NATIONAL REPORT
OF THE REPUBLIC OF BELARUS
ON THE BELARUSIAN NPP
OBJECTIVE SAFETY REASSESSMENT (STRESS TESTS)

Minsk, 2017

TABLE OF CONTENTS

LIST OF ABBREVIATIONS

AC	Alternating Current
AOO	Anticipated Operational Occurrences
APCS	Automated Process Control System
AR	Absorbing Rod
ARMS	Automated Radiation Monitoring System
BDBA	Beyond Design Basis Accident
BDBAMG	Beyond Design Basis Accidents Management Guideline
BRU-A	Atmospheric Steam Dump Valve
BRU-K	Turbine Bypass Valve
CPS CR	Control Rod of the Control and Protection System
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DG	Diesel Generator
DiD	Defense in Depth
EPSS	Emergency Power Supply System
ECB	Electrical Connection Block
ECCS	Emergency Core Cooling System
ECR	Emergency Control Room
EFWEP	Emergency Feed Water Electrical Pump
EOP	Emergency Operation Procedure
EPS	Emergency Protection System
FA	Fuel Assembly
FE	Fuel Element (synonym of fuel road)
FR	Fuel Rod (synonym of fuel element)
GSZ	General Seismic Zoning
HPH	High Pressure Heater
I&C	Instrumentation & Control
LPH	Low Pressure Heater
MCR	Main Control Room
MES	Ministry of Emergency Situations
MSIV	Main Steam Isolation Valve
NO	Normal Operation
NPP	Nuclear Power Plant
PHRS	Passive Heat Removal System
POSV	Pilot-Operated Safety Valve
PPE	Personal Protective Equipment

PSA	Probabilistic Safety Analysis
PES	Possible Earthquake Source
RC	Reactor Compartment
RCP	Reactor Coolant Pipeline
RCPU	Reactor Coolant Pump Unit
RP	Reactor Plant
SAMG	Severe Accident Management Guidelines
SAR	Safety Analysis Report
SG	Steam Generator
SFP	Spent Fuel Pool
SMA	Seismic Margin Assessment
SNF	Spent Nuclear Fuel
SS	Safety System
SSE	Safe Shutdown Earthquake
TCCP	Technical Code of Common Practice
UHL1	Climatic design: moderate and cold climate (from -60 ^{0}C to +40 ^{0}C)

INTRODUCTION

On March 11, 2011 a serious accident happened at Fukushima-1 NPP (Japan) triggered by an earthquake and following tsunami. In this regard operating organizations and regulatory bodies faced the need for a detailed analysis of the causes and lessons learned as well as for the development and implementation of actions to prevent serious accidents caused by extreme and low probability events and to mitigate their negative impact on members of the public and environment.

On March 25, 2011 the European Council announced that safety at European nuclear power plants should be reviewed on the basis of a comprehensive and transparent risk assessment (stress tests). During the fifth Review Meeting of Convention on Nuclear Safety (April 4 to 14, 2011), the state parties within their joint declaration on Fukushima Daiichi accident outlined the need to reassess NPP safety and develop additional measures to improve NPP safety.

On May 13, 2011 the European Nuclear Safety Regulatory Group (ENSREG) and the European Commission agreed upon the technical requirements for stress tests of European nuclear power plants (ENSREG Declaration, Annex 1: EU stress tests specifications). In accordance with ENSREG, the technical requirements of these stress tests are an objective reassessment of nuclear power plants in the light of the events at Fukushima-1. Stress tests should include a detailed analysis of very severe natural phenomena and any combinations that could affect the NPP safety functions and that can result in a severe accident.

In June 2011, the Republic of Belarus acceded to the Joint Declaration of the European Union and neighboring states on comprehensive risk and safety assessments of nuclear plants (stress tests) and made commitments for their implementation.

As a part of the preparation for the Belarusian NPP stress tests, the Ministry of Emergency Situations of the Republic of Belarus has established their regulatory framework. In particular, TKP 566-2015 Evaluation of the Frequency of Severe Damage to the Reactor Core (for external source of natural and man-made events) [22] was developed and approved by a Resolution of the Ministry of Emergency Situations dated April 28, 2015 No. 21. To improve the stress tests regulatory requirements taking into consideration international recommendations in 2017, norms and regulations on nuclear and radiation safety "Requirements to stress tests (objective safety reassessment) of a nuclear power plant" approved by the Resolution of the Ministry of emergency situations of the Republic of Belarus dated 12.04.2017 No. 12 [23] were adopted. This document was developed with the support of the European Union International Technical Assistance Project.

Under the guidance of the above technical regulatory legal acts, the operating organization (Belarusian NPP State Enterprise) performed stress tests under the project in 2016-2017 and prepared a relevant report. Belarusian NPP design development organizations (ASE, JSC Engineering Company and Atomproekt, JSC) were involved in the safety reassessment activities. A report on the objective safety reassessment (stress tests) of the Belarusian NPP was agreed with the Head designer of the Belarusian NPP reactor unit – JSC OKB Gidropress, and with the Research Supervisor of the NPP and reactor unit projects development –National Research Center Kurchatov Institute.

This report has been prepared by the interagency working group lead by the head of the Department for nuclear and radiation safety of the Ministry of emergency situations of the Republic of Belarus (hereinafter referred to as Gosatomnadzor), founded on May 4, 2017 by order of the Prime Minister of the Republic of Belarus.

The interagency committee for coordinating and monitoring implementation of the plan of key organizational actions for the nuclear power plant construction in the Republic of Belarus , approved by Resolution of the Council of Ministers of the Republic of Belarus dated November 5, 2012, No. 1010, agreed to the Action Plan (road map) for establishing and implementing activities developed based on the results of the objective safety reassessment of the Belarusian NPP (Protocol No.03/37pr-dsp dated 27.09.2017) which are intended to improve the safety level of the Belarusian NPP.

1. LEGAL AND REGULATORY FRAMEWORK

1.1 Establishing and Maintaining the Legal and Regulatory Framework for Nuclear Power Plant Safety Assurance.

The Republic of Belarus has announced adoption of global nuclear safety procedures and commitment to the implementation of nuclear power program in compliance with international conventions and treaties by their ratification and adoption [1-14].

Nuclear safety activities are carried out in the Republic of Belarus in accordance with international practice and IAEA recommendations. Safety principles defined in IAEA Safety Fundamentals publication No. SF-1 “Fundamental Safety Principles” as well as the general provisions of other IAEA safety standards are built into the fundamental Law of the Republic of Belarus “On the use of nuclear energy” and regulations of the Republican state administration bodies and other state organizations applicable to nuclear energy use.

In June 2011, the Republic of Belarus acceded to the Joint Declaration of the European Union and neighboring states for comprehensive risk and safety assessments of nuclear plants (stress tests) and made commitments for their implementation.

1.2 National Requirements and Regulations in the Sphere of Nuclear and Radiation Safety

Legal safety regulation in the sphere of nuclear energy use in the Republic of Belarus features a hierarchical structure and provides for the subordination of lower legal force documents to the relevant requirements of the higher legal force documents, and is applied based on the Constitution of the Republic of Belarus and in accordance with:

laws of the Republic of Belarus;

normative legal acts of the President of the Republic of Belarus;

resolutions of the government of the Republic of Belarus;

normative legal acts of the Republican state administration authorities executing state regulation of nuclear power application safety;

rules and regulations related to ensuring nuclear and radiation safety, standard practice technical codes and other technical regulations.

Since a decision on the nuclear power program implementation in Belarus [15] was made in 2008, the regulatory framework in the sphere of nuclear and radiation safety has undergone significant changes. By 2017, top-level legislation (Decrees of the President of the Republic of Belarus, laws of the Republic of Belarus, Resolutions of the Government of the Republic of Belarus) had been established. The basis is formed by:

Law of the Republic of Belarus No. 122-3, dated January 5, 1998 “On radiation safety of members of the public” [17];

Law of the Republic of Belarus No. 426-3, dated July 30, 2008 “On nuclear energy use” [16];

Decree of the President of the Republic of Belarus No. 450, dated September 1, 2010 “On the licensing of specific types of activities” [18];

Decree of the President of the Republic of Belarus No. 62, dated February 16, 2015 “On safety during the construction of the Belarusian nuclear power plant” [19];

Law of the Republic of Belarus “On radiation safety to members of the public” defines fundamental legal regulations in the sphere of radiation safety to the members of the public and is intended to create conditions which ensure protection of people’s lives and health from harmful ionizing radiation exposure.

Law of the Republic of Belarus “On nuclear energy use” regulates the matters related to the design, allocation, construction, commissioning, operation, limits on operational characteristics, extending operational life and decommissioning a nuclear plant and/or storage facility, as well as any matters related to nuclear materials handling in a nuclear plant and/or storage facility, spent nuclear materials and/or operational radioactive waste, and other matters in the sphere of nuclear power application.

Decree of the President of the Republic of Belarus No. 450, dated September 1, 2010 “On the licensing of specific types of activities” regulates the licensing of activities related to nuclear energy and ionizing radiation sources; determines the powers of state administration authorities for issuing, suspending, terminating and renewing licenses as well as to verify compliance with licensing requirements and conditions by licensees; establishes general and specific licensing requirements and conditions in the sphere of nuclear power use.

Decree of the President of the Republic of Belarus No. 62, dated February 16, 2015 “On safety during the construction of the Belarusian nuclear power plant” establishes a special procedure of control arrangement and implementation related to safety assurance during the plant construction and commissioning.

Considering the functions assigned by the Resolution of the Prime Minister of the Republic of Belarus No. 158r, dated 04.05.2017 [21], the Ministry of Emergency Situations (MES), in the person of Gosatomnadzor, acted as a supervisor for preparing the National report on the Belarusian nuclear power station objective safety reassessment (stress tests).

As part of preparatory activities for the stress tests of the Belarusian NPP, the MES established the legal and regulatory framework their performance. In particular, TKP 566-2015 “Evaluation of the frequency of severe damage to the reactor core (for external source of natural and man-made events)” [22] was developed and approved by a Resolution of the Ministry of emergency situations dated April 28, 2015 No. 21 [22]. Regulatory requirements to improve stress tests taking into consideration the international recommendations in 2017, norms and regulations on nuclear and radiation safety “Requirements to stress tests (objective safety reassessment) of a nuclear power plant” approved by the Resolution of the Ministry of emergency situations of the Republic of Belarus dated 12.04.2017 No. 12 [23] were adopted. This document was developed with the support of the European Union international technical assistance project.

1.3 Licensing System

The system of licensing activities related to nuclear energy and ionizing radiation sources, ensuring nuclear and radiation safety has been established in the Republic of Belarus. Basic regulatory requirements related to nuclear safety are determined by the regulation of licensing of specific types of activities, approved by Decree of the President of the Republic of Belarus No. 450, dated September 1, 2010 “On licensing of specific types of activities” [18]. As licensed, activities related to nuclear energy and ionizing radiation sources are defined. Licensing authority is MES.

Major licensed activities include:

activities related to nuclear power use;

activities related to sources of ionizing radiation use;

activities related to radioactive waste management;

activities about the design and manufacture of processing equipment for the facilities using nuclear energy, design and manufacture of radiation protection equipment;

activities for the examination of safety in using atomic energy and sources of ionizing radiation.

Activities related to nuclear energy use include the following work and/or services:

design, siting, construction, operation, and decommissioning of nuclear plants;

design, allocation, construction, operation, and decommissioning of nuclear materials storage facilities

handling of nuclear materials, nuclear fuel, spent nuclear materials, spent nuclear fuel and operational radioactive waste;

performance of works and provision of services to operating organizations affecting safety including construction of facilities.

Unlicensed implementation of these activities shall be illegal.

MES has delegated the functions of organizing of licensing activities related to nuclear energy and ionizing radiation sources to Gosatomnadzor, being its structural division.

The Government of the Republic of Belarus approved the examination procedure for the documents substantiating provision of nuclear and radiation safety when carrying out activities related to nuclear energy and ionizing radiation use [24].

Documents substantiating provision of nuclear plant safety, design, process and operational documentation for nuclear facility to be submitted by the license applicant to the MES, are subject to a nuclear plant safety examination. Appointed by the MES, the safety examination is carried out to determine the safety level of a nuclear facility by matching the design concepts adopted and their implementation results with the requirements of normative legal acts, including technical normative legal acts in the sphere of nuclear and radiation safety.

Nuclear facility safety assessment shall be performed on a regular basis over the entire life cycle of the nuclear facility by the operating organization and Gosatomnadzor [6].

License shall be issued only when the results of the nuclear facility safety evaluation are positive. At the construction stage of nuclear facilities, design concepts and measures are taken to ensure their safety is being evaluated when licensing.

Issues addressed in the stress tests are not subject to review under the standard licensing procedure.

Belarusian NPP operating organization which has obtained, in line with established procedures, a license to operate in the field of nuclear energy in terms of construction is the Belarusian nuclear power plant Republican Unitary Enterprise.

Belarusian nuclear power plant Republican Unitary Enterprise (Belarusian NPP State Enterprise) was founded by a Decree of the President of the Republic of Belarus No. 583, dated December 30, 2013 “On the reorganization of state institution ‘Directorate for nuclear power plant construction’”, Order of the Ministry of energy of the Republic of Belarus No. 1, dated January 10, 2014 as a result of reorganizing state institution “Directorate for nuclear power plant construction” by its transformation into a Republican unitary enterprise. Ministry of energy of the Republic of Belarus is the enterprise founder and its Republican state administration body.

The enterprise is the operator (operating organization) of the nuclear power plant and assumes all responsibility for its safety during the entire life cycle stages of the Belarusian NPP [33]

1.4 Regulatory Control and Assessment System

As per the provisions of Law of the Republic of Belarus “On nuclear energy use” [16] MES shall:

establish state supervision for ensuring nuclear and radiation safety as well as physical protection of nuclear power facilities;

arrange and perform state supervision of spent nuclear materials management and radioactive waste management;

monitor compliance with nuclear and radiation safety legislation;

participate in organizing and performing activities for assessing equipment, products and technologies for nuclear power facilities;

provide for a functioning unified state accounting and control system for ionizing radiation sources and a state accounting and control system for nuclear materials in the Republic of Belarus;

arrange safety examinations of the nuclear plant and/or storage facility, as well as their projects, including the involvement of independent experts.

Pursuant to the Decree of the President of the Republic of Belarus No. 756, dated 29.12.2006 “On several issues of the Ministry of emergency situations" [20], Gosatomnadzor, being a structural division acting as a legal entity of the MES headquarters, has been delegated to establish special functions in connection with providing nuclear and radiation safety, such as: providing state supervision of nuclear and radiation safety; monitoring of nuclear and compliance with radiation safety legislation.

Safety related regulatory control is based on fundamental functions of safety assurance and defense-in-depth principles which correspond to up-to-date international practices in this area, as reflected in IAEA documents.

The Belarusian NPP is the only facility of its kind in the country. As a consequence, special procedures for establishing and organizing safety control during construction and commissioning of the plant [19] has been established at the construction site enabling supervising authorities to perform continuous monitoring (supervision) in their sphere with application of sanctions and other measures. Its organization and implementation procedure is established by the Government of the Republic of Belarus [25]. Such a control organization to ensure safety during plant construction and commissioning is being allowed to proceed with further steps for implementation of the IAEA recommendations in national legislation to achieve the highest standards of safety.

Overall coordination of the supervision over the Belarusian NPP construction is carried out by the Ministry of emergency situations through the Working group to coordinate supervision over the Belarusian NPP construction. The above-mentioned group is under the leadership of the first Deputy Minister of emergency situations of the Republic of Belarus includes representatives of all monitoring (supervisory) authorities. The Working group authority is established by the Government resolution [26]. It is noted that the above coordinating mechanism of monitoring and supervision activities has been valued as a “good practice” by the IAEA mission experts of the Integrated Regulatory Review Service (IRRS) held on October 3 to 14, 2016 in the Republic of Belarus.

Areas of nuclear and radiation safety provision to be assessed by the regulatory authority are based on established international practice as shown in the IAEA documents. They include: site characteristics for NPP construction, the NPP project, the operating organization management system, personnel competence, emergency planning and response, defense-in-depth concept, consideration of global best practices, etc.

The scope and contents of documents substantiating nuclear and radiation safety are subject to submission to the regulatory authority within the licensing procedure as defined in the terms of the legislation. The regulatory authority has sufficient powers to carry out expert assessment of the documents submitted in respect to nuclear and radiation safety issues. The current regulatory control and assessment system provides the regulatory authority with fairly extended powers both to implement the Report provisions in the Belarusian NPP safety system and to monitor the results of such implementation.

National legislation provides the basis for the establishing measures aimed at increasing functional stability and security of the organizations in conditions of natural and man-made factors that could result in emergency situations. Development of these measures is mandatory for organizations [27]. The regulatory authority is empowered for the development and distribution to the organizations, operating nuclear energy facilities, proposals necessary for consideration related to issues to improve the stability of such facilities [20].

The Report on the Belarusian nuclear power station safety reassessment (stress tests) evaluation (hereinafter – the Report) was carried out as part of existing procedures for interaction with the operating organization and technical support organization, involving experts having the required level of technical competence.

The estimates presented in this Report are derived and calculated using verified and validated software tools [31].

1.5 Enforcement of Applicable Regulations and License Conditions

Pursuant to the Statute of Gosatomnadzor (approved by Decree of the President of the Republic of Belarus No. 756, dated 29.12.2006 [20]), Gosatomnadzor has the authority to monitor compliance with the nuclear and radiation safety legislation. Should any violations of the applicable legislation be discovered in the process of supervisory activities, either a protocol on an administrative offence(s) can be issued and/or a ruling on the case of an administrative offense can be delivered. On the basis of the report issued, a judgment or a formal request shall be rendered on the elimination of violations which were discovered during the inspection. Judgment as well as formal request on the inspection report shall be binding.

As provided for in the legislation, when violations constitute a threat to national security, inflict harm to human life and health, the environment, then a formal request shall be rendered to suspend the activities of the inspected entity, workshops (production sites), equipment, operation of vehicles, manufacture and sales of goods (works, services) on the day when the violation was discovered. This formal request should specify the time period of the suspension and the timing for informing the supervisory authority of eliminating the violations x. When there is evidence proving essential element of offence availability the supervisory authority is authorized to pass the inspection materials to criminal prosecution authority.

The entity under inspection has the right to challenge decisions of the supervisory authorities rendered based on the inspection report as well as formal requests for the eliminating violations, actions or inactions of inspectors.

Should violations of the licensing legislation, license requirements and conditions by the licensee (his employee and separate subdivision) be discovered, licensing or other regulatory (supervising) authority shall issue the licensee, in line with established procedures, a formal request (improvement notice) to eliminate the violations discovered and shall set a remedial period. This period cannot last longer than 6 months.

If the licensee has not eliminated the violations set forth in the formal request (improvement notice) on elimination of the violations discovered, or a written notice on elimination of the violations was not submitted to the licensing or other regulatory (supervising) authority, or licensing or other regulatory (supervising) authority discovered violation of special licensing requirements and conditions by the licensee (his employee and separate subdivision) within the set period, the licensing authority either on its own initiative or at the instance of the other regulatory (supervising) authority shall make a decision on the license suspension for up to 6 months.

When the violations resulting in the license suspension are not eliminated by the licensee within the set period, or a written notice on elimination of the violations has not been submitted to the licensing or other regulatory (supervising) authority, the licensing authority which issued the license shall decide on terminating the license.

Failure to comply with requirements of the legislation related to nuclear and radiation safety shall be subject to administrative accountability pursuant to the Code of the Republic of Belarus on administrative offences or criminal liability under the Criminal code of the Republic of Belarus [28, 29]. Criminal liability will be incurred when the facts indicating essential elements of offence stipulated in the Criminal Code of the Republic of Belarus, are revealed during the inspection.

In addition, within the scope of licensing and permitting activities, there is a possibility to complement the licenses issued for activities related to nuclear energy use with specific licensing requirements and conditions for nuclear energy using facilities. These specific licensing requirements and conditions may include, in particular, the activities targeted at improving safety, which are developed within review of the report on stress tests (objective safety reassessment) of the Belarusian nuclear power plant.

1.6 Transparency and Public Awareness

Pursuant to the Statute on the Ministry of emergency situations of the Republic of Belarus (approved by Decree of the President of the Republic of Belarus No. 756, dated 29.12.2006) [20], the MES shall inform (notify) in accordance with the law the Republican state administration authorities, local executive and administrative authorities, other organizations and the members of the public on nuclear and radiation safety issues. Public awareness of the radiation facilities, nuclear plants, nuclear power facilities safety performance is also included in the Statute on Gosatomnadzor.

In the process of preparing the report on the objective safety reassessment (stress tests) of the Belarusian nuclear power plant Gosatomnadzor in line with its information and communication strategy [30] performed active information work. The National report is planned to be placed also on the publicly available Gosatomnadzor website.

2. GENERAL INFORMATION ON THE NPP SITE AND GENERAL CHARACTERISTICS OF NPP UNITS

2.1 Brief Description of Characteristics of the NPP site

Administratively, the Belarusian NPP construction site is located in the North-East of Grodno region, in the Ostrovets district, 19 km North-East of Ostrovets town and 23 km East of the border with the Republic of Lithuania.

Geographical position of the Belarusian NPP site is localized by coordinates as follows: latitude 54°46', longitude 26°07'.

The site is graded. Absolute elevation is 174.5 to 182.7 m BES.

The shortest distance to the nearest administrative borders of the neighbor regions are:

Minsk region is 28 km to the East;

Brest region is 150 km to the South;

Vitebsk region is 29 km to the North.

The closest neighboring states are: the Republic of Lithuania is 23 km, the Republic of Latvia is 110 km, the Republic of Poland is 200 km, the Russian Federation is 150 km and the Ukraine is 320 km.

The nearest administrative districts of the Grodno region are: the Smorgon district is 12.5 km, Oshmyany district is 31 km; Minsk region: Myadel district is 16.5 km and the Postavy district of the Vitebsk region is 22.5 km.

In a 100 km area surrounding the NPP site, one city with a population of over 100 thousand people is located 49.5 km away, i.e. Vilnius (the capital of the Republic of Lithuania).

a West-Belarusian subarea within the Naroch-Vileika lowland and represents low-hilly and undulating Poozersky plain , raised at 140 to 180 m above sea level. The highest point of Ostrovets district with an elevation of 309 m is 30 km to the West of the NPP location site.

The hydrographic system includes the major river Viliya (5 km to the North) with its tributaries: the Stracha (4 km), the Oshmyanka (5.5 km), the Losha (9 km) is a tributary of the Oshmyanka, the Gozovka, the Polpe and the Sorochanskie group of lakes (10 km), the Olkhovskoe reservoir (7 km), and finally shallow rivers and ditches. The Viliya river also flows through the territory of Lithuania (the Neris), into the Neman river. In the NPP surveyed zone, the river Vilia (5 km to the North) and 3 its small tributaries flow on the left: the Gozovka, 17 km long, Polpe, 9.3 km long, Losha, 9 km long.

The Vilia river is the process water source.

According to official sources, there are 595 settlements with a population totaling 46097 in the 30 km area around the plant.

2.2 Basic Characteristics of the Units

The Belarusian NPP includes 2 units which are presently under construction. The Belarusian NPP unit consists of a water cooled water moderated thermal neutrons reactor plant VVER -1200 and turbine-generator plant . For conversion and transfer of energy from the reactor to the turbine generator the design provides for the two-circuit heat diagram. . Heat generated by the nuclear fuel fission in the reactor core is transferred to the primary coolant. The primary coolant enters the steam generators through four MCP circulating loops. The steam produced in the steam generators is delivered to the turbine.

The NPP is composed of two power units with thermal power of 3200 MW and electrical output of 1194 MW. Pressure inside the 1st circuit is 16.2 MPa.

It is estimated that the power unit No. 1 of the Belarusian NPP will be brought to the critically for the first time in 2019.

Basic characteristics of the NPP unit with VVER-1200 are given in Table 2.2.1.

Table 2.2.1 - Basic characteristic of the NPP unit with VVER-1200

Parameter name	Value
1 Unit's structure	Monoblock
2 Operation lifetime, years: - Unit - reactor plant	 50 60
3 Unit power , MW: - electrical (gross), under warranty conditions - thermal	 1194 3200
4 Unit heating power, MW	46.6
5 Capacity factor, relative units	0.9
6 Auxiliary power consumption (including recycling water supply cost and on-site costs), %	7.0
7 Unscheduled reactor automatic shutdowns, min., 1/year	0.5
8 Average annual duration of scheduled shutdowns (reactor refueling , routine maintenance, PM), days, max.	25
9 Process-related manpower for the first Unit (specific), pers./MW	0.66
10 Number of FAs in the core, pcs.	163
11 Number of FAs in CPS CE.	121
12 Maximum fuel burnup, average across FA, MW·day/kgU	60
13 Fuel life time, year	4 (3)
14 Refueling frequency, month	12 (18)
15 Fuel average enrichment with isotope - U^{235}, %	4.79
16 Average fuel burnup in unloaded FAs for stationary refueling mode, MW·day/kgU	55.5
17 Coolant basic parameters: Primary circuit : - core inlet temperature, °C - core outlet temperature, °C - coolant heating across the core, °C - coolant flow rate across the reactor, m^3/h - core outlet pressure, MPa Secondary circuit: - steam pressure at the SG outlet, MPa - PG steam capacity, t/h	 298.2 328.6 30.4 88000 16.2±0.3 7.0±0.1 1602^{+112*}

- feed water temperature, °C - steam humidity at SG outlet, %, max. - generated steam temperature at the SG steam header outlet, ºC	225±5 0.2 285.8 ± 1 * maximum allowable deviation
18 Turbine plant	К-1200-6.8/50
19 Turbine plant structural formula	2LPC+HPC+2LPC
20 Diagram of regenerative heating	4LPP + D + 2HPP
21 Number of main feedwater pumps, and type of drive	Provisionally: 5 FEP. (electric drive)
22 Generator	Т3В-1200-2
23 Type of generator cooling	Complete water-type
24 Turbine plant circulating water supply circuit	Recycling water supply with evaporative cooling towers
25 Industrial water supply circuit of safety-critical systems	Recycling water supply with spray cooling ponds
26 Fresh fuel and solid radioactive waste storage facility, including: - fresh fuel storage facility - solid radioactive waste storage facility: - solidified liquid RAW (low-level and intermediate-level): - very low-level solid RAW - low-level solid RAW	 499.3 m^2 777.5 m^2 (1st unit) 673.5 m^2 (2nd unit) 38 non-retrievable shielding casks (57 m^3) annually including possible emergency situations Storage facility area is 194.0 m^2 and is designed for 448 casks 27 drums (5.42 m^3) annually 161 drums (32.3 m^3) annually Storage facility area is 133.7 m^2 is designed for 1626

- intermediate-level solid RAW	drums (including very low-level and low-level RAW) Long component storage facility area is 28.5 м^2 Filter temporary storage facility area is 11.76 m^2 25 drums (2.5 m^3) annually Storage facility area is 54.6 v^2 and is designed for 546 drums
- high-level solid RAW	0.5 m^3 annually Storage bay area is 18.6 m^2
27 Availability of spent fuel storage facility	Reactor spent fuel pool (storage pool), spent FA storage system description located in the reactor compartment, as well as systems that provide fuel transportation and installation are given in [31]

28 Double protective containment of the reactor building	
outer protective concrete containment	
- internal diameter, m	50.0
- dome top elevation, m	70.2
- thickness (cylindrical part /dome), m	0.8/0.6
internal leak-tight reinforced concrete containment	
- internal diameter, m	44.0
- dome top elevation, m	67.7 (68.5 including buttress)
- thickness (cylindrical part /dome), m	1.2/1.0
- design overpressure, MPa	0.4
- design temperature, °C	150
emergency annulus air cleaning system from radioactive leaks provides cleaning level not lower than:	
- elemental iodine, %	99.9
- organic iodine, %	99
- aerosols, %	99.99

2.3. NPP SAFETY CONCEPT

2.3.1. Basic NPP Safety Principles and Criteria

2.3.1.1. General Safety Objective

NPP meets safety requirements provided its radiation effect on the personnel, public and environment in normal operation conditions and during design basis accidents does not exceed established exposures of personnel and public as well as normative-based emissions and radioactive substance content in the environment, and limits this impact during beyond design-basis accidents, too.

NPP safety must be ensured by consistent implementation of the defense-in-depth protection based on application of physical barrier system across the path of propagation of ionizing radiation and radioactive substances into the environment and system of engineering and organizational measures to protect the barrier and maintain their performance as well as for protection of personnel, public and the environment.

The main objective of safety assurance at all NPP life cycle stages is implementation of efficient actions aimed at prevention of accidents and protection of the personnel and public by means of prevention of emission of radioactive products into the environment under any circumstances. NPP is considered safe provided :

– its radiation effect on the personnel public and environment in normal operation conditions and during design-basis accidents does not exceed established values;

– in case of BDBA, the radiation exposure is limited to acceptable values.

2.3.1.2. Basic Safety Principles.

At this construction stage the nuclear safety of RP and NPP are determined by the design's technological state of the art, required quality of fabrication, installation, adjustment and testing of safety-critical components and systems, their operational reliability, diagnostics of technical condition of equipment, relevant organization of work , personnel qualifications and discipline.

Belarusian NPP safety concept as defined by the design involves application of a system of technical and organizational measures of the defense-in-depth, including:

- application and development of inherent safety characteristics;
- application of safety systems based on principles of redundancy, spatial and functional independence, single-mode failure;
- application of reliable, practice-proven engineering solutions and validated procedures, design analyses and experimental studies;
- meeting requirements of laws and regulations on RP and NPP safety as well as strict compliance with the requirements set forth by RP and NPP design ;
- stability of technological processes;
- implementation of quality assurance systems at all stages of the NPP construction and operation;
- safety culture development and implementation at all stages of the NPP construction and operation.

2.3.1.3. Design Limits

The RP design is developed for the whole range of NPP conditions.

Depending on the frequency of occurrence of initiating events, the design conditions are divided into four categories:

- category 1: normal operation;

- category 2: faulty conditions (reoccurrence rate of initiating events per year: $f \geq 10^{-2}$);

- category 3: class 1 postulated accidents (reoccurrence rate of initiating events per year is in the range $10^{-2} > f \geq 10^{-4}$);

- category 4: class 2 postulated accidents (reoccurrence rate of initiating events per year is in the range $10^{-4} > f \geq 10^{-6}$).

In addition to the design provisions on fulfilling the requirements of deterministic design conditions, specific provisions on additional design conditions have been considered, which inherently corresponds to beyond design-basis accidents. This is provided to be able to identify the need (based on the analysis results) to implement measures, including upgraded or additional equipment, or accident management procedures to be applied for:

- complex sequences that involve failures beyond those considered in the deterministic design conditions, but do not result in the core melting;

- core melting accidents.

The RP design provides for actions to control the development and mitigation of consequences of such accidents (for beyond design-basis accidents without core melting these are actions to prevent melting; for beyond design-basis accidents with core melting these are actions to reduce pressure in the reactor vessel at the time of its destruction).

For the initiating events of categories 1 and 2 respectively, the operational limit and safe operation limit for the fuel rods that lost tightness in the course of an accident are set according to the relevant rules and regulations:

- permissible amount of fuel rods with a “gas leakage” type defects:

1) 0.2 % of fuel elements – operational limit;

2) 1.0 % of fuel elements – safe operation limit;

- permissible amount of fuel elements in direct contact with fuel and coolant:

1) 0.02 % of fuel elements – operational limit;

2) 0.1 % of fuel elements – safe operation limit.

For design-basis accidents the following eligibility criteria are set basing on the number of fuel rods that lost tightness during an accident

- for category 3 design conditions – max. 1 % of the total number of fuel rods ;
- for category 4 design conditions – max. 10 % of the total number of fuel rods .

For category 3 design basis accidents related to:

- loss-of-coolant accidents (the relevant analysis determines the scale of the primary circuit leakage without boiling crisis);
- design basis accidents of depressurization of the secondary circuit;
- failure to close of one pilot-operated safety valve of the pressurizer ,

safe operational limits of the fuel elements must not be exceeded by core subcriticality, keeping it under a coolant and ensuring crisis-free cooldown taking into account provided design margins as well as response rate and effectiveness of safety systems.

For the design basis accidents and complex sequences the maximum design limit of the fuel elements damage is established which corresponds to the following boundary parameters:

- maximum cladding temperature of the fuel rods is 1200 ºC;
- maximum local oxidation rate of the fuel rod cladding is 18 % of the initial wall thickness;
- maximum portion of reacted zirconium is 1% of its weight in the fuel rod cladding;
- maximum fuel temperature must not exceed the melting point.

The following acceptance criteria are set for severe beyond design-basis accidents with the core melting and escape of the corium outside the reactor vessel:

- concentration of gas mixture generated in the reactor and in space under the reactor after the drop-over of corium shall not reach a hazardous explosive value;
- pressure in the primary and secondary circuits must not exceed the respective pressure values of hydraulic strength test;
- if the core debris cannot be cooled down inside the reactor vessel then, at the time of the melt-through of the reactor vessel , the pressure in the primary circuit coolant system must not exceed 1 MPa;
- permissible impact of the pressure surge on the elements of the concrete cavity is 150 kPa·sec.;
- maximum permissible pressure inside the concrete cavity is 2.0 MPa;
- corium boiling must be excluded;
- subcriticality of the destroyed and molten core must be provided.

2.3.2 Design Concept for Safety Systems

The design provides for the safety systems and special technical devices that perform the following main safety functions under faulty conditions, design-basis accident or beyond design-basis accident:

- System for the reactor emergency shutdown and maintaining it in a subcritical state;
- System for emergency heat removal from the reactor;
- System of nuclear fuel cooling;
- System of confinement of radioactive substances inside established boundaries;
- System of maintaining integrity of the primary circuit boundaries.

Safety systems and their components ensure performance of assigned functions under all external effects considered in the design:

- protective safety systems that perform functions of emergency core cooldown and residual heat removal include active and passive portions; the localizing safety system also includes active and passive systems and components;

– the design provides for a spacial separation of the safety system channels and channel structural protection thus excluding the possibility of common cause failures (due to fire, flooding);

– Safety systems are supplied with electrical power from independent sources (diesel generators) designed in accordance with the requirements for supporting safety systems.

In the normal operation modes of the Unit the protective systems do not function and are in a standby state, they are subject to routine inspections and tests according to the technical regulations.

Safety systems are designed to be fail-safe , including dependent failures and common cause failures, and able to perform functions upon loss of a power supply.

Basic principles applied in relation to the safety systems:

– Single failure principle;
– Channel redundancy (4-channel structure is applied);
– Diversity;
– Physical (spacial) separation;
– Passive action (applies to individual systems, for example: ECCS accumulator tank).

2.3.2.1. Defense- in-Depth Principle

Application of the defense -in -depth principle allows to take into consideration all potential emergency initiating events at a NPP and provide a sufficient and required safety level. The defense -in -depth principle implies application of the barrier system across the way of propagation of ionizing radiation and radioactive substances into the environment, a system of engineering and organizational measures to protect the said barriers and preserve their efficiency when carrying out protection of the population. The engineering design of the Belarusian NPP provides for five barriers preventing unacceptable release of radioactive substances and ionizing radiation.

Barrier system for NPP:

– Fuel matrix;
– Fuel element cladding;
– Reactor plant primary coolant circuit boundaries;
– Reactor plant containment ;
– Biological protection.

To ensure an efficient protection of barriers of the Belarusian NPP, the design provides for a multilevel defense .

Each defense level of NPP provides certain protection of barriers against impacts typical for that level. For each level the design provides relevant technical and/or organizational activities that prevent and/or mitigate results of the above impacts effects by limiting normal operation until the termination of NPP operation in order to prevent the NPP transition from a higher level of defense to a lower one or to provide mitigation if such prevention fails as well as to bring back the NPP from a lower defense level to a higher one.

Application of the multilevel defense ensures implementation of the requirements of accounting completeness of all emergency initiating events possible at the NPP and safety measures as low as reasonably practicable.

– Level 1: NPP siting conditions and prevention of anticipated operational occurrences ;

– Level 2: Prevention of design-basis accidents by the normal operation systems;

– Level 3: Prevention of beyond design-basis accidents by the safety systems;

– Level 4: Control of beyond design- basis accidents;

– Level 5: Emergency planning.

Technical resources provided at Level 1

The objective of Level 1 is to ensure the NPP safety by running the normal operation within the specified normal operation limits and conditions, by means of a technical devices designed for normal operation, and organizational measures aimed at maintaining efficiency of the barriers.

Operation of systems at level 1 is aimed at ensuring a cost-effective mode of power delivery to customers and maintaining the operational limits and conditions. Normal operation of the Unit is ensured by automatic control systems, shutdowns and interlocks implemented by NO I&C. The first level of defense features predominantly an automated control, i.e. control by means of automation equipment with involvement of the personnel. The design provides for a system of control, monitoring, and diagnostics of the reactor plant that perform diagnosis in the process of operation of the reactor plant main process equipment.

Objectives and functions of defense provided at level 2

Main objectives to be solved at defense of level 2:

– identification of deviations from normal operation and their correction ;

– control in the process of operation with deviations.

Level 2 technical resources are represented by the normal operation systems. Basic functions of defense of level 2 are:

– Bringing the reactor into a subcritical state;

– Maintaining the steam generator feedwater inventory , heat removal and RP cooldown via the secondary circuit;

– Maintaining a coolant inventory in the reactor core, heat removal and RP cooldown via the primary circuit;

– Limiting release of radioactive substances into the environment;

– Providing the backup power supply for normal operation systems;

– Ensuring monitoring and control of the normal operation process systems, including safety-related systems.

Objectives and functions of defense provided at level 3

Main objectives to be solved at defense of level 3:

– prevention of escalation of initiating events into design-basis accidents, and design basis accidents into beyond design basis accidents by means of safety systems;

– in case of a failure to prevent an accident, mitigation of the accident consequences by confinement of released radioactive substances.

Main functions of defense of level 3 and relevant technical devices :

– Bringing the reactor into a subcritical state;

– Maintaining the coolant inventory in the reactor core under a high pressure in the primary circuit;

– Maintaining the coolant inventory in the reactor core and the residual heat removal under a low pressure in the primary circuit;

– Pressure reduction the in the primary circuit during the primary-to-secondary circuit leakage;
– Discharge of non-condensable gases from the primary circuit;
– Heat removal from the reactor core through the steam generators;
– Heat removal from the containment ;
– Protection of the containment against under-pressure ;
– Isolation of the containment
– Isolation of reactor compartment systems along the seismic resistance boundary ;
– Isolation of the emergency steam generator from the secondary circuit systems and the external environment;
– Protection of the primary circuit against overpressure protection;
– Protection of the secondary circuit against overpressure;
– Removal of hydrogen removal from the containment ;
– Providing backup power supply from the emergency diesel generators;
– Providing emergency power supply from the batteries;
– Ensuring monitoring and control of safety-related process systems.

Objectives and functions of defense provided at level 4

Main objectives to be solved at defense at level 4:

– prevention of progression of beyond design-basis accidents and mitigation of their consequences;
– protection of the containment against destruction during beyond design basis accidents and maintaining its operation capability;
– restoration of a controllable state of NPP under which the fission chain reaction is terminated, nuclear fuel continuous cooldown is ensured, and radioactive substances are confined within the design boundaries.

Level 4 technical devices are represented by the normal operation systems, the safety systems as well as additional technical devices controlling BDBA.

Basic functions of defense level 4:

– Bringing the reactor into a subcritical state;
– Heat removal and reactor cooldown via the steam generators;
– Heat removal from the containment;
– Pressure reduction in the primary circuit to prevent a scenario with a corium exit from the reactor vessel under a high pressure;
– Reduction of gas-aerosol emissions through double containment leakages;
– Suppression of the hydrogen inside the containment;
– Confinement of volatile forms of the iodine inside the containment;
– Confinement of the corium during an accident with the reactor vessel destruction;
– Corium cooldown in the catcher.

Arrangements provided at level 5.

Emergency documentation is developed based on the Unit’s process regulations, the NPP safety case reports, and in full compliance with the design documentation.

The Belarusian NPP project provides for the development of the emergency procedures as follows:

<u>Event-oriented procedures:</u>

– emergency mitigating procedures (EOP) “Anticipated operational occurences ”, while the design provides only event oriented format of the data reporting that is fully justified by the nature of mitigated events: anticipated operational occurrences mean simple, easily recognizable events, considered in the design. EOP “Design basis accidents”;

– Control guidelines for beyond design basis accidents.

Symptom-based emergency procedures:

– EOP “Design basis accidents”;

– Control guidelines for beyond design basis accidents.

– Control guidelines for severe accidents, symptom-based format is only provided.

Emergency procedures in the event oriented format will be developed and handed over as part of a documentation package to obtain a license for the Belarusian NPP operation.

Currently, the development of a documentation package required to obtain a license for the Belarusian NPP operation, including emergency response documentation in the event oriented format, is at the final stage. Activities required for development of emergency procedures in the symptom-based emergency procedure format, funding sources and performers of these works are defined. It should be noted that the development of procedures in symptom-based emergency procedure format provides for design-basis justifications for multiple scenarios of emergency procedures.

2.3.2.2. Inherent Safety Principle

Advanced third-generation reactors include features of passive or “inherent safety”.

Inherent safety principle implemented in the project is expressed in the RU capability of initiating events and accidents development prevention, limitation of their effects without personnel involvement, power consumption and external assistance during long time period. This time will be used by personnel to evaluate the situation and take corrective actions.

Inherent safety criteria include “non-interference period” in various situations, self-imposed limiting level of critical safety parameter values, and response rate of emergency processes.

The reactor inherent safety properties are aimed to constrain the energy release and self-restraint, limiting of the reactor pressure and temperature, heating rate, the primary circuit depressurization extent and blowdown rate, extent of fuel damage.

The above properties are provided within the project by the following:

– core energy release self-restraint properties due to negative reactivity factors in all critical states possible over the entire range of the reactor parameter changing during normal operation and at normal operation troubles, including design basis accidents;

– control elements triggering the emergency protection mode based on the gravitational forces (transfer of control elements into the core under their own weight);

– equipment configuration and the reactor primary circuit physical dimensions, allowing to provide conditions for development and maintenance of the coolant natural circulation in the primary circuit and heat removal off the core in case of forced circulation loss or absence;

– application of steam blanket pressurizer in the project providing “soft” limitation of pressure deviations from the nominal value during the steam blanket compression or expansion in the pressurizer in the process of operation. Water volume due to back links within “evaporation-condensation as pressure function” process is also involved in the pressure maintaining process: with increasing steam volume (level decreasing) water in the pressurizer is evaporated, thereby contributing to pressure retention and with vapor phase compression there is its condensation occurring on water surface thus limiting pressure buildup;

– lack of tie-ins and holes in the reactor vessel below the inlet piping and, respectively, below the upper mark of the core creating a sealed vessel around it that prevents its exposure in case of accidents with the coolant loss and the core impoundment;

– sufficient volume of water in steam generator of the secondary circuit enabling to residual heat removal from the core by steam discharge via SG disgorging devices within the timeframe required for PHRS startup;

– implementation of "leak before break" concept with respect to the primary circuit pipelines for early detection of leaks to prevent escalating them to the reactor primary circuit leakages;

– application of coastdown of special MCP rotating masses to provide required flow rate slump through the core upon electric power loss;

– application of direct action devices and passive safety systems;

– implementation of passive systems to catch and cool down the fuel melt outside the reactor vessel and providing its subcriticality.

Reactors of this generation within the nuclear safety framework feature the following advantages compared to the PWR reactors:

– Retention of safe operation conditions for a loner time;

– Longer time of the operator non-interference;

– Reduced probability of the core melting accidents;

– Low demand for nuclear fuel due to integration of neutron absorbers in it and a higher burnout level;

– Reduced amount of spent fuel (i.e. radioactive waste).

2.3.3. Special-Purpose Equipment and Facilities

Figure 2.3.3.1 shows the reactor building diagram which contains the essential reactor equipment with elevations of its installation.

The primary circuit is radioactive. It is limited by the reactor plant (RP) consisting of a reactor, four coolant circulation loops, four RCPU, tube side of four SG and one steam pressurizer.

The RP equipment is located in the RC containment. Spent fuel is also stored in the reactor compartment in the spent fuel pool in the nuclear fuel storage racks. Capacity of the spent fuel pool is designed proceeding from the following conditions:

- storage of spent fuel during ten years;
- arrangement of the FAs removed from the reactor core under emergency conditions;
- arrangement of leak-tight bottles for damaged FAs.

The fuel is stored in the spent fuel pool under protective water layer with boric acid concentration 16-20 g/dm^3.

The spent cooling pool consists of one compartment, designed for spent FA storage, and refueling cavity where transportation containers are filled with spent FAs and transportation casks are unloaded.

The spent fuel pool is adjacent directly to the reactor cavity and is connected with it via the canal for FA supplying. Spent FAs removed from the reactor core are stored in the cooling pool racks.

The spent fuel pool is cooled by the relevant system (FAK) designed for:

residual heat removal from the spent FA located in the cooling pool in all design operation modes as well as under DBA and BDBA;

formation of the anti-radiation layer over the FAs in the reactor cavity, spent fuel pool and refueling cavity;

filling the reactor cavity and spent fuel pool during refueling operations;

discharge of the reactor cavity and internals inspection cavity;

discharge of the spent fuel pool and refueling cavity during repair of the cladding;

FAK system pipelines are used to supply water from the sprinkler pumps for filling the internals inspection cavity during refueling operations and post-accident activities following the BDBA with core melting and release of the molten core material out of the reactor vessel.

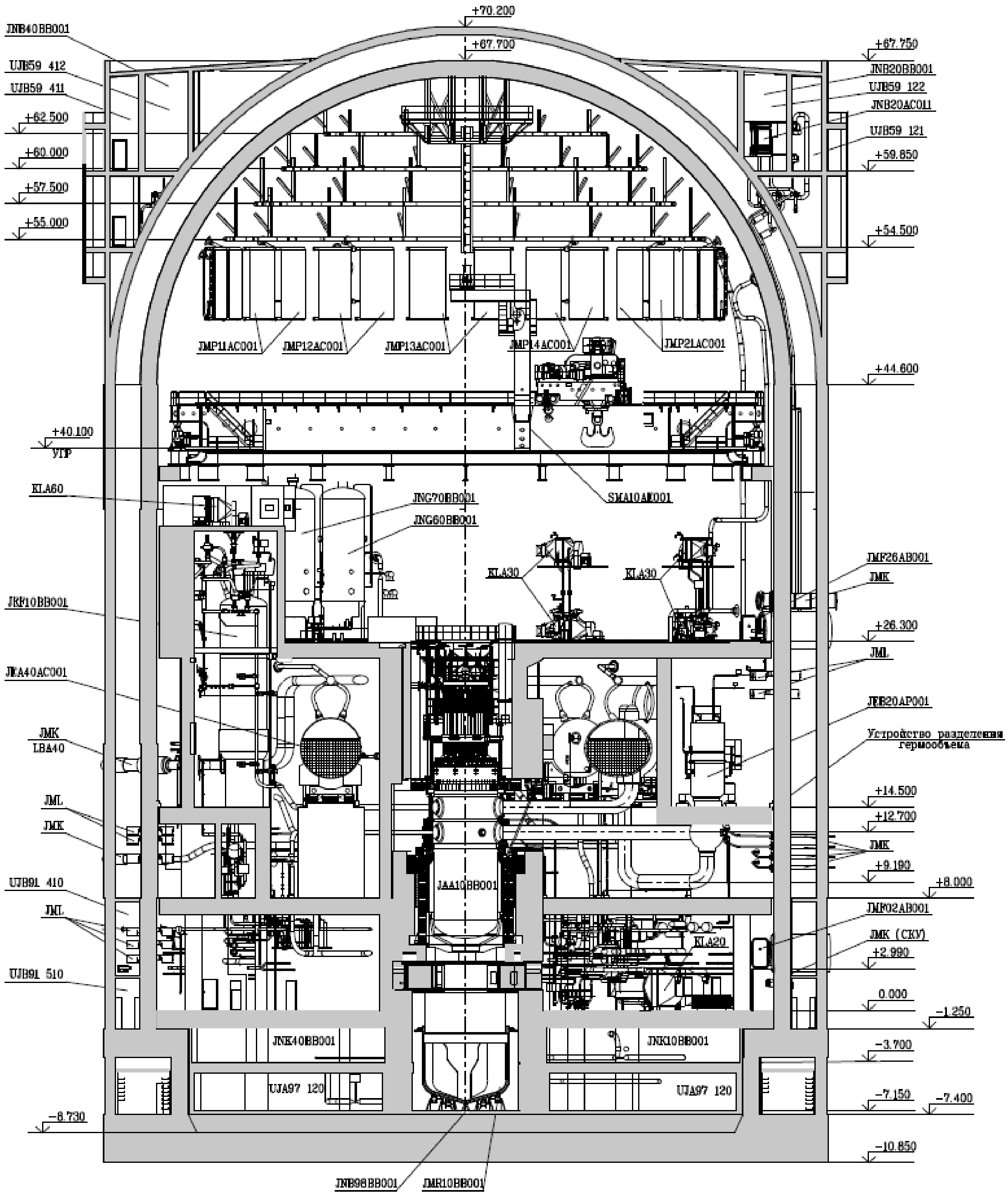

Figure 2.3.3.1 Reactor building with elevations of the NPP unit equipment installation.

Устройство разделения гермообъема	Containment separation device

Safety during spent nuclear fuel management at the Belarusian NPP site is ensured by the organization and technical measures and solutions provided by the design documentation. The main SNF handling systems at the Belarusian NPP are the following:

Reactor core refueling system;

SNF reactor storage system;

FE integrity control system;

On-site nuclear fuel transportation system;

Nuclear materials control and recording system.

The main SNF handling equipment at the Belarusian NPP is as follows:

Racks of the spent fuel pool designed for SNF cooling and storage prior to its removal out of the NPP;

Refueling machine;

Leak-tight bottles for storage of damaged spent fuel assemblies (SFA);

Equipment set for detection of the damaged FAs;

Multi-purpose seat in the refueling cavity where SNF is loaded into a transportation container;

Containers for SNF transportation to the fuel recycling plant providing transportation of spent fuel assemblies of VVER-1200 reactors after the preliminary SNF cooling in the cooling pool;

140 t truck-trailer for the SNF containers transportation within the NPP site for their reloading to the container car;

Container car (ТК-U-141 type) for SNF railway transportation to the fuel recycling plant. The container car provides accommodation, fastening and transportation of the SNF transportation container in accordance with the regulations for SNF transportation.

Other transportation and handling equipment.

Nuclear and radiation safety during SNF reloading is ensured by the organizational and technical measures specified in the project, namely:

equipment is designed to provide sub-criticality at least 0.05;

SFA transportation package sets are designed according to the requirements for their integrity under external extreme impacts;

equipment is designed to withstand natural phenomena (earthquakes) and other impacts during the NPP operation;

technical means are provided to avoid uncontrolled and spontaneous movements of the fuel transportation equipment;

continuous monitoring of water level and temperature in the spent fuel pool is performed;

radiation situation in the reactor room is monitored.

The on-site storage facilities for spent nuclear fuel (wet or dry) are not available.

In the reactor core the conditions under which thermal-neutron nuclear reaction with the release of thermal energy are created.

The primary circuit coolant passing through the reactor core is heated and supplied to the SG tube side through four circulation loops installed in parallel. There it gives up its energy producing steam of the secondary circuit. Downstream of the SG the coolant is returned to the reactor for re-heating. Circulation in the loops is forced by four RCPUs. Pressure fluctuations and temperature changes of the primary circuit coolant are controlled by the pressurizer. Under significant pressure increase in the primary circuit (during anticipated operational occurrences) steam from the pressurizer is discharged via the pulse safety devices into the surge tank which is cooled by the intermediate component cooling circuit. Layout of the RP equipment is given in Figures 2.3.3.2 and 2.3.3.3.

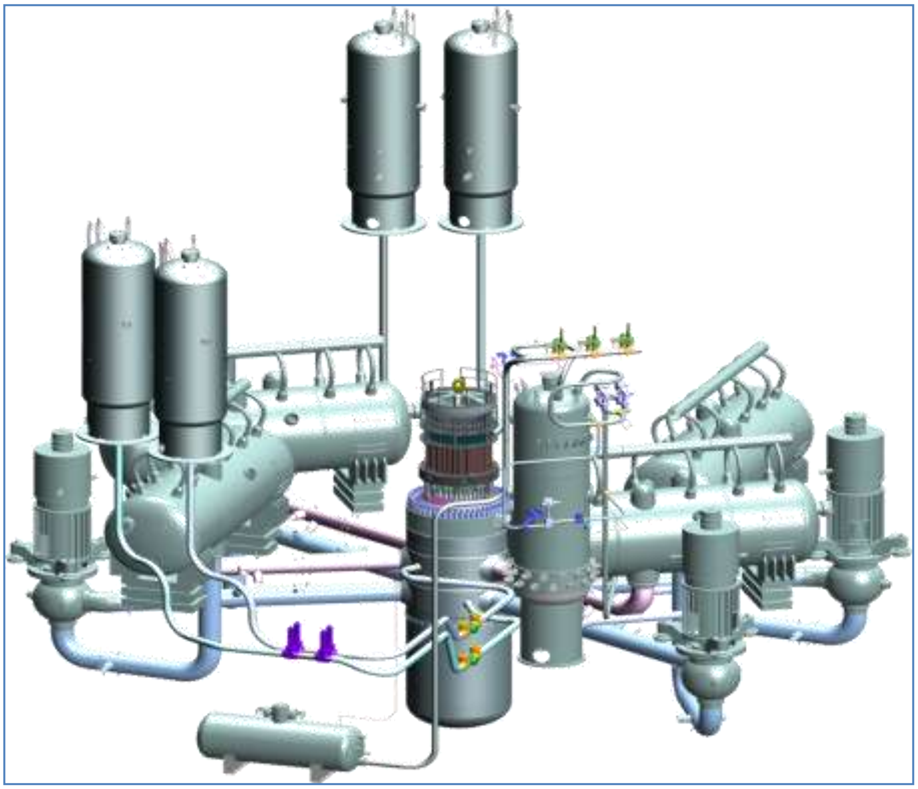

Figure 2.3.3.2 Layout of the Reactor Plant main equipment

The second circuit is non-radioactive. It includes: SG steam-generating assembly; fresh steam pipelines; one turbine plant consisting of turbine set and turbogenerator; condensate pumps, LPH system, main condensate system, deaerator, feed water system including feed water pumps and HPH systems.

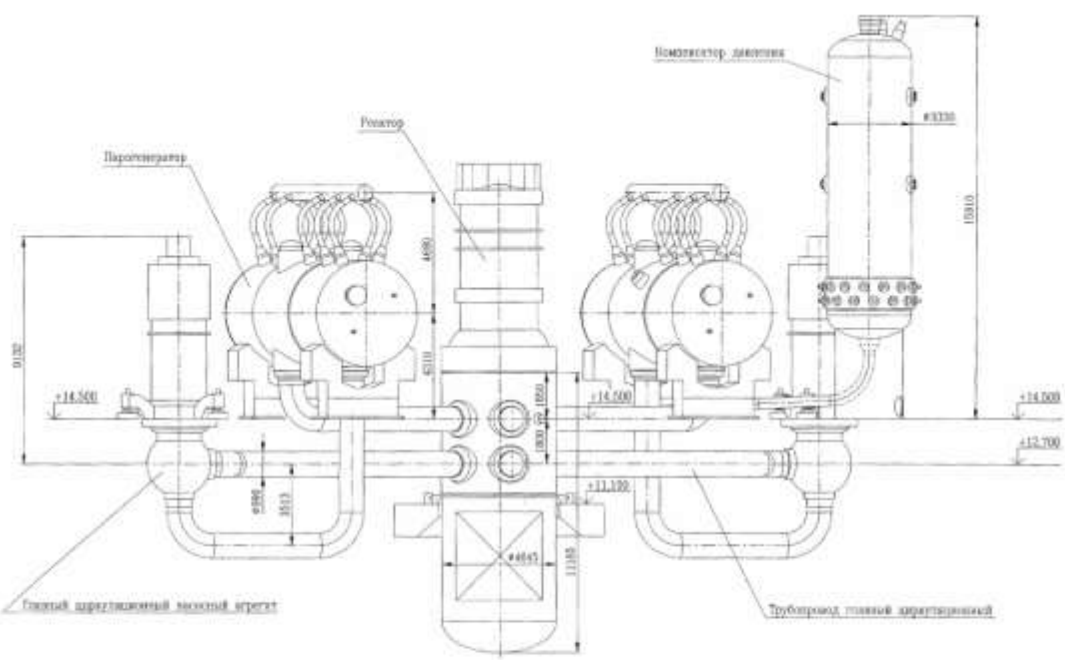

Figure 2.3.3.2 Layout of the Reactor Plant equipment with elevations

Парогенератор	Steam generator

Реактор	Reactor
Компенсатор давления	Pressurizer
Главный циркуляционный насосный агрегат	Main circulation pump
Трубопровод главный циркуляционный	Main coolant pipeline

From the SG steam generating assembly the steam is supplied to the turbine via the main steam pipelines through the pulse safety devices. Passing through the high-pressure cylinder and through four low pressure cylinders the steam gives up energy to the turbine. Thermal energy turns into mechanical energy of the turbine rotor rotation. Generator which rotor is on the same shaft with the turbine rotor converts mechanical energy into electrical energy.

The Belarusian NPP Project provides the radioactive waste management system. The system is designed for the collection, treatment, processing, conditioning, transportation and storage of the radioactive waste generated during the nuclear power plant operation. During the NPP operation gaseous, liquid and solid radioactive waste are generated. Technical solutions are provided to reduce LRW generation compared with water-cooled power reactors of the previous generations due to implementation of the low-waste technologies.

The main targets of the radwaste handling at the NPP:

gaseous waste handling: collection and purification from radionuclide up to the condition meeting the sanitary standards prior to discharge into the atmosphere;

LRW handling: LRW storage for short-lived radionuclides decay, radionuclide concentration in the minimum volume and LRW solidification providing safety during storage;

SRW handling: minimization of radwaste volumes and their safe, reliable storage during the entire design lifetime.

Each unit is provided with specially equipped SRW storage facility of the above-ground type.

Capacity of the storage facility is designed to store:

very low-level, low-level and intermediate-level waste (SRW in drums, solidified LRW in non-returnable containers) generated during 10 years of the power unit operation;

high-level waste in steel capsules for the entire lifetime of the power unit.

High-level radwaste is stored at the nuclear power plant site throughout its entire service life.

Conditioned very low-level, low-level and intermediate-level radwaste is stored in the storage facility during 10 years.

When the period of interim storage is over very low-level, low-level and intermediate-level radwaste is transported to the disposal site for long-term storage and/or disposal.

The safety systems and additional technical means for BDBA management are listed in Table 2.3.3.1.

Table 2.3.3.1 - Safety systems and additional technical means for BDBA management

Name	Number of channels and their efficiency
Protective, localizing, support and control safety systems	
1. High pressure safety injection system	4 x 100 %
2. Low pressure safety injection system	4 x 100 %
3. Emergency boron injection system	4 x 50 %
4. Emergency feed water system and heat removal through BRU-A system	4 x 100 %
5. Sprinkler system	4 x 50 %
6. Residual heat removal system	4 x 50 %

Name	Number of channels and their efficiency
7. Essential consumer intermediate cooling circuit system	4 x 100 %
8. Components cooling system for essential loads	4 x 100 %
9. SS ventilation systems	4 x 100 %
10. Containment isolation valve system	2 x 100 %
11. Borated water storage system	2 x 100 %
12. Emergency gas removal system	2 x 100 %
13. Primary circuit overpressure protection system	2 x 100 %
14. Secondary circuit overpressure protection system	2 x 100 %
15. Main steam line isolation system (MSIV)	2 x 100 %
16. Emergency diesel generator and supporting systems (EPS)	4 x 100 %
17. Startup system for safety systems	4 sensors/parameter, 4 logic channels each with 2/4 logic
18. Reactor emergency shutdown system	4 sensors/parameter, 4 sets of 2/4 logic at the 1st voting level and 2 sets of 2/4 logic at the 2nd voting level
Passive safety systems	
19. ECCS accumulator tanks system	4 x 50 %
20. Reactor containment system	1 x 100 %
21. System of hydrogen removal from the containment (1st subsystem)	1 x 100 %
BDBA control additional technical resources	
22. System of passive residual heat removal via SG	4 x 33 %
23. System of passive residual heat removal from the containment	4 x 33 %
24. Melt localization system	1 x 100 %
25. System of hydrogen removal from the containment (2nd subsystem)	1 x 100 %
26. Volatile iodine forms chemical binding system	1 x 100 %
27. Ventilation system to maintain vacuum in the inter-shell space	2 x 100 %
28. Internals inspection shaft water emergency use system.	2 x 100 %

NPP safety systems include protective, localizing, support and control systems.

Protective safety systems are designed to prevent or limit damage of nuclear fuel, fuel element, equipment and pipelines containing radioactive substances.

Principle diagram of safety systems, equipment and facilities for BDBA control is given in Figure 2.3.3.4.

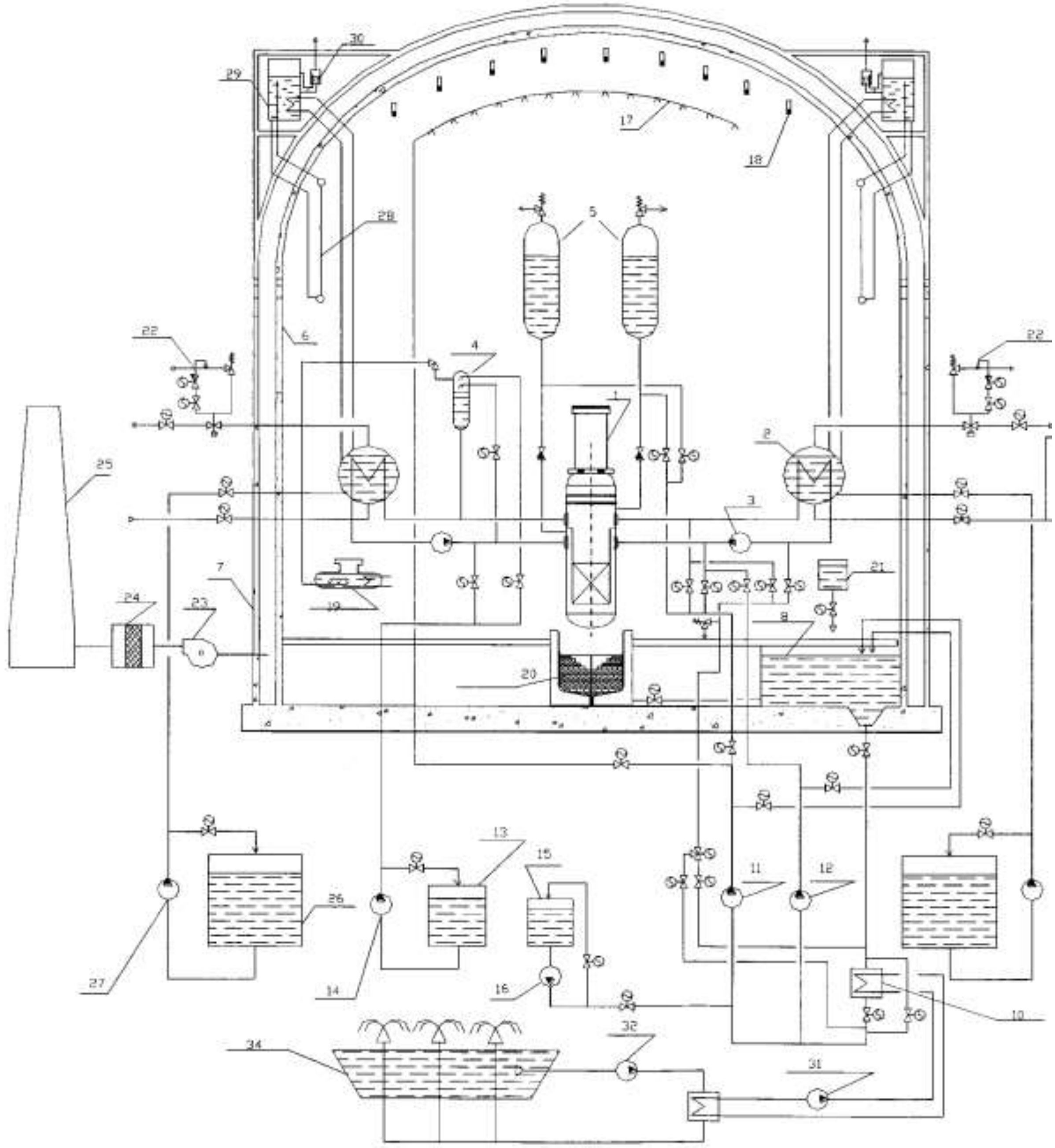

Figure 2.3.3.4. Principle diagram of safety systems, equipment and facilities for BDBA control.

The following protective safety systems are provided in the Belarusian NPP project:

- Reactor preventive and emergency protection system. The system is designed for emergency and preventive reactor protection, parameter monitoring and control rods position control, registration of the events and signals exchange with integrated I&C subsystems. The emergency protection system is a special reactor system and it limits the reactor power level under accidents and reactor shutdown under emergency conditions and accidents;

- Emergency boron injection system (JDH). The system is designed to inject boric acid solution into the pressurizer under loss-of-coolant accidents from the primary circuit to the secondary one, to supply highly concentrated boric acid solution (40 g/dm3) to the primary circuit for quick switching the reactor to the subcritical state under anticipated operational occurrences accompanied by failure of the reactor emergency protection as well as for switching the reactor to the subcritical state and compensation of the primary coolant decrease to ensure the power unit safe shutdown;

In addition, a part of pipelines and equipment of the system performs the function of a barrier preventing radioactivity emission outside the containment;

- High pressure safety injection system (JND). The system is designed for boric acid supply to the reactor coolant system under loss-of-coolant accidents exceeding compensation capacity of the normal makeup system, when the coolant system pressure is below the JND system operating pressure (below 7.9 MPa). Part of the system pipelines and equipment perform the function of a barrier preventing radioactivity emission outside the containment;
- Low pressure safety injection system (JNG1). The system performs the following functions: the reactor cooldown during the plant normal shutdown under anticipated operational occurrences and DBA provided that integrity of the primary circuit is maintained (together with the JNA system); residual heat removal from the core in case of accidents; residual heat removal from the core and maintenance of the temperature in the primary circuit of max. 60°C during transportation and installation operations during refueling; residual heat removal from the core and maintenance of the temperature in the primary circuit of max. 70°C under cooldown mode for repair; residual heat removal from the spent fuel pool if one of the channels of the FAK spent fuel pool cooling system is failed under complete and emergency reactor core unloading in conjunction with one of the FAK system channels;
- Borated water storage system (JNK). The system is designed to store borated water of low (16 g/dm^3) and high (40 g/dm^3) concentration required for NPP operation in all modes of operation. The system ensures storage of low-concentration borated water used for the following needs: core emergency cooling during the accident with coolant leakage; injection under the contamination during loss-of-coolant accidents or under steam line rupture within the containment; make-up water supply to the coolant system under power operation and reactor cooldown at the power unit shutdown; borated water supply for initial filling of the core catcher heat exchanger under DBA; borated water supply to fill the reactor cavity, internals inspection cavity, cavities between hydraulic locks in the refueling mode. The system ensures storage of high-concentration borated water used for the following needs: boron concentration adjustment in the primary circuit under normal operation and anticipated operational occurrences; injection of borated water into the pressurizer at the leakages from the primary circuit to the secondary one; borated water injection into the reactor under anticipated operational occurrences accompanied by the anticipated transients without scram (ATWS).
- Emergency core cooling system, passive part (JNG2). Passive part of the emergency core cooling system is designed for boric acid supply (with min. 16 g/dm^3 concentration and min. 20°C temperature) to the reactor under 5.9 MPa pressure in the primary circuit and in the quantity sufficient for cooling the reactor core before connecting low pressure emergency injection system pumps under loss-of-coolant design based accidents;
- Emergency gas removal system (KTP). The system is designed for steam and gas mixture removal from the RP primary circuit (reactor, pressurizer and SG headers) and pressure reduction in the primary circuit together with pressurizer pulse safety device in order to reduce DBA and BDBA effects;
- Residual heat removal system (JNA). The system is designed for residual heat removal and the reactor cooldown during the unit normal shutdown under anticipated operational occurrences and under DBA provided that integrity of the primary circuit is maintained together with JNG1 system as well as to protect the primary circuit against overpressure in cooling and residual heat removal modes at low temperatures of the primary circuit;

- Primary circuit overpressure protection system. The system is designed to protect the reactor equipment and pipelines from overpressure in the primary circuit under design and beyond design basis accidents due to operation of the pressurizer pulse safety devices installed on the steam discharge pipeline from the pressurizer steam space into the bubbler tank. To protect the primary circuit equipment from overpressure pressurizer pulse safety device opens automatically when reaching pressure opening set point. Control pressurizer pulse safety device ensures steam discharge from pressurizer when the primary circuit reaches 18.11 MPa pressure. Two operating pressurizer pulse safety device provide steam discharge from pressurizer when the primary circuit reaches 18.6 MPa pressure. Besides the primary circuit protection function against overpressure pressurizer pulse safety device ensures BDBA control procedure implementation and mitigation of its consequences due to the main valve additional control line, being part of pulse safety device, enabling to reduce pressure in the primary circuit down to 1.0 MPa when MCR and ECR remote control;

- Secondary circuit overpressure protection system. The system is designed to prevent overpressure in the steam generators and fresh steam piping in excess of the value not exceeding 15% of the SG operating pressure. This function is ensured by SG header as well as the second circuit discharge devices which provide heat removal from the reactor by steam discharge into the atmosphere in case of the unit emergency cooldown;

- Heat removal through BRU-A. BRU-A is designed to discharge steam into the atmosphere to prevent safety valves pulse operating mode in case of sharp load release and NPP blackout as well as steam discharge for residual heat removal, the reactor cooldown under anticipated operational occurrences and emergency conditions together with the emergency feed water system;

- Main steam line isolation system. The system is designed for quick and reliable isolation of steam generators from leaks in all emergency conditions which require SG isolation: at steam line breakages from SG to turbine stop valves in SG isolated and non-isolated sections, at feed pipelines breakage along the section from SG to check valve, in case of leakage from the primary circuit to secondary one. The system includes main steam isolation valves (MSIVs) being part of the main steam valve unit and motor operated valves installed in series;

- Emergency feed water system (LAR/LAS). The system is designed to feed SG with feed water under anticipated operational occurrences and DBA when feed water supply from the regular system and the auxiliary system is impossible. The system should function with the initiating events associated with water level lowering in SG and requiring emergency cooling down or maintaining the unit in hot standby.

System of passive residual heat removal via SG (JNB). The system is designed for long-term core residual heat removal to ultimate heat sink via the second circuit during beyond design basis accidents. The JNB system duplicates the corresponding active heat removal system to the ultimate heat sink in case of its design functions performance failure and represents technical resource to overcome beyond design basis accidents (as discussed below). The JNB system is designed to operate with the following beyond design basis accidents: residual heat removal and the reactor cooldown in the modes of all AC power sources failure; residual heat removal and the reactor cooldown in complete loss-of-feed water modes; ensuring backup for active safety systems (BRU-A) in case of their failure during accidents with the primary circuit leaks; accidental release reduction with leakages from the primary circuit to the secondary one.

Localizing safety systems are designed to prevent or limit radioactive substances released outside the accident confinement area.

The following localizing safety systems are provided in the Belarusian NPP project:

- reactor containment system. The system is designed to protect the reactor from external effects as well as to prevent radioactive substances from emission into the environment in all modes including emergency ones. The system consists of the following component parts: metal containment vessel with anchoring system, concrete building envelope, including the containment pre-tension system, gateways (for maintenance personnel and transport) with embedded plates, sealed penetrations (process, cable, ventilation, etc.) with embedded plates, isolation devices, piping and air duct system sections crossing containment barrier (within isolation devices).

The project uses double containment.

Inner containment is made of prestressed concrete with a steel sealing cladding, the containment is designed for the design basis accidents (DBA) parameters in combination with safe shutdown earthquake (SSE) and is able to limit the release of radioactive substances generated at the same time.

Outer containment is made of reinforced concrete and is designed to protect the reactor building from external effects. The inter-containment space is connected to the vent system which provides vacuum and cleaning of the environment;

- containment isolating valve system. The system is designed to isolate pipelines with various process media which pass through the protective containment boundary to prevent the fission products release in case of the primary circuit loss-of-coolant accident. The containment confinement system includes isolation valves located on the pipelines crossing the containment boundary;

- Sprinkler system (JMN). The system is designed to perform the following functions:

pressure reduction in injection mode under the containment after the accident in order to maintain pressure in the containment below its design pressure at accidents;

fission products withdrawal from the containment atmosphere, thus reducing the total concentration of fission products in the air to prevent their leakage into the environment;

water chemical composition control in a sump tank by adding chemical reagents for long-term iodine retention and corrosion prevention;

residual heat removal from the spent fuel pool during complete reactor core unloading together with one of the spent fuel pool cooling system (FAK) channels;

ensuring redundancy of the spent fuel pool cooling system (FAK);

Internals inspection cavity filling during the refueling operations;

Internals inspection cavity filling in 24 hours after the BDBA related to the reactor core melting and melt release outside the reactor vessel;

- System of hydrogen removal from the containment (JMT). The system of hydrogen removal from the containment is designed to provide for hydrogen explosion safety in the accident localization area (ALA) during design and beyond design basis accidents. Hydrogen removal system from the containment during DBA and BDBA prevents the generation of explosive mixtures in ALA by maintaining hydrogen volumetric concentration in the mixture at a safe level;

- leakage localization system of the containment KLC11/21/31/41;

- safety building ventilation system valves and air ducts KLG01AA101, KLG01AA102, KLG02AA101, KLG02AA102.

Support safety systems are designed to supply safety systems with power, operating medium and to create conditions for their functioning.

The following support safety systems are provided in the Belarusian NPP project:

Components cooling system for essential loads (KAA/KAB). The system is designed to supply cooling water and remove heat from the reactor equipment, the reactor auxiliary systems and systems which ensure the NPP safety under NO, ANO and DBA conditions, as well as to provide a barrier between auxiliary systems containing radioactivity and process water system for essential loads;

Emergency power supply system (EPS). The emergency power supply system provides power supply to electrically driven components of the safety systems under NO, ANO and DBA conditions;

cold supply system for the ventilation system SAC10/20/30/40 QKC10/20/30/40;

cold supply system fir the air condition system SAC12/22/32/42 QKC12/22/32/42;
cold supply system fir the air condition system SAC17/27/37/47 QKC17/27/37/47;
combined exhaust-and-plenum system for channels 1, 2, 3, 4 in SAC10/20/30/40;
MCR personnel life support system SAC11/21/41;
air condition system of MCR premises SAC12/22/32/42;
extract ventilation system of channels 1, 2, 3, 4 in the control building battery rooms SAC15/25/35/45;
ECR personnel life support system ECR SAC16/26/46;
ECR personnel life support system SAC17/27/37/47.
recirculation cooling system KLG11/21/31/41;
fire dumpers and air ducts of the system KLG13, KLG23, KLG33, KLG43 located on the safety channels boundary;
controlled access area electric heating system of the building SBH13;
combined exhaust-and-plenum system for channels 1, 2, 3, 4 in SAS10/20/30/40;
extract ventilation systems of cable rooms SAS11/21/31/41;
cold supply system for the ventilation systems SAS10/20/30/40 QKS10/20/30/40;
inter-containment rooms emergency cooling system in the reactor building KLC10/20/30/40;
passive heat removal room ventilation system of channels 1, 2, 3, 4 SAA10/20/30/40, SAA11/21/31/41;
steam cell electric heating system elements SBH12.

Control safety systems are designed to actuate safety systems, perform their monitoring and control during task performance.

The following control safety systems are provided in the Belarusian NPP project:

- Reactor emergency protection system (EPS). Reactor emergency protection system is part of the control and protection system (CPS). The reactor EPS is designed to perform the following functions:

monitoring of the neutron flux density, its change rate, process parameters and parameters characterizing the equipment condition required for reactivity protection and control as well as the reactor power;

EPS signal generation upon reaching the respective set point controlled parameters which ensure the CR drives de-energization followed by the CR fall in order to transfer the core into a subcritical state and maintain it in this state, at the same time EPS priority over other types of control must be provided;

reporting of the MCR and ECR controlled neutronic parameters and MCR process parameters as well as this information release to other APCS subsystems;

generation of control signals to other NPP APCS subsystems.

Apart from the reactor EPS, CPS also includes:
DBE warning safety system;
group and individual control system;
power offload and limitation device;
automated power control;
reactor CR drive complete with position transducers;

- Actuating system for protective safety systems. Actuating system for protective safety systems includes I&C and controls required to automatically initiate actions in order to mitigate the core and coolant system damage.

Normal operation systems performing safety functions:

- Make-up and boron control system (KBA). The system is designed to perform the following safety functions: boron solution injection into the pressurizer to reduce pressure in the primary circuit in case of the NPP blackout, the primary circuit coolant loss compensation in the event of leakage up to 80 m^3/h, boric solution supply in the primary circuit in case of EPS actuation.

Spent fuel pool cooling system (FAK). The system is designed for residual heat removal from the spent FA, located in the spent fuel pool in all design operating modes including DBA and BDBA states.

Systems and tools to control the BDBA which are operable and are able to perform their functions in BDBA conditions.

- Melt localization system (JMR). Melt localization device under a severe accident with the core and reactor vessel destruction ensures holding of the melt and destroyed core solid fragments, the reactor vessel components and internals. Melt localization and cooling is carried out within the concrete vault under-reactor space. While ensuring heat removal from the containment in the accident progress the melt retention and CC operation reliability is ensured during unlimited time period.

- Internals inspection shaft water emergency use system (JNB90). The internals inspection shaft water emergency use system is designed to supply borated water from the internals inspection cavity into the melt localization device during beyond design basis accidents related to the reactor core melting and the melt release outside the reactor vessel, for NaOH alkali solution supply into the containment sump tanks to reduce the rate of iodine volatile forms generation inside the containment, as well as fill and drain the Internals inspection cavity during internals refueling and inspection operations.

As regards the residual heat removal from the spent fuel pool, the system provides the spent fuel pool makeup. In addition, the spent fuel pool makeup can be implemented by connecting non-routine emergency resources (fire truck with pump unit 40 l/s capacity and 100m head) to two technological emergency heat removal tank makeup line connectors located outside the steam chamber building (at +0.690 and +0.730 elevations, while water intake is made from the water make-up system tanks via the fire truck pump unit and further is delivered to the spent fuel pool through EHRT tank makeup line piping the lines feeding the emergency heat removal tanks,) with mounted flanges with stubs.

- System of passive residual heat removal from the containment (JMP). The system (JMP) is a part of the BDBA overcoming technical means and is designed for long-term (off-line operation for at least 24 hours) heat removal from the containment under BDBA. The system provides for reduction and maintenance of the project defined pressure range inside the containment and the heat removal to the ultimate heat sink which is released under the containment under BDBA, including accidents with severe damage of the core;

- System of passive residual heat removal via steam generators (JNB). The system is designed for the core residual heat long-term removal to the ultimate heat sink through the second circuit during beyond design basis accidents.

Under beyond design basis accidents the JNB system performs the following functions:

- residual heat removal and the reactor cooldown in the NPP full blackout modes;
- residual heat removal and the reactor cooldown in loss-of-feed water modes;
- provision active safety systems backup in case of their failure.

The system performance has been selected considering the redundancy principle, based on the most likely scenarios of beyond design basis accidents considered in the project. The system includes 4 channels completely independent one from the other with 4x33.3% capacity, i.e. three working PHRS circulation circuits are sufficient for the system to perform its functions in full in any mode requiring its operation.

The selected system design ensures its fully off-line operation without the operator intervention for at least 24 hours in accidents resulting in complete blackout and the SG feed water failure. The emergency heat removal tank makeup line (JNB50 subsystem) is used for emergency heat removal tanks makeup and spent fuel pool makeup within the 24 to 72 hour period. In case of active safety systems redundancy provision upon their failure, LCU07,08AP001 system pump unit delivering water of makeup water system tanks (LCU system) to consumers, is used for emergency heat removal tanks makeup and spent fuel pool makeup.

Section 6 of this report presents the design analysis results for the BDBA with all AC power sources failure during SG PHRS operation. The data obtained show that the RP off-line operation capacity by the residual heat removal is 72 hours from the beginning of the accident, provided the 4 EHRTs water supplies are used. But given that the monitoring and control system in full blackout BDBA mode can be powered by BDBA consumer channel batteries (power supply channel 7) for 24 hours only (without battery recharge) from the beginning of the accident, SG PHRS is able to operate for 24 hours without the operating personnel intervention. For the following SG PHRS tanks filling and continuation of the SG PHRS monitoring and control system operation a regular mobile diesel generator plant is connected which is required for the power supply channel 7 batteries' recharge and for powering the EHRT makeup pump from LCU tanks.

Makeup of PHRS tanks and the spent fuel pool is performed with JNB50AP001 low-power high-pressure pump of PHRS tank makeup system. This pump unit is located in the steam chamber and connected to the LCU system tanks. The pump power supply is delivered from the BDBA power channel (connected mobile diesel generator plant of the power supply channel 7). BDBA power channel is designed for 24 hours off-line operation and has an option to connect the mobile diesel generator plant to recharge the batteries and continue the system functioning.

Based on the recommendations provided following development of the Report on the stress tests (objective safety reassessment) of the Belarusian nuclear power plant, two 500 kW mobile diesel generator plant (one per each NPP unit) which are supposed to be located openly on the NPP site, will be provided for two units.

The mobile diesel generator plant operating performance is provided with ambient temperatures from - 50ºC to + 41ºC. In the event of the accident resulting in complete power loss (all DG failure), BDBA power channel switchgears will be disconnected from 0.4/0.23 kV sections of the EPS channels. BDBA power supply is with their UPS batteries. Monitoring and control is carried out through BDBA panel located on the MCR.

Mobile diesel generator plant is delivered to its connecting point and is prepared for operation within 24 hours. The connecting point to the channel 7 switchgear is a cabinet (I seismic category as per NP-031-01, dustproof-and-moisture proof version– IP54, UHL1, hufter-proof, lockable) located on the outer wall of the UJE building at elevation 1.400. If the mobile diesel generator plant cannot be delivered there is a possibility to connect the mobile diesel generator plant on regular location using special switching equipment and additional cables.

The mobile diesel generator plant monitoring and control are carried out directly from local control panels installed on the equipment.

EHRT is filled out of the LCU demineralized water tanks until water depletion in the tanks (at around 72 h from the emergency process commencement). This is followed by re-emptying SG PHRS tanks and LCU demineralized water tanks re-filling. In order to keep the RP safe stable state while maintaining the SG PHRS operating performance LCU tanks are to be fed regularly (followed by EHRT filling) with any water sources available at the NPP site using off-site mobile equipment (e.g., fire water make up tanks).

Units No. 1 and No. 2 are constructed in accordance with the Belarusian NPP project documentation establishing the same basic technical requirements to all the systems and equipment of both units. All the differences of units No. 1 and No. 2, their systems and equipment, implemented based on the above design requirements, will be defined on further stages of the Belarusian NPP project. The basic specifications of the NPP unit with VVER-1200 are shown in Table 2.2.1.

2.3.4 Application of Probabilistic Safety Analysis as a Constituent Part of the Safety Assessment

NPP Safety Analysis Report is included into the scope of the documents substantiating nuclear and radiation safety which are submitted by the operating organization to obtain the license on activities related to the use of nuclear power and ionizing radiation. Requirements for the structure and contents of Safety Analysis Report are established in the legislation of the Republic of Belarus. One of these requirements is submission of a probabilistic safety analysis within Safety Analysis Report.

To assess the Belarusian NPP safety, a probabilistic safety analysis (PSA) of the 1st and 2nd level is applied. For the Belarusian NPP, comprehensive PSA-1 (for internal initiating events, internal fires and flooding, seismic PSA and PSA for external impacts) and comprehensive PSA-2 based on PSA-1 are developed.

PSA-1 is developed on the basis of the following regulatory documents:

- NP-095-15 Basic requirements for the NPP unit probabilistic safety analysis;
- RB-024-11 Regulations on basic recommendations for developing level 1 probabilistic safety analysis for internal initiating events under all operating modes of the NPP unit;
- RB-021-14 Basic recommendations for developing level 1 probabilistic safety analysis of the NPP unit for initiating events caused by external natural and man-induced impacts.

PSA-2 is developed on the basis of the following regulatory documents:

- NP-095-15 Basic requirements for the NPP unit probabilistic safety analysis;
- RB-044-09 Basic recommendations for level 2 probabilistic safety analysis of nuclear power plants with VVER-type reactors.

The average value of frequency of nuclear fuel damage in the reactor obtained from PSA-1 for internal initiating events is as follows:

- at power operation: 7.7×10^{-7} 1/year;
- in standby modes: 2.42×10^{-7} 1/year.

The average value of total frequency of nuclear fuel damage in the spent fuel pool is as follows:

- at power operation: 3.32×10^{-10} 1/year;
- in standby modes: 3.19×10^{-8} 1/year.

Based on the results of the internal flooding analysis it is established that equipment failure resulting from the flooding does not lead to an accident with nuclear fuel damage.

3. EARTHQUAKE

The area of the Belarusian NPP site refers to the Belarusian-Baltic seismotectonic region which is characterized by rather low seismic activity.

3.1 Design Basis

The level of seismic danger for the NPP site within the near-field region is defined, generally, by seismicity of the Belarus platform territory .

As a normative basis (TKP 45-3.02-108-2008 "High-rise buildings. Design buildings rules") for assessment the NPP site seismic level, the map of the general seismic risk zoning of the North Eurasia (including the Belarus territory (Fig.3.1.1)) with a scale of 1:10000000 GSZ-97-D was accepted. The GSZ map corresponds to frequency of seismic impacts on average once in 10 000 years and probable occurrence (P = 0.5%) and exceedance of the seismic effects during 50 years, specified in MSK-64 scale points. The map is designed to assess seismicity of the NPP sites, radwaste disposal sites and the sites with the other extremely important structures. According to the GSZ-97-D map the Belarusian NPP site refers to the 7-points zone. This assessment corresponds to the level of safe shutdown earthquake, SSE. For assessment of the design basis earthquake, DBE, the value of 6 points respectively is accepted (the frequency period is 1000 years with 5% exceedance probability during 50 years).

Buildings and structures as well as process pipelines, other communications and engineering structures of the Belarusian NPP are designed proceeding from the following seismic impacts:

- Maximum horizontal acceleration of the SSE level – 0.12 g (7 points as per the MSK-64 scale)

- Maximum horizontal acceleration of the DBE level – 0.06 g (6 points as per the MSK-64 scale).

The design-base justification of seismic resistance of the building structures is performed taking into account soil conditions at the Belarusian NPP site.

The maximum peak (horizontal) accelerations (PHA) received by the results of field research during seismic risk zoning is < 0.1g (0.069g). At the same time according to NP-031-01 for the new NPPs designed irrespectively of the site seismicity the accelerations corresponding with the SSE level are minimum 0.1g. In the design bases the PHA value is 0.12g (VVER-1200 Project, 2006). (Water cooled water moderated power reactor). In the IAEA recommendations NS-G-3.3, item 5.26 and SSG-9 item 2.11, the recommended minimum level is the maximum horizontal acceleration equal to 0.1g.

In IAEA recommendations of NS-G-3.3, item 5.26 and SSG-9, item 2.11., the recommended minimum level also is the maximum horizontal acceleration of soil 0.1g.

In the design bases the value PGA=0.12g (The project VVER-1200, 2006) with a reserve 0.01g, i.e. 0.13g is accepted. Thus, for an extreme earthquake which exceeds the maximum values provided by the project of the Belarusian NPP actually the reserve of exceeding of seismic influences makes 0.03g or 30% in relation to the corresponding MDBE value.

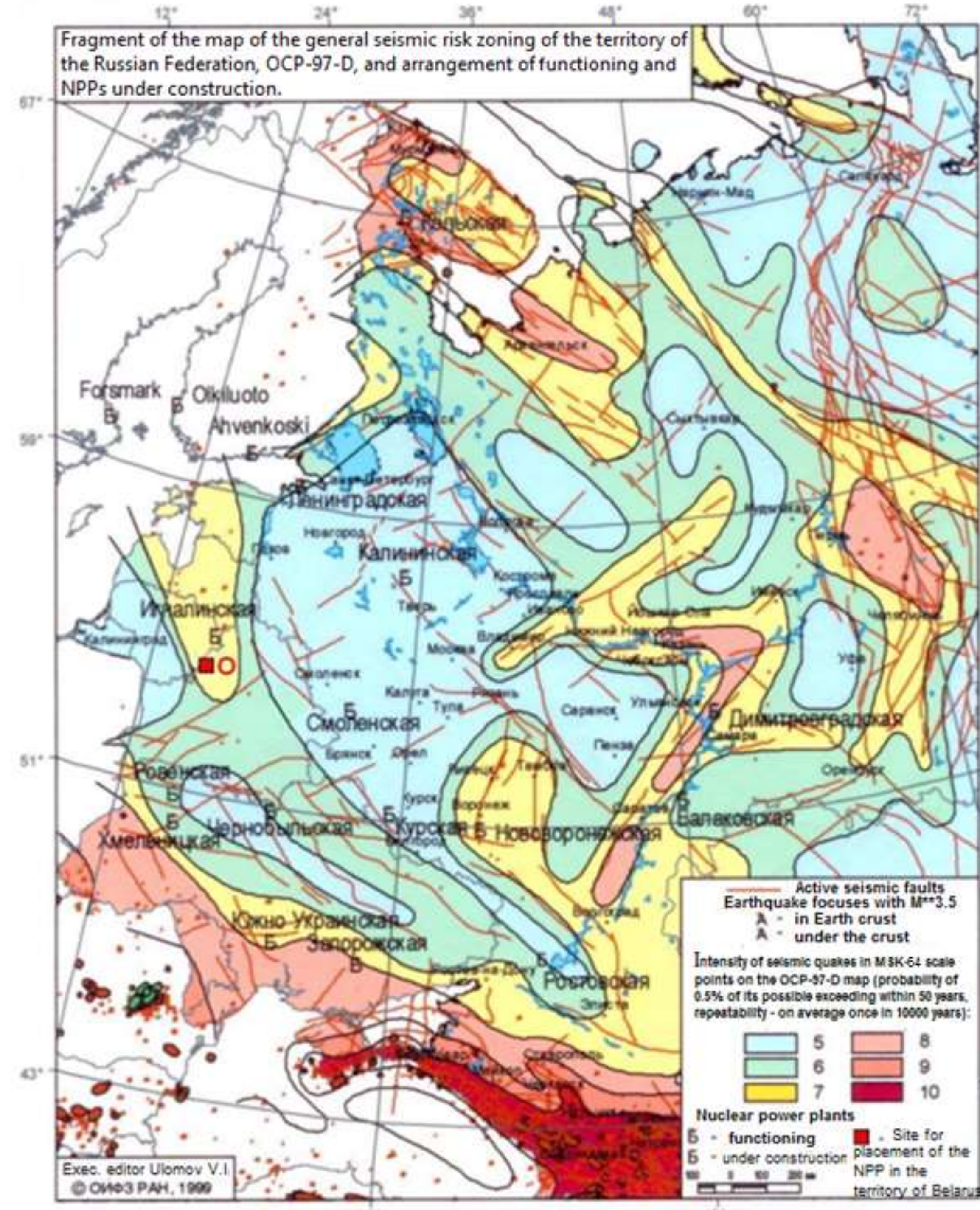

Figure 3.1.1 - A fragment of the map of the general seismic risk zoning of North Eurasia OCP-97-D, scale 1:10 000 000 (the Belarusian NPP site is marked by a small square)

3.1.1 Design-Basis Earthquake

At the stages of seismic hazard assessment, seismic impacts caused by distant intensive Carpathian earthquakes and the closest seismic origin zones were analyzed.

To determine the DBE and SSE levels the following works were performed.

1) The impact of distant origins of deep-focus earthquakes of the Vrancea area in the East Carpathians in Romania was analyzed to assess seismic activity at the site. The analysis results showed the following level of seismic impacts:

- Design basis earthquake - 4 points;
- Safe shutdown earthquake - 5 points of MSK-64 scale for the generalized soil conditions.

2) The seismic activity of the areas within a radius of 300 and 30 km from the Belarusian NPP site was determined.

3) Seismotectonic researches were performed within the region of the NPP site location and in the neighboring areas to detail the regional tectonic structure and to specify the initial seismic impacts determined according to the general seismic zoning maps GSZ-97 using the methods and approaches specified by regulations.

4) Historical and measured felt earthquakes of the western part of the East European Platform during 1602-2007 were listed based on literary and archival data.

5) Geodynamic active zones within the region of the NPP site location and in the neighboring areas were determined. For the NPP site, 23 XV-level geodynamic zones and 185 XIV–level geodynamic zones have been determined (according to classification of RB-019-01 "Seismic assessment of the regions of location of nuclear and radiation hazardous sites on the basis of geodynamic data") activated within the N-Q period (the latest tectonic movement). 55 zones of 185 XIV–level geodynamic zones activated in the N-Q period are considered as sub-active, for which no significant tectonic movements are determined. The potential XIV-level PES zones (possible earthquake source) that are the nearest to the NPP site and significant for seismic assessment are potential PES zones 78 and 147.

Velocity gradient of quaternary-Neogene movements (deformation velocity) in the geodynamic zones of the described area that were activated within the quaternary-Neogene period is a characteristic of low-active areas and is $4.45 \cdot 10^{-9}$ per year, thus, the described area can be referred to hazard level II (Appendix 1 (mandatory) NP-064-05 "Accounting external, natural and man-induced impacts on nuclear facilities").

6) The parameters of significant geodynamic zones of the NPP site and the neighboring areas were considered. According to the geodynamic conditions, the site is located within the XIII-level monolithic crustal block at a minimum distance (4km) from the XIII–level inter-block border (XIII-14 zone). Deformation velocity (velocity gradient of quaternary movements) in the geodynamic zones of the neighboring areas that were activated within the quaternary period changes from $1.2 \cdot 10^{-8}$ to $5.3 \cdot 10^{-8}$ per year that is a characteristic of low-active areas, thus, the described area can be referred to hazard level II (Appendix 1 (mandatory) NP-064-05.

7) The PES zones were determined according to the seismic and geological data and evaluation (value) of their seismotectonic potential (M_{max}). The value of M_{max} for each zone has been defined as follows: by the magnitude of the strongest earthquake for this structure (if there are measured earthquakes) and by analogy with similar structures of other ancient platforms or with geological structures of the described region (if there are no measured earthquakes).

The depth of a possible seismic origin with M_{max} for each zone has been determined proceeding from the specifics of geological structure, prevailing depths of intensive earthquakes under the similar tectonic conditions of other ancient platforms, vertical dimensions of the seismic origin (under condition of no fault outcrop that is a characteristic of the crustal earthquakes with $M \leq 5.5$ on platforms), or by the hodograph curve for the measured earthquakes.

The integrated approach enables to determine the PES zones nearest to the NPP site, to define their main characteristics and to evaluate the level of potential hazard. The PES zones nearest to the Belarusian NPP site are the Oshmyany seismogenic zone located 19km to the south, and the Daugavpils seismogenic zone located 67.5km to the north.

The Oshmyany seismogenic zone is the continuation of the Vilnius zone. This zone is in vicinity of the active fault intersection of the first level. Given the kinematics, the fault zone is defined as a strike-slip fault. The structure belongs to the seismogenic type with seismotectonic potential of M_{max} = 4.5, and a possible seismic origin located at a depth of H = 5km.

The Daugavpils zone is quite actively shown at the latest stage. Territorial isolation of a seismogenic zone is caused by wrong determination of the coordinates of the seismic origin, and it could be localized both in the latitudinal fault zone, and in the fault zone of the north-eastern direction. Therefore both of these faults were taken for a seismotectonic framework of a seismogenic zone. During the formalized seismotectonic zoning, four cells with M_{max}= 4.5 were established here. The obtained data allowed to classify the described zone as a seismogenic type with a seismotectonic potential of M_{max}= 4.5 and H = 10-12 km.

Also the Kaliningrad-Lithuanian seismogenic zone was considered, taking into account the felt earthquakes of 2004 from this zone measured in the territory of Belarus. The Kaliningrad-Lithuanian seismogenic zone is located on the western continuation of a large Kurzeme-Polotsk fault zone, and comprises three subzones: Northern, Central and Southern. All three subzones are characterized by the seismotectonic potential of M_{max} = 4 and H = 8 km.

8) When defining the seismic activity of the Belarusian NPP site, the following procedures were performed:

- deterministic evaluations of earthquakes magnitude on the basis of geodynamic data (an assessment of the maximum magnitudes M_{max} of PES potential zones of the region of the Belarusian NPP site location and the neighboring areas), and relevant evaluations of intensity of shocks on average soils of the site for the DBE and SSE levels that did not exceed 7 points (SSE);

9) Probabilistic evaluations according to the available lists of earthquakes (probabilistic values of seismic hazard were obtained based on the list of historical earthquakes of the region of NPP site location within 1602-2012 prepared by the Center of Geophysical Monitoring of National Academy of Sciences of Belarus taking into account the list made by Gryuntal).

When using a probabilistic approach, the evaluations of intensity of SSE reaching 6 points of MSK-64 scale for average soils were obtained.

For a hypothesis of the scattered seismic activity, earthquakes of M_{DBE} and M_{SSE} level can occur anywhere within the described area, including directly under the NPP site. Taking into account M_{DBE} and M_{SSE} and the most probable depth of the seismic origin, the intensity values of DBE- and SSE-induced shocks for average soils of the site can reach 4 and 6 points of MSK-64 scale respectively.

For a hypothesis of the structured seismic activity, taking into account the minimum distance from the site of the potential PES zones capable to cause similar seismic events, the most probabilistic intensity values of DBE- and SSE-induced shocks for average soil conditions are 4.6 and 7.2 points that does not exceed deterministic values of the maximum intensity.

Thus, according to the data of integrated seismological and geodynamic researches of the NPP site location area (scale 1:500 000) and the neighboring area (scale 1:50 000), the values of intensity of DBE (SL1) and SSE (SL2) equal to 6 and 7 points of MSK-64 scale, respectively, are obtained for average soils.

The studies aimed to specify the seismic hazard, taking into account soil conditions at the NPP site, and performed during seismic microzoning by methods of stiffness analysis, recording microseisms and explosions did not reveal significant increasing of seismic intensity in points.

The man-induced changes of conditions, i.e. rising of groundwater level, excavating a pit, and soil bedding, etc. will not cause significant changes in seismic activity values of the Belarusian NPP site determined for natural soil conditions. (Technical report. The Belarusian NPP. Power Units 1&2. Specification of seismotectonic conditions of the region of the Belarusian NPP location and the Belarusian NPP site, and preparation of materials for the Safety Analysis Report (SAR). Book 1. Seismotectonic description and seismic activity of the region, area and the site of the Belarusian

NPP location. BLR1.C.752.&.&&&&&&.&&&&&.002.HG.0001. Contract No.5808/1. as of 23.04.2012 LLC ENERGOPROJECTTEKHNOLOGIYA, Moscow, 2013.)

Soil liquefaction at the site under DBE- and SSE-induced seismic loads does not occur. Data on laboratory study of the soils are reported in the Special laboratory studies on defining physical and mechanical properties (including dynamic) and seepage-suffusion properties of the subsoil of Power unit No. 1 of the Belarusian NPP. Technical report. (Final). Volume 1. Dynamic tests of soils. JSC VNIIG of B. E. Vedeneyev. St.-Petersburg - 2012.

For control of geodynamic stability of the environment conditions at stages of justification of investments, designing and construction of the Belarusian NPP, a 24 h seismic monitoring network has been provided in the region of the Belarusian NPP site location since 2008 to this day. The local seismic network developed in the region of the Belarusian NPP site location consists of 7 points of monitoring located 15-25km far from the site, and provides recording of seismic events within the required range of epicentral distances and energy. During monitoring, full data processing, including drawing up bulletins and lists of earthquakes in the neighboring area (up to 300km), regional earthquakes (up to 1000km) and distant earthquakes is provided. No local earthquakes were recorded.

The performed monitoring investigations did not give the grounds for changing parameters of seismic impacts at the Belarusian NPP site. In particular, seismic magnitude at the Belarusian NPP site and the obtained values were not revised: a DBE value is 6 points, a SSE value is 7 points corresponding to the seismic levels SL-1 and SL-2 respectively, according to the European macroseismic scale EMS-98.

Thus, the results of integrated seismotectonic, structural and geodynamic, neotectonic, geomorphological, and field studies in the territory of the monolithic block of Earth crust placement, on which the Belarusian NPP site is located (Belarusian NPP. Power unit 1. Safety Analysis Report. Chapter 1, General description of Nuclear Power Plant and Chapter 2, Characteristics of the area and the NPP site: BLR1.B.130.1.01&&&&.01&&&.000.HD.0001), exclude probable normal faults, thrusts, upcast faults, strike-slip faults and other crustal faults accompanied with strong oscillations and seismotectonic troubles.

Taking into account the performed analysis, the substantiation of the design-basis solutions providing seismic stability of the NPP under the conditions of the Belarusian NPP site is confirmed.

3.1.2 NPP Protection under DBE and SSE

All equipment of the NPP required for the RP safe shutdown refers to seismic category I (designed for SSE).

All NPP equipment which damage can affect operation of the safety-related equipment either refers to seismic category I or physically separated from the safety-related equipment. Thus, protection It follows that under an earthquake with intensity up to the SSE level the safety-related equipment is not failed. Also operational emergency situations do not occur.

The systems and components required for the RP safe shutdown and their functions (depending on operation conditions - NO, AOO, DBA and BDBA) are given in Table 3.1.2.1.

Table 3.1.2.1 - Systems and the elements components required for the RP safe shutdown

KKS code	System	System components	Performed functions	Buil-ding
FAK*)**)	Fuel pool cooling system	Heat exchangers, pumps	Heat removal from spent fuel	UJA UKD
JAA*)**) ***)	Reactor system	Reactor Pressure Vessel	Coolant inventory maintenance	UJA
JEA*)**) ***)	Steam-generators system	Steam-generator housing, pulse safety device	Coolant inventory maintenance	UJA
JEC*)**) ***)	Reactor coolant piping system	Reactor coolant pipeline Reactor coolant pump,	Coolant inventory maintenance	UJA
JEB*)	Reactor coolant pump system	Reactor coolant pump, spherical pump housing, Independent Cooling Circuit	Coolant inventory maintenance	UJA
JEF*)**) ***)	Pressurizing and steam relief system	Pressurizer , heaters, pulse safety device	Coolant inventory maintenance, primary circuit Overpressure protection of the primary circuit	UJA
JNG50,60,70,80 **) ***)	Passive part of the emergency core cooling system	Hydraulic accumulators, pipelines, valves	Coolant inventory maintenance. primary circuit Primary circuit make-up.	UJA UKD
KTP**) ***)	Emergency gas removal system	Pipelines, valves	Coolant inventory maintenance	UJA UKD
CL*)**) ***)	<u>**Instrumentation and control system**</u>	The equipment performing functions of reactor emergency protection	Reactivity control	UJA, UCB
JDH***)	Emergency boron injection system	Pumps, pipelines	Coolant inventory maintenance. primary circuit Primary circuit make-up. Reactivity control	UJA UKD
JET*)	Reactor coolant leakage collection system	Pipelines, valves	Coolant inventory maintenance.	UJA UKD
JEW*)	RCP sealing water system	Pipelines, valves	Coolant inventory maintenance.	UJA UKA
JMN*)**)	Spray system	Pumps, pipelines	Pressure limitation in the containment. Standby heat removal from spent fuel.	UJA UKD
JMK*)**) ***)	Containment isolation system leak tight	Isolation valves, leak tight penetrations	LTV localization	UJA

	penetrations			
JMP***)	System of passive heat removal from Containment	Tanks, pipelines, valves	Pressure limitation in the containment.	UJA UJB
JMR***)	Corium Localization System	Melt catcher	In standby mode In standby mode.	UJA
JMT**) ***)	Hydrogen removal system from containment	Recombiners	In standby mode In standby mode.	UJA
JMU**) ***)	Hydrogen concentration monitoring system.	Hydrogen concentration sensors	In standby mode In standby mode.	UJA
JNA*)**)	Residual Heat Removal System	Pipelines, valves	Heat removal from the primary circuit.	UJA UKD
JNB***)	System of passive heat removal through steam-generators	Tanks, pipelines, pump, mobile diesel-generator set.	Heat removal from the secondary circuit.	UJA UJB
JNB90***)	System of emergency water usage from the reactor internals inspection well	Pipelines, valves.	In standby mode In standby mode.	UJA
JND**)	High-Pressure Safety Injection System	Pumps, pipelines.	Coolant inventory maintenance. Feed of the primary circuit.	UJA UKD
JNG10,20,30,40*)**)	Low-Pressure Safety Injection System	Heat exchangers, pumps.	Coolant inventory maintenance. primary circuit Primary circuit make-up. Heat removal from the primary circuit.	UJA UKD
JNK*)**) ***)	Borated water storage system referred to seismic category I.	Tanks, pipelines.	Coolant inventory maintenance. primary circuit Primary circuit make-up.	UKA UKD UKA
KAA10,20,30,40*)**)	Intermediate cooling circuit for essential loads	Heat exchangers, pumps.	Heat removal from the primary circuit.	UJA UKA UKC UKD
KAB10,20,30,40*)	High-pressure intermediate cooling circuit for important Consumers	Heat exchangers, pumps.	Heat removal from the primary circuit.	UJA UKA
KLA10,20,30,50,60,80*)	Ventilation system of the reactor building	Fans, heat exchangers.	Life support.	UJB
KLA13*)	Recirculation air-purification system of the containment rooms	Fans, filters.	Life support.	UJB
KLC10,20,30,40*)**)	Emergency air-cooling system of the between-	Recirculating cooling unit.	Life support.	UJB

	reactor-shell rooms			
KLC11,21,31,41*)**)	System of leakage removal from the reactor building containment and safety building	Fans, filters	In standby mode	UJA UJB
KLD10*)	Plenum and exhaust ventilation systems for vacuum creation in the reactor building containment	Fans, filters	In standby mode In standby mode.	UJA UKA
KLD20*)	Plenum and exhaust ventilation systems for repair works in the reactor building	Fans, filters, air-conditioner	Life support.	UJA UKA
KLE10,20,30*)	Components of the main plenum and exhaust ventilation systems of the rooms in the controlled-access area referred to seismic category I .	Fans, filters, air-conditioner	Life support	UKA
KTA*)	Components of the reactor building equipment drainage system referred to seismic category I	Pipelines, valves.	Coolant inventory maintenance.	UJA UKA UKD
KTB*)	Components of the Vent system of Reactor building equipment referred to seismic category I	Pipelines, valves, regulators.	Coolant inventory maintenance.	UJA
LAR*)**) ***)	Emergency feed water system.	Pipelines, valves, regulators.	Coolant inventory maintenance. Heat removal from the secondary circuit.	UJA UJE
LAS*)**) ***)	Pumps of the emergency feed water system.	Pumps.	Coolant inventory maintenance. Heat removal from the secondary circuit.	UJE
LBA*)**) ***)	Components of the main steam pipelines referred to seismic category I.	MSIV (Main steam isolation valve).	Coolant inventory maintenance.	UMA UJE
LBU*)**) ***)	Main steam pressure relief system	BRU-A (Fast acting steam dump valve for steam discharge into atmosphere).	Coolant inventory maintenance. Heat removal from the secondary circuit.	UJE
LCU*)**) ***)	Components of the Make-up water system referred to seismic category I	Tanks, pipelines	Coolant inventory maintenance. Heat removal from the secondary circuit.	UMA UJE
PEA*)**)	System of mechanical	Spray cooling pools,	Heat removal from	UQC

	cleaning devices for cooling water of the essential loads	pipelines, valves.	the primary circuit.	URR
PEB*)**)	Cooling water piping system for essential loads	Pipelines, valves.	Heat removal from the primary circuit.	UKD UQC URR URS UQZ URZ
PEC*)**)	Pump units of the cooling water system for essential loads	Pumps	Heat removal from the primary circuit.	UQC
QKA*)**)	Components of the system supplying cooling water to the power unit consumers, operating under loss of power referred to seismic category I	Pumps	Life support.	UGB UKA UCB UBN
QKC*)**)	Systems of chilling medium supply for ventilation systems of Control building	Refrigeration machines (air cooled condenser).	Life support.	UCB
QKD*)**)	Systems of chilling medium supply for ventilation systems of SDGS building .	Refrigeration machines (air cooled condenser).	Life support.	UBS
QKS*)**)	Chilling medium supply system of Steam cell ventilation systems .	Refrigeration machines (air cooled condenser).	Life support.	UMA UGB
SAA	Ventilation systems for PHRS tanks rooms	Air-conditioners with built-in chilling machine .	Life support.	UJB UJG
SAC*)**)	Ventilation, air-conditioning systems of Control Building rooms	Air-conditioners, electric heaters.	Life support.	UCB
SAD*)**)	Ventilation and chilling medium supply systems of Standby diesel-generator station building of EPSS	Air-conditioners, electric heaters.	Life support.	UBS
SAS*)**)	Components of the Ventilation systems of Steam cell referred to seismic category I	Air-conditioners, electric heaters.	Life support.	UJE
SAG*)**)	Ventilation systems and cooling of rooms of the pumping building of responsible consumers.	Fans	Life support.	UQC
SAG*)**)	Ventilation systems for the rooms of Pumping station of essential loads	Fans	Life support.	URS

SGA*)**)	Components of the Fire-fighting water supply system referred to seismic category I	Pumps, pipelines	Fire protection.	UFG
SGB*)**)	Modular units system for automatic sprayed water fire-fighting	Modules, pipelines, spray nozzles.	Fire protection.	UBS UBN UXS
SGE*)**)	Components of the Modular units system for automatic sprayed water fire-fighting referred to seismic category I	Modules with standby control panel, pipelines, spray nozzles.	Fire protection.	USG UXR UCB UKD UKT UXS USG
XJ10,20,30,40*)**)	Standby diesel generator	Diesel generators.	Power supply	UBS
XJA10,20,30,40HA 001	Diesel engine	Diesel engine.	Generator actuation	UBS
XKA10,20,30,40AG 001	Generator	Generator	Power generation	UBS
XJG10,20,30,40	Cooling system.	Pumps, electric heaters, heat exchangers, tanks, pipelines, valves.	Diesel engine cooling.	UBS
XJN10,20,30,40	Fuel supply system.	Pumps, heat exchangers, tanks, filters, pipelines, valves.	Fuel supply to the diesel engine	UBS
XJV10,20,30,40	Diesel fuel storage and supply system	Pumps, heat exchangers, tanks, pipelines, valves.	Lubrication of diesel engine and generator bearings	UBS
XJP10,20,30,40	Startup system.	Compressors, compressed-air flasks, pipelines, valves.	Diesel engine start-up	UBS
XJR10,20,30,40	Gases exhaust system.	Dampers, pipelines, valves.	Exhaust gases discharge	UBS
XJQ10,20,30,40	Air intake system.	Filters, pipelines, valves.	Air intake for combustion in the diesel.	UBS
BE BN BS BW BT XKA10-40*)**)	Emergency power supply.	Accumulator batteries and SDPP.	Power supply.	All buil-dings
BS BT XKA70 ***)	Power supply under BDBA	UPSU (uninterruptible power supply unit), accumulator batteries of the 7th channel and MEDGS.	In standby mode In standby mode.	All buil-dings
CL*)**) ***)	Instrumentation and control system of the safety system (SS I&C)	I&C cabinets.	Control	UJA UCB

CL*)**) ***)	Components of the CPS electric equipment complex referred to seismic category I: - EPS - DBE - APC - Group and individual control system	I&C cabinets	Control	UJA UCB
CN*)**)	Components of the Monitoring, Control and Diagnostics System referred to seismic category I	Computer complexes.	Control.	UCB
CW*)**) ***)	Components of the On-line Supervisory Control Hardware referred to seismic category I.	MCR / ECR panels	Control and monitoring	UCB
CYE*)**)	Instrumentation and control system for fire protection system (including fire signaling) (FP I&C)	FP I&C cabinets.	Control	All buil-dings

Analysis of the main systems taking into account influence of the SSE-level earthquake is given in Table 3.1.2.2, 3.1.2.3.

In addition, during assessment of the earthquake influence on operation and condition of 360(205)/32+10t electric travelling polar crane for the Belarusian NPP units 1 and 2 the following was defined:

The crane refers to seismic category I as per NP-031-01.

The electric equipment used in the crane meets the requirements of GOST 17516.1 and GOST 16962.2 by shock and vibration resistance. The crane is equipped with the cabinets (AK1.1, AXT1, AXT2, AXT3 and AXT4) which metalwork meets the requirements of seismic category I. To protect the crane parts from falling under seismic impacts fastening of the electric equipment and crane metalwork meets the requirements of seismic category I.

The electric equipment installed on the crane and in the cabinets and boxes withstands the mechanical impacts caused by crane operation and maintains operation under shock loads not exceeding the DBE level. Under seismic impacts of the SSE level falling of the electric equipment installed on the crane and loss of the cabinet integrity is avoided.

Table 3.1.2.2. An earthquake with intensity above the DBE level and up to the SSE level, RP initial condition under power operation

Object (parameter, element, construction, equipment, etc.)	Explanation	Consequences
Reactor building		
	CPS CRs switch the reactor to the subcritical state following the relevant signal (without use of boric acid in the primary circuit). Under power loss the MCP is switched off. Increase of boric acid concentration in the primary circuit due to injection of boric acid solution with high concentration – JDH system.	Reactivity change is avoided
	Natural circulation in the primary circuit (under tripped MCP). Heat removal from SG via the BRU-A.	Heat is removed from the reactor core after SSE.
Safety systems are available.	Safety systems refer to seismic category I as per NP-031-01. Water-supply (~ 2150 m^3) for SG make-up under cool-down is sufficient for switching the RP to the "cold" state.	Heat is removed from the reactor core after SSE.
Supporting systems		
Intermediate component cooling circuit(КАА/КАБ) is available The PE system is available	The KAA/KAB/PE systems refer to seismic category I and II as per NP-031-01. The systems and components referred to seismic category II and III are isolated from the systems and components referred to seismic category I as per NP-031-01 following the relevant automatic signal. The KAA/KAB/PE safety-related components under power loss are powered from the emergency power supply system (DG is put in operation according to the step-by-step start-up program)	Heat removal from the safety systems and safety-related systems is provided.
Available: - Ventilation and air conditioning systems	Ventilation and air cooling systems refer to seismic category I as per NP-031-01. The systems and components referred to seismic category II and III are isolated from the systems and components referred to seismic category I as per NP-031-01 following the relevant automatic signal.	The required ambient conditions are provided

Table 3.1.2.2 (continued)

Name of object (parameter, element, construction, equipment, etc.)	Explanation	Consequences
Systems important for safety.	The systems and components referred to seismic category I are isolated from the systems and components referred to seismic category II, III following the relevant automatic signal	The components which are not powered from the emergency power supply system are not available (under loss of normal operation power supply)
Parameters of the primary circuit are stable.	MCPs are tripped (under loss of normal operation power supply). Leakage through the MCP seals is max.0.05 m^3/h during the first 24 hours. Then it is max. 0.5m^3/h. Make-up of the primary circuit is performed from the JDN system. The pressure is maintained: - Injection from system JDH is provided to reduce pressure - Pressurizer PS**D** is provided to protect from over-pressure Heat is removed from the secondary circuit through the BRU-A.	Heat removal from the reactor core after SSE
Fire-fighting		
Fire-fighting systems.	Fire-fighting systems refer to seismic category II and III as per NP-031-01. The exception is made for fire-fighting water tanks in buildings 01UGF and 02UGF referred to seismic category I as per NP-031-01 maintaining water-supply under an earthquake with intensity from the DBE level to the SSE level. The systems and components referred to seismic category II and III are isolated from the systems and components referred to seismic category I	Fire-fighting water tanks are available.
Cooling pool		
Parameters of the cooling pool are stable.	Parameters of the cooling pool are maintained by the FAK system. As required, the JMN system channel can be put in operation. The JMN and FAK systems refer to seismic category I as per NP-031-01 and are powered from the emergency power supply system.	Residual heat removal from spent fuel is provided

Table 3.1.2.2 (continued)

Name of object (parameter, element, construction, equipment, etc.)	Explanation	Consequences
<u>Turbine compartment</u>		
Turbine, systems of the condensate and feed water path and systems supporting their operation under unavailable state	The turbine and systems of the condensate-feed path refer to seismic category II as per NP-031-01.	Function of power generation and heat removal through the BRU-K is not provided.
Parameters of the secondary circuit are stable.	Pressure is maintained by the BRU-A. The BRU-A refers to seismic category I. The water level in SG is maintained by the LAR/LAS system (emergency feed water system).	Heat is removed from the reactor core through the BRU-A
<u>Main buildings and structures</u>		
It is guaranteed that the buildings referred to seismic category I are saved (see s. 2.1.2).	The main safe-related buildings and structures refer to seismic category I as per NP-031-01	There is no radiation effect.
<u>Emergency power supply sources</u>		
Emergency power supply systems are available.	The emergency power supply systems refer to seismic category I as per NP-031-01; they are put in operation under loss of normal operation power supply.	All components powered from the emergency power supply system are in operation. DG is put in operation according to the step-by-step start-up program.
<u>Other power supply sources</u>		
Power supply system of normal operation (including UDPS) is not available.	Systems and elements of NO power supply refer to seismic category II as per NP-031-01.	Systems and components powered from the normal operation power supply sources are not available.
DBA power supply system	The DBA power supply system refers to seismic category I as per NP-031-01.	Available

Table 3.1.2.2 (continued)

Name of object (parameter, element, construction, equipment, etc.)	Explanation	Consequences
<u>Mobile power supply sources</u>		
Mobile power supply sources are available	Mobile diesel-generator plant, no effect during and after seismic impacts.	Available to be connected to the regular connectors under DBA.

Table 3.1.2.3. An earthquake with intensity above the DBE level and up to the SSE level, RP "cold" initial condition

Object (parameter, element, construction, equipment, etc.)	Explanation	Consequences
Reactor building		
The reactor is in subcritical state	CPS CRs are in the lower abutment stop position.	Reactivity change is avoided Heat removal from the reactor core is provided.
	In the reactor and in the system pipelines boron acid solution is only concentration min. 16g/dm^3.	
	The safe systems removes residual heat from fuel to the terminal absorber.	
Safety systems are available.	The safety systems refer to seismic category I as per NP-031-01. Components of the safety systems are powered from the emergency power supply system.	Heat removal from the reactor core is provided.
Supporting systems		
Intermediate component cooling circuit(KAA/KAB) is available	According to Table 3.1.2.1.	Heat removal from the reactor core is provided. Heat removal from the safety systems is provided.
PE system is available	According to Table 3.1.2.1.	
Available: - Systems providing ventilation and conditioning of rooms.	According to Table 3.1.2.1.	The required ambient conditions are provided
Parameters of the primary circuit are stable.	MCPs are disconnected (including under loss of normal operation power supply). The primary circuit make-up is not required. Residual heat is removed from fuel the safety systems.	Heat removal from the reactor core is provided.
Fire-fighting		
Fire-fighting systems.	According to Table 3.1.2.1.	Fire-fighting water tanks are available.

Table 3.1.2.3 (continued)

Name of object (parameter, element, construction, equipment, etc.)	Explanation	Consequences
<u>Fuel cooling pool</u>		
The cooling pool parameters are stable	According to Table 3.1.2.1.	Residual heat removal from spent fuel is provided
<u>Turbine compartment</u>		
Turbine, systems of the condensate and feed water path and systems supporting their operation under unavailable state	The turbine and systems of the condensate-feed path refer to seismic category II as per NP-031-01.	Function of power generation and heat removal through the BRU-K is not provided.
<u>Main buildings and structures</u>		
It is guaranteed that the buildings referred to seismic category I are saved (see s. 2.1.2).	According to Table 3.1.2.1.	There is no radiation effect.
<u>Emergency power supply sources</u>		
Emergency power supply systems are available.	The emergency power supply systems refer to seismic category I as per NP-031-01; they are put in operation under loss of normal operation power supply.	All components powered from the emergency power supply system are in operation. DG is put in operation according to the step-by-step start-up program.
<u>Other power supply sources</u>		
Power supply system of normal operation (including UDPS) is not available.	Systems and elements of NO power supply refer to seismic category II as per NP-031-01.	Systems and components powered from the normal operation power supply sources are not available.
DBA power supply system	The DBA power supply system refers to seismic category I as per NP-031-01.	Available. Not required.
<u>Mobile power supply sources</u>		
Mobile power supply sources are available	Mobile diesel-generator plant, no effect during and after seismic impacts.	Available to be connected to the regular connectors. Not required.

The accepted design solutions provide the relevant seismic inventory of power unit buildings and structures according to the accepted SSE level.

3.1.3 License Compliance of NPP

Comparison of the seismic levels accepted for the Belarusian NPP design with the seismic levels obtained by the seismic zoning data shows that the design levels have a margin minimum 10% in all PES zones. At the same time, the design response spectrum for the NPP site is completely covered by the standard spectrum as per NP-031-01 used in the basic project (see Fig. 3.1.3.1 and 3.1.3.2). Thus, seismic stability of the NPP meets the regulatory requirements.

Compliance of the NPP with the license requirements means obligatory meeting the requirements in the field of nuclear energy and the requirements of the operational manuals and operating organization procedures developed thereupon.

To confirm compliance of the Belarusian NPP construction to the license requirements inspections are performed by the following organizations:

- Federal inspectorate for nuclear and radiation safety authority (Gosatomnadzor) in the field of compliance with nuclear and radiation safety requirements;
- Other republican state bodies in the field of compliance with construction, industrial, sanitary and fire safety requirements;
- General contractor in the field of compliance with engineering and design documentation requirements;
- Operation organization in the field of compliance with the regulatory requirements, quality assurance program and requirements of the design and operational documentation.

Operational organization documentation provides scheduled maintenance, repair and tests of the equipment and monitoring of the building structures. The maintenance and repair programs are also applied to the portable equipment and expendables used for the emergency operation to provide their availability.

Requirements for arrangement and documentation of the NPP equipment maintenance and repair are specified in the relevant standards of the operating organization. The procedures of taking the equipment out of service for repair and its acceptance after repair as well as quality assessment are specified in the relevant instructions.

The regular actions of the operating organization personnel under seismic impacts are specified in the operational documentation.

Additional actions of the operating organization personnel to provide operation ability of the systems and components required for the RP safe shutdown after an earthquake are not required.

Actions of the operating organization personnel to provide availability of the portable equipment and resources after an earthquake are described in the operational documentation, for example instructions of the Severe Accident Management Guidelines (SAMG) and the Beyond Design Basis Accidents Management Guideline (BDBAMG).

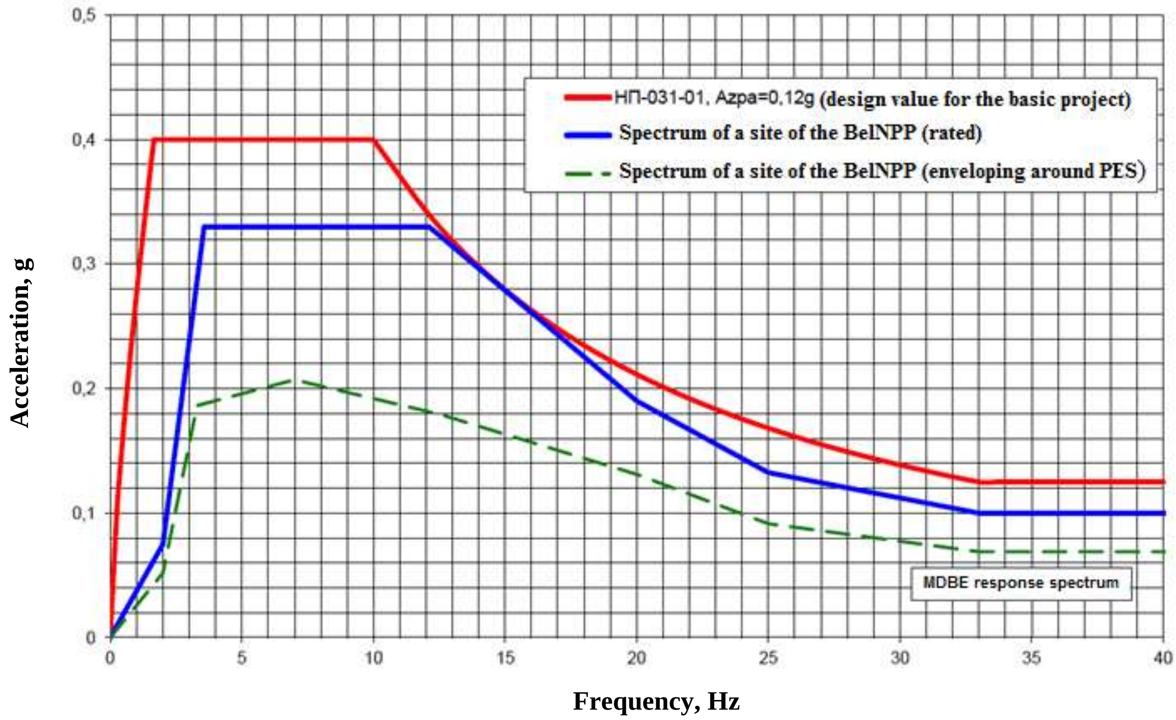

Figure 3.1.3.1. Comparison of the response spectrum, adopted for the basic project SSE with the response spectrum of the Belarusian NPP site under SSE (horizontal component, attenuation 5%).

НП-031-01	NP-031-01

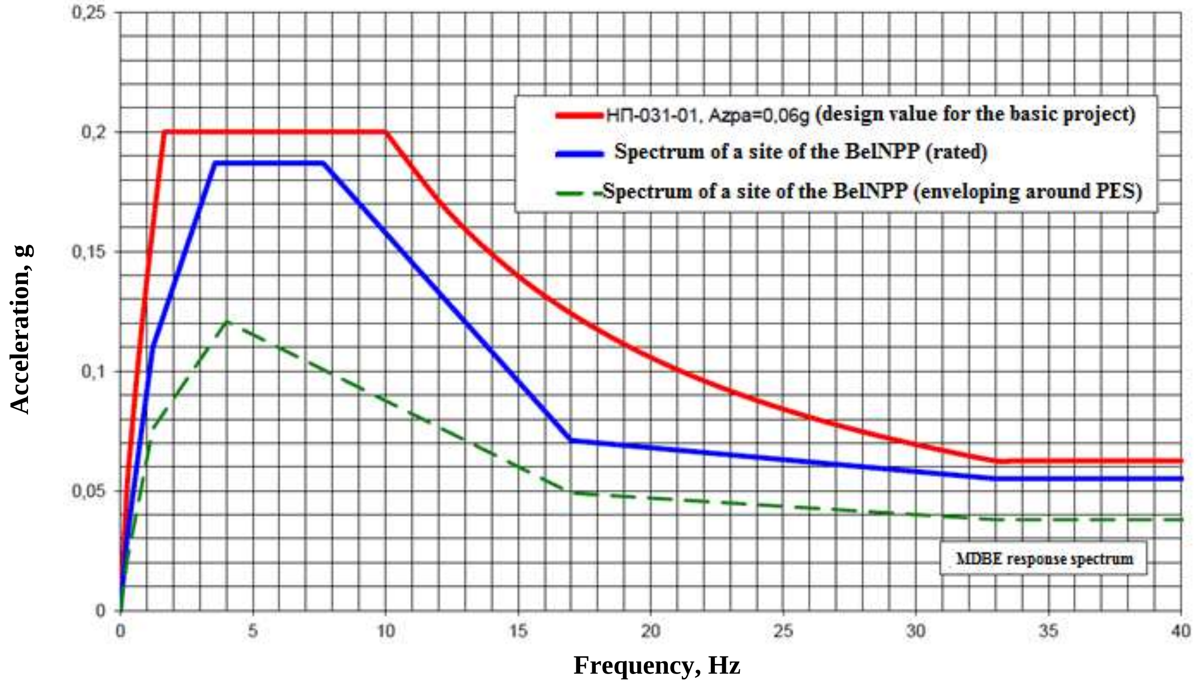

Figure 3.1.3.2. Comparison of the response spectrum adopted for the basic project DBE with the response spectrum of the Belarusian NPP site under DBE (horizontal component, attenuation 5%).

НП-031-01	NP-031-01

3.2 Assessment of Safety Margin

3.2.1 Earthquake Intensity Leading to Accident with Severe Nuclear Fuel Damage

3.2.1.1 RP Equipment Fixation Safety Margin

The reactor coolant circuit includes: reactor, four loops of the reactor coolant pipeline, steam generators, reactor coolant pump units. The reactor consists of a vessel with internals, a core and an upper unit.

The main bearing support of the reactor is a supporting truss which bears the reactor vessel load and all other types of operational loads, including seismic and emergency impacts. To withstand horizontal dynamic load and to protect the vessel from turning over a trust truss is installed at the level of the reactor vessel flange connector.

The upper unit includes a reactor head with 121 CPS branch pipes with solenoid drives, in-core monitoring branch pipes and prefabricated metalwork consisting on a truss and six bars with the upper and lower plates limiting movements of the CPS casings in the horizontal plane.

Fixation of the upper unit in case of dynamic impacts is performed at the truss level by metalwork of the electrical connection block (ECB).

The RCP loops are identical by routing and equipment installation. Steam generators of the horizontal type are provided with cylindrical bodies and elliptic bottoms. They are installed on two roller supports. As a seismic supports for each SG four pairs of R-450 hydraulic snubbers located at the SG edges are used. The upper and lower hydraulic snubbers in couple are installed by height symmetrically to the SG central axis.

The RCPU is of the vertical type and consists of an electric motor, upper and lower space plates, spherical body and a roller support. As a seismic supports for each RCPU two R-300 hydraulic snubbers on the motor and one rod on the body are used. SG and RCPU fasteners do not interfere with temperature expansions of the equipment and RCP.

The pressurizer system is designed to maintain pressure in the primary circuit and consists of a pressurizer and pipelines (connecting, injection and discharge).

The pressurizer is a vertical cylindrical vessel with two elliptic bottoms and a set of tubular electric heaters filled with coolant. The pressurizer is provided with two supports: lower supporting shell welded to the support on the pressurizer lower bottom and the upper thrust unit representing a metalwork and withstanding only horizontal loads.

The connecting pipeline connects the "hot" line of loop No. 4 of MCC with the pressurizer and it is designed for coolant flow from the primary circuit into the pressurizer and back under change of temperature and coolant in the primary circuit. The injection pipeline is designed to supply coolant from the “cold” line of loop No. 3 to the pressurizer steam space during pressure maintenance in the primary circuit. The discharge pipeline connects the pressurizer steam space with pressurizer bubbler and is designed for steam relief via the presurrizer pulse safety devices.

Fasteners of the connecting pipeline consist of two R-50 hydraulic snubbers designed to withstand the dynamic loads. On the injection pipeline fixed supports and viscoelastic dampers to withstand the dynamic loads are installed. The discharge pipeline is also equipped with fixed supports. Hydraulic snubbers are also installed to withstand the dynamic loads.

The ECB is a part of the reactor equipment and is designed to arrange and transport the loops of the power utilities for servicing the upper unit and retaining it and the loops under seismic impacts.

The ECB includes a metalwork and power utilities loops placed on it. The metalwork consists of the support plate, two maintenance platforms and flooring connected by sixteen vertical poles. The support plate is fixed on an inset ring of the concrete vault by ten M64-bolts. The ECB is also equipped with two M64-pins for reliable fixation. The power utilities loops are fastened on handrails and floorings of these two platforms.

For the mentioned-above RP equipment seismic analysis of the equipment, pipelines and their supports under an earthquake of 8 points intensity [32] gas been performed. The main method of the RP equipment seismic analysis is a response-spectrum method using floor-by-floor response spectrums as initial seismic impacts.

The method of seismic margin analysis represents design assessment of the maximum loads on the equipment expressed in terms of an earthquake intensity of 8 points for basic design spectra.

If seismic stability of the equipment is not provided, then the reduction factor considering the difference between the basic design spectra and the NPP site spectra is used for the main natural frequencies of the equipment.

The analysis has shown than the main RP equipment – reactor, SG, RCPU, RCP, pressurizer , ECB and connecting pipeline are provided with the required margin to withstand loads under an earthquake with 8-points intensity. Resistance conditions under an 8-point earthquake are not provided for the emergency core cooling system (ECCS), injection and discharge pipelines and pressurizer system, metalwork of the reactor upper unit, spent fuel pool, RCPU anti-seismic fixation rod. At the same time, the reactor upper unit is provided with a 10% seismic margin regarding the SSE level of 7 points, the ECCS has a 35% safety margin, and spent fuel pool storage rack are provided with a 20% seismic margin regarding the design SEE level [32].

3.2.1.2. Condition of the Safety-Related Seismic Resistant Components of the Safety Systems after Seismic Impact with Intensity Exceeding the Threshold Value

The passive part of the emergency core cooling system is designed for automatic supply of cooling water to the reactor core in case of coolant leakage and is put in operation under emergency leakage of the primary circuit.

The ECCS consists of four tanks and four pipelines connecting them to the reactor. The ECCS tank is a high-pressure thick-walled cylindrical vessel with two elliptic bottoms installed vertically on a support.

Immovable supports are installed on all four loops of the ECCS pipelines between isolation valves. The immovable support represents a metalwork with support assemblies on which pipelines are fixed. To withstand weight and earthquake loads the vertical rods are installed on these four pipelines. Horizontal rods are installed on pipelines No. 2, 3 at elevation+19.500m and on pipelines No. 1, 4 at elevation+16.100m. On the pipeline No. 3 at elevation+16.100m an immovable support is installed. Also to withstand dynamic and emergency loads two R-50 hydraulic snubbers are installed on pipelines No. 2 and 3 at elevations +12.700m and +14.500m.

The ECCS tank support shell, the ECCS pipelines and their fasteners are not provided with the sufficient margin to withstand loads from the 8-point SSE. The design loads in the ECCS tank shell under normal operation and 8-point SSE exceed the acceptable values by 64%. The maximum exceeding of the acceptable loads in the ECCS pipelines is 40%. For the cooling pool racks under 8-points SSE the resistance criteria are exceeded by 48%. Thus, the ECCS has a 35% safety margin, and spent fuel pool storage rack are provided with a 20% seismic margin.regarding the 7-points SSE level.

The cooling pool racks are designed to arrange and store in the vertical position the spent FA and leak-tight bottle with the damaged FA as well as fresh FA prior to their loading into the reactor. The racks are made in the form of separate transport sections which design provides easiness of their installation and removal from the cooling pool. The cooling pool rack sections are installed on the supports welded to the embedded parts of the cooling pool floor. The limit stops separating sections from each other and from the cooling pool walls are welded to the top plates of the racks. The limit stops control the rack movement in the horizontal direction along the long side of cooling pool under external dynamic impacts.

The FA rack is a part of refueling system and is installed in the fresh fuel storage facility

(FFSF). The FA rack is designed to arrange and store in the vertical position the fresh FA and CPS AR frames. The FA rack represents a welded box-shaped metalwork in the form of a framework in which "borated" hexagon pipes with FAs are vertically placed.

12 pads for rack supporting on the embedded parts of the FFSF floor are welded to the rack bottom plate. To prevent the rack from moving under seismic impacts up to the SEE level, aircraft crash and shock wave the peripheral pads are welded to the embedded pads.

Under seismic impacts the major power factors determining resistance of the equipment and pipelines are bending moments due to inertial seismic forces. Allowable σ_2 – category stress is 1.3 [σ] under normal operation and 1.8 [σ] under normal operation +SSE where [σ] is an allowable stress.

Thus, under seismic impact stress increase by 1.8/1.3 = 1.4 times due to nonlinear behavior of the material in the point of plastic deformations is allowed. Considering the design margin for resistance by 1.5 times (i. 3.4 of PNAE G-002-86 "Equipment and pipelines strength analysis norms for nuclear power plants") it is possible to make the following conclusion – during an earthquake of the SSE level determined for the Belarusian NPP site the equipment and pipelines structures can remain in the zone of linear-elastic behavior of the material, i.e. as for the mode of normal operation.

As it was noted that allowable σ_2 – category stress (taking into account bending) is $k_{[\sigma]}$ where a form factor k = 1.3, is defined for a pipe. For the equipment and tanks this factor is, as a rule, theoretically equal to 1.5 (ACI359 "ASME BPVC-ASME Boiler and Pressure Vessel Code, Part III, Division 2"). In that case, there is an additional margin 15% for a shell.

During the design of penetrations the seismic impact is considered in a similar way as for the pipelines. Thus for penetrations the same approach is applied as for the pipelines. During design of the civil construction part the loads from penetrations, including seismic loads, were applied to the design model of the civil construction part, thus determination of margin for bearing ability of the building structures is appropriate and for these units.

The safe-related electrical equipment refers to seismic category I as per NP-031-01 and maintains operation ability under an earthquake of the 7-points SEE level as per the MSK-64 scale.

The threshold value of the equipment and pipelines resistance under seismic impacts is the SSE level with acceleration 0.12g accepted in the Belarusian NPP Project. When exceeding this threshold level there is a safety margin up to plastic deformation 0.2% min. 1.07 times (i. 3.6 of PNAE G-002-86) due to determination of the allowable pressure. Taking into account the accepted resistance margin for the equipment and pipelines the maximum admissible acceleration is 0.12 x 1.07=0.13g. Under further strength increase, zones with plastic deformation can be expected on the equipment and pipelines. In case of the relevant earthquake it is required to perform a careful inspection of all components condition in order to replace the damaged or deformed components.

3.2.1.3 General Assessment of Structures and Buildings Seismic Safety Margins

Seismic margins of the NPP buildings and structures calculated for the SSE level are ensured by the following conditions.

1) Design ground response spectrum is set with the margin respective to the site response spectrum with a probability of 84%, and the margin is about 10%.

During Seismic Margin Assessment (SMA) for buildings and structures the ground response spectrum with lower probability (50%) is accepted. At the same time, zero acceleration is maintained and in the peak zone acceleration is reduced by 25 – 15% (e.g. ref. NUREG/CR-0098 "US Nuclear Regulatory Commission, Development of Criteria for Seismic Review of Selected Nuclear Power Plants").

2) Since during the SMA large inelastic deformations in structures are allowed in comparison with the SSE seismic analysis, large attenuation values are used when calculating

seismic loads for building structures and equipment. E.g. for the reinforced concrete structures relative attenuation value is 0.10, not 0.07 as for the SSE analysis.

3) The margin is ensured by the conservative approach for material strength assessment (0.999 non-exceedance probability). During the SMA the standard material characteristics with 0.95% non-exceedance probability can be accepted.

4) Conservative approach for strength assessment of the structures and equipment materials. During the *SMA* in this case the factors reducing seismic load due to inelastic deformation can be used. The load may be reduced by several times.

For the structured referred to seismic category I as per NP-031-01 the recommend inelastic deformation ratio is K_1=0.625.

SP 14.13330.2011 (Updated edition of SNiP II-7-81* "Construction in Seismic Areas") for in-situ reinforced concrete structures specifies the value 0.22. IBC-2000 "International Building Code" for reinforced concrete structures with shearing walls specifies the value 0.18 – 0.22, and UBC-97 "Uniform Building Code" for the same structures gives the value 0.22.

When K_1=0.625 the seismic load reduction factor is $R = 1/K_1 = 1/0.625 = 1.6$, plasticity factor (ratio of admissible deformation to elastic deformation) is $\mu = 1.78$, when $K_1 = 0.22$, $R = 1/0.22 = 4.5$, and $\mu = 10.6$.

At the same time, the buildings bars are fractured under $\mu \approx 70$ (elongation at fracture is minimum 0.14, while elastic elongation is 0.002). Respectively, even with $K_1 = 0.22$ there is considerable margin before damage of the structure.

5) Safety margin of the structures under construction. Usually during development of the design documentation, the quantity of bars or element section in metalwork are accepted with some margin during calculation of the design loads. That is why when checking the integrity during the SMA this margin, if available, may be considered.

The mentioned-above margins prevent the possibility of cliff-edge effect (immediate failure) of the NPP building structures when loads are higher than under the SSE level.

Margins for the main structures are approximately 1.1 times when the design seismic criteria are applied (margin from i. 1 is only considered) and 1.1*4.5 = 4.95 when high inelastic deformations are accepted (margins from i. 1 and 4 are considered). The rest of margins are not used for this assessment. In this case the maximum acceleration value must not exceed 0.62g.

3.2.1.4. Comprehensive Assessment of the NPP Safety under Seismic Impact

According to the method of safety margin assessment specified in [22], an earthquake level leading to a severe accident must be determined. For this purpose it is required to analyze: "...vulnerable points and threshold effects: assessment of maximum horizontal acceleration on the ground, once exceeded, loss (failure) of the main safety functions or significant damage of nuclear fuel (in reactor vessel and (or) spent fuel storage pool) is imminent. To meet this requirement, it was necessary to assess the minimum seismic load that results in failure of the building structures, reactor unit, equipment and piping of the safety systems and safety-related systems.

For the building and structures referred to seismic I the ground seismic acceleration which exceeding may result in imminent damage is 0.62g [31].

For the reactor unit, including the cooling pool equipment, a special topical report [32] has been developed with the assessment of the possibility to adapt the V-491 RP Project for the Belarusian NPP site with 8-point earthquake intensity by the MSK-64 scale. The report shows that the main RP equipment – reactor, steam generator, reactor coolant pump, reactor coolant pipeline, pressurizer and connecting piping –is provided with the required margins to withstand the 8-point SSE loads. For the ECCS, injection and discharge pipelines of the pressurizer system, metalwork of the reactor upper unit and spent fuel storage pool the strength conditions under the 8-point SSE are not provided. For these elements the reactor developer gives recommendations to improve seismic resistance specified in [32] and section 7.3.1.

For the equipment and piping the maximum admissible acceleration is 0.13g considering the accepted safety margin.

Thus, the determining factor in assessment of the maximum horizontal acceleration on the ground resulting in "failure of the building structures and reactor unit, as well as equipment and piping of the safety systems and safety-related systems" is failure of the safety systems pipelines. In this case, the seismic impact must not exceed 0.13g.

The condition of roads inside and outside the NPP site during earthquakes is described in i. 5.2.1 of this report.

3.2.2. Earthquake Intensity Leading to Loss of Containment Integrity

The containment is aimed to prevent radioactivity emission into the environment under the SSE, to limit the emissions under the BDBE, as well as to protect the equipment and reactor building internal structures from the possible external impact. As a building structure, the internal pre-stressed containment refers to radiation and nuclear safety category I as per PiN AE - 5.6 "Norms of Construction Design of a Nuclear Station with Reactors of Different Types", and to seismic category I as per NP-031-01. As a mechanical system it refers to the safety system of class 2 as per TKP 170-2009 (02300) "General Provisions to Ensure Safety of Nuclear Stations (OPB AS)", with classified designation 2L.

To reach the mentioned above goals the containment is of double-type. Its structure consists of the internal leak-tight containment with localization functions and the additional outer containment.

The inner containment is designed in the form of a pre-stressed reinforced concrete structure. It is designed according to the requirements of the American regulations ACI Standard 359-13 "ASME BPVC – ASME Boiler and Pressure Vessel Code, Part III "Rules for Construction of Nuclear Facility Components", Division 2 "Code for Concrete Containments". This standard is the most comprehensive and well-developed international document in the field of containment design. The inner containment is designed also according to the Russian regulations PNAE G-10-007-89, Regulations for design of the reinforced concrete structures of localizing safety systems of nuclear plants (NP-010-98) and Rules to design and operate localizing safety systems of nuclear plants (NP-031-01). The inner containment is designed and installed according to the ASME regulations considered to be the tightest ones by combinations of loads and acceptance requirements.

In terms of resistance, the determining combination of loads during containment design is the combination of emergency overpressure and temperature during an accident. According to the requirements of ACI Standard 359-13 the overpressure 0.39 MPa is accepted with the safety factor of 1.5. The SSE+DBA combination is also considered in the Project.

Engineering seismic assessment of the inner containment has been performed by two methods to determine the impact threshold value. The first method includes strength calculations for the containment by the linear-spectral theory of seismic resistance during step-by-step increase of the acceleration level (design response spectrum is applied in accordance with NP-031-01). The second method is an assessment based on the design experience and strength inspection of this type of structures performed according the recommendations of EPRI-NP-6041 "A Methodology for Assessment of Nuclear Power Plant Seismic Margin").

By calculations the containment withstands the load 0.324g (2.7 times higher than the SSE load) under design strength criteria (operation of the structure at the elastic stage). If minor inelastic deformations are allowed (load reduction ratio of the containment is accepted as R=1/K=1.6 as for the structures referred to seismic category I as per NP-031-01), the threshold value is 0.51g (4.3 times higher than the SSE load).

According to EPRI-NP-6041 the threshold value of acceleration without direct inspection of the building structure the value= 0.6g is accepted priory. During seismic analysis of the building structures, the following margins were used:

1) Seismic response spectrum on the site is approximately 10% lower than the design one. It is justly the for low frequencies which are typical for the building natural frequencies.

2) The factors reducing seismic load due to inelastic deformations are used. According to SNiP II-7-81* for the in-situ reinforced concrete structures inelastic deformations ratio K_1 0.22 and seismic load reduction ratio R=1/K1=4.54 are used.

Thus, threshold seismic acceleration Amax is 0.12g * 1.1 * 4.54 = 0,6 g

Note that not all margins of the structure bearing capacity described in [31] are used.

When the minimum acceleration from the two methods mentioned above is accepted the threshold acceleration on the ground for the leak-tight containment is 0.51g (response spectrum as per NP-031-01, scaled to the zero acceleration level 0.51g).

3.2.3 Earthquake Exceeding the DBA Level for the NPP and Subsequent Flooding of the NPP Site

The NPP site is not subject to flooding during high-waters, flashes, blockages and ice jams since the site design elevation is 179.3 m BES, i.e. more than 51 m above the Viliya river water level with 0.01% probability in the area of Mikhalishki.

Based on the calculations made in 1972 by the Central Research Institute for Complex Use of Water Resources and the Institute of Hydrodynamics (Siberian department of the USSR Academy of Science, Novosibirsk) [29] the maximum water levels due to break of the Vileisk water basin dam located upstream do not exceed the level mark with 1% probability as the break wave from the dam location to the supposed water intake point is mainly quiet. It happens due to considerable remoteness of the water intake point from the dam location (140 km) as well as due to the excising structures (roads, bridges, etc.) in the section between the dam and the water intake point which are the natural barrier for the break wave and accumulate considerable amount of water in the upstream territories.

There is no information about seismic characteristics which would guarantee break of the dam.

Break of the Vileisk water basin dam is not threat to the NPP and its water intake facilities for the following reasons:

- design elevation of the NPP site 179.3 m BES is higher than the maximum design upstream water line of the Vileisk water basin (159.8 m BES) by 39.5 meters, i.e. the NPP site is not flooded under any conditions, and the NPP safety is guaranteed.
- calculations of the maximum water levels caused by break of the Vileisk water basin dam performed by the Central Research Institute for Complex Use of Water Resources and Novosibirsk Institute of Hydrodynamics of the Russian Federation show that the break wave running from the dam to the water intake point is quiet; the design time for the wave to reach the intake structures is 3.5 days, and the Viliya river water level by the intake structures does not exceed 123.52 m BES.
- design elevation of the water intake structure site is accepted as 130.3 m BES.

Temporary loss of water make-up source for the turbine equipment cooling system under rear natural events (maximum water level of the Viliya river, wind, waves on the Viliya river) and man-induced event "break of the Vileisk water basin dam) (with probability less than 10^{-8} 1/year) does not affect nuclear and radiation safety of the NPP and is compensated by organizational and technical measures. Operation of the safety systems does not depend on the turbine equipment cooling system, loss of water make-up from the Viliya revier does not violate safe operation of the NPP. When the natural events resulting in temporary loss of water refill resource in the turbine equipment cooling system are over it is necessary to inspect and, as required, repair the water intake facilities.

Possible measures to improve resistance of the NPP to floods are specified in section 4.2.2.

3.2.4 Possible Measures to Improve NPP Seismic Resistance

Buildings and structures of the Belarusian NPP are designed with consideration of the specified design impacts in accordance with the current regulatory base, therefore, there are no radiation consequences during DBE and SSE, and no extra measures for improvement are required. For the structures referred to seismic category I the margin is minimum 4.9 times higher (0.62g) relative to the design SSE level. The inner containment remains its integrity till the seismic impacts 4.3 times exceeding the SSE level (0.51 g). There are no threshold effects for the buildings and structures during such seismic impacts.

Buildings and structures of the Belarusian NPP are provided with considerable safety margin relative to the design seismic impact, and no extra measures to improve their seismic resistance are required.

The threshold acceleration value specified above for the structures referred to seismic category I is determined with a certain level of conservatism. Without the conservative approach, the threshold level can be further increased. The structures are made of in-situ reinforced concrete to avoid cliff-edge effect under increase of seismic impact.

The following organizational and technical measures are proposed to moderate the consequences of earthquakes exceeding the design values:

– To perform analysis of the documents under development on the personnel actions under accidents when seismic impact exceeds the design one. As required, to add the documents on the personnel actions providing diagnostic of the NPP, restoration of normal operation conditions, restoration of safety functions and prevention or limitation of the core damage consequences to the Process Regulations, RP Emergency Operating Procedure, Severe Accident Management Guidelines (SAMG) and the Beyond Design Basis Accidents Management Guideline (BDBAMG) as well as to the Personnel Protection Plan under Accidents;

- PSA development.

It is proposed to reassess seismic margins for the equipment and pipelines referred to seismic category I by the results of the Belarusian NPP finished construction and commissioning using the SMA method specified in EPRI-NP-6041 and NS-G-2.13.

4. FLOODING

The hydrographic system of the Belarusian NPP site is a part of the basin of the Baltic sea and Nyoman river.

The Vileyka reservoir is by Viliya river, and part of its water is directed into the Svisloch river by the Vileyka-Minsk water system to supply Minsk with water.

Major tributaries: right: Servech (75 km), Naroch (75 km), Stracha (59 km); left: Dvinosa (54 km), Iliya (66 km), Usha (75 km), Oshmyanka (105 km); this system of major tributaries and reservoirs of Viliya is shown in Figure 4.1.1.

In the NPP control area are 3 minor tributaries in the left of the Viliya: 17 km-long Gozovka, 9.3 km-long Polpe, and 9 km-long Losha (tributary of Oshmyanka).

4.1 Design Basis

Rivers of the district are of lowland snow-fed type which determines the general nature of the annual course of changing water levels– high snow-melting season, low summer-autumn season almost annually interrupted by rain flows, and late winter season due to frequent thaws.

Long-term estimated values of water levels, long-term amplitude of water level variations, and the maximum water level during spring and rainfall flows are determined with the estimated water flow data applied to the cross-sections of the Viliya at indicative sections Mikhalishki – Outlet of Stracha and Malye Sviryanki – Muzhily.

Estimated water levels corresponding to average annual water flows, highest flood and rainfall levels with set confidence, including 0.01%, with ice blockages, wind gales and other hazardous factors were considered; and lowest estimated winter and summer monthly and daily water levels with set confidence, including 97%, are provided in tables 4.1.1-4.1.2. The calculation was completed by the Central Research Institute for Complex Use of Water Resources [31]. Estimated water levels are confirmed by estimated water flows under the results of mathematical modeling of the water regime using all cross-sections of the Viliya measured by the Central Research Institute for Complex Use of Water Resources in 2008-2012 and Belgiprovodkhoz in 2012 [31].

Table 4.1.1 – Estimated water levels corresponding to average annual water flows, m BES. Viliya – water intake section, distance to the river mouth – 262.58 km.

Probability of exceedance	Water level, m BES
Long-term average annual	117.40
50 %	117.39
75 %	117.30
95 %	117.17
97 %	117.14

Table 4.1.2 – Estimated water levels corresponding to maximum spring water flows, m BES. Viliya – water intake section, distance to the river mouth – 262.58 km.

Probability of exceedance	Water level, m BES
0.01 %	127.80
0.1 %	125.70
1.0 %	123.52
3 %	122.60
5 %	122.06

10 %	121.54
25 %	120.44
50 % (close to bed building)	119.12
at maximum observed water flow (1958, 1.69% confidence)	123.49

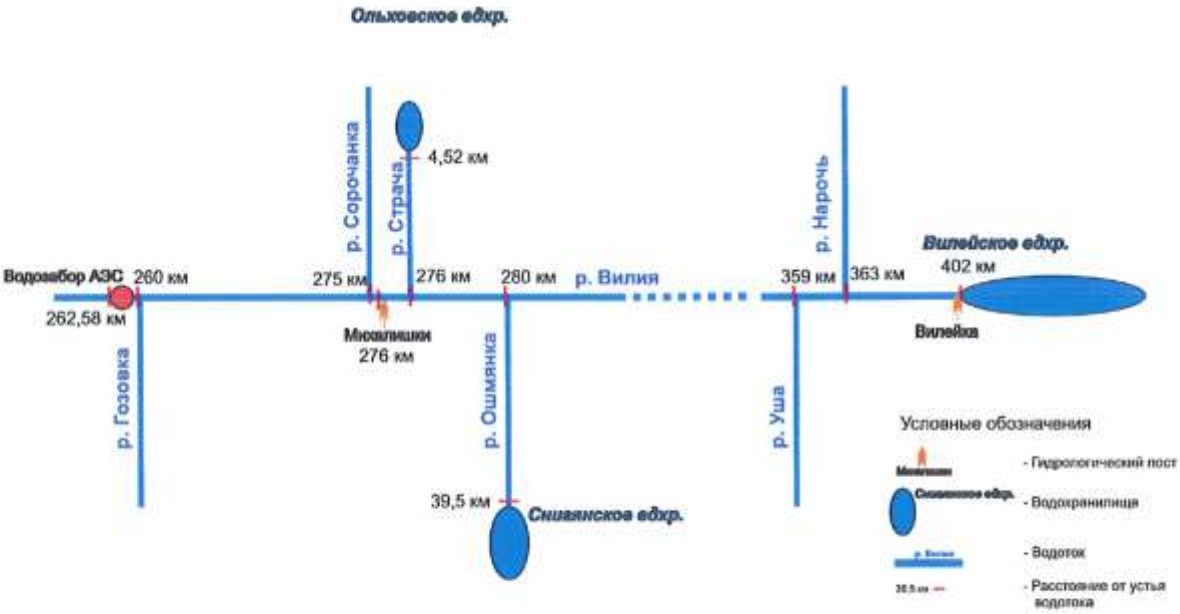

Figure 4.1.1 – Scheme of major tributaries and reservoirs of Viliya

4.1.1 NPP Design-Basis Flooding

Figure 4.1.1.1 shows the areas that could be flooded at maximum water levels during spring floods of 0.01% confidence. As the figure shows, the NPP site cannot be flooded.

Maximum water levels are conditioned by the wave after the break of the Vileyka reservoir which is located higher, based on the calculations made in 1972 by the Central Research Institute for Complex Use of Water Resources and the Institute of Hydrodynamics (Siberian department of the USSR Academy of Science, Novosibirsk) [29] will not exceed the level elevation with 1% confidence as the break wave from the dam location to the supposed water intake point will mostly calm. This will happen due to considerable remoteness of the water intake point (Malye Sviryanki) from the dam location (150 km), as well as due to the excising structures (roads, bridges, etc.) in the area between the dam and the water intake point which would be the natural barrier for the break wave and will accumulate a considerable amount of water in the higher-level territories.

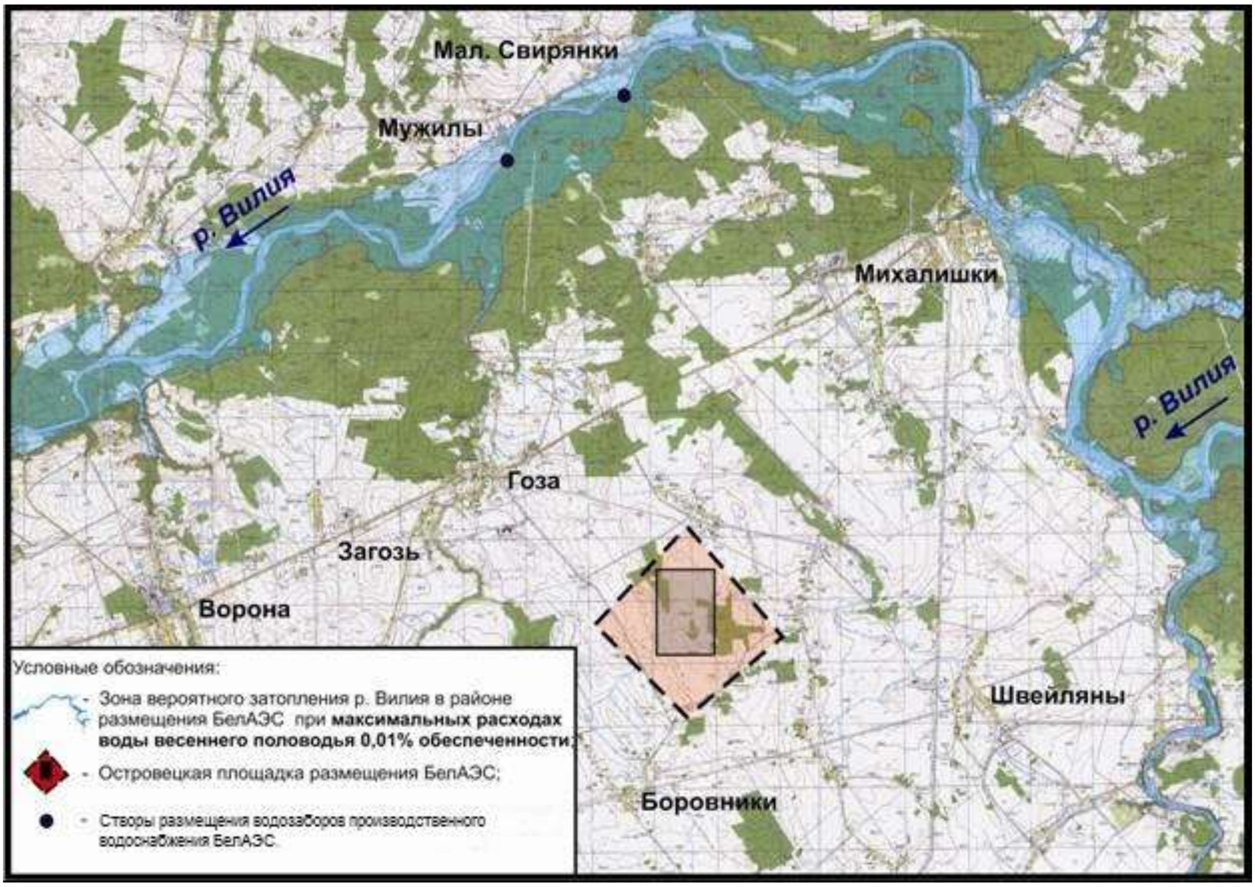

Figure 4.1.1.1 – Areas of possible flooding at maximum water flows during spring floods of 0.01% confidence

Условные обозначения	Legend
Зона вероятного затопления р. Вилия в районе размещения БелАЭС при **максимальных расходах воды весеннего половодья 0,01% обеспеченности**	Zone of possible flooding by Viliya in the Belarusian NPP location area at **maximum spring water flows with 0.01% confidence**
Островецкая площадка размещения БелАЭС	Ostrovets site of the Belarusian NPP
Створы размещения водозаборов производственного водоснабжения БелАЭС	Locations of water intake for industrial water supply for the Belarusian NPP

The top aquifier of Sozh terminal moraine debris determines hydrogeological conditions of the construction and operation of the NPP. Its special features are artesian and non-artesian waters, water head at the locations where subface of terminal moraine stratum deflects, and closed loops of non-artesian areas. The aquifier is fed by the influent seepage of precipitation.

It is drained by small rivers, streams and a melioration system into the Gozovka, Viliya, Oshmyanka and Losha, and flows into the lower aquifiers. The underflow is directed to the rivers and their tributaries.

The standing level depth is 17.2-22.1 m, the absolute elevations being 157.18-162.67 m. The highest absolute elevations of the levels are in the north-eastern part of the site, while the water table slopes to the south-west. The stream slope is 0.002-0.009. Where the subface of the top moraine stratum level becomes lower, the water head is 0.7-10.7 m. Sections with non-artesian water level are mostly found in the southern part of the site.

0.01% confidence level of the Viliya is 127.80, which is more than 30m below the standing ground water level, and there is no impact of the top aquifier on the ground water level.

Fluctuations of the level due to natural regime factors are observed to be no more than 0.6 m (mostly fluctuations are within 0.13-0.26 m range). Regardless of what the observations show, to protect the foundation and to prevent possible flooding of the underground power unit buildings, as well as the general, administrative and production operating basements from ground water, the design provides for stratum drainage. At the same time, the analysis below leads to the conclusion that ground waters cannot reach the bottom of the foundation.

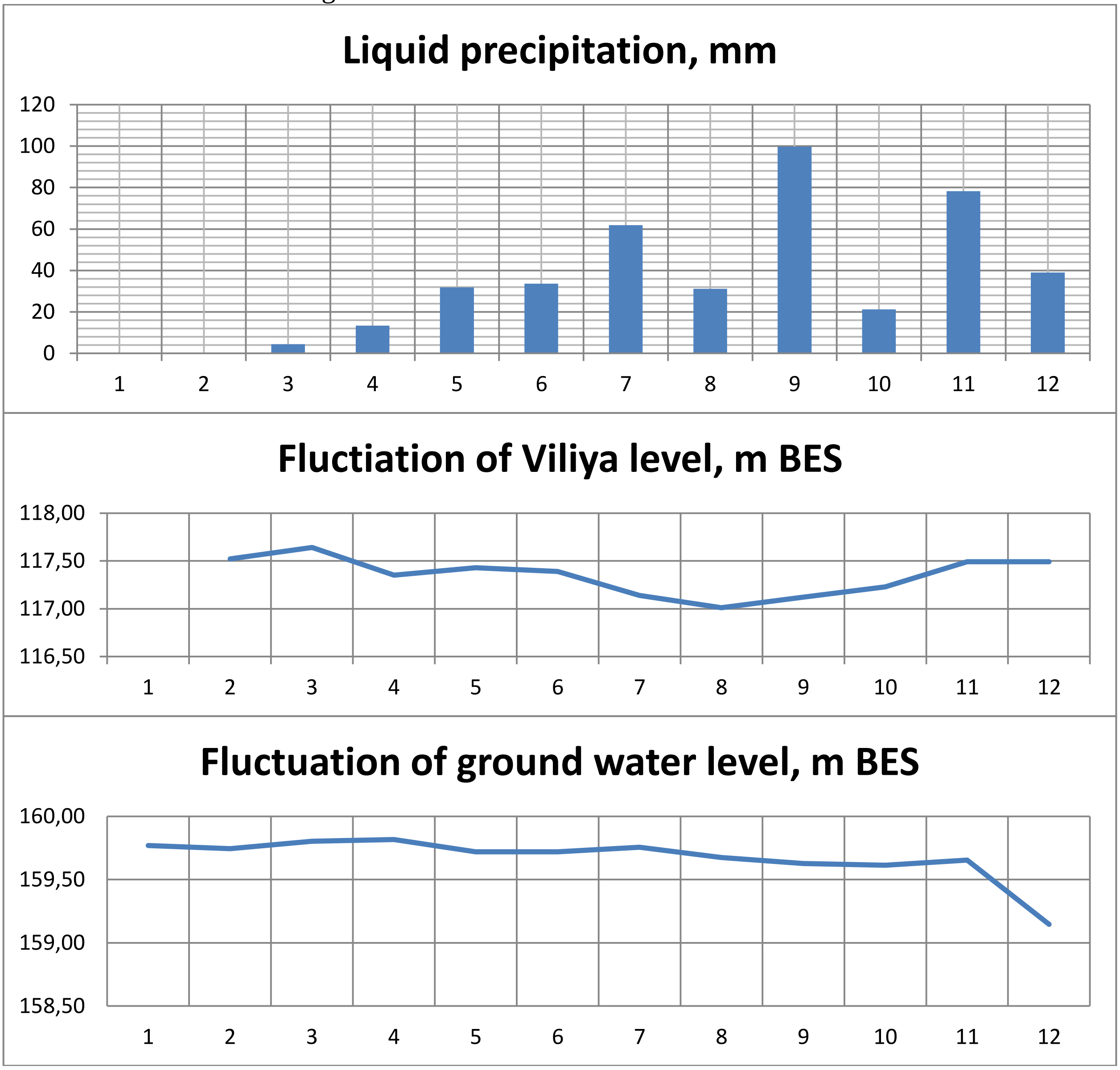

Figure 4.1.1.2 – Levels of precipitation, curves of Viliya level fluctuation and ground water level fluctuation throughout a year.

For this analysis of reciprocal influence of surface and ground waters affected by precipitation observations from the full calendar year 2015 were used.

Combined The combined figure 4.1.1.2 shows that temporary charts changes in precipitation, level fluctuations of the Viliya and fluctuations of ground water level do not correlate. Also, the curve of ground water level fluctuations shows that ground water level does not exceed 0.6 m throughout the year. The maximum level is observed in April, while the minimum is in December;

Analysis of the curve shows that the ground water level on site is not connected to the water level in the Viliya, and has no connection to precipitation either, i.e. natural physical principles of forming ground water run-off in conditions of considerable remoteness of the site from natural drainage, absence of localized ground water on site (hydrogeological windows) and any significant feed from influent seepage are fully complied with.

This is confirmed by groundwater levels remaining unchanged during construction in conditions of land run-off and longstanding dead drain ditches filled with water, and that leads to the conclusion that there is no possibility of a flood during operation.

During the observation period beginning in 2012, there have been no clear abnormalities in dynamics of ground waters and temperature regime, and there is no tendency towards overall decrease of increase of the level.

Maximum ground water levels in the upper part of terminal moraine stratum section higher than the basic aquifier level were detected at depths of 12.0--19.85 m (absolute elevations 159.69-167.88 m).

Design and construction measures and means to prevent flooding at the NPP are specified in section 4.1.2 of the National Report.

An assessment of the level of flooding which may have a significant impact on normal operation of electric power systems or heat transfer to ultimate heat sink is provided in section 4.2 of the National Report.

The Belarusian NPP site is not subject to flooding, as the design elevation of the site is 179.4 m BES, that is, 51.5 m higher than the water elevation level at 0.01% confidence; ground waters and heavy precipitation has no affect on NPP safety [31].

Section 4.2 of the National Report shows that NPP safety is guaranteed during a flood up to 0.0 relative elevation, i.e. when the water level rises by 51.5 m.

4.1.2 Measures and Means to Protect NPP during Design-Basis Flooding

To prevent flooding of basic buildings and structures, the design provides for the following systems:

stratum drainage,

catchwater ditch,

storm-water drain.

Stratum drainage intercepts ground waters flowing from the lower soil column.

Catchwater ditch is meant to prevent flooding of the site during operation of the NPP by floods and rainfall to ensure normal operation of safety category I-III structures under "Norms of Structural Design of Nuclear Power Plants with Reactors of Different Types" (PiN AE-5.6).

Catchwater ditches from the western and north-western sides of the industrial site of the Belarusian NPP, in accordance with SP 58.13330.2012 "Hydraulic Structures. Basic statements. Updated edition of SNiP 33-01-2003", are designed for flood and rainfall flows of 0.1% confidence and are checked to pass water flows of 0.01% confidence.

The design solutions for ditches (their section, minimum 3‰ longitudinal slope, strengthening of the banks and bed of the catchwater ditches by 20-40 mm grain size crushed stone of 0.20 m thickness laid on a geotextile base) ensure passing of the estimated water flow while eliminating the possibility of ditch silting and clogging during operation that may result in failure of this water drainage system.

To collect and drain storm water from the territory surrounding the industrial site of the Belarusian NPP there is a GU storm-water drain system.

External gravity drainage networks of GU system ensures intake of storm water from roofs of the buildings and territory of the site, as well as production flows close by their composition, with their further drain into the UGU pump station located near general buildings and structures; around the first power unit and second power unit.

During normal NPP operation, pump stations pump the aforementioned flows to the UGV treatment facilities for production and storm water flows and oil-containing flows. Treatment facilities continuously take in and treat water flows, then are pumped back into the in-plant water recycling system.

Productivity of the UGV treatment facilities is selected taking into consideration the sediment pond (accumulator)which is meant to receive storm flows at a maximum daily precipitation of 24 mm (with 86% confidence).

Treatment facilities with a pond for temporary drain storage are accepted with 10 mm sediment layer to be cleaned in accordance with SP 32.13330.2012. When the sediment layer exceeds 10 mm, this drain will be collected in the storm water drain system and gradually pumped into the WWTP, with the clean treated water directed into the blowing pipes of the cooling towers via by-pass line for further discharge into the Viliya.

Calculations for the storm water drain system are made in accordance with SNiP 2.04.03-85 with consideration of a single exceedence of the estimated storm intensity P=1.00.

Pump stations 01UGU, 02UGU and 03UGU are considered to be of reliability category I under SNiP 2.04.03-85 as stations that do not allow waste water feed to stop or decrease.

UGU pump stations have underground steel reservoirs a form of 0.2 m diameter and up to 9.0 m depth and above ground control cabinets that operate in automatic mode under level sensors data. The data are displayed in the local control room of the pump station. Deviation of the most important parameters within design limits is communicated and rectified by means of emergency communication in the central control room. Control cabinets are powered from the normal electric power system.

Drainage pump station 01UGU is accepted with productivity of 280 m^3/h; two immersion pumps (operating and back-up) are installed.

Drainage pump station 02UGU is accepted with productivity of 190 m^3/h; two immersion pumps (operating and back-up) are installed.

Drainage pump station 03UGU is accepted with productivity of 180 $^m3/h$; two immersion pumps (operating and back-up) are installed.

Each pump station has a separate discharge line connected to the 00UGV treatment facilities.

To eliminate flooding from external premises, sumps with pump installations are inside the buildings at marks below ground level designed to collect and drain production waste water. Waste water from the sumps is pumped into the external gravity drainage networks. Pump installations operate automatically under the level sensors data. The data are displayed in the MCR.

Deviation of the most important parameters within design limits is communicated and rectified by means of emergency communication in the MCR. Control units are powered from the normal electric power system.

At the outlet of sewage of the open access zone and controlled access zone (of GQA, GQD systems) from buildings 10UKC and 20UKC where edges of sanitary fixtures are located lower than the nearest manhole cover, there are latches with an electric drive installed. In water treatment buildings, backup diesel electric plant and steam camera where sanitary fixtures are located at -4.000 to.-8.000 m, sewage is drained into the external network by sololift and liftaway.

The storm water treatment system and drainage systems at the industrial site of the Belarusian NPP are designed for normal operation conditions.

In case of electric power failure, the storm water treatment system and drainage systems will not operate.

The maximum daily volume of storm water calculated under paragraph 7.2.2 of SP 32.13.330.2012 is 61804 m^3.

In case of electric power failure and the drainage pump station and treatment facilities become inoperable, part of this volume – 6908 m^3 will be stored in pumps and wells of the drainage systems. The rest of the volume, which is 54896 m^3, will be distributed over the NPP area, with the sediment layer to be 5.3 mm. Taking into consideration the relief on the industrial site of the NPP, this waste water will partially soak up into the soil and partially accumulate around storm water tanks by the roads. Also, as the perimeter pavement around the building is

150 mm high and the buildings have waterproof walls in their underground sections, such a flood will not affect the equipment stored in the buildings.

4.1.3 NPP Compliance with License Requirements

The NPP complies with the regulatory requirements regarding protection from floods.

Compliance of the NPP with the license requirements means obligatory compliance with the requirements in the field of nuclear power use, as well as with the operating documents and procedures of the operating organization are developed for this reason.

To confirm the compliance of the Belarusian NPP with the license requirements, inspections are conducted by:

Gosatomnadzor ensures compliance with the legislative requirements in the field of nuclear and radiation safety,

other republican government bodies ensure compliance with requirements in the field of construction, industrial, sanitary and fire safety,

the general contractor ensure compliance with the requirements of technical documents and the design,

the operating organization ensures compliance with the requirements of legislation, quality assurance programs, as well as design, technical and operation documents.

Activities of the staff that are meant to keep buildings and structures in operating mode are stipulated in the operation documents. The operation documents of the operating organization specify, inter alia, regular technical maintenance, repairs and tests of equipment, as well as monitoring building structures. Technical maintenance and repair programs cover, among other things, mobile equipment and consumable materials to be used in emergency operations to ensure their availability and in ready-for-use condition.

Under the conditions of determining the Belarusian NPP site, no design basis external flood is provided for in the design. With that considered, no special actions of the staff to ensure operability of systems, structures and components of the NPP to reach and maintain the conditions for safe RU shutdown are required.

Inspections initiated by the licensee following the accident at the Fukushima NPP include stress-tests of the Belarusian NPP and initiation of the IAEA SEED mission for siting the Belarusian NPP.

4.2 Assessment of Safety Margins

4.2.1 Assessment of Flood Safety Margin

To initiate an event, i.e. flooding of the NPP site to the relative elevation of 0.00 (absolute elevation 179.3), water level would need to rise from its absolute elevation of 127.8 (Viliya river) by 51.5 m.

Borders of the surrounding areas that could possibly be flooded with maximum water flows during spring floods of 0.01% confidence show that there is no possibility of access routes of to the NPP and main roads would be flooded, eliminating hampered or delayed access of the staff and equipment delivery to the NPP site.

Condition of the roads inside and outside the NPP site during floods is described in para. 5.2.1.

In case of extreme precipitation, even if considering failure of the UGU pump stations, the level of water on site can only rise5.3 mm, which due to 150 mm perimeter pavement around the buildings which eliminates the possibility of a design basis flood.

Flooding is not the principle cause of an UGU pump station failure, therefore ensuring, operability of the flood protection systems. In case of several successive failures (including power failure which may result in possible failure of the systems required for flood protection), there are systems required to transfer the plant into controlled mode that keep operating.

A pump station is part of the turbine equipment water refill system, so it does not affect the NPP safety. The NPP is designed to operate with an in-plant water recirculation system, so that the operating organization has enough time to shut the NPP down and transfer it into the safe mode.

Figure 4.2.1 shows the schematic layout of the Belarusian NPP relative to an external flood source.

To assess safety margins though there is no design basis flood threat foreseen in the design, this section conservatively applies a deterministic approach and considers the flooding of all NPP buildings located below 0.00 level. This flood affects safety systems critical for heat transfer from RU and spent nuclear fuel. A conservative analysis [31] of the flood regime with affected SS and SCS systems and elements located below 0.00 level has shown that flooding results in the loss of the following major functions critical for NPP safety:

- transfer of heat from spent nuclear fuel (FAK and JMN systems are not operating),
- transfer of heat from the primary circuit (JNG, JNA, KAA, KAB systems are not operating),
- coolant inventory maintenance (JND system is not operating),
- the primary circuit feed (JND system is not operating).

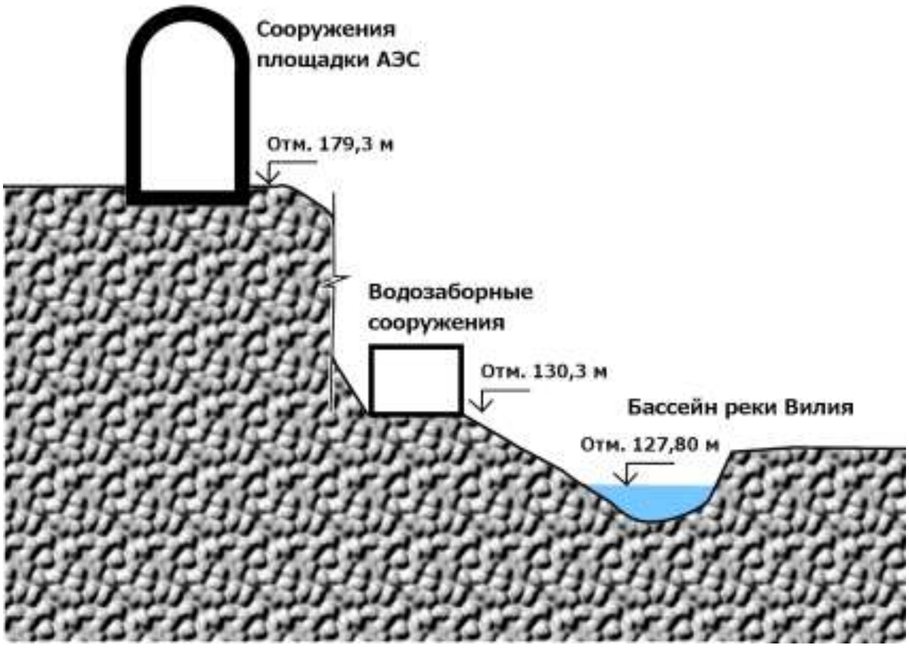

Figure 4.2.1. Schematic layout of the Belarusian NPP relative to the external flood source.

Сооружения площадки АЭС	NPP site structures
Отм.	Elevation
Водозаборные сооружения	Water intake installations
Бассейн реки Вилия	Viliya river basin

During a flood, the following basic functions critical for NPP safety will be preserved:

- reactivity control (EPS, preventive protection, systems JDH are operable),
- coolant inventory assurance (reactor systems, reactor coolant circuit, pressurizer are operable),
- primary circuit makeup (system JDH operable).
- heat removal from the secondary circuit (emergency feedwater, BRU-A, SG PHRS systems operate),
- control assurance (SS I&C and reliable power supply cabinets operable),
- provision of electric power supply (systems of normal and emergency power supply operate, except accumulator batteries and RP of the 7th power supply channel),
- life support (basic ventilation and air conditioning systems operable),
- protection of the primary and secondary circuits from overpressure (pressurizer POSV and SG POSV systems operable).

In case of a flood threat, RP is transferred into safe mode where it can be maintained for the following periods of time:

- in the SFP section: afterheat is transferred due to water boiling (in case of system FAK failure during the flood) for at least 41 hours,

- in the RP section: afterheat is transferred by BRU-A or SG PHRS for at least 72 hours.

To guarantee hot standby for 72 hours, 2201 m^3 of chemically demineralized water are required.

With total effective tank volume for ensuring hot standby for 72 hours considered, minimum supply level of 50% shall be maintained in LCU02 and 03BB001 tanks.

The time calculated (72 hours for RU and 41 hours for SFP) makes it possible to fully implement the proposed measures.

Thus, for the considered initiating event during 41 hours for SFP and 72 hours for RP, no extra technical and organizational measures are required to maintain safety of the NPP.

The analysis of internal floods has shown that equipment failure due to flooding will not result in an accident with nuclear fuel damage.

Thus, the analysis leads to the conclusion that to prevent consequences of floods at the Belarusian NPP site, no extra measures are required to improve safety of the unit.

4.2.2 Potential Measures to Improve NPP Resistance to Flooding

Design basis flood with water rising to site level is impossible. That is why the design does not specify any special measures to prevent floods.

With flooding of the NPP site postulated as an initiating event, to increase the period of time when the NPP is in the safe mode, it is necessary to implement the following measures to improve tolerance of the NPP to floods:

- in terms of heat transfer from RP, in 72 hours arrange feed of LCU tanks from the site (e.g. URR spray cooling pond, URX backup tank) and off-site (e.g. fire department tanks) water sources.

- in terms of afterheat transfer from the cooling pool:

in 41 hours (following the results of calculations in section 5.1.2) arrange feed of SFP. This can be made by connecting in an unconventional means (fire engine with 40 l/s pump and 100 m head) to two process connectors of JNB50 system located at the external side of the UJE building (at +0.690 and 0.730, with water pumped from LCU tanks via pump of the fire engine to JNB50 system piping and further to the coolant pool) with flanges and plugs installed;

the process scheme of the JNB50 system can be altered by cutting a bypass of the back-pressure valve at the EHRT feed line. This will ensure feed of the cooling pool from EHRT performed by operating staff after 41 hours have passed.

For unconventional technical means used to feed EHRT and SFP, there are the following limitations:

- minimum flow 10 l/s,
- minimum head 90 m,
- maximum head 200 m.

5. EXTREME WEATHER CONDITIONS

5.1 Design Basis

5.1.1 Dangerous Meteorological Phenomena

The following dangerous meteorological phenomena were analyzed as part of the stress tests:

strong winds (instantaneous speed> 25 m/s);
squalls (short-term wind speed increase up to 21 – 35 m/s);
tornados;
heavy rain (precipitation> 50 mm within 12 hours or less);
large hail (diameter> 20 mm);
dust storms;
strong snowstorms (with a wind speed of 15 m/s);
heavy snowfalls (precipitation> 20 mm within 12 hours or less);
heavy fogs (visibility – less than 100 m);
thick ice coating and hard rime (diameter> 20 mm);

drought;

extreme temperatures: according to the Lyntupy weather station, maximum air temperature of the warmest five-day period from August 13 to August 17, 2010 was 33.2 ºC.

Additionally, various combinations of these weather conditions were considered during the stress tests.

Strong winds. For the period analyzed, from 1961 to 2000, strong winds in Vitebsk and Grodno regions were observed in 25 out of 33–35 years at least in one of the points in the region. The frequency of strong winds at least in one of the points of the region was 69% and 71%, respectively. In the Minsk region, such winds were noted in 20 years at least in one of the points of the region. The frequency of strong winds at least in one point was 63%.

Squalls are recorded on average every 5 years, mainly during a warm summer. The wind speed exceeds 10 m/s, but generally reaches 16-20 m/s. The greatest danger by squalls is when wind speeds exceed 25 m/s, leading to the destruction of structures, communication and electric transmission lines. Due to the infrequency of squalls, the available data is incomplete. Destructive squalls are rare: in January 1993, the maximum wind gust was recorded in Lyntupy (25 m/s), in April 1967 – in Ashmyany (36 m/s).

In general, throughout the country there are approximately four days with destructive squalls during the warm period, which effect individual farms of 5 – 10 administrative districts (Climate of Belarus, Minsk, 1996).

Tornadoes. The NPP is located in the sub-zone A–L. The values for the impact of tornadoes for the Belarusian NPP site are shown in Table 5.2.1.1.

Through 1961 – 2016, waterspouts have not been registered in the territory of the Republic of Belarus nor in the vicinity of the Belarusian Nuclear Power Plant.The sub-zone A-L mostly extends in the territory of Belarus and Lithuania and covers minor parts of Latvia, Ukraine and Russia. In 1988 – 2016, Scientific Production Association “Gidrotekhproyekt”, LLC collected data on tornadoes in the subzone A-L. Data source – Belhydromet (archive data).

Heavy rain. Areas with heavy rainfall are generally small. It spreads over large areas only in a few cases. In the Minsk and Grodno regions, heavy rain was registered at least in one area in 16 out of 34-35 years however in the Vitebsk region heavy rain was registered in 26 out of 34-35 years. Heavy rain frequency at least in one of the points in Vitebsk region is 74%, in Minsk region – 47% and in Grodno region – 46%.

Large hail. This is a rare phenomenon noted in separate points of the country every 40-50 years on average. In Vitebsk region, large hail with a diameter exceeding 20 mm was observed in 23 years, in Minsk region – in 12 and in Grodno region – in 11 out of 35 years at least in one of the regional districts. Maximum hail dimensions reach up to 8-10 cm, the size of the goose egg. Such hail was observed in particular on July 11, 1953 in the Braslav district of the Vitebsk region.

Dust storms. In the region under consideration, there was no such dangerous phenomenon as dust storms.

Strong snowstorms. Strong snowstorms occurs when the prevailing wind lasts 12 hours or more with a wind speed in excess of 15 m/s. Snowstorms are characterized by snow drifts on roads, as well as poor visibility.

Such snowstorms are rare in this region. In the Minsk and Grodno regions at least in one of the districts of the region they were noted in 6 years, in the Vitebsk region, in 7 out of 24 years.

Heavy snowfall. In the territory of Belarus, heavy snowfall can be observed from November to March, but it most often occurs in January and February. The frequency of heavy snowfall on the territory of the Minsk region is 6%, that is, they may happen once in 17 years. In Grodno region dangerous snowfalls occur once in 9 years (frequency – 11%), in Vitebsk region – once in 6 years (frequency – 17%).

(visibility – 50 m or less for 6 hours or more). Classification of fog by their origin for the NPP is of no fundamental importance. Independent of the fog origin, its presence does not contribute to dispersal of impurities in the air.

In some areas, dangerous fog are extremely rare. At least in one of the districts of Vitebsk and Minsk regions their frequency is 33%, meaning that at any point in the region it can be observed once in 3 years. The frequency of such fogs in Grodno region is 25% (once in 4 years).

Thick ice coating and hard rime ($D_{ice\ coating} \geq 20$mm, $D_{hard\ rime} \geq 35$mm). Over the past 35 years, dangerous ice coating and hard rime on the territory of the Minsk region were recorded in 2 years, in the Vitebsk region3 years and in the Grodno region 8 years. Their frequency is 6.9 and 23% respectively.

Drought. The plant is designed to work with a circulating water supply system, so that a prolonged drought does not lead to threshold phenomena. As the makeup water source is shallowing, the operating organization has an opportunity to keep the NPP in safe mode (see section 6.2 of this report).

Combinations of loads and impacts for buildings and structures are adopted in accordance with PiN AE-5.6, SP 20.13330.2011 “Loads and Impacts”. A detailed analysis is carried out as described in PSA-1.

Calculation of structures is performed taking into account unfavorable combinations of loads or their corresponding effects. These combinations are established from the analysis of real types of various simultaneous loads at the relevant stage of construction.

Depending on the composition of loads, the following are considered:

a) basic load combinations, consisting of permanent, long-term and short-term loads;

b) special load combinations, consisting of permanent, long-term, short-term and one of the special loads.

5.1.2 Selective Analysis of Possible Combinations of Initial External Effects

Combinations of various external effects may have a more pronounced impact on the safety of the plant than each effect considered separately. The frequency of combinations of various external effects can be compared with the frequencies of individual effects. The process of identifying external effects will include identification of all their possible combinations that might be relevant to the risk.

The basic principles for analyzing external effects are based on general recommendations specified in the IAEA’s SSG-3 manual:

determining possible combinations of external effects is based on the list of individual (single) external effects;

useing the entire list of potential external effects to identify combinations of external events prior to any screening analysis of individual events;

as a rule, combined effects are natural (for instance, a combination of strong wind and high sea level). However, combinations of natural effects and risks caused by human activities are also possible, and cannot be excluded *a priori* (for instance, an increased risk of ship accidents in severe weather conditions);

effects taking place under the same conditions and at the same time (for instance, strong winds and snow/precipitation) should be considered for possible combinations;

single external effects causing other hazards (for instance, seismically induced external flooding followed by dam destruction) needs to be considered for possible combinations;

the impact of combinations of different external effects on safety functions shall be reviewed, as they may influence different safety functions (or one and the same function) more profoundly than a single effect.

There are no possible sources of external fire and smoke in the two-kilometer zone of the NPP industrial site.

Table 5.1.2.1 presents an analysis of possible combinations of external effects. A detailed analysis will be given in the scope of PSA-1.

Information on the development of a full-scale PSA-1 (including PSA-1 for internal IEs, fire and flooding PSA, seismic PSA, PSA of external effects) and PSA-2 are given in section 2.4 of the national report.

Based on the analysis of the consequences, it may be concluded that in the event of various combinations of extreme natural effects, the worst possible consequences are those leading to the simultaneous power loss and final heat absorber failure. Under this scenario, turning on a safe mode of the NPP and its maintenance are provided by the operation of passive safety systems (RCCS, PHRS SG, containment PHRS). Any combinations of events specified in Table 5.1.2.1 do not affect the performance of these systems. This scenario is described in more detail in the section of the national report "Loss of Heat Dissipation from the Primary Coolant Circuit in Combination with Complete NPP Deenergization".

Table 5.1.2.1 – Analysis of Combinations of External Effects

Event 1	Event 2	Event 3	Analysis of the Impact on Safety
Lightning (by the power lines)	Tornado	-	It is assumed that these events have the following consequences: - partial loss of high-voltage power lines, external deenergization, loss of power supply for own needs; The NPP cooldown is carried out in DG EPS safety systems.
Lightning (in the area of power lines)	Wind > 54 m/s	-	It is assumed that these events have the following consequences: - partial loss of high-voltage power lines, external deenergization, loss of power supply for own needs; The NPP cooldown is carried out in DG EPS safety systems.
Wind > 54 m/s	Low air temperature (less than -41.5°C)	-	It is assumed that these events have the following consequences: - partial loss of high-voltage power lines, external deenergization, loss of power supply for own needs; - failures of spray ponds due to freezing. The NPP safe mode is maintained due to SG PHRS and containment PHRS operation.
Wind > 54 m/s	Rain > 101 mm/24 hours	-	It is assumed that these events have the following consequences: - partial loss of high-voltage power lines, external deenergization, loss of power supply for own needs; - failure of drainage systems due to water ingress into buildings of nuclear power plants or ventilation ducts. The NPP safe mode is maintained due to SG PHRS and containment PHRS operation.
Wind > 54 m/s	Snowfall with the snow cover thickness > 72 cm	-	It is assumed that these events have the following consequences: - partial loss of high-voltage power lines, external deenergization, loss of power supply for own needs; - snow clogging of the plant air intakes (impact on the DG EPSS is considered). The NPP safe mode is maintained due to SG PHRS and containment PHRS operation.
Wind >	Sludge ice	Low air	It is assumed that these events have the following

54 m/s	in the area of water intake	temperature (less than -41.5°C)	consequences: - partial loss of high-voltage power lines, external deenergization, loss of power supply for own needs; - failure of the normal heat dissipation system through circuit 2; - failures of spray ponds due to freezing. The NPP safe mode is maintained due to SG PHRS and containment PHRS operation.
Wind > 54 m/s	Snowfall with the snow cover thickness > 72 cm	Sludge ice in the area of water intake	It is assumed that these events have the following consequences: - partial loss of high-voltage power lines, external deenergization, loss of power supply for own needs; - failure of the normal heat dissipation system through circuit 2; - snow clogging of the plant air intakes (impact on the DG EPSS is considered). The NPP safe mode is maintained due to SG PHRS and containment PHRS operation.

5.1.3 NPP Resistance to Extreme Weather Conditions

The analysis of natural impact taking into account in the project of the Belarusian NPP shows that the plant is resistant to extreme weather conditions.

Analysis The analysis of combinations of extreme weather conditions shows that there is an option to turn on and maintain the NPP safe mode regardless of any possible hazards in any combination.

Stress tests did not reveal additional external extreme effects and their combinations (not included in the Report on Safety Case of the Belarusian NPP Units 1,2).

5.2 Assessment of Safety Margins

5.2.1 Assessment of Safety Margins in Extreme Weather Conditions

Extreme natural impacts on building structures and facilities constitute special load combinations. In accordance with the current regulations, requirements for building structures are imposed exclusively as of the first limit state (strength, stability). Therefore, after heavy rains, snowfall, hurricane winds, earthquakes and other natural phenomena, unscheduled general or partial technical inspections of buildings and electrical power units will be conducted in accordance with the regulatory requirements and provisions of the operating organization internal documents.

Objectives of technical inspection:

- gathering information about the environment in the working area of industrial buildings and structures;
- timely detection of structural defects and measures to eliminate them.

During scheduled and unscheduled general technical inspections, the commission should perform a full inspection of buildings and structures, including various types of finishes and protective coatings.

For all observations made during inspections of buildings and facilities (deformations, damage, violations of maintenance and operating instructions), employees of the technical inspection department make relevant notes in inspection record books.

The survey should be carried out according to the developed and approved operating organization program, using visual and instrumental methods.

If necessary, samples will be taken for mechanical testing of building structure materials. A non-destructive method may replace it.

The results of the survey are presented in the form of an act, opinion or technical report about the condition of buildings and facilities and the technical possibility of their further operation with recommendations and technical solutions for restoration of the structures with defects.

For the Belarusian NPP project, the seismic impact and that of the aircraft, whose intensity is several times higher than the extreme impacts of a hurricane, a tornado, snow load, etc. are determining in terms of structural strength (security category I, according to PiNAE-5.6). Therefore, there are considerable safety margins with regard to site-characteristic extreme effects.

The plan of protective measures in case of a radiation accident at the Republican Unitary Enterprise “Belarusian Nuclear Power Plant” (external emergency plan) will involve the road service for road maintenance planning and organization, as well as for repair and restoration works on routes in case of damage.

There is a developed transportation system in the area of the Belarusian NPP, providing for a considerable number of access roads to the NPP site. The existing network of highways ensures access to nuclear power plant from three different directions:

from Polotsk – along the P-45 highway;

from Ashmyany – along the P-52 highway “Goza (from P-45) – NPP – Ostrovets – Ashmyany”;

from Vilnius (Republic of Lithuania) along the P-45 highway.

Additionally, there is a railway track aimed at transportation of goods and people from Ashmyany railway station to the NPP site. Railway access is provided from the production base. Also, to ensure transport operations for the shipment of spent fuel and the supply of fresh fuel in the eastern part of the industrial site, the railway station “NPP” consists of two tracks may be used for personnel access, material and technical support.

From the industrial site, two exits to public roads are planned for personnel access, material and technical support.

The network of on-site roads provides access to buildings and facilities of the industrial site. On-site roads are looped to organize unimpeded and free movement of vehicles and personnel.

According to the plan “Measures to Protect Personnel in the Event of an Accident at the Belarusian NPP”, of the general set of extreme phenomena, processes and events, both natural and man-made, the safety of access roads may be affected by: MDBE, tornadoes and an air shock wave (ASW). In case of seismic impacts (MDBE intensity – 7 points), only partial damage to transport communications is expected with preservation of their functions. With seismic forces above MDBE level, there is a possibility of damage to the engineering structures at points of access. At the same time, depending on the degree of damage to the personnel access transportation routes, material and technical support, their restoration by the above-mentioned road services is planned. If it is not possible to restore engineering structures, alternative options of personnel access and maintenance will be provided.

In case of extreme effects of natural origin (tornadoes) – both at the design level and above it – no damage of the engineering structures in points of access and on the tracks is envisaged. At the same time, in these conditions, roadblocks are anticipated resulting from destruction of nearby structures, fallen trees, etc. In this case, engineering measures will be taken to clear the transportation lines.

Other processes and events equal to or exceeding the design level will not lead to damage of the tracks.

Influence on System Design

In the project of the Belarusian NPP, systems and elements are required to operate with various load combinations, depending on the need for these systems and elements to perform safety functions, as well as on their role in ensuring safety in the event of external effects.

Taking into account different natural and man-made effects on systems and elements located on the NPP premises, building structures are characterized by various loads. These loads have been compared, their maximum values – determined and recorded in the Belarusian NPP documentation.

For instance, in piping systems, to determine maximum loads on system elements, response spectra values have been compared for various effects (MDBE, aircraft impact, shock wave, etc.). As a result, maximum load values (or envelope response spectrum) have been determined followed by the introduction of the concept of “External Dynamic Effects”. This concept is successfully used when the “Initial Technical Requirements” are reviewed, including the NPP pipelines and equipment, developed as part of the Technical Project of the Belarusian Nuclear Power Plant. Examples of comparisons for MDBE, explosion shock wave and aircraft impact for UKD buildings (el. + 23.400) and UJA buildings (el. + 25.400) are shown in Figures 5.2.1.1 and 5.2.1.2 respectively.

The loads on equipment from the pipeline systems as a result of EDE were set in accordance with NP-068-05 for the maximum bearing capacity of the pipelines. Thus, the loads exceeding the maximum load-carrying capacity of the pipelines were not considered due to the impossibility of the pipeline system functioning as a whole. Within this approach, provided the condition of the pipeline system strength is met, the condition for the strength of the equipment branches is automatically ensured.

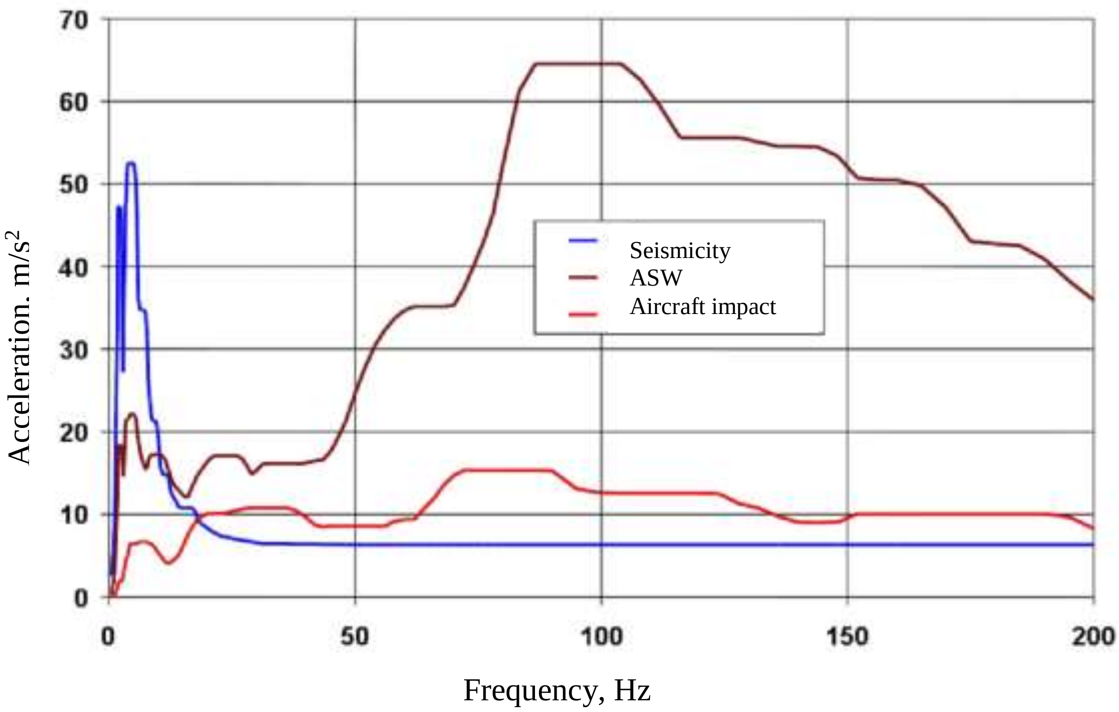

Figure 5.2.1.1 Response spectra for the UKD building, el. +23 400 (example)

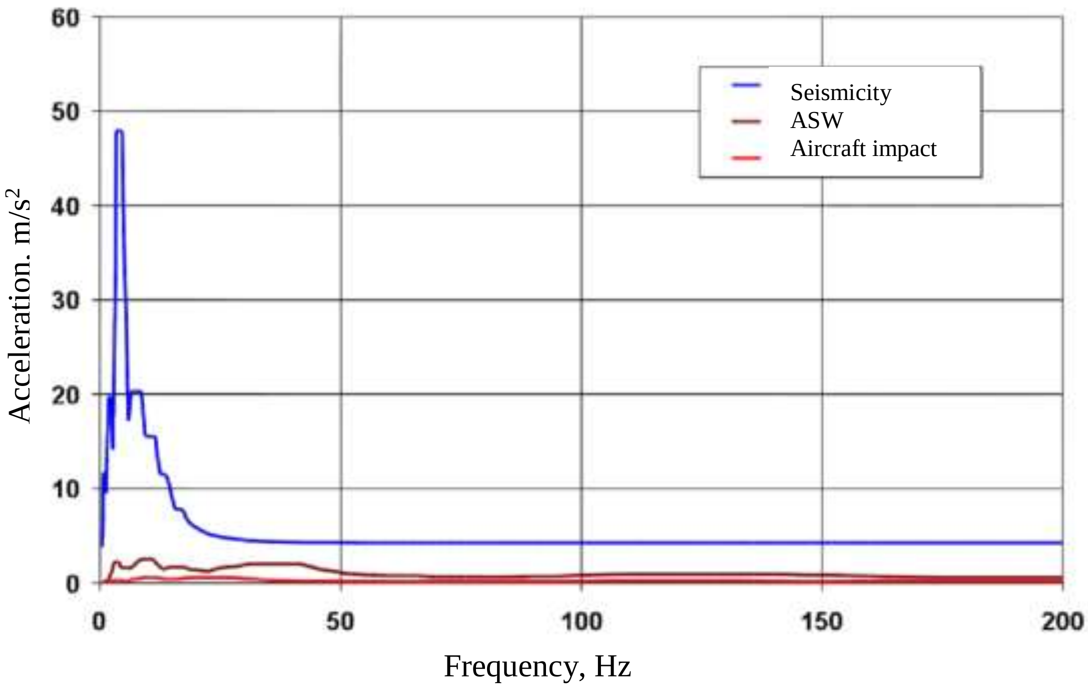

Figure 5.2.1.2 Response spectra for the UJA building, el. +23 400 (example)

The following requirements were formulated and reflected in the BS after determining maximum values of loads on equipment from pipeline systems:

1. Equipment in the category of seismic resistance I and II remaining operative in the following conditions:

- normal operation (NO);
- abnormal operation (ANO);
- normal operation with seismic forces up to DBE and including it (NO + DBE);
- abnormal operation with seismic forces up to DBE and including it (ANO + DBE).

2. Equipment with the seismic resistance of category I shall perform safety-related functions in the following conditions:

- design basis accident (DBA);
- normal operation combined with external dynamic effects (NO + EDE);
- abnormal operation combined with external dynamic effects (ANO+EDE);
- normal operation combined with design basis accident and seismic forces up to DBE and including it (NO + DBA + DBE).

The design of ventilation systems has a 20% reserve with regard to standard values for minimum and maximum ambient temperatures. In this case, if these values are exceeded, the ventilation systems can be switched to recirculation mode, thus minimizing the influence of external temperatures.

For containment PHRS and SG PHRS, the effect of extremely low outdoor air temperatures on the system performance was analyzed. The performed calculation showed the functioning of containment and SG passive heat removal systems at extremely low temperatures (up to – 61°C for more than 20 days). The estimated air temperature in the EHRT and pipeline facilities remained positive during the entire base period. Water temperature in EHRT decreased to a minimum of 0.9 °C.

Functionality of the especially affected PE system (spray ponds), including during waterspouts, is described in section 6.2 hereof.

Supply and discharge pipes of the cooling water system of the PE essential consumers are placed in underground passageway tunnels of the UQZ and URZ safety systems, which excludes their freezing.

Analysis of the possibility of flooding in extreme rainfall is described in section 4.1.2 hereof.

Thus, systems and elements, necessary and sufficient to ensure safety, are in an efficient state in the entire range of loads from the external influences under consideration. Operability of the remaining systems and components is guaranteed at loads depending on the category of seismic resistance.

Threshold Analysis

Comparison of design loads with loads specific to the site

The maximum values of extreme climatic conditions determined for the site are much lower than those used in the design. The values of natural loads are determined with a frequency period of 1 time per 10 000 years according to PiN AE-5.6. Loads and their numerical values included in the project are discussed in detail in the SAR.

A comparison of extreme natural impacts used in the design with the effects identified for the Belarusian NPP site is presented in Table 5.2.1.1 below.

Table 5.2.1.1 Values of extreme effects for the baseline design and for the Belarusian NPP site.

Recurrent extreme effects	Value used in the Belarusian NPP design	Values of extreme natural impacts with a frequency of 1 time per 10 000 years, typical for the Belarusian NPP site
Minimum temperature	– 61°C	-50 °C
Maximum temperature	+52°C	+37.4 °C
Extreme snow load	4.3 kPa	3 kPa
Extreme wind speed	61 m/s	54 m/s
Tornado	Class F3,6	Class F2,5
- maximum wind speed in the vortex	V_m=95 m/s	V_m=70 m/s
- maximum subatmospheric pressure in the tornado eye	ΔP_{max} = 11.1 kPa	ΔP_{max} = 5.55 kPa
- maximum wind pressure	P_{max} = 8.7 kPa	P_{max} = 3.2 kPa
- flying objects	Considered	No flying objects

Threshold value

All facilities and structures (category of radiation and nuclear safety – I) are designed in accordance with loads from ASW and aircraft impact. The loads and their numerical values incorporated in the project are discussed in detail in the SAR and in the engineering analysis [31].

In the event of ASW, reduced loads on building structures constitute 87 kPa – on the face area and 36 kPa – on the roof.

Reduced load from the aircraft impact in the impact spot is 1300 kPa.

If we compare values given in the table with the loads from the air shock wave and the aircraft impact, it is obvious that both snow load and wind pressure reserves are about 8 times bigger than the loads from ASW and more than 400 times bigger than the loads from aircraft impact.

Extreme temperature load from the point of view of the strength of the enclosing structures is not dangerous, since it is neither power nor self-compensating due to increased cracking followed by the reduction of the structures' rigidity.

Thus, threshold values for the above-mentioned natural loads are not achievable.

Natural impacts in the form of precipitation are considered in the design of the following ventilation systems:

<u>In the reactor building</u>

- Emergency cooling system between the shells in the reactor building;
- PHRS tank cooling system: channels 1,2,3,4;
- System for localization of leaks in the containment building and security building;

In the safety building

- Recirculation cooling system for technological rooms: channels 1,2,3,4 in the security building;
- Electric heating system of the rooms located within the controlled access zone of the security building;

In the administrative building

- Supply and exhaust ventilation system in the rooms: channels 1,2,3,4 in the administrative building and a corresponding cooling system;
- An air-conditioning system in MCR complex and a corresponding cooling system;
- An air-conditioning system of ECR and a corresponding cooling system;
- Personnel life support system in MCR;
- Personnel life support system in ECR;

In SDPP building

- Supply and exhaust ventilation system in the rooms: channels 1,2,3,4 in the SDPP building and a corresponding cold supply system;
- Supply and exhaust ventilation system in the rooms: diesel channels 1,2,3,4 in the SDPP building;

In the pumphouse of essential consumers

- Recirculation cooling system, supply and exhaust ventilation systems in the operating room: channels 1,2,3,4 in the pumphouse of essential consumers;
- Electric heating system in the pumphouse of essential consumers;

In the switching chamber

- Exhaust ventilation system in the switching chambers;
- Electric heating system in the switching chambers;

In the steam cell

- Supply and exhaust ventilation system in the rooms: channels 1,2,3,4 in the steam cell building and a corresponding cold supply system.

To protect air intake systems from precipitation, immovable louvered grills are installed in the outer walls.

No threshold effects arise for ventilation systems, taking into account the temperature margin and the possibility of recirculation mode.

Information on margins of safety associated with extreme precipitation is provided in section 4.1.2 of the national report.

5.2.2 Possible Measures to Improve the NPP Resistance to Extreme Weather Conditions

Extreme weather conditions do not affect the safety of nuclear power plants and are overlapped by other effects with a significant margin.

Information on safety and efficiency of structures intended for electrical equipment installation and their protection from possible natural and man-made effects in the NPP area is provided in 6.1.1.

Please note that it is impossible to predict the reliable supply of diesel fuel at a late stage of an accident through pipelines laid from the NPP oil fuel storage to the Block Diesel Power Plant (diesel fuel freezing and damage due to extreme wind force). Diesel fuel for refilling shall be delivered in tanks. Means to preserve the functions of access roads in the extreme weather conditions is described in paragraph 5.2.1.

Information on additional personnel protection in extreme conditions was taken into account in the plan "Measures to Protect Personnel in the Event of an Accident at the Belarusian NPP", BLR1.E.534.&.&&&&&&.&&&&&.000.YN.0001.

6. POWER LOSS AND FINAL HEAT ABSORBER FAILURE

6.1 Power Loss

6.1.1 Loss of External Power Supply

For the Belarusian NPP, loss of external power supply is a design basis condition analyzed in the SAR on the Belorussian NPP. The design provides for the following backup AC power supplies for each NPP Unit (constantly available for use:

- Emergency backup transformer with a power of 16 MVA, seismic category I, voltage 110/10 kV, powered from the "Vilia" substation through a cable line laid in the ground. The power of this transformer was selected so as to supply power to one EPSS (emergency power supply system) channel of each Unit (feeders from 110/10 kV substation are provided for all 10 kV sections of the Unit reliable power supply system. If operating personnel decide to use an additional 110/10 kV power source to supply power to essential loads of the unit, the circuit is assembled manually. The 10 kV section (including EPSS) was selected in accordance with the NPP emergency response manual;
- Unit DG with a power of 6300 kW;
- 4 EPSS DGs with a power of 6300 kW each;

Calculation and selection of the equipment take into account both the external climatic conditions of the site and the external conditions associated with human activities. The structures of buildings and facilities are designed taking into account natural and man-caused impacts such as seismic activity, tornado, snow, ice and wind loads, outside air temperature, humidity, air shock wave, aircraft crash. Based on the calculated data, appropriate design solutions are applied for the facilities and equipment. The equipment reliability analysis shows that the above-mentioned natural impacts do not lead to accidents.

The equipment of the main power output system allows for cutting off the electrical equipment of Unit 1 and Unit 2 with 330 kV circuit breakers of the units, therefore the power supply system of one Unit is independent of the other Unit.

Design accident management measures are independent of the operation of the other Unit. These measures are sufficient and resistant to impacts caused by earthquakes and flooding.

Analysis of the accident with loss of external power supply for "power operation" and "cold" operating conditions of the Unit RP is provided in [31]. The analysis shows that stabilization of the parameters and residual heat removal prior to restoration of the normal auxiliary power supply of the Unit is ensured. The Unit can be in the safe mode for up to 7 days with active safety systems operating. Safety limits of the Unit are not violated in this mode.

Operation of the Unit with power supply from DG is guaranteed for 72 hours due to the following features:

- each DG has its own self-contained auxiliary systems;
- tanks with diesel fuel are provided for each DG to ensure a 53-hour operation of the EPSS DG at rated power (the storage tank capacity is 100 m^3, the supply tank capacity is 8 m^3), a 29-hour operation of the normal operation reliable power supply DG at rated power (the storage tank capacity is 50 m^3, the supply tank capacity is 8 m^3);
- alarms for levels in the supply tank and the storage tank are displayed on the control panel;
- oil system is designed for independent operation for at least 240 hours.

- An additional diesel fuel amount of 1160 m^3 is stored at site in the central diesel fuel warehouse (00UEJ) of (290 m^3 for DG of one EPSS channel of each Unit) to ensure additional stock for 7 days for DG of one EPSS channel of one Unit (this calculation is based on a flow rate of 204 g/kWh for one DG).

The fuel tanks are refilled from tank trucks and through pipelines from the diesel fuel storage warehouse to the supply tanks so that quantitative and qualitative fueling is ensured. In case of NPP blackout, if the NPP auxiliary power supply is not restored within two days (48 hours) with DG in standby mode, the main and intermediate warehouses are refilled with diesel fuel of the required quality delivered from the regional oil supply points by road.

6.1.2 Loss of External Power Supply and Standard Redundant Alternating Current Power Supplies

The initiating event for the considered beyond design basis accident is the failure of all AC power sources (NPP blackout), including EPSS DG, Unit diesel generator and emergency standby transformer powered from the "Vilia" substation.

The facilities designed for electrical equipment installation meet the requirements for ensuring their safety and operability in accordance with their classification and ensure protection against possible natural and man-caused impacts in the NPP area. The technical means are resistant to impacts caused by earthquakes and flooding.

Calculation and equipment selection of the equipment take into account both the external climatic conditions of the site and the external conditions associated with human activities. The structures of buildings and facilities are designed taking into account natural and man-caused impacts such as seismic activity, tornado, snow, ice and wind loads, outside air temperature, humidity, air shock wave, aircraft crash. Based on the calculated data, appropriate design solutions are applied for the facilities and equipment.

In case of loss of power supply to the normal operation systems and safety systems (failure of all DGs), functioning of the normal operation systems ensuring the residual heat removal to the ultimate heat sink and cooling of the spent fuel pool is stopped. At the same time failure of the active safety systems occurs.

The condition of the Unit at the initial stage of the accident is characterized by:

1. complete unavailability of AC power supply (external and internal);
2. availability of power supply from the EPSS UPS for some valves (isolating valves of the sealed enclosure, BRU-A, MSIV) and I&C. Power supply from the UPS is designed for 2 hours of operation without battery recharge;
3. subcritical state of the reactor;
4. primary circuit leaks – 2.15 m^3/h, which corresponds to the maximum possible leakage during operation at the rated power;
5. tight secondary circuit;
6. full reserve of cooling water for the BDBA management systems in four SG PHRS (passive heat removal system) tanks;
7. water level in the fuel pool – 8.7 m (level at fuel storage).

As a result of blackout, the RCPS (reactor coolant pump sets) are tripped, the turbogenerator stop valves are closed, feed water supply to the steam generators is stopped, the primary circuit makeup-blowdown is tripped, the pressurizer heating elements, injection to the pressurizer , BRU-K are out of operation. As a result of the DG startup failure (initiating event), ECCS and EFWEP pumps are out of operation.

The emergency protection is actuated upon the first signal: de-energization of more than two RCPS at power exceeding the rated value by more than 5%. CPS ARs start to move in 1.9 sec (1.4 sec of technological delay and 0.5 sec – time for the signal to pass through the electrical circuits).

Closing of the turbogenerator stop valves leads to a pressure rise in the secondary circuit and actuation of BRU-A on the steam generators; further, BRU-A operate in the SG pressure maintaining mode.

Upon de-energization of the SS sections and DG startup failure (with a delay of 30 sec), the SG 1-4 PHRS are activated and within 80 sec. the SG PHRS reaches the full design capacity.

Operation of four PHRS channels decreases pressure in the steam generators in accordance with the PHRS performance parameters. As a result, BRU-A on all the SG steam lines are closed and loss of boiler feed water in the steam generators is stopped.

Reaching of the design parameters by the PHRS and reduction of residual heat generation in the reactor core lead to pressure decrease in the primary circuit. When the pressure in the primary circuit decreases to 5.89 MPa, boric solution is supplied from the ECCS hydro accumulators to the reactor chambers. During operation of the hydro accumulators (4046 – 259850.0 sec), the supply from them is pulsating to ensure compensation of leakages and cooling of the primary circuit coolant (along with the PHRS).

Thus, the PHRS ensures the removal of residual heat within three days. Parameters of the plant are stabilized due to PHRS operation in the cooldown mode.

As a result of the emergency heat removal tank dehydration, after three days, PHRS operation stops, which leads to an increase of the parameters in the secondary circuit up the setpoints for opening of the pilot-operated safety valves (POSV) of all the SGs. Deterioration of heat removal by the secondary circuit (due to level decrease in the SGs) causes an increase in the primary circuit parameters up to the setpoints for actuation of the pressurizer control POSV.

Later, continuous leakages of the primary circuit coolant, operation of the pressurizer control POSV, lack of supply from the ECCS hydro accumulators, stopping of PHRS operation can lead to dehydration of the FA upper part, as well as fuel heating and melting. Time allowance substantiated by the thermohydraulic analysis prior to the start of heating is about 310 000 sec (86 hours).

Mass yield of the primary circuit coolant through leakages after 72 hours is approximately 41 tons. Mass of steam discharged through the steam dump devices of the second circuit during the considered time interval is about 210 tons.

Reactor plant

As a result of loss of the power supply to the normal operation systems, the RCPS are tripped, the turbine stop valves are closed, feed water supply to the steam generators is stopped, the primary circuit makeup-blowdown, the pressurizer heating elements, injection to the pressurizer , BRU-K are tripped. Upon de-energization of more than two RCPS the emergency protection is actuated.

As a result of DG startup failure, the safety systems, including EFWEP, are out of operation.

Closing of the turbine stop valves leads to a pressure rise in the secondary circuit resulting in BRU-A opening. Further, BRU-A operate in the mode of pressure maintaining in the secondary circuit.

Upon loss of the power supply to the normal operation systems and failure to start all the diesel generators, the SG PHRS SG is activated.

The level of boiler feed water in the steam generators after some reduction resulting from steam discharge through BRU-A is stabilized due to SG PHRS operation. The presence of boiler feed water in the SG in case of loss of external power supply and design backup AC power supplies is substantiated by the operating organization in [31].

When RCPS are tripped and their run-down is over, natural circulation of the primary circuit coolant is established.

Heat is removed to the ultimate heat sink – the environment – through the following chain: reactor – steam generator – SG PHRS – atmospheric air. Heat is removed to the atmosphere by evaporation of water from the SG PHRS tanks.

Stable cooldown of the reactor plant is performed with a pressure decrease in the primary and secondary circuits.

In 4046 seconds (1 hour 7 minutes) from the accident beginning, the pressure in the primary circuit decreases to 5.89 MPa. Supply of boric solution from the ECCS hydro accumulators to the reactor chambers is started.

During SG PHRS operation, even a short-term dehydration of the core does not occur and the FR (fuel rod) temperature does not exceed the design limit of 1200 °C. Pressurizer POSV and emergency gas removal systems remain closed.

Initial data for design analysis are structural parameters of the reactor plant components, thermophysical properties of materials, fuel and coolant, Unit parameters, as well as neutron and physical characteristics of the reactor core.

It is assumed in the analysis that the PHRS reaches its full power in 80 sec from the time of its activation. It is assumed in the analysis that the PHRS operation time determined by the time of water evaporation in the emergency heat removal tanks in this accident is three days.

The analysis presented by the operating organization takes into account both controlled and uncontrolled primary circuit leakages in the amount of 2.15 m^3/h, which corresponds to the maximum possible leakage rate during operation at the rated power.

It is assumed that in 24 hours from the start of the accident Unit becomes uncontrollable because the reliable power supply batteries are discharged.

Analysis of the reactor plant parameters and the temperature conditions of the core in the considered mode was performed using DYNAMICS-97 software [31].

Acceptance criteria were assessed by comparing the values of the parameters determined as a result of design analysis with the acceptance criteria adopted in the design for the considered initiating event.

If the PHRS operates in the cooldown mode for three days, maximum pressure of the primary and secondary circuits does not exceed 115% of the design pressure (16.20 and 8.74 MPa, respectively). When PHRS operation is completed, maximum pressure in the primary circuit is 18.12 MPa and maximum pressure in the secondary circuit is 8.80 MPa.

Thus, during the accident design period (about 3.5 days), the maximum pressure values of the primary and secondary circuits are not reached, the acceptance criterion is met, the fuel pellets do not melt even locally (the temperature is less than 2540 °C for spent fuel and less than 2840° C for fresh fuel).

Analysis of the blackout accident development in the course of three days demonstrates the following:

- maximum fuel temperature is reached at the start of the process and is equal to 1426 °C, which does not exceed the criterion value. Thus, the acceptance criterion is met;
- criteria for emergency core cooling are not violated. Maximum temperature of the FR cladding is 352.5 °C, which is less than 1200 °C. There are no conditions for FR oxidation. There are no conditions for melting of control rods and/or FA and FR deformation.

The analysis results show that in the considered time interval (72 hours) the requirement to reach the safe state of the core is fulfilled. There is no damage of the reactor core and internals. After activation of the emergency protection the reactor is in a subcritical state.

To prevent the accident development to a severe stage, it is required to take measures within not more than 3 days from the accident beginning to restore and maintain water reserve in the emergency heat removal tanks for ensuring the PHRS operation. In this case, the time allowance before the beginning of the reactor core heating and exceeding of the maximum design damage limit for the fuel rod of maximum power will be determined by the amount of boric solution in the ECCS hydro accumulators. The engineering assessment shows that the time allowance before the beginning of the reactor core heating may vary from 13 to 15 days from the beginning of the accident, provided that the rate of pressure decrease in the primary circuit (and, correspondingly, in the ECCS hydro accumulators) is stable [31]. Later, it will be required to restore the design NPP power supply and supply boric solution to the primary circuit using design means.

Fuel pool

Loss of power supply to the normal operation systems and failure of all DGs result in the failure of the fuel pool design cooling system (FAK) and the standby cooling channel through the emergency low pressure injection system (JNG) and the sprinkler system (JMN).

Cooling of spent nuclear fuel in the fuel pool is performed by heating and evaporation of water above the fuel assembly level.

Calculations of water evaporation time before fuel uncovering due to the initiating event associated with the stopped heat removal (cooling water supply) were performed for two options:

option 1: full unloading of the core into the spent fuel pool, taking into account availability of spent FAs for 10 years of operation;

The total time of the spent fuel pool boiling-off to the FA heads from the beginning of the accident will be at least 41 hours.

option 2: power operation at the beginning of the reactor campaign (after refueling).

The total time of the spent fuel pool boiling-off to the FA heads from the beginning of the accident will be at least 89 hours.

The time for personnel emergency response was determined: from the initiating event to FA damage (uncovering) for the options considered. It is assumed that fuel damage does not occur if uncovering of the FA fuel portion does not occur.

Characteristics of the technical means for makeup of the spent fuel pool were selected taking into account the prevention of heavy fuel damage in the spent fuel pool.

APCS (automated process control system)

In case of loss of power supply to the APCS equipment for more than 24 hours:

- I&C and BDBA panel are in operation (channels 7-8);
- it is necessary to supply power from mobile DG set within 24 hours;
- power supply interruption is allowed.

After restoration of the power supply from DG of the power supply system for the BDBA monitoring and management equipment, I&C of the 7th and 8th channels will start automatically. Serviceability of the I&C (7th channel) and BDBA panel will be restored within 10 minutes after restoration of the power supply.

Basic directions of the personnel actions in case of complete AC loss:

- reactor plant transfer to and maintaining in the safe condition in accordance with the requirements of the Process Regulations, Instructions for the Reactor Plant Emergency Response, BDBA Management Guidelines, Severe Accident Management Guidelines;

- prompt assessment of the condition of the NPP power supply equipment (including emergency equipment), as well as availability and operability of systems and equipment;

- arranging for priority (urgent) works on restoration of power supply, including putting a mobile DG set into operation;

- putting a mobile DG set into operation to ensure water supply to the PHRS tanks and spent fuel pool;

- implementation of the plan “Measures to Protect Personnel in the Event of an Emergency at the Belarusian NPP” (if necessary).

6.1.3 Loss of External Power Supply, Regular Redundant AC Power Supplies and Various Stationary Backup AC Power Supplies

In case of loss of the power supply to the normal operation systems and safety systems (failure of all DGs), functioning of the normal operation systems ensuring the removal of the reactor residual heat to the ultimate heat sink and cooling of the spent fuel pool. At the same time, failure of active safety systems occurs.

The Unit condition at the initial stage of the accident is characterized by:

- complete unavailability of AC power supply (external and internal);
- availability of power supply from the EPSS UPS for some valves (isolating valves of the sealed enclosure, BRU-A, MSIV) and I&C. Power supply from the UPS is designed for 2 hours of operation without battery recharge. After 2 hours I&C technical means powered from the EPSS will be tripped;
- availability of power supply from UPS of the system for power supply to the BDBA monitoring and management equipment (channel 7). The battery capacity is 2030 A*h. Power from UPS is designed for 24 hours without recharge of the batteries (with no regard to the operation of communication systems) constituting a part of the UPS. Connection of a mobile DG set (power 500 kW) within 24 hours to the switchgear of channel 7 – the cabinet (seismic category I according to NP-031-01, dust and moisture proof design – IP54, UHL1, hufter-proof, with a lock) located on the outer wall of building UJE at el.+1.400. The power calculation for the mobile DG (XKA70) takes into account a current equal to the current of a 10-hour battery discharge (203 A). At this current the fully discharged battery (at the estimated discharge time of 93-95 hours) will be charged to a full capacity of 2030 A*h in 10 hours. As DG is planned to be connected for a time less than that required for a full battery discharge, the time of recovery to the full capacity will be significantly less than 10 hours and will be determined by the discharge mode of the battery;
- subcritical state of the reactor;
- primary circuit leaks – 2.15 m^3/h, which corresponds to the maximum possible leakage during operation at the rated power;
- tight secondary circuit;
- full reserve of cooling water for the BDBA management systems in the four SG PHRS tanks;
- water level in the fuel pool – 8.7 m (level at fuel storage).

Under these conditions, the technical means and organizational arrangements for BDBA management are presented below.

Professional level of the operational shift to establish the necessary electrical connections, as well as the time required to perform these actions, are the tasks for the personnel emergency training. These procedures are described in section 7.1.3.4.

Residual heat removal and reactor plant cooldown

Removal of residual heat and cooldown of the reactor plant in the BDBA mode with blackout are performed using the system of passive heat removal through the steam generators.

SG PHRS is technical means for BDBA management.

The system consists of four independent channels – one for each steam generator. The efficiency of one channel is 33.3%.

The system channels are natural circulation circuits. Each circuit includes one water tank, sixteen heat exchangers, pipelines of the steam-and-condensate path with startup, control and isolating valves.

In all modes requiring the system operation, heat removal to the ultimate heat sink – the environment – is performed through the following chain: reactor – steam generator – SG PHRS SG – atmospheric air. Heat is removed to the atmosphere by evaporation of water from the tanks.

Large and small startup valves are installed in parallel to each other on the downflow pipe of each SG PHRS channel upstream of the steam generator. The startup valves ensure automatic connection of JNB system in the respective cooldown mode. The startup valves are closed in the standby mode. A solenoid valve is used as a small startup valve, which opens upon power failure during an accident with loss of all power supplies in case of DG startup failure. The small valve also opens automatically in case of an accident with a complete loss of feed water, as well as in case of failure of all BRU-A in the cooldown mode (30oC/h) upon APCS signals in accordance with SG PHRS control algorithms. A motor-operated valve is used as a big valve. The valve opens automatically upon the APCS signals in accordance with the SG PHRS control algorithms in case of an accident with primary circuit leakage and HP ECCS pump failure. Automatic opening of the big valve is also provided in case of an accident with primary circuit leakage to the secondary circuit with failure of all BRU-A in the cooldown mode at a rate of 60oC/h.

Automatic starting of SG PHRS in case of an accident with blackout of the reactor plant is based on a passive principle. For this purpose, mechanical passive opening of the startup valve is implemented following the initiating event without operator or automatic control system involvement. In other BDBAs where SG PHRS operation is provided, the signal for opening the valves has to be generated in the APCS for automatic startup of the system. SG PHRS valves are powered from the power supply channel 7.

In case of BDBA with blackout, water in the four SG PHRS water tanks is sufficient to ensure effective heat removal from the reactor plant within 72 hours (provided that water from the four emergency heat removal tanks is used). To continue cooldown with SG PHRS, the water tanks shall be refilled.

Heat removal from the containment

To ensure long-term heat removal from the containment during BDBA, the system of passive heat removal from the containment (containment PHRS) is used.

Containment PHRS is technical means for BDBA management.

During BDBA, including accidents with severe core damages, the containment PHRS reduces and maintains temperature and pressure inside the containment within the design limits, as well as removal of the heat released in the sealed volume to the ultimate heat sink.

The system consists of four independent channels. The efficiency of one channel is 33.3%. Each channel consists of four circuits of natural circulation.

Each circuit has one heat exchanger-condenser, circulation pipelines with isolating and protective valves and a steam dump device. Heat exchangers-condensers are installed in the sealed volume at the containment dome.

In all modes requiring the system operation, heat removal to the ultimate heat sink – the environment – is performed through the following chain: atmosphere of the sealed volume – containment PHRS – SG PHRS tanks – atmospheric air. Heat is removed into the atmosphere by evaporation of water from the SG PHRS tanks.

In order to stabilize the flow, reduce pulsations and prevent condensation hydraulic shocks inside the SG PHRS tanks, steam dump devices are installed at the inlet of the system pipelines.

During SG PHRS operation, pressure and temperature slightly increase in the sealed volume due to leakages through RCPS seals and evaporation of water from the fuel pool. In this situation, operation of the containment PHRS is not required.

Personnel actions to diagnose the NPP state, restore the damaged safety functions and prevent or mitigate the consequences of the core damage in BDBA will be specified in the BDBA Management Guidelines, Severe Accident Management Guidelines and the plan “Measures to Protect Personnel in the Event of an Accident”.

BDBA with loss of external power supply and design backup AC power supplies (DG sets) result in failure of ventilation systems, including those supporting the operation of I&C equipment, sources of their power supply, cooling and supply of outdoor air to the MCR. Since the I&C equipment is powered from UPS (from batteries through inverters), their operability will be maintained during the estimated battery discharge time (for normal operation I&C and safety systems – 2 hours). As a result, the air temperature and concentration of carbon dioxide increase in the MCR.

Based on the calculation results for this mode, it was established:

1) allowed time of personnel stay (before PPE use) is 52 hours for 8 people, and 34.5 hours for 12 people;
2) temperature in the MCR will not exceed 43 °C during 72 hours.

Basic directions of the personnel actions in case of complete AC loss:

- reactor plant transfer to and maintaining in the safe condition in accordance with the requirements of the Process Regulations, Instructions for the Reactor Plant Emergency Response, BDBA Management Guidelines, Severe Accident Management Guidelines;
- prompt assessment of the condition of the NPP power supply equipment (including emergency equipment), as well as availability and operability of systems and equipment;
- arranging for priority (urgent) works on restoration of power supply, including putting a mobile DG set into operation;
- putting a mobile DG set into operation to ensure water supply to the PHRS tanks and spent fuel pool;
- implementation of the plan “Measures to Protect Personnel in the Event of an Emergency at the Belarusian NPP” (if necessary).

6.1.4 Conclusion on Sufficiency of NPP Protection from Loss of Power Supply

To prevent fuel damage in the reactor in case of an accident involving the loss of all AC sources at the NPP at power operation of the reactor plant it is required to take measures not later than within 72 hours from the beginning of the accident to restore and maintain water reserve in the emergency heat removal tanks for ensuring PHRS operation (with all the emergency heat removal tanks involved). In this case, the time allowance before the beginning of the reactor core heating and exceeding of the maximum design damage limit for the fuel rod of maximum power will be determined by the amount of boric solution in the ECCS hydro accumulators. The time allowance before the beginning of the reactor core heating may vary from 13 to 15 days from the beginning of the accident, provided that the rate of pressure decrease in the primary circuit (and, correspondingly, in the ECCS hydro accumulators) is stable [31]. Later, it will be required to restore the design NPP power supply and supply boric solution to the primary circuit using design means.

To prevent fuel damage in the spent fuel pool in case of an accident involving the loss of all AC sources at the NPP at power operation of the reactor plant, it is necessary to supply water to the spent fuel pool at a flow rate of min.4.5 kg/s within not more than 89 hours [31].

To prevent fuel damage in the spent fuel pool in case of an accident involving the loss of all AC sources at the NPP under conditions of the complete core unloading, it is necessary to supply water to the spent fuel pool at a flow rate of min. 7 kg/s within not more than 41 hours [31].

As additional technical means to manage accidents involving loss of power supply exceeding the design requirements (more than 72 hours from the beginning of the accident with the loss of all power supplies), the design provides for a make-up system for the SG PHRS water tanks and spent fuel pool.

Making-up of the PHRS tanks and spent fuel pool is provided by low-power high-pressure pump JNB50AP001 of the make-up system for the emergency heat removal tanks. This pump unit is located in the steam chamber and connected to the tanks of the LCU system and to the sump tanks of the containment. The pump is powered from the BDBA power supply channel (from the connected mobile DG set of power supply channel 7). The BDBA power supply channel is designed for 24 hours of independent operation and allows for connection of a mobile DG set for recharging the batteries and further operation of the system.

In accordance with the recommendations resulting from development of the Stress Test Report (target reassessment of safety) for Belarusian NPP, two mobile DG sets (one per NPP Unit) with a power of 500 kW will be provided, which presumably will be located outdoors at the NPP site.

Monitoring and control are performed from the BDBA panel located in the MCR.

Within 24 hours a mobile DG set is delivered to the point of its connection and prepared for operation. According to the design documentation, the location and design of the connection points for the mobile DG set as power supplies ensures protection from flooding, extreme precipitation and other unfavorable weather conditions. The mobile DG set is connected to the switchgear of channel 7 – the cabinet (seismic category I according to NP-031-01, dust and moisture proof design – IP54, UHL1, hufter-proof, with a lock) located on the outer wall of building UJE at el.+1.400.

The mobile DG set is controlled directly from the local control panel located on the equipment.

Facilities designed for electrical equipment installation meet the requirements for ensuring their safety and operability in accordance with their classification and ensure protection against possible natural and man-caused impacts in the NPP area.

Calculation and equipment selection of the equipment take into account both the external climatic conditions of the site and the external conditions associated with human activities. The structures of buildings and facilities are designed taking into account natural and man-caused impacts such as seismic activity, tornado, snow, ice and wind loads, outside air temperature, humidity, air shock wave, aircraft crash. Based on the calculated data, appropriate design solutions are applied for the facilities and equipment.

6.1.5 Measures to Improve the NPP Stability in case of Power Supply Loss

To prevent fuel damage in the reactor in case of an accident involving the loss of all AC power supplies at the NPP at the reactor plant power operation, it is required to take measures not later than within 72 hours from the beginning of the accident to restore and maintain water reserve in the emergency heat removal tanks for ensuring PHRS operation (with all the emergency heat removal tanks involved). The time allowance before the beginning of the reactor core heating may vary from 13 to 15 days from the beginning of the accident, provided that the rate of pressure decrease in the primary circuit (and, correspondingly, in the ECCS hydro accumulators) is stable.

To prevent fuel damage in the spent fuel pool in case of an accident involving the loss of all AC sources at the NPP at power operation of the reactor plant, it is necessary to supply water to the spent fuel pool at a flow rate of min. 7 kg/s within not more than 41 hours (the most conservative option) [31].

Based on the information provided in the report [31], it can be concluded that the means available in the NPP design are sufficient, adequate and stable to protect against loss of power supply, including impacts caused by earthquakes and floods.

To mitigate the consequences of accidents with a complete loss of power supply, the following organizational and technical measures are provided:

- in terms of organizational measures for preparation of operation and commissioning of a mobile DG set - it is required to develop appropriate operational instructions and sections of emergency procedures for their use in case of complete loss of AC power supply;
- in terms of organizational measures for preparation of operation and commissioning of an emergency standby auxiliary transformer with a power of 16 MVA 110/10 kV, it is required to develop appropriate operational instructions and sections of emergency procedures for its use with a complete loss of AC power supply. The relevant instructions and procedures shall indicate the actions of operating personnel, including: a procedure for taking decision by the operating personnel to use an additional 110/10 kV source, a procedure for manual preparation of the circuit for power supply of the Unit essential loads; a procedure for selection of 10 kV section (including EPSS);
- in terms of organizational measures to allow for power supply from the neighboring Unit (if possible) through 10 kV assemblies of 330/10 kV standby transformers connected

together with cable jumpers, it is required to develop appropriate operational instructions and sections of emergency procedures for its use at full loss of AC power supply.

- in terms of relevant operational documentation - develop additional sections on the actions of personnel in the event of an accident with complete loss of AC power supply of the NPP with regard to:

  - strengthening the monitoring of the Unit process parameters;
  - strengthening the monitoring of the safety-related systems operation;
  - preparation for operation and commissioning of the designed safety systems;
  - preparation for operation and commissioning of the mobile DG set.

- develop a list of consumables and material resources (fuel, cable, pipes, tools, etc.) for repair and restoration works, determination of the storage locations and conditions for these materials to perform the actions specified in the sections of BDBA Management Guidelines, Severe Accident Management Guidelines in case of complete loss of power supply at the NPP.

- in terms of heat removal from the reactor plant - to arrange for making-up of LCU tanks from the on-site and off-site sources of water after 72 hours;

- in terms of removal of residual heat from the spent fuel pool:

  - arrange for making-up of the spent fuel pool after 41 hour. This measure can be implemented by connecting non-standard facilities (a fire engine with a pump unit having a capacity of 40 liters/s and a head of 100 m) to two process connectors of JNB50 system located on the outside of building UJE (at elevations +0.690 and +0.730 elevations, with water intake from LCU tanks through the pump unit of the fire engine and further through the pipelines of system JNB50 the water, is supplied to the spent fuel pool) having flanges with plugs installed on them;
  - modify the process flow diagram of JNB50 system by adding tie-in of a check valve bypass to the make-up line for the emergency heat removal tanks. This solution will allow for making-up of the spent fuel pool by the operating personnel after 41 hours.

6.2 Loss of Residual Heat Removal/Loss of the Ultimate Heat Sink

6.2.1 Design Measures and Means to Prevent Loss of Ultimate Heat Sink, Resistance of Provided Measures and Means to Earthquakes and Flooding

Cooling system for essential loads

The cooling water system for essential loads (PE) operates in all operating modes of the Unit (including blackout), except for the mode with loss of external power supply, design backup AC power supplies and various fixed backup AC power supplies.

As initial design data determining the required characteristics and parameters of the cooling water system for essential loads, the following criteria and requirements are applied for the system:

- in the modes of normal operation at the reactor power operation, the PE system must ensure heat removal from the consumers through two channels PE 10 (20) and PE 30 (40) at a cooling water temperature from 4 to 28°C;
- in the modes of planned cooldown of the primary circuit, the PE system must ensure heat removal from the consumers of intermediate component cooling circuit KAA through two channels, for example PE10 and PE30, and from the consumers of buildings UJA, UKA - through one channel, for example PE20, with a cooling water temperature not exceeding 28°C;
- in the modes of design basis accidents, the system must provide heat removal from the consumers of safety systems through any two channels at a cooling water temperature of 31°C.

In accordance with the structure of the safety systems, the PE system consists of four channels which are independent in terms of process and electrical connections, as well as in terms of I&C systems. To perform safety functions in emergency modes with a loss of coolant, it is sufficient to operate two of the four channels with an efficiency of 50% each.

The water flow in each operating channel is 1700 or 3400 m3/h - depending on the operation mode of the Unit and the cooling water temperature.

Pumps of each channel of the system are located in isolated rooms of buildings UQC, therefore in all emergency situations a common-cause failure of not more than one of the four channels of the system is possible.

The channels of the system are completely independent of each other: process parts, control systems, locations of equipment, pipelines, cables, controls, etc. are independent. Thus, a dependent failure in another channel is impossible.

For the PE system of each Unit, two spray pools are provided - one spray pool per two channels. Accordingly, the spray pool is divided into two sections. Water cooled in the spray pools is supplied via gravity water conduits through URS switching chambers to the water receivers of the pump stations for essential loads UQC (two per Unit), passes through rotating water purification grids to the suction of the pumps. From the pump stations, the water from the water supply conduits the water is supplied to the heat exchangers of the intermediate component cooling circuit for essential loads in building UKD. The heated water is diverted through the tailwater conduits to the switching chambers of the spray pools and then to the distributing pipelines of the spray pools for cooling. All the water conduits are laid in tunnels.

Consumers of the safety systems requiring uninterrupted supply of cooling water with a temperature not exceeding + 31°C are connected to each channel of the PE system.

Intermediate component cooling circuits of the cooling systems for essential loads

At the NPP power operation, the KAA system must ensure heat removal from 18 to 43 MW from consumers of the normal operation systems, and in case of failures - through two channels at a cooling water temperature of the intermediate component cooling circuit of 33°C and a service water temperature of 28°C.

In the cooldown modes of the intermediate circuit, the KAA system must provide heat removal from 86 to 103 MW from consumers through three channels at a cooling water temperature of the intermediate component cooling circuit of 33°C and a service water temperature of 28°C.

The KAA system must ensure heat removal from consumers when maintaining the reactor plant in the "hot" state, as well as when cooling the reactor plant down to the "cold" state and ensure heat removal when maintaining the reactor plant in the "cold" state.

SG PHRS, Containment PHRS

The passive heat removal systems SG PHRS and Containment PHRS are technical means of BDBA management.

The system of passive heat removal through steam generators (JNB) is designed for the long-term removal of the core residual heat to the ultimate heat sink through the second circuit in case of BDBA involving total loss of all AC power supplies, total loss of feed water, as well as a part of the range of accidents with the primary circuit coolant leakage in case of failure of the active safety systems.

The system of passive heat removal from the containment (JMP) reduces and maintains the pressure inside the containment within the design limits and removes the heat released under the containment in case of BDBA, including accidents with severe core damage, to the ultimate heat sink.

Heat is removed to the ultimate heat sink by evaporation of water from the four emergency heat removal tanks, which are a single storage of cooling water of the SG PHRS and Containment PHRS.

The tanks are reinforced concrete structures lined with stainless steel located in separate rooms of the ring structure of the reactor building, the total water volume in each of the four emergency heat removal tanks being not less than 540 m3.

The capacity of the SG PHRS and Containment PHRS was selected taking into account the principle of redundancy, based on the conditions of the most probable BDBA scenarios considered in the design. Each of the systems consists of 4 channels totally independent of each other with a capacity of 4x33.3%. Three operable channels of the SG PHRS and Containment PHRS are sufficient for the systems to perform their functions in full scope in any mode requiring their operation.

Assessment of operability of the available facilities at external impacts:

Assessment of the design measures for resistance to earthquake in situations with loss of heat removal to the ultimate heat sink is provided in Section 2, resistance to flooding - in Section 3.2.1. The design measures exclude the possibility of the impact of destruction of the pipelines and equipment referring to seismic category II on the NPP components referring to seismic category I.

The design systems of heat removal to the ultimate heat sink described in the section are able to perform their functions at external impacts assumed for this design. The components of the systems for heat removal to the ultimate heat sink are protected from external disasters: earthquakes, tornadoes, aircraft crashes, and also from the impact of an air shock wave. This is ensured by the design of the buildings and structures intended to accommodate components of the systems.

The equipment and pipelines of the systems for heat removal to the ultimate heat sink refer to seismic category I and fulfill their functions in the event of an earthquake up to the level of the safe shutdown earthquake (SSE).

In case of accidents with NPP blackout, the systems for heat removal to the ultimate heat sink remain operable during the operation time of the EPSS DG and the presence of a water level in the emergency heat removal tanks, the independent operation time of the PE system spray pools is indicated in Section 6.2.2.

6.2.2 Loss of Heat Removal for Different Operation Modes of the Reactor Plant Modes

The main ultimate heat sink in the normal operation mode is cooling water towers. The ultimate heat sink in the emergency mode is spray pools. An alternative ultimate heat sink in the BDBA mode is emergency heat removal tanks.

In case of loss of cooling water from the condensers of the turbine plant, the process of the Unit cooldown and maintaining in a safe state is performed using other ultimate heat sinks - the atmosphere (through BRU-A) and further - the spray pools.

Heat removal from the primary circuit to the atmosphere through BRU-A allows (at a rated RP cooldown rate of 15°C/h) for reducing the primary circuit temperature to the values sufficient for connection of the JNG-JNA system in 8-10 hours. The SG make-up to ensure the BRU-A operation is provided by emergency feed water pumps of the safety systems.

Spray pools.

The design characteristics of the spray pool (capacity, overall dimensions, type and nozzle arrangement) are determined by the thermal hydraulic analysis of all operation modes of the system, based on the necessity to ensure the cooldown of the reactor plant in the mode of the maximum design basis accident at a temperature of cooling water supplied to the reactor compartment not higher than + 31°C.

In this case, any two channels of the PE system can be in operation, and the entire thermal load can be applied to one of the spray pools.

The temperature of the cooling water is determined for the weather conditions of the hot five-day period of 10% probability, when the Unit is transferred from the nominal operation mode to the emergency cooldown mode, taking into account the heat-storage capacity of the spray pool.

For make-up of the spray pools, the design provides for the use of water downstream of the first stage of reverse osmosis of the water treatment plant. As a standby source of make-up for the spray pools, the capacity of the cooling tower pool and the capacity of the water intake chamber of the pump station for the consumers of the turbine building (URD) from which the water is pumped to the water treatment building (UGB) and then pumped to the spray pools are adopted. In case of common-cause failure of both make-up sources, for example, in the event of SSE, the capacity of each spray pool ensures the operation of the system without making-up for a long time (not less than 8 days) during which the technical measures for supply of make-up water must be arranged.

For accidents involving the loss of heat removal from the primary circuit and the loss of an alternative heat removal, the operation of the automatic equipment and the analysis of the final states for various operation modes of the reactor plant are presented in [31] in Tables 5.2.2.1 to 5.2.2.5.

To ensure the supply of additional water to the Unit site in order to compensate for losses in the service water supply systems, the additional water piping system (GAC) supplies the required amount of water to the GA system for filling and make-up of the recirculating water supply system with cooling towers. Water is supplied to the GAC system by the shore pump station of the service water supply system. In case of anticipated operational occurrences and design basis accidents, the additional water is supplied in the required volumes, provided that the system remains operable.

The NPP design does not provide for any dependence in the operation of different NPP Units on each other.

Failure of the main and auxiliary cooling systems in the "cold" initial state of the reactor plant does not affect safety and does not change the NPP operation mode.

Mutual support between the Units is possible by pumping water of the spray pools from one Unit to the other by means of a standby tank, which is functionally designed for water storage when one section of the spray pool is empties for cleaning, repair and inspection. The standby tank is designed as a common tank for Units 1 and 2, it is located between the adjacent spray pools of Units 1 and 2. The standby tank is connected with each section of the spray pool by means of drain and filling piping network. The standby tank is emptied by pumps.

Also, rooms of the Units allow for storage of chemical reagents for water chemistry adjustment for tanks of the process systems. Therefore, the need for chemical reagents can be promptly satisfied by transporting them from one Unit to the other.

6.2.3 Loss of Heat Removal and Loss of Alternative Heat Removal for Different Operating Modes

Loss of the main ultimate heat sink (water of the system PA) results in the loss of vacuum in the turbine condenser. If the vacuum is lost, the high-pressure stop valves of the turbine are closed. During the reactor plant unloading, with BRU-A and BRU-K in operation, a pressure setpoint of 8.1 MPa can be reached in the steam generator followed by generation of the emergency protection signal. Loss of the ultimate heat sink (spray pools) leads to the inability to use the residual heat removal systems for the primary circuit cooldown (JNG-1, JNA, JND). Thus, heat removal from the primary circuit is ensured by the BRU-A operation, as well as by the main and auxiliary feed water systems.

In case of failure of all BRU-A, which is an unlikely event, SG PHRS tanks (JNB10-40) may be regarded as an additional emergency ultimate heat sink for cooling of nuclear fuel in the reactor. According to the calculations presented in the report [31], the SG PHRS can remove residual heat of the reactor plant in the self-sufficient mode for 72 hours from the beginning of the accident, provided that the water reserves of the 4 emergency heat removal tanks are used. If 3 out of the 4 emergency heat removal tanks are used, the self-sufficient operation for not less than 24 hours is provided. Further operation of the SG PHRS is ensured by making-up of the emergency heat removal tanks with JNB50 pump from the LCU tanks.

Loss of all the systems for heat removal to the ultimate heat sink results in failure of the design fuel pool cooling system (FAK) and the standby channel for cooling by the low pressure emergency injection system (JNG) and the sprinkler system (JMN). Spent nuclear fuel cooled in the fuel pool by heating and evaporation of water above the level of the fuel assemblies (FA).

In order to avoid fuel damage in the spent fuel pool in case of an accident with loss of all the systems for heat removal to the ultimate heat sink at the NPP during power operation of the reactor plant or complete core unloading in the cold state, it is required to make up the spent fuel pool from the LCU tanks with tJNB50AP001 pump through the FAK70 line.

The main directions of the personnel actions in case of complete loss of the design ultimate heat sinks are as follows:

- reactor plant transfer to and maintaining in the safe condition in accordance with the requirements of the Process Regulations, the Reactor Plant Emergency Response Manual, the BDBA Management Manual.
- putting the SG PHRS into operation, monitoring the operation of the system;
- prompt assessment of the equipment condition for the NPP design ultimate heat sinks (PA, PC, PE systems), as well as the availability and operability of the systems and equipment;
- preparation for operation of additional technical means for SG and PHRS making-up;
- arranging for priority (urgent) works to resume the operation of the NPP ultimate heat sink systems (PA, PC, PE systems);
- implementation of the Action Plan for personnel protection in the event of an accident at the Belarusian NPP (if required).

6.2.4 Conclusion on Sufficiency of NPP Protection in Case of Loss of the Ultimate Heat Sink

In case of loss of the ultimate heat sink there are enough technical means to prevent damage of nuclear fuel and transfer the reactor plant to a safe state.

Depending on the failures postulated in section 6.2, heat is removed from the primary circuit by means of the operation of BRU-A, main and auxiliary feed water systems, spray pools or SG PHRS. The controlled and uncontrolled leaks of the primary circuit can be compensated from the JNG-2 system and from the tanks of the JNK system by the pumps of the JDH system. The spent fuel pool is made up from the JNK tanks by the design means using pumps of the FAK or JMN systems.

Assessment of the design measures for resistance to earthquake in situations with loss of heat removal to the ultimate heat sink is provided in Section 3, resistance to flooding - in Section 4. The design measures exclude the possibility of the impact of destruction of the pipelines and equipment referring to seismic category II on the NPP components referring to seismic category I.

The design systems of heat removal to the ultimate heat sink described in the section are able to perform their functions at external impacts assumed for this design. The components of the systems for heat removal to the ultimate heat sink are protected from external disasters: earthquakes, tornadoes, aircraft crashes, and also from the impact of an air shock wave. This is ensured by the design of the buildings and structures intended to accommodate components of the systems.

The equipment and pipelines of the systems for heat removal to the ultimate heat sink refer to seismic category I and fulfill their functions in the event of an earthquake up to the level of the safe shutdown earthquake (SSE).

In case of accidents with NPP blackout, the systems for heat removal to the ultimate heat sink remain operable during the operation time of the EPSS DG and the presence of a water level in the emergency heat removal tanks, the independent operation time of the PE system spray pools is indicated in Section 6.2.2.

6.2.5 Measures to Improve the NPP Stability in Case of a Loss of the Ultimate Heat Sink

As additional technical means to manage accidents involving loss of the ultimate heat sink, the design provides for a make-up system for the SG PHRS water tanks and spent fuel pool.

Making-up of the PHRS tanks and spent fuel pool is provided by low-power high-pressure pump JNB50AP001 of the make-up system for the emergency heat removal tanks. This pump unit is located in the steam chamber and connected to the tanks of the LCU system and to the sump tanks of the containment. To improve the NPP stability, the measures similar to those proposed in paragraph 6.1.5 are proposed in regard to the making-up of the LCU tanks and the spent fuel pool.

To maintain the controlled state after BDBA for more than 72 hours in case of loss of the ultimate heat sink at two NPP Units at the same time, the respective measures will be proposed.

6.3 Loss of the Ultimate Heat Sink in Combination with the NPP Blackout

6.3.1 Time of the NPP Site Self-Sufficiency before Loss of Normal Cooling of the Reactor Core and the Spent Fuel Pool

The results of the analysis of the BDBA with the blackout are valid for the BDBA with complete loss of the ultimate heat sink due to complete loss of the systems for heat removal to the ultimate heat sink at the initiating event associated with the NPP blackout.

For management of the BDBA with complete loss of the ultimate heat sink the same means and measures are applied as for the management of the BDBA with the blackout.

The calculations presented by the operating organization demonstrate the following time reserves for personnel to take measures for preventing a severe accident stage with the blackout.

The time to fuel uncovering in the spent fuel pool is:

- 41 hours at the initiating event: full unloading of the core into the spent fuel pool, taking into account the availability of spent fuel assemblies for 10 years of operation;
- 89 hours at the initiating event: power operation at the beginning of the reactor campaign (after refueling).

Residual heat is removed from the reactor plant by the SG PHRS within not less than 72 hours. If after 72 hours from the beginning of the accident the emergency heat removal tanks are not made-up with cooling water, gradual dehydration of the reactor plant will occur and the accident will pass to a severe phase.

6.3.2 Actions to Prevent Nuclear Fuel Damage

The main directions of the personnel actions in case of complete loss of the design ultimate heat sinks (PA, PC, PE systems) are as follows:

- reactor plant transfer to and maintaining in the safe condition in accordance with the requirements of the Process Regulations, the Reactor Plant Emergency Response Manual, the BDBA Management Manual.
- putting the SG PHRS into operation, monitoring the operation of the system;
- prompt assessment of the equipment condition for the NPP design ultimate heat sinks (PA, PC, PE systems), as well as the availability and operability of the systems and equipment;
- preparation for operation of additional technical means to make up the SG PHRS and the spent fuel pool.

As additional technical means for management of accidents exceeding the design requirements (more than 72 hours from the beginning of the accident with the loss of all power sources), the design provides for a system for make-up of the SG PHRS water storage tanks and the spent fuel pool.

The PHRS tanks and the spent fuel pool are made up by a low-power high-pressure pump JNB50AP001 of the make-up system for the PHRS tanks. This pump unit is located in the steam chamber and connected to the tanks of the LCU system and to the sump tanks of the containment. The pump is powered from the BDBA power supply channel (from the connected mobile DG set of 7th power supply channel). The BDBA power supply channel is designed for 24 hours of self-contained operation and allows for connecting a mobile DG set for recharging the batteries and further operation of the system.

In accordance with the recommendations resulting from development of the Stress Test Report (target reassessment of safety) for Belarusian NPP, two mobile DG sets (one per NPP Unit) with a power of 500 kW will be provided, which presumably will be located outdoors at the NPP site.

Monitoring and control are performed from the BDBA panel located in the MCR.

Within 24 hours a mobile DG set is delivered to the point of its connection and prepared for operation. The mobile DG set is connected to the switchgear of channel 7 – the cabinet (seismic category I according to NP-031-01, dust and moisture proof design – IP54, UHL1, hufter-proof, with a lock) located on the outer wall of building UJE at el.+1.400.

The mobile DG set is controlled directly from the local control panel located on the equipment.

6.3.3 Measures to Improve the NPP stability in Case of Loss of the Ultimate Heat Sink in Combination with the NPP Blackout

As additional technical means for management of accidents with losses of the power supply and the ultimate heat sink, the design provides for a system for make-up of the SG PHRS water storage tanks and the spent fuel pool.

The PHRS tanks and the spent fuel pool are made up by a low-power high-pressure pump JNB50AP001 of the make-up system for the PHRS tanks. This pump unit is located in the steam chamber and connected to the tanks of the LCU system and to the sump tanks of the containment. The pump is powered from the BDBA power supply channel (from the connected mobile DG set of 7th power supply channel). The BDBA power supply channel is designed for 24 hours of self-contained operation and allows for connecting a mobile DG set for recharging the batteries and further operation of the system. In accordance with the recommendations resulting from development of the Stress Test Report (target reassessment of safety) for Belarusian NPP, two mobile DG sets (one per NPP Unit) with a power of 500 kW will be provided, which presumably will be located outdoors at the NPP site.

In the normal operation mode, each of the 2 switchgears of the BDBA power supply channel is connected through 2 circuit breakers installed in series to the 0.4/0.23 kV sections of the two EPSS channels. In the normal operation mode, the switchgear feeder circuit breakers are constantly closed.

In the event of an accident related to the total loss of power supply (failure of all DGs), the switchgears of the BDBA power supply channel are disconnected from the 0.4/0.23 kV sections of the EPSS channels. In case of BDBA the power is supplied by the UPS batteries.

Monitoring and control are performed from the BDBA panel located in the MCR.

Within 24 hours a mobile DG set is delivered to the point of its connection and prepared for operation. In case of BDBA the mobile DG set is connected for power supply to the consumers of Unit 1 using the design means and methods of connection to the BDBA panels. The mobile DG set is connected to the switchgear of channel 7 – the cabinet (seismic category I according to NP-031-01, dust and moisture proof design – IP54, UHL1, hufter-proof, with a lock) located on the outer wall of building UJE at el.+1.400.

The mobile DG set is controlled and monitored directly from local control panels located on this equipment.

To improve the NPP stability in case of loss of power supply and the ultimate heat sink, the following measures are provided: two mobile DGs are included into the delivery set (one DG per NPP Unit).

The use of the mobile DG sets in the amount of 1 mobile DG set per NPP unit improves the NPP stability in case of accidents leading to complete loss of power supply at once at all NPP Units.

7. MANAGEMENT OF SEVERE ACCIDENTS

7.1 Licensee's Organization and Activities for Accident Management

7.1.1 Organizational Measures of the Operating Organization for Accident Management

For the management of accidents at the Belarusian NPP, the Emergency Operation Procedure (EOP), Beyond Design Basis Accident Management Guidelines (BDBAMG) and Severe Accident Management Guidelines (SAMG) that describe the order and criteria for transition from one procedure or instruction to another, as well as their application and mutual relations are under development and will be put into force.

The procedure of development of EOP, BDBAMG and SAMG is as follows:

Stage 1. Development of the draft content of EOP, BDBAMG, SAMG for the Belarusian NPP.

Stage 2. Based on the drafts of EOP, BDBAMG, SAMG for the Belarusian NPP, a specialized organization develops the following documents:

- Substantiating calculation of EOP of the power unit No. 1 of the Belarusian NPP;
- Substantiating calculation of BDBAMG of the power unit No. 1 of the Belarusian NPP;
- Substantiating calculation of SAMG of the power unit No. 1 of the Belarusian NPP.

Stage 3. Development of EOP, BDBAMG, SAMG for the Belarusian NPP on the basis of documents developed at Stages №1 and 2.

Stage 4. Agreement and approval of EOP, BDBAMG, SAMG for the Belarusian NPP in accordance with the established procedure.

Design basis accidents (considered in EOP) include accidents that are initiating events for activation of the reactor protection system and/or resulting in activation of the safety systems or creating conditions for their activation.

The algorithm of personnel actions is detailed in a tabular step-by-step format. The actions are based on the reactor plant's condition character:

- process parameters.
- display of the alarm actuation and equipment status.

Management of a beyond design-basis accident (BDBA) establishes one of levels of protection of physical barriers across the way of propagation of radioactive materials and includes actions preventing escalation of any design basis accidents into beyond design-basis accidents and mitigating consequences of beyond design-basis accidents. For these actions are used any available operating technical facilities, which are designed for safety assurance as during design basis accidents as for normal operation.

The main purpose of management of BDBA is to prevent an uncontrolled release of radioactive substances outside the design boundaries. This goal is achieved by consistent implementation of the defense-in-depth concept, called also as the multiple barriers concept.

The integrity of the protective barriers is ensured by the following conditions:

- quick shutdown of the reactor and maintaining the reactor core in a subcritical state;
- heat removal from the reactor core during an accident, as well as after stabilization of parameters in the post-accident state, heat removal from the reactor plant;
- protection of the primary circuit against overpressure, water hammers, thermal loads; localization of the consequences of the accident by means of shut-off the of the containment to minimize the radiological consequences, retain radioactive substances within the specified boundaries and in specified quantities.

More detailed organizational and technical activities are considered in BDBAMG and SAMG.

BDBAMG provides criteria for transition onto SAMG.

The main goal of SAMG is to implement a set of procedures and instructions for managing severe accidents and mitigating their consequences. The technical devices for managing severe accidents are mostly engineered based on the passive principle of actuation. During the first 24 hours following transition of an accident into a severe stage, the automated controls help the operator perform a minimum amount of actions to provide integrity of the container.

The main tasks of management of severe accidents are as following:

- maintaining tightness and integrity of the fourth physical barrier (i.e. the containment shell) as the last barrier across the way of release of radioactive substances;
- restriction of release of radioactive substances into the environment.

The design provides the following strategy for the severe accident management:

- protection of the containment against all physical impacts caused by a severe accident potentially leading to loads resulting in a its breach of tightness and integrity;
- brining NPP to a controlled state during the initial stage of an accident (immediately after the initiating event) to prevent damage of the containment and carry out proactive measures to reduce the radiation exposure;
- brining NPP into a manageable and safe state, as soon as functioning of necessary systems is restored; the main task at this stage is to reduce the pressure and temperature inside the containment, as well as to reduce a concentration of radioactive substances in the containment's atmosphere for minimizing radioactive releases into the environment in an event of a breach of the integrity of the containment shell.

To mitigate consequences of BDBA, the operating organization shall perform the following organizational and technical actions:

- assessment of scenarios of development of emergency situations during beyond design basis earthquakes based on analysis of seismic effects above the design values;
- development of documentation detailing actions of the personnel to evaluate the nuclear plant's condition, restore normal operation conditions and disarranged safety functions, prevent or limit effects of the reactor core damage. This documentation include: Technological Regulations of Safe Operation, Instruction for Elimination of Accidents at the Reactor Plant, Guidelines for Management of BDBA; Guidelines for Management of Severe Accidents, as well as the Action Plan for Protection of the Personnel in Case of an Accident. The Action Plan includes chapters detailing measures to solve the following issues:
- control of operation of the algorithms of safety functions;
- ensuring shutdown of the power unit following actuation of the reactor emergency protection signal, both by automatic controls (the main way) and relevant actions of the personnel;
- implementation of the personnel' s corrective actions, if necessary;
- maintaining the reactor core under the coolant layer, ensuring the coolant circulation;
- bringing the power unit to its final safe state by actions of the personnel thus allowing to start the recovery activities;
- control of equipment of systems of normal operation and systems important for safety for bringing the power unit to the final safe state and limiting radiation consequences resulting from operational occurrences and accidents;
- in case of failure of main controls, maintaining reliably the specified NPP mode by means of backup controls during a time established to correct the failure;
- assessment of degree of NPP destruction caused by an earthquake;
- preservation or recovery of functions of systems and equipment under destructions caused by an earthquake;
- localization of development of accidents simultaneously ensuring the limits of safe operation by means of ensuring subcriticality of the reactor core and keeping it under the coolant inventory (taking into account the foreseen design margins, actuation speed and efficiency of the protective systems);
- ensuring integrity of undamaged physical barriers;
- reaching the final state under which the chain reaction is terminated, the subcriticality is ensured and the recriticality of the core is prevented (also for a potentially damaged core);
- prevention (reduction) of a severe damage to the fuel, both by automatic controls and by relevant actions of the personnel;
- prevention of damage to the reactor vessel and primary circuit equipment;
- prevention of damage to the containment;
- prevention of severe damage to the core and mitigation of the consequences of the severe core damage;

- limitation of the radiation exposure on the personnel, population, and the environment.
- shipment to the site and putting into operation of mobile water pumps.

More detailed description of organizational and technical activities is provided in SAMG and BDBAMG.

Destruction or failure of the containment is a serious hazard in terms of a large emission of fission substances. This radioactive emission requires immediate measures to protect the health and safety of the population and NPP personnel.

The containment integrity may be potentially assessed by means of the site radiation monitoring. In case the radiation background is higher than the design values , it may be concluded that the containment integrity is under threat or already damaged, thus requiring immediate measures to limit the release and spread of the radiation substances.

Mitigation of the consequences of a severe accident (the power unit after such accidents does not return to operation) includes brining the emergency power unit to a safe state, processing a large amount of resulting liquid radioactive waste (i.e. the water of the emergency tanks of the containment for cooling the fuel), and development of a long-term project for mothballing the suffered power unit.

The decision to introduce the emergency state (announcing the state of "emergency situation") is to be adopted by the Director (or Chief Engineer); in case of their absence is made by the NPP shift supervisor. After announcing the emergency situation at NPP, the emergency operations shall be managed by the General Director of the Belarusian NPP who performs as the Emergency Response Supervisor (ERS) with a help of the Commission for Emergency Situations of NPP (NPP CES). Members of the NPP CES shall participate in activities of the prior nominated emergency response teams to identify causes of deviation of the normal operation mode, assess the situation, forecast potential radiation consequences, and work out proposals for normalizing the situation.

Centralized assistance during a NPP emergency is carried out by the State Emergency Prevention System. The Republican Special Operations Detachment of the Ministry of Emergency Situations of the Republic of Belarus will provide the relevant assistance in the emergency situation for mitigation of the NPP emergency situation consequences.

Prior to shipment of the nuclear fuel to NPP, the documents including the “Action Plan for Protection of Personnel in the Event of Accidents at the Belarusian NPP" and the "Protective Measures against the Radiation Accidents at the State Company "Belarusian NPP" (i.e. the in-house and external emergency plans) shall be developed and enacted. These documents address the radiation consequences of BDBAs. At the same time will be established the main and backup communication channels with the relevant Ministry, State Regulatory Authorities and permanent management bodies authorized ad-hoc to solve problems related to protection of the population and territories during emergencies.

These plans are the main guiding document for implementation of protective, organizational, engineering and technical, medical, preventive, and other measures in the event of an accident in order to protect the personnel of the NPP and the population, confine and mitigate an accident.

The in-house emergency plan is developed by the NPP operating organization on the basis of the results of the analysis of BDBA with the worst consequences for the personnel and population; this plan addresses also phases of development of the accident. The plan provides for coordinated activities of NPP with external organizations, members of the State System for Prevention and Mitigation of Emergencies.

The in-house emergency plan shall be submitted to the Supervisory Authority as part of a set of documents that substantiate assurance of the nuclear and radiation safety and help obtain the power unit operation license.

The NPP management is responsible for maintaining the permanent readiness and implementation of the plan.

The external emergency plan shall be developed by the Ministry of Emergency Situations with the participation of the Ministry of Natural Resources and Environmental Protection, Ministry of Health, Ministry of Internal Affairs, State Security Committee, local government and self-government bodies. The plan outlines coordination of the activities of the facility-based, departmental and territorial units that are incorporated into the resources of the State System of Prevention and Elimination of Emergencies.

In-house and external emergency plans are interlinked regarding a timely notification of a potential or actual hazard of an accident, the volume and frequency of the transmission of the current information, as well as in coordination of actions and mutual assistance in the implementation of the activities.

The Belarusian NPP has a training center equipped with the necessary simulators, techniques and training materials for training and exercising the personnel in emergency situations.

7.1.2 Capability to Use the Available Equipment

Using mobile units

In accordance with the recommendations based on the results of the development of the Stress Test Report (target reassessment of safety), for the Belarusian nuclear power plant consisting of two power units the design shall provide two mobile DG sets (one DG set per each power unit) with a capacity of 500 kW. The mobile DG sets can be operated under an ambient temperature range of -50 to +41 C^{o}.

Within 24 hours the mobile DG set shall be delivered to the place of its connection and get ready for operation. The connection point of the mobile DG set to the switchgear of the 7th channel is a cabinet (I category of seismic resistance according to NP-031-01, dust-proof-and-moisture-proof version IP54, UHL1, vandal-proof, with a lock), located on the outer wall of building UJE at elevation +1.400. If it is impossible to ship the mobile DG set, connection can be provided from its original location.

The mobile DG set is controlled and monitored directly from local control panels located on this equipment.

Makeup of SG PHRS tanks and the spent fuel pool is provided by a high-pressure pump of the PHRS tank makeup system. The pump unit is located in the steam chamber and connected to desalinated water tanks of system LCU.

The emergency heat removal tanks must be replenished from LCU tanks before depletion of water (72 hours from the beginning of the emergency process). After this, the SG PHRS tanks are re-emptied and the desalinized water tanks of system LCU are refilled with water. In order to further maintain the stable and safe state of the reactor plant, maintaining also operability of SG PHRS, it is necessary to periodically makeup tanks LCU from any sources of water available at the NPP site using an off-site mobile equipment (for example, from fire water storage tanks).

On the territory of the Belarusian NPP a fire-fighting and rescue unit with a total number 116 men has been established. This unit is equipped with 12 fire-fighting and rescue vehicles, including cars for chemical and radiation protection and reconnaissance, as well as cars intended for supplying fire extinguishing agents and putting out fires. Fire-fighting and rescue unit # 2 of the Ostrovets Regional Emergency Department for protection of the Belarusian NPP facilities was put into operation in February 2016.

Of 116 people, 29 are middle and senior command staff (officers). Every day 30 people go on alert duty using 6 units of basic, special and auxiliary machines. Depending on the emergency situation, rescuers can use various types of technical equipment for emergency situations. There are three duty shifts, with a shift duration of 24 hours. In accordance with the republican plan for mobilization of resources and manpower for mitigation of an emergency situation, up to 47 machines can arrive from the neighboring units of the Ministry of Emergency Situations of the Republic of Belarus with a personnel and relevant equipment.

The combat team of fire-fighting and rescue unit-2 has the following vehicles:

- Firefighting truck on chassis MAZ 6317X9 AC 8.0-40;
- Air and foam extinguishing vehicle on chassis MAZ-6317x9 AB 8,0 50 (6317);
- Emergency and rescue vehicle in configuration of chemical and radiation reconnaissance ERV (Mercedes);
- Vehicle with a ladder with a lifting height of at least 50 m, DL.CS5 CS,
- Firefighting truck on the chassis MAZ 530905 AC 5, 0-50 / 4;
- Firefighting truck on the chassis MAZ 530905 AC 5, 0-50 / 4;
- Pump and hose vehicle on the chassis MAZ -6317x5 AHP-133 (6317);
- Emergency and rescue vehicle in configuration of the emergency and rescue vehicle ERV (Mercedes).
- Firefighting truck on the chassis MAZ 530905 AC 5, 0-50 / 4;

- Powder extinguishing truck on chassis MAZ-6317x5 AP 5000 (6317),
- Automotive sky lifts MAZ-6516V8-555-001;
- Carbon dioxide extinguishing trailer.

All buildings and facilities of the nuclear plant can be accessed by roads with a roadway of at least 5 m; the design provides for entrances for transport cars, fire trucks and sidewalks for the personnel. The power units are surrounded by circular roads with a roadway width of 7.00 m.

The design provides for two automobile entrances to the NPP site. The main entrance is from the north side of the NPP site. The reserve entrance is from the construction and installation base located near power unit № 2. The railway entry is provided from the direction of the construction and installation base.

The distance from the fire-fighting and rescue unit building (intended for protection of the Belarusian NPP facilities) to the territory of the construction and installation base is 100 meters.

Supply of water for fire-fighting purposes of NPP buildings and facilities is provided by the internal and external fire-fighting water supply systems (internal and external fire water pipelines, spray pools, etc.). The volume of water in 4 spray pools is 144 000 m^3. The NPP site has a circular network of external fire-fighting water supply, consisting of 60 fire hydrants.

The buildings of NPP will use automatic modular fire-fighting systems with a finely dispersed spray, automatic gas fire-fighting systems, automatic water extinguishing systems for the main buildings and facilities of the power unit.

On the territory of FERU-2 there is a fire reservoir with a volume of 50 m and a network of fire hydrants.

In an event of an emergency situation, the personnel of the power unit shall be sheltered in a protective building for 60 people.

The main source of cooling water makeup for essential loads with spray pools is the water from the river Vilya.

The drainage basin of cooling tower URA and the water intake chamber in the pumping station of the consumers of turbine building URD serve as a source of reserve makeup of the spray pools. The volume of water in the drainage basin of the cooling tower and the water intake chamber is 23600 m^3.

In case of failure for a common reason of the main and backup makeup sources of essential loads system makeup - for example: SSE - the capacity of each spray pool ensures operation of the system without replenishing for at least 8 days. The water inventory in each spray pool provides a capability for emergency cooldown during the hot five-day period of the 10% occurrence (i.e. with a repeatability of once in 10 years) and transfer of the power unit into a "cold" state. During this time, the main makeup system must be restored, or technical arrangements are to be engineered to supply the spray pools with a make-up water from another source (for example, from a fire-fighting and production water supply reservoirs UGF with a total capacity of 2400 m^3 ; from water storage reservoirs for household and drinking water supply UGG with a total volume of 912 m^3 ; or from water wells for Civil Defense Shelters UZM, the latter for a special period, with a capacity of at least 36 m^3/h).

Spray pools of power unit No. I and power unit No. 2 are not connected with each other; consequently, shutdown of one power unit does not influence in any way on operation of the spray pools of the other power unit. But, if necessary, water can be supplied by mobile vehicles from the spray pool of the companion power unit.

Providing resources and supply management

In terms of accident management, delivery of resources (fuel for the diesel generators, water, etc.) will be carried out within the framework of the SAMG. Relevant activities will also be provided for emergency planning.

Radioactive releases

In case the level of external effects exceeds the threshold values, the state of the NPP systems located in different buildings (except the reactor building) is not predicted. For regimes that are accompanied by loads exceeding threshold values, taking into account the existing reserves in the reactor building, the integrity of the containment systems presented in Section 6.3.9 of the Report on the targeted safety reassessment (stress tests) of the Belarusian NPP is ensured, and radioactive emissions are prevented.

The management of radioactive emissions and their limitation under seismic impacts, floods, extreme weather conditions, and loss of power supply are described in the relevant sections of the Report on the targeted safety reassessment (stress tests) of the Belarusian NPP. Measures to limit emergency emissions are also considered in section 6.4.1 of the Report on the targeted safety reassessment (stress tests) of the Belarusian NPP.

The Automated Radiation Situation Monitoring System (ARSMS) includes facilities for on-line monitoring and forecast of the radiation situation after an accident; ARSMS includes instruments of meteorological observation and radiation monitoring, it is equipped with a dedicated software for simulation of spreading of radioactive emissions in the atmosphere and forecast of population exposure doses.

Based on the results of radiation monitoring and the forecast of the radiation situation, decisions are adopted to introduce protective measures for the personnel and public in accordance with the criteria set forth in the *Action Plans for the protection of personnel and population in the event of an accident*.

The actions of operational personnel to mitigate consequences of the accident are set forth in the operational instructions and emergency procedures, including:

- "Technological Regulations";
- "Instructions for Mitigation of Accidents";
- "Guidelines for Management of Beyond Design Basis Accidents".

To avoid an overexposure and anyway reduce personnel exposure doses, the design provides for arrangement of an efficient sanitary and access mode, use of personal protective equipment (PPE), dosimetry and radiometric control, thorough planning of work with regard to its maximum duration, shelters for the maximum contingent of the shift personnel, shielded emergency control post.

To ensure the personnel permanent readiness for actions during accidents, are developed a procedure and curriculum for training and conducting regular emergency response drills.

Reactor plant operator announces a class of emergency basing on a previously introduced Emergency Action Levels. These Emergency Action Levels are developed by the Management of the Belarusian NPP in accordance with the approaches set forth in the IAEA GSR documents Part 7 "Preparedness and Response for a Nuclear or Radiological Emergency", GSG-2 "Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency".

The general response criteria for implementation of protective measures to prevent deterministic effects and reduce the risk of stochastic effects are applied as an emergency response to nuclear or radiological emergencies at NPP. The general response criteria for carrying out protective measures in an event of emergency situation are established by the Hygienic Standard "Radiation Exposure Evaluation Criteria" approved by the Decree of the Ministry of Health No. 213 of December 28, 2012. Forecasted exposure doses for the whole body, thyroid gland and fetus serve as criteria for protection of the population during a radiation accident. These values are established based on a presumption that the reference level of the residual dose of population exposure must not exceed 100 mSv. The general response criteria established by the hygienic standard meet requirements of the IAEA GSR Part 7 "Preparedness and Response for a Nuclear or Radiological Emergency", GSG-2 "Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency".

Communication and information systems

The BelNPP communication and warning system is designed as part of the Integrated Communication System for Operations Management and Emergency Response and is based on its structure and the structure of the operational and technological management of the NPP, as well as the requirements for the organization of personnel management and its prompt warning in emergency situations.

Composition and scope of equipment of communication and warning of BelNPP, as well as their characteristics are determined in accordance with the guidance documents requirements in the chapter relating to NPP’s communication and warning.

The BelNPP communication and warning system is designed to provide efficient, reliable and sustainable operational and administrative management in daily operations and emergency situations; receiving and sending centralized warning signals to the personnel; timely notification on occurrence of emergency situation of NPP personnel and Civil Defense forces , including the relevant duty services, as well as the population, managers and personnel of facilities located within the coverage area of a local warning system.

The system of communication and warning of NPP includes complexes of external and internal communications.

Organization of external communication of BelNPP

To ensure survivability of the system of external communications in an event of man-made impacts which may be potentially accompanied by earthquakes, the system includes independent main and backup communication channels.

The BelNPP external communication system includes the following technical facilities:

- above-ground transmission systems,
- truncking radio communication (based on the TETRA digital protocol);
- radio communications with the emergency action units (specially designated radio facilities);
- local alarm system;
- dispatcher communication system with power system facilities.

To provide communication during emergency situation, the BelNPP design provides for two geographically separated communication nodes in the shielded emergency response control post:

- NPP shielded emergency response control post located at the NPP area,
- City shielded emergency response control post located a city close to NPP.

Systems of shielded emergency response control post communication facilities are interconnected by communication channels (main and backup) on the basis of a high-speed digital transport network and fiber-optic communication lines.

All shielded emergency response control posts have communication channels (main and backup) allow the NPP personnel to enter the public telecommunications network of the common telecommunications network of the Republic of Belarus, as well as other special communication networks.

Communication channels that ensure safety of NPP operation are located at the communication nodes of the shielded emergency response control post with a capability of switching them to the common NPP communication center (the central hub of the common plant communication).

The external communication system includes also digital channels based on fiber-optic cable lines for dispatcher communication with power system facilities.

Wire communication channels are used as the main channels ensuring the safety of NPP operation and control of emergency response; satellite communication channels and radio channels are used in case of damage to the above-ground channels following an earthquake, in special cases and during insufficient performance of the above-ground terrestrial channels.

The area of coverage of the territory adjacent to the BelNPP with the trunking radio communication is up to 30 km, and the area of reliable coverage is limited by the NPP surveillance zone.

The communication channel that provide the emergency management has the highest priority when transmitting and receiving information for emergency response actions.

Systems of communication facilities of the shielded emergency response control post ensures operability in case of failure of the common plant and power unit's communication nodes.

Basis of organization of BelNPP internal communication

BelNPP internal communication system includes operational and common plant components.

Internal common plant communication is intended for administrative and economic management of NPP and communication between the personnel of various services (workshops) of NPP; it serves also for retransmitting the radio broadcasting, announcing the exact time, providing documentary information exchange , and TV monitoring of the main process equipment (including power unit's main equipment).

Internal operational communication is intended for transfer of commands of the operative personnel of the main management units (boards) to the subordinate operational personnel, maintain communication, make announcements, seek contact with the needed personnel; the transmitted information (including video of actions of the operating personnel) can be automatically recorded.

To implement the above types of communication at the BelNPP, the design provides for systems and networks of internal communication, consisting of technical devices and software packages, channels and communication lines that provide functioning of the dispatching, emergency response and administrative management at the NPP.

The internal common plant communication includes six systems that consist of the technical devices common at NPP;

- Common plant telephone system;
- Master clock system.

7.1.3 Assessment of Factors Potentially Complicating Accident Management and related Contingencies

7.1.3.1. Impact on presence of personnel in MCA and ECR

Life support systems cover technological systems, additional equipment, supplies and instructions provided at the plant create safe normal conditions under which operators can operate the power unit and also maintain it in a safe state under extreme and emergency conditions, including loss-of-coolant accidents .

Life support systems must support the power unit in a safe state under extreme and emergency conditions, including loss-of-coolant accidents.

The main control room (MCR) is designed to support its operation under normal operating conditions of the power unit, anticipated operational occurrences, and accidents accounted for in the design. In case MCR cannot function due to a damage caused by an effect not considered in the design (such as sabotage, military actions, etc.), an emergency control room (ECR) is provided at a distance from MCR. MCR allows to provide an independent monitoring and control of the most important functions that are necessary to carry out a shutdown and complete cooldown of the reactor, and confinement of radioactive substances.

The personnel life support systems of MCR (ECR) shall meet the following requirements:

- ensuring conditions permissible for operation of the on-duty personnel;

- protection of the personnel from external penetrating radiation;
- protection of the personnel that is engaged in mitigation actions following a radiation accidents at the power unit or after appearance of toxic, chemical and radioactive substances in the outdoor air;
- protection of personnel in case of fire.

Systems supporting operators' activities in MCR (ECR) in all design modes include:

- ventilation systems;
- radiation protection system;
- fire protection system.

MCR (ECR), Personnel Life Support Systems of MCR (ECR), as well as their controlling and supporting systems are located in the management building, which is referred to Category I of seismic resistance in terms of nuclear and radiation safety.

The systems are reliably protected by the building structures from external design impacts, including flying items, shock wave, and seismic impacts.

The MCR / ECR life support system is equipped with an efficient plenum air purification system on the basis of aerosol and iodine filters. The building structures of double containment and management building UCB ensure a permanent stay of the personnel at the MCR / ECR to manage an accident.

The Safety Analysis Report of the Belarusian NPP demonstrates that the exposure of the personnel in the premises of MCR / ECR following severe design-accounted accidents will not exceed the target limit of the effective equivalent dose of 25 mSv, which is defined for the Belarusian NPP project. This value will not be exceeded during the entire time of the accident and mitigation of its consequences.

The radiation situation at the site allows short-term access of the personnel with use of personal protective equipment for skin and respiratory organs; the new personnel will assist and replace the personnel of the previous shift. To reduce the personnel exposures when moving across the NPP site, the design provides for special vehicles.

In case the personnel cannot stay in MCR, the design provides for relocation into ECR.

In an event of damage or complete destruction of MCR and ECR, the accident management activities can be carried out from the power plant’s shielded emergency control post.

7.1.3.2. Adverse Working Conditions caused by High Radiation Levels and Contamination, Destruction of Buildings and Structures.

In case of damage or destruction of NPP buildings and structures, the power plant’s shielded emergency control post premises may become inaccessible to the personnel (or working their becomes impossible). The city shielded emergency control post , or other crisis centers (for example, the Ministry of Emergency Situations of the Republic of Belarus) can be used as an alternative place to accommodate the emergency works management.

In case of loss of the NPP shelters, it will be necessary to evacuate an excessive personnel from the NPP site and organize rotations of necessary specialists (with necessary personal protective equipment and special vehicles). At the same time, the control of individual radiation control must be arranged, in full compliance with the current legislation of the Republic of Belarus regarding emergency recovery activities.

7.1.3.3. Potential Failures of Measuring Instruments

In case of severe and beyond design basis accidents the design provides for application of special instrumentation and implementation of organizational measures. In case of severe and BDBA accidents, control of integrity of the containment is performed from the main control room. The information received by I&C regarding integrity of the containment is displayed at panel CWL01 (the segmented panel of post-accident monitoring).

Sensors monitoring containment integrity are located in reactor building UJA and in annulus UJB. The sensors of different channels and relevant soft & hardware complex devices are located at different rooms. The soft & hardware complex devices are located in APCS facilities rooms, the segmented panels are located at MCR in control room building UCB.

All the containment integrity monitoring equipment is of category 1 of seismic stability.

The containment integrity sensors are designed to measure an appropriate range of changing parameters and are resistant to adverse impacts of BDBA.

Temperature inside rooms containing instrumentation for monitoring the containment integrity shall not exceed the admissible instrumentation operation values. An operator controls the containment integrity by means of the following I&C devices and signals displayed at MCR panels:

1) Indications of status of the containment penetration isolating valves (closed/opened).

2) Indications of the air lock automated tightness control system (inner/outer hatch is not leak tight, the air lock automated tightness control system is disabled or malfunctions).

3) Indications of the automated radiation monitoring system (ARMS) for measurement of the gamma radiation dose rate inside the containment which allow to assess the degree of fuel damage (complete melting. The design provides for four detectors to measure the gamma radiation dose rate inside the containment; these detectors have operation capability also during the worst DBA phase. The detectors are of category 1 of seismic stability and safety class 3. Three detectors are powered from the relevant channel of the emergency power supply system (EPSS). One detector is powered from the BDBA power supply system . The measuring range of these detecting devices is 10^{-3}-10^{5} Gy/h. The detectors perform reliably at a temperature of +180 °C during 2 days, and at a temperature of +205 °C within 30 minutes.

4) Indications of ARMS sensors of the inert radioactive gases (IRG) volume activity in the containment annulus atmosphere, which allow to assess the containment leakage with operability of annulus ventilation system KLC11/21/31/41 (four channels of the system are powered from four EPSS channels).

5) The indications of ARSMS sensors located at the NPP site which allow to assess the extent of the accidental emission from under the containment into the environment, provided they remain functioning (powered from the safety-related power supply system of NO).

The indications of status the air lock personnel hatch and transport air lock hatch are also displayed at the automated workstation of the unit upper level control system (inner/outer hatch is leak tight or leaking, the inner/ outer door is open/ closed).

Measurement of overpressure in the containment during the emergency modes is carried out by 8 sensors.

Hydrogen concentration control system JMU controls the hydrogen explosion safety.

The hydrogen concentration control system consists of two similar measurement channels - JMU10 and JMU20 - which are completely independent from each other.

Each channel of the hydrogen concentration control system includes eight gas analyzers of hydrogen and oxygen combined with temperature sensor WS85PLUS, two pressure sensors DAE-100T and hardware & software complex “Station of data processing of hydrogen concentration control system ” .

The number of control points and location of sensitive elements of the hydrogen analyzers are selected on the basis of the analysis on hydrogen distribution, accumulation and modes of potential inflammation inside rooms under the containment.

The measurement range of hydrogen concentration is from 0 to 25% , by volume.

Functioning of components of the emergency parameters monitoring system is assessed and qualified according to the inspection, verification, and test procedures. The systems that ensure monitoring of parameters during a severe accident are resistant to BDBA conditions.

7.1.3.4. Practicability and Efficiency of Measures to Control Accidents under External Impacts (Earthquake, Flooding)

To ensure implementation of safety function, the design provides for development of strategies and procedures aimed at prevention of nuclear fuel damage and minimization of consequences if this damage occurs (including the Emergency Operation Procedure (EOP), Beyond Design Basis Accident Management Guidelines (BDBAMG), Severe Accident management Guidelines (SAMG)). Due to application of the symptom-oriented approach towards actions in emergency situations, applicability of these instructions and guidelines is not limited by external conditions.

According to the requirements, anti-emergency training, emergency response drill and fire-fighting drills are carried out in order to:

- train skills on how to act under conditions of operation occurrences, at pre-emergency conditions, and NPP accidents;
- train skills of the shift personnel interaction, interaction with fire-fighting, medical, civil defense brigades and other relevant teams;
- teach the staff on how to prevent accidents, reduce their escalation, notify about accidents, and mitigate consequences;
- check the skills of pre-medical assistance and use individual and collective protection devices, fire extinguishing devices, etc.;
- drill an arrangement of personnel evacuation;
- check the personnel readiness to act in an independent, fast and correct manner;
- drill interaction with the management of the operating organization following announcement of the "Emergency readiness" / "Emergency situation" status.

The senior management of the Republican Unitary Enterprise "Belarussian Nuclear Power Plant" and NPP operations shops must be stuffed with a qualified and experienced personnel with high and/or secondary vocational education in the respective area and related spheres of knowledge and also with work experience in the respective area.

In order to conduct the personnel education and emergency training, "The common - plant set of anti-emergency trainings for operational personnel" is under development "(BLR1.E.534.0.&&&&&&.&&&&&.022.PZ.0001).

These documents, in particular, will focus on assessment of equipment condition and extent of its damage as a result of a seismic impact, fastest recovery of power supply both from the power plant and from the off-site sources, connection and use of mobile electric power generating units. In case of difficulties with access of the shift personnel to the pant and delivery of equipment (destructed road, etc.), the manpower and resources of Ministry of Emergency and the Ministry of Defense will help to restore the transport communication.

7.1.3.5. Availability of Premises to Manage Accidents

To provide control of manpower and resources of mitigation activities, stationary control posts are established and kept in constant readiness at the Belarussian NPP. The stationary control posts include the shielded emergency response control posts of the nuclear power plant (NPP SERCP) and the shielded emergency response control post of the town (T SERCP). All these control posts are equipped with independent engineering life supporting systems for working under conditions of radiation and chemical emergency; they are also equipped with information systems, software & hardware complexes and communication devices, data transmission system; supplied with necessary technical documentation and office equipment for operations of the emergency response participants (the Commission on Emergency Situations (CES) of NPP, the Emergency Response Team, the Nuclear Power Plant Emergency Response Team (NPPERT).

NPP SERCP is arranged in a separate standing shelter (coordinates 01UYX at the NPP plan: 11A, 6B) with a capacity of 100 people. It is designed to manage the NPP divisions and all forces involved in emergency activities during localization and mitigation of the consequences of emergency.

T SERCP is established in a sheltered structure (with a capacity of 100 people) at the territory of the Ostrovets town. It is intended for management of NPP divisions and the forces that are involved in emergency activities in case the current radiation situation at the NPP site does not allow to provide control from NPP SERCP. T SERCP is equipped with workstations for representatives of local authorities, emergency response teams of various institutions and organizations participating in the emergency activities according to the established procedure.

NPP SERCP and T SERCP are supplied with:

- personal protection equipment;
- medical stock (including iodide potassium);
- food stock, for 5 days.

Crisis centers located in NPP SERCP and T SERCP are stationary information & control centers of anti-emergency planning and emergency response, they are linked with each other organizationally. NPP SERCP and T SERCP equipped in the same way.

The transfer of the control function of the emergency actions from NPP SERCP to T SERCP is carried out by a relevant order of the Emergency Response Supervisor (Director of NPP or a person in charge).

By the same order the information support of the emergency recovery activities is transferred from NPP Crisis Center to the Town Crisis Center. The Town Crisis Center is equipped in the same way as the NPP Crisis Center, and does not require any special commutations for transfer of the control function.

7.1.3.6. Significant Infrastructural Damage or Flooding Preventing Access to NPP Site

According to the design, the NPP site is not subject to flooding during the high waters season, floods, jams and waters under the snow.

However, in case of infrastructure disruption, the Belarusian NPP site is self-contained. It means that NPP can be transferred to a safe state in a self-contained manner and maintained in this state during the following periods of time:

1. In the mode of failure of the safety systems and systems-related systems (including failure of the supply systems located below 0.00, and in case of loss of the external power supply):

- Reactor Plant: the residual heat is removed by BRU-A or SG PHRS for at least 72 hours;
- Spent Fuel Pool: the residual heat is removed by means of boiling of the water (in case of failure of system FAK as a result of a flooding) for at least 41 hours;

2. In the mode of NPP blackout: operation of a power unit fed by diesel generators is reliably provided within 72 hours due to the following reasons:

- each DG has its own autonomous auxiliary systems;
- each DG has tanks with the diesel fuel stock that ensures EPS DG function for 53 hours at the nominal power rate, and NO DG function for 29 hours at the nominal power rate;
- levels in the feed tank and the stock tank are displayed at the control panel;
- the lube system is designed to operate in a self-contained manner for at least 240 hours.
- the design provides for storage of an additional stock of diesel fuel at the diesel fuel central warehouse (00UEJ) in the amount of 1160 m^3 , (290 m^3 for DG of one channel od emergency power supply of one power unit). This amount provides for additional 7 days of operation of one DG of one EPS of one power unit. This calculation is made for consumption of 204 g / kWt·h per one DG).

The design provides for replenishment of tanks with fuel from tanker trucks and/or through the pipelines from the warehouse of diesel fuel stock into the feed tanks; provision is made to ensure quality of fuel and accurate intake of relevant amounts. If it becomes impossible to restore the power supply of NPP auxiliaries during 48 hours (and return DGs to the standby mode) in the mode of NPP blackout, the design provides to replenish the main and intermediate warehouse with diesel fuel of the required quality from the regional oil product depots by means of oil tanker trucks.

7.1.3.7. Communication System Failure

Reliability of the external and internal communications are considered in section 7.1.2.

7.1.3.8. Impact of Companion Nuclear Units on Each Other and Sufficiency of Trained Staff to Manage Multiple Accidents

According to the design, each power unit of the Belarussian NPP is self-contained, that also means that one power unit does not influence another one.

The number of trained personnel is sufficient to carry out simultaneous accident control at two power units.

7.1.3.9. Loss of Power Supply

Efficiency of the design measures in case of a loss of electric power supply is considered in section 6.1.1.

7.1.4 Conclusion on Conformity of Organizational Measures on Accident Management

To ensure the 4th level of the defense-in-depth, comprehensive activities are under way at the Belarussian NPP to establish the beyond design basis accidents (including severe ones) management system. The functioning and links between the accident control systems and the emergency preparedness are based on the solid background of administrative and engineering measures, commitment of the staff to achieve high level of safety culture, and personal responsibility, too. The plans on timely employment and training of highly qualified human resources in sufficient amount are approved and carried out. According to the decision of the Council of Ministers of the Republic of Belarus the system of situation-dependent and crisis centers is established.

The instructions for accident mitigation and the guidelines on management of beyond design basis and severe accidents are under development.

The training center equipped with necessary simulators, techniques and training materials for staff training in emergency situations is established at the Belarusian NPP. The design provides for capability to manage the severe beyond design basis accidents from the main control room , emergency control room, and the NPP crisis center. The design provides for necessary and sufficient measures to support the survivability and self-sufficiency of these facilities. The necessary measures regarding the beyond design basis accidents control and emergency readiness and response are provided. The strategy of management of beyond design basis accidents and minimization of their consequences is detailed in the Manuals on the Beyond Design Basis and Severe Accidents Management.

As part of BDBAMG and SAMG, procedures are drafted for the power unit shutdowns (including the dismantled reactor head), and management of accidents caused by the fuel damage in SFP. The measures of emergency response at the site and beyond its boundaries are foreseen, deployment of manpower (from outside NPP) and resources for minimization of consequences of the severe accidents are provided.

The information on personnel education and training, the scheduled trainings and the specialists and anti-emergency trainings are provided in “The set of programs of anti-emergency trainings for operational staff”, BI.RI F. 534 &&&&&&&&&&& 022 P7.0001.

The design of the Belarusian NPP provides for necessary and sufficient amount of measures which will be developed as part of BDBAMG and SAMG. The civil defense facilities are located according to the requirements of normative reference documents of the Republic of Belarus and with due account for the normative radius of concentration of the shielded men in the places of the greatest personnel concentration and the damaged areas.

Each shelter is equipped with extra and emergency exits. Debris removal at the evacuation pathways and the protective constructions entrances clearing are carried out by the available means and means of joint NPP rescue team, and also by the engaged forces and forces of rescue and other emergency actions.

Thus, the applied efforts to provide operability of 4th and 5th levels of the defense- in-depth protection are sufficient.

7.1.5 Measures to Improve Capabilities for Accident Management

Despite availability of diverse systems to implement each strategy of accidents management , still there are areas which require further improvement of measures of beyond design basis accidents management (including severe accidents, too). Incorporation of additional engineering measures and instructions to provide the safety functions in case of the design systems failure will enhance the NPP capabilities to manage development of the beyond design basis accidents at a severe stage of their progression.

In terms of administrative issues related to management of severe accidents: after determination of a complete list of the initial events of the severe accidents according to the probabilistic safety analysis results, it becomes possible to further develop BDBAMG and SAMG for all statuses of the power units, including the reactor with removed head, and the management of accident in case of fuel damage in SFP. The final objective of these measures is to ensure cooldown of the reactor core and the spent fuel in SFP, as well as prevention of radioactive emissions. A regular update of developed EOP, BDBAMG and SAMG is also feasible.

In order to improve measures of beyond design basis accidents management, a number of actions is recommended, for example:

- organizational measures for a more effective usage of available capabilities or determination of additional measures - so-called crisis plans - their material and staff support for management of unforeseen situations, which, nevertheless, can hypothetically arise at NPP. In particular, the simultaneous impact of several factors over the entire NPP site, the situation-dependent and crisis centers and control centers failure, the communication announcement systems failure, the forced need to carry out high risk measures and decisions, etc.;
- implementation of additional engineering measures to ensure administrative support of accident management measures, in particular, availability of resources of various civil structures and organizations outside the plant, fire-fighting equipment, civil defense shelters and facilities suitable for allocation of crisis centers ;
- development and implementation of additional measures for the long term stability of communication channels and interaction between different components of systems to manage accidents and emergency responding, both at the NPP site and outside it.

7.2 Accident Management with Loss of Ultimate Heat Sink at Various Stages of the Accident

This section provides general information on organization of accident management and elimination of accident consequences with loss of ultimate heat sink.

7.2.1 Accident Management with Loss of the Ultimate Heat Sink before Beginning of Nuclear Fuel Severe Damage

The Belarusian NPP design considers measures for managing beyond-the-design-basis accidents. At the stage of development before the beginning of the accident severe phase, identified by exceeding of the temperature above the core, a certain limiting value, the management measures are regulated by the BDBA Management Guidelines. BDBA management activities before the start of nuclear fuel severe damage shall be aimed at:

- ensuring the reactor subcriticality (quick shutdown and maintaining of the reactor core in the subcritical state). The maximum temperature of re-criticality when all the CPS CRs (except for the most efficient one) are inserted at the end of the campaign at steady-state poisoning without boric acid in the coolant is given in [31].
- reactor plant cooldown;
- ensuring reliable heat removal from the core and the fuel in the spent fuel pool during the accident as well as after stabilization of the parameters in the post-accident state;
- ensuring integrity of the primary circuit system (protection from overpressure, hydraulic shocks and thermal loads);
- ensuring integrity of the containment;
- minimizing the release of radioactive substances.

To ensure the performance of the safety function - core cooling, the operator action strategy shall be focused on:

- ensuring the long-term serviceability of SG PHRS by maintaining the water reserve in the tanks of the system;
- in case of depletion of the water reserve in the PHRS tanks, heat can be removed from the primary circuit by evaporating water from the SG using BRU-A or SG POSV;
- protection of the primary circuit against overpressure, as well as pressure decrease in the primary circuit, which ensures the supply of boric solution to the reactor from the ECCS tanks, due to the operation of the pressurizer POSV and the emergency gas removal system;
- after depletion of the boric solution reserves in the ECCS tanks and if the reactor makeup by alternative sources is impossible, the pressure in the primary circuit is reduced to 1.0 MPa and less due to the operation of the pressurizer POSV and the emergency gas removal system in order to ensure the conditions for reliable cooling of the molten core material in the molten core catcher.

To prevent the transition of the accident to a severe stage, the main task at this phase of the accident is recovery of the safety function - cooling of the core.

The application of passive safety systems in the design increases the NPP reliability, because only uncompensated leakages of the primary circuit as the initiating events can lead to the accidents with core damage. Failed heat removal from the secondary circuit in case of SG PHRS failure is a rather unlikely event that reduces the probability of an accident by 3 orders of magnitude. Thus, the main task at this phase of the accident development is the restoration of the coolant reserve in the reactor core.

The water is supplied to the reactor core until the time when the transition of the accident to a severe phase is diagnosed.

In the Belarusian NPP design it is assumed that the transition of the accident to a severe phase of its development occurs when the temperature reaches + 650 ° C (according to preliminary estimates) above the core (the limiting value). From this moment, the tasks of operational control are changing. The tasks of compensation of the primary circuit coolant loss and restoration of the core cooling function are canceled; the operator’s actions are aimed at ensuring that the containment performs its localizing functions. The operative management objectives after the onset of this phase of the accident are to monitor the accident process, operation of the equipment involved in severe accident management and adjust the equipment operation, if required.

Exceeding of the specified temperature leads to further heating of the nuclear fuel, its temperature rise to the values at which an intense steam-zirconium reaction begins as an additional heat source and destroys the fuel.

To monitor the coolant temperature, as well as reactor (core) temperature in an emergency condition, in-core detector assemblies are used, where temperature is measured with thermoelectric cable converters.

DBA management is regarded as:

- event-oriented procedures of the personnel actions to eliminate accidents;
- symptom-oriented procedures of the personnel actions to eliminate accidents (as a part of optimal recovery procedures).

The event-oriented procedures are applied when reactor emergency protection and\or safety systems are actuated or conditions for their actuation appear and before failure of the critical safety functions.

The symptom-oriented procedures are applied after actuation of the reactor emergency protection and\or safety systems or appearance of conditions for their actuation and before failure of the critical safety functions, but only in the following cases:

- the operating personnel failed to determine which event-oriented procedure shall be applied;
- overlapping of initiating events occurred and the operating personnel failed to determine which event-oriented procedure shall be applied first of all;
- application of the event procedures does not lead to expected results.

The purpose of the personnel step-by-step actions is to monitor the design behavior of the mode, algorithm of the automatic equipment operation, diagnose a fault or an accident and transfer of the Unit to a safe state, and in case of automatic equipment failure these actions are performed remotely from MCR (ECR) or manually on site.

The location and actions of the shift operating personnel are determined by the duty and operational instructions and orders of the operational manager. The diagram of the operational subordination of the duty personnel of the shift of the Belarusian NPP is shown in Figure 7.1.1.1.

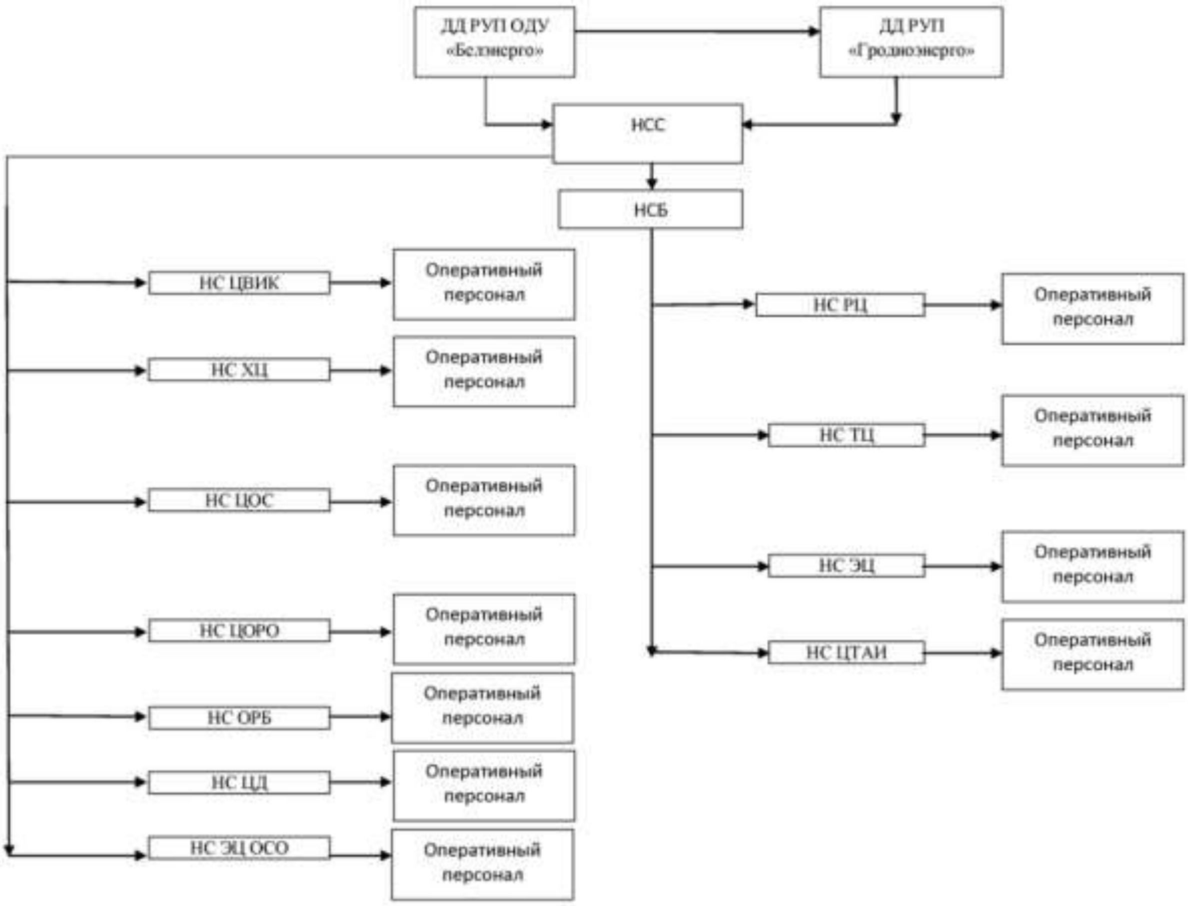

Figure 7.1.1.1. - Diagram of the operational subordination of the duty personnel of the shift of the Belarusian NPP

ДД РУП ОДУ «Белэнерго»	Duty dispatcher of the Republican Unitary Enterprise Operational Dispatcher Department “Belenergo”
ДД РУП «Гродноэнерго»	Duty dispatcher of the Republican Unitary Enterprise “Grodnoenergo”
оперативный персонал	Operational personnel
НСС	Head of the shift of the NPP
НСБ	Head of the shift of the NPP unit
НС ЦВИК	Head of the shift of the NPP ventilation and conditioning shop
НС ХЦ	Head of the shift of the chemical shop of the NPP
НС ЦОС	Head of the shift of the supplying NPP systems shop
НС ЦОРО	Head of the shift of the NPP radioactive waste management shop
НС ОРБ	Head of the shift of the NPP radiation safety department
НС ЦД	Head of the shift of the NPP decontamination shop
НС ЭЦ ОСО	Head of the shift of the NPP electrical shop for general station facilities
НС РЦ	Head of the shift of the reactor shop
НС ТЦ	Head of the shift of the NPP turbine shop
НС ЭЦ	Head of the shift of the NPP electrical shop
НС ЦТАИ	Head of the shift of the NPP thermal automation and measurements

Shift acceptance and transfer before restoration of the reactor plant safe condition or the appropriate order of the operational manager is prohibited. The shift personnel is put at the disposal of the operational manager and follows his instructions.

Further actions after transfer of the reactor plant to a stable safe state are determined by a separate decision of the authorized bodies based on the results of the investigation of the accident causes and consequences.

Based on the analysis of the accident causes, an accident elimination schedule is prepared, which is approved by the NPP Chief Engineer.

Elimination of the accident consequences in "contaminated" process systems begins with decontamination of premises and equipment in order to ensure an acceptable radiation situation for repair work. After elimination of the accident, the NPP systems undergo commissioning, tests, and on the basis of their results a certificate for putting the system (equipment) into operation is issued. The certificate for putting the system (equipment) into operation after the accident elimination is approved by the NPP Chief Engineer

In accordance with the instructions considered in the Emergency Operation Procedure, application of BDBA Management Guidelines is provided (if certain criteria are exceeded).

BDBA Management Guidelines define BDBA management actions of the operating personnel to prevent destruction of the physical barriers on the way of propagation of the fission products and mitigate BDBA effects, and is intended for use in the MCR and ECR.

The main goal of the BDBA management is to prevent uncontrolled release of radioactive products beyond the design limits. This goal is achieved by the consistent implementation of the defense-in-depth concept or multiple barriers concept.

The integrity of the protective barriers is ensured by meeting the following conditions:

- quick shutdown of the reactor and maintaining of the core in a subcritical state;
- heat removal from the core during the accident, as well as after stabilization of the parameters in a post-accident state;
- heat removal from the reactor plant;
- protection of the primary circuit from overpressure, hydraulic shocks and thermal loads;
- localization of the accident consequences by sealing the reactor containment to minimize radiological impacts,
- retention of radioactive products within the established boundaries and in the established amounts.

In the case of failure of a critical safety function, immediate actions shall be taken to restore this critical safety function until the integrity of the respective barrier is damaged.

To mitigate the BDBA consequences, the following organizational and technical measures are suggested:

- assessment of the documentation on personnel actions in case of development of emergency situations at earthquakes, seismic impact exceeding the design value;
- the documentation on personnel actions shall include sections providing for measures to diagnose the NPP state, restore the normal operation conditions, failed safety functions and prevent or limit the effects of the core damage: Process Regulations, Reactor Plant Emergency Operation Procedure, BDBA Management Guidelines, Severe Accident Management Guidelines, as well as Action Plan for Personnel Protection in Case of an Accident, which will contain sections providing for measures to solve the following tasks:
- to monitor operation of the safety function algorithms;
- to ensure safe shutdown of the Unit, when the emergency protection signal is initiated, both by automatic actions (as the main method for solving the task), and by the control actions of the personnel;
- to use corrective actions by the personnel if required;
- to keep the core under the coolant and ensure the coolant circulation;
- to bring the Unit to a final state by actions of the personnel, allowing for performing the restoration activities;

- to control the equipment of the normal operation systems and safety-related systems to bring the Unit to a safe final state and limit the radiation effects of malfunctions and accidents;

- to use the standby controls for reliable maintaining of the NPP predetermined mode in case of failure of the main controls for the specified time period required to eliminate the failure;

- to assess the degree of destructions at the NPP caused by an earthquake;

- to maintain or restore functions of the systems and equipment in case of destructions caused by an earthquake;

- to localize development of accidents and provide the limits of safe operation by ensuring subcriticality of the core, keeping it under the coolant (with account for the provided design reserves, speed and efficiency of the protective systems);

- to ensure the integrity of unbroken physical barriers;

- to achieve the final state where the fission chain reaction is discontinued, the reactor subcriticality is ensured and the core re-criticality is prevented, with account for its possible damage;

- to prevent (mitigate) severe damage of the fuel by both automatic actions of the systems and control actions of the personnel;

- to prevent damage of the reactor vessel and primary circuit equipment;

- to prevent damage of the containment;

- to prevent severe damage of the core and mitigate the effects of the core severe damage;

- to limit the radiation impact on the personnel, population and environment;

- to arrange for the accident management at the NPP site.

More detailed organizational and technical measures will be considered and presented in BDBA Management Guidelines, Severe Accident Management Guidelines.

BDBA Management Guidelines define BDBA management actions of the operating personnel to prevent destruction of the physical barriers on the way of propagation of the fission products and mitigate BDBA effects, and is intended for use in the MCR and ECR.

BDBA management forms one of the protection levels of the physical barriers on the way of propagation of the fission products and includes actions to prevent transition of any design basis accidents to beyond design basis accidents and to mitigate consequences of beyond design basis accidents. All available serviceable technical means intended both for ensuring safety at design basis accidents and for normal operation are applied for such actions.

With a large number of possible scenarios for beyond design basis accidents, the application of the "event-oriented" approach for BDBA management is impossible. It is basically impossible to cover the entire list of beyond design basis accidents, taking into account possible overlapping of multiple failures and the complexity of accident diagnosing. The "event-oriented" approach is applicable to management of simple easily recognizable accidents provided for in the design.

Therefore, for development of BDBA Management Guidelines the "symptom-oriented" approach is applied. The advantage of the ""symptom-oriented" approach in such modes is that the personnel actions are aimed at protection of the physical barriers without reference to specific events and failures and on the basis of the actual state of the reactor plant and equipment in terms of the key parameters.

The main tasks of severe accident management are as follows:

- to maintain the tightness and integrity of the fourth physical barrier (containment) as the last barrier to the way of propagation of radioactive substances;

- to limit the release of radioactive substances to the environment.

The severe accident management strategy implemented in the design provides for:

- protection of the containment against all physical impacts that occur during a severe accident and can lead to loads affecting its integrity and tightness;

- NPP transfer to a controlled state at the initial period, in order to prevent damage to the containment and implement preventive measures for reducing radiation exposure;

- NPP transfer to a controlled and safe state, as soon as the functioning of the systems required for it is recovered; the main task here is to reduce pressure and temperature in the containment, as well as to reduce the concentration of radioactive substances in the atmosphere of the containment, in order to reduce radioactive releases to the environment in case of the containment integrity damage.

The design provides for 5 levels of the defense-in-depth. For severe accident conditions, the following levels of the defense-in-depth are relevant:

- Level 4. Limitation of emissions at severe accidents. At the fourth level, consequences of severe accidents are mitigated in order to limit the release of radioactive substances to the environment;

- Level 5. Mitigation of consequences. At the fifth level, measures are taken to limit the population exposure to radioactive radiation in cases where significant releases of radioactive substances to the environment have occurred.

To manage severe accidents, the design provides for a set of technical and organizational measures aimed at transferring the NPP to a controlled state. The means applied are, as far as possible, independent of the means applied at levels 1-3 of the defense-in-depth.

7.2.2 Management of Accidents with Loss of Ultimate Heat Sink after the Beginning of Nuclear Fuel Severe Damage

In the adopted concept of severe accident management the operator's actions are specified in Severe Accident Management Guidelines. Severe accidents are expected to be managed by the personnel actions. Diagnostics of the reactor plant state on the basis of which a decision will be made to proceed to the severe accident management is implemented from the MCR, the diagnostic tools are provided with reliable power supply allowing them to operate for: 24 hours from the batteries, for not less than 72 hours from the mobile DG.

When Severe Accident Management Guidelines are developed for NPPs with VVER reactor plant, the focus is made on the following strategies:

- supplying water to the primary circuit;
- ensuring subcriticality of the molten core material;
- reducing pressure in the primary circuit;
- reducing pressure in the steam generator and cooling-down through BRU-A (in case of SG PHRS non-operability);
- cooldown by heat removal through SG PHRS;
- supplying water to the spent fuel pool;
- supplying water to the Containment PHRS;
- localizing the molten core material in the molten core catcher;
- reducing the release of fission products.

The limited action strategies are not considered.

The next stage in implementation of the strategies of Severe Accident Management Guidelines is to ensure the integrity of the containment and to limit the releases.

Also, one of the severe accident management measures is prevention of explosive hydrogen-steam-air mixture presence by inertizing the atmosphere with steam (increasing the concentration of steam in the containment).

Approaches to ensure water supply to the reactor at severe accidents:

1. Water supply to the reactor plant primary circuit at significant degradation of the core in case of restoring the primary circuit makeup function can lead to activation of the steam-zirconium reaction and increase of hydrogen release to the containment.

2. At significant destruction of the core and formation of molten material pools, water supply to the core cannot stop the destruction of the reactor vessel, and water supply to the core is dangerous in terms of the risk of a steam explosion when the reactor vessel is destroyed.

3. The decision to supply water to the partially destroyed core is taken by the operator depending on the temperature above the core, which characterizes the degree of its destruction. This algorithm is substantiated in Severe Accident Management Guidelines.

In accordance with the adopted severe accident management strategy, the primary circuit pressure reduction to prevent the molten core material releasing beyond the reactor vessel at high pressure is performed by opening the valves of the emergency gas removal system and the pressurizer POSV by the operator. The procedure for the operator's actions to open the valves of the emergency gas removal system and the pressurizer POSV is specified in Severe Accident Management Guidelines.

7.2.3. Management of Accidents with Loss of Ultimate Heat Sink after Failure of the Reactor Vessel

After destruction of the reactor vessel, the severe accident passes to the ex-vessel accident stage.

The main task of the management after the failure (melt-through) of the reactor vessel is to ensure transfer of the molten core material to the molten core catcher and its long-term retention until the full cooldown.

In case of a severe beyond design basis accident, the main functions ensured by the molten core catcher are as follows:

- prevention of the molten core material release beyond the established boundaries of accident localization area;
- guaranteed cooldown of the molten core material;
- ensuring subcriticality of the molten core material in the concrete shaft;
- minimizing the release of radioactive substances and hydrogen into the space of the containment.

The design substantiation of the molten core catcher operational efficiency in case of severe beyond design basis accidents was made using an example of DN850 leakage as a scenario characterized by the maximum energy of the molten core material delivered to the molten core catcher. The operating organization submitted to Gosatomnadzor the results of the design analysis, from which it follows that the molten core catcher is able to perform its design functions, namely:

- the maximum heat flux from the cylindrical part of the vessel, observed at 19220 sec. from the beginning of the accident does not exceed 0.56 MW/m^2, which ensures more than 2.5 times reserve before the critical heat flux in the heat exchanger shell and guarantees efficient cooldown of the molten core material;
- free space inside the vessel of the molten core catcher is sufficient to accommodate the entire volume of the molten core material from the reactor. The provided level of the molten material does not exceed the minimum level of the shell wall cooling;
- water is supplied to the surface of the molten material by passive methods after the inversion of molten materials.

Pressure is reduced in the containment during the operation of the Containment PHRS due to evaporation of the water in the emergency heat removal tanks. The water reserves in the emergency heat removal tanks are designed for a period of 24 hours after the beginning of the accident. At the end of this period, it is necessary to take measures to replenish the water reserves from sources located outside the containment- LCU tanks This measure is necessary to maintain pressure in the containment for an unrestricted period of time.

The makeup of the emergency heat removal tanks is arranged by the operator from the MCR and ECR. The cooling water level in the emergency heat removal tanks is monitored with a level gauge.

For fixation of radioactive iodine isotopes and reduction of radioactive release from the containment, injection of alkali solution into the sump tanks of the containment is provided. Alkali supply is implemented by the operator’s actions.

One of the main threats to the integrity of the containment last safety barrier is the detonation of the hydrogen-steam-air medium in the containment spaces.

Formation of explosive concentration of hydrogen-steam-air mixture is prevented by operation of the JMT system. The hydrogen removal system (JMT) is completely passive, and the autocatalytic recombiners included in the system do not require electric power. Another measure to prevent formation of explosive concentration of hydrogen-steam-air mixture is inertization of the atmosphere with steam (steam concentration increase in the containment) due to the following actions:

- sprinkler system operation control;
- opening of the emergency gas removal system and pressurizer PORV;
- monitoring of the reactor unit cooldown rate.

The most conservative scenario in terms of the early destruction of the reactor vessel is "Double-ended break of the reactor coolant circuit (DN 850) with failure of the ECCS active part". For this scenario, the time of the core top uncovering is 860 sec., the time of the reactor vessel destruction and release of the first portion of the molten core material into the molten core catcher is 6330 sec. Thus, the time reserve from the moment of the core top uncovering till the molten core material release beyond the reactor vessel is 5470 sec.

7.3 Maintaining the Containment Integrity after Severe Nuclear Fuel Damage (including the Core Melting) in the Reactor Core

7.3.1 Prevention of Nuclear Severe Damage (Melting) at High Pressure

High pressure in the reactor vessel at the time of its destruction can cause direct heating of the containment as a result of fuel dispersion as it comes out to the reactor cavity.

Cooldown of the primary circuit using BRU-A or SG PHRS, and the use of the pressure relief systems in case of severe accidents (emergency gas removal system and pressurizer POSV) allow for intensive pressure reduction in the primary circuit of the reactor plant and exclude the possibility of the severe accident scenario with the vessel destruction at high pressure.

The Safety Analysis Report on the Belarusian NPP contains analysis of the thermophysical parameters in the reactor plant and in the containment at severe accidents for scenarios of accidents with loss of the primary circuit coolant into the containment. In all the considered scenarios, the vessel damage at high pressure is excluded because pressure difference between the containment and the reactor vessel at the time of the reactor wall destruction is very small (less than 10 Pa).

If the above measures are not enough to reduce the pressure after the beginning of the accident severe phase (e.g. failure of the pressure reducing systems), it is possible to reduce the pressure in the reactor plant primary circuit by opening the valves of the emergency gas removal system and the pressurizer POSV. The respective analysis is provided in the Safety Analysis Report on the Belarusian NPP.

The emergency gas removal system consists of pipelines with valves and removes steam-gas mixture to the bubbler tank or under the containment:

- from under the reactor head;
- from the primary circuit headers of the steam generators;
- from the pressurizer.

The removed steam-gas mixture is sent to the bubbler tank by opening the appropriate valves.

7.3.2 Control of Hydrogen Concentration Inside the Containment

Release of hydrogen and other combustible gases into the containment through a leakage leads to formation of explosive mixtures. The most unfavourable situation may arise when the steam-gas mixture components reach a concentration at which the detonation is possible.

The detailed analysis of the possible combustion modes is presented in the Safety Analysis Report on the Belarusian NPP and in the technical report.

The hydrogen amount in the containment is reduced due to catalytic oxidation at the hydrogen recombiners (JMT). The existing system of sensors allows for obtaining information on concentrations of oxygen and hydrogen, and assessing the risk of high hydrogen concentration. The hydrogen removal system (JMT) is completely passive, location and total amount of the recombiners are determined based on the results of the design analysis.

In addition, to control the hydrogen situation, some measures are applied to increase the water vapour concentration in the containment and to redistribute hydrogen among the containment rooms:

- controlling the sprinkler system;
- opening the valves of the emergency gas removal system and pressurizer POSV;
- disabling the SG PHRS to increase the amount of steam in the containment;

Disabling the sprinkler system is one of the measures for managing the hydrogen situation by controlling the amount of steam in the containment with the sprinkler system. The steam amount in the containment is increased with the sprinkler system controlled by the operator.

Opening the valves of the emergency gas removal system and pressurizer POSV

The design provides for the emergency gas removal system and pressurizer POSV to redistribute hydrogen between the containment boxes. Opening the valves of the emergency gas removal system and pressurizer POSV allows for redistributing the hydrogen formed in the reactor during the zirconium-steam reaction into a box which is remote from the leakage. A more uniform hydrogen distribution improves the hydrogen situation in the containment rooms by reducing the probability of the air-hydrogen mixture detonation.

The hydrogen situation at this stage is controlled by: disabling the sprinkler system, decreasing the rate of the reactor plant cooldown through the secondary circuit by means of disconnecting a part of the SG PHRS channels, opening the valves of the emergency gas removal system and pressurizer POSV.

The hydrogen situation control with the sprinkler system and by opening the valves of the emergency gas removal system and pressurizer POSV is performed by the operator. The valves of the emergency gas removal system are remotely controlled by the operator. The condition of the steam-gas medium in the containment is monitored using hydrogen and oxygen sensors, pressure and temperature sensors installed in the containment rooms.

Disabling the SG PHRS

The SG PHRS in the Belarusian NPP design is used for long-term removal of the core residual heat to the ultimate heat sink through the secondary circuit at beyond design basis accidents; its operation is taken into account during severe accidents, since the system uses a passive operation principle. Heat removal through the secondary circuit to the emergency heat removal tanks during the operation of the SG PHRS reduces the amount of the steam generated in the primary circuit, therefore disabling the SG PHRS causes the opposite effect.

Upon receiving the signal "Threat of a severe accident," the operator disables the SG PHRS.

If the SG PHRS ensures the cooldown of the primary circuit (e.g. at scenarios with a blackout), the SG PHRS is disabled automatically after the pressure drop in the reactor plant primary circuit to the values excluding development of the scenarios involving severe damage to the reactor vessel at high pressure.

At the final stage of the severe accident where the operation of the SG PHRS is efficient, inertization of the containment medium is irrelevant.

Thus, to control the hydrogen situation in the containment, the following sequence of actions is recommended:

1. At severe accidents such as the "Major leakage of the primary circuit with failure of the ECCS active part", the sufficient hydrogen situation control measure is to disable the sprinkler system; for accidents of the same type but with a blackout, the control is not required.
2. At severe accidents such as "Medium and minor leakage of the primary circuit with failure of the ECCS active part", disabling the sprinkler system is not sufficient to control the hydrogen situation; therefore, additional measures for inertizing the containment with steam are required – reactor plant cooldown rate control via BRU-A or by reducing the number of the SG PHRS channels contributing to the reactor plant cooldown. Similar measures to control the reactor plant cooldown rate may be required at severe accidents such as "Medium and minor leakage of the primary circuit with a blackout".

7.3.3 Prevention of the Containment Destruction due to Pressure Increase

There are several potential hazards leading to the containment destruction by high pressure: steam explosions at the in- and ex-vessel stage of the severe accident; loading of the containment due to mass and energy release at the in- and ex-vessel stage of the severe accident.

Steam explosions at the in- and ex-vessel stage of the severe accident

A steam explosion within the reactor vessel can result in damaging the containment through its penetration with flying objects, which may be classified as an early radioactive emission to the atmosphere.

Steam explosion hazard arises as a result of the core degradation due to interaction of heated fuel fragments and coolant residues in the reactor vessel.

The design and organizational measures adopted in the design are aimed at minimizing the possibility of intensive interaction of the molten core material with water and preventing the possibility of the molten material dispersion after it comes out of the reactor vessel.

The minimization of the intensive interaction arising from the contact of the molten core material with water is achieved by the design features of the reactor vessel and the prohibition for water supply to the core when the onset of the accident severe phase is diagnosed.

To prevent ex-vessel steam explosions, no water shall be present inside the molten core catcher when the first portions of the molten core material enter the catcher. This is ensured by the design of the safety membrane on the vessel of the molten core catcher.

Loading the containment due to mass and energy release at the in- and ex-vessel stage of the severe accident

For accidents with major and medium leakages, there exists an initial pressure peak determined by the steam flow from the reactor plant. As release from the reactor plant decreases, the condensation processes start to prevail, and the pressure in the containment rooms begins to reduce.

The volume of the containment is designed for release of the coolant during an accident involving a rupture of the reactor coolant circuit, therefore no control actions are required for pressure reduction at the initial stage of the accident. The temperature at this stage of the accident does not exceed the saturation temperature at the respective pressure.

The results of the design analysis have shown that the absolute pressure in the containment does not exceed 0.5 MPa, which corresponds to the maximum permissible pressure for the containment under severe accident conditions. From the analysis it is known that the maximum initial peak pressure (up to 400000 Pa) arises at the scenario with DN 850 leakage; at the accident with DN 346 leakage the maximum pressure is about 350000 Pa throughout the accident; DN 80 leakage is characterized by a lower maximum pressure.

The pressure increase is limited by operation of the containment PHRS removing a part of the heat from the containment to the atmosphere through evaporation of the water in the emergency heat removal tanks.

The design analysis of the containment late loading made for severe accidents with coolant leakage under the containment and evaporation of the spent fuel pool has shown that operation of the containment PHRS allows to limit the pressure rise at the ex-vessel stage of the accident without exceeding the design pressure limits. The analysis took into account the coolant release from the rupture (in-vessel stage), steam release from the molten core catcher (ex-vessel stage) and an additional source of steam due to boiling of the spent fuel pool.

Containment strength analysis for the NPP 2006 design

In case of overpressure under the containment the following transition from the elastic to the plastic state of the pre-stressing cables is observed:

- from 0 to 0.5 MPa: linear stress increase is observed;
- from 0.5 to 0.94 MPa: stress dependence on pressure is non-linear;
- at 0.94 MPa: the stress reaches the yield limit, theoretical cable break.

At a pressure of 1.17 MPa, stresses in the reinforcement reach the yield point, which is the moment when the containment loses its bearing capacity. Damage to the concrete reaches 1 at a pressure of about 0.8 MPa. Damage to the internal surface reaches 1 at a pressure of about 0.98 MPa. As the cladding is a welded structure, it can be conservatively assumed that at a pressure corresponding to the appearance of through cracks in the concrete, local loss of cladding tightness is possible due to its rupture. Taking into account a damage margin factor of 1.3, the critical pressure for tightness is 0.89 MPa.

7.3.4 Prevention of Re-criticality

The operating organization has submitted an analysis of severe accident accompanied by complete destruction of the core, melt-through of the reactor vessel, the molten core material entering the molten core catcher with its subsequent cooling, with a global change in configuration and temperature of the fissile material.

To perform the criticality analysis, an approach published in the papers of the Research Center "Kurchatovsky Institute" [31] was applied. It includes the following:

- a number of the basic states which degradation of the core and cooling of the molten core material in the molten core catcher go through are identified;
- for each basic state, a set of parameters affecting the K_{eff} value is determined, and the range of their possible values is assessed;
- if there is no reliable information on values of the parameters, the most conservative approach is applied;
- the possibility of external impact resulting in increase of K_{eff} value, e.g. injection of non-borated water, shall be taken into account;

The following states are considered as basic states at the in-vessel stage of the accident:

- a state corresponding to the maximum reactivity margin is selected as the initial state of the core prior to the accident;
- the core after actuation of the emergency protection;
- the drained core, heating-up, the rod structure is retained;
- formation of the ceramic filling, FA residues remain;
- melting of the ceramic filling;
- melting on the bottom of the reactor vessel.

The following states are considered at the ex-vessel stage of the accident:

- molten core material before the inversion;
- molten core material after the inversion;
- cracking of the solidified molten core material at the final stage of the cooling.

The maximum fuel enrichment for 235U in the NPP-2006 design is 4.95% (percentage of weight units). Fuel with such enrichment in the amount loaded into the core in the absence of a moderator is characterized by K_{eff} values below 1, regardless of the fuel temperature and configuration. Presence of structural material impurities in the molten fuel leads only to decrease of K_{eff}.

Conclusions on criticality at the ex-vessel stage of an accident, obtained previously for nuclear power stations of NPP-2006 type, given the maximum enrichment of 4.95%, are as follows:

- the core will be in a subcritical state at all characteristic stages of its degradation in the absence of water inside it at the fuel level;
- the core may only be reflooded with water containing boric acid at not less than 16 g/dm^3.

Conclusions on criticality at the ex-vessel stage of the accident, obtained previously for nuclear power plants of NPP-2006 type, given a maximum enrichment of 4.95%, are as follows:

- the core will be in a subcritical state at all typical stages of its degradation in the absence of water in it at the fuel location level;
- only water with boric acid content not lower than 16 g/dm3 may be injected to the core.

The criticality analysis of the ex-vessel stage was performed [31] using TDMCC software which implements Monte Carlo method for neutron transfer simulation. The software was produced at the Federal State Unitary Enterprise "Russian Federal Nuclear Centre – All-Russian Research Institute of Experimental Physics".

It was noted above that three stages can be identified in operation of the molten core catcher. The first two of them are very similar in terms of K_{eff} calculation. Indeed, at these stages the fissile material with an admixture of sacrificial and structural material is a homogenous medium, water is present between the vessel of the molten core catcher and the concrete shaft, later water is also fed to the surface of the molten core material to cool it.

The starting moment of the last stage is uncertain. It is known that during solidification of the molten core material it cracks. The formed cavities are filled with water supplied earlier to the surface of the molten core material. As a result, a random heterogeneous medium is formed of the molten core material and water. Lack of reliable information on the weight, composition and structure of the resulting medium makes it necessary to vary the characteristics of the medium in a wide range of possible values in order to detect the configurations corresponding to the maximum K_{eff} value.

The stage of the homogeneous molten core material is characterized by hard neutron energy spectrum. For systems with such neutron energy spectrum there is a relatively small number of benchmark experiments, i.e. test experiments with clearly fixed conditions, for the calculation of K_{eff}. Most experiments were performed at room temperature, which does not allow using them to assess errors of the software products simulating neutron transfer at melting temperatures of the molten core material. To solve this task, the method of calculation result uncertainties is applied. Uncertainty of the calculation results is defined as a mean-square deviation of the cumulative calculations in all the software products in the considered class of tasks from the experimental values, if any, or an average value in the solution of test tasks. For nuclear systems with low enrichment, hard neutron energy spectrum and a temperature of 2000-3000 K, the uncertainty of K_{eff} calculation is about 3%. Such high uncertainty of K_{eff} calculation is not essential for substantiation of the subcriticality at the stage of the homogeneous molten core material. In this case the system is in a deeply subcritical state, and such uncertainty of the estimates cannot change this result.

The isotopic composition of the fuel is estimated at the beginning of the first and sixth (stationary) fuel loads. Calculations were also made for super-conservative approximation when enrichment of the total fuel for 235U was assumed as 4.95%.

Figure 7.3.4.1 shows the results of the following calculations:

- various temperatures of the molten core material (mixture of oxides of the uranium and plutonium heavy isotopes);

- molten core material configuration before the inversion;

- various isotopic compositions of the fuel, including the conservative estimation - enrichment of 4.95% for 235U.

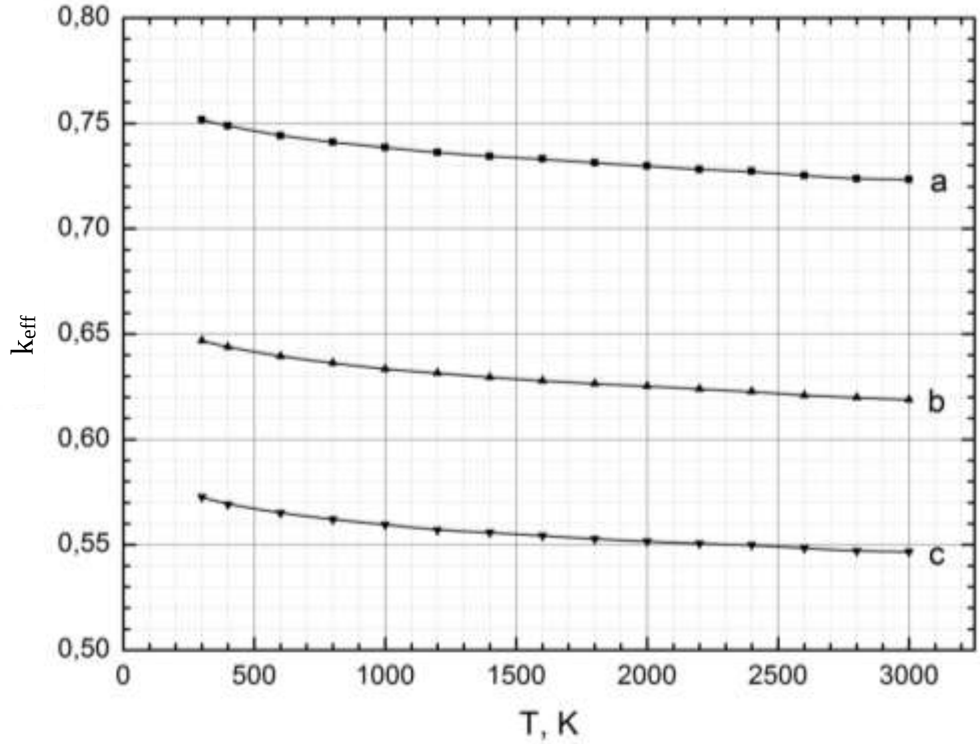

Figure 7.3.4.1- K_{eff} versus temperature of the molten pure fuel for various isotopic compositions: a – 4.95% enrichment for 235U; b - beginning of stationary fuel load; c - beginning of the first fuel load

K_{eff} maximum value does not exceed 0.752 in the most conservative approximation. Given the 3% uncertainty of the estimation, K_{eff} is <0.81 with 95% probability.

Presence of structural material impurities in the molten fuel results in reduction of K_{eff} value. It is caused by several factors: reduction of the medium density, increase of the medium surface area for neutron leakage, increase in the amount of neutron absorbers in the medium. It should be noted that gadolinium oxide added to the sacrificial materials does not significantly affect the K_{eff} value at this stage of the molten core material cooling due to the hard neutron energy spectrum. Figure 7.3.4.2 shows that this statement is true for the entire temperature range considered.

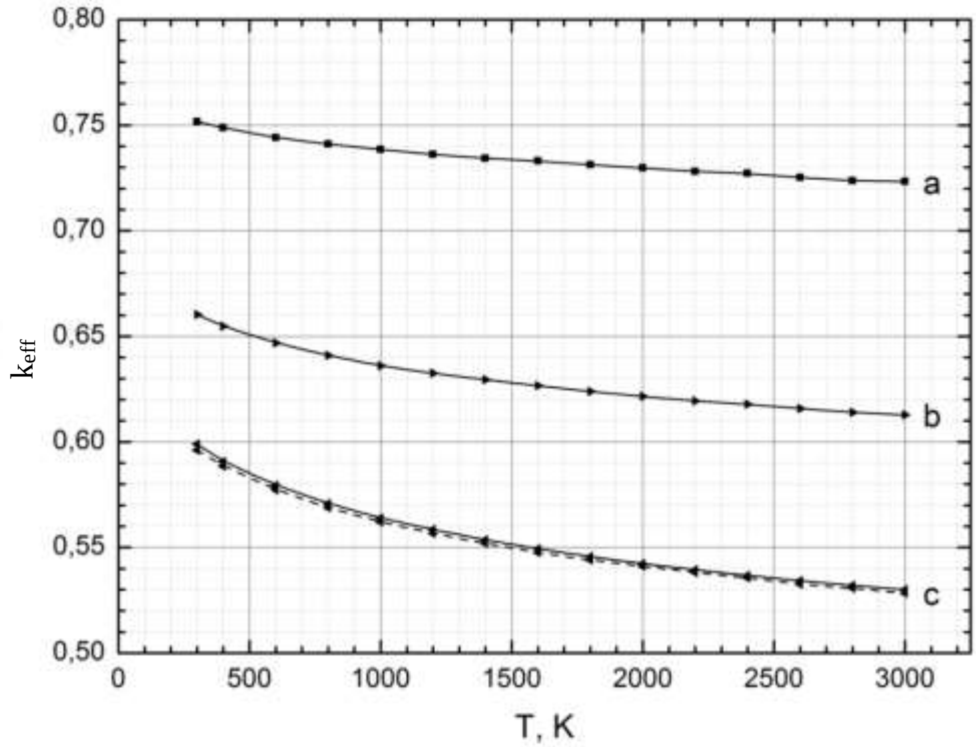

Figure 7.3.4.2 - K_{eff} versus temperature of the molten fuel and various impurities, isotopic composition of the fuel corresponds to a conservative approximation, a - pure fuel; b - fuel mixed with ZrO_2; c - fuel mixed with ZrO_2 and sacrificial materials, the dotted curve corresponds to the presence of gadolinium oxide in the sacrificial material in the amount of 0.1% of the sacrificial material weight.

Similar results were obtained for various isotopic compositions of the fuel and for the molten core material configuration before and after the inversion (when the layer of uranium, zirconium, aluminium and iron oxides mixture is higher than the iron layer). In general, K_{eff} value increases with cooling of the molten core material. K_{eff} maximum values obtained in all the calculations are presented in Table 7.3.4.3. It can be seen that at this stage of the molten core material cooling, the system is in a deeply subcritical state. It was noted above that a conservative estimate gives K_{eff} of <0.81 with 95% confidence interval. A more realistic estimate obtained for the stationary fuel load and the fuel mixed with ZrO_2 and sacrificial materials suggests that for 95% confidence interval K_{eff} does not exceed 0.62.

The detailed information on the distribution of materials and temperatures throughout the molten core catcher does not affect the conclusions. The calculations performed previously for NPP of this type showed that consideration of actual distribution of materials and temperatures usually reduces K_{eff} value.

Table 7.3.4.1 –Maximum K_{eff} values for various compositions of the molten core material and isotopic compositions of the fuel

Impurities in the fuel	Isotopic composition of the fuel		
	4.95% enrichment by 235U	First fuel load	Stationary fuel load
	k_{eff}		
-	0.7516	0.5726	0.6468
ZrO_2	0.6604	0.4879	0.5553
ZrO_2+sacrificial material	0.5989	0.4257	0.4916

ZrO_2+ sacrificial material $+Gd_2O_3$	0.5960	0.4235	0.4897
Inversion	0.6029	0.4511	0.5120
Inversion $+Gd_2O_3$	0.5828	0.4143	0.4792

The final stage of the molten core material cooling is accompanied by its cracking, with formation of voids which can be filled with water supplied to the molten core catcher for cooling. As a result, water-uranium medium with a random distribution of voids is formed. Currently, there is no reliable data on the potential volume of the voids. There are only rough estimates of the possible maximum of this value. Thus, estimation performed at the Research Center "Kurchatovsky Institute" was based on UO_2 density change with a temperature from the melting point to a normal state. On this basis it was concluded that the volume of the voids does not exceed 12% of the total volume of the molten core material. At the same time, preliminary experimental data on cooling of the molten core material allow to assume that the volume of the voids can reach 30%. These estimates do not consider uniformity of the voids distribution throughout the volume of the molten core material. It may be assumed that in certain parts of the molten core material, the volume of the voids can significantly exceed the estimates.

For criticality analysis, a 3D model of the molten core catcher was used as shown in Figure 7.3.4.3. Given the high dependency of K_{eff} on the relative proportion of the voids - VH_2O / V_0, it appears dangerous to limit its maximum possible value in the absence of reliable data. The second parameter characterizing the voids is the average length of the neutron path crossing the void area in a random direction. Regarding the characteristic values of this parameter, the reasoning is the same as that regarding the maximum possible volume of the formed voids. There is no actual information on the characteristic linear dimensions of the voids, at the same time there is a dependency of K_{eff} on this parameter, which leads to the necessity to vary its values in a wide range.

There are two circumstances common for the two considered parameters - VH_2O / V_0 and $\langle L \rangle$, which characterize a random heterogeneous medium: firstly, a gradual dependency of K_{eff} value on these parameters is observed, and secondly, as the calculations showed, K_{eff} value decreases at excessive increase of any of the two parameters. Thus, Figures 6.3.4.4 and 6.3.4.5 show K_{eff} dependency on the proportion of the voids and the average size of the voids, respectively. The calculations were performed for the conservative assumptions regarding the isotopic composition of the fuel. It can be seen that K_{eff} begins to decrease steadily at VH_2O / V_0> 0.4 for all the considered average void sizes, and for $\langle L \rangle > 8$ for various VH_2O / V_0 values. Thus, the selected ranges of the void proportion and average sizes overlap the range of maximum K_{eff} values.

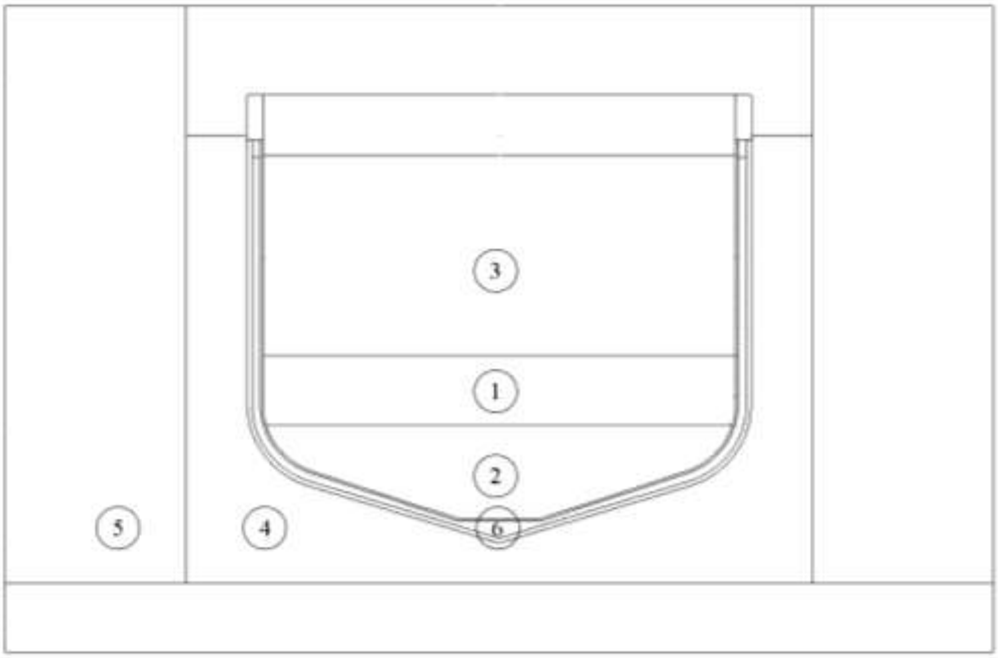

Figure 7.3.4.3 - Model of the molten core catcher used in the K_{eff} calculations: 1- molten core material layer with voids filled with water; 2, 4 - water; 3 - metal layer; 5 - concrete protection; 6 – vessel of the molten core catcher

The resulting limits for the range of the two parameters of the model describing the random heterogeneous medium, allowed to perform systematic calculation of K_{eff} values. The calculations were made for three fuel compositions corresponding to the three load options: the conservative approximation, the stationary fuel load and the first fuel load. For each fuel composition, various extent of crack filling with water and various extent of fuel mixing with the sacrificial material were considered.

The calculation result shows that within 50% - 100% of water content in the cracks, the maximum K_{eff} varies slightly. With decrease of the water content in the cracks, the VH_2O / V_0 value, corresponding to the maximum K_{eff} value, increases.

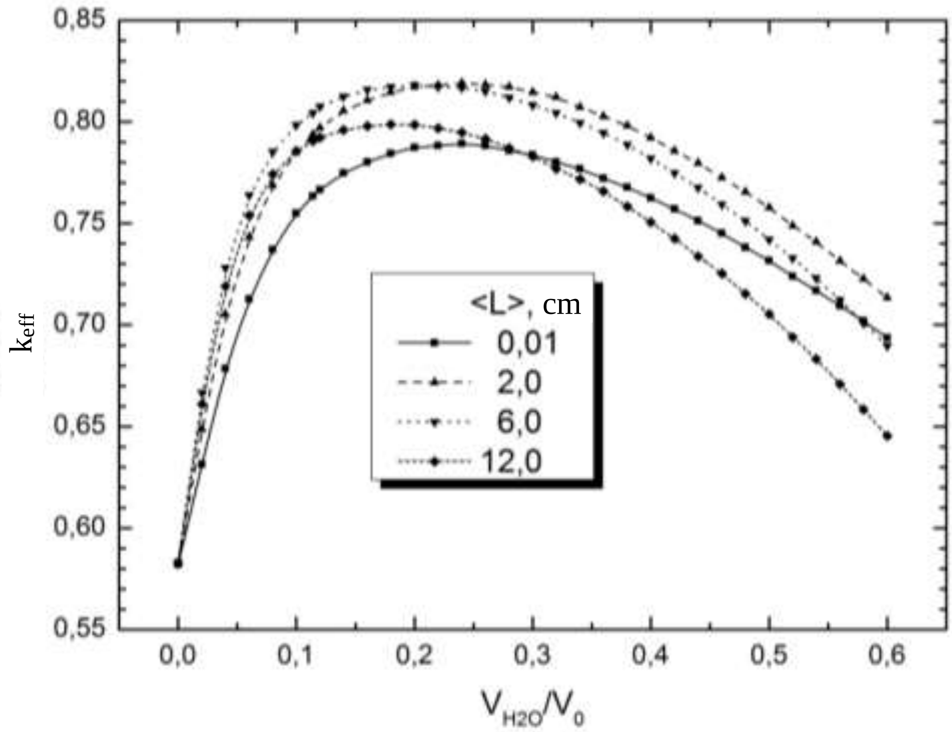

Figure 7.3.4.4 - K_{eff} versus proportion of the voids completely filled with water for various average void sizes; isotopic composition of the fuel corresponds to the conservative approximation, 100% of the sacrificial materials containing Gd_2O_3 are included into the molten core material.

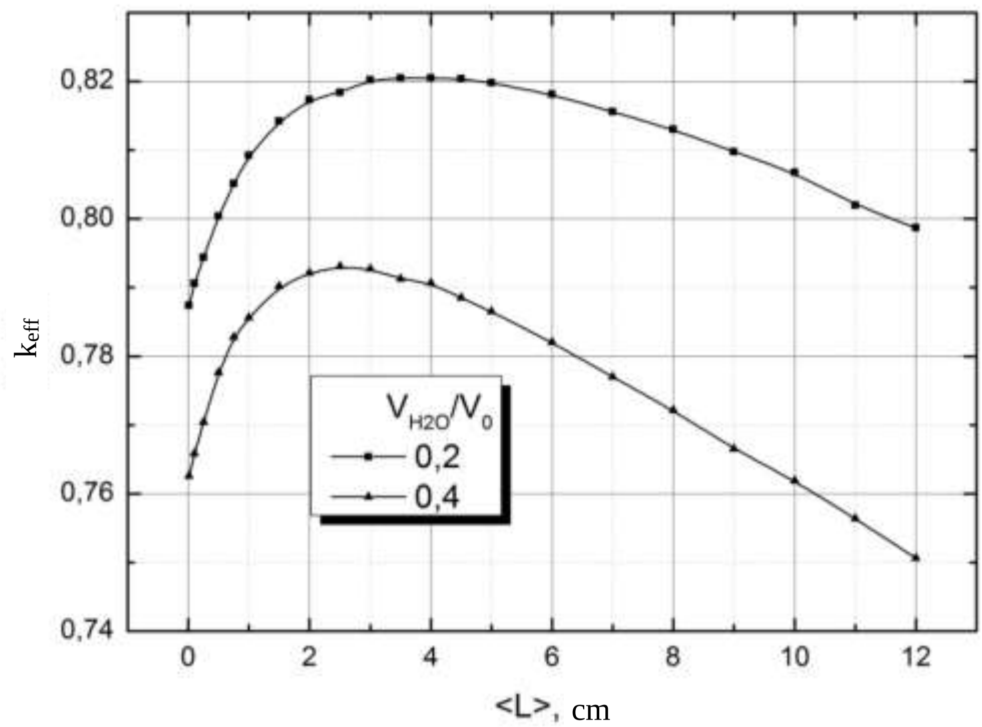

Figure 7.3.4.5 - K_{eff} versus average size of the voids completely filled with water for two void proportion values; isotopic composition of the fuel corresponds to the conservative approximation, 100% of the sacrificial materials containing Gd_2O_3 are included into the molten core material.

Table 7.3.4.2 – Maximum K_{eff} values obtained for various compositions of the fuel, the extent of the fuel mixing with the sacrificial material and water content in the voids.

Fuel composition	Mixing with sacrificial material%	Water content, %	Maximum K_{eff}	V_{H2O}/V_0
Conservative approximation	100	50	0.816	0.36
	100	80	0.821	0.26
	100	100	0.822	0.22
	90	100	0.847	0.26
	80	100	0.878	0.28
	70	100	0.910	0.30
Stationary load	100	50	0.740	0.38
	100	80	0.743	0.29
	100	100	0.744	0.24
	90	100	0.767	0.26
	80	100	0.795	0.28
	70	100	0.824	0.30
First load	100	50	0.591	0.28
	100	80	0.593	0.20
	100	100	0.594	0.18
	90	100	0.616	0.18
	80	100	0.643	0.20
	70	100	0.670	0.24

From the obtained data shown in Table 7.3.4.2 it follows that in an extreme case: conservative approximation of the fuel composition and only 70% of the sacrificial materials mixing with the molten core material, the maximum K_{eff} value is 0.910. For a more realistic fuel composition corresponding to the stationary fuel load, the maximum K_{eff} value does not exceed 0.825. Thus, when gadolinium oxide content in the sacrificial material is not less than 0.1% of the sacrificial material weight, the solidified molten core material will be in a subcritical state, regardless of its degree of cracking and filling of the cracks with water.

7.3.5 Prevention of the Containment Destruction Due to Melting of the Foundation

Bottom slab

After the melt-through of the reactor vessel the molten core material drains to the bottom slab, the main task of which is to ensure transfer of the molten core material to the molten core catcher. The bottom slab has a layered structure - the top layer of the slab consists of special concrete which forms a liquid underlayer at thermal contact with the hot molten core material; this underlayer prevents solidification of the molten core material during its movement. The remaining layers of the bottom slab are made of reinforced refractory concrete.

After flowing down the bottom slab, the molten core material enters the space confined at the sides and at the bottom with water-cooled steel walls of the molten core catcher vessel located in the space under the reactor in the concrete shaft, where the long-term cooldown is ensured.

Water supply to the shaft of the molten core catcher

For successful functioning of the molten core catcher, it is necessary to supply water to the shaft of the molten core catcher. This action can be performed passively – at any accident with coolant leakage by filling the shaft of the molten core catcher with water supplied from elevation +4.900 of the containment.

If the passive filling is unavailable, the water supply to the shaft of the molten core catcher (transfer to the "hot standby" state) is backed up by opening the valves through which the shaft of the molten core catcher will be filled with borated water from the ECCS sump tanks. To do this, upon receiving the signal "temperature at the core outlet has reached 600 ° C", the operator opens the respective valves from the MCR.

Subsequently, the residual heat from the molten core material is removed during water boiling at the outer wall of the molten core catcher vessel in a fully passive mode for an indefinitely long time.

Water supply to the surface of the molten core material

The main water supply method is passive. Water is supplied from the shaft of the molten core catcher when the passive water supply valve is actuated upon reaching of the preset temperature by the heat-conducting insert. Opening of the valve ensures water supply from the reactor cavity to the melt mirror by gravity.

The vertical section of the passive water supply valve is located in a cylindrical channel with cooling water from the shaft of the molten core catcher. The passive water supply valve is located below the minimum water level in the concrete reactor cavity, which ensures reliable cooling water supply to the surface of the molten core material after its actuation.

The calculation results show that at a preset melting temperature of the valve solder plug, water is supplied to the melt mirror under the following conditions:

- 90% of the molten core material have entered the molten core catcher;
- Inversion has occurred in the molten core catcher, and the oxide layer appeared to be on the top.

Water supply to the surface of the molten core material from the reactor internals inspection shafts is a backup function which is considered only in case of failure of all valves of the passive system. In this case, due to the effect of thermal radiation from the surface of the molten core material, the temperature of the metal structures in the service area can increase.

7.3.6 Sources of AC and DC Power Supply and Compressed Air for the Equipment Used to Maintain the Containment Integrity

In case of loss of power supply to the normal operation systems and failure of all diesel generators, the normal operation systems ensuring the reactor residual heat removal to the ultimate heat sink and cooling of the spent fuel pool stop functioning. Simultaneously, failure of the active safety systems occurs.

The Unit state at the initial stage of the accident is characterized by:

- total lack of AC power supply (on-site and off-site);
- availability of power supply from the UPS of the emergency power supply system for some valves (isolating valves of the containment, BRU-A, MSIV) and I&C. Power supply from the UPS is designed for 2 hours without recharging the batteries included into the UPS;
- availability of power supply from the UPS for the BDBA I&C equipment (Channel 7). Power supply from the UPS is designed for 24 hours without recharging the batteries (excluding pump JNB50AP001 and communication systems) included into the UPS. In addition, a mobile diesel generator can be connected;
- subcritical state of the reactor;
- leakages from the primary circuit - 2.15 m^3 / h, which is the maximum possible amount of leakage during operation at nominal power level;
- tight secondary circuit;
- full reserve of cooling water for the BDBA I&C in the four SG PHRS tanks;
- water level in the fuel pool - 8.7 m (fuel storage level).

Under these conditions, facilities and organizational measures mentioned in paragraphs 5.1.2 and 5.1.3 of the Report [31] shall be used for BDBA management.

For power supply to the BDBA I&C system under the conditions of total loss of AC power supplies, including all diesel generators, a special power supply system (BDBA power supply system) is provided. It is able to supply power to the BDBA I&C for the required time (24 hours or more, up to 72 hours), as Unit I&C is powered by batteries installed at the Unit for 2 hours only.

The design provides for a dual-channel structure of the BDBA I&C system. BDBA power supply system has also a dual-channel structure. Each power supply channel for the BDBA equipment includes the following power sources:

- UPS;
- batteries.

In the normal operation mode the BDBA power supply system is connected to two out of four EPSS channels. Thus, the batteries are maintained in a fully charged state (floating charge mode). Also, as required, the operator can use the required equipment connected to the BDBA power supply system. In case of loss of power supply to the auxiliary power supply system, including the loss of all the DGs, the BDBA power supply batteries start discharging. To ensure the pump starting after a long battery discharge and to increase the duration of the system operation during BDBA, a mobile diesel generator is provided, which is stored in a special location at the NPP site during normal operation, anticipated operational occurrences and design basis accidents.

All electrical equipment of the BDBA power supply system refers to seismic category I.

The electrical equipment of the BDBA power supply system is located in the rooms at elevation -7.20 in steam chamber building UJE. The building refers to seismic category I. The temperature in the rooms, where the electrical equipment is located, is within the range permissible for operation of electrical equipment at extreme (both positive and negative) outdoor air temperatures.

Compressed air systems are not used to maintain the containment integrity.

The hydrogen removal system is entirely passive, and the autocatalytic recombiners included into this system do not require electric power.

To monitor the hydrogen explosion safety, the hydrogen concentration monitoring system (JMU) is provided in the design.

The system for monitoring hydrogen concentration consists of two identical and completely independent measurement channels JMU10, JMU20.

Each channel of the monitoring system includes eight combined hydrogen and oxygen analysers with a temperature sensor (WS85PLUS), two pressure sensors ДАЭ -100Т and a hardware-software complex "Data Processing Station of the Hydrogen Concentration Monitoring System".

The number of monitoring points and locations of the sensing elements of the hydrogen analysers are selected based on the results of the analysis of hydrogen propagation, accumulation and potential burning modes in the rooms under the containment. Control areas are identified in the rooms under the containment where hydrogen concentration increases rapidly during the accident. Each control area has at least one hydrogen analyser sensor of one of the measuring channels.

The range of measured hydrogen concentrations is from 0 to 25% by volume.

7.3.7. Instrumentation Required for Maintaining the Containment Integrity

In case of beyond design basis accidents the design provides for application of special instrumentation and implementation of organizational measures. The containment integrity during BDBA is monitored from the MCR. The information from the instruments characterizing the containment integrity is displayed by the indicating instruments of the segmented panel CWL01, which is located in the MCR. There is no such panel in the ECR.

The sensors monitoring the containment integrity are located in reactor building UJA and in annulus UJB. The sensors of different channels and relevant soft & hardware complex devices are located in different rooms. The soft & hardware complex devices are located in the APCS facilities rooms; the segmented panels are located in the MCR in control room building UCB

All the containment integrity monitoring equipment refers to seismic category 1.

The containment integrity monitoring sensors are designed for a wide range of parameters and are resistant to adverse impacts of BDBA.

An operator monitors the containment integrity by means of the following instruments and signals displayed at MCR panel.

1) Indications of status of the containment penetration isolating valves (closed/opened).

2) Indications of the air lock automated tightness monitoring system (inner/outer hatch is not leak tight, the air lock automated tightness monitoring system is disabled or malfunctions).

3) Indications of the automated radiation monitoring system (ARMS) for measurement of the gamma radiation dose rate inside the containment which allow to assess the degree of fuel damage in the core up to its complete melting. The design provides for four detectors to measure the gamma radiation dose rate inside the containment; these detectors have operation capability also during the worst DBA phase. The detectors refer to seismic category 1 and safety class 3. Three detectors are powered from the relevant channel of the emergency power supply system (EPSS). One detector is powered from the BDBA power supply system. The measuring range of these detectors is 10^{-3}-10^{5} Gy/h. The detectors perform reliably at a temperature of +180 °C during 2 days, and at a temperature of +205 °C within 30 minutes.

4) Indications of the ARMS sensors for measuring volumetric activity of inert radioactive gases in the air of the annulus which allow to assess the degree of the containment leakage provided that annulus ventilation system KLC11/21/31/41 is in operable condition (four channels of the system are powered from four EPSS channels).

5) Indications of the ARMS sensors at the NPP site allowing to assess the extent of emergency release from the containment to the environment provided that they remain in operable condition (powered from the power supply system for the safety-related normal operation systems).

Also, indications of the personnel and equipment lock hatch status are displayed at the workstation of the unit upper level control system (inner/outer hatch is leak tight/not leak tight, inner/outer door open/closed).

Excessive pressure in the containment in the emergency modes is measured by eight sensors. The sensors are powered from the respective EPSS channel (10UJA10CP811, 10UJA10CP812 – from channel 1; 10UJA20CP821, 10UJA20CP822 – from channel 2; 10UJA30CP831, 10UJA30CP832 – from channel 3; 10UJA40CP841, 10UJA40CP842 – from channel 4).

For hydrogen explosion safety, the design provides for the hydrogen removal system (JMT) and the hydrogen concentration control system (JMU).

The JMT system prevents formation of explosive mixtures in the containment by maintaining the volumetric hydrogen concentration in the mixture at a safe level. At design basis accidents the JMT system maintains the hydrogen concentration at a level not allowing for detonation, including local detonation, as well as for rapid fire propagation in large volumes (commeasurable with dimensions of the containment main rooms).

The system performance is substantiated for severe accidents leading to the core melting.

The system uses passive autocatalytic hydrogen recombiners as the main functional components. The recombiners start to function when high hydrogen concentration is formed in the room and continue to operate until the hydrogen concentration is reduced to a safe value.

There are 44 recombiners in the containment rooms. In order to ensure maximum efficiency of the system, the recombiners are installed in places where the hydrogen concentration during the accident can reach maximum values, as well as on the ways of the steam-gas medium movement.

The hydrogen concentration monitoring system (JMU) is an information system designed to monitor the hydrogen explosion safety in the containment during BDBA.

The hydrogen concentration monitoring system consists of two identical and entirely independent measurement channels JMU10, JMU20.

Each measurement channel includes eight temperature, hydrogen and oxygen concentration monitoring points, and two pressure, temperature, hydrogen and oxygen concentration monitoring points located in the containment rooms, as well as secondary equipment located outside the containment.

Data on hydrogen and oxygen concentration, temperature and pressure in the containment rooms obtained by direct measurements, and steam concentration data obtained by calculation (using equipment of the hydrogen concentration monitoring system) is transmitted to the Unit APCS and then displayed at the upper level monitors (in a form convenient for the operator), in the backup area of the BDBA segmented panels, and also used for generating respective signals to the alarming equipment in the case of formation of explosive hydrogen-containing steam-gas mixture during BDBA.

The hydrogen concentration monitoring system (JMU) is powered from channel 7 of the BDBA power supply system.

7.3.8 Management of Severe Accidents in Case of Simultaneous Core Melting and Nuclear Fuel Damage in the Spent Fuel Pool at Different Units of the NPP Site

The design provides for the spent fuel pool cooling system (FAK) to remove heat from the spent fuel pool. In the modes with design basis accidents, it removes heat from the spent fuel pool and performs the makeup function in the modes with loss of water. If this system fails, residual heat of the spent fuel assemblies is removed by water accumulating the heat. Water in the spent fuel pool within the heat-generating part of the FA and in the spent fuel pool volume above the FA is heated up to the saturation temperature; thereafter the water level in the spent fuel pool decreases due to boiling caused by power of the spent FA residual heat. In this case the heat is removed from the containment to the ultimate heat sink - atmosphere - by steam condensation on the heat exchangers of the containment PHRS.

To avoid damage of the spent fuel pool, the design shall provide for such measures which will allow to monitor temperature conditions in the spent fuel pool. These measures shall ensure monitoring of the water level in the spent fuel tanks. The fuel must be kept under water. To meet this requirement, making-up of the spent fuel pool with water shall be provided.

During BDBA with failure of all power sources which ensure making-up of the emergency heat removal tanks and spent fuel pool, the system of emergency water use from the reactor internals inspection shaft is used (JNB90). For the period of 24 hours and more, pump set JNB50AP001 ensures making-up of the emergency heat removal tanks and spent fuel pool from the demineralized water tanks (LCU). Active components of the JNB90 system have reliable power supply from batteries and a mobile DG set and can ensure the reactor plant safety in a hot state cooldown in case of NPP blackout.

During severe accident the operator shall monitor water level in the spent fuel pool.

Simultaneous accidents in the reactor core and spent fuel pool are analysed in terms of their mutual impact in [31, Section 5.1.1].

Severe accident management in case of simultaneous core melting and nuclear fuel damage in the spent fuel pool is analysed in [31, Section 6.3.8].

Simultaneous accidents in the spent fuel pool and the reactor have no impact on each other, because different safety systems are used to manage accidents in the spent fuel pool and the reactor. For example, FAK or JMN system is used for heat removal from the spent fuel pool, while JNG1,2, JND systems are used to remove heat from the reactor [31].

Operative time reserve in case of nuclear fuel damage in the spent fuel pool depends on the loading of the pool with spent fuel assemblies.

The following options are considered:

- option 1: complete unloading of the reactor core to the spent fuel pool, with the presence of the spent fuel assemblies for 10 years of operation;
- option 2: power operation at the beginning of the reactor campaign (after refuelling).

The Units of the Belarusian NPP are technically and structurally independent. Emergency coordination and management at both Units will be performed by the NPP administration with the crisis center being involved. Analysis of the initiating event with the core melting is presented in Sections 7.1 - 7.3.

Emergency response activities at each of the Units will be performed by the personnel of the respective Unit or general-plant personnel in accordance with orders of the administration. Depending on the situation development, the manpower and resources can be transferred from one Unit to the other. In case of an accident at one Unit only, the personnel of the other Unit will have instructions on actions in such situations. In case of a severe accident at both Units, identical accident management guidelines will be applied at both Units, but the situations at each of the Units will be assessed independently, and the NPP administration will coordinate the works at the Units. The requirements for accident management at several Units are specified in the BDBA Management Guidelines, Severe Accident Management Guidelines.

7.3.9 Conclusion on the Sufficiency of the Systems Required for Severe Accident Management to Ensure the Containment Integrity

Accident management facilities

At the in-vessel stage of the severe accident the operator opens the valves of the emergency gas removal system and the pressurizer POSV. Opening of these valves shall ensure the reliability level of the pressure reduction functions required to prevent the scenario of the molten core material releasing from the reactor vessel at high pressure.

In order to manage severe accidents at the ex-vessel stage, the Belarusian NPP design provides for the following facilities:

- molten core catcher (JMR);
- emergency alkali supply system (JNB91);
- system of passive heat removal from the containment (JMP);
- PHRS emergency heat removal tanks (JNB);
- emergency makeup system for the emergency heat removal tanks (JNB50);
- hydrogen monitoring and removal system (JMT, JMU);
- BDBA emergency power supply system (channel 7);
- inner and outer containments of the reactor building (UJA, UJB).

At severe accident with the reactor core and reactor vessel destruction, the molten core catcher retains the molten core material and solid fragments of the destroyed core, parts of the reactor vessel and internals. The molten core material is localized and cooled within the concrete cavity section under the reactor for an unlimited time, provided that heat is removed from the containment.

The heat is removed from the containment by the system of passive heat removal from the containment (JMP). The system reduces and maintains the pressure in the containment within the design limits, and removes heat from the containment to the ultimate heat sink at design basis accidents, including accidents with severe core damage.

The selected design of the system ensures its fully self-contained operation without operator involvement for at least 24 hours. During the period after 24 hours up to 72 hours or more, if required, it is possible to use a mobile DG set to ensure functioning of the system (makeup of the emergency heat removal tanks).

When the thermal protection of the upper part of the molten core catcher vessel is destroyed, the passive valves for water supply to the molten core material are actuated. Water is supplied to the surface of the molten core material through the passive valves after heating of the thermosensitive element located behind the special shielding insert in the upper part of the thermal protection of the molten core catcher vessel to a temperature of 650°C. The selection of the shielding element design ensuring timely water supply to the surface of the molten core material is substantiated in the design analysis of thermophysical processes in the molten core catcher (Technical Report "Design-Basis Substantiation for the Molten Core Catcher of the Belarusian NPP, BLR1.B.110. & .&&&&&. &&&&&. 022.HG.0003, 2016").

If the valves fail, the operator opens the valves of water supply from the reactor internals inspection shafts.

After the molten core material is released to the molten core catcher, the emergency alkali supply system (JNB91) supplies sodium alkali to the containment pits for fixation of the iodine volatile forms.

To ensure the integrity of the containment system and protect the components of the localizing safety systems from hazardous and harmful factors arising from the combustion and explosions of hydrogen-containing mixtures, the design provides for the system for hydrogen removal from the containment (JMT).

The system for hydrogen removal from the containment (JMT) prevents formation of explosive mixtures in the accident localization area by maintaining the volumetric concentration of hydrogen in the mixture at a safe level.

To monitor hydrogen explosion safety, the design provides the hydrogen concentration monitoring system (JMU).

BDBA is monitored and controlled from BDBA panel CWL01 located in the MCR. Instruments located within the containment are designed for parameters of the external impacts during BDBA.

BDBA I&C system is powered from the BDBA power supply channel (Channel 7). UPS of channel 7 have batteries designed for 24 hours of discharge in the BDBA mode with blackout. For further operation of channel 7, it is possible to connect a mobile DG set.

Making-up of the emergency heat removal tanks is performed low-power high-pressure pump JNB50AP001 of the makeup system for the PHRS emergency heat removal tanks. This pump is located in the steam chamber and connected to the tanks of the LCU system.

Accident management approaches

In the adopted concept of severe accident management the operator's actions are minimized and shall be specified in the Severe Accident Management Guidelines. For severe accident management, operation of the automatic equipment having reliable power supply from the power supply system for severe accidents is provided.

Based on the practice of Severe Accident Management Guidelines development for operating and designed NPPs with VVER reactor plant, and taking into account the IAEA recommendations, a "symptom-oriented approach" is applied.

Severe Accident Management Guidelines are developed on the basis of the severe accident management strategies and related design analyses. Severe Accident Management Guidelines are applicable for NPP regardless of the operational status. Priority and strategies are selected on the basis of NPP state monitoring, with the appropriate selection of the operating personnel’s actions.

Based on the practice of Severe Accident Management Guidelines development for NPPs with VVER reactor plant, the focus is made on the following strategies:

- supplying water to the primary circuit;
- ensuring subcriticality of the molten core material;
- reducing the pressure in the primary circuit;

- reducing the pressure in the steam generator and ensuring cooldown through BRU-A (if SG PHRS is in non-operable condition);
- ensuring cooldown by heat removal through SG PHRS;
- suppling water to the spent fuel pool;
- suppling water to the containment PHRS;
- localizing the molten core material;
- reducing the release of the fission products.

Strategies of limited action are not considered.

The next stage in the implementation of the strategies specified in Severe Accident Management Guidelines is to ensure the integrity of the containment and to limit the releases.

Types of the containment failure can be grouped as follows:

- detonation of hydrogen;
- failure of the containment at high pressure.

The main threat from the containment destruction and failure is high release of the fission products. High release of the fission products requires immediate actions to ensure health protection and safety of the population and NPP personnel.

The containment integrity can be assessed through radiation monitoring at site. If the radiation background exceeds the design values, it is assumed that a threat to the containment integrity is created or implemented, and this threat requires immediate measures to limit the release and propagation of the fission products at site.

Elimination of a severe accident (after accidents of this grade the Unit is not put back into operation) is reduced to transfer of the affected Unit to a safe state, treatment of large quantities of the formed liquid radioactive wastes (water of the containment emergency sumps of the container for fuel cooling), development of the long-term moth-balling project for the affected Unit.

The hydrogen concentration monitoring system consists of two identical and completely independent measuring channels JMU10, JMU20.

Each channel of the monitoring system includes eight combined hydrogen and oxygen analysers with a temperature sensor (WS85PLUS), two pressure sensors ДАЭ -100T and a hardware-software complex "Data Processing Station of the Hydrogen Concentration Monitoring System".

The number of monitoring points and locations of the sensing elements of the hydrogen analysers are selected based on the results of the analysis of hydrogen propagation, accumulation and potential burning modes in the rooms under the containment. Control areas are identified in the rooms under the containment where hydrogen concentration increases rapidly during the accident. Each control area has at least one hydrogen analyser sensor of one of the measuring channels.

The range of measured hydrogen concentrations is from 0 to 25% by volume.

The diagram of location of the JMU system monitoring points in the rooms under the containment is provided.

Ensuring the containment integrity after severe fuel damage is described in Sections 7.2.3, 7.3.2-7.3.4.

Despite the fact that there are several different systems for implementing each of the accident management strategies, there are areas for further improvement in terms of the measures for management of beyond design basis (including severe) accidents. Taking additional technical measures and introduction of instructions for their use to ensure safety functions in case of loss of the design systems will improve the NPP ability for management of beyond design basis accidents at their severe stage.

In regard to administrative issues of severe accident management, BDBA Management Guidelines and Severe Accident Management Guidelines are updated for shutdown states of the Units, including those with the reactor head removed, accident management in case of fuel damage in the spent fuel pool. The ultimate goal of this activity is to ensure cooling of the reactor core and spent fuel in the spent fuel pool, as well as to prevent radioactive releases. It is also advisable to review regularly the developed Emergency Operation Procedure, BDBA Management Guidelines and Severe Accident Management Guidelines.

BDBA Management Guidelines and Severe Accident Management Guidelines provide for necessary and sufficient measures to manage severe accidents in terms of ensuring the integrity of the fourth safety barrier as well.

7.4 Measures for Severe Accident Management Aimed at Limiting Releases of Radioactive Substances

7.4.1 Releases of Radioactive Substances in Case of Loss of the Containment Integrity

The design provides for measures to prevent loss of the containment integrity.

Implementation of these measures for BDBA management ensures mitigation of severe accident effects by:

- preventing the core destruction at an early stage of the accident by using the pressure relief systems for the primary circuit;
- suppressing explosive concentrations of hydrogen by the combustion system to maintain the integrity of the containment;
- using the system passive heat removal from the inner containment (containment PHRS) to prevent exceeding of the maximum design pressure in the containment (0.49 MPa) and the design level of the containment leakage (0.2% of the volume per day);
- using the molten core catcher in case of the molten core release from the reactor vessel to prevent melt-through of the reactor building foundation slab;
- using the normal operation systems and actions of operating personnel to prevent significant radioactive releases.
- Safety Analysis Report on the Belarusian NPP shows that with account for implementation of the above measures for BDBA management with the containment integrity maintained, severe accident radiation effects do not exceed level 5 as per INES scale:

The estimated release to the environment is as follows: xenon-133 - 10^5 TBq; iodine-131 -100 TBq; cesium-137 -10 TBq.

The life-support system of the MCR/ECR equipped with efficient treatment of the supply air at the aerosol and iodine filters, as well as the civil structures of the double containment and control room building UCB allow for permanent stay of the personnel at the MCR/ECR to manage the accident. Safety Analysis Report on the Belarusian NPP shows that in case of severe accidents considered in the design, the personnel exposure in the MCR/ECR will not exceed the target limit of the effective equivalent dose of 25 mSv for the entire period of the accident and elimination of its effects specified in the Terms of Reference for the Belarusian NPP. The NPP site area, which can have significant levels of radiation contamination as a result of emergency release, shall not be used in the post-accident period to avoid unreasonable overexposure of the personnel. Radiation conditions at the site allows for short-term access of the personnel for a limited time with the use of personal protective equipment for skin and respiratory organs to assist and change the working personnel. To reduce the dose loads on the personnel when moving through the site, special vehicles can be used.

With additional failures in implementation of the BDBA management measures in case of a severe accident, loss of the containment integrity and localizing properties can occur due to pressure increase in the containment above 0.7 MPa, steam explosions or hydrogen explosions (with water boil-off in the spent fuel pool), which will lead to releases to the environment of a significant part of the radioactive substances accumulated in the reactor core and the spent fuel pool (probability is much lower than 10^{-7} 1/year). Loss of integrity of the containment - the last protective barrier of the defense-in-depth, leads to uncontrolled propagation of radioactive substances released during the accident from the damaged fuel. As a result of such an accident, urgent measures will be required to protect the personnel and the population in the Belarusian NPP area. Further measures to eliminate the consequences of the accident are developed based on the results of radiation survey and taking into account the actual radiation situation.

7.4.2 Management of Severe Accidents after Uncovering of Nuclear Fuel in the Spent Fuel Pool

Technical design solutions and operational measures to prevent uncovering of the fuel in the spent fuel pool are described in detail in Section 6.3 of this report.

In case of NPP blackout with failed starting of the standby diesel generators, complete drainage of the spent fuel pool can occur, resulting in heating-up and melting of the fuel assemblies located in the pool. At the maximum possible level of activity accumulated in the spent fuel pool for 10 years (420 spent fuel assemblies), taking into account complete emergency unloading of the reactor core (163 fuel assemblies with 3-day storage), the following typical results are expected:

- the time of water level reduction to the upper part of the FA is ~ 41 h, and the complete drainage of the spent fuel pool occurs in ~ 60 h after the beginning of the accident;
- the FA melted material start to reach the bottom of the spent fuel pool in ~ 60.5 h after the beginning of the accident;
- the spent fuel pool is in a subcritical state at any time during the accident;
- the release of the main significant radioactive nuclides (I-131, Cs-137, Xe-133, etc.) from the spent fuel pool to the containment is at the release level during the core melting in the reactor, taking into account that during the accident a part of the unloaded FA fuel is not heated to the melting point and located on the bottom of the spent fuel pool. The remaining FA groups in the spent fuel pool are damaged and partially melted, with a significant part of the fission products remaining in the closed porosity of the undestroyed fuel.

During BDBA with failure of all power sources which ensure making-up of the emergency heat removal tanks and spent fuel pool, the system of emergency water use from the reactor internals inspection shaft is used (JNB90). For the period of 24 hours and more, pump set JNB50AP001 ensures making-up of the emergency heat removal tanks and spent fuel pool from the demineralized water tanks (LCU). Active components of the JNB90 system have reliable power supply from batteries and a mobile DG set and can ensure the reactor plant safety in a hot state cooldown in case of NPP blackout.

7.4.3 Conclusion on the Sufficiency of Measures to Limit Releases of Radioactive Substances

The main measures implemented in the design to limit the releases of radioactive substances at BDBA are aimed at preventing severe damage to the fuel in the reactor core and in the spent fuel pool, as well as maintaining the integrity and localizing functions of the double containment.

Efficiency of the measures to limit emergency releases will be confirmed within the framework of implementation of PSA-2 (based on the results of full-scale PSA-1), where the probability of large radiation releases leading to global contamination of the area around the NPP set as the target criteria is not more than 10^{-7} (reactor/year).

In case of implementation of the technical and organizational measures for BDBA management aimed at maintaining the integrity and localizing functions of the double containment, the radiation effects of severe accidents will not exceed level 5 as per INES scale. Contamination of vast areas with radionuclides is excluded and mandatory introduction of protective measures affecting significantly the social and economic conditions and vital activity of the population (evacuation, resettlement) is not required. Protective measures for the population are limited to temporary sheltering, preventive iodine intake and restricted of consumption of local contaminated food in the NPP surrounding area. Provision is made for the personnel to stay in the MCR/ECR to manage the accident and transfer the Unit to a safe state.

8. GENERAL CONCLUSIONS

8.1 Basic Measures Implemented to Improve the NPP Reliability

The safety systems of the Belarusian NPP are designed with comprehensive consideration of the external, and the buildings, structures and equipment of the Belarusian NPP are designed with account for the design impacts specified in the report in accordance with the current regulatory framework.

There are no radiation consequences of DBE and SSE, and no additional improving measures are required. The assessment of the load-bearing capacity reserves available in the building structures (security of strength characteristics, reserves due to elastoplastic behavior of the structures, etc.) showed that for the structures of seismic category I the reserve relative to the SSE level adopted in the design is not less than 4.9 times (0.62 g), for the inner containment - not less than 4.3 times (0.51 g).

The threshold value of the accelerations specified above for the structures of seismic category I is determined with a sufficient degree of conservatism. If the conservatism is not considered, the threshold level can be further increased. The structures are made of cast-in-situ reinforced concrete, which excludes the brittle instantaneous failure (*cliff-edge effect*) in case of increased seismic impact.

The main equipment of the reactor plant: reactor (except for the metal structures of the spent fuel pool), SG, RCPS, reactor coolant pipeline, pressurizer, electrical connection block, connecting pipeline have the necessary reserves for load accommodation at 8-point SSE.

Strength conditions at 8-point SSE are not provided for ECCS, injection and discharge pipelines of the pressurizer system, metal structures of the reactor upper unit and the spent fuel pool. The RCPS anti-seismic fixation rod does not have sufficient strength for the combination of loads NO + DBE (7 points) + DBA.

The strength threshold value for the equipment and pipelines under seismic impact is the SSE level of 7 points adopted in the Belarusian NPP design with an acceleration of 0.12 g. If this threshold level is exceeded, there is a safety margin of not less than 1.07 times before a relative plastic deformation of 0.2% due to the determination of the permissible stress as $[\sigma]=\max\{R^T_m/2.6, \sigma^T_{p0,2}/1.5\}$. Taking into account the adopted safety margin for the equipment and pipelines, the maximum permissible acceleration is 0.12x1.07 = 0.13g. With a further increase of stresses, areas of significant plastic deformation can be expected in the equipment and pipelines.

The NPP site is not exposed to flooding caused by the surrounding rivers and water basins, as the grading elevation of the site (179.3 m BES) is 51.5 m higher than the water rise level with 0.01% probability. This scenario takes into account breakthrough of dams, the highest levels of seasonal floods and rainfall floods, ice blocking, wind surges and other dangerous factors.

Upstream water drainage systems (stratum drainage, storm water sewerage and catch drain) are designed with account for extreme precipitation.

The design solutions for the catch drain ensure throughput of the design water flow, while the possibility of the sludge setting and overgrowing of ditches with vegetation (clogging), leading to a failure of this water drainage system is excluded during operation.

When power supply is cut off and the pumping systems of the storm water sewerage are in non-operable condition, some of the precipitation volume will be in the pipelines and wells of the sewerage systems. The remaining estimated volume will be distributed over the whole area of the NPP and the precipitation layer will be 5.3 mm. In view of the fact that the paving around the buildings is 150 mm, and the buildings themselves have waterproofing of the walls of the underground part, this underflooding will have no impact on the equipment located in the buildings.

Thus, flooding with water rising to the first floor level of the buildings is impossible. No special flood prevention measures are required in the design.

The maximum values of extreme climatic conditions determined for the site are much lower than those used in the design.

In order to avoid fuel damage in the reactor in an accident involving the loss of all AC power supplies at the NPP during the power operation of the reactor plant, it is required to take measures to restore and maintain the water reserve in the emergency heat removal tanks not later than within 72 hours from the beginning of the accident (provided that the water reserves of the four emergency heat removal tanks are used) in order to ensure the operation of the PHRS (with all the emergency heat removal tanks involved). In this case, the time allowance before the beginning of the reactor core heating and exceeding of the maximum design damage limit for the fuel rod of maximum power will be determined by the amount of boric solution in the ECCS hydro accumulators. The engineering assessment shows that the time allowance before the beginning of the reactor core heating may vary from 13 to 15 days from the beginning of the accident, provided that the rate of pressure decrease in the primary circuit (and, correspondingly, in the ECCS hydro accumulators) is stable [31]. Later, it will be required to restore the design NPP power supply and supply boric solution to the primary circuit using the design means.

In order to avoid fuel damage in the spent fuel tank in case of an accident with loss of all AC power supplies at the NPP during the power operation of the reactor plant, it is required to supply water to the spent fuel tank at a rate of not less than 7 kg/sec within not more than 41 hours.

On January 16-20, an IAEA mission for the safety assessment of the Belarusian NPP (SEED-mission) was held in the Republic of Belarus. In the course of the mission, both natural and man-caused external impacts were analyzed and characterized, the design parameters of the construction site were studied, the site and the environment were monitored and the lessons learned at the Fukushima NPP accident were taken into account.

Based on the results of the mission, IAEA experts noted that the NPP design parameters take into account external threats typical for the site such as earthquakes, floods and extreme weather conditions, as well as man-caused events. International experts noted that the threat monitoring programs to be implemented throughout the life cycle of the Belarusian NPP are sufficient and properly provided in the NPP design. It was also noted that the Belarusian side took additional measures related to external events in view of the lessons of the Fukushima NPP accident.

8.2 Safety Issues

The buildings and structures have a significant safety margin in relation to the design seismic impact. Threshold effects for the buildings and structures do not occur at seismic impacts.

To enhance the NPP seismic stability, the measures presented in Section 8.3.1 can be provided.

The NPP site is not exposed to flooding. To determine the safety margins with respect to flooding, a conservative analysis of the postulated flooding conditions for systems and components of the safety systems and safety-related systems located below elevation 0.00 was performed. The results of the analysis show that in case of flooding of the equipment of the safety systems and safety-related systems located below elevation 0.00, the NPP can be transferred to a safe state and maintained in this state for the following period of time:

- for the reactor plant: residual heat is removed by BRU-A or SG PHRS within not less than 72 hours.
- for the spent fuel pool: residual heat is removed by water boiling (in case of FAK system failure due to flooding) within not less than 41 hours.

With the flooding of the NPP site postulated as an initiating event, the measures listed in section 8.3.2 can be provided to increase the time of the NPP remaining in a safe state.

The report confirms the NPP safety at the initiating events related to the loss of power supply and the loss of the ultimate heat sink, including the combination of the initiating events which occurred at the Fukushima-1 NPP.

However, to improve the NPP stability in case of events occurring at several NPP Units simultaneously, measures to improve the NPP stability in case of the loss of power supply and the loss of the ultimate heat sink listed in Section 8.3.3 can be provided.

Measures taking into account the results of the Belarusian NPP safety reassessment will be implemented within the framework of the Belarusian NPP Safety Enhancement Program with account for the degree of their impact on safety.

Within the framework of the Safety Enhancement Program, the Technical Council of the Enterprise will propose solutions in regard to the proposed measures, their necessity, dates and procedures of their implementation, in accordance with the prescribed procedure. The stages of the Security Enhancement Program are as follows:

1. Substantiation of the modification.
2. Development of the modification project.
3. Preliminary approval of the modification project.
4. Solving the issues of the modification financing.
5. Informing the involved state administrative bodies (Gosatomnadzor and the Ministry of Energy) and, if required, having the modification project approved by them.
6. Final approval and including of the modification project into the list of activities under the Program.
7. Selection of a contractor for implementation of the modification.

8.3 Possible Measures to Improve the NPP Safety in the Future

8.3.1 Possible Measures to Improve the Safety of NPP in Case of an Earthquake

In order to improve the seismic resistance, the following measures can be provided:

1. Seismic resistance of the ECCS and pressurizer system, as well as pipeline systems of small diameter, can be improved by installing additional anti-seismic supports along the length of the pipelines.

2. To improve the seismic resistance of the ECCS hydro accumulators and the racks of the spent fuel pool, the support design can be modified. For the racks of the spent fuel pool, stops can be installed to limit the rack movement in the horizontal plane. To improve the seismic resistance of the spent fuel pool metal structure, additional anti-seismic fixation can be applied. It is planned to install the fixation on the metal structure of the electrical connection block in order to limit the structure movement.

3. To improve the seismic resistance of the RCPS anti-seismic fixation rod at high earthquake level (8-point SSE, 7-point DBE) for combination of loads NO + DBE + DBA, the rod shall be strengthened.

4. For the safety systems and the safety-related systems, additional measures to improve the seismic resistance can be determined by the operating organization after the NPP startup, based on the SMA methodology [31]. When applying this methodology to the process flow diagram, the components critical for safe shutdown of an operated NPP are identified and estimated. The analysis is performed on the basis of engineering experience using the results of the NPP walkdown inspections for seismic stability, data on the actual state of the equipment fixation, etc. The necessity to improve the NPP seismic stability level will be determined by the operating organization based on expert findings of national or international organizations for supervision of atomic energy safe use.

5. To mitigate the consequences of earthquakes exceeding the design requirements, the following organizational measures are proposed:

5.1. assess the documentation on the personnel actions for diagnostics of the NPP state in emergency situations caused by earthquakes when the seismic impact exceeds the design level;

5.2. finalize the documentation on the personnel actions for diagnostics of the NPP state, restoring of the normal operation conditions and disordered safety functions, preventing or limiting the effects of the core damage, including Process Regulations, Reactor Plant Emergency Response Manual, BDBA Management Manual, Severe Accident Management Manual, as well as Action Plan for Personnel Protection in Case of an Accident, which will contain sections providing for the measures to solve the following tasks:

- ensure safe shutdown of the Unit, when the emergency protection signal is initiated, both by automatic actions (as the main method for solving the task), and by the control actions of the personnel;
- use corrective actions by the personnel if required;
- keep the core under the coolant and ensuring the coolant circulation;
- bring the Unit to a final state by actions of the personnel, allowing for performing the restoration activities;
- control the equipment of the normal operation systems and safety-related systems to bring the Unit to a safe final state and limit the radiation effects of malfunctions and accidents;
- use the standby controls for reliable maintaining of the NPP predetermined mode in case of failure of the main controls for the specified time period required to eliminate the failure;
- assess the degree of destructions at the NPP caused by an earthquake;
- maintain or restore functions of the systems and equipment in case of destructions caused by an earthquake;
- localize accidents, limit their development and ensure the limits of safe operation by providing the core subcriticality, keeping it under the coolant (with account for the provided design reserves, speed and efficiency of the protective systems);
- ensure the integrity of unbroken physical barriers;
- achieve the final state where the fission chain reaction is discontinued, the reactor subcriticality is ensured and the core re-criticality is prevented, with account for its possible damage;
- prevent (mitigate) severe damage of the fuel by both automatic actions of the systems and control actions of the personnel;
- prevent damage of the reactor vessel and primary circuit equipment;
- prevent damage of the containment;
- prevent severe damage of the core and mitigate the effects of the core severe damage;
- limit the radiation impact on the personnel, population and environment.

5.3. arrange a permanent (fixed) local seismic monitoring network for the NPP operation period to obtain current objective data on changes of the geodynamic situation in the area of the facility location by detecting weak earthquakes which occur on activated tectonic structures; obtain new data on the resonance properties of the soils at the NPP site for protection against future maximum earthquakes by introducing changes in the natural periods of oscillations of the facilities and essential structures in order to avoid the resonance effects; obtain more precise and additional data for determining the quantitative parameters of the design basis earthquake (DBE) and safe shutdown earthquake (SSE) from local and remote seismically active areas for making more precise estimated accelerograms and response spectra; obtain current data on the parameters of seismic impacts on buildings and essential structures during intense local and remote earthquakes for making decisions on the necessity of additional verification of reliability of the facilities and equipment which have experienced significant seismic impacts; obtain additional data for setting the parameters of the vibration protection sensors.

8.3.2 Possible Measures to Improve the Safety of NPP in Case of Flooding

To improve the NPP stability in case of flooding, the following measures can be provided:

1. With regard to the heat removal from the reactor plant, after 72 hours it is necessary to arrange for making up the LCU tanks from the on-site and off-site water sources.

2. With regard to the residual heat removal from the spent fuel pool:

after 41 hours it is necessary to arrange for making up the spent fuel pool. This measure can be implemented by connecting non-standard facilities (a fire engine with a pump set with a capacity of 40 l/s and a head of 100 m) to the two process connectors of the JNB50 system located on the outside of the UJE building (at elevations +0.690 and +0.730, with the water taken from the LCU tanks through the pump set of the fire engine and then supplied through the pipelines of the JNB50 system to the spent fuel pool) with blind flanges installed on them;

the process flow diagram of the JNB50 system shall be modified by adding a tie-in of the check valve bypass on the make-up line for the emergency heat removal tanks. This solution will allow the operating personnel to make up the spent fuel pool after 41 hours.

8.3.3 Possible Measures to Improve the NPP Safety in Case of Loss of Power Supply and Loss of the Ultimate Heat Sink

In order to increase the time of the NPP remaining in a safe state in case of loss of the power supply and loss of the ultimate heat sink, it is necessary to implement the measures presented in Sections 6.1.5, 6.2.5, 6.3.3 of this report, and the measures similar to those presented in Section 8.3.2 may be also recommended.

In addition, the possibility of UPS recharging shall be considered.

In order to ensure reliable connection of a mobile DG set, it can be provided with standby facilities to allow for its connection at its design location.

Also, to improve the NPP stability state in case of loss of the power supply and the ultimate heat sink at two NPP Units at the same time, two mobile DG sets will be provided, one set per NPP Unit.

In addition, it is required to assess the possibility and necessity of arranging for emergency power supply from the Yanovskaya HPP and Vileyskaya HPP and assess the efficiency of their use in case of complete loss of the power supply.

8.3.4 Possible Measures to Improve the NPP Safety in Terms of Accident Management

For the accident management at the Belarusian NPP Emergency Operation Procedure, BDBA Management Guidelines and Severe Accident Management Guidelines (symptom-oriented) are developed and introduced. These manuals include the procedure and criteria of transfer from one manual to the other, as well as scopes of their application and mutual links

The organizational and technical measures for management of accidents, including severe ones are described in Section 7 of this document.

REFERENCES

1. Convention on Early Notification of a Nuclear Accident (since 1987).

2. Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency (since 1987).

3. Treaty on the Non-Proliferation of Nuclear Weapons (IAEA Safeguards) (since 1993).

4. Convention on the Physical Protection of Nuclear Material (since 1993).

5. Agreement between the Republic of Belarus and the International Atomic Energy Agency on the application of safeguards in connection with the Treaty on the Non-Proliferation of Nuclear Weapons (1995).

6. Convention on Civil Liability for Nuclear Damage (Vienna Convention) (since 1998).

7. Convention on Access to Information, Public Participation in Decision-making and Access to Justice in Environmental Matters (Aarhus Convention) (since 1999).

8. Convention on Nuclear Safety (since 1999).

9. Comprehensive Nuclear-Test-Ban Treaty (since 2001).

10. Protocol amending the Vienna Convention on Civil Liability for Nuclear Damage (since 2003).

11. Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (since 2003).

12. International Convention for the Suppression of Acts of Nuclear Terrorism (since 2005).

13. Convention on Environmental Impact Assessment in a Transboundary Context (Espoo Convention) (since 2005).

14. Additional Protocol to the Agreement between the Republic of Belarus and the IAEA on the Application of Safeguards in Connection with the Non-Proliferation Treaty (signed in 2005, the ratification procedure is not completed).

15. Resolution of the Security Council of the Republic of Belarus of January 31, 2008 No. 1 "On the development of nuclear energy in the Republic of Belarus".

16. The Law of the Republic of Belarus of July 30, 2008 "On the Use of Atomic Energy".

17. Law of the Republic of Belarus of January 5, 1998 "On Radiation Safety of the Population".

18. Decree of the President of the Republic of Belarus of 1 September 2010 No. 450 "On licensing of certain types of activities".

19. Decree of the President of the Republic of Belarus No. 62 dated 16.02.2015 "On ensuring security in the construction of the Belarusian Nuclear Power Plant".

20. Decree of the President of the Republic of Belarus No. 756 of December 29, 2006 "On Certain Issues of the Ministry of Emergency Situations".

21. Ordinance of the Prime Minister of the Republic of Belarus of May 4, 2017 No. 158r "On the establishment of an interdepartmental working group".

22. TCCP 566-2015 "Assessment of the frequency of severe damage to the reactor core (for external source events of natural and man-made nature)."

23. Norms and regulations for ensuring nuclear and radiation safety "Requirements for carrying out stress tests (targeted reassessment of safety) of the nuclear power plant", approved by the resolution of the Ministry of Emergency Situations of the Republic of Belarus dated 12.04.2017 No. 12.

24. Resolution of the Council of Ministers of the Republic of Belarus of December 7, 2010 No. 1781 "On approval of the Regulations on the procedure for the examination of documents substantiating the provision of nuclear and radiation safety in the implementation of activities in the field of the use of atomic energy and sources of ionizing radiation."

25. Decree of the Council of Ministers of the Republic of Belarus No. 133 dated February 25, 2015 "On Approval of the Regulation on the Organization and Implementation of Control

(Supervision) of Providing Security in the Construction and Commissioning of the Belarusian Nuclear Power Plant".

26. Resolution of the Council of Ministers of the Republic of Belarus of December 30, 2011 № 1791 "On the establishment of a working group to coordinate the implementation of state control (supervision) for the construction of a nuclear power plant."

27. The Law of the Republic of Belarus of May 5, 1998 "On Protection of the Population and Territories from Emergencies of Natural and Man-Caused Nature".

28. The Criminal Code of the Republic of Belarus of 9 July 1999

29. The Code of Administrative Offenses of the Republic of Belarus of April 21, 2003.

30. Information and communication strategy of the Gosatomnadzor for 2016-2018 and for the period until 2020, approved by the decision of the board of the Gosatomnadzor on January 28, 2016.

31. Report on the conduct of a targeted reassessment of safety (stress tests) of the Belarusian NPP" BL-11752.

32. Analysis of seismic resistance of the main equipment of the reactor unit of units 1,2 of Belarusian NPP at 8-points MDBE, 491-Pr-1975.

33. Decree of the President of the Republic of Belarus of 29 March 2011. No. 124 "On measures to implement international agreements in the field of civil liability for nuclear damage".