder ОФ-10 (Layer 1) and Св-08X19H10Г2Б (Layer 2). The weld overlay covers all the surfaces except for the zone next to the weld seam between dome and flange. After welding the dome and flange together, the weld seam internal surface is cladded with the same corrosion-protective weld overlay which has been applied to the above vessel head components. The total nominal weld overlay thickness is 8 mm.

The head dome has 92 penetrations in which the RPS, temperature control, neutron flux measurement, and vent ports are installed. Those nuzzles are welded to the vessel head by means of strong and massive weld seams at the lower ends of the penetrations. The inner surfaces of the RPS, temperature control, and neutron flux measurement ports are protected by claddings made of chromium-nickel stainless steel of austenitic class 08X18H10T. Installation of two sealing O-rings on the nozzle flanges is provided for. Nozzle flange leak-tightness monitoring tubes connect the hollow space between them. The head flange has 54 penetrations, 180 mm in diameter, intended for the studs used to couple flange and vessel.

Reactor vessel head overall dimensions:

- Outer diameter at the head flange – 4580 mm;
- Distance between the nuzzles – 236 mm.