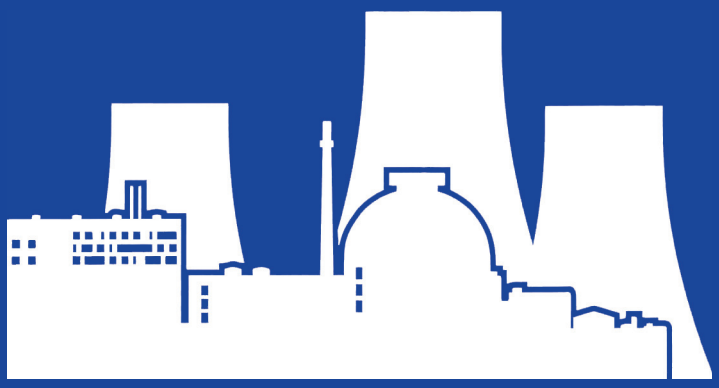
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SAFETY SYSTEMS OF NPP-2006 DESIGN



POCATOM

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SAFETY SYSTEMS OF NPP-2006 DESIGN

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ABBREVIATIONS

Acc.	—	Accumulators
ALA	—	Accident Localization Area
ALZ	—	Accident Localization Zone
APCS	—	Automated Process Control System
AR	—	Absorber Rod
APP	—	Accelerated Preventive Protection (UPZ)
ATWS	—	Anticipated Transient without Scram
BA	—	Burnable Absorber
BDA	—	Beyond Design Basis Accident
BPD	—	Blast Protection Damper
BRU-A	—	Atmospheric Dump Valves
BRU-K	—	Turbine Bypass Valve
BZOK	—	Fast Acting Isolating Valve
CCF	—	Common Cause Failure
CCTV	—	Closed Circuit TV system
CIU	—	Command Issue Units
CDA	—	Complex of electricity Distributon Arrangement (KRU)
CPS	—	Control and Protection System
CRDM	—	Control Rod Driven Mechanism

DACA-V – Design Acceptance Criteria for Accidents
DBA — Design Basis Accident
DBC — Design Basis Condition
DCB — Direct Current Board
DCS — Digital Control System
DCV — Double Check Valve
DEC — Design Extension Conditions
DGIS — Galvanic Isolation of Signals
DNB — Departure from Nucleate Boiling
DNBR — Departure from Nucleate Relative
DPS — Diversion Protection System
ECCS — Emergency Core Cooling System
ECR — Emergency Control Room
EPSS — Emergency Power Supply System
ESFAS — Engineered Safety Feature Actuation System (USBoT)
FA — Fuel Assembly
FC — Fuel Cycle
FCS — Federal Code Safety (rules)
FE — Fuel Elements
FMCS — Functions of the Monitoring and Control System (FSKU)
FPGA — Field Programmable Gate Array
FSAR — Final Safety Analysis Report
FTA — Failure Tolerance Assessment
Gd — Gadolinium
GT — Guiding Tube
I&C — Instrumentation and Control
IAEA — International Atomic Energy Agency
ICSS — Initiating Controlling Safety Systems

IOPRS — Important Operating Parameters Record System
LER — Large Emergency Release of Radioactive Substances Into The Environment
LOCA — Loss of Coolant Accident
MCC — Main Circulation Circuit
MCPL — Minimum Controllable Power Level
MCR — Main Control Room
MDBA — Maximum Design-Basis Accident
MDL — Maximum Design Limit of Fuel Elements Damage
MSIV — Main Steam Isolation Valve
NAA — Non-accessible area
NF — nuclear fuel
NFCE — Neutron Flux Control Equipment
NFMS — Neutron Flux Monitoring System (AKNP)
NO — Normal Operation
NOS — Normal Operating Systems
NPP — Nuclear Power Plant
NVNPP-2 — Novovoronezh Nuclear Power Plant -2
OL (OC) — Operating Limits (Operating Conditions)
OLTC — On-Load Tap Changer
OPB — Basic Safety Regulation Rules for NPP
OPV — overpressure valve
PCHR — Passive catalytic hydrogen recombiner
PCM — Priority Control Modules
PHRS — Passive Heat Removal System
PLE — Priority Logic Equipment
PORV — Power operated relief valve
PP — Preventive Protection

PPPE — Process Parameter Protection Equipment
PRZ — Pressuriser
PSCS — Pre-Stressed Containment System
PZ — Preventive Protection
RAA — Restricted access area
RCP — Reactor Coolant Pump
RCPFS — Reactor Core Passive Flooding System
RCPS — Reactor Control and Protection System
RCS — Reactor Coolant System
RF — Russian Federation
ROM — Regulator of Power Limiting Controller
RPM — Pressure-Reducing and Protection Module
RPP — Reactor Preventive Protection
RPV — Reactor Pressure Vessel
RS — Reactor Scram
RSWS — Reactor Shop Work Station
SB — Storage Battery
SCSPP — Safety Control System by Process Parameters
SFP — Spent Fuel Pool
SG — Steam Generator
SG ESC — Steam Genelabor Emergency Cooling System
SG SV — Steam Generator Safety Valve
SHS LP — Software and Hardware Systems of Local Protection
SHSM — Standard Hardware and Software Means
SOL (SOC) — Safe Operation Limits (Safe Operation Conditions)
SP — Safety Panels
SPU — Signal Propagation Unit
SVU — Steam Valve Unit (PAB)

SW — Software
TG — Turbogenerator
TPTS — Automatic Control System (Type TELEPERM-XS)
TSWS — Turbine shop work station
UPS — Uninterruptible Power Supply Unit
UPZ — Accelerate Preventive Protection
USBI — ICSS
USBT — ESFAS
UULS — Upper Unit Level System

FOREWORD

Nuclear Power Plants (NPP) with VVER type reactors are under construction in Russia and abroad since 60-th of the past century (the first VVER type power unit was commissioned at Novovoronezh NPP in 1965).

At present Nuclear power Plants (the design NPP – 2006) with reactor plant VVER-1200 are successfully operating at Novovoronezh NPP-2 (Units 1 and 2) and Leningrad NPP-2 (Unit 1).

NPPs with VVER —1200 reactors are under construction not only in Russia but in many foreign countries such as Belarus, Bangladesh, Turkey. Negotiations and contracts awards are underway with Egypt, Hungary, Finland and Uzbekistan. In addition there are plans for construction of NPP - 2006 units in China and India (new sites). The shaping factor enabling contract award and VVER type NPPs construction is availability of advanced (generation 3+) VVER-1200 reference power units with proved high safety level. However, these plans have the right to exist only under one condition – unconditional NPP safety assurance.

High safety level is assured by reliability of the main equipment and availability of special purpose safety systems as well as beyond design accident management facilities (not available in previous generations power units).

NPP acquiring the following features can be referred to 3+ generation:

— extensive safety systems including those which are based on passive operation principles;

— double containment to mitigate accidents consequences and protect against external impacts;

— special systems and equipment for severe accidents management (corium catcher or reactor pressure vessel cooling system);

— improved performance indicators.

The authors express most heartfelt thanks to JSC “Atomenergoproject”, in particular, the deputy director of Moscow branch Gorshkov V.I. and Kudankulam NPP design director Vorontsov V.V. for interest in work and provided support.

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INTRODUCTION

The main goal of the book is to provide description of safety principles, equipment layout and safety systems operation of reactor plants with VVER-1200.

In further description by "NPP safety" we mean nuclear safety and not physical security or radiation safety or industrial safety, otherwise it will be stipulated separately.

This workbook is intended, first of all, for specialists of VVER type reactors NPPs and specialized technical support organizations in nuclear infrastructure "Nuclear Safety". Authors believe that the book will be helpful as the work book for students of "nuclear" field as well as for specialists in the field of nuclear power use.

In preparation of the present book efforts were made to make the book friendly user and interesting for readers without special background in this area, but experiencing interest in advanced and state-of-the-art technologies.

In the book safety systems of Novovoronezh NPP-2 Units 1 and 2, designed within the design NPP-2006 with reactor plant B-392M, are considered.

It should be noted that the difference between reactor plants of Novovoronezh Units 1 and 2 and Leningrad NPP-2 is minor. At the same time the difference in technical solutions for safety systems and general layout of Units is quite significant.

As for gage pressure values it will be mentioned a priori, if it was not stipulated separately.

1 NUCLEAR SAFETY CONCEPT OF NVNPP-2 WITH VVER-1200 TYPE REACTORS

1.1 GENERAL APPROACH TO NUCLEAR SAFETY OF NPP

1.1.1 GENERAL TERMS AND DEFINITIONS

Prior to start presentation of NPP safety issues it is required to introduce main terms and definitions.

In compliance with NP-001-15/1/ **NPP safety** (nuclear and radiation safety of the NPP) - the NPP characteristic to provide reliable protection of the personnel, the public and the environment against any radiation exposure unacceptable in accordance with the federal rules and regulations in the area of nuclear energy use. This definition implies division into nuclear and radiation safety, adopted in the Russian Federation.

In IAEA documents the above listed description covers the term “nuclear safety” (its definition is given in IAEA Safety Glossary/2/, the same understanding of this term is used in Convention on Nuclear Safety), considering that the sources of all specific hazards at NPP are nuclear processes.

In Russian nuclear related terminology the term “nuclear safety” historically had narrow meaning, first of all, as prevention of self-sustainable non-controllable fission chain reaction.

Safety concept is closely related to accident risk.

In compliance with NP-001-15 **accident at the NPP (accident)** is an operational occurrence at the NPP with a release of radioactive substances and (or) ionizing radiation beyond the borders stipulated in the NPP design documentation for normal operation in the amounts exceeding the established safe operation limits;

an accident is characterized by the initiating event, development scenarios and consequences.

Nuclear accident is an accident accompanied by damage of fuel elements (FE) exceeding the established safe operation limits or an accident without FE damage caused by: any disturbance in the chain fission reaction monitoring and control; criticality occurrence in the course of refueling, transportation and storage of nuclear fuel.

Nuclear accident is the accident involving a fuel rod damage in excess of the safe operation limits and/or personnel exposure in excess of the permissible limits, as initiated by:

— disruption of monitoring and control over the nuclear fission reaction in the reactor core;

— criticality during refueling, transportation and storage of nuclear fuel;

— disruption of heat removal from FEs;

— other causes leading to FE damage.

Thus, definition of nuclear accident at NPP as per NP-001-15 is very similar to IAEA definition and includes not only accidents related to criticality but any accidents related to fuel rod damage.

The beyond design basis accident is an accident caused by initiating events not considered for design basis accidents or accompanied by additional failures of safety system components as compared to design basis accidents in excess of a single failure and implementation of erroneous personnel decisions.

Special-purpose hardware for beyond design basis accident management - systems (components) are provided in the NPP design for beyond design basis accidents management. In addition, for beyond design basis accident management all available hardware in proper conditions intended for normal operation and belonging to safety systems, can be used.

The severe accident is a beyond design basis accident with damage of fuel elements exceeding the maximum design limit.

Large-scale emergency release is a release of radioactive substances into the environment in case of any accident at the NPP when it is necessary to implement measures for protection of the public within the initial stage of the accident at the border of the protective actions planning zone established in accordance with the requirements of NPP siting regulations and rules and outside it.

NPP safe operation limits are the values of the process parameters established in the NPP design, any deviations from which, can lead to an accident.

There are safe operation limits with regard to radiological parameters and safe operation limits related to other process parameters.

Exceedance of the safe operation limits with regard to radiological parameters is an accident.

Violation of safe operation limits, characterized by other process parameters, results in the accident under additional conditions.

Safe operation limits are established for the purpose of emergency measures to be taken to control technological process, intended for accident prevention. In other words, in order to protect physical barriers along the way of radioactive substances spreading into the environment. Safe operation limits shall be established for all stipulated in the design states and normal operation states, reflected in the NPP Technical Specification.

Safe operation conditions are the minimum requirements for quantity, characteristics, operability, scope, frequency and other conditions of maintenance, control and testing of safety-related systems (components) established in the NPP design when compliance with the safe operation limits and (or) safety criteria is assured.

Together with safe operation limits, safe operation conditions determine the area for which the safety justification of the nuclear power plant is performed (both the absence of accidents is justified, and the NPP readiness to limit design accidents to the limits established for such accidents in case of IE design basis accident).

Safe operation conditions include, first of all, minimum requirements for the number and operability states of systems and components important for safety. In particular, these are requirements to the minimum number of operable systems (trains of systems) and safety elements, where safety is ensured in case of design basis accidents (taking into account the normative principle of single failure).

Operation limits are the values of parameters and characteristics of the systems (components) and the entire NPP, established in the NPP design for normal operation.

Operation conditions are conditions for the quantity, characteristics, operability and maintenance of systems (components) established in the NPP design and required for operation without any deviations from operation limits.

Design limits are the values of parameters and characteristics of the systems (components) and the entire NPP established in the NPP design for normal operation and anticipated operational occurrences.

Design limits include operation limits, safe operation conditions as well as other limits (including design limits for design basis accidents). Limits shall be established in order to identify boundaries of different plant conditions and phases of processes development.

The design basis accident is an accident with the initiating events and end states defined in the NPP design and the provided safety systems aimed to assure mitigation of its consequences to the limits established for such accidents in case of any failure of a safety system component independent from the initiating event and considered in the NPP design or in case of a human error independent from the initiating event.

The term “design basis accident” is stipulated not by the fact that such accidents are covered by the design, but they are considered by the design for further safety systems development.

1.1.2 FUNDAMENTAL SAFETY OBJECTIVE AND SAFETY FUNDAMENTALS

The fundamental safety objective is to protect people and the environment from harmful effects of ionizing radiation. The fundamental safety objective applies for all facilities and for all stages over the lifetime of a facility starting from siting and ending with decommissioning.

For Nuclear Power Plants **fundamental safety principles** are defined in IAEA documents and can be grouped in the following way:

1) Management responsibilities;

2) Defense in-depth;

3) Technical principles .

To achieve fundamental safety objectives it is required to implement a number of **fundamental safety principles**, such as:

— safety culture;

— application of proved engineering practices (proved operability);

— quality assurance;

— operating experience;

— defense- in- depth;

— accident prevention and mitigation; radiation protection; human factor; safety assessment and verification;

— operational excellence of all safety related executors;

— etc.

In compliance with modern safety standards (the RF and IAEA) probability of the reactor core damage shall not exceed the value $1*10^{-5}$ reactor year, probability of radiactivity release off the containment shall not exceed the value $1*10^{-7}$. Radiological impact on population and environment shall not exceed 1 % of dose received from natural background radiation sources.

1.1.3 NPP OPERATION MODES

NPP safety assurance is closely related to the main production and business objectives and does not contradict to the current operation of the plant. Measures, taken at the plant on accidents and failures prevention, are aimed at Unit Capability Factor and economic efficiency improvement.

In terms of fulfilment of fundamental operation objectives, listed in the previous section, it is very important to establish compliance between modes and operation limits.

Modes of operation in simple cases are defined in the following way. Normal operation is the NPP operation within the operation limits and conditions defined in the NPP design. Abnormal operation is the NPP operation with deviations from the operation limits or conditions. Definitions of design basis accidents and beyond-design basis accidents are given in the section “Main terms and definitions”.

Interrelation between modes of operation and operation limits, process protections and interlocks actuation and reactors scrams are shown on Figure 1.1.1.

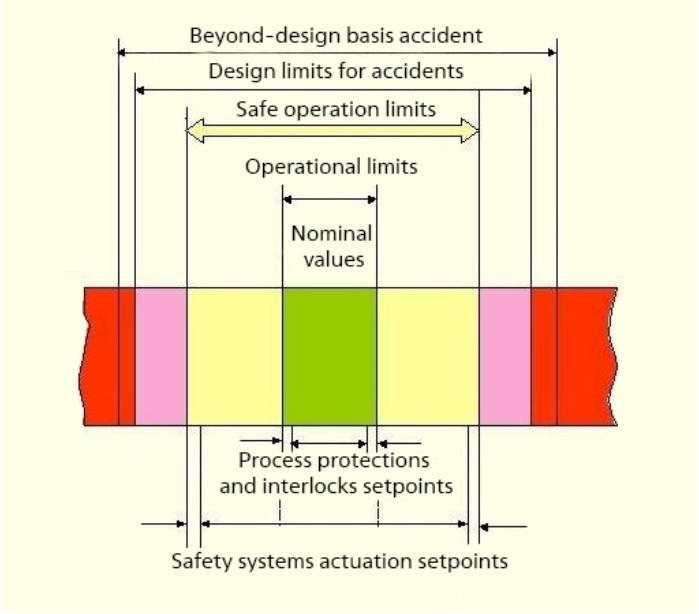


Figure 1.1.1 - Interrelation between modes of operation and operation limits

Design modes depending on frequency of initiating events occurrence are divided into four categories (see Figure 1.3.3):

— Category 1 – normal operation;

— Category 2 – failures conditions (frequency of initiating events per year: $f \geq 10^{-2}$, initiating events correspond to operation with deviations in compliance with OPB);

— Category 3 – postulated accidents of Class 1 (frequency of initiating events per year: $10^{-2} > f \geq 10^{-4}$);

— Category 4 – postulated accidents of Class 2 (frequency of initiating events per year: $10^{-4} > f \geq 10^{-6}$).

In addition to provisions stipulated in the design, in order to meet requirements of deterministic design modes, certain provisions about additional conditions of

designing were defined, which is according to the definition of OPB corresponds to beyond design basis accidents. It is done in order to focus on necessity to implement measures, including upgraded or additional equipment, or accident management procedures for:

— Complex sequences, including failures in addition to deterministic design conditions, but not leading to the reactor core melting;

— Accidents with the reactor core melting.

Beyond design bases accidents are divided into 2 categories:

— DEC-A – emergency scenarios for which it is possible to prevent accident progression into severe stage by means of technical facilities stipulated in the design

— DEC-B – emergency scenarios of severe accidents.

In accordance with EUR standards the first of the above mentioned categories of beyond design basis conditions (DEC-A) has identification “Complex sequences”, and the second (DEC-B) – “severe accidents”.

In accordance with NP-001-15 the NPP safety targets are as follows:

— non-exceedance of cumulative severe accident probability for each NPP unit within a year equal to 10^{-5};

— non-exceedance of cumulative probability of a large-scale emergency release for each NPP unit within a year equal to 10^{-7};

— non-exceedance of cumulative severe accident probability for nuclear fuel storage facilities at the NPP (not included into the NPP units) within a year equal to 10^{-5}.

Probabilistic safety indicators of NVNPP-2 by significant emergency release is in line with NP-001-15 (less than 10^{-7} for one year interval), and for severe accidents are more strict (less than 10^{-6} for one year interval), which allows to refer NVNPP-2 Units to generation “3 +”.

The NPP safety shall be ensured by consistent implementation of defense-in-depth based on application of:

— a system of physical barriers on the way of ionizing radiation and radioactive substances spreading into the environment;

— a system of technical and administrative measures for protection of barriers and maintenance of their efficiency as well as for protection of the personnel, public and environment.

These topics will be discussed in details.

1.2 SYSTEM OF PHYSICAL BARRIERS

In order to ensure NPP safety, the concept of a system of physical barriers to the spread of ionizing radiation and radioactive substances into the environment is introduced.

The physical barriers system in accordance with the NVNPP-2 design includes:

Barrier 1: Fuel matrix

The main characteristic of the fuel matrix state is its ability to retain radioactive fission products, including gaseous and volatile ones.

In particular, in case of change of the fuel aggregate state when the melting point is reached, the fuel matrix loses the ability to retain radioactive products, which is considered as damage to the first physical barrier.

Such achievement of the fuel maximum temperature of melting in a local zone, first of all in the central part of fuel element/gadolinium fuel element means transition to a pre- emergency situation, or to accident. At the same time, both from the point of view of temperature effect value on the fuel matrix and from the point of view of the physical barrier state, a single limit of safe operation, characterizing the possibility of damage to the fuel matrix as the first physical barrier, can be considered.

Barrier 2: Fuel element/fuel element with Gd

The main characteristic of the fuel element/fuel element with Gd condition is its ability to retain inside radioactive fission products, including gaseous and volatile, coming out of the fuel element/fuel element with Gd cladding in the process of operation.

Damage of the fuel element/fuel element with Gd cladding as the second physical barrier, in particular, as a result of deterioration in heat removal from the fuel and effect of considerable pressure differences between space inside the fuel element/fuel element with Gd cladding and coolant at high temperatures of the fuel element/fuel element with Gd cladding as a result of the irreversible plastic yield leading to fuel element/fuel element with Gd cladding damage, or occurrence of steam — zirconium reaction of fuel element/fuel element with Gd cladding at high temperatures.

The limit of safe operation, characterizing impact on the fuel element/fuel element with Gd cladding shall be specified by technological process parameters, reflecting fuel element/fuel element with Gd cladding cooling in the reactor core and lack of significant temperature increase in fuel element/fuel element with Gd cladding.

Water chemistry within operational limitations ensures absence of the second physical barrier damage due to long-term physical and chemical processes. Therefore, safety operation limit associated with water chemistry is not established.

Barrier 3: The primary circuit boundary

Damage of the primary circuit coolant pressure boundary limits will lead to coolant release into the containment system or, depending on the area of damage, beyond its boundaries (for instance, pulse tubes breaks, blowdown and boron shim system, inter loop).

The primary coolant pressure boundary integrity is ensured by keeping admissible pressure in the primary side, whereby durability of the reactor plant components and equipment is assured. In particular it is required to take into consideration admissible pressure decrease in case of coolant temperature decrease and, accordingly, to avoid embrittlement of the reactor plant equipment construction materials .

Considering admissible leak from the primary to the secondary side within operational limits as well as occurrence of leak from the primary side to the secondary, the physical barrier to prevent radioactive substances spreading is the secondary coolant pressure boundary. In addition, it is necessary to consider positive reactivity insertion and postulated nuclear accident evolution in case of the secondary circuit ruptures. In this case provision is made for introduction of safe operation limit for the primary and secondary sides.

Barrier 4: Containment

Under normal operation conditions and Category 2 operation modes it should ensure acceptable degree of fission products keeping due to low pressure inside and release of radiative fission products, entering the containment with uncontrolled leaks. Consequently, it is not required to stipulate for the containment certain values of fission products activity as safe operation limits in the medium as normal operation boundaries.

Barrier 5: Biological protection

Civil structures of the reactor building as much as possible weaken radiation impact on personnel under normal operation conditions, abnormal operation, design basis accidents and beyond basis based accidents.

Physical barriers are shown on Figure 1.2.1.

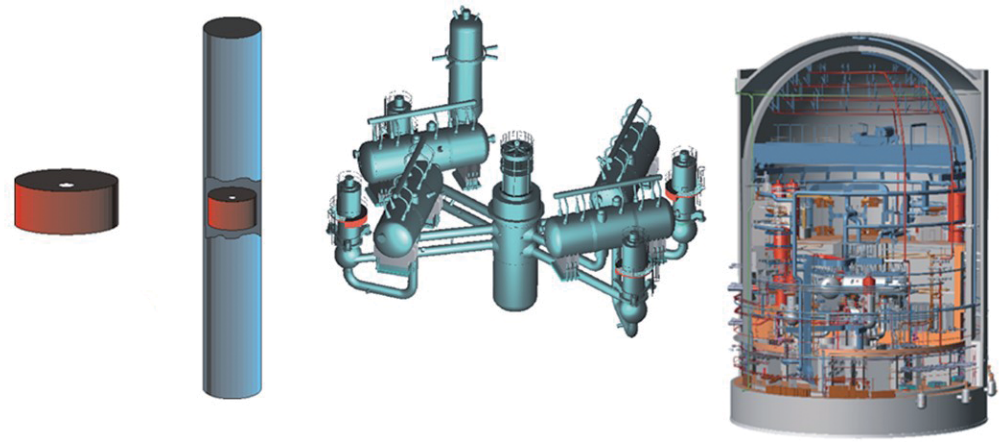


Figure 1.2.1 – Physical safety barriers

Note: The fifth barrier is not shown on the picture

Physical barriers are established to prevent and weaken radiation substances spreading into environment. However, radiation substances are in the spent fuel pool.

In the context of the spent fuel pool it is proper to establish monitoring of Cesium-134 and Cesium-137 long lived isotopes activity levels in the coolant with recommendations to the personnel to ensure efficient water chemistry in the spent fuel pool in order to meet radiation safety requirements (for instance, personnel exposure), as well as arrangement of leaky fuel assemblies in sealed casings to be placed in the spent fuel pool. To introduce such actions it is not required to introduce safe operation limits in regard to fission products activity in the spent fuel pool.

Each physical barrier is to be designed and manufactured considering specific norms and rules to assure enhanced reliability. Number of barriers between different radioactivity products and environment, as well as characteristics, are specified in the NPP designs.

In the course of operation physical barriers conditions are controlled by direct techniques (for instance, visual inspection of fuel assemblies prior to loading into the reactor core), as well as indirect techniques (for instance, coolant and air medium activity measurement under containment). If it is detected that any physical barrier is not efficient or damaged, it is required to take measures to address causes and restore its serviceability.

1.2.1 CONDITIONS OF BARRIERS OPERATION

Conditions of barriers operation are defined by design based operation modes (including emergency) and characterized by impact of different factors on their physical properties.

The effect (or consequences of effect) on the barrier means a change in the barrier material structure, strength characteristics of the material and/or the structure of the barrier, and geometric characteristics of the barrier structure.

Effects, having impact on the barrier, are taken into account in the design.

Prevention of impact on barriers at each level of NPP defense means technical and organizational measures aimed at maintaining amplitude/frequency and duration of impacts within the limits determined by the corresponding level of NPP defense.

Mitigation of effects on barriers means technical and organizational measures aimed at keeping radioactive substances within the boundaries of barriers and minimizing releases into the environment and the permissible impact of radioactive substances and ionizing radiation on personnel and the population

Under normal operation conditions, the NPP design shall ensure the design level of impacts on physical barriers.

Under abnormal operation conditions the NPP design shall ensure limitation of the effects on physical barriers and integrity of all four barriers (except for events postulating damage to one of the barriers, which do not lead to an accident).

In design basis accident modes, advanced design conditions modes (see 1.3.3) and in case of severe accidents, the design shall ensure limitation of impact and integrity of, at least, the fourth barrier (the containment).

The design shall consider measures to eliminate dependent failures of the third and fourth barriers, taking into account external impacts.

The design shall also take into account the impact that could effect the NPP operation, external environmental and climatic conditions, as well as human activities throughout the life cycle of NPP.

1.2.2 SOURCES OF IMPACT ON SAFETY BARRIERS

Sources of impact on barriers determined by the NPP main technological process are as follows:

— nuclear fission reaction;

— fission products radioactive decay;

— mechanical interactions; chemical interactions;

— electric and electromagnetic interactions.

These sources have a permanent effect on barriers during normal and abnormal operation (including accidents). The following measures shall be applied in the design to mitigate the impact and/or reduce the effect of the impact:

— use of special materials with characteristics providing maximum resistance to undesirable effects; the increased design margins;

— use of thermal protection and other design shielding against ionizing radiation;

— thermal protection and cool down of heated surfaces; parameters tracking system, automatic actions of control system;

— permanent diagnostics and timely preventive actions; water chemistry maintenance;

— equipment design solutions, mitigating effects on barriers;

— etc.

Natural climatic conditions determining the sources of impacts can be: ambient air temperature, chemical composition of the air, wind, whirl, snowfalls, and snowstorms, ice, storm, tsunami, extreme precipitation, tides, ice events on watercourses, abnormal reduction of water level, flooding, seismic activity, etc. Section 6 of this document describes in details the approaches to accounting for external impacts in the design. Analysis of external impacts and design measures to ensure resistance to impacts are given in Section 4 of the Design.

Man-made factors determining sources of impact can be: aircraft crashes, explosions at facilities, toxic substances emissions, fires, smoke, dam breakouts, etc. Analysis of man-made impacts and design measures to ensure resistance to them are described in the Design.

1.2.3 PROTECTION OF BARRIERS

The NPP design in design modes shall meet the following goal to protect physical barriers:

To ensure fuel integrity:

— during normal and abnormal operation probability of fuel destruction shall be very low;

— during postulated accidents quantitative indicators of fuel destruction should be kept low and fuel cooling should be maintained; the likelihood of an accident with an uncontrolled chain reaction is extremely small.

To ensure the integrity of the primary and secondary circuits:

The primary circuit system must be designed and manufactured in accordance with high quality standards, so that the probability of failures in structures and equipment that threaten their integrity is minimized, and all possible defects can be diagnosed with a high degree of reliability during examinations and inspections;

— the primary circuit shall be designed with margin to withstand stresses occurring during normal operation, abnormal operation, postulated accidents and advance design modes;

— the primary circuit and systems directly related to it, as well as secondary side safety systems important for safety, shall be reliably protected under all abnormal operation and accident scenarios to prevent damage caused by overpressure and;

— sufficient means shall be provided to ensure leak detection.

To ensure the containment integrity:

— the containment shall be designed so that it retains tightness under the forecast operational events and with high probability of accidents;

— when designing the containment, it is necessary to take into account pressure drops, radiation and thermal loads, radiation levels in the rooms were equipment is installed, combustible gases, drops and short-term powerful energy emissions as a result of the accident;

— the possibility of loss of tightness of the containment due to reactor vessel damage shall be bare.

The nuclear power plant shall be equipped with systems that ensure stabilization and cooling of the molten reactor core in the event of a severe accident. Direct exposure of the molten core to the basement or walls of the containment must be prevented in a reliable manner.

1.2.4 CHARACTERISTIC OF IMPACT ON PHYSICAL BARRIERS

The main safety objective is to maintain physical barriers integrity, and doing so, it shall be taken into account that barriers are exposed to various types of effects from the sources of impact mentioned in Section 1.2.2. These effects, depending on their nature and intensity, can lead to various changes in the structure of the material, decrease in strength characteristics, change in the geometric dimensions of structures, increase in stresses in the material and, as a result, loss of the barrier(s) integrity.

By using design measures, it is possible to minimize the likelihood of the barriers destruction against damaging effects caused by sources during normal operation.

To determine measures in order to protect barriers under any operating conditions (including emergency), a special list of design conditions and initiating events was developed.

1.3 LEVEL, OBJECTIVES AND MEANS OF DEFENCE-IN-DEPTH

As it was mentioned in Section 1.1, to achieve safety objectives it is required to meet a number of fundamental principles, one of them is defense-in-depth principle (Defense-in- Depth = DiD).

This fundamental safety principle plays a special role (sometimes it is called a strategy), because it is a basis of nuclear power plants safety technology. The implementation of this principle leads to construction of special systems to ensure safety.

1.3.1 DEFENSE-IN-DEPTH LOGICS

The safety philosophy is aimed at preventing accidents, as well as mitigating the consequences of accidents that can lead to significant releases of radioactive substances into the environment. The goal is to reduce both the likelihood of such events and their consequences.

The defense-in-depth principle is applied in various fields of activity that require a high level of safety, for instance, IT protection, security and air defense. This principle is based on the idea of successive barriers in the way of spreading and protection of these barriers by various means.

The essence of defense-in-depth principle is based on the use of a system of physical barriers to the spread of ionizing radiation and radioactive substances into the environment and a layered system of technical and organizational measures to protect barriers, as well as the direct protection of personnel, the public and the environment. Thus, the main task implemented in the project on ensuring safety is the design of means for maintaining the integrity of physical barriers in various modes.

The design defines safety functions that ensure protection of barriers by reducing the intensity of harmful effects (or eliminating them), as well as engineering and technical means for their implementation.

Thus, logics of defense-in-depth principle requires to:

1) Identify physical barriers system

2) Identify physical conditions of barriers operation (parameters, loads design cycles, materials selection, operational limits etc.)

3) Identify sources of harmful effects on barriers

4) Determine the spectrum (nature) of harmful effects;

5) Specify design conditions and initiating events that entail (or may entail) the loss of integrity of protective barriers;

6) Specify levels of defense-in-depth.

7) Define safety functions to reduce the intensity of harmful effects (or its exclusion) for each level of protection;

8) Develop a multi-level system of engineering and technical means to fulfill safety functions, guided by defense in-depth principles set forth in this concept, as well as requirements for systems classification, components and structures in terms of the performed safety functions and quality assurance requirements;

9) Develop a set of organizational protective measures for accident management.

To compensate for potential human and mechanical failures, a defense-in-depth concept is implemented, centered on several levels of defense including successive barriers preventing the release of radioactive material to the environment. The concept includes protection of the barriers by averting damage to the plant and to the barriers themselves. It includes further measures to protect the public and the environment against harm in case these barriers are not fully effective.

All safety activities are carried out on the basis of repeatedly overlapping measures so that in the event of a failure it is possible to take compensatory or corrective measures and prevent damage to individuals or the general public. This idea of numerous levels of protection is the main feature of defense-in-depth principle.

Defense-in-depth concept for nuclear power plants is described in Section 1.2.4 of general provisions for nuclear power plant safety assurance (NP-001-15)/1/.

The NPP safety shall be provided by consistent implementation of defense-in-depth based on a system of physical barriers in the way of ionizing radiation and radioactive substances spreading into the environment and a system of technical and administrative measures for protection of the barriers and maintenance of their efficiency as well as for protection of the personnel, public and environment.

The defense-in-depth principle involves set up of a series of consistent levels of protection against probable failures of equipment and human errors, including:

— establishing consistent physical barriers on the way of radioactive products propagation into the environment;

— early identification of technical and administrative measures to maintain the integrity and effectiveness of these barriers;

— early identification of measures to protect personnel, public and environment in the event of the barriers destruction.

The defense-in-depth principle provides limitation within each level of consequences of probable equipment failures and human errors and ensures that a single failure of equipment or human error do not lead to dangerous consequences. In the event of multiple equipment failures and / or human errors, the application of this principle reduces the likelihood of negative effects of radiation on personnel, public and environment.

1.3.2 DEFENSE-IN-DEPTH LEVELS

The purpose of the defense-in-depth concept is to ensure safety of the nuclear power plant meeting requirements of completeness of possible conditions of the nuclear power plant and the reasonable sufficiency of safety measures.

The content of the defense-in-depth concept, described in NP-001-15, is based on the publications INSAG-10 and INSAG-12, in compliance with requirements of the IAEA Safety of Nuclear Power Plants: design (NS-R-1). However, there are some differences in levels identification and content. Detailed explanations on application of the concept are given in the "Comments to federal norms and rules "General Provisions for Ensuring the Safety of Nuclear Power Plants" (RB-152-18)/3/.

The requirement to take all reasonably achievable measures to ensure defense-in-depth levels is one of the key effectiveness and reliability factors of multilevel protection, ability to withstand common cause failures of levels. In addition, this is ensured by application of various technical means or working on different principles for different levels of protection. Besides, in order to fulfill this requirement, the design of the NPP should consider measures, so that impacts (internal and external) that can disable several levels of defense at once, are all the less likely, the more of defense-in-depth levels they make ineffective

Defense-in-depth is structured in the following five levels:

The first level of defense is the high quality NPP design where all design solutions are substantiated and based on a conservative approach from the point of view of safety, and high quality training and qualification of operating personnel. When conducting the technological process, the first level of defense of physical barriers is ensured by maintaining the NPP operating parameters within the specified design limits, under which barriers are not exposed to the risk of damage. The effectiveness of the first level of defense is significantly affected by the maturity of the internal self-protection features of the reactor plant, that is, the features that determine the resistance to dangerous deviations of the process parameters and the ability to restore parameters within acceptable values.

The second level of NPP defense is to ensure availability of equipment and safety related systems by identification and elimination of failures. The importance at this level of defense is to correctly control the plant in case of deviations from normal operation modes and taking timely measures by personnel to eliminate them. Technically, the second level is ensured by reliable redundancy of equipment and systems for monitoring the state of components and equipment.

The third level of NPP defense is prevention of beyond design basis accidents by safety systems:

— prevention of escalation of initiating events into design basis accidents, and escalation of design basis accidents into beyond design basis accidents;

— mitigation of accidents consequences that could not have been prevented by confinement of the released radioactive substances

The forth level of NPP defense is beyond design basis accidents management:

— prevention of progression of beyond design basis accidents and mitigation of their consequences;

— return of the NPP to the controlled state in which the fission chain reaction stops and continuous fuel cooling and confinement of radioactive substances within the established boundaries are ensured;

The last, **the fifth level of NPP defense** is emergency planning: preparation and execution of action plans for protection of the personnel and the public at the NPP site and beyond it.

The main purpose of this level is mitigation of accidents consequences in regard to decrease radiological impact on the population and environment. This level of protection is ensured by emergency actions on the NPP site and implementation of emergency action near the site.

Table 1.3.1 presents levels, objectives and Defense-in-depth means.

Table 1.3.1 - Levels, objectives and means of NPP Defense-in-depth

Levels of Defense-in-depth	Objective	Means to achieve objectives
Level 1	NPP siting conditions and prevention of abnormal operation	— assessment and selection of a site suitable for the NPP location; — arrangement of a sanitary protective area, a supervised area around the NPP as well as a protective action planning zone; — development of the NPP design documentation (hereinafter — the NPP design) based on a conservative approach with well-developed inherent self-protection of the RP and measures aimed at prevention of the cliff edge effect; — assurance of the required quality of the NPP systems (components) and works performed; — operation of the NPP in accordance with the process regulations and production guidelines developed with due regard for the requirements of regulations and other regulatory documents; — maintenance of safety-related systems and components in operable condition by timely detection of defects, implementation of preventive measures, replacement of equipment with exhausted life span, arrangement of an efficient maintenance and repair system, documenting of the operation and monitoring results; — selection and provision of the required qualification level of the NPP personnel for working under normal operation conditions and in abnormal operation including pre-accident situations and accidents, formation of safety culture

Levels of Defense-in-depth	Objective	Means to achieve objectives
Level 2	Prevention of design basis accidents by normal operation systems	— timely identification of deviations from normal operation and their elimination; — control during operation with deviations.
Level 3	Prevention of beyond design basis accidents by safety systems:	— prevention of escalation of initiating events into design basis accidents, and of escalation of design basis accidents into beyond design basis accidents through the use of safety systems; — mitigation of consequences of accidents that could not have been prevented by confinement of the released radioactive substances
Level 4	Beyond design basis accident management	— return of the NPP to the controlled state in which the fission chain reaction stops and continuous fuel cooling and confinement of radioactive substances within the established boundaries are ensured; — prevention of progression of beyond design basis accidents and mitigation of their consequences, including through the use of special engineering features to manage beyond design basis accidents as well as any systems (components) including normal operation systems (components) and safety systems (components) capable of performing the required functions under the given conditions; — protection of the RP containment from destruction during beyond design basis accidents and maintaining its operability
Level 5	Emergency planning	Emergency planning: preparation and execution of action plans for protection of the personnel and the public at the NPP site and beyond it.

The defense-in-depth shall be implemented at all stages of activities related to the NPP safety assurance to the extent covered by this type of activity. The adverse events prevention strategy shall take priority, with special attention paid to Levels 1 and 2. All reasonably achievable measures shall be taken to ensure independence of defense-in-depth levels from each other. The applied measures shall be substantiated.

Each level of protection of the NPP shall ensure a certain level of effectiveness of protecting barriers against impacts. For each level appropriate technical and/or organizational measures shall be provided to prevent and/or mitigate consequences of impacts or sources of impacts.

In this case, the objective of technical means stipulated at various levels of defense-in-depth, in case of failures or emergency situations, is to bring back the plant into one of three states:

— normal operation mode;

— controlled state;

— safe state.

Controlled state is characterized by the NPP condition, **when the chain fission reaction is stopped, continuous fuel cooling and confinement of radioactive substances within the established borders is assured**.

Safe state is characterized by the NPP condition when the fission chain reaction is stopped, residual heat removal is ensured and there is no overpressure within physical barriers 3 and 4.

The NPP final state and requirements to achieve it are defined in the design, depending on initial events characterizing the mode and state of the reactor plant as well as the NPP in general.

1.3.3 DEFENSE-IN-DEPTH LEVELS INDEPENDENCE

Defense protection levels shall be independent from each other so that the failure of the next level is not a consequence (or the same cause) of the failure to perform safety function at one of the previous levels. The degree of independence should be reasonably achievable.

The main ways to ensure independence are **the functional and physical separation of means** that perform similar safety functions at different levels of protection, as well as the application of diversity principle. All these methods provide protection against common cause failures during internal and/or external impacts.

By functional separation it is meant the absence of common connections between systems that exercise similar functions at different defense levels, so that they cannot simultaneously fail due to a failure in a common system, structure or component. For example, they shall not have common links with supporting and/or control systems. If a system, component or structure exercise various functions at different levels of defense-in-depth, then this is also a case of functional separation.

By physical separation is meant separation in order to avoid simultaneous failure of systems exercising similar functions at different levels of defense, due to common cause associated with external or internal impact. Physical separation can be implemented in the following ways:

— with application of geometrical factors (distance, layout, location etc.);

— by means of barriers;

— due to combination of the above mentioned measures.

Selection of separation measures depends on impacts resulting from fires, chemical explosions, air crashes, flying objects ingress, flooding, temperature impact, humidity etc, to the extent applicable.

By diversity principle it is meant application of technical means having distinctive features for exercising the specified safety function. Thus, the distinctive features of technical means may be:

— different principles of operation;

— different physical variables;

— different operating conditions;

— different manufacturers of equipment;

— etc.

Principles of independence at different levels of defense shall be applied both to the main systems, exercising safety functions, and to the supporting and control systems. In other words, when analyzing the implementation of main safety functions, their dependence from auxiliary systems at each level of defense-in-depth shall be taken into account.

Thus, implementation of defense-in-depth principle allows achieving the main safety goals during operation - preventing failures and accidents, and in case of their occurrence, provides means for eliminating and mitigation of accidents consequences.

Analysis of major accidents causes showed that accidents path and consequences were directly dependent on the correct application of measures stipulated by defense-in-depth. It is required to be provided by the principle of defense-in-depth. In order to implement this principle and to make it fully operational, it is necessary to ensure the effectiveness of all five levels of defense-in-depth.

A graphical presentation of defense-in-depth algorithm is shown on Figure 1.3.2.

It is necessary to note critical points depicted on Figure 1.3.2.

1. At the level of design basis (3rd level) and beyond design basis accidents (4th level) you can get right away, bypassing the previous levels of defense-in-depth.

For instance, the cause of beyond design basis accidents may be initiating events that are not included in the list of initiating events for design basis accidents analysis.

In compliance with NP-001-15 internal events with the estimated probability of occurrence within a year of 10^{-6} or less may be excluded from the list of initiating events for analysis of design basis accidents.

Such initiating events are possible, but they cannot lead directly to a severe accident, long time is required for the development of the emergency process to exceed maximum design limit of fuel elements damage (MDL).

MDL of fuel rods damage is more than 24 hours in the NVNPP-2 design). During

this time, personnel have the opportunity to deploy and use additional technical means to maintain the reactor plant into controlled state (for example, emergency diesel generators and alternative closed cooling circuit).

2. Not every severe accident leads to a large emergency radioactive substances release into the environment. The design provides for special technical means to mitigate the consequences of severe accident inside the containment (core melt catcher, hydrogen recombines).

3. There are no initiating events with a probability more than 10^{-7} reactor/year leading directly to significant radiation exposure. For example, in NVNPP-2 design it is justified that the probability of the reactor pressure vessel destruction of one unit over an interval of one year does not exceed 10^{-7}.

Figure 1.3.3 shows defense-in-depth levels in accordance with the design and beyond design modes of NVNPP-2.

The use of defense-in-depth is aimed at limiting accidents and mitigating their consequences.

The ways of protecting the population and the environment against the possibility and consequences of accidents at NPP depend on the impacts on physical barriers that occur in various abnormal events.

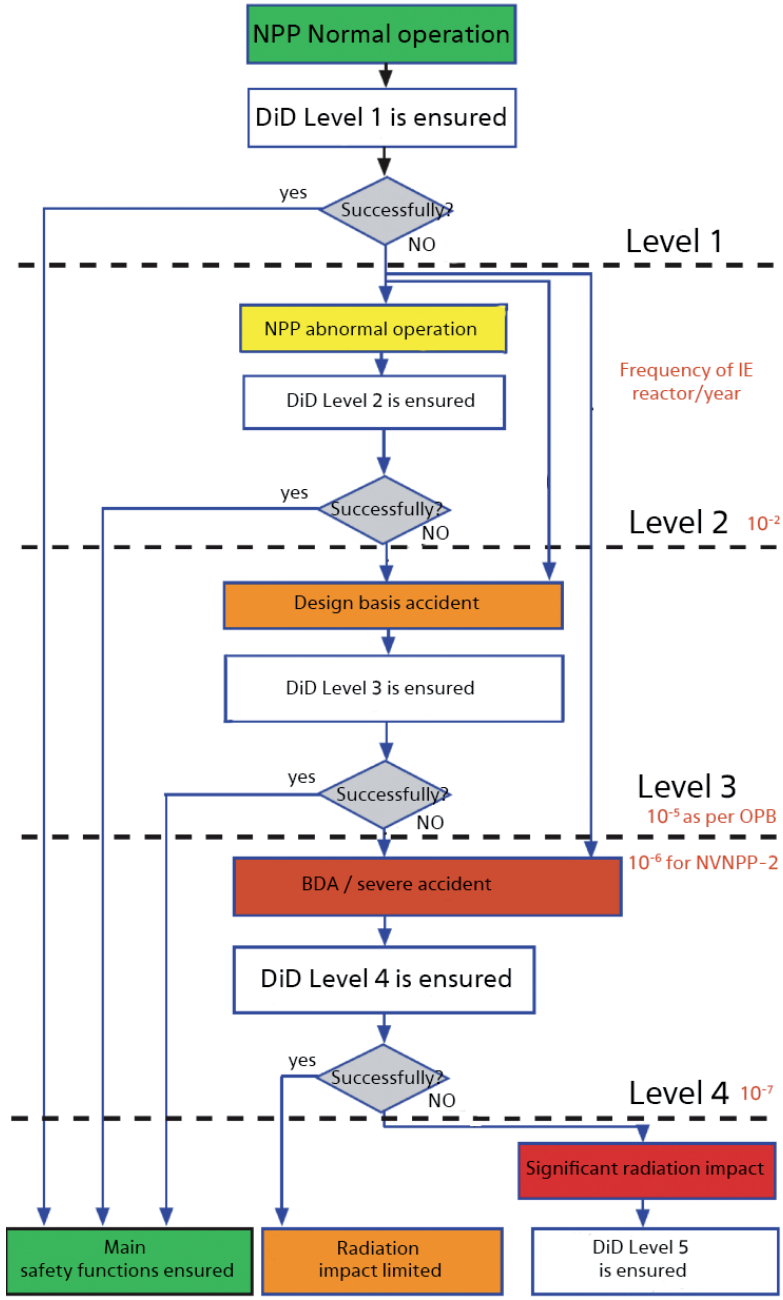


Figure 1.3.2 shows defense-in-depth levels in accordance with the design and beyond design modes of NVNPP-2.

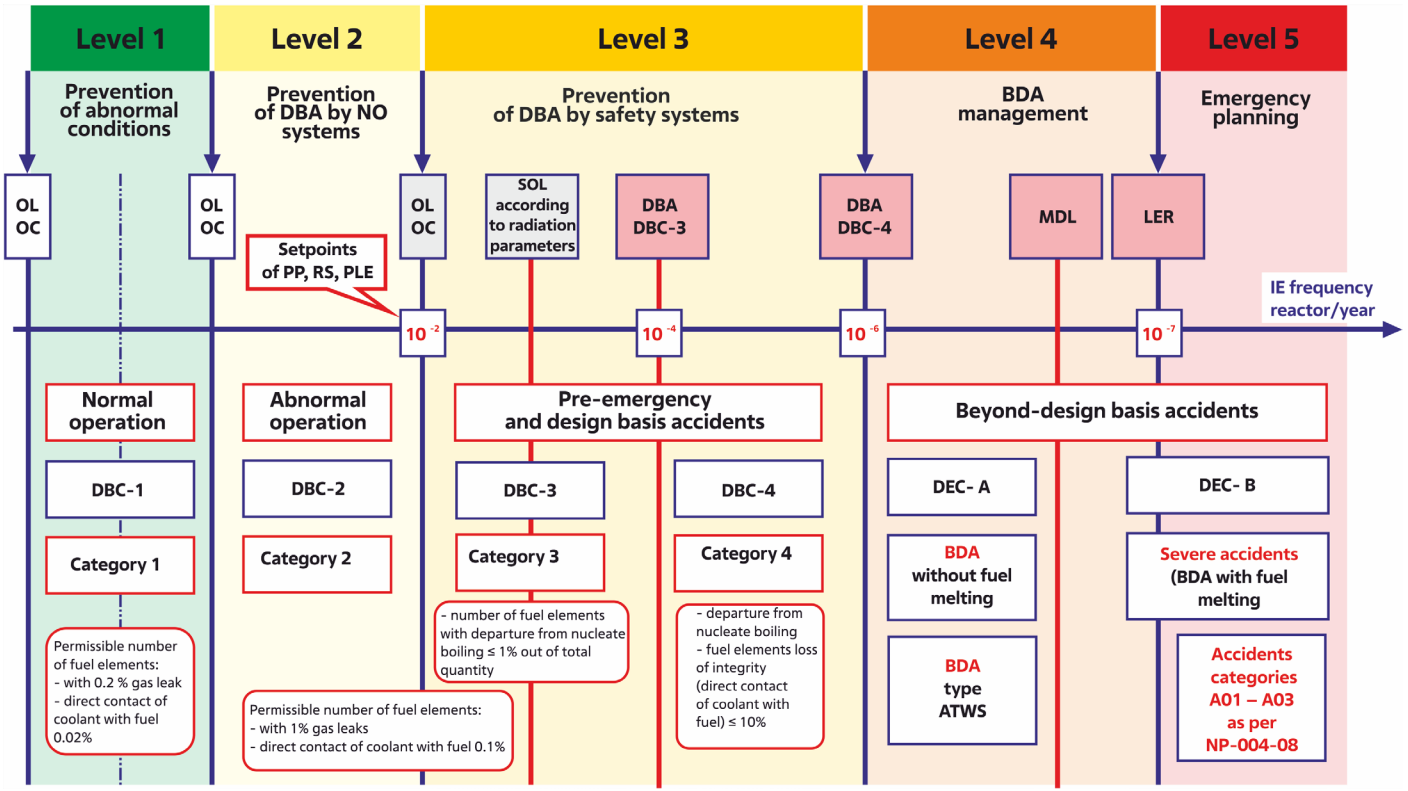


Figure 1.3.3 - Compliance of DiD levels with categories of operation modes

NO – normal operation	DBA - design acceptance criteria for accidents
OL (OC) - operating limits (operating conditions)	MDL - maximum design limit of fuel elements damage
SOL (SOC) - safe operation limits (safe operation conditions)	LER - large emergency release of radioactive substances into the environment

1.4 LIMITS AND CONDITIONS OF SAFE OPERATION

Design limits and safe operation conditions for NPP with reactor type V-392M for the first time in national practice were established for all power unit conditions implemented during their operation.

1.4.1 DESIGN LIMITS AND CONDITIONS FOR NORMAL AND ABNORMAL OPERATION

The following design limits and safe operation conditions for states of normal and abnormal operation of NPP with reactor type V-392M are established:

— State 1 (C1) — operation at power;
— State 2 (C2) — achievement of the minimum-controllable power level;
— State 3 (C3) — hot state;
— State 4 (C4) — cold state;
— State 5 (C5) — shut down for maintenance;
— State 6 (C6) — refuelling;
— State 7 (C7) — heating up;
— State 8 (C8) — cool down.

In addition, boundaries between states are clearly defined and set up in the following way:

1) Between the state C6 (refueling) and state C5 (shutdown for maintenance) — removal/installation of hydraulic locks between the fuel pool and reactor pit;

2) Between the state C5 (shutdown for maintenance) and state C4 (cold) — untightening of one of the primary circuit joints/sealing of all the primary circuit joints;

3) Between the state C4 (cold) and state C7 (heating up) — the coolant temperature at the reactor outlet is less/more than 70 °C;

4) Between the state C7 (heating up) and state C3 (hot state) — the coolant temperature at the reactor inlet is less/more than 260 °C;

5) Between the state C3 (hot state) and state C2 (achievement of minimum controllable power level) - all reactor control and protection system rods groups are completely injected into the reactor core/beginning of reactor control and protection system rods groups extraction from the reactor core;

6) Between the state C2 (achievement of minimum controllable power level) and state C1 (operation at power) — neutron flux power is less/more than 1 % N_{nom};

7) Between the state C3 (hot state) and C8 (cooldown) — the coolant temperature at the reactor inlet is more/less than 260 °C;

8) Between the state C8 (cooldown) and state C4 (cold state) – the coolant temperature at reactor outlet is more/less than 70 °C.

Parameters and conditions of normal operation states of NVNPP-2 Units 1 and 2 for all operational states are given in Table 1.4.1.

Table 1.4.1- Parameters and conditions of normal V-392M reactor operation.

№	Reactor state	Multiplication factor K_{eff}	Thermal power, % N_{nom}	Boron concentration in the primary coolant, g/dm³	Coolant temperature, °C		Further conditions
					inlet	outlet	
C1	Operation at power	1	> 1	C_{crit}	> 284	—	— Pressure in the primary circuit P_l = (16,1 ± 0,3) MPa — SG outlet pressure (6,8 – 7,0) ± 0,1 MPa; — at least two RCP_s running
C2	Minimum controllable power level (MCPL)	≤ 1	≤1	20 to C_{crit}	> 274	—	— one CR bank lifted from a lower limit position — P_l = (16,1±0,3) MPa; — L_{Przr} = (5100 ± 150) mm — P_{SG} = (5,8 – 6,7) ± 0,1 MPa — L_{SG} = (L_{nom} ± 50) mm — H_3BO_3 concentration difference between the reactor and the pressurizer < 0,5 g/dm³ — at least two RCPs running
C3	Hot state	< 0,99 excl. CRs inserted in the core	—	C_{min} to 20	260 ÷ 284	—	— all CR_s in a low limit or fully inserted bottom position — P_l = 16,1+0,3 MPa; L_{PRZR} = (5100 ± 150) mm — P_{SG} = 4,7 – 6,8 MPa; L_{SG} = (L_{nom} ± 50) mm — H_3BO_3 concentration difference between the reactor and the pressurizer < 0,5 g/dm³

№	Reactor state	Multiplication factor K_{eff}	Thermal power, % N_{nom}	Boron concentration in the primary coolant, g/dm³	Coolant temperature, °C		Further conditions
					inlet	outlet	
C5	Outage for maintenance	< 0,98 excl. CRs	—	16 ÷ 20	—	< 70	— primary circuit open — reactor level 300 – 400 mm lower than the main RPV flange or dropped for maintenance down to at most 600 ± 50 mm above the cold leg nozzle axis — H_3BO_3 concentration difference between the reactor and the pressurizer < 0,5 g/dm³ — atmospheric pressure in the primary circuit; — all CRs in a fully inserted bottom position
C6	Refuelling	< 0,98 excl. CRs	—	16 ÷ 20	—	< 60	— the upper block and the block of guide tubes removed — water seal removed, the reactor cavity and the SFP connected — refuelling pool level = 25,45 ± 0,1 m
C7	Heating up	< 0,98 excl. CRs	—	16 ÷ 20	< 260	≥70	— H_3BO_3 concentration difference between the reactor and the pressurizer < 0,5 g/dm³
C8	Cool down	< 0,98 excl. CRs	—	16 ÷ 20	< 260	≥70	— H_3BO_3 concentration difference between the reactor and the pressurizer < 0,5 g/dm³

C_{crit}, critical concentration of H_3BO_3 for a specific fuel loading pattern, power level, CR position, xenon and samarium poisoning, length of fuel campaign;

C_{min}, minimum permissible value to ensure subcriticality of at least 1 % (irrespective of CR_s in the core) at the current point of fuel campaign (as per a reference book of reactor physics pa-rameters for the current loading pattern)

Table 1.4.2 denoted physical barriers which have safety limits incorporated into the design coupled with corresponding operational states.

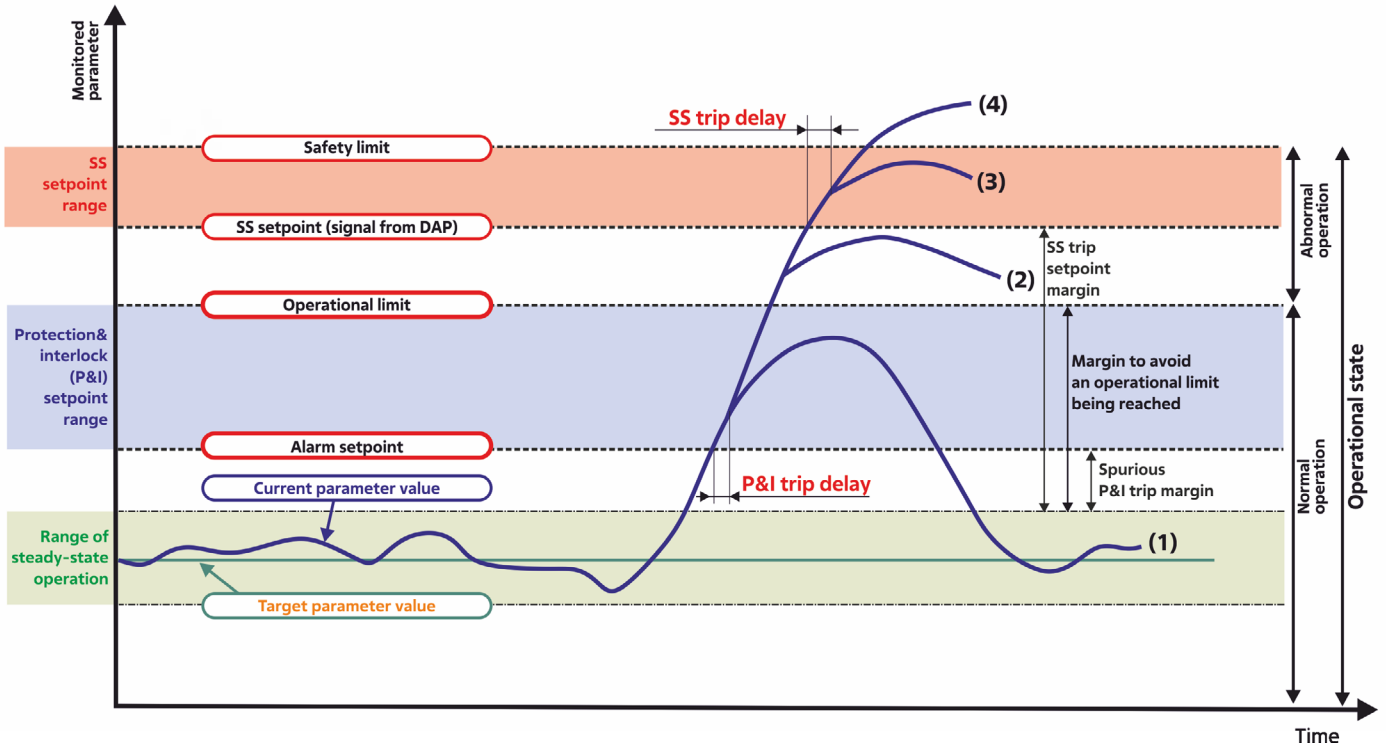
Table 1.4.2 - Physical barriers which have safety limits incorporated into the design coupled with corresponding operational states.

Physical barrier	Operational state							
	C1	C2	C3	C4	C5	C6	C7	C8
Fuel matrix	+	+	–	+	+	+	+	+
Fuel cladding	+	+	+	+	+	+	+	+
Reactor coolant system boundary	+	+	+	+	–	–	+	+
Containment and biological shield[1]	–	–	–	–	–	–	–	–

[1] Operational safety conditions are established for the containment and biological shield.

Variables on which safety limits have been placed define factors affecting the first three barriers, condition of the barriers and influences on fuel cladding.

Figure 1.4.1 depicts design limits and transient parameters with regard to normal operation (1), anticipated operational occurrences without safety system settings being reached (2), preaccident conditions without safety limits being exceeded (3) and accidents leading to violation of a safety limit (4).



(1) transients (category 1 modes);

(2) anticipated operational occurrences without safety system settings being reached (category 2 modes);

(3) pre-accident conditions without safety limits being exceeded (category 2 modes);

(4) pre-accident conditions and accidents leading to violation of a safety limit (category 3&4 modes).

Figure 1.4.1 - Changes in plant parameter during transients for various operational states

1.4.2 OPERATIONAL SAFETY LIMITS AND CONDITIONS FOR ACCIDENTS

Fuel damage limits

How do operational safety limits and conditions reflect the condition of nuclear fuel and fuel rod damage?

Limits for fuel (rod) damage are adopted as the principal reactor core design criteria. However, the distinction is made between gas leaks relative to the number of fuel rods in the core, and leaks exposing fuel to coolant (direct contact fuel to coolant) relative to the number of fuel rods in the core.

A s**afety limit** with regard to the character and size of a fuel leak is known as the first design limit (see Table 1.4.3).

A **maximum design limit** for fuel damage is known as the second design limit (see Table 1.4.3).

Assurance is provided that the first fuel damage design limit won't be exceeded in any normal and abnormal operation mode (see Figure 1.3.3. — categories 1&2), while the second design limit — under accident conditions.

Highly unlikely initiating events whose safety review principles are consistent with the definition of a design-basis accident belong to a category of design-basis accidents.

Table 1.4.3. denotes mode-related limits adopted for safety analyses as part of the RU-V 392M design FSAR.

Maximum fuel damage design limit

The maximum fuel damage design limit is set for design-basis accidents (DBC-2, DBC-3, DBC-4) and complex sequences (DEC-A) to meet the following boundary parameters:

— temperature of fuel cladding: max. 1200 ºC;

— equivalent cladding oxidation: max. 18 % of the initial cladding thickness;

— share of steam-reacted zirconium in the core: max. 1 % of the mass of zircaloy in fuel cladding;

— maximum fuel temperature no greater than the melting temperature.

The following criteria are established for core melt accidents (DEC-B):

— concentration of gas mixtures generated in and under the reactor after the RPV meltthrough shall stay below the explosion hazard value;

— primary and secondary side pressure shall not exceed corresponding pressure values for hydrostatic strength tests;

— if molten core can't be retained in the reactor vessel, pressure in the reactor coolant system shall be up to 1 MPa at the point of melt-through;

— allowable pressure pulse acting on elements of the concrete reactor cavity: 150 kPa/sec;
— maximum allowable pressure in the concrete cavity: 2,0 MPa;
— molten core boiling shall be precluded;
— damaged or molten core shall be maintained subcritical.

Table 1.4.3 - Acceptance criteria for designed modes of categories DBC-2, DBC-3, DBC-4

Criterion no.	Description
1	Pressure in the primary and secondary circuits will be max.115 % of the design value, which is max. 20,29 MPa on the primary side, and max. 9,315 MPa - on the secondary side.
2	Adequate I&C shall be provided to help reveal potential fuel loading errors in power ascension.
3	There's a 95 % probability that the hottest fuel rod won't reach the critical heat flux (minimal departure from nucleate boiling = DNB in the core is over 1,0)
4	No local fuel pellet melting. A U-fuel rod melting temperature is 2540 °C for 'burnt-up' fuel and 2840 °C for 'fresh' fuel. The melting temperature of U-Gd fuel rods equals 2260 and 2300 °C correspondingly [1])
5	The following emergency core cooling criteria shall be observed: — the maximum fuel damage design limit is not exceeded; — coolant flow channels in FAs are not clogged to a degree that will inhibit cooling ability due to ballooning, rupture of fuel, as well as deformation of other FA parts and reactor vessel internals[2]; — control rod melting is unacceptable[2]; — control rod motion in the reactor is not disrupted by potential deformation of fuel assemblies, control rods and reactor vessel internals[2]; — interaction of various FA components doesn't cause melting of those components[2]; — the reactor core shall be brought to a safe state so that the reactor can be maintained subcritical, cooled down in a post-accident shutdown, and the core and reactor vessel internals can be dismantled.
6	When an alarm is first received to warn the operator about an unintended boron concentration drop, there's minimum time before criticality is achieved: — during refuelling – 30 min.; — during start-up, cold shutdown, hot state and power operation – 15 min.
7	Radial average fuel enthalpy is not in excess of 586 J/g
8	Radial average fuel enthalpy is no greater than 963 J/g for 'fresh' fuel with burnup of under 50 MWd/kgU and 691 J/g for fuel having a radial burnup distribution of over 50 MWd/kgU along fuel rods

Criterion no.	Description
9	For designed modes of category 3. The ratio of leaking rods shall be less than 1 % of the total number of fuel rods in the core
10	For category 3 modes. During an accident the ratio of leaking rods 3) shall be less than 10 % of the total number of fuel rods in the core

1) When space-time kinetics calculations are performed, fuel melting temperature (Tm) should be identified as follows (according to FSAR):

— U-Gd rods $T_m = T_m\,0 - 0{,}56\,B$,

where B is burnup, MWd/kgU;

$T_m\,0$ = 2405, 2300 °C – non-irradiated U-Gd fuel melting temperature, respectively for UO_2 it's + 5,0; 8,0 % Gd_2O_3;

— U rods $T_m = 2840 - 0{,}56B$,

where B is burnup, MWd/kgU.

2) Compliance with the stated criteria is verified in FSAR.

Note: In case of a design-basis accident of category 3 associated with:

— primary LOCA (a leak rate which doesn't escalate to the boiling crisis shall be determined in relevant analyses);

— secondary LOCA (design-basis);

— failure to seat one pressurizer PORV,

Assurance is provided that core subcriticality, continued core flooding and secure crisis-free cooling will be maintained with due respect of stipulated design margins, fast-responding and efficient safety features, so the fuel damage safety limit shouldn't be exceeded.

Table1.4.4 - Acceptance criteria for design extension modes (BDBAs before core damage) DEC-B)

c	Description
1a	Pressure in the primary and secondary systems will be max. 115 % of the design value, which is max. 20,29 MPa on the primary side and max. 9,315 MPa on the secondary side.
1b	In case of an ATWS emergency (anticipated transient without scram) pressure in the primary and secondary systems will be max.135 % of the design value, which is max. 23,8 MPa on the primary side and max. 10,9 MPa on the secondary side.
2	No local fuel pellet melting. A U-fuel rod melting temperature is 2540 °C for 'burnt-up' fuel and 2840 °C for 'fresh' fuel. The melting temperature of U-Gd fuel rods equals 2260 and 2300 °C correspondingly[1)]

c	Description
3	The following emergency core cooling criteria shall be observed: — the maximum fuel damage design limit is not exceeded; — coolant flow channels in FAs are not clogged to a degree that will inhibit cooling ability due to ballooning, rupture of fuel, as well as deformation of other FA parts and reactor vessel internals[2]; — control rod melting is unacceptable[2]; — control rod motion in the reactor is not disrupted by potential deformation of fuel assemblies, control rods and reactor vessel internals[2]; — interaction of various FA components doesn't cause melting of those components[2]; — the reactor core shall be brought to a safe state so that the reactor can be maintained subcritical, cooled down in a post-accident shutdown, and the core and reactor vessel internals can be dismantled.

[1] When space-time kinetics calculations are performed, fuel melting temperature (Tm) should be identified as follows (according to FSAR):

— U-Gd rods $T_m = T_{m0}$–0,56*B, where B is burnup, MWd/kgU;

— $T_{m\ 0}$=2405, 2300 °C – non-irradiated U-Gd fuel melting temperature, respectively for UO_2 it's+ 5,0; 8,0 % Gd_2O_3;

— U_{fuel} T_m = 2840 – 0,56*B, where B is burnup, MWd/kgU.

[2] Compliance with the stated criteria is verified in FSAR.

[3] This criterion is out of scope of ATWS emergencies

1.4.3 OPERATIONAL SAFETY LIMITS INDICATIVE OF EFFECTS ON PHYSICAL BARRIERS

As per IAEA guidance NS-G-2.2/4/ Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants violation of safety limits (SLs) must be prevented by ensuring acceptable margins between established safety system trip setpoints and parameter values at which the safety limits are reached. In individual cases, Russian NPP designs introduce safety system trip settings equivalent to safety limits. The decision is associated with additional challenges arising in justification of the limits, seeing that in any case settings are be validated by calculations carried out for emergency conditions.

On the one hand, it's contrary to the requirements set forth in international and national regulations. On the other hand, it defeats the main purpose of establishing the limits, i.e. 'taking urgent actions to gain control of the process and prevent an accident'. Moreover, exceedance of safety limits results in

a higher incident category on both the scale found in the national NP-004-08/5/ document and the international INES/6/scale. In IAEA documents a term 'safety limits' is used. In foreign NPP designs safety system settings notably differ from safety limits. For instance, in most VVER-1000 plants a safety system trip setpoint for the primary system pressure is numerically equal to the safety limit and makes 180 kgf/cm^2. When it comes to VVER-1000 units operated by the Tianwan NPP the safety system setting is 17,8 MPa, while the safety limit is 20,3 MPa.

It was Novovoronezh-2 where the issue of coinciding safety limits and trip setpoints was resolved for the first time across Russian NPPs.

Table 1.4.5 summarizes operational safety limits of Novovoronezh-2 indicative of effects on physical barriers.

Table 1.4.5 - Operational safety limits of Novovoronezh-2 indicative of effects on physical barriers

Parameter	Barrier	SL value	Setting			Parameter measurement method and location
			Scram (AZ)	(PZ-2)	Alarm	
1. Local DNB heat flux ratio, relative units = DNBR	— fuel matrix — fuel cladding	* **1,0** at reactor power of over 20 % N_{nom}	**1,35**	**1,6**	**1,7**	Computation with PTK-Z SVRK (in-core instrumentation system protection hardware and software complex) based on measured self-powered neutron detector currents and common detector indications.

*In the C1 state fuel heatup capable of starting a fuel matrix damage is likely in the event of an uncontrolled power excursion resulting in a boiling crisis and subsequent heatup of fuel rods and their cladding. Therefore, the safety limit which indicates an impact on the first physical barrier is a local DNB heat flux ratio (DNBR = 1, at power over 20 % N_{nom}).

The determining factor at power of under 20 % N_{nom} is fuel cladding integrity. Because, should heat removal from the reactor core deteriorate with subsequent heatup of fuel rods, fuel cladding would reach the clad damage temperature earlier than conditions could develop for the damage of fuel matrix.

The **DNBR=1** value considers the correlation uncertainty used for calculating critical heat flux. With the given uncertainty disregarded the DNBR limit amounts to **1,274**.

PTK-Z SVRK application programs calculate critical heat flux based on GYDROPRESS's correlations and NIKIET's methodology depending on applicability. The in-core instrumentation system computes and displays the actual minimum DNBR for operators 24/7.

DNBR is calculated by a correlation **DNBR = (q_{acc} / q_z)**, where:

q_z = actual axial heat flux;

q_{acc} = allowable heat flux = q_{cr}·(1 - δ);

q_{crit} = predicted critical heat flux;

δ = of GYDROPRESS's correlation uncertainty.

By virtue of subcritical reactor and small linear heat flux from fuel cladding, DNBR is not appropriate as a safety limit for such states as C3, C4, C5, C6, C7, C8.

In case of an unwanted power rise under C2 (MCPL) conditions this safety limit gets monitored in the C1 state when power reaches over 20 % N_{nom}.

Departure from nucleate boiling on the fuel rod surface shall be continuously monitored at reactor power of over 20 % N_{nom}.

Operator actions after loss of controls:

— monitor/perform an unscheduled reactor shutdown at a specific speed to bring the reactor to a 'hot' state.

Parameter	Barrier	SL value	Operational state	Parameter measurement method and location
2. RPV water level by SKUT (level measurement system) at the upper generatrix of inlet nozzles	fuel cladding	Level drop signal from a discrete sensor	C1 (power operation) C2 (MCPL) C3 (hot) C4 (cold) C7 (heatup) C8 (cooldown) C5(outage for maintenance) with SKUT detectors on	SKUT

Level monitoring using SKUT shall be non-stop.

Operator actions after loss of controls:

— C1 (power operation) & C2 (MCPL): monitor core outlet temperature. Once noted core outlet temperature corresponds to the saturation line under existing pressure, unscheduled outage is to be monitored/performed by safely bringing the reactor to a hot state;

— C3 (hot), C4 (cold), C5 (outage for maintenance) and C6 (cooldown): stop tasks that might trigger a level drop in the reactor

— C7 (heatup): discontinue heatup procedure. Stop tasks that might trigger a level drop in the reactor. Resume heatup once level monitoring has been restored.

Parameter	Barrier	SL value	Operational state	Parameter measurement method and location
3. RPV water level at the lower generatrix of inlet nozzles, m	fuel cladding	0 (corresponds to the lower generatrix of the inlet nozzle)	C5 (outage for maintenance) with SKUT detectors off and FAs in the core C6 (refuelling) if no operations associated with movement of fuel assemblies	In-vessel level detector with a 4-m filling height

Fuel cladding may heat up due to an RPV level drop below the top of fuel assemblies.

Judgement of whether the safety limit is violated shall lie with operations staff who consider design level detector indications and indirect evidence of a decreasing coolant level (intense steaming, reactor coolant temperature rises up to saturation temperature etc.), including TV surveillance capabilities.

Level monitoring by SKUT shall be non-stop.

Operator actions after loss of controls:

C5 (outage for maintenance):

— stop tasks potentially contributing to a level drop in the reactor.

C6 (refuelling):

— finalize fuel handling by placing FAs in their final position in the SFP or reactor.

— stop refuelling.

Justification of the RPV level safety limit with regard to the inlet nozzle lower generatrix.

C5:

1. A scenario was considered when operators go on with draining of the primary circuit even upon reaching a level of 600 ± 50 mm above the inlet nozzle centreline. If primary coolant is drained at a rate of 5 m^3/h, the reactor level safety limit will be reached in 10 hours.

2. When heat removal fails (accident with a total loss of AC power sources - blackout) with a reactor level of 550 mm above the inlet nozzle centreline, the coolant level will take 1,94 hours to go down to the lower inlet nozzle generatrix. In light of unavailability of active SS trains operating personnel, within the given timeframe, shall ensure the reactor gets water from hydroaccumulator-2 to prevent the RPV level safety limit from being reached.

Parameter	Barrier	SL value	Operational state	Trip setpoint				Parameter measurement method and location
				PZ-1	Scram (AZ)	Opening of PRZ PORVs (from USBT (DAP units))		
						Major	Auxiliary	
4. RCS pressure, MPa	RCS boundary	* 20,29= =17,64 x 1,15	C1 (power operation) C2 (MCPL) C3 (hot)	17,2	17,5	18,01 (BE11)	18,5 (BE11)	Reactor pressure tap, in-vessel measurement
		Derived from Figure 1	C4 (cold) C7 (heatup) C8 (cooldown)	BE12: close valves in ECCS hydroaccumulator connecting pipelines at T1c < 100 °C & P_{1c} > 3,3 MPa				
		25,0	C7 (during hydrostatic tests).	RCS strength hydrostatic test pressure = 24,5 MPa The RCS is visually inspected for integrity				

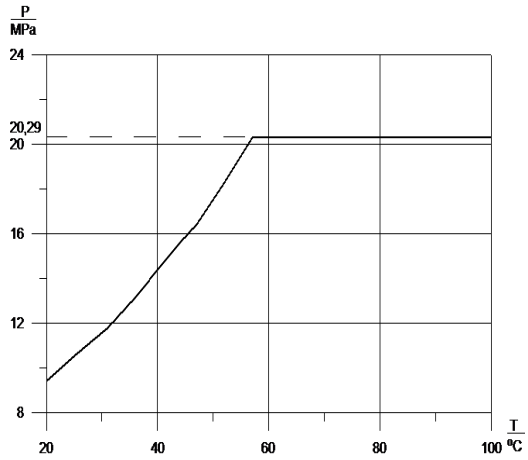


Figure 1.4.2 – Correlation between the RCS pressure limit and coolant inlet

This safety limit is a boundary beyond which the integrity of the reactor coolant system may be jeopardized:

— directly → in case of rising primary pressure with no other deviations

— indirectly → in case of an anticipated operational occurrence, including a designbasis accident.

The introduced safety limit value related to primary pressure is greater than a scram setpoint and PORV trip setpoints (from DAP units). Though, it's in agreement with the acceptance criterion for the design modes of categories 2, 3 & 4.

Primary pressure below the given safety limit shall ensure that the reactor coolant system will stay intact and damage won't go beyond the relevant design limits, should initiating events of anticipated operational occurrences, including designbasis accidents, occur.

