UKRAINE

NATIONAL REPORT
First Topical Peer Review
AGEING MANAGEMENT

State Nuclear Regulatory Inspectorate of Ukraine

Kyiv
2017

FOREWORD

According to the Association Agreement signed between the European Union and Ukraine, the Action Plan on Implementation of the Association Agreement between Ukraine on the one side and the European Atomic Energy Community and its member states on the other side has been underway since 2014. The Nuclear Safety Directive 2014/87/EURATOM establishing a Community framework for nuclear safety envisages topical peer reviews to be conducted by member states in identified technical safety areas.

The ageing management area was selected for the first topical peer review by the European Commission based on proposals of the Western European Nuclear Regulators Association (WENRA) that were approved by the European Nuclear Safety Regulators Group (ENSREG).

The review shall cover NPP units and research reactors with power more than 1 MW to be in operation as of 31 December 2017 or under construction as of 31 December 2016.

According to the Schedule of Measures to Implement Council Directive 2014/87/Euratom of 8 July 2014 amending Directive 2009/71/Euratom establishing a Community framework for nuclear safety of nuclear installations approved by SNRIU Order No. 54 dated 24 March 2015, Ukraine will also participate in the first topical peer review.

The State Nuclear Regulatory Inspectorate of Ukraine has developed the National Report on the First Topical Peer Review on Ageing Management to be analyzed by the EU member states. The National Report results from persistent, continuous and diligent efforts of a huge team. In this regard, I would like to express my sincere gratitude for the support provided to SNRIU by SSTC NRS, Energoatom and Nuclear Research Institute of the National Academy of Sciences of Ukraine.

Kyiv, November 2017

Chairman of the State Nuclear Regulatory Inspectorate of Ukraine **Hryhorii Plachkov**

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INTRODUCTION

The National Report has been developed to support the process of Ukraine's European integration aimed at implementation of EU legislation into the national regulatory and legal framework in the field of nuclear and radiation safety.

Ukraine joined the European process of implementing the Nuclear Safety Directive 2014/87/EURATOM. According to Article 8e of this Directive, each EU member state shall at least every six years participate in topical peer reviews in specific technical safety areas. The first topical peer review has started in 2017. It is devoted to ageing management issues and covers nuclear power plants and research reactors with power more than 1 MW (in operation as of 31 December 2017 or under construction as of 31 December 2016).

The objective of the first topical peer review is to exchange information between countries on the ageing management of structures, systems and components of nuclear installations, identify the best practices and general issues and develop an action plan to improve the regulatory and legal framework and practices in this area.

The National Report has been developed in full compliance with the WENRA Technical Specification [1].

ABBREVIATIONS

AAMS	–	Automated Ageing Management System
AM	–	Ageing Management
AMIAS		Ageing Management Information Analysis System
AMP	–	Ageing Management Program
BFR	–	Brittle Fracture Resistance
C(I)SIP	–	Comprehensive (Integrated) Safety Improvement Program
CAMP	–	Cable Ageing Management Program
CAMPU		Cable Ageing Management Program for Nuclear Power Unit
CE	–	Control Element
CPS	–	Control and Protection System
CPSS	–	Containment Prestressing System
CSS	–	Containment Safety System
DSTU (GOST)	–	State Standard of Ukraine
ECCS	–	Emergency Core Cooling System
EIF	–	External Impact Factor
Energoatom	–	National Atomic Energy Generating Company “Energoatom”
ENSREG	–	European Nuclear Safety Regulators Group
ESWS	–	Essential Service Water System
Euratom	–	European Atomic Energy Community, EAEC
FA		Fuel Assembly
HAZ		Heat Affected Zone
HPP	–	Hydroelectric Power Plant
HTC	–	Horizontal Test Channel
I&C	–	Instrumentation and Control
IAEA	–	International Atomic Energy Agency
IGALL	–	International Generic Ageing Lessons Learned Program
ISP		Integrated Surveillance Program
KhNPP	–	Khmelnitsky Nuclear Power Plant
KIEP	–	Kyiv Research and Design Institute “Energoproject”
LTO	–	Long-Term Operation
MCP	–	Main Coolant Piping
MGV	–	Main Gate Valve
MHT	–	Mechanical Hardness Testing
NDI	–	Nondestructive Inspection
NMC	–	Neutron Instrumentation Channel
NPP	–	Nuclear Power Plant
NRI	–	Nuclear Research Institute of the National Academy of Sciences of Ukraine
NRR		Nuclear Research Reactor
NRS	–	Nuclear and Radiation Safety
PRZ PORV	–	Pressurizer Pilot Operated Relief Valve
PSPP	–	Pumped Storage Power Plant
PSRR	–	Periodic Safety Review Report
RC	–	Reactor Compartment
RCE	–	Reinforced Concrete Enclosure
RCP	–	Reactor Coolant Pump
RHWG	–	Reactor Harmonization Working Group

RNPP	–	Rivne Nuclear Power Plant
RPV	–	Reactor Pressure Vessel
RVCH	–	Reactor Vessel Closure Head
RVMF	–	Reactor Vessel Main Flange
SAR	–	Safety Analysis Report
SDGS	–	Standby Diesel Generator Station
SG	–	Steam Generator
SNRIU	–	State Nuclear Regulatory Inspectorate of Ukraine
SSTC NRS	–	State Scientific and Technical Center for Nuclear and Radiation Safety
SUNPP	–	South Ukraine Nuclear Power Plant
TCA	–	Technical Condition Assessment
TCP	–	Technical Condition Parameter
TLAA	–	Time Limited Ageing Analysis
TM	–	Temperature Monitoring
TS (TU)	–	Technical Specifications
URDB	–	Ukrainian Reliability Database for Nuclear Power Plants
UTM	–	Ultrasonic Thickness Measurement
VI	–	Visual Inspection
VVER	–	Water-Cooled Water-Moderated Power Reactor
WANO	–	World Association of Nuclear Operators
WENRA	–	Western European Nuclear Regulators Association
ZNPP	–	Zaporizhzhya Nuclear Power Plant

01 GENERAL INFORMATION

01.1 Identification of nuclear installations

According to the Technical Specification [1], this report provides information on all nuclear power plants (units) and research reactors (> 1 MW).

The National Nuclear Energy Generating Company "Energoatom" (Energoatom) was established in October 1996. Energoatom operates four Ukrainian nuclear power plants including 15 nuclear power units, 13 VVER-1000 and two VVER-440, with a total installed capacity of 13,835 MW, two hydraulic power units of the Tashlyk PSPP with an installed capacity of 302 MW and two hydraulic power units of the Oleksandriv HPP with an installed capacity of 11.5 MW.

Four nuclear power plants – Zaporizhzhya, Rivne, South Ukraine and Khmelnitsky – are stand-alone subdivisions of Energoatom.

In compliance with the Law of Ukraine "On Nuclear Energy Use and Radiation Safety" [2], Energoatom is entrusted with functions of the operating organization responsible for the safe production of electricity. The Energoatom headquarters is located in Kyiv. Energoatom deals with general management over its policy, activities, and development.

As of 2017, there are 15 VVER power units in operation in Ukraine:

- VVER-1000/320 (11 power units);
- VVER-1000/302 (1 power unit);
- VVER-1000/338 (1 power unit);
- VVER-440/213 (2 power units).

Energoatom operates all the above NPP units.

Data on the number of reactors on NPP sites, installed capacities, beginning of commercial operation and expiration of the design-basis life and LTO periods are summarized in Table 1.1.

Table 1.1 List of Operating Nuclear Power Units

NPP	Power unit	Electrical power, MW	Reactor type	Beginning of commercial operation	Expiration of design-basis life/LTO period*
ZNPP	1	1000	V-320	10.12.1984	23.12.2015/**23.12.2025**
	2	1000	V-320	22.07.1985	19.02.2016/**19.02.2026**
	3	1000	V-320	10.12.1986	05.03.2017/**05.03.2027**
	4	1000	V-320	18.12.1987	04.04.2018
	5	1000	V-320	14.08.1989	27.05.2020
	6	1000	V-320	19.10.1995	21.10.2026
SUNPP	1	1000	V-302	31.12.1982	02.12.2013/**02.12.2023**
	2	1000	V-338	09.01.1985	12.05.2015/**31.12.2025**
	3	1000	V-320	20.09.1989	10.02.2020
RNPP	1	420	V-213	22.12.1980	22.12.2010/**22.12.2030**
	2	415	V-213	22.12.1981	22.12.2011/**22.12.2031**
	3	1000	V-320	21.12.1986	11.12.2017
	4	1000	V-320	10.10.2004	07.06.2035
KhNPP	1	1000	V-320	22.12.1987	13.12.2018
	2	1000	V-320	07.08.2004	07.09.2035

NPP	Power unit	Electrical power, MW	Reactor type	Beginning of commercial operation	Expiration of design-basis life/LTO period*
* this means the LTO period established at the time this report was under development, which may change later upon the operator's separate application and associated justification in compliance with regulations, rules and standards on nuclear and radiation safety in force in Ukraine.					

The total installed capacity of Ukrainian NPPs is 13,835 MW. In 2016, Energoatom's share in the national production of electricity was 52.4%.

Based on review of PSRRs for seven nuclear power units in Ukraine, SNRIU made a decision on their LTO with further implementation of ageing management measures:

- in 2010 for RNPP units 1 and 2 (20 years);
- in 2013 for SUNPP unit 1 (10 years);
- in 2015 for SUNPP unit 2 and ZNPP unit 1 (10 years);
- in 2016 for ZNPP unit 2 (10 years);
- in 2017 for ZNPP unit 3 (10 years).

In a period from 2017 to 2020, PSRRs will be submitted to SNRIU to make a decision on LTO for further five units:

- in 2017 for RNPP unit 3;
- in 2018 for ZNPP unit 4 and KhNPP unit 1;
- in 2020 for ZNPP unit 5 and SUNPP unit 3.

01.2 Process to develop the national assessment report

Considering that the Technical Specification [1] requires a significant amount of specific technical information on ageing management of NPP components to be provided in the National Reports, the operating organizations (Energoatom, Nuclear Research Institute) and the regulator's technical support organization were involved in the process and provided SNRIU with all needed data in appropriate format and within established timeframes.

To organize and coordinate the process to develop the National Report, SNRIU established Coordination and Working Groups. The Coordination Group was entrusted with overall coordination of the development and review of the draft National Report, while the Working Group directly developed the National Report using information provided by the operating organizations.

The Coordination and Working Groups included representatives of various management levels from SNRIU, SSTC NRS, Energoatom, NPPs and NRI.

At the first meeting of the Coordination Group, a detailed action plan to develop the National Report on the First Topical Peer Review was prepared. The plan included appropriate actions and responsible individuals:

No.	Action	Responsible	Deadline
1.	Translation of the WENRA Technical Specification into Ukrainian	SSTC NRS	Two weeks after approval of the Technical Specification

2.	Meeting with SNRIU, SSTC NRS, Energoatom and NRI for: 1. Introduction into the topical peer review process. 2. Approval of the National Report format. 3. Discussion of the composition, responsibilities and working procedure of the Coordination Group	SNRIU	7 November 2016
3.	First meeting of the Working Group for discussion and approval of work schedule	SNRIU	30 November 2016
4.	First meeting of the Coordination Group for discussion and approval of work schedule	SNRIU	21 December 2016
5.	Development of ageing management reports by Energoatom and NRI (to serve as the basis for the National Report)	Energoatom, NRI	28 April 2017
6.	Development of the first draft of the National Report	Working Group	28 July 2017
7.	Review of the first draft of the National Report by the Coordination Group	Coordination Group	30 August 2017
8.	Revision of the National Report upon review of the Coordination Group	Working Group	29 September 2017
9.	Review and approval of the National Report by the SNRIU Board	SNRIU	October-November 2017
10.	Translation of the National Report into English	SSTC NRS	December 2017
11.	Publication of the National Report on the SNRIU website (in Ukrainian and English)	SNRIU	December 2017

In the National Report development process, the analytical material provided by Energoatom and NRI as individual reports was addressed by SSTC NRS experts and submitted as proposals to the first draft of the National Report to SNRIU.

Following a series of revisions, the National Report was agreed by the Coordination Group and submitted for approval to the SNRIU Board.

After approval by the SNRIU Board, the National Assessment Report is published on the SNRIU website (in Ukrainian and English).

02 REQUIREMENTS AND IMPLEMENTATION OF THE OVERALL AGEING MANAGEMENT PROGRAM

The main objective of ageing management is to ensure the required safety level of a power unit over its entire lifetime (including LTO) and to achieve the maximum efficiency of its operation through technically and economically feasible measures intended to detect degradation of power unit components caused by ageing in a timely manner and keep it within acceptable limits.

02.1 National regulatory framework

The development of the national regulatory framework on ageing management was started in the early 2000s with implementation of measures under the Comprehensive Program of Activities for Long-Term Operation of Nuclear Power Plants [4].

One of the main principles of regulatory control in Ukraine is to apply a systematic hierarchic approach in the development and revision of regulations.

This principle is implemented in practice as a hierarchic pyramid of nuclear and radiation safety regulations, including documents of several levels:

Level I represents Laws of Ukraine.

Level II represents regulations on the safety of nuclear facilities adopted by the Government of Ukraine.

Level III represents regulations on the safety of nuclear facilities adopted by state regulatory bodies for nuclear and radiation safety and other central executive bodies.

Level IV represents recommending documents and operator's documents.

The hierarchic pyramid representing the legislative and regulatory framework on nuclear and radiation safety is schematized in Figure 2.1.

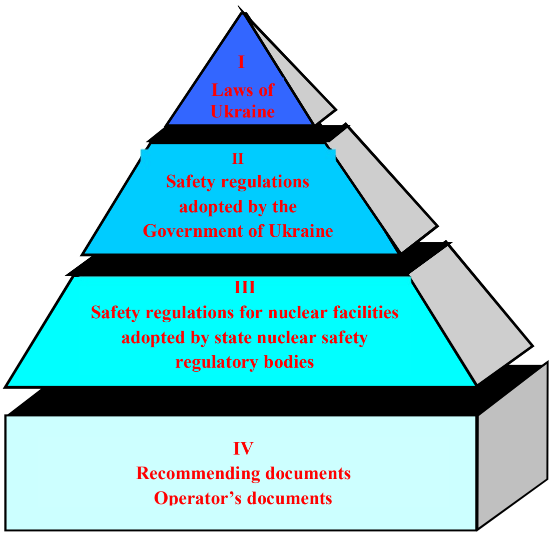

Figure 2.1 Hierarchic pyramid representing the NRS legislative and regulatory framework

The main legal documents and regulations that establish requirements for NPP LTO and AM include:

Level I – Laws of Ukraine:

- On Nuclear Energy Use and Radiation Safety [2];
- On Authorizing Activity in Nuclear Energy Use [3].

Level II – Resolutions and ordinances of the Cabinet of Ministers of Ukraine:

– On Immediate Measures for Improvement of Nuclear Energy Safety and Reliability [24];

– Comprehensive Program of Activities for Long-Term Operation of Nuclear Power Plants [4];

– Energy Strategy of Ukraine until 2035 “Safety, Energy Efficiency, Competitiveness” [87].

Level III – Regulatory documents:

– NP 306.2.141-2008. General Safety Requirements for Nuclear Power Plants (Ageing Management Section) [5];

– NP 306.2.099-2004. General Requirements for NPP Long-Term Operation Resulting from Periodic Safety Review [6];

– NP 306.2.210-2017. General Requirements for Ageing Management of Components and Structures and Long-Term Operation of NPP Units [7];

– SOU-N YaEK 1.004:2007. Requirements for the Structure and Contents of Periodic Safety Review of Operating NPPs [10].

Level IV – Operator’s standards:

– PM-D.0.03.222-14. Standard Program for Ageing Management of NPP Components and Structures [11];

– PM-T.0.08.121-14. Cable Ageing Management Program for Nuclear Power Plants [12];

– PL-D.0.03.126-10. Provisions on the Procedure for Long-Term Operation of Equipment in Systems Important to Safety [13];

– SOU NAEK 080:2014. Operation of Technological System. Long-Term Operation of NPP Units. General Provisions [14];

– SOU NAEK 109:2016. Operation of Technological System. Monitoring of NPP Structures. General Provisions [38];

– SOU NAEK 141:2017. Engineering, Scientific and Technical Support. Ageing Management of NPP Components and Structures. General Requirements [75], etc.

02.1.1 Overview of regulatory requirements for ageing management

A general scheme of the main regulatory documents for ageing management and their subordination is shown in Figure 2.2.

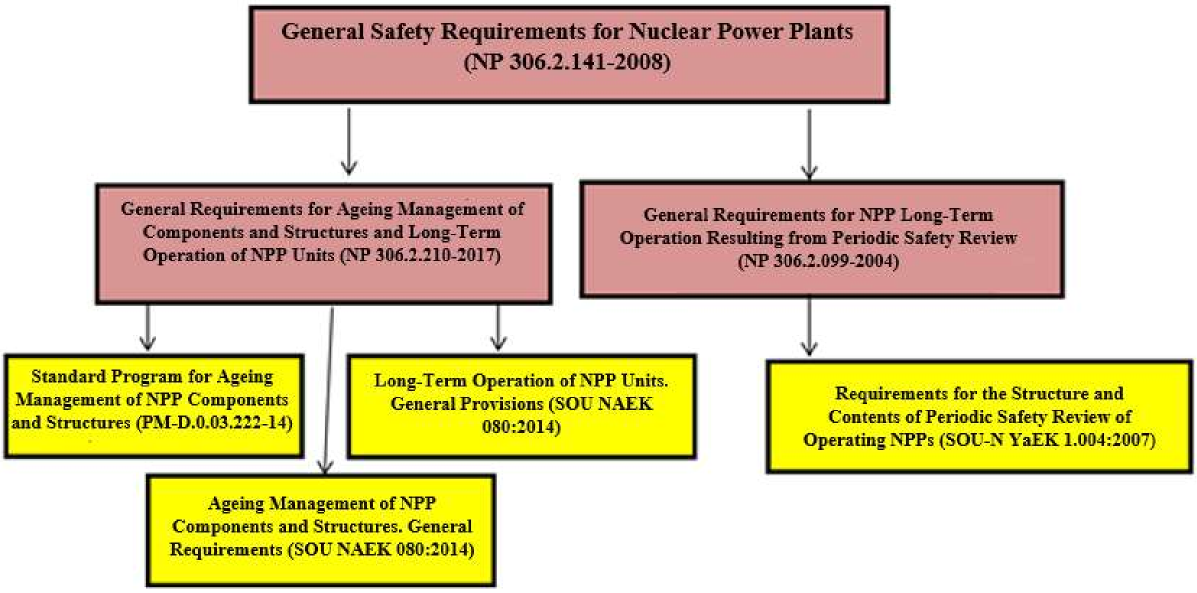

Figure 2.2 General scheme of regulations for ageing management

02.1.1.1 Requirements of NP 306.2.141-2008 General Safety Requirements for Nuclear Power Plants

NP 306.2.141-2008 [5] establishes general requirements and criteria for safety of nuclear power plants and general technical and administrative measures for their implementation. The general safety requirements for nuclear power plants are based on Ukrainian legislative provisions and take into account IAEA recommendations and national and international experience in safe operation of nuclear power plants.

According to its status, NP 306.2.141-2008 [5] contains general requirements on ageing management of power units components and structures and LTO.

In compliance with NP 306.2.141-2008 [5], to keep degradation (caused by ageing, wear, corrosion, erosion etc.) of equipment, systems and components important to safety within acceptable limits, the operator develops AMP and takes necessary actions to maintain equipment, systems and components in operable condition. Considering AMP, the operator makes a list of equipment, systems, components and structures to be analyzed in AMP and agrees it with SNRIU.

Taking into account the ageing effects and equipment degradation for each power unit, the operator assesses the capability of systems and components important to safety to perform their safety functions over the service life of power units.

The general requirements of NP 306.2.141-2008 [5] related to ageing management of power unit components and structures are detailed in NP 306.2.210-2017 [7].

02.1.1.2 Requirements of NP 306.2.210-2017 General Requirements for Ageing Management of Components and Structures and Long-Term Operation of NPP Units

NP 306.2.210-2017 [7] establishes requirements for AM and LTO of components and structures important to safety of nuclear power units.

According to NP 306.2.210-2017 [7], the operator arranges ageing management of components and structures at the following life stages of nuclear installations: design, construction, commissioning, operation (including LTO) and decommissioning.

Ageing management is arranged on a systematic basis and recorded. For systematic ageing management, the operator creates a dedicated division. This division is provided with sufficient staff and resources and entrusted with necessary authorities.

In compliance with NP 306.2.210-2017 [7], ageing of components and structures is divided into two types: physical ageing that leads to degradation and obsolescence that results from the development of knowledge and technologies and the modification of international and national requirements and standards. The operator applies an approach to ageing management that is based on the understanding of ageing effect and prediction of degradation in components and structures.

For ageing management activities, the operator identifies:

1) ageing management policy;

2) administrative and technical measures on ageing management to ensure the required safety level in compliance with regulations, rules and standards on nuclear and radiation safety;

3) measures intended to analyze ageing mechanisms and detect and prevent degradation of components and structures caused by ageing in a timely manner;

4) measures intended to predict degradation of components and structures within acceptable limits;

5) requirements for AMP for a nuclear power unit, AMPs for individual components and structures and/or programs for management of individual ageing mechanisms;

6) requirements for recording of AMP results and reports.

The ageing analyses carried out to assess and predict ageing effects of components and structures are based on conservative prognosis and operating experience (to compensate uncertainties of input data). The results of these analyses are reflected in SAR and agreed with SNRIU.

In operation of NPPs, ageing management of components and structures is carried out through AMP coordination with existing programs, such as safety improvement, in-service inspection, maintenance, surveillance, water chemistry, testing and inspection and equipment qualification programs. Results of analysis that determines the lifetime and information on operating experience are taken into account.

The development and implementation of AMP is a necessary condition for LTO of a nuclear power unit. Reports on AMP implementation are submitted by the operator to SNRIU.

02.1.1.3 Requirements of NP 306.2.099-2004 General Requirements for NPP Long-Term Operation Resulting from Periodic Safety Review and SOU-N YaEK 1.004:2007 Requirements for the Structure and Contents of Periodic Safety Review of Operating NPPs

Results of the operator ageing management activities are described in PSRR – safety factor of ageing.

Requirements for the structure and contents of this safety factor are provided in the following documents:

- *General regulatory* – in NP 306.2.099-2004 [6];
- *Technical* – in SOU-N YaEK 1.004:2007 [10].

The main objectives of the safety factor of ageing are to:

- determine that NPP has developed and efficiently implements AMP for structures, systems and components important to safety;

– justify that AMP can ensure required safety functions in further operation of a power unit.

In analysis of the safety factor of ageing, the operator demonstrates that NPP has developed and efficiently implements AMP for components important to safety. Results of activities carried out in compliance with plant-specific AMPs for power unit components and structures are briefly describes, in particular:

– operator's ageing management policy, ageing management arrangement and resources;

– methods and criteria to identify systems and components to be included into the list of critical components;

– list of systems and components included into the list of critical components (critical components of a nuclear power unit are identified individually);

– data that support ageing management;

– analysis and information on degradation mechanisms that can affect the design functions of systems and components important to safety;

– analysis of dominating degradation mechanisms caused by ageing;

– data needed to assess degradation caused by ageing, particularly those contained in design, operational and maintenance documents;

– efficiency of the maintenance program for ageing management of irreplaceable components;

– measures to monitor and mitigate ageing mechanisms and effects;

– safety criteria and safety margins for systems and components;

– prediction of technical condition of systems and components, including design safety margins and other conditions that limit the service life of a nuclear power unit.

In analysis of the safety factor of ageing, the operator and SNRIU take into account equipment qualification in compliance with NP 306.2.141-2008 [5], STP 0.03.050-2009 [32] and SOU-N YaEK 1.004:2007 [10] (safety factor of equipment qualification).

Requirements for ageing management are described in greater detail in the operator's documents.

02.2 International standards

The main documents of international organizations whose recommendations are considered by the regulator and operator in the revision of existing documents on ageing management and development of new ones are as follows:

– Report. WENRA Safety Reference Levels for Existing Reactors [15];

– Pilot Study on Long-Term Operation of Nuclear Power Plants. WENRA Harmonization Working Group. March 2014 [16];

– Safety of Nuclear Power Plants: Design. Specific Safety Requirements No. SSR-2/1, IAEA, Vienna, 2012 [17];

– Methodology for the Management of Ageing of Nuclear Power Plant Components Important to Safety. Technical Reports Series No. 338, IAEA, Vienna, 1992 [18];

– Periodic Safety Review for Nuclear Power Plants. Specific Safety Guide No. SSG-25, IAEA, Vienna, 2016 [19];

– Ageing Management for Nuclear Power Plants. Safety Guide No. NS-G-2.12, IAEA, Vienna, 2014 [20];

– Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL), Safety Reports Series No. 82, IAEA, Vienna, 2015 [21];

– Safe Long Term Operation of Nuclear Power Plants. Safety Reports Series No. 57, IAEA, Vienna, 2008 [22];

– Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants, draft Safety Guide No. DS485, IAEA, Vienna, 2017 [88];

– IAEA Safety Glossary. Terminology Used in Nuclear Safety and Radiation Protection, IAEA, Vienna, 2007 Edition [23].

Practically all new regulations of the regulator and operator are developed considering latest IAEA and WENRA recommendations and analyzed in detail in the framework of international technical assistance to Ukraine. The Ukrainian regulatory framework relating to AM and LTO was analyzed for completeness and adequacy, as well as compliance with advanced international experience and practices and IAEA recommendations, within the following international projects:

– Support to the State Nuclear Regulatory Committee of Ukraine in Assessing Implementation of Safety Improvements and Ageing Programmes at NPPs. Project TACIS U3.01/06 (UK/TS/38) [25];

– Technical Support to the SNRIU and its TSOs to Develop Their Capabilities on the Basis of Transferred Western European Safety Principles and Practices. Project INSC U3.01/08 (UK/TS/40) [26].

The analysis results provided in Reports [27] and [28] indicate that the Ukrainian regulatory requirements form a strong framework for AM and LTO of NPPs. The Ukrainian regulations and rules on AM and LTO were developed considering IAEA and WENRA recommendations and advanced international experience and practices.

Regarding specific examples of using international regulations/recommendations on NPP safety, the latest Ukrainian regulations developed from 2013 to 2017 and harmonized with international standards/recommendations [15]-[23] are as follows:

– NP 306.2.210-2017. General Requirements for Ageing Management of Components and Structures and Long-Term Operation of NPP Units [7];

– SOU NAEK 080:2014. Operation of Technological System. Long-Term Operation of NPP Units. General Provisions [14];

– SOU NAEK 109:2016. Operation of Technological System. Monitoring of NPP Structures. General Provisions [38];

– SOU NAEK 141:2017. Engineering, Scientific and Technical Support. Ageing Management of NPP Components and Structures. General Requirements [75];

– PM-D.0.03.222-14. Standard Program for Ageing Management of NPP Components and Structures [11].

02.3 Description of the overall ageing management program

The overall AMP for Energoatom is "PM-D.0.03.222-14. Standard Program for Ageing Management of NPP Components and Structures" [11], which is periodically revised to incorporate updated international recommendations and standards, SNRIU requirements, as well as operating experience and practices, and is agreed with SNRIU.

02.3.1 Scope of the overall AMP

02.3.1.1 Distribution of responsibilities within the licensee to ensure development and implementation of the overall AMP

The operator conducts ageing management of components and structures important to safety in compliance with respective SNRIU regulations. For systematic ageing management activities, the operator established an appropriate department. At the level of Energoatom Company, ageing management issues are within competences of the Department for Management of Long-Term Operation (Company Department). At the level of NPPs, ageing management issues are within competences of respective divisions, such as Service for Reliability and Lifetime and Long-Term Operation (NPP AM Service).

According to NP 306.2.141-2008 [5], to keep degradation (caused by ageing, wear, corrosion, erosion etc.) of equipment, systems and components important to safety within acceptance limits, the operator develops AMP.

To implement this requirement:

– Company Department developed “PM-D.0.03.222-14. Standard Program for Ageing Management of NPP Components and Structures” (Standard AMP) [11], which contains general requirements for implementation of ageing management measures, and agreed it with SNRIU;

– NPP AM Services developed the following unit-specific AMPs in compliance with requirements of the Standard AMP [11] and agreed them with Energoatom and SNRIU:

- Ageing Management Program for Components and Structures of ZNPP Units 1-6, 123456.1020.00.MR.PM.23-16.1N.2N,L,L,Y.3N,L,O,U;
- Ageing Management Program for KhNPP Units 1 and 2, No. 0.NR.4965.PM-16;
- Ageing Management Program for Unit 1, PM.1.3812.0196, and Ageing Management Program for Unit 2, PM.2.3812.0166;
- Ageing Management Program for RNPP Units 3 and 4, 191-220-PR-US-10;
- Ageing Management Program for RNPP Units 1 and 2, 191-136-PR-US-08.

Ageing management of components and structures is carried out by NPP personnel with involvement, if necessary, of specialized organizations, manufacturers, designers and technical support organizations.

Energoatom carries out ageing management of NPP components and structures on a systematic basis beginning from the design stage.

02.3.1.2 Methods used to identify components and structures within the overall AMP

The list of components and structures subject to ageing management is developed using current safety classification of power unit components, data of design documentation, installation and operation charts, certificates and other technical and operating documents.

According to NP 306.2.210-2017 [7], to identify components and structures for ageing management within the overall AMP, it is required to select:

1) systems that include components and structures (safety classes 1, 2 and 3 according to the power unit design) intended for:

– reactor emergency shutdown and subcriticality control;

– emergency heat removal;

– prevention or limitation of radioactive releases during accidents beyond design limits;

2) other systems that include components and structures whose failure or damage may lead to failure to perform design functions by systems, components and structures identified in item 1 of this subsection.

02.3.1.3 Methods for grouping of components/structures in the screening process

The approach to grouping of components and structures is described in SOU NAEK 080:2014 [14] with indication of the following groups:

– thermal mechanical equipment and piping;

– electrical equipment;

– I&C equipment;

– civil structures of buildings and facilities;

– hoisting machines and mechanisms.

For power unit components that include thermal mechanical and electrical equipment, individuals groups may be established according to equipment type, for example, thermal mechanical equipment and electrical equipment.

The procedure for grouping and screening of power unit components and structures subject to ageing management in accordance with SOU NAEK 141:2017 is shown in Figure 2.3.

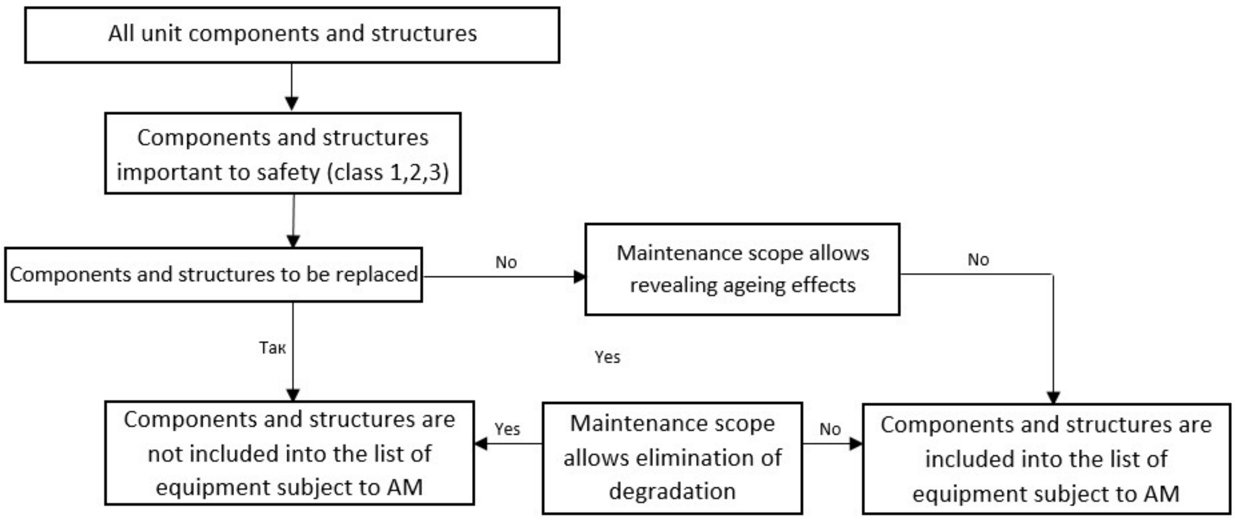

Figure 2.3 Procedure for screening and grouping components and structures subject to ageing management

* – maintenance scope envisages inspection of technical condition of vessel parts;

– location of equipment, doses for personnel and other features allow maintenance of all components and structures;

– maintenance documentation complies with current NRS requirements.

According to NP 306.2.141-2008 [5], NP 306.2.099-2004 [6], NP 306.2.210-2017 [7] and SOU NAEK 141:2017 [75], AMP includes:

– all critical components and structures of NPP unit[1] that include but are not limited to:

- RPV, closure head, internals and support elements;
- reactor coolant piping, particularly RCP casing and MGV and surge line;
- steam generators;
- pressurizer;
- ECCS accumulators;
- pressure relief tank, emergency cooldown heat exchange, regular heat exchanger of spent fuel pool;
- high-pressure system piping;
- feedwater system piping;
- emergency and auxiliary feedwater system piping;

– noncritical components important to safety whose regular scope of inspection and maintenance does not allow identifying ageing affects and mitigating degradation or whose maintenance revealed defects;

– buildings and structures containing systems and components important to safety;

– instrumentation and control systems (I&C), cables (power and control) and cable structures are included into the list in one line with references to:

- equipment qualification programs for I&C;
- AMPs for cables (power and control) and cable structures and equipment qualification programs for cables (power and control).

The operator may also decide to include normal operation components into AMP that has no impact on safety and cannot be replaced or recovered for technical or other reasons, such as:

– turbine facilities with a condenser;

– turbine generators;

– power transformers.

The list of components and structures subject to ageing management is developed individually for each power unit and included into NPP AMP or unit-specific AMP.

Based on AMP implementation and operating experience, the operator may revised the list of components and structures subject to ageing management (not more than once a year).

02.3.1.4 Methodology and requirements to assess available maintenance practice and develop ageing management programs according to identified significant ageing mechanisms

According to the overall practice of ageing management at Ukrainian NPPs, AMPs identify ageing management measures for each component and structure included in AMP. It is allowed to develop programs for individual components, but decision on the need to develop such a program for a selected component can be made by each NPP individually.

For example:

ZNPP developed AMP for containment structures "Working Ageing Management Program for Structures and Components of the Containment Safety System of Unit 1

[1] According to NP 306.2.099-2004 and SOU NAEK 141:2017, critical components of a power unit are those whose lifetime limits the power unit life and whose replacement or recovery is impossible for technical or other reasons.

Reactor Compartment. 01.RO.KhA.PM. 255-16/N.2L, 2ZL". This program was finalized and updated upon results of TCA and LTO of the containment.

RNPP developed and implemented "Ageing Management Program and Ageing Management Measures for Elements of Reactor Pressure Vessel and Reactor Vessel Main Flange. 191-232-O-PSE-16. Part 1" and "Ageing Management Program and Ageing Management Measures for Elements of Reactor Vessel Closure Head. 191-231-O-PSE-16. Part 2".

Erosion and corrosion damage of secondary piping resulting from interaction with the coolant is one of the significant ageing mechanisms. To decrease negative effects on piping and prevent its damage, the operator developed and implemented the document "SOU NAEK 040:2017. Engineering, Scientific and Technical Support. Ageing Management of NPP Equipment and Piping Susceptible to Erosion and Corrosion Wear. General Requirements" [37], which addresses ageing management measures to deal with the degradation mechanism such as erosion and corrosion wear.

According to SOU NAEK 033:2015 "Maintenance and Repair. Rules for Maintenance and Repair of Systems and Equipment at Nuclear Power Plants" [35] and SOU NAEK 113:2016 "Maintenance and Repair. Rules for Condition-Based Maintenance and Repair of Equipment at Nuclear Power Plants" [36], condition-based maintenance is carried out upon results of inspection and/or technical condition assessment of a system (component) using parameters that determine the technical condition through technical observation, technical diagnosis, degradation monitoring, testing with special procedures and/or in case when the system (component) becomes inoperable because of failure.

Failure is accompanied with detection of significant malfunctions in the system (component) during operation, maintenance, degradation monitoring, testing and technical diagnosis and/or when the parameters that determine the technical condition reach their boundary values).

To keep degradation (caused by ageing, wear, corrosion, erosion, fatigue and other mechanisms) of structures, systems and components important to safety within acceptable limits and implement necessary actions to keep them operable and reliable in operation, AMPs for individual components and structures and ageing effects may be developed (for example, AMP for cables).

02.3.1.5 Quality assurance of the overall ageing management program

The quality assurance requirements for ageing management are established in the Standard AMP [11] and in the operator's document "Quality Assurance Program for Ageing Management at NPP Units. PK-Ch.0.08.410-07" [29].

Collection and storage of data and analysis of trends in the maintenance history and operating data

To ensure effective ageing management of components and structures, all data important for ageing management are collected and processed. For this purpose, an electronic database that represents an ageing management information analysis system (AMIAS) was developed and implemented. A separate module was developed for each NPP as a software application integrated with lists, directories and classifiers of the Ukrainian reliability database for NPP equipment. The interface for this module is shown in Figure 2.4.

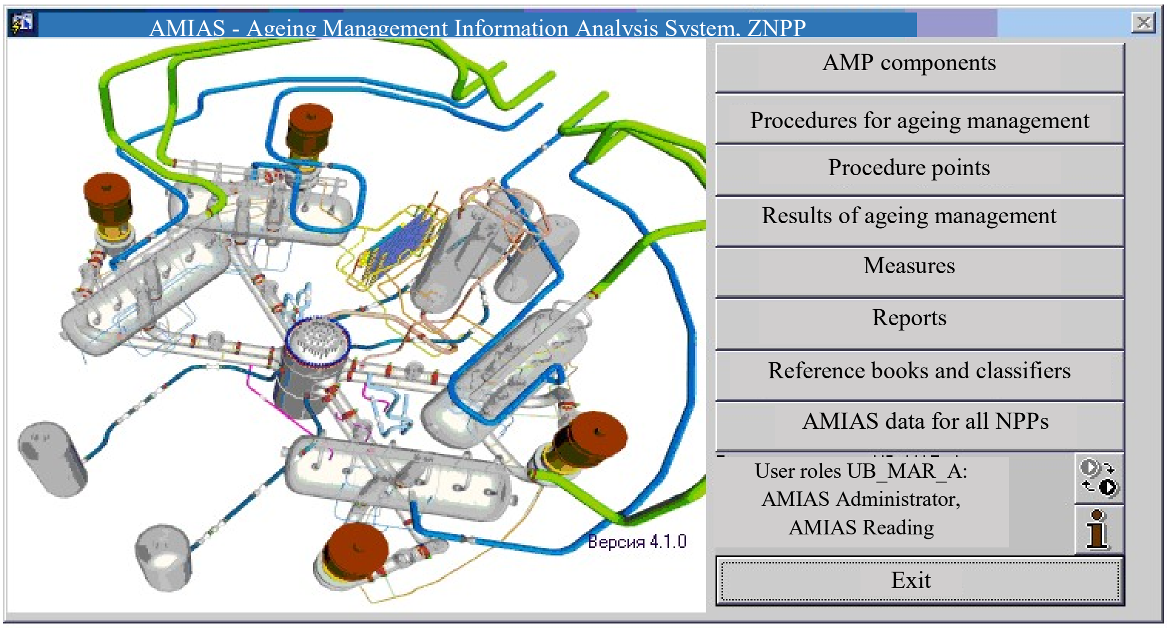

Figure 2.4 Ageing management information analysis system

AMIAS is intended to ensure effective ageing management of components and structures of all NPP units.

AMIAS provides information support of the main functions, such as:

– analysis of design, operational and maintenance documents, operating and maintenance history of components and structures;
– monitoring of ageing processes and technical condition;
– technical condition assessment of components and structures and prediction of their changes resulting from ageing relative to safety functions;
– development and technical and administrative measures to keep degradation of power unit components and structures resulting from ageing;
– collection and recording of data on components;
– recording of the actual number of loading cycles and those caused by actuation of components;
– planning, recording and monitoring of ageing management activities for components and structures;
– reporting on ageing management;
– assessment of ageing management efficiency using corresponding performance indicators;
– exchange and provision of information on ageing management of single-type components and structures of NPP units in the industry.

AMIAS collects and stores information containing design data (in particular, regulatory requirements), data on manufacturing (in particular, properties of applied materials and required operating conditions), data on maintenance history, inspection results and research findings.

Energoatom scientific and technical support ensures AMIAS maintenance considering IAEA "Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL). Safety Report Series No. 82, IAEA, Vienna, 2015" [21].

Performance indicators used to assess ageing management efficiency

Results of AMP efficiency assessment are annually included in the operator's documents individually for each NPP. For example, results of AMP efficiency assessment for 2016 are provided in the following documents:

– Annual Report on Implementation of the Ageing Management Program for ZNPP Units 1-6 for 2016 [104];

– Report on Implementation of Ageing Management Program Measures for RNPP Units 1, 2 (191-136-PR-US-08) and Units 3, 4 (191-220-PR-US-10) in 2016, No. 191-264-O-US-16, No. 191-264-O-US-16 [105];

– Annual Report on Ageing Management Measures for SUNPP-1-3 Components Subject to Ageing Management for 2016 [106];

– Report on Ageing Management of KhNPP Units 1 and 2 in 2016, No. 0.NR.0005.OT-16 [107].

All these reports are reviewed by Energoatom management and summarized for all NPPs. The summary results are provided in the operator's annual reports. The results for 2016 are provided in the "Summary Industry Report on Ageing Management Activities for NPP Components in 2016" [108].

The following data are used as performance indicators to assess efficiency of ageing management for NPP units:

– forced outages of the power unit caused by failures of components resulting from ageing;

– change in costs for scheduled outages and maintenance of components and structures and for recovery repair;

– deviations from operating parameters of power unit components and structures covered by ageing management from allowed values in operational documentation;

– change in periodicity of maintenance upon results of ageing management relative to periodicity originally established in the operational and design documentation.

The ageing management performance indicators (K_{AM}) is determined by the following equation:

$$K_{AM} = \frac{N_F - n_A \text{ (for the last four quarters)}}{N_F \text{ (for the last four quarters)}} \cdot 100\%$$

where

n_A is the number of failures caused by equipment ageing (direct causes);

N_F is the total number of failures.

In summary, data are provided on the presence or absence of failures caused by equipment ageing during each year and improvement or deterioration of the performance indicator for efficiency of ageing management at each NPP unit.

Performance indicator for efficiency of AMP implementation at Ukrainian NPPs for 2016 are considered to be satisfactory. Failures or abnormal operation due to ageing effects were not observed. AMPs are recognized as effective.

02.3.2 Ageing assessment

02.3.2.1 Use of basic standards, guidelines and production documents to develop the overall ageing management program

Based on regulatory requirements set forth in NP 306.2.141-2008 [5], NP 306.2.099-2004 [6] and NP 306.2.210-2017 [7], the operator developed and introduced the following main documents pertaining to ageing management:

– PM-D.0.03.222-14. Standard Program for Ageing Management of NPP Components and Structures [11];

– PM-T.0.08.121-14. Cable Ageing Management Program for Nuclear Power Plants [12];

– PK-Ch.0.08.410-07 Quality Assurance Program for Ageing Management at NPP Units [29].

The following guidelines and production documents were used to develop the overall Standard AMP [11]:

– PM-T.0.03.061-13. Standard Program for Periodic Inspection of Base Metal, Welds and Claddings of Equipment and Piping of VVER-1000 Nuclear Power Plants (TPPK-13) [64];

– AIEU-10.09. Standard Program for In-Service Inspection of Base Metal, Welds and Claddings of Equipment and Piping of VVER-440 Nuclear Power Plants [31];

– STP 0.03.050-2009. Qualification of NPP Equipment and Technical Devices. General Requirements [32];

– STP 0.03.078-2009. Quality Management System for Equipment Qualification. General Requirements [33];

– PM-D.0.03.476-09. Program of Activities for Equipment Qualification at Energoatom NPPs [34];

– SOU NAEK 033:2015. Maintenance and Repair. Rules for Maintenance and Repair of Systems and Equipment at Nuclear Power Plants [35];

– SOU NAEK 113:2016. Maintenance and Repair. Rules for Condition-Based Maintenance and Repair of Equipment at Nuclear Power Plants [36];

– SOU NAEK 040:2017. Engineering, Scientific and Technical Support. Ageing Management of NPP Equipment and Piping Susceptible to Erosion and Corrosion Wear. General Requirements [37];

– SOU NAEK 109:2016. Operation of Technological System. Monitoring of NPP Structures. General Provisions [38];

– SOU-N YaEK 1.013:2014. Primary Coolant of VVER-1000 Nuclear Power Reactors. Technical Requirements and Quality Assurance Methods [39];

– SOU-N YaEK 1.012:2014. Primary Coolant of VVER-440 Nuclear Power Reactors. Technical Requirements and Quality Assurance Methods [40];

– SOU-N YaEK 1.028:2013. Secondary Water Chemistry of VVER Nuclear Power Plants. Technical Requirements for Secondary Water Quality [41];

– SOU NAEK 067:2013. Management of Chemical Technologies. Water Chemistry of Essential Service Water Systems at VVER NPPs. General Requirements [42].

02.3.2.2 Key components used in NPP ageing assessment programs

According to the procedure for grouping and screening of components and structures (see para. 02.3.1.2) for ageing management (in compliance with NP 306.2.210-2017 [7]), the key components used in NPP ageing assessment programs include (for more detail, see para. 02.3.1.3):

– thermal mechanical reactor part (main primary components, such as RPV, reactor internals, reactor support elements, MCP, RCP casing, steam generator, pressurizer etc.);

– electrical part (power and control cables, cable structures, main plant generators and transformers, diesel generators);

– construction part (main civil structures of buildings and facilities, such as main building with containment, liquid radwaste treatment building, SDGS, elevated walkways, spray ponds, unit pump station, essential pump houses, cooling towers, radioactive waste storage facilities);

– handling equipment (primarily polar crane of the reactor compartment, located under the containment dome).

02.3.2.3 Processes/procedures to define ageing mechanisms and their possible consequences

The ageing mechanisms and their possible consequences are defined through technical and administrative measures intended to reveal ageing effects and establish degradation mechanisms of components and structures considering their materials, parameters of the working environment and factors that influence change in the characteristics of a component or structure with time.

The procedure to define ageing mechanisms and their possible consequences covers maintenance, metal inspection, check of water chemistry, equipment qualification, safety analysis etc.

To identify ageing mechanisms and their possible consequences, the following data are collected, analyzed and systemized:

– design basis (in particular, regulations, rules and standards);

– materials and their properties;

– operating conditions;

– safety functions;

– design and manufacture (analysis of adverse residual effects after manufacture, such as residual stresses from cold working or welding, inspection and testing in the production process);

– equipment qualification;

– operating and maintenance history (in particular, in-service metal inspection, technical examination, maintenance and upgrading);

– national and international operating experience (for example, international generic ageing lessons learned program (IGALL));

– research findings;

– ageing effects (potential or actual for a specific component or structure).

In case of change or unforeseen deviations of operational parameters or identification of new degradation mechanisms, NPP AMP is amended.

02.3.2.4 Establishment of acceptance criteria

The acceptance criteria used by the regulator to assess whether ageing management is effective and whether the operator's activities allow degradation of components and structures to be kept within acceptable limits include:

– all actions are implemented to detect ageing effects in a timely manner, predict their development and mitigate degradation;

– current values of technical condition parameters of components and structures do not exceed acceptance criteria established in regulations and standards on NRS;

– components and structures can perform their functions over the service life justified upon analysis.

02.3.2.5 Use of research and development programs

Feedback between operating experience and research findings is one of the AMP efficiency indicators. Ukraine implements a number of research programs, whose results are further considered in development of ageing management measures. These programs include but are not limited to the following:

1) Individual industry programs on erosion and corrosion wear

A series of research efforts have been made to determine the effect of erosion and corrosion factors on metal condition and minimize their consequences. Upon these research efforts:

– piping areas that are most susceptible to erosion and corrosion processes in compliance with operating experience have been identified;

– additional monitoring and inspection has been arranged for these locations;

– inspection results are introduced into certificates and analyzed;

– "Methodology for Determining the Allowed Thickness of NPP Carbon Steel Piping Elements Susceptible to Erosion and Corrosion Wear. MT-T.0.03.224-11" has been developed and agreed with SNRIU [109];

– "Engineering, Scientific and Technical Support. Ageing Management of NPP Equipment and Piping Susceptible to Erosion and Corrosion Wear. General Requirements. SOU NAEK 040:2017" has been developed [37].

2) Analysis of temperature stratification in the pressurizer surge line

Conceptual Decision "On Monitoring and Treatment of Temperature Stratification of Pressurizer Surge Line at VVER-1000 NPPs" has been developed and agreed with SNRIU. This decision is intended to:

– implement a monitoring system for temperature stratification of the pressurizer surge line at the pilot unit (ZNPP-2 is selected);

– install seven surface-type thermocouples with a step no more than 30° and introduce the monitoring system into trial operation;

– based on monitoring and instrumentation data, perform refinement calculation of surge line cyclic strength.

Upon implementation of this research program, a decision will be made on introduction of the stratification monitoring system at other power units of Ukrainian NPPs.

3) Assessment of RPV metal embrittlement based on surveillance specimens

RPV neutron fluence is monitored through the surveillance specimen program. Since Ukrainian NPPs transfer to long-term operation and there are no data on RPV metal inspection in LTO period, the following research programs have been developed for Ukrainian NPPs.

1. Integrated Program for KhNPP-2, RNPP-3, 4 and ZNPP-6 Pressure Vessel Metal Properties Using Surveillance Specimens Irradiated in Conditions Reproducing RPV Irradiation in the Beltline Region (Integrated Program).

The integrated program has been developed to obtain additional data for regular, upgraded and new surveillance programs to increase reliability in the assessment of changes in RPV metal properties. The integrated program applies to KhNPP unit 2, RNPP units 3 and 4, and ZNPP units and identifies activities to obtain additional information on the main characteristics of RPV metal for these power units. The integrated program envisages irradiation of the surveillance specimens of the above RPVs at Telelin NPP VVER-1000, analysis of their mechanical properties and systematization and summary of tests on KhNPP-2, RNPP-3&4, and ZNPP-6 RPV surveillance specimens considering new data.

2. Upgrading programs for surveillance specimen container assemblies

These programs include:

– Working Program for In-Service Inspection of SUNPP Unit 1 RPV Weld Metal Properties Using Surveillance Specimens;

– Program for Upgrading of Irradiated Container Assemblies with ZNPP Unit 2 RPV Metal Surveillance Specimens. 02.RO.00.PM.205-14/N;

– Program for Upgrading of Irradiated Container Assemblies with ZNPP Unit 4 RPV Metal Surveillance Specimens. 04.RO.NS.PM.276-17/N 3N.

These program are aimed at materials science support of RPV safe operation in LTO period in unloading of all two-layer (for prognosis) container assemblies with surveillance specimens.

4) Integrity assessment of welds of steam generator headers and heat-exchange tubes

To assess the integrity of welds of steam generator headers, a series of research programs have been developed and planned as measures under unit-specific AMP.

Methods to assess the integrity of header welds are to:

– perform comprehensive analysis of stress-strain state of SG weld No. 111 critical crack sizes;

– consider all possible operating states;

– perform elastoplastic calculation;

– verify calculation results with known strain measurements and records of MCP movement;

– calculate thermal hydraulic parameters in weld pocket.

To implement the above measures, the operator developed the following documents and submitted them to SNRIU for agreement:

– MT-T.0.03.308-14. Methodology for Assessing the Acceptability of Discontinuity Flaws Detected in the Weld Area between the Header and Nozzle DN1200 of Steam Generators PGV-1000M [110];

– SOU NAEK 060:2014. Maintenance and Repair. Procedure for Calculated Justification of Strength and Reliability of Welds between Headers and Casings of VVER-1000 Steam Generators with Discontinuity Flaws Detected in Operation. Guidance [111].

In summary, it should be noted that findings of research programs are directly reflected in AMPS for power units of Ukrainian NPPs.

02.3.2.6 Use of internal and external operational experience

Energoatom revises AMP to incorporate operating experience feedback and research, as well as results of self-assessments and expert judgements.

In determination and analysis of degradation mechanisms, summarized national and international operating experience is taken into account (for example, international generic lessons learned program (IGALL)). In particular, Ukraine implemented a number of international projects, including:

– Support to the State Nuclear Regulatory Committee of Ukraine in Assessing Implementation of Safety Improvements and Ageing Programmes at NPPs. Project TACIS U3.01/06 (UK/TS/38) [25];

– Technical Support to the SNRIU and its TSOs to Develop Their Capabilities on the Basis of Transferred Western European Safety Principles and Practices. Project INSC U3.01/08 (UK/TS/40) [26];

– Project INSC U2.01/07. Development of the Strategy for the Long Term Ukrainian NPP Safety Management [112].

The conclusions and recommendations of foreign partners made in the implementation of these projects are considered in unit-specific AMPs and regulations on ageing management.

Operating experience may also be used to revise the lists of components and structures subject to ageing management and determine ageing management measures.

For higher efficiency, planning of ageing management activities for components and structures includes collection and analysis of information obtained in the industry and from foreign sources on operating experience of components and structures and uses results of ageing management analysis.

NPPs also systematically collect information on operating experience and research findings and assesses their applicability to a specific NPP.

Acceptance criteria, operating experience and research and development findings are described in detail in AMPs for individual components and structures.

02.3.3 Monitoring, testing, sampling and inspection activities

02.3.3.1 Programs for monitoring of indicators and parameters of the condition and analysis of trends

To monitor the technical condition of power unit components and structures and assess ageing effect, in-service inspection, maintenance, testing and inspection programs for components and structures were developed and are systematically optimized. The specific examples include programs and documents that are regularly revised and updated:

– PM-T.0.03.061-13. Standard Program for Periodic Inspection of Base Metal, Welds and Claddings of Equipment and Piping of VVER-1000 Nuclear Power Plants (TPPK-13) [26];

– AIEU-10.09. Standard Program for In-Service Inspection of Base Metal, Welds and Claddings of Equipment and Piping of VVER-440 Nuclear Power Plants [31];

– SOU NAEK 033:2015. Maintenance and Repair. Rules for Maintenance and Repair of Systems and Equipment at Nuclear Power Plants [35];

– SOU NAEK 113:2016. Maintenance and Repair. Rules for Condition-Based Maintenance and Repair of Equipment at Nuclear Power Plants [36];

– SOU NAEK 040:2017. Engineering, Scientific and Technical Support. Ageing Management of NPP Equipment and Piping Susceptible to Erosion and Corrosion Wear. General Requirements [37].

Based on in-service and inspection programs for power unit components and structures:

– additional scope of condition-based maintenance and lifetime monitoring of NPP components is determined;

– necessary changes to in-service inspection programs are introduced or new inspection programs are developed in compliance with the NPP procedures.

Based on monitoring of ageing processes and analysis of tendencies in condition parameters of power unit components and structures, measures are implemented to optimize:

– frequency of inspection to check operability of components;

– frequency and scope of measurements and testing of components and structures;

– in-service metal inspection of components and structures;

– maintenance of components and structures.

02.3.3.2 Inspection programs

To analyze ageing management activities, the operator implements topical or target internal inspection programs and/or oversight programs. SNRIU permanently monitors the operator's ageing management activities in inspections, operation and preparation for LTO. These inspections are intended to check:

– compliance with requirements indicated in the operating license and individual written permits for power unit startup after scheduled outage regarding ageing management measures;

– operation and maintenance of thermal mechanical equipment of systems important to safety;

– long-term operation and ageing management of the power unit;

– measures on equipment qualification, strength and seismic resistance of equipment, piping, buildings and civil structures;

– development and implementation of SAR and PSRR, systems for accounting of operating experience upon investigation of NPP operational events;

– implementation of C(I)SIP measures;

– implementation of SNRIU prescriptions.

02.3.3.3 Provisions to define unpredictable degradation

In case of changes in operating parameters, unpredictable deviations of operating parameters from values established in regulations, standards and rules on NRS and in design documentation or identification of new ageing mechanisms/degradation effects, respective changes are introduced into AMP and agreed by the operator with SNRIU.

02.3.4 Preventive and remedial measures

Ageing management of power unit components and structures is carried out at all NPP life cycle stages and envisages:

– development of AMPs for power unit components and structures;

– identification and analysis of ageing processes for power unit components and structures (ageing understanding);

– implementation of measures to monitor ageing of power unit components and structures (ageing monitoring);

– implementation of measures to mitigate degradation;

– analysis of residual life and reliability indicators for structures, systems and components;

– continuous improvement of AMPs to incorporate operating experience feedback and research findings, as well as results of self-assessments and expert judgements (development and implementation of additional monitoring and degradation mitigation measures).

Monitoring and degradation mitigation measures are identified considering technical condition assessment of components and structures and paying special attention to the following aspects:

– support reliability of components and structures in compliance with requirements of technical documentation;

– replace components with expired lifetime in a timely manner;

– implement additional measures for condition monitoring and diagnostics (if necessary).

There are the following examples of preventive and remedial measures.

1. Recovery annealing of weld No. 4 of RNPP unit 1 RPV (see Figure 2.5).

Figure 2.5 Annealing facility

2. Upgrading of surveillance programs to justify RPV safe operation in LTO period (relocation of the existing surveillance specimens into position to allow increase in the average fluence accumulation rate by the surveillance specimens by two to three times) (see Figure 2.6).

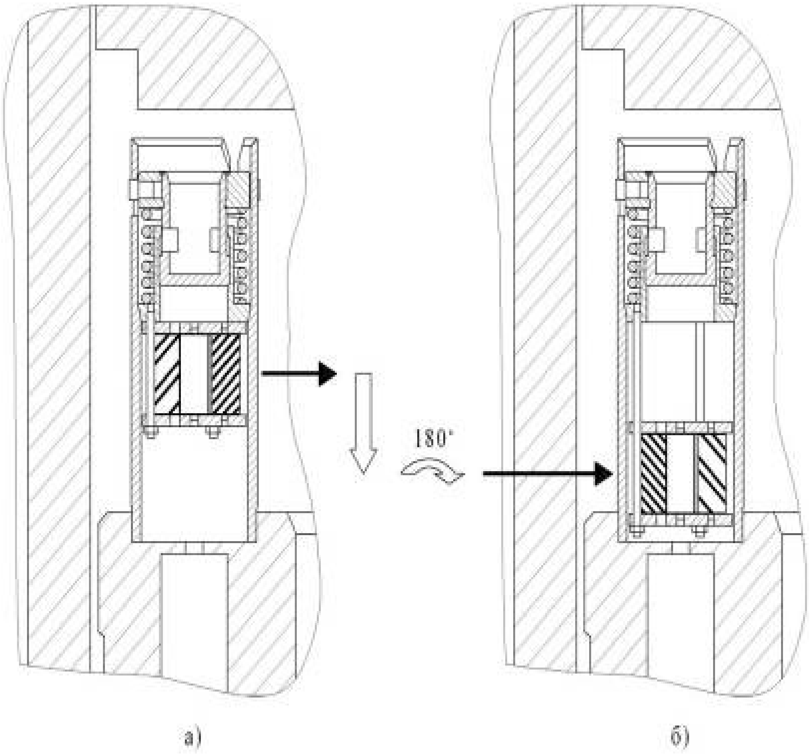

Figure 2.6 Upgrading of surveillance programs

a) location of containers with surveillance specimens in the reactor before upgrading;

b) location of containers with surveillance specimens in the same container assembly in the reactor after upgrading.

3. Measurement of geometrical dimensions of the baffle to assess its swelling under irradiation and consider the results in TLAA (see Figure 2.7).

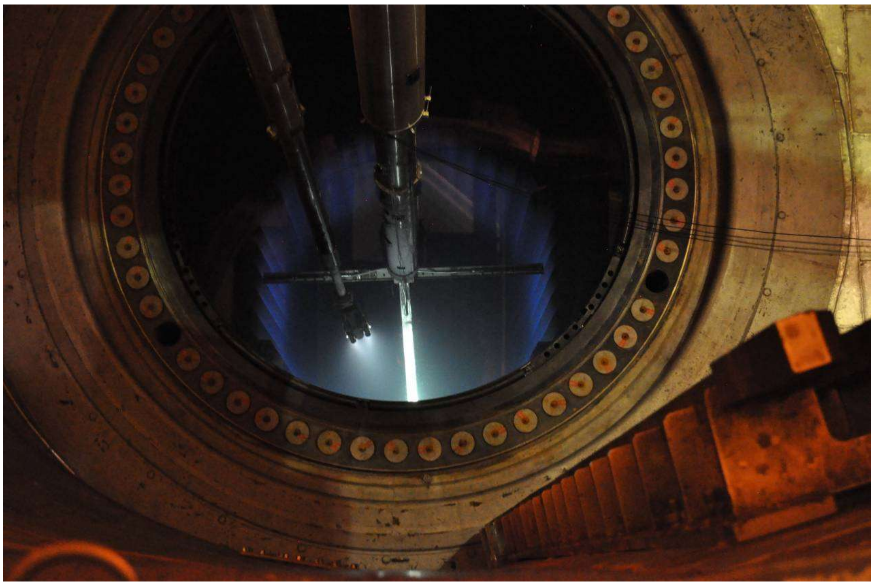

Figure 2.7 Reactor measurement system

02.4 Review and update of the overall AMP

02.4.1 Use of audit results and checking of licensee for AMP analysis and update, definition of ageing management efficiency

According to regulatory requirements, the operator ensures effective ageing management and considers ageing effects in safety justification of a nuclear power unit. The operator periodically (not less than once a year) assesses AMP efficiency using the attributes provided below.

1. For all components and structures included into AMP, ageing effects are identified and degradation mechanisms are established.
2. Ageing management activities are intended to monitor, prevent or mitigate degradation of components and structures.
3. TCPs of component and structures are monitored by the operator with the frequency established in technical and regulatory documents. Ageing effects are revealed before TCPs reach criteria established in regulatory, operational and design documents (acceptance criteria).
4. Current TCP values for components and structures are comparable with results of previous monitoring, degradation rates are determined and predicted, scope and frequency of monitoring is revised (if necessary).
5. Ageing effects are monitored under in-service metal inspection programs, TCPs comply with acceptance criteria. If acceptance criteria are not compiled with, corrective actions or replacement of a component or structure are envisaged.
6. Corrective actions are implemented to ensure performance of design functions by a component or structure over the entire operating period. Acceptance criteria are met after implementation of corrective actions.
7. Actions to be taken if acceptance criteria are not met are identified and described in detail. Corrective actions that cover determination of root causes and mitigation of degradation are performed in a timely manner.
8. AMP incorporates results of operating experience analysis, metal inspection, testing and research on identification of ageing effects and mitigation of degradation.
9. NPP administration continuously monitors recording of the ageing management process. The management system for NPP activities ensure AMP implementation, updating and systematic analysis and monitoring for compliance with all efficiency criteria.

AMP is also improved in accordance with the current level of science and technology considering operating experience, results of ageing management activities, as well as results of safety review and self-assessment.

Self-assessment carried out at NPP includes:

- self-assessment of management staff (including management staff involved in ageing management activities);
- independent assessment of ageing management activities and interrelation with other technical programs (such as maintenance, equipment qualification etc.);
- assessment of ageing management activities in periodic safety review of a nuclear power unit.

If self-assessment revealed AMP drawbacks, measures are developed to identify responsible performers and deadlines.

Results of AMP implementation are also considered in SNRIU inspections. According to established practice, SNRIU conducts two comprehensive inspections in power unit preparation for LTO upon completion of the design-basis life. The first inspection is conducted six to eight months before the power unit starts LTO and is intended to check preparations for LTO. Significant attention is paid to implementation of ageing management measures. The second comprehensive inspection is conducted about one month before the power unit starts LOT and focuses on completed measures and power unit preparedness for LTO. Based on inspections, AMP may be amended in compliance with SNRIU prescriptions.

02.4.2 Review of the time limited ageing analysis (TLAA)

Time limited ageing analysis demonstrates that TCPs for a specific component or structure will not exceed criteria established in NRS regulations, standards and rules and design documentation over the entire period for which analysis is carried out. A specific list of TCPs considered in time limited ageing analysis is made upon TCA results and operating experience.

The following TLAAs are used in Ukraine in compliance with NP 306.2.210-2017 [7]:

1. For thermal mechanical equipment and piping:

- brittle fracture resistance of the reactor pressure vessel and elements of the main reactor piping;
- cracking in reactor pressure vessel cladding;
- radiation and thermal embrittlement of reactor support elements;
- fatigue (low-cycle and high-cycle);
- stratification of piping (where this mechanism is identified upon analytical or experimental studies);
- radiation swelling, vibration fatigue, change in mechanical properties of reactor internals;
- fatigue of heat-change tubes, change in mechanical properties of steam generator tubing;
- thermal embrittlement (for components where this mechanism is identified upon analytical or experimental studies);
- fatigue of penetrations;
- corrosion resistance;

– erosion and corrosion wear of piping;
– fatigue of RCP flywheel;
– stress corrosion cracking;
– change in metal mechanical properties.

2. For structures and buildings:

– radiation and thermal embrittlement of reactor support frame;
– change in mechanical properties of concrete and metal;
– corrosion damage (loss of material);
– relaxation of containment tendon wire;
– fatigue of lining and welds;
– change in settling of buildings and structures (prediction of these changes).

3. For electrical equipment and instrumentation:

– environmental impact on elements of electrical equipment and I&C (environmental qualification).

4. For cables:

– thermal degradation of insulation materials;
– loss of isolating properties of materials under the action of electrical field and moisture.

Time limited ageing analysis is considered acceptable if it meets one of the following criteria:

– analysis remains valid for the intended period of LTO. TCPs of components and structures will not exceed the value accepted in ageing analysis that determines the lifetime;

– analysis has been projected for the design-basis/long-term operation period. Analyzed TCP value changes with time but continues to meet the acceptance criteria at the end of the design-basis/long-term operation period;

– effects of ageing on the intended function of a structure or component will monitored for the design-basis/long-term operation period. AMP ensures that a component or structure performs design functions by monitoring of TCP compliance with acceptance criteria over the entire power unit operating period and timely implementation of compensatory measures, in particular, degradation mitigation measures.

The above-mentioned analyses are carried out in power unit preparation for LTO. Results of the analysis serve as a basis for developing additional (if necessary) ageing management measures, which in turn leads to AMP revision/updating.

Measures added to AMP based on TLAA results may also serve as a basis for new or continuation of existing research activities. For example, based on experience in AMP implementation at RNPP-1&2, SUNPP-1&2 and ZNPP-1&2, the operator developed the “Action Plan for Justification of Long-Term Operation of Ukrainian VVER-1000 and VVER-440 NPPs” [71] and agreed it with SNRIU. This action plant, in particular, includes the following measures:

– meeting of scientific and technical council to discuss methodological aspects of RPV operation at Ukrainian NPPs;

– development of a new standard program for technical condition assessment and LTO of RPV and its elements at Ukrainian NPPs and its agreement with SNRIU;

– development of methodology for determining the degree of RPV embrittlement and the period for unloading of the surveillance specimens;

– upgrading of the surveillance program etc.

Completion of these and other measures and research efforts will be considered in revision and updating of the overall AMP and unit-specific AMPs.

02.4.3 Strategy of periodic assessment of the overall AMP according to the periodic safety review frequency

Regulatory safety requirements (NP 306.2.141-2008 [5]) provide for periodic safety review of power units every 10 years or upon the regulator's request. Periodic safety review is an instrument for analysis of power unit safe operation over the entire life. In the framework of periodic safety review, Energoatom in particular assesses effect of ageing on NPP safety.

The objective of periodic safety review is to determine:

– compliance of the current safety level of a power unit with NRS regulations, standards and rules, as well as design and operational documents and SAR;

– adequacy and efficiency of existing conditions that maintain the proper safety level of the power unit until the next periodic safety review or until completion of its operation (if power unit operation is terminated before the next scheduled outage);

– list and timeframes of power unit safety improvement measures required to eliminate or mitigate drawbacks if revealed by safety analyses.

The objective of ageing management within periodic safety review is to confirm the efficiency of AMPs for components and structures important to safety and their capability to ensure performance of design functions by these components and structures in further operation of the power unit.

The operator updates (if necessary) AMPs upon technical condition assessment of power unit components and structures important to safety. AMP is an inseparable part of the PSRR safety factor of ageing.

This approach has been implemented at all Ukrainian NPPs in periodic safety review and all unit-specific AMPs have been revised and updated upon results of this review.

02.4.4 Inclusion of unpredicted or new issues into AMP

In planning of ageing management of components and structures, in particular, in case of unforeseen or new issues, it is required to:

– identify ageing management activities to be carried out to ensure safe and effective operation of components, including additional measures for lifetime management and replacement of components (parts of components) if their technical condition does not meet established requirements or their life has expired;

– identify inspection and diagnosis methods and means to be used to assess technical condition of components and structures;

– determine specific scopes of activities for each component (system), in particular, scopes and sequence of tests and measurements;

– specify conditions for repair, maintenance and upgrading, including development of NRS measures, fire safety measures and occupational safety measures;

– determine the effect of proposed activities on power unit safety;

– establish requirements for qualification of personnel involved in activities and determine the need for additional special staff training.

For each power unit, schedules for ageing management activities are developed. The schedules cover:

– measures intended for timely detection of ageing effects for components and structures;

– measures for monitoring of ageing processes and technical condition of power unit components and structures;

– corrective measures related to ageing management.

02.5 Summary of ageing management processes and key findings by the licensees

Upon summary of the ageing management processes by the licensee, the following conclusions are made:

– in compliance with regulatory requirements, the operator developed detailed technical requirements that cover all aspects of ageing management activities. These requirements were developed considering international and national experience and practices, IAEA recommendations and provisions of WENRA documents;

– operator established subdivisions at each NPP that carry out ageing management activities. These subdivisions are provided with adequate financial, material and human resources;

– regulatory documents were developed to clearly determine requirements for the selection of components and structures for ageing management;

– regulatory requirements were established for timely implementation of preventive and remedial measures to mitigate degradation;

– AMP efficiency is assessed and the operator's self-assessment and independent assessment of ageing management activities are carried out on a permanent basis;

– results of ageing management activities are properly recorded and included into AMIAS.

Results of independent assessments by international organizations indicate that current Ukrainian regulatory requirements on ageing management form a strong framework for the solution of these issues. The Ukrainian regulations in this area were developed considering IAEA and WENRA recommendations and the best international experience and practices. The regulations are revised and improved in a planned manner.

02.6 Regulatory oversight

The main regulatory requirements for ageing management and associated processes are set forth in NP 306.2.141-2008 [5], NP 306.2.099-2004 [6] and NP 306.2.210-2017 [7].

During development of the National Report, the regulatory document "Requirements for Periodic Safety Review of Nuclear Power Plants (NP 306.2.214-2017)" [89] came into force.

To implement overall regulatory requirements, the operator developed a number of documents with detailed technical requirements for ageing management and Standard AMP [11], which were assessed by SNRIU for compliance with NRS regulations, standards and rules, international experience and practices and IAEA and WENRA recommendations. All documents developed by the operator were favorably evaluated after resolution of SNRIU comments and agreed for application.

The Standard AMP [11] was developed by the operator in 2004, and implementation of ageing management approaches at Ukrainian NPPs has begun since that time. Ageing management components (maintenance, in-service nondestructive and destructive metal inspections, definition of degradation mechanisms and development of measures for their prevention and mitigation) have been present in the operational practices since NPP commercial operation. The Standard AMP [11] is regularly revised by the operator with introduction of changes and updates to incorporate experience in implementation of ageing management measures, results of new research activities on ageing management and other measures.

It should be separately noted that regulatory oversight of completeness and adequacy of the operator's detailed technical requirements is conducted on a permanent basis.

02.7 Regulator's assessment of the overall ageing management program

At present, the Standard AMP [11] is the main document of the operator and establishes overall requirements for the procedure for ageing management of components and structures and determines the scope and sequence of LTO activities. The main drawback of the Standard AMP [11] is that it combines aspects of AM and LTO, while they should be governed by separate documents of the operator, as required in NP 306.2.210-2017 [7]. This drawback is noted in the SSTC NRS Report "Generalized Analysis of Documentation on TCA and LTO and Ageing Management of NPP Units. Recommendations on Revision of Regulatory Framework" [72], prepared upon SNRIU request to find drawbacks in the regulatory framework related to AM and LTO. Currently, this drawback has been practically removed by the operator through development of two separate industry standards that govern AM and LTO: SOU NAEK 080:2014 [14] and SOU NAEK 141:2017 [75].

SOU NAEK 141:2017 [75] is currently in the final agreement stage. When this standard comes into force, the Standard AMP [11] will become invalid.

SNRIU conducts continuous oversight and monitoring of AMP implementation at Ukrainian NPPs. According to NP 306.2.210-2017 [7], the operator annually submits reports on AMP implementation to SNRIU. SNRIU assesses and checks information provided in the operator's reports during scheduled inspections at NPPs, particularly in assessment of issues related to ageing management.

03 ELECTRICAL CABLES

03.1 Ageing management for electrical cables

The main objective of cable ageing management is safe, reliable and economically efficient operation of cables to perform their functions during the design lifetime and long-term operation.

Figure 3.1 shows a methodological algorithm for cable ageing management.

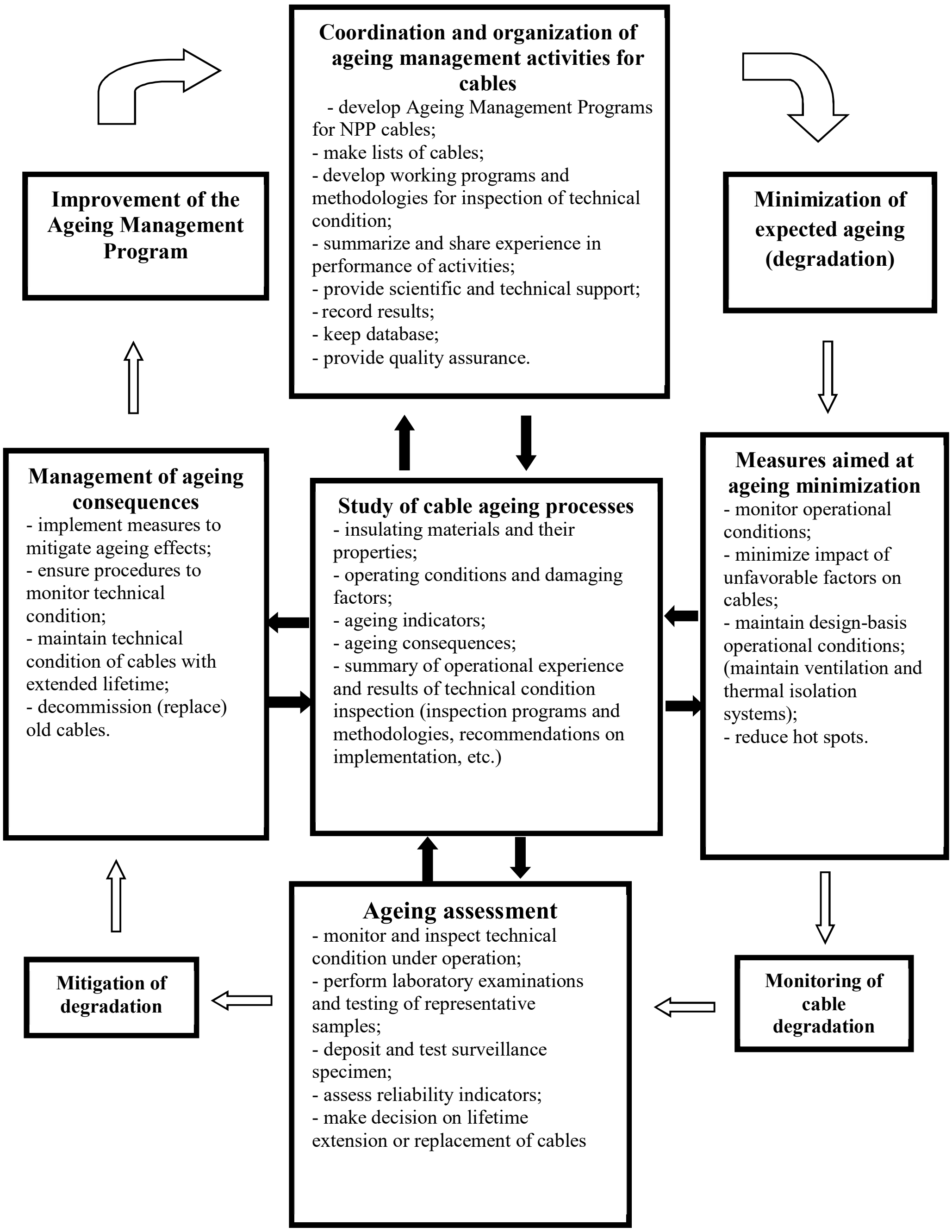

Figure 3.1 Methodological algorithm for cable ageing management

03.1.1 Scope of ageing management for electrical cables

Cable ageing management envisages the following activities:

- develop cable AMP for power units considering features of the NPP cable system;
- monitor operating conditions;
- identify cables and make a list of representative cables for inspection;

– place (deposit) surveillance specimens;

– develop TCA working programs and methodologies for groups of single-type cables;

– perform TCA of cables;

– record results and keep databases;

– ensure quality.

"Cable Ageing Management Program for Nuclear Power Plants" PM-T.0.08.121-14 (CAMP) [12] was developed for cable ageing management. This Program is standard and applies to control (management, instrumentation, alarm and interlock) cables with plastic and rubber insulation and to medium-voltage (from 380 V to 3 kV) and high-voltage (over 3 kV) cables with plastic and paper-oil insulation, which are operated at NPP units and belong to the systems and components important to safety (safety classes 2 and 3) in compliance with NP 306.2.141-2008 [5].

CAMP [12] specifies:

– content of ageing management activities;

– requirements for the "Cable Ageing Management Program for Nuclear Power Plant Unit" (CAMPU);

– requirements for the contents of "Working Programs for Technical Condition Assessment of Cables";

– requirements and principles for the development of lists of cables subjected to TCA;

– procedures to monitor operating conditions of cables and detect hot spots;

– methods for cable TCA;

– requirements for cable TCA aimed at their lifetime extension;

– requirements for databases and information on cables to be included into databases on cable operation;

– records on ageing management measures;

– content of scientific and technical support and assistance to activities;

– requirements for quality assurance.

Based on CAMP [12], NPP develops CAMPU for several or each power unit considering cable operation features, for example:

– 191-04-PR-SNRiPE, RNPP Cable Ageing Management Program [113];

– 123456.1020.00.MR.00.PM.11-16, Cable Ageing Management Program for ZNPP Units 1-6 and Common-Plant Equipment [114].

The types of cables at different NPPs are the same, thus the approaches to cable AMP and LTO are similar.

During cable ageing management activities, the following lists were developed at each NPP:

– general list of cables subject to AM (General List), the recommended form of which is provided in Table 3.1;

– "List of Representative Cables of NPP Unit Subject to Technical Condition Assessment", the recommended form of which is provided in Table 3.2 (List of Representative Cables);

– list of hot spots;

– list of cable surveillance specimens.

The General List of cables is made in electronic form based on information of the Ukrainian NPP Reliability Database (URDB).

Table 3.1 Form for General List

No.	Cable type	Design mark	Identification place	Classification according to General Safety Provisions	Affiliation with systems important to safety	Regulation that defines cable lifetime	Cable lifetime according to regulation	Commissioning date	Lifetime expiration date according to regulation	Document that extends lifetime	Date of operation termination (design)

Table 3.2 Form for List of Representative Cables

No.	Power unit No.	Type of representative cable, length	TS, GOST	Service life, years	Type of cable insulating material		Type of cables in operation	Mounting name of representative cable	Cable boundaries (devices connected to cable)	Routing in rooms	Safety class according to General Safety Provisions	Year of commissioning	Shop (section)	Parameters of hot spot		
					Cable core	Cable sheath								Location (unit compartment, elevation, room, etc.)	Ambient temperature, °C	Level of gamma radiation, µSv/h
1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17

The automated database was developed and implemented for practical use to arrange and provide collection, accumulation and systematization of information on cable AM, accounting and management of work related to cable AM at NPP units.

The database includes information on technical condition of cables, their life and AM procedures and LTO measures.

The database provides information support for the following procedures:

– submission and record of information on cables;
– planning, accounting and management of work related to cable TCA and AM;
– reporting on cable TCA and AM;
– development of technical and organizational cable ageing management measures;
– information exchange on cable operation between NPP units and the Energoatom Company's Directorate.

The General List serves as the basis for making the List of Representative Cables.

It is allowed to develop one list of representative cables for several power units provided that positions are associated with the NPP unit number.

The List of Representative Cables includes cables whose lifetime expires prior to unit LTO expiration and which are subjected to harsh operating conditions (in hot spots).

It is allowed not to include cables whose replacement is economically and technologically possible and advisable before expiration of their lifetime into the List of Representative Cables.

As an example, Table 3.3 provides a List of Representative Cables for RNPP unit 4 subject to TCA. The source is Report "RNPP Unit 4. PSRR" (381306.203.006.OB.01.04)

[115]. Representative cables were selected for TCA in compliance with requirements of 191-04-PR-SNRiPE [113] and CAMP [12].

Table 3.3 List of Representative Cables for RNPP Unit 4 Subject to TCA

Cable No.	Cable grade		Cable boundaries (devices connected by cable)	Routing (rooms with cable routing)		Commissioning date	Cable type	Regulation (service life)
	factory	installation		design	rooms			
27337K (1900067)	KVVGng 7x2.5	TX12S05k335	LV05-8 – sk TZ12S05	1080/4 1099/4 1078/51003/19 102/10 1013/7 1016/4 1062/3 235/3	AE607/1-AE038/1	2004	control	GOST 1508-78 (basic regulation) TU 16-705.426-86 25 years
20609K (65)	KVVGEng 7x1.5	UM10KK902	HZ10S (UVS) – sk UM10XK08	303.4/2 314/1 304/2 326/2 326.3/4 326.2/4 326.1/6 104/0 104.2/6 8/6 12/5 12.6/6 12.8/6 28/3 18.5	EK-1203-turbine hall	2004	control	GOST 1508-78 (basic regulation) TU 16-705.426-86 25 years
27036k-10-1	KPoBVng 7x1.5	TZ50L17k500	TZ50L17B1-PEG1 Tr73 GA-315/1	1170/1143/1110	GA406-GA315/1	2002	control	TU 16-705.432-86 30 years
54743k, 113	KPoEVng 7x1.5	VB14T01k500	VB14T01B1 – PEG116 Cx9 SU1	1169/1 1134/3 1133/3 1131/5 1110/8	GA603/3-GA-315/1	2002	control	TU 16-705.432-86 30 years
54333k, 32	KUGVVEng x0.5	SUZ-1784	HQ128 – HD25	–	AE726-AE128/2	2004	control	TU 16-505.856-75 (basic regulation) TU 16-705.426-86 20 years
54743k, 113	KUGVEVng x0.5	HN22k013	HD23 – HN22	3370 E-299a/3368 E-295a/3295/3280 3273/3119/3279/3121 3003/3126/3025/3238	AE 128/2-AE725/2	2004	control	TU 16-505.856-75 (basic regulation) TU 16-705.426-86 20 years

Grouping criteria comply with CAMP requirements [12] and inspection methodologies:

– control or power cables;
– cable type;
– voltage (1 kV or 6 kV) was also considered for power cables.

According to the “Methodology for Adaptation of Cable Laboratory Testing Results to Other NPP Units for the Purpose of Ageing Management” MT-D.0.03.530-11 [90], cables are combined into groups of single-type cables in compliance with the following features:

– regulatory documents on cable. Cables were manufactured according to the same TS (GOST);

– manufacturer. One group includes cables produced by the same manufacturer. If cables were produced by different manufacturers but in compliance with the same TS (GOST), it is allowed to combine cables into one group;

– functional purpose. One group includes cables with the same functional purpose (options: control, 1 kV power cables, 6 kV power cables, etc.);

– design and design features. Identical type of core insulation and sheath insulation (options: polyethylene, irradiated or vulcanized polyethylene, polyvinyl chloride (PVC), rubber, fluorine plastic, paper-oil insulation, etc.);

– identical type of cable protection (sealing) insulation (options: sheath made of lead or aluminum; armor or shield). Groups of single-type cables according to MT-D.0.03.530-11 [90] are presented in Table 3.4.

Table 3.4 Groups of Single-Type Cables

Group No.	Cable type	Regulatory document for cable
1	6 kV AAShv	GOST 18410-73. 6 kV Power Cables
2	6 kV AABnlG, TsAABnlG	TU 16-505.840-84. 6 kV Power Cables
3	1 kV AVVG, VVG, AVVGng, VVGng	GOST 16442-80 (basic regulation for all cables). 1 kV Power Cables TU 16-705.426-86. Flame Retardant Cables for Nuclear Power Plants
4	1 kV PvSG	TU 16-505.948-81. 1 kV Power Cables
5	6 kV PvSG	TU 16-505.948-81. 6 kV Power Cables
6	6 kV PvVng	TU 16-705.431-86. 6 kV Power Cables
7	1 kV PvBVng	TU 16-705.431-86. 1 kV Power Cables
8	KUGVEV, KUGVVE, KUGVV, KUGVEVng, KUGVVEng, KUGVVng	TU 16-505.856-75 (basic regulation for all cables). Flexible Control and Instrumentation Cables TU 16-705.426-86. Flame Retardant Cables for Nuclear Power Plants
9	KVVG, KVVGE, KVVGng, KVVGEng	GOST 1508-78 (basic regulation for all cables). Control Cables TU 16-705.426-86. Flame Retardant Cables for Nuclear Power Plants
10	KPoSG, KPoESV	TU 16-505.949-81. Control Cables
11	KPoBOV, KPoEOV	TU 16-505.949-81 (amendment 1 of 1984). Control Cables
12	KMPEV, KMPV, KMPVE, KMPVEV, KMPEVE, KMPEVEV, KMPVVE, KMPEVng, KMPVng, KMPVEng, KMPEVEng	TU 16-705.169-80 (basic regulation for all cables). Small cables TU 16-705.426-86. Flame Retardant Cables for Nuclear Power Plants
13	KPETI, KPETIng, PETI	TU 16-505.883-76. Instrumentation Cables and Wires
14	MKSh, MKShM, MKESh,	GOST 10348-80. Installation Cables
15	KNRETE, KNRE, KNRT, KNRTE, KNRET	GOST 7866.1-76. Shipboard Cables
16	SPVr	TU 16-705.126-80. Special Cables for Radiation Monitoring
17	SPOVr	TU 16-705.126-80. Special Cables for Radiation Monitoring
18	TPV, TPVng	GOST 22498-88 (basic regulation for all cables). Local Telephone Cables

19	TSV	TU 16.K71-005-87. Office Telephone Cables
20	KPoBVng, KPoEVng	TU 16-705-432-86. Control Cables
21	MSTP, MSTPE	TU 16.505.554-81. Heat- and Radiation-Resistant Installation Wires
22	PVTF-5, PVTFE-5	TU 16-505.287-71. High-Voltage Heat-Resistant Wires

03.1.2 Ageing management for electrical cables

Cable ageing mechanisms are identified mainly according to the type of cable insulation material and affect cable through damage factors. The main damage factors at NPPs are thermal, electric, mechanical and radiation ones. These factors can affect a cable separately and in combination. As a result of this effect, the following basic ageing mechanisms can be developed:

– thermal degradation of cable core and sheath insulation due to the environmental temperature impact and ohmic heating of cores (for power cables);

– loss of surface insulation features under the impact of electric field;

– elongation at tension through cable slipping, residual deformation during compression, accidental physical damage during operation;

– for cables in the containment – loss of insulating properties of the material due to the break of molecular chains with formation of free radicals under radiation impact and elevated temperatures.

Additional ageing mechanisms include:

– for power cables with paper-oil insulation - ageing of permeable material and paper tapes, formation of air pores through viscous leaks, formation of waxy deposits under the action of partial discharges in oil layers and trapped air between paper layers;

– for cables with polyethylene insulation - formation of tree-like paths, dendrites, caused by partial breakdown: electric paths, which are formed under the impact of a strong electric field; water paths resulting from water penetration into the insulation in places of cable damage;

– for cables with PVC insulation - decrease of insulation elasticity due to migration and evaporation of plasticizers, destruction of PVC resin with release of hydrogen chloride (especially intensive after hydrogen chloride saturation of lead-based stabilizers included into PVC insulation).

The recorded changes in cable insulation properties that can be observed in in-service inspection of cables are as follows:

– changes in appearance of external sheath;
– reduction in electrical resistance of insulation;
– increase in dielectric loss tangent;
– change in electrical capacity;
– increase in partial discharges (for power cables);
– change in renewable voltage parameters (for power cables);
– changes in relaxation current parameters (for power cables);
– increase in hardness in PVC cable sheath.

Taking into account the accumulated international and national experience in TCA of control and power cables at nuclear power plants, as well as the availability of modern diagnostic equipment to determine the above parameters in operating conditions, the following methodologies were developed and implemented:

– Methodology for Inspection of Technical Condition of Control Cables at Nuclear Power Plants by Nondestructive Methods, MT-T.0.03.160-16 [116];

– Methodology for Inspection of Technical Condition of Power Cables at Nuclear Power Plants by Nondestructive Methods, MT-T.0.03.172-12 [117].

In addition, the following methodologies are in force at NPPs:

– Methodology for Determining the Period of Long-Term Operation for NPP Cables, MT.0.03.339-14 [118];

– Methodology for Adaptation of Laboratory Testing Results for Cables at Other NPP Units for Ageing Management Purposes, MT-D.0.03.530-11 [119].

In developing the above methodologies, the following sources of information and provisions were used:

– Analytical Review of Scientific and Technical Information on Ageing of Nuclear Power Plant Cables, Kharkiv, I&C Certification Center, 2003, 81 p. [91];

– IEC 216 Standard, Guide for the Determination of Thermal Endurance Properties of Electrical Insulating Materials. Part 1: General Guidelines for Ageing Procedures and Evaluation of Test Results; Part 2: Choice Criteria; Part 3: Instruction for Calculating Thermal Endurance Characteristics; Part 4: Ageing Ovens; Part 5: Guidelines for Application of Thermal Endurance Characteristics. Fourth Issue (1990 - 1994) [92];

– IEC 1026 Standard, Guidelines for Application of Analytical Test Methods for Thermal Endurance Testing of Electrical Insulating Materials, First Issue, 1991 [93];

– Assessment of Cable Ageing Using Condition Monitoring Techniques, ICONE-8, Proceedings of the 8th International Conference on Nuclear Engineering (2000), 79–89 [94];

– Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: In-Containment Instrumentation and Control Cables, TECHDOC-1188, Vol. 1-2, IAEA, Vienna, 2000 [95];

– Safety Evaluation Report Related to the License Renewal of Arkansas Nuclear One, Unit 1. Docket No. 50-313, Entergy Operations, Inc., U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, April 2001, 494 pp. [96];

– Equipment Qualification in Operational Nuclear Power Plants: Upgrading, Preserving and Reviewing, IAEA Safety Reports Series No. 3, 1998 [97];

– Safety of Nuclear Power Plants: Design, Safety Standards Series No. NS-R-1, 2000 [98];

– Instrumentation and Control Systems Important to Safety in Nuclear Power Plants, Safety Standards Series No. NS-G-1.3, 2002 [99];

– IEC 60780:1998, Nuclear Power Plants. Electrical Equipment of the Safety System – Qualification [100];

– IEEE 323-2003, Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations [101].

NPPs incorporated an approach according to which the subdivision responsible for cables starts TCA activities three years before lifetime expiration. This allows planning of activities in time and sufficient time to agree the Decision with the regulator.

The developer of CAMP [12], Engineering and Technical Center for Equipment Lifetime Qualification and Assessment “KORO” (Kharkiv, Ukraine), has access to the SCAP Cable international database and R&D programs, considers international experience in its activities and is a corporate supplier of services to the Energoatom Company related to cable ageing management.

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

One of the main elements in cable ageing management at power units is monitoring of cable operating conditions.

Cable operating conditions are monitored at NPPs in compliance with the requirements of plant-specific programs/working programs in two stages:

- initial monitoring;
- regular monitoring.

The monitoring objective is to determine the actual operating conditions for cables.

The monitoring scope is determined with consideration of EIFs that make the main contribution to the cable ageing process. Such factors include:

- environmental temperature;
- ionizing radiation.

Temperature is dominant EIF under these conditions.

The initial monitoring is performed in all rooms (places) with cable routes to identify hot spots.

The regular monitoring is performed only in hot spots identified upon the initial monitoring for each type of operated cable on a regular basis until the power unit is decommissioned.

The following data are additionally used to specify hot spots:

- operating experience;
- technical inspection of cable routes.

Technical inspection of cable routes as part of cable maintenance is an important and efficient element of ageing management aimed at identification of hot spots according to the following features:

- change in color of cable sheath;
- condensation on cable sheath resulting from plasticizer desorption;
- presence of moisture drops on surface;
- presence of oil;
- sheath cracking.

During technical inspection, proper attention is paid to the environmental conditions along the cable route: presence of thermal and radiation sources, water or oil leaks, boron compounds and other active chemical compounds, condition of thermal insulation at adjacent pipework and vessels. Proper attention in the containment is paid to sealing of cable penetrations.

During activities related to cable ageing management, a list of revealed hot spots is kept on a mandatory basis, and measurement results obtained in monitoring of operating conditions are recorded in the operating condition monitoring logbook and in the database.

The list of revealed hot spots includes the following information:

- number of a hot spot;
- room (identifier for database);
- place in the room;
- values of measured parameters of EIFs;
- nature of additional factors;
- measurement date.

Upon analysis of the monitoring results and verification of actual cable operating conditions for compliance with regulatory requirements, a technical report is developed. Based on this technical report:

- list of hot spots is made (revised);
- list of cable surveillance specimens is made (revised);
- list of representative cables for technical condition inspection is made;
- compensatory measures to decrease the effect of EIFs on cables are taken (if necessary);
- further cable lifetime is specified.

Cable TCA at NPPs is performed in compliance with requirements of appropriate working programs. The development of new programs in compliance with the established procedure through amendments to CAMP [12] is planned for the types of cables requiring initial TCA.

Cable specimens are placed at NPPs in compliance with requirements of the working program for the preparation and placement of cable surveillance specimens.

The measurement results obtained in the monitoring of operating conditions are recorded in the logbook for monitoring of operating conditions and in the electronic database according to the procedure accepted at NPP.

The results obtained in monitoring of operating conditions are considered in the development and implementation of activities at NPPs aimed at mitigation of cable ageing effects at power units and determination of their long-term operation period.

Changes in the average annual temperature determined in regular monitoring are assessed during the entire long-term operation. The assessment of changes in the average annual temperature is performed no more than every two years (in one scheduled outage). If the average annual temperature around the cables increases by more than 2°C since the last LTO activities, the long-term operation period for cables needs to be revised. If the specified cable long-term operation period is two and more years shorter than the previously established one, an additional decision on LTO should be drawn up.

The working programs for monitoring of cable operating conditions also include information on monitoring periodicity.

Instead of reports on cable laboratory studies, a package of documents on adaptation of TCA results at other VVER-1000 units of Ukrainian NPPs is drawn up when the adaptation approach for LTO is used in compliance with MT-D.0.03.530-11 [119].

As mentioned in section 03.1.2 of this Report, the development of TCA methods based on the understanding of ageing for insulation materials subjected to the action of several factors is the key component of cable ageing management.

Based on the main ageing management principles, methods for technical condition assessment of cables are divided into:

- methods for assessment of cables in operating conditions;
- methods for assessment of cables in laboratory conditions.

Methods for assessment of cables in operating conditions:

- ensure NDI of technical condition;
- are quite simple;
- allow trending of insulation parameters in the operating process.

Diagnostic methods are selected to ensure registration of changes in insulation properties of cables in the ageing process in operating conditions. The efficiency of cable diagnostics in the operating conditions is determined by the knowledge of the ageing mechanisms of cable insulating materials, sensitivity of nondestructive methods for inspection of ageing and damage to electrical insulation along the cable route, and scope of tests using modern, proven and compact diagnostic systems and devices.

Unlike the methods of cable TCA in operating conditions that are used at NPP units, methods in laboratory conditions provide for testing and inspection of samples and isolation materials for cables selected from NPP units, in particular, accelerated thermal and radiation ageing.

General requirements for tests:

– testing conservatism (testing parameters and assessment criteria are accepted considering margins that lead to the most unfavorable results);
– testing focuses on the ageing mechanism that dominates in operating conditions;

Tests are performed in the following sequence:

– inspection of technical condition of cable samples in initial state;
– accelerated thermal and radiation (for containment cables) ageing of cable samples, which is equivalent to cable ageing in normal operation during the needed number of future years of operation;
– inspection of technical condition of cable samples after accelerated thermal and radiation ageing.

For inspection in laboratory conditions, the following samples may be provided:

– deposited surveillance specimens installed at hot spots at NPPs;
– cable samples aged in the operation process. Cable samples decommissioned and left in cable routes together with working cables and cable sections obtained in recovery and maintenance activities.

The inspection methods are selected according to functional purpose of cables, type of insulation materials and cable design.

Examples of recommendations on use of methods for inspection of technical condition of individual cable types operated at NPPs in operating and laboratory conditions in compliance with CAMP [12] are provided in Table 3.5 and Table 3.6.

Table 3.5 Example of recommendations on use of TCA methods for representative cable types in operating conditions

Group No.	**Cable type**	**Core insulation type**	**Methods**							
			Assessment of cable external condition	**Assessment of cable insulation resistivity**	**Assessment of magnetic loss tangent*)**	**Assessment of cable core electrical capacity*)**	**Assessment of level of partial discharges**	**Assessment of recoverable voltage parameters*)**	**Assessment of relaxation current parameters**	**Assessment of PVC cable sheath hardness**
1.	1 kV AAShv	Paper -oil	+	+	-	-	-	-	-	+

Group No.	Cable type	Core insulation type	Methods							
			Assessment of cable external condition	Assessment of cable insulation resistivity	Assessment of magnetic loss tangent*)	Assessment of cable core electrical capacity*)	Assessment of level of partial discharges	Assessment of recoverable voltage parameters*)	Assessment of relaxation current parameters	Assessment of PVC cable sheath hardness
2.	6 kV AAShv	Paper -oil	+	+	-	-	+	-	-	+
3.	KVVG, KVVGE, KVVGng, KVVGEng	PVC	+	+	-	-	-	-	-	+ KVVG, KVVGng,

Table 3.6 Example of recommendations on use of TCA methods for representative cable types in operating conditions

Group No.	Cable type	Core insulation type	Methods											
			Assessment of insulation resistivity				Assessment of core electrical capacity	Assessment of partial capacitance parameters			Assessment of recoverable voltage	Assessment of the number of double bends of cable insulating paper	Assessment of elongation in break of insulation material samples	Check of radiation integrity of protective sheath
			Core	Internal sheath	Core at 90°C (100°C)	Testing for resistance at elevated temperature		Dielectric loss tangent	Correlation factor	Electrical capacity between cores of pairs (core-shield)				
1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
	1 kV AAShv	Paper-oil	+	–	–	–	–	–	–	–	–	+	+	–
	6 kV AAShV	Paper-oil	+	–	–	–	+	–	–	–	+	+	+	–
	KVVG KVVGE KVVGng KVVGEng	PVC	+	–	–	–	–	+	+ KVVG KVVGng	–	–	–	+	+

When LTO is performed by adaptation in compliance with MT-D.0.03.530-11 [119], instead of reports on laboratory analysis of cables, a package of documents on adaptation of TCA results at other VVER-1000 NPP units is prepared.

03.1.4 Preventive and remedial actions for electrical cables

To improve and increase efficiency of the ageing management procedure for cables, the following measures are taken:

– regular monitoring (on a permanent basis) of operating conditions for extended types of cables;
– TCA of cables with implementation of new inspection methods and instruments;
– decrease of EIF effects;
– location (disposal) of surveillance specimens (if necessary);
– measures for improvement of maintenance procedures (technical inspection, detection and elimination of defects etc.).

Based on the main ageing effects (thermal degradation of insulation materials, thermal degradation of insulation materials, loss of insulating properties under the action of electrical field and humidity), NPPs implement measures to decrease ageing-caused degradation rate:

– decrease in environmental temperature in rooms with cables (change in ventilation conditions, change in cable route);
– decrease in current load on power cables (use of cable with large cross-section, connection of parallel cable).

In the framework of equipment qualification in compliance with NP 306.2.141-2008 [5], STP 0.03.050-2009 [32], and PM-D.0.03.476-09 [34] and respective plant programs, assessment of cable qualification for harsh environments in the long-term operation period was carried out by testing or a combination of testing and analysis (including testing of specimens).

The methodologies for qualification for harsh environments developed at NPPs apply to power and control cables that ensure operation of process, electrical and I&C systems which belong to systems important to safety according to NP 306.2.141-2008 [5] and perform safety functions such as:

– safe reactor shutdown and keeping it in this state for required time;
– removal of residual heat from the reactor core during required time;
– limitation of accident consequences by confinement of radioactive releases within established boundaries (for elements of the containment safety system).

Qualification requirements for cables are parameters of the most harsh environments that occur in one of the initiating events in rooms with cables. These parameters are used in qualification of cables operated at NPPs.

Qualification requirements are based on technical reports on development of lists of initiating events for NPPs. As an example, the document “List of Initiating Events Leading to Harsh Environments for Components and Structures of Zaporizhzhya NPP Unit 3. No. 03.MR.00.PR.39” contains a list of initiating events for ZNPP unit 3.

Cables are qualified by testing in the sequence that allows justification of cable resistance to harsh environments considering ageing.

The standard sequence of tests includes:

– visual inspection od cables;
– initial functional tests (measurement of electrical parameters, mechanical tests of samples, increased voltage tests);
– accelerated thermal and radiation (for containment cables) ageing;
– testing in modeled accident and post-accident conditions;
– final functional tests.

03.2 Outcomes of experience and reviews of ageing management for electrical cables

In the framework of measures associated with replacement of equipment in instrumentation and control systems, control and power cables were replaced. As a result, some control cables (for example, of KMPEVE, KPoSG, KPOESV types) are no longer operated in rooms where harsh environments may occur.

The assessment of cable qualification for harsh environments did not reveal ageing effects that deteriorated qualification characteristics.

Efficiency of cable AMPs is periodically assessed in compliance with NP 306.2.210-2017 [7] and SOU NAEK 141:2017 [75].

During these activities and their analysis, editorial changes were introduced into the Cable Ageing Management Program for Nuclear Power Plants (CAMP [12]). The monitoring procedure was specified (procedure for primary and permanent monitoring was specified). There were no substantial changes or reasons for them.

03.3 Licensee's findings and conclusions on ageing management of electrical cables

Based on activities related to cables, the following conclusions can be made:

– respective documents have been developed to govern AM and LTO of cables;

– working programs for technical condition inspection for single-type cables have been developed at NPPs for all cable types subject to ageing management;

– operating conditions of cables at power units have been monitored in all cable rooms, permanent monitoring is conducted on a permanent basis only at hot spots identified in primary monitoring. Primary monitoring is conducted at individual power units;

– cables have been identified at power units, lists of representative cables for inspection have been made;

– representative cables have been analyzed in laboratory and operating conditions. Inspection findings for cables used in the containment are mainly positive. Some cables that show unsatisfactory mechanical and capacity characteristics of insulation in laboratory tests after accelerated thermal and radiation ageing are replaced;

– in replacement of cables in rooms with where hot spots are revealed, surveillance specimens are placed (deposited);

– database on cables is kept for information support of ageing management processes. The database is a model of the URDB "Automated Database on Cable Operation" intended for information support of ageing management activities for power unit cables, in particular:

- analysis of design, operational and maintenance documentation;
- preparation of lists of cable for their technical condition assessment;
- analysis of monitoring of cable operating conditions and technical condition;
- reporting.

NPPs also implemented and efficiently operate the Automated Ageing Management System for Power Unit Components (AAMS), which is a separate software application integrated with the lists, directories and classifiers of the Ukrainian equipment reliability database.

The main AAMS objective is to ensure efficiency of AM and LTO measures for NPP components and spread positive experience in these areas at the industry level.

Ageing management measures on cables of NPP units are carried out in compliance with current regulatory and working documents. The main scope of these activities is carried out within regular operations and technical condition assessment and lifetime reassignment. The results are finalized as technical reports and decisions to be agreed with SNRIU.

Introduction of cable ageing management at NPPs allows timely response to changes in cable operating conditions (which is one of the important parameters in determination of the residual life) and optimum planning of LTO.

03.4 Regulator's assessment of ageing management of electrical cables and conclusions

The following can be stated after analysis of information on ageing management cables provided in this section: ageing management of cables at NPP units is paid proper attention both during the design-basis life and in the LTO period.

The main objective of ageing management is to ensure the required safety level through technically and economically feasible, proven measures intended for timely detection of deterioration in properties of cable insulation materials to predict their further operation period. Besides inspection of representative cables in laboratory and operating conditions, it is also important to monitor cable during operation. Introduction of cable ageing management at NPPs allows timely response to changes in cable operating conditions and optimum planning of LTO.

Ageing management measures on cables are implemented in compliance with current regulations and working documents developed at NPPs. The main scope of these activities is carried out during regular operations and in TCA and LTO. Cables were qualified for harsh environments. The results are finalized as technical reports and decisions, which , in accordance with NP 306.2.210-2017 [7], are submitted by the operator to SNRIU for agreement for permanent oversight and monitoring of AMP implementation and particularly CAMP [12] implementation at NPP units. The information provided in the reports is assessed and checked during scheduled inspections, including aspects related to cable ageing management.

The implementation of CAMP [12] along with other ageing management programs for NPP equipment is a necessary condition for power unit LTO. They are periodically revised and improved to incorporate national and international experience, practices and technical capabilities for ageing management measures. As an example, Energoatom developed "Notification No. 03-43-17-I on Amendment of Cable Ageing Management Program for NPPs PM-T.0.08.121-14 (Notification No. 2)" in 2017, which has been agreed with SNRIU.

The evaluation of TCA and cable qualification for harsh environment is mainly positive. Separate cables that showed unsatisfactory results in tests are replaced: for example, cables KMPEVE, KPoSG and KPoESV that are laid in rooms with harsh environments. In addition, in the framework of measures related to replacement of equipment in instrumentation and control systems and electrical equipment, control and power cables have been or are going to be replaced with fire retardant ones and those in automated firefighting systems and emergency power supply systems with fireproof ones. As an example, technical decisions agreed with SNRIU are provided below:

– in the framework of Technical Decision No. TR.3.0009.1728 of 18 March 2016 "On Replacement of 220 V Direct Current Board in the SUNPP Unit 3 Emergency and Reliable Power Supply Systems", it is also envisaged to replace power and control cables by fireproof ones to improve reliability of the SUNPP unit 3 emergency and reliable power supply systems: AVVGng by VVGngd-FR and KVVG by KVVGngd-FR. This requirement is provided in NP 306.2.205-2016 "Requirements for Power Supply Systems Important to Safety of Nuclear Power Plants" (para. 3, Chapter 5, Section V);

– in the framework of Technical Decision No. 03.ETs.UJ.TR.2410 "On Installation of Equipment of the Automated Fire Alarm at ZnPP-3 Reactor Compartment Safety Systems 1, 2, 3 and 3SDGS 1, 2, 3", KVVGE control cables are to be replaced by KVVGngd-FRLS cables (fireproof, retardant) in compliance with para. 5.3.2.2 of VBN V.1.1-034-03.307-2003 (NAPB 03.005-2002) "Fire Safety Standards for Designs of Nuclear Power Plants with Water-Cooled Water-Moderated Power Reactors".

Efficiency of AM is periodically assessed in compliance with NP 306.2.210-2017 [7] and SOU NAEK 141: 2017 [75].

04 CONCEALED PIPEWORK

04.1 Description of ageing management for concealed pipework

04.1.1 Scope of ageing management for concealed pipework

At Ukrainian NPPs, concealed piping include underground piping of the essential service water system (ESWS).

At Ukrainian NPPs, underground piping does not contain radioactive drain water, is not used for fuel pumping, but only contain essential service water. Other types of concealed piping are not used at Ukrainian NPPs.

ESWS piping is laid underground, along reactor buildings of power units at a depth from 2.0 m to 6.0 m. The base and backfill of piping are represented by small loose damp sand with different density: from loose to medium density.

The detailed description of this system piping, as well as generalized list of ESWS piping is presented in Annex A.

All underground piping of the essential service water system is included to unit-specific ageing management programs. The criterion to select it for inclusion in AMP is its lying underground and limited availability. Taking into account the same operation conditions, the criterion to group piping is its nominal diameter (Ø 2440, Ø 1620, and Ø 820).

The following procedures are used to define the ageing mechanisms of concealed piping:

- technical examination;
- assessment of current technical condition;
- monitoring of technical condition.

ESWS piping is operated under the same pressure and temperature, and the places of the most stress concentration are located at tee-joints, branch conductors, adapters of diameter and in the places of stop valve cutting-in. A special attention is paid to these components.

According to TCA results, failures were not registered at Ukrainian NPPs during operation of ESWS piping.

The examples of underground piping location are presented below in Figure 4.1 – Figure 4.3.

Figure 4.1 ESWS piping

Figure 4.2 ESWS piping near well

Figure 4.3 ESWS piping (visualization after excavation)

04.1.2 Ageing management for concealed pipework

TCA activities are performed to define/specify the ageing mechanisms of the concealed pipework. The basis for TCA for inspected piping is comparative analysis of technical condition actual parameters and design requirements.

The main collectors are laid along the reactor buildings of power units under conditions of dense development full of communications and in close vicinity to existing highways, piping racks, railways to transport fuel, cargoes and other underground and aboveground communications and structures. This aspect is taken into account when determining location of piping opening (test pitting) taking into account process diagrams of its tracing and availability of places with the most stress concentration.

Piping is divided according to observation groups. In each of the observation groups, piping is operated under the same pressure and temperature. Particular attention is paid to the places with the most stress concentration. Control elements (CEs) subjected to observation are determined in the observation groups. Observation results for CEs are applied to the entire observation group.

Piping of the “observation group” operated under worse ground conditions and having the most operation cycles is subject to inspection.

The following is applied to inspect and define technical condition of concealed piping:

– visual examination (conducted to define its compliance with the requirements of technical documents with the application, if necessary lighting and optical devices;

– nondestructive metal inspection (visual inspection, ultrasonic thickness measurement, ultrasonic inspection, and in questionable places liquid penetrant inspection and magnetic particle inspection)

– defining mechanical properties of metal by hardness (conducted to define compliance of the actual values of mechanical properties with the values established in the regulatory and technical documents);

– inspection of anticorrosion piping coating condition;

– contactless diagnostics of piping (method of contactless magnetometric diagnostics, method of acoustic tomography). These methods reveal loss of piping integrity and do not require direct access to the outer piping surface. Diagnostics is carried out from ground surface above the piping. 100% of piping from all observation groups are inspectable;

– assessment of corrosion hazard of external and internal media (specific electrical resistance of piping is measured, elementary soil composition is determined, hazard of ground current is assessed).

Resulting from TCA activities, current technical condition of piping is determined and administrative and technical measures are developed regarding piping ageing management due to which technical condition of piping should be maintained within the required limits by monitoring and slowing down:

– ageing of materials: worsening of their properties;

– ageing of structures that increases load on materials.

Ageing management measures may include:

– correction of monitoring and maintenance;

– correction of external loads;

– correction of piping structures;

– improvement and development of inspection means.

To provide additional monitoring of underground piping, AMP includes measures for annual piping contactless diagnostics with the finalization of the reporting documents. In case of identifying doubtful areas according to the contactless diagnostics, activities are planned and performed on test pitting of these areas for NDI and other measures to identify areas susceptible to degradation and to identify appropriate measures to restrain degradation.

The ageing mechanisms of piping metal and controlled effects of their development are presented in Table 4.1.

Table 4.1 Ageing mechanisms and controlled effects

Controlled effect of metal ageing	**Metal ageing mechanisms**			
	Low- and high-cycle fatigue	**General corrosion**	**Local corrosion**	**Flow accelerated corrosion**
Thinning	-	×	-	×
Change of mechanical properties	×	-	-	-
Pitting corrosion	-	-	×	-
Erosion (local thinning)	-	×	-	-
Electrochemical corrosion	-	-	-	×

The determining parameters of piping metal condition are specified in Table 4.2.

Table 4.2 The determining parameters of piping metal condition

No.	**Controlled effect of metal ageing**	**Determining parameters of piping metal condition**
1	General thinning	Wall thickness
2	Change of mechanical properties	Safety margin, yield point, relative elongation, relative reduction
3	Crumbling	Number of defects per area unit, depth of defects
4	Erosion (local thinning)	Damage area, wall thickness
5	Electrochemical corrosion	Cracks

The list of studied ageing mechanisms that affect piping service life specified in Table 4.3 is developed based on analyzing operation conditions complemented by the results of studying piping degradation mechanisms with the assessment of the need to develop additional measures to control and restrain degradation.

Table 4.3 List of studied ageing mechanisms that affect piping service life

No.	**Degradation mechanisms**	**Ageing effect**	**Technical condition parameters (TCP)**	**TCP inspection methods**	**Periodicity**	**Assessment of TCP impact on component service life**
1	Low- and high-cycle fatigue	Change of metal mechanical properties	Safety margin, yield point, relative elongation and reduction	MHT by individual inspection program	During TCA for LTO	Main ageing mechanism
2	General corrosion	Thinning	Wall thickness		During TCA for LTO	Impacts

No.	Degradation mechanisms	Ageing effect	Technical condition parameters (TCP)	TCP inspection methods	Periodicity	Assessment of TCP impact on component service life
3	Local corrosion	Crumbling	Wall thickness	UTM by individual program	During TCA for LTO	Impacts
4	Corrosive wear	Thinning	Wall thickness		During TCA for LTO	Main ageing mechanism
5	Mechanical damage	Loss of local resistance, cracks in the main metal and weld joints	Condition of structures	Visual inspection	During TCA for LTO	Depending on condition of inspected piping

ESWS piping is included into unit-specific ageing management lists and correspondingly it is covered by the requirements of Standard AMP [11] due to which:

– technical condition during LTO is assessed;
– ageing management measures are implemented;
– technical examination and monitoring according to operating procedures is conducted.

Technical condition is assessed according to unit-specific working programs developed based on Standard AMP requirements [11], for example:

– Working Program for Technical Condition Assessment and Determination of Residual Life/Lifetime Extension of Underground Piping at ZNPP Unit 1 and Common-Plant Facilities. 01.GTs.VF.VG.PM.08 [43];

– Working Program for Technical Condition Assessment and Life Extension of ESWS Underground Piping at ZNPP Unit 2. 02.GTs.VF/VG.PM.19-13 3OH [44].

Similar programs are developed for RNPP, KhNPP and SUNPP units.

It should be noted separately that over the past 5 years, the methods for contactless diagnostics (method of contactless magnetometric diagnostics, acoustic tomography) according to the requirements of [53] - [55] is widely used during diagnostics of underground piping and prior to their implementation a large-scale complex of activities was conducted to verify these methods and determine their ability to detect and classify defects with the necessary precision.

In addition, the issue of strength calculation of the concealed (piping deepen into the ground) piping arose separately. The specificity of calculation is determined by conditions of its operation and additional loads caused by ground impact, namely:

– internal pressure;
– vertical ground pressure;
– alternating pressure of ground surface;
– shock load of surface;
– buoyancy force;
– thermal expansion;
– piping displacement relative to ground;
– piping moving and bending;
– lowering of terrestrial rocks;

– seismic impacts.

Separate “Methodology for Strength Calculation of Underground Piping of the Zaporizhzhya NPP Essential Service Water System Considering Seismic Impacts” was developed for strength calculation of concealed (underground) piping. Current international experience described in [57] - [62] was considered in developing this methodology. This methodology belongs to TLAA guideline for underground piping and according to calculation results, LTO period is established and additional measures on ageing management are assigned (if necessary).

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

The condition of NPP underground piping is monitored according to the requirements of “Standard Procedure for Technical Examination of NPP ESWS Underground Piping. IN-T.0.03.325-13” [63], which specifies general assessment rules of technical examination and monitoring of NPP underground piping.

Standard Procedure IN-T.0.03.325-13 [63] is used as the basis for development of unit-specific (working procedures) that considers site design features.

The unit-specific procedures contain:

– diagrams and lists of ESWS underground piping specifying access points, control elements and NDI areas;

– list of piping sections to be drained for internal examination;

– coordinates of sample cutting zones for destructive control;

– scope and sequence of the preparatory activities and individual operations before technical examination of ESWS underground piping;

– scope and sequence of conducted activities and inspections during technical examination of ESWS underground piping;

– description (reference to relevant documents) of methodologies used for technical diagnostics of the linear part of piping and metal condition inspection, in particular measurement errors and determination of values;

– criteria of piping proper condition;

– instructions on the ways to process the results and reporting documentation;

– requirements of safety rules;

– instructions on administrative issues of technical examination;

– need for personnel of NPP subdivisions and contracting organizations;

– information on additional activities to prepare for technical examination, scope and methods for restoration and control of equipment initial condition (dismantling, detachment, cutting, welding, etc.);

– scope and sequence on performed activities on bringing ESWS underground piping to operable condition after technical examination;

– need to perform activities (depending on their nature) according to orders and instructions (of check-lists);

– safety requirements for excavation activities during management of electrical devices, equipment and tools, fire safety requirements, labor protection requirements (in particular, when examining the inner surface of piping).

Figure 4.4 presents a standard diagram for technical examination of ESWS underground piping.

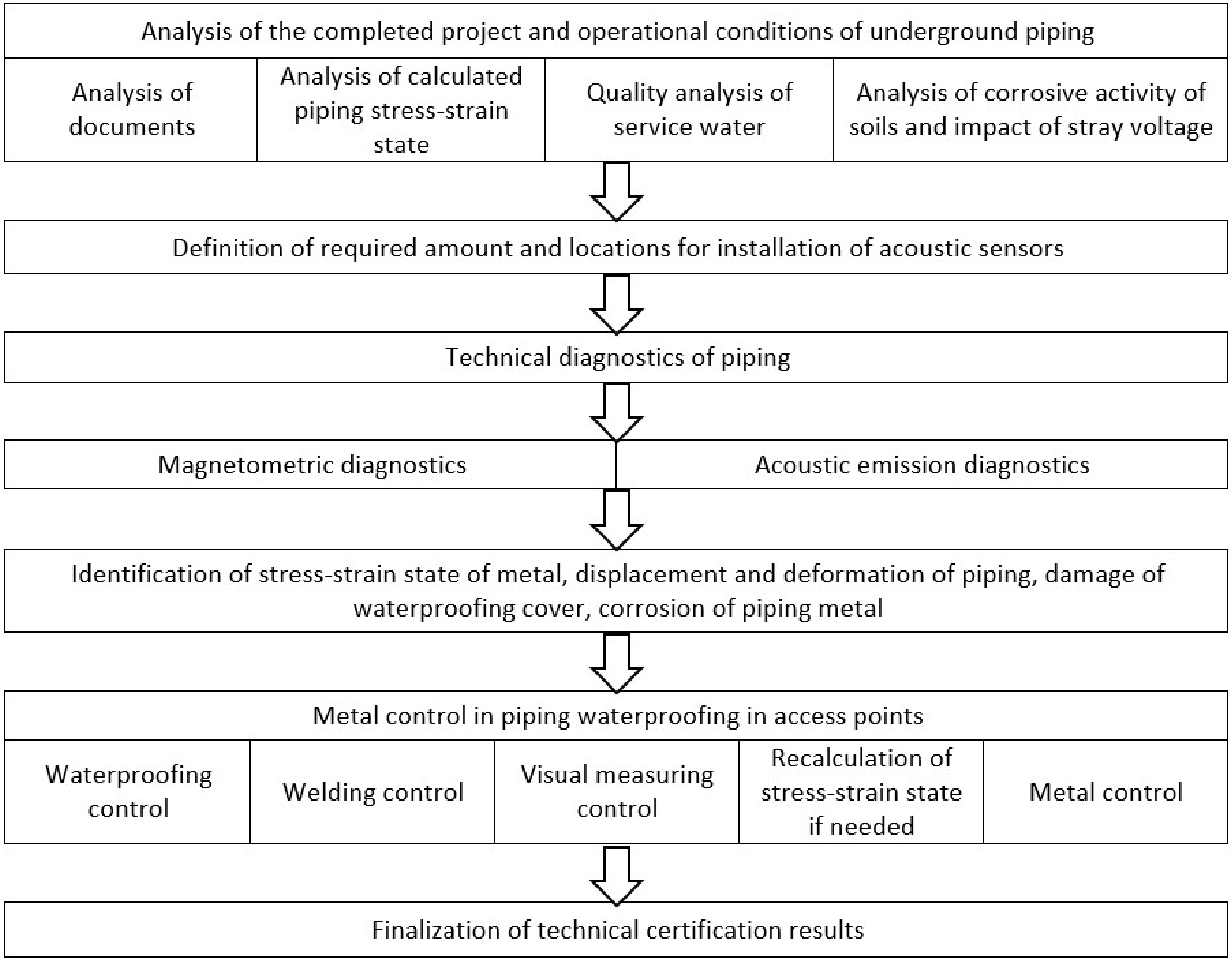

Figure 4.4 standard diagram for technical examination of ESWS underground piping

Monitoring of technical condition of NPP underground piping consists of the following main components:

– system of technical condition examination, monitoring and diagnostics of underground piping parameters;
– TCA methods of underground piping;
– technical condition forecasting of underground piping and assessment of its service life;
– ageing management of underground piping.

Since the most of piping is not accessible for external and internal examination, the main monitoring methods are methods of contactless diagnostics for underground piping, namely:

– contactless magnetometric diagnostics method;
– acoustic tomography method.

These methods reveal loss of integrity of piping and do not require direct access to piping external surface. Diagnostics is carried out from ground surface above the piping. 100% of piping from all observation groups are inspectable.

Contactless magnetometric diagnostics implies using portable devices with sensors on magnetoresistive structures, which provide high selectivity and interference resistance for activities in the industrial area of an enterprise with the use of gradiometry method to exclude the impact of man-made contamination of piping route and with the possibility to monitor magnetograms during examination, as well as piping finders, which allow determining the axis and depth of piping.

The objectives of acoustic tomography are to:

– determine the places of local defects;
– classify defects by hazard degree.

The acoustic tomography method determines the places of local defects resulting from:

– thinning of pipe wall due to external and internal corrosion;
– leak.

Combined use of the contactless magnetometric and acoustic diagnostics may increase the probability to detect areas with various defects of piping metal as a whole, which is extremely important for determining technical condition of underground piping sections that are not available for inspection by nondestructive methods.

Forecasting of technical condition of underground piping allows determining permissible period of long-term operation.

Ageing management of underground piping is determined by a number of administrative and technical measures to restrain degradation of components and retain operational characteristics during piping LTO.

04.1.4 Preventive and remedial actions for concealed pipework

Preventive and remedial measures for concealed pipework are established based on TCA activities, technical examination and monitoring individually for each power unit.

Thus, for example, according to TCA results for ZNPP unit 1-3 it was determined that worsening of ZNPP ESWS underground piping condition is not significant. Repair measures are not required for this equipment that is confirmed by TCA with the activities on piping opening, as well as performed strength and seismic resistance calculation. Piping operation conditions meet the design values and are adequate to maintain an appropriate safety level.

When developing reports on TCA of ESWS underground piping, measures were selected to reduce degradation rate of piping material, for example:

– restore anticorrosion coating of piping in accessible places and places of NDI;
– perform periodic diagnostics of ESWS piping sections with low-level anomalies to detect changes in its condition by contactless methods;
– exclude piping operation under beyond design modes.

The list of ageing management measures established for ZNPP units 1, 2 according to TCA for LTO is presented below as an example. This list of measures is typical for all Ukrainian NPP sites.

Ageing management measures for ZNPP-1:

1. Perform intermediate TCA (each 10 years during periodic review).

2. Operate piping according to operating instructions and technical specifications for safe operation of ZNPP-1.

3. Restore protective anticorrosion coating on control elements and in accessible places.

Ageing management measures for ZNPP-2:

1. Perform intermediate TCA (each 10 years during periodic review).

2. Operate piping according to operating instructions and technical specifications for safe operation of ZNPP-2.

3. Restore protective anticorrosion coating on control elements and in accessible places.

4. Perform contactless diagnostics of certain piping sections with detected “anomalies”.

04.2 Outcomes of experience and review of ageing management for concealed pipework

The activities on TCA of ESWS underground piping show that the following may be referred to the main ageing management measures:

– restore anticorrosion coating of piping in accessible places and places of NDI;

– perform diagnostics of ESWS piping sections with low-level anomalies to detect changes in its condition by contactless methods.

These measures are included into unit-specific AMP and their adequacy is confirmed by operating experience and practice.

04.3 Licensee’s findings and conclusions on ageing management of concealed pipework

Monitoring of technical condition of NPP underground piping is performed on a regular basis in compliance with the operator’s technical regulations and envisages to:

– determine technical condition, monitor and diagnose underground piping parameters;

– apply current methods of TCA for underground piping;

– forecast technical condition of underground piping and assess its service life;

– provide ageing management of underground piping.

Since the most of piping is not accessible for external and internal examination, the main monitoring methods are methods of contactless diagnostics for underground piping, namely:

– contactless magnetometric diagnostics method;

– acoustic tomography method.

These methods reveal loss of integrity of piping and do not require direct access to piping external surface. Diagnostics is carried out from ground surface above the piping. 100% of piping from all observation groups is inspectable.

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

To provide activities on ageing management of concealed piping and assess its current technical condition the operator developed Standard Procedure IN-T.0.03.325-13 [63], which is used as the basis for development of working programs, such as [43]-[44]. In addition, taking into account peculiarities of concealed piping operation, namely its underground location and special loads caused by this location, the operator developed the "Methodology for Strength Calculation of Underground Piping of the Zaporizhzhya NPP Essential Service Water System Considering Seismic Impacts" [56] and agreed it with SNRIU. In developing this methodology, international experience and approaches to strength calculation of concealed piping were taken into account, namely [57]-[62]. Strength calculation of underground piping in fact is one of TLAA types according to whose results LTO period is established and ageing management measures are determined.

Preventive and remedial measures for concealed pipework are established based on TCA activities, technical examination and monitoring individually for each power unit. TCA activities performed at Ukrainian NPPs revealed insignificant worsening of underground piping condition.

The main ageing management measures of concealed piping are:

– restore anticorrosion coating of piping in accessible places and places of NDI;

– perform periodic diagnostics of ESWS piping sections with low-level anomalies to detect changes in its condition by contactless methods.

The activities performed by the operator regarding ageing management of concealed pipework meet the regulatory requirements at the same time taking into account that the contactless diagnostics methods are constantly improved, in particular in terms of improving accuracy of determining parameters, the SNRIU recommended the operator to continue the following measures on a permanent basis:

– analyze current research and development (methods, methodologies, equipment), whose purpose is to perform adequate assessment (diagnostics) of current technical condition for piping, which is deepened in the ground and is not easily accessible for examination;

– analyze current international experience in assessing the current technical condition of piping that is deepened in the ground and is not easily accessible for examination

– involve specialized organizations having experience in designing, operating and repairing similar piping in other industries, etc.

05 REACTOR PRESSURE VESSEL

05.1 Description of ageing management for RPV

The information on design and main design components of reactor pressure vessels (RPV) with VVER-1000/320, 302, 338 and VVER-440/213 is presented in Annexes A and B.

05.1.1 Scope of ageing management for RPV

Reactor vessel closure head and reactor vessel are a safety barrier on the path of radioactivity spread. Due to this, the following components of each RPV type are included to AMP:

- actually RPV:
  - reactor vessel flange with threaded seats;
  - lower ring of nozzle area;
  - upper ring of nozzle area;
  - MCP, ECCS and instrumentation nozzles;
  - separating ring;
  - support cylindrical ring;
  - lower cylindrical ring;
  - upper cylindrical ring;
  - elliptic bottom;
  - brackets;
  - welds.
- reactor vessel closure head:
  - flange, ellipsoid, welds;
  - CPS, temperature monitoring, energy release monitoring and air vent nozzles including fastening in the closure head, seal assemblies and flange joints.

The procedures described below are used to define ageing mechanisms of the reactor pressure vessel and reactor vessel closure head.

05.1.1.1 Nondestructive inspection of base (clad) metal and welds

The requirements for NDI of the reactor pressure vessel during operation are established in document “PM-T.0.03.061-13. Standard Program for Periodic Inspection of Base Metal, Welds and Claddings of Equipment and Piping of VVER-1000 Nuclear Power Plants (TPPK-13)” (Standard Program PM-T.0.03.061-13) [64]. To inspect metal condition of the reactor pressure vessel and main reactor flange in compliance with the Standard Program PM-T.0.03.061-13 [64], the following is used:

- visual inspection;
- liquid penetrant inspection;
- magnetic particle inspection;
- ultrasonic inspection;
- eddy current inspection of anticorrosive cladding.

Standard Program PM-T.0.03.061-13 [64] includes the list of areas, scope, methods and periodicity for NDI of material condition of RPV and RVMF components, as well as the list of documents regulating quality assessment standards. The requirements of Standard

Program PM-T.0.03.061-13 [64] are the main for developing working programs of NDI at power unit.

05.1.1.2 RPV metal control using surveillance specimens

Control of exposed materials, RPV components is conducted in compliance with the "Standard Program for Monitoring of VVER-1000 RPV Metal Properties Using Surveillance Specimens. PM-T.0.03.120-08" [65], agreed by SNRIU.

Stages and periods for testing of the surveillance specimens are determined by individual technical solutions for each reactor pressure vessel.

05.1.1.3 Internal/external inspection of equipment

Within technical examination of the reactor pressure vessel and reactor vessel closure head, examinations (internal and external) are conducted at least once per four years in compliance with the requirements of PNAE G-7-008-89 [8].

05.1.1.4 Control of reactor vessel main flange density

Reactor vessel main flange density is controlled according to pressure occurrence in inter-gasket cavities.

05.1.2 Ageing assessment for RPVs

05.1.2.1 Basic ageing mechanisms and definition of their significance

According to the results of applying the methods and procedures specified above, the following degradation mechanisms were defined for RPV:

- radiation embrittlement (for RPV);
- thermal ageing (for reactor vessel and reactor vessel closure head);
- fatigue (reactor vessel and reactor vessel closure head);
- stress corrosion cracking (for reactor vessel and reactor vessel closure head);
- boron corrosion (for reactor vessel and reactor vessel closure head);
- crushing/mechanical damages (for reactor vessel and reactor vessel closure head).

A detailed description of each ageing mechanism is presented below.

Radiation embrittlement

Radiation embrittlement is the main process that limits lifetime of RPV made of 15Kh2NMFA(-A) grade ferrite steel and welds No. 3, No. 4 (located in core area) made using Sv-10KhGNMAA grade weld cable. For these materials, ductile to brittle transition is typical at a certain temperature. Under the impact of neutron exposure there is a temperature shift of brittle ductile transition to a higher temperature area, which increases the probability of reactor vessel brittle fracture.

Radiation embrittlement does not cause cracks, but it reduces metal resistance to the formation of cracks resulting from fatigue, stress corrosion or during RPV manufacturing.

The main damaging factor: neutron flux.

Thermal ageing

Thermal ageing causes worsening of material mechanical properties that is explained by structural changes in the material under the impact of various loads; especially if the material is used under high temperatures.

The main damaging factor: high temperature.

Fatigue

Metal fatigue starts from local plastic deformation, which causes shear strips and microcracks, one of which may transform into the main crack. Cracks are formed in the places of stress concentration, for example on surface defects, geometric cavities.

The main damaging factors: pressure and temperature change, cyclic load change (for example, tightening of studs, forces and moments from piping to nozzles).

Stress corrosion cracking

There are two forms of stress corrosion cracking: intercrystalline (along grain boundaries) and transcrystalline (inside the grain) corrosion.

Austenitic stainless steels (even made, for example cladding, jackets of RPV components) are extremely susceptible to stress corrosion cracking in the chloride medium; high temperature and the presence of oxygen increases stress corrosion cracking in stainless steels.

Stress corrosion cracking occurred in such medium is called transcrystalline corrosion. It is known that combined presence of chlorine and oxygen in solution contributes to transcrystalline corrosion. Even under high chlorine content, corrosion damage under stress does not occur in the material, if there is no oxygen in water. Moreover, with an increase of oxygen concentration in the medium, the boundary value of chlorine ion concentration for transcrystalline corrosion is reduced.

Consequently, the radical measure to prevent transcrystalline cracking is strict regulation of primary coolant quality by oxygen and halogen content.

The main damaging factors: corrosion medium (primary coolant), temperature and mechanical stress.

Boron corrosion

Boric acid solution may damage carbohydrate and low-alloy steels. Boric acid is a relatively weak acid, but when it is on hot surface, water evaporates and concentrated boric acid solution remains, which ultimately turns into crystals

The main damaging factor: corrosion medium (primary coolant).

Crushing/mechanical damages

During operation on RVMF sealing surfaces, defects may occur due to force impact of the gaskets. Mechanical damage of threaded joints may occur if studs are tightened not correctly.

The main damaging factors: mechanical loads.

Controlled parameters and acceptance criteria

The controlled parameters and acceptance criteria are presented in Table 5.1.

Table 5.1 Parameters and acceptance criteria

No.	Parameter	Description
1	Critical temperature of RPV metal brittleness	During the whole RPV lifetime, the actual value of critical temperature of RPV metal brittleness should not exceed the values of the maximum allowable critical temperature of brittleness, which is defined by the calculation of brittle fracture resistance.
2	Mechanical properties of exposed RPV materials	The acceptance criteria for strength characteristics obtained within the surveillance specimen program: the actual strength value and yield stress ratio should not be equal to one. Acceptance criteria for plasticity characteristics obtained within the surveillance specimen program: ensuring RPV safe operation taking into account degradation of plasticity characteristics under radiation impact during operation.
3	Mechanical properties of not exposed RPV materials	The actual values of RPV and RVMF material mechanical properties obtained by measurements (for example, hardness measurement) should meet the requirements of regulatory and technical documentation.
4	Condition of base (clad) metal and welds	The actual condition of base (clad) metal and welds should meet the requirements of documents regulating quality assessment standards specified in the Standard Programs PM-T.0.03.061-13 [64] (for VVER-1000) and AIEU-10-09 [31] (for VVER-440).
5	Reactor loading cycles	Accumulated fatigue damageability of RPV components does not exceed the allowable value established by the standards of PNAE G-7-002-87 [9]. The actual number of reactor loading cycles should not exceed the predictive one specified in the table of the technical specifications for safe operation.
6	Boron corrosion caused by primary coolant leak	Primary coolant leaks are not allowed.

The analysis, control and degradation mitigation measures for RPV of VVER-1000 / B-320, 302, 338 i VVER-440/B-213 are presented in Table 5.2 and Table 5.3.

Table 5.2 RPV ageing assessment

No	Component/ area	Affected function	Degradation mechanisms/ ageing effects	Preventive measures	Identification of ageing effects	Monitoring and analysis of trends for ageing effects	Degradation mitigation	Acceptance criteria	Corrective measures
1	Welds No. 3 and No. 4; core cylindrical rings, support ring	Reactor vessel integrity	Radiation embrittlement and strengthening/ change of material properties	Use of fuel loading with a reduced neutron leakage in the reactor	Not applied	Inspection of RPV metal properties with the modernization of 4-6 sets of surveillance specimens (testing of exposed sets of specimens). Monitoring of neutron fluence on RPV components	Not applied	Actual T_k value during the entire operation period should not exceed the maximum allowable values (T_k^a). The actual safety margin and yield point ratio should be over 1	BFR calculations (for example with a smaller postulated crack). Annealing implementation
2	Entire RPV and RVMF	Reactor vessel integrity	Thermal ageing/ change of mechanical properties	Not applied	Not applied	Inspection of metal properties by testing of temperature sets of surveillance specimens. Measuring metal hardness of RPV flange and RVMF fastening components to determine properties	Not applied	Actual T_k value during the entire operation period should not exceed the maximum allowable values (T_k^a). Compliance of the actual values of mechanical properties with the requirements of regulatory and technical documentation	BFR calculations (for example with a smaller postulated crack). Strength calculations taking into account the actual values of mechanical properties in compliance with the requirements of current regulations

No	Component/ area	Affected function	Degradation mechanisms/ ageing effects	Preventive measures	Identification of ageing effects	Monitoring and analysis of trends for ageing effects	Degradation mitigation	Acceptance criteria	Corrective measures
3	RPV internal surface (RPV cladding)	Protection of reactor vessel base metal against aggressive primary coolant medium	Corrosion, stress corrosion cracking/ defects	Monitoring of water chemistry parameters in the primary system	Periodic NDI according to standard program TPPK-2013 for VVER-1000 (AIEU-10-09 for VVER-440).	Monitoring of water chemistry parameters in the primary system according to current regulations. Periodic NDI according to standard program TPPK-2013 for VVER-1000 (AIEU-10-09 for VVER-440).	It is necessary to observe: - optimum value of pH; - oxygen-free regime with a minimum oxygen concentration and a sufficient hydrogen concentration; - minimum concentration of undesired ions of Cl, F, SO_4, etc.	Actual condition of clad metal and welds meets the requirements of documents according to quality assessment standards	Repair or replacement
4	Entire RPV. Components of RVMF seal assembly (studs, washers, nuts)	Reactor vessel integrity/tightness of RVMF flange joint	Fatigue/ defects	Not applied	Periodic NDI according to standard program TPPK-2013 for VVER-1000 (AIEU-10-09 for VVER-440).	Monitoring of the number of reactor loading cycles. Periodic NDI according to the requirements of standard program TPPK-2013 for VVER-1000 (AIEU-10-09 for VVER-440)	Try to exhaust as little as possible number of normal operation, abnormal operation and emergency modes. Hydraulic strength testing under decreased pressure of 207 kgf/cm^2 (for VVER-1000), 195 kgf/cm^2 (for VVER-440) according to technical specifications for safe operation	Accumulated fatigue damageability of RPV and RVMF components does not exceed the allowable value specified in PNAE G-7-002-87 and is equal to 1. The number of reactor loading cycles does not exceed the established predictive one. Actual condition of base (clad) metal and welds meets the requirements of documents according to quality assessment standards	Repair or replacement. Cyclic strength calculation taking into account crack growth, for example according to the VERLIFE methodology

No	Component/ area	Affected function	Degradation mechanisms/ ageing effects	Preventive measures	Identification of ageing effects	Monitoring and analysis of trends for ageing effects	Degradation mitigation	Acceptance criteria	Corrective measures
5	RPV flange: internal surface of cladding in grooves for RVMF gaskets	Tightness of RVMF flange joint	Crushing, mechanical damage/defects	Not applied	Periodic NDI according to standard program TPPK-2013 for VVER-1000 (AIEU-10-09 for VVER-440).	Periodic NDI according to standard program TPPK-2013 for VVER-1000 (AIEU-10-09 for VVER-440).	Assembly of RVMF in strict compliance with design documentation on RPV	No leakage. Actual condition of clad metal meets the requirements of documents according to quality assessment standards	Use of gaskets of a larger cross-section. Repair
6	Threaded seats of RPV flange for studs. RVMF nuts, studs, washers	Tightness of RVMF flange joint. Integrity	Crushing, mechanical damage/thread stripping, defects	Not applied	Periodic NDI according to standard program TPPK-2013 for VVER-1000 (AIEU-10-09 for VVER-440).	Periodic NDI according to standard program TPPK-2013 for VVER-1000 (AIEU-10-09 for VVER-440).	Assembly of RVMF in strict compliance with design documentation on RPV	Actual condition of base metal meets the requirements of documents according to quality assessment standards	Repair or replacement

No	Component/ area	Affected function	Degradation mechanisms/ ageing effects	Preventive measures	Identification of ageing effects	Monitoring and analysis of trends for ageing effects	Degradation mitigation	Acceptance criteria	Corrective measures
7	Threaded seats of RPV flange for studs, RPV flange. RVMF nuts, studs, washers	Integrity	Boron corrosion/ defects	Not applied	Periodic NDI according to standard program TPPK-2013 for VVER-1000 (AIEU-10-09 for VVER-440). Inspection of reactor main flange tightness	Periodic NDI according to standard program TPPK-2013 for VVER-1000 (AIEU-10-09 for VVER-440). Integrity inspection of reactor main flange	Assembly of RVMF in strict compliance with design documentation on RPV	No leakage. Actual condition of base metal meets the requirements of documents according to quality assessment standards	Repair or replacement

Table 5.3 Ageing assessment of reactor closure head assembly

No	Component/ area	Affected function	Degradation mechanisms/ ageing effects	Preventive measures	Identification of ageing effects	Monitoring and analysis of trends for ageing effects	Degradation mitigation	Acceptance criteria	Corrective measures
1	Reactor vessel closure head, RVCH nozzles	Integrity	Thermal ageing/change of mechanical properties	Not applied	Not applied	Inspection of mechanical properties of safety system components (according to hardness measurement)	Not applied	Compliance of the actual values of mechanical properties with the requirements of regulatory and technical documents	Strength calculations taking into account the actual values of mechanical properties
2	Reactor vessel closure head, safety system nozzles including cladding and welds	Protection of reactor closure head base metal and nozzles against aggressive primary coolant medium	Fatigue/defects	Monitoring of cooldown rate under mode “Scheduled cooldown to “cold” condition under a rate not over 30 ^{0}C/h”	Periodic NDI according to standard program TPPK-2013 for VVER-1000 (AIEU-10-09 for VVER-440).	Monitoring of the number of reactor loading cycles. Periodic NDI according to standard program TPPK-2013 for VVER-1000 (AIEU-10-09 for VVER-440).	Try to exhaust as little as possible number of normal operation, abnormal operation and emergency modes. Hydraulic strength testing under decreased pressure of 207 kgf/cm^2 (for VVER-1000), 195 kgf/cm^2 (for VVER-440) according to technical specifications for safe operation	Accumulated fatigue damageability of safety system components does not exceed the allowable value specified in PNAE G-7-002-87 and is equal to 1. The number of reactor loading cycles does not exceed the established predictive one. Actual condition of base (clad) metal and welds meets the requirements of documents according to quality assessment standards specified in standard program TPPK-2013 for VVER-1000 (AIEU-10-09 for VVER-440).	Repair or replacement. Cyclic strength calculations taking into account crack growth, for example according to the VERLIFE methodology.

No	Component/ area	Affected function	Degradation mechanisms/ ageing effects	Preventive measures	Identification of ageing effects	Monitoring and analysis of trends for ageing effects	Degradation mitigation	Acceptance criteria	Corrective measures
3	Internal surface of reactor vessel closure head with safety system nozzles (cladding, nozzle jackets)	Protection of reactor vessel closure head base metal and nozzles against aggressive primary coolant medium	Corrosion, stress corrosion cracking/ defects	Monitoring of water chemistry parameters in the primary system	Periodic NDI according to standard program TPPK-2013 for VVER-1000 (AIEU-10-09 for VVER-440).	Monitoring of water chemistry parameters in the primary system according to current regulatory documents. Periodic NDI according to standard program TPPK-2013 for VVER-1000 (AIEU-10-09 for VVER-440).	It is necessary to observe: - optimum value of pH; - oxygen-free regime with a minimum oxygen concentration and a sufficient hydrogen concentration; - minimum concentration of undesired ions of Cl, F, SO_4, etc.	Actual condition of base (clad) metal and welds meets the requirements of documents according to quality assessment standards presented in standard TPPK-2013 for VVER-1000 (AIEU-10-09 for VVER-440).	Repair or replacement
4	Internal surface of reactor vessel closure head, flange joints of nozzles, spacer ring	Integrity	Boron corrosion/ defects	Not applied	Periodic NDI according to standard program TPPK-2013 for VVER-1000 (AIEU-10-09 for VVER-440). Inspection of inter-gasket cavities for TM, CPS flange leakage	Periodic NDI according to standard program TPPK-2013 for VVER-1000 (AIEU-10-09 for VVER-440). Inspection of inter-gasket cavities for TM, CPS flange leakage	Quality improvement of repair activities to prevent leakage of flange joints of safety system nozzles: assembly in strict compliance with design documentation on RPV, examination of flange joints in hydraulic testing	No leakage. Actual condition of base metal and welds meets the requirements of documents according to quality assessment standards presented in standard program TPPK-2013 for VVER-1000 (AIEU-10-09 for VVER-440)	Repair or replacement

No	Component/ area	Affected function	Degradation mechanisms/ ageing effects	Preventive measures	Identification of ageing effects	Monitoring and analysis of trends for ageing effects	Degradation mitigation	Acceptance criteria	Corrective measures
5	Threaded seats and sealing surfaces of RVCH nozzle flanges	Tightness of nozzle flange joints	Mechanical wear, crushing, mechanical damage/defects	Not applied	Periodic NDI according to standard program TPPK-2013 for VVER-1000 (AIEU-10-09 for VVER-440).	Periodic NDI according to standard program TPPK-2013 for VVER-1000 (AIEU-10-09 for VVER-440).	Tightening of studs of flange joints of CPS nozzles (TM, EB) should be performed according to the instructions on gasket mounting and for flange joint of air valve according to the maintenance instructions	Actual condition of base (clad) metal meets the requirements of documents according to quality assessment standards presented in standard program TPPK-2013 for VVER-1000 (AIEU-10-09 for VVER-440)	Repair or replacement
6	Components of metal structures	Retaining CPS drives from release under break of RVCH nozzles. Protection against flying objects. Biological protection	Boron corrosion, wear, mechanical damage/defects	Not applied	Periodic examination	Periodic examinations in compliance with TS for repair	Strict compliance with in equipment assembling and disassembling	No defects and damages	Repair and replacement

05.1.2.2 Use of national and international experience

Accounting and implementation at Ukrainian NPPs of operating experience and findings of current research and development is ensured by meeting the requirements of NP 306.2.141-2008 [5], NP 306.2.210-2017 [7], NP 306.2.099-2004 [6], Standard AMP [11], as well as provisions on the system of applying NPP operating experience developed taking into account WANO and IAEA requirements and recommendations for information exchange on operational events and NPP operation experience feedback.

The main information source on NPP operating experience are documents containing the information on:

- internal NPP operating experience;
- external operating experience;
- performed assessments of the system for NPP operating experience accumulation, analysis and use;
- examples of positive operating practice;
- findings of new research and development;
- meeting of the Council of Chief Engineers, Council of Operating Experience Experts.

The main documents containing the information on internal operating experience:

- preliminary message on abnormal operation, operational events, inconsistencies at NPPs;
- reports on investigation of abnormal operation and operational events, certificates on investigation of category 1 and 2 inconsistencies at NPPs;
- reports on investigation of insignificant and undeveloped events at NPPs;
- reports on efficiency of corrective measures assigned after investigation of abnormal operation, operational events, and inconsistencies, insignificant and undeveloped events at NPPs.

The main documents containing the information on external operating experience:

- Documents of the Energoatom:
    - preliminary messages on events (operational events, abnormal operation, and inconsistencies) at NPPs of the Energoatom;
    - reports on investigation of abnormal operation and operational events, certificates on investigation of category 1 and 2 inconsistencies at NPPs of the Energoatom;
    - feedback cards received from the Energoatom;
    - industry-wide corrective measures according to the analysis of level 2 root causes.
- Documents containing the information on equipment condition, reliability and safety of Ukrainian NPPs, improvement of NPP components and systems, on other:
    - equipment failures and reliability at Ukrainian NPPs;
    - nuclear fuel failures and reliability at Ukrainian NPPs;
    - improvement of planning and performing maintenance and repair at NPPs;
    - improvement of nuclear, radiation, technical safety at Ukrainian NPPs;
    - improvement of chemical technologies at Ukrainian NPPs;
    - positive (good) practice at Ukrainian NPPs.
- WANO-MC documents:
    - significant operating experience report (SOER);
    - significant event report (SER);

- just-in-time operating experience report (JIT);
- WANO event report (WER);
- report on positive practice (MC-TEMP).

– information documents of IAEA and other foreign organizations regarding NPP operating experience.

– Information documents of Ukrainian electricity generation facilities not included into the Energoatom.

– Information documents and reports of the regulators and their organizations (SNRIU, SSTC NRS).

– Information documents of design and research organizations (KIEP, KhIEP, NRI), suppliers, manufacturers.

The main documents on performed assessments:

– self-assessment reports;
– WANO-MC per review reports;
– OSART mission reports;
– reports on quality system audits;
– orders of inspections of regulatory and oversight bodies.

Examples of "good practice" should be identified and distributed within the experience exchange. A focus should be brought on the activities aimed at safety improvement at NPPs, prevention of inconsistencies during their operation, improvement of operating procedures, methods and ways for diagnostics of structures, systems and components taking into account ageing and wear.

The positive practice of operation is based on successful application of new equipment, technology, modernization of systems and installations, new approaches in the system of maintenance and repair, work with personnel and allows safety and reliability improvement of components, systems, and NPP in general.

Findings of research and development were also considered when developing AMP of RPV and RVMF fastening components. The following reports on inconsistencies in RPV and RVMF operation were considered:

– PWER ATL 13-0274. Reactor Shutdown after Detection of a Crack;

– PWER PAR 15-0563. Registered Indications Detected in Nondestructive Inspection of Reactor Pressure Vessel;

– PWER PAR 12-0104. Detection of Reactor Pressure Vessel Defect at Doel NPP Unit 3;

– MER PAR 09-021. Defect on Reactor Pressure Vessel;

– 650-2. 4HB0-P11-10-10-07. Defects in Welds No. 11 (8/4) between Protective DN500 Nozzles of Reactor Pressure Vessel Detected in In-Service Metal Inspection in Scheduled Outage-2007;

– 738. MER ATL 09-210. Leakage of Reactor Vessel Closure Head Flange;

– 954-40. MER TYO 11-117. Crack in Weld of Reactor Pressure Vessel Outlet Nozzle;

– 904. 1KLN-P10-06-12-10. INES-0. EAR MOW 11-005. Failure of Reactor Main Flange Sealing Caused by Insufficient Tightening of RVMF Screws.

In developing AMP of the reactor closure head assembly, the following reports on inconsistencies in safe operation of the reactor closure head assembly were considered:

– WER MOW 13-0019. Defects in Metal of the Reactor Closure Head Assembly at Kola NPP Unit 2;

– 2ROS-P10-03-04-13. WER MOW 13-0048. Unscheduled Outage of Unit 2 to Replace Nozzles in the Reactor Closure Head Assembly having Flow-Accelerated Corrosion;

– 833. 63AP-P07-003-05-10 (revised). INES-0. Failure of Equipment Important to Safety (EV-4 Nozzle of the Reactor Closure Head Assembly);

– UNITED STATES NUCLEAR REGULATORY COMMISSION, NRC Bulletin 2002-01, Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity, US NRC, Washington, D.C., March 18, 2002;

– UNITED STATES NUCLEAR REGULATORY COMMISSION, NRC Information Notice 2003-02, Recent Experience with Reactor Coolant System Leakage and Boric Acid Corrosion, US NRC, Washington, D.C., January 16, 2003;

– UNITED STATES NUCLEAR REGULATORY COMMISSION, NRC Regulatory Issue Summary 2003-13, NRC Review of Responses to Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," US NRC, Washington, D.C., July 29, 2003;

– SOER 2003-2: Reactor Pressure Vessel Head Degradation at Davis-Besse Nuclear Power Station – Rev.1.

05.1.2.3 Use of research and development programs

In VVER-1000 reactors of Ukrainian NPPs, surveillance specimens are located above the core as required by design documentation – on the internals baffle.

According to the world practice, irradiation of the surveillance specimens in containers on the RPV wall mostly corresponds to irradiation conditions on the VVER-1000 RPV wall in the beltline region. This condition was implemented in the Integrated Surveillance Program (ISP) at Czech Telelin NPP.

In this regard, SNRIU imposed requirement in 2004 to implement ISP developed by the Řež Nuclear Research Institute (Czech Republic) as one of the conditions for KhNPP-2 and RNPP-4 commissioning.

ISP covers RPV of KhNPP-2, RNPP-4, RNPP-3 and ZNPP-6 for which the Řež Nuclear Research Institute produced surveillance specimens from archive metal that was previously purchased at the RPV manufacturer and is property of the institute. Compliance of the archive material with chemical composition of RPV of the above power units was confirmed by certificates.

The main objective of this program is to confirm identify of changes in properties of the surveillance specimens according to the regular program for VVER-1000 and RPV material. In addition, it will allow additional changes in properties of RPV materials of KhNPP-2, RNPP-4, RNPP-3 and ZNPP-6.

In implementation of the first ISP part, the surveillance specimens from two main containers (1EU and 2EU) and additional container U1 were tested. The operator plans to further continue ISP activities.

05.1.3 Monitoring, testing, sampling and inspection activities for RPV

05.1.3.1 Description of surveillance specimen program

VVER-1000

In accordance with the requirements of regulatory and design documentation for inspecting the properties of RPV metal with destructive methods, the surveillance program is implemented. In compliance with this program, surveillance specimens made from the base metal, weld metal and heat affected zone (HAZ) metal are placed into the reactor before it is put into operation. Typically, specimens of the base metal are cut out of surplus metal of one of the standard RPV rings, and the weld and HAZ samples are cut out from a weld sample produced by the same performers and with the same methods using the welding materials of the same grade as for one of the welds located in the beltline region.

To determine the actual changes in the mechanical properties of the RPV metal (yield stress, ultimate strength, elongation, contraction) and brittle fracture resistance characteristics (metal nil ductility temperature, fracture toughness), surveillance specimens are periodically removed from reactor and tested.

In all Ukrainian reactors of VVER-1000 type, the surveillance specimens are located near the inner wall of the core barrel in the space between the baffle and upper internals (Figure 5.1). In total, six sets of surveillance specimens are loaded into the reactor, each of which consists of five container assemblies. The containers may be located in an assembly in one or two layers. The neutron flux falling on the specimens of the upper layer is approximately equal to the neutron flux falling on the inner surface of the RPV in the beltline region. The specimens in the lower layer assume a slightly larger (approximately 2.5 times) fast neutron flux and are intended to predict the condition of RPV metal.

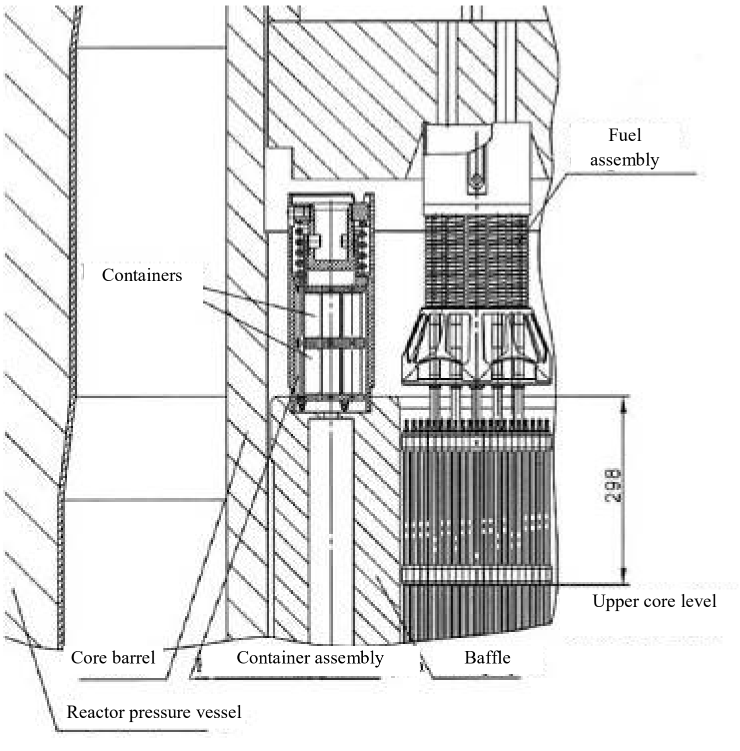

Figure 5.1 Location of container assemblies in the reactor

Besides surveillance specimens, from three to six temperature sets of surveillance specimens are loaded into the rector; they are located near the upper internals plate at the level of outlet nozzles where coolant temperature is the highest. They are intended to determine change in metal properties associated with temperature ageing.

The nomenclature of the reference surveillance specimens should correspond to the nomenclature of the irradiation and temperature sets loaded into RPV, and their number should be sufficient to accurately assess the initial properties of the RPV metal in accordance with PNAE G-7-008-89 [8] and PNAE G-7-002 -87 [9].

Irradiated (1L - 6L) and temperature (1M - 6M) surveillance specimen sets are removed during scheduled outage. The surveillance specimens (sets 1L and 1M) can be first removed after operation of the reactor for five years (fuel cycles). This period is minimum and may be revised upon justification of a later date based on experience in implementation of the surveillance programs at Ukrainian and foreign power units with VVER-1000 reactors, as well as data on fast neutron fluence accumulated by RPV elements.

Most power units in Ukraine implement regular surveillance program; one of its disadvantages is that one layer accumulates neutron fluence in a range that exceeds the requirements of PNAE G-7-002-87 [9]. In this regard, to select representative groups of surveillance specimens and, as a consequence, increase the reliability in determining the properties of RPV metal, the surveillance specimen sets should be tested under the regular program using the reconstruction technology for tested specimens and then reconstructed surveillance specimens should be tested.

To reconstruct impact bending or V-notched specimens, fragments (halves) of surveillance specimens made of RPV material tested for impact and three-point bending are used (Figure 5.2).

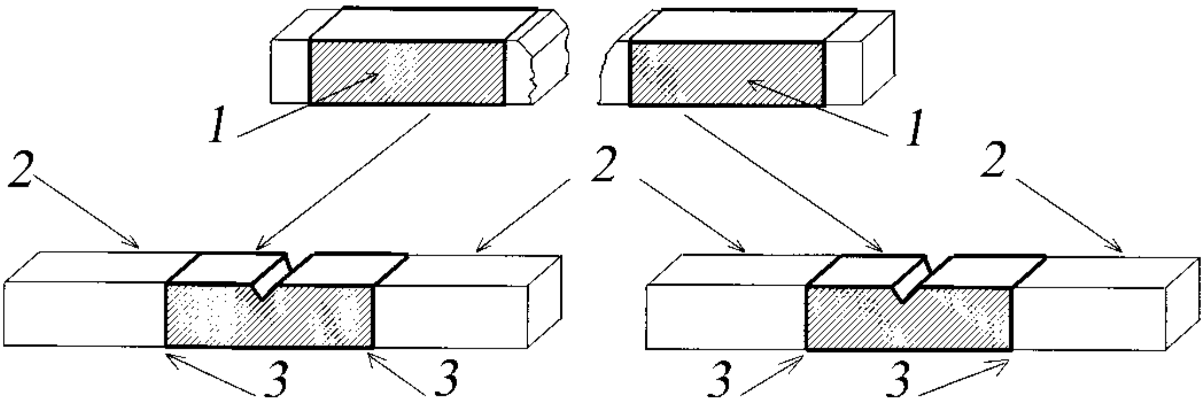

Figure 5.2 Scheme for reconstruction of Charpy specimens

1 – inserts made of halves of bending or V-notched surveillance specimens; 2 – end pieces, 3 – welds

End pieces of pressure vessel steel are welded to the processed flat-parallel ends of the specimens, a notch imitating a crack is made at the center and a fatigue crack is grown in fracture toughness specimens.

The reconstruction technology allows increasing the number of specimens and selecting groups of specimens to plot serial curves for impact bending tests, which should meet the requirements for both the number of experimental points per curve and the uniformity of irradiation.

The surveillance program used for Ukrainian NPPs ensures complete monitoring over changes in properties of RPV materials.

VVER-440 (RNPP unit 1)

The RPV of RNPP unit 1 was covered by the surveillance program over the entire service life. Long-term operation of the power unit was justified with requirement to implement recovery annealing of weld No. 4, which determines the radiation life of RNPP-1 RPV. For RPV operation after recovery annealing, a new RPV metal inspection program was implemented for the entire service life using surveillance specimens.

The "Surveillance Program to Support Operation of Rivne NPP Unit 1 Reactor Pressure Vessel after Recovery Annealing" determines the procedure for monitoring of changes in metal properties in operation of weld No. 4 and metal of the core upper and lower rings of RNPP unit 1 RPV that were subjected to recovery annealing at 475 ± 15 ^{0}C for 150 h.

All regular surveillance specimens of RNPP-1 RPV were removed and examined. Since there is no archive RPV metal, material of previously tested regular surveillance specimens installed for irradiation as inserts fabricated in compliance with the reconstruction technology were used as the base metal of RNPP-1 RPV. It is not sufficient to use only metal of the surveillance specimens of RNPP-1 weld No. 4. Analysis of radiation embrittlement of VVER-440 RPV material after repeated (after annealing) irradiation shows that shift of the nil ductility temperature substantially depends on phosphorus content. As was shown, the surveillance specimens of the RNPP-1 RPV weld did not comply with the phosphorus content of 0.041% accepted for the vessel weld.

In this case, the surveillance program for the weld uses the database ideology. This approach involves verification of the repeated radiation embrittlement model used in RPV strength calculations for conservatism for all materials used in this surveillance program.

Several weld materials with phosphorus content varying from 0.027 to 0.051%, produced using the regular VVER-440 RPV technology, are installed for repeated irradiation after annealing. The surveillance program includes materials that were originally irradiated at different rates of the damaging dose (different fast neutron flux densities).

The same concept of the surveillance program was implemented at Loviisa NPP unit 1 (Finland).

VVER-440 (RNPP unit 2)

The control effects of ageing on the mechanism of radiation embrittlement are changes in mechanical properties.

The main characteristics for estimating the residual life in terms of BFR are the critical stress intensity coefficient, critical temperature of brittleness, and yield stress.

This mechanism is monitored during operation by the results of tests on surveillance specimen sets, based on which dependencies were obtained for assessing the radiation embrittlement of the base metal and weld metal at the end of the service life up to 60 years. Based on monitoring of the surveillance specimens, RPV components do not reach the limit state after completion of the service life up to 60 years.

The main nodes used to assess the residual life of RPV in terms of BFR is weld No. 4, base metal in the beltline region and nozzle area. The criterion for reaching the limit state for RPV is allowed temperature T_{ka}, which is specific for weld No. 4.

The calculation results showed that the operation of RNPP-2 RPV in terms of BFR is ensured during the design-basis life up to 40 years and during LTO up to 60 years.

Testing of surveillance specimens

The Ukrainian NRI is the center where surveillance specimens are tested in Ukraine. The NRI has hot cells (Figure 5.3) equipped with necessary equipment and intended for work with irradiation materials. In the recent decade, the hot cells have been upgraded and provided with modern experimental equipment for testing and reconstruction of irradiated surveillance specimens.

Figure 5.3 Hot cells at the Nuclear Research Institute

To improve the testing of surveillance specimens, equipment for testing and reconstruction of surveillance specimens was purchased and transferred to NRI. Along with the NRI equipment, the purchased equipment allows successful testing of RPV surveillance specimens (successful tests of RPV surveillance specimens) for Ukrainian NPPs in the timeframes determined by schedules for testing of regular and reconstructed surveillance specimens of NPP RPVs.

Energoatom will upgrade the modern test facility for radiation materials science and additionally equip hot cells at NRI at its own expenses in accordance with the plan agreed with NRI.

05.1.3.2 Hydraulic testing and in-service inspection

Hydraulic tests

Hydraulic tests are performed to assess the strength and integrity of NPP equipment and piping. The frequency of hydraulic tests is determined in PNAE G-7-008-89 [8] and is every four years for strength test and every year for leak-tightness tests.

Since 2014, technical decisions have been implemented at Ukrainian NPPs to change parameters of hydraulic tests (decrease pressure and temperature):

for strength of the primary system:

– for VVER-1000 – from 250 kgf/cm^2 (24.5 MPa) to 207 kgf/cm^2 (20.3 MPa) and with inspection at 180 kgf/cm^2 (17.6 MPa);

– for VVER-440 – from 195 kgf/cm^2 (19.1 MPa) to 161 kgf/cm^2 (15.7 MPa) and with inspection at 140 kgf/cm^2 (13.7 MPa).

for strength of the secondary system:

– for VVER-1000 – from 110 kgf/cm^2 (10.78 MPa) to 92 kgf/cm^2 (9.02 MPa) and with inspection at 80 kgf/cm^2 (7.85 MPa);

– for VVER-440 – from 78 kgf/cm^2 (7.6 MPa) to 65 kgf/cm^2 (6.4 MPa) and with inspection at 56 kgf/cm^2 (5.5 MPa).

The pressure of hydraulic tests was changed because it allow decreasing the fatigue of RPV material and other equipment and piping.

Nondestructive inspection

Periodic inspection is conducted to reveal and record defects in metal and assess metal condition.

In-service NDI (periodic inspection) of RPV metal is conducted within working inspection programs developed in compliance with the following standard programs:

– for VVER-1000 – Standard Program for Periodic Inspection of Base Metal, Welds and Claddings of Equipment and Piping of VVER-1000 Nuclear Power Plants (TPPK-13). PM.T.0.03.061-13 [64];

– for VVER-440 – Standard Program for In-Service Inspection of Base Metal, Welds and Claddings of Equipment and Piping of VVER-440 (V-213) Nuclear Power Plants. AIEU-10-09 [31].

Visual inspection

Visual inspection of RPV elements is conducted to detect surface cracks, imperfections, detachments, gaps, surface inclusions, clusters and other imperfections. This inspection is carried out in accordance with SOU NAEK 009:2013 “Maintenance and Repair. Nondestructive Visual and Instrumented Inspection. Methodology for Inspection of Base Materials (Semi-finished Products), Welds and Cladding of NPP Equipment and Piping” [77].

The scope of visual inspection for RPV components is determined in TPPK-13 [64] and is performed every four years, except for a number of inspection areas on the flange and reactor sealing and closure head (annual inspection in case of over-sealing).

Ultrasonic inspection

Ultrasonic inspection is carried out to detect cracks, shells, floats, flakes, stratifications, non-metallic inclusions, and other imperfections in the main materials (semi-finished products, parts, assemblies) and base metal, which cause or result in the appearance of echoes with amplitude more than a given value, the so-called fixation level, or reduction in the previous signal to a value lower than specified fixation. This inspection is carried out in accordance with “SOU NAEK 027:2014. Maintenance and Repair. Nondestructive Ultrasound Inspection. Methodology for Inspection of Base Materials (Semi-finished Products)”.

The scope of visual control of RPV elements is determined in TPPK-13 [64] and is conducted every four years.

Eddy current inspection

Eddy current inspection is used to detect subsurface and surface imperfections on anticorrosive cladding. This inspection is carried out in accordance with 104-4-04-M-SKM "Methodology for Automated Eddy-Current Inspection of Inner Corrosion-Resistant Austenitic Cladding of VVER-440 and VVER-1000 Pressure Vessels Using CMM-SAPHIRplus System."

The scope and frequency of eddy current inspection are determined in TPPK-13 [64].

Liquid penetrant and magnetic particle inspection

Liquid penetrant inspection of RPV elements is used to detect imperfections that extend to the surface: cracks, pores, sinks, imperfections, intercrystalline corrosion and other imperfections. Liquid penetrant inspection is applied to the surface of products accepted on the basis of visual inspection, in accordance with current regulatory requirements. For welds and claddings, this inspection is conducted in accordance with “SOU NAEK 014:2013.

Nondestructive Liquid Penetrant Inspection. Methodology for Inspection of Materials (Semi-finished Products), Welds and Claddings of NPP Equipment and Piping”.

Nondestructive magnetic particle inspection is intended to detect surface and subsurface metal imperfections (cracks, scratches, floats, inclusions, flakes, etc.) for products made from ferromagnetic materials with relative magnetic permeability of at least 40. This inspection is conducted according to “SOU NAEK 066:2015. Nondestructive Magnetic Powder Inspection. Methodology for Inspection of Materials (Semi-finished Products), Welds and Claddings”.

The scope of liquid penetrant and magnetic particle inspection of RPV and RVCH is defined in TPPK-13 [64] and is conducted every four years, except for a number of inspection areas on the flange and sealing of the RPV and RVCH (inspection in case of over-sealing).

05.1.3.3 Monitoring of neutron fluence on RPV wall

At all Ukrainian NPPs, irradiation conditions and current and accumulated neutron fluence in typical RPV areas are determined.

Typical areas of VVER-100 RPV include:

- welds in the upper and support RPV rings (welds No. 3 and 4);
- RPV area where current and/or accumulated neutron fluence is maximum. As a rule, this area is located at or above the most energy-intensive layer of the reactor core.

Current RPV neutron fluence is the maximum fast neutron fluence on the inner surface of typical areas of the pressure vessel or maximum displacements per atom (dpa) caused by neutrons with En>0.5 MeV per fuel cycle. Accumulated neutron fluence is the maximum total neutron fluence on the inner surface of typical areas of RPV or maximum dpa caused by neutrons with En>0.5 MeV for the entire operating period of the power unit.

Equipment for installation of neutron activation detectors near the outer RPV surface is a part of the RPV neutron fluence monitoring system. In the development of main equipment components, needs for dose measurements at the level of welds No. 3 and 4 and at the level of maximum current neutron fluence were taken into account. Besides this, the necessary condition was to ensure minimum doses for personnel directly working with the equipment.

Knowledge of irradiation conditions and current neutron fluence in specific RPV areas during each fuel cycle allows determining the accumulated neutron fluence and:

- obtaining data required to transfer the results of surveillance specimen tests directly to RPV metal;
- assessing the time when specific RPV areas reach the maximum allowed neutron fluence.

For all Ukrainian NPPs, radiation fluence on RPV wall is permanently monitored to confirm the maximum safe operating period for RPVs.

05.1.3.4 Consideration of load cycles

Monitoring of the number of reactor load cycles allows recording all changes in operating parameters (pressure and temperature) required to correctly assess fatigue (by calculation) of RPV critical areas.

Analysis of operating conditions and actual load cycles to calculate RPV operating modes is carried out in accordance with the “Procedure for Analysis of Load Cycles for Equipment and Components of Energoatom VVER-1000 NPPs. PL-D.0.03.588-16”.

To obtain data on actual operating impacts on RPV, RPV operating conditions are monitored and analyzed to:

– determine the number and sequence of operating modes that were observed in the reactor primary system for the entire operating period;
– determine the number and sequence of actual normal operation modes and transients;
– determine the number of reactor load cycles in normal operation, abnormal operation, emergencies and accidents, which is compared to the design number of cycles over the entire design-basis life of the reactor;
– classify operating modes into load groups.

05.1.3.5 Primary water chemistry control

Water chemistry control in the primary system is a standard preventive program. The objective of this program is to prevent and minimize ageing effects caused by coolant quality.

Primary water chemistry is monitored and controlled in accordance with SOU-N YaEK 1.013:2014 [39] and SOU-N YaEK 1.012:2014 [40]. Ammonia potassium water chemistry-1 with boron regulation is maintained in primary systems of all NPPs with VVER compliant to these regulatory documents.

These documents establish values for the process parameters of the primary water chemistry to ensure performance of fuel assemblies in the core during the design lifetime and storage after operation at NPPs, ensure design lifetime of reactor equipment and minimize the accumulation of activated corrosion products in the reactor primary system in all design modes.

In 2016, there was one incompliance of controlled quality parameters for the primary system coolant with the range of allowable values within the first level of activities at ZNPP-3 at the total molar concentration of alkali metal ions.

The analysis of results of in-service chemical inspection of NPP primary system coolant demonstrates the steady maintenance of water chemistry -1 and ensuring of safe operation conditions at NPP units.

05.1.4 Preventive and remedial actions for RPV

05.1.4.1 Preventive actions

Preventive actions aimed at the mitigation of ageing effects include measures presented in Table 5.4.

Table 5.4 Preventive and corrective actions

Inspection component/area	Ageing effect or degradation mechanism	Ageing management measures (program)	Corrective action
The whole RPV: - elliptical bottom; - lower cylindrical ring; - weld No. 3 (between lower and upper rings); - upper cylindrical ring; - weld No. 4 (between upper and support rings);	Radiation embrittlement and strengthening; thermal embrittlement and strengthening; occurrence of defects; corrosion, stress corrosion cracking; fatigue; mechanical wear.	Operation of reactor components (pressure vessel) shall be carried out in accordance with regulatory and operational documents in force at NPP.	Continue maintenance of primary water chemistry parameters at the power unit in accordance with requirements of current regulatory documents. Continue using the procedure for core loading with low neutron leakage. Implement the leak-before-break concept to prevent scenarios related to MCP rupture (LOCA) at the relevant power unit.

Inspection component/area	Ageing effect or degradation mechanism	Ageing management measures (program)	Corrective action
- support cylindrical ring; - lower ring of the nozzle area, including nozzles (DN850) and ECCS nozzles; - upper ring of the nozzle area, including nozzles DN850, ECCS nozzles and nozzle for pulsed lines; - flange ring.		Perform technical examination of reactor components (pressure vessel) in the scope and conditions according to current requirements and approved schedule for technical inspection of NPP equipment and piping	During hydraulic test at a reduced pressure, it is necessary to maintain temperature not lower than the minimum allowable temperature established in the approved technical solution.
		Conduct periodic inspection of the base metal, welds and cladding of reactor components (pressure vessel) in accordance with TPPK-13 [64].	In addition to the TPPK-13 [64], during each major repair of the power unit with the core unloading, starting from the scheduled outage in 2020, inspection of the reactor pressure vessel ring shall be performed from inside by means of RPV-1000 inspection system according to the following scope: - anticorrosive cladding on the inner surface of nozzles DN850, including the cladding over the bottom of welds in welding of the primary coolant system to reactor nozzles – eddy current inspection – 100%, ultrasonic inspection – 100%; - radial transitions of nozzles DN 850, a section of 150 mm wide along the perimeter (circle): a) anticorrosive cladding eddy current inspection – 100%, ultrasonic inspection – 100%; b) base metal ultrasonic inspection - 100%; - anticorrosive cladding in the area of nozzles DN 850, DN 300 eddy current inspection – in accessible places, ultrasonic inspection – in accessible places; - anticorrosive cladding on the ring flange in the technically possible scope, eddy current inspection – 100%, ultrasonic inspection – 100%; -anticorrosive cladding of reactor pressure vessel bottom, eddy current inspection – 100%; - anticorrosive cladding of the reactor pressure vessel on the section from weld No. 2 and to the support ledge eddy current inspection – 100% except the section in the area of vibration dampers brackets inaccessible to inspection; - anticorrosive cladding on welds No. 2, 3, 4, 5, 6, 7 eddy current inspection – 100%.

Inspection component/area	Ageing effect or degradation mechanism	Ageing management measures (program)	Corrective action
		Continue implementation of Standard Program PM-T.0.03.120-08 [65]	Develop and submit to the SNRIU the technical decision on the next removal of RPV surveillance specimens of the respective power unit. The period for the next removal of surveillance specimens shall be justified taking into account results of the state review of nuclear and radiation safety with regard to calculations for resistance to brittle fracture at the respective power unit, in the part of neutron fluence accumulation by the surveillance specimens and RPV wall. Based on tests of the surveillance specimens of the third unload batch, develop an upgrading program for single-layer container assemblies to provide surveillance specimens to justify long-term operation of respective RPV.
		Continue dosimetric measurements of neutron fluence on external surface of RPV and calculation of neutron fluence on RPV according to the standard "Quality System. Definition of Neutron Fluence on VVER-1000 Pressure Vessel" SOU 73.1-23724640-004-2014.	If till the next safety review the core loading will be performed with increase of neutron release, it would be necessary to correct neutron fluence calculation and reassess RPV safe operation period according to brittle fracture criteria.

05.1.4.2 Remedial actions

Annealing of RNPP-1 RPV (VVER-440)

Recovery annealing was carried out for weld No. 4 of RNPP-1 on the basis of the relevant calculation justification, which confirmed the need for such a measure.

The annealing was performed from 20 September 2010 to 27 September 2010. The scope of activities included the following:

- modernization of annealing device (inspection and control system);
- calculation justification (calculation of temperature in pressure vessel components, support structures and shaft; material science justification of the annealing mode; definition of RPV material characteristics);
- inspection of RPV metal before annealing;
- preparation of unit process systems to annealing;
- mounting of the annealing device;
- conduct of recovery annealing;
- dismantling of the device and inspection of metal after annealing.

Annealing conditions:

- annealing temperature: 475 ^{0}C;

– duration: 150 hours.

A new surveillance program was developed as a result. The new surveillance specimen set was loaded into RPV before the beginning of long-term operation of the power unit (after scheduled outage of 2010).

Use of dummy assemblies in RNPP-1 (VVER-440) reactor

RNPP-1 reactor core consists of 276 working assemblies, 36 dummy assemblies and 37 control assemblies, which can move in vertical direction. The dummy assemblies installed since the beginning of unit operation to reduce neutron fluence on the base metal and metal of RPV welds and thereby limit ageing in view of radiation embrittlement.

The dummy assembly is a welded structure of details made of corrosion-resistant steel and consisting of head, end section, casing and displacer.

Formation of zones with small or low leakage

The formation of fuel loadings with low neutron leakage began with the implementation of profiled working assemblies with enrichment of 4.21% at RNPP-1,2 (starting from 19th fuel cycle at RNPP-1 and starting from 17th fuel cycle starting from RNPP-2).

New fuel, in particular, allows reducing of neutron fluence on RPV due to the core structure. Carries out calculations for the accumulation of neutron fluence on RPV showed additional reduction of fluence by 12% for RNPP-1 (by 25% for RNPP-2) due to the use of fresh assembly loading into the second (third) row from the core periphery.

Measures to prevent cold overpressure

In the process of operation, under neutron fluence on RPV, RPV material suffers from the shift in brittleness temperature to the higher temperature area, and RPV metal and weld resistance to brittle fracture is reduced. The shift of the brittleness temperature to the higher temperature area increases the risk of RPV overpressure in a situation where the reactor is in the cold shutdown state. The causes leading to the cold overpressure include the failure of the residual heat removal system, the imbalance between the supply and removal of the medium of the makeup and blowdown system, etc.

Measures to increase reliability of the primary system from high pressure in the cold state have been implemented at all NPP units according to the C(I)SIP. The measure has been implemented in accordance with the report IAEA-EBP-WWER-05 “Safety Issues and Their Ranking for VVER 1000 Model 320 Nuclear Power Plants”, which identifies an issue of category II: S1 “Primary Circuit Cold Overpressure Protection”.

PRZ PORV was replaced (modernized) to increase reliability of the primary system protection from high pressure in the cold state.

05.2 Outcome of experience and reviews of ageing management for RPV

Ageing management of reactor pressure vessel and its components is carried out taking into account results of performed activities on TCA and on the basis of national and international experience.

Depending on TCA results or during periodic safety review of reactor pressure vessel, the corresponding list of ageing management measures is formed.

The following general measures applicable for all RPVs are included into these measures:

– carry out technical examination of RPV components according to the scope and schedule in compliance with current regulations and rules, and pursuant to the approved schedules of the technical examination;

– conduct periodic inspection of the base metal, welds and cladding of reactor components (reactor pressure vessel) in accordance with TPPK-13 [64];

– continue to maintain the primary water chemistry parameters at an appropriate level;

– continue to use fuel loading with low neutron leakage;

– continue implementation of the surveillance program etc.

In addition to individual measures that are specific to a particular RPV, for example:

– recovery annealing of weld No. 4 was conducted for RNPP-1 as an ageing management measure, and a new surveillance program was developed and implemented after the annealing;

– upgrading of the surveillance program is envisaged for SUNPP-2;

– use of dummy assemblies to reduce neutron fluence on RPV is envisaged for RNPP-1;

– measurement of RPV and RVMF component hardness to control mechanical peculiarities and comparison of data for 2012, 2016 and 2024 is envisaged for RNPP-3.

It should be separately noted that for the reactor pressure vessel, reactor main flange and upper internals of unit 3, there are specific AMP:

– Ageing Management Program and Ageing Management Measures for Elements of Reactor Pressure Vessel and Reactor Vessel Main Flange. 191-232-O-PSE-16 [73];

– Ageing Management Program and Ageing Management Measures for Elements of Reactor Vessel Closure Head. 191-231-O-PSE-16 [74].

These programs were developed within the international cooperation between Energoatom and Řež Nuclear Research Institute. These programs are pilot projects for the implementation of AMP for specific components. According to results of their efficiency assessment, a decision will be made on the use of good practices at other Ukrainian NPP units.

05.3 Licensee's findings and conclusions on ageing management of RPVs

The following procedures are used to define ageing mechanisms of reactor pressure vessel and reactor closure head:

– NDI of the base (clad) metal and welds;
– monitoring of RPV metal properties using surveillance specimens;
– internal/ external inspection of equipment;
– density control of the reactor main flange.

According to the results of the procedure on revealing ageing mechanisms, analysis and assessment of the technical state, the following degradation mechanisms were defined for RPV:

– radiation embrittlement (for RPV);
– thermal ageing (for reactor pressure vessel and reactor closure head);
– fatigue (for reactor pressure vessel and reactor closure head);
– stress corrosion cracking (for reactor pressure vessel and reactor closure head);
– boron corrosion (for reactor pressure vessel and reactor closure head);
– creasing/mechanical damage (for reactor pressure vessel and reactor closure head).

Preventive and corrective measures are established for all degradation mechanisms aimed at degradation mitigation. Continuous monitoring and analysis of ageing effect trends are carried out. AMP is supplemented by the results of activities within TCA and on the basis of industry summary reports on conducted activities with ageing management of NPP components.

The results of performed activities on RPV TCA and LTO considering TLAA results indicate that in a number of cases the operator faces a lack of representative data based on tests of surveillance specimens. Certain actions are taken in order to exclude such a situation. In particular, a surveillance program for the RNPP-1 RPV after annealing has been developed and implemented and the design-basis container assemblies for surveillance specimens are under upgrading to locate them more favorably relative to the core.

Ageing management efficiency is periodically performed in accordance with NP 306.2.210-2017 [7] and SOU NAEK 141:2017 [75].

05.4 Regulator's assessment of ageing management of RPV and conclusions

RPV is a component that cannot be replaced and its current and estimated technical condition affects long-term operation of the power unit. Given this issue, both the operator and regulator pay special attention to RPV ageing management.

General requirements for ageing management are presented in NP 306.2.141-2008 [5], NP 306.2.210-2017 [7] and in relation to RPV they are specified in the operator's basic technical requirements [64], [65], [76], [77] and [78].

These technical requirements cover and regulate the following ageing management aspects:

- NDI of the base (clad) metal and welds, periodicity and scope of technical examination and hydraulic tests;
- controlled parameters and acceptance criteria;
- revision of the maintenance system;
- detection and monitoring of ageing effects for RPV, RVCH and RVMF;
- prediction of possible progression of the revealed ageing mechanisms;
- adjustment/loading of the core with fuel with low neutron leakage fuel;
- use of national and international experience;
- start, implementation and use of the results of research and development programs to improve operational practice and improve safety.

According to the specified aspects, new systems of remote NDI of RPV metal condition (such as CMM-SAPHIRplus, RPV-1000, etc.) are implemented at Ukrainian NPPs. There are improvements in the methodology for calculation of fluence, thermohydraulic parameters and strength calculation, which are reflected in TLAA used to justify safety of reactor pressure vessel long-term operation.

To provide more reliable results of tests for the surveillance specimens already removed from the reactor, the operator uses the reconstruction technology to increase the number of specimens to plot serial curves of bending tests and improve the accuracy and reliability of the mechanical properties of irradiated RPVs.

The operator developed and is implementing the Integrated Program [79] in order to receive additional data on regular, modernized and new surveillance programs to improve reliability of the assessment of changes in RPV metal properties (see para. 02.3.2.5 of this Report). According to this program, the surveillance specimens are irradiated in the beltline region. At the same time, the applied use of the results of implementing this program is complicated by a number of factors that are still not resolved by the operator (compliance with the conditions in which the results of surveillance specimen tests conducted in accordance with the Integrated Program [79] can be used for specific conditions of RPV operation in Ukraine has not been demonstrated).

Over the years of AMP implementation, the operator identified the main ageing mechanisms, determined parameters to be monitored and established acceptance criteria. All these aspects are continuously and carefully monitored by the operator under regulatory supervision.

The process of RPV AM continue to be improved on the basis of accumulated national and international experience and results of the implementation of research and development programs.

06 CONCRETE CONTAINMENT STRUCTURES

06.1 Description of ageing management for the concrete containment structures

06.1.1 Scope of ageing management for the concrete containment structures

The building of the reactor compartment consists of the following components: foundation; basis; containment; containment prestressing system; support plate; embedded details and metal structures for adjustment of reactor equipment and piping; walls and ceilings in the unsealed part.

The description of building structures in the reactor compartment of VVER-1000/320, 302, 338 and VVER-440/213 is presented in Annexes D, E, F.

The activities on ageing management of reactor compartment building structures (including the containment) is based on the provisions of industry documents: Standard AMP [11], SOU NAEK 109:2016 [38], PM-T.0.03.181-13 [67], MT-T.0.03.171-14 [68], PM-T.0.03.126-14 [69] and PM-T.0.03.127-13 [70].

The list of building structures and constructions to be included into AMP is defined on the basis of classification according to the safety impact taking into account data of the design, engineering and operational documents. The list of such structures is made for each power unit and for each building.

For example, the AM list for VVER-1000/320 includes the following components:

– inner structures of the containment;
– reactor compartment basis;
– foundation part (foundation plate, walls and ceiling) of the reactor compartment;
– reactor containment;
– reactor shaft;
– spent fuel pool;
– auxiliary building and ventilation pipe of the reactor compartment;
– main building (turbine hall, deaerator compartment).

Metal and reinforced concrete structures of each building shall have a list with information on possible degradation mechanisms, ageing affects, parameters of the technical state, materials, operational conditions, ageing management measures, etc. An example of the list of AM buildings and structures approved for the foundation plate of ZNPP-1 is presented in Annex D.

Ageing mechanisms are defined on the basis of monitoring results and activities on TCA of building structures.

The monitoring algorithm and TCA of supporting building structures are defined in SOU NAEK 109:2016 [38].

The methodology of activities on VVER-1000 containment according to the Standard Program PM-T.0.03.181-13 [67] is presented in Figure 6.1.

Standard SOU NAEK 109:2016 [38] defines:

– general rules for monitoring of NPP building structures;
– classification of the technical state of building structures, and NPP buildings and structures in general;

– TCA procedure for NPP buildings and structures according to results of periodic examination, continuous monitoring and diagnostics of parameters of building structures using automatic system for technical diagnostics;

– criteria for boundary condition of building structures, basis and foundations of NPP buildings and structures;

– procedure for the development and implementation of I&C at NPP;

– procedure for prediction and management of the technical condition of building structures, basis and foundations for design period and long-term operation of NPP buildings and structures.

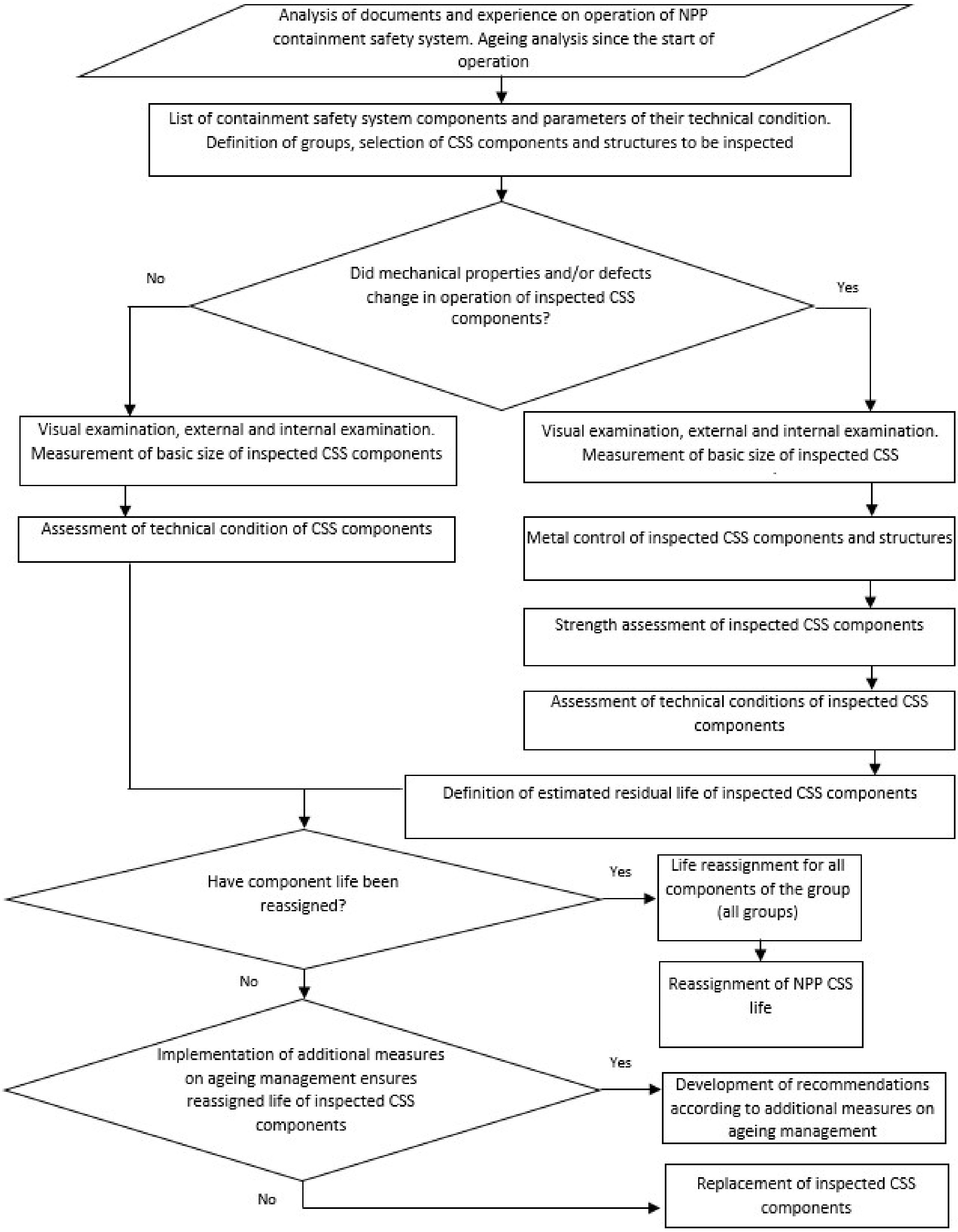

Figure 6.1 Methodology of activities on VVER-1000 containment systems according to the standard program PM-T.0.03.181-13

06.1.2 Ageing assessment for concrete containment structures

Standard AMP [11], SOU NAEK 109:2016 [38], PM-T.0.03.181-13 [67], MT-T.0.03.171-14 [68], PM-T.0.03.126-14 [69] and PM-T.0.03.127-13 [70] are used for ageing assessment.

AM measures shall be defined on the basis of continuous monitoring of building structures, results of activities with TCA and taking into account results of research and development programs.

Examples of using the results of research and development programs include the following.

1. Experimental studies of containment behavior under the combination of loads – maximum design-basis accident and safe shutdown earthquake in order to define possibility of the containment to fulfil its functions in the specified conditions in LTO period.

The objective of studies is to:

– carry out experimental studies of specimens that model containment fragments with regard to preserving integrity of the sealing lining (welds in particular) in case of lining stability loss under impact of corresponding power and temperature loads;

– analysis of confining ability (resistance to penetration) of concrete in reinforced concrete structures of containment enclosure, taking into account natural porosity of concrete and the presence of structure microdamage in it;

– implementation of numerous studies on the definition of rational level of CPSS tendon stressing along rows to provide the necessary bearing capacity, crack resistance and deformability of the containment.

The specified activities are under implementation and are implemented in accordance with the program approved by the SNRIU "Working Program for Examination of Building Structures of SUNPP-1,2 Containments. PM.1.3812.0248". Corresponding changes in the structure of the containment will be implemented according to results of implementation of this research and development program into AMP.

2. Arrangement of additional loading of the reactor compartments of ZNPP-1 and ZNPP-3 to prevent the influence of building heeling on the position of reactor flange (heeling values for other power units are within the design boundaries).

Monitoring settlement and heeling of buildings of ZNPP-1 and ZNPP-3 reactor compartments started from the moment of pouring reinforced concrete foundation plates of reactor compartments at the elevation 4.200 in 1980 and continues till now. It includes monitoring of general heeling and settlement of structures and heeling of RVMF. Additional examination of ground bases of the reactor compartments was conducted in 1995 and 2004. These examination showed that heeling increases and can go beyond the design values. The Prydniprovia State Academy of Civil Engineering and Architecture conducted a number of research and development activities. The results of these activities proposed to arrange additional surcharging of the reactor compartment (see Figure 6.2 and Figure 6.3) for stabilization of building heeling and the relevant design has been developed. The design envisages placement of one-sided surcharge on structures of the reactor compartment auxiliary building from the side opposite to the building heeling. Installation of surcharge at ZNPP-1 and ZNPP-3 power units has been completed and put into trial operation in 2012. In a year (in 2013), it was put into commercial operation. The arrangement of additional surcharge was one of obligatory conditions in transitions of power units to LTO, which was fulfilled by the operator. The containment was included into the AM list and monitoring of condition of the reactor compartment with surcharge is included into the ageing management measures. Currently, there is a process of stabilization and reducing of heeling due to surcharging.

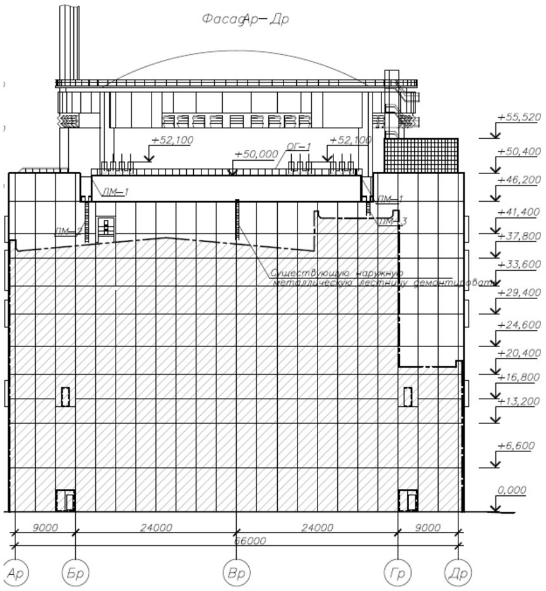

Figure 6.2 Layout of surcharge on reactor compartment auxiliary building

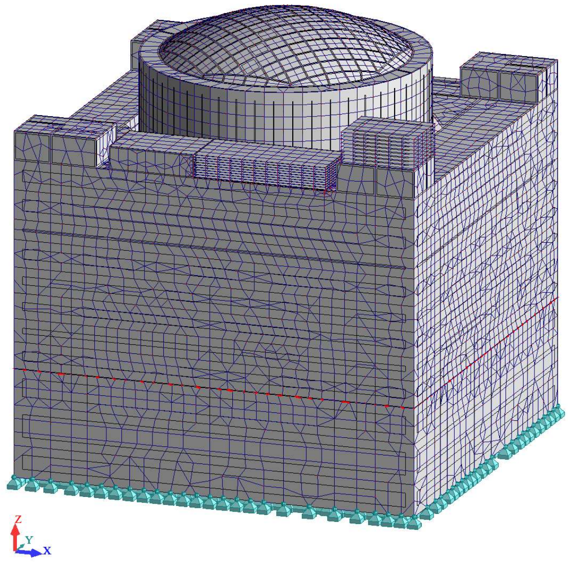

Figure 6.3 Computer finite-element model of the reactor compartment

3. Ukraine received significant contribution to the development of AM system within international cooperation and exchange of experience. The examples of using international experience include the implementation of the project "Creating a System of Monitoring Technical Condition of NPP Buildings and Structures Based on Modern Methods and Technologies" (international project U1.05/12, participants - Empresarios Agrupados Internacional/EGIS/SITES, Ministry of Energy and Coal Industry of Ukraine, Energoatom).

The general objective of the project was to create improved system of continuous monitoring, assessment and management of structural integrity of NPP buildings and systems important to safety in order to ensure their ability to fulfil envisaged operational functions and safety functions.

The project aimed at the assessment of available methods and rules of monitoring practice and technologies proposed for their improvement. Expected output of the project was to create the corrected and improved system for monitoring structural integrity of NPP buildings and structures important to safety to ensure their ability to comply with design functions. The monitoring procedure shall be used in accordance with the best international practice. It shall also define differences between Ukrainian and international rules.

Creation of the system for monitoring technical condition of NPP buildings and structures will give an opportunity to control main parameters that ensure reliable operation of buildings and structures, which is especially relevant at the stage of LTO and AM of NPP buildings and structures.

Ageing mechanisms that affect life of reactor compartment components with measures for monitoring and deterrence of degradation for VVER-1000/320 and VVER-440/213 are presented in Table 6.1 and Table 6.2.

Table 6.1 List of ageing mechanisms affecting the lifetime of the containment safety system and reactor compartment components including degradation monitoring and mitigation measures (for VVER-1000/320)

Components and documents for operation		Degradation mechanisms	Ageing effect	Technical condition parameter	Methods for inspection of technical condition parameters	Frequency	Assessment of effect of technical condition parameters on component lifetime	Assessment of need for additional measures for degradation monitoring and mitigation
1	Reactor compartment base	Change in structure and mechanical properties	Deformations and displacement. Settling	Change in profile and elevations of ground surface. Formation of dents and holes	Visual inspection	During TCA in scheduled outage	Insignificant effect	Upon results of visual inspection, develop, if needed, a program for the next visual inspection. Ensure soil level, structure and mechanical properties
2	Foundation plate	Leaching. Change in structure and mechanical properties. Corrosion	Cracking. Corrosion of reinforcement	Change in size. Decrease in mechanical properties. Increase in defects	Visual inspection	During TCA in scheduled outage	Insignificant effect	Upon results of visual inspection, develop, if needed, a program for the next visual inspection. Recovery repair, protective covering
3	Walls and coverings of foundation part	Leaching. Local corrosion	Cracking. Corrosion of reinforcement	Change in size. Decrease in mechanical properties. Increase in defects	Visual inspection	During TCA in scheduled outage	Insignificant effect	Upon results of visual inspection, develop, if needed, a program for the next visual inspection. Recovery repair, protective covering
4	Embedded details and metal structures for fixing of reactor equipment and piping	Local corrosion. Cyclic ageing	Thinning. Cracking. Loss of stability	Change in shape and size. Change in mechanical properties	Visual and instrumented inspection	During TCA in scheduled outage	Possible effect (for embedded parts)	Upon results of visual inspection, develop, if needed, a program for the next visual inspection. Recovery repair, protective covering

Components and documents for operation		Degradation mechanisms	Ageing effect	Technical condition parameter	Methods for inspection of technical condition parameters	Frequency	Assessment of effect of technical condition parameters on component lifetime	Assessment of need for additional measures for degradation monitoring and mitigation
5	Sealing steel lining	Stress corrosion cracking, general corrosion of components, local corrosion	Cracking, change in mechanical properties, thinning, pitting	Change in shape and size. Thinning. Change in mechanical properties	Visual and instrumented inspection. MHT	During TCA in scheduled outage	Affects	Upon results of visual and instrumented inspection, develop, if needed, a program for the next MHT and VI. Recovery repair, recovery of pain coating where damaged
6	RCEs (in particular, support plate)	Stress corrosion cracking, general corrosion (destruction) of concrete, carburization of concrete in protective layer, corrosion of reinforcement	Cracking, change in mechanical properties and structure, decrease in protective properties of concrete layer, concrete destruction along reinforcement	Change in shape and size. Change in mechanical properties	NDI of outer surface	During TCA in scheduled outage	Strong effect	Upon results of visual and instrumented inspection, develop, if needed, a program for the next NDI and VI. Recover protective layer where damaged
7	Containment prestressing system (CPSS)	Stress corrosion cracking, general corrosion of components, local corrosion	Cracking, change in mechanical properties, thinning, pitting	Change in shape and size. Change in mechanical properties	Visual inspection. Tendon tension inspection according to established procedure	During TCA in scheduled outage	Affect in period between repairs	Upon results of inspection, develop, if needed, a program for the next NDI and VI. Optimize tendon tension. Recovery repair of irreplaceable anchor elements

Components and documents for operation		Degradation mechanisms	Ageing effect	Technical condition parameter	Methods for inspection of technical condition parameters	Frequency	Assessment of effect of technical condition parameters on component lifetime	Assessment of need for additional measures for degradation monitoring and mitigation
8	Main lock	Stress corrosion cracking, general corrosion of components, local corrosion	Cracking, change in mechanical properties, thinning, pitting	Thickness and size. Operability of actuation mechanisms	Visual inspection in 100% accessible places in scheduled outage. MHT. Inspection of actuation mechanisms	During TCA in scheduled outage	Affects	Upon results of inspection, develop, if needed, a program for the next NDI and VI. Recovery repair of metal, rotary and stop mechanisms
9	Emergency lock	Stress corrosion cracking, general corrosion of components, local corrosion	Cracking, change in mechanical properties, thinning, pitting	Thickness and size. Operability of actuation mechanisms	Visual inspection in 100% accessible places in scheduled outage. MHT. Inspection of actuation mechanisms	During TCA in scheduled outage	Affects	Upon results of inspection, develop, if needed, a program for the next NDI and VI. Recovery repair of metal, rotary and stop mechanisms
10	Operational lock	Stress corrosion cracking, general corrosion of components, local corrosion	Cracking, change in mechanical properties, thinning, pitting	Thickness and size. Operability of actuation mechanisms	Visual inspection in 100% accessible places in scheduled outage. MHT. Inspection of actuation mechanisms	During TCA in scheduled outage	Affects	Upon results of inspection, develop, if needed, a program for the next NDI and VI. Recovery repair of metal, rotary and stop mechanisms

Components and documents for operation		Degradation mechanisms	Ageing effect	Technical condition parameter	Methods for inspection of technical condition parameters	Frequency	Assessment of effect of technical condition parameters on component lifetime	Assessment of need for additional measures for degradation monitoring and mitigation
11	Handling transport lock	Stress corrosion cracking, general corrosion of components, local corrosion	Cracking, change in mechanical properties, thinning, pitting	Thickness and size. Operability of actuation mechanisms	Visual inspection in 100% accessible places in scheduled outage. MHT. Inspection of actuation mechanisms	During TCA in scheduled outage	Affects	Upon results of inspection, develop, if needed, a program for the next NDI and VI. Recovery repair of metal
12	Penetrations	Stress corrosion cracking, general corrosion of components, local corrosion	Cracking, change in mechanical properties, thinning, pitting	Wall thickness. Decrease in mechanical properties	Visual inspection in 100% accessible places in scheduled outage. MHT.	During TCA in scheduled outage	Affects	Upon results of inspection, develop, if needed, a program for the next NDI and VI. Recovery repair, recovery of paint coating where damaged

Table 6.2 List of ageing mechanisms affecting the lifetime of reactor compartment civil structures including degradation monitoring and mitigation measures (for VVER-440/213)

Components and documents for operation		**Degradation mechanisms**	**Ageing effect**	**Technical condition parameter**	**Methods for inspection of technical condition parameters**	**Frequency**	**Assessment of effect of technical condition parameters on component lifetime**	**Assessment of need for additional measures for degradation monitoring and mitigation**
1	Reactor compartment base	Change in structure and mechanical properties	Deformations and displacement. Settling	Change in profile and elevations of ground surface. Formation of dents and holes	Visual inspection	Periodically in compliance with SOU NAEK 033:2015	Insignificant effect	Upon results of visual inspection, develop, if needed, a program for the next visual inspection. Ensure soil level, structure and mechanical properties
2	Foundation plate	Leaching. Change in structure and mechanical properties. Corrosion	Cracking. Corrosion of reinforcement	Change in size. Decrease in mechanical properties. Increase in defects	Visual inspection	Periodically in compliance with SOU NAEK 033:2015	Insignificant effect	Upon results of visual inspection, develop, if needed, a program for the next visual inspection. Recovery repair, protective covering
3	Walls and coverings of foundation part	Leaching. Local corrosion	Cracking. Corrosion of reinforcement	Change in size. Decrease in mechanical properties. Increase in defects	Visual inspection	Periodically in compliance with SOU NAEK 033:2015	Insignificant effect	Upon results of visual inspection, develop, if needed, a program for the next visual inspection. Recovery repair, protective covering
4	Embedded details and metal structures for fixing of reactor equipment and piping	Local corrosion. Cyclic ageing	Thinning. Cracking. Loss of stability	Change in shape and size. Change in mechanical properties	Visual and instrumented inspection	During TCA in scheduled outage, periodically in compliance with SOU NAEK 033:2015	Possible effect (for embedded parts)	Upon results of visual inspection, develop, if needed, a program for the next visual inspection. Recovery repair, protective covering

Components and documents for operation		Degradation mechanisms	Ageing effect	Technical condition parameter	Methods for inspection of technical condition parameters	Frequency	Assessment of effect of technical condition parameters on component lifetime	Assessment of need for additional measures for degradation monitoring and mitigation
5	Sealing steel lining	Corrosion. Mechanical impact	Cracking. Mechanical damage. Local loss of stability	Change in shape and size. Thinning. Change in mechanical properties	Visual and instrumented inspection. MHT	During TCA in scheduled outage, periodically in compliance with SOU NAEK 033:2015	Affects	Upon results of visual and instrumented inspection, develop, if needed, a program for the next NDI and VI. Develop and implement a program for recovery of paint coating. Recovery repair protective coating
6	Penetrations	Corrosion. Accumulation of cyclic damage	Residual deformation. Thinning. Cracking	Wall thickness. Decrease in mechanical properties	Visual inspection in 100% accessible places. MHT	During TCA in scheduled outage, periodically in compliance with SOU NAEK 033:2015	Affects	Upon results of visual inspection, develop, if needed, a program for the next NDI and VI. Recovery repair protective coating

06.1.3 Monitoring, testing, sampling and inspection activities for concrete containment structures

06.1.3.1 Monitoring scope

Monitoring of NPP buildings and structures consists of the following components:

– information system;

– system for inspection of the technical condition, observation and diagnostics of parameters of building structures;

– methods for assessing the technical condition of building structures;

– models for predicting the technical condition of building structures and assessment of their life;

– methodology to manage the technical condition of building structures, NPP structures and buildings.

The information system ensures collection, systematization, processing, storage, access, display and dissemination of data on the state of building structures in different periods of construction and operation to the interested NPP subdivisions.

The system for inspection of the technical state, examination and diagnostics of parameters of NPP building structures is intended to receive data on actual geometric parameters, physical and mechanical properties, carrying capacity of structural components and changes in the condition of building structures.

The methods of assessing the technical condition establish the principles for comparison of controlled parameters (characteristics) with design criteria of safe operation. TCA resulted in the conclusions on possible further operation of NPP buildings and structures in normal conditions.

The models for prediction of the technical condition are intended to define life of NPP building structures. The prediction will result in the obtaining of initial data on the development of organizational and technical decisions on ensuring operational suitability of building structures and management of their technical condition, TCA of building structures for the design period and long-term operation.

The methodology for the management of the technical condition consists of a number of specific techniques aimed at maintaining the necessary operational qualities during the design life of buildings and structures.

06.1.3.2 Methodology of control and observation

The arrangement of control and observations of building structures consists of the following procedures:

– carry out periodic control of parameters of building structures;

– install systems for control of building structures that include technical means for measuring geometrical (spatial), power, vibration and other parameters (if necessary).

Periodic control of parameters of building structures consists of the following procedures:

– geodetic observations of parameters of the basis and components of buildings and structures;

– geotechnical observations of parameters of the basis (if necessary);

– hydrogeological observations;

– visual inspection of building structures;

– instrumental inspection of building structures;

– collection of data and drawing up a report on the actual state of observed building structures, on loads and operating conditions.

06.1.3.3 Structures and components for control and observation

The following structures and components of the reactor compartment shall be subject to control and observations:

– foundation and basis;
– bearing structures;
– premises of the containment safety system and in the unsealed part of the reactor compartment;
– foundation under equipment.

Parameters of building structures that shall be obligatory subject to the periodic control procedure are as follows:

– average settlement of the foundation;
– heeling of the reactor compartment building;
– uneven settlement;
– actual geometrical parameters in general;
– deformations of bearing structures of the reactor compartment building;
– physical and mechanical properties of materials of structures and components;
– parameters of cracks in reinforced concrete structures recorded during inspection.

The periodicity and parameters of building structures that are obligatory for the procedure of continuous monitoring shall be defined at the stage of monitoring procedure development depending on the peculiarities of buildings and structures, and they shall be specified after verification calculations taking into account data obtained in the instrumental and visual inspections.

The analysis of defects and damages and verification calculations define the technical condition of certain structures. According to the carrying capacity and operational characteristics, the condition of structures is recommended to be classified as one of the following conditions:

I – normal. *Structure or component is in operable condition. There are no defects and damages that prevent normal operation or reduce carrying capacity or durability.*

II – satisfactory. *Structure or component is in operable condition. There are defects and damages that can reduce the durability of the structure. Measures are needed to ensure the durability.*

III – unsuitable for operation. *Structure is overloaded or there are defects and damages that indicate a decrease of its carrying capacity. Based on verification calculations and analysis of damages, one can ensure the integrity of the structure for the period of reinforcement.*

IV – emergency. *The same as in the condition III. However, based on verification calculations and analysis of defects, one cannot ensure the integrity of the structure for the period of reinforcement, especially if the brittle nature of the destruction is possible.*

The conclusions on TCA results contain measures aimed at ensuring reliable and safe operation of building structures and components of the containment during the period of long-term operation.

Monitoring of the technical condition of building structures and components of the containment is carried out on a continuous basis and is a combination of regular technical inspections, visual and specialized instrumental observations, and continuous monitoring of parameters of the boundary conditions of building structures.

06.1.3.4 Sampling of metal and reinforced concrete metal structures (when substantiating the need)

The need for sampling, metallographic studies and chemical analysis of metal is determined by the results of visual inspection and instrumental diagnostics using NDI method in case if the results of NDI appeared to be below the design values or in case the received data do not allow reliable determination of measured parameters for the corresponding structures.

The selection of places for cutting specimens is based on the inspection results under agreement with the relevant NPP services. Tests are carried out on special facilities that have undergone metrological calibration and that have appropriate certificates of quality (see examples in Figure 6.4, Figure 6.5).

Figure 6.4 Cutting of specimens

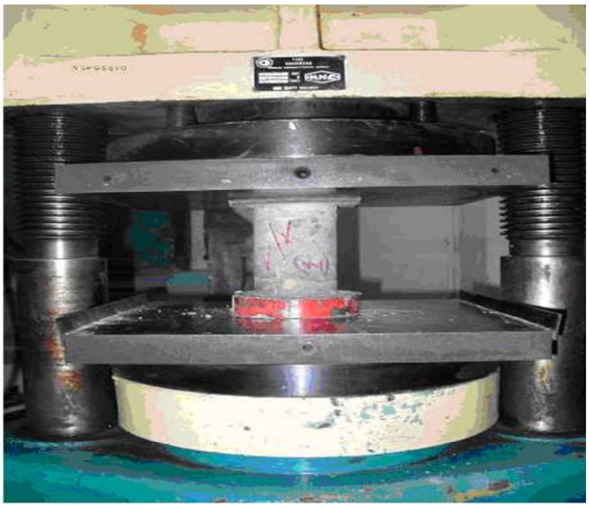

Figure 6.5 Tests

Places for cutting specimens are selected in a way that the cut sections of building structures do not weaken their carrying capacity and allow unimpeded restoration.

The surface of lining is cleaned from the anticorrosive cover in the established places. Relevant requirements for cleanliness of surface are established depending on the type of monitoring (MHT, UTM).

For preliminary assessment of the condition of concrete under the lining and in the walls of internal containment, holes are made where samples of concrete from reinforced concrete structures are taken.

Restoration of the protective layer of concrete after removal of test samples is performed by filling of formed cavities with the relevant solution or certain mixture.

Concrete mixture shall ensure defect-free filling of the cavities formed during the tests. Density and strength of “new” concrete shall be at least equivalent to the values of “old” concrete.

The restoration of the lining in places of cutting specimens is performed by a metal sheet with the preparation of sheet edges and lining for welding. Welding and control of welds are performed in accordance with PNAE G-7-010-89 [102].

Chemical analysis of metal specimens, sampling (sampling of metal chips), metallographic studies on the determination of non-metallic inclusions are carried out in accordance with requirements of relevant standards.

Preventive and corrective measures for concrete structures of the containment.

AM measures are planned on the basis of results obtained from the implementation of AMP for unit components (ageing analysis, monitoring and retention of ageing) and operational experience.

AM measures shall not duplicate procedures implemented within regular operational activities.

Measures on AM of components due to ageing can be aimed at the following aspects:

– correction of technical documents on the component;
– changes in scope and periodicity of technical inspection, repair and maintenance;
– improvement of operating system for control and diagnostics of the technical condition;
– replacement of components;
– changes in safe operational limits and conditions and introduction of relevant changes to the unit design;
– mitigation of external conditions of component operation.

Based on the performed assessment of degradation mechanisms for the basis of the reactor compartment of ZNPP-1 and ZNPP-3 and prediction for changes in the operability according to the heeling of the reactor compartment building taking into account peculiarities of degradation mechanisms for the ground basis of the reactor compartment building of ZNPP-1 and ZNPP-3, measures were implemented on the heeling stabilization. They included installation of reinforced concrete surcharging on the auxiliary building cover, which consists of:

– prefabricated reinforced concrete beams of a rectangular section with a total weight of 2287.5 t for ZNPP-1;
– prefabricated reinforced concrete beams of a rectangular section with monolithic surcharging with a total weight of 4080 t for ZNPP-3.

06.2 Outcomes of experience and reviews of ageing management for concrete containment structures

AM measures are performed for unit components included into the list of components subject to AM.

Actions on the prevention of unacceptable degradation of components due to ageing are implemented during:

– regular NPP operating activities (tests and inspections, monitoring and process parameters, replacement, etc.);
– additional (in comparison with regular) procedures on AM of power unit components.

The main task of AM is to identify and eliminate deficits in regular operating procedures in terms of their adequacy for efficient retention of component degradation due to ageing within acceptable limits.

Measures on the implementation, performance and assessment of AM program efficiency are defined in accordance with IAEA recommendations.

AM objective is to ensure retention of degradation and wearing within acceptable limits by means of the implementation of technical and operational measures to:

– maintain or improve necessary level of power unit safety and reliability through prevention of unacceptable impact of ageing on safety functions that are fulfilled by the component;

– improve economic effect from the operation of components through the creation of condition for its LTO (in particular, in beyond design-basis period).

Retention of degradation due to ageing of other power unit components is performed with the regular operating activities.

The ability of systems and components important to safety to fulfil prescribed safety functions during unit operation period taking into account impact of degradation due to ageing is performed for the power unit.

Such an assessment includes analysis of reliability of systems and components that is performed within regular operating activities, as well as additional analysis that is performed within AM.

06.3 Licensee's findings and conclusions on ageing management of concrete containment structures

Ageing management activities of containment concrete structures are performed in accordance with NPP AMP for NPP components and structures.

Revealed defects of power unit building structures are recorded. Life and repair characteristics of unit components are introduced into the relevant databases.

Analysis of degradation mechanisms of containment building structures and components is performed on the basis of results of familiarization with technical documents, visual and instrumental inspection, and verification calculations for the reactor compartment.

The following reactor compartment structures are especially subject to the ageing processes:

– reinforced concrete enclosures (RCEs);
– sealing steel lining with all fastening details and welds;
– main hatch;
– emergency hatch;
– operational hatch;
– process (tube) penetrations, emergency process penetrations;
– electrical penetrations;
– penetrations of ionization chamber channels;
– sections of piping communications, process penetrations of drain systems.

The following ageing mechanisms of containment structural components were defined as dominating:

– reduction in the values of the protective layer of reinforced concrete structures;
– strength reduction of reinforced concrete structures;
– carbonization of reinforced concrete structures;
– deformation of steel components;
– reduction in thickness of steel components;
– strength reduction of steel components.

Possible failures of structures prone to ageing shall be considered as gradual degradation processes, which appear in places of specified damages. The end stage of such failures is the reduction of cross sections of structures, reinforcing rods and the reduction of carrying capacity to boundary values.

The general factor defining the progression of degradation processes includes operating conditions, namely the temperature fluctuations of the environment and the quality of construction works.

The assessment of the condition of structures and conditions is carried out by means of visual and instrumental observations. According to observation results, measures are developed to put containment components and structures into condition that ensures durability of structures for the period of LTO by eliminating the defects and damages revealed during observations:

- current repair of defective structure or components;
- reinforcement of structural components adjoining the area of impact of the defect;
- keep operational situation without changes with the development of LTO measures (AMP).

Measures on AM of building structures and components are implemented in accordance with the approved schedules based on the results of instrumental and visual observations.

The gained experience in performing activities on TCA based on results of instrumental, visual observations and calculations of strength and carrying capacity indicates that revealed defects and damages do not affect the carrying capacity of structures. Further operation (LTO period) of building structures is allowed in design mode without any restrictions, but on condition of implementation of ageing management measures.

06.4 Regulator's assessment of ageing management of concrete containment structures and conclusions

General requirements for AM are presented in NP 306.2.141-2008 [5] and NP 306.2.210-2017 [7], and requirements for the containment are specified in the operator's main technical requirements such as Standard AMP [11], SOU NAEK 109:2016 [38], PM-T.0.03.181-13 [67], MT-T.0.03.171-14 [68], PM-T.0.03.126-14 [69] and PM-T.0.03.127-13 [70].

Measures on AM shall be defined based on continuous monitoring of the condition of building structures, results of activities on TCA and taking into account results of the research and development programs.

The list of building and structures to be included into AMP shall be defined based on the classification by safety impact taking into account data of design, engineering and operational documents. The list of such structures is made for each specific power unit and for each specific building.

Based on established requirements for AM of the containment components and structures, the operator:

- defines the list of containment components and structures subject to AM;
- establishes the list of ageing mechanisms and develops measures on monitoring and deterrence of degradation;
- develops and implements AAMS database that ensures collection, processing, storage, access and display of data on the current state of building structures;

– arranges continuous monitoring of the condition of building structures;

– implements AM measures according to the schedule.

Gained experience of conducted activities on TCA based on the results of instrumental, visual inspection and calculation of strength and carrying capacity indicates that the revealed defects and damages have no effect on the carrying capacity of the structures. Further operation (for the period of LTO) of containment structures is allowed in the design mode without restrictions, but on condition of the implementation of ageing management measures.

To ensure safe operation of the containment, the operator developed "Schedule for Implementation of Measures on Safe Operation of Containment at NPPs with VVER-1000" and agreed it with the regulator. This plan provides for the implementation of measures to 2020 and includes:

– implementation of a remote inspection system for CPSS tendon tension at NPP units;

– implementation of activities on TCA (in particular, activities on TCA and calculation justification of containment reliability to check compliance with requirements of regulatory documents (with determination of minimum allowed tendon tension)).

The operator and regulator pay special attention to calculations intended to justify containment reliability with determination of minimum allowed tendon tension since Ukrainian NPPs are currently operated at tendon tension different from the design-basis one. One of the important factors affecting the determination of tendon tension is the level of design-basis earthquake, since containment reliability calculation considers the combinations of loads as MDBA+DBE and MDBA+SSE. In this case, it is necessary to note that the seismic level of NPP sites was reevaluated over the past 10 years and new level is actually two or three times higher than the design level. Such a calculation, as a rule, is performed with activities on power unit preparation to LTO separately for each power unit, since the seismic level of sites varies and each containment has its own peculiarities, so the calculation is performed individually. Relevant measures on AM are developed according to the calculation results.

Summarizing the results based on the assessment of AM concrete structures, it is necessary to state that the system for AM and monitoring of the technical condition of NPP buildings and structures at Ukrainian NPPs makes it possible to control main parameters that ensure reliable operation of buildings and structures, which is especially relevant at LTO stage.

07 NUCLEAR RESEARCH REACTOR

According to the Technical Specification [1], information on ageing management of the VVR-M nuclear research reactor operated in Kyiv is provided below. The scope of data on NRR ageing management is smaller than that for nuclear power plants; hence all information on NRR ageing management is set forth in this section. The following aspects are considered for NRR:

- general information on NRR;
- overall AMP for NRR and its implementation;
- electrical cables;
- vessel (tank);
- protective concrete structures;
- general conclusions.

Concealed pipework (concealed in soil, concrete or closed trenches) is not used at the VVR-M NRR and thus is not considered in this report.

07.1 General information

The VVR-M water-cooled water-moderated nuclear research reactor has been in operation since 12 February 1960 in Ukraine. NRI is the VVR-M operating organization. The VVR-M lifetime is not established in the design. VVR-M has a license issued by SNRIU for reactor operation until 31 December 2023.

The NRI VVR-M nuclear research reactor (see Figure 7.1) is used for a wide range of research and applied tasks that require high neutron fluence.

Figure 7.1 Reactor general view (building, control room, reactor)

Research in the following areas of science and technology are conducted at the reactor:

- nuclear physics;
- solid state physics;
- radiation materials science;

– radiation biology;
– production of radiation sources for industry;
– production of radiation sources for medicine.

The main technical characteristics of the reactor are summarized in Table 7.1.

Table 7.1 Reactor main technical parameters

No.	Parameter	Unit	Value
1	Reactor power	MW	10
2	Maximum thermal neutron flux	n/cm^2 sec	$1.2 \cdot 10^{14}$
4	Nuclear fuel UO_2-Al: FA type	-	VVR-M2
5	Enrichment	%	19.7
6	Number of FAs (depending on core state)	pcs.	156-262
7	Water pressure after primary pumps	kgf/cm^2	≥1.5
8	Primary water temperature at core outlet	^{0}C	≤55
9	Temperature difference in the core	^{0}C	≤6.9
10	Secondary water consumption	m^3/h	≥920
11	Secondary water pressure	kgf/cm^2	2.0-4.0
12	Secondary water temperature (before heat exchangers)	^{0}C	≤35

07.2 Requirements for the overall AMP of the nuclear research reactor and its implementation

07.2.1 National regulatory and legal framework and international standards

The general approach to development of the regulatory framework for NRR is identical to that applied to NPPs. The only difference is that there are a number of regulations that address NRR specific operating features. These documents include:

– General Safety Provisions for Research Reactors (OPB IR) [80].
– PBYa-03-75. Nuclear Safety Rules for Research Reactors [81].

Regarding ageing management of NRR components as recommended by SNRIU, NRI is governed by the provisions set forth in NP 306.2.141-2008 [5] for ageing management of NPP components taking into account design features of the research reactor. The application of individual provisions from regulations and standards related to NPPs as recommending ones to NRR was also agreed with SNRIU. These are provisions of NP 306.2.099-2004 [6], PNAE G-7-008-89 [8], PNAE G-7-002-87 [9], SOU-N YaEK 1.004:2007 [10] etc.

The main documents that govern NRR ageing management are as follows:

– Ageing Management Program for Vessel (Tank) and Primary Equipment and Piping of the NRI VVR-M Nuclear Research Reactor, P-2-134-09/10/11 [82];

– Working Program for Periodic Inspection of Base Metal and Welds of VVR-M Reactor Equipment and Piping, KM2-001/4 [83];

– General Provisions on the Structure and Contents of the Report on Technical Condition Assessment and Lifetime Extension of the Vessel (Tank) with Part of Primary

Piping (Aluminum Alloy SAV-1) and Primary Piping (Stainless Steel 1X18H9T and 12X18H10T) with NRI VVR-M NRR Equipment [84].

The main international documents applied by the NRR operator in planning and implementation of ageing management activities include: NS-G-2.12 [20], SSR-3 [85], Safety Glossary [23] and SSG-10 [86].

These documents were used in the development of the "Ageing Management Program for Vessel (Tank) and Primary Equipment and Piping of the NRI VVR-M Nuclear Research Reactor, P-2-134-09/10/11" [82] (Nuclear Research Reactor AMP).

07.2.2 Description of the overall ageing management program

General NRS requirements and criteria for NRR, as well as the main technical and administrative measures aimed at meeting them as it is mentioned above are specified in documents OPB IR [80] and PBYa-03-75[81]. When addressing the issues on ageing management of NRR components by SNRIU recommendation, NRI is guided by the provisions presented in NP 306.2.141-2008 [5] taking into account design features of the research reactor.

According to the requirements of NP 306.2.141-2008 [5] in order to retain degradation of equipment, systems and components important to safety (resulting from ageing, wear, corrosion, erosion) within the established limits, NRI develops an ageing management program (AMP) and takes the necessary actions to maintain operability and reliability of equipment, systems and components during operation. Taking into account AMP, NRI determines and agrees with the SNRIU a list of equipment, systems and components to be analyzed within AMP. Equipment covered by ageing management is subject to systematic analysis of service life and reliability indexes by NRR personnel. Depending on the results of this analysis, a decision is made on the possibility of its further operation, renewal of its service life or replacement.

Taking into account the impact of ageing and degradation of NRR equipment, the operator evaluates the ability of systems and components important to safety to perform their safety functions during reactor lifetime.

The NRR AMP envisages that during operation the operator will take into account loading cycles, periodicity of scheduled outage and maintenance, scope and results of testing of equipment, systems and components important to safety, results of equipment strength calculations, etc.

AMP implementation for NRR equipment, systems and components allows efficient maintenance of NRR safety functions at the level provided by the design. In its practical activities on ageing management of components and structures important to NRR safety, NRI is also guided by document NP 306.2.210-2017 [7].

07.2.3 Ageing assessment

The NRR AMP [82] considers the requirements of such documents as: NP 306.2.141-2008 [5], NP 306.2.210-2017 [7] and Standard AMP [11] (with the consent of the SNRIU, Standard AMP for single NRR in Ukraine was not developed), as well as provisions of IAEA documents SSR-3[85], NS-G-2.12 [20], SSG-10 [86].

The basis of the ageing management system of research reactor components is based on implementing the following measures:

- develop AMP of reactor components;
- develop the list of reactor components (critical components) covered by ageing management;
- perform TCA of reactor components;

- detect and study ageing processes of reactor components;
- provide long-term operation of components;
- develop and implement measures to mitigate ageing processes;
- develop and implement monitoring of ageing processes;
- maintain reliability of components according to the requirements of technical documentation;
- compare the costs for putting components out of service and replacing them by new ones with the costs for long-term operation;
- timely replace reactor components with expired life;
- perform qualification of components;
- record and form an efficient information system of reactor component ageing management.

The following approach is used when compiling the list of components for inclusion in AMP:

a) systems containing components and structures, specific components and structures (safety class 1, 2 according to reactor Technical Safety Justification) designed to provide emergency shutdown and maintenance of the reactor in subcritical state, emergency heat release, prevent or confine spread of radioactive substances in accident beyond the limits provided in the design;

b) other systems containing components and structures, specific components and structures whose failure or damage may cause failure to perform design functions by systems, components and structures listed above (a).

According to the specified approach, the ageing management list includes the following constituents:

- List of critical components:
  - vessel (tank) with devices;
  - part of primary piping with equipment.
- list of normal operation components that do not impact on safety:
  - secondary piping and equipment.
- Additional list:
  - components of active ventilation system;
  - components of emergency automatic electric supply system;
  - electric cables (power and control);
  - building structures.

An example of the list of components and structures included in the NRR AMP is presented in Annex H.

Determination and study of the ageing mechanisms and their possible consequences in the NRR AMP [82] is based on:

- design documents taking into account rules and standards;
- materials and their properties;
- operation conditions;
- safety functions;
- manufacturing quality;
- equipment qualification;
- operation and maintenance history including inspection performed during operation, technical examination, adjustment, repair and modernization;

– generalized national and foreign operating experience of similar research reactors;
– ageing effects (potentially possible or visible for this component or structure).

According to the results of determination and study of the degradation mechanisms and their consequences, technically and economically justified measures are implemented to prevent failures of reactor components and structures caused by ageing processes of these components.

Moreover, based on studies and strength calculations performed, degradation mechanisms and ageing effects are specified, technical condition parameters and assessment criteria for them are defined.

The examples of such efforts developed in NRI are as follows:

– Report on Mechanical Tests of Specimens from Irradiated Vessel and Primary Piping of NRI VVR-M Reactor, Kyiv, 2001;
– Record of Visual Optical Corrosion Inspection of Specimens from Alloy SAV-1 of the Control Channel from the VVR-M Reactor Core Center, Kyiv, 2001;
– VVR-M Nuclear Research Reactor. Report on Inspection and Technical Condition Assessment of VVR-M NRR Civil Structures, Buildings and Facilities Important to Safety. IYaI 01.01.00.000 OT, Kyiv, 2008;
– Report “Strength Calculation of Piping and Structural Elements of NRI VVR-M NRR” (Kyiv, 2014);
– Report “Strength Calculations of the NRI VVR-M NRR Vessel (Tank) and Its Components”. Agreed by SNRIU, No. 18-19/6-3918 of 17 June 2014;
– Report of Nuclear Research Institute (Operator). Results of Additional Safety Assessments of VVR-M NRR under External Hazards That May Cause an Accident (Stress Tests) No. ZST.03-014-13/14.

Concerning acceptance criteria, NRI demonstrated in the AMP implementation process that:

– practical actions are taken on timely detection of ageing effects, forecasting their development and retaining degradation;
– current parameters of technical condition of components and structures do not exceed the acceptance criteria established for them;
– components and structures are able to perform design functions within a justified period of time based on the analysis determining service life.

These acceptance criteria are established based on the following:

– analyses of technical documents on reliability, operation conditions, repair and maintenance of components and structures;
– established parameters and criteria of technical condition;
– analysis of failures and damages;
– established ageing mechanisms;
– results of in-service and other inspections.

The AMP for NRR components, systems and equipment is revised based on operating experience feedback and results of self-assessments or expert judgement. Generalized national and foreign NRR operating experience is considered in the determination and study of degradation mechanisms.

During VVR-M NRR operation, NRI uses operating experience of similar NRRs located in the territory of the Russian Federation, and to prevent faults, performs preventive measures recognized acceptable by the SNRIU, namely:

– revise operating manuals of systems and components taking into account new requirements and operating experience;

– provide modernization and repair of NRR systems and components important to safety;

– implement additional measures on the inspection of systems important to safety, etc.

Based on the operating experience the list of components and structures subject to ageing management may be revised. The information on operating experience is systematically collected at the reactor, and its possible application for NRR is also assessed.

07.2.4 Monitoring, testing, sampling and inspection activities

In order to monitor technical condition of reactor components and structures and assess ageing impact, the programs of in-service inspection, maintenance and repair of components and structures, test and inspection programs are developed and systematically optimized. For example, in recent years, a number of program documents were developed, according to which technical condition was monitored and the results were verified during inspections. For example the following documents:

– Working Program to Justify further Safe Operation of Vessel (Tank), Primary Piping and Equipment of VVR-M Research Reactor, Kyiv, 2001, agreed by the Ministry of Ecology, No. 13/2-11/141 of 16 February 2001;

– Report on Mechanical Tests of Specimens from Irradiated Vessel and Primary Piping of NRI VVR-M Reactor, Kyiv, 2001;

– Record of Visual Optical Corrosion Inspection of Specimens from Alloy SAV-1 of the Control Channel from the VVR-M Reactor Core Center, Kyiv, 2001;

– Working Program for Periodic Inspection of Metal and Welds of VVR-M Equipment and Piping (KM2-001/3), Kyiv, 2003;

– Working Program for Periodic Inspection of Metal and Welds of VVR-M Equipment and Piping (KM2-001/3), Kyiv, 2003, etc.

According to the results of in-service and other inspections of reactor components and structures, in-service inspection programs are amended or, if necessary, new programs are developed.

The results of technical condition assessment of primary piping, heat exchangers and pumps are presented in the Report “Technical Condition Assessment and Determination of Further Lifetime of Primary Equipment and Piping of NRI VVR-M Nuclear Research Reactor” No. ZAB.04-069-14 and demonstrate the following:

– completeness, composition and content of the technical documents is adequate to assess technical condition of reactor primary components;

– abnormal operation specified in the design documents is not registered in the technical specifications and technical characteristics of piping, pumps, gates and heat exchangers;

– maintenance and in-service inspection of NRI VVR-M NRR primary piping and equipment was performed in compliance with the regulatory and technical documents.

SNRIU permanently supervises NRI ageing management activities during inspections.

07.2.5 Preventive and remedial actions

Ageing management of NRR components and structures included in the AMP list is carried out all operating stages. In addition, the following activities are envisaged:

– identify and analyze ageing processes for components and structures;

– implement measures to monitor ageing processes for components and structures (ageing monitoring);

– implement measures to mitigate degradation;

– analyze the lifetime and reliability indicators of components, systems and equipment;

– improve AMP based on operating experience feedback and research, as well as results of self-assessment and expert judgement (development and implementation of additional degradation monitoring and mitigation measures);

– record activities and develop an effective information system for ageing management of reactor components and structures.

Preventive and remedial actions are developed upon detection of degradation mechanisms and identification of ageing effects. If such actions are not feasible, a component or structure is to be replaced.

There are the following examples of remedial and preventive actions::

1. For primary equipment and piping of VVR-M NRR, strength calculations were performed in compliance with PNAE G-7-002-87 [9] and PNAE G-5-006-87 [103] and set forth in the "Report. Strength Calculation of Primary Piping and Structural Elements of NRI VVR-M Nuclear Research Reactor", considering new piping sections. These calculations indicate that these reactor components meet strength conditions and that there are adequate strength margins for reactor safe LTO.

2. Individual sections of primary system were replaced and a defect (crack) that occurred in the primary piping pressure side flange was eliminated. The replacement was carried out in compliance with the procedure for modifications at NPPs. The piping sections were introduced into commercial operation (Technical Decision TR3.5-031-14, agreed by SNRIU with letter No. 17-18/5-7270 of 5 November 2014).

3. At NRI VVR-M NRR, heat exchangers and their piping, gate vales and individual primary piping sections were replaced.

4. Primary piping and equipment:
   - primary coolant treatment system was upgraded, thermal oxide filter was introduced, ion-exchange filters were upgraded (1968-1971);
   - heat exchangers and part of stainless steel piping (heat exchanger piping) were replaced with new ones (1989);
   - upgraded reactor emergency cooldown system was commissioned (1989);
   - piping sections made of aluminum alloy SAV-1 were replaced by sections of aluminum alloy AMG-3 (2014);
   - gate valves that isolated the vessel (tank) from primary piping were replaced by new ones of the same type.

5. Reactor power supply system:
   - emergency storage battery was replaced by a new one (2011);
   - diesel generator station (120 kW) was introduced as an auxiliary emergency power supply source (1997) and diesel generator station (42 kVA) as a standby emergency source (2011);
   - emergency steam generators were replaced by new ones (energy converters AB 110 V to 0.4 kV 50 Hz (2009);

- electrical boards were replaced by cabinets with new automatic circuit breakers, starters, relays (2007);
- power and control cables were replaced with new ones with flame retardant insulation (2002-2007) etc.

07.2.6 Analysis and update of AMP

AMP is analyzed and updated in accordance with advances in science and technology, operating experience, results of ageing management activities, as well as safety review and self-assessment of the operator.

Regarding the nuclear research reactor, the NRI provides periodic assessment of ageing management activities using the following indicators:

– reactor management's self-assessment;

– independent external assessment of ageing management activities and associated programs;

– assessment of ageing management activities within NRR periodic safety review.

The NRR AMP contains a list of components and structures subject to LTO. LTO of components and structures important to safety is determined in a technical decision agreed with SNRIU. A decision on possible LTO of a specific component is made upon positive TCA results, operating experience and TLAA. The TCA results are incorporated in SAR, which is a mandatory annex to the technical decision on LTO of a specific component.

AMP is appropriately amended if there are changes in operating parameters, unforeseen deviations of operating parameters from those established in NRS regulations and standards or in design documentation or identification of new ageing mechanisms and degradation effects.

07.2.7 Summary of AM processes and licensee's main conclusions

Ageing management of components and structures has been actually underway since the beginning of NRR operation.

AMP of NRR components and structures is based on regulations, standards and industry documents developed for NPPs but with account for design features of the research reactor. This approach is agreed with SNRIU because no special regulations were developed for the only research reactor in Ukraine.

According to AMP, ageing management activities are entrusted by NRI management to personnel involved in operation of NRR components, systems and equipment.

AMP describes the procedure for selecting components and structures for ageing management and establishes requirements for timely implementation of preventive and remedial actions to mitigate degradation.

Assessment of AMP efficiency, self-assessment of NRR management, and independent outer assessment of the operator's ageing management activities are carried out on a permanent basis.

The results of ageing management activities are recorded in compliance with NRI procedure.

According to the results of independent external assessment of NRI, no significant comments were made on AMP of NRR components and equipment.

07.3 Electrical cables

Electrical cables (power and control) are intended to ensure reliable power supply and monitoring of parameters and automatics of systems important to safety, such as:

- power supply system for main reactor loads: engines of pumps and gate valves of primary and secondary systems, engines and gate valves of reactor emergency cooldown system, electric engines of special ventilation system fans, and main and emergency lighting of reactor rooms;
- reactor instrumentation and control system;
- reactor radiation monitoring and protection system;
- automatic fire alarm system;
- reactor physical protection system.

The main objective of ageing management of cables is to ensure their reliable functions in normal operation and, in particular, in emergencies.

07.3.1 Scope of ageing management for electrical cables

The ageing mechanisms of cables are mainly determined by types of insulation materials and damaging factors for cables. The main damaging factors at NRR are electrical, mechanical and radiation. These factors may affect cables separately or in combination. This may cause the following ageing mechanisms:

- thermal degradation of core insulation and sheath through ohmic heating of cores (for power cables);
- loss of surface insulation properties under the action of electric field;
- accidental physical damage in the operating process.

Changes in cable insulation properties that may be revealed in cable in-service inspection are as follows:

- change in the outer sheath;
- decrease in insulation electrical resistance;
- increase in dielectric loss tangent.

As distinct from NPPs, thermal damaging factors are not considered for NRR because there are no conditions at the research reactor that may lead to increase of power or control cables above 40°C, which does not exceed the certificate value.

All power and control cables of normal operation systems important to safety are included in AMP. In addition, all these cables were replaced by new ones from 1997 to 2012. Their life is 25-30 years according to technical specifications.

Replacement of components and upgrading of the reactor emergency power supply system allowed the cables and emergency power supply system to be brought into compliance with current regulations, standards and rules. Therefore, maintenance and monitoring are the main ageing management measures for cables.

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

Maintenance and monitoring are conducted periodically in compliance with annual schedules developed on the basis of rules for technical operation of electrical facilities.

Operating conditions of cable at NRR are monitored in compliance with working programs for monitoring of cable operating conditions on a permanent basis with involvement of a specialized organization (after replacement of old cables with new ones). Further activities are carried out by reactor personnel.

The monitoring objective is to determine actual conditions for operation of electrical cables.

External factors of influence on cables during operation of NRR, such as air temperature and ionizing radiation in the room where the cables are located, are constantly monitored by regular measuring devices. Measurement results are recorded in the computer memory.

Maintenance includes visual inspection and preventive maintenance and testing. Power cables are subject to inspection every three months. Preventive maintenance and testing of power cable lines are carried out once a year, and those of electrical valves and other equipment every six years.

07.3.3 Preventive and remedial actions for electrical cables

Ageing management of cable includes the development and implementation of the following measures:

- regular monitoring;
- technical condition assessment of cables immediately after periodic maintenance.

Based on analysis of technical condition of electrical cables and considering SNRIU recommendations, all electrical cables (power and control) were replaced by new ones in 1997-2007.

Reliability of electric cables was improved through use of flame retardant insulation (power cables of VVGng type and control cables of KVVGng type) and they were laid in compliance with nuclear and fire safety requirements.

Operational reliability of cables is analyzed using statistical methods in accordance with maintenance rules and including the following activities:

- collection of information on failure of cables in the operating process;
- assessment of operational reliability indicators for cables based on processing of statistical data and comparison of these indicators with reliability indicators provided in TS for cables;
- assessment of failure flow trend;
- analysis of cable failure causes and development of recommendations for their decrease.

Upon operational reliability analysis, a report is developed to provide conclusions and recommendations based on assessment of activities. Assessments of failure flow trends show that cable reliability does not decrease, which is evidence of high operational reliability of NRR cables and possibility of their further operation.

07.3.4 Licensee's findings and conclusions on ageing management of electrical cables

Ageing management measures on NRR electrical cables are carried out in compliance with regulatory and working documents under regular operational activities and during TCA of cables.

All power and control cables of normal operation systems important to safety are included into AMP. In addition, all these cables were replaced by new ones from 1997 to 2012. The lifetime of these cables is 25-30 years according to technical specifications.

Replacement of components and upgrading of the reactor emergency power supply system allowed the cables and emergency power supply system to be brought into compliance with current regulations, standards and rules. Therefore, maintenance and monitoring are the main ageing management measures for cables.

07.4 VVR-M nuclear research reactor vessel (tank)

The description of vessel main structures and component is provided in Annex G.

07.4.1 Scope of ageing management

The main factors of RPV metal ageing results from operating conditions. These conditions for the VVR-M vessel (tank) are as follows:

- radiation (neutrons of different energies, gamma and other types of reactor radiation);
- cyclic stress from temperature differences and change in hydraulic pressure (in connection and disconnection of primary water pumps);
- mechanical load on vessel components;
- corrosion wear.

Considering these factors, AMP addresses metal ageing (alloy SAV-1) for vessel (tank) components such as support grid and bottoms of nine HTCs subjected to the highest neutron flow during reactor operation at rated power.

07.4.2 Assessment of reactor pressure vessel (tank) ageing

In radiation damage to aluminum, the main role is played by transmutation products, the most important of which is silicon. Transmutation silicon occurs due to the reaction with thermal neutrons.

X-ray analysis of changes in the parameters of the crystallographic lattice indicate almost double increase in the concentration of silicon in the alloy (~0.7 wt.% of transmutation silicon at fluence ~ 3.6×10^{26} neutrons$\times$m^{-2}).

NRI tested samples of aluminum alloy SAV-1 made of the change of ones of the control rods that was located at the core center for 27 years and had fluence 2×10^{26} neutrons$\times$m^{-2} (neutron energy $E\geq1.2$ MeV). The data show that neutron irradiation of alloy SAV-1 leads to its strengthening, resulting in increase in cyclic strength, but decreases is ductility.

At the same time, the maximum neutron fluence accumulated at HTC bottoms is 11.2×10^{25} neutrons$\times$m^{-2}, and fluence on the support grid is 1.643×10^{25} neutrons$\times$m^{-2}. Experimental data show (Figure 7.2) that elongation for alloy SAV-1 at HTC bottoms is ~ 2%. Considering that the vessel (tank) design and operating conditions make impact loads on HTC bottoms impossible, decrease in ductility does not limit further operation of the NRI VVR-M reactor.

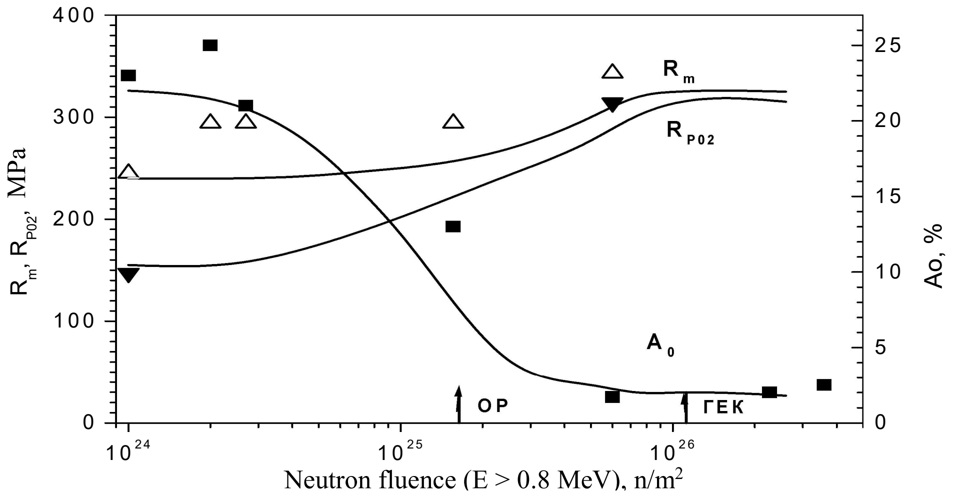

Figure 7.2 Change in mechanical properties of alloy SAV-1 depending on neutron fluence

The results of NRI research are marked as follows:

□ – ultimate strength R_m; ▼ – yield stress R_{P02}; ■– elongation A_0

The above data on SAV-1 ageing resulting from neutron irradiation of the vessel metal indicate that the vessel has strength margin sufficient for its further operation (SAV-1 strength is sufficient up to fluence 3.6×10^{26} n/m^2).

Another important ageing factor is cyclic loads on vessel metal resulting from temperature differences, mechanical loads on support grind because of FA weight and hydraulic pressure in operation of primary pumps.

In operation of primary pumps, water pressure under the grid is 0, so pressure difference is $\Delta P \leq 0.5$ kgf/cm^2 (water column height in reactor tank). The weight of all FAs ~ 230 kg and the weight of all experimental devices on the support grid ~ 130 kg.

Cyclic stresses in the support grid and HTC bottoms are due to changes in hydraulic cycles or pressure difference and are below the endurance limits, and loads in the operational years are low-cycle ones.

Concentration of the mechanical stress on a limited area or individual section may results from impact (dynamic) load, for example, impact of the reloading machine mast. This impact is almost impossible in reactor normal operation: there are FAs in the core and, hence, access to the support grid and HTC bottoms is impossible. Impact on the HTC bottom form the external side is excluded by engineered safety features (mast length is limited).

One of the important factors of vessel ageing is corrosion of SAV-1 metal. SAV-1 surface wear (corrosion, erosion) is regularly inspected at the reactor using different methods. During two years of operation, accessible vessel areas were visually inspected, tank walls were subjected to ultrasonic thickness measurement and surveillance specimens were examined (the surveillance specimens were made of a control rod channel produced from alloy SAV-1 and located at the core center). Optical visual inspection of the samples made of the control channel (control rod) revealed no signs of corrosion. Upon the inspection, it was concluded that the average corrosion of the vessel and reactor components made of alloy SAV-1 did not exceed 22±5 μm and the maximum corrosion did not exceed 37 ±5 μm during operation of the VVR-M reactor. The thickness of tank wall is 16 mm, bottom is 20 mm, and HTC bottoms is 13 mm.

07.4.3 Monitoring, testing, sampling and inspection activities for reactor pressure vessel

The VVR-M NRR was constructed taking into account rules and standards that were valid at the time of construction and commissioning in 1960. The loading of surveillance specimens was not envisaged by regulations in force at that time. Samples made of one of the control rod channels were used as surveillance specimens for VVR-M NRR-M. Channel material is SAV-1 and it was located in the core center, where the density of the neutron flux (in particular, fast neutrons) is much higher than in the location of HTC bottoms and support grid. Strength of reactor pressure vessel (tank) and its components is confirmed by calculations, which used mechanical properties of samples of SAV-1 irradiated to fluence 3.6×1026 n/m^2.

The operation of the research reactor includes the following:

- maintenance of the reactor pressure vessel (tank);
- monitoring of neutron fluence on support grid and HTC bottom;
- accounting of fluence cycles;
- compliance of the water chemistry with the established standards.

Maintenance of the reactor pressure vessel (tank) is performed in accordance with requirements of current rules and regulations: technical examination is performed to implement activities on control of metal, visual inspection and hydraulic tests. Control of metal is performed in accordance with the specially developed program.

The coolant (distilled water) in the reactor pressure vessel (tank) is under pressure of the height of water column in it – ≤5.1 m. Hydraulic tests are carried out by the method of full filling of the reactor pressure vessel (tank) by water with keeping of it during a certain period of time (the period is defined by the hydraulic test program), during which leakage of water is not allowed.

Neutron fluence is defined according to the reactor energy production, which is recorded in the relevant technical documents (operational log, energy production log).

The cycles of neutron fluence on the reactor pressure vessel (tank) and its components are recorded by reactor personnel in the technical documents.

The chemical composition of the coolant is regulated by the relevant standards and procedures. Water performance is monitored by reactor personnel: values of pH indicators and specific electrical conductivity are measured by devices during the reactor power operation. All the other performance indicators are measured by means of chemical analysis of water samples that are taken from the primary system once in a week.

Continuous monitoring of ageing processes is conducted for reactor. There are also periodic TCA of the reactor pressure vessel (tank), according to which the long-term operation may be ensured. The reactor has complete information on the operating conditions, maintenance and repair of reactor pressure vessel (tank) components. Based on such data, considering mechanical peculiarities of the irradiated SAV-1 alloy, there were relevant TLAA for reactor pressure vessel (tank) and its components. Positive results of strength calculation made it possible to ensure LTO of the reactor pressure vessel (tank).

07.4.4 Preventive and remedial actions for RPV

It is not possible to stop the ageing process. The operator's main objective is to create conditions for the tank metal to be functional as long as possible – tank technical parameters to be within design values. Such conditions envisage strict compliance with reactor vessel (tank) maintenance schedules, compliance of primary circuit water chemistry, temperature and flow with technical specification requirements, monitoring of the support grid mechanical loading value and prevention of the support grid mechanical loading value to be more than the value specified by design documentation, and prevention of the accidental load on HTC bottoms both from the core side and from the inside of “cylinders” of each of the nine HTCs.

In reactor normal power operation it is not possible to reduce the value of reactor vessel (tank) radiation load, that is why the objective of the operational personnel is to control and fix the neutron fluence value (SAV-1 fluence safety margin $>3.6\times10^{26}$ n/m^2) and to carry out periodic visual inspection of accessible spaces of tank walls (from inside), vessel bottom, HTC bottoms and support grid. The fluence is specified on the basis of reactor generated power that is recorded in corresponding technical documentation (operations log, generated power log) and visual inspection is carried out once in two years in compliance with the schedule.

The support grid mechanical loading value (mass of FAs, movable beryllium blocks, experimental devices) is controlled by the operational personnel and does not exceed the value that makes 320-360 kg, accidental mechanical loads on the support grid and HTC bottoms are not possible provided that it is filled with nuclear fuel. In periodic inspection of the support grid and at ultrasonic wall thickness test, FAs are unloaded from the core and placed to the spent fuel storage facility, which increase the danger of support grid and HTC bottoms mechanical damage. Currently, additional organizational measures are taken to prevent mechanical damage of “bared” HTC bottoms and support grid. Such measures include minimization of works inside the core and if the work is planned it is carried out by corresponding personnel with not less than 3 years of operational experience.

07.4.5 Licensee’s findings and conclusions on ageing management of RPV (tank)

Reactor vessel (tank) is unreplaceable element that identifies the possibility of further operation of the reactor as a whole.

Ageing management of the vessel (tank) is paid special attention over the entire reactor lifetime.

Owing to the operator’s measures to mitigate ageing of the vessel (tank), such as assurance of strict compliance with maintenance schedules, compliance of the primary circuit water chemistry, temperature and flow with technical specification requirements, monitoring and prevention of the support grid mechanical loading value to be more than the value specified by design documentation, the vessel (tank) is in operable condition.

The positive results of ageing management activities carried out during NRR operation, including inspection of reactor vessel (tank) metal (technical inspection, periodic metal inspection and nondestructive testing, definition of metal corrosion, integrity testing) and the positive results of strength calculations of the vessel (tank) and its components using results of mechanical tests of irradiated metal samples allowed the operator to make a decision (in 2014) on LTO of the reactor vessel (tank) until 31 December 2023. This technical decision was agreed with the SNRIU.

07.5 Concrete containment structures

Ageing management of civil structures of containment building, facilities and elements is carried out considering certain ageing mechanisms and is characterized by the peculiarities of the collection of the data on their technical condition.

07.5.1 Scope of ageing management for concrete containment structures

The ageing management scope includes inspection and specification of the technical condition of the elements of civil structures. These measures are included into the NRR AMP [82].

Concrete structure ageing management is carried out considering certain ageing mechanisms for civil structures. Review and assessment of technical condition of civil structures is carried out in compliance with general building regulations and rules considering NRS regulations with periodicity that makes two times a year.

07.5.2 Monitoring, testing, sampling and inspection activities for concrete structures of the nuclear research reactor containment

Monitoring of civil structures of NRR buildings and facilities is one of the regulated procedures that include inspection of civil structures of NRR buildings and facilities, and also the evaluation of the possibility of intended usage of structures under design conditions and during specified long-term operation period.

Monitoring of NRR buildings and facilities includes the following constituents:

- information system;
- inspection system for technical condition of civil structures;
- methods of TCA of civil structures and assessment of their lifetime.

The system for technical condition inspection of NRR buildings and civil structures, parameter monitoring and diagnostics is designed to obtain the data on actual geometry, physical and mechanical characteristics and capabilities of structural elements to withstand the loads and changes of the state of civil structures.

TCA methods set the principles of comparison of controlled parameters (characteristics) with regulatory criteria of safe operation. The conclusion on the possibility of further operation of NRR buildings and facilities is the result of TCA.

Monitoring and observation of civil structures of NRR buildings and facilities include periodic control of parameters of civil structures of buildings and facilities. Periodic control of parameters of civil structures of buildings and facilities include the following procedures:

- geodesic observations of the parameters of basements and structures of buildings and facilities;
- hydrogeological observations;
- visual inspection of structures;
- instrumental inspection of structures;
- data collection and preparation of the report on actual state of examined structures, loads and operation conditions.

Concrete structure basement and load-bearing structures undergo monitoring and observation.

Monitoring of the technical condition of containment structures and elements is carried out permanently and represent a combination of the regulatory technical inspections, visual and specific instrumental inspections and also permanent monitoring of the boundary condition of parameters of structural elements.

07.5.3 Preventive and remedial actions for concrete containment structures

In the operational years since 1960, inspections, TCA, strength and seismic calculations, etc. were conducted. In particular, the following activities were carried out:

- the main reactor building was inspected and a technical certificate was drawn up (1981);
- additional analyses of geotectonic conditions and seismic safety of the NRI VVR-M research reactor (1992) were carried out. It was concluded that the reactor site belongs to the seismic area with design-basis earthquake of 5 magnitudes and safety shutdown earthquake of 6 magnitudes;
- strength calculations of VVR-M civil structures under seismic impacts were performed (Kyiv Research and Design Institute “Energoproject”: Report “Strength Calculation of Reactor Building Structures under Seismic Impacts”, 1996);

– in 2008, inspection and TCA of VVR-M civil structures, buildings and facilities important to safety were carried out. The inspection was conducted in compliance with the working program agreed by SNRIU (with letter No. 15-18/06-6491 of 13 November 2008): “NRI VVR-M. Working Program for Inspection and Technical Condition Assessment of NRI VVR-M NRR Civil Structures, Buildings and Facilities Important to Safety. IYaI01.01.00.000RP, 2008”;

– activities were carried out to calculate the strength, bearing capacity and operability of civil structures, buildings and facilities of VVR-M NRR important to safety (2008, Association “Reliability of Machines and Buildings”).

For inspections, strength calculations and technical condition assessment of civil structures, foreign specialized organizations are involved. They have the right to carry out such activities in compliance with NRS regulations and standards. The frequency of such inspections is every 10 years. For these inspections, a special working program is developed and agreed with SNRIU. The results of periodic inspection and TCA are recorded in certificates drawn up by the operator’s technical commission.

07.5.4 Licensee’s findings and conclusions on ageing management of concrete structures of the nuclear research reactor containment

Ageing management activities of NRR containment structures are carried out according to AMP [82] existing at NRI (a special AMP for NRR civil structures was not developed).

The main scope of activities on ageing management of reactor components and structures is carried out under regular operations and during TCA of structures and buildings.

During VVR-M NRR operation (since 1960), no failures have been revealed in structures or facilities.

07.6 Regulator’s assessment of ageing management and conclusions

Operation of the NRI VVR-M research reactor was started on 12 February 1960. The design did not establish its lifetime, and not the research reactor is in LTO. NRI has a license issued by SNRIU for VVR-M NRR operation until 31 December 2023. The possibility of NRR operation after 2023 will be decided by SNRIU upon the operator’s application, based on periodic safety review and AMP implementation. Preliminary safety review was carried out in 2008-2013. In 2005-2008, the operator upgraded individual systems and replaced some equipment with new one to bring the reactor compartment and systems into compliance with current safety requirements:

– heat exchangers and part of the primary and secondary reactor coolant systems were replaced;

– CPS and I&C were replaced by hardware and software for automatic control, instrumentation and protection;

– emergency control board was introduced;

– power supply system and emergency power supply systems were upgraded: emergency generators and their control equipment, control equipment for electric engines of primary and secondary pumps, electrically driven gate valves of primary and secondary systems, fans and electric valves of special ventilation system, and cooling tower fans were replaced;

– power and control cables were replaced with copper cables whole isolation is flame retardant (VVGng type), radiation monitoring equipment was replaced with equipment based on automated radiation monitoring devices AKRB-06 etc.

Based on periodic safety review, AMP was updated [82]. AMP implementation is under strict supervision of SNRIU and AMP results are analyzed in NRR inspections by SNRIU.

Ageing management has been introduced into the NRR life cycle of operation (ageing management at the NRR decommissioning stage will be decided by the operator).

The main scope of ageing management activities on NRR components and structures is carried out under regular operations and during TCA of components and structures.

SNRIU takes active part in the WENRA topical working group for development of reference levels for research reactors. It is planned to review and update the regulatory requirements for research reactors considering the WENRA reference levels, IAEA standards and operating experience.

08 OVERALL ASSESSMENT AND GENERAL CONCLUSIONS

Ageing management is considered to be one of the main safety factors for design-basis and long-term operation of nuclear installations. The design-basis life has expired or is going to be completed for most units of Ukrainian NPPs. The Energy Strategy of Ukraine until 2035 “Safety, Energy Efficiency, Competitiveness” identifies long-term operation of power units based on periodic safety review of one of the priority areas in nuclear energy development.

Seven units of Ukrainian NPPs have already obtained licenses for LTO, the operator plans to obtain LTO licenses for five more units in 2018-2020.

The main objective of ageing management is to ensure safe and effective operation through implementation of technically and economically feasible measures and upgrades intended to detect degradation of power unit components caused by ageing in a timely manner and keep it within acceptable limits.

The term of ageing management was introduced into operational practice at the beginning of the 2000s, but ageing management components were implemented since the beginning of power unit operation:

- timely maintenance;
- identification of significant degradation and implementation of compensatory measures;
- equipment upgrading and replacement;
- change in operational modes (if necessary).

The national regulatory framework on ageing management began its development to implement measures under Cabinet Resolution No. 263-r of 29 April 2004 “On Approval of the Comprehensive Program of Activities for Long-Term Operation of Nuclear Power Plants”.

Once of the main principles of Ukraine’s regulatory control is a systematic hierarchic approach to the development and revision of regulatory documents. In practice, this principle is implemented through development of a hierarchic pyramid of NRS regulations, which includes documents of several levels, from legal regulations to detailed technical standards.

The main documents that establish requirements for ageing management are currently as follows:

- regulator’s documents on nuclear and radiation safety:
  - ✓ General Safety Requirements for Nuclear Power Plants (NP 306.2.141-2008) [5];
  - ✓ General Requirements for Ageing Management of Components and Structures and Long-Term Operation of NPP Units (NP 306.2.210-2017) [7];
  - ✓ General Requirements for NPP Long-Term Operation Resulting from Periodic Safety Review (NP 306.2.099-2004) [6];
  - ✓ Requirements for the Structure and Contents of Periodic Safety Review of Operating NPPs (SOU-N YaEK 1.004.2007) [10].
- operator’s documents agreed with the regulator:
  - ✓ Standard Program for Ageing Management of NPP Components and Structures (PM-D.0.03.222-14) [11];
  - ✓ Cable Ageing Management Program for Nuclear Power Plants (PM-T.0.08.121-14) [12];
  - ✓ Provisions on the Procedure for Long-Term Operation of Equipment in Systems Important to Safety (PL-D.0.03.126-10) [13];

- ✓ Operation of Technological System. Long-Term Operation of NPP Units. General Provisions (SOU NAEK 080:2014) [14];
- ✓ Operation of Technological System. Monitoring of NPP Structures. General Provisions (SOU NAEK 109:2016) [38];
- ✓ Engineering, Scientific and Technical Support. Ageing Management of NPP Components and Structures. General Requirements (SOU NAEK 141:2017) [75].

The above documents were developed and are periodically revised to incorporate international experience, practices and recommendations of international organizations.

In the development of basic regulatory documents, detailed analysis for compliance with international experience and practices is carried out. In some cases, the regulator and operator carry out this activity within international assistance to Ukraine for harmonization of the national regulatory framework with European Union requirements and IAEA recommendations. The results of this activity indicate that Ukrainian standards and regulations that govern principle aspects of ageing management have been developed considering IAEA and WENRA recommendations and advanced international experience. A modern regulatory framework has been developed to conduct ageing management activities at a proper international level. The regulator and operator continuously develop new regulations and improve the existing documents.

Ageing management is conducted on a systematic basis. For this purpose, respective subdivisions have been established at all NPPs and provided with sufficient competent personnel with required authorities and resources.

Two types of ageing management are identified for components and structures: physical ageing and obsolescence. Management of physical ageing that leads to degradation is based on the understanding of ageing effects and prediction of degradation for components and structures and is arranged as follows: detection of degradation mechanism – identification of ageing effect – location of ageing effect on components – methods and means to monitor degradation – analysis of monitoring results – measures to mitigate/limit degradation – analysis of AMP effectiveness.

Management of obsolescence is based on the development of knowledge, technologies and amendments of national requirements and standards and implementation of new technical decisions.

Analyses carried out to assess and predict ageing effects for components and structures are based on conservative prediction and operating experience (to compensate uncertainty of input information).

AMP and ageing management database are developed for each power unit. In operation of power units, ageing management of components and structures is carried out though coordination of AMP with existing programs, such as safety improvement, in-service inspection, maintenance, surveillance, water chemistry, inspection and testing and equipment qualification programs. Results of the analysis that determine the lifetime and information on operating experience are considered as well.

The operator developed reports on AMP implementation for each power unit and submit them to SNRIU. AMP development and implementation are necessary conditions for LTO.

Administrative and technical ageing management activities carried out at Ukrainian NPPs comply with NRS regulations, standards and rules and ensure effective implementation of ageing management tasks.

According to the Technical Specification [1], the following components were selected for detailed analysis of the existing ageing management process considering the features of power units operated in Ukraine:

– electrical cables;
– concealed pipework;
– reactor pressure vessel;
– containment concrete structures.

The NRR ageing management activities are based on approaches identified for NPPs considering the NRR design features and graded approach.

Results of the analysis provided in sections 02-07 of this report show that requirements of national regulations, rules and standards on NRS are of proper level.

Considering analysis of ageing management at NPPs and NRR, the following can be concluded:

1) Existing Ukrainian regulatory and legal framework on ageing management is of the level that complies with IAEA and WENRA documents and safety recommendations. This was confirmed by independent analyses carried out by Western experts within international projects.

2) Ageing management is carried out on a systematic basis and properly recorded with inclusion of data into electronic databases.

3) Approach to ageing management is based on the understanding of ageing effects of prediction of degradation for components and structures.

4) AMP development and implementation are necessary conditions for LTO of power units.

5) Safety factor of ageing is a part of the Periodic Safety Review Report in compliance with IAEA standards.

The following potential good practices have been identified:

1) Accumulation and summary of ageing management experience in the ageing management information analysis system (AMIAS).

2) Implementation of performance indicators to assess effectiveness of the ageing management process.

3) Inclusion of ageing management measures into the program documents approved by the Ukrainian Government.

At the same time, potential areas for further activities have been determined:

1) Improvement of the regulatory and legal framework to consider new IAEA and WENRA documents and best experience and practices, develop regulations for nuclear research reactors, including that within cooperation with international organizations.

2) Improvement of TCA and LTO of RPVs through extension of representative data based on tests of surveillance specimens.

3) Involvement of international peer reviews, in particular, IAEA SALTO mission.

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ANNEX A

Information on piping of the essential service water system

The service water system of the reactor compartment is designed for core heat removal to the ultimate heat sink, heat removal from the spent fuel pool, intermediate circuit, a number of ventilation systems and coolant of safety system mechanisms. The system is important to safety and combines the functions of the safety system and the system of normal operation. The system is designed according to the following requirements:

– the system fulfills its functions in any emergency that occurs in the conditions of NPP total blackout;

– the system has a three channel structure that corresponds to the structure of process safety systems;

– the system can be controlled and tested in any modes of normal operation without disturbing its functional properties;

– the system ensures a cooling water temperature in the range of 5-33 °C in all modes of operation.

The cooling system is circulating, isolated from external water reservoirs, groundwater and other water supply systems with cooling of service water in the spray ponds. Each NPP unit envisages three independent channels of reactor service water supply, which provide power unit operation in the rated mode, scheduled outage and also in emergencies.

The layout of underground ESWS piping on an example of ZNPP-4 is presented in Figure A.1.

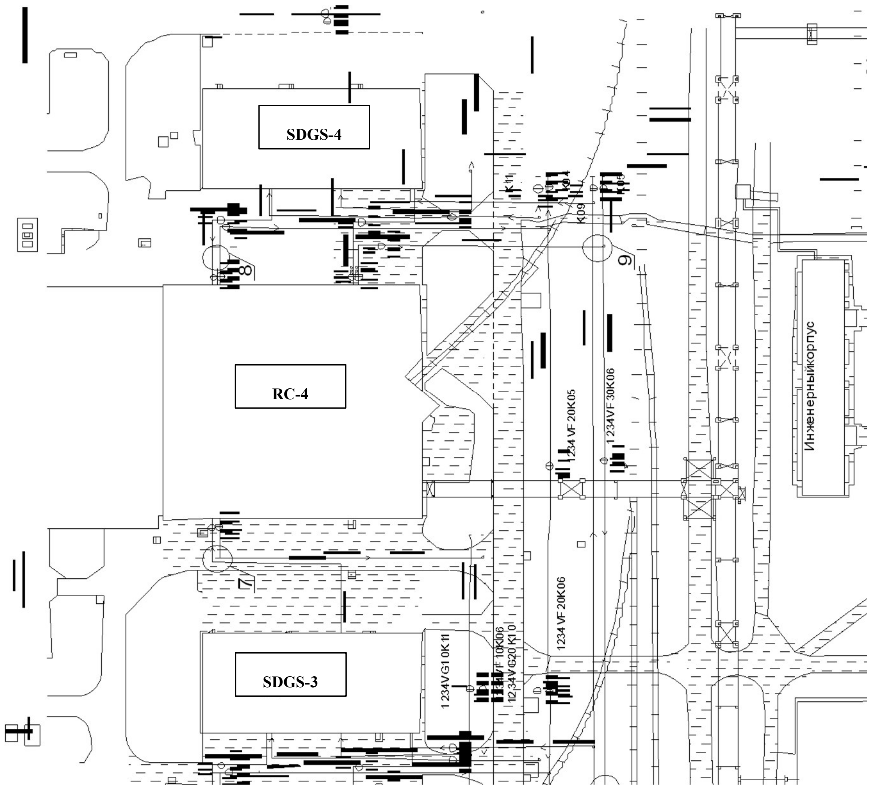


Figure A.1 Layout of underground ESWS piping of ZNPP-4

The list of ESWS piping is presented in Table A.1.

Table A.1 The list of ESWS piping (example for VVER-1000/V-320 design)

No.	Piping name (indication)
1	Piping of reactor compartment, outflow (VF10, VF20, VF30)
2	Piping of reactor compartment, inlet (VF10, VF20, VF30)
3	Tailrace (VF10, VF20, VF30)
4	Headrace (VG10, VG20, VG30)
5	Piping of spray ponds, outflow (VF10, VF20, VF30)
6	Piping of spray ponds, inlet (VG10, VG20, VG30)
7	Distributing pipes of ESWS spray ponds (12VF10, 12VF20, 12VF30)

Basic technical characteristics of ESWS piping are presented in Table A.2.

Table A.2 Basic technical characteristics of ESWS piping

No.	Characteristics	Value
1	Working medium	Service water
2	Temperature of working medium	5 ÷ 65 °C
3	Working pressure of inlet piping	0.6 kg/cm^2
4	Working pressure of outflow piping	4.5 kg/cm^2
5	Hydraulic testing pressure of inlet piping	2.0 kg/cm^2
6	Hydraulic testing pressure of outflow piping	7.5 kg/cm^2
7	Minimum wall temperature during hydraulic testing	> 10 °C
8	Testing medium	water
9	Duration of testing	10 min.
10	Material of production	Steel 20
11	Service life	30 years (262800 hours)

The main structural characteristics of piping (generalized, may vary depending on the design of a particular unit) are presented in Table A.3.

Table A.3 Main structural characteristics of piping

No.	Specified outer diameter, mm	Piping wall thickness, mm	Length of piping section, m	Depth of underground laying, m	Length of piping under building structures, m
1	Ø 2440 (with stiffening rings)	10	854.0	5.3 ÷ 5.8	82.5
2	Ø 2240	10	381.6	5.3 ÷ 5.8	
3	Ø 2040	10	285.6	5.3 ÷ 5.8	
4	Ø 1620	10	1 405.0	5.0 ÷ 5.4	82.5
5	Ø 1420	10, 14	502.0	5.0 ÷ 5.4	
6	Ø 1220	10, 11	337.0	5.0 ÷ 5.4	
7	Ø 820	9, 10	1 296.0	5.0 ÷ 5.8	

The main components of piping that affect piping life include welds and stress concentration areas (piping sections with a sharp change in geometry: bends, T-pipes, etc.).

ANNEX B

Information on VVER-1000 RPV structure and components

VVER-1000/320, 302, 338 RPVs

RPV for VVER-1000/320, 302, 338 has no differences in the design and is intended for the placement of in-vessel internals and core.

RPV is a vertical welded cylindrical vessel of a high pressure. Together with the closure head and details for sealing of the main joint and flanges of the closure head, RPV provides a sealed volume. RPV (Figure B.1) consists of a flange, nozzle area, support ring, rings of a cylindrical part and elliptical bottom.

On the upper end of the flange, there are 54 threaded jacks M170×6 for studs of the main joint and two ring grooves for the placement of rod sealing gaskets. There is a special drilling in the flange between the ring grooves ending with a thread with the installed fitting for connecting the leakage control system tube in order to control leakage of the main seal.

On the inner surface of the RPV flange, there is a ledge for barrel support. On the outer surface of the flange, there is a transitional cladding for welding of the dividing bellows and a ledge for installation of the thrust ring. The nozzle area consists of two rings, each with four main circulation nozzles DN 850 – in the lower ring for coolant inlet and in the upper ring for coolant outflow. There are two holes with nozzles DN 300 at the level of the axes of the upper row of nozzles DN 850 for the organization of emergency core cooling and a hole with nozzle DN 250 for the outflow of internal reactor I&C pulsed pipes. At the level of the lower row of nozzles DN 850, there are two holes with nozzles DN 300 for the organization of emergency core cooling.

On the inner surface of the upper ring in the nozzle area, there is a ring welded for spreading coolant flow. The upper ring of the nozzle area envisages installation of two channels for the placement of resistance thermometers intended for measuring temperature of RPV outer surface in the process of operation. Eight stainless steel brackets are welded to the inner surface of RPV cylindrical part for fastening of the lower part of the barrel.

The outer surface of the support ring has a support ledge with key grooves for fastening of the reactor on the support ring. All internal RPV surface is covered with anticorrosive cladding. Anticorrosive cladding has a thickness of not less than 15 mm in the places of contact with the closure head, barrel, gaskets, welding of brackets, details of I&C tube fastening, on flow spreader surface.

RPV components are welded together by automatic submerged welding by ring welds No. 2 – No. 7.

RPV bottom of KhNPP-2 and RNPP-4 are produced by single piece forging – weld No. 1 is absent.

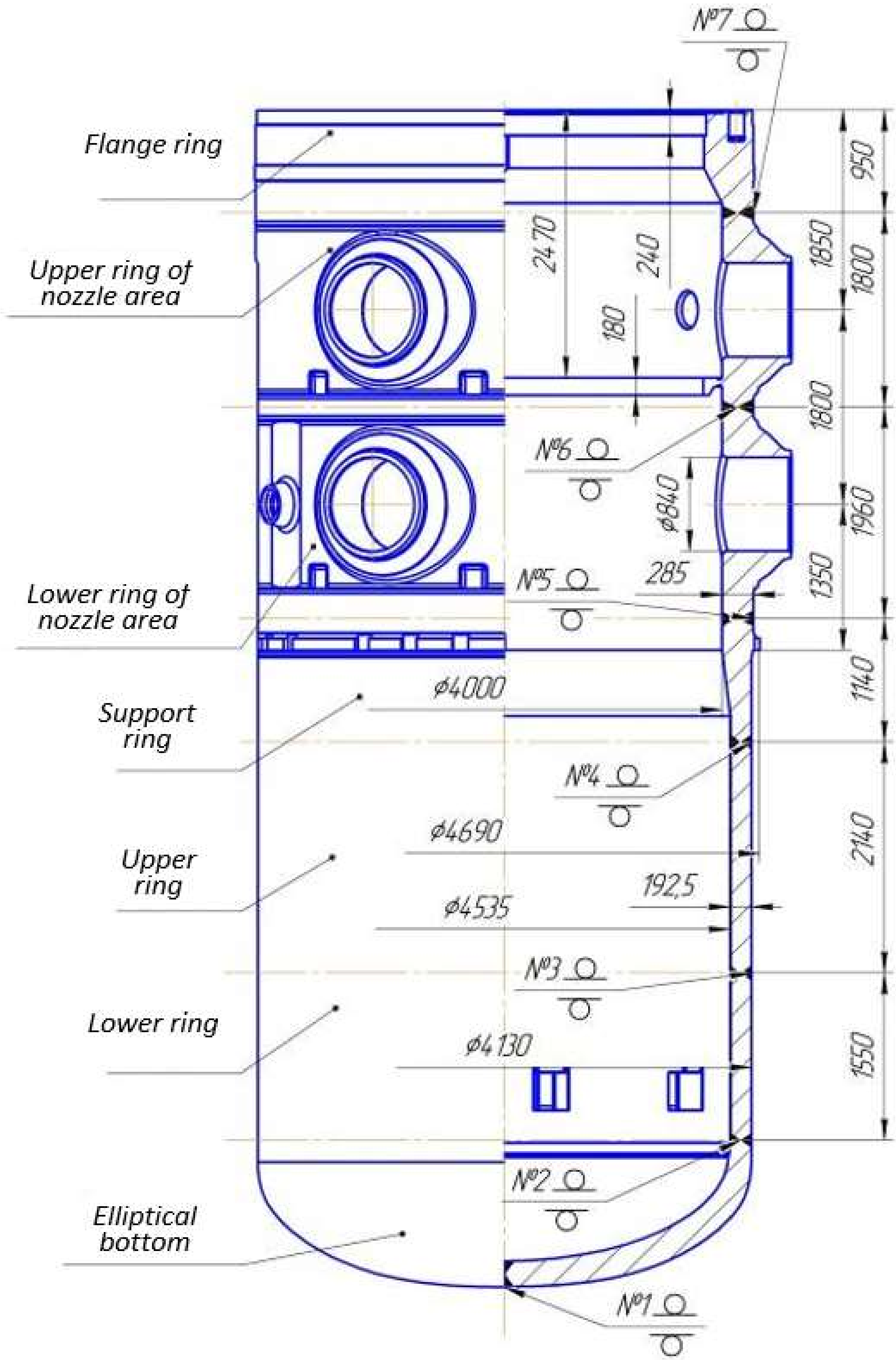


Figure B.1 Reactor pressure vessel structure

Reactor closure head

Rector closure head is intended for:

- RMJ sealing;
- outlet of communications of in-vessel monitoring system and their sealing;
- placement of step electric drives;
- securing assemblies, upper internals and in-vessel core barrel from emerging.

The structure of the reactor closure head is presented in Figure B.2.

Reactor closure head has a plate shape and is a tamp-welded structure consisting of a truncated ellipsoid and a flange. On the upper end of reactor closure head, there is an intermediate ring consisting of individual sectors and intended for increasing the area for contact of the wheel with closure head flange surface in its operation. The inner and end surfaces of the closure head are covered with anticorrosive cladding.

On the outer surface of the closure head, the cladding is adjusted under lugs and under the ring of biological protection. A contact surface is envisaged at the lower end surface of the flange for rod gaskets of the main sealing. There are cross-holes under closure head flanges for passage of pins of the main sealing and threaded nests for fastening of the system for centering of the upper internals and intermediate ring.

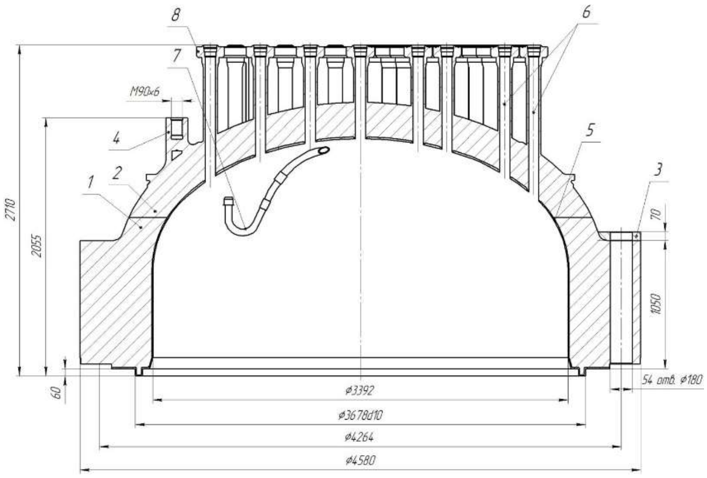


Figure B.2 Reactor closure head

1 – closure head flange
2 – elliptical part of closure head
3 – intermediate ring
4 – lug
5 – cladding
6 –CPS, TM and NIC nozzles
7 – air vent tube
8 – tapped flange of CPS, TM and NIC nozzles

On the closure head, there are CPS nozzles welded to the inner surface of the closure head and intended for the fastening of housing of drives and passage of pins of control rods, TM nozzles, NIC nozzles, nozzle of air vent and six cylindrical lugs with threaded nests for installation of rods of upper internal metal structures. All CPS, TM and NIC nozzles are made with the same connecting flange sizes.

ANNEX C

Information on VVER-440 RPV structure and components

VVER-440/213 RPV

RPV structure consists of the following main components (Figure C.1):

- bottom welded from two parts by an electro slag weld No. 1;
- three rings – one-piece-forged, turned;
- rings of the nozzle area – lower and upper;
- flange.

The components are welded together by automatic submerged welding by ring welds No. 2 – No. 7.

Each ring of the nozzle area has six nozzles DN 500, which serve for the inlet (lower row) and outflow (upper row) of the coolant and two nozzles DN 250 for water supply from ECCS.

Besides, on the upper ring there is a nozzle DN 250 for pulse tubes used for measuring the pressure difference between the inlet and outflow of the coolant during reactor operation and coolant level in reactor shutdown. Pulse tubes for measuring the pressure difference and coolant level are laid on the inner surface of RPV and are strengthened with protective plates. At the nozzle outlet, pulse tubes are equipped with shutoff devices for the event of rupture of tubes outside nozzle DN 250. The nozzle with shutoff devices for tubes intended for measuring coolant level and pressure difference welded to the nozzle of the upper ring in the nozzle area.

In the upper part of the outer surface of RPV flange, there is a cladding for welding of concrete console bellows. On the inner surface of the upper ring, there is a ring welded below nozzles 500 to ensure centering of in-vessel core barrel and spreading of inlet coolant flow and coolant outflow.

On the outer surface of the lower ring in the nozzle area, there is a circular ledge of 55 mm width, by means of which RPV is secured on the support ring installed on the support frame.

Eight brackets for fastening of keyholes intended for securing of the in-vessel core barrel bottom are evenly welded along the perimeter to the ring.

Due to technological features, piping DN 500 and DN 250 are welded to RPV nozzles through transition sockets, which are welded through cladding made at nozzle edges.

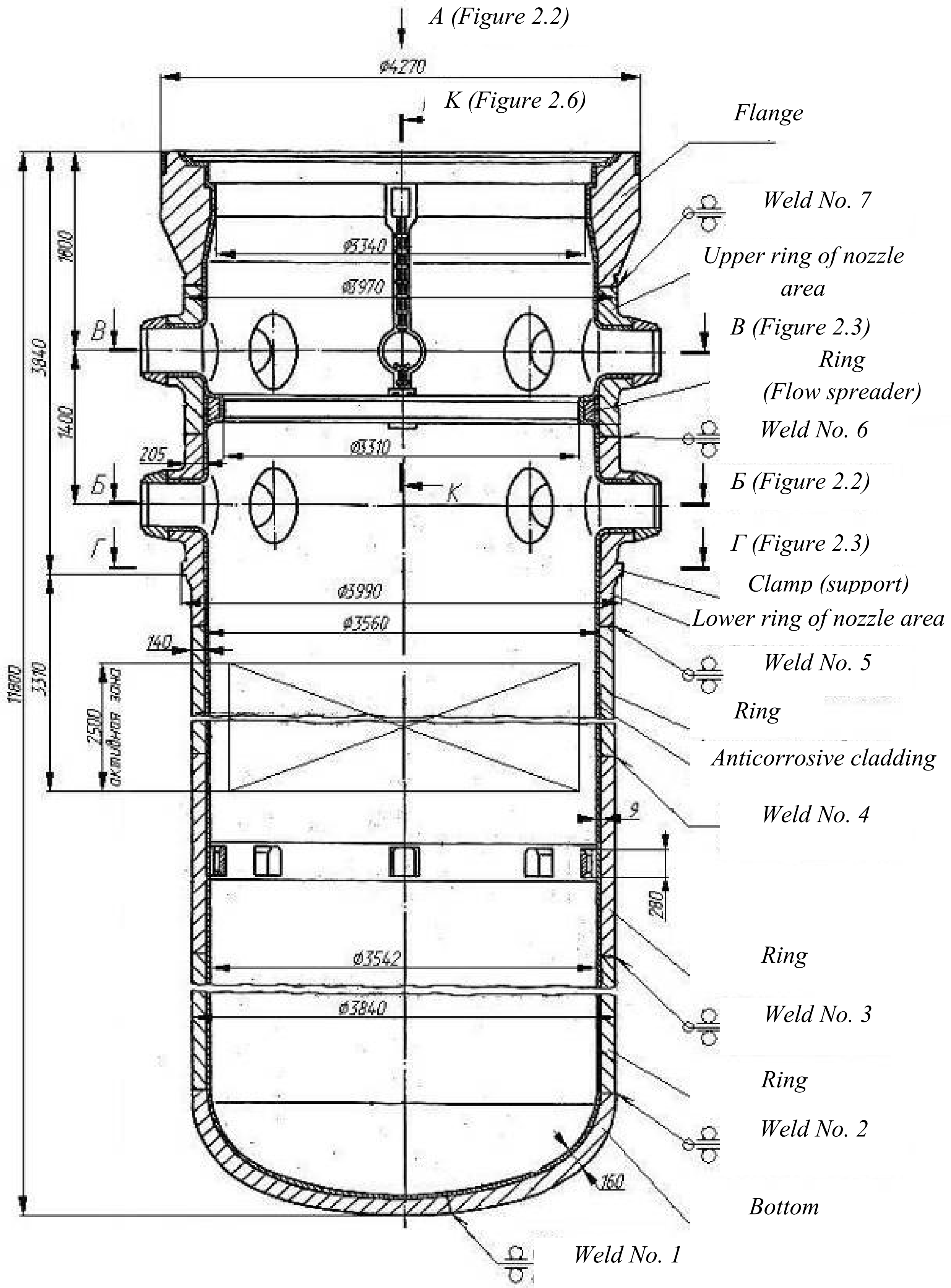


Figure C.1 Reactor pressure vessel

Reactor pressure vessel

The structure of reactor closure head is presented in Figure C.2.

Reactor closure head consists of the following main joints and components:

- reactor closure head;
- sealing joints;

– thermal isolation of reactor closure head.

Reactor closure head (flange, sphere) is made of alloyed steel of the perlite class. The inner and outer (partially) surfaces of the closure head are covered with claddings of corrosion resistant steel of the austenitic class.

37 nozzles of CPS of carbon steel with flanges for connection to the casings of the automatic regulation and control system, 12 thermal control nozzles and 6 nozzles of NIC for the installation of sensors are welded to the closure head. The inner surface of nozzles and holes in the spherical part of the closure head (TM, NIC nozzles) are protected against corrosion by stainless protective jackets. The toroidal compensator of RPV sealing joint is welded on closure head flange. The toroidal compensator is intended for the creation of a cavity to monitor possible (postulated) leakages.

Six lugs, into which six rods M90 are threaded, are welded to the outer surface of the sphere for the transport of the upper internals.

The thermal insulation of the closure head is made in the form of granules and can be removed.

Reactor upper internals are admitted for further operation, if their technical condition complies with requirements of current regulatory documents and TS for delivery.

In case of incompliance of the technical condition with requirements of current documents and TS for delivery, the decision is made on repair of relevant components or their replacement, changes in conditions and operating modes, or calculation justification of the strength of components, structure of the upper internals.

Data obtained during the TCA and assessment of upper internals life shall be included into the certificate. The certificate shall be accompanied by a solution of TCA and RVCH life assessment.

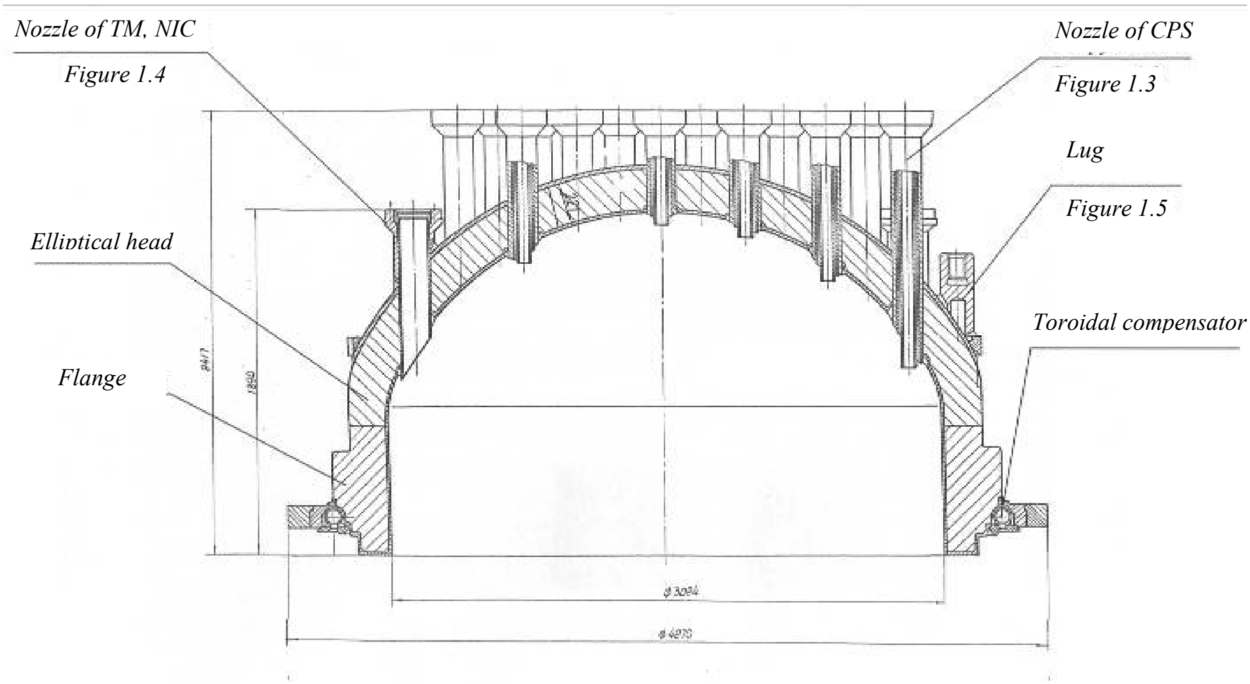


Figure C.2 Reactor closure head

ANNEX D

Description of building structures of VVER-1000/320 reactor compartment

The reactor compartment consists of the shell and auxiliary building (see Figure D.1, Figure D.2) located on a common foundation. The sealed cylindrical shell with an internal diameter of 45.0 m is placed centrally symmetrically in the auxiliary building of 66.0×66.0. The auxiliary building structure was designed on the principle of zoning safety systems and categories of production, and the modularity of prefabricated monolithic structure.

Building structures of the reactor compartment are intended for the placement of equipment and systems of the reactor compartment

Building structures prevent the impact of radiation and spreading of radioactive substances, help to separate NPP premises according to different categories, create necessary climatic and temperature conditions in the premises, organize ventilation, arrange the collection and removal of leakages, adjust piping and equipment depending on the impact of emergency and seismic loads, etc.

Foundation plate

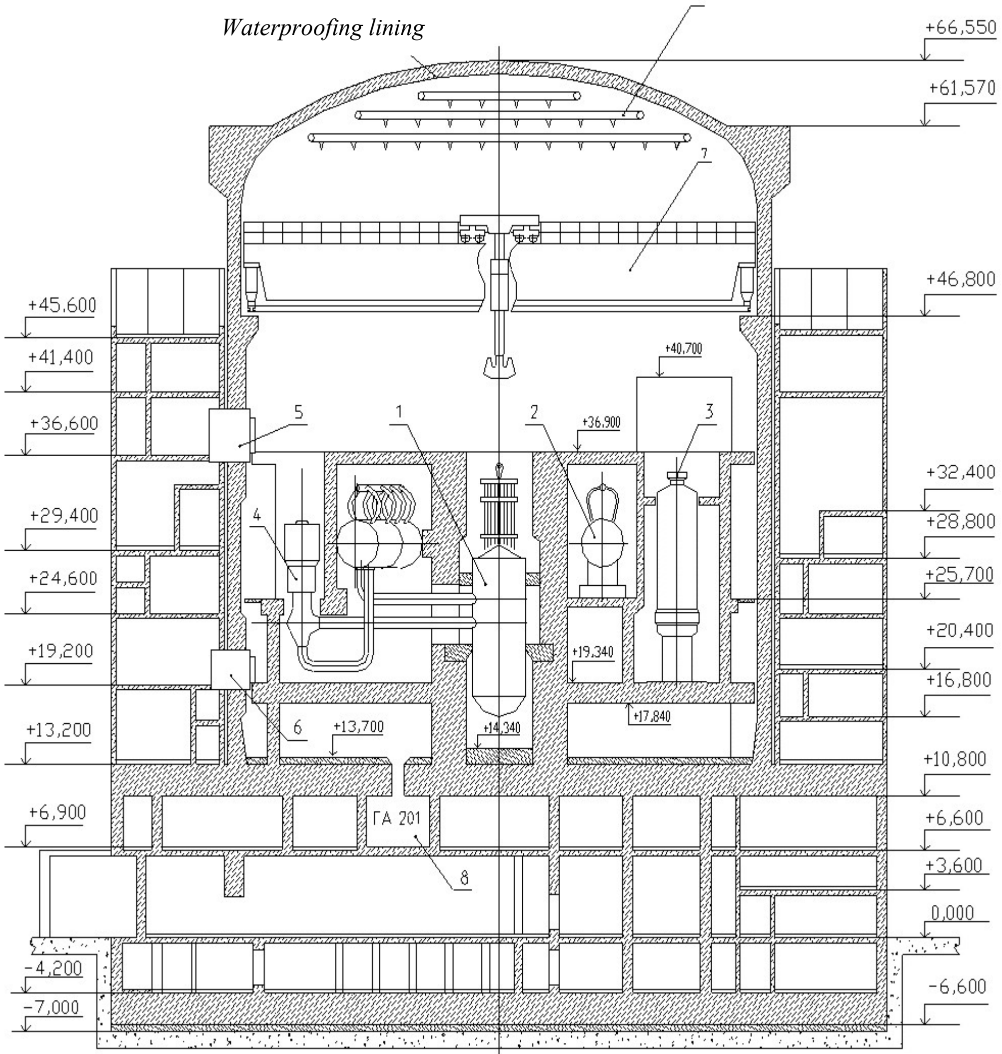
The design envisages the reinforced concrete foundation plate under the reactor compartment with a thickness of 2.4 m at the bottom indication of 6.600.

The foundation plate is designed taking into account asymmetric loading relative to the geometric axes of the reactor compartment and its area is more than the area of the reactor compartment to align the center of mass and the geometric center of the foundation plate in order to reduce lurch of the reactor compartment.

Concrete of the foundation plate is of B15 class and its waterproofness indication is W-6. Fittings are of class AIII with a diameter up to 40 mm inclusive.

There is a waterproofing made of profiled polyethylene for the protection from groundwater.

Prefabricated reinforced concrete plates with waterproofing made of profiled polyethylene are used as formwork for the end surfaces of the plate.



1 Reactor
2 Steam generator
3 Pressurizer
4 RCP
5 Main lock
6 Emergency lock
7 Polar crane
8 Sump tank
9 Spray system

Figure D.1 Section of the reactor compartment on axes I-III

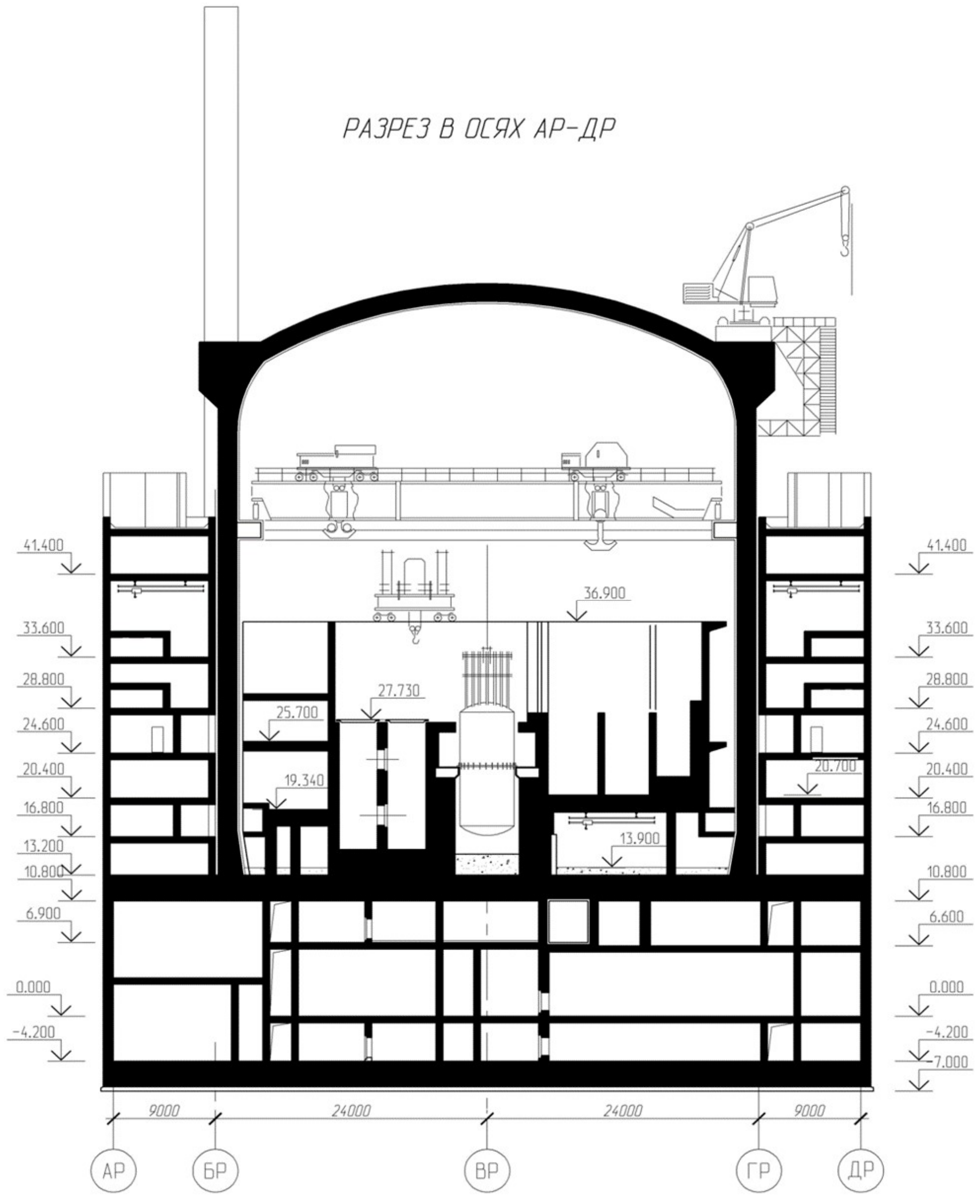


Figure D.2 Section of the rector compartment on axes II-IV

Example of the list of buildings and structures with AM approved for ZNPP-1 for the foundation plate is presented in Figure D.3.

ZNPP	AGEING MANAGEMENT PROGRAM for Components and Structures of ZNPP-1	Page 457
01.10.00.MR.PM.03-15		

Foundation plate, walls and overlapping of the foundation part RV-1. Classification indication according to NP 306.2.141-2008 – 2N. Category according to PiN AE-5.6 - I. Seismic resistance according to PNAE G 5-006-87 -I. Basis: Organizational measures on the ageing management of building structures. Table 1. "Ageing Management Program for Building Structures of Foundation Plate, Walls and Overlapping of Foundation Part RV-1 of ZNPP" (01.3C.00.PM.32-14)

Name of component and structure	Function	Possible degradation mechanism	Ageing effect	Parameter of technical condition to assess degradation process	Value of technical condition parameter		Material	Operational conditions	Ageing management measures		Corrective actions
					Actual	Regulated			Organizational measures	Technical measures	
Steel structures of lining	Enclosing building structures	Welds – generation of cracks	Reduced strength	Parameters of crack	Absent	Not allowed	VSt3kp2 according to GOST 380-71* (steel S235 according to GOST 27772-88).	Temperature increased to 70°C. Small doses of radioactive radiation (neutron flow 1013 neutron / cm²).	- Specification of working program for survey, assessment of the technical condition and reassignment of the life of components of the foundation plate, walls and overlapping of foundation part RV-1 of ZNPP - Specialized inspection of building structures - review and specification of AMP provisions - Monitoring of building structures - observation of groundwater level	Constructive measures aimed at the elimination of revealed defects and damage of components and structures of the foundation plate, walls and overlapping of the foundation part of RV-1 of ZNPP to manage such dominating ageing mechanisms for steel structures – general and local corrosion, possible availability of different defects and damages in welds	Not required
		Destruction of anticorrosive coating – local corrosion	Embrittlement	Sheet thickness	3.9mm	4.0mm					
		General corrosion, flow accelerated corrosion	Thinning	Sheet thickness	3.9mm	4.0mm					
Monolithic foundation plate	Bearing building structures	Insufficient protective layer	Uncovering of reinforcement	Value of protective layer	20.7mm	20mm	Heavy concrete M200	Temperature increased to 70°C. Small doses of radioactive radiation (neutron flow 1013 neutron/cm²). Load: own weight of structures and other process equipment. The waterproofing is made under the foundation plate.			
		Destruction	Reduced strength	Strength	31.7MPa	24.8MPa					
		Damage of protective layer	Uncovering of valves	Visually	Not detected	Not allowed					
		Corrosion of reinforcement	Reduced bearing capacity	Diameter	20mm	20mm					

Figure D.3 Example of buildings and structures with AM

Auxiliary building of the reactor compartment

The auxiliary building of the reactor compartment is located around the containment on the support plate at the elevation 13.200. The auxiliary building that is separated by an anti-seismic joint from the containment, is a multi-story box-shaped structure with the transfer of seismic loads and loads from the external shock wave through the ceiling slabs to the shear walls.

There are ladders and elevator shafts in the corners of the building. Structures of walls and ceilings are similar to the foundation part.

Containment Safety System

CSS is intended to hold active fission products that are generated in different design modes of NPP operation and to reduce effect of radiation.

The containment system consists of the following components:

- sealing metal lining;
- RCE, including the containment prestressing system;
- hatches;
- isolation devices;
- sealed penetrations;
- sections of process and ventilation piping that cross the containment within the isolation devices.

Sealing metal lining

There is a metal lining on the inner surface of the containment made of steel VSt3pc5 with a thickness of 8 mm according to GOST 380-71*, which is in general available for inspection and checking tightness during installation and operation, used to ensure tightness. The lining from the side of accident confinement area is covered by the anticorrosion protection – an epoxy coating on aluminum sublayer (Al250).

There is a concrete layer of not more than 500 mm (elevation 13.700) and additional protection by a steel sheet VSt3sp5 with a thickness of 8 mm made to protect the protective lining at the elevation 13.2 m (foundation plate) during operation from possible damage by equipment, instruments, etc.

Steel sheets were connected by automatic welding in the factory conditions for pre-assembling of components and manual welding during installation of such components at the facility before concrete. Installation joints of cylindrical part and dome part can be checked for waterproofness, if needed, during operation.

Welding joints of the lining in places not accessible for inspection during operation are carried out at the stage of manufacturing and installation with increased control scope according to VU-2S-83. Metal lining is anchored in concrete by means of welded corners with rod anchors. Failure of one rod anchor does not lead to failure of the anchoring system.

Layout of boxes provides for the arrangement of walls that fulfil functions of the screen for protection against flying objects to protect the lining.

Reinforced concrete enclosure including the containment prestressing system

RCEs are a reinforced concrete cylinder closed from the top by a spherical dome and by the foundation plate from the bottom. RCE height is 53.25 m, internal diameter is 45.00

m, thickness of the cylindrical part wall is 1.20 m, thickness of the dome is 1.10 m and thickness of the foundation plate is 2.40 m. The connection of the cylinder and the spherical dome is made using a thickened support ring. The following materials were used for RCE construction: heavy concrete M-400, unstressed fittings of class AIII steel.

RCEs are intended for:

– protection of the reactor primary side and its systems from external effects and special loads;

– protection of sealing steel lining from possible deformations, which can lead to unsealing of the sealed enclosure;

– performance of the function of biological protection of personnel, the public and the environment against radiation and radioactive substances located in the accident confinement area.

There is the containment prestressing system for preliminary stressing of RCE.

The cylindrical part and the dome part of RCE are reinforced by tendons with a length from 80 to 180 m. Each tendon consists of 450 (456) high-strength wires with a diameter of 5 mm and two thimbles. The design provides for installation of 96 tendons in the cylindrical part and 36 tendons in the dome part.

The tendons are installed in ducts, which are pipes made of dense polyethylene 219 mm in diameter.

About the RCE cylindrical part, tendons are located on helix lines (spiral loop system) with right and left directions. Each tendon is bent in the lower plate of the containment at the elevation 10.8 m, changing the direction to the opposite. Both ends of each tendon are fixed to the upper ledge of the containment support ring. Across the containment wall thickness, tendons are located in three rows.

In the dome part, tendons are located in two mutually perpendicular directions in two rows across thickness. A tendon passes through the entire dome, bends to the opposite edge of the support ring, and again returns to the anchorage area, forming an elongated loop.

Within conduct of activities with TCA and LTO of ZNPP-1,2 containment, there was a calculation justification performed with regard to reliability of the containment and its compliance with requirements of regulatory documents (with determination of minimum allowed tendon tension).

Calculation results confirmed operability of the containment under stated combined loads. General overview of RCE with the image of the trajectories of tendons in the cylindrical (a) and dome (b) parts is presented in Figure D.4.

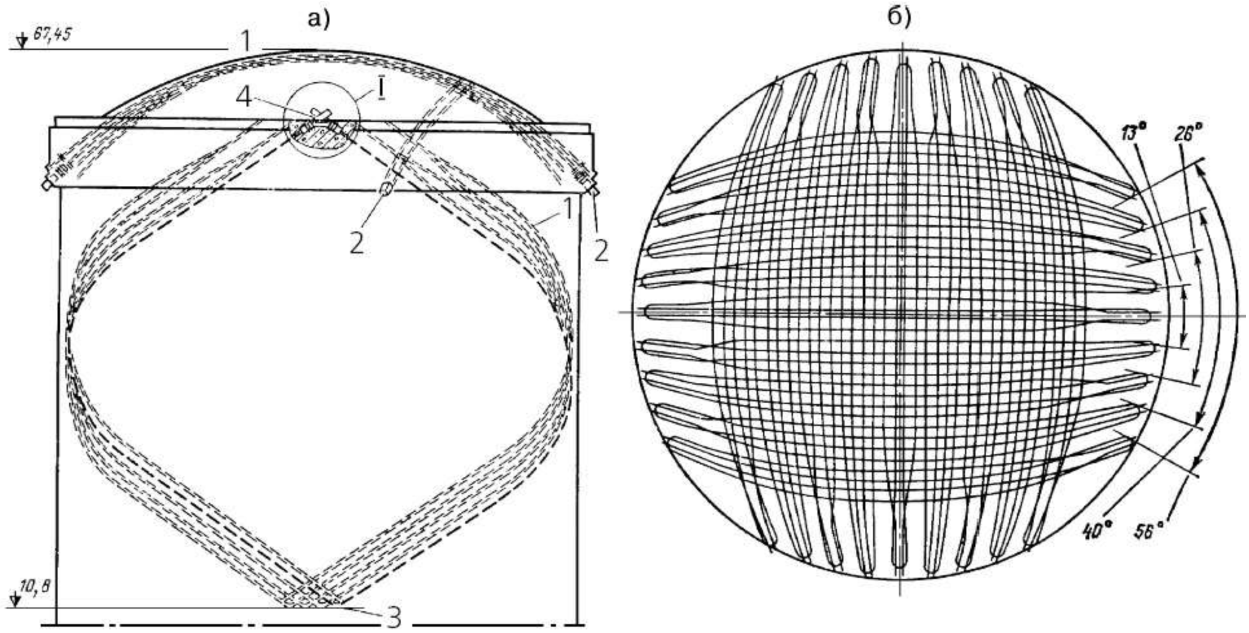


Figure D.4 General overview of RCE with the trajectory of tendons

Pressure tight hatch of the transport corridor

The design solution for ensuring radiation safety for transport communications that link the containment with the external environment is to use the special pressure tight hatch at the elevation 13.700.

The pressure tight hatch (see Figure D.5) that consists of two parts (operational and installation) is intended to maintain tightness of the containment and handling operations with the transfer of equipment and fuel to/from the accident confinement area.

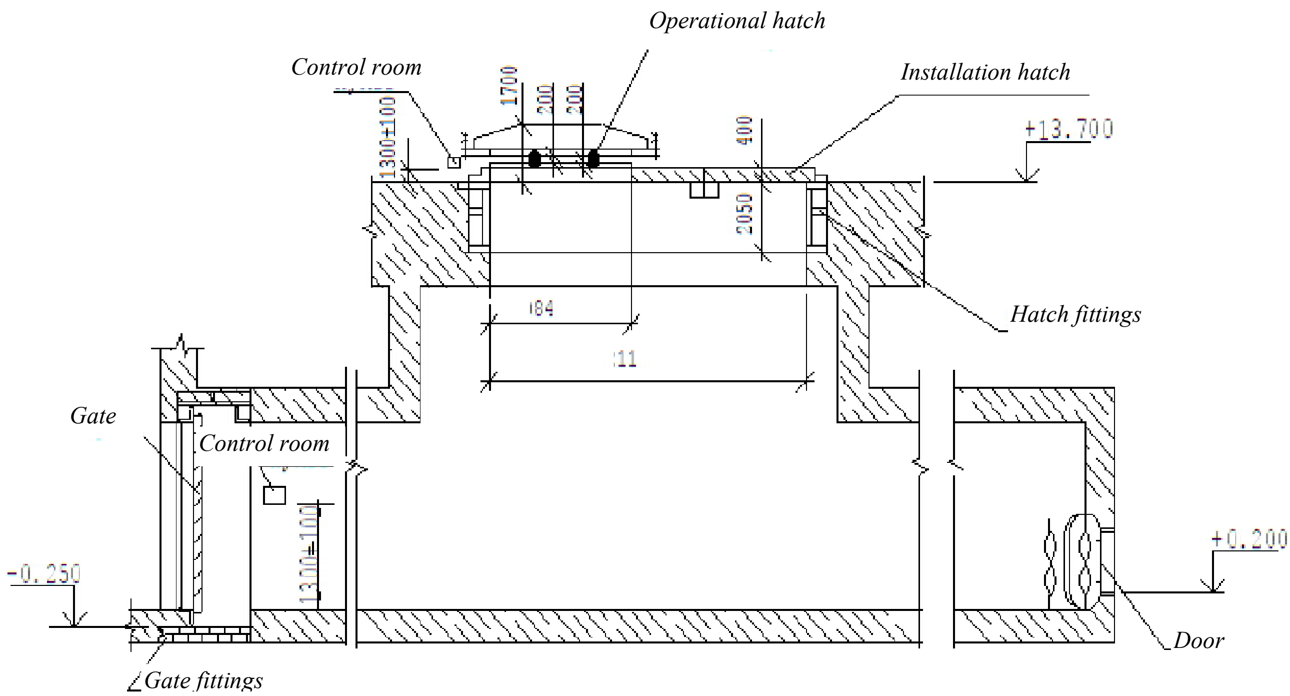


Figure D.5 Location of the pressure tight hatch and gates of the transport corridor

Main and emergency hatches

According to the principle laid in the CSS design, there are two hatches envisaged in the containment for the entry of personnel: main hatch at the elevation 36.600 and one emergency at the elevation 19.300.

The hatches (see Figure D.6) are the passive confining systems of CSS, which ensure required tightness of the containment and remain operable in all design modes of unit operation.

The hatches are provided with locking devices made in the form of gears installed on the side walls of the holes and wedge grooves made in the hatch. Along the casing, there is the mechanism for distribution of motion. Hatch is controlled from flywheels, which are installed from the side of unsealed area, and in the inner cavity of the hatch from the side of sealed area.

The kinematics of opening (closing) mechanisms of hatches in normal mode of operation does not allow simultaneous opening position of both hatches of hatches (mechanical locking). The hatches have the communication with the main control room.

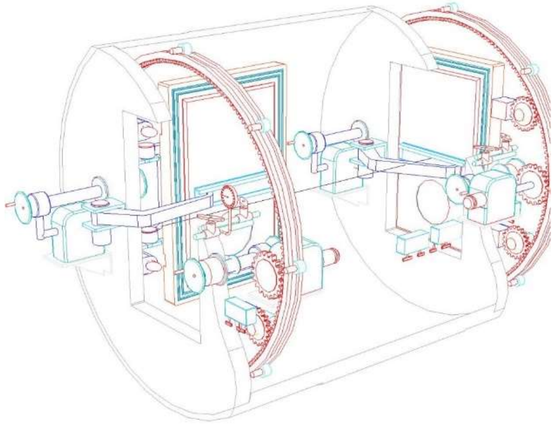


Figure D.6 Main hatch

Sealed penetrations

The connection of the containment with different communications is performed by means of penetrations. Penetrations ensure tightness of the containment when crossed by the following communications: electrical, process, ventilation, channels of ionization chambers.

The design provides for:

– sealed penetrations for process and ventilation communications of 33 types (piping, drains, pulsed lines, etc.);
– embedded details of sealed penetration of 23 types;
– sealed cable penetrations of PGKK type for control cables, penetrations of VGU type, penetrations of Eloks for forced and control cables;
– channels of ionization chambers.

Tube penetrations are located in the cylindrical part, in the foundation plate and bottom of the drainage systems of pumps in the spray system (GA201 tank). PGKK, VGU and Eloks penetrations are grouped in six groups (PEG systems) with different penetrations in each group and they are located at different elevations along the entire height of the containment. Channels of ionization chambers are located in the RCE foundation plate.

Tube process sealed penetrations (Figure D.7) are installed into the embedded detail put into the reinforced concrete enclosure and is welded to the embedded detail.

Al electrical sealed penetration (VGU, PGKK, Eloks) (Figure D.8) is installed into an embedded detail put into the reinforced concrete enclosure. The penetration is welded to the embedded detail through a ring (VGU, PGKK) or nozzle (Eloks), connected to relevant cables. Elements of biological protection are placed between penetration and the embedded detail.

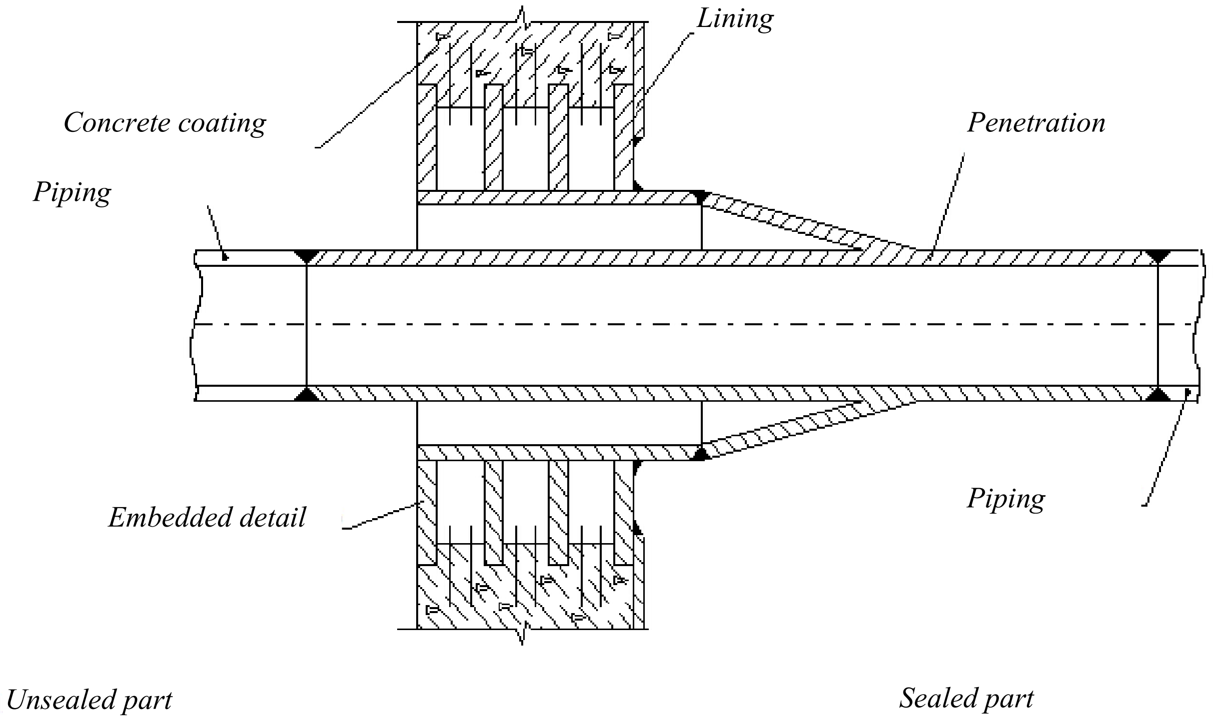


Figure D.7 Process (tube) penetration

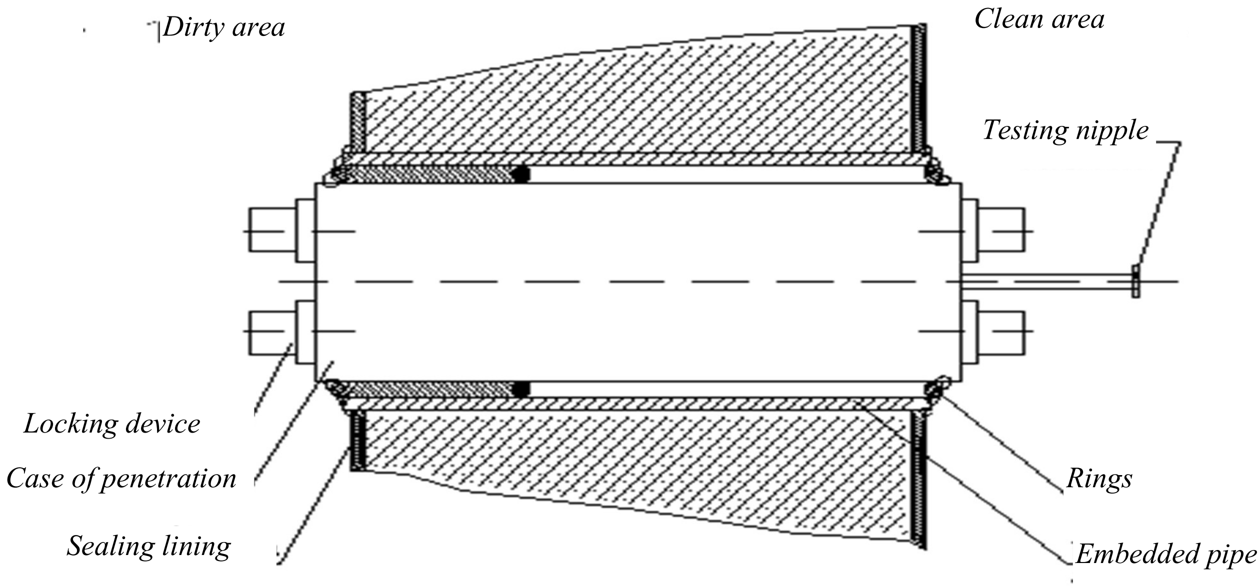


Figure D.8 PGKK sealed penetration

ANNEX E

Description of building structures of VVER-1000/302, 338 reactor compartment

The reactor compartment (see Figure E.1) is a round building with a diameter of 47.4 m, a height of 76.0 m. It consists of two main volumes: unsealed foundation part from the elevation 0.000 to the elevation 11.800 and sealed part higher than the elevation 11.800 m.

Building structures of the reactor compartment are intended for the placement of equipment and systems of the reactor compartment.

Building structures prevent the impact of radiation and spreading of radioactive substances, help to separate NPP premises according to different categories, create necessary climatic and temperature conditions in the premises, organize ventilation, arrange the collection and removal of leakages, arrange ventilation, perform collection and removal of leakages, adjust piping and equipment depending on the impact of emergency and seismic loads, etc.

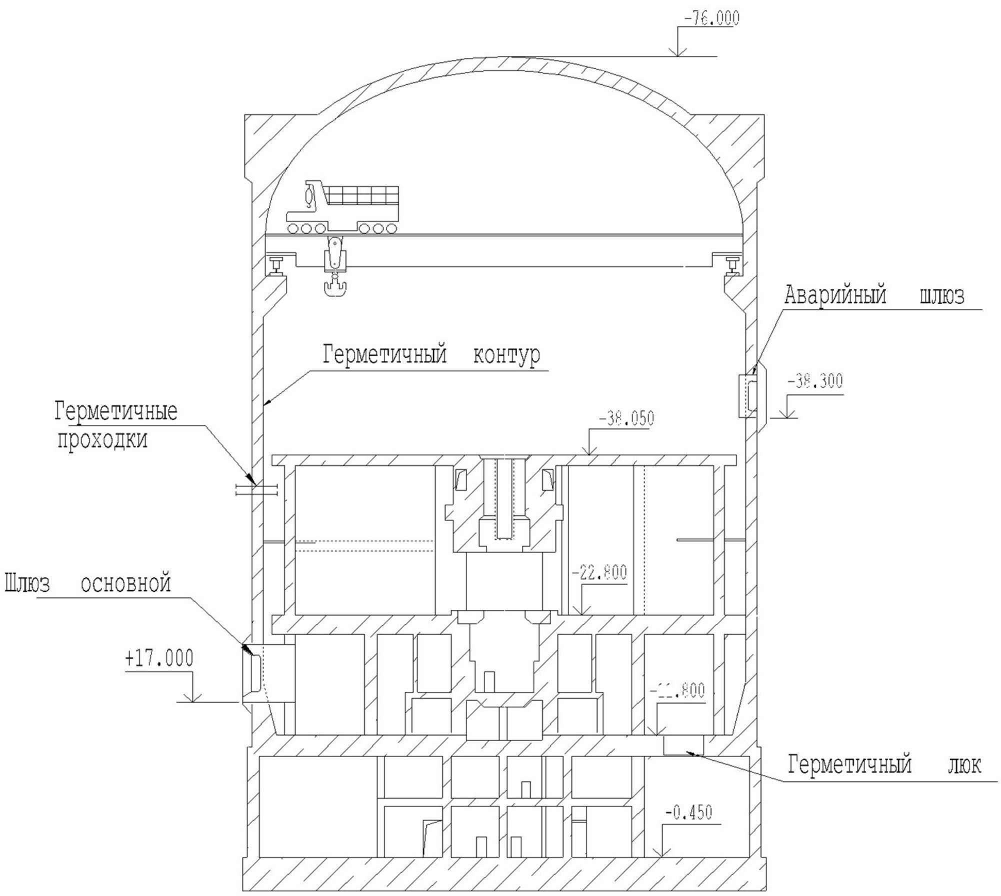


Figure E.1 Layout of the containment

Foundation plate

The design envisages the monolith reinforced concrete foundation plate under the reactor compartment with a thickness of 3.0 m and a diameter of 48.4 m. The elevation of the plate sole is minus 3.45 m, taking into account concrete footing with a thickness of 1.5 m, the depth into the soil is 4.95 m.

The plate is made of concrete M300 (C25), on sulfate-resistant portland cement, waterproofness indication W6, fittings of classes AI and AIII, with a diameter up to 40 mm inclusive.

The reactor compartment bottom plate is located on a solid concrete footing with a thickness of 1500 mm, concrete M100 (C7.5).

There is a waterproofing made of profiled polyethylene under the foundation plate for the protection from groundwater. The same waterproofing is provided on the outer sides of the foundation part of the reactor compartment to the elevation 0.000.

Internal structures of the containment

The containment of the reactor compartment consists of the protective prestressed monolithic reinforced concrete cylindrical shell covered by a prestressed spherical dome.

Internal structures of the reactor compartment due to the complex configuration and a large number of process penetrations are performed in monolith reinforced concrete design. Concrete is M400 (C30).

All concrete surfaces in the containment are lined with carbon steel, and in premises where leakages of radioactive media are possible they are lined by stainless steel.

There are support structures specially designed for adjustment of main RPV equipment, steam generators, RCP, MCP, ECCS and other for the case of seismic loads, and there are restraining supports that perceive the load at the rupture of large diameter pipes.

These supports and embedded details are reactor structures and they are supplied by the manufacturing plant.

Containment Safety System

CSS is intended to hold active fission products that are generated in different design modes of NPP operation and to reduce effect of radiation.

The containment system consists of the following components:

– sealing metal lining;
– RCE, including the containment prestressing system;
– hatches;
– isolation devices;
– sealed penetrations;
– sections of process and ventilation piping that cross the containment within the isolation devices.

Sealing metal lining

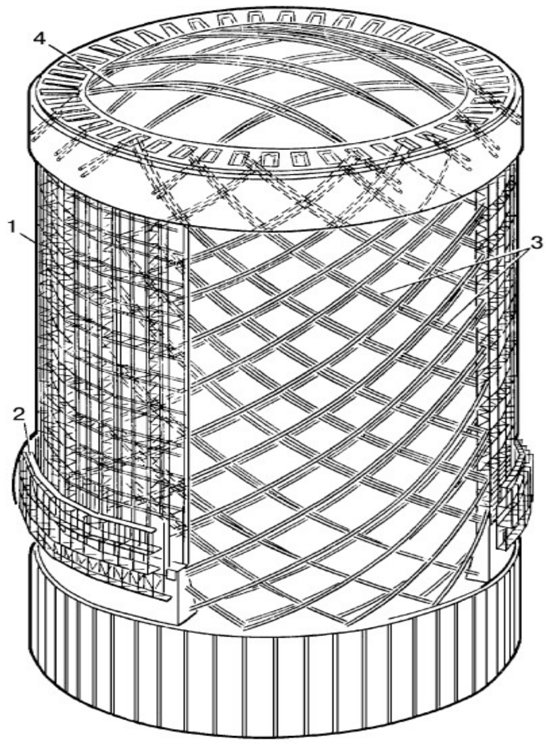
There is a metal lining on the inner surface of the containment made of steel VSt3pc6 with a thickness of 8 mm according to GOST 380-71*, which is in general available for inspection and checking tightness during installation and operation, used to ensure tightness. The lining from the side of accident confinement area is covered by the anticorrosion protection – an epoxy coating on aluminum sublayer (Al250).

Checking of tightness in the places of mounting joints is performed with the help of a special bar filled with air at excess pressure of 0.415 MPa and covering of welds with foam indicating penetrant. Welds of the containment elements made and controlled in factory conditions (in particular, by physical inspection methods) are subject to visual inspection in the process of construction, acceptance and operation. In case of revealing defects and after their correction, it is necessary to check tightness by means of vacuum chamber or welding of a bar filled with air at excess pressure of 0.415 MPa.

Welding joints of the lining in places not accessible for inspection during operation are carried out at the stage of manufacturing and installation with increased control scope according to VU-2S-83. Metal lining is anchored in concrete by means of welded corners with rod anchors. Failure of one rod anchor does not lead to failure of the anchoring system.

Layout of boxes provides for the arrangement of walls that fulfil functions of the screen for protection against flying objects to protect the lining.

The containment is the main component of sealed enclosures (Figure E.2).



1. Tendon cage
2. Frame
3. Polyethylene channel former of the cylindrical part
4. Polyethylene channel former of the dome

Figure E.2 VVER-1000/V-302, 338 containment

The prestressing system includes two-loop tendons of continuous winding, upper and dome anchor blocks, lower support blocks, hydraulic jacks, pumping units to control systems, channel formers in the cylinder and dome. The prestressing value was established on the condition of perception of membrane stresses in moment-free areas from the impact of test pressure of 0.515 MPa.

Unstressed fittings were defined from the condition of perception of temperature influences, regulation of crack formation and perception of forces caused by moments and transverse forces in boundary zones of the cylinder and dome.

Pressure tight hatch of the transport corridor

The design solution for ensuring radiation safety for transport communications that link the containment with the external environment is to use the special pressure tight hatch at the elevation 12.300 and sealed gates in the special transport corridor at the elevation 0.000.

Gates have seals to exclude release of contaminated media outside the transport corridor and income of precipitations and storm water to it.

The hatch is built into the support ceiling plate 11.800 and is an integral part of the containment and active confining device designed to perform confining functions in any design modes, including maximum design-basis accident.

The hatch belongs to the seismic resistance category I and preserves strength and operability under safe shutdown earthquake of 6 magnitudes according to MSK-64, and it is designed for all types of loads that accompany normal and emergency modes.

The greatest pressure on the hatch is 0.5 MPa. Allowable estimated leakage is no more than 0.1 m^3/day in all emergency modes.

Main and emergency hatches

According to the principle laid in the CSS design, there are two hatches envisaged in the containment for the entry of personnel: main hatch at the elevation 17.850 and one emergency at the elevation 39.600.

Allowable leakages through components of hatches are indicated in the technical documents of the hatch manufacturers.

The hatches belong to seismic resistance category I and are made taking into account seismic impacts. They are designed for all types of loads that accompany normal and emergency modes.

Sealed penetrations

The connection of the containment with different communications is performed by means of penetrations (see Figure E.3). Penetrations ensure tightness of the containment when crossed by the following communications: electrical, process, ventilation, channels of ionization chambers.

Embedded details of penetrations with metal lining and RCE joints are connected by welding. Welding of the penetration with its own embedded detail is also performed by welding.

The design provides for:

– sealed penetrations for process, heat and cooling supply and ventilation communications of 31 types according to technical specifications TU 108-11-294-78;
– tube penetrations for ventilation communications DN=1620x14 according to working drawings KhO TEP No. YuAT-661-1132 and YuAT-661-1124;
– embedded details of sealed penetration of 23 types according to technical specifications TU 108-11-293-78;
– sealed cable penetrations of PGKK type for control cables according to TU 4112.000;
– sealed cable penetrations for power cables of 6 and 0.4 kV according to technical specifications 0AA.195.581-77 of VDU type);
– channels of ionization chambers according to drawing 1117.27.000.

Types of tube penetrations were defined by features of reactor compartment structure and parameters of the media passing in piping.

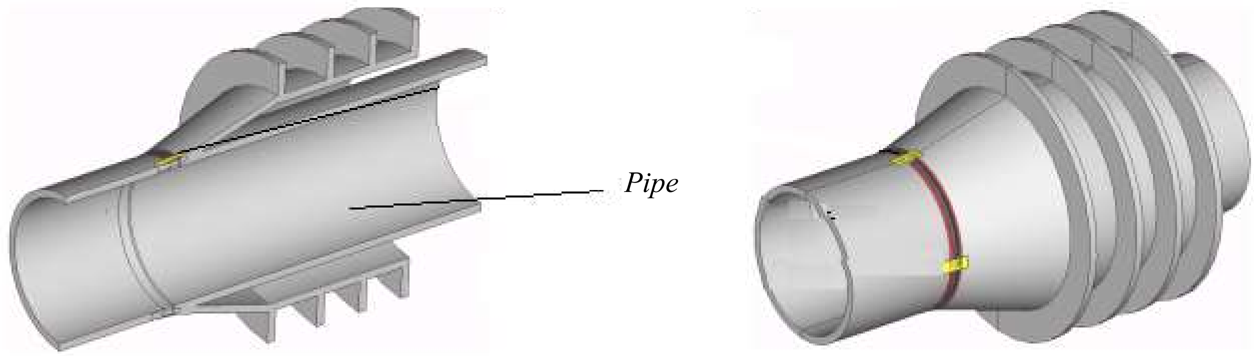
Sealed penetrations belong to seismic resistance category I.

Tube penetrations are fixed and are designed for all loads from piping and loads in case of their possible rupture. Under such conditions, the tightness of connection with penetrations is not disturbed.

The use of sealed cable penetrations ensures proper tightness in case of pressure increase inside sealed premises.

The criteria for penetration failure is penetration rupture and loss of tightness of accident confinement system through this penetration.

Isometric diagram of penetration



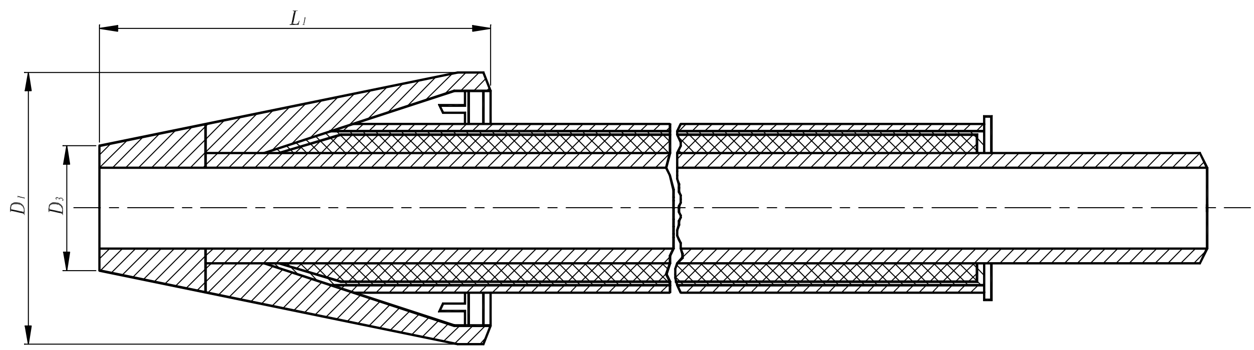


Figure E.3 Structure of sealed process penetration

ANNEX F

Description of building structures of VVER-440/213 reactor compartment

Reactor compartment design solution

Reactor compartment is a rigid reinforced concrete box system (see Figure F.1 - Figure F.3) located on a solid foundation plate with a thickness of 1.70 m reinforced by rod barrels. The elevation of the foundation plate with footing is minus 8.500.

The load from upper structures is put on the plate through the massive walls of box rooms and monolithic reinforced concrete columns. Up to the elevation 18.900, structures of the frame, bearing structures of columns, ladders and elevator shafts are reinforced concrete, prefabricated and monolithic. Walls and ceilings are made in prefabricated and monolithic reinforced concrete using steel cells where metal lining performs the role of formwork during concrete operations and sheet fitting during operation.

Above the elevation 18.900, the bearing structures are made in the form of steel frame.

The steel frame of the reactor compartment is a U shaped frame with a bolt in the form of trapezoidal farm with a triangular lattice, whose elements are made of single rolling angles. The longitudinal deck with electrical devices of unit 1 and machine hall of unit 1 adjoin U shaped frame of the reactor compartment. Insulated panels in the form of profiled flooring with 12.00 m passages are leaning in the joints of the upper row of the farms. In constructive terms, the covering is made of mounting blocks in the form of two farms with 12.00 m panels.

After installation of covering mounting blocks, belts and grids of trussed frames were not structurally linked. The blocks are autonomous and have horizontal and vertical connections.

Columns are made in the upper and lower parts in the form of welded I-beam. The column is based on the reinforced concrete stack. The crane beams of the bridge electric crane with a carrying capacity of Q = 250/30 t are based on the console and the ledge of the column. The crane beams are made of welded I-beam sections. The brake system for the crane Q = 250/30 t is made in the form of a complex beam, which has a brake sheet made of corrugated steel and a belt of rolled channel based on stanchions of main and support columns. The bridge crane of the lower tier with a carrying capacity Q = 30/5 t is based on the special console in the lower part of frame columns. Crane beams are welded I-beam with the braking system of the similar braking structure of the crane Q = 250/30 t.

The reactor compartment is conventionally divided into four volumes:

- box part;
- central part;
- accident confinement shaft;
- ventilation center and special ventilation building.

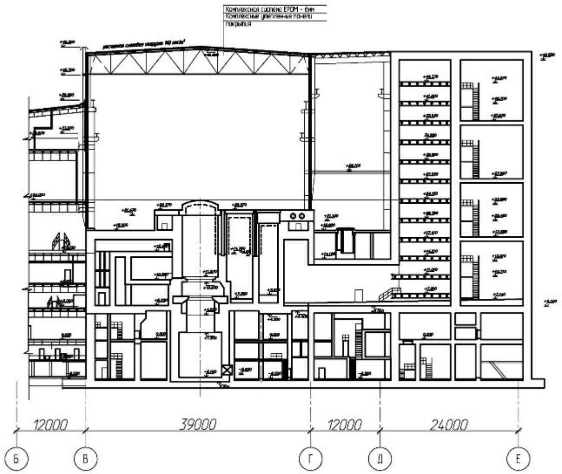


Figure F.1 Reactor compartment, section 1-1

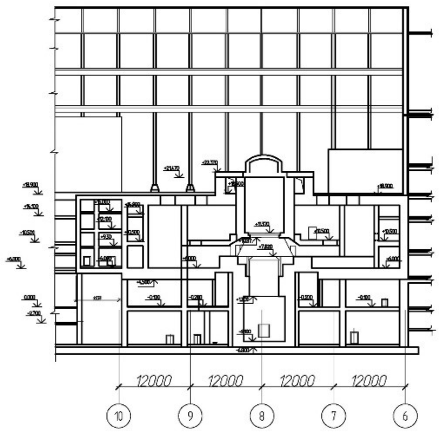


Figure F.2 Reactor compartment, section 2-2

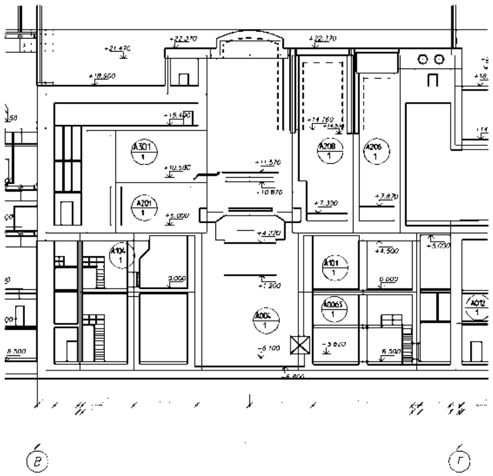


Figure F.3 Cross-section of the reactor shaft

Box part

The reactor compartment consists of a reactor shaft located in the center along the axis 8 at the elevations from minus 6.100 to 22.370, different process rooms and sealed volume around the reactor shaft.

Structures of the box part of the reactor compartment from the elevation minus 6.500 to the elevation 18.900 are monolithic reinforced concrete, massive walls and ceilings.

ECCS rooms from the elevation minus 6.500 to the elevation 6.000 are made in monolithic reinforced concrete with wall thickness of 1000 – 1500 mm, ceiling of 800-1000 mm.

At the elevation 18.900, there is the reactor service compartment. The frame of the reactor service compartment is made of metal columns and beams. Walls of the reactor compartment frame are made of prefabricated reinforced concrete panels.

Coating of the reactor compartment is made of integrated insulated panels using a steel zinced profile sheet on steel beams.

The containment is based on the ECCS room at the elevation 6.400 and consists of:

– reactor shaft;
– spent fuel pool and well;
– box of steam generators with internal rooms;
– room of recirculation facilities;
– corridor for release of steam-gas mixture from RCP rooms and steam generators to the accident confinement shaft.

Structures of the containment are made in monolithic reinforced concrete with the thickness of walls and ceilings of 1000-1500 mm.

Central part

The lower part of the building from the elevation minus 6.500 to 18.900 along the axes 6-10 is an independent block, the general premises for units 1 and 2 are located in the central part of the reactor compartment along the axes 10-14.

The structure of the central part of the reactor compartment in unit 1 from the elevation minus 6.500 to 18.900 is monolithic and prefabricated monolithic massive walls and ceilings with a thickness from 400 mm to 800 mm.

The surface part is made in the form of steel frame from the elevation 18.000.

Accident confinement shaft

The structures of the accident confinement shaft are made from monolith reinforced concrete with a thickness of 1000-1500 m with the use of reinforced concrete components and steel cells in the area of the containment. The structures of the ceilings are made of prefabricated monolithic reinforced concrete with a thickness of 800-1000 mm. Continuous inner or outer steel lining is made on the perimeter of enclosing structures of sealed rooms. The outer lining of the accident confinement shaft facades (along the E axis and partially on the transverse walls) is made of prefabricated reinforced concrete plates with a thickness of 85 mm. The plates are fastened to concrete walls with metal embedded details. The gap between the wall and the lining is 80 mm.

Ventilation center and special ventilation building

The structure of the special ventilation building from the elevation minus 6.500 to 18.900 consists of monolithic reinforced concrete walls and ceilings.

The surface part from the elevation 6.000 is made in the form of a steel frame.

Sealing metal lining

There is a steel lining in general available for inspection and checking in the process of mounting and operation is made on the inner and outer surface of the containment to ensure tightness.

Lining of the containment from the side of rooms is covered with anticorrosion protection.

Checking tightness in the places of mounting joints is performed with the help of air filling at excess pressure 2.0 kgf/cm2 (0.2 MPa) into chambers formed by special bars, or with the help of dilution 0.2 kgf/cm2 (0.02 MPa) by means of vacuum suction cups.

Welds of the containment elements, which were produced and inspected in the factory conditions (in particular, by means of physical inspection methods), are subject to visual inspection in the process of construction, acceptance and operation. In case of revealing defects and after their correction, it is necessary to check tightness by means of vacuum chamber or welding of a bar or filling of the generated chamber with air at excess pressure of 2.0 kgf/cm2 (0.2 MPa).

Welds of the sealed lining in places not accessible for inspection during operation are performed with increased scope of inspection in accordance with requirements of VU-2S-83.

Special walls functioning as a screen are envisaged in the box structure for the protection of the sealing lining from flying object.

Reinforced concrete enclosing structures

RCEs of the containment system include:

- steam generator box with internal premises;
- corridor for release of steam-gas mixture from premises of RCP and steam generators to the accident confinement shaft;
- premises of recirculation facilities;
- reactor shaft;
- accident confinement shaft.

Reinforced concrete enclosing structures are designed for the perception of the following loads:

- in normal operation:
  - own weight of building structures and process equipment;
  - mounting loads;
  - temperature inside the containment up to 60 ^{0}C (333 K);
  - wind and snow loads;
  - temperature climatic impacts;
- in emergency operation:
  - own weight of building structures and process equipment;
  - excess pressure of 1.725 kgf/cm2 (0.1725 MPa) or a dilution 0.2 kgf/cm2 (0.02 MPa);

- short-term temperature increase up to 127 ^{0}C (400 K);

– in a special combination of loads considering seismic impact with magnitude 6 intensity.

Penetrations, hatches, manholes, doors

The walls of sealed premises where they are intersected by piping or electric cables are equipped with a sealed penetration preventing release of aerosols beyond the sealed premises.

Types of tube penetrations are determined by structural peculiarities of piping layout.

Penetrations are made fixed for isolated and non-isolated piping.

Tube penetrations are designed to withstand the flow in case of piping complete break. However, tightness of the penetration is not disturbed.

Laying of cables through walls and ceilings is performed through special tight cable penetrations. At the same time, penetrations with current carrying rods are used for power cables that pass through special electroceramic pipes and tight cable penetrations are used for I&C cables.

ANNEX G

Nuclear research reactor

1. An example of the list of components and structures included in the NRR AMP

INR of NASU	Ageing Management Program for Reactor Pressure Vessel	Page 51
P-2-134-09/10	(Tank), Piping and Equipment of VVR-M Primary Side	Pages in total 63

Annex 1

APPROVED
Reactor Chief Engineer
V.M. Makarovskyi

LIST
of VVR-M Components Subject to Ageing Management

No.	Indication	Class	Spreading rules, group, category, etc.	Type, factory number (limit, passport No. for piping, limit for cables)	Commissioning date	Design lifetime expiry date	Notes
1	2	3	4	5	6	7	8
1	List of critical components						
1.1	List of critical components and systems important to safety, replacement or recovery of which are not possible due to high radiation and economic inexpediency						
1.1.1	Pressure vessel (tank) with devices Sb. 01-74	1		Passport of reactor pressure vessel (tank), factory No. 2, registration No. 3IR, SNRIU	1959	Life not established by the design	List of equipment and limits for piping sections are presented in para. 2.1.1.1 of AMP
1.1.2	Part of primary side piping with equipment Sb. 27-VVR-M	1		Passport of piping Sb27-VVR-M Sb27k-VVR-M Piping does not have factory number. SNRIU Reg. No. 82 (4IR)	1959 1989	Life not established by the design	List of primary side equipment is presented in para. 2.1.1.2 of AMP
				Main equipment of primary side registered by the SNRIU: heat exchanger, 1st under No. 83 heat exchanger, 2nd under No. 84 Pump casing: factory No. 38 – 1st under No.6IR factory No. 40 – 2nd under No.8IR factory No. 36 – 3rd under No. 5IR factory No. 41 – under No. 9IR factory No. 39 – under No. 7IR	 1989 1989 1959 1959 1959 1959 1959	 to 1000 cycles to 1000 cycles Life not established by technical documents	
2	List of normal operation components that does not affect safety, replacement of which is impossible due to technical conditions						
2.1	Piping and equipment of the secondary side			Passport of piping Piping does not have factory number. Registered in the INR of NASU. Reg. No. 9	1959	Life not established by the design	List of equipment is presented in para. 2.1.2.1 of AMP
3	Additional list of reactor components (systems)						
3.1	Special ventilation system V-2			Passport of the ventilation facility V-2A – factory No. 7713 V-2B – factory No. 01/11 – 09511-001	January 1960 2011	Life not established by the design and technical documents	List of equipment is presented in para. 2.2.1 of AMP

2. Description of main structures and components of pressure vessel

The main structural component of the reactor is the reactor pressure vessel (tank) (

Figure G.1). The reactor pressure vessel (tank) has a cylindrical form and is made of aluminum alloy SAV-1; thickness of cylindrical part walls is 16 mm; diameter of the tank is 2300 mm and the height is 5340 mm. Elliptical bottom with a height of 270 mm and thickness of 20 mm is welded to the cylinder from below (argon-arc welding). From above, the pressure vessel (tank) is covered by process closure head made of aluminum alloy SAV-1 with a thickness of 40 mm. The tank cavity up to the level of 5000 mm is filled with unsalted water (distillate), which serves as a coolant, neutron moderator and biological protection. The capacity of water in the tank is 22 m3 (40 m3 of distillate in total in the primary side). Different purpose nozzles are welded to the tank bottom: three nozzles with a wall thickness of 20 mm, two of which have an internal diameter of 385 mm, to which piping of the primary side is connected (suction and discharge) and one with an internal diameter of 310 mm (according to the design, it was intended for the return of water after deaerator, and when deaerator switches off, it is used for the return of water to the tank after the filtration system); four nozzles with an internal diameter of 124 mm, one of which is used for the connection of overflow pipe, second one is connected to the drainage pipe, the other two are used for penetration of experimental devices from the upper closure head under the bottom); four nozzles, one of which with inner diameter of Ø 68 mm, three - Ø 90 mm – are used as transport channels (after irradiation of samples) to hot cells; three nozzles Ø 32 mm, two of which are used for the connection of a device for measuring water level in the tank, and one for water sampling for chemical analysis.

The following components are fastened on the process closure head (top of the tank): a cross with control rod channels (9 pcs), isotopic channels and channels for radiation materials science activities; transport channels, experimental channels (2 pcs) going to flanges under the tank bottom; channels of ionization chambers (9 pcs); channel for light inside the tank; device for transportation of spent FAs from the core to spent fuel storage facility and vice versa; channels for initial loading of fresh FAs and irradiated samples; two channels that end in the bottom of the heat column.

In the lower part of the pressure vessel (tank), there is a core that is a cylindrical bearing structure made of aluminum alloy SAV-1 – pipe with a diameter of Ø 940 mm, height of 1080 mm, wall thickness of 12 mm: the pipe in the lower part has a taper, which passes into the cylinder, which by its lower part is welded into the reactor pressure vessel (tank) bottom. There is a suction nozzle filter in the cylinder, which ends at the bottom of the tank by a nozzle with flange, to which the piping of the primary side is connected.

In the lower part of the cylindrical structure of the core, on the upper edge of the conical part, there is a support grid of the core and guide grid. There are 262 holes with a diameter of Ø 14 mm for FAs and 690 holes with a diameter of Ø 16 mm for the circulation of water for cooling in the support grid with a diameter of Ø 760 mm and thickness of 40 mm (material – SAV-1). There are 9 holes with a diameter of Ø 35 mm through which control rod channels pass in the grid center.

The support grid is an important component of the reactor in terms of operational safety.

There is a guide grid with a diameter of 760 mm and thickness of 20 mm located above the support grid at a distance of 20 mm from it.

The guide grid has 93 holes of triangular shape with rounded vertices (instead of angle – arc R=12 mm) and three holes with a diameter of Ø 27mm, which are cells for the installation of single FAs; 96 holes with a diameter of Ø 16 mm, through which all water passes for grid cooling.

Horizontal experimental channels (9 pcs) are located in the center (along the height) of the core and they are fastened in the pressure vessel (tank) by welding (sealed weld). In the pressure vessel (tank), there is also a heat column, whose bottom directly adjoins the core. Parts of horizontal experimental channels and thermal column are components of the reactor pressure vessel (tank).

Under the tank, there are radial ribs welded to the bottom used for stability during installation of the tank on the support plate.

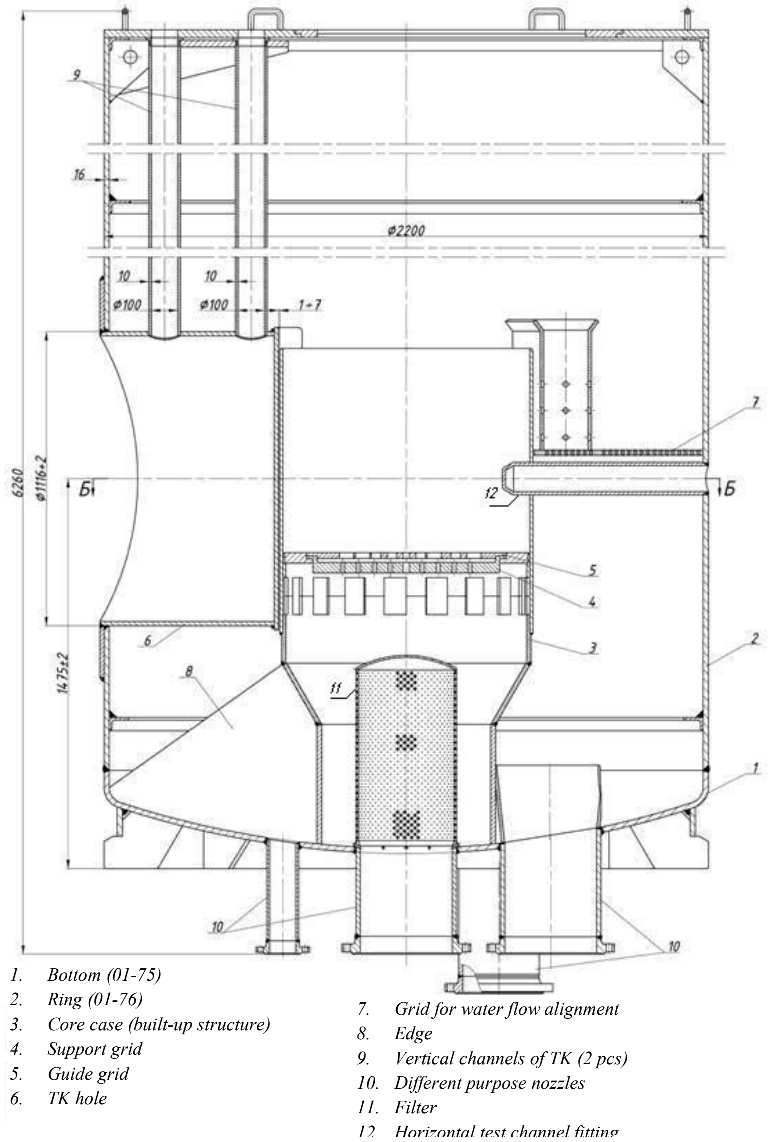


Figure G.1 Vertical section of VVR-M pressure vessel (tank)