In operational states C5 & C6 the coolant system is open. Therefore, the primary pressure safety limit is superfluous for these modes.

For C1, C2 and C3, breach of the third physical barrier is feasible when pressure in the primary circuit rises beyond the acceptable limit verified by strength calculations. However, with FCS requirements for safety relief valves in mind, a conservative pressure constraint of up to 115 % of the design pressure is in place.

When it comes to C4, C7, C8, the primary pressure safety limit is derived from the curve in Figure 1.4.2 which is based on the results from analysis of residual heat removal speeds of over 60 °C/h.

The transient state C7 suggests conduct of hydrostatic tests for primary components and pipelines to confirm their integrity and strength. Over duration of the tests higher safety limits are out of use. Hydrostatic test procedure respects permissible metal temperatures in case primary pressure rises up to the hydrostatic test pressure, which is P_i = 24,5 MPa. At that, a trip setpoint value for safety relief valves installed on the discharge side of hydrostatic test pumps serves as the primary pressure safety limit, which is 25,0 MPa.

Pressure in the primary system shall be continuously monitored.

Operator actions after loss of controls:

C1 & C2:

— Monitor/perform a manual scram and bring the reactor to a ‘hot’ state.

C3, C4, C7 and C8:

— isolate and lock out the equipment whose operation could cause an increase in the primary system pressure.

Parameter	Barrier	SL value	Operational state	Trip setpoint				Parameter measurement method and location
				BRU-A opening	Scram (AZ)	SG POSRVs (opened by solenoid valve)		
						Major	Auxiliary	
5. Pressure in the secondary system, MPa	* RCS boundary	** **9,32** = = 8,1 x 1,15	C1 (power operation) C2 (MCPL) C3 (hot) SG shell temperature over 100 °C - C7 (heatup) - C8 (cooldown)	7,7 (BC11)	**8**	8,7 (BF11)	8,9 (BF11)	Pressure tap at the SG outlet
		12,0	C7 (during hydrostatic tests)	Hydrostatic strength test pressure on the secondary side is 11,5 MPa				

* In view of an acceptable primary to secondary leak the steam circuit boundary acts as a barrier limiting distribution of radioactive substances.

This safety limit is a boundary beyond which the integrity of the secondary circuit may be jeopardized:

— directly → in case of a rising secondary pressure with no other deviations,

— indirectly → in case of an anticipated operational occurrence, including a design-basis accident.

The established safety limit value for secondary pressure is greater than a scram setpoint and SG POSRV trip setpoints. Though, it's in agreement with the acceptance criterion for the design modes of categories 2, 3 & 4.

SG pressure below the given safety limit shall ensure that the SG and main steam lines will stay intact, and damage won't go beyond the relevant design limits, should initiating events of anticipated operational occurrences, including design-basis accidents, occur.

For C1, C2 and C3, breach of the third physical barrier is feasible when pressure in the SG rises beyond the acceptable limit verified by strength calculations.

However, with FCS requirements for safety relief valves in mind, a conservative pressure constraint of up to 115 % of the design pressure is in place, which

is P_{II} = 115 % x 8,1 MPa = 9,32 MPa

In case of C7 & C8, at a SG shell temperature of over 100 °C the secondary pressure safety limit is set similar to C1, C2, C3 with due account of relief valve specifications.

At a temperature of max. 100 °C the secondary pressure safety limit is useless due to low primary and secondary coolant parameters under these conditions and, by extension, negligible likelihood of a primary-to-secondary leak.

A transient state C7 suggests conduct of hydrostatic tests for secondary components and pipelines to confirm their integrity and strength. Over duration of the tests higher safety limits are out of use. Hydrostatic test procedure respects permissible metal temperatures in case secondary pressure rises up to the hydrostatic test pressure, which is P_{II} = 11,5 MPa. At that, a trip setpoint value for safety relief valves installed on the discharge side of hydrostatic test pumps serves as the secondary pressure safety limit, which is 12,0 MPa.

Pressure in the secondary circuit shall be continuously monitored.

Operator actions after loss of controls:

C1 & C2:

— Monitor/perform a manual scram and bring the reactor to a 'hot' state.

C3, C4, C7 and C8:

— Isolate and lock out the equipment whose operation could cause an increase in secondary side pressure.

Parameter	Barrier	SL value	Operational state	Parameter measurement method and location
6. Water level in the spent fuel pool, metres from the bottom	- fuel cladding	**6**	C1 (power operation) C2 (MCPL) C3 (hot) C4 (cold) C5 (outage for mainteance) C7 (heatup) C8 (cooldown) C6 (refuelling) - no fuel handling operations associated with FA movement	Monitored by a design level meter.
		16	C6 (refuelling) - ongoing fuel handling operations associated with FA movement	

High-density racks in the SFP provide for K_{eff} < 0,95 as per regulatory requirements without consideration of boric acid in the SFP coolant and reactivity control devices as part of FAs. Therefore, criticality of spent and fresh fuel located in the SFP can be ruled out. With this in mind, a safety limit on fuel in the SFP is factored into the design solely for the second physical barrier, fuel cladding, and is applied to all reactor states.

Potential fuel damage in the SFP might result from fuel heatup to the melting temperature following a SFP water level drop below the top of FAs. Further damage to fuel cladding is not expected provided water is maintained over the top of FAs, even if decay heat removal is lost.

Thus, the safety limit with regard to the second barrier is a 6-m water level from the SFP bottom and applied to all operational states, except C6 (refuelling).

Under C6 conditions a failure of the second physical barrier might be triggered by a drop of the SFP water level so that the top of FAs being moved becomes uncovered.

Consequently, safety limits for C6 defining effects on the second barrier are as follows:

— a 16-metre level from the SFP bottom during fuel movement operations, which is equivalent to the top of the FA being moved;

— a 6-metre level from the SFP bottom with no fuel movement operations.

The SFP water level shall be monitored non-stop.

Operator actions after loss of SFP level controls:

C1, C2, C3, C4:

— Visually monitor the water level in the SFP (via CCTV).

— Monitor the temperature of the SFP water.

— Monitor operability of the spent fuel cooling system.

C6 (refuelling):

— Suspend refuelling until an alternate monitoring method is in place.

— Stop fuel movement by placing the FA in its final position in the core or the SFP if controls are not available.

— Perform visual monitoring of the water level in the SFP, and keep track of the water temperature using adequate instrumentation.

Parameter	Barrier	No. of running RCPs	SL value	Neutron flux scram setpoint			Parameter measurement method and location
				Variable setpoint*	Scram setpoint, adjusted by reactor operator**	End of power range -2.	
7. Reactor neutron power, % **% N_{nom}**	fuel matrix	four	**108**	107	107	107	NFMS (neutron flux monitoring system)
		three	**75**	74	** N_{actual} +7 ≤ 74		
		two	**48**	47	** N_{actual} +7 ≤ 47		

A neutron flux density constraint is dictated by the need to limit core heat generation and balance the heat generation (as a function of neutron power) and heat removal from the core (as a function of the flow rate of coolant circulating through the core, or the number of running RCPs).

Neutron flux safety limits agree with scram setpoint values in view of the measurement error margin and/or relevant parameters obtained from instrumentation and controls.

Neutron power shall be constantly monitored during power operation.

Operator actions after loss of neutron power controls:

— Monitor/perform a manual scram.

— Bring the reactor to a hot state (C3).

***** A variable power setpoint as a function of the number of running RCPs is generated as follows:

— with a 1 out of 4 RCP trip the variable setpoint linearly decreases from 107 to 74 % N_{nom} at 0,5 % N_{nom}/sec.;

— with a 1 out of 3 RCP trip the variable setpoint linearly decreases from the current level to 47 % N_{nom} at 0,33 % N_{nom}/sec.;

Setpoint decrease rate differences are attributed to different reactor runback rates, as 1 out 4 RCPs (UPZ-accelerated preventive protection) or 1 out of 3 RCPs (ROM-power limiting controller) trips.

****** A scram setpoint adjusted by the operator during operation with one or several loops unavailable.

Parameter	Barrier	NFMS range	SL value	Setting		Operational state	Parameter measurement method and location
				Scram (AZ)	PZ-1		
8. Reactor period, sec.	fuel matrix	**Power range -1. Power range -2.**	**9**	10	20	C1 (power operation)	NFMS
		Startup range	**8**	10	20	C2 (MCPL)	

A significantly shorter reactor period poses a risk of approaching to the reactor controllability boundary when the core reactivity and the effective delayed neutron fraction become close in value. Upon reaching the above boundary reactor power starts increasing at a colossal speed and can reach high values before the reactor protection system is actuated in response to having reached settings for some other parameters.

Another potential threat associated with a rapid reactivity insertion consists in fuel enthalpy rise up to the values conducive to fuel matrix degradation.

The following fuel enthalpy acceptance criteria are established:

— category 2: radial average fuel enthalpy is no greater than 586 J/g

— categories 3 & 4: radial average fuel enthalpy is max. 963 J/g for 'fresh' fuel with burnup of up to 50 MWd/kgU and 691 J/g for fuel having radial burnup distribution of over 50 MWd/kgU along the fuel rod.

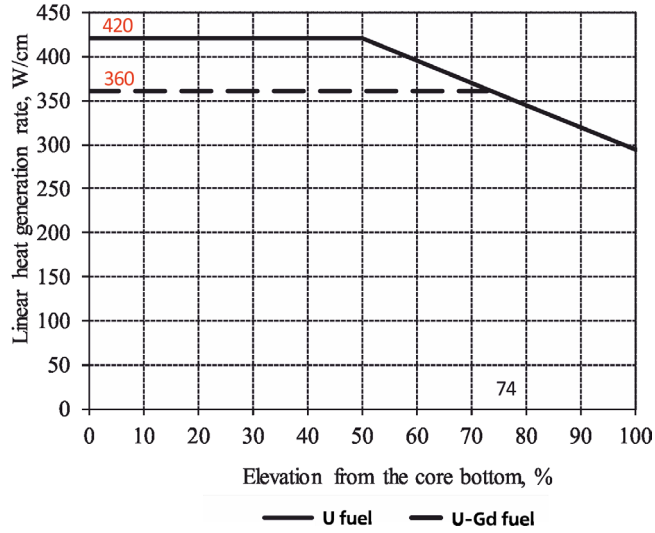
Reactor period safety limits agree with scram setpoint values in view of the measurement error margin and/or relevant parameters obtained from instrumentation and controls.

Neutron power shall be constantly monitored during power operation.

Operator actions after loss of controls:

— Monitor/perform a manual scram and bring the reactor to a 'hot' state.

Parameter	Barrier	SL value	Operational state	Parameter measurement method and location
9. Linear heat generation rate for U & U-Gd fuel rods, W/cm	fuel matrix	* A 5% exceedance of the limit curve 'linear heat rate – elevation from the core bottom'	C1 (power operation)	PTK-Z



* A 5% exceedance of the limit curve 'linear heat rate – height from the core bottom'

* Safety limits on the linear heat generation rate agree with scram setpoint values in view of the measurement error margin and/or relevant parameters obtained from instrumentation and controls.

Continuous monitoring at reactor power of over 20 % N_{nom}.

Operator actions after loss of controls: Monitor/perform an unscheduled reactor shutdown at the predetermined speed to bring the reactor to a 'hot' state.

Parameter	Barrier	SL value	Operational state	Parameter measurement method and location
10. Boron concentration in the primary coolant, g/dm³	fuel matrix	A value ensuring reactor subcriticality is min 0,02 (irrespective of CRs inserted in the core), plus 1 g/dm³	C4 (cold state) C5 (outage for maintenance) C6 (refuelling) C7 (heatup) C8 (cooldown)	Boron meter readings and intermittent sampling

The states C4, C5, C6, C7, C8 are characterized by a lower primary coolant temperature as compared to the states C1, C2, C3: therefore, in the event of the feed & bleed malfunction causing boric acid dilution in the coolant the reactor might become critical.

That being said, under the C4, C5, C6, C7, C8 conditions the boron concentration safety limit valid for the first physical barrier relies on the minimum subcriticality margin.

As defined in NP-082-07, in refuelling the boron concentration in the reactor coolant system shall ensure (in view of potential errors) reactor subcriticality of at least 0,02 without consideration of CRs inserted in the core.

Continuous monitoring with the help of boron meters and periodic verification by sampling

Operator actions after loss of controls:

C4 (cold state), C5 (outage for maintenance), C6 (refuelling):

— Make sure that 'clean' condensate is not present in/prevented from entering the primary circuit

C7 (heatup), C8 (cooldown):

— Stop heatup / cooldown.

— Make sure that 'clean' condensate is not present in/prevented from entering the primary circuit.

— Resume heatup/cooldown once controls are back in service.

Parameter	Barrier	SL value	Operational state	Parameter measurement method and location
11. Primary coolant outlet temperature, °C	fuel cladding	*** 80**	C5 (outage for maintenance) C6 (refuelling)	Resistance temperature detectors in 'cold' and 'hot' legs

* Non-exceedance of the temperature limit guarantees that primary coolant will remain far from boiling under C5, C6 conditions, which otherwise could lead to fuel uncovery in the reactor core.

The primary coolant outlet temperature shall be monitored non-stop.

Operator actions after loss of controls:

— Monitor performance of the normal and emergency residual heat removal systems in a decay heat removal mode.
— Monitor the reactor coolant level.

Parameter	Barrier	SL value	Operational state	Parameter measurement method and location
12. SFP water temperature, °C	fuel cladding	*** 80**	C1 – C8	Temperature sensors mounted in the SFP

* Non-exceedance of the temperature limit guarantees that the SFP water will remain far from boiling in all operational states, which otherwise could lead to fuel uncovery in the reactor core.

The SFP water temperature shall be monitored non-stop.

Operator actions after loss of SFP temperature instrumentation:

— Monitor performance of the SFP cooling system.
— Monitor the water level in the SFP.

1.4.4 SAFETY LIMITS INDICATIVE OF THE CONDITION OF PHYSICAL BARRIERS

Parameter	Barrier	SL value	Operational state	Note
1. Primary coolant activity determined by I-131-132 reference isotopes, expressed as dose equivalent I-131, Bq/kg	— fuel matrix-fuel cladding	$1,5 \cdot 10^6$ including the acceptable primary-to-secondary leak rate	C1 (power operation)	The limit providing assurance that annual public dose quotas at the boundary of or outside the buffer area are not exceeded The above level corresponds to 25 leaking fuel rods (0,048 % of the total fuel rod number in the core) provided that homogeneous failed fuel distribution is postulated in terms of power density and burn-up and fuel matrix doesn't show any significant micro- and macrostructure changes to a considerable degree deteriorating its capability as the first physical barrier to retain the spread of radioactive fission products. In addition, the ratio of the number of gas-leaking fuel rods to fuel rods with leaks exposing fuel to coolant shall be 10:1, which is in line with the ratio stated in NP-082-07 /7/

An operational limit for the activity of dose-contributing fission products in the primary coolant with the reactor online is stipulated in the design of units 1&2, Novovoronezh-2, based on the public dose limit of 10 μSv/y per unit.

With the activity levels of fission products in the reactor coolant system within the established safety limits the maximum acceptable quota of the annual effective dose for the public per unit shouldn't be exceeded. The quota must be established for a single unit considering quotas allocated for other units on the plant site.

Safety limits and operational limits on the fission product activity in the primary coolant should be selected in view of the overall health of physical barriers, viz. their capability to retain radioactive fission products, as well as parameters of ventilation and radioactivity confinement systems intended for normal operation.

The fuel cladding integrity is key to whether fuel becomes a source of radioactive fission products exposing personnel and the public to radiation. On this account, a safety limit is established to indicate the condition of the barrier. General

requirements for safety limits in terms of the amount of leaking fuel are specified in NP-082-07. At that, the numeric value assigned to the safety limit for specific activity of reference nuclides in the primary coolant relies on the requirements for non-violation of the safety limits outlined in NP-082-07. Moreover, the safety limit for the primary coolant activity based on the public dose quotas at the boundary of and outside the buffer area is characterized by even more stringent requirements for the quantity of leaking fuels, as opposed to NP-082-07.

When establishing a safety limit for the acceptable fission product source term in the reactor coolant burnup system, allowance is made for variations in the release of fission products from leaking fuel depending on.

Build-up of radioactive fission products in the fuel occurs solely during power operation, or in the C1 mode. In the C2 mode (MCPL) build-up of fission products due to splitting of uranium nuclei (and other splitting transuranic isotopes) can be ignored. Leaking fuel is detected as part of online cladding tightness testing where specific activity of reference nuclides is measured in the primary coolant. A safety limit value should be provided for the activity of reference nuclides in the primary circuit.

In addition to the above limits, another safety limit associated with the second physical barrier under C1 conditions is activity of reference fission products (in view of Novovoronezh-2 conditions the safety limit value for the primary coolant I-131-135 activity amounts to $1{,}5 \cdot 10^6$ Bq/kg dose equivalent I-131).

Given that the quantity of fission products escaping into the coolant of the operating reactor from leaking fuel is contingent on the condition of the first radioactive release barrier, this safety limit is also instrumental in identifying the integrity of the first barrier.

Monitoring is done via intermittent sampling. Should the design sampling system fail, the reactor shall be shut down immediately and brought to a hot state within the 24-hour time window since first noted.

Parameter	Barrier	SL value	Operational state	Justification
2 Primary leak rate, t/h	RCS boundary	2,0	C1 (power operation) C2 (MCPL) C3 (hot state)	The limit value ensures that the containment activity limit is not exceeded.

Parameters indicative of the condition of the RCS boundary are established with a focus on consequences of coolant leaks originating in various primary components. Having said that, a coolant leak may contribute to radiation exposure of the public and plant personnel, which is to say the location of the leak within or beyond the fourth physical barrier, the containment system, is of paramount importance. Containment bypass may be resorted to where a coolant leak develops in a SG header or heat-exchange tube or for example in a blowdown pipe of the chemical and volume control system, or the coolant flows out through a sensing line in the I&C compartment, which is out of containment.

Thus, the parameters representative of the health of the primary coolant boundary are as follows:

— activity in the containment (the value of the maximum coolant leak rate acts as a safety limit on activity in the containment). The limit introduced for C1, C2 and C3 is a primary coolant leak rate of 2,0 t/h;

— secondary side activity (the maximum primary-to-secondary leak rate is used as a safety limit on activity in the secondary circuit). The limit established for C1 is a primary-to-secondary leak rate of 5 kg/h;

— activity of noble gases and particulates discharged from the plant.

Parameter	Barrier	SL value	Operational state	Note
3. Primary-to-secondary leak rate, kg/h	RCS boundary	5	C1 (power operation)	The limit value ensures that the secondary side activity limit is not exceeded.

The unintended primary-to-secondary leak rate is a 'critical' parameter which governs radiological consequences of category 2 modes.

If the SG primary side leak rate of 2 kg/h serves as an operational limit to make sure the maximum acceptable radiation exposure of the public is not exceeded in category 2 modes (100 μSv per event), the safety limit on iodine-131 specific activity in the primary coolant shall be at most $7{,}3 \cdot 10^5$ Bq/kg, which is equivalent to 25 leaky fuel rods.

If the SG primary side leak rate of 5 kg/h serves as a safety limit to make sure the maximum acceptable radiation exposure of the public is not exceeded in category 2 modes (100 μSv per event), the operational limit on iodine-131 specific activity in the primary coolant shall be at most $2{,}91 \cdot 10^5$ Bq/kg, which is equivalent to 10 leaky fuel rods.

Given the improved robustness of Russian VVER fuel (manufacturer promises the likelihood of fuel failure of under $5 \cdot 10^{-6}$ events per year) this constraint is feasible. Though, specification of overly stringent fuel failure limits without consideration of potential operational conditions around them can actually lead to stricter limiting conditions of operation and a lower capacity factor, as well as add to challenges in implementation of new fuel types and fuel cycles.

As defined in NRB-99/09/9/, in design-basis accidents expected exposure of the critical group (limited number of locals) at the boundary of and outside the buffer area, shall not go beyond 5 mSv (0,5 rem) for the entire body and 50 mSv (5 rem) for individual organs over the first year following the accident. The above doses which comply with Criteria for Decision Making on Protecting the Public in an Accident at a Nuclear Facility found in NRB-99/2009 don't require protective action.

1.4.5 DESIGN CONDITIONS AND INITIAL CONDITIONS

Initial conditions are postulated events that result in anticipated operational occurrences and emergency conditions.

As initial events, the plant design shall consider both internal and external threats.

By the level of possible negative consequences and probability of occurrence, the list of conditions considered in the design was split into the following categories:

DBC1, normal operation;

DBC2, abnormal operation;

DBC3, design-basis accidents (DBA) level 3;

DBC4, DBAs level 4.

The DBC2 are anticipated operational occurrences initiated by design initial conditions, which can be expected once or more times every one hundred years of operation at a frequency of above 10^{-2} per reactor*year. In the worst case, they can result in reactor shutdown, after which plant operation can be restored upon eliminating failure causes. Such conditions do not tend to propagate and create threat of more severe failures, i.e. the DBC3,4 design conditions.

The DBC3 are DBAs level 1 initiated by design initial conditions, which can be expected less frequently that once every one hundred years of operation but at least once every thousand years of operation at a frequency of 10^{-2} to 10^{-4} per reactor*year.

The DBC4 are DBAs level 2 initiated by design initial conditions, which can be expected less frequently than once every thousand years of operation at a frequency of

10^{-4} to 10^{-6} per reactor*year. These events are postulated as, apart from other consequences, they result in a large radioactive release. As a rule, these conditions are caused by severe equipment damage and are the severest of all design conditions, for which protective measures shall be provided for in the design.

In addition to deterministic design conditions, certain design extension conditions (DEC) have been considered, which correspond to beyond-design accidents (BDBAs) as per General Provisions of Safety Assurance for Nuclear Power Plants. This was done to define a set of process measures and parameters of additional engineered features to control BDBAs and validate that they are effective in case of:

— complicated consequences, which include failures apart those ones provided for in deterministic design conditions but which do not result in core melt, i.e. DEC A;

— severe accidents with core melt, i.e. DEC B.

1.5 SAFETY FUNCTIONS, STRUCTURE OF ENGINEERED SAFETY FEATURES

1.5.1 FUNDAMENTAL SAFETY FUNCTIONS

As already mentioned, the fundamental safety objective is to mitigate and/or, if possible, completely exclude the destructive effect by influencing the source of effect and/or by combating the consequences of the effect /10/.

To this end based on the deterministic analysis, for every initial event defined are safety functions and features with account of the defense-in-depth principles.

As per NP-001-15, **the safety function is a specific objective and actions to achieve it aimed at preventing accidents and/or limiting their consequences.**

In accordance with It. 2.6.2 of the RB-152-18 /3/, the safety function is requirements for actions/operation of plant systems/elements and plant staff aimed to achieve a specific objective of preventing accidents or limiting their consequences.

During normal and abnormal operation, safety functions to prevent accidents are fulfilled by normal operation systems related to safety, for example primary equipment and pipelines - when they operate normally - prevent reactor-coolant-flowrate-reduction-accidents and loss-of-coolant accidents (LOCA).

The design of Novovorenezh NPP-2 is fitted with the safety systems intended to perform the following main safety functions:

— reactor scram and subcriticality maintenance;

— emergency heat removal from the reactor and spent fuel pool (SFP);

— keeping radioactive substances within the established boundaries;

— maintaining reactor pressure circuit integral.

The main operation objective is to simultaneously and constantly perform these fundamental functions, i.e. under all conditions, including unit refueling outage.

Depending on the plant state or initial conditions, fundamental safety functions are ensured by specialized safety functions (hereinafter, main functions).

For the Novovorenezh NPP-2 design, a list of main and specialized safety functions was developed given in Table 1.5.1.

Table 1.5.1 Main and specialized safety functions

A. Reactivity control	B. Emergency heat removal from the primary circuit and SFP to the heat sink	C. Radioactive substance retention within the established boundaries
AA: Emergency termination of the fission reaction	BA: Reactor coolant makeup	CA: Containment pressure control
AB: Subcriticality maintenance in the shutdown state	BB: Primary heat removal to the heat sink	CB: Containment fission product isolation
AC: Reactor power control	BC: Primary heat removal through the steam generator (SG)	CC: Removal of media from the annulus with cleaning
	BD: SG coolant makeup	CD: SG isolation in case of the primary-to-secondary leak
	BE: Primary circuit protection from overpressure and fragile embrittlement	CE: Supporting system isolation
	BF: Secondary circuit overpressure protection	CF: Containment hydrogen removal
	BG: SFP heat removal	

The plant design shall provide for the features and procedures to fulfill safety functions depending on the initial event. These features and procedures shall be developed taking into account independence of the defense-in-depth levels. In so doing, to perform specific safety functions engineered safety features shall include all necessary elements, systems, and structures, i.e. process ones, distributed control system, power supply system, cooling media, etc.

Further, it is expedient to clarify how plant elements are classified as related to safety.

1.5.2 SAFETY SYSTEMS

To perform safety functions in case of accidents, the plant design has a separate group of special systems named **safety systems.**

Plant systems and elements differ by:

— purpose;

— safety impact.

In addition, safety systems and elements, differ by the nature of safety functions performed by them.

By purpose, plant systems and elements are split into:

— normal operation systems and elements;
— safety systems and elements;
— systems and elements of special engineered features to control BDBAs.

By the safety impact, plant systems and elements are split into:
— safety related systems;
— other non-safety related systems.

By the nature of functions performed, safety systems and elements are split into:
— protective systems;
— isolation systems;
— supporting systems;
— control systems.

Protective safety systems serve to prevent or limit damage of fuel matrix, fuel cladding, and primary coolant boundaries, i.e. the reactor vessel, primary pipelines and other primary equipment. Protective safety systems protect first three physical safety barriers.

Isolation safety systems are intended to prevent or limit propagation of radioactive substances in case of plant accidents. An example of an integrated and most effective isolation safety system at the state-of-the-art plants is the containment.

Control safety systems actuate other safety systems, control, and monitor them when they perform functions assigned to them.

Supporting safety functions supply power and working media to the safety systems. These systems create necessary conditions for reliable functioning of safety systems.

Safety related systems include:
— safety systems/elements;
— normal operation systems/elements, failure of which violates plant normal operation or interferes with fixing plant abnormal operation if the conditional probability that this failure will transfer to the severe accident is 10^{-6} or higher;
— normal operation systems/elements, failure of which results in exceedance of maximum allowable emissions, permissible radioactive waste discharge or acceptable radioactive contamination of plant working premises;
— systems/elements provided for in the plant design to control accidents for the first three days after an initial accident or for other time interval set in the design, which should last at least three days;
— radiation control systems/elements.

The classification scheme described is shown in Figure 1.5.1.

The specific INSAG safety principles with specifying relations between them are given, for example, in INSAG documents.

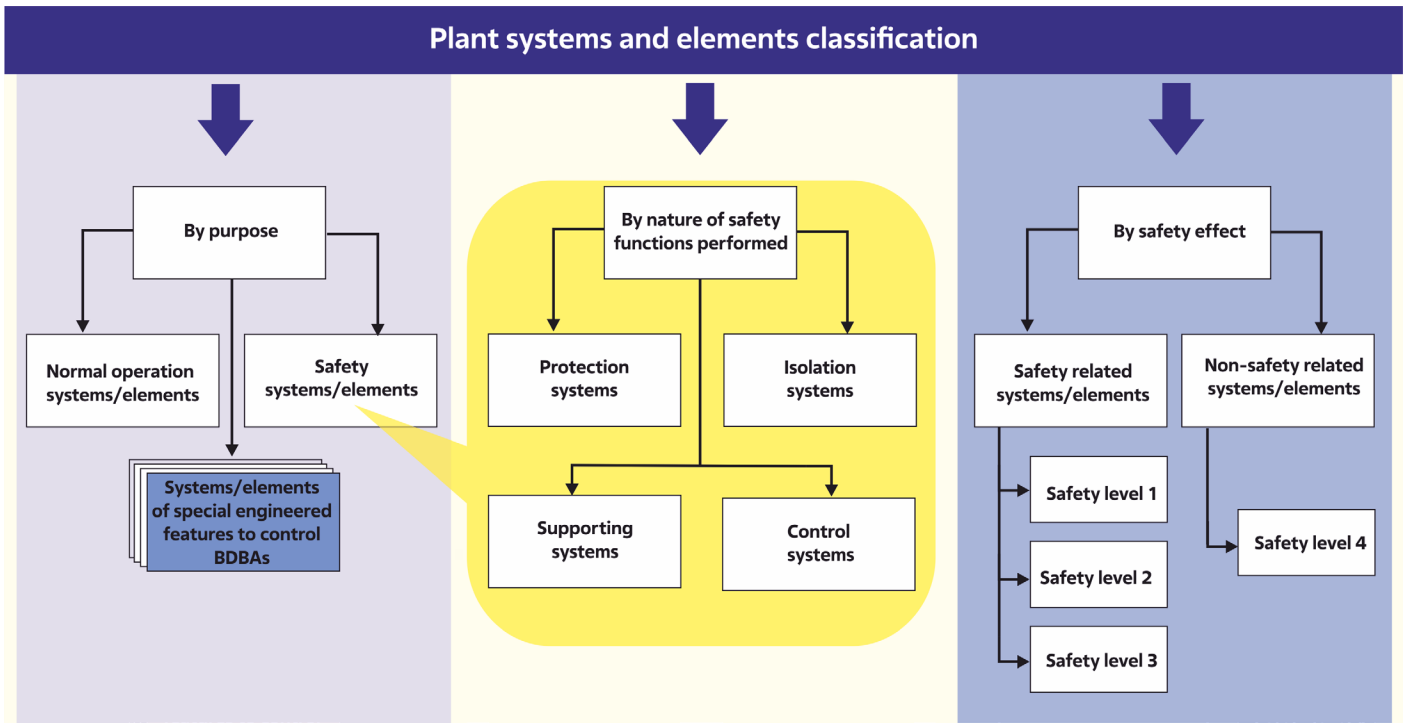


Figure 1.5.1 Classification of plant systems/elements

Note: Level 4 can include systems and elements of special engineered features to control BDBAs, for example, the core melt

1.5.3 SAFETY FEATURES

Reliability of Functions Performed

Safety features intended to perform safety functions shall be designed in compliance with principles that ensure required safety owing to failure resistance and by using redundancy, independence, and diversity principles.

Failure Resistance

Failure resistance is ensured by preventing abnormal operation resulted from plant equipment failures, external effects, and abnormal operation conditions becoming DBAs. To this end, the design has:

— required margins between normal operation limits and setpoints of safety systems;

— redundant components;

— automatic control systems that regress processes to normal parameters.

During design, a failure tolerance assessment (FTA) is done to demonstrate reliability of safety functions with account of common-cause failures.

Redundancy

The redundancy principle includes the following:

— redundancy of systems by using several channels;

— redundancy of elements and elements inside system channels.

A single failure is an independent failure occurring in addition to the failure that resulted in an accident or plant malfunctioning.

The design of the safety systems shall exclude loss of functions performed by safety systems as a result of failure of one component. It is called a **single failure criterion** for safety systems.

In accordance with this principle, the safety system shall function in case of any initial event and in case of a failure of any element of this system independently of the initial event.

Limits set for DBAs shall not be exceeded at any initial event considered in the plant design with supposition over the initial event in accordance with the single failure principle of one failure that is independent of the initial event for any such safety system element as an active or passive element fitted with the mechanical rotating parts or a passive element without rotating parts, for which there is a possibility of not fulfilling the safety function of 10^{-3} or higher, or one human error that is independent of the initial event.

In addition to a failure of one of the above components that is independent from the initial event, considered shall be all failures resulted from this single failure, failures resulted from the initial event, and failures of elements that are not

detected during plant operation affecting the accident propagation.

The passive elements are actuated immediately by the initial event. An active element is an element, for actuation of which it is required to take some active actions, for example, to supply power to the drive, supply compressed air or other actions. In electric systems, all the elements are considered to be active.

The design excludes failures by making respective requirements for the quality of the design, manufacture, and maintenance of systems and elements that include inspections and testing. As a rule, the quality of systems, structures, and components is defined by their safety class as per NP-001-15.

The single failure principle should be considered not only as related to active equipment of systems such as pump units and valves but also as related to passive elements of the systems that include mechanical elements like check valves.

The degree and approach to redundancy during system designing shall be defined in accordance with regulatory safety requirements and based on deterministic and probabilistic analysis. Depending on the level of the functions performed and level of the defense-in-depth, redundancy requirements are different for systems.

When an initial event originates, all standby safety channels are actuated to retain functions in case of a single failure and failure that is dependent on the initial event.

Reliability of the safety functions shall be ensured by making redundant all systems and elements involved in this function, including features of the DCS, power supply, etc. So, configuration of supporting and control systems – level of their redundancy – shall correspond to that of the safety systems.

Independence

The system channels shall be functionally independent and physically separated to exclude failures caused by mutual interaction between these channels in case of internal action or by using components common for two or several channels.

Functional independence is ensured when:

— components are shared with other channels.

— Power is supplied to channel components by a separate power source different from that for other channels.

— To control active components, a special set of the DCS equipment is used that includes measurement channels, signal processing and generation modules, annunciators, priority control modules, etc. with the DCS equipment powered by the same power supply channel as active components.

— All auxiliary systems for heat removal, media supply, ventilation, cleaning, etc. needed to perform safety functions also have a channel structure with independent channels.

Control safety systems are functionally and physically separated from normal operation DCSs.

The design uses control systems designed to function at different levels of the defense in depth.

To make them more independent – especially as related to some common-cause failures – used is to the extent reasonably practical a layout and design of systems using physical separation principles. These principles include:

— separation using geometrical factors, i.e. distance, orientation, etc.;

— separation by barriers;

— separation by combination of the above arrangements.

Which separation method will be used depends on what effects were considered in the design, i.e. those resulted from fires, chemical explosions, aircraft crashes, projectiles crashes, flooding, temperature actions, humidity impacts, etc., to the extent applicable.

The design implements principles of physical separation for buildings, structures, and equipment in the nuclear island using layout, distance, location, and physical barriers.

Arrangements for physical separation of safety channels and equipment in containment are designed for loads by internal and external effects provided for in the design. Safety systems are protected from dependent failures by locating system channels in separate premises. Safety channels are separated by physical fire-resistant barriers designed for internal and external effects.

Diversity

The diversity principle means usage of features based on different principles of action to perform the assigned function.

The diversity principle is applied for diverse systems and units with the same function by introducing different distinctive features into these systems or units. These distinctive features can be different principles of operation, different physical variables, different operation conditions, different equipment manufacturers, etc.

While designing, considered should be all possible similarities of materials, elements or manufacturing processes, as well as implicit operation principles or those of shared auxiliary systems and features. In case when unlike elements and systems are used, a sound degree of confidence shall be ensured that introducing them will be, as a whole, useful with account of such deficiencies as additional complicacy of operation procedures, maintenance procedures, testing checking procedures with the resulting usage of less reliable equipment.

The diversity principle in the design shall be applied to engineered accident management features, supporting features, and control features.

In accordance with the regulatory documents, the following shall be ensured based on diversity:

— The reactivity control function shall be ensured by introducing both solid neutron absorbers and liquid absorbers.

— The function of heat removal from the reactor core and containment shall be ensured both by active safety systems and passive ones.

— The function of containment isolation shall be performed by active elements, i.e. electrically-driven isolation valves, and passive elements, i.e. check valves.

— The function of programmable features control shall be diversified by hardware features.

Initiating signals to trigger certain safety functions shall be generated using at least two different parameter measurement methods.

When it is not possible to use passive systems and elements, the diversity principle is implemented by using active equipment of different types – electrically-driven valves, electric pumps, and diesel generators – and equipment of different manufacturers.

1.6 INHERENT REACTOR SAFETY AND NUCLEAR SAFETY

1.6.1 INHERENT REACTOR SAFETY

The first level of defense-in-depth assumes development of plant design documentation based on the conservative approach with a developed inherent reactor safety property and arrangements aimed to exclude a cliff edge effect.

The cliff edge effect is a significant abrupt plant safety deterioration caused by small changes in parameters.

The inherent plant safety is an ability of the reactor plant to prevent initial events and accidents, limit their consequences without personnel, power, and external support for a long time, which will be used by personnel to judge the situation and carry out corrective actions.

The criteria of the inherent safety level are a duration of the accepted non-interference period in different situations, level of self-limitation of most important safety parameters, and inertia of emergency processes.

The design of Novovorenezh NPP-2 excludes necessity of immediate emergency actions by personnel in case of abnormal occurrences. The time margins for corrective actions by personnel is given in Table 1.6.1.

Table 1.6.1 Time margins for corrective actions by personnel

	Corrective actions	Time margin, hr.
1	Actions by operations personnel in the control room (CR), minimum	0,5
2	Actions by operations personnel in the emergency CRs and at locations of special equipment – jumper cables, valve drives, etc., minimum	1
3	Actions by plant personnel using portable equipment, minimum	6
4	External support, minimum	24

Inherent safety is ensured in the B-392M reactor plant (RP) by the following design solutions:

1. Core power release is self-limited using negative reactivity coefficients by specific coolant volume and fuel temperature, reactor power, aggregate reactivity coefficient by fuel temperature and coolant temperature in all critical states possible within all reactor parameter range during normal and abnormal operation, including DBAs.

Necessary temperature and density reactivity coefficients are ensured in the design by using gadolinium burnable absorber (BA) integrated into the fuel. Concentration of the fuel BA is selected so as to ensure almost complete burnup of BA during one fuel cycle (FC).

2. Control rod drive mechanisms (CRDMs) are actuated in emergency conditions by gravity, i.e. control rods drop by gravity.

3. Direct action devices and passive safety systems are widely used. As related to main safety functions, the V-392M design uses both active and passive principles. Action of active and passive safety channels is cumulatively considered in the design for both DBAs and BDBAs.

4. Runout by inertia is used for special rotating masses of primary coolant pumps (PCPs) to ensure necessary flowrate decrease through the core in case of power loss.

The V-392M design uses the GTsNA-1391 RCP with removable electric motor and flywheel installed on the rotor. The flywheel is a stainless-steel disc with the mass of 10,500 kg. The mass was increased as compared with that of the GTsN-195M RCP due to usage of water as a lubricating oil for the radial thrust bearing – the minimum radial gap in the hydrodynamic wedge decreased by 10 times as compared with the oil lubrication.

The design sets requirements for the coolant flowrate in the loop in case of inertial RCP runout. In case of loss of power for four RCPs at nominal primary parameters:

— The time of RCP runout to full stop shall be minimum 85 s.

— The pump flow for 30 s after power loss shall decrease maximum by 2,7 times.

5. The layout of equipment and geometry of the RP primary circuit ensure creation and maintenance of natural circulation for the primary coolant and core heat removal up to 10 % of the nominal power under natural circulation.

For the V-392M RP, the vertical displacement of the SG middle portion with respect to the reactor core middle portion is above 10 meters (Figure 1.6.1).

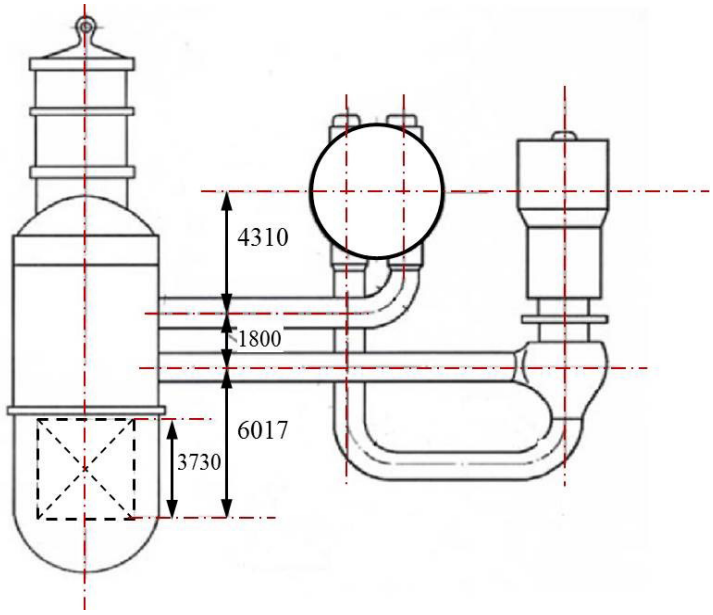


Figure 1.6.1 Primary equipment layout and geometry

6. Absence of the cut-in pipes and holes in the reactor vessel below the upper core level creates a leak-tight vessel around the core that prevents fuel assembly uncovering in case of LOCA.

As compared with the V-320 RP, in the V-392M RP the core coolant volume is larger as the inner diameter of the reactor vessel was increased by 100 mm and the length of the support barrel, by 300 mm.

7. The V-392M RP is fitted with the PGV-1000MKP SG. As compared with the PGV-1000M SG of 127 m^3, the volume of the secondary water in the PGV-1000MKP SG of 150,2 m^3 is larger as the inner diameter of the SG was increased by 200 mm. The increased volume of SG secondary water makes it possible to remove residual heat from the core by discharging steam through SG dumps during the time needed to actuate the SG passive heat removal system (PHRS).

8. The V-392M RP design is fitted with a passive system for catching, cooling, and ensuring subcriticality of core melting outside the reactor vessel. The core catcher is installed above the reactor vessel (see Chapter 3) and ensures protection of the containment foundation plate from damage in case of reactor vessel melting.

9. In the primary circuit, the V-392M RP conventionally uses pressurizer with the steam blanket thanks to which during operation pressure deviation from the

nominal values is softly limited in case of steam blanket compression or expansion in the pressurizer. In the process of evaporation/condensation as a function of pressure, the water volume due to return coupling is also involved in pressure maintenance, i.e. when steam volume increases – level decreases – pressurizer water evaporates to maintain pressure, and when the steam phase is compressed it condenses on the water level to limit pressure buildup.

1.6.2 PRINCIPLES OF NUCLEAR SAFETY ASSURANCE FOR THE VVER-1200 RP

The RP nuclear safety is defined as a whole by the technical maturity of the design and is primarily based on the reactor inherent safety in the form of internal core nuclear feedbacks characterized by reactivity coefficients. As applied to the VVER reactors, considered are reactivity coefficients for coolant temperature (**dp/dtH_2O**), fuel temperature, coolant density (**dp/dyH_2O**), reactor power, and coolant boron concentration.

In accordance with the NP-082-07 /7-8/ Nuclear Safety Rules for Nuclear Power Plants, core inherent safety is ensured using negative reactivity coefficients for specific coolant volume and fuel temperature, reactor power, cumulative reactivity coefficient for fuel and coolant temperature in all critical states possible under all range of reactor parameters during normal and abnormal operation, including DBAs.

Necessary signs of temperature and density reactivity coefficients are ensured in the design taking into account usage of the Gd BA integrated into the fuel.

The negative fuel temperature reactivity coefficient is conditioned by the Doppler's effect, i.e. by resonance neutron absorption in ^{238}U and thus is ensured for the entire temperature range.

The power coefficient of reactivity is defined by the sum of two constituents, one of which is associated with the coolant temperature change, and another, with that of fuel temperature. The negativity of each constituent ensures negativity of the cumulative reactivity power coefficient.

The boron concentration reactivity coefficient is negative in all states.

Another important aspect of nuclear safety assurance in the process of RP operation is selection and validation of the instrumentation and control systems. The RP design uses two independent reactivity control systems – boron control and mechanical control systems. Both systems comply with requirements imposed for them in the NP-082-07. In all the states considered, which are associated with the reactivity buildup when CR are withdrawn at the working speed and when boron is diluted in the coolant, the rate of positive reactivity insertion does not exceed the limit set in the NP-082-07 equal to 0,07 β_{eff}/s.

The efficiency of the protection was selected so that with account of most effective CR failure reactor shutdown is ensured in all possible operation states, pre-emergency situations and DBAs, and subcriticality is maintained before the boron systems starts to operate effectively with account of released positive effects of reactivity. This condition was confirmed by analysis of reactivity balances and by safety analysis.

Characteristics of the core and reactivity control features ensure negative reactivity insertion when they are inserted to the core at any point of movement for any combination of them in case of normal, abnormal operation, and DBAs.

To validate safety, presented is information about reliability of separate reactivity control systems and subsystems as related to:

— CRDMs;

— CRs;

— primary circuit blowdown/makeup system and boron control system;

— emergency boron injection system.

To prevent local criticality insertion at refueling, nuclear fuel (NF) transportation and storage, the RP design provides for the following:

— design/engineering solutions and organizational features to exclude conditions that can result in critical states during fuel transportation and storage;

— FA handling along the entire fuel handling path at the plant using conventional handling equipment.

Nuclear safety analysis and validation were performed in accordance with requirements of the NP-061-05 Safety Rules of NF Storage and Transportation at Nuclear Installations for fuel with maximum enrichment of 5 % by ^{235}U used in the reactor at the maximum design load upon transportation and storage facilities.

It was shown in the design that under normal conditions with account of process deviations for design elements the required subcriticality of $F_{eff} < 0,95$ is ensured for fresh FAs.

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2 SAFETY SYSTEMS

2.1 PROTECTIVE SAFETY SYSTEMS

Protective safety systems/elements are safety systems/elements that prevent or limit damage of the NF, fuel cladding, equipment, and pipelines with radioactive substances as per the NP-001-15/1/.

These systems/elements directly protect first three physical barriers on the way of radioactive substance propagation. To protect the fourth barrier – containment – isolation safety systems are used/2/.

Table 2.1.1 gives safety functions performed by protective safety systems embedded into the Novovorenezh NPP-2 design in all normal operation, abnormal operation conditions, including DBAs and BDBAs.

Table 2.1.1 Protective safety functions of the Novovorenezh NPP-2 design

A. Reactivity control	B. Emergency heat removal from the core, primary circuit, and SFP to the heat sink
AA: Emergency termination of the fission reaction	**BA:** Reactor coolant makeup
AB: Subcriticality maintenance in the shutdown state	**BB:** Primary heat removal to the heat sink
	BC: Primary heat removal through the SG
	BD: SG coolant makeup
	BE: Protection of the primary circuit from overpressure and fragile embrittlement
	BF: Secondary circuit overpressure protection
	BG: SFP heat removal

The Novovorenezh NPP-2 design is fitted with the following protective safety systems:

— emergency reactor protection system, AA and AB safety functions;— JND emergency boron injection system, AA and AB safety functions;

— JNA emergency core cooling system (ECCS) and emergency SFP cooling system (active portion of the ECCS), BA, BB, AB, and BG safety functions;

— JNG50-80 system of hydraulic accumulators of stage 1 (passive portion of the ECCS), BA, AB, and BE safety functions;

— JNG10-40 system of hydraulic accumulators of stage 2 (passive portion of the ECCS), BA and AB safety functions;

— KTP emergency gas removal system, BA and BE safety functions;

— JNB10-40 SG emergency cooldown system, BC and BB safety functions;

— JNB50-80 passive heat removal system, BC and BB safety functions;

— main pipeline cutoff system, AB and BD safety functions;

— pressurizer pilot-operated relief valve (PORV), i.e. primary circuit overpressure protection system, BE safety function;

— secondary circuit overpressure protection system, BF safety function.

Configuration of protective safety systems is based on usage of interredundant passive and active systems and/or usage of active systems with elements of different design to perform safety functions.

The BB safety function – primary heat removal to the heat sink – is performed jointly by the JNA, JNB10-40 active protective elements and supporting safety systems, i.e. the KAA that is an intermediate circuit for essential consumers in the reactor building and the PEB/PEC that is a cooling water system for essential consumers of the unit. The SG PORV passive system removes heat to the heat sink – air – without the KAA and PEB.

The BG safety function – SFP heat removal – is done by evaporation. In so doing, SFP makeup is done by the FAK that is an isolation safety sprinkling system or by the JNA in case of BDBAs with FAK failure.

2.1.1 REACTOR PROTECTION SYSTEM

The reactivity control safety function means emergent termination of the fission reaction and maintaining reactor subcritical within the entire range of working parameters.

This function is done by interredundant safety systems:

— reactor protection system;

— JND emergency boron injection system.

The systems are based on different principles of action to ensure the functional and design diversity and high reliability level.

2.1.1.1 General Data about the Reactor Protection System

The working element in the reactor protection system is absorber rods (ARs) of the CPS connected to the CRDM mast.

As the CPS actuator, the ShEM-3 electromagnetic drive is used.

ARs are a mechanical reactor CPS intended to:

— function as preventive and emergency reactor protection;

— maintain the critical state when operating at the stationary power level and control core power distribution;

— change reactor power and compensate for the reactivity change during xenon poisoning/depoisoning.

In the VVER-1200 design, ARs can be located in all FAs, excluding those at the border of the core baffling, i.e. 121 FAs.

The core power is regulated when ARs are displaced by the CRDMs at the constant rate of 2 cm/s in FA GTs.

Protection is actuated when ARs drop upon emergency signals by gravity.

ARs contain eighteen absorbing elements, a mast with a gripper, springs, and nuts (Figure 2.1.1.1).

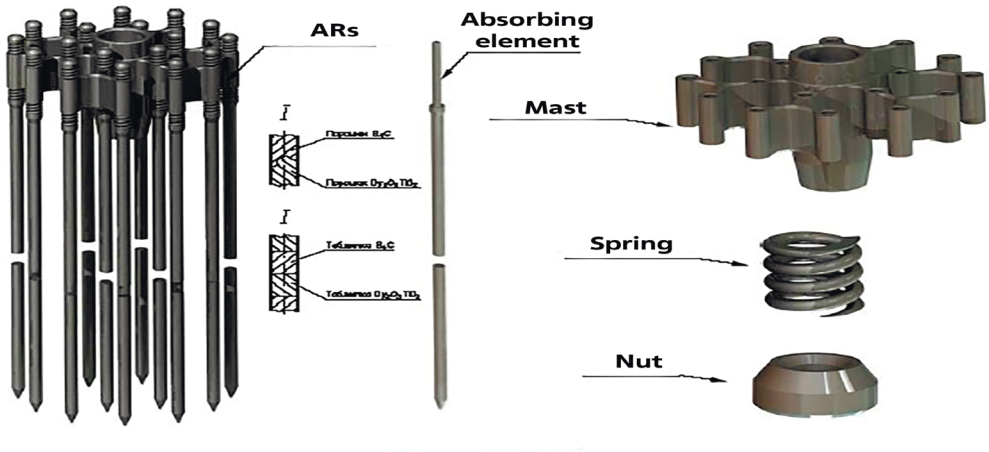


Figure 2.1.1.1 ARs design

The ARs are fitted with a combined absorber (Fig. 2.1.1.2) with the upper portion containing the B_4C boron carbide with boron of the natural isotope composition and the lower portion ~500 mm long containing the Dy_2O_3-TiO_2 dysprosium titanite.

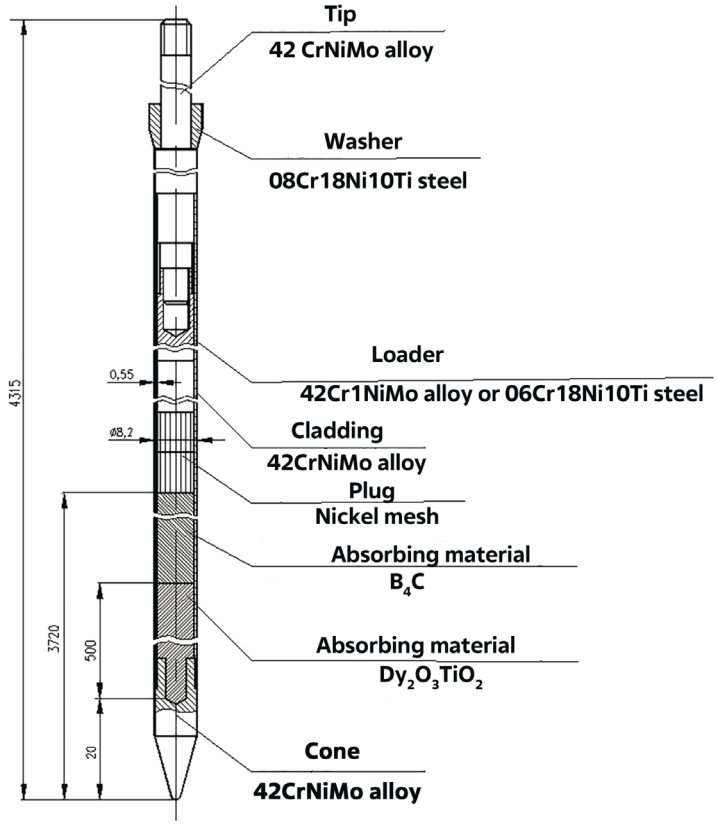


Figure 2.1.1.2 ARs design

Dysprosium is an (n,y) absorber and, to the smaller extent, undergoes the radiation-induced damage as compared to the boron-containing (n,α) absorber. The dysprosium titanite in the AR lower portion makes it possible to reduce gas release in the location of most neutron fluxes and thus extend the AR lifetime while retaining sufficient efficiency of protection.

The absorbing material is fixed against displacement by plugs made of the nickel mesh. The AR tip is connected by welding to the loader, which ensures the required AR weight.

The ARs are fixed using individual suspension springs at the gripper mast. The gripper ensures possibility that ARs engage with the CRDM mast and handling equipment. In so doing, the design excludes the possibility that the mast in the unauthorized manner rotates with respect to the AR gripper and disengages with ARs.

Table 2.1.1.1 gives main characteristics of the ARs.

Table 2.1.1.1 Main characteristics of the ARs

Parameter	Value
Quantity of absorbing elements in the ARs, pcs.	18
AR height, mm, nominal	4315
Outer diameter of the absorbing element cladding, mm, nominal	8,2
Cladding thickness of the absorbing element, mm, nominal	0,55
Cladding material of the absorbing element	42CrNiMo (steel)
Absorbing material of the absorbing element	$B_4C + (Dy_2O_3 \times TiO_2)$
Density of the absorbing material, g/cm^3, minimum: — boron carbide (B_4C); — dysprosium titanite ($Dy_2O_3 \times TiO_2$)	 1,7 4,9
Height of the absorbing material, mm, nominal: — dysprosium titanite; — total (with account of the cone)	 500 3720
AR weight, kg	18,5±0,5
Assigned AR lifetime, years	10, of which maximum 3 as part of emergency control
Assigned AR lifetime, eff. h	82000, of which 25500 as part of emergency control

The maximum efficiency of single ARs is limited so that in case of an accident with ARs drop with account of protection actuation the assigned limits of fuel damage are not exceeded. Efficiency of single ARs ejection with account of the design error is maximum 0,66 % and 0,21 % for, respectively, minimum controlled power and nominal power.

All ARs are combined into 12 groups. ARs in every group are located at the same height and are displaced simultaneously.

Four Groups 12, 11, 10 and 9 that contain, respectively, six, seven, nine, and twelve ARs are used for control. They were selected based on the possibility of reactor shutting down from the nominal power at the lowest power perturbance. These groups can be also used during design reactor load shedding conditions and power control in xenon transients. Group 12 is the working one.

Grouping of ARs is given in Figure 2.1.1.3. Six ARs in Group 7 marked in the cartogram function as accelerated preventive reactor protection.

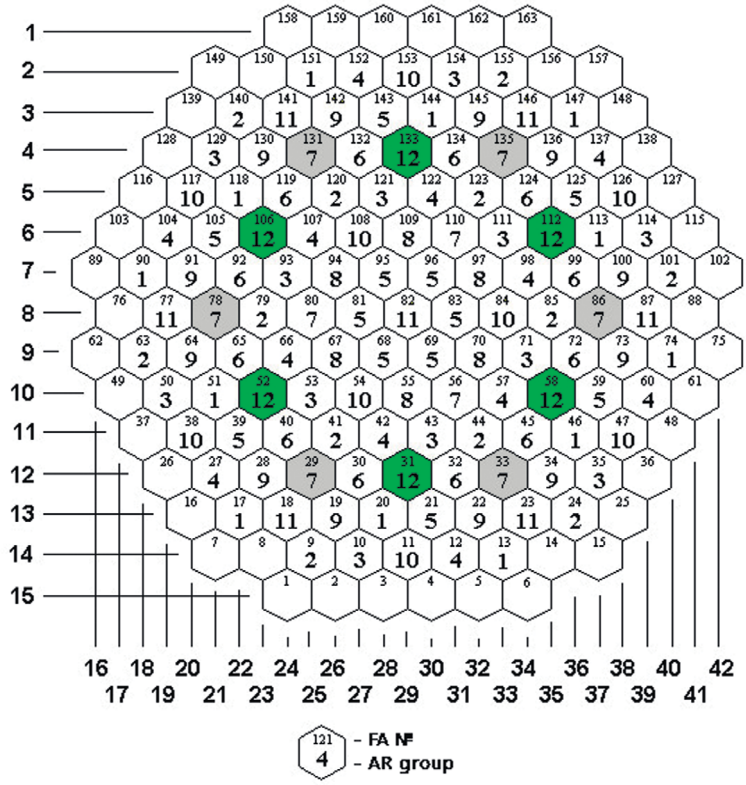


Figure 2.1.1.3 AR distribution by groups in the VVER-1200 core

In normal and abnormal operation, the maximum reactivity insertion rate conditioned by ARs withdrawal does not exceed 0,07 β_{eff}/s, where β_{eff} is the effective fraction of delayed neutrons in its limiting value in accordance with requirements of the NP-082-07.

Reactor scram is done by the protection by inserting all ARs to the reactor core by gravity after power loss at the ShEM drive holding coils (Figure 2.1.1.4). The AR drop time is rated and is maximum 4,0 s and minimum 1,2 s.

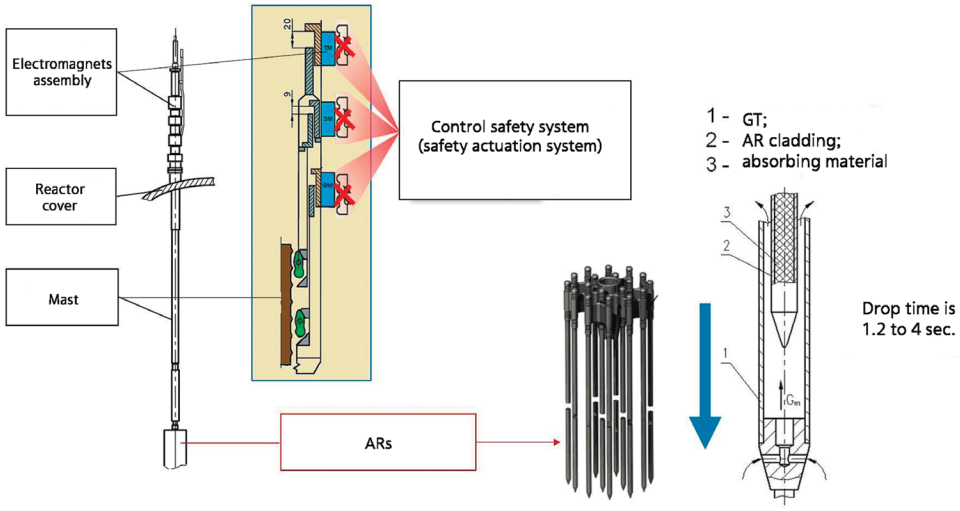


Figure 2.1.1.4 Principle of operation for reactor protection

Protection must make the reactor subcritical at any instance of operation from any power and maintain subcriticality in the process of emergency cooling down accounting for the most effective AR stuck in the uppermost position. Protection must be efficient enough to make the core subcritical and maintain it subcritical during cooldown with the temperature of repeated criticality of up to 100 °C at the current primary coolant boron concentration at any instant of cycle without one most effective AR.

For every fuel loading, experimentally defined is efficiency of the protection with all CRs functioning and with one most effective CR stuck.

The results of testing the first fuel loading at Unit 1 of Novovorenezh NPP-2 are as follows:

— The overall efficiency of the protection was —10,3 % (—13,9 β_{eff}).

— Efficiency of the protection without the most effective CR (14—33) was — 7,9 % (— 10,7 β_{eff}).

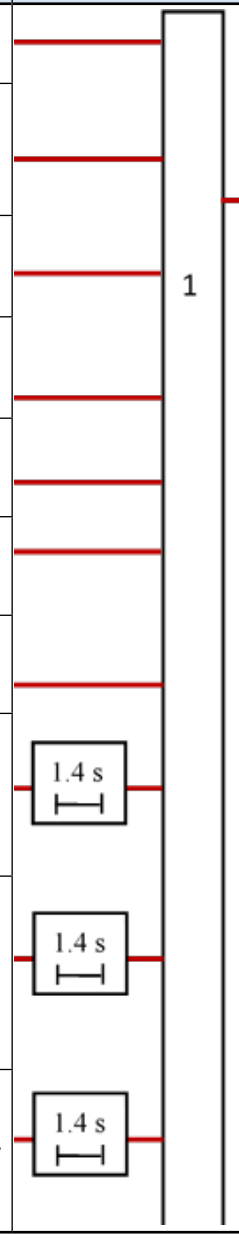
2.1.1.2 Algorithms of Emergency Signal Triggering by the Control Safety System

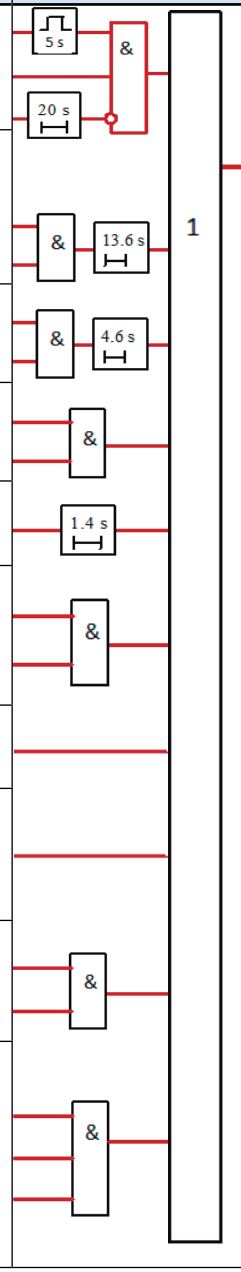
To prevent violation of safety operation limits, the design has setpoints actuation for the scram and process safety systems.

Protection prevails over other types of protection and control. Protection has two independent identical sets in its structure. Protection is completed to the full even when the signal of the initial cause is silenced.

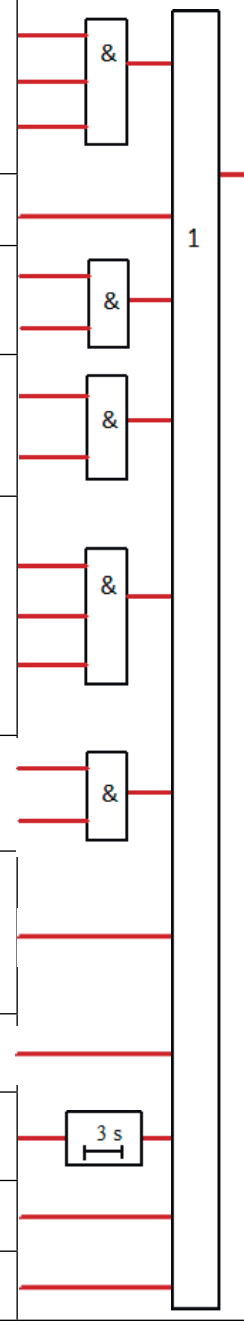
The list of actuating signals and logics of the protection are given in Table 2.1.1.2.

Table 2.1.1.2 List of signals and logics of protection

Parameter	Actuation setpoint	Logics	Protective action
1. From the scram button in the CR or emergency CR			CRDMs power loss
2. Period of neutron power change in the source range, actuation and working range, minimum	10 s		
3. Neutron power in the measurement range of 10^{-8} to 1 $\%N_{nom}$, above	Allowed value (assigned by the operator)		
4. Neutron power in the measurement range of 10^{-1} to 120 % N_{nom}, above	Allowed value (assigned by the operator)		
5. Neutron power, above	107 % N_{nom}		
6. Neutron power, above	Sliding scram set-point		
7. Neutron power, above: — for four RCPs;	107 % N_{nom}		
— for three RCPs;	Reduction of the setpoint from 107 to 74 by the linear law at the rate of 0,5 % N_{nom}/s		
— for two RCPs;	Reduction of the setpoint from the current value to 47 by the linear law at the rate of 0,33 % N_{nom}/s		
— for less than two RCPs	Reduction of the setpoint from any current value to 5 by the linear law at the rate of 0,33 $\%N_{nom}$/s		

Parameter	Actuation setpoint	Logics	Protective action
8. RCP pressure drop, below RCP power, below	0,392 MPa 0,245 MPa 180 kWt		CRDMs power loss
9. RCP power below 3000 kWt (signal delay of 1,4 s): — at one out of four RCPs reactor power, % N_{nom}, above	90		
— at two out of four RCPs reactor power, % N_{nom}, above	75		
— at above two RCPs reactor power, % N_{nom}, above	5		
10. Power rate of three out of four RCPs, below	46 Hz		
11. Pressurizer coolant level, below Coolant temperature of any hot loop threads, above	4,0 m 150 °C		
12. Containment excess pressure, above	30 kPa		
13. Boling margin, i.e. difference between saturation temperature and temperature of any four hot loop threads, below	8 °C		
14. Excess pressure at the reactor outlet, below Neutron power, above	15,1 MPa 75 % N_{nom}		
15. Excess pressure at the reactor outlet, below Coolant temperature of any hot loop threads, above Position of the cooldown switch	14,1 MPa 260 °C Off		

16. Excess pressure at the reactor outlet, above Coolant temperature of any hot loop threads, above Position of the cooldown switch	17,5 MPa 150 °C Off	CRDMs power loss
17. Coolant temperature of any hot loop threads, above	$(T_{nom}$ + 8) °C	
18. SG water level, below Coolant temperature of any hot loop threads, above	$(H_{nom}$ – 0,65) m 150 °C	
19. Excess pressure at the reactor outlet, MPa, above Coolant temperature of any hot loop threads, above	8,0 150 °C	
20. Excess pressure at the SG outlet, below Difference between saturation temperatures of the primary and secondary circuit, above Coolant temperature of any hot loop threads, above	5,53 MPa 70 °C 150 °C	
21. Pressurizer coolant level, above Position of the cooldown switch	9,4 m Off	
22. Rate of the gamma-radiation at the steam pipeline surface, above (at the temperature of four hot loops below 150 °C, the scram signal is blocked)	Allowed value, μZv/h	
23. Horizontal acceleration, above	0,67 m/s^2	
24. Loss of voltage at two power-lead terminals of the CRDMs		
25. Critical power ratio of the FA surface, below	Allowed value	
26. Linear power density, above	$Q_{al\ core}$ Wt/cm	



Actuation scram signals are grouped by the type of the initiating event and form eight control and protection functions, i.e. AA11–AA18 (Table 2.1.1.3).

Table 2.1.1.3 Reactor protection algorithms

Code	Control and protection function
AA11	Failed reactor reactivity and power control
AA12	Primary coolant flowrate reduction
AA13	Loss of primary coolant
AA14	Loss of feedwater/steam pipeline rapture
AA15	Primary-to-secondary leak
AA16	Seismicity
AA17	Loss of 380 V power voltage
AA18	Failed power distribution

All initiating events in every set of protection are implemented through the RS trigger (Figure 2.1.1.5).

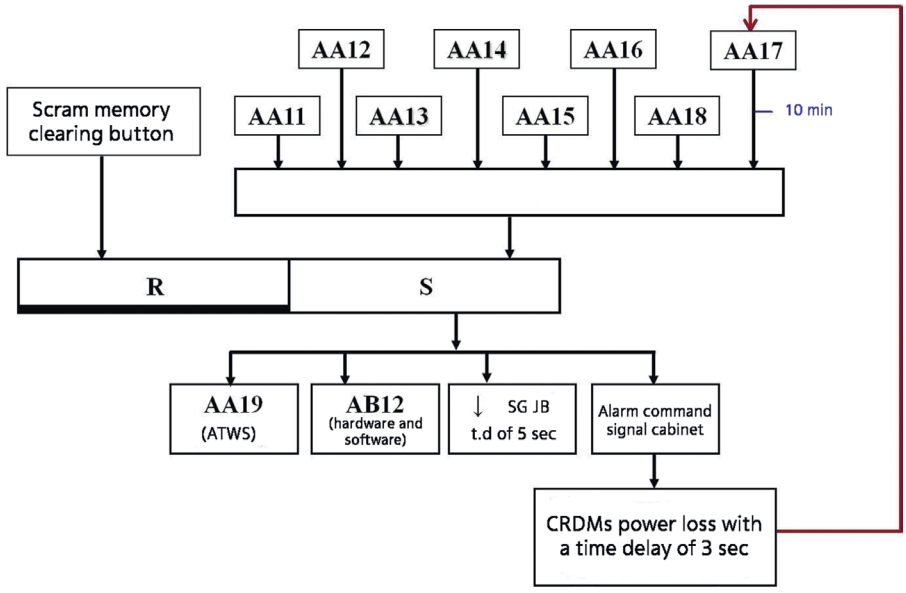


Figure 2.1.1.5 Algorithms of reactor protection

After actuation of the scram trigger upon any signal, the following commands are generated:

— to the cabinet of emergency commands → de-energize the CPS magnets (loss of power);

— to the system of turbines protections → close the SG turbogenerator stop volve at a time delay of 5 s;

— to the AB12 algorithm → block distillate supply;

— to the AA19 algorithm → actuate scram to emergently inject boron in case of protection failure, i.e. actuate the JND to the primary circuit in case of ATWS;

— to the CR/emergency CR alarm system → cumulative signal to actuate the scram trigger and scram root cause alarm.

Time of the scram alarm command from actuation of the threshold devices to de-energization of CRDMs shall not exceed 0,3 s.

After scram actuation, engineered features exclude the possibility that operators manipulate CRs for 10 min.

To this effect in case of any scram actuation, in 10 min the AA17 algorithm generates the CRDMs De-Energization impulse signal at a time delay of 3 sec (Figure 2.1.1.5).

The AA11 Function: Failures in Reactivity and Power Control

In any scram set, the scram alarm generates by the following parameters:

— scram command by the CR operator;

— scram command by the emergency CR operator;

— flux period below 10 s;

— neutron reactor power above the allowed value, which is set by the operator within the source range, startup or working range;

— neutron reactor power within the working range of above 107 % nominal that is a hardware setpoint;

— neutron reactor power above the sliding setpoint;

— exceedance of the reactor neutron power above the allowed value depending on the number of working RCPs, i.e. a variable setpoint.

In addition to the neutron power setpoint manually set by the operator, the Novovorenezh NPP-2 design has automatically generated sliding and variable setpoints.

The measurement range of the scram sliding setpoint is 7 % N_{nom} to 107 % N_{nom}. Under the stationary conditions, the sliding setpoint shall exceed the current neutron power by 7 % N_{nom}.

When the reactor power rises, the sliding setpoint changes at the rate defined in the design for this power range (Table 2.1.1.4).

Table 2.1.1.4 Rate of sliding setpoint change at power increase depending on the power level

Algorithm mode	Reactor power, % N_{nom}	Allowed rate of power uprate, % N_{nom}/min
1	45, below	3
	45, above	1
2	50, below	3
	50 to 80	0,17
	80, above	0,017

At the rate higher than specified in Table 2.1.1.3, at first the alarm system will actuate and after it fails, scram.

State 1 is the main one.

State 2 is actuated under the following conditions:

— reactor operation for above 12 days in a stationary condition at any reduced power;

— after refueling at reactor operation for above 12 days at the end of the previous cycle at the power reactivity effect of below 85 % N_{nom};

— after connection of the out-of-operation loop.

When power decreases, the rate of sliding setpoint decrease is 0,5 % N_{nom}/sec.

Depending on the number of operating RCPs, the variable neutron power scram setpoint is generated in the following way:

— When one out of the four RCPs is disconnected, the variable setpoint linearly decreases from 107 % N_{nom} to 74 % N_{nom} at the rate of 0,5 % N_{nom}/sec.

— When one out of the three RCPs is disconnected — two adjacent or two opposite RCPs continue to operate – the variable setpoint linearly decreases from the current value to 47 % N_{nom} at the rate of 0,33 % N_{nom}/sec.

— When three and more RCPs are disconnected, the variable setpoint linearly decreases from the current value to 5 % N_{nom} at the rate of 0,33 % N_{nom}/sec.

When the RCP is connected, the value of the variable scram setpoint shall instantly restore to the maximum variable setpoint that corresponds to the number of operating RCPs.

AA12 Function: Reduction of the Primary Coolant Flowrate

Emergency trip of RCPs results in reduction of the coolant flowrate through the reactor core. To prevent fuel rod damage as a result of insufficient core cooling, it is necessary that scram actuates under certain conditions.

Every scram set generates the tripping signal by the following parameters:

— RCP pressure loss decreases from 0,392 MPa to 0,245 MPa in less than 5 seconds in case of RCP shaft jamming or runaway.

— One out of four RCPs is disconnected at the power of above 90 % N_{nom}. The alarm is generated at a time delay of 15 seconds. This time is given to reduce RP power by the accelerated protection system.

— Simultaneously, two out of four RCPs are disconnected at the power of above 75 % Nnom. The alarm is generated at a time delay of 6 seconds. This time is given to reduce the RP power by the accelerated protection system.

— Three out of four, two out of three or one out of two PCPs are disconnected at different power levels of above 5 % N_{nom}.

— The frequency of three out of four RCP power supply boards is below 46 Hz. The alarm is generated at a time delay of 1,4 seconds.

The RCP disconnection alarm is generated at active power reduction of the respective PCP below 3000 kW.

The 1, 2 RCPs trip alarms are generated by the first set, and the 3, 4 RCPs trip alarms, by the second one.

AA13 Function: Primary Coolant Loss

In case of LOCA it is necessary that reactor protection actuates, and core is made subcritical followed by cooldown.

Every protection set generates emergency alarms by the following parameters:

— The pressurizer level is below 4,0 m, AND primary coolant temperature is above 150 °C.

— Containment excess pressure is above 30 kPa.

— The difference between the primary circuit saturation temperature and temperature of any hot loop is below 8 °C.

— The reactor outlet excess pressure is below 15,1 MPa, AND reactor neutron power is above 75 % N_{nom}.

— The reactor outlet excess pressure is below 14,1 MPa, AND primary coolant temperature is above 260 °C, AND the cooldown switch is OFF.

At the power of above 5 % N_{nom}, the cooldown command is blocked.

AA14 Function: Loss of Feedwater/Steam Pipeline Rapture

Operational failures of secondary systems – steam supply system or feedwater system – violate heat removal from the primary coolant and thus reactor core cooling. In this connection, scram actuation is required.

Every protection set generates emergency alarms by the following parameters:

— The excess coolant pressure at the reactor outlet is above 17,5 MPa, AND primary coolant temperature is above 150 °C, AND the cooldown switch is OFF.

— The coolant temperature in any primary hot loops is above (T_{nom} + 8) °C.

— The pressurizer level in any SG is below (L_{nom} – 0,65) m at the primary coolant temperature of above 150 °C.

— The excess pressure of any steam pipeline is above 8,0 MPa at the primary coolant temperature of above 150 °C.

— The excess pressure of any steam pipeline is below 5,53 MPa, AND the difference between the primary circuit saturation temperature of the primary and secondary loops is above 70 °C, AND the temperature of any hot loop is above 150 °C.

— The pressurizer level is above 9,4 m, AND the cooldown switch is OFF.

AA15 Function: Primary-to-Secondary Leak

At the primary-to-secondary leak, activity goes out of the primary circuit. In this case, it is necessary to trip the reactor fast followed by RP cooldown and failed SG isolation (CD11 control and protection function).

In every set, a signal of reactor protection is generated when there is a discrete signal on activity higher than the allowed value in any out of four SG pipelines upon condition that the primary coolant temperature is above 150 °C.

AA16 Function: Seismicity

At the seismic impact that achieved the level of the design-basis earthquake, reactor scram shall be initiated.

To generate protection signals, seismic sensors are used that are a part of the industrial anti-seismic system.

The industrial anti-seismic system includes two independent sets, each of which has three seismic sensors and three commutation units, i.e. it has a three-channel structure.

The seismic sensors are located in periodically maintained process compartments in the reactor building at the level of the unit foundation plate, i.e. at elevation of –5,400.

Every set of reactor protection generates the reactor protection alarm when there are discrete alarms on the level of horizontal acceleration that exceeds the allowed value in two out of three sensors of the industrial anti-seismic system. The scram actuation setpoint for every channel is 0,67 m/s^2 that corresponds to the seismic impact of 7 points by the INES scale.

AA17 Function: Loss of 380 V Power Voltage

When the 380 V power is lost at both lead-in terminals of CRDMs, the reactor scram alarm is generated.

In every set, binary power loss alarms ($U < 0,8\ U_{nom}$) are generated by relay contacts and are used for scram at a time delay of 3 seconds.

In every set, an impulse signal is generated that ensures scram blockage for 10 minutes after actuation of any AA11-AA18 function.

Upon the fact of sound and light indication of scram alarms, the CR reactor

operator must do the following:

— repeat the scram actuation;

— by neutron flux control equipment indications, monitor reactor neutron power reduction and negative period availability;

— monitor all CRs position at lower abutment stops;

— monitor neutron flux control equipment range switching;

— monitor/ensure boron injection of 40 g/dm^3 to the reactor core, i.e. connect the primary makeup to the boron concentrate system, if impossible, inject boron solution from the JND emergency boron injection system;

— in case of scram actuation failure – in 4 sec after generation of the scram root cause alarm, neutron reactor power is above 15 % nominal – check actuation of the AA19 emergency boron injection upon the ATWS function.

AA18 Function: Violation of Power Distribution

Scram upon departure nucleate boiling ratio on the surfaces of FAs in the core ensures reactor trip to prevent violation of safe operation limits that characterize fuel and fuel cladding integrity and/or maximum design limit of fuel rod damage at initial events accompanied with the change of fuel rods power release, an increase in coolant enthalpy at the reactor inlet, reduction in PCP power supply frequency, and primary pressure reduction.

Scram upon an increase in linear power distribution ensures reactor trip to prevent violation of safe operation limits that characterize fuel and fuel cladding integrity in case of significant local inequalities in reactor core power distribution.

Algorithms of diverse protection systems

To reduce the consequences of a common cause failure due to the availability of the same tools, including software, the principle of diversity was applied in both AZ-USBI packages, and the subsystem of diverse (additional) protections in a hardwired implementation was introduced in SUZ-USBT. For more information about additional protections see section 2.4.2.

2.1.1.3 Safety function AB: Reactor subcriticality in a shutdown state.

Function AB11: Cutoff of the SG feedwater supply in case of emergency

This control function allows shutdown of the steam generator in case of the leak of the steam line or feedwater pipeline in the non-isolated section of the steam generator.

Implementation of this function allows timely stopping of uncontrollable cooling of the primary circuit through the emergency SG and introduction of positive

reactivity due to the feedback reactions.

The actuating signal for triggering AB11 function is formed when there are two factors:

— difference between the primary and secondary circuit saturation temperatures in the given steam generator more than 70 °C;

— excessive pressure in this steam generator less than 5,04 MPa;

— no inhibit signal from the task “Primary-to-secondary leak” in any steam generator;

— no inhibit signal “Temperature of primary coolant less than 150 °C”;

— no inhibit signal “Boiling margin in hot legs less than 8 °C”.

When there is an actuating signal, the executive part of the protection performs the following:

— switches off the RCP of the given leg;

— closes the valve on the main and auxiliary feedwater lines to the SG;

— closes the valve from the steam generator blowdown;

— loses the valve from the of SG emergency cooling system.

FSKU AB12: Inhibiting supply of pure condensate.

This control function prevents supply of pure condensate after the triggering of reactor protection system.

The control command is the signal of reactor protection system formed by an initial event leading to the scram in any set of the control system.

The function is implemented in TPTS by closing valves on the pipelines of pure condensate supply to the primary coolant makeup.

2.1.2 EMERGENCY BORON INJECTION SYSTEM JND

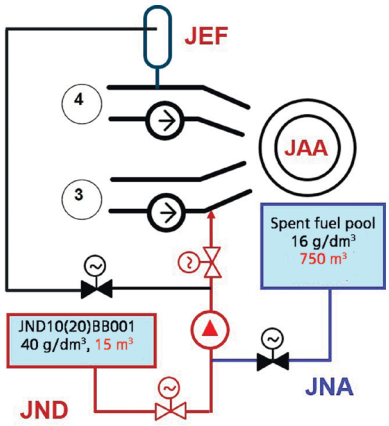
2.1.2.1 Purpose and layout of the system

Emergency boron injection system is designed to perform the following safety functions:

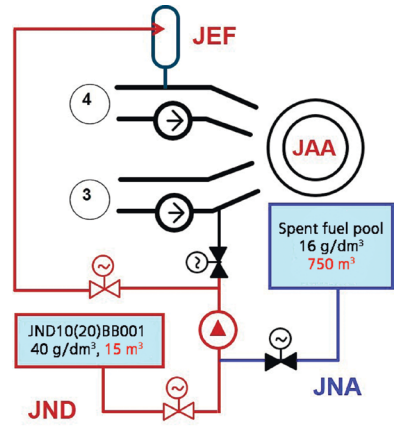
— supply of a high concentration of boric acid (40 g/dm³) to the primary circuit for quick transfer of the reactor installation to a subcritical state in the modes accompanied by a failure of the reactor emergency protection (ATWS) - safety function AA19 “Emergency injection of boron during ATWS” (Figure 2.1. 2.1);

— injection of boric acid solution into the steam part of the pressurizer in case of primary-to-secondary leak accidents, in order to reduce pressure in the primary circuit system - safety function CD12 "Control of injection into the pressurizer during primary-to-secondary leakage" (Figure 2.1 .2.1);

— switching to the supply of a boric acid solution with a concentration of 16 g/dm³ after the supply of a higher concentration solution is over (when the level in the tank JND10 (20) BB001 is less than L <0,15 m) - the safety function AB13 (Figure 2.1.2.2).



AA19. Emergency boron injection without scram (ATWS)



CD12.Control of injection into the pressurizer in case of primary-to-secondary leakage

Figure 2.1.2.1 – Modes of emergency boron injection JND in case of reactor protection failure and primary-to-secondary leakage

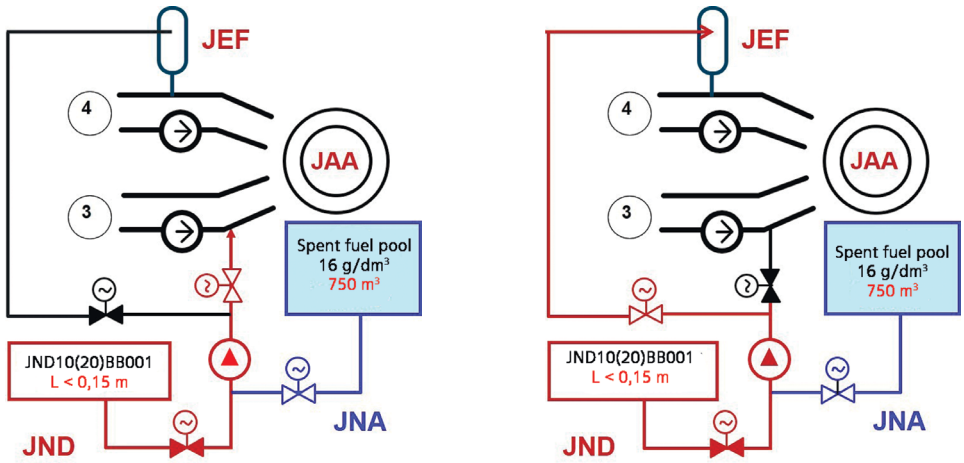


Figure 2.1.2.2 - Function AB13: switching pumps JND to supply boric acid solution with a concentration of 16 g/dm³

The emergency boron injection system performs its functions for any initial event of an accident requiring its operation, with the simultaneous failure, independent of the initial event, of any active or passive element having mechanical moving parts, or one personnel error independent of the initial event (the principle of a single failure).

The JND system doesn't have a failure which depends on the initial event. The system design incorporates a two-channel structure with redundancy of channels 2 x 100 %. Inside each channel there are two threads with an efficiency of 50 % (Figure 2.1.2.3). Thus, one available channel or two available threads in different channels ensure that the system fully performs its functions.

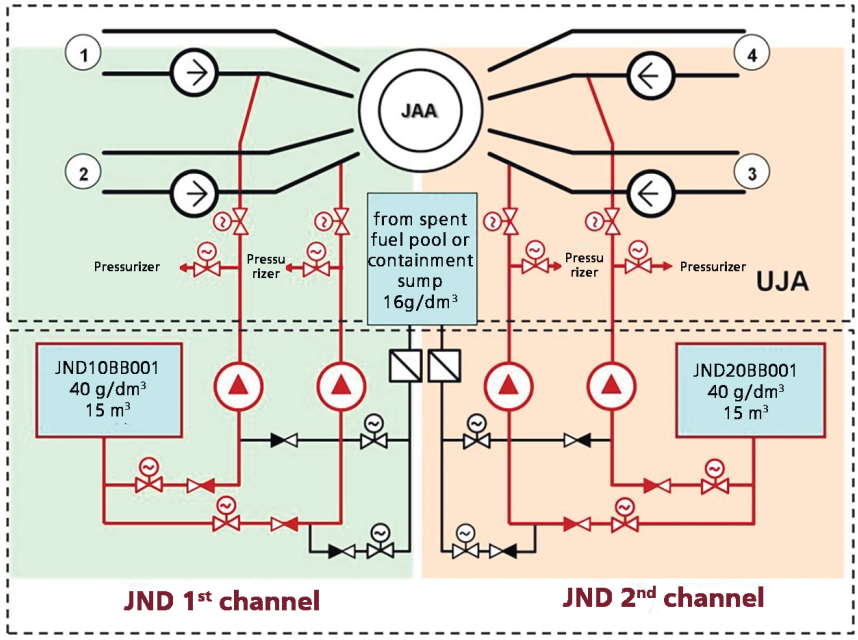


Figure 2.1.2.3 – Two-channel structure of emergency boron injection system

One channel of the system is allowed to be unavailable, due to repair or maintenance, for a period of not more than 72 hours. In this case, there should be an unplanned inspection on the remaining available active elements of this system and supporting systems. In this case, a failure independent of the initiating event is not taken into account in the safety analysis.

The system design allows monitoring and periodic testing of system elements in order to verify their operability, thus eliminating undetectable failures of elements which are not monitored during operation of nuclear power plants.

In the event of a failure of the elements reserved inside the channel, it is possible to isolate this element for repair, provided that the availability of the other backup part of the channel is confirmed. The permissible repair time, in this case, is 240 hours.

2.1.2.2 System flowchart.

Emergency boron injection system consists of two channels identical in the equipment and fully independent from one another. Each channel consists of two lines and common emergency tank of concentrated boron solution.

Each line consists of: a plunger pump, pipelines, valves and instrumentation. A simplified flow chart of the first channel of the emergency boron injection system is shown in Figure 2.1.2.4.

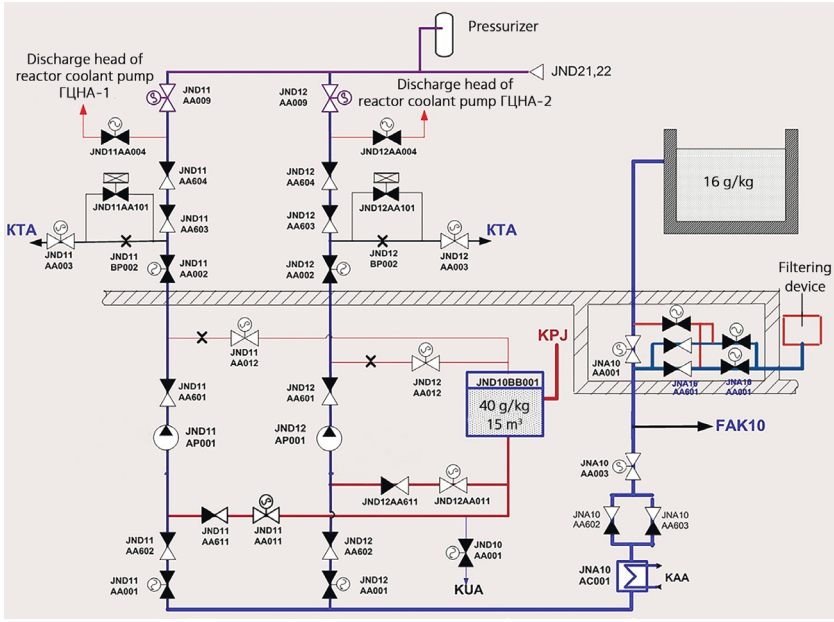


Figure 2.1.2.4 - simplified flow chart of the first channel of the emergency boron injection system

The suction pipelines of the emergency boron injection pumps are connected to the JND10 (20) BB001 tanks and to the pipelines of the corresponding channels of the JNA system (emergency and planned cooling of the primary circuit and of the spent fuel pool).

JND10 (20) BB001 tanks are for a stock of concentrated boron solution of 40-44,5 g/dm^3 for the JND system.

The stock of boron solution with a concentration of 16-20 g/dm^3 for the JNA and JND systems is the excessive (more than it's required for the spent fuel storage) volume of water in the spent fuel pool (not less than 750 m^3).

To prevent the overflow of JNA systems into JND and backflow, there are check valves on the inlet pipelines to the pumps.

The tanks JND10,20BB001 are sufficient to transfer the reactor installation to a subcritical state at any time of the fuel cycle (for all possible core subcriticality values). Further maintaining the core subcritical is achieved by introducing a solution of boric acid from the spent fuel pool.

First JND10 (20) BB001 tanks are filled from the KPJ chemicals preparation and supply system.

To protect the pumps and pressure pipelines from pressure exceeding the working pressure by 15 %, two safety valves with 100 % capacity are installed on the output pipelines of each pump (not shown in the diagram). Medium from safety valves is discharged into tanks JND10 (20) BB001.

To avoid the operation of the pumps in a nonflow mode and to periodically check their operability during operation, there is a recirculation pipeline for each pump with an expansion device and shut-off valves which are open when the system is in the “standby” mode. Recirculation piping are connected to tanks JND10 (20) BB001.

The pipelines of each channel of the JND system are connected to the cold pipelines of the main circulation loops and to the steam section of the pressurizer. At the same time, each line of the system supplies boron solution into its loop of MCC: line 11JND11 into loop No. 1, line 11JND12 into loop № 2, line 12JND21 into loop № 3, line 12JND22 into loop № 4.

The pipelines of the system are isolated from the main circulation circuit by two successive check valves and one stop valve installed in series. Between the stop valves and the first upstream check valve, a drain is installed in the controlled leakage system.

Shutoff valves are installed on the corresponding pipelines to separate the flows in the MCC and the pressurizer.

Based on a probabilistic safety analysis, an accident with a primary-to-secondary leakage circuit is a more likely initiating event than an ATWS accident. Therefore, when the system is in the “standby” state:

— valves on the supply line to the pressurizer must be open,

— valves on the supply line to the MCC must be closed.

2.1.2.3 Short description of the equipment

Pumps and tanks are located in UKA building at “minus 5,4 meters”.

The pump of JND system consists of a high-pressure horizontal five-plunger single-acting pump with a crank mechanism (60 mm stroke), a damper, a coupling and an asynchronous electric motor (Figure 2.1.2.5).

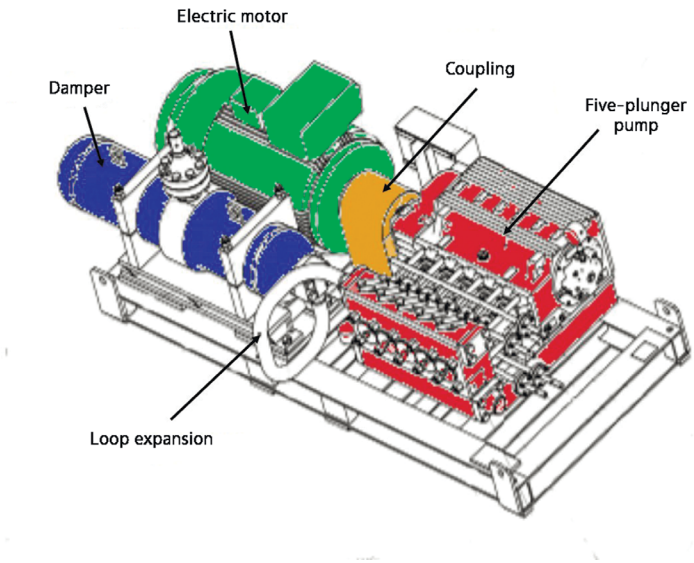


Figure 2.1.2.5 – General layout of the pump of the emergency boron injection system.

Due to the design of the pump there can be no resonance phenomena and pulsation is limited during its operation. There is a damper at the pump outlet that is used for reducing pressure pulsations. Moreover, the pulsation value is ± 1,5%.

To compensate the temperature expansion of the metal between the pressure flange of the pump and the damper, a compensation pipeline (compensation loop) is used.

ATB SEVER d.o.o. asynchronous electric motor, Serbia, type 23.ZKIT315MD-4 with air cooling, is used as a pump drive. The electric motor is powered by the emergency supply of the second reliability group.

The main characteristics of the pump are given in table 2.1.2.1.

Table 2.1.2.1 - Main characteristics of the JND system pump

Characteristics	Value
Nominal flow, m^3/h	14,5
Nominal pressure at the outlet, MPa	19,6
High-side design pressure, MPa	24,5
Minimal pressure at the inlet during operation, MPa	0,016
Operating temperature, °C	Not higher 90
Power input with the pressure at the pump outlet 19,6 MPa, kW	98
Nominal motor capacity, kW	200
Nominal voltage, V	380
Nominal rotations, rot/min	1486

The emergency tank of concentrated boron solution JND10 (20) BB001 is a vertical cylindrical vessel under atmospheric pressure. The tank capacity is 15 m^3, the diameter is 2 meters, the total height with the filler neck = 6,15 m. The nominal water level in the tank is 4,6 m.

2.1.2.4 Monitoring of parameters of JND emergency boron injection system

The operator monitors the parameters of the emergency boron injection system with the help of UULS snapshots (Figure 2.1.2.6) and indicating devices on the safety control panels of the MCR / ECR (Figure 2.1.2.7)

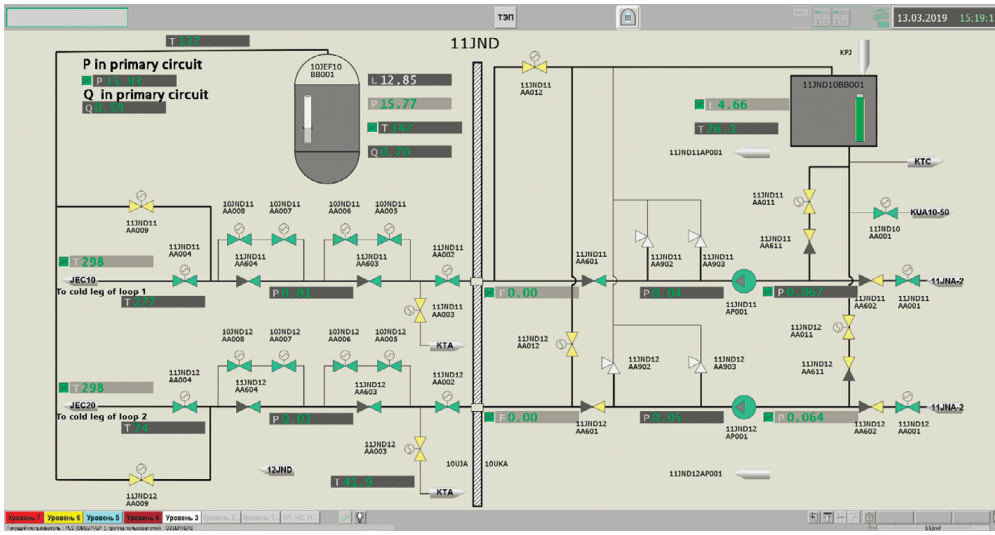
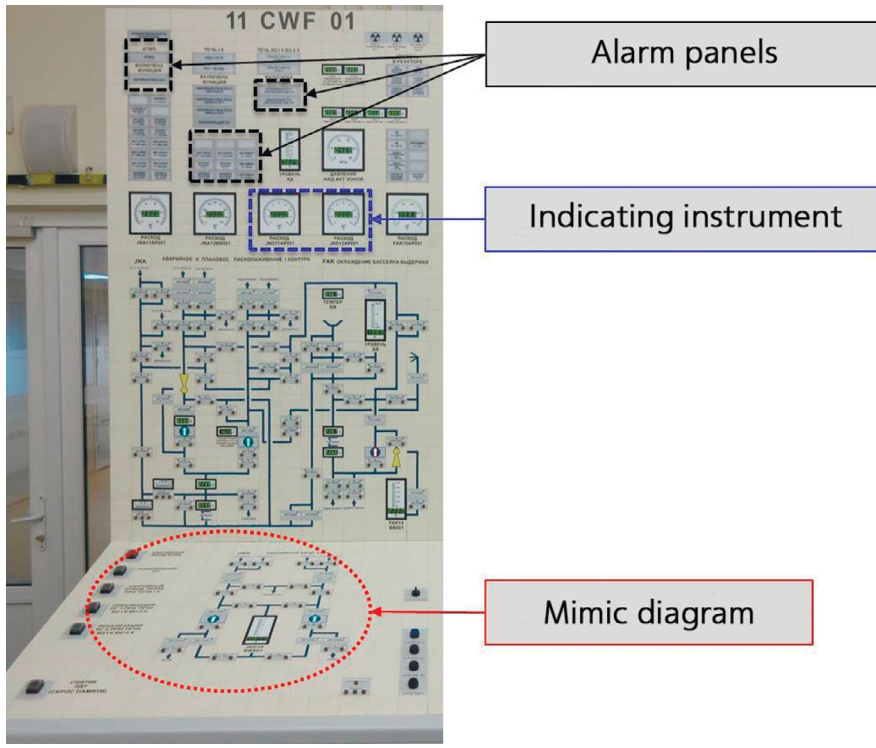


Figure 2.1.2.6 – Snapshot of the first train of emergency boron injection system



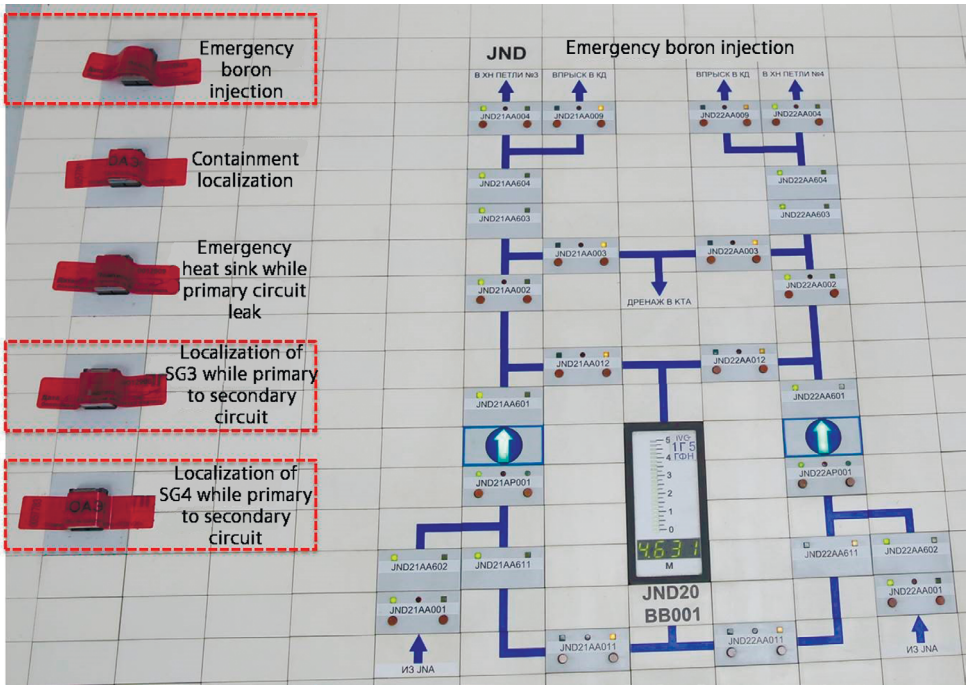


Figure 2.1.2.7 - Emergency boron injection system on MCR safety panels

Monitoring is carried out for the valves, the level and temperature of boron concentrate in the tanks, the pressure at the suction and discharge of the pumps, the flow rate through each pump, the surface temperature of the pipeline upstream of the MCC loop.

Leaks from the primary circuit with the leaking valves are monitored by the pressure between the check valves and by the temperature in the control drainage line.

The operational limits of the JND system are:

— The minimum flow rate of the emergency boron injection pump is 14,3 m^3/h;

— the minimum level in the tank is 0,15 m;

- the minimum temperature of the medium in the tanks is 20 ° C;

— minimum pressure at the inlet to the pumps 0,015 MPa;

— maximum pressure at the inlet to the pumps 2,1 MPa

There is a unique UULS snapshot for each pump of JND system (figure 2.1.2.8).

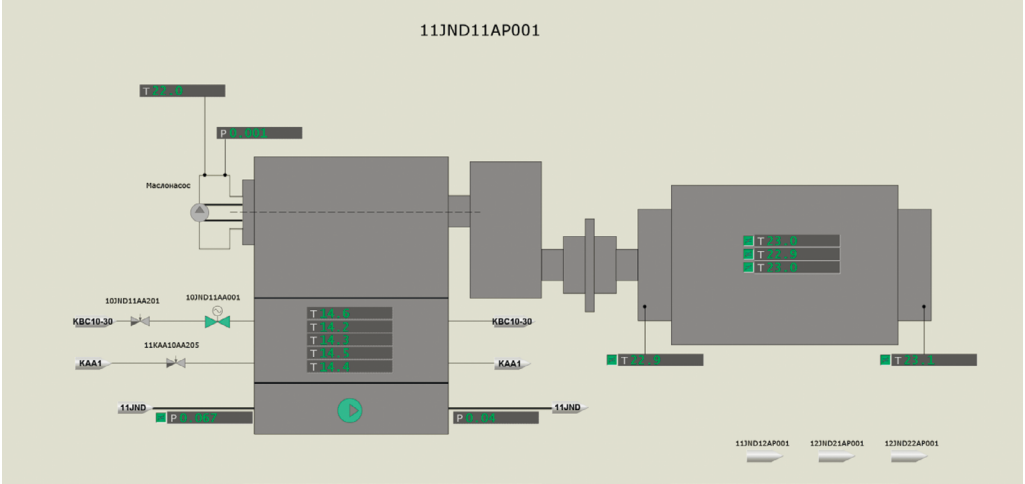


Figure 2.1.2.8 – Snapshot of 11JND11AP001 pump

Monitoring parameters:

— pressure at the inlet and outlet of the pump;

— temperature of the stator windings of the electric motor (I, II, III phases);

— temperature of the front and rear bearings of the engine;

— oil temperature;

— pressure on the oil pump discharge;

— temperature of the oil seals of the pump.

It is forbidden to operate pump units in case of exceeding emergency setpoints for pump parameters:

— oil temperature in the mechanical part: more than 80 °C;

— temperature of seals: more than 70 °C;

— oil pressure: less than 0,3 MPa;

— temperature of the electric motor bearings: more than 90 °C;

— temperature in the stator windings: more than 150 °C;

— leaks through the seal (one) during pump operation: more than 2 l/h

— leaks through the seal (one) when the pump is in standby mode: more than 0,15 l/h;

— Average mean square (AMS) vibration speed of more than 7,0 mm/s.

2.1.2.5 Modes of operation of emergency boron injection system

In normal operation modes, the system does not function and is in a standby mode. Periodic testing is carried out.

JND system operation in case of a power loss.

The system has reliable power supply from diesel generators of the emergency power supply system of consumers of the second group. On a signal of a power loss, the switches of the pumps of the system do not switch off. When there is voltage from diesel generators on sections of the emergency power supply (zero stage of the step-by-step load program):

— JND pumps operating at the time of a power loss — resume operation;

— JND pumps that were not operating at the time of a power loss — do not come into operation, but the operator can put them into operation, if necessary.

JND system operation in case of primary-to-secondary leakage.

In emergency conditions associated with a leak from the primary circuit to the secondary, in order to reduce the pressure of the primary circuit, the concentrated boron solution is injected into the pressurizer steam bubble by pumps of the JND system - function CD12.

The CD12 function is actuated when the pressure in the primary circuit is above 8,4 MPa and the level in the pressurizer is below 7,97 m. Two signals from the function CD11 “Localization of SG during the primary-to-secondary leak” (Figure 2.1.2.9).

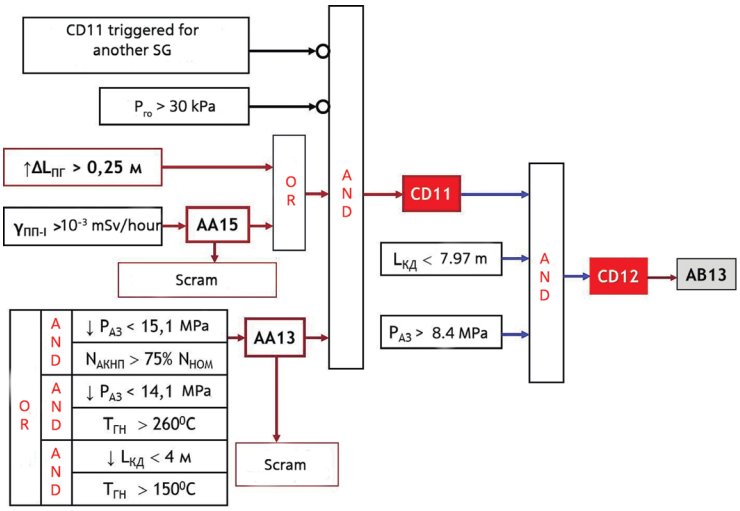


Figure 2.1.2.9 – Actuating signals of CD11 and CD12 functions

One of the CD11 signals provides control of pumps and valves and prohibits remote control from individual controls for 30 minutes. After 30 minutes, the operator can reset this command.

Another signal prohibits the commands of normal operation. The operator can reset this command only when the temperature of the primary coolant is less than 70 °C.

Upon activation of the CD12 function, the operator must check (Figure 2.1.2.10):

— pumps of system of emergency input of boron are in operation;

— opening of valves on the line of water supply to the pressurizer;

— closing of valves on the lines of recirculation of pumps, control drains and water supply to the cold legs of primary circuit loops;

— flow rate through the pump - not less than 14,5 m^3/h;

— decrease of the level in JND system tanks

— “SF completed” signal triggering for FS12 CD12 on the UULS snapshot

— parameters of operating pumps JND (according to UULS and on the spot)

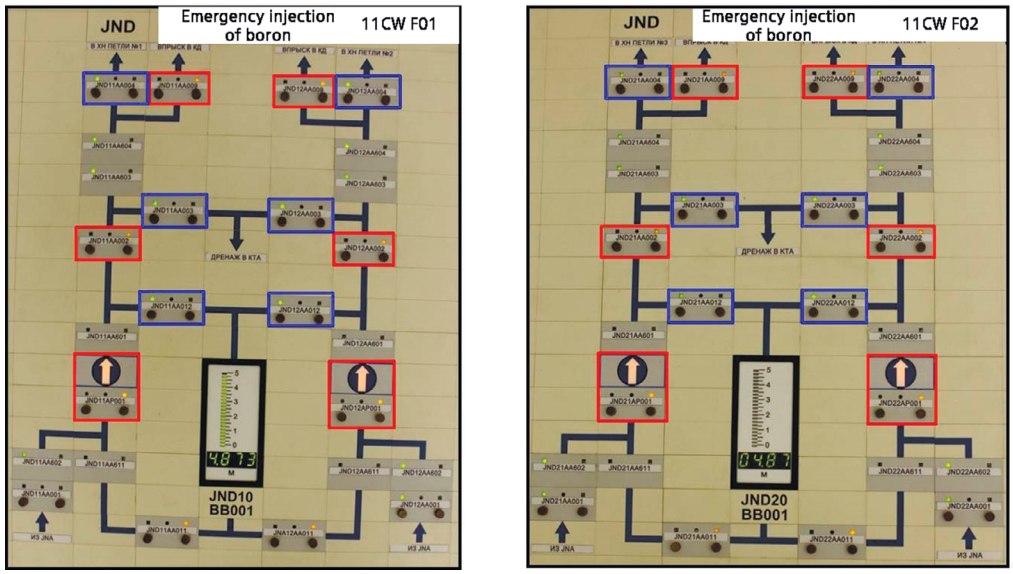


Figure 2.1.2.10 - Control of actuation of CD12 by mimic diagram of JND system

With a decrease in pressure above the core less than 8,2 MPa or an increase in the pressure level over 8,37 m, the operator must control the shutdown of the JND11,12AP001, JND21,22AP001 pumps. The position of the valve remains unchanged.

If the a level in the tank of less than 0,15 m, a signal is generated to close the

valves from the emergency tank of boron concentrate and open the valves from the pipeline of the emergency and planned cooldown system of the primary circuit (function AB13).

JND system operation in case of a reactor trip and feed-and-bleed system failure

After the reactor protection is actuated, it is necessary to increase the concentration of boric acid in the primary coolant using feed-and-bleed system and boron concentrate systems.

When closing localizing valves or in case of a failure of the feed-and-bleed system, emergency procedures require injection of at least 30 m^3 of boron concentrate 40 g/dm^3 from the JND system into the primary circuit.

To do this, the operator must push the button “Emergency injection of boron” and check (Figure 2.1.2.11):

— switching on the pumps of emergency injection of boron;

— opening of the valves on the lines of boron concentrate supply from tanks to cold loops of the MCC;

— closing of valves on the control drainage line and on the line of water supply to the pressurizer;

— closed state of valves on the pump recirculation line;

— flow rate through the pump - not less than 14,5 m^3/h;

— decrease of the level in the tanks of the JND system;

— parameters of the pumps in operation.

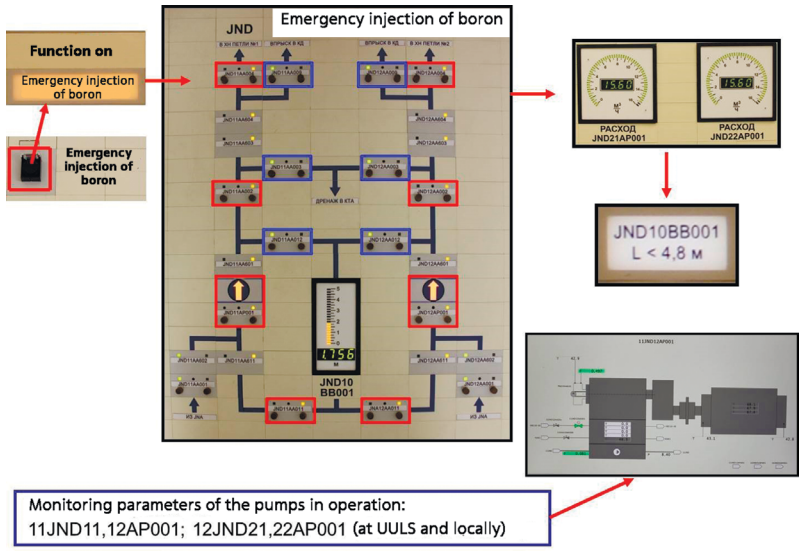


Figure 2.1.2.11 – Actions of the MCP operator when remotely putting JND into operation for the primary circuit

JND system operation in case of BDBA of ATWS type

In case of beyond design basis accidents of the ATWS type (failure of reactor protection in emergency modes), the JND system performs the function of supplying boron solution to the primary circuit to transfer the reactor installation to a subcritical state.

The initial quick shutdown of the reactor (hot shutdown) is achieved by introducing a concentrated boron solution with a concentration of 40-44,5 g/dm^3 (FMCS AA19), and a long cold shutdown is achieved by introducing an additional supply of boron solution with a concentration of 16-20 g/dm^3 from the spent fuel pool (FMCS AV13).

The criteria for fulfilling the AA safety function: “Emergency termination of the fission reaction” are:

— supply of a concentrated solution of boric acid to the reactor - not less than 30 m^3/h.

— transport delivery time after the actuation signal is generated - no more than 100 seconds;

— temperature of the supplied solution is more than 20 °C;

— prohibition of the normal operation commands and remote operator commands for 10 minutes.

The monitoring and control function (FMCS AA19) is actuated when the following signals are generated and match:

— any signal requiring emergency protection of the reactor with a time delay of 4 s;

— neutron power of the reactor is more than 15% of the nominal neutron power.

When AA19 function “Emergency boron injection during ATWS” is actuated, the operator must check (Figure 2.1.2.12):

— switching on JND pumps;

— opening of valves on the lines of boron concentrate supply from the tanks to the MCC cold loops;

— closing of valves on the control drainage line and on the line of water supply to the pressurizer;

— closed state of valves on the pump recirculation line;;

— flow rate through the pump – not less than 14,5 m^3/h;

— decrease of the level in JND system tanks;

— parameters of JND pumps in operation (by UULS and on the spot).

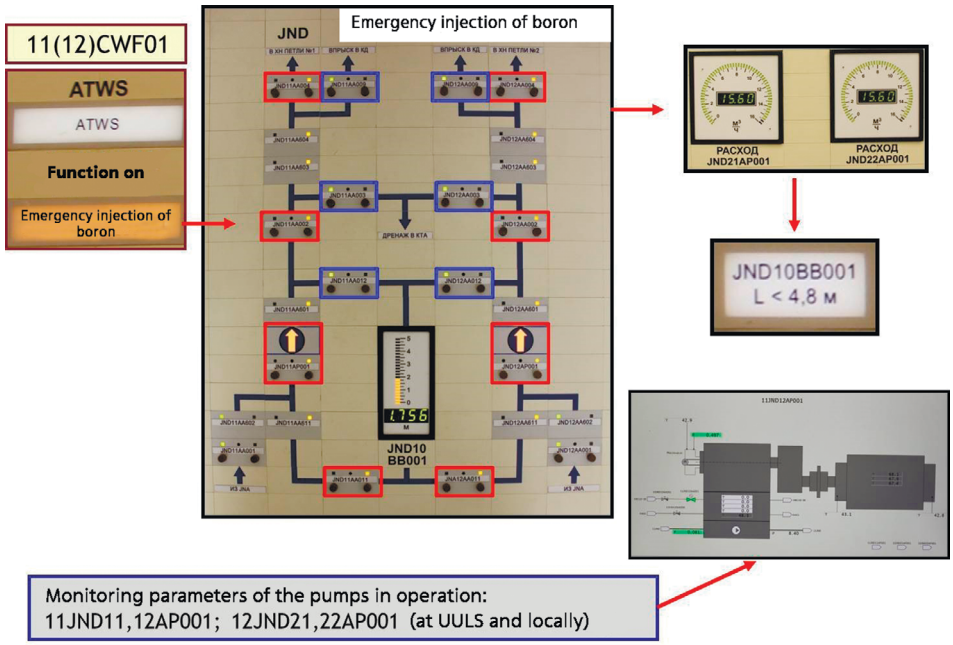


Figure 2.1.2.12 – Actions of MCR operator during operation of FMCS AA19

When the level in the tank is less than 0,15 m, a signal is generated to close the valve from the tank and open the valve from the pipeline of the emergency and planned cooling of the primary circuit (function AB13).

The AA19 function has the priority over the CD12 function: during the initiation of both functions, JND pumps will supply water to the MCC, and not to the pressurizer. When the level in the tank is less than 0,15 m, a signal is generated to close the valve from the tank and open the valve from the pipeline of the emergency and planned cooling of the primary circuit (function AB13).

The AA19 function has the priority over the CD12 function: during the initiation of both functions, JND pumps will supply water to the MCC, and not to the pressurizer.

JND system operation during beyond design basis accidents with loss of all AC sources and simultaneous leakage from primary circuit.

Within 24 hours from the accident start the personnel must provide:

— switching of an alternative diesel generator and restoration of power supply to pumps and valves of the first train of the emergency boron injection system;

— introduction of alternative intermediate circuit with a cooling tower;

— supply of intermediate circuit water to the heat exchangers of JNA system (first

train) and to the cooling of 11JND11AP001 and 11JND12AP001 pump seals (first train);

— supply of water to the cold legs of MCC with the flow rate of 2 x 14,5 m^3/h from two emergency boron injection pumps (first train) from the sump with the temperature below 70 °C.

In addition, in this accident, personnel provide water supply from one pump of SFP cooling (1 train) to the spent fuel pool and reactor with flow rates of 50 m^3/h and 270 m^3/h, respectively.

These actions provide reliable cooling of the reactor and spent fuel pool for an unlimited time.

2.1.2.6 Main design solutions for safety function “Reactivity control”

Table 2.1.2.2 shows the main design solutions for the implementation of the safety function “Termination of the fission reaction” and “Subcriticality in a shutdown mode”.

Table 2.1.2.2 - Safety function “Reactivity control” in NVAES-2 design

Design solution	Action	Actuation	Mode
The use of uranium-gadolinium fuel elements	Self-limitation of core energy release due to negative values of reactivity coefficients for specific coolant volume and fuel temperature, reactor power, total reactivity coefficient for fuel temperature and coolant temperature in all critical states that are possible over the entire range of reactor parameters during normal operation and in case of violations from normal operation, including design basis accidents	—	NO NOF DBA BDBA

Design solution	Action	Actuation	Mode
Reactor control and protection system: — 121 control rods in 12 groups — Limitation of maximum effectiveness of a single rod — Limitation of the rate of input of positive reactivity when extracting the group of rods (less than 0,07 β_{eff} /s) — Re-criticality temperature of less than 100 °C with one of the most effective rod stuck in the highest position	According to the signal from the reactor protection system (AZ-USBI), electromagnets are de-energized and control rods fall into the core under their own weight	AA11 AA12 AA13 AA14 AA15 AA16 AA17 AA18	DBA
	During a safety trip, by a signal from a diverse protection system, the electromagnets are de-energized and the control rods fall into the core under their own weight.	Diverse protection system	BDBA (failure of EPS)
Emergency boron injection system (JND): — Two-channel structure with back of channels 2 x 100 % and backup inside the channels 2 x 50 %	In case of a scram failure (ATWS) by the signal from USBT the system supplies concentrated boric acid solution (40 g/dm^3) into the core.	AA19	BDBA (APCS failure)
	In case of failure of the KVA / KVS40-60 system or when closing the localizing valves, the operator introduces at least 30 m^3 of boron concentrate 40 g / dm^3 from the JND system into the primary circuit	operator	DBA
	Transfer to the supply of boron solution 16 g/dm^3 from SFP when the level in the system tank is below 0,15 m.	AB13	DBA BDBA
The feed-and-bleed system (KVA) of the primary circuit and boron concentrate system (KVS40-60)	During safety trip, by the signal from operational I&C, the systems supply concentrated boric acid (40g/dm^3) into the primary circuit.	Oper. I&C	DBA
The feed-and-bleed system	Closing with the block of opening valves on the pipelines of pure condensate supply into the primary makeup system after the scram	AB11	DBA

Design solution	Action	Actuation	Mode
Limitation of positive reactivity during uncontrolled cooling of the reactor installation through the secondary circuit	In case of leakage of the steam or feedwater pipelines in the non-isolated section of the steam generator, according to the signal from the USBT, emergency steam generator is localized from the feed water for its evaporation and termination of uncontrolled cooling of the reactor	AB12	DBA
Use of borated water (16 g/dm^3) in SFP and hydraulic tanks of 1st and 2nd stage to compensate leakages of the primary circuit	By the signal of primary leakages (LOCA-type accidents) JNA pumps supply borated water (16 g/dm^3) into the primary circuit. If hydraulic tanks of 1 or 2 stages are passively activated, borated water with a concentration of 16 g/dm^3 is also supplied to the reactor	—	DBA BDBA

2.1.3 PRIMARY CIRCUIT AND FUEL COOLING POOL EMERGENCY AND MAINTENANCE COOLDOWN SYSTEM (JNA)

2.1.3.1 Application and design of the primary circuit and fuel cooling pool maintenance cooldown system

The primary circuit and fuel cooling pool emergency and maintenance cooldown system (JNA) is designed for:

— reactor plant cooldown to 70 °C after reactor shutdown, when heat removal through steam generators becomes ineffective ($P_{primary\ circuit}$ < 2,1 MPa, T < 150 °C) in all unit opration modes (maintenance, repair, emergency cooldown);

— residual heat removal from the fuel in the reactor to the essential consumers intermediate circuit system (KAA) in all unit operation modes, requiring the operation of the JNA system;

— residual heat removal from the fuel in the cooling pool in all unit operation modes (as a reserve for FAK10-20);

— maintaining coolant reserve in the reactor during a "large leak" event;

— primary circuit emergency makeup during a "small leak" event.

You can see the composition of the primary circuit and fuel cooling pool emergency and maintenance cooldown system in Figure 2.1.3.1.

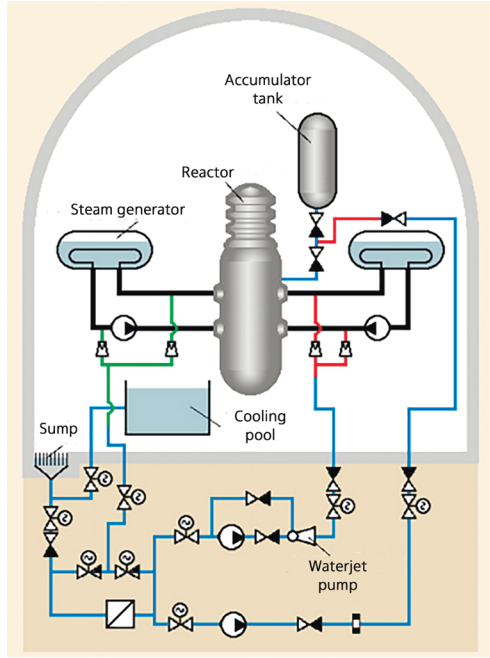


Figure 2.1.3.1 – Primary circuit and fuel cooling pool emergency and maintenance cooldown system

The primary circuit and fuel cooling pool emergency and maintenance cooldown system (JNA) consists of two fully independent channels with identical equipment.

Each channel includes:

— emergency and maintenance cooldown heat exchanger JNA10(20);

— emergency cooldown pumps JNA11,21(12,22);

— main waterjet pump JNA12(22)BN;

— filtering device JNA01AT001(002);

— check valves, safety valves, isolation and control valves;

— pipelines.

Pressure pipelines for pump units JNA11,21 consist of two legs DN 300. One leg connects check valves in the first stage hydro-accumulators' system drain line, which supplies boric acid solution to the reactor collection chamber. Another leg connects check valves in the first stage hydro-accumulators' system drain line, which supplies boric acid solution to the reactor pressure chamber.

In the primary circuit maintenance cooldown mode, JNA11 pump draws coolant from the **“hot” leg of 2nd MCC loop** to the reactor pressure chamber under the reactor core. In the primary circuit repair cooldown mode JNA11 pumps draw coolant from the “cold” leg of 2nd MCC loop to the reactor collection chamber above the reactor core. Suction pipelines are connected to “hot” and “cold” legs of 2nd MCC loop via leak limiters.

In the primary circuit maintenance cooldown mode, JNA21 pump draws coolant from the **“hot” leg of 4th MCC loop** to the reactor pressure chamber under the reactor core. In the primary circuit repair cooldown mode JNA21 pumps draw coolant from the “cold” leg of 4th MCC loop to the reactor collection chamber above the reactor core. Suction pipelines are connected to “hot” and “cold” legs of 4th MCC loop via leak limiters.

Pressure pipelines for pump unit JNA12 consist of two legs DN 300, and are connected to the "cold" and **"hot" leg of the 1st MCC loop** via leak limiters.

Pressure pipelines for pump unit JNA22 consist of two legs DN 300, and are connected to the "cold" and **"hot" leg of the 3rd MCC loop** via leak limiters.

Any of the pump units of the first and second channels of the JNA system can operate using cooling pool circuit.

The JNA system maintenance cooldown pipelines, connected to the "cold" and "hot" legs of the main circulation circuit, can be disconnected from it with two sequentially located isolation valves with drainage between them.

In order to protect the primary circuit, as well as the suction pipelines and equipment of the JNA system from overpressure during cooldown, the shared maintenance cooldown collector has two safety valves. The capacity of each of the safety valves allows operation of two feed-and-bleed pumps, as well as high pressure emergency and maintenance cooldown pump with maximum total capacity no less than 400 m^3/h, opening pressure of 2,2 MPa.

Isolation valves and check valves are installed on the pressure line of pumps JNA11,21 (12,22). During the unit power operation, the pressure pipeline is disconnected from the primary circuit pipelines by shut-off valves and two serially installed check valves. Leakage removal from check valves is provided by drainage between the shut-off valves and the check valve. To ensure the required flow rate in the maintenance reactor shutdown cooling mode, there are orifice gages at the discharge pipeline of pumps JNA11,21.

The excessive volume of spent fuel pool equal to 750 m^3 above the normal level required for spent fuel, serves as a storage of the emergency reserve of boric solution with concentration from 16 to 20 g/dm^3. In case of emergency cooldown with primary circuit decompression, water is initially supplied to the reactor from the fuel pool, and then from the containment sumps.

Pipelines downstream of heat exchangers JNA10,20 serve as collectors combining suction pipelines of primary circuit emergency and maintenance cooldown pumps JNA11 (12), JNA21 (22) and emergency boron injection pumps JND11,21 (12,22). Pump recirculation lines are installed in the pipeline upstream of the heat

exchanger. There are orifice gages installed to ensure the operation of pump units JNA11 (12), JNA21 (22) in the operating pressure range during operation along the recirculation line.

Heat removal from emergency and maintenance cooldown heat exchangers is provided by the essential consumers intermediate circuit system (KAA). Heat from the intermediate circuit is removed by the essential cooling water system (PE).

Heat removal from the reactor plant in emergency modes is carried out in the following order: fuel cooling pool emergency and maintenance cooldown system (JNA) → essential consumers intermediate circuit system (KAA) → essential cooling water system (PE) → spray cooling pool.

When the unit is de-energized, power for each train is supplied from the diesel generator plant of the corresponding second group loads emergency power supply system train.

Technical characteristics of the JNA system equipment are given in Table 2.1.3.1.

Table 2.1.3.1 – Technical characteristics of JNA system equipment

Name	Quantity	Main technical characteristics of equipment
Primary circuit and fuel cooling pool emergency and maintenance cooldown heat exchanger	2	Design pressure on cooling side (gauge) – 1,0 MPa. Design pressure on the cooled side (gauge) – 2,1 MPa. Flow rate on the cooling side – 2500 m^3/h. Flow rate on the cooled side – up to 1800 m^3/h. Medium on the cooling side – demineralized water in the intermediate circuit KAA. Medium on the cooled side – 1st circuit coolant.
Emergency cooldown pump (HP)	2	Type – ACN-230-65A. Flow rate in the operating point – 230 m^3/h. Head in the working point – 740 m. Inlet pressure – no more than 2,1 MPa. The temperature of the borated water to be pumped – from 13 °C to 130°C.
Main water jet pump	2	Supply – from 230 m^3/h to 900 m^3/h.
Emergency cooldown pump (LP)	2	Type – ACN-800-15A. Flow rate in the operating point – 800 m^3/h. Head in the working point – 160 m. Inlet pressure – no more than 2,16 MPa Inlet temperature – not more than 50 °C

The JNA system combines the functions of the protective, localizing safety system, as well as the normal operation system; in accordance with the design, all its elements are of the second safety class.

Heat exchangers JNA10,20 are modular plate structures, where heat is transferred from one medium to another through thin metal plates pressed into a special model (Figure 2.1.3.2a-b).

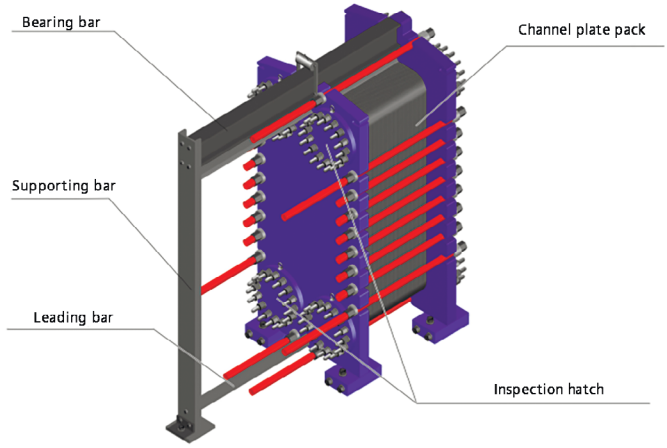


Figure 2.1.3.2a – General view of the emergency and maintenance cooldown heat exchanger, with the main elements indicated.

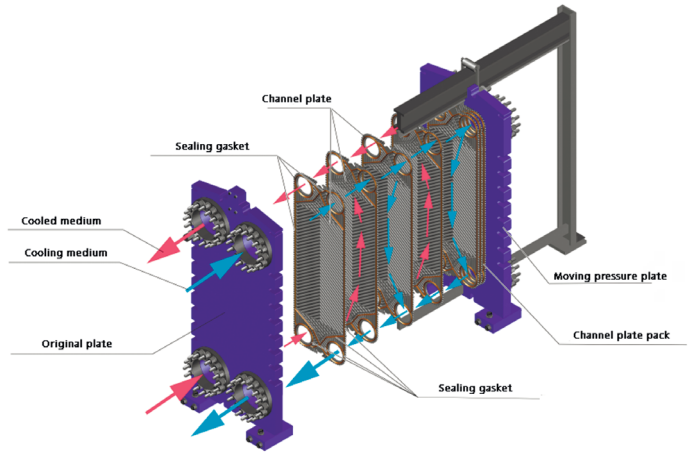


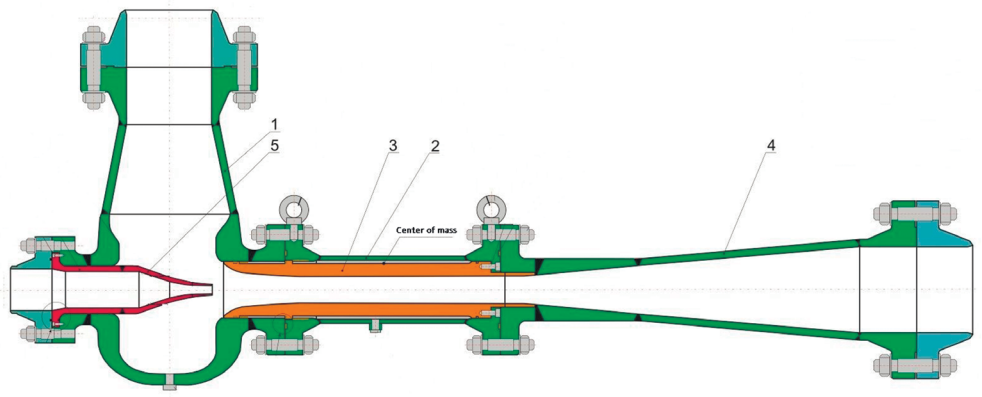
Figure 2.1.3.2b – Coolant movement diagram and design of emergency and maintenance cooldown heat exchanger

Heat from the primary circuit coolant (spent fuel pool water) is transferred through the heat exchanger channel plates to the intermediate circuit medium. Media movement direction is arranged in opposite directions. Two media are separated by ring and common seals. To prevent media mixing in the corner parts of the seal, where the common and annular gaskets are located next to each other, the connections have grooves connected to the atmosphere. Medium leakage through any gasket thus occurs through these grooves.

The "ejector pump" serves as the high pressure pump unit in the system. That unit is a combination of high pressure pump 11JNA12AP001 (12JNA22AP001) of ACN 230-65A type and water jet pump (ejector) installed on its pressure pipeline.

Working fluid flow is generated by high-pressure centrifugal pump. The working flow exits the nozzle into the receiving chamber of the jet blower and entrains the "ejectable" medium which has a lower pressure. The entrained flow is called "injected". The jet blower converts the potential energy of the working flow into kinetic energy. The kinetic energy of the working flow is partially transferred to the ejected flow. When flowing through the jet blower, the velocities of the mixed flows are equalized and kinetic energy of the mixed stream is converted back into potential.

The design of the water jet pump is shown in Figure 2.1.3.3.



1 — inlet chamber
2 — nozzle pipe
3 — neck seal ring
4 — diffuser
5 — nozzle

Figure 2.1.3.3 – Water jet pump design

The working nozzle is used to convert potential pressure energy into kinetic energy of the working flow and create a certain swirl in the jet. The inlet chamber

serves to supply the ejected flow to the working medium jet. Streams of working and ejectable media enter the mixing chamber, where velocity is equalized, followed by pressure increase. From the mixing chamber the flow enters the diffuser, where further pressure increase occurs. The pressure of the mixed flow at the outlet of the diffuser is higher than the pressure of the mixed flow entering the inlet chamber.

Flow and pressure characteristics of water jet pump JNA12 are shown in Figure 2.1.3.4.

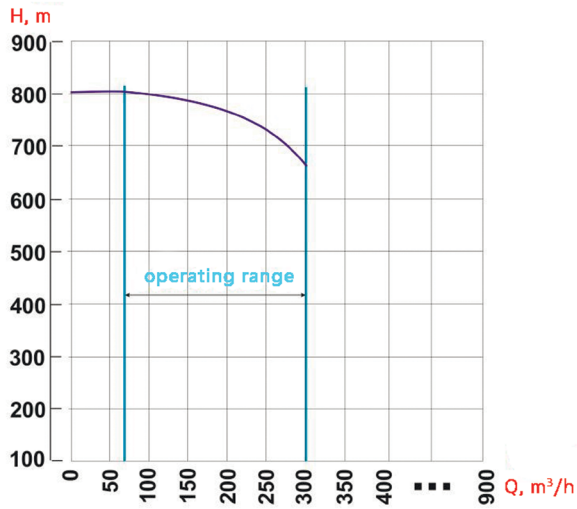


Figure 2.1.3.4–Flow(Q) and pressure(H,meter of water) characteristics of water jet pump JNA12.

In case of back pressure from 1,6 to 8 MPa, the ejector pump operates as a resistance on the pump head JNA12,22. When the back pressure is not less than 1,5 MPa, the injected medium is sucked when operating from the fuel pool, when the back pressure is not less than 1,35 MPa – when operating from the sump.

Technical characteristics of the ejector pump equipment are given in Table 2.1.3.2.

Table 2.1.3.2 – Technical characteristics of ejector pump equipment

Parameter	Value
Pressure at ejector pump inlet	up to 10,1 MPa
Pressure at ejectable medium branch inlet	up to 2,1 MPa
Pumped medium temperature	from 14 °C to 100 °C

The ACN 230-65A pump is of centrifugal, horizontal, multi-stage double-body type and is equipped with end seals. The electric motor is connected to the pump with a plate coupling.

The ACN 230-65A pump has a cartridge structure (Figures 2.1.3.5a-b) consisting of an outer housing, into which a recessed part is inserted, formed from the housing of the inner housing and adjoined seal casings, as well as brackets with rolling bearings. Seals on the connectors of the outer housing and recessed part are rings made of heat-resistant rubber. The auger is designed to provide cavitation margin. The pump piston is designed to compensate the axial force acting on the thrust bearing. The throttle device is designed to provide the required pressure behind the piston in the unloading chamber.

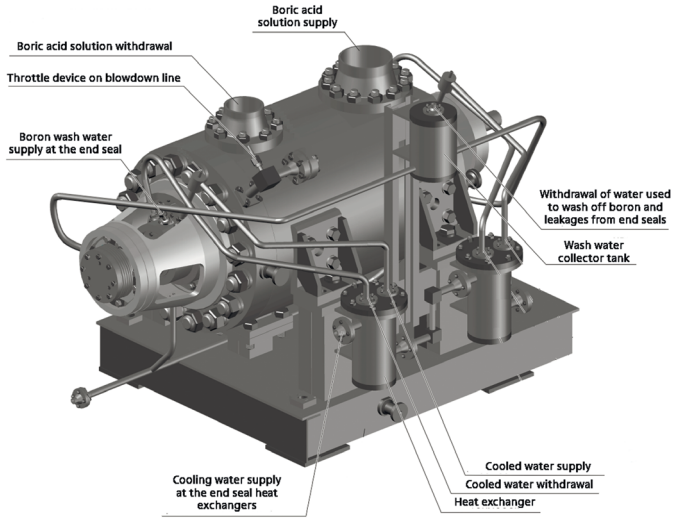


Figure 2.1.3.5a – ACN 230-65A pump

Remote heat exchangers are designed to cool the pumped medium that ensures the operation of end seals. The volume of the tank on washing water discharge is designed based on the calculation to provide an eight-hour cycle of gland seal operation in a wetted state, but is not less than 3,7 litres.

Cylindrical roller bearing on the suction side takes radial load.

Spherical ball bearing with four-point contact on the head side takes radial load and residual axial force from hydraulic force acting on the flow part of the pump.

End seals are of end type with friction pair of silicated graphite SG P 0,5.

The inner housing comprises a set of sections connected to each other by studs, as well as rotor, pressure cover and supply chamber. The sections are equipped with guide devices locked by special stoppers. The joints of the sections are compacted due to metal contact, and have heat-resistant rubber rings additionally installed.

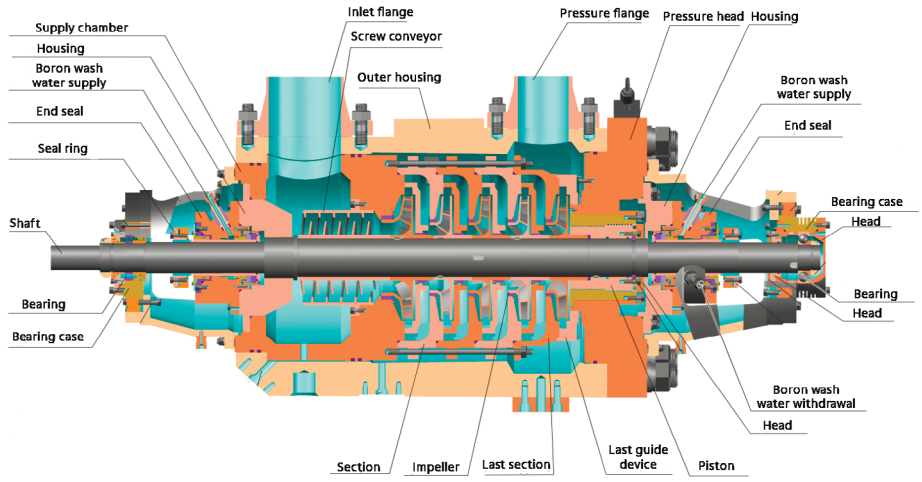


Figure 2.1.3.5b – ACN 230-65A pump structure

Impeller front slotted seal rings are pressed into section disks, while impeller rear seal rings are pressed into guide vane disks.

Impellers are installed on the forged shaft. The first stage impeller has an extended inlet. All other wheels have the same flow part.

There is a fixing pin and a guide key in the lower part of the outer casing to ensure the directed thermal expansion of the pump.

Pump unit JNA11,21 of ACN 800-15A type serves as a low pressure pump in system 1 JNA. The design of the pump unit is similar to that of ACN 230-65A type units.

According to the conditions of placement in the containment sump, the filtering device structurally consists of two identical filtering module units. A filtering module unit consists of a collector with four filtering modules mounted on its front surface in two levels.

The 1st and 2nd channels' equipment (pump units JNA11 (21), JNA12 (22), plate heat exchangers JNA10 (20)) are located in different physically separated rooms.

The filtering device is installed in the containment slab sump on emergency pump unit suction.

2.1.3.2 System operation in normal operation modes

Unit power operation mode

When the fuel pool cooling system FAK10-20 is brought to repair, in the normal operation mode heat removal from fuel located in the fuel pool is performed by the JNA system.

When the FAK10-20 system is operable, the JNA system is in standby mode. To remove heat from the electric motor, the cooling water system PE is used, while end seals are cooled with the intermediate circuit KAA water. Boron is washed from the end seals of the pump with distilled water.

To maintain the system operable, periodic tests and trials are carried out in accordance with the process regulations.

Maintenance and repair cooldown mode

In primary circuit maintenance cooldown mode the JNA11 pump inlet is connected from **the "hot" leg of the 2nd MCC loop**, and supplies coolant under the reactor core (to the reactor pressure chamber). In primary circuit repair cooldown mode the JNA11,21 pump inlet is connected from the "cold" leg of the 2nd MCC loop, and supplies coolant above the reactor core (to the reactor collection chamber). Suction pipelines are connected to “hot” and “cold” legs of 2nd MCC loop via leak limiters.

In the primary circuit maintenance cooldown mode, JNA21 pump draws coolant from the “hot” leg of 4th MCC loop to the reactor pressure chamber under the reactor core. In the primary circuit repair cooldown mode JNA21 pumps draw coolant from the “cold” leg of 4th MCC loop to the reactor collection chamber above the reactor core. Suction pipelines are connected to “hot” and “cold” legs of 4th MCC loop via leak limiters.

In maintenance unit cooldown mode reactor plant cooldown is performed by steam discharge to turbine condensers through a turbine bypass valve until the temperature in the primary circuit decreases to 150 °C and the pressure reaches 1,96 MPa. Primary circuit cooldown below 150 °C is performed via the emergency and maintenance cooldown system JNA. At the same time, the extraction of the primary circuit coolant is carried out from the "hot" part of the 2nd or 4th MCC loop via a pipeline DN 300 through the heat exchangers JNA10,20 to the inlet of the primary circuit emergency and maintenance cooldown pumps JNA11,21 (JNA12,22) providing coolant circulation along the maintenance cooldown circuit. The cooled medium is returned via one of the JNA channels to either "cold" parts of the respective loops (1st or 3rd) or pipelines connecting ECCS tanks to the lower reactor chamber (under the core). The power extracted this way by JNA system in the repair cooldown mode equals to at least 15,46 MW.

The pipelines taking coolant from 2nd and 4th loops during cooldown have a common header for two channels. Not more than two pumps can be connected in one or different system channels at the same time.

Refueling mode

When the fuel is fully unloaded from the core into the fuel pool, the boron solution temperature (up to 45 °C) is ensured by the joint operation of one channel of the FAK10-20 system and one channel of the 1 JNA system.

In the de-energizing mode the pumps automatically start recirculation after the diesel generators start-up according to the step-by-step start-up program.

2.1.3.3 System operation in case of a design basis accident

In case of accidents with a large leak (rupture of the pipeline with a diameter of more than 100 mm, including rupture of the main coolant pipeline DN 850) at the initial stage the core cooling and preservation of the required coolant reserve in the reactor is ensured by the reactor core pouring with boric acid solution when the 1st and 2nd stage hydro-accumulators are activated. At the next stage, heat removal from the core and maintaining the required coolant reserve in the reactor are carried out via the operation of the primary circuit emergency and maintenance cooldown pumps, which are started automatically when the process parameters reach the values indicating the primary circuit leak.

Boric acid solution from the fuel pool is supplied to the JNA system pumps' inlets at the first moment of the accident.

In case the fuel pool level is lower than 14,83 m, boric acid solution is supplied to the pumps' inlets only from the containment sump.

After switching from the fuel pool to the containment sump, the residual heat is continuously removed via closed circuit (containment sump → primary circuit and fuel cooling pool emergency and maintenance cooldown heat exchanger → primary circuit emergency and maintenance cooldown pump → MCC → containment sump). Heat removal from JNA system heat exchangers is performed via the KAA building's essential consumers intermediate circuit system.

In case of accidents with primary coolant leak within the pressure range from 7,9 to 2,1 MPa, boron solution supply is performed by pumps JNA12,22. When the pressure drops below 2,5 MPa, the JNA11,21 pump is connected, and the flowrate from the JNA12,22 unit increases due to the medium suction into the mixing chamber of the ejector pump.

In case of "small leaks" (DN 25-80) the pressure in the primary circuit is maintained in the range from 7,9 MPa to 5,5 MPa due to the operation of the JNA12,22 pump. The fuel pool cooldown line can be used for further pressure reduction in the primary circuit. Part of the medium from the JNA11,21 pump supply line is bypassed along the recirculation line, makeup of the primary circuit is temporarily stopped (until the pressure in the primary circuit drops below 2,2 MPa, at which moment the supply from low pressure pumps starts). In order to ensure the cooldown after pressure drops below 1,96 MPa it is necessary to connect one of

the JNA channels to the primary circuit with preliminary disconnection of suction pipelines from the fuel pool and sump. In the other channel of the system, the valves on the fuel pool cooldown line shall remain opened to avoid pressure rise in the primary circuit.

Each channel of the system provides a line for fuel pool makeup in emergency mode, if it is impossible to feed it according to the normal operation scheme. Heat removal from the fuel pool is carried out by the medium evaporation.

2.1.3.4 System operation in case of a beyond design basis accident

In case of a beyond design basis accident with FAK10-20 system pumps failure boric solution supply to sprinkler headers is performed from JNA11,21 pumps.

In case of a severe beyond design basis accident corium catcher heat exchanger is cooled with boric acid solution taken from the sump via electrically driven JKM10 valves.

To protect the primary circuit in the standby mode and cooldown mode, as well as the equipment of the JNA system from overpressure, two safety valves JNA13 are installed on the maintenance cooldown header common for both channels.

2.1.4 1ST STAGE HYDRO-ACCUMULATORS SYSTEM (PASSIVE PART OF ECCS) JNG50-80

The first and second stage hydro-accumulators make up the passive part of the reactor emergency core cooling system (Figure 2.1.4.1).

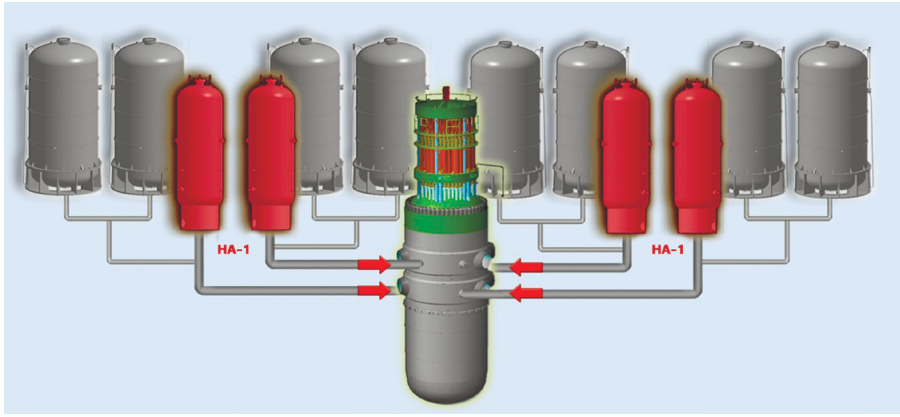


Figure 2.1.4.1 – Schematic diagram of passive part of ECCS

Based on the single failure principle, the water reserve in the 1st and 2nd stage hydro-accumulators is sufficient for re-flooding of the reactor core and removal of residual heat within for at least 24 hours in case of leak of the primary circuit coolant with equivalent diameter DN 850 mm and failure of the active part of ECCS.

2.1.4.1 Application and structure of the 1st stage hydro-accumulators system

The first stage hydro-accumulators system is designed to supply boric acid solution with concentration of at least 16 g/dm^3 and temperature of at least 20 °C to the reactor at primary circuit pressure less than 5,9 MPa in amount sufficient for reactor core cooling until the connection of low pressure part of emergency and maintenance cooldown and fuel pool cooling system and/or 2nd stage hydro-accumulators system in case of design and beyond design basis accident modes involving loss of primary circuit coolant.

Each tank contains a reserve of 50 m^3 boric acid solution (16 g/dm^3) under a pressure of 5,9 MPa, which is created by a nitrogen cushion.

ECCS hydro-accumulators shall actuate passively and cut off from the reactor in a timely manner when the boric acid solution reserve is exhausted.

The system consists of four identical and completely independent channels. Two channels provide boric acid solution supply to the reactor pressure chamber under the core, while the other two supply it to the reactor collection chamber above the core (Figure 2.1.4.2.)

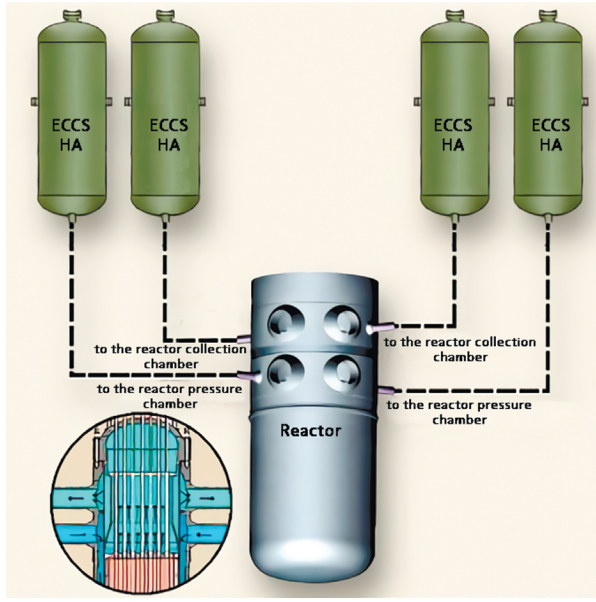


Figure 2.1.4.2 – Diagram of ECCS hydro-accumulators' connection to the reactor

ECCS hydro-accumulator channels are separated pairwise (Figure 2.1.4.3) and are securely fixed to exclude common cause failures

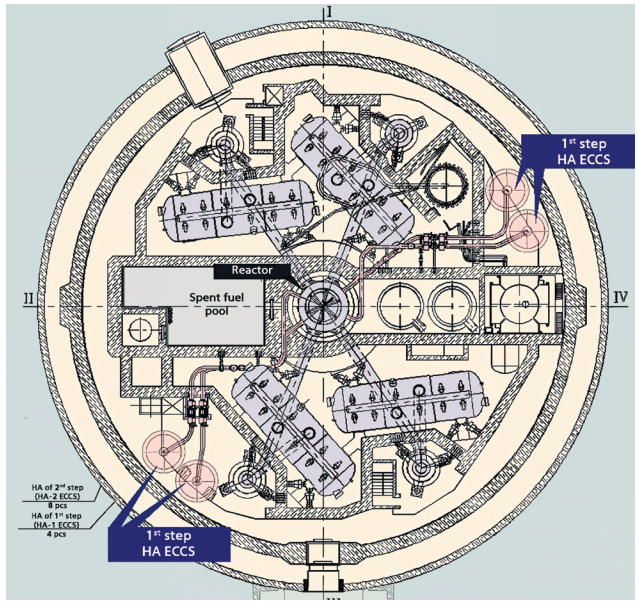


Figure 2.1.4.3 – Pairwise separation of ECCS passive part's channels

The system was designed based on the single failure principle and with conservative assumption that part of boron solution from activated ECCS hydro-accumulators enters the leak bypassing the reactor core.

2.1.4.2 Process diagram of the system.

Each channel of the system includes: ECCS tank, valves and pipelines located inside the containment. All ECCS hydro-accumulators are situated at the same height – in the central hall at elevation + 26,3 m (Figure 2.1.4.4).

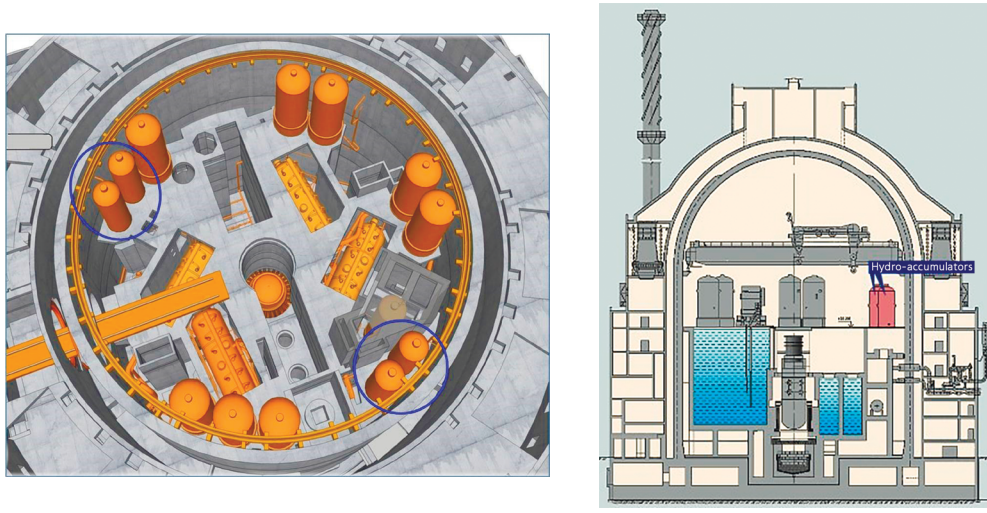


Figure 2.1.4.4 – ECCS hydro-accumulators' location in the containment

The tank is connected to the reactor by ECCS pipeline (DN 300). The ECCS pipeline is welded to a specific connection on the reactor vessel (Figure 2.1.4.5).

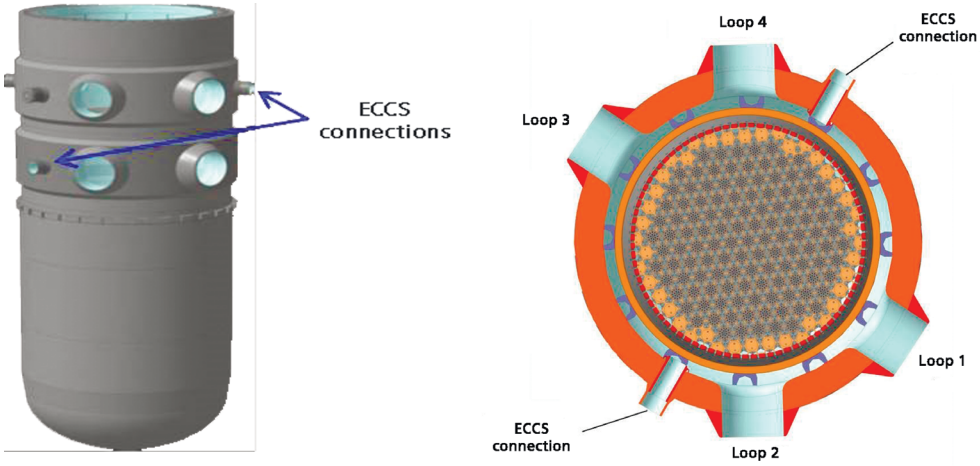


Figure 2.1.4.5 – Location of ECCS connections on the reactor vessel

Each connection has a thermal jacket for protection against rapid temperature changes when water enters the reactor.

Simplified diagram of such channel is shown in Figure 2.1.4.6.

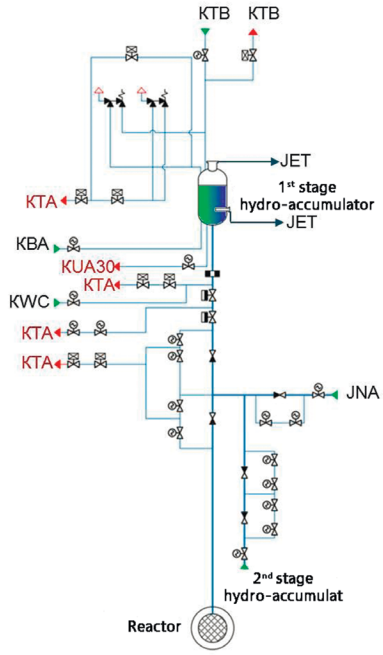


Figure 2.1.4.6 – Diagram of the 1st stage hydro-accumulator channel

The connection pipeline "ECCS hydro-accumulator – reactor" has the following parts sequentially installed: throttle device, two fast-acting isolation valves and two check valves.

Fast-acting isolation valves are installed due to the fact that when emptying an ECCS tank during an accident it is necessary to quickly cut off the boric solution flow and prevent nitrogen from entering the reactor. During normal operation fast-acting isolation valves stay open.

Check valves are designed to cut off the tank from the reactor during normal operation of the reactor plant. ECCS pipelines contact with boric acid solution from ECCS tanks until the second check valve, and with primary circuit coolant – after the check valve.

ECCS hydro-accumulators deploy water to the reactor passively in case of positive pressure swing at the check valve (more than 0,03 MPa) and are cut off from the reactor in a timely manner at a low level in the hydro-accumulators by closing a fast-acting isolation valve in less than 10 seconds.

The pipeline DN 300 between fast-acting isolation valves has a drain line DN 20, which serves to remove possible leaks through the closed fast-acting isolation valves (when a hydro-accumulator is disconnected from the reactor) to the drainage and controlled leakage system (KTA).

Bypass lines parallel to check valves are provided for pipelines heating and check valves density control. Each bypass line is equipped with bellows isolation valves DN 15 with electric drive (two for each bypass). During the reactor plant power operation, those valves are closed.

Between the check valves there is a common tie-in from the low-pressure part of the JNA (primary circuit and fuel cooling pool emergency and maintenance cooldown) system and 2nd stage hydro-accumulators (JNG10-40). This solution reduces the number of tie-ins into the reactor vessel.

ECCS hydro-accumulators system is connected to the following process systems:

— primary circuit blowdown and makeup system (KBA), which supplies boric acid solution to hydro-accumulators;

— radioactive media drains and controlled leakage system (KTA), which provides hydro-accumulators drainage and leakage removal through closed fast-acting isolation valves and valves on the check valve bypass lines;

— gas blow-off system from Reactor Shop equipment (KTB), which provides:

- air removal during ECCS hydro-accumulator filling;
- nitrogen supply to ECCS hydro-accumulators under 5,9 MPa pressure from KRA21 high pressure nitrogen system;

— Reactor Shop equipment sampling system (KUA30);

— system for hydraulic testing and flushing I&C sensors with boron-containing water (KWC);

— primary circuit connections' leakage monitoring system (JET).

2.1.4.3 Short description of the equipment

An ECCS hydro-accumulator is a vertical cylindrical vessel with elliptical head and bottom mounted on a cylindrical support (Figure 2.1.4.7).

Cylindrical support, in turn, is installed on a ring embedded in concrete with anchors.

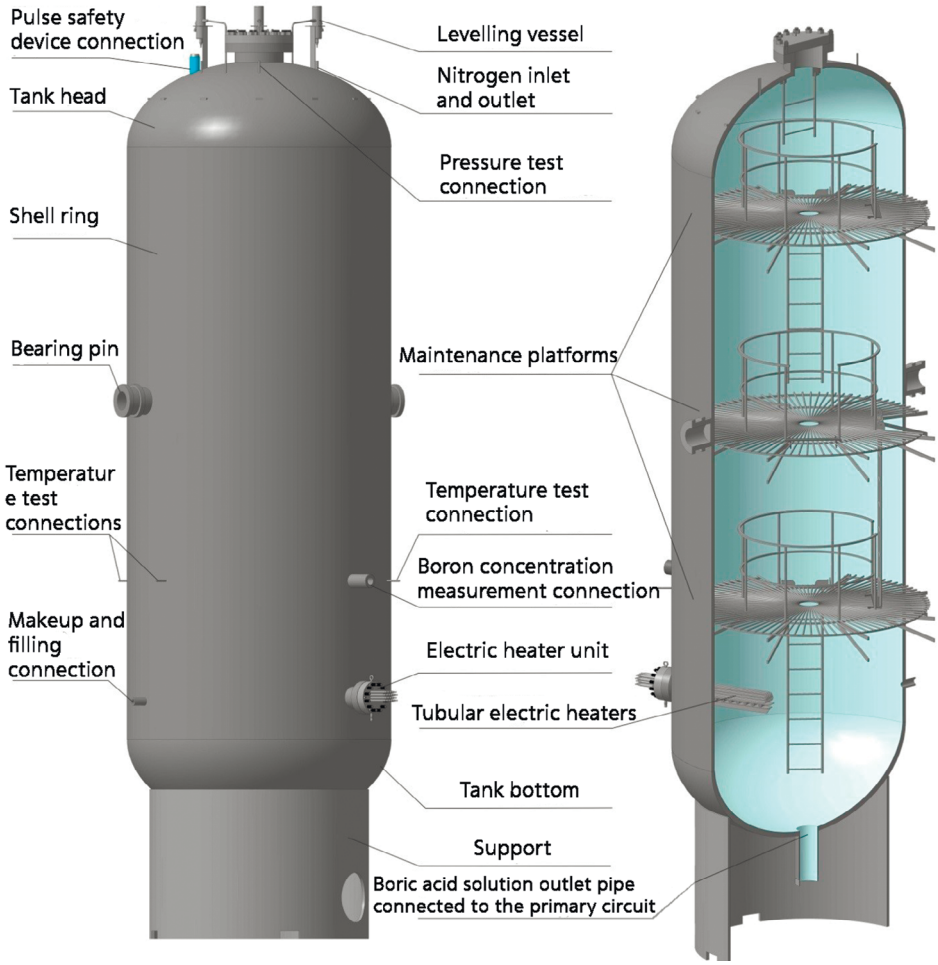


Figure 2.1.4.7 – ECCS hydro-accumulator design

At the lower bottom there is an inlet pipe DN 300 for supplying solution to the reactor. That pipe is connected to the ECCS pipeline via an adapter DN 300. Exit for the ECCS pipeline and access to the inlet pipe are provided via two 600 mm diameter holes in the support.

The cylindrical part of an ECCS hydro-accumulator body is made of three sections welded together.

In the lower section there is an electric heater unit, which heats the solution and maintains its temperature within the required range. The tubular electric heater unit is sealed with by a lid using two sealing graphite gaskets. The cavity between the gaskets is connected to the sealing control connector.

The tank head has a hatch with an internal diameter of 450 mm, which serves to provide access to the ECCS tank internals. The hatch cover is sealed with two graphite gaskets, the cavity between which is connected to the sealing control connector.

Inside the ECCS tank there are metal maintenance platforms installed at different heights. Stairs are available to move from one platform to another.

Main technical characteristics of an ECCS tank are given in Table 2.1.4.1.

Table 2.1.4.1 - Main technical characteristics of an ECCS hydro-accumulator

Parameter name	Value
Operating environment	Boron solution with concentration 16-20 g/dm^3 + nitrogen
Operating pressure, MPa	5,9 ± 0,3
Design pressure, MPa	6,37
Solution temperature in an ECCS tank during operation, °C, within the following limits	20-70
Volume of ECCS tank, m^3	60 ± 1
Volume of solution in an ECCS tank in operating conditions, m^3	50 ± 0,5
Nominal level of solution in an ECCS tank, mm	7350 ± 50
Number of units of tubular electric heaters, pcs	1
Assigned service life of an ECCS hydro-accumulator, years	60

The boric acid solution volume in the tank is selected on the condition that the solution reserve in three vessels is sufficient for reactor core cooling in case of design basis accidents with coolant loss in the period prior to connection of the low-pressure part of the emergency and maintenance cooldown and fuel pool cooling system and/or the 2nd stage hydro-accumulators.

The volume and pressure of nitrogen in the tank, as well as the hydraulic resistance of the pipelines connecting hydro-accumulators to the reactor, were selected based on the condition of providing the flooding rate required for the core cooling.

Boron concentration in the tanks was selected based on the condition of ensuring reactor core subcriticality.

Pulse safety devices

Each ECCS tank has two pulse safety devices – control and operating – installed.

Pulse safety device capacity was selected based on the condition of not exceeding the design pressure by more than 15 % and is at actuation pressure: 6 t/h for nitrogen and 30 t/h for water, which exceeds the maximum flow rate of the tank filling systems with nitrogen or water.

The opening start pressure of a control / operating valve is equal to 6,18/6,24 MPa, respectively. The closing start pressure of the control and operating valve is 5,48 MPa.

2.1.4.4 System operation monitoring and control.

Reliable operation of the instrumentation and control system for the passive part of ECCS is ensured with the following solutions adopted in the design:

— redundancy of measurement channels by protection-related parameters;

— power supply of measurement channels and actuating parts (fast-acting isolation valves) of the passive part of ECCS is provided from the sources of the emergency power supply system for first category essential consumers. In addition, each of the actuators and corresponding measurement channels are powered by their own reliable power supply system.

Fast-acting isolation valves are controlled automatically by ESFAS signals (Table 2.1.4.2) or remotely by MCR / ECR operator.

Table 2.1.4.2 - List of instrumentation and control system functions for ECCS hydro-accumulators

Actuation condition	Setting	Action	I&C system function
1. Signals match: — difference between primary circuit saturation temperature and temperature in any of loops' hot legs, °C, less than; — overpressure in ECCS tank, MPa, more than — no signal on level decrease in ECCS tank, m, less than	8 2,35 1,25	Opening fast-acting isolation valves on pipelines connecting hydro-accumulators to the reactor. The logic is implemented using a memory element (trigger), which is loaded when there is a signal to open fast-acting isolation valves and provides valve control, while blocks individual remote control for 30 minutes. After 30 minutes the operator can reset this memory and access control of the valves, which have individual controls on safety panels. In addition, the memory automatically resets 30 minutes after receiving a signal for opening the fast-acting isolation valves due to coolant temperature decrease in loops' hot legs to 70 °C.	ESFAS BA13
2. Signals match: — overpressure in the containment, kPa, more than; — overpressure in ECCS tank, MPa, more than — no signal on level decrease in ECCS tank, m, less than	30 2,35 1,25		ESFAS BA13
3. Signals match: — a signal from the MCR / ECR button "Emergency heat removal in case of primary leak" — overpressure in ECCS tank, MPa, more than — no signal from BE13 "$P_{primary\ circuit}$ > 3,3 MPa & $T_{emergency\ protection}$ < 100°C" — no signal on level decrease in ECCS tank, m, less than	2,35 1,25	Opening fast-acting isolation valves on pipelines connecting hydro-accumulators to the reactor. The logic is implemented using a trigger without a 30-minute reset block. The trigger is reset: — by operator via "Memory reset" button — automatically due to coolant temperature decrease in loops' hot legs to 70 °C.	ESFAS BA13
4. ECCS tank level, mm, less than	1250	Closing fast-acting isolation valves for a corresponding ECCS tank	ESFAS BA13

Actuation condition	Setting	Action	I&C system function
5. Signals match: — overpressure above the reactor core, MPa, more than; — coolant temperature in hot legs of loops, °C, less than	3,3 100	Closing fast-acting isolation valves for a corresponding ECCS tank with remote opening blocked	ESFAS BE12
6. ECCS tank level, mm, less than	1150	Closing fast-acting isolation valves for a corresponding ECCS tank with remote opening blocked	DPS

The system status is monitored by operating personnel using UULS snapshot "JNG50-80" (Figure 2.1.4.8).

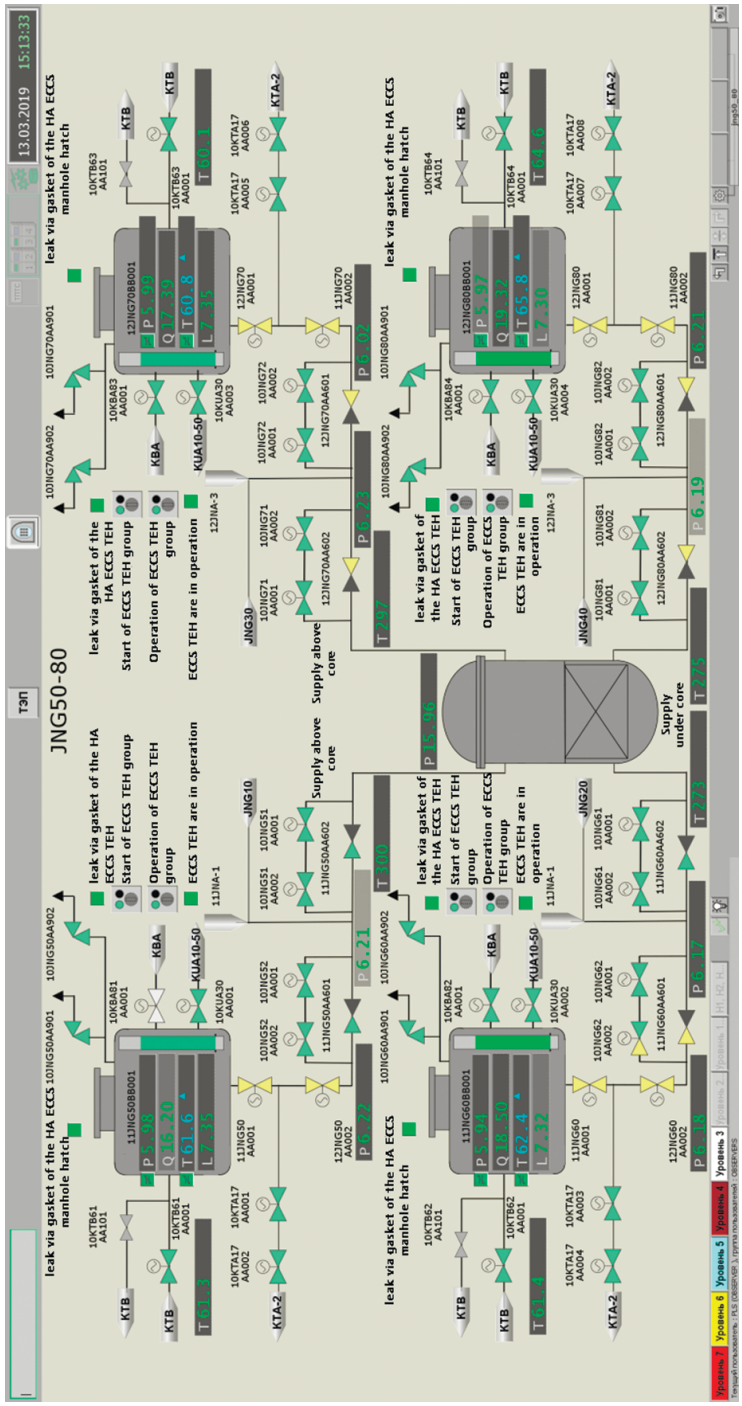


Figure 2.1.4.8 – UULS snapshot for JNG50-80 system

The main parameters indicating system's normal operation are given in Table 2.1.4.3.

Table 2.1.4.3 – ECCS hydro-accumulator system's controlled parameters

Controlled parameter	Nominal value
Level in ECCS tank, m	7,35
Pressure in ECCS tank, MPa	5,9
Water temperature in ECCS tank, °C	20 – 70
Mass concentration of boric acid, g/dm^3	16 – 20
Pressure on bypass lines of check valves, MPa	5,9

Alarm is provided to inform the operator about deviation of these parameters beyond the permissible limits.

ECCS tank level is regulated by draining borated water into the drainage and controlled leakage system and supplying concentrated boric acid solution to a tank from the primary circuit makeup and blowdown system.

Nitrogen pressure in tanks is maintained by the operator by supplying nitrogen or discharging it into the gas blow-off system.

Boric acid solution temperature in ECCS tanks is maintained at 55-70 °C level due to automatic activation – disconnection of starting and operating group of ECCS tubular electric heaters.

Boric acid concentration is maintained by draining borated water into the drainage and controlled leakage system and supplying concentrated boric acid solution to the tank from the primary circuit makeup and blowdown system.

In emergency modes, when ECCS hydro-accumulators deploy water to the reactor, the operator uses indicating instruments and mnemonic diagram of the 1st stage hydraulic accumulators system on the MCR/ECR safety panels to monitor changes in hydro-accumulator parameters and control the subsequent closure of fast-acting isolation valves (Figure 2.1.4.9).

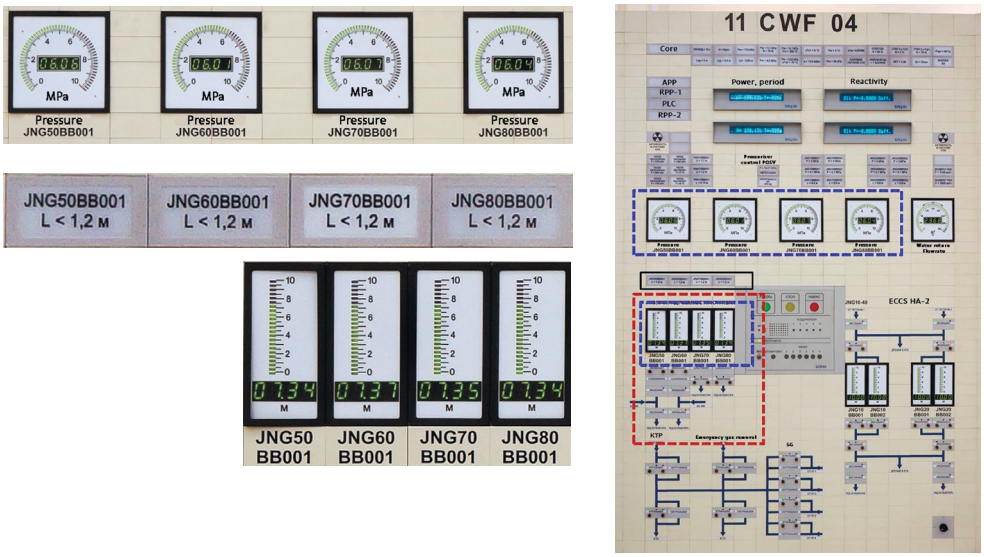


Figure 2.1.4.9 - Location of instrumentation for monitoring and control of the 1st stage hydro-accumulators system on MCR safety panels

2.1.4.5 The working modes of ECCS accumulators

Normal operation of the system

When heating up the reactor an operator connects accumulators to the reactor at primary pressure of more than 6,9 MPa(g). If there is a reactor coolant leakage before the operator connects the system, the system switches to the working mode automatically by the signal of the primary circuit breaking protections: «ΔT_{S-1} < 8 °C» or «$P_{containment}$ > 30 KPa» (Figure 2.1.4.10).

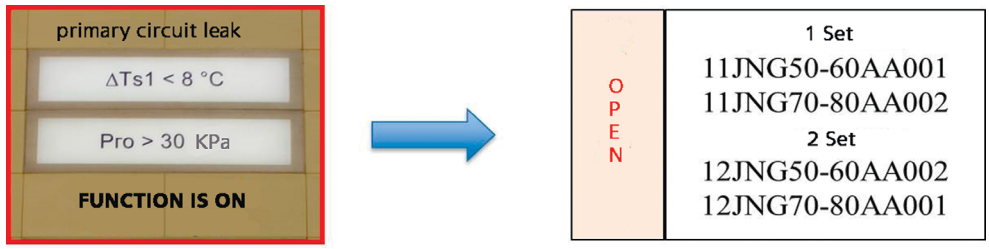


Figure 2.1.4.10 – Opening of quick-acting shut-off valves of ECCS accumulators by the signal of the primary circuit breaking protections «ΔT_{S-1} < 8 °C» or «$P_{containment}$ > 30 KPa»

When the reactor is operating at power, the system is in standby: quick-acting shut-off valves are open, each accumulator is isolated from the reactor by two tandem check valves Dn 300.

In scheduled reactor cooling down to prevent actuation of the system caused by pressure drop, an operator isolates the system from the reactor in advance at pressure of 9,8 – 8,8 MPa (Excess) above the reactor core. If there is a reactor coolant leakage after the accumulators were isolated, the system switches to the working mode automatically by the signal of the primary circuit breaking protections: «ΔT_{S-1} < 8 °C» or «$P_{containment}$ > 30 KPa» (Figure 2.1.4.10).

System operation during large break in the primary circuit (LBLOCA)

At initial stage of large break accidents (break of a pipeline larger than 100mm diameter including main circulation pipeline Dn 850) passive actuation of I stage accumulators provides for cooling of the reactor core and maintaining the level of the reactor coolant.

The estimates show that the reactor and the entire primary circuit are almost completely drained in the first seconds of the accident. As a result a heat exchange crisis arises on the surface of the fuel cladding and the temperature begins rising. At the 7th second the cladding temperature of the most heat-stressed fuel rod in the hot spot reaches 1094 °C with maximum design limit of 1200 °C.

To supply boron solution to the reactor in time there are I stage accumulators designed. Parameters of ECCS accumulators are based on conditions of maximum design basis accident. The working pressure of 5,9 MPa triggers water supply from accumulators to the reactor in approximately 7 seconds after the guillotine rupture of Dn 850.

It is not reasonable to supply water from the accumulator to the reactor earlier because much of it will flow to the leak at initial stage. In this case to maintain sustainable supply from ECCS accumulators to the reactor, we would have to ramp up its volume.

At further stages of the accident starting with its 40th second, heat removal from the reactor core and reactor coolant supply in the reactor are provided by continuous circulation and cooling of boron solution with pumps and heat exchangers of JNA system (emergency and planned cooling of the primary circuit and spent fuel pool).

In large leak mode an operator shall monitor the work of I stage accumulators based on the following features:

— shut-off and check valves in ECCS pipeline are open;

— pressure and level in the accumulators are reducing according to the indicators;

— green color of the accumulator bar chart has changed to orange one when the level in the accumulator falls below 7 meters;

— alarm display «L_{HA} < 1,2 м» triggers on MCR/ECR safety panels and quick-acting shut-off valves close automatically when the level in the accumulator falls to 1250 mm (Figure 2.1.4.11).

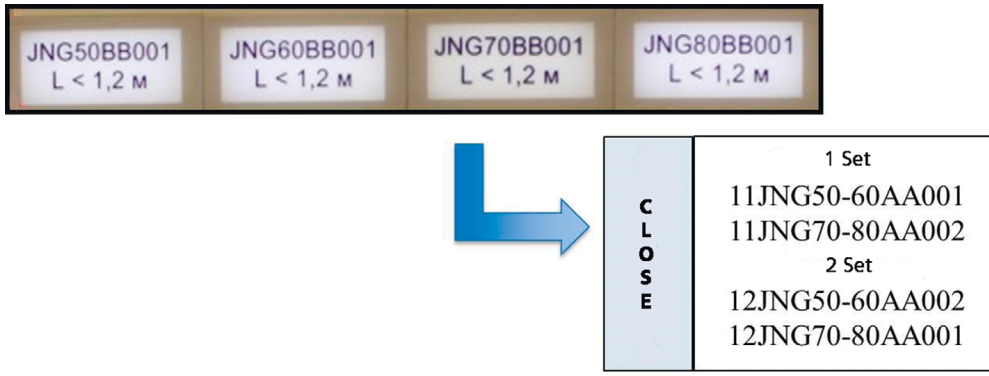


Figure 2.1.4.11- Closure of quick-acting shut-off valves when the level of ECCS accumulators falls

System operation during small break in the primary circuit (SBLOCA) and failure of active part of high pressure ECCS.

During small break in the primary circuit (Dn25 – Dn100) in the first and third loops of the main circulation pipeline there may be a failure of two trains of active part of high pressure ECCS due to subsequent single and dependent failures of subsystem trains.

In this case personnel shall provide for the reactor cooling down with 60 °C/h rate and reduction of pressure above the reactor core below 5,9 MPa. After that ECCS accumulators supply the primary circuit until ECCS low pressure pumps (2,1 MPa) start.

System operation during LOCA with loss of all AC power sources including diesel generators

If all AC power sources are lost, ECCS active part fails. In this case reactor coolant supply is provided by I stage accumulators (passive flooding of the reactor core at the initial moment) and II stage accumulators (long-term water supply to the reactor in line with the reduction of the residual heat).

To remove heat from the reactor core robustly, water supply shall not be interrupted. There are throttle units installed in the pipelines connecting ECCS accumulators and the reactor (Figure 2.1.4.12). They can limit the flow rate of the water supplied from the accumulators to maintain reactor core cooling until II stage accumulators start.

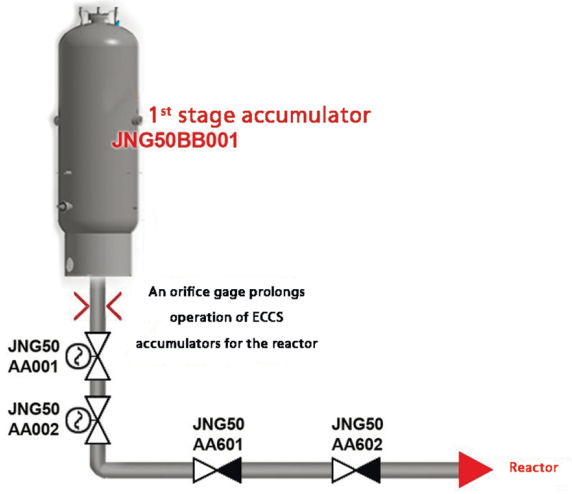


Figure 2.1.4.12 A throttle unit in the ECCS pipeline

The throttle unit prolongs supply from ECCS accumulators to the reactor up to 185 - 195 seconds at maximum design basis accident, and boron solution starts flowing into the reactor from II stage accumulator (see the next chapter) at 116th second (Figure 2.1.4.13)

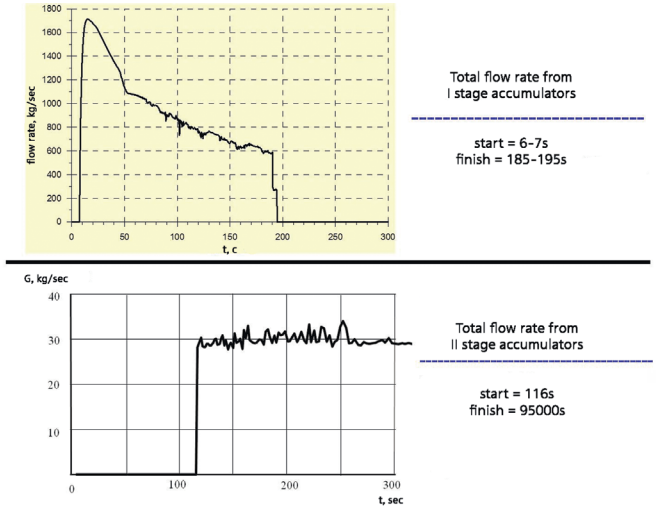


Figure 2.1.4.13 –Overlap of the flow rates of I and II stage accumulators at maximum design basis accident

2.1.5 II STAGE ACCUMULATORS (REACTOR CORE PASSIVE FLOODING SYSTEM) JNG10-40

2.1.5.1 The function and structure of the system

The system of the second stage accumulators JNG10-40 is designed to perform the following functions:

— to supply boron solution with 16-20 g/dm^3 concentration to maintain the coolant level in the reactor core at LOCA and pressure drop below 1,5 MPa in the primary circuit;

— to store boric acid in order to fill the spent fuel pool compartments at shutdown reactor during refueling.

The system JNG10-40 acts passively providing residual heat removal at LOCA when primary pressure is less than 1,5 MPa.

The system has four independent and physically separated groups. Each group contains two accumulators of passive flooding system. They supply boron solution to the pressure/ or collecting chamber of the reactor via appropriate ECCS pipelines of I stage accumulators (Figure 2.1.5.1).

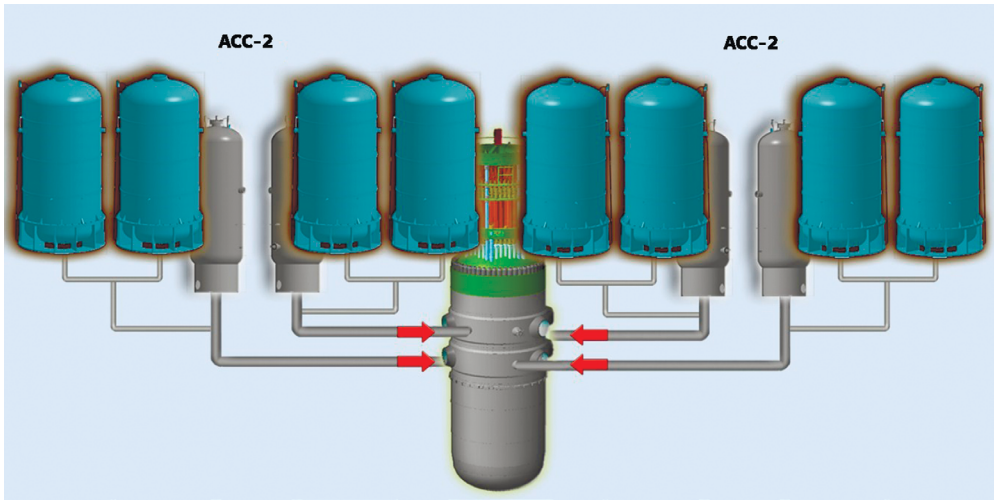


Figure 2.1.5.1 – Basic layout of the ECCS passive part

The system design considers time profiling of the flow rate supplied to the reactor core in accordance with the power reduction of the residual heat (table 2.1.5.1).

Table 2.1.5.1 – Relationship between boric acid flow rate of one group and duration of the profiling stage

Duration of stage, s	100 - 5430	5431 - 10860	10861 - 29000	29001 - 86400
Boric acid flow rate of one group, kg/s	10,0	5,0	3,3	1,6

According to the accepted design characteristics of the equipment three available groups are enough for the system to carry out its design functions in full in any mode that requires its operation. The water supply in the accumulators of three groups (720 m^3) with passive heat removal system in operation is sufficient to remove residual heat within 24 hours.

The system groups are physically separated and totally independent: they have independent process equipment, control systems, supporting systems, locations of equipment, pipelines, cables, control elements etc. Thus a failure in one group cannot lead to a failure in another one. Such circuit design and layout of equipment eliminates dependent failures and common mode failures caused by location of the equipment as well as the influence of any activity performed on the equipment in one group (repair, maintenance) on another group.

The design of the system provides for checking and periodic testing of system elements in order to verify their operability which eliminates undetectable failures of elements that are not monitored during NPP operation.

2.1.5.2 Process flow diagram of the system.

The system contains four groups (trains). Each system train includes two accumulators of the reactor core passive flooding system (RCPFS), valves and pipelines located inside the containment. All accumulators of passive flooding system are installed in the reactor hall at elevation of +26,3 m (Figure 2.1.5.2).

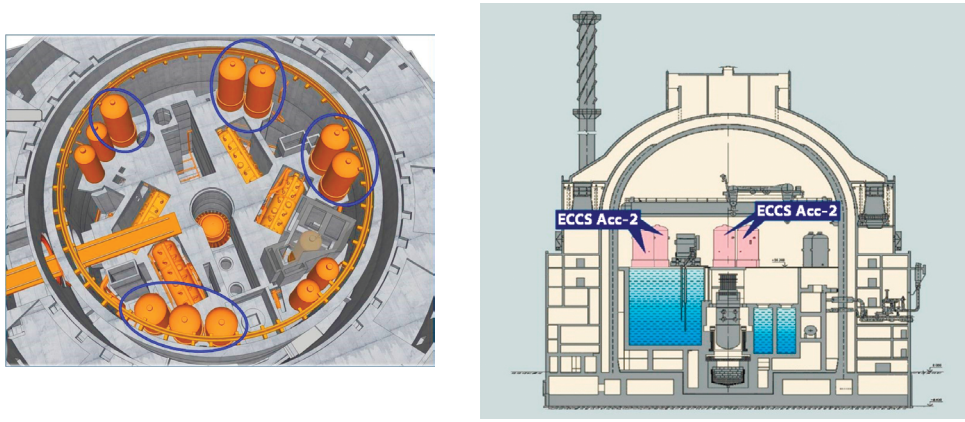


Figure 2.1.5.2 – Location of ECCS accumulators in the containment

The leveling mark for installation of passive flooding accumulators above the reactor is at least 3 times higher than the level necessary to provide the coolant flow from the accumulator to the reactor. This is confirmed by both calculations and experiment in which the flow rate exceeded the design value by 70 %.

The system accumulators and drain lines are totally filled with boric acid solution.

To protect the equipment and pipelines of II stage accumulators from overpressure, there are two safety valves installed in each system train: 2 x 100 %.

The II stage accumulators of the RCPFS (Acc-2) of one train are connected with a drain line to the connecting line of ECCS I stage accumulators (Acc-1) in the section non-isolated from the reactor– between the check valves. In addition the drain line of II stage accumulators is cut into the connecting pipeline of ECCS accumulators from the bottom upwards creating a hydraulic lock higher than 0,5 m to eliminate steam inflow to the drain line during an accident.

Figure 2.1.5.3 shows a simplified flow diagram of the system.

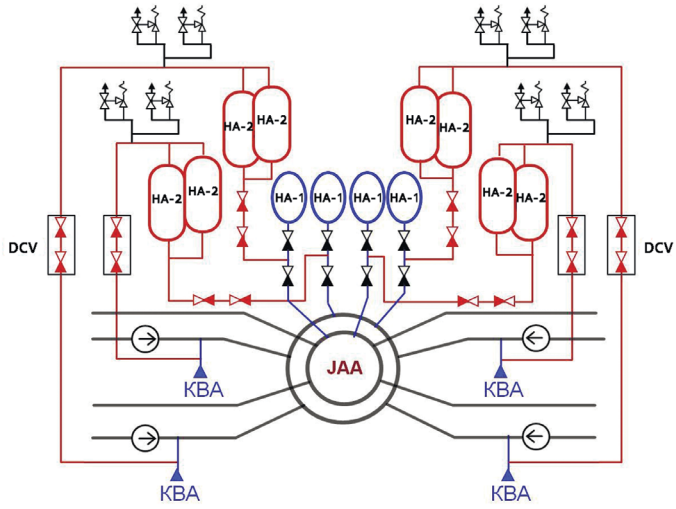


Figure 2.1.5.3 - A simplified chart of the 2 stage accumulators

Legend: DCV - double check valve, HA-1 – Acc-1 (I stage accumulators), HA-2 – Acc-2 (II stage accumulators), KBA - makeup and blowdown system

Along the steam supply line the PCPFS accumulators are connected to the cold lines of main circulation pipelines via Dn100 lines with special double check valves and electric-driven valves installed in them (Figure 2.1.5.4). The connecting line is cut in the sectors of the makeup and blowdown pipeline.

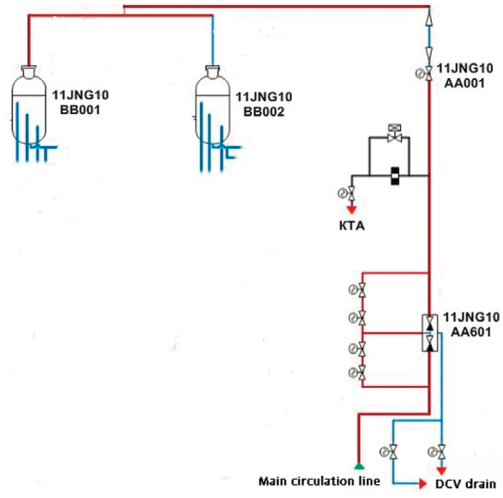


Figure 2.1.5.4 – Connecting lines of the reactor core passive flooding accumulators and the main circulation line

Double check valves (DCV) open when pressure in the primary circuit drops from 1,9 to 1,5 MPa. When they open, pressure in the accumulators rises up to the pressure in the primary circuit and boron solution flows in the reactor along the drain lines under the action of water head.

Isolation valves and check valves are installed in the drain line (Figure 2.1.5.5).

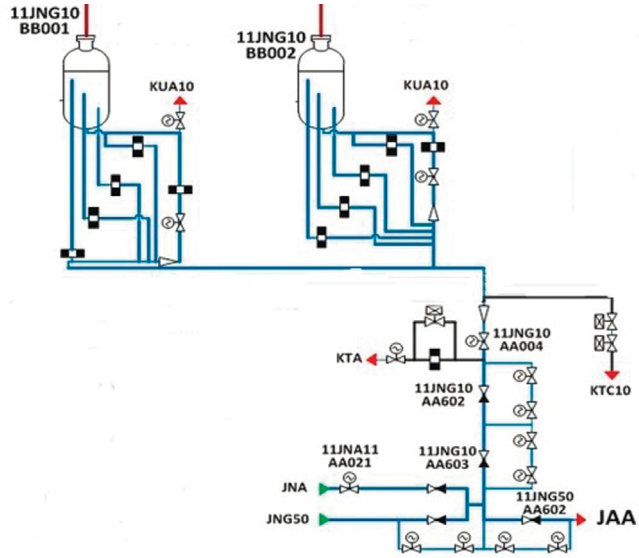


Figure 2.1.5.5 – Drain lines of the II stage accumulators

Isolation valves are necessary to isolate accumulators of the reactor core passive flooding system from the primary circuit during the outage (only after reactor shutdown). Check valves are designed to avoid pressure growth in the accumulators in standby and automatic passive opening of the discharge line in emergency in case of positive pressure difference at the check valve. To remove possible leakages of the check valves there is a drain line between the valves and the check valves to KTA system (primary circuit drains and controlled leaks).

Figure 2.1.5.6 shows a process diagram of the II stage accumulators (without a pilot operated valve). The system JNG10-40 has process communications with the following systems:

— water supply from the spent fuel pool to purification (FAL);

— emergency and scheduled cooling down and cooling of the spent fuel pool (JNA);

— makeup and blowdown system (KBA);

— the system of primary circuit drains and controlled leaks (KTA);

— off gas system (KTB);

— sampling system from the equipment of the reactor building (KUA10-50);

— hydraulic testing and blowdown of instrumentation sensors with boron water (KWC);
— spent fuel pool cooling (FAK10-20).

2.1.5.3 Summary of equipment

II stage accumulator.

An accumulator of the reactor core passive flooding system (RCPFS) is a vertical cylindrical tank with oval bottoms (Figure 2.1.5.7).

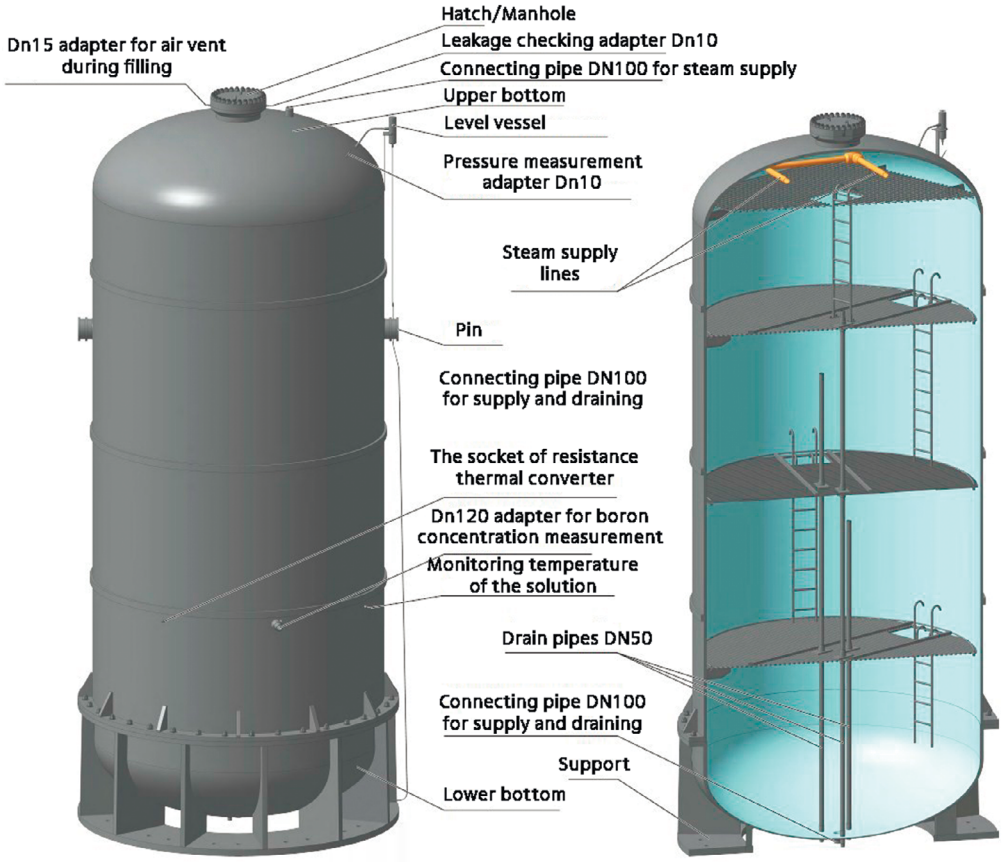


Figure 2.1.5.6 – The design of RCPFS accumulator

The fastening elements of the tank form a support that bears static and dynamic loads. The support is made of ribs that are welded to the plate and the flange. The flange is attached to the tank ring with studs. Facing sheets are welded to

the end side of the ribs which are, in turn, welded to the flange and the plate. The plate is welded from the inside to the embedded ring and it is also attached to it with studs. The ring is embedded in the concrete.

In the lower bottom there are connecting pipes to supply boron solution to the reactor and drain the accumulator of the reactor core passive flooding system.

To ensure the necessary change of the flow rate to the reactor in time, the boric acid solution is drained from the accumulators through a series of parallel pipelines with throttle washers (Figure 2.1.5.5). Each flow rate at a certain stage corresponds to a specific range of the level change in accumulators. The flow rate at each stage depends on the location height of the pipelines with throttle washers. These pipelines are located directly in the accumulators. For four-stage profiling three outlet pipes Dn 50 and one pipeline Dn 100 with throttle washers are provided.

As long as the level is above the upper pipeline, the water flow is maximum. As the water level falls from the upper pipeline to the second-to-last one, the water flow rate is determined by the total throughput capacity of those parallel lines with throttle washers that are connected below the current water level in the accumulator.

The minimal water flow from the accumulators enters the reactor after the level falls below the second-to-last nozzle, this flow is determined by the throughput capacity of the last washer only.

At the upper bottom of the accumulator there is a nozzle to weld the pipeline connecting the accumulator with the main circulation pipeline (Dn100) and a supporting bracket to fix a level vessel.

Besides at the top bottom there is a hatch of 480 mm inner diameter providing access inside the RCPFS tank. The hatch is closed with a lid. The lid is also sealed with two gaskets made of expanded graphite and installed in the flange boring of the hatch. The leakage is monitored with a leakage checking adapter installed at the flange and connected with the space between the gaskets.

In the upper part of the tank there is a perforated plate with 0,5 m^2 total area of the holes. The volume of the boric acid solution separated by the perforated plate in the upper part of the tank equals to 1,5 m^3. Above the perforated plate there is a distributing steam header providing distribution of steam delivered from the main circulation pipeline along the steam supply line mentioned above to the RCPFS tank if the system actuates.

Inside the tank there are standard metal service platforms installed at different height.

Main characteristics of the RCPFS accumulator are given in table 2.1.5.2.

Table 2.1.5.2 – Technical characteristics of the RCPFS accumulator

Name of parameter	Unit	Value
Pressure, rated	MPa	3,0
Pressure of hydraulic testing: — lower limit; — upper limit.	MPa	 4,3 4,8
Volume of solution in working conditions	m^3	120
Volume of the tank	m^3	120
Height	mm	10450
Inner diameter	mm	4120
Design lifetime of an accumulator, years	year	60

Safety valve of the RCPFS accumulators

The opening setting of the safety valve is 3,0 MPa. At this pressure one safety valve maintains throughout capacity of 200 m^3/h.

This throughout capacity corresponds to the capacity of an emergency cooling pump of JNA high pressure system at 3,0 MPa pressure in the primary circuit. Operation of this pump for the primary circuit and non-closure of a double check valve in any group of JNG10-40 system is considered by the design as one of the reasons for pressure growth in the II stage accumulator system.

The pilot operated valve consists of a main valve, a pilot valve and a pipe connecting the main valve with the pilot one. The main valve is designed to release overpressure of the working medium in the protected system and is equipped with a piston drive for relief.

The pilot valve is designed to create open/close commands of the main valve by releasing the working pressure of the space above the main valve piston or supplying the working medium to it. The pilot valve is a direct operated pressure relief valve supplying the medium to the slider valve and the sensor element - bellows. The pilot valve is also equipped with an electric magnet for opening.

Double check valve (DCV)

Double check valve is designed:

— to isolate II stage accumulators from the primary circuit when its pressure

increases from 2,75 to 4,0 MPa in all operating modes of the power unit including design-basis and beyond-design-basis accidents;

— to connect passively II stage accumulators at 1,9 – 1,5 MPa pressure in the primary ciurcuit in all operating modes of the power unit.

Double check valve consists of two tandem shutoff units which can be opened and closed simultaneously with the same drive. The valve is actuated (opens/closes) passively with the working medium.

DCV includes the following units interconnected with pneumatic and hydraulic communications (Figure 2.1.5.7a):

— actuator valve;

— control valve;

— leak tight tank;

— hydraulic lock.

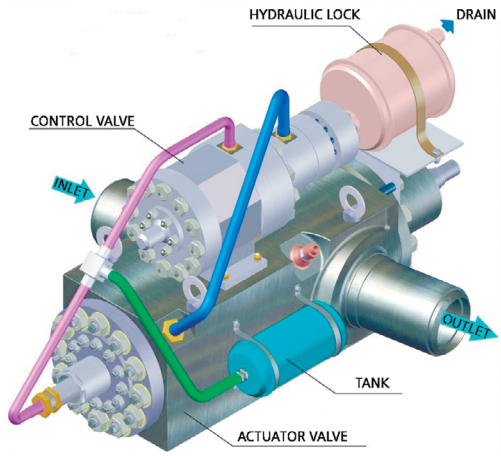


Figure 2.1.5.7a – Overview of the double check valve

An actuator valve (Figure 2.1.5.7b) consists of two tandem shutoff units – the plates connected pivotally with the piston; the bellows drive operating the ball lock and the end indicators of open/close position of the plates.

The plates close under the action of spring in the piston cavity and open under the action of pressure difference at the piston throttle.

The gas cavity of the bellows drive is connected with a leak tight tank by a pipeline.

The open/close position of the actuator valve is monitored by electric signals from the sealed contacts of the end indicators. Response of the sealed contacts is determined by position of the magnet connected rigidly with the plates.

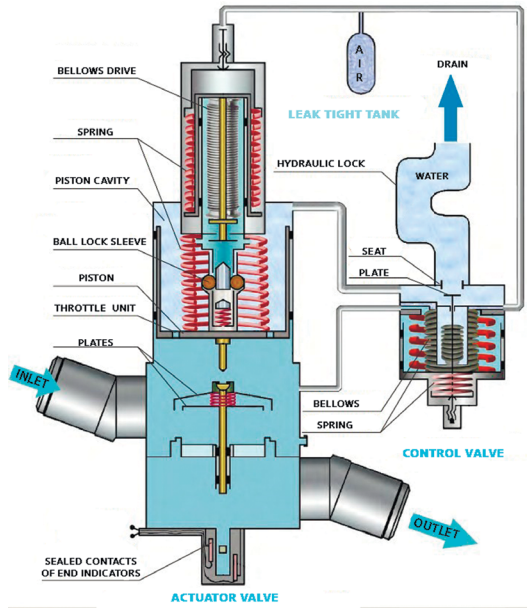


Figure 2.1.5.7b Composition and design of a double check valve

A control valve contains a bellows unit, a plate pivotally connected with it and control springs. The inlet of the control valve is connected with the piston cavity of the actuator valve by the pipeline; and the outlet of the control valve – with hydraulic lock and further on drain line. The cavity between the bellows is connected with the leak tight tank. The leak tight tank is filled with the air of atmospheric pressure. The hydraulic lock maintains the working medium (water) at the outlet of the control valve which eliminates formation of deposits at contact between the plate and the valve seat.

When the reactor is at power, the inlet-outlet line is closed pressure tight with the plates of the actuator valve due to the spring force and the inlet pressure on the plate surface. The tightness towards the drain is provided by the compression of the control valve plate to the seat by the pressure force at the DCV inlet onto the surface of the control valve bellows.

Passive opening of DCV when pressure drops in the reactor is implemented as follows:

The springs of the control valve are adjusted such a way that the pressure force on the surface of the control valve bellows at 1,7 ± 0,2 MPa pressure is less than the force of the control spring. Therefore when pressure reduces below the set point pressure at the DCV inlet, the springs stretch the bellows and the plate moves off the seat. Through the open seat of the control valve the working medium starts

running to the drain line from the piston cavity of the actuator valve. As a result the pressure difference generated at the throttle of the actuator valve is enough to overcome the force of the spring of the actuator valve and compress it to the size providing full opening of the plates and locking them with the ball lock. The actuator valves switches to open position.

Passive closure of DCV is implemented as follows

When pressure in the primary circuit rises above 1,7 ± 0,2 MPa, the control valve closes and the working medium stops flowing through the drain line. As a result the pressure difference at the throttle of the actuator valve falls to zero.

The spring of the bellows drive of the actuator valve is adjusted such a way that at 2,75 — 4,0 MPa pressure the force of this pressure on the surface of the bellows drive exceeds the force of the spring.Therefore when pressure at DCV inlet exceeds the presure setting of the spring, the drive bellows start to compress moving the sleeve of the ball lock. As a result the lock opens releasing the piston. Under the influence of the spring in the piston cavity the piston together with plates closes the inlet-outlet line. The actuator valve switches to close position.

2.1.5.4 Monitoring and control of the system

Using video frames of the Upper Unit Level System (UULS) an operator can monitor valves state and parameters of JNG10-40 system (Figure 2.1.5.8):

— level in the accumulators;

— temperature of the solution in the accumulators;

— temperature of the accumulators vessels;

— pressure in the accumulators;

— H_3BO_3 concentration in the accumulators;

— discrete signal of leaks through the hatches - manholes of the accumulators.

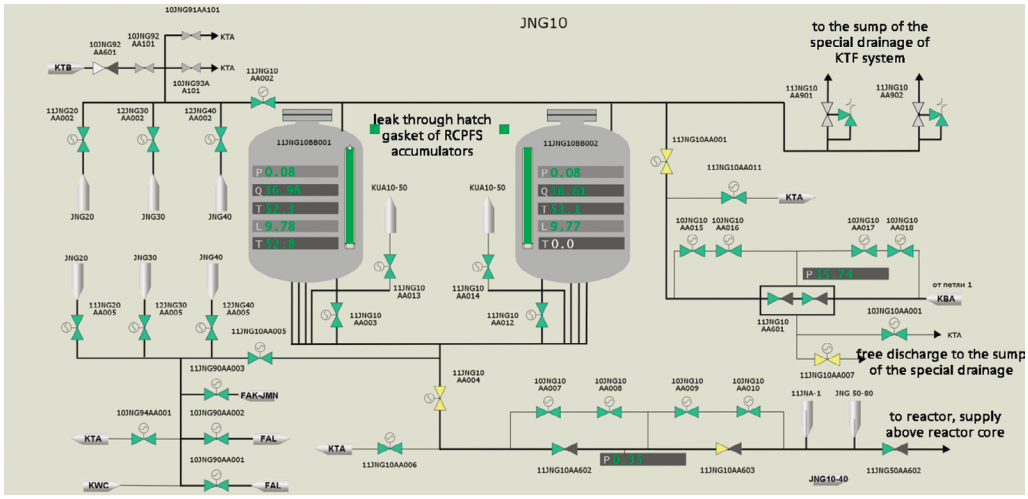


Figure 2.1.5.8 - Monitoring the first train of the II stage accumulators with UULS video frame (standby mode).

When II stage accumulators start water supply to the reactor in emergency conditions, the operator monitors operation of the system using symbolic circuit at MCR/ECR safety panels (Figure 2.1.5.9).

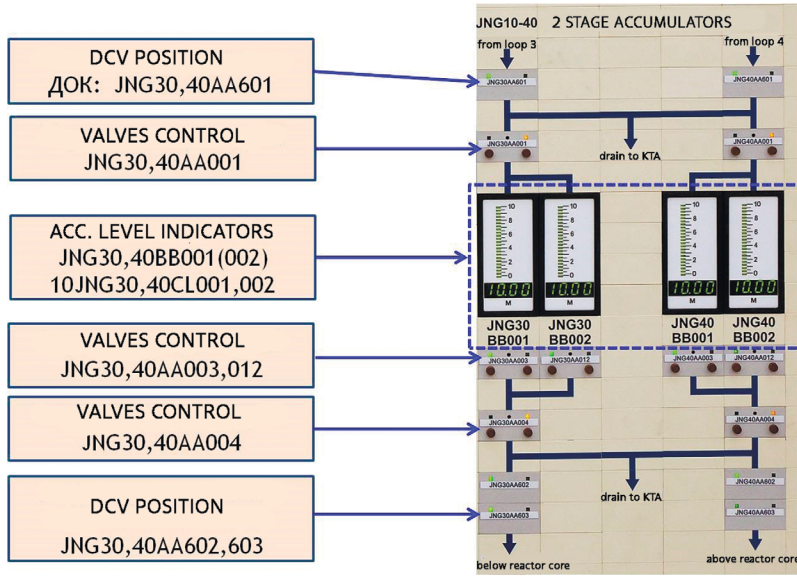


Figure 2.1.5.9 – Symbolic diagram of 3 and 4 trains of II stage accumulators at MCR safety panels

2.1.5.5 Operating modes of the system

Normal operation of the system

In reactor heating-up mode an operator connects II stage accumulators to the primary circuit at more than 3,3 MPa (excess) pressure above the core after closing the double check valve.

When the reactor operates at power, II stage accumulators are in standby. The double check valve is closed. Shut-off valves are opened in connection lines between the accumulators and the main circulation line except for 10JNG10 (20,30,40)AA003, 10JNG10(20,30,40)AA012 valves (see Figure 2.1.5.8).

To maintain the system in working condition, periodic checks and inspections of the elements are carried out.

When the reactor is in scheduled cooling down after the primary pressure drops below 3,5 MPa, personnel must isolate RCPFS accumulators from the primary circuit.

In refueling mode at shutdown reactor boric acid solution in the II stage accumulators is used to fill the upper part of the reactor vault and internals inspection cavity (the accumulators are drained to FAL system tanks and then pumped with FAL pumps to the refueling pool).

When spent fuel handling is finished, the II stage accumulators are filled with boron solution with the help of FAL system pumps. The boron solution is drained from different sectors of the refueling pool: lower part of the internals inspection cavity, fuel handling wells (tank sector of the spent fuel pond, upper part of the reactor vault, upper part of the internals inspection cavities).

System operation during large break in the primary circuit (Large)

During an accident with large break in the primary circuit as pressure is decreasing above the reactor core, I stage accumulators are connected first (when pressure drops below 5,89 MPa) and then II stage accumulators (when pressure is 1,9 — 1,5 MPa).

When double check valve opens, II stage accumulators supply boric acid solution to the reactor by means of hydrostatic fluid column. There are two supply modes possible:

— coolant recirculation «II stage accumulators – primary circuit – I stage accumulators» when the coolant level in the reactor vessel is higher than the cut-in place of the pressure balance line;

— draining of II stage accumulators when the coolant level in the reactor vessel is lower than the cut-in place of the pressure balance line. In figure 2.1.5.10 there is a flow rate diagram of three trains of II stage accumulators at maximum design-basis accident.

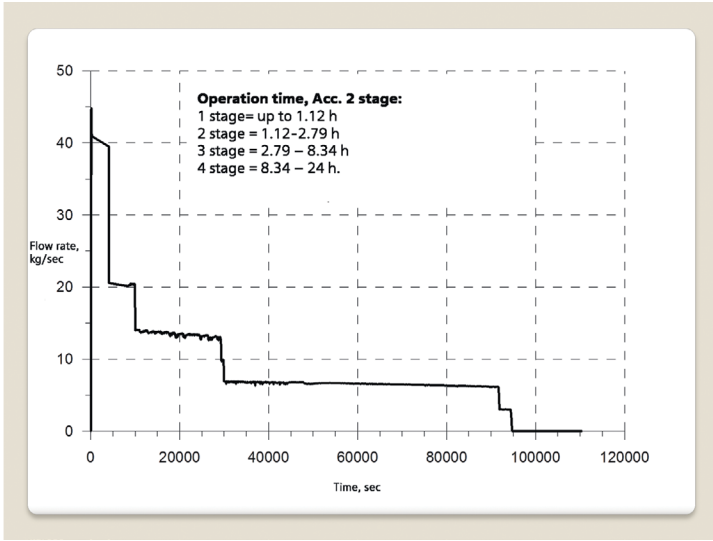


Figure 2.1.5.10 – Total flow rate of three trains of II stage accumulators to the reactor at LBLOCA

System operation during large break in the primary circuit (LBLOCA) with total failure of ECCS active part.

With total failure of ECCS active part, safety function to maintain reactor coolant supply is performed by its passive system: I and II stage accumulators as well as SG passive heat removal system.

I stage accumulators provide reactor core flooding at the initial stage of an accident.

II stage accumulators provide further supply of borated water to the reactor to remove residual heat in evaporating mode. Additional function for the II stage accumulators is automatic passive removal of non-condensable gases of the primary circuit from the SG tubes which promotes improving of the heat exchange and provides for long-lasting operation of SG in steam-condensing mode. The main non-condensable gas in the reactor core is nitrogen dissolved in the water of 1 stage accumulators as well as radiolytic oxygen and hydrogen.

Design analysis of accident with full cross section rupture of the main circulation line Dn 850 at the reactor inlet and subsequent failure of ECCS active part demonstrates (Figure 2.1.5.11):

— in this mode there is maximum initial loss of reactor coolant;

— cooperation of 3 trains of I-II stage accumulators and SG passive heat removal system provides for guaranteed removal of residual heat during ~ 24 — 26 hours (time for drainage of II stage accumulators);

— the fuel temperature grows above 1200 °C after depletion of water in II stage accumulators.

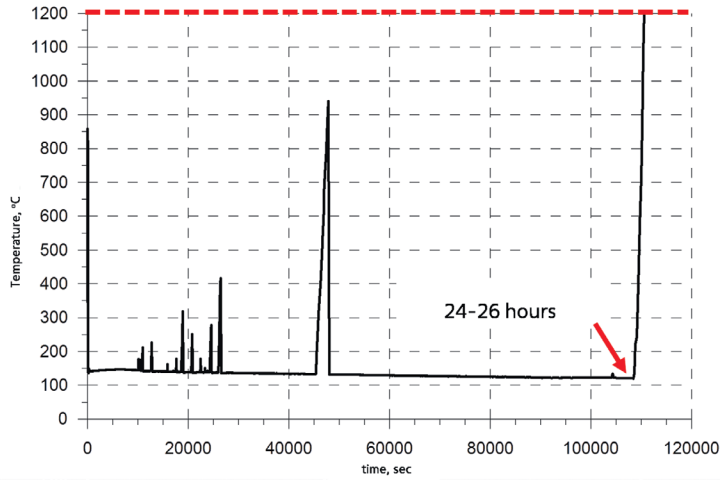


Figure 2.1.5.11 – Time-temperature history of fuel cladding during full cross section rupture of the main circulation line with subsequent failure of ECCS active part

To perform safety function on maintaining reactor coolant supply in the reactor after 24 hours there are additional measures with alternative equipment designed (self-contained diesel generator, alternative intermediate circuit with a cooling tower) and additional pipeline from the 1 channel of spent fuel pool system (FAK) to the drain line of II stage accumulators.

2.1.6 EMERGENCY OFF GAS SYSTEM (KTP)

2.1.6.1 Function, composition and process flow diagram of KTP system

The emergency off gas system is designed to perform the following functions:

— to remove steam-gas mixture from the equipment of the reactor island (reactor, pressurizer and SG headers);

— to reduce pressure in the primary circuit together with pilot operated relief valve (PORV) in order to mitigate consequences of design-basis and beyond-design basis accidents;

The steam-gas mixture is removed:

— to prevent creation of explosive mixture in the primary circuit equipment;

— to avoid failure of the coolant circulation in the primary circuit in natural circulation mode.

The system is supplied by the I group emergency power supply sources and performs its functions in all unit operating modes including emergencies.

The equipment of the emergency off gas system is located inside the containment of the reactor building UJA

KTP system is a set of electric driven shut-off valves with connecting pipes located between the equipment of the reactor island (reactor, steam generators, pressurizer) on the one hand and the pressure relief tank (BB) and the containment on the other one (Figure 2.1.6.1).

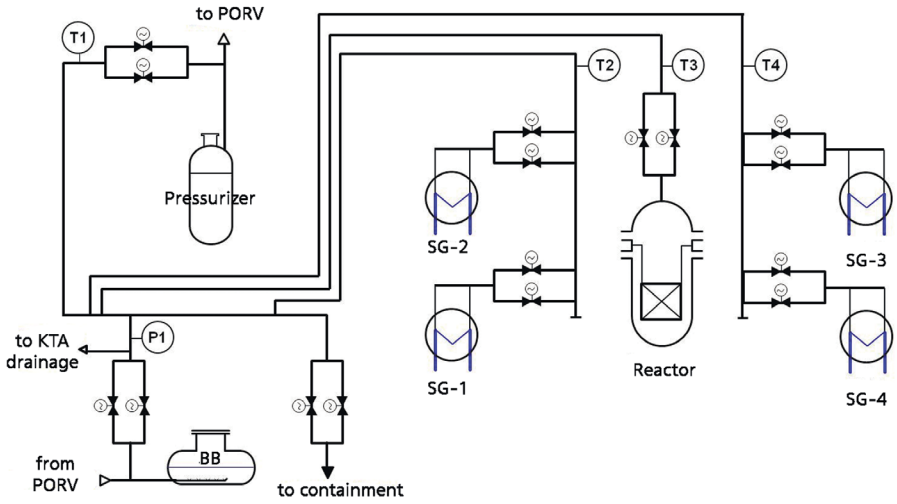


Figure 2.1.6.1 - Process flow diagram of KTP system

Reliability of the system performance is ensured by the valve redundancy adopted in the design. The valve sets consist of two tandem shut-off valves installed in the pipelines from each SG, in the pipelines from the reactor and pressurizer as well as in front of the pressure relief tank and before dumping the steam-gas mixture to the containment. And to perform system functions it is enough to open either of two tandem valves in the appropriate discharge lines.

To drain possible leaks of the reactor coolant through the shut-off valves, a drainage line is provided to KTA controlled leakage collection system. The leaks are monitored using a pressure gauge in the common pipeline for venting the steam-gas mixture.

The design of emergency off gas system for Novovoronezh NPP-2 has the following differences from the system design for V-320 reactor:

— redundancy of valves in discharge lines from the reactor and pressurizer is harmonized with redundancy of valves in discharge lines from steam generators (there are two tandem valves in each discharge line in the design);

— discharge lines connected pairwise downstream the reactor and steam generators are laid separately upstream the pressure relief valve;

— in the room with pressure relief tank there is an additional discharge line directly to the containment bypassing the pressure relief tank;

— there is pressure monitoring in the pipelines between the valves and temperature monitoring to monitor leaks in the shut-off valves.

2.1.6.2 Monitoring and control of the system

The following parameters are measured to get the information about the emergency off gas system:

— metal temperature of pipelines for steam-gas mixture discharge from the reactor, pressurizer and SG headers;

— pressure in the common pipeline for steam-gas mixture discharge.

System valves are controlled by an operator remotely from MCR or ECR. And the tandem valves are controlled in two independent safety trains. The symbolic circuit of the second channel of KTP system is given in Figure 2.1.6.2.

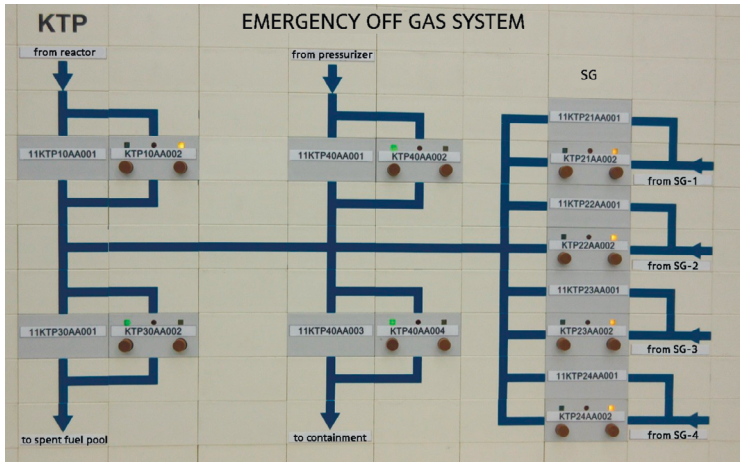


Figure 2.1.6.2 – Symbolic circuit of KTP system at MCR panel (second train)

2.1.6.3 Operating modes of the system

In normal operation modes except for reactor scheduled cooling down the system is in standby: all shutoff valves are closed, the pipelines between discharge valves from the reactor to the pressure relief tank and the containment are filled with nitrogen.

In reactor scheduled cooling down mode an operator uses the system for cooling the reactor head and SG headers by opening appropriate lines from the equipment to the pressure relief tank according to the reactor operating procedure.

The emergency off gas system can be used in cooling down modes in case of power supply loss to cool the upper part of the reactor and avoid sticking of the reactor head temperature.

In emergency modes an operator can use the system to remove non-condensable gases from the top of the primary circuit (reactor, SG headers, pressurizer).

In beyond-design basis accidents with reactor core melt the system is designed to reduce pressure in the primary circuit up to 1MPa by releasing the medium from the pressurizer to the containment.

2.1.7 STEAM GENERATOR EMERGENCY COOLING SYSTEM (1 JNB10-40)

SG emergency cooling system 1 JNB10-40 is an active system of SG cooling down. It is designed to:

— remove residual heat of the reactor core and cool down the reactor in emergencies with loss of power supply or failure of normal heat removal by the secondary circuit including leaks in steam lines and SG feed water lines;

— remove residual heat of the reactor core and cool down the reactor in emergencies with primary circuit leaks including break of the primary circuit line and leak from primary to secondary circuit.

SG emergency cooling system contains two channels: 11JNB10 and 12JNB30 (Figure 2.1.7.1).

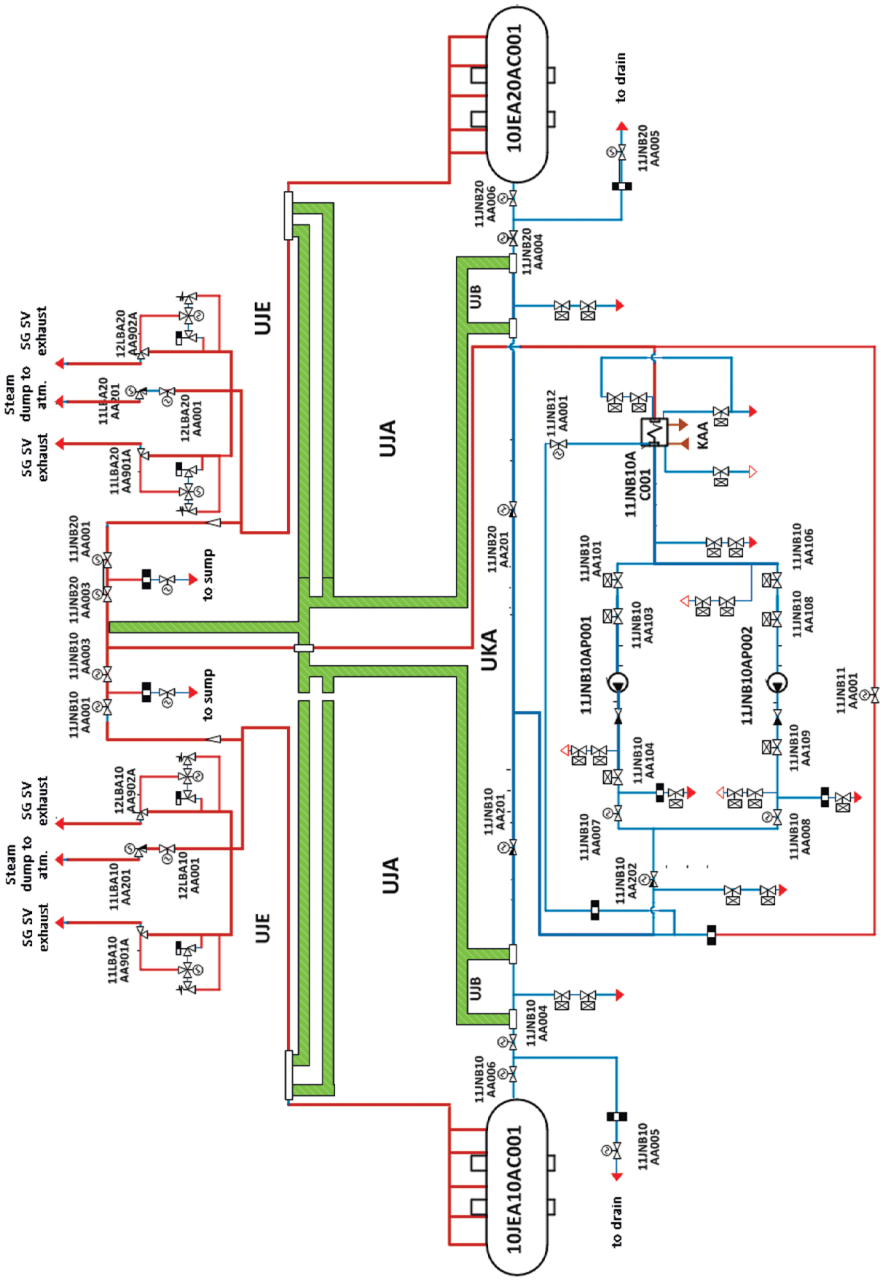


Figure 2.1.7.1 – Steam generator emergency cooling system

Each system channel is connected to two steam generators accordingly SG 1,2 and SG-3,4.

Each channel includes communication lines with steam lines of two steam generators, emergency cooling heat exchanger 11(12)JNB10(30), emergency cooling pumps and pipelines of condensate return to appropriate steam generators.

There are quick acting shut off valves installed in the steam supply pipelines from each steam generator to emergency cooling heat exchanger.

There are check valves installed in pressure pipeline to protect pumps from the medium backflow. To check the pump operation within the whole working range there is a test line with a throttle unit and shutoff valves.

To remove heat from emergency cooling heat exchangers (process condensers) the cooling water is supplied from appropriate channels of the intermediate circuit of essential consumers of reactor building 10UJA (1 KAA).

The system of remote and automatic control supports necessary actions to make system functioning and monitor the state of its elements.

The system 1 JNB10-40 has two channels, a failure in one channel does not lead to a failure to perform set functions in another channel.

The pumps of SG emergency cooling system JNB10,30 are redundant, one pump is working, another is backup. A failure of a pump of SG emergency cooling system does not lead to a failure of the system to perform its functions because the pump which stayed in operation is able to supply water to the SG.

The equipment, valves and pipelines in the system 1 JNB10-40 are made of corrosion—resistant austenitic steel.

In all modes that require operation of the system, the heat is removed to the final absorber (spray pools) along the following route: reactor — steam generator — system 1 JNB10-40 — intermediate circuit of essential consumers in 10UJA 1 KAA building — cooling system of essential consumers 1 PE — spray pools.

The system of SG emergency cooling is an active part of the emergency heat removal via steam generators for both leak tight and leaky primary or secondary circuit.

In normal operation mode the system does not work. It is in standby and is regularly checked.

In emergencies with failure to remove heat via secondary circuit the active system of SG emergency cooling down as well as passive heat removal system remove heat equal to at least 2 % of reactor thermal power (64 MW) at nominal parameters of the reactor.

In emergencies with primary circuit break including rupture of the main circulation pipeline the system cools down steam generators and removes residual heat of the reactor core to cool the primary circuit and limit pressure growth in the containment due to reduction of energy release from hot steam generators to the containment.

The main components of the system are located in building 10UKA at elevation + 4,8 m.

The elements of 1 JNB10-40 system are safety important in terms of their safety impact. They belong to protective and localization safety systems in terms of their functions. All system elements belong to safety class 2 and are classified as 23L, 23.

Summary of the equipment

SG emergency cooling pump 11JNB10A, 12JNB30 is a centrifugal single-stage pump with a face seal of the shaft.

The emergency cooling heat exchanger is a vertical shell-and-tube heat exchanger with coiled heat transfer surface and single-flow along tube and intertube space (Figure 2.1.7.2a-b).

The heat exchanger is made of stainless steel.

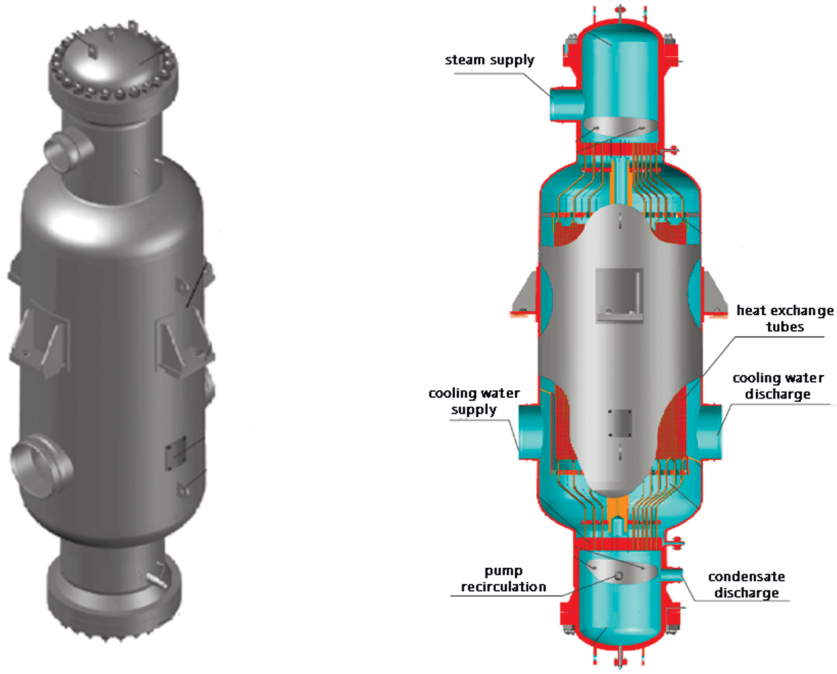


Figure 2.1.7.2a-b - Heat exchanger of emergency cooling down

The heat exchanger 11,12JNB10,30 consists of a bottom and upper collection chamber, two removable covers, a vessel, connection pipes, supports, slinging equipment.

Inside the vessel made of a shell ring and two bottoms there is an intertube space with cylinder multistart coils and a baffle plate in it dividing the volume between the shell ring and the intertube space in two parts.

The cooled medium (steam) moves from top to down along the intertube space (cylinder multistart coils, collection chambers).

The steam is supplied via the top connection pipe. It condenses in the tube space and condensate is drained through the bottom connection pipe.

The cooling medium is supplied to the vessel of the heat exchanger through the connection pipe. It comes to the intertube space through an opening in the low baffle plate. Then it moves up cooling the medium of the secondary circuit going through the coils and comes out through openings in the upper baffle plate and is drained via the connection pipe in the bottom of the vessel.

Technical characteristics of pumps and heat exchangers for SG emergency cooling 11JNB10,30 are given in table 2.1.7.1.

Table 2.1.7.1 - Technical characteristics of pumps and heat exchangers for SG emergency cooling 11JNB10,30

	Name of parameter	Value
The pump A125-NED-265-22-A8-FE		
	Pumped medium	2 circuit steam condensate
	Temperature of the pumped medium, °C, max	130
	Nominal suction head, m	80
	Rated feed, m^3/h	130
Cooling medium– intertube space		
	Medium type	Intermediate circuit of essential consumers (KAA)
	Flow rate, t/h	2500
	Temperature of the cooling water, °C:	
	at inlet	13÷46

	Name of parameter	Value
	at outlet	74 max
	Pressure of the medium, MPa	1,0
Cooled medium – tube space		
	Medium type	steam, steam-water mixture, water
	Flow rate, t/h	130
	Temperature (steam, water), °C	
	at inlet	296÷100
	at outlet	90 max
	Pressure of the medium, MPa	8,1

The systems of monitoring and control are provided to monitor, adjust and control the main processes in system 1 JNB10-40 and to maintain parameters representing processes within design limits.

Control of all electric driven elements, monitoring position (state) of the elements and process parameters as well as warning and emergency alarms is implemented at MCR and ECR in full (Figure 2.1.7.3.).

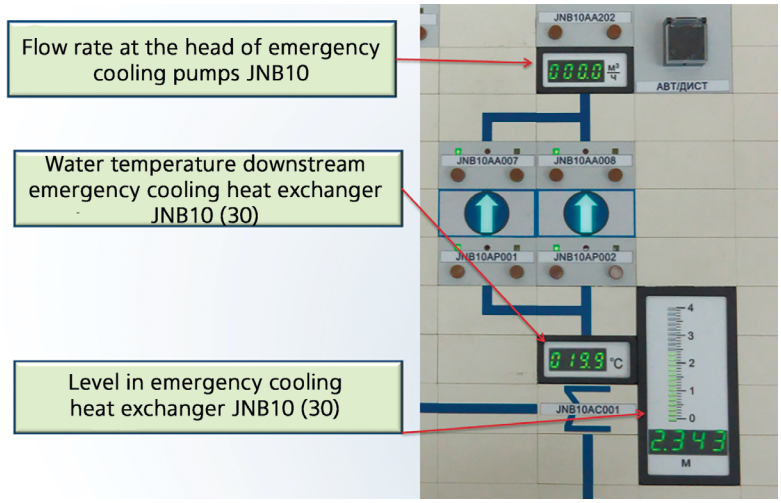


Figure 2.1.7.3 – Monitoring diagram of SG emergency cooling process parameters at MCR/ECR

For initiating events of an accident that require the system to be triggered, the design provides automatic connection of the system without operator's interference.

In addition to automatic control the remote control of system equipment and valves by an operator is also provided by the design. For this at MCR and ECR (safety panels and UULS) there are controls of SG emergency cooling system. They allow for remote triggering of any system channel to perform its safety function.

In emergencies the system is controlled remotely if its channel failed to start automatically as well as at the final stage of an accident when the unit was brought to a stable, safe, cooled down state and it was possible to withdraw excess equipment.

In normal operation the system of SG emergency cooling 1 JNB10-40 is in stand-by mode.

In standby mode the system channels 1 JNB10-40 shall be connected to the steam header. The valves shall be closed in the lines of condensate return to SG.

In standby mode the valves shall be open in the drain lines from the heat exchangers (for its warming up), and the level in the heat exchangers of the system 11,12JNB10,30 shall be maintained within 4 to 4,1 m. When the system starts operation, the valves close in drain lines from the heat exchangers.

1 JNB10-40 system operation in emergencies

Operation of 1 JNB10-40 system in emergencies is required to perform the following functions:

— remove residual heat of the reactor core and cool down the reactor in emergencies with loss of power supply and failure if normal heat removal by the secondary circuit including leaks in steam lines and SG feed water lines;

— remove residual heat of the reactor core and cool down the reactor in emergencies with primary circuit leaks including break of the primary circuit line and leak from primary to secondary circuit.

The system starts automatically by the following signals:

— leak in primary circuit (reduction of boiling margin in any hot leg of the loops up to 8 °C);

— loss of power supply of emergency bus;

— leak in secondary circuit (SG pressure drops up to 5,53 MPa at $\Delta T_{s,I-II} > 70$ °C and it rises further on up to 7,2 MPa in SG);

— leak in secondary circuit (uncompensated) (SG pressure drops up to 5,04 MPa, the temperature in the hot legs of the loops is more than 150 °C at $\Delta T_{s,I-II} > 70$ °C);

— signal coincidence: SG water level drops by 900 mm from H_{nom}, the temperature in the hot legs of the loops is more than 150 °C;

— leak from primary to secondary circuit;

— pressure growth up to 8,1 MPa in SGs.

In this case operating personnel shall follow the “Procedure on mitigation of design-basis accidents”.

Mode selection of SG emergency cooling regulator

The function of ESFAS «Mode selection of SG emergency cooling regulator» is designed to select necessary mode of SG emergency cooling regulator depending on emergency conditions.

There are the following modes of SG emergency cooling regulator:

— maintain SG constant pressure equal to 6,7 MPa;

— maintain SG constant cooling rate 30 °C/h;

— maintain SG constant cooling rate 60 °C/h;

— maintain temperature at the outlet of process condenser within 60 °C - 70 °C;

— maintain SG current pressure.

The mode selection switch of the regulator is located at MCR/ECR (Figure 2.1.7.4)

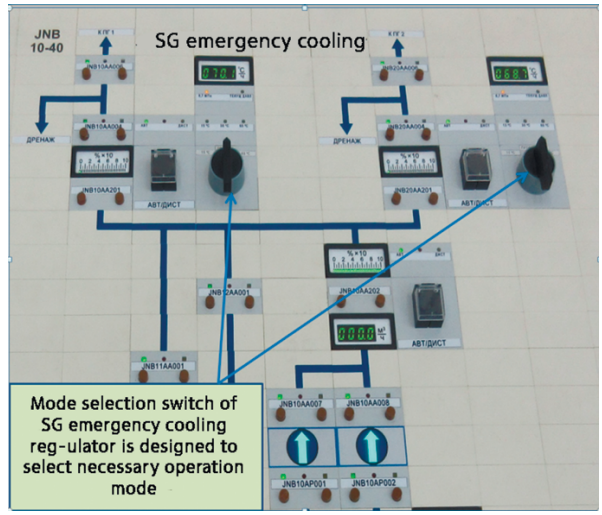


Figure 2.1.7.4 Mode selection switch of the regulator

The temperature at the outlet of the SG emergency cooling heat exchanger is maintained within 60 °C — 70 °C with the help of control valves located in the common pressure pipelines of SG emergency cooling pumps.

The rest of parameters of appropriate SG are maintained with the help of control valves located in the pipelines of condensate return to appropriate steam generator.

Additionally this function controls maximum cooling rate of the steam generator. When it exceeds 65 °C/h, the control valve is urged into closed position by the protection.

The regulator impacts the control valve to maintain constant temperature at the outlet of the process condenser if the temperature exceeds 75°C or if an operator switches the regulator to the automatic mode.

The regulator is switched off (transfer to remote mode) when the temperature is below 60 °C at the outlet of the process condenser.

The regulator impacts control valves to maintain other conditions on the signal «Actuation of emergency cooling» from BC12 ESFAS control function or if an operator switches the regulator to automatic mode.

The set point for SG constant pressure equal to 6,7 MPa actuates if there are no other signals on actuation of other set points.

The set point for SG constant cooling rate 30 °C/h is actuated by an operator and it acts if there are no signals «60 °C/h», «15 °C/h» and «Current pressure».

The set point for SG constant cooling rate 60 °C/ч actuates on the signal «Leak from primary to secondary circuit (T_1 > 220 °C)» or «Primary circuit leak» or it is actuated by an operator and it acts if there are no signals on actuation of set points «15 °C/h» and «Current pressure».

The set point for SG constant cooling rate 15 °C/h is actuated by an operator and it acts if there is no signals on actuation of the set point «Current pressure».

The set point for SG current pressure actuates on the signal «Leak from primary to secondary circuit (T_1 > 220 °C)» regardless of any signals on actuation of other set points.

2.1.8 PASSIVE DECAY HEAT REMOVAL SYSTEM (PHRS) 1 JNB50-80

2.1.8.1 PHRS configuration and functions

The system for emergency heat removal via the steam generators consists of two parts, i.e. the steam generator emergency shutdown cooling system as an active part presented above and the PHRS as a passive part.

The PHRS is a protective safety system based on the passive operating principle ensuring the decay heat removal from the reactor core via the secondary circuit.

The PHRS performs the specified functions during the design-basis and the beyond-the-design-basis accidents resulting in the need to passively remove heat from the reactor facility in order to keep it in a safe state.

The PHRS has four independent loops.

According to the accepted design specifications of the equipment three operable loops are sufficient for the system to perform its functions in full in any mode requiring its operation.

The PHRS equipment shall be capable of withstanding the loads imposed by the seismic impact of the safety shutdown earthquake level, impact of the external shock wave penetration and impact of the aircraft crash.

The PHRS ensures the continuous removal of decay heat (up to 2 % of the rated power of the reactor facility) during the accidents involving a loss of the trains of the SG ECS active safety system with normal or emergency power supply available and in the case of the beyond-the-design-basis accidents involving a loss of all the AC power sources.

The PHRS comes into operation by the action of the passive drive in 30 seconds after the loss of all the AC power sources.

In case of the primary coolant leakages the PHRS regulator automatically opens (in response to the signal “Difference between the saturation temperature and the temperature of the hot legs of the loops less than 8 °C”).

The trains of the system are physically separated and completely independent of each other, i.e. process parts, control systems, support systems, locations of equipment, pipelines, cables, control components, etc. are independent hereby the failure in one train cannot result in the failure in another.

These design and the equipment layout make it possible to rule out dependent failures and common-cause failures related to the location of the equipment, as well as the influence of any work performed on the equipment of one of the trains (repairs, maintenance) on another train.

The PHRS main components are as follows:

— steam-and-condensate path pipelines with valves,

— air ducts,

— heat exchange modules.

Each heat exchange module comprises the following:

— air gate valves (at inlet and outlet of the module);

— a heat exchanger,

— a regulator with a passive-action drive,

— a heat exchange module shell.

The air duct is intended to transport heated air from the PHRS heat exchangers to the outlet header and is designed as a collapsible structure made of individual links and thermally insulated on the outside (Figure 2.1.8.1).

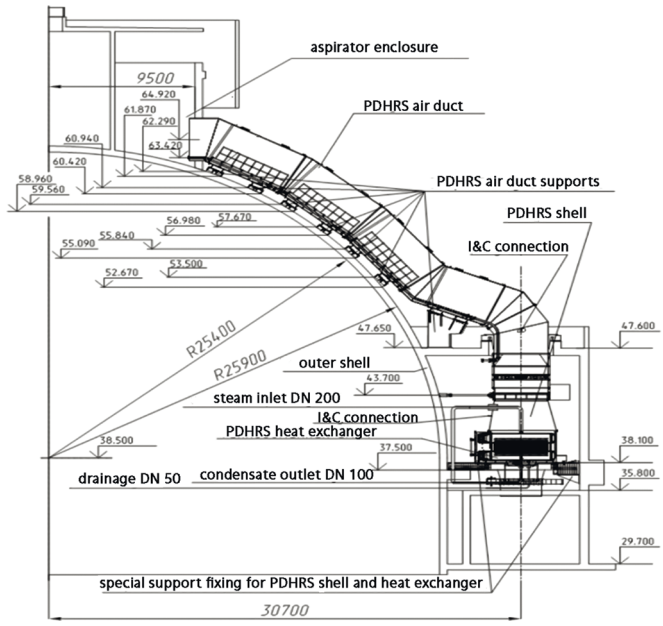


Figure 2.1.8.1 PHRS air duct cross-section

Figure 2.1.8.2 shows the cross-section of the heat exchanger.

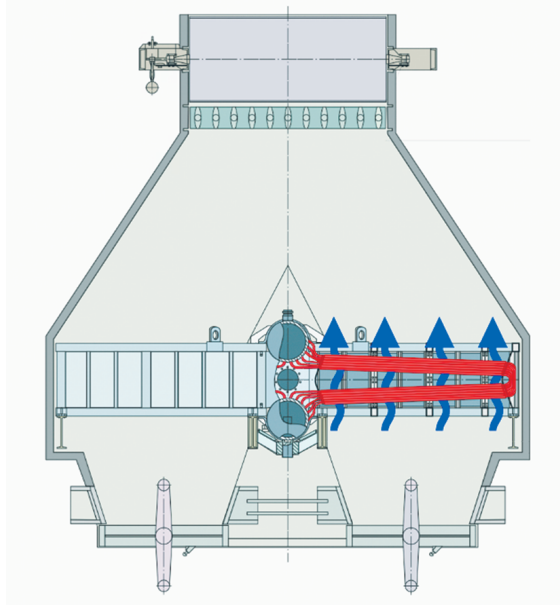


Figure 2.1.8.2 PHRS heat exchanger cross-section

The heat exchange module shell is intended to organize the air flow through the PHRS heat exchanger and to reduce the heat loss in rooms accommodating the heat exchange modules.

The heat exchange module shell is a collapsible structure made of metal plates and thermally insulated on the outside.

Figure 2.1.8.3 shows a general view of the PHRS heat exchange module.

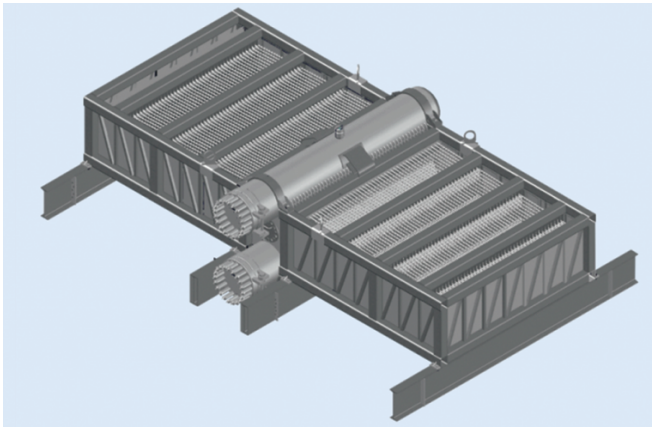


Figure 2.1.8.3 General view of the PHRS heat exchange module

The steam from the steam generator is condensed in the PHRS heat exchanger during the system operationas a result of its cooling by atmospheric air.

The technical characteristics of the PHRS heat exchanger are presented in Table 2.1.8.1.

Table 2.1.8.1 Technical characteristics of the PHRS heat exchanger

Name	Quantity, pcs	Basic technical characteristics of the equipment
Passive cooling system heat exchanger	8	Thermal capacity: 8MW (cooling temperature + 38 °C); Operating medium: — on tube side: steam, condensate, — on shell side: air. Air pressure: atmospheric. Pressure in the steam and condensate path: 8,1 MPa. Temperature: — steam/condensate: 300 °C, — air at inlet/outlet of the heat exchanger: 38/300 °C.

The air gate valves are installed at the heat exchange module inlet upstream the heat exchanger and at the heat exchange module outlet downstream the heat exchanger (Figure 2.1.8.4).

At the heat exchange module inlet two single air gate valves are installed, and at the outlet of the module one double gate valve is installed.

The gate valves are designed to maintain the integrity and thermal insulation of the heat exchanger which is in the waiting mode under the normal operation of the reactor facility. The upper and the lower gate valves are equipped with the electromagnets that keep them closed.

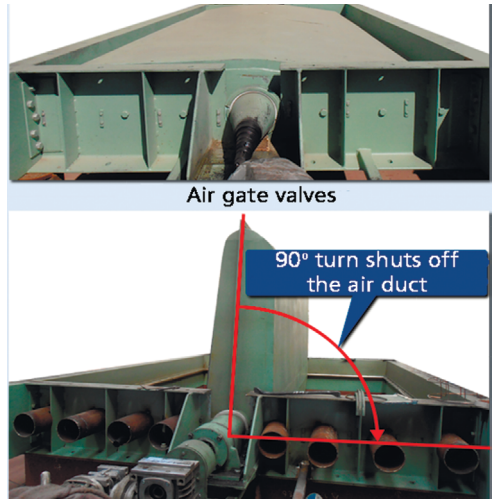


Figure 2.1.8.4 Air gate valves of the heat exchange module

The regulator is equipped with two drives, i.e. a passive-action drive and a back-up active drive.

The active drives are powered from the emergency power supply system of category 1 which makes it possible for the operator to control the regulator.

The regulator operation is ensured by the passive-action drive operated by the secondary circuit fluid.

The regulator has the following drive fluid pressure-dependent operation algorithm:

— at a pressure of (6,75+0,1) MPa and more the gate blades are in the open position,

— at a pressure drop of between 6,751 and 6,5 MPa, the gate blades are in the open position or start to close,

— at a pressure drop of between 6,5 and 5,8 MPa, the gate blades are closing,

— at a pressure of 5,8 MPa and less the gate blades are in the closed position,

— at a pressure increase of between 5,8 and 6,05 MPa the gate blades are in the closed position or start to open,

— at a pressure increase of between 6,05 and 6,75 MPa the gate blades are opening.

The regulator gate is capable to be remotely and locally controlled, the gate blades being opened at whatever secondary circuit pressure.

The time of the full opening of the gate blades in the remote control mode is 15 s or less.

The passive-action drive is designed for long-term operation during an accident. The passive-action drive of the regulator is a piston mechanism. The piston is made both of the steel bellows with a rod and a helical spring unit.

The piston rod is moved by the action of the steam pressure onto the piston on one side and by the action of the spring on the other side.

The rod movement direction and accordingly the regulator opening or closing depend on the ratio of the forces created by the spring and by the steam pressure onto the rod.

The PHRS includes a leak detection system intended for performing information functions.

The PHRS leak detection system consists of the acoustic sensors with the waveguides (16 waveguides, 2 waveguides on each PHRS shell).

The steam leakage of the PHRS heat exchanger is accompanied by a specific sound which is detected by the acoustic sensors.

The PHRS leak detection system has three operating modes two of which are system self-diagnostics modes and one of which a leak detection mode.

In the Leak detection mode (permanent system operation mode) the readings of all the acoustic sensors are compared with the preset threshold, and if it is exceeded the timer which measures the duration of the leak signal is activated. When the timer exceeds the preset value of the leak signal duration the date and the time of the onset of the leakage are entered into the controller memory, and the leakage onset alarm is shaped for the upper level of the computer-aided plant control system.

The PHRS is a safety-related system and, by the nature of its functions, a safety protection system. The system components are categorized as safety class 2 according to NP¬001-15. According to NP-089-15 the piping and the heat exchangers of the system belong to group B. The classification code is 2ZL.

No system functioning is required during normal operation and the PHRS is in the waiting mode.

In this mode the air gate valves are closed and the regulator position depends on the magnitude of pressure in the respective steam generator.

In emergencies the PHRS design permits two operating modes of the system:

— steam generator pressure maintenance within the range of 5,8 to 6,75MPa,

— the reactor facility shutdown cooling.

In response to the initiator the PHRS automatically starts to operate, the air gate valves open, the heat exchangers reach the design capacity, and the regulator keeps the secondary circuit pressure in the range of 5,8 to 6,75 MPa.

The regulator is changed over to the reactor shutdown cooling mode by the operational personnel.

The design thermal power of one PHRS heat exchanger is 8 MW and it decreases with the steam generator pressure decrease (as the reactor facility is cooled down).

The emergency power supply system of the first category provides the motor-driven components of the PHRS with electric power under the emergency conditions requiring the PHRS operation in the steam generator shutdown cooling mode.

The steam generator pressure maintenance mode is implemented by the passive self-operated (own fluid-driven) PHRS regulators. In such a case the power supply is not required.

The PHRS heat exchange modules (located at the elevation of + 36,55) with the valves are located in the free-access area in building 10UJA. In the radiological controlled area in building 10UJA the valves of the system are located.

2.1.8.2 Operating modes of the PHRS 1 JNB50-80 system and equipment

In **normal operation** the air gate valves are closed while the regulators open. The system is under standby operating conditions.

When the electromagnets are de-energized in response to the engineered safety feature actuation system (ESFAS) signals or an operator command the gates open by gravity (the gate blade is an eccentric workpiece) and the PHRS starts to operate.

Operation of the system in emergency modes

The system can automatically start operation in case of a failure of the respective train of the emergency shutdown cooling system (1 JNB10-40) to maintain the steam pressure in the secondary circuit.

Operation of the system during beyond-the-design-basis accidents

PHRS operation in case of the loss of all the AC power sources with intact primary circuit

When the 0.4 kV section of the second category emergency power supply system is deenergized the air gate valves open after a delay of 30 s.

The natural draught makes atmospheric air which cools the PHRS air heat exchangers to move through the air draught shafts.

When the steam pressure in the steam generator is greater than nominal the regulator remains open.

When the steam pressure in the steam generator decreases down to 6,5 MPa by the action of the secondary circuit steam pressure-actuated passive drive the regulator dampers start to close, and they fully close at a steam pressure of 5,8 MPa in the steam generator.

If the secondary circuit pressure is further reduced the regulator remains closed.

When the steam pressure in the steam generator increases up to 6,05 MPa the regulator dampers start to open, and the regulator is fully open at a pressure of 6,75 MPa.

When the secondary circuit steam pressure increases further the regulator remains open.

When the PHRS is reaching the rated power the decay heat is mainly removed with steam dumped via the atmospheric pressure relief valve.

When it is impossible to replenish the steam generators without an AC power supply sources the steam generator level decreases.

When the level decreases by 900 mm below the nominal value the quick-closing isolation valves and the motor-operated shut-off valves automatically close on interlock.

The atmospheric pressure relief valves are disconnected from the pressure source and also close. Therefore the water inventory in the steam generator is sufficient for the PHRS operation, and further decay heat removal is carried out through the PHRS operation.

The regulator maintains the secondary circuit pressure in the range between 5,8 and 6,75 MPa. At the time of the quick-closing isolation valve closure the decay heat is no longer more than 2 % of the reactor rated power, which prevents the steam generator safety valves from operating.

The reactor facility can be kept by PHRS under the hot standby conditions during the time necessary for the recovery of the AC power supply sources but not less than for 24 hours from the onset of the accident.

If necessary, the PHRS can be changed over by an operator to the reactor shutdown cooling mode by opening the regulator supplied from the first category emergency power supply system.

PHRS operation in case of loss of all the AC power supply sources with steam generator steam line rupture within or beyond the containment boundaries

When the initiator occurs, the reactor scrams and the quick-closing isolation valves (BZOK) on the damaged steam line close on interlock. The quick-closing isolation valves also close and the heat exchange modules also start operating on three intact loops after the time delay determined by the pressure reduction rate in the same way as described in the previous subsection. After the quick-closing isolation valve closure on the intact steam generators the PHRS maintains the

reactor parameters of the hot standby mode through operation with three intact loops connected.

The duration of this mode of the reactor is unlimited.

The location of the rupture of the steam line is relevant only for the parameters within the containment.

PHRS operation in case of loss of all the AC power supply sources with primary circuit rupture (small and large leaks)

When the 0,4 kV section of the second category emergency power supply system is de-energized, the air gate valves open and PHRS starts operating after a delay of 30 s.

At the first stage of PHRS operation the steam generator is cooled down only on the secondary side and its duration depends on the outdoor temperature. The PHRS is not involved in the decay heat removal at this stage. This task is performed by the hydraulic accumulators used to replenish the core. The quick-closing isolation valves close only if the level in the steam generator lowers (for example, due to the steam leakage to the auxiliary header, etc.).

At the second stage of PHRS operation (when the temperature of the secondary circuit becomes lower than the temperature of the primary circuit) the PHRS on the intact loops change over to the mode of primary circuit steam condensation and condensate return to the core. Without the sprinkler system the pressure in the containment is practically equalized with the pressure in the primary circuit. Over time the changes in the daily temperature of the outdoor air create the conditions under which the condensing power of the PHRS can temporarily exceed the steam generating power of the core. During this period a reverse steam flow is formed through the rupture, which will lead to a partial return of the coolant from the containment to the primary circuit.

The operation of the steam generator in the steam condensation mode is associated with the arrival of non-condensable gases in the tube bundle.

In case of the loss of all the AC power supply sources with the rupture of the primary circuit the 1 JNB50-80 passive decay heat removal system operates together with the 1 JNG1040 first-stage hydraulic accumulator system.

If the initiator with a small leak occurs the PHRS of which the purpose is to reduce the pressure down to these points of actuation of the first- and second-stage hydraulic accumulators is involved both in the decay heat removal and reactor shutdown cooling. The PHRS performs the same task in case of the primary-to-secondary circuit leaks (for example, leaks due to the loss of integrity of the steam generator header covers).

In order for the passive decay heat removal system to be operable in case of the beyond-the-design-basis accidents, the following related systems should be in operation:

— steam generator systems (1 JEA),

— the first category emergency power supply systems,

— I&C systems.

In order for the passive decay heat removal system to be operable, the related systems should meet the following conditions:

— no leakage of medium (steam or water),

— the steam generator should be equipped with a separate connection to receive the PHRS condensate, and the end of the condensate line should be brought in under the water level at the depth of the lower third of the tube bundle,

- the PHRS steam lines should be connected to the steam headers of the steam generators on the side of the their blind ends.

The first category emergency power supply system supplies electric power to the motor-driven components of the 1 JNB50-80 system in case of the accidents requiring the PHRS operation in the mode of the steam generator shutdown cooling. The steam generator-pressure maintaining mode is implemented by the passive self-operated (own fluid-driven) PHRS regulators without the operation of the electrical system being required.

PHRS operation-related algorithms of the engineered safety feature actuation system

Function BC13 — Engineered safety feature actuation system: "Full opening of the PHRS regulator"

The BC13 function ensures the PHRS change-over to the shutdown cooling mode by fully opening the regulator located in the air path downstream the PHRS heat exchanger.

The logic of shaping output commands and signals in control sets 1 and 2 is the same.

Function BC14 — Engineered safety feature actuation system: "PHRS initiation with the steam generator emergency shutdown cooling system failure to come into operation"

The BC14 function ensures the PHRS of a separate steam generator to come into operation when an active steam generator emergency shutdown cooling system should operate and it does not perform its functions for this steam generator.

The logic of shaping output commands and signals in control sets 1 and 2 is the same.

Provision is made for active and passive initiation of the PHRS.

Passive initiation of the PHRS (de-energization of the electromagnets keeping the air gate valves closed) is carried out when the uninterruptible power supply section de-energizes and the power supply fails to restore within 30 seconds (the time sufficient for the diesel generator to start up and be stepwise loaded). The voltage in the de-energized electromagnet power supply circuit is maintained for 30 seconds by a capacitor.

The BC14 control function under consideration assumes the active initiation of PHRS operation.

If there is a command from the TXS hardware or from the alternative hardware, the AC circuit feeding the electromagnets which keep the air gate valves closed is opened.

In order to prevent spurious initiation of the PHRS during bringing the steam generator emergency shutdown cooling system to design performance, the delayed opening of the circuit for 118 seconds is provided for.

2.1.9 PRIMARY CIRCUIT OVERPRESSURE PROTECTION SYSTEM

2.1.9.1 Purpose and structure of the system

The primary circuit overpressure protection system is intended for protection of the reactor facility equipment and pipelines against overpressure in the primary circuit during the design-basis and beyond-the-design-basis accidents through the operation of the pilot-operated relief valve of the pressurizer installed on the line for steam discharge from the pressurizer steam space into the relief tank.

Throughput, number of the pilot-operated relief valves and their opening setpoints were determined such that under the design conditions at any point of the protected equipment the pilot-operated relief valve operation pressure does not exceed the rated one by more than 15 % taking into account the response time and transient dynamics.

The pressure adopted for the primary circuit by the designer as the rated one (the maximum excess pressure in the equipment or pipelines used during strength analysis when choosing the main dimensions) is 17,64 MPa. In addition it was determined that the rated pressure is equal to the respective operating pressure.

The criterion for the pressurizer pilot-operated relief valve to perform its functions during the design operation of the reactor emergency protection system is that the primary circuit pressure does not exceed 115 % of the rated one, i.e. 17,64 x 1,15 = 20,29 MPa.

The criterion for the pressurizer pilot-operated relief valve to perform its functions during the beyond-the-design-basis accidents of the ATWS type is that the primary circuit pressure does not exceed 135 % of the rated one, i.e. 17,64 x 1,35 = 23,81 MPa.

In case of the severe accidents the primary circuit pressure should not exceed the pressure of the primary circuit hydraulic test for strength, i.e. 17,64 x 1,39 = 24,5 MPa.

The number of the pilot-operated relief valves to be installed was determined according to the n+1 principle (where n is the number of the pilot-operated relief valves ensuring the discharged medium flow-rate necessary for the pressure reduction) assuring that the primary circuit pressure would not exceed the operating pressure by more than 15 % even in case of the failure of the pilot-operated relief valve with the least opening setpoint. Thus, the structural redundancy of the pressurizer pilot-operated relief valves is determined by the relationship 3 x 50 %.

The turbine stop valve closure mode accompanied by a maximum increase in the primary circuit pressure was chosen as the design mode for determining the throughput of the pilot-operated relief valves.

2.1.9.2 Process flow diagram of the system

The primary circuit overpressure protection system consists of the following:

— three pressurizer pilot-operated relief valves which can operate on both active and passive operating principles,

— discharge pipelines from the pressurizer to the pilot-operated relief valves with the fastening components,

— instrumentation.

The pressurizer pilot-operated relief valves are a part of the pressurizer system (JEF) and are located in the reactor building (UJA) at the elevation of + 29,41.

Figure 2.1.9.1 shows the process flow diagram of the pressurizer pilot-operated relief valves.

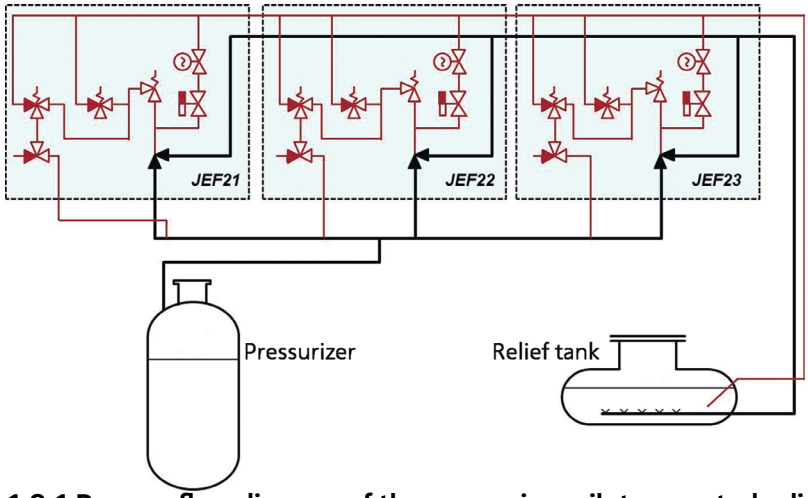


Figure 2.1.9.1 Process flow diagram of the pressurizer pilot-operated relief valves: JEF21-23

The first pilot-operated relief valve is the reference one and ensures that the excess pressure of the operating medium is released from the pressurizer into the relief tank when the opening setpoint of 18,01 MPa is reached. The other two pilot-operated relief valves are the operating ones ensuring that the excess pressure of the operating medium is released when the opening setpoint of 18,5 MPa is reached.

The components installed on the discharge pipeline (245x19 mm) are as follows:

— three 38 x 3,5 mm pipe connections to discharge steam via the pilot valves of the pilot-operated relief valves to the relief tank,

— three 133 x 14 mm pipe tees to discharge steam via the main valve of the pilot-operated relief valve into the relief tank,

— a 76 x 7 mm pipe connection to discharge the steam gas mixture from the pressurizer to the emergency gas removal system (KTP),

— three 18 x 2,5 mm pipe tees to connect to the gas blow-off system (KTB20),

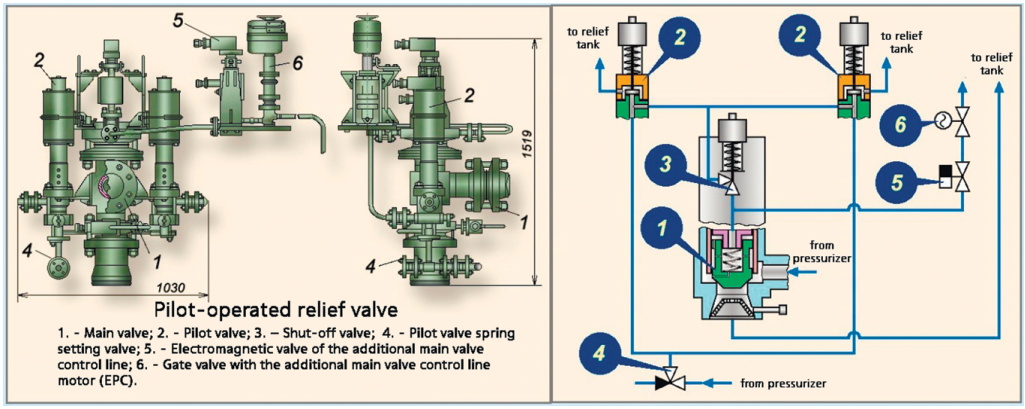
— a 14 x 2 mm pipe tee to extract steam to the sampling system (KUA).

To bear the weight loads and ensure the seismic stability of the discharge pipeline the rigidly fixed supports in the area of the pilot-operated relief valves as well as the hangers, the braces and the hydraulic shock absorbers on the pipeline sections are used.

The design and the layout of the discharge pipelines make it possible to inspect and, if necessary, repair them.

2.1.9.3 Brief description of the design of the pressurizer pilot-operated relief valves

Each pilot-operated relief valve represents a modular structure and consists of the main valve with a piston drive, two pilot valves, impulse pipelines, a valve for the pilot valve spring setting from the external pressure source, one shut-off valve and an additional main valve control line with two gate valves in series, one of which has an electromagnetic drive and the other a motor drive (Figure 2.1.9.2).



1. - Main valve; 2. - Pilot valve; 3. – Shut-off valve; 4. - Pilot valve spring setting valve; 5. - Electromagnetic valve of the additional main valve control line; 6. - Gate valve with the additional main valve control line motor (EPC).

Figure 2.1.9.2 Exterior and basic diagram of the pressurizer pilot-operated relief valve UF50024-100-12

The main valve is the main component of the pilot-operated relief valve and is designed to discharge the operating medium from the primary circuit system into the relief tank. The valve operates according to the relief principle. In the standby mode the main valve is kept closed by the operating medium pressure. When the dome pressure is released the main valve opens. The dome pressure is released in the following cases:

— when any pilot valve opens (on the signal from the engineered safety feature actuation system or from the spring),

— when an operator opens the additional main valve control line valves.

In case of the pilot valve closure (the additional main valve control line valve closure) the main valve dome is filled with the operating medium via the orifice and it closes.

The pilot valve is intended for controlling the main valve opening by discharging the operating medium from the main valve drive into the relief tank and by shutting off this discharge in order to close the main valve. The pilot valve is a spring valve equipped with an electromagnetic drive for opening and closing. Normally, the pilot valve is controlled by the electromagnetic drive on the signals from the pressure sensors above the core.

The principle of control of the pilot valve electromagnetic drive is as follows: at an operating pressure in the primary circuit, in order to ensure the necessary gate tightness, a closing winding is incorporated in the pilot valve electromagnet. When the pressure above the core increases up to the pilot valve opening setpoint, on the command from the engineered safety feature actuation system, the closing winding is switched off and the opening winding is switched on. When the system pressure decreases down to pilot valve closing setpoint the opening electromagnet is switched off and the closing electromagnet is switched on to operate first under forced conditions, for closing the pilot valve, and then under the conditions of keeping the pilot valve closed.

When the electromagnetic drive fails to open or close, the pilot valves perform their function of protecting the primary circuit against excess pressure as the direct-acting spring-loaded relief valves, also with the electromagnetic drive switched on to operate for closing.

The shut-off valve is located between the main valve and the pilot valves and is intended to automatically or remotely disconnect the main valve from the pilot valve in case of the pilot valve failure to seat or if the pilot valve spuriously opens. In addition the shut-off valve allows the pilot valve springs to be checked and set without actuation of the main valve.

The shut-off valve is structurally integrated into the main valve cover, is normally open and has an electromagnetic drive which operates only for closing.

The additional control line is designed to remotely reduce primary circuit pressure down to 1 MPa from the main/emergency control room. Two gate valves are installed on this line, i.e. one with an electromagnetic drive, the other with a electro-motor drive. Both valves should be opened to release pressure.

The valve with an electromagnetic drive which operates only for opening is normally closed. In the normal position the electric magnet is de-energized, and the valve is kept closed with the spring.

The valve with a motor drive is the standby one, normally closed and remotely put into operation (opened and closed) from the main/emergency control room simultaneously with the gate valve with the electromagnetic drive.

The valve for the pilot valve spring setting from the external pressure source makes it possible to check and set the pilot valve springs at a pressure in the primary circuit of 0,7 of the operating pressure. This valve is included in the pilot-operated relief valve delivery set.

2.1.9.4 Technical characteristics of the pressurizer pilot-operated relief valve

Table 2.1.9.1 shows the main characteristics of the pressurizer pilot-operated relief valve and its settings.

Table 2.1.9.1 Pressurizer technical characteristics

Parameter name	Measurement unit	Value
Operating environment	—	Saturated steam, steam-water mixture, water
Nominal diameter of inlet/outlet branch pipe of the main valve	mm	100/200
Design (operating) pressure	MPa	17,64
Design temperature	°C	350
Design flow-rate at P=18,01 MPa: — saturated steam — water	kg/s (t/h)	 50 (180) 222 (800)
Full opening/closing time, not more than	s	1/5
Pressure of opening from electromagnet: reference pilot-operated relief valve	MPa	18,01
1st and 2nd operating pilot-operated relief valves	MPa	18,5
Closing pressure from electromagnet: reference pilot-operated relief valve	MPa	17,1
1st and 2nd operating pilot-operated relief valves	MPa	17,6

Parameter name	Measurement unit	Value
Closing pressure from spring: reference pilot-operated relief valve	MPa	16,6
1st and 2nd operating pilot-operated relief valves	MPa	17,0
Opening pressure from spring: reference pilot-operated relief valve	MPa	18,0
1st and 2nd operating pilot-operated relief valves	MPa	18,5
Full opening pressure from spring: reference pilot-operated relief valve	MPa	18,5
1st and 2nd operating pilot-operated relief valves	MPa	19,1

2.1.9.5 Control and monitoring of the system performance

The control system of the pressurizer pilot-operated relief valve is implemented as part of the unit engineered safety feature actuation system and is designed for the following:

— to shape control commands for the pilot valve and main valve,

— to receive signals from the position sensors of the gates of the pilot-operated relief valve from the main control room and emergency control room and to form information signals to the main control room, emergency control room and power unit digital control system.

The power supply to the pilot-operated relief valve control system is provided from the sources of uninterruptible power supply to the first-group consumers.

The pressure measurement for processing and delivering signals for controlling the electromagnetic drives of the controlled elements is carried out by the standard reactor measuring tools. The pressure is measured at the outlet of the core. The pulse is shaped according to the two of three principle.

Table 2.1.9.2 shows the pressure thresholds of the pilot valve opening/closing from the electromagnet and the shut-off valve actuation conditions.

Table 2.1.9.2 List of the monitoring and control system functions for the primary circuit overpressure protection system

Actuation condition	Setpoint	Action	Monitoring and control system
Increased excess pressure above the core, MPa, more than	18,01	Opening of the pilot valve of the reference pressurizer pilot-operated relief valve	Engineered safety feature actuation system BE11
Decreased excess pressure above the core, MPa, less than	17,1	Closing of the pilot valve of the reference pressurizer pilot-operated relief valve	Engineered safety feature actuation system BE11
Increased excess pressure above the core, MPa, more than	18,5	Opening of the pilot valve of the operating pressurizer pilot-operated relief valve	Engineered safety feature actuation system BE11
Decreased excess pressure above the core, MPa, less than	17,6	Closing of the pilot valve of the operating pressurizer pilot-operated relief valve	Engineered safety feature actuation system BA11
Coincidence of the signals: — excess pressure above the core, MPa, less than, — failure of any of the pilot valves of the pressurizer pilot-operated relief valve toclose	16,1	Closing of the shut-off valve with prohibited opening on this pressurizer pilot-operated relief valve (the setpoints for the reference and operating pressurizer pilot-operated relief valves are the same)	Engineered safety feature actuation system BA17
Presence of any of the signals: — excess pressure above the core, MPa, more than, — both pilot valves of the reference pressurizer pilot-operated relief valve are closed	18,01	Cancel the prohibition of opening and open the shut-off valve on the reference pressurizer pilot-operated relief valve	Engineered safety feature actuation system BA17
Presence of any of the signals: — excess pressure above the core, MPa, more than, — both pilot valves of the of the perating pressurizer pilot-operated relief valve are closed	18,5	Cancel the prohibition of opening and open the shut-off valve on this operating pressurizer pilot-operated relief valve	Engineered safety feature actuation system BA17

For the pilot valve and main valve, apart from automatic control, the manual remote control from the main or emergency control room is provided for.

The main valve can be opened by the operator by remotely opening the pilot

valve at a pressure in the pressurizer not less than 0,8 of the operating one.

The control of the additional control line valves is carried out only remotely by the operator using the key from the main or emergency control room.

The condition of the system is monitored by the operating personnel using the digital control system image 10JEF (Figure 2.1.9.3).

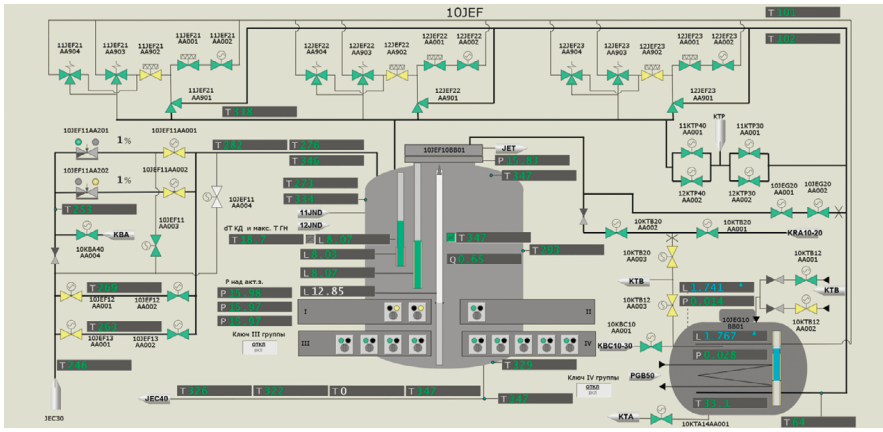


Figure 2.1.9.3 Digital control system image 10JEF

The measured parameters characterizing the normal functioning of the system (absence of leakages through the pilot valve and main valve) are the metal temperatures of the discharge pipelines downstream the pilot valve and main valve.

The mnemonic diagram of the reference pressurizer pilot-operated relief valve is presented on the main/emergency control room panels of the first safety train; the mnemonic diagrams of the operating pressurizer pilot-operated relief valves are presented on the panels of the second safety train (Figure 2.1.9.4).

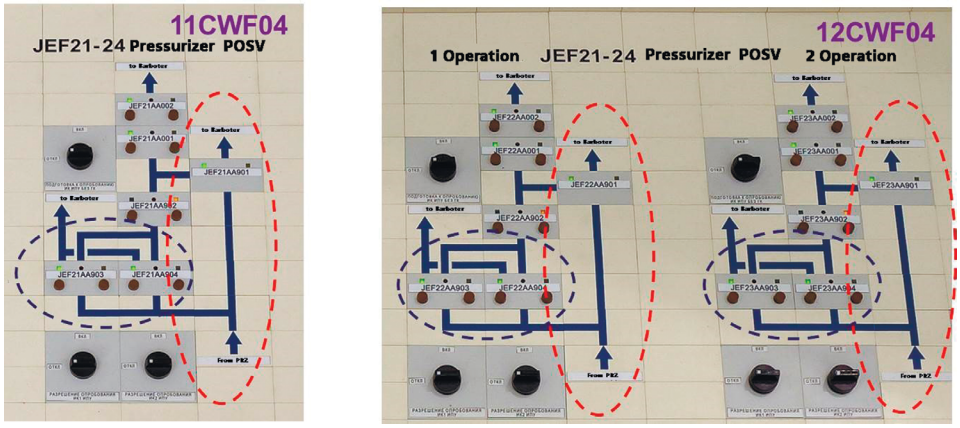


Figure 2.1.9.4 Mnemonic diagram of the pressurizer pilot-operated relief valve on the main control room safety panels

In order to make it possible to test the pilot valves on the mnemonic diagram of each pressurizer pilot-operated relief valve the following keys are provided for:

— “Pilot valve 1 of the pilot-operated relief valve is permitted to be tested” and “Pilot valve 2 of the pilot-operated relief valve is permitted to be tested”. The keys disable the command “Close” for the given pilot valve at low pressure above the reactor core (less than the closing setpoint).

— “Testing of the pilot valve of the pilot-operated relief valve without the main valve to be prepared”: when this key is activated by the operator the shut-off valve automatically closes.

The inspection of the pressurizer pilot-operated relief valve is carried out every four years.

2.1.9.6 Operating modes of the system

Operation of the system during normal operation

Primary circuit heating and hydraulic testing modes

Prior to the start of the primary circuit heating operations, the pressurizer pilot-operated relief valves should be operable and in standby mode.

In standby mode:

— the main valves are closed by the pressure of the medium,

— the pilot valves are closed and ready to operate on excess pressure pulse and as direct

— acting valves,

— the shut-off valves are open, their electromagnets are de-energized,

— the additional control line valves are closed.

During the primary circuit hydraulic testing for strength the hydraulic testing pressure is 24,5 MPa, which is higher than the actuation setpoint of the pressurizer pilot-operated relief valve. Therefore, in this mode the personnel should lock the pilot valve. The function of the primary circuit overpressure protection is then performed by the pressure relief valves of the hydraulic testing pump which are set to the actuation pressure corresponding to the pressure of 25 MPa above the core.

During the primary circuit hydraulic testing for integrity (hydraulic testing pressure equal to 17,64 MPa) it is also permitted to block operation of the pressurizer pilot-operated relief valve for the duration of the hydraulic testing.

Reactor facility operation at power

During reactor facility operation at power the pressurizer pilot-operated relief valves should be operable and in standby mode.

In case of a failure, the working technical guidelines for safe operation require the following:

1. If there is a failure of one pilot valve of one pressurizer pilot-operated relief valve, continue to operate the power unit at rated power and repair the defect during the next outage.

2. If there is a failure of one pressurizer pilot-operated relief valve, proceed immediately with the planned unloading of the power unit down to 50 % of rated power. Within 24 hours the operability of this pressurizer pilot-operated relief valve should be restored, otherwise the reactor should be properly brought to the cold condition.

3. If there failures of more than one pressurizer pilot-operated relief valves, the reactor should be properly brought to a cold condition as, taking into account a single failure, in this case the system does not ensure the accomplishment the safety function to protect the primary circuit against overpressure.

Operation of the system during the design-basis accidents

The designer provided for the multi-level protection of the primary circuit in case of the pressure growth above the core (Figure 2.1.9.5).

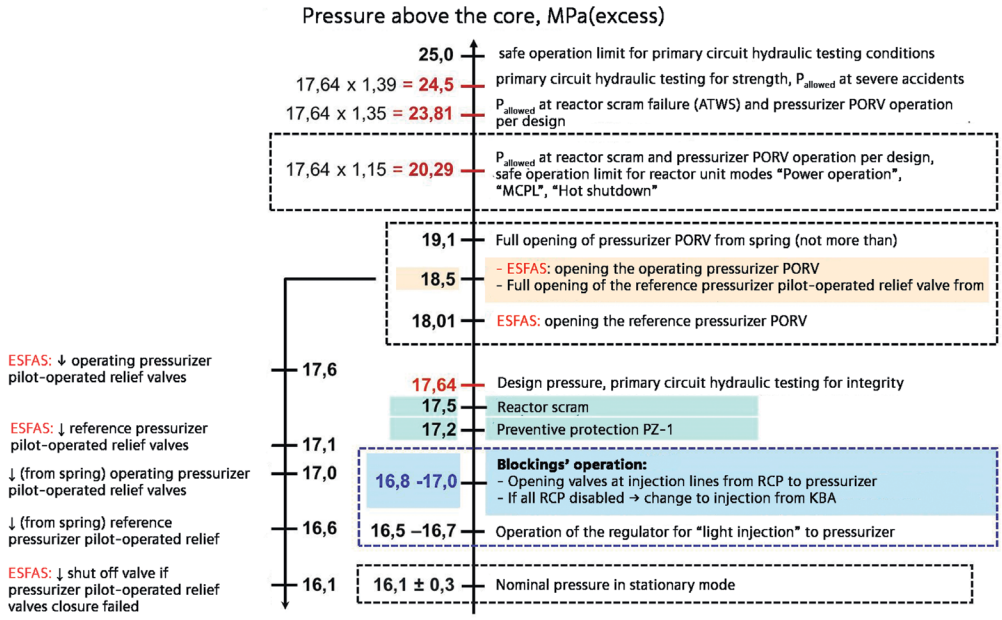


Figure 2.1.9.5 Operation of the safety systems at increased pressure above the core

In the case of minor perturbations, the pressure growth in the primary circuit should be compensated by the water supply to the pressurizer steam space via the injection lines, i.e. from the outlet of the main cooling pump of loop 3 or, in the case of the disconnection of all the cooling pumps, from the primary circuit makeup system pumps.

For a number of the postulated initiators during which the primary circuit pressure grows up, the designer provided for the proactive control, without waiting for the pressure buildup:

— unloading of the reactor from the accelerated preventive protection (APP) at a reactor power more than 75 % of the rated one and the proactive opening of the quick-acting pressure reducing plant of air discharge to the condenser in response to the turbogenerator load dropping signals: "Closing of 2 of 4 turbogenerator stop valves", or "Trip of the generator circuit breaker", or "Trip of the power unit circuit-breaker";

— unloading of the reactor from the accelerated preventive protection in response to the signals "Trip of 1 of 4 main cooling pumps and reactor power higher than 80 %" or "Trip of 2 of 4 main cooling pumps and reactor power higher than 75 % of the rated one";

— reactor scram according to the AA12 algorithms “Reduction of the primary coolant flow-rate”.

At a pressure above the core of more than 17,2 MPa the PZ-1 preventing protection comes into operation in response to the signal of the engineered safety feature actuation system (monitoring and control system function AC21), and the reactor is unloaded until the pressure drops below 16,95 MPa.

At a pressure above the core of more than 17,5 MPa, the reactor is taken subcritical by the emergency protection system.

If the pressure above the core increases further, first the reference and then the operating pressurizer pilot-operated relief valves open. If the pressure above the core decreases, they close in the reverse sequence.

Operation of the system in case of the severe accidents

The strategy of management of the severe accidents provides for the reduction of the emergency processes to the “low pressure scenarios” in the system reactor – containment.

The additional control line of the pressurizer pilot-operated relief valve makes it possible to remotely reduce the pressure in the primary circuit down to 1 MPa from the main or emergency control room.

The operator can also use the emergency gas removal system KTP in combination with the pressurizer pilot-operated relief valve to reduce the pressure in the primary circuit down to 1 MPa (through the discharge of the medium from the primary circuit to the containment).

2.1.10 SECONDARY CIRCUIT OVERPRESSURE PROTECTION SYSTEM

The secondary circuit overpressure protection system includes: Pilot Operated Relief Valves of Steam Generators (SG PORV), Steam Dump Relief Valves Discharged to Atmosphere (BRU-A), isolation electric driven valves in front of the BRU-A, Main Steam Line Isolation Valves (MSIV and isolation electric driven valves located behind them).

The steam pipelines from each Steam Generator pass through the reactor building containment 10UJA into the steam chambers, and then in the turbine building 10UMA they are led to the Stop-Control Valves of the turbine.

On each steam pipe within the steam chamber of the reactor building, a steam valve unit is installed, consisting of a Pressure-Reducing and Protection Module (RPM) and of a Shut-off and Isolation module

Let's consider these devices in more detail.

2.1.10.1 Main Steam Line Isolation Valve (MSIV)

Main Steam Line Isolation Valve (MSIV) is designed for quick and reliable isolation of Steam Generators from steam in emergency situations.

MSIV consists of a body with a piston moving inside it, which is driven through four-way valves of the steam drive control system.

At the same time, to actuate the MSIV, the inlet and outlet steam pipelines with the shut-off valves installed in them are provided. A general view of a MSIV is shown in Figure 2.1.10.1.

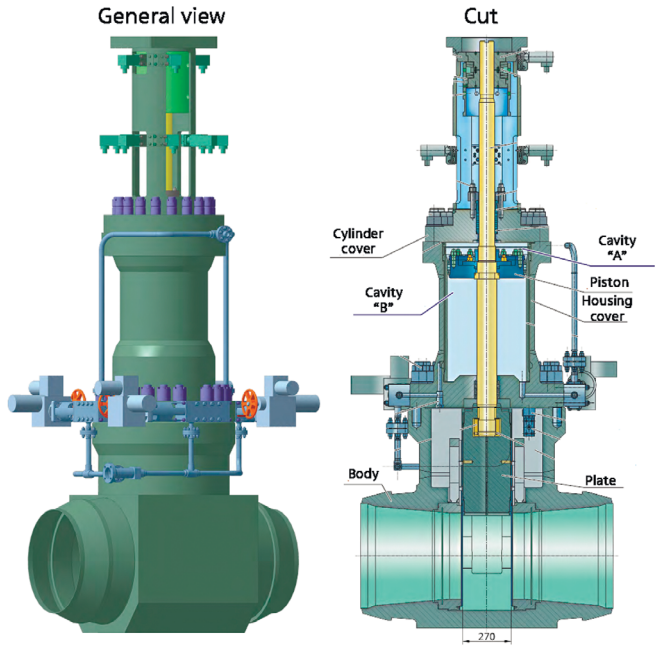


Figure 2.1.10.1 - General view of a MSIV

The main technical characteristics of the MSIV and the isolation electric driven valve behind the MSIV are given in the table 2.1.10.1. The valve is designed for a full operational pressure fall on the shut-off element.

Table 2.1.10.1 — Main technical characteristics of the MSIV and of the isolation electric driven valves behind the MSIV

No.	Name	Quantity	The main technical characteristics of the equipment
1	Main Steam Line Isolation Valve (MSIV)	4	Operating pressure — 8,1 MPa; Pressure in the rated mode — 6,9±0,1 MPa; Temperature in the rated mode — 287±1 °C; Flow rate of the medium — 1600 —1712 t / h; Emergency closing time —10 s; Slow closing (opening) time — 150 s; Parameters of steam from an external source (steam from plant service collector): Operating pressure — 1,08 MPa; Specified temperature — 180 °C;
2	Isolation electric driven valves behind the MSIV	4	Operating pressure — 8,6 MPa; Specified temperature — 300 °C; Closing time — 135 s

To ensure emergency shutdown of the MSIV two control 4-way valves are provided, each of which simultaneously supplies the working medium to the chamber above the piston and dumps the medium from under the piston. Each control valve has an individual steam supply. Each control valve is controlled by an independent control channel using an electric actuator. To prevent steam leakage from the cavities above and below the MSIV piston, check valves are provided on the supply steam lines to the MSIV. The design and control system of the MSIV ensures its reliable operation in normal operation and during accidents.

Emergency closing of the MSIV is provided regardless of the position of the control valves that affect the opening of the MSIV in technological modes. Closing of the MSIV is carried out from its own steam by supplying steam to the MSIV piston and steam dump from under the MSIV piston through two pairs of electric driven valves (on different power supply channels). These valves first are opened, and then, after closing of the MSIV and actuation of the limit position sensors, they are closed. To exclude the loss of the medium from the Steam Generator when the above valves are not closed, the electric driven valves on the dump collector are closed (on different power supply channels).

To ensure slow opening and closing of the MSIV, throttle washers are provided in the supply and dump channels. The dump lines of the control valves are connected to a drain header.

The design of the MSIV provides, with the control valves closed, the guaranteed location of the MSIV in extreme positions (open or closed) for an unlimited time,

regardless of the pressure in the system.

In order to guarantee the elimination of the flow of the working medium into the atmosphere through the drain header after the emergency shutdown of the MSIV, it is equipped with four isolation valves. In the event of a failure of one MSIV, the isolation electric driven valve of this Steam Generator takes over its functions. In the event of a failure of one 4-way control valve of MSIV (including the failure of the supporting systems), the 4-way control valve remaining in operable condition ensures closing of the MSIV.

Steam supply from an external source for the slow opening of the MSIV is provided from a system of inside steam pipelines (1 LBG). On each steam supply line, high and low pressure separation unit is installed.

In various operating modes of the MSIV, a different position of the control valves is provided.

The scheme of the MSIV quick closure is shown in Figure 12.1.10.2.

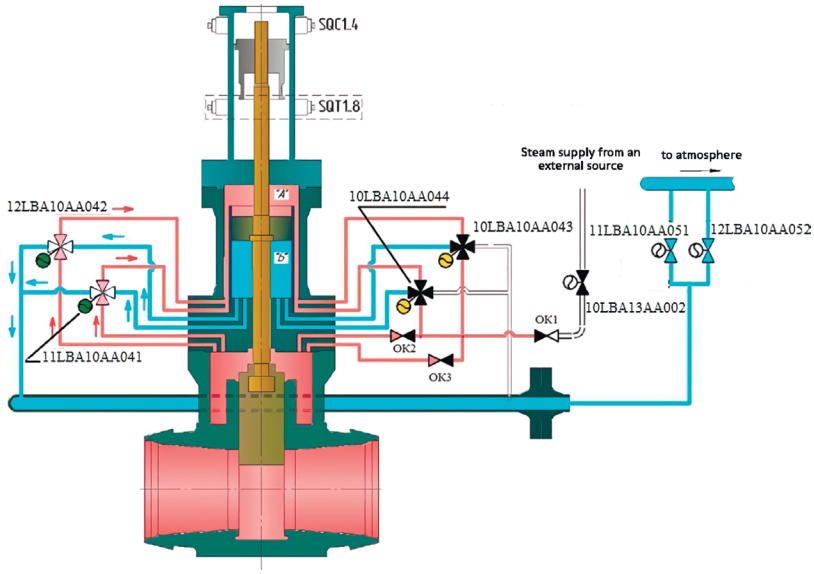


Figure 12.1.10.2 –MSIV quick closing scheme.

In emergency conditions, the emergency shutdown of the MSIV and of the electric driven valve on the pipeline behind the MSIV occurs under the conditions:

Coincidence of the signals:

— the difference between the current values of the saturation temperature of the primary and secondary circuits is more than 70 ° C;

— the pressure in the corresponding Steam Generator is less than 5,53 MPa;

— the primary circuit temperature is more than 150 °C;

Coincidence of the signals:

— reduction of the level in a Steam Generator for 900 mm from the rated;

— primary circuit temperature is more than 150 °C;

Coincidence of the signals:

— increase of the level in the Steam Generator for 300 mm from the rated when the reactor installation is operated at power;

— the reserve before boiling in each of the hot legs of the primary circuit loops is more than 15,5 °C or the neutron power of the reactor is less than 20 % Nrated.

In the case of closing of the MSIV due to reduction of the level in the Steam Generator for 900 mm from the rated, the conditions are created for operation of the SG Passive Heat Removal and Emergency Cooldown systems with the remaining level in the SG.

In the case of passage of the signal, the MSIV and the high-speed electric driven valve are actuated on the Steam Generator, where the signal was generated.

Opening of the MSIV and of the electric driven valve behind the MSIV is made by the operator after cancellation of the prohibition for their opening.

In the event of a simultaneous failure to close the MSIV and the isolation electric driven valve, the medium loss is possible only from one Steam Generator, because other Steam Generators will be cut off by their BZOKs and/or valves. The fragment of the indication of the MSIV position in the Main Control Room is shown in Figure 1.2.10.3.

In case the active safety systems fail to perform the function of heat removal through the Steam Generator from the core, this function will be performed by the Passive Heat Removal System (system 1 JNB50-90).

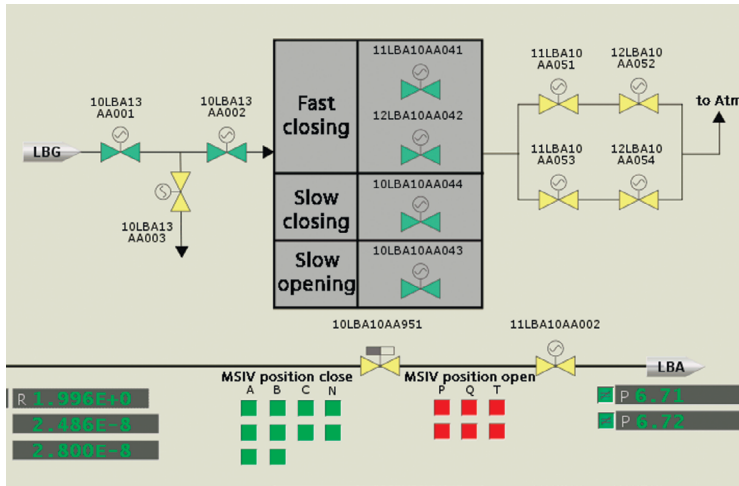


Figure 2.1.10.3 – The fragment of the indication of the MSIV position in the Main Control Room.

During a leak from the primary circuit to the secondary, a sharp decrease in pressure occurs in the primary circuit, an increase of pressure and of level in the emergency Steam Generator, and an increase of activity in the steam. In this case, Reactor Control and Protection System is actuated, Emergency Stop Valves of the turbine are closed.

Cooling of the reactor installation in the mode of leakage from the primary circuit to the secondary is carried out through non-emergency Steam Generators in accordance with the procedures for accidents confinement.

2.1.11 STEAM DUMP RELIEF VALVE DISCHARGED TO ATMOSPHERE -BRU-A

2.1.11.1 Purpose and structure of BRU-A

BRU-A are included in the steam valve unit SVU (PAB) and are located in the building of the steam chamber (UJE) at the level + 17,4.

The RPM reduction and safety module includes: two SG Pilot Operated Relief Valves of Steam Generator, BRU-A and an electric shut-off valve in front of the BRU-A.

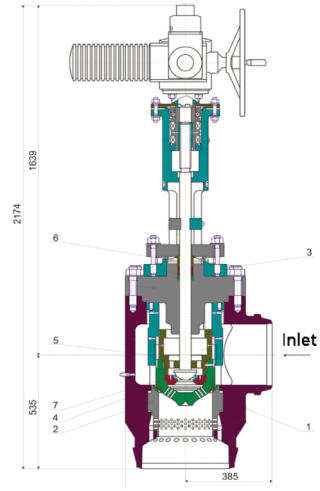
Also, on each steam line in 10UJE steam chambers, steam outlets to the SG Emergency Cooldown System (1JNB10 40) are provided.

Part of the 1 LBA system: steam pipelines from the Steam Generators to the first welded joint outside of the 1 UJE building (including installed on them the SG Pilot Operated Relief Valves, BRU-A, isolation electric driven valves in front of the BRU-A, MSIV and isolation electric driven valves installed behind them) according to the safety impact are the safety related elements, according to the functions performed — elements of the protective and localizing safety system. These elements of the main steam piping system in accordance with NP-001-15 belong to safety class 2. The classification label of SG PORV, BRU-A, electric driven valves in front of the BRU-A, MSIV and the isolation electric driven valves installed behind them on steam pipelines according to NP-001-15 is 2NZL.

BRU-A are designed to dump steam into the atmosphere in the modes, associated with excess of pressure in a Steam Generator and at the simultaneous prohibit for dump of steam into the turbine condenser or with blackout. BRU-A also provides regulation of steam pressure in Steam Generators within the specified limits during transient modes, as well as in the modes of violation of normal operating conditions and in emergency conditions.



Figure 2.1.11.1- The general view of a BRU-A. The BRU-A section is shown in Figure 2.1.11.2.



1 — Valve body assembly;
2 — Seat;
3 — Yoke;
4 — Gate valve;
5 — Barrel;
6 — Gland;
7 — Gate valve seat

Figure 2.1.11.2 – the BRU-A section

The throttle devices Du 400/600, 600/800, 800/1000 are installed on the BRU-A exhaust pipelines, designed to throttle the working medium (reduce pressure) to the specified parameters.

The BRU-A is a part of the steam valve unit. The main technical characteristics of the BRU-A and of the electric driven valve in front of the BRU A are given in the Table 2.1.11.1.

When the stem moves up, the flow cross section opens in the gate valve seat. During the movement of the stem for 9 ± 0,5 mm steam is dumped from the upper part of the barrel, the pressure of the medium in it decreases to values close to the pressure of the medium behind the valve and, as a result, the decrease of pressure drop on the main gate valve occurs. In this case, the stem support flange comes into contact with a barrel connected by a thread with gate valve.

With further movement, the stem displaces the unloaded gate valve. The gate valve during its movement opens or closes the flow cross section, thereby regulating the volume of the medium flowing through the valve.

Table 2.1.11.1 - The main technical characteristics of the BRU-A and of the electric driven valve in front of the BRU-A

Name	Quantity	The main technical characteristics of the equipment
Steam Dump Relief Valve Discharged to Atmosphere -BRU-A.	4	Throughput capacity of each BRU-A in totally open position – 1000 t/hour; Opening pressure - 7,7 MPa; Closing pressure - 6,95 MPa; Regulation pressure- 7,3 (± 0,1) MPa Time of opening/closing of valve - 15/15 s
The electric driven valve in front of the BRU-A	4	Closing pressure - 6,7 MPa; Opening pressure - 6,9 MPa;

Control of the BRU-A and of the electric driven valve in front of the BRU-A is implemented using Automatic Process Control System equipment or by an operator.

The BRU-A is designed for operation in the following modes:

basic mode, providing operation in:

— protective mode;

— automatic pressure control in a Steam Generator;

— Unit cooldown.

Selection of the BRU-A operating mode is determined by an operator of the Main Control Room.

Operation of the BRU-A in a protective mode - this function provides actuation of the system of steam dump to atmosphere through the BRU - A at the first stage of the transient until the design capacity of the SG Emergency Cooldown System (SAR) or Passive Heat Removal System (SPOT) is reached.

The output signal for control (opening) of the BRU A comes into effect when the pressure in the steam line becomes more than 7,7 MPa. At a pressure of less than 7,7 MPa, the opening signal disappears and the valve remains in the degree of opening that it has reached. The closing signal comes into effect when the pressure in the steam line is less than 6,95 MPa, in case there is no signal the “Cooldown mode”.

When operating in automatic control mode, the BRU-A maintains the constant steam pressure at the outlet of the Steam Collector of the Steam Generator, equal to 7,3 MPa with an accuracy of ± 0,1 MPa. The automatic BRU-A control algorithm is as follows.

At coincidence of the conditions:

— the pressure in the steam line of the corresponding Steam Generator is 7,6 MPa and less;

— the BRU-A 11 (12) LBA10,20 (30,40) AA201 open for more than 6%;

The BRU-A switches to the automatic mode «Pressure holding» at 7,3 MPa in the steam line of the corresponding Steam Generator.

If in the «Pressure holding» mode the BRU-A is closed, then the time delay of 100 seconds is turned on - the "Guard mode", during which the BRU-A is in automatic mode with a pressure control of 7,3 MPa.

If after this time delay the regulator never opened, it switches to the remote mode.

Further actuation of the BRU-A will be in the protective mode.

In the cooldown mode, the BRU-A provides cooling of the reactor installation at a speed of 30 °C/h or 60 °C/h, this value is selected by an operator of the Main Control Room, by inputting overlays on video frames of the Unit Upper Level Control System (ULCS). Overlays for selecting of a mode on ULCS video frames are shown in the Figure 2.1.11.3.

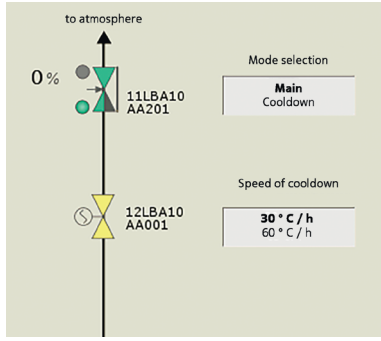


Figure 2.1.11.3 - Overlays for selecting the BRU-A mode on the ULCS video frames

When Unit is operating at the rated power level, the pressure of 6,9 MPa is maintained in the Steam Generators. In this case, the SG Pilot Operated Relief Valves are in standby mode, i.e. safety valves are closed; MSIV and isolation electric driven valves behind the MSIV are open and in standby mode are in a state of availability for operation; BRU-A and BRU-K are closed, the electric driven valve in front of the BRU-A is open.

The electric driven valve in front of the BRU-A:

— closes under the condition that the BRU-A is not closed or the BRU-A is unexpectedly opened and the pressure at the outlet of the Steam Generator collector is below 6,7 MPa if there is no signal the "Cooldown mode"

— opens when the pressure at the outlet of the steam collector of the Steam Generator is more than 6,9 MPa.

2.1.11.2 Modes of violation of normal operating conditions

In the modes of violation of normal operating conditions, not associated with loss of power supply for auxiliary needs or prohibition of steam dump into the turbine condenser, when the pressure in the secondary circuit rises and the corresponding pressure setpoint is reached, first open the BRU-Ks, while maintaining pressure in the Steam Generator within the specified limits. If the further increase of pressure has stopped and the pressure has decreased to the closing setpoint, then the BRU-Ks will close. If the pressure increase has not stopped, the BRU-Ks have not actuated, or there is a prohibition for their opening, then the BRU-As start operation. If the BRU-A fails to open, the safety valves of the SG PORV are actuated.

In the modes of violation of the normal operating conditions associated with the loss of power supply to busbars of normal operation, when the turbine shut-off valves are closed, the steam pressure in the secondary circuit increases sharply. Steam dump to the turbine condensers is prohibited in this mode, since cooling water is not supplied to the condensers. When the pressure at the outlet of the SG steam collector reaches the value of 7,7 MPa BRU-As are actuated.

In the modes of violation of normal operating conditions, associated with the loss of power supply to the busbars of normal operation, according to the step-by-step start-up program, the Motor Driven Auxiliary Feedwater Pump is put into operation, the BRU-A is opened, the pumps of the SG Emergency Cooldown System (1 JNB10-40) are turned on for recirculation. In case of non-closure of the BRU-A, in order to exclude the loss of the medium of the secondary circuit from the Steam Generator, the electric driven valves in front of the BRU-A are closed. Closing of the MSIV in this mode does not occur.

If during blackout of the station there is need to cooldown the reactor installation or the blackout occurred at the first stage of the cooldown mode, the removal of the residual heat and the cooldown of the reactor installation is provided through the BRU-A system and the SG Emergency Cooldown System (1 JNB10-40) until actuation of Reactor Coolant Circuit Emergency and Planned Cooldown System (1 JNA).

In the modes of violation of normal operating conditions that are not related to blackout, when prohibiting the steam dump into the turbine's condenser, the Unit maintaining in hot reserve or the reactor installation cooldown is performed similarly to the NPP blackout mode.

2.1.11.3 BRU-A operation at design basis accidents

At rupture of a steam pipe in the non-cut-off part - from the Steam Generator to the isolation electric driven valve – on the basis of the NPP reactor protection signals is actuated Reactor Emergency Protection, the turbine stop-control valves are closed, the MSIV and isolation electric driven valve on the steam pipe from the emergency Steam Generator are closed, the supply of feed water to this Steam Generator stops, the RCP of the emergency loop is turned off, the Steam Generator blowdown is closed. Thus, the emergency Steam Generator is localized, and the remaining Steam Generators remain in operation. After that, the Unit is put into a cooldown mode using standard systems.

At rupture of the Main Steam Header or steam pipelines behind the isolation valves (in the isolated part), by the NPP protection signals the Reactor Emergency Protection is actuated, the MSIV and isolation electric driven valves on the steam lines from all Steam Generators are automatically closed, and Emergency Stop Valves of the turbine are closed. After that, the Unit is put into the cooldown mode using the BRU-A system and the SG Emergency Cooldown System (1 JNB10-40).

2.1.12 THE SG PILOT OPERATED RELIEF VALVES (SG PORV)

2.1.12.1 Function and structure of SG PORV system

The secondary circuit overpressure protection system is designed to prevent overpressure in Steam Generators and main steam pipelines in excess of 15 % of the operating pressure of the Steam Generators.

The system provides its functions in any initial event of the accident with the imposition of one, independent of the initial event, failure of any active or passive component, having mechanical moving parts, or of one personnel error, independent of the initial event.

The main elements of the system are SG Pilot Operated Relief Valves (SG PORVs). Each Steam Generator has two PORVs of the same design, but having different actuation setpoints.

The number of the installed Pilot Operated Relief Valves is determined according to the principle n + 1 (where n is the number of PORVs, providing the flow rate of the dumped medium necessary to reduce the pressure), which ensures that the pressure in the Steam Generator does not exceed the operational pressure

for more than 15 %, even at failure of PORV with the lowest setpoint for opening.

The throughput capacity of each valve is at least 1050 t/h at the actuation pressure, what allows to fully perform the function of Steam Generator protection from overpressure by actuation of one PORV.

PORVs are qualified for operation on steam, water and steam-water mixture.

In the design it is accepted that the design pressure for the secondary circuit is equal to the corresponding value of the operational pressure and is equal to 8,1 MPa.

The pressure in the steam pipelines of the Steam Generator during the design basis operation of reactor scram and of SG PORV shall not exceed 115 % of the calculated value: 8,1 x 1,15 = 9,32 MPa.

The pressure in the SG steam pipelines in failure of reactor scram and design work of SG PORV shall not exceed 135 % of the calculated value:

$$8{,}1 \times 1{,}35 = 10{,}94 \text{ MPa}.$$

In severe accidents, the pressure in the SG steam pipelines must not exceed the value of the pressure at hydrotests for strength equal to 11,5 MPa.

2.1.12.2 Flow diagram of the system.

The SG PORV are part of the steam valve unit SVU (PAB) and are located in the building of the steam chamber (UJE) at +17,433.

For each SG, a steam valve unit is installed in a room separated from the equipment and pipelines of other Steam Generators by building structures, designed to absorb the dynamic effects of pipeline rupture, flying objects and seismic effects of the safe shutdown earthquake level.

The exhaust pipelines of the SG PORVs are equipped with continuously operating drains to remove moisture, generated as a result of precipitation and steam leaks through the PORV control valves.

The steam valve assembly (PAB) is a part of the main steam piping system LBA. The location of the main steam pipelines and of PAB is shown in the Figure 2.1.12.1.

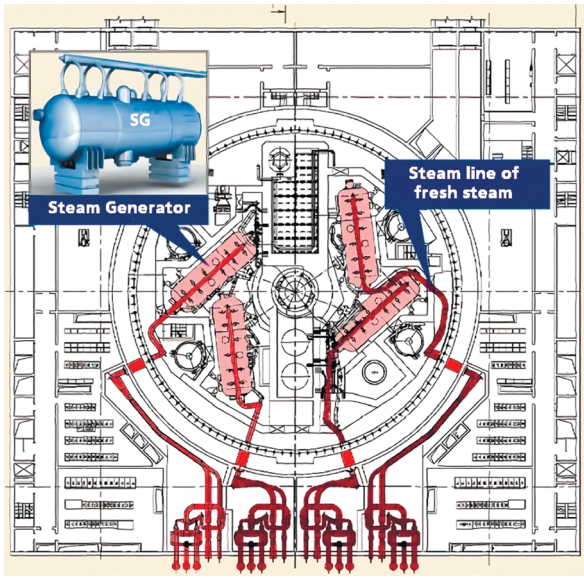


Figure 2.1.12.1 - Location of the main steam pipelines

The flow diagram of the steam valve unit is shown in the Figure 2.1.12.2.

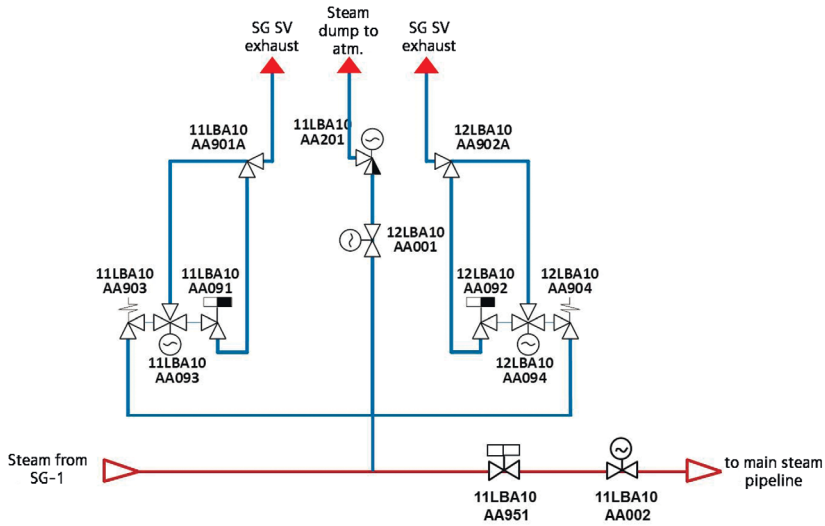


Figure 2.1.12.2 – The flow diagram of steam valve unit of SG-1

LBA10AA001 – the shut-off valve in front of the BRU-A
LBA10AA201 – the BRU-A

LBA10AA091(092) – the electromagnetic valve of SG PORV
LBA10AA903(904) – the pilot valve of SG PORV
LBA10AA093(094) – the switching device of SG PORV
LBA10AA901A(902A) – the main valve of SG PORV
LBA10AA951 – the Main Steam Isolation Valve (MSIV)
LBA10AA002 – the isolation valve behind the MSIV
On the steam line of each Steam Generator in the non-cut-off part (before MSIV) two PORVs are installed: one control, the second – operational.

2.1.12.3 Brief description of the SG PORV design

The SG PORV has a modular type design, which includes (Figure 2.1.12.3):
— the main safety valve (Pilot Operated Safety Valve - Posy);
— the pilot valve with tuning valve;
— the electromagnetic valve (EV);
— the switching device;
— the c (Steam filter).

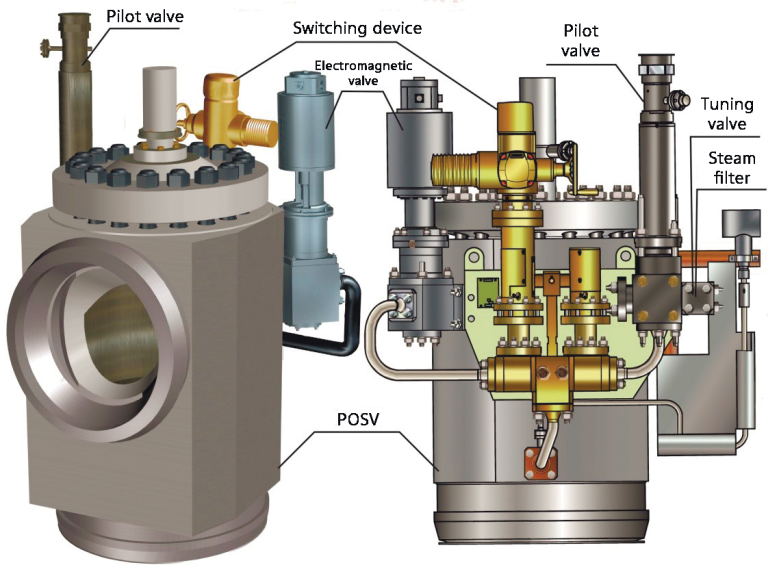


Figure 2.1.12.3 - The appearance of a SG PORV

The main valve is opened by the working medium when controlled by an electromagnetic valve actuator or, in case of electromagnetic valve failure, by control of the direct-acting pilot valve.

The main function of the switching device is to turn off one of the channels for supplying the working medium to the piston cavity of the main valve in case of failure of electromagnetic valve or of pilot valve to close (unauthorized pressure reduction in the system).

The switching device may have three positions: middle (open) and two limit (closed). If the electric actuator of the switching device is de-energized, then its shut-off element is in the middle position and then the pilot valve and electromagnetic valve are connected to the main valve in parallel. When voltage is supplied to the electric actuator of the switching device, its shut-off element can be moved to one of the limit positions, and then one of the valves (pilot valve or electromagnetic valve) is then disconnected from the main valve, what allows to disconnect the supply channel of the control medium to the main valve in case of freezing of the pilot valve or of the electromagnetic valve and to exclude the pressure drop in the Steam Generator below the permissible level.

The tuning valve provides the ability to test the pilot valve function without actuating of the main valve. In this case, the channel from the pilot valve to the main valve must be closed using the switching device. The test is performed at the shutdown power Unit at the reduced (up to atmospheric) pressure in the Steam Generator.

The springs, which are parts of the pilot valve, the electromagnetic valve, the switching device are located in the bodies of the corresponding elements and are protected from the direct exposure to the environment and overheating.

The steam filter prevents solid particles, which present in the steam, from entering the pilot valve using the removable metal grid, made in the shape of a cylinder.

The pilot valve and the switching device are assembled on a plate, that is mounted directly on the main valve body. The steam filter is attached to the tuning valve, which is a part of the pilot valve. The electromagnetic valve is mounted directly on the main valve body.

To monitor the position of the shut-off device, the position sensors are installed on the electromagnetic, pilot and main valves.

The pressure for the electromagnetic valve is taken from the main valve body. The pressure for the pilot valve - from the tee through the pilot pipeline. Drainage from the switching device is provided to the outlet pipe of the main valve.

The SG PORV scheme in standby mode is shown in the Figure 2.1.12.4.

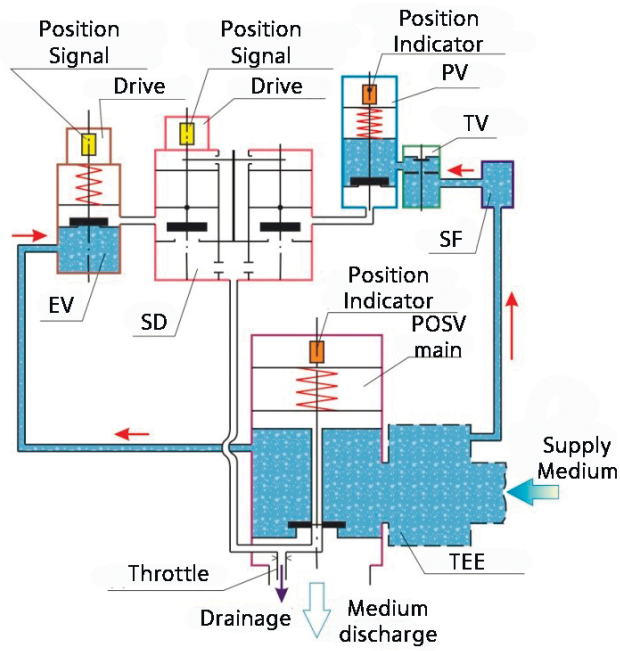


Figure 2.1.12.4 – Scheme of SG PORV in standby mode

In the standby mode, the main valve is closed by the pressure of the working medium and by the force of the spring. The electromagnetic and pilot valves are closed by the spring force. The switching device is in the middle position (the channels from the electromagnetic valve and the pilot valve are open), the electric drive of the switching device is de-energized, the shut-off elements of the switching device are in the "open" position. The output pipe of the switching device is connected by a pilot tube with the piston chamber of the main valve.

Actuation of the main valve is provided by supplying of the working medium pressure to the Main Safety Valve through the electromagnetic valve, which opens by the signal of the Engineered Safety Features Actuation System (ESFAS) when the pressure in the SG steam line increases to the opening setpoint.

When the electromagnetic valve is opened, steam from the steam pipe enters the piston chamber of the main safety valve (the Figure 2.1.12.5).

At the same time, in the piston chamber, due to the difference in the area of the seat and the piston, forces arise that compress the spring and lift the plate up, as a result of which the steam from the main steam line begins to be dumped into the atmosphere.

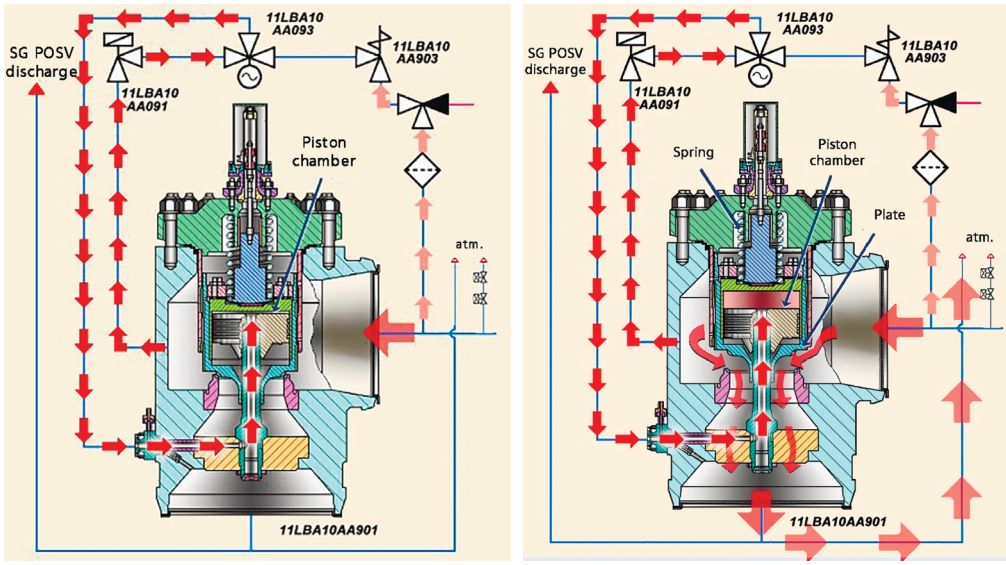


Figure 2.1.12.5 The SG PORV actuation at opening of the electromagnetic valve

When the pressure of the working medium in the steam line is reduced to the closing setpoint by the ESFAS signal the electromagnetic actuator of the electromagnetic valve is de-energized, and the shut-off element of the electromagnetic valve, under the action of the spring, moves to the “closed” position. The pressure from the switching deice and the piston cavity of the main valve is dumped through the throttle device into the drainage. The shut-off element of the main valve under the action of the spring moves to the "closed" position (Figure 2.1.12.6).

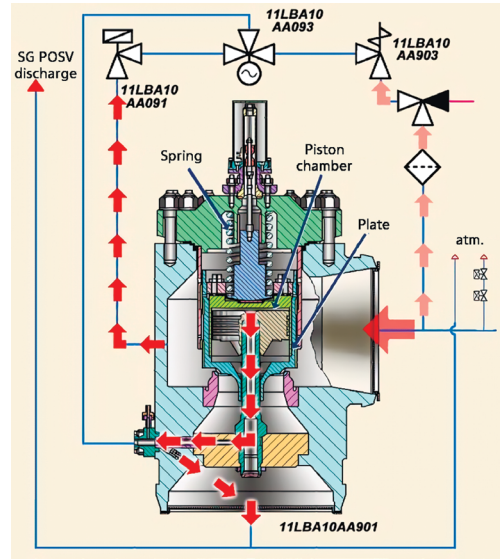


Figure 2.1.12.6 – The scheme of closing of the SG PORV main valve at closing of elecrtromagnetic valve

Figure 2.1.12.7 shows the operation diagram of the switching device in case of electromagnetic valve failure to close.

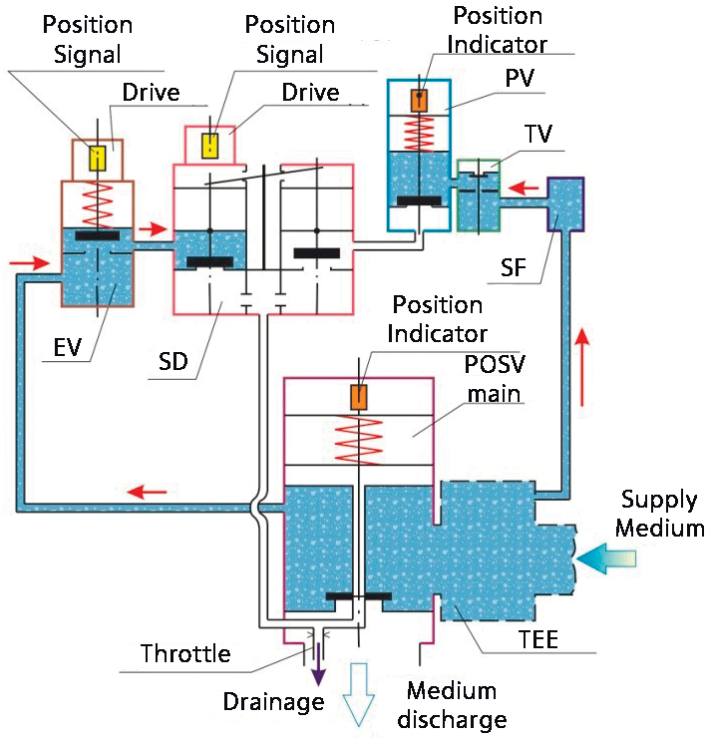


Figure 2.1.12.7 – The scheme of the switching device closing in case of electromagnetic valve failure to close

If, after the main valve actuation and sending of the signal for closing to the electromagnetic valve a further pressure drop occurs in the steam line below the SG PORV closing setpoint (the event of failure to close of the electromagnetic valve), then the control signal is automatically sent to the electric actuator of the switching device to “close” (to move the stem of switching device down) . In this case, the left shut-off element of the switching device moves to the "closed" position, disconnecting the channel from the electromagnetic valve, while blocking the right shut-off element of the switching device (the channel from the pilot valve) in the "open" position.

The pressure from the switching device and the piston cavity of the main valve is dumped through the throttle device into the drainage. The shut-off element of the main valve under the action of the spring moves to the closed position.

Figure 2.1.12.8 shows the diagram of the Main Safety Valve actuation from the pilot valve.

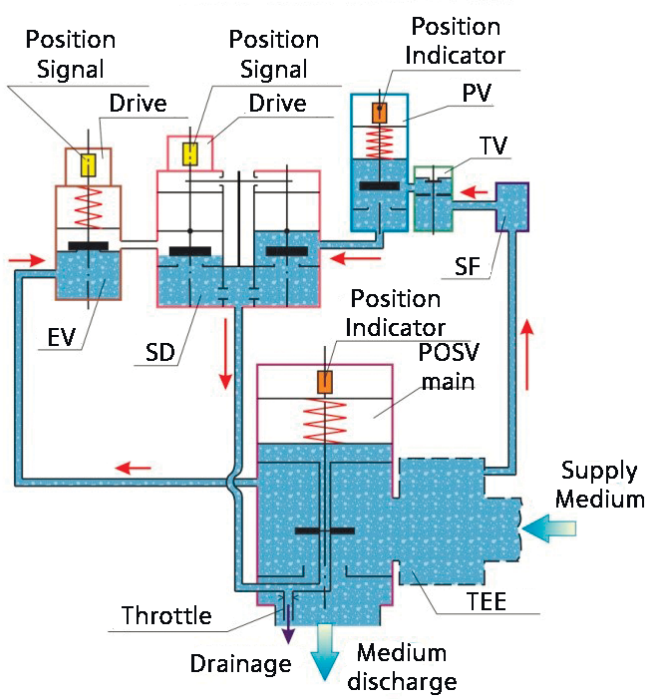


Figure 2.1.12.8 - Scheme of actuation of the Main Safety Valve (POSV) from the pilot valve (at the failure of the electromagnetic valve to open)

At failure of **Electromagnetic Valve to open** and at further increase of the pressure in the steam line, the spring, compressing, opens the Pilot Valve and the working medium enters through the switching device into the piston cavity of the main valve. The main valve opens.

With a subsequent decrease of pressure in the steam line, the pilot valve is closed under the action of a spring and the main valve fits.

Figure 2.1.12.9 shows the operation scheme of the switching device in case of failure of the to close.

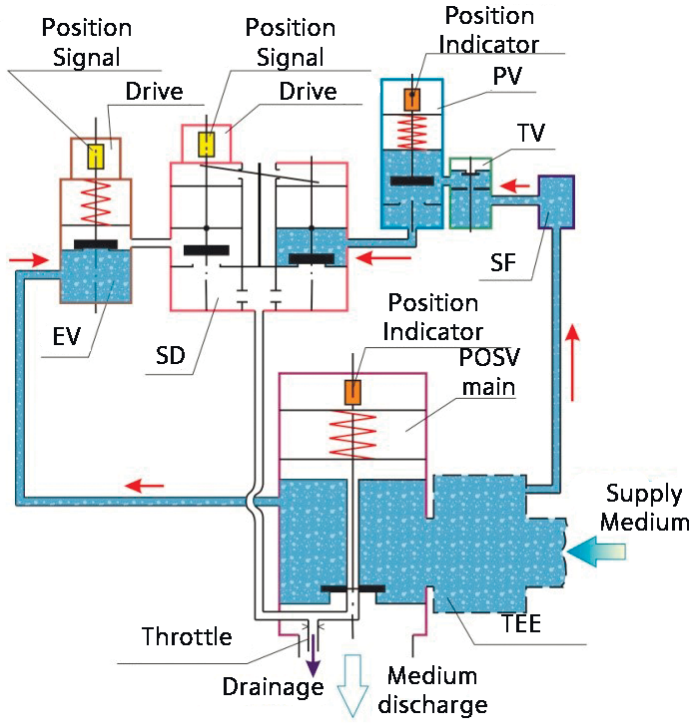


Figure 2.1.12.9 - Scheme of operation of the switching device in case of failure of the pilot valve to close

If, after the Main Valve is actuated at non-closure of the pilot valve, a further pressure decrease occurs in the steam line (below the pilot valve closing setpoint), then the control signal is sent to the electric actuator of the switching device to "open" (to move the stem of the switching device up). In this case, the right shut-off element of the switching device moves to the "closed" position, disconnecting the channel from the pilot valve, while blocking the left shut-off element of the switching device (the channel from the electromagnetic valve) in the "open" position.

The pressure from the switching device and the piston cavity of the main valve is dumped through the throttle device into the drainage. The shut-off element of the main valve under the action of the spring moves to the "closed" position.

Technical characteristics of the SG POSV are given in the table 2.1.12.1.

Table 2.1.12.1 - Technical characteristics of the Main valve SG PORV

Parameter	U/M unity of measurement	Parameter value
The working medium	—	Steam, steam-water mixture, water
The cross section, Du — The Main Safety Valve (GPK); — The pilot valve, the tuning valve, the electromagnetic valve, the switching device	mm	450/600 25
The steam flow capacity of each device at the opening pressure, not less than	t/h	1050
ESFAS: the opening pressure of the electromagnetic valve of the reserve PORV	MPa	8,7
ESFAS: the opening pressure of the electromagnetic valve of the operating PORV	MPa	8,9
The opening time from the moment, when the signal is sent to the EV up to the complete opening of the MSV, no more than	s	1
ESFAS: the closing pressure of the electric valve EV of the operating and reserve PORV	MPa	7,85
The opening pressure of the pilot valve	MPa	9,23
The closing pressure of the pilot valve	MPa	7,29

2.1.12.4 Monitoring and control of the system.

The power supply, control and monitoring of two SG PORVs, installed on the steam line of one Steam Generator is carried out from different channels of reliable power supply systems of category I with the territorial, information and electrical separation of automation equipment within the Safety Systems train.

Elements control, monitoring over the position (state) of elements, monitoring of the technological parameters, as well as the warning and emergency alarm system in full are displayed in the Main Control Room and the Emergency Control Room. In the MCR and the ECR information on the limit positions of the main valve and the indication of the position of the pilot valve are provided (Figure 2.1.12.10).

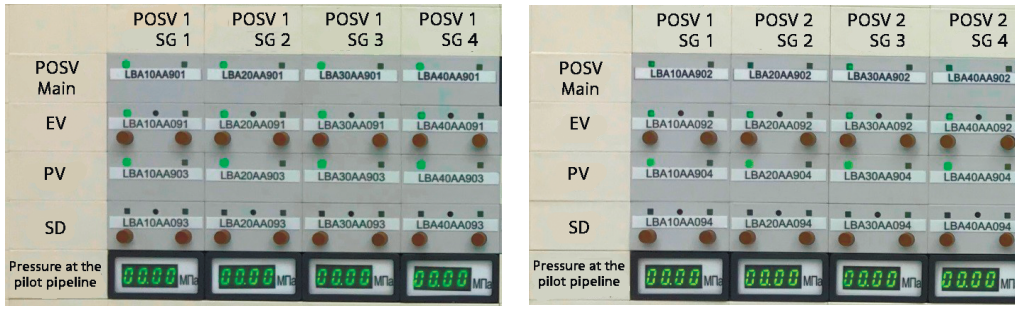


Figure 2.1.12.10 - The mimic diagram of SG PORV on the MCR panels

An automatic putting of the system into operation, which does not require the operator intervention, when the pressure in the secondary circuit rises to the setpoints for actuation of the steam dump devices and remote (by key) — by the operator from the MCR or ECR are realized. At the same time, the automatic control takes precedence.

The list of FMCS for the secondary circuit overpressure protection system is presented in table 2.1.12.2.

Table 2.1.12.2 - The list of Functions of the Monitoring and Control System (FMCS) for the secondary circuit overpressure protection system

The actuation condition	The setpoint	The action	FMCS
Increasing of the overpressure in the main steam line of the SG, MPa, more than	8,7	Opening of the first SG PORV	ESFAS BF11
Increasing of the overpressure in the main steam line of the SG, MPa, more than	8,9	Opening of the second SG PORV	ESFAS BF11
Reduction of the overpressure in the main steam line of the SG, MPa, less than	7,85	Closing of the first and the second SG PORV	ESFAS BF11

The points of technological control for the secondary circuit protection system against overpressure are pressure measurements in the SG steam pipelines and the pressure in the pilot line for supplying steam from the switching device to the Main Steam Valve.

The pressure sensor in the steam supply line from the switching device to the Main

Valve is used to monitor leaks through the pilot valve and the electromagnetic valve. The critical value of pressure in the steam supply line from the switching device to the Main Steam Valve = 0.495 MPa. This value (at which the flow rate in the steam supply line from the switching device to the Main Steam Valve will be equal to the flow rate through the drainage hole) is the initial value for deciding on an urgent shut-off of one of the control lines to the Main Valve.

2.1.12.5 The system operation modes

System functioning during normal operation

In normal operation modes the secondary circuit overpressure protection system is in the standby mode in the closed position.

For hydrotests of SG for the 2-ry circuit the mechanical intermediate stop device is installed in the pilot valve.

System functioning during design basis accidents

A sharp increase of the pressure in the Steam Generator occurs when there is the imbalance between the power, supplied to the SG by the primary coolant, and the power, removed from the SG by the secondary circuit.

This imbalance can occur, for example, when the electric load of the TG is dropped, the TG Emergency Stop Valves are closed or at false closure of the MSIV.

The power imbalance can be reduced in two ways:

1. due to reduction of the reactor power – actuation of the reactor preventive protection-1, reactor scram

2. due to increase in heat removal from the Steam Generator — actuation of steam dump devices (BRU-K, BRU-A, SG SV) or by switching on the emergency cooldown systems SG Emergency Cooldown System (Passive Heat Removal System).

The design provides the multi-stage protection of steam pipelines at increasing of pressure in the Steam Generator (Figure 2.1.12.11).

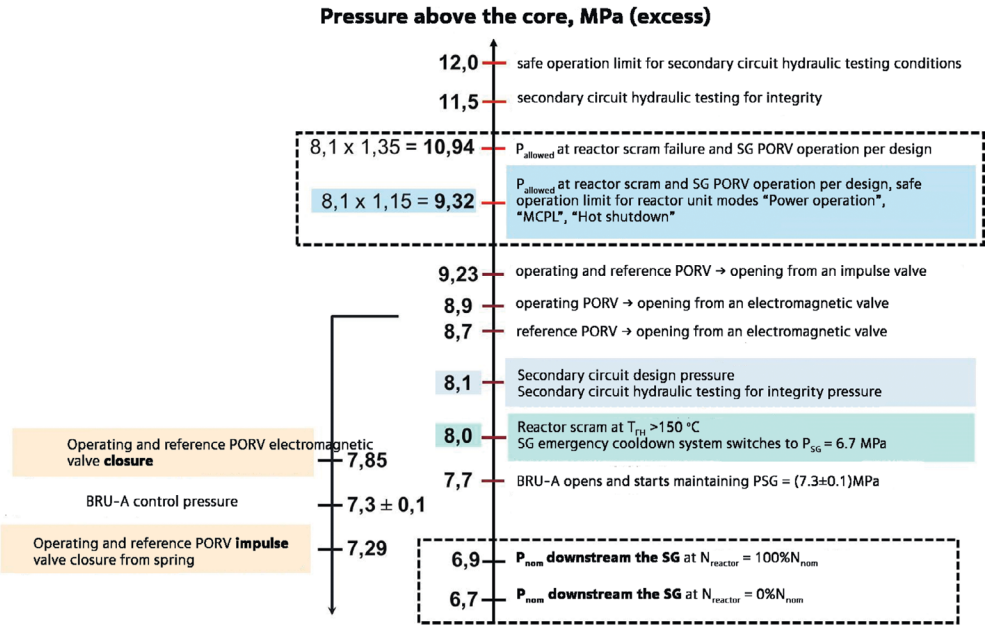


Figure 2.1.12.11 - Operation of safety systems at increasing of pressure in the SG

When the pressure in the steam line increases to 7,7 MPa, according to the signal of the ESFAS (FMCS BC11), the BRU-A opens and switches on to the mode of pressure maintaining in the SG = (7,3 ± 0,1) MPa.

When the pressure in the Main Steam Valve increases to 7,6 MPa, the PZ-1 reactor preventive protection is actuated and the reactor is unloaded till the PZ-1 signal is removed at the pressure in the MSV below 7,2 MPa or when the power of the Reactor Installation is less than 3% Nrated.

When the pressure in the SG steam line increases to 8 MPa, the reactor scram (AZ) is actuated and the SG Emergency Cooldown System is switched on in the mode of maintaining pressure of 6,7 MPa in the Steam Generator. The reactor AZ setpoint is higher than the BRU-A opening setpoint.

When the pressure in the steam line increases to 8,7 MPa, the first SG PORV opens, and at the pressure of 8,9 MPa, the second SG PORV opens. Both valves close when the pressure in the secondary circuit drops to 7,85 MPa.

In the event of failure of actuation of the SG POSV on the signal from the ES-FAS and the subsequent increase of overpressure in the Steam Generator to 9,23 MPa — the SG PORV are actuated by opening of the direct spring pilot valve.

2.2 LOCALIZING SAFETY SYSTEMS

Localizing safety systems (components) are intended for preventing or localizing radioactive materials and ionizing radiation released beyond the designed boundaries into the environment in the event of accidents.

An accident localization area (ALA) is a space confined with a leak-tight enclosure (or other confining safety system components) within which boundaries are provided to retain radioactive materials released during an accident.

The containment (or a leak-tight enclosure) is a set of plant components including construction structures that confine the space around the reactor facility or other components with radioactive materials which makes the designed boundary and prevents the spread of radioactive materials and ionizing radiation into the environment in the amount exceeding the specified limits.

2.2.1 CONTAINMENT SYSTEM

A leak-tight enclosure is a physical barrier against the spread of radioactive materials and ionizing radiation into the environment.

In the design of Novovoronezh NPP-2 the leak-tight enclosure consists of the following components:

— reinforced concrete enclosing structures;

— containment metal lining;

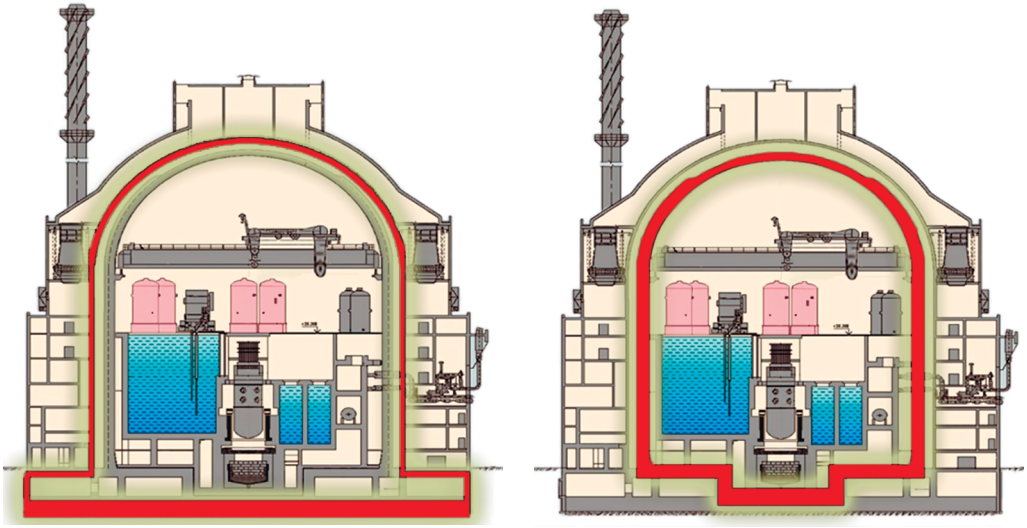
— transport, main and emergency locks;

— containment penetrations;

— isolating structures with pipelines and air ducts between them.

2.2.1.1 Reinforced concrete enclosing structures

The reactor building consists of a double containment and an auxiliary building (Figure 2.2.1.1).



a). External containment **b). Internal containment**

Figure 2.2.1.1 – Reactor building sectional view

The reactor building foundation slab is a rectangle of size 79,2 x 73,2 m with its upper part at elevation minus 5,4 m. The foundation part with internal stiffening baffles lies on the foundation slab in a form of a rigid reinforced concrete frame (Figure 2.2.1.2).



Figure 2.2.1.2 – Construction of the reactor building

The foundation part is covered with a 1,800 mm slab at upper elevation 0,000 m. The cover slab is a support for the double containment (Figure 2.2.1.3).

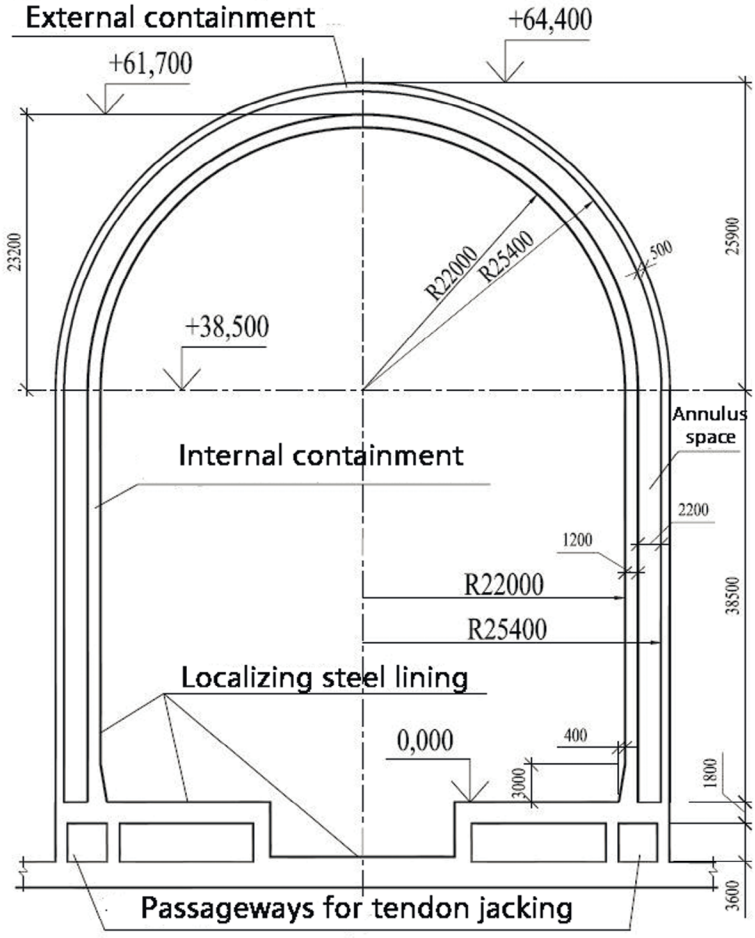
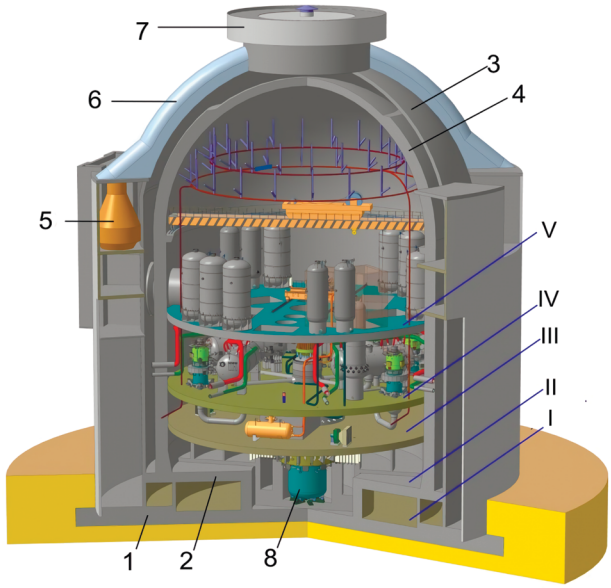


Figure 2.2.1.3 – Double containment sectional view

In the centre of the foundation part under the reactor vault there provided a place for the molten core catcher structures beginning at elevation minus 4,9 m (Figure 2.2.1.4).



1.Foundation slab
2. Support slab (cover slab)
3. External containment
4. Internal containment
5. Heat exchanger of the passive heat removal system
6. Air ducts of the passive heat removal system
7. Outlet header of the passive heat removal system
8. Molten core catcher
I Elevation «— 5,4 m»
II Elevation «0,0 m»
III Elevation «+ 8,14 m»
IV Elevation «+ 14,5 m»
V Elevation «+ 26,3 m»

Figure 2.2.1.4 - Layout of the reactor equipment inside the containment

The foundation slab as a containment system component at elevation 0,000 m, the double containment, walls and the molten core catcher bottom form a reinforced concrete enclosure.

The double containment consists of internal and external containments which together with the cover slab at elevation 0,0 confine the annulus space.

The volume of the annulus space is 18,000 m^3. The gap between internal and external containments is 2,2 m.

Vacuum is maintained in the annulus space to collect and purify potential steam-gas mixture leaks from the internal containment in the event of the primary coolant leaks.

The external containment (external enclosure) is made of cast-in-situ concrete and intended for the protection of systems and components in the reactor building against the direct effect of external natural and man-induced impacts such as an aircraft crash, an external air-shock wave, and extreme wind and climate impacts (Table 2.2.1).

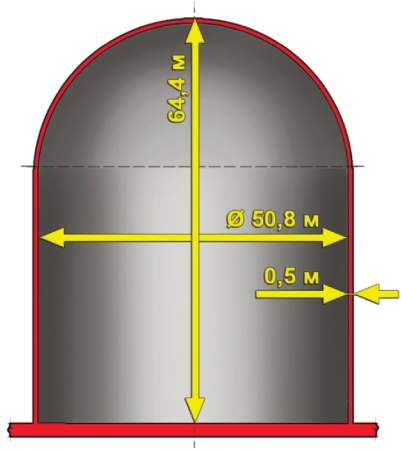
Table 2.2.1 – List of external natural and man-induced impacts at NV NPP-2 site

N	Name of processes, phenomena, factors	Quantitative value of parameters and characteristics	Estimated frequency
1	Earthquakes and tectonically active faults	Design basis earthquake is force-six on the MSK-64 scale. Safe shutdown earthquake is force-seven on the MSK-64 scale.	
2	Aircraft crashes	Building structures for 1st category buildings are rated for a crash of an aircraft of not more than 5,7 tons in weight. The speed at the moment of contact with building enclosure and structures equals to 100 m/s.	
3	External air-shock wave	Pressure in the air-shock wave front is 30 kPa; duration of the contraction phase is not more than 1 s; direction of propagation is horizontal.	
4	Rated values of snow cover weight	1,26 kPa	
5	Annual average air temperature	5,7 °C	
	Rated maximum air temperature	50 °C	Once in 10 000 years
	Rated minimum air temperature	minus 46 °C	Once in 10 000 years
6	Rated maximum of wind velocity (considering wind gusts)	38 m/s	Once in 100 years
		56 m/s	Once in 10 000 years
7	Rated maximum high-water level in spring	92,3 m	Once in 100 years
		94,7 m	Once in 10 000 years

N	Name of processes, phenomena, factors	Quantitative value of parameters and characteristics	Estimated frequency
8	Annual average rate of precipitation	557 mm	—
	Rated daily maximum of precipitation	141 mm	Once in 100 years
		175 mm	Once in 10 000 years

The external containment has a form of a cylinder with a hemispherical dome. The 500 mm thickness of walls and a dome is conditioned by crash loads of an aircraft of 5 tons and a speed of 100 m/s. For cylinder parts protected against the aircraft crash, the thickness is decreased down to 400 mm (Figure 2.2.1.5).

The inner surface of the containment is covered with effective polymeric coating which ensures the required leak-tightness degree.



Inner diameter	50,8 m
Thickness of walls and a dome	0,5 m
Elevation of the dome top	+ 64,400

Figure 2.2.1.5 - Geometry of the external containment

Outside of the upper part of the containment at the elevation of the auxiliary building roof (above elevation + 29,7 m) there is a gallery of compartments for passive heat removal system heat exchangers and metal thrust air ducts with thermal insulation and coating made of stainless steel plate to protect them against weather impacts. In the dome vertex of the external containment there are building structures of the passive heat removal system outlet header (Figure 2.2.1.4).

The internal containment (internal enclosure) is a boundary of the accident

localization area (ALA). It is designed for parameters of the whole spectrum of accidents estimated in the design including accidents with fuel meltdown.

In the accident localization area (Figure 2.2.1.6) there is a reactor facility, a spent fuel pool, systems operating under primary circuit parameters, transport and process equipment, and ventilation systems. Under the reactor vault (below the support slab) there are molten core catcher structures.

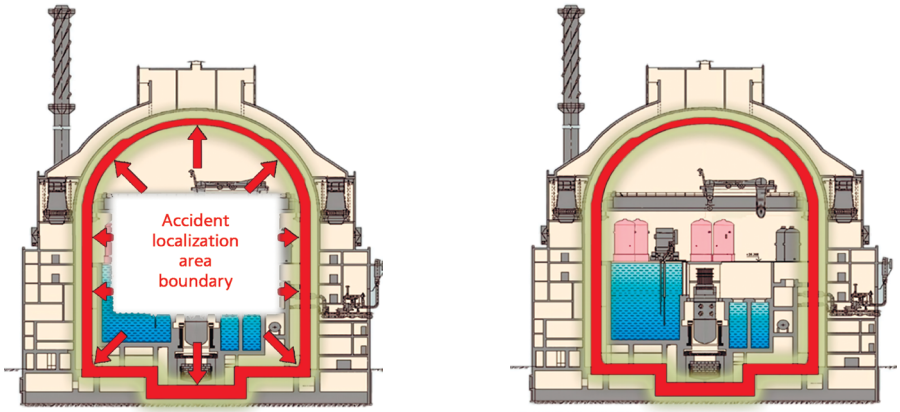


Figure 2.2.1.6 - An accident localization area boundary

The design provides for the separation of the internal containment into two areas:

— Non-accessible area (NAA) is an area where process equipment and pipelines with highly potential primary coolant (a reactor vault, steam generator compartments, a volume compensator compartment, and etc.) are located. The radiation environment in these compartments excludes the personnel access when the reactor is in power.

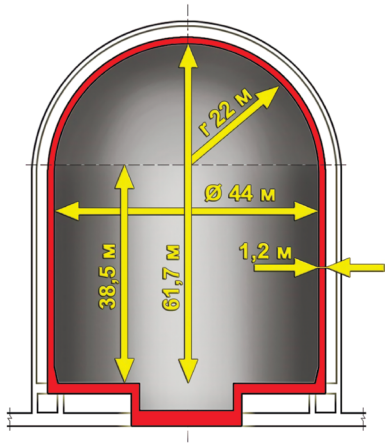
— Restricted access area (RAA) is an area with periodically serviceable compartments where the personnel access is permitted during only limited periods of time.

Internal building structures of the accident localization area are designed to consider pressure differential loads appearing during postulated leaks.

Analyses of compartment geometric characteristics and computational results show that maximum values of pressure differential in LOCA accidents appear between NAA and RAA compartments.

To protect building structures against exceeding the permissible pressure differential, the design provides for special pressure relief facilities installed in penetrations at the border between NAA and RAA. Pressure relief facilities look like louver structures which in the open state provide a clear flow passage of approximately 36 m^2. Louvers are opened when pressure differential between NAA and RAA reaches 0,01 MPa.

The internal containment is made of pre-stressed reinforced concrete in a form of a cylinder with inner diameter of 44 meters covered with a spherical dome with inner radius of 22 meters (Figure 2.2.1.7).



Inner diameter	44 m
Wall thickness:	
— cylindrical parts	1,2 m
— dome	1,2 m
Free space inside the containment	64,558 m^3
Elevation of the dome vertex	+ 61,700
Relative elevation of the cylinder bottom	+ 0,000
Elevation of the foundation slab top	— 5,400

Figure 2.2.1.7 - Geometry of the internal containment

The height of the cylindrical part is 38,5 m. The total height of the containment from the foundation slab is 61,7 m. The thickness of the cylinder and the dome walls is 1,2 m due to conditions of the biological shielding and design requirements.

In the part of the rigid connection of the cylindrical part with the foundation slab, the containment is strengthened with a haunch of 3 m in height and 1,6 m in maximum width.

A pre-stressed containment system (PSCS) consists of tendons (AP-55), support anchor blocks, channel formers, the equipment for tendon mounting and jacking, and a tendon diagnostics system.

Each tendon (AP-55) consists of 55 strands. Each pre-stressing strand has a polyethylene jacket and is located in a channel former filled with special injecting solution. Thus tendon-containment bonding is excluded.

Channel formers are metal and polyethylene tubes with outer diameter of not more than 225 mm.

A pre-stressed containment system is designed to ensure minimum permissible level of containment pre-stressing during the whole NPP operational life without strand tightening and replacing. For pre-stressing they use the orthogonal layout of reinforced components (Figure 2.2.1.8).

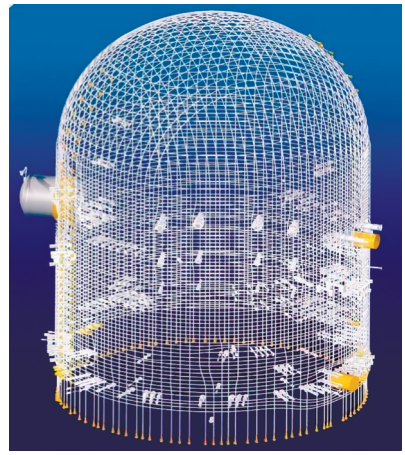


Figure 2.2.1.8 - Arrangement of pre-stressed containment system tendons

Jacking of annular pre-stressed tendons located in the cylindrical part of the containment and in the bottom part of the dome is performed on two vertical pilasters of 0,65 m in thickness endwise and 5,0 m in width.

Annular pre-stressed tendons in the cylindrical part of the containment are located with mean spacing of 700 mm in height, in the dome part with mean spacing of 1 300 mm along a generating line.

Anchoring of annular tendons is performed in turn, first on one pilaster, then on the other one (i.e. each successive tendon relative to the previous one is turned around 180 degrees) which allows to perform more homogeneous compression. A total number of tendons AP-55 is the following:

— horizontal in the cylinder – 53 tendons;

— horizontal in the dome – 15 tendons;

— vertical – 60 tendons.

For vertical cylinder and dome pre-stressing U-tendons are used covering the containment in orthogonally related directions, going through the dome and anchoring in the bottom annular gallery which is located in the foundation part of the reactor building. Annular spacing of vertical tendons is 3 degrees.

In the containment cross-sectional thickness horizontal pre-stressed tendon components are arranged in two rows at the outer containment border and vertical tendons with correction of 400 mm at the inner containment border. Annular tendons in the dome are arranged in one row.

The internal containment is additionally reinforced with common bars which are partially take force in stress concentration areas at various types of impacts including post-tensioning strand jacking and overpressure.

Monitoring of the containment deformation and the compression level includes a measurement of PSCS pre-stressed strand jacking force, stresses in containment tendons, containment concrete temperature and deformation, and wall displacement in the cylindrical part.

Advantages of the double containment

Main advantages of the double containment are the following:

1. reduction of emergency releases and the level of potential radiation effect on population and the plant personnel;

2. separation of functions: the external containment serves as a physical barrier against external impacts, the internal containment ensures leak-tightness during all modes of the plant operation including emergency modes;

3. availability of space between containments (the annulus) which allows:

— to maintain vacuum inside and confine leaks;

— distribute penetrations thus improving conditions for operation and inspection;

— clearly separate safety system channels and arrange equipment sensors, pipelines and cables;

— ensure stable annulus temperature conditions thus reducing thermal stresses in the internal containment.

2.2.1.2 Localizing steel lining

The design specifies high requirements for accident localization area leak-tightness. The value of the integral leak through the leak-tight containment of the accident localization area (ALA) should be not more than 0,3 % a day of the plenum under the containment at the design emergency pressure 0,4 MPa (overpressure).

To ensure these requirements the inner surface of the internal containment is lined with carbon steel with anticorrosion polymer protective coating.

The steel lining is a beamless plate structure connected with containment concrete by anchor components. Anchor components are rolled angles with welded anchors.

The lining thickness is 6 mm in places accessible for inspection and leak-tightness checks during operation and 8 mm in non-accessible for inspection places (along the containment foundation slab protected by the reinforced concrete plate of 600 mm).

The layout of ALA internal structures is made in a manner to protect the leak-tightening lining against potential water jets, shock waves and missiles during emergency modes.

In specific containment spots they install welded to the lining plate embedded items, locks and penetrations which are also leak-tight components.

The floor and walls of compartments intended for radioactive media or filled with boron solution at refueling (spent fuel pools, a reactor vault, pressure vessel internals vault) are lined with leak-tight corrosion-resistant steel 08X18H10T (5 mm thick for the floor and 4 mm thick for walls).

2.2.1.3 Transport, main and emergency locks

To maintain the design level of ALA leak-tightness during load transfer inside the containment or personnel entry into/exit out of the containment restricted access area, the design provides three leak-tight locks:

— transport lock for fuel and large equipment transfer (elevation + 26,3);

— main operational lock for personnel access and small loads supply (elevation + 8,14);

— emergency (standby) lock for personnel access and small loads supply (elevation + 8,14).

All locks are categorized as safety class 2 according to standard rules NP-001-15 and classified as 2NL.

Locks are designed for the following loads:

— a maximum design earthquake of force 7 on the MSK-64 scale;

— a crash of an aircraft of 5 tons in weight and 100 m/s speed on the reactor building enclosure;

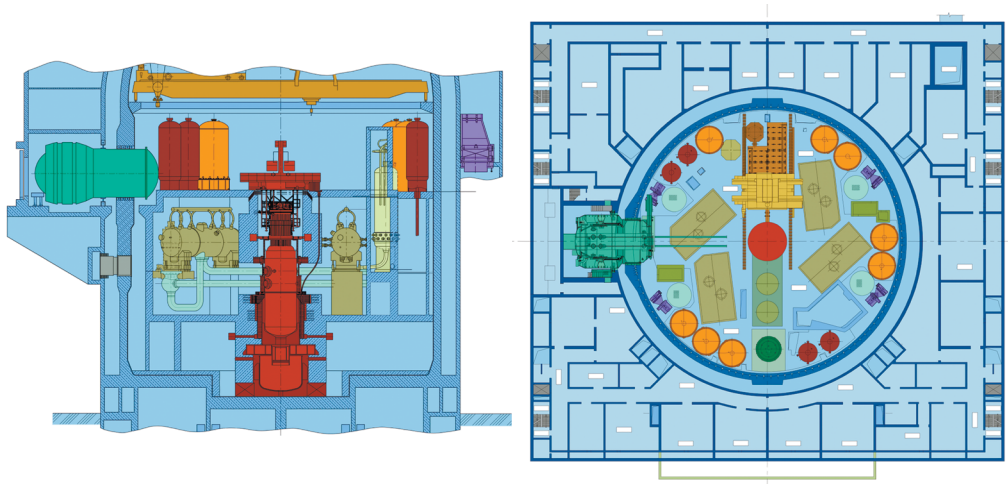
— an external air-shock wave with front pressure of 30 kPa that corresponds to the explosion of 2,2 tons of explosive material at 39-meter distance away from the building structure.

Transport lock

The transport lock provides load transfer to the Reactor Hall maintenance elevation (+ 26,3) during reactor equipment maintenance, reactor core refueling and spent fuel transfer from the spent fuel pools.

The lock axis is located at elevation + 28,9 m transversely to the spent fuel pool longitudinal axis (Figure 2.2.1.9).

Internal containment walls in the transport lock area are reinforced by thickening external and internal faces.



Reactor building.
A transport lock cross-section

Reactor building.
A scheme at elevation +26,300

Figure 2.2.1.9 - Transport lock location in the reactor building

A general view of the transport lock is given in Figure 2.2.1.10.

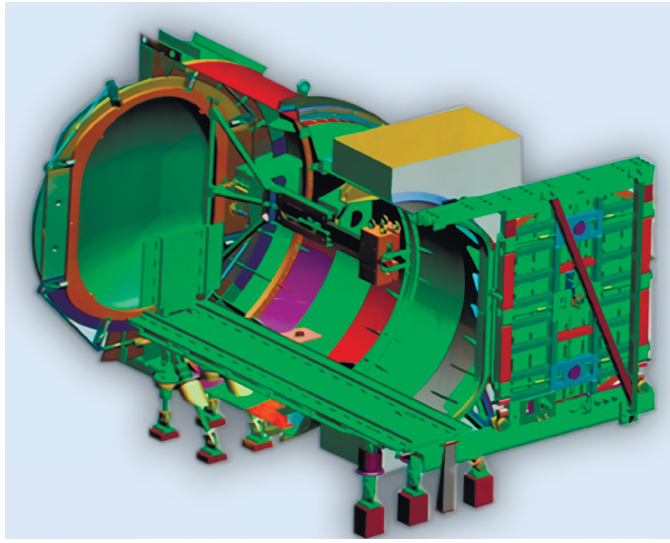


Figure 2.2.1.10 - A general view of the transport lock

An inner diameter of the lock embedded detail is 7,26 m. The embedded detail is anchored inside the internal containment body. The load of a transport trolley with equipment is transferred to external and internal containment structures. The maximum weight of the transferred load during Unit operation is 140 t, the trolley weight is about 50 t.

Permissible leaks via connections of all lock components (seals, welds) at any modes of operation and emergency modes do not exceed 0,1 m^3/day. The lock is equipped with pressure relief valves to equalize pressure before gate opening.

A gate is opened inside the leak-tight area. Lock gate opening and closing is activated by the electric-driven hydraulic actuator. Control of the electric-driven hydraulic actuator is performed via control panels located outside of each side of the lock and one panel is located inside the lock.

Manual control is provided as a standby actuator. Manual control is performed with a help of levers (handwheels) located outside of each side of the lock and inside the lock. The lock's design provides for an escape manhole from the unpressurized area side.

The main and emergency locks for personnel access

These locks allow organizing the personnel passage inside the containment without violating its leak-tightness.

The lock's design implements the following requirements:

— lock's gate opens inside the containment;

— each gate can be controlled both from inside and outside the lock;

— gate opening and closing is performed both with a help of an electric motor (main control) and manually (standby control);

— the design excludes the possibility to open one gate when another one is still open (a mechanical interlock);

— the lock is equipped with pressure relief valves to equalize pressure;

— the lock is equipped with an emergency air feeder;

— inside the lock there is emergency lighting and communications facilities;

— the lock is provided with the gate position and leak-tightness control system from the outer side towards ALA, the signals go to the Main Control Room and the Emergency Control Room.

The locks are located on different sides of the spent fuel pool (Figure 2.2.1.11):

— at elevation + 8,140 m from the internal containment side (UJA)

— at elevation + 8,400 m from the reactor building rigging side (UKA)

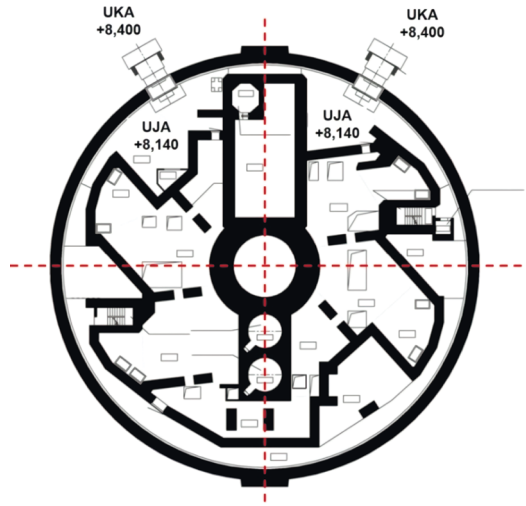


Figure 2.2.1.11– Location of the main and emergency locks for personnel access. A scheme of the internal containment at elevation + 8,140.

Axes of the main and emergency locks are located at elevation + 9,10 m. The inner diameter of embedded details of both locks is identical and equals to 2,82 m.

The lock's vessel consists of a shell to which two flanges are welded at ends with apertures of a rectangular shape 900 mm in width and 1800 mm in height for personnel passage through the lock. The vessel apertures are air-tightly closed by doors opening inside the leak-tight area. The door thickness is 120 mm.

The connection of the lock's vessel and lock's inner embedded detail installed in the leak-tight internal containment is made via welding. The leak-tightness of the welded connection is checked according to a work program after installation during containment testing as well as during Unit operation.

A gap between the lock's vessel and embedded detail in the external containment is pressurized with a help of lock's sealing.

The check of leak-tightness of all locks and doors on the external side towards ALA shall be performed after each opening-closing cycle by checking the value of pressure when air is supplied inside special cavities of the door sealing.

2.2.1.4 Leak-tight penetrations

Process and electric routes cross the containment through special penetrations ensuring leak-tightness of cross points.

The connections of penetrations with embedded details and embedded details with the leak-tight steel lining are welded.

The design uses:

— leak-tight tube penetrations for process and ventilation communication lines;

— embedded details of leak-tight tube penetrations for process and ventilation communication lines;

— leak-tight cable penetrations for control cables;

— leak-tight cable penetrations for power cables.

Types of tube penetrations are designed due to layout peculiarities of the reactor building and parameters of media flowing in pipelines.

Penetrations stay integral and functional both in normal conditions and in the whole range of seismic impacts up to the maximum design earthquake of force 7 on the MSK-64 scale occurring simultaneously with the maximum design basis accident.

Tube penetrations are made fixed and designed for all possible pipeline forces as well as forces appearing at their potential ruptures. At the same time the leak-tightness of penetration connections is not failed.

According to design features and media characteristics, tube penetrations are divided into cooled and non-cooled penetrations, made of carbon and stainless steel, intended for high and low pressure, vertical and horizontal, intended for pipeline bundles, for drains and discharge.

Power and control cable routes through ALA enclosure structures are laid via special leak-tight cable penetrations which are designed for the same pressure as the construction part of the internal containment. Penetrations are equipped with special cameras for monitoring their leak-tightness.

According to design features and media characteristics, leak-tight control and power cable penetrations are selected according to thickness of containment structures, a number of current distributors, a cable grade, nominal voltage and current, dynamic and thermal stability.

Cable lines go through the containment via leak-tight cable penetrations. The penetration design has two leak-tight barriers. For the routine visual check of leak-tightness the penetration is installed under internal overpressure of gas with a pressure gage mounted on the side from the so-called “clean area”.

Figures 2.2.1.12 and 2.2.1.13 show rough drafts of various penetration types.

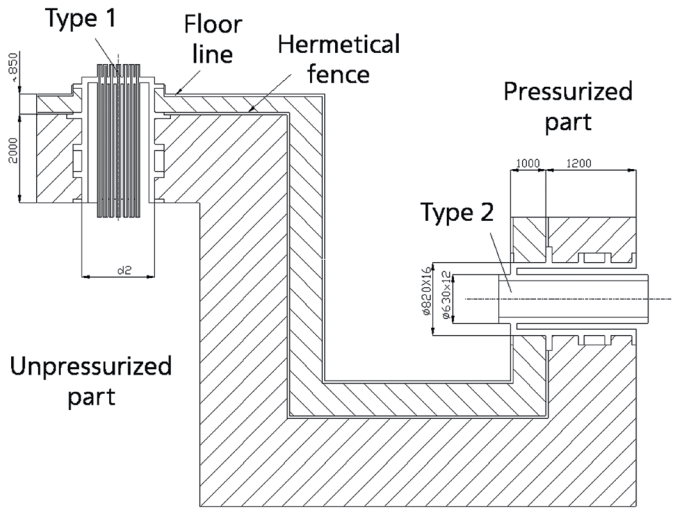
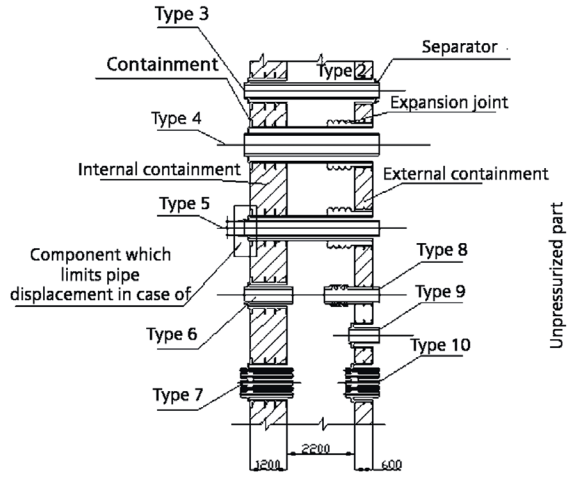


Figure 2.2.1.12 – Location of penetrations in slab at elevation 0,000 Figure



2.2.1.13 – Rough drafts of various penetration types

Figure 2.2.1.14 shows a rough draft of arrangement of tube penetrations for steam lines and feedwater pipelines.

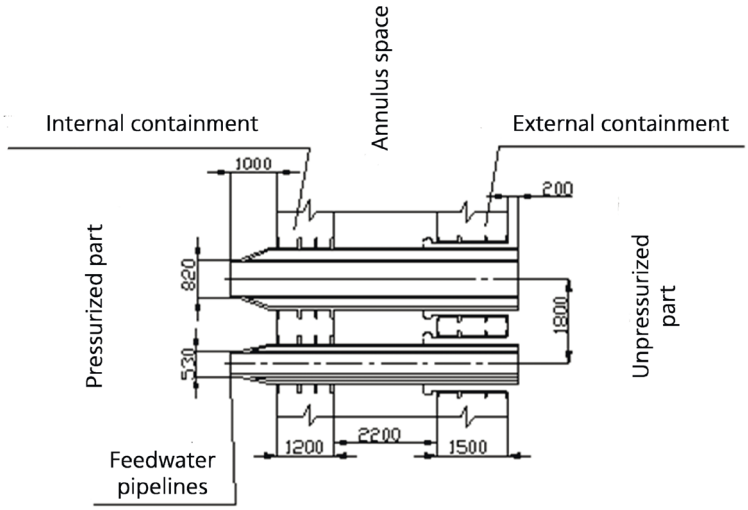


Figure 2.2.1.14 – A rough draft of tube penetrations for steam lines and feedwater pipelines

A failure criterion of penetrations is a penetration tube rupture and (or) loss of exhaust gas purification system integrity via the penetration.

Standby tube penetrations are plugged. In group penetrations intended for instrumentation systems non-used (standby) lines are plugged too.

Embedded details of penetrations are welded to the lining with lap and fillet welds forming test chambers that allow checking weld integrity by air overpressure during containment structures manufacture and installation. Service accessibility of these welded joints allows, if necessary, controlling them during Unit operation.

2.2.1.5 Isolating facilities

All pipe communications crossing the containment which are to be isolated in case of an accident to prevent radioactive material release beyond ALA, are equipped with isolating facilities.

A number of isolating facilities and places for their installation are conditioned by the demand to protect at any initial event (including a body rupture of one isolating facility) at least one barrier which prevents radioactive material release beyond ALA considering a single failure principle and a common cause failure.

The main criterion for timing the isolating facility closure is prevention of increased activity release beyond the containment limits.

2.2.1.5.1. Localizing valves

Localizing valves include isolation valves and shut-off gate valves with electric drives, pneumatic actuators or manual drives (with a lock).

When the reactor is in power, **localizing valves with a manual drive** stay in the closed position and their opening is performed according to a special permit when the reactor is in «cold shutdown» or «outage» modes.

Pneumatic localizing valves have not more than 10 seconds for closing and they are mounted beyond the containment.

To supply localizing valve pneumatic actuators with compressed air of 4,5-5,0 MPa the design provides for a compressed air system QFA whose components are categorized as safety class 3 according to standard rules NP-001-15 and classified as 3N.

In the event of the control air pressure drop in valves (failure of receivers or pipelines of the QFA system):

— a pneumatic actuator stem stays in the position «open» for not less than 10 hours (if no electric signal for valve closing comes);

— a pneumatic actuator ensures one actuation for closing the valve at the remote transmission of an electric signal to the electromagnetic switch of the air control valve responsible for air discharge from pneumatic actuator under-piston cavities.

Figure 2.2.1.15 shows general and sectional views of localizing valves with an air bottle and a pressure lock.

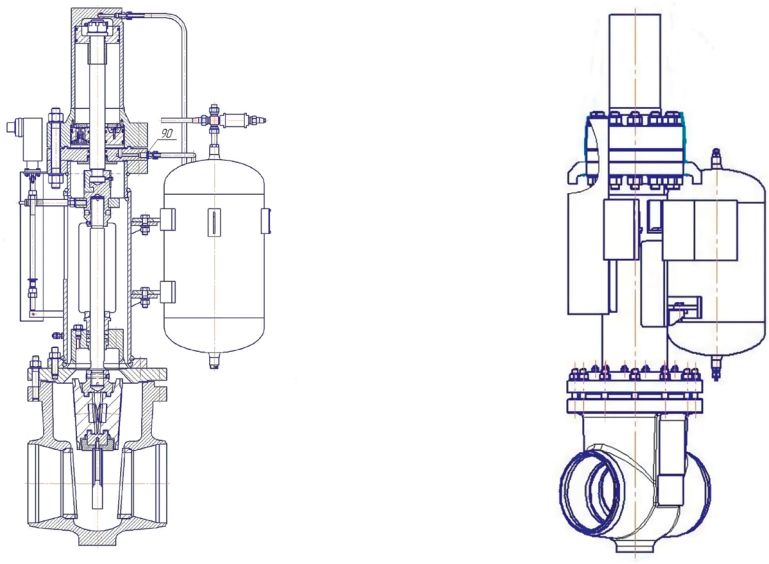


Figure 2.2.1.15 – General and sectional views of localizing valves

Electric localizing valves have not more than 60 seconds for closing and they are mounted inside the internal containment.

Figure 2.2.1.16 shows a localizing group with pneumatic and electric valves.

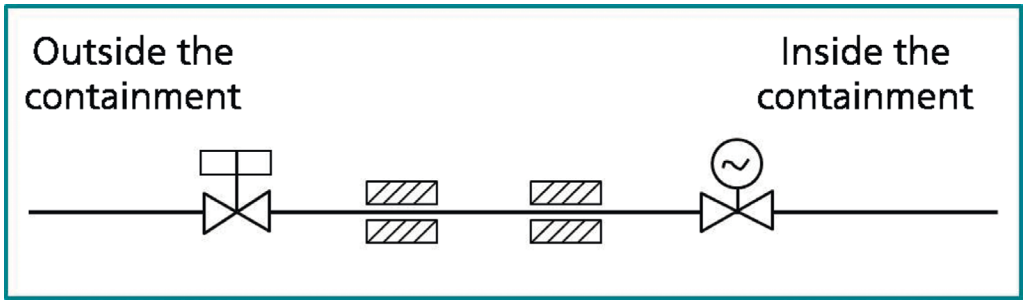


Figure 2.2.1.16 – A localizing group with pneumatic and electric valves

2.2.1.5.2 Tripping devices

The border of the leak-tight part is crossed with a great number of instrumentation sensor pulse tubes which do not have any localizing valves when passing through penetrations and walls.

To restrict leaks in leak-tight compartments and in instrumentation sensor compartments where they may appear in case of pulse line damages or sensor leakages in pulse tubes located in pressurized radioactive medium, tripping devices are installed near sampling systems.

Tripping devices are intended for multiple automatic pulse line disconnection if a leak appeared in a pulse tube behind the tripping device exceeds the installed flowrate for actuation.

Tripping devices consist of the following main details (Figure 2.2.1.17):

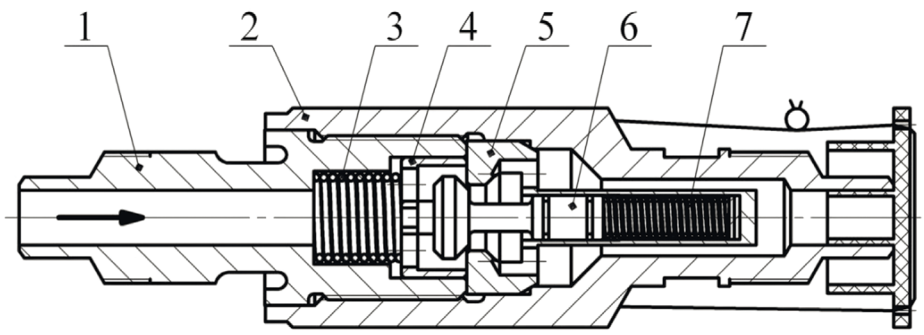
— a body and a bonnet which maintain working medium flow when the device flap is open;

— a slide valve (a sensitive component) which closes a seat opening when the installed flowrate for actuation is exceeded.

— seats with openings;

— bushings which increase medium flowrate during tube blowdown;

— springs which provide the needed flowrate for actuation.



1 - bonnet; 2 - body; 3, 7 - springs; 4 - bushing; 5 - seat; 6 – slide valve
Figure 2.2.1.17 – A drawing of a tripping device

Principle of tripping device operation

In case of no flow in a pulse tube or flowrate is less than needed for actuation, a slide valve (6) under the spring action (7) is withdrawn away from a seat (5), and it means that a tripping device is in normal work position. When the flowrate starts exceeding the installed limits due to a pulse tube rupture or an excess leak, a slide valve (6) under force closes a seat opening (5).

2.2.1.5.3 Containment localization during loss of coolant

The control of the electric and pneumatic localizing valve position in emergency is carried out by an operator via safety panels in the Main Control Room/Emergency Control Room.

Localizing valve closure is performed automatically on initiating signals of the safety control system (FSKU SV11).

At pressure increase in the internal containment up to 0,3 kPa, localizing valves in air ducts of air-supply and exhaust ventilation systems of internal (systems KLA) and external (systems KLC) containments are automatically closed (Figure 2.2.1.18).

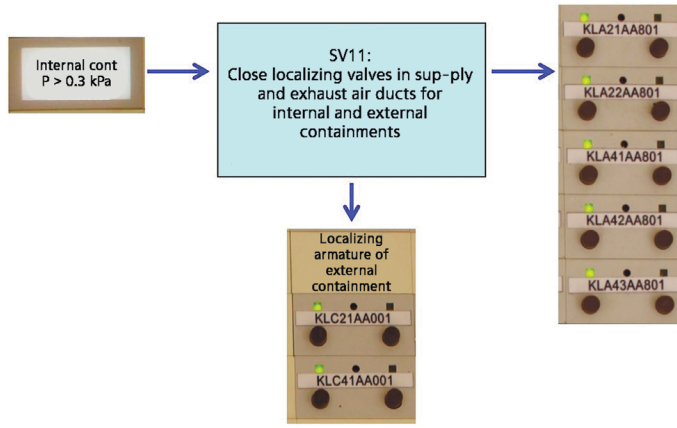


Figure 2.2.1.18 - SV11 – localizing valves closure in air ducts of ventilation systems

Valves in air ducts are closed at relatively low overpressure inside the containment as air ducts have a large cross-section, and medium outlet through exhaust ventilation systems is made directly beyond boundaries of the double containment.

Localizing valves **in process pipelines** are closed when one of the following criteria confirming the primary coolant leak, is achieved:

— the limiting temperature differential (ΔT_{s-1}) between the primary circuit saturation temperature and the hot leg maximum temperature is less than 8 °C

or

— the containment overpressure is more than 30 kPa.

To exclude the storage battery overload due to large starting current of simultaneously closing localizing valves, a part of localizing valves in process pipelines is closed with a 5-second delay (Figure 2.2.1.19).

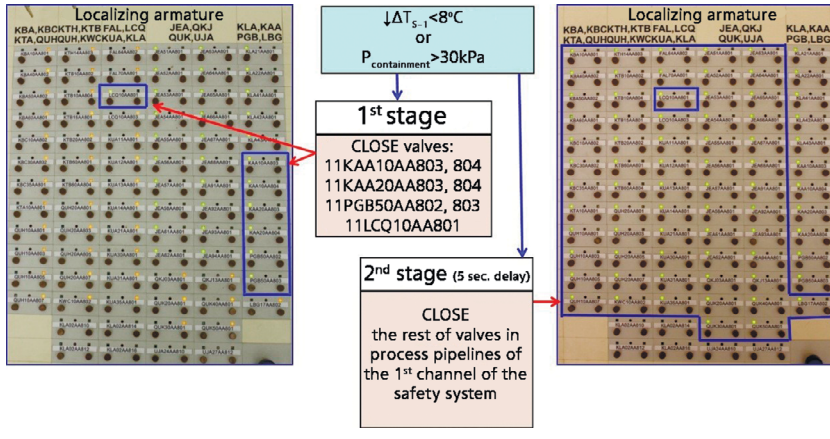


Figure 2.2.1.19 - SV11: closure algorithm for localizing valves in process pipelines (1st channel of the safety system)

The above-mentioned specific feature is designed only for the first channel of the safety system. The second channel uses mostly pneumatic valves which have low starting current, so all valves are closed simultaneously (Figure 2.2.1.20).

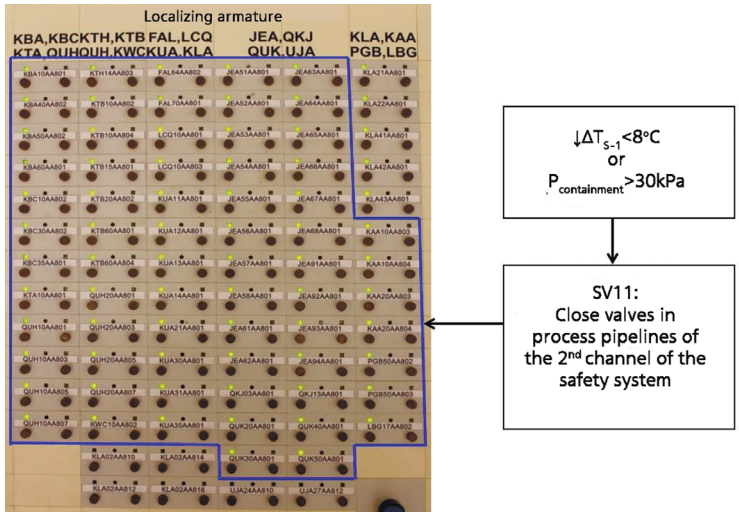


Figure 2.2.1.20 - SV11: closure algorithm for localizing valves in process pipelines (2nd channel of the safety system)

To exclude erroneous actions of the plant personnel, there is a prohibition on human intervention in controlling localizing valves for 30 minutes from the moment of initiation of the SV11 control function. The prohibition is cancelled at coincidence of three conditions:

— 30 minutes have passed;

— containment overpressure is less than 20 kPa;

— no-signal «ΔT_{S-1} less than 8 ºC».

Containment localization at failure of safety control system by process parameters (SCSPP)

At SCSPP failure the following initiating signals on localizing valve closure are generated:

— “Containment overpressure more 0,4 kPa” for valves in air ducts of air-supply and exhaust ventilation systems of internal and external containments;

— “Containment overpressure more 0,4 kPa” for valves in process pipelines.

2.2.1.6 Operational modes of a leak-tight enclosure system

A leak-tight enclosure is available in the following modes:
— normal operation;
— violation of heat removal;
— «minor» break emergency;
— «large» break emergency;
— beyond design basis accident;
— severe accident with fuel meltdown.
Parameters of medium in the leak-tight enclosure at various modes of operation are provided in Table 2.2.1.2.

Table 2.2.1.2 – Parameters of medium in the leak-tight enclosure

Name	Dimension	Value
1. Normal operation		
Temperature	°C	15 — 60
Absolute pressure	MPa	0,085 – 0,103
Humidity	%	90
Absorbed dose rate	**Gy/h**	**0 – 1,0**
Volumetric activity	Bq/l	$7,4 \times 10^4$
2. Violation of heat removal		
Temperature	°C	30 — 75
Absolute pressure	MPa	0,069 – 0,118
Humidity	%	up to 100
Mode duration	h	up to 15
Mode frequency	1/year	1
Absorbed dose rate	Gy/h	0 – 1,0
Volumetric activity	Bq/l	$7,4 \times 10^4$
3. Minor break: 25 – 100 mm in bore		
Steam-gas medium temperature in containment volume	°C	up to 90

Name	Dimension	Value
Maximum permissible overpressure in containment	MPa	up to 0,07
Humidity	%	steam-gas mixture
Emergency mode duration	h	up to 5
Post-accident temperature	°C	20 — 60
Post-accident pressure	MPa	0,09 – 0,12
Mode frequency	1/year	0,5
Absorbed dose rate	Gy/h	0 – 1,0
Volumetric activity	Bq/l	2×10^7
4. Large break: 100 – 850 mm in bore		
Steam-gas medium temperature in containment volume	°C	up to 150
Maximum permissible overpressure in containment	MPa	0,4
Vacuum in containment	kPa	not more than 10
Humidity	%	Steam-gas mixture)
Hydrogen concentration	%	less than 2 %
Emergency mode duration	h	up to 24
Post-accident temperature	°C	20 — 60
Post-accident pressure	MPa	0,09 – 0,12
Post-accident mode duration	day	30
Mode frequency	1/year	0,033
Absorbed dose rate	Gy/h	less than 1000
5. BDBA without fuel meltdown		
Maximum possible steam-gas medium temperature in containment volume	°C	— during 5 min: 210 °C — long-term: 150 °C (linear decrease down to 100 °C during 24 hours)
Maximum permissible overpressure in containment	MPa	0,4 (linear decrease down to 0,15 during 24 hours)

Name	Dimension	Value
Humidity	%	steam-gas mixture
Emergency mode duration	h	up to 24
Post-accident mode duration	day	30
Mode frequency	1/year	less than 10^{-7}
6. Severe accident		
Maximum possible steam-gas medium temperature in containment volume	°C	— short-term: 210 °C. — long-term: 180 °C
Maximum permissible overpressure in accident localization area *	MPa	0,6
Humidity	%	steam-gas mixture
Emergency mode duration	h	up to 72
Post-accident mode duration	day	30
Mode frequency	1/year	less than 10^{-7}

* Maximum permissible pressure in ALA in the event of severe accidents is 50 % higher than the estimated pressure for design basis accidents:

0,4 MPa (excessive) x 1,5 = 0,6 MPa (excessive).

To implement acceptance criteria according to ALA parameters, the design provides for active and passive safety systems:

— for the internal containment: pressure suppression and heat removal system (a sprinkler system), emergency hydrogen removal system;

— for the external containment: exhaustion and gas-aerosol purification system.

For severe accidents the design provides for a molten core catcher and corresponding heat removal systems.

2.2.2 SPRAY SYSTEM (JMN)

2.2.2.1 Spray system (JMN) application and its flow diagram

Spray system JMN is meant to decrease the parameters inside the containment (pressure and temperature) and fixation of radioactive iodine contained in steam-gas atmosphere containment.

The system is designed with no single point of failure and provides the carrying out of its functions at any accidents initial event with application of one active or passive element, which is independent of failure initial event, which contains mechanical moving parts, or one independent of personnel failure initial event. The system does not contain failures that depend on initial events.

The design presupposes two-channel system structure 2 x 100 % — one operational channel carries out the designed functions to full extend.

The system channels are physically separated and fully independent of each other. Pipelines and system valves are located in reactor building KA and UJA (Figure 2.2.2.1).

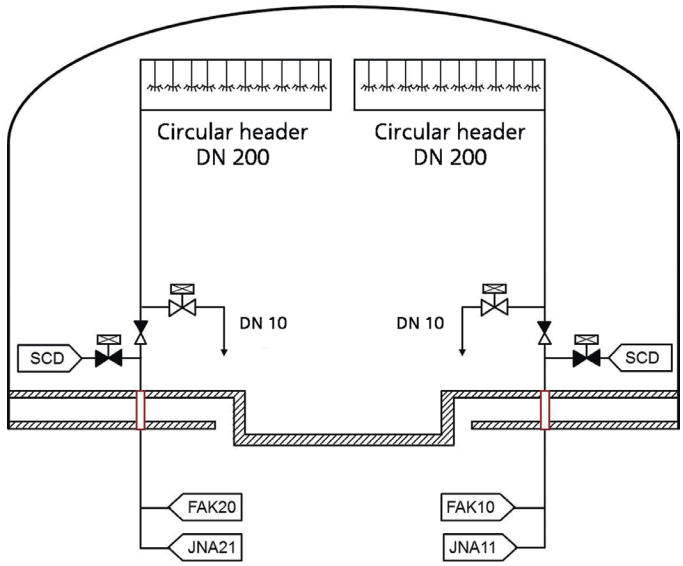


Figure 2.2.2.1 – JMN flow diagram

Each channel includes:

— ring header (DN 200) with 10 spraying jets installed on it (DN 50);

— pipeline (DN 200) used for borated water supply on ring header (check valve is

installed on the pipeline);

— pipeline (DN 200) used for borated water supply from FAK 10-20 system;

— pipeline (DN 200) used for borated water supply from JNA system;

— drain pipeline DN 10 with normally open manual valve;

— pipeline DN 25 used for air pressure supply from SCD system with normally closed manual valve – for pneumatic test performance;

— moisture control sensor – for monitoring the water situation in pipeline used for borated water supply on ring header after check valve.

Drain pipeline DN 10 with normally open manual valve is anticipated on the pressure part of the spray pipeline after the check valves. It is meant to remove the boric acid solution after the system shut down in order to exclude the crystallization of boron on the check valve.

Existence of continuously open drainage does not detract the characteristics of spray system because supply flow rate through the drainage does not exceed 3 m^3/hour when the total supply flow rate of boric acid solution in every system channel is 300 — 310 m^3/h.

Distributing headers with spraying jets are located under the dome of reactor building at elevations + 53,150 (1st channel) and + 52,35 (2nd channel) (Figure 2.2.2.2).

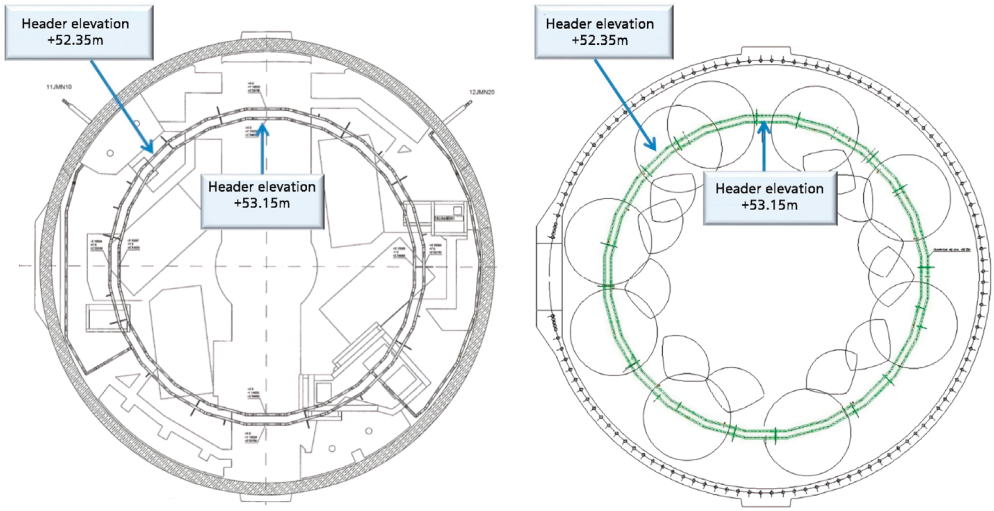


Figure 2.2.2.2 – Spray system headers layout plan and spraying jets location map

The annual arrangement of the sprayers in liaison with particular direction of spray provides the most effective containment reflux.

In this case in the reactor building center (elevation + 26,3) the maximum reflux is reached with radially cosinusoidal reflux distribution.

2.2.2.2 Flow diagram of spent fuel pool (SFP) FAK10-20 cooling system

Spray system JMN does not include pumps, heat exchangers and tanks. It consists of pipelines, spraying jets and valves.

The function of spray pumps in the NVNPP 2 design is performed by the pumps of SFP FAK10-20 cooling system (figure. 2.2.2.3).

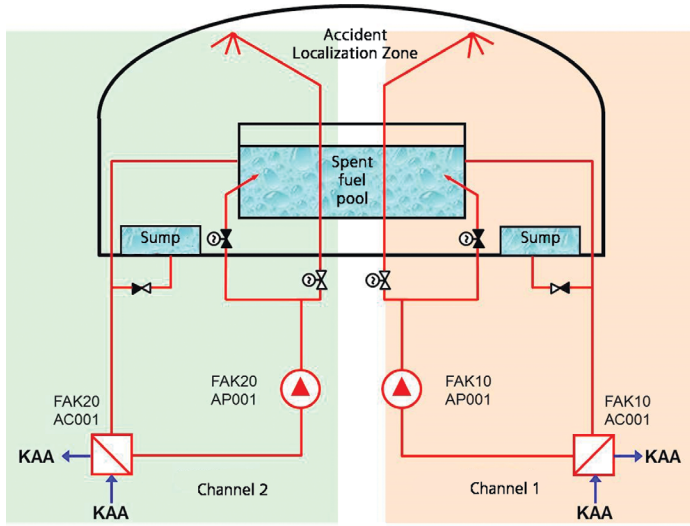


Figure 2.2.2.3- Simplified flow diagram of water supply to the spray system spraying jets from the pumps of spent fuel pool FAK10-20 cooling system

The pump of spent fuel pool FAK10-20 cooling system provides the injection of boric acid solution to the containment enclosure airspace with the flow rate of no less than 300 m^3/hour for one spray header.

The heat exchanger of FAK10-20 system is used for the purpose of spray water temperature decrease up to 70 °C. In its turn, it is cooled by the intermediate circuit system of UJA (KAA) building essential consumer.

Alkaline solution from alkali tanks FAK14(24)BB001 is mixed with boric acid solution, which is supplied to the spraying jets, with the purpose of chemical fixation of iodine radioactive isotopes ($^{131}I \div ^{135}I$). Mixing is produced in favour of spent fuel pool cooling system water pump jets operating (figure. 2.2.2.4.)

Heat exchangers FAK10(20)AC001, pumps FAK10(20)AP001, alkali solution tans for radioactive iodine mixing FAK14(24)BB001, water pump jets are installed in building UKA at elevation "— 5,400 m".

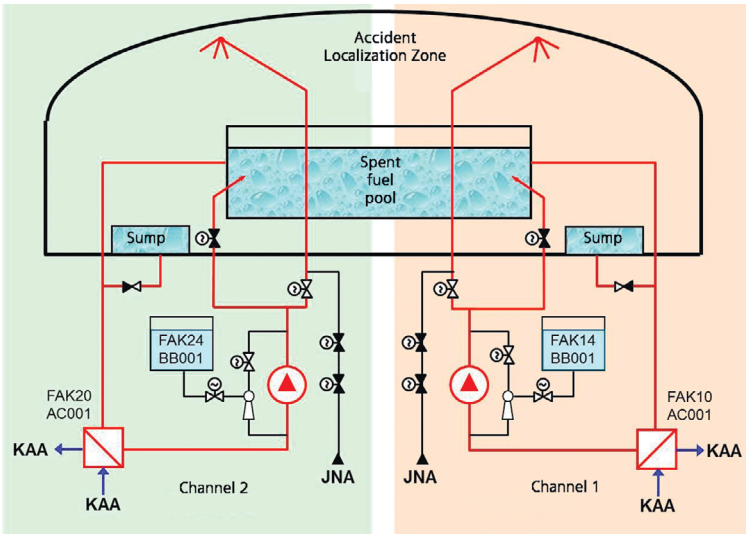


Figure 2.2.2.4 – Simplified flow diagram of chemical reagent supply to the FAK10-20 spent fuel pool cooling system suction

The operator has an opportunity to deliver the water to the spraying jets from JNA system – emergency and planned cooldown of primary circuit and spent fuel pool cooling during the beyond design accidents with system FAK 10-20 pumps failure. JNA system heat exchanger is used when the spray system operates in liaison with JNA system, but the function of radioactive iodine mixing is not carrying out due to the lack of water pump jets and alkali tanks in the JNA system composite.

Flow diagram of JMN system spray first channel and FAK10 spent fuel pool cooling system is demonstrated in Figure 2.2.2.5. The flow diagram includes:

— line of borated water intake from spent fuel pool or containment sump (through filter device) ;

— line of water supply to the spring jets – system operating during the accidents such as LOCA;

— line of cooling water return to the spent fuel pool with the flow rate of no less than 400 m^3/hour through the valve 11FAK10AA002 (system operating in cooling the spent fuel pool mode) or with the flow rate of no less than 20 m^3/hour through the valve 11FAK10AA001 (system operating in spent fuel pool emergency make-up mode);

— line of FAK10AP001 pump recycle – in order to exclude the pump operating in non-flow mode;

— line of sample taking KUA40;

— line of chemical reagents supply to the alkali storage tank from KPJ system;

— line of water supply to the reactor through the interconnecting pipelines of hydraulic tanks of the 2nd stage (JNG10-40) in the mode of the beyond design basis accident management (only on the 1st channel of FAK10).

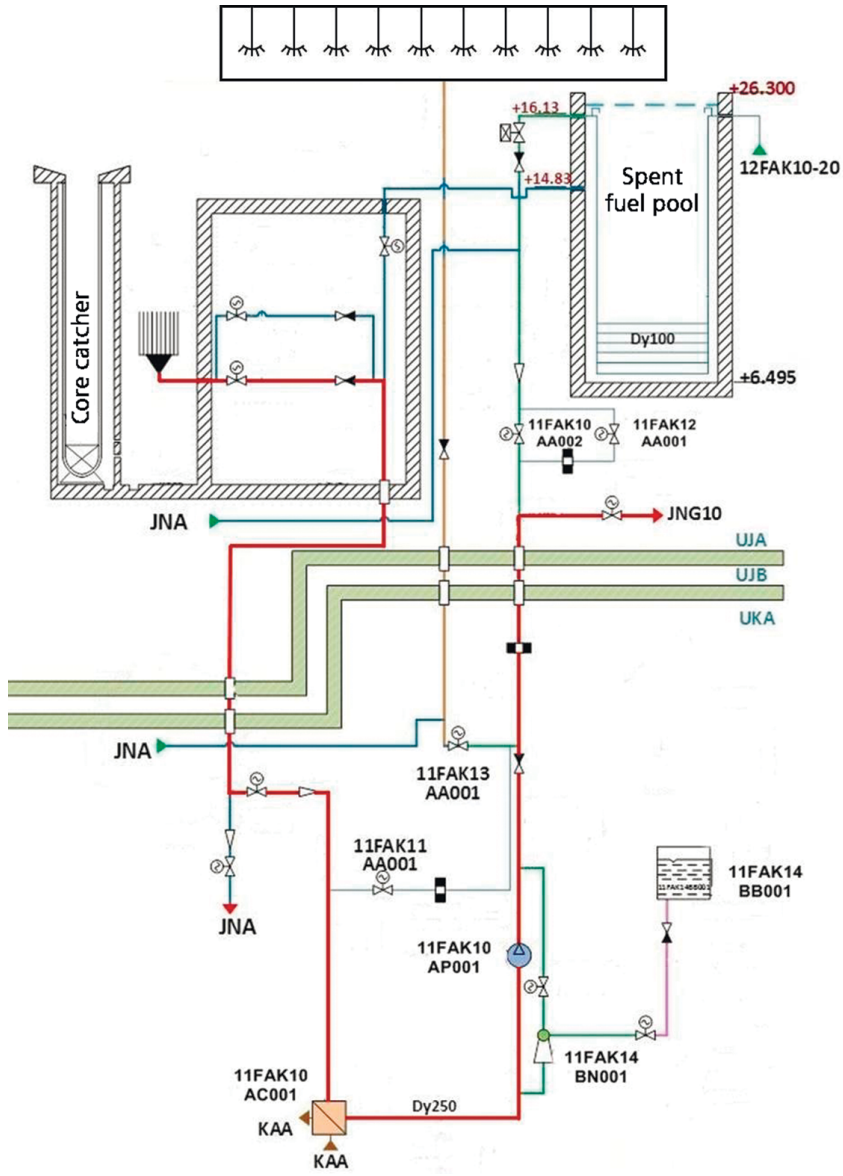
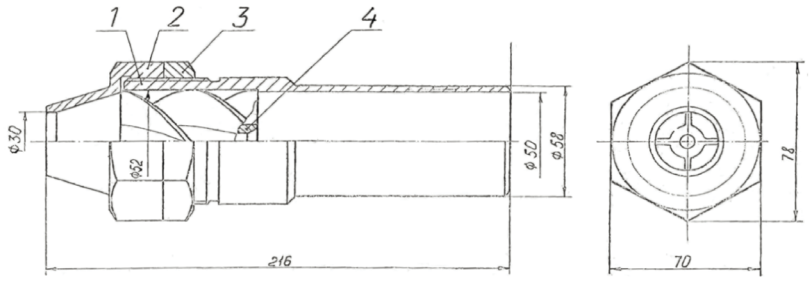


Figure 2.2.2.5 – First channel of JMN spray system and FAK10 spent fuel pool cooling system flow diagram

2.2.2.3 Design and technical specifications of JMN and FAK systems equipment

Spray system spraying jet.

The design of spraying jet is demonstrated in Figure 2.2.2.6. The nut and the nozzle are rotated on the spraying jet shell. The nozzle diameter is 30 mm.



1. Shell
2. Nozzle
3. Nut
4. Whirler



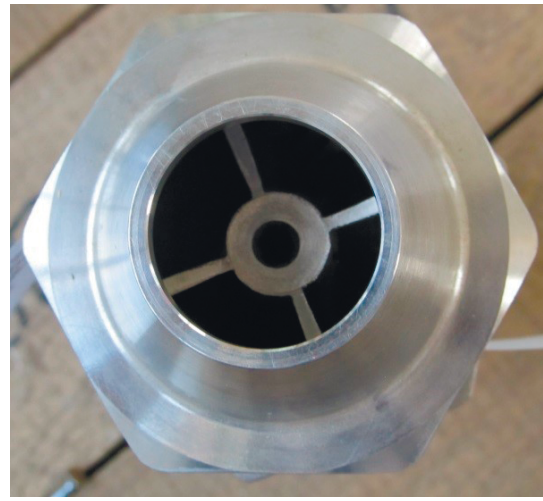


Figure 2.2.2.6 – Spray system spraying jet design

Inside the shell, there is the screw-type swirler with a length of 64 mm and the outer radius of 52 mm (Figure 2.2.2.7).

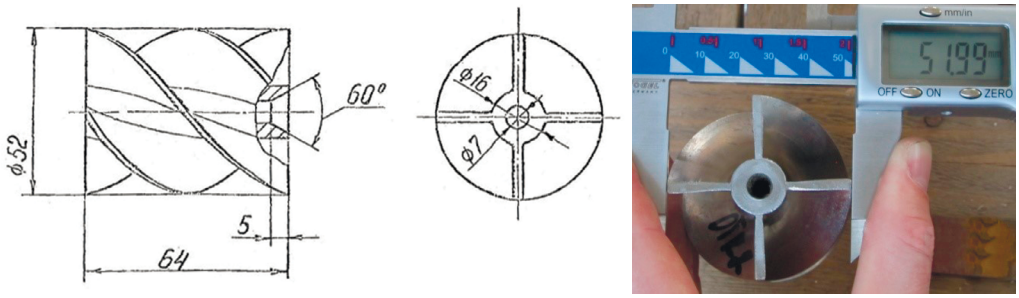


Figure 2.2.2.7 - Spraying jet whirler

Design flow rate through one spraying jet is 30 m^3/hour while the pressure differential on the spraying jet is 0,1 MPa.

Spent fuel pool cooling system pump FAK10-20AP001.

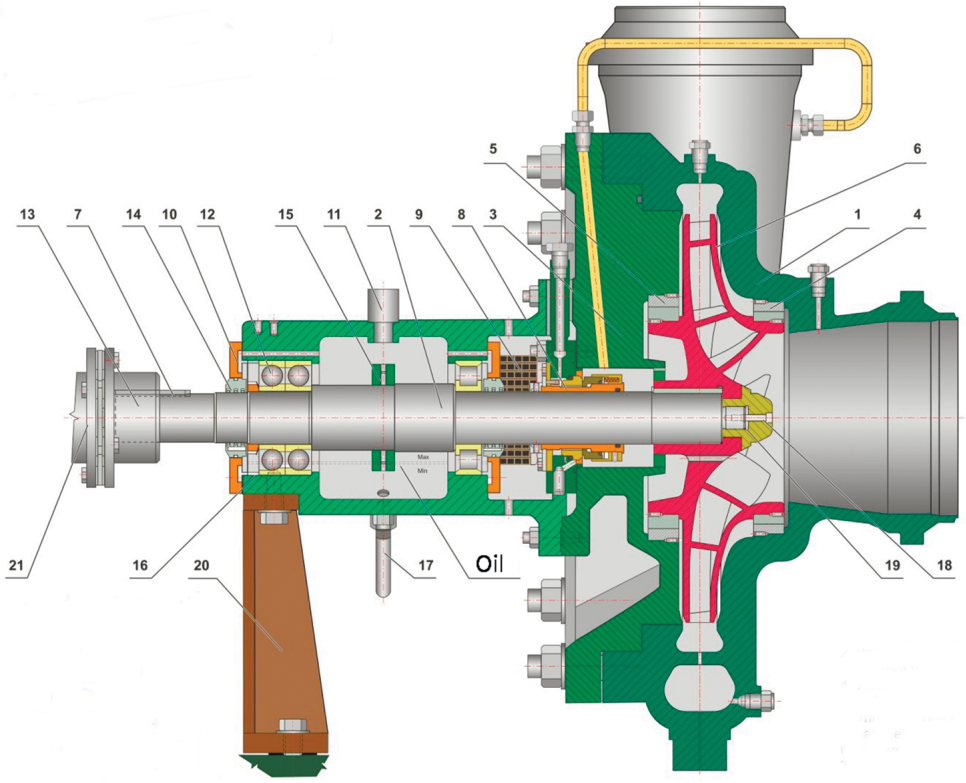
Spent fuel pool cooling system pump is designed for:

— borated water supply to the spray system spraying jets with the flow rate of 300 m^3/h during the design basis accidents which are connected with the increase of overpressure of more than 30 kPa in the containment;

— establishing the circulation in the spent fuel pool cooling circuit with the flow rate of no less than 400 m^3/h under all operating conditions without the activation of the primary circuit burst protection;

— spent fuel pool makeup with the flow rate of 20 m^3/h while the heat is abstracted from the fuel to the spent fuel pool in the evaporative mode.

Pump FAK10(20)AP001 – rotodynamic, horizontal, overhang, one-stage pump with axial liquid supply and spiral liquid extraction (Figure 2.2.2.8).



1. Spiral pump casing
2. Pump rotor shaft
3. Pump head
4. Front sealing ring of working wheel
5. Back sealing ring of working wheel
6. Working wheel
7. Key
8. End sealing
9. Protective casing of end sealing shaft unit
10. Bearing cap
11. Bleeder plug
12. Bearing
13. Joint
14. Unit of beating sealing
15. Bearing forced lubrication ring
16. Distance ring
17. Plug
18. Bolt
19. Nut
20. Strut support
21. Electric motor semi-coupling
22. Bearings casing

Figure 2.2.2.8 - FAK10-20 pump design

Rotor is installed in one neck radial bearing and in two radial-axial bearings with oblique touches that perceive radial and axial force.

The single-type butt seal with cleansing and cooling by the own pumped medium densifies the shaft. Cleansing and rinse pipeline connects the pump discharge with the seal.

Pump technical specifications are depicted in table 2.2.2.1.

Table. 2.2.2.1 FAK10-20AP001pump specifications

Parameter	Value
Supply operating range	68 - 500 m^3/hour
Operating range at the head	68 - 84 m
Guaranteed supply	430 m^3/hour
Guaranteed head	80 m
Maximum pressure of operation medium on intake	no more than 1 MPa
Allowable cavitation margin	no less than 3,5 m
Pumped liquid temperature	20 - 150°C
Pump capacity at 20 °C	131 kW
Pump efficiency at 20 °C	78,7 %
Rotation frequency	1487 rpm

Pump unit is delivered together with bed frame of supporting construction which is attached to the foundation by anchor bolts.

Unit pumps are equipped with I&C for their safe operating, monitoring and management while performing the tasks.

Spent fuel pool cooling system pump operating is prohibited:

— during the non-flow mode if it is lasts more than 2 minutes;

— at pumped medium temperature is no higher than 150 °C;

— at absolute pressure on the intake is less than 0,035 MPa (Cavitation margin);

— at pump delivery flow is less than 68 m^3/h and more than 500 m^3/h;

— at pump bearings temperature is no higher than100 °C;

— at motor bearings temperature is no higher than 90°C;

— at stator coil temperature is no higher than 120 °C;

— during the bleeding of more than 0,03 l/h through the butt seal while the pump is operating;

— during the bleeding of more than 0,015 l/h through the butt seal while the pump is not operating;

— while the pump bearings vibration speed is higher than 4 mm/s, motor – higher than 4,5 mm/s;

— while the pressure at the operating pump head is lower than 0,6 M.

Spray solution tank FAK10-20BB001.

The tank is designed for the storage of alkaline solution that fixates radioactive iodine.

The tank has the form of cylindrical vessel of fully welded design with elliptic roofing and bottom.

The tank has the following constructional parts: air ventilation, filler, pump nozzle, two level measurement nozzles, hydroseal overflow and access hatch used for checking and cleaning the inside area with a diameter of 800 mm (Figure 2.2.2.9).

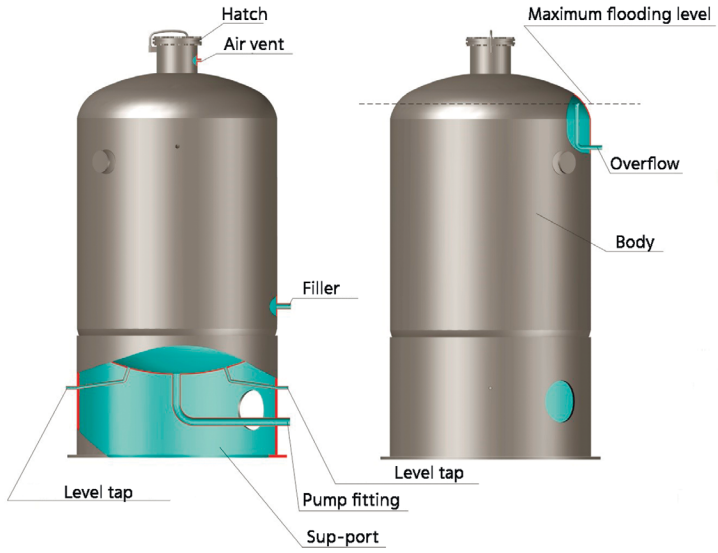


Figure 2.2.2.9 – Spray solution tank design

Spray solution tank technical specifications:

— Total storage capacity: 7 m^3

— Available storage capacity: 6,3 m^3

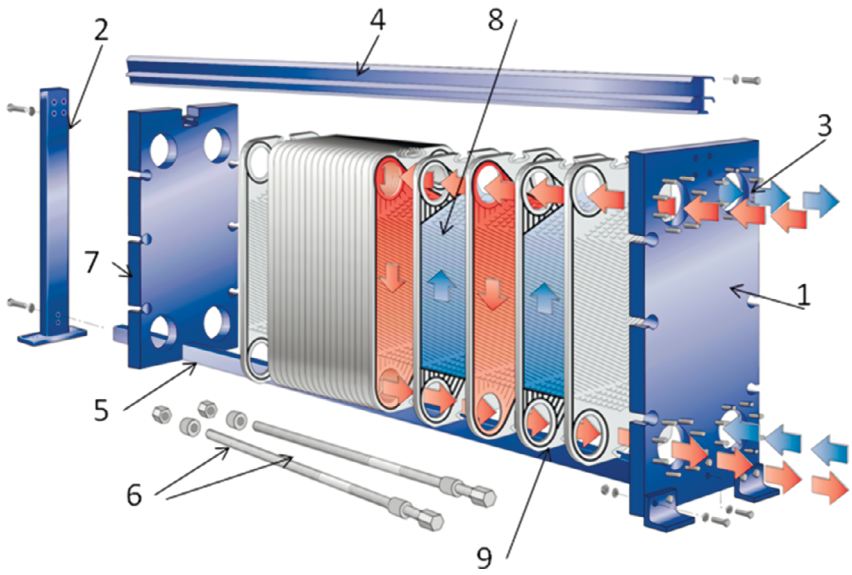
— Operating pressure: 0,024 MPa.

— Operating temperature: 40 °C.

Spent fuel pool cooling heat exchanger FAK10(20)AC001.

T20-PFG plate-type heat exchanger is used as a spent fuel pool cooling heat exchanger in the NVNPP-2 design.

The heat exchanger is made of a row of similar plates, which are collected into the packet (channel plates) and tighten between the original and pressure plate with the help of tie bolts (Figure 2.2.2.10).



1.	Original plate	4.	Bearing bar	7.	Pressure plate
2.	Supporting bar	5.	Leading bar	8.	Channel plate
3.	Connection	6.	Tie bolts	9.	Sealing

Figure 2.2.2.10 – Plate heat exchanger design

Channel plate has four holes for heating and heated mediums. There is a groove on perimeter of the channel plate, in which the joint packing is installed.

The packing has a special shape that allows to block the medium supply to two out of four plate holes.

Channel plates that are flipped through 180° one by one form counter channel for medium moving. Channels for medium moving have curly form in section.

Curly form of the plates is the reason for turbulent stream in heat exchanger channels. The turbulence offers big resistance to the fallout formation on the plate surface.

Heating and heated mediums inputs and outputs are located on the part of the original plate. During the passing through the apparatus warm medium carries off the heat to the thin barrier which, in its turn, is cooled by the cool medium from the other side.

Heat-exchanger technical specifications are depicted in table 2.2.2.2.

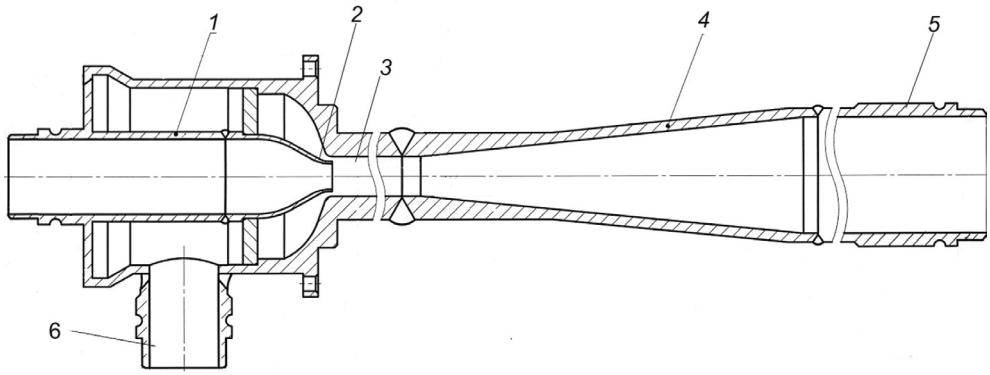
Table. 2.2.2.2 FAK10-20AC001 heat-exchanger specifications

Parameter	Value
Heat exchanger type	T20-PFG
Heating side (spent fuel pool water, containment sump)	
Discharge	410 t/hour
Pressure drop	24,64 kPa
Input/end medium temperature	48 / 30°C
Design pressure	1 MPa
Design temperature	150°C
Heated side (KAA system distillate)	
Discharge	700 t/hour
Drop	70,54 kPa
Input/end medium temperature	28 / 38,5 °C
Design pressure	1 MPa
Design temperature	150°C

Water-jet pump FAK14(24)BN001

Water-jet pump is designed for reagents supply for the suction of the solution spent fuel pool pump in order to bind the radioactive iodine in case of the primary coolant leak accidents.

Water-jet pump design is depicted in Figure 2.2.2.11. The pump has no moving parts and it is totally enclosed.



1. Motive fluids supply
2. Nozzle
3. Mixing chamber
4. Diffuser
5. Discharge nozzle
6. Pumped liquid inlet

Figure 2.2.2.11 – Water-jet pump design

Operating principle. Active flow is transferred to the water-jet pump inlet from FAK10 (20)AP001 pump discharge. Active flow potential energy (pressure) converts into kinetic energy in the contracting nozzle (pose2). Vacuum is produced due to the pressure decline in nozzle and it provides the reagents solution from the alkali storage tank (passive flow) inflow through the intake nozzle (position 6).

In the mixing chamber the active flow, mixing with the passive flow, gives its' energy with passive flow (position 3). After mixing liquid goes to diffuser (position 4), where the kinetic energy of active and passive flows mix transfers into the potential energy. Thus the flow speed drops while the pressure increase takes place at the pump discharge.

Process fluid supply to the pump is carried out along the pump axis, pumped fluid supply – perpendicularly to the pump axis, pumped fluid extraction – along the pump axis (Figure 2.2.2.11).

Pump operation point depends on pressure and supply of operating and pumped medium, and it depends on pump discharge pressure.

Pump technical specifications are depicted in table 2.2.2.3.

Table. 2.2.2.3 FAK14(24) BN001 water-jet pump technical specifications

Parameter	Value
Power medium consumption	50 m^3/hour
Pumped medium consumption	10 m^3/hour

Parameter	Value
Process fluid overpressure at pump inlet	1,5-0,83 MPa
Pumped fluid overpressure at pump inlet	0,03 MPa
Back pressure at pump discharge	0,7-0,03 MPa
Process fluid design pressure: — at pump inlet —at pump discharge	 2,0 MPa 1,0 MPa
Nominal diameter: — of operating medium nozzle —of pumped liquid inlet —of discharge nozzle	 57 57 89

2.2.2.4 JMN and FAK10-20 system parameters monitoring on MCR and Stand-by Shutdown panels

The following control and management instruments are available on safety panels in MCR and Stand-by Shutdown panels for JMN and FAK10-20 systems (Figure 2.2.2.12):

— alarm (FAK pump accident condition, absent of overpressure at pump inlet, level recession in alkali storage tank – less than 2 m, poor borated water supply to the spent fuel pool, low water level in the spent fuel pool, high water temperature in the spent fuel pool– higher than 70 °C);

— indicator «Raskhod FAK10(20)AP001»;

— system mimics diagram.

Pump and motordriven valves system remote control switchers, also the monitors of: level in the spent fuel pool, level in the alkali storage tank, water temperature in the spent fuel pool, temperature before and after the heat exchanger of FAK system are positioned on the mimics diagram.

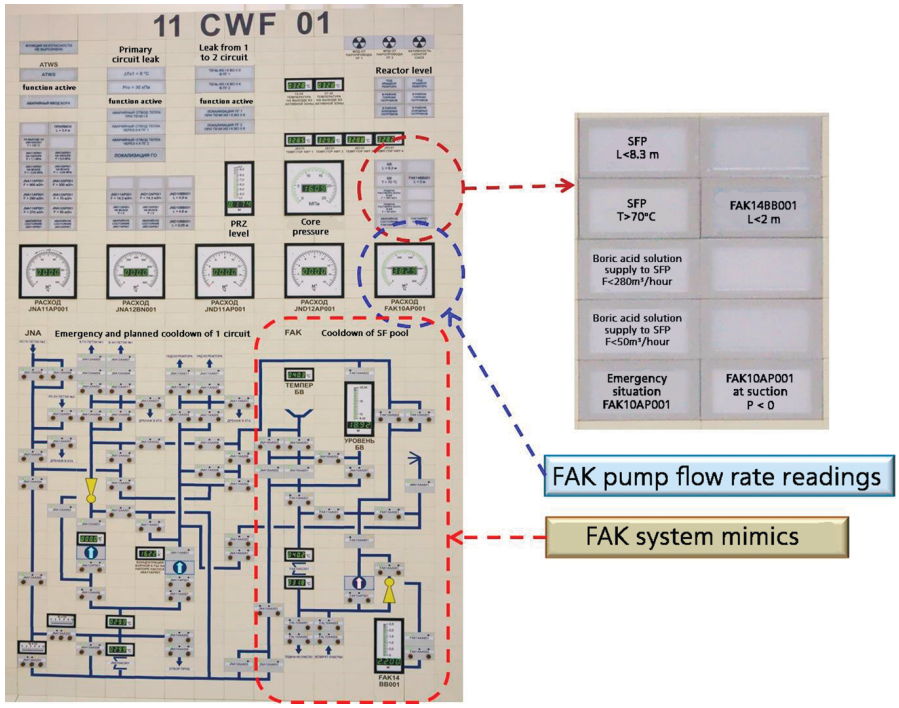


Figure 2.2.2.12 – Controls of JMN and FAK10-20 system operation on MCR/ Stand-by shutdown panels.

2.2.2.5 Spray system (JMN and FAK1020) operation under normal operating conditions

Under normal operating conditions, when unit operates at full power and unit shutdown, FAK10-20 system provides the heat removal from fuel which is located in the spent fuel pool, When there is no need for two channels of the system to operate simultaneously in this mode, one of the channels does not work but it remains in stand-by mode. Permanent cooling water passage through the heat exchangers of spent fuel pool is provided. At this time, the valve that is responsible for water supply to spraying jets and reagent solution for radioactive iodine fixation supply is in good repair, electric circuit is complete, I&C system and alarm are in working order.

When taking FAK10-20 system equipment out of service for repair, heat removal from the spent fuel pool can be carried out by JNA system. FAK 10-20 system channel and JNA system channel must operate in order to abstract heat from fuel in spent fuel pool in full core offload mode

2.2.2.6 Spray system (JMN and FAK10 20) operation in the mode of design basis accidents like LOCA

Water is supplies to the containment spraying jets automatically after trigging signal “overpressure containment is higher than 30kPa” from the safety control system. This signal initiates the operation of control function such as CA11 and BG11, CA12, CB12.

CA11 and BG11 functions provide the water supply from spent fuel pool cooling pumps to spraying jets (Figure 2.2.2.13).

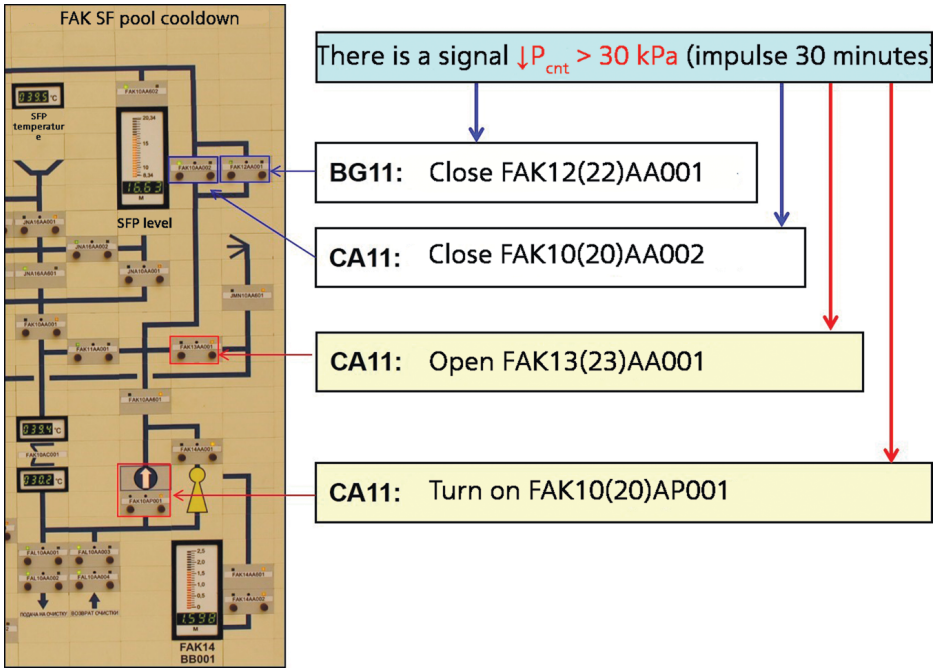


Figure 2.2.2.13 – Management of water supply from spent fuel pool cooling pumps to spraying jets

CA12 function operates the valve on FAK1020pump recircularization line and does not allow the pump to operate in non-flow mode (Figure 2.2.2.14).

Protection opens the valve when the flow rate through the pump is less than 60 m^3/hour with delay in 30 seconds and closes the valve when the flow rate becomes more than 120 m^3/hour.

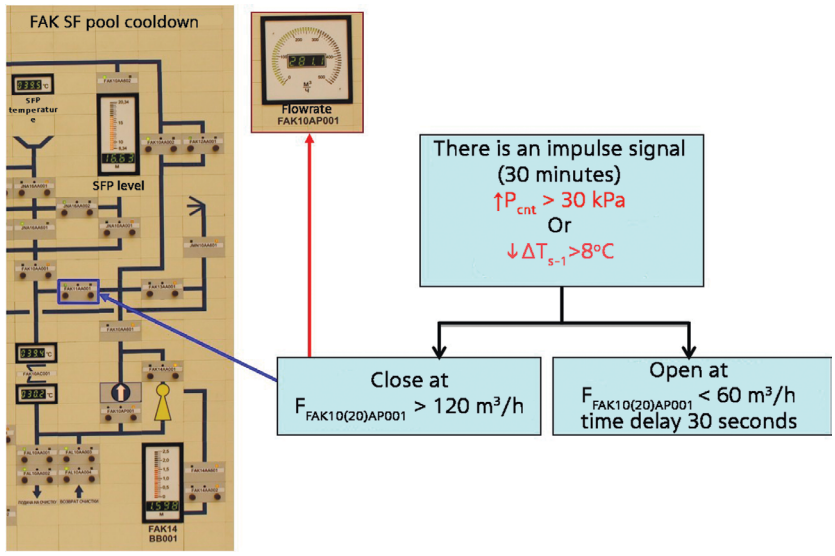


Figure 2.2.2.14 - CA12: spent fuel pool system recirculation pumps management

CB12 function provides the chemical reagents supply from alkali storage tank to spray pumps suction and well-timed closure when the level in tank is low – lower than 0,14 m. (Figure 2.2.2.15).

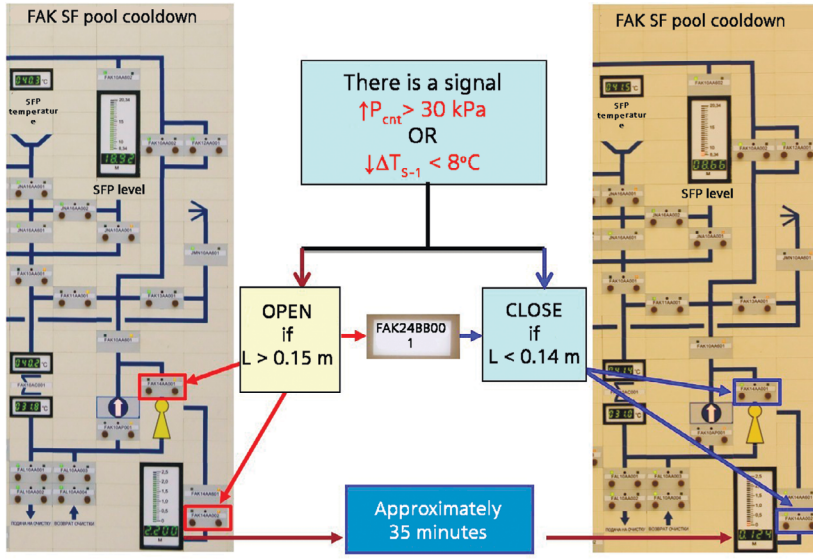


Figure 2.2.2.15 - CB12: solution for iodine fixation supply management

If the pressure in containment falls to the point of less than “minus 2 kPa” – in order to avoid the containment degassing – valves, that are located on the lines of water supply to the spraying jets from FAK and JNA systems, close. (Figure 2.2.2.16).

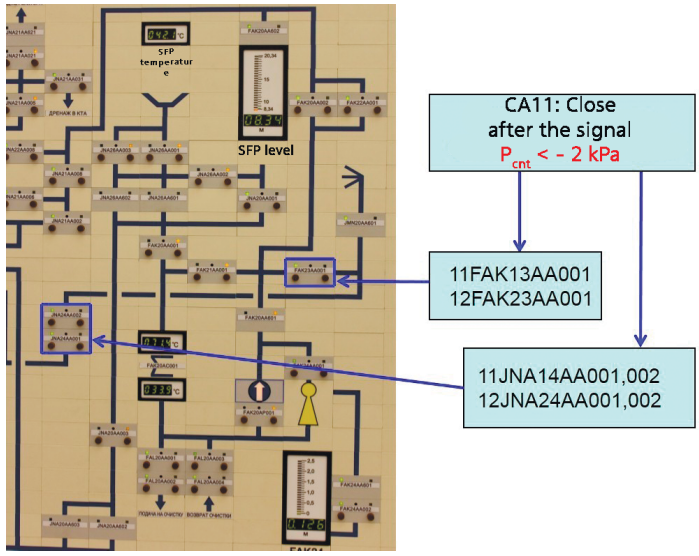


Figure 2.2.2.16 - CA11: containment degassing protection

After the valve close on the water supply to the spraying jets lines, FAK10(20) AP001pumps start operating in recycling mode - on a signal that the flow rate through the pump drops to the point less than 60 m^3/h with 30 seconds time delay.

2.2.2.7 JMN and JNA spray system operation during the beyond design basis accidents connected with spent fuel pool pump system failure

Emergency and design standby cooling system and JNA spent fuel pool cooling system, pressure pipelines of which have the special leg with JMN system pipelines, can be used in case of the beyond design basis accidents with the failure of FAK 10-20 system pumps which are used for water supply to the containment airspace. In this case the spray system in liaison with JNA system perform the function of parameter decreasing (pressure, temperature) inside the containment and does not perform the function of radioactive iodine fixation.

2.2.3 CREATING DEPRESSION SYSTEM IN OUTER CONTAINMENT KLB22

Exhaust ventilation system KLB22 is designed for:

— creating and maintaining of vacuum no less than 200 Pa in annulus in case of leak of primary circuit in inner containment including Dy 850 mm main circulation pipeline rapture ;

— annulus exhaust air cleaning.

The system must remain functional and provide the functions performance under the following loads:

— loads during the accidents taking place in primary containment including maximum design basis accidents;

— seismic load of maximum credible earthquake level;

— outer shockwave impact load;

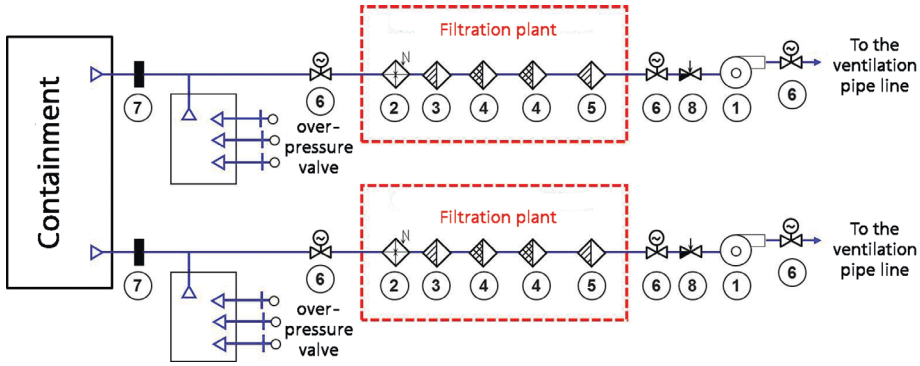
— loads arising due to the aircraft crash to the containment shell;

— loads connected with operating medium parameter variation.

KLB22: 2L in liaison with NP-001-15 system elements classified designation

KLB22 system flow diagram is depicted in Figure 2.2.3.1.

The system has the 2 x 100 % structure and consists of two installations: 11KLB22 and 12KLB22.



1- air-moving unit;
2- air heater section;
3- aerosol purifying section;
4- iodine purifying section;
5- aerosol purifying section;
6- cut-off valve;
7- fire-stop valve;
8- control-slide valve

Figure 2.2.3.1 – Flow diagram of depression system creating in outer containment KLB22

Every filtration plant includes:

— air-moving unit;

— filtration plant;

— isolation hermetic valves;

— fire-stop valve;

— control-slide valve;

— flues.

Installations 11KLB22, 12KLB22 are located in exhaust air ventilation apartments of controlled access area in UKA building (additional building of UJA building). At the same time the filtration units and appropriate fans are located in one apartment at point + 12, 450: 11KLB22 - 10UKA22R008, 12KLB22 - 10UKA22R012. Fans are isolated from the filtration units by the wall that acts as a screen.

Air, taken from the outer containment, goes to the filtration unit and throws into the ventilation pipe through the blast protection damper (BPD) after the cleaning from radioactive aerosols and iodine. BPD stops shock-wave penetration into KLB 22 system.

Unit rated capacity is 3500 m^3/hour. Air rated capacity maintenance is provided by operation of control valve, located on fan suction.

Air removal efficiency on KLB22 system filters is:

— from radioactive aerosols (from the most penetrating particles 0,3 microns) – 99,99 %

— from molecular iodine – 99,9 %;

— from organic iodine compounding– 99 %.

Filtration unit is a metal frame with replaceable modules/sections installed in it (Figure 2.2.3.2).

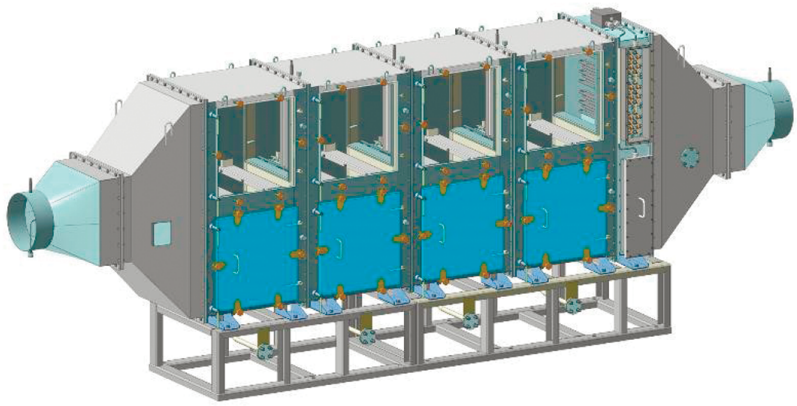


Figure 2.2.3.2 - KLB22 system filtration unit

The filtration unit consists of:

— air heater section

— section of high performance aerosol cleaning section секция аэрозольной очистки высокой эффективности;

— two sections of iodine cleaning with granular coal sorbent acting as filter-adsorber;

— section of aerosol, cleaning at unit output, designed for secondary aerosol entrapment, appearing of which is not excluded during the coal sorbent detrition.

Nozzles for measurement and control of operation parameters and for aerosol and iodine filters operation monitoring are available in filtration unit construction.

List of equipment and general technical specifications of KLB22 system are depicted in table 2.2.3.1.

Table 2.2.3.1 - List of equipment and general technical specifications of KLB22 system

Name of equipment	Diagram coding	Quantity in one unit, pcs	Equipment main technical specifications
Radial fan located on the same axis with electromotor and distributor	11KLB22AN001 12KLB22AN001	1	F = 3500 m^3/hour, H= 4600 Pa, N= 7,5 kW
Aerosol and iodine purification filtration unit, made up of:	11KLB22AT101 12KLB22AT101	1	P_{min}= 2400 Pa, P_{max}= 3500 Pa
1. air heating section	11KLB22AH001 12KLB22AH001	1	Q = 9,6 kW (total capacity)
2. aerosol purification section– FVEA-3500-2	11KLB22AT001 12KLB22AT001	1	
3. iodine purification section – FAI-2000-1E	11KLB22AT002 11KLB22AT003 12KLB22AT002 12KLB22AT003	2	
4. aerosol purification section – FVEA-3500-1	11KLB22AT004 12KLB22AT004	1	

The operator monitors the system parameters by DCS still frame (Figure 2.2.3.3).

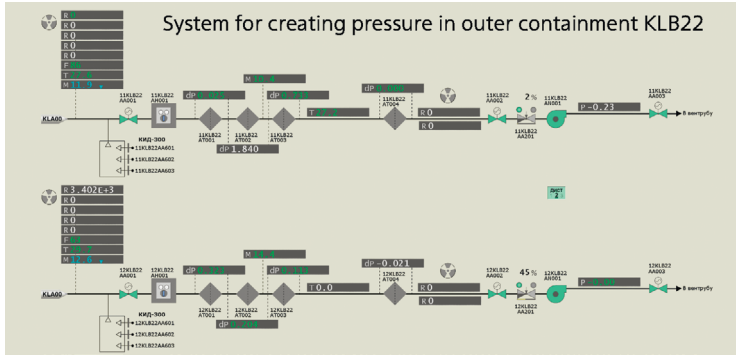


Figure 2.2.3.3 - DCS still frame of creating pressure system in outer containment KLB22

KLB22 system monitors the following parameters:

— Difference in pressure between the input and output of every cleaning element;
— head after the fans;
— air temperature before the filters and after the iodine filter;
— air consumption;
— radioactivity concentration before and after filters;
— air humidity before the filtration units and in front of the iodine filters;
— pressure in inner containment;
— control valve opening degree;
— cut-off valve position.

Fan operability is determined by the pressure existence in pressure air line, valve operability – by the signals from the limit stops.

Aerosol filters trans-shipment is carried out when there is the increase in filters pressure difference, which is higher than design value. Trans-shipment is carried out into the polyethylene bags, which are heal-sealed, put into standard barrels and then sent for future recycling and repository. Iodine filters trans-shipment into the polyethylene bags, which are put into standard barrels, is carried out in accordance with radiation protection monitor readings.

System active elements energized from emergency power supply of IInd group proper safety channel except from 11KLB22AA001, 12KLB22AA001 sealed valves which are energized from emergency power supply of Ist group proper safety channel and also from Emergency diesel-generator.

KLB22 system operation under normal operating conditions.

Depression in containment, under normal operating conditions, is supported by

draw-in air system (11,12KLC41) and exhaust ventilation system (11,12KLC21). KLB22 system remains in stand-by mode. In this case one unit is chosen to be operating, the second one – to be the standby unit.

KLB22 system operation while overpressure in inner containment is rising.

Priority Logic Equipment initiates CB11 control function, which closes the valve on combined extract-and-input ventilation systems air pipings of inner (KLA) and outer (KLC) containments, when overpressure in inner containment rises and reaches the point more than 0,3 kPa.

KLB 22 operating unit system switches on automatically after the cut-off valve closing on combined extract-and-input ventilation systems air pipings of the outer containment (11,12KLC21,41AA001) (Figure 2.2.3.4).

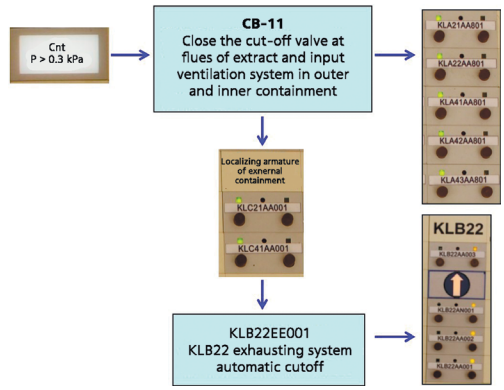


Figure 2.2.3.4 – Algorithm of KLB 22 exhaust system automatic switching-on

In case of non-start or shutting down of operating unit – the stand-by unit switches on automatically. In case of block system malfunction – the operator switching on the stand-by unit in 2 minutes with the help of control switch from MCR or Stan-by shutdown safety panel.

2.2.4 SYSTEM OF CONCENTRATION MONITORING AND HYDROGEN EMERGENCY DELETE UNDER THE CONTAINMENT (JMT–JMU)

According to the regulatory documents the following statements are the hydrogen explosion safety criteria:

— under normal operating conditions – prevention of explosive hydrogen-containing mixture creation in systems, elements and facilities, located in the containment (provided by design options for normal operating conditions and procedures and instructions observation);

— in case of design basis accidents – exclusion of hydrogen-containing mixings detonation and deflagration in facilities, located in the containment;

— in case of beyond design basis accidents – hydrogen-containing mixings detonation is excluded, while deflagration is possible when the safety localization systems fulfill the functions which are defined by the power plant design.

According to the NVNPP-2 design, the hydrogen explosion safety is provided by two independent systems:

— system of hydrogen emergency delete under the containment JMT;

— system of hydrogen concentration monitoring under the containment JMU.

Lay out of hermetically sealed compartments inside the containment, character and structure of connection between them provides the hydrogen quasi-uniform compartition in emergency operations thanks to the dead areas absence.

System of emergency hydrogen remove under the containment.

JMT system is designed for emergency operating which is connected with coolant loss inside the containment and it provides the hydrogen explosion safety with the help of passive catalytic hydrogen recombiner (PCHR).

JMT system does not operate under normal operating conditions and it remains in stand-by mode. During the beyond design basis accidents, JMT system prevents the creation of explosive hydrogen concentrations above the design limits that determines the deflagration burning. During the beyond design basis accidents, JMT system prevents the creation of explosive hydrogen concentrations above the design limits that determines the detonation.

JMT system is a passive safety system and it does not require any external power sources and control actions from automatic equipment or operator for its operation.

JMT system consists of:

— autonomous PCHR models RVK-1000 - 200 pcs

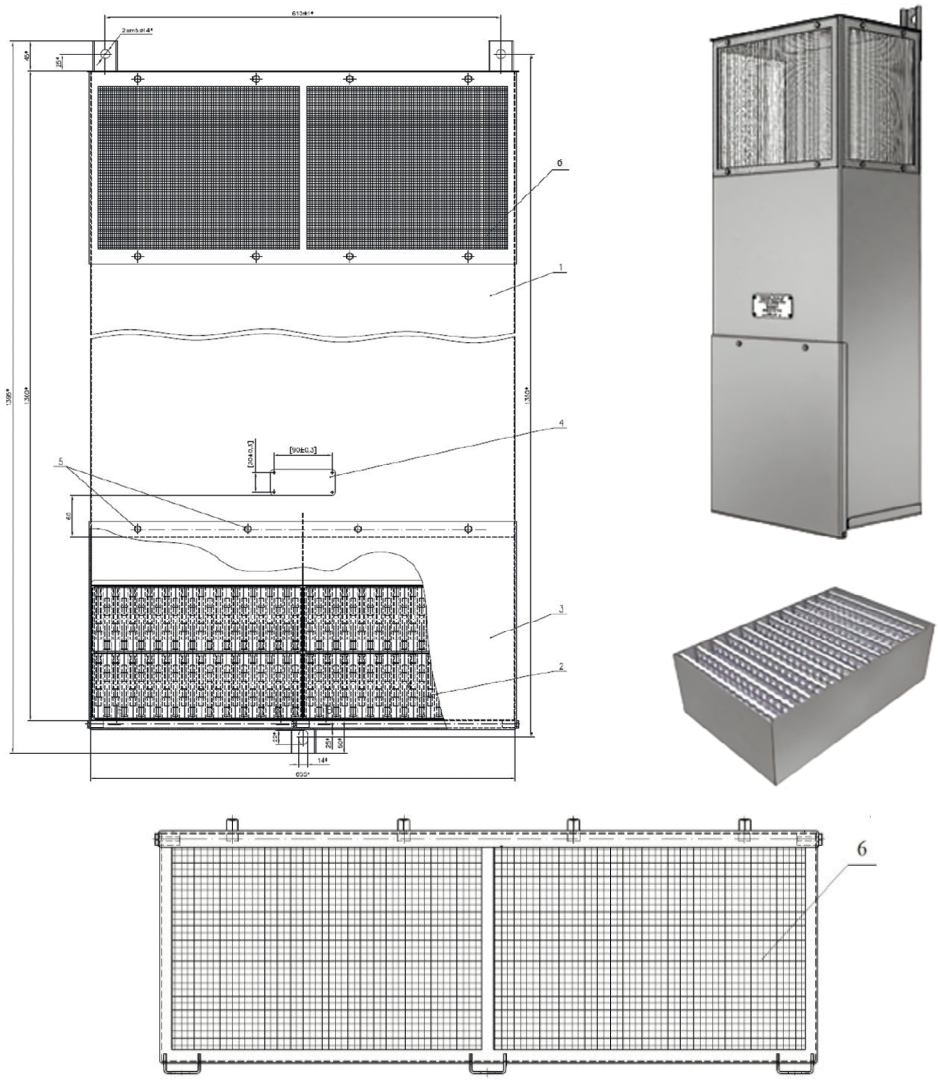
— autonomous PCHR models RVK-500 - 7 pcs

— setting of hydrogen recombiners operational monitoring and random tests (for annual monitoring of PCHR catalyst operability).

The determining factor while choosing the specified PCHR models was threshold value of transition from the mode of hydrogen flameless, catalytic recombination to the mode of ignition with formation of open flame and burning transition to the containment volume (11% bulk hydrogen concentration).

The flow diagram of the hydrogen emergency removal system represents the distribution of recombiners by the accident localization zones facilities in places of possible hydrogen accumulation.

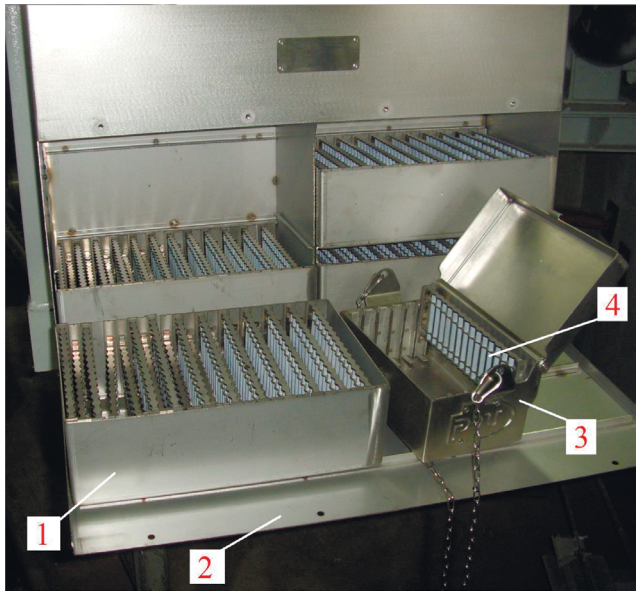
PCHR includes the catalysts assembly, convection section with guard vessel, loops for fixing with embedded plates and protection grids at input and output from PCHR (Figure 2.2.4.1).



1. Casing
2. Catalyst assembly
3. Door
4. Sign
5. Clamp bolt
6. Protection grid

Figure 2.2.4.1- Passive hydrogen catalytic recombiner of RVK model

24 frames are installed in each catalyst assembly. 15 or 14 catalytically active elements are assembled in each frame. Catalytic elements look like wires made from aluminum oxide with finely crystalline spread them. (Figure 2.2.4.2).



1 – catalyst assembly; 2 – door; 3 – box for catalytic frames transportation;
4 – catalytic frame (withdrawn for operational monitoring and random tests).

Figure 2.2.4.2- RVK-100 hydrogen recombiner photo at the moment of catalyst frame withdrawal for operational monitoring and random tests

Hydrogen emergency remove under the containment system starts operating if the following criteria are congruent:

— hydrogen bulk concentration is more than 0,45 %;

— oxygen bulk concentration is more than 0,45 %;

— containment medium temperature is higher than 20 °C.

Hydrogen-containing mixture goes through the grids of the PCHR casing butt to the catalyst assemble. Disposable hydrogen exothermic oxidation happens on contact with catalyst. Heated air and moisture vapor, produced as a result of hydrogen oxidation, rises from the catalyst assembly up to the convective cap and goes away through the casing outer protective grid to the compartment environment. Stable environment convective flow that goes through PCHR facilitates the atmosphere agitation under the containment what reduces the possibility of local explosive hydrogen concentrations formation.

Hydrogen recombiners specifications are depicted in table 2.2.4.1

Table 2.2.4.1 – Hydrogen recombiners specifications

Parameter	PCHR type	
	RVK-500	RVK-1000
Identification according to NP-001-15	2L	2L
Overall dimensions: — height, mm — width, mm — depth, mm	 950 335 225	 1400 655 220
Weight, kg	22	50
Capacity by recombined hydrogen (at a temperature of 100°C and pressure of 0,1 MPa), kg/hour: — at bulk hydrogen concentration of 3 %; — at bulk hydrogen concentration of 5 %; — at bulk hydrogen concentration of 8 %	 0,16 0,32 0,67	 0,39 0,77 1,58
Threshold bulk hydrogen concentration that characterizes the transition from flameless recombination mode to ignition mode with open flame output outside the PCHR casing, %	11	

In order to confirm the function of hydrogen safety assurance by the emergency hydrogen removal system in the conditions of design basis accidents, the appropriate design analysis was carried out for the designed scenario of the accident with two-way break of Main Circulation Pipeline DN 850 mm at the reactor inlet. The change of average bulk hydrogen concentration in containment without taking into account the recombiners operation and also taking into account the combiners operation is demonstrated in Figure 2.2.4.3

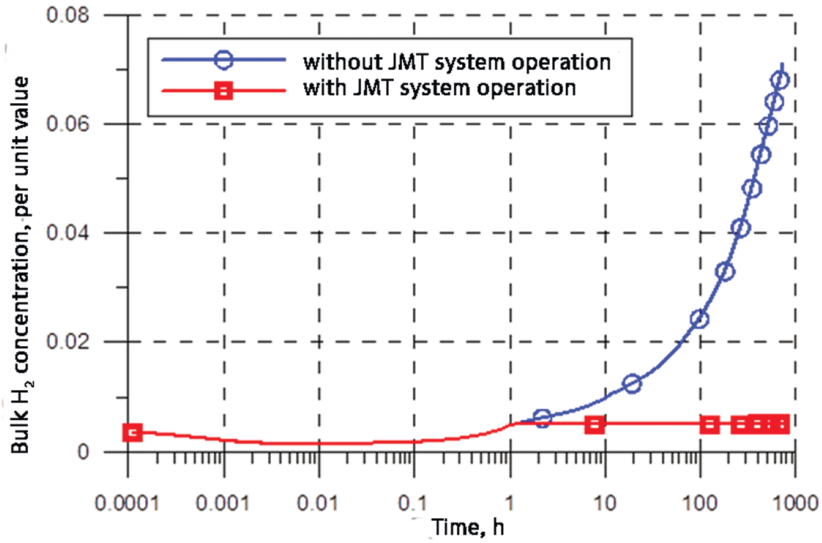


Figure 2.2.4.3 - Change of average bulk hydrogen concentration in containment without taking into account the recombiners operation and also taking into account the combiners operation

It follows from the figure that without taking into account the recombiners operation, the bulk hydrogen concentration averaged by the free containment volume, reaches the value of the lower concentration limit of flame propagation which is 4,1% during the post accident period after 340 hours (14 days) from the moment of design basis accident.

Maximum value of bulk hydrogen concentration in containment with account for hydrogen emergency removal system operation does not exceed 0,52 %.

Monitoring instrumentations are not included into JMT system. Monitoring is carried out by the operator with the help of JMU system.

System of hydrogen concentration monitoring under the containment (JMU).

JMU system is designed for conveying the information about the composition of explosive hydrogen-consisting mixtures in containment compartments in case of design basis and beyond design basis accidents with the aim of taking measures connected with beyond design basis accidents management to operations personnel.

The means of monitoring system are related to the elements of control safety and normal operating systems, important for safety (group 2 NC according to NP-001-15) and they are built according to 2-channel principle.

Safety channel equipment consists of:

— 9 hydrogen gas analyzers of GV-01 type and 9 temperature detectors from GV-01 set;

— 3 hydrogen and oxygen gas analyzing monitors of GVK type which consist of 2 hydrogen gas analyzers of GV-01 type, 1 oxygen gas analyzer of GK type and 1 temperature indicator;

— 1 air-vapor mixture in containment pressure indicator (not included into gas analyzers set);

— connector cables;

— secondary equipment for processing, displaying and recording of measured parameters: hardware and software analyzer, signals display unit.

Gas analyzer concsists of the measurement indicator which is installed in places of possible hydrogen accumulation, and of the measurement unit which is located in I&C facility.

Bulk hydrogen concentration measurement ranges:

— for normal operating conditions, normal operating conditions violation and beyond design basis accidents - from 0% to 5 %;

— for beyond design basis accidents conditions – from 0% to 25 %.

Bulk hydrogen concentration measurement range under normal operating conditions, normal operating conditions violation, design basis and beyond design basis accidents is from 0% to 25 %. Vapor-air medium temperature measurement range is from 0 °C to 250 °C. Pressure measurement range is from minus 0,02 MPa to 0,6 MPa (gauge.)».

Information on the pressure in containment, necessary for gas analyzers readings correction, is entered into the hydrogen concentration monitoring equipment.

The hardware and software analyzer is installed in rooms 1 and 2 of safety system channels and is designed for:

— measurement of hydrogen and oxygen gas analyzers analog signals, pressure indicators and temperature;

— conversion of received information and forming of output data in digital and analog form.

Alarms display unit (BOS) is designed for:

— displaying the current parameters values of hydrogen concentration, oxygen and temperature on control points;

— archiving of the data on hydrogen and oxygen concentration in containment compartments during the campaign;

— displaying and recording of measured parameters above burning/detonation zone (Shapiro-Moffetti diagram).

Alarms Display Unit is located in adjacent equipment rooms of MCR. The information collected from all sensors of JMU system and 2 channels of safety system is depicted there (Figure 2.2.4.4).

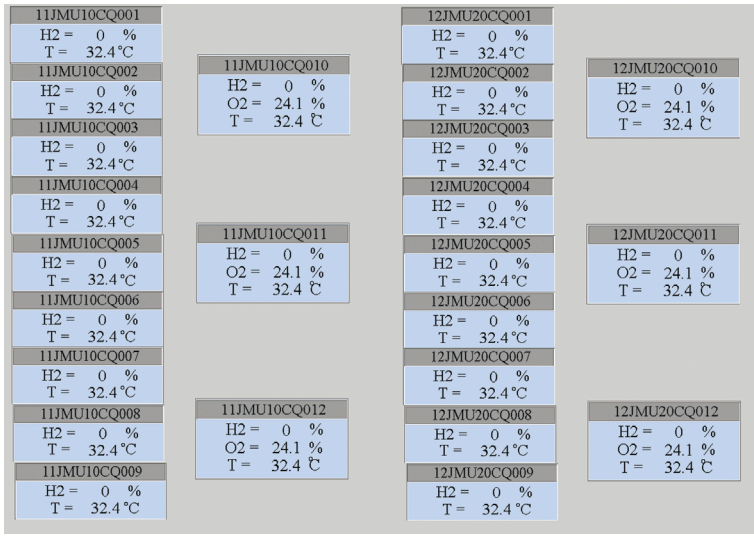


Figure 2.2.4.4 – BOS video frame «JMU system control points»

The results of hydrogen explosion safety state in containment are depicted on the operator display in a form of a point of the steam-air-hydrogen containing mixture current state (Shapiro-Moffetti triangular diagram in the following coordinates «vapor concentration, air, hydrogen concentration»), where the deflagration curve is simultaneously displayed at the current parameters inside the containment with the alarm in case of exceeding the permissible parameters of the design (Figure 2.2.4.5).

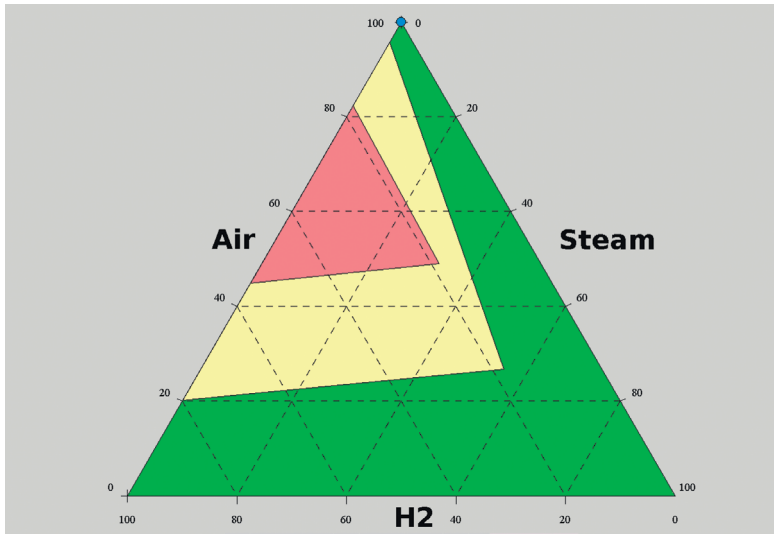


Figure 2.2.4.5 – BOS video frame «Shapiro-Moffetti diagram»

Vapor-gas medium components concentration in containment under normal operating condition, in case of design basis and beyond design basis accidents including severe accidents, are determined in the following way:

— **hydrogen concentration is less than 2 %** (normal operating conditions, design basis accidents):

— vapor (not required);

— air (not required);

— **hydrogen concentration is from 2 to 4 %** (beyond design basis accidents):

— hydrogen (sensor measurement result at control point);

— air – concentration is conservatively accepted equal to 100% (the oxygen gas analyzer GK of of the measuring complex of hydrogen and oxygen gas analyzers GVK, the turning time of which is ≈ 20 min, starts operating while the hydrogen concentration is more than 2 %. During this time the oxygen concentration is accepted equal to 21%);

— vapor (not required);

— **hydrogen concentration is more than 4 %** (severe accidents):

1) hydrogen (sensor measurement result at control point);

2) air (oxygen sensor measurement result at control point);

3) vapor (100 % - hydrogen - air).

Information on the current values of hydrogen concentration in controlled points is used for critical safety function monitoring KFB-5 “Containment Integrity” (Figure 2.2.4.6)

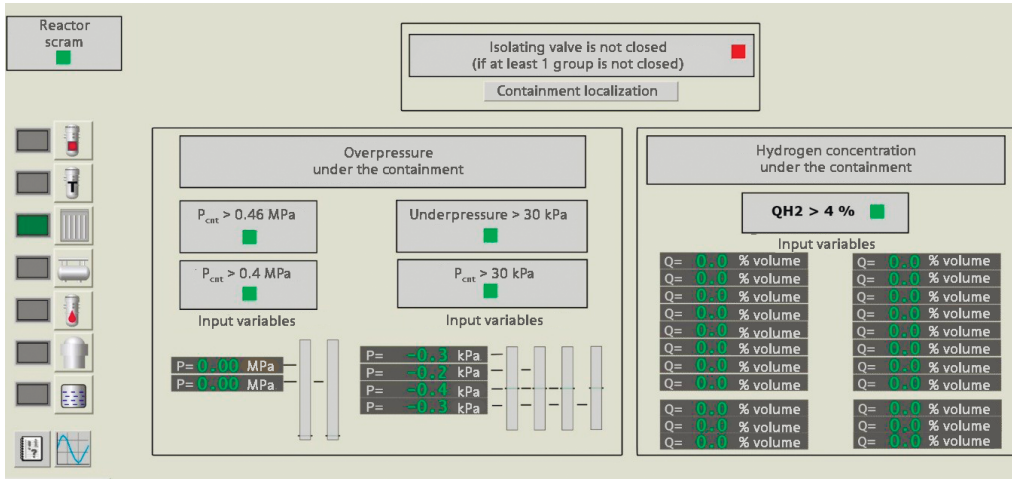


Figure 2.2.4.6 -KFB-5 “Containment integrity”parameters monitoring video frame

There is an alarm on hydrogen deviations from design limits on safety panels of MCR and Standby Control Panel:

— warning → in case of hydrogen holdup of 2 %;

— emergency → in case of hydrogen holdup of 4 %.

2.2.5 LOCALIZING PROTECTION FUNCTION

Table 2.2.5.1 demonstrates the localizing safety functions that have been implemented in NVNPP-2 design.

Table 2.2.5.1 – Localizing protection functions classification

Safety aim	C. Retention of radioactive substances in the established boundaries
Safety function	CA: Pressure limitation in containment
	CB: Fission products localization in containment
	CC: Medium removal from containment with cleaning
	CD: Steam generator localization in case of leak from primary to 2 circuit
	CE: Support systems localization
	CF: Hydrogen delete from containment

2.2.5.1 Pressure limitation in containment and abstract of heat from accident localization area

The emergency mode, that determines the maximum parameters of steam-gas medium in containment and the temperature state of building constructions of containment in the NVNPP-2 design, is the maximum design basis accident:

guillotine rupture of cold line DN 850 at the reactor inlet with the power unit blackout, failure of one diesel generator SAE and failure of 1 of 4 channels of reactor core emergency cooling system (SAOZ) passive part.

The signal for closing the localizing valve is generated at 0,18 sec in the design analysis of maximum design basis accident. The containment complete tightness after the localizing valve closing is conservatively accepted.

Absolute pressure in containment during maximum design basis accidents does not reach the maximum permitted value (0,5 MPa) and starts declining 25 seconds into. Spraying jets start operating later – 90 seconds into (Figure 2.2.5.1).

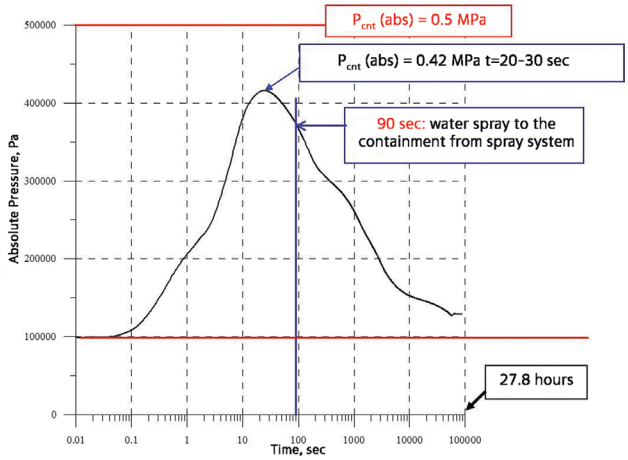


Figure 2.2.5.1 – Absolute pressure change in containment during the maximum design basis accident

Vapor-gas medium temperature in containment compartments during the maximum design basis accidents starts decreasing before the start of water supply to the spraying jets (Figure 2.2.5.2).

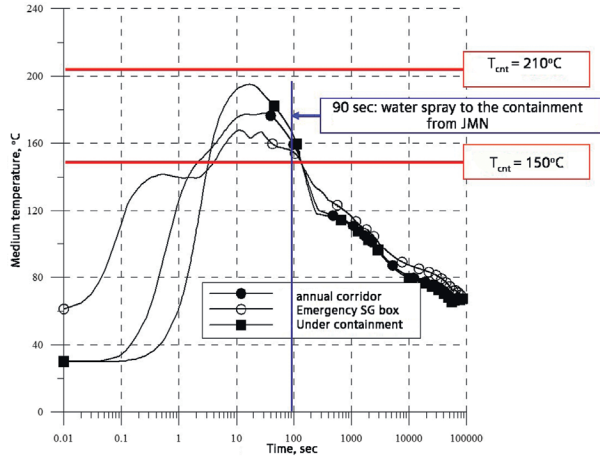


Figure 2.2.5.2 - Vapor-gas medium temperature change in containment compartments during the maximum design basis accidents

Thus, the accumulating capacity of building constructions of containment exceeds the energy of primary coolant, entering the accident localization zone at the initial stage of maximum design basis accident (release stage).

During the next accident stages, the permanent heat removal from the accident localization zone is required in order to keep the containment parameters (pressure and temperature) in permissible by the design limits

At the stage "reactor refilling" the heat removal from ALZ is carried out by means of cool water supply to the primary circuit (temperature is less than 70 °C) from the passive and active part of reactor core emergency cooling system and water supply from spent fuel pool cooling system pumps to spraying jets **(300 m³/hour with temperature less than 70°C).**

At the stage of the core long-term cooldown - after the safety system pumps transition to the containment sump - the heat removal is carried out by means of FAK and JNA heat exchangers operation. In this case the ultimate heat sink is external air that cools spraying pools (Figure 2.2.5.3).

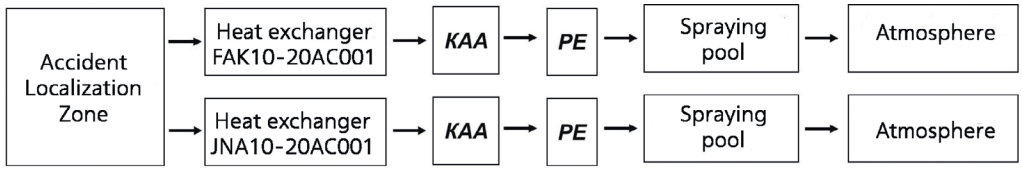


Figure 2.2.5.3 - Heat removal from accident localization zone to the ultimate heat sink at the stage of long-term core cooldown during the maximum design basis accident.

In case of beyond the design basis accidents with primary circuit leak and failure of reactor core emergency cooling system passive part, the heat removal from the accident localization zone is carried out by means of cooperation between second stage accumulator and steam generator passive heat removal system in condensation mode. In this case, air is the ultimate heat sink (Figure 2.2.5.4).

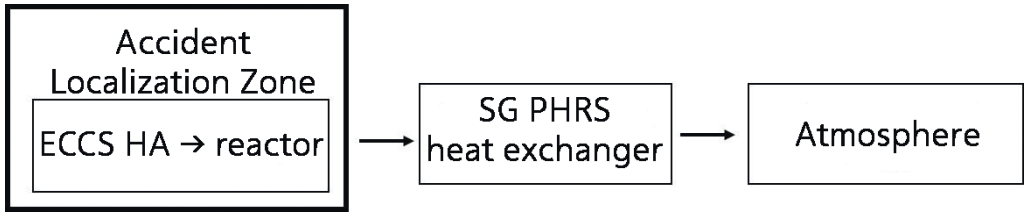


Figure 2.2.5.4 - Heat removal from the accident localization zone to the ultimate heat sink in case of beyond the design basis accident with primary circuit leak, failure of reactor core emergency cooling system active part and passive safety systems operation (second stage hydro-accumulator-2 and steam generator passive heat removal system)

Prior to the water exhaustion in second stage accumulator (24 hours in case of DN 850 leak) the personnel must arrange the heat removal from the accident localization zone with the use of the air cooling tower of the alternative closed cooling circuit (Figure 2.2.5.5).

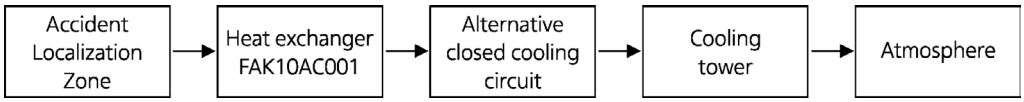


Figure 2.2.5.5 - The heat removal from the accident localization zone to the ultimate heat sink with the help of alternative closed cooling circuit

Thermal capacity, removed by the air-cooling tower, is equal to **21,2 MW** at the temperature of + 24 °C.

For comparison: thermal capacity, removed by containment 4 channels of passive heat removal system at LNPP, equally comprises **21,2 MW.**

Thus, in order to mitigate pressure in the containment and to abstract heat from the accident localization zone, the following safety systems and technical facilities are specified in NVNPP design (table 2.2.5.2).:

— JMN spray system that operates together with FAK or JNA system - air is the ultimate sink;

— system of passive heat removal from the steam generator, operating in condensation mode in cooperation with second stage accumulator - air is the ultimate sink;

— alternative intermediate circuit with air cooling tower, connected to the first channel of FAK system - air is the ultimate sink .

Usage of 2 previous ways of heat removal from ALZ allowed not to use the system of passive heat removal from containment in NVNPP design.

Table 2.2.5.2 - Implementation of safety function of SA "Containment pressure limit" in Novovoronezh NPP-2 design

Safety system (elements)	Structure	Operating principle	Management	Action	In-depth protection level
Inner containment and reinforced concrete structures inside the accident localization zone	—	Passive	Not required	Building and concrete structures in containment accumulate the primary circuit coolant energy that goes to the accident localization zone in case of primary circuit leak	3
Spray system JMN operates in liaison with FAK10-20 system	2x100%	Active	CA11 CA12	Water supply to the containment spraying jets on alarm from ESFAS in case overpressure increase more than 30kPa in containment. Spraying water flow rate is 300 m^3/hour.; temperature - less than 70°C.	3
Spray system JMN operates in liaison with JNA system	2x100%	Active	Operator action	In case of pressure increase in containment and FAK 10-20 system pumps failure. the operator carries out the water supply to the spraying jets from JNA system (300 m^3/hour with temperature no less than 70°C)..	4
Spray system JMN operates in liaison with FAK10-20 or JNA system	2x100%	Active	CA11	On alarm from ESFAS when the pressure is less than “minus 2 kPa” in containment, the water supply to the spraying jets stops - JNA system is cut off from FAK 10-20 system. JNA	3

Safety system (elements)	Structure	Operating principle	Management	Action	In-depth protection level
System of passive heat removal from steam generator, operating in condensation mode together with second stage accumulator	4x33%	Passive	Not required	Beyond the design basis accident: Big leak of the primary circuit coolant in case of Main Circulation Pipeline rupture with all sources of alternating current loss (up to 24 hours). The second stage accumulator provides the borated water supply to the reactor in order to remove the residual heat in evaporative mode. Steam generator passive heat removal system operates in primary circuit steam condensation and condensate return to the core mode	4
FAK10 spent fuel pool cooling system first channel: pump and valves are energized from Emergency Diesel Generator, the heat exchanger is cooled by alternative closed cooling circuit	1x100%	Active	Operator action	Beyond the design basis accident: Big leak of the primary circuit coolant in case of Main Circulation Pipeline rupture with all sources of alternating current waste (up to 24 hours). Stop of water boiling in spent fuel pool and reactor due to the water supply from containment sump through the FAK 10 system heat exchanger to the spent fuel pool and reactor with a flow rate of 50 m^3/hour and 270 m^3/hour and temperature of less than 70 °C respectively	4

2.2.5.2 Localization of fission products in the containment

Fission products are localized in the containment by prompt closing of isolation devices and feeding of chemicals to the inlet of the sprinkler pumps, which provides for the efficient binding and removal of radioiodine from the atmosphere inside the containment (Table 2.2.5.3).

Table 2.2.5.3 - Implementation of safety functions of CB "Localization of fission products in the containment» in Novovoronezh NPP-2 design

Safety system (components)	Structure	Operation principle	Control	Action	In-depth protection Level
Double containment, gateways, containment penetrations	—	Passive	Not required	Form a leak-tight enclosure system, as a whole	3 4
Isolation valves with electric and pneumatic drives	2 x 100 %	Active	SV11	Closing of isolation valves in ducts of ventilation systems (KLA. KLS) by ESFAS signal "Excessive pressure in the containment exceeding 0,3 kPa" (P_{cnt}> 0,3 kPa)	3
				Closing of isolation valves on the process pipes by ESFAS signal 'P_{cnt} > 30 kPa OR ΔT_{s-1} < 8 °C "	
			DSZ (diverse protection system) DPS	Closing containment valves in ducts of ventilation systems (KLA. KLS) by DSZ signal" $P_{го}$> 0,4 kPa"	4
				Closing of isolation valves on the process pipes by DSZ signal "Excessive pressure in the containment exceeding 40 kPa"	

Safety system (components)	Structure	Operation principle	Control	Action	In-depth protection Level
JMN srinkler system, which operates in conjunction with FAK10-20 system and FAK14-24 subsystem	2x100%	Active	CV12	Feed chemicals from a water pump of the alkali reserve tank to suction pumps cooling system BW — for chemical binding of radioactive iodine isotopes by the signal from the ESFAS "$P_{ГО}$ > 30 kPa" OR «ΔT_{s-1} <8 °C"	3
Tripping device (TD)	1x100%	Passive	Not required	Automatic trip of the impulse lines in the event of leak in a pipeline downstream a tripping device and when a set response flow rate is exceeded.	3

2.2.5.3 Removing medium from Space Between Containments purification

This function is implemented by operation of an exhaust ventilation system with KLB22 filtering unit (Table 2.2.5.4).

Table 2.2.5.4 - Implementation of safety function SS "Removing a medium from Space Between Containments (the annulus) and purification» in Novovoronezh NPP-2 design

Safety system (components)	Structure	Operation principle	Control	Action	In-depth protection Level
Vacuum system in an outer containment KLB22	2x100%	Active	Automatic activation by TPTS (standard software-hardware tools)	Automatic activation of KLB22 system working unit by a signal "Excessive pressure in the containment exceeding 0,3 kPa" after closing of shutoff valves in ducts of KLS21,41 system. Air taken from the outer containment is supplied to the filter unit and is discharged to a vent pipe (3500 cubic meters per hour) after purification from radioactive iodine and aerosols.	3
			operator action	In case of accidents with loss of all AC sources – feeding of active components of KLB22 system from a standalone diesel generating unit	4

2.2.5.4 Localization at the steam generator in case of leak from the primary circuit to the secondary circuit

In case of accidents with rupture of a heat exchange tube or a steam generator header, a boundary of accidents localization goes along the secondary circuit. To localize a failed steam generator, it is necessary to shut down a main circulation pump in this loop and to close:

— Isolation valve on feed water pipelines;

— shut-off valve on blow-down water pipelines;

— shut-off valves on pipelines of a passive steam generator heat removal system;

— BRU-A and the shutoff valve upstream BRU-A (steam dump relief valve to atmosphere);

— MSIV and stand-by motor driven valves downstream MSIV.

A signal from ESFAS "Heat exchange steam generator tube rupture" initiates shutdown of a main circulation pump in the emergency loop and automatic localization of failed steam generator in all flows - except for MSIV and stand-by valves downstream of MSIV.

Prior to closing of MSIV/valves downstream of MSIV, it is necessary to reduce pressure above the reactor core to less than 8,2 MPa to prevent a possible release of the primary circuit coolant through safety valves of the failed steam generator (Figure 2.2.5.6).

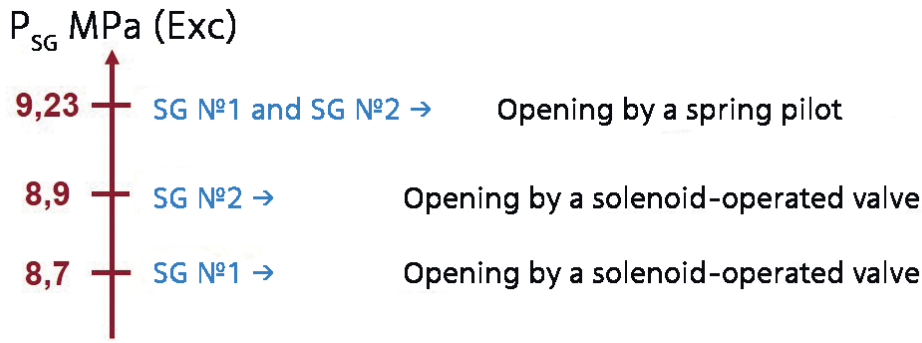


Figure 2.2.5.6 – Set point of steam generator safety valves

The necessary pressure reduction the primary circuit is achieved by injection of boron to a pressurizer from pumps of JND emergency boron injection system (Figure 2.2.5.7) and by emergency cooling of the reactor at a rate of 60 °C/hour by activation of the steam generator automatic control system or passive steam generator heat removal system on non-accident loops of the system.

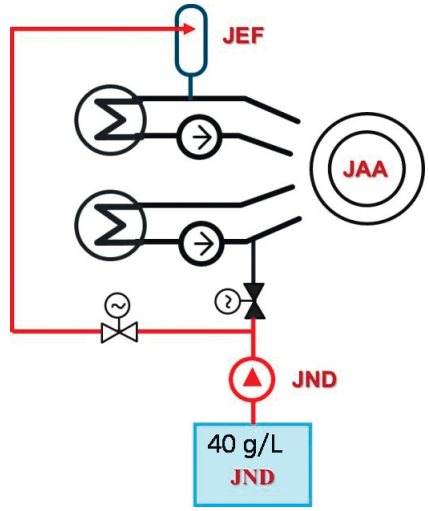


Figure 2.2.5.7 – Operation of the emergency boron injection system upon receipt of a signal "Leakage from circuit 1 to circuit 2"'

2.2.5.5 Localization of support systems

This safety function provides for localization of the support systems or any parts thereof - in order to avoid the activity breakthrough the design limits.

For example, in the event of inter-circuit leakage in a heat exchanger containing a primary circuit coolant (main circulation pump independent circuit heat exchanger, aftercooler of a primary circuit flushing) – by binary signals indicating a spark level increase in the UJA building consumer intermediate circuit – isolating PGV50 system valves are closed and a part of an intermediate circuit system within a containment is isolated.

2.2.5.6 Removing of hydrogen from a containment

Removing of hydrogen from the containment is a safety function which provides for the prevention of pressure growth in the containment. In design of Novovoronezh NPP-2 this safety function is implemented by using passive hydrogen recombiners and does not require control actions on the part of the safety control system and/or staff (Table 2.2.5.5).

The staff shall monitor condition of hydrogen explosion safety in the containment through control of hydrogen concentration within the containment (JMU).

Table 2.2.5.5 - Implementation of safety function CF «Removal of hydrogen from the containment" in Novovoronezh NPP-2 design

Safety system (components)	Structure	Action	In-depth protection level
Emergency hydrogen removal system in the containment (JMT)	RVK-1000 (200 units) RVK-500 (7 units)	Catalytic oxidation of free hydrogen in Accident Localisation Area, stirring containment atmosphere due to convective flow through PKRV	4
System control of hydrogen concentration in the containment (JMU)	2 channels Each channel includes: — GV-01 (9 units). — GVK (3 pcs.) — APA — BOS	Signals from sensors are processed in APA and demonstrated on a display of a signal processing unit (SPU). Results of analysis of hydrogen safety condition in the containment are shown on a display as a point of a current condition on a triangular Shapiro-Moffett diagram.	4

2.3 SAFETY SUPPORT SYSTEMS

Safety support systems (components) – is a safety system (components) designed to supply safety systems with energy, working environment and to create conditions for their functionality.

In accordance with the requirements of NP-001-15 (Basic Safety Rules for Nuclear Plants) the design of Nuclear Plants must provide for the existence of necessary safety systems able to supply safety systems with working environment, energy and to create the conditions required for their functioning, including heat transfer to a final absorber.

Support safety systems must demonstrate reliability indices for the performance of these specified functions, which should be sufficient to ensure that in conjunction with reliability indices of safety systems, the necessary performance reliability of safety systems as defined in the nuclear plant design is also achieved.

Execution of safety functions by the support systems shall have priority over functionality of internal protections of components of the safety support system if it does not result in more serious negative effect for nuclear plant safety.

2.3.1 A CONCEPT OF EMERGENCY POWER SUPPLY

In accordance with general requirements to reliability of electric power supply for the NPP's auxiliary consumers, all consumers are divided into the following groups in respect of its effect on safety and nature of its functions:

Group 3 Consumers – consumers of CDA (KRU-10) sections and 0,4 kV normal operation, which do not have high demands as to reliability of electric power supply and allow for interruption of power supply for the duration of an automatic circuit breaker. De-energizing of consumers of this group does not affect safety of the reactor plant, and can only result in reduction of power, but this group of consumers use three sources of power:

— turbine-generator set during the Unit operation at power;

— unit transformer when a turbine-generator is disconnected;

— stand-by transformer when the Unit is shutdown.

Group 2 Consumers – consumer of safety systems, which are responsible for emergency cooling of the reactor plant and for its transfer into a subcritical state in the Unit blackout mode. Consumers of this group allow for power interruptions at the time of start-up and connection of an emergency power supply source (diesel-generating set), but not more than for 15 seconds.

Group 1 Consumers – consumers that provide control and management of safety systems and do not permit any power interruption in all modes including power blackout of a power unit. The emergency source of power for this group of consumer is battery energy storage;

Normal operation reliable power supply consumers – consumers that ensure safety of expensive equipment such as a turbine and a generator in case of de-energizing of the power unit. This system consists of two redundant channels using a diesel-generating set as an emergency source of power.

2.3.2 AUXILIARY POWER SUPPLY SYSTEM

An auxiliary power supply system (home power system) is used to supply power to normal operation auxiliary consumers of a nuclear power plant unit, and normal operation reliable power supply system and emergency power supply system in all operating modes of a nuclear plant unit, except for the conditions of loss of voltage on auxiliary busbars of NPP from working or redundant auxiliary transformers.

General power supply diagram for auxiliary power supply with elements of a normal operation reliable power supply system is shown in Figure 2.3.2.1.

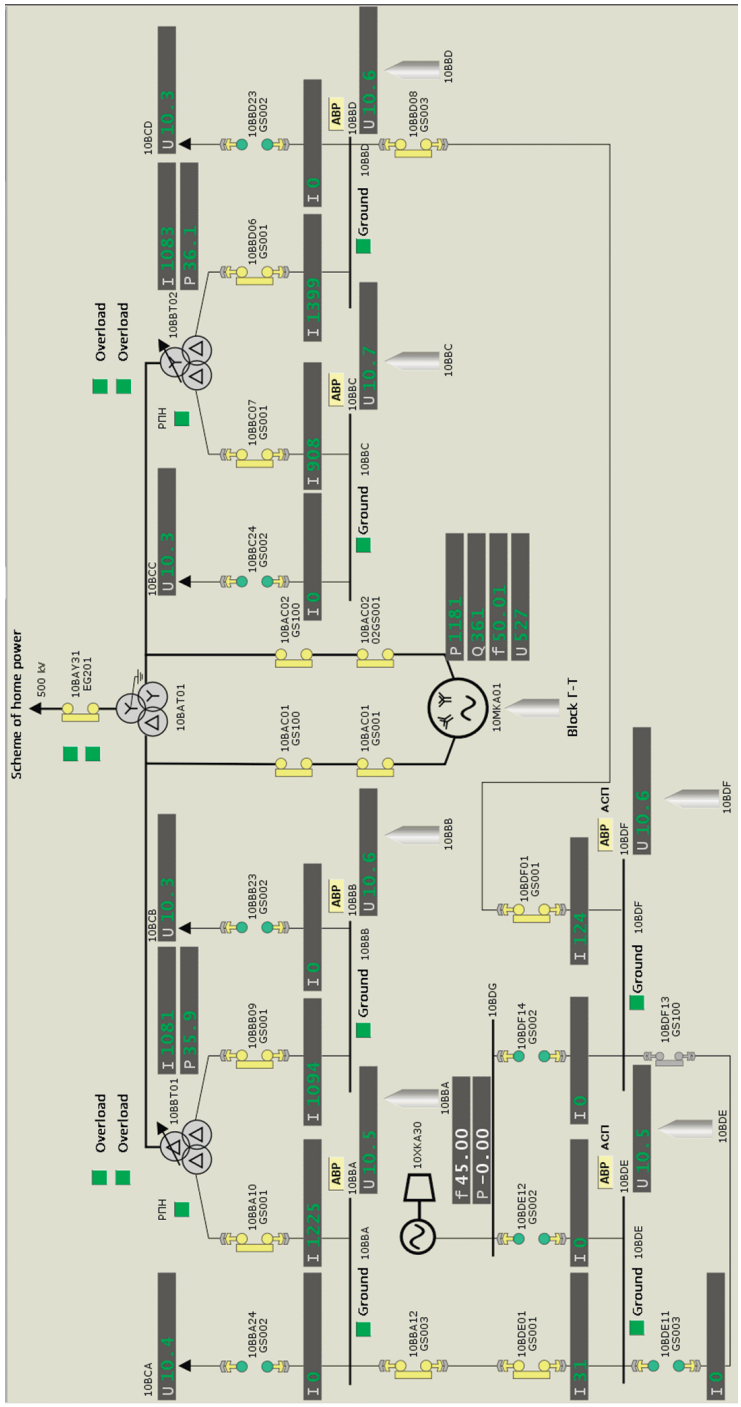


Figure 2.3.2.1 - General power supply diagram for auxiliary power supply

For power supply of 10 kV normal operation auxiliary consumers a system includes two working auxiliary transformer (BBT01,02) TRDNS-80000/24, capacity - 80 MVA, voltage - 24/10,5-10,5 kV, with two split 10.5 kV windings. Each secondary winding has a capacity of 40 MVA and feeds 10 kV sections (BBA, BBB, BBC, BBD).

Consumers of 10 kV switchgear sections shall support normal operation of the power unit, provided that each 10 kV switchgear section is connected to one main circulation pump, which has a rotating mass mounted on its shafts to utilize inertial overrun when a power unit is deenergized. The rotating mass mounted on RCP shafts increase the starting current, but within a limit of eight fold of set parameters, therefore each section of 10kV switchgear cannot be connected to more than one RCP and thus a number of 10 kV normal operation switchgear sections shall be determined by a number of RCPs.

It is known that asynchronous motors are used as a pump drive in all process systems of a power unit, which asynchronous motors have a simple configuration and are reliable in its operation, but they have two major disadvantages: low starting torque and sensitivity to voltage fluctuations, i.e., rotation torque of asynchronous motors is proportional to a squared voltage. Therefore, devices, such as OLTC (on-load tap changer), for maintaining constant voltage on 10 kV switchgear sections are installed on primary windings of working auxiliary transformers in order to maintain constant nominal voltage supplied to electric motors. OLTC device helps to ensure that voltage on 10 kV switchgear sections is always by 5 % higher than nominal voltage, i.e. 10,5 kV.

To improve reliability of power of normal operation power unit auxiliary consumers, in case of an emergency de-energizing of a turbine plant by initiation of technological or electrical protections, working auxiliary transformers are connected upstream generator switches, by so called taps, i.e. without using any switching devices, therefore main electrical protections of the transformers (gas and differential protections) operate to de-energize a power unit from a high voltage line by using switches of 500 kV switchgear.

The figure 2.3.2.2 shows a section of a video snapshot with a BBT02 auxiliary transformer that energizes sections BBC, BBD

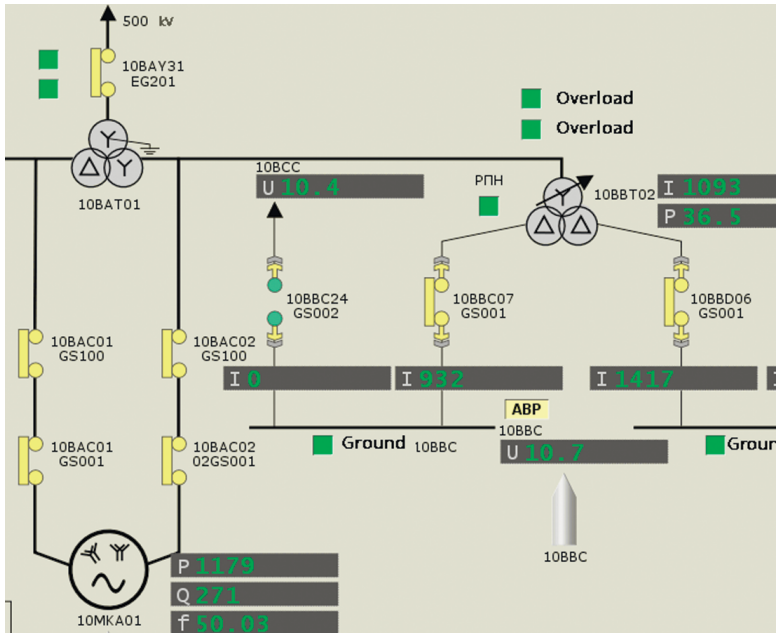


Figure 2.3.2.2 - Section of a video snapshot with a BBT02 auxiliary transformer that energizes sections BBC, BBD

Upon starting-up of a power unit, power supply of sections 10 kV BBA, BBB, BBC and BBD CH is provided by external source (grid) through CH transformer and a unit transformer when a generator circuit breaker is disconnected.

During normal operation, sections CH BBA, BBB, BBC and BBD are powered by a generator-transformer unit and grids through working CH transformers. Upon disconnection of a power unit due to damage of a step-up transformer or working CH transformer, power supply of normal operation sections is automatically switched to redundant transformers.

This scheme provides for the remote switching from the working power supply to the redundant power supply from a unit control panel.

Power redundancy of 10 kV CH normal operation sections is implemented by two groups of redundant transformers 2 x 80 MVA each, 230/10,5-10,5 kV (BCT01,02).

The figure 2.3.2.3 shows a schematic diagram of power redundancy for BBA, BBB, BBC и BBD CH 10 kV sections from the redundant auxiliary transformers (RTCH) BCT01, BCT02.

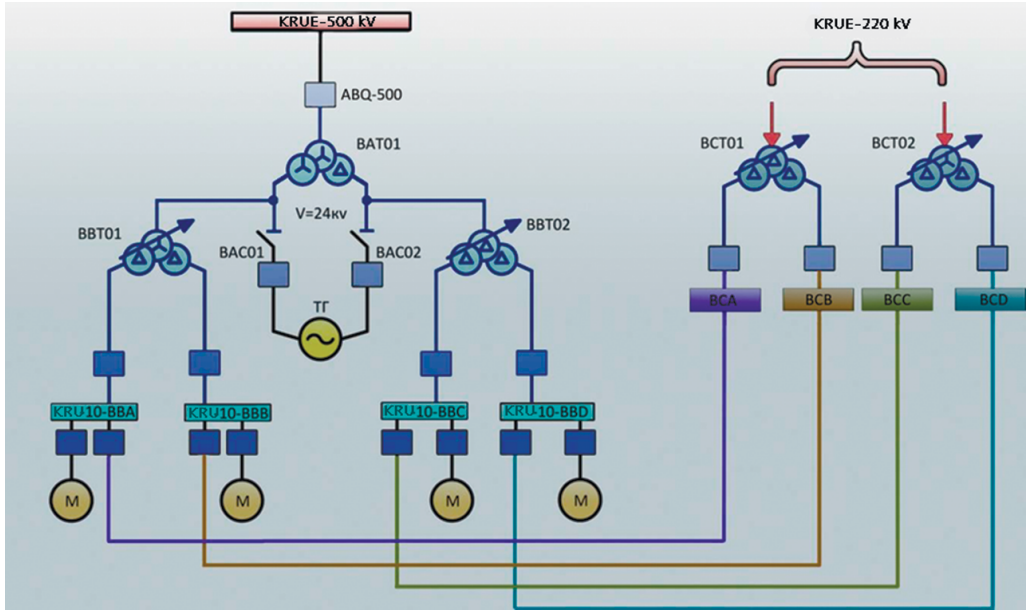


Figure 2.3.2.3 - A schematic diagram of power redundancy for BBA, BBB, BBC and BBD CH 10 kV sections from the redundant auxiliary transformers (RTCH) BCT01, BCT02

For the purposes of power supply to common plant consumers, each section of CDA (KRU) — 10 kV switchgear is connected to 10/0,4 kV step-down transformers with electric power of 1600 kVA and 1000 kVA each.

Sections of KRU — 10 kV switchgear are composed of separate cabinets that are completed with necessary high-voltage equipment at a manufacturing factory in accordance with the design and are delivered to an installation site as a completely built unit.

Cabinets are combined into sections, which are assigned with specific operational designation. Each cabinet consists of a frame, such frame is divided by metal partitions into separate compartments.

Gas-insulated switches are installed on sliding elements of the cabinets. The use of sliding elements with high voltage equipment enables quick replacement in case of failure of an equipment which is installed on such sliding elements.

2.3.3 NORMAL OPERATION RELIABLE POWER SUPPLY SYSTEM FOR AUXILIARY CONSUMERS

Normal operation reliable power supply system for group 2 users consists of two sections of KRU-10-BDE and KRU-10-BDF, which are connected to the following consumers:

— Water supply pump for RDES - 10RSS03 consumers;

— Water makeup pumps 10KVA;

— Fire pump 00SGA;

— Auxiliary feedwater pump 10LAJ;

— Intermediate circuit pumps - 10PGB50

Each section of KRU-10-BDE and KRU-10-BDF is connected to two step-down transformer of 1600 kVA each, and secondary windings of such step-down transformers supply power to sections of KRUZA- 0.4 kV.

Major consumers sections KRUZA- 0.4 kV include:

— Distilled water pump - 10LCP10;

— Fire pumps - 10SGD01;

— Lubrication system pumps - 10MAV;

— Controlled leakage pumps - 10KTA;

— Fire extinguishing unit;

— Turbine Rotor turning gear unit;

— Other consumers.

In normal operating conditions, a section of KRU-10-BDE is energized by sections of normal operation KRU-10-BBA, and section KRU-10-BDF is energized by KRU-10-BBD. 10 kV sections (BDE, BDF) are backed up by a common plant diesel generator. In normal operating conditions, a sectional circuit breaker is opened, the diesel-generating plant stays in "Hot stand-by" mode.

Automatic activation of stand-by power supply

An automatic stand-by power supply activation is implemented in order to ensure the supply of 10 kV normal operation system for auxiliary consumers in the event of failures of working power sources and its disconnection for each section of 10 kV normal operation system.

In the event of de-energizing of one of sections of normal operation reliable power supply system, a start of a diesel generator is not performed – power supply for this section is restored by closing of 10BDE11GS003 circuit breaker.

Preliminary conditions for activation of the automatic activation of a stand-by power supply:

— activation of a power unit relay protection

— when a level of voltage in 10 kV section of normal operation system drops below 80% within 20 seconds;

— when a level of voltage in 10 kV section of normal operation system drops below 25% within 0,9 seconds;

— in case of erroneous or inadvertent opening of a circuit breaker;

— in case of opening of circuit breakers of 500 kV power unit;

— in case of opening of circuit breakers of 24 kV generator;

- when a switch of the automatic stand-by power supply activation is in "On" position.

Initiation of the automatic stand-by power supply activation leads to opening of a circuit breaker for a working input of 10 kV section (or the fact of its opening is acknowledged) and an impulse to close a stand-by power supply circuit breaker for this section.

The stand-by power supply circuit breaker is closed without checking of whether there is synchronism with the control voltage executed by the stand-by power supply source.

After elimination of damages pertinent to the working power source, a reverse transfer from the stand-by power supply source to the working one shall be performed by an operator from a control panel. The procedure is as follows, a circuit breaker of the working power supply is opened in the first place, then the stand-by power supply circuit breaker is opened automatically.

Automated diesel power plants

Automated diesel-generating electric power stations "ZVEZDA-6000BC-MTU-01" are intended for use as a stand-alone sources of electric power supply for an equipment of a normal operation system in the event of de-energizing of a power unit.

The design B-392M provides for the use of a common plant diesel-generating stand-by electric power station to supply electric power to the equipment of normal operation system, which are referred to as critical to safety. Here is an algorithm for its step-by-step start-up. The designs of common plant and emergency power supply diesel-generating stand-by electric power station are identical (only external cooling systems are different), therefore description of the diesel-generating stand-by electric power station will be given in Article 2.3.4 below “Emergency power supply system”, and the same Article describes an algorithm of a step-by-step startup of the diesel-generating stand-by electric power station for emergency power supply system.

To exclude a abrupt voltage reduction in the section of reliable power supply for normal operation (emergency power supply) due to connection of large load consumers at the same time, connection of normal operation system mechanisms, which are critical to safety (safety systems), shall be spaced in time and implemented according to the specific program. This program has 5 levels and ensures even connection of mechanisms as required from time to time and

taking into account its impact on safety (for safety system mechanisms, taking into account its impact on critical safety functions).

In the event of de-energizing at the nuclear plant, operation of diesel generators is isolated for the selected load of 10 kV sections of normal operation reliable power supply.

Both 10 kV sections of normal operation reliable power supply provide for the possibility of step-by-step startup of loads after starting of the diesel generator.

Description of the equipment of the diesel-generating stand-by electric power station is given in the Article “Emergency power supply system”.

Diesel-generating set	channel № (diesel-generating stand-by electric power station)	Building	KKS code	Electrical power supply system	Quantity (unit)
"ZVEZDA-6000VS-MTU-01"	3	13UBN	10XLA30	Normal operation power supply system	1

Startup algorithm for diesel-generating stand-by electric power station

In case of de-energizing of both sections of reliable power supply system for normal operation, power supply of these sections is switched over to a diesel generator. De-energizing of both sections is recorded upon satisfaction of one of the following conditions:

— reduction of voltage on both sections down to 0,25 U_{nom} within 1,3 seconds;

— reduction of voltage on both sections down to 0,8 U_{nom} within 21 seconds;

— reduction of frequency on both sections down to 48,5 Hz or lower within 20 seconds.

Signals to be generated:

— signal of blackout of 10 kV 10BDE, 10BDF sections in Electrical I&C with activation of alarm display “10BDE AND 10BDF blackout" on 10CWG09 panel

— signal of opening of circuit breakers 10BBA12GS003, 10BDE01GS001, 10BBD08GS003, 10BDF01GS001, 10BDE11GS003

— signal of opening of circuit breakers of connected consumers that are not switched on at the step 0 of the Stepped Program.

Signals to be generated:

A signal to start a diesel generator 10HKA30 provided that at least one of the circuit breakers 10BBD08GS003 (or 10BDF01GS001) and 10BBA12GS003 (or 10BDE01GS001) is opened and 10BDE11GS003 circuit breaker is opened.

Signals to be generated:

— a signal to prevent a remote and/or automatic switching on of connected normal operation consumers (except for Stepped Program)

In the event of violation of normal operation modes of NPP that are not attributed to loss of voltage on 10 kV buses, reliable power supply systems (diesel-generating set) stay in a "waiting" mode.

In case of satisfaction of the following conditions:

10HKA30 diesel generator is ready to receive the load and there is no prohibition on energizing of 10 BDE section (Protection of section or 10VVA12GS003 (or 10BDE01GS001) is not opened)

Signals to be generated:

— signal to close the circuit breaker of 10BDE12GS002 diesel generator

In case of satisfaction of the following conditions:

— 10HKA30 diesel generator is ready to receive the load and there is no prohibition on energizing of 10 BDF section (Protection of section or 10BBD08GS003 (or 10BDF01GS001) is not opened)

Signals to be generated:

— signal to close the circuit breaker of 10BDF14GS002 diesel generator

In case of satisfaction of one of the following conditions:

— voltage in section 10 BDE ≥ 0,9 U_{nom} and 10BDE12GS002 is closed

— voltage in section 10 BDF ≥ 0,9 U_{nom} and 10BDF14GS002 is closed

Signals to be generated:

— signal to switch on mechanisms according to the Stepped Program together with initiation of the alarm display "STEP BY STEP START" on 10CWG09 panel.

Automatic controls of the diesel generator maintains steady voltage and speed of its rotation during a stepped load increase.

Testing of the diesel generator under about 30 % of nominal load and testing of automatic controls of the stepped start with disconnection of 10 kV section of Emergency Power Supply Systems from the power source at least once per month when a reactor is at full load. Testing of a parallel operation at 100 % load is tested once a year. In both cases, a procedure provides for a priority of automatic emergency command signals, which would stop a testing mode and transfer a system to the emergency regime.

The extent of automation of the diesel generator is such that it provides for an automatic maintenance of the equipment in a "stand-by" mode, automatic start and load increase, stabilization of a diesel engine rotational frequency and voltage at the generator terminals during the stepped load increase and continuous operation in standalone mode.

Time interval from the moment of a start command to the moment of readiness to receive a load does not exceed 15 seconds.

Stepped start program for 10HKA30 diesel generator:

1. **0-second** after occurrence of voltage - Step 0 of the Stepped Program is initiated:
— auxiliary consumers of a diesel generator;
— ventilation unit;
— SGB fire waterpipe system;
— communication and physical protection reliable power supply unit;
— boron concentrate pump;
— turbine lubrication system pump;
— hydrolift rotor system pump;
— controlled leakage pump;
— working lighting units;
— fuel storage pond feeding pump;
— service water pump;
— engine-turning gear device;
— standalone air conditioner of a turbine hall recirculation cooling system;
— electric air heater of a tank supply unit;
— air supply system for I&C power storage premises.

2. **10th second** after occurrence of voltage – Step 1 of the of the Stepped Program is initiated, which means connection of mechanisms to the reliable power supply sections:
— auxiliary electrical feeding pump10LAJ;
— make up pump10kVA.

3. **20th second** after occurrence of voltage – Step 2 of the of the Stepped Program is initiated, which means connection of mechanisms to the reliable power supply sections:
— water pump for consumers of a stand-by diesel electrical power plant (SDEPP);
— demineralized water pump LCP;
— CPS (control and protection system) actuators cooling system;
— unattended containment area cooling system;
— pressure suit air supply system;
— restricted area cooling system;
— distillate feed pump;
— electric heaters 10KPL;

— electric air heater 10 KPM;
— supply/exhaust air ventilation plant.

4. **30th second** after occurrence of voltage – Step 3 of the of the Stepped Program is initiated, which means connection of mechanisms to the reliable power supply sections:
— intermediate circuit pump 10PGB50;
— electric air heater of an air supply plant;
— supply/exhaust air ventilation plant;
— standalone air conditioning plant.

5. **40th second** after occurrence of voltage – Step 4 of the of the Stepped Program is initiated, which means connection of mechanisms to the reliable power supply sections:
— electric heaters of a pressurizer10JEF;
— rectifier;
— fire pump 00SGA;
— operation of an alarm display "STEPPED START IS COMPLETED" on 10CWG09 panel.
Note: this regime activates pumps (mechanisms) that are switched to "Operation" mode.

10 kV and 0,4 kV switchgears are located in separate rooms of UBN buildings. 10HLA30 diesel-generating plant is automatically connected through separate 10 kV circuit breakers, in case of de-energizing of electrical sections 10BDE 10BDF, and further, via auxiliary transformers 10/0,4 kV, provides autonomous power supply for equipment of normal operation systems that are critical to safety.
When a "Testing" mode is activated, a diesel-generating plant for normal operation systems is connected to either of the two 10 kV normal operation sections, whereby connection to the other section is blocked upon its de-energizing. A standard connection of 10HLA30 diesel-generating plant is shown in Table 2.3.2.1.

Table 2.3.2.1 - Standard connection of 10HLA30 diesel-generating plant

Electrical section 10 kV		Electrical section 10 kV		
10BDE		10BDF		
Electrical section 0,4 kV		Electrical section 0,4 kV		
10BGA	10BGS	10BGV	10BGD	10BGE

Normal operation reliable power supply system for Group 1 auxiliary consumers is intended to supply power to normal operation consumers in case of de-energizing of a power unit.

To supply power to consumers of this group this system utilizes an uninterruptible power supply unit (UPS), which is a source of DC 220 V and AC 380 V. An emergency power source for UPS is a rechargeable battery which is maintained in a mode of constant recharge.

Normal operation DC scheme intended for the power supply of DC normal operation consumers relating to a turbine and a reactor systems, includes four 220 V batteries with DC shields (BUA, BUB, BUC and BUD), rectifiers (BTL11, BTL12; BTL21, BTL22; BTL31, BTL32 and BTL41, BTL42) that are responsible for recharging and charging of batteries from sections 10 kV normal operation reliable power supply system, and inverters (BRT11, BRT12, BRT21 and BRT31, BRT32, BRT41).

Rechargeable batteries are used to power the following loads:

— automatic direct current RUSN-10 and KRUZ-0.4 kV;

— emergency lighting;

— electromagnets of a turbine hall and a turbine plant;

— electromagnets of MCP (main circulation pump) bearings;

— turbine oil pumps;

— standard hardware and software means;

— relay protection and excitation of a generator.

Rectifiers provide for the constant load of a DC panel and recharging of batteries.

A normal operation system utilizes four stationary earthquake-resistant lead-acid rechargeable batteries. Each 220 V battery consists of 106 elements with a nominal capacity of 1740 Ah (10BTB10 ÷ 40).

In addition to the above batteries, there are two 110V batteries consisting of 54 elements with a nominal capacity of 2320 Ah, they are used as a stand-by power source to maintain the CPS control rod drives in a set position during voltage dips in the auxiliary electric network (10BTC21,41).

In the normal operation mode, power supply of a reactor control and protection system, to the extent relating to power supply of the control rod drives, shall be provided by two step-down transformers 10/0,4 kV.

In accordance with NP-001-15, normal operation reliable power supply system for auxiliary consumers shall be classified as safety class "3" and has a classification designation "3H".

2.3.4 EMERGENCY POWER SUPPLY SYSTEM

For the purposes of reliable power supply for consumers of this group, there are two independent systems with a physical separation in respect of technological and electrical sections, each of such systems having all necessary equipment to perform emergency cooling of a power unit.

In respect of the nature of their functions and in accordance with the requirements of NP-001-15, this system is classified as a support system.

The main consumers of the second group — motors. Acceptable power supply interruption according to safety requirements of the reactor plant is up to 40-60 seconds. Motors used in safety systems are asynchronous with a short-circuited rotor. Motors are able to operate with fluctuations ± 10 % of rated voltage and frequency change ± 2,5% of nominal frequency, however a simultaneous voltage and frequency deviation with a sum of the absolute values of deviation should not exceed 10%, if a frequency deviation is not greater than a designed value.

Motors are responsible for the rotation of a mechanism when voltage at startup is reduced down to 80% (down to 75% for the main circulation pump) and down to 60% in case of self-running.

The following equipment is included in this group, specifically Channel 1 of the safety system:

— cooling water pumps for essential consumers 11PEC10AP001, 002;

— safety system intermediate circuit pumps 11KAA11,12AP001;

— primary circuit emergency and planned cooling pumps 11JNA11,12AR001;

— steam generator 1 and steam generator 2 emergency cooling pump 11JNB10AP001,002;

— emergency boron feeding pump 11JND11,12AP001;

— supply air ventilation system for electrical facilities;

— other consumers.

The structure of each system includes the following equipment:

— Diesel generator;

— Section of KRU–10–11BDA switchgear;

— Transformer 11BMT11,12,13;

— Section of KRU-0,4-11BMA, 11BMC;

— Section KRU-0,4-11BME - to supply auxiliary consumers of a diesel generator.

Figure 2.3.4.1 shows a schematic emergency power supply diagram

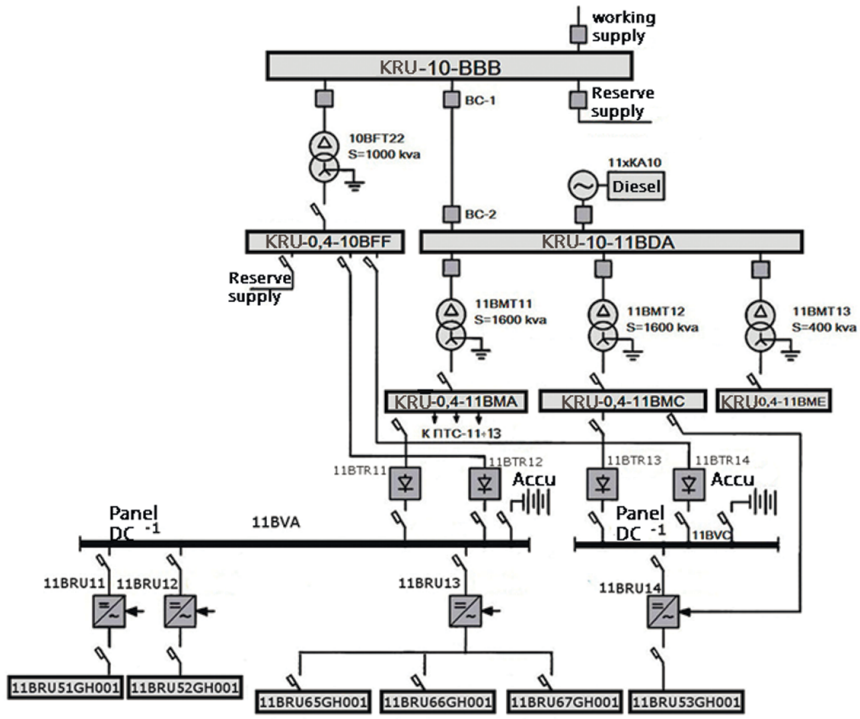


Figure 2.3.4.1 - Schematic emergency power supply diagram

At the time of start-up, shutdown and normal operation, electric power for 10 kV sections of the emergency power supply system is supplied from a 10 kV normal operation section. 10 kV and 0.4 kV switchgears are located in separate rooms of UBN buildings. Each standby diesel electrical power plant provides redundancy of the electrical power for corresponding 10 kV electrical sections:

— 11HLA10 is automatically connected through a 10 kV circuit breaker to the de-energized electrical section 11BDA;

— 12HLA20 automatically connected through a 10 kV circuit breaker to the de-energized electrical section12BDV.

Sections 11BDA and 12BDB of the emergency power supply system are connected to the corresponding normal operation sections via two circuit breakers.

Two circuit breakers are installed to ensure reliable disconnection of a section of an emergency power supply system in the event of loss of power on normal operation sections.

Figure 2.3.4.2 shows a part of a tile control panel installed in the main control room with emergency power supply system control elements.

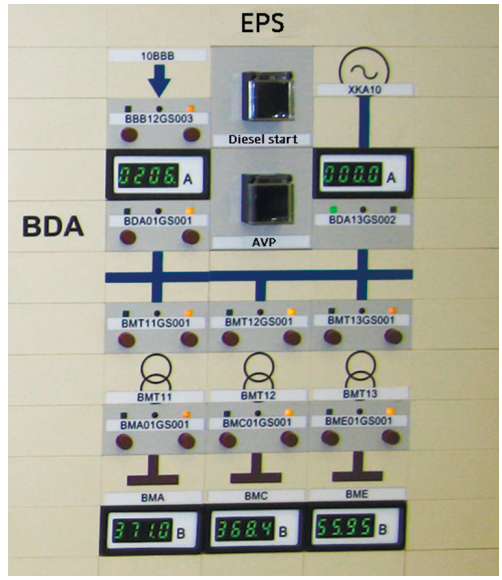


Figure 2.3.4.2 - Part of a tile control panel installed in the main control room with emergency power supply system control elements

In emergency conditions that are not associated with the loss of voltage on 10 kV auxiliary buses, electric power for sections of the emergency power supply system is also supplied from 10 kV normal operation sections.

The design and parameters of switchgears are the same as those of the similar normal operation devices.

Diesel generators are used in the emergency power supply system as emergency power supply sources.

A diesel generator is provided for each channel, and all consumers of that channel are connected to such diesel generator.

In the power supply mode the diesel generator triggers an activation of a working gear only, whereas activation of an automatic stand-by power supply circuit breaker is prohibited.

Each channel has 10 kV and 0,4 kV sections to which all channel consumers are connected.

When an emergency power supply system is energized by a working auxiliary transformer, mutually redundant mechanisms have an automatic stand-by power supply circuit breaker installed between them.

Diesel-electric plant

The design of Novovoronezh NPP-2 includes automated diesel electric plants "ZVEZDA-6000VS-MTU" intended for use as autonomous sources of electric power supply for equipment of the safety system channels in the power unit blackout mode. The design provides for the required response time for activation of technological system related to the safety system. A time interval from receipt of a command for start of a diesel generator to its readiness to accept the load must not exceed 15 seconds.

Channels are located in cells of the segregated buildings. Configuration of each channel includes a diesel-generating set (DGS) and systems that provide for the start-up and maintenance of DGS in "Waiting" mode. Configuration and designation of channels is shown in Table 2.3.4.1.

Table 2.3.4.1 - Configuration and designation of channels

DGS	Channel № (SDEPP)	Building	KKS code	Electrical supply system	Quantity (units)
"ZVEZDA -6000VS-MTU"	1	11UBN	11XLA10	Emergency power supply system	1
"ZVEZDA -6000VS-MTU"	2	12UBN	12XLA20	Emergency power supply system	1

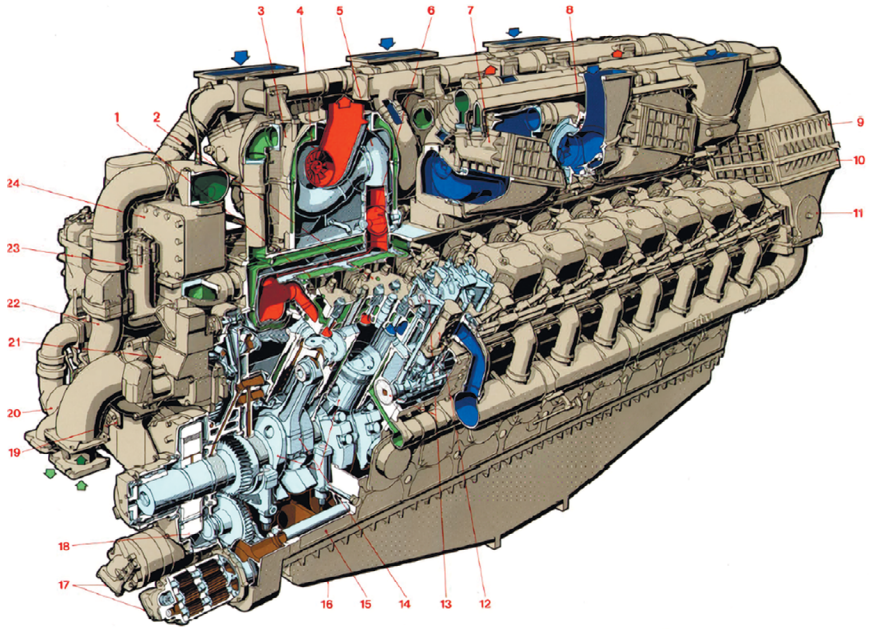
A diesel power plant consists of the following components:
— diesel generator with attachable equipment and pipelines;
— diesel fuel system (XJN);
— diesel engine lubrication system (XJV);
— diesel engine cooling system (XJG);
— diesel engine starting air system (XJH);
— PEB external diesel engine cooling system (PCB);
— air intake (XJQ) and gas exhaust (XJR) system;
— oily wastewater drainage and collection system (GMP);
— generator lubrication system (XKV);
— automatic control and monitoring system;
— primary protection system.

Main characteristics of DGS are described in Table 2.3.4.2.

Table 2.3.4.2 – DGS main characteristics

Pos.	Parameters	Measuring unit	Value
1	Overall nominal power	kW	6070
2	Maximal allowable electric power	kW	6677
3	Current type	—	3-phase alternating
4	Nominal current frequency	Hz	50
5	Nominal voltage	V	10500
6	Rotational speed at nominal electric load	rot/min	1500
7	Minimal continuous output power	kW	1200
8	Startup time (until readiness to load acceptance)	Second	15

A diesel engine is shown in Figure 2.3.4.3.



1. Constant pressure exhaust manifold
2. Cylinder heads
3. Carrier housing
4. Low pressure turbokompressor
5. High pressure turbo-compressor
6. Exhaust flap
7. Intermediate low pressure heat exchanger
8. Air flap
9. Intermediate high pressure heat exchanger
10. Turbine air heater
11. Emergency air shutoff damper
12. Nozzle-pump
13. Valve assembly
14. Crank mechanism (crankshaft).
15. Crankcase
16. Crankcase oil tray
17. Engine oil pumps
18. Vibration damper
19. High-temperature cooling circuit pump
20. Thermostatic coolant valve
21. Speed control measuring unit
22. Low-temperature cooling circuit pump
23. Engine oil filter
24. Engine oil heat exchanger

Figure 2.3.4.3 - Diesel engine

DGS (XLA) consists of a diesel engine (hereinafter — diesel), — XJA, generator — XKA, primary sensors and actuators installed on DGS and systems that ensure correct operation of the plant. A diesel engine and a generator are connected to each other by means of an elastic coupling.

Diesel engine: four-cycle engine, 20-cylinder, V-shaped, non-reversing rotation with liquid cooling and direct injection, gas turbine charging and cooled charge air – this diesel engine is designed to drive the AC generator (10 kW) with an automatic dual excitation.

The engine is started by delivery of compressed air from two air distributors directly into the cylinders, where the compressed air moves apart the pistons and creates an angular velocity on the engine crankshaft.

A synchronous generator is a driving electrical machine and is a brushless assembly with self-excitation systems and automatic regulation of voltage.

The generator is an alternating current machine with isolated neutral conductor. The generator has an open design with open cycle forced ventilation. Cooling of the stator windings is provided by forced air circulation from a machinery hall in a diesel plant building. Ventilation is provided by a radial fan mounted on a drive shaft between a rotor and a generator protective cover.

Description of SB-1 algorithms.

"BDA blackout" signal is generated by electrical I&C system and is delivered to FMCS (FSKU) DA11 (1) upon satisfaction of one of the following conditions:

— reduction of voltage on 11BDA section to 0,25 U_{nom} within 1,3 seconds;

— reduction of voltage on 11BDA section to 0,8 U_{nom} within 21 seconds;

— reduction of frequency on 11BDA section to less than 48,5 Hz within 20 seconds.

An electrical I&C system gives a signal to open the circuit breakers of the working input of 11BDA 10BBB12GS003 and 11BDA01GS001 sections. Failure to open both circuit breakers will result in generation of prohibition to close a circuit breaker DG 11BDA13GS002 (i.e. a permission to open the circuit breaker DG 11BDA13GS002 is generated upon closure of at least one of the circuit breakers).

Connection of the load to 10,5 kV and 0,4 kV sections of EPSS (emergency power supply system) is accompanied by control of a voltage level on EPSS section ($\geq 0{,}99U_N$) according to the stepped startup algorithm as prescribed by a load characteristics of a diesel generator. Operator intervention into operation of plug-in mechanisms during a preset time interval (30 minutes) is not allowed during a stepped startup.

Recovery of power for 10,5 kV EPSS section from a 10,5 kV block section of an automatic stand-by power supply circuit breaker of a reliable power supply system is possible after voltage recovery on a block section and after removal of a ban on operational actions (30 minutes) after de-energizing. An operator from the main control room or the emergency control room will send a command signal to stop a diesel generator with its disconnection from 10,5 kV section of EPSS. Further, power line circuit breakers are automatically closed under control over reduction

of voltage on 10,5 kV section of EPSS down to 0,25 UN and with electrical voltage on the 10,5 kV block section of reliable power supply system.

FSKU DA11 (1): SCSPP (engineered safety feature actuation system) "Activation of DG upon de-energizing" generates the following commands:

1. Signal to start 11XKA10

2. Normal operation command inhibition for mechanisms involved in SSP (Stepped Start Program)

3. Generation of "Memory 1" signal: Disables mechanisms and closes valves, prohibits remote control within 30 minutes. A memory reset by a button on a safety system panel in the main control room is possible after 30 minutes.

4. Generation of "Memory 2" signal: initiates SSP and prohibits remote disabling of safety system mechanisms until "Memory 1" has been reset or 11HKA10GS001 DG circuit breaker has been opened.

5. Control of activation of FSKU DA12 (1): SCSPP "Stepped connection of SB-1 mechanisms to DG", when the section 11BDA is under voltage ≥ 0,9 Unom and the DG 11HKA10GS001 circuit breaker is closed:

Note: Stand-by mechanisms are enclosed in brackets. They are disabled by a de-energizing signal, but they do not participate in the program of automatic stepped loading of a diesel- generator. In case of failure of the main mechanisms to start, stand-by mechanisms can be started by an operator after lifting of a ban at the end of the DG stepped start program.

6.1 Opening of the circuit breakers of SB-1 mechanisms, except for the circuit breakers of mechanisms to be connected at 0-stage of SSP and fittings.

6.2 Occurrence of voltage (**0 second**) initiates the start of the following mechanisms, if they have been operating prior to de-energizing:

— Emergency boron input pump 11JND11AP001, 11JND12AP001

— Air supply unit of indoor air cooling sector for safety system penetrations 11KLB02AN001 (11KLB02AN002)

— Air supply unit of a vacuum system create in the outer containment 11KLB22AN001,

— Other ventilation systems;

— Other consumers of 0,4 kV assemblies, including armature;

— Auxiliary load of a diesel generator;
— Pump station of critical electrical consumers;
— Assemblies of working lighting in the main control room and unit premises.

6.3 **10th second** after occurrence of voltage – initiation of 1st step of SSP – connection of mechanisms to emergency power supply sections:
— planned and emergency cooling shutdown cooling high pressure pump 11JNA12AP001;
— planned and emergency cooling shutdown cooling low pressure pump 11JNA11AP001;

6.4 **20th second** after occurrence of voltage – initiation of 2nd step of SSP – subsequent connection of other mechanisms to emergency power supply sections:
— Spent fuel cooling pool cooling pump 11FAK10AP001
— Steam generator emergency cooling pump 11JNB10AP001 (11JNB10AP002)
— Control and protection system cooling system air supply unit 11KLA03AN001
— Recirculation system fan 11SAD01AN001 (11SAD01AN002)
— Pump of the critical electrical consumers cooling water system 11PEC10AP001 (11PEC10AP002)

6.5 **30th second** after occurrence of voltage – initiation of 2nd step of SSP – subsequent connection of other mechanisms to emergency power supply sections:
— Intermediate circuit pump 11KAA11AP001 (11KAA12AP001)
— Air supply unit of the indoor cooling system of the containment unattended area 11KLA01;
— Electric heater of air supply plant 11SAC;
— Cold supply pump 11QKB.

6.6 **40th second** after occurrence of voltage – initiation of 2nd step of SSP – subsequent connection of other mechanisms to emergency power supply sections:
— Air supply unit of the indoor cooling system of the containment 11KLA04;
— Refrigerating plant 11QKB01.

Group 1 consumers include consumers that are responsible for monitoring and control over operation of safety systems and do not allow for any interruption in power supply in all conditions, including a major blackout of a power unit.

The design of NVNPP-2 provides for two independent channels of the emergency power supply system connected to Group 1 consumers that have a physical separation, whereas each of such consumers has every necessary equipment making it capable of providing reliable power supply to safety systems in case of major blackout of a power unit.

Group 1 consumers are supplied with both – 220 V direct current voltage and 380V alternating current.

An operating principle of an uninterruptible power supply unit (UPS) is to convert AC to DC by using rectifying devices and to supply the direct current to the direct current board buses to which consumers are connected.

Figure 2.3.4.4 shows the operating principle of an uninterruptible power supply unit (UPS)

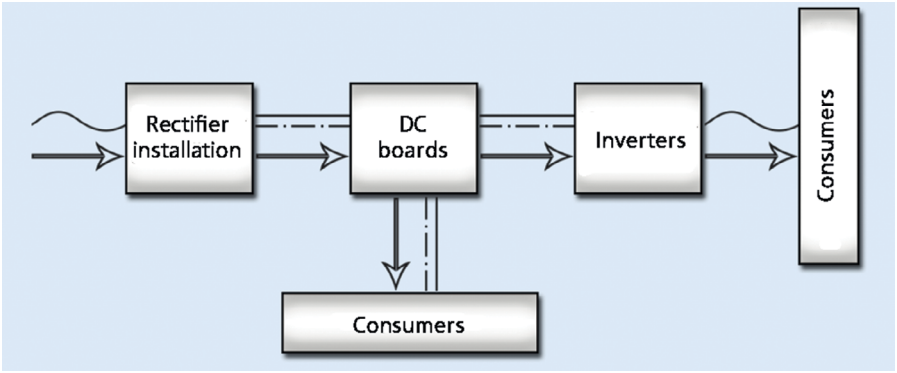


Figure 2.3.4.4 – Operating principle of an uninterruptible power supply unit (UPS)

Consumers of the first reliability group of EPSS (emergency power supply system) are:

— neutron flux control equipment (NFCE);

— process parameter protection equipment (PPPE);

— radiation control technical aid;

— remote control equipment;

— safety control system equipment;

— consumers of MCR (main control room) and ECR (emergency control room) of EPSS:

- Workstation of SB-1 MCR panels, workstation of SB-1 ECR panels
- Workstation of SB-2 MCR panels, workstation of SB-2 ECR panels

— pressurizer pilot operated pressure relief valve, forced opening valve of a pressurizer pilot operated pressure relief valve: 11JEF21AA001, 11JEF21AA002, 12JEF22AA001 12 JEF22AA002, 12JEF23AA001, 12JEF23AA002

— isolating valves, KTR system valves

— 11 (12) JNG10-40AA003,012, 11 (12) JNG50-80AA001,002

— axial compressor remote position compensator for JNA, JND, JNG, KAA systems

— steam generator pilot operated pressure relief valve, BRU-A, valves of MSIV (main steam isolation valve) and 11 (12) LBA10-40AA001 piping

— valves on steam generator blowdown pipelines 11 PG (12) LCQ14 (24,34,44), 17 (27,37,47) and pipelines for supply of steam generator blowdown water to feed water for the steam generator 11LCQ15 (25) AA001, 12 LCQ35 (45) AA001

— valves of KLC21,22,25,26,41, 11KLM10 (20,30,40) AA001

— valves of 11JKM10AA001, 12JKM10AA002

— drives of gate valves of the passive heat removal system (they open in 30 seconds after the loss of power of Group 1 consumers).

Let us consider the list of the equipment and its function in case of the first system. The first system of UPS comprises the following equipment:

• Rectifier unit 11VTR11, 11VTR13 - working;

• Rectifier unit 11VTR12, 11VTR14 - stand-by;

• DC boards 11BVA, 11BVS;

• Batteries 11VTA11, 11VTA12;

• Inverters BRU11 ÷ BRU14

• BRU assemblies.

In normal operating conditions, alternating current 380 V is supplied from KRU-0,4-11BMA and KRU-0,4-11BMC switchgear sections to the working rectifier units 11BTP11 and 11BTP13, respectively. The rectified DC 220 V is supplied to DC boards 11BVA and 11BVC to feed the first group consumers.

Rectifier unit

The rectifier unit is a three-phase silicon controlled bridge that converts an alternating current 380 V into direct current 220 V.

The rectifier unit is a part of an uninterruptible power supply unit (UPS) and is designed to:

• feed the DC board (DCB) consumers;

• recharge the storage battery (SB) on a continuous basis;

• charge SB during routine maintenance on the power unit.

The rectifier unit consists of metal cabinets, which are composed of: input transformer for galvanic isolation of the power circuit, a three-phase rectifier bridge;

A rectifier control panel consists of AC and DC instrumentation.

Direct Current Board (DCB)

Design of the direct current boards may have either a form of panels, which have devices and circuit breaker drives as well as signaling boards installed on the front of such panels, or cabinets with a two-side service access. Figure 2.3.4.5 shows DC boards.

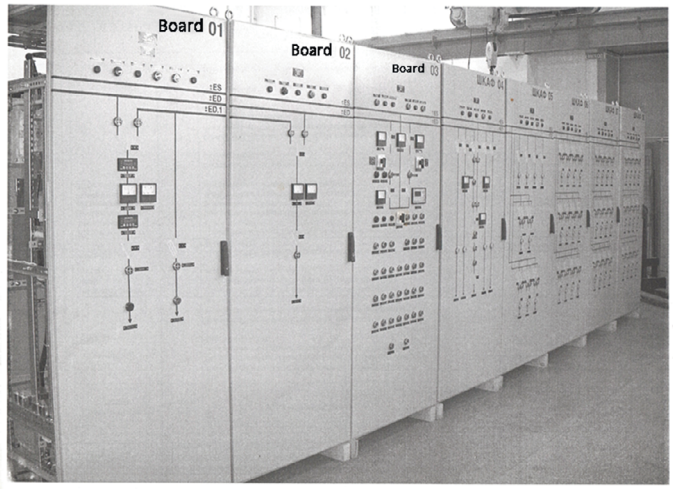


Figure 2.3.4.5 - DC boards

The direct current board (DCB) is designed to feed the first group of consumers with direct current 220 V and is one of the critical elements of the emergency power supply system.

As to its designation, DCB buses consist of two bus systems:

Main (power) buses — ± ED 220 V, rated current 1000 A, to which the following devices are connected: rectifier unit, storage battery, power consumers such as BRU-A, MSIV, pressurizer pilot operated pressure relief valve, other consumers;

Auxiliary (low-voltage) buses — ± ES 220 V, rated current 250 A, are used to supply power to electrical receivers with damaged insulation in a search mode "grounding". Figure 2.3.4.6 shows a DCB videoframe

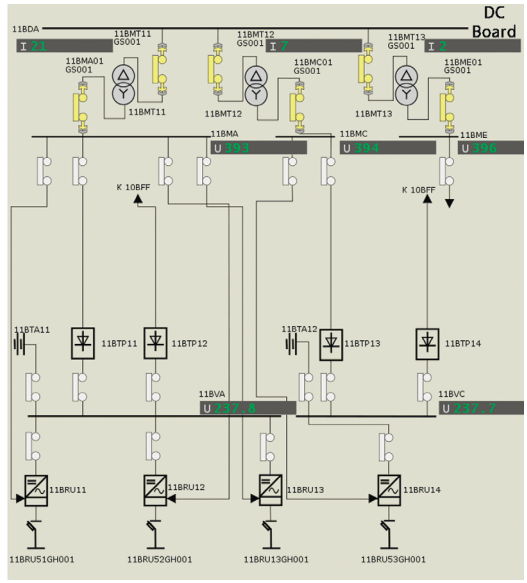


Figure 2.3.4.6 - DCB videoframe

Storage battery.

Each channel has two storage batteries with DC distribution boards and rectifiers for charging and recharging the storage battery. A discharge period of one of the batteries - 2 hours, the other - 24 hours.

Figure 2.3.4.7 shows the storage battery



Figure 2.3.4.7 - Storage battery

Each 220 V battery is composed of 106 elements. The first channel of the Emergency Power Supply System (EPSS) uses one battery with a nominal capacity of 3480 Ah (BTA11) and one battery with a nominal capacity of 1740 Ah (BTA12). The second channel of EPSS uses two batteries with a nominal capacity of 1740 Ah (BTA21,22).

This distinction is based on the fact that DCB of Channel 1 energizes a large number of powerful isolating valves (such as dy 500) that isolate pipelines and steam lines inside the containment. While DCB of Channel 2 energizes much fewer number of isolating valves (a major amount of isolating valves has a pneumatic actuator) and they isolate pipelines and steam pipes outside the containment.

Assumed battery discharge time:

— two hours without recharging for design basis accidents, given normal power supply or power supply provided by EPSS, in which case the batteries are used to energize isolating valves and APCS (automated process control system);

— 24 hours without recharging for off-design accidents, case that no power supply sources for APCS are available.

Inverters

TC-type inverters (thyristor converter) to convert (invert) DC 220 V into AC 380 V are used to supply the first group consumers with alternating current 380 V.

Inverter transformers are used to supply power to the most critical to safety consumers of the first group that must always be in operation, even in case of a power unit blackout.

For example, inverter 11VRU14 with capacity of 220 kVA is used to supply power to drives of the isolating valves;

11VRU11 inverters (12,13) with a capacity of 30 kVA each are used to supply the following consumers:

- SHSM - Standard hardware and software means;
- ICSS – initiating controlling safety systems;
- DCS – digital control system;
- IOPRS - important operating parameters record system (black box);
- Radiation monitoring;
- drives of gate valves of the passive heat removal system (PHRS);
- neutron flux monitoring apparatus (NFMS);
- Other consumers.

A stand-by power supply is provided from the diesel-generating sections KRU-0,4-11BMA and KRU-0,4-11BMC, taking into account importance of the first group consumers which receive power from the inverters.

The transition to the stand-by power supply can be performed manually by an operator by means of a switching device or automatically in case of emergency de-energizing of the inverter.

To maintain operability of logic elements of safety systems, transition to the stand-by power supply (bypass) should be performed within a time interval not exceeding 10-15milliseconds. For the purposes of successful transition to the stand-by power source within a specified time, output voltage of the inverter is constantly synchronized with a stand-by power source.

The UPS can operate both in parallel to a power network and as a stand-alone unit in case of de-energizing of a power unit.

Thus, in case of de-energizing of a power unit, SHSM (standard hardware and software means) initiates a start of the diesel generators and a program of automated stepped start (ASP) of consumers participating in a power unit shutdown cooling is activated in 15 seconds after its connection to the sections of KRU-10BDA and KRU-10BDB 15 seconds.

During the starting time of diesel generators and operation of ASP, consumers will be supplied with DC and AC power from the storage battery.

An automatic recloser of the rectifier units 11BTP11, 11BTP13, which have been working prior to the blackout, will be activated in 60 ± 10 seconds (after occurrence of voltage on 11BDA from a diesel generator 11XKA10), and thereafter the storage batteries are switched to a continuous recharge mode.

Elimination of causes resulting in the blackout of the power unit provides for the ability to restore normal operation of the EPSS (emergency power supply systems) of consumers by de-energizing of a diesel-generating plant and to make their transition to a “hot reserve” mode.

2.3.5 ESSENTIAL CONSUMER INTERMEDIATE CIRCUIT SYSTEM IN 10UJA (KAA) BUILDING

Essential consumer intermediate circuit system in 10UJA (KAA) building is designed to remove heat from consumers, as listed in the table below, to the essential consumer cooling water system (PE) in all operating modes of the power unit. In addition, this system acts as a barrier preventing the release of radioactive substances into the environment. Besides this, in case of off-design accidents KAA system can be used as an instrument of control of off-design accidents by organizing an alternative intermediate circuit in the first channel with connection of a fan cooling tower and a pump unit.

List of KAA system equipment is presented in Table 2.3.5.1

Table 2.3.5.1 - List of equipment of the essential consumer intermediate circuit system in 10UJA (KAA) building

Pos.	Diagram marking	Equipment	Flow rate, cubic meters per hour	Quantity
1	JNA10,20	Heat exchanger of emergency and planned shutdown cooling of the primary circuit and spent fuel cooling pond	2487	2
2	JNB10 JNB30	Steam generator emergency cooling heat exchanger	2500	2
3	FAK10 FAK20	Heat exchanger for cooling of a spent fuel cooling pond	700	2
4	JNA11,12 JNA21,22	Emergency and planned primary circuit cooling pump	6,5	4
5	KLA01 KLA01	KLA ventilation system air cooler	206,4	2
6	KLA03 KLA03	KLA ventilation system air cooler	101,8	2
7	KLA04 KLA04	KLA ventilation system air cooler	66	2
8	KLA04 KLA04	KLA ventilation system air cooler	32,8	2
9	JEA71÷74	Sampling heat exchanger	3,5	4
10	JEA90	Sampling heat exchanger	9,0	1
11	JND11,12 JND21,22	Emergency boron injection pump	0,3	4
12	QUN10÷40	Steam generator blowdown water sample cooler	9,15	4

Note: Flow rate is specified in relation to each customer

According to the safety system architecture, a system has two independent and physically separated channels. Each channel is responsible for the performance of safety functions to the fullest extent in the design basis accidents conditions.

Figure 2.3.5.1 contains a video-frame of a system of the essential consumer intermediate circuit system in 10UJA (KAA) building.

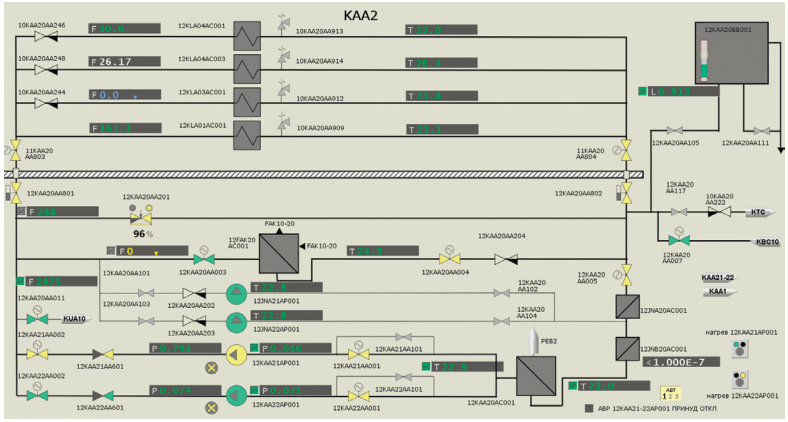


Figure 2.3.5.1- Video-frame of a system of the essential consumer intermediate circuit system in 10UJA (KAA) building

KAA system includes the following equipment:

— two heat exchangers KAA10,20 (-5,4m level);

— two breather tanks KAA10,20;

— four intermediate circuit pumps KAA11,12, KAA21,22 (-5.4 level);

— pipes and valves.

KAA system has connections with the following systems:

— cooling water pipeline system (PEB);

— system for sampling from the equipment of building 10UJA (KUA10-50);

— boron-containing water collection system (KTC10-20);

— cooling system of the spent fuel cooling pond (FAK10-20);

— primary circuit emergency and planned shutdown cooling system and spent fuel cooling pond cooling system (JNA);

— steam generator emergency shutdown cooling system (JNB10-40);

— building ventilation system 10UJA (KLA);

— emergency boron injection system (JND);

— level control system for steam generator and steam humidity control system for steam lines (JEA50-90);

— distillate system (KBC10-30);

— level control system for steam generator and steam humidity control system for steam lines (JEA50-90).

First channel of KAA system also includes pipes and valves required for connecting to the alternative intermediate circuit equipment in case of beyond-design accidents, as well as pipelines and valves supplying water to the level control system of the steam generators and steam humidity control system in the steam lines JEA50. The alternative intermediate circuit includes a fan cooling towers KAA1 and pump unit of the alternative intermediate circuit KAA13.

Figure 2.3.5.2 shows the KAA system schematic diagram with the connection to the alternative intermediate circuit.

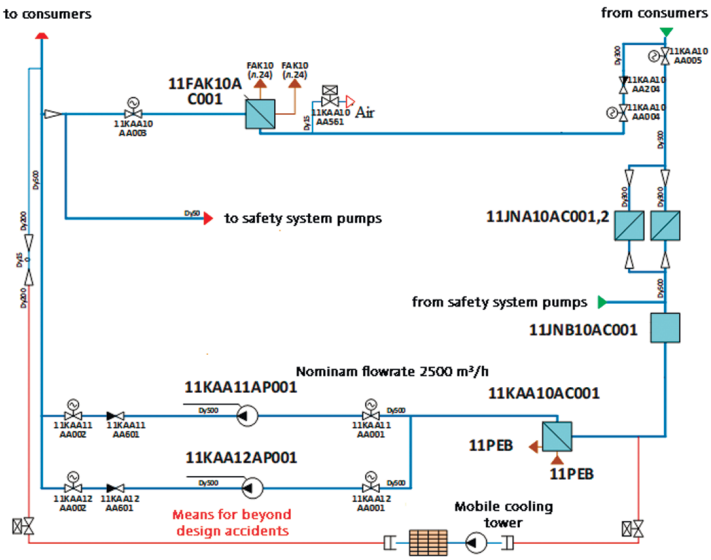


Figure 2.3.5.2 - KAA system schematic diagram with the connection to the alternative intermediate circuit

Isolating valves are installed in the pressure and drain pipes of each channel that cross the containment boundary.

For the purposes of better reliability of elements of the accident localization system, pneumatically and electrically driven isolating valves are installed in the pipelines of KAA system. Pneumatically driven isolating valves are installed outside the containment, electrically driven isolating valves are installed within the containment.

Filling the intermediate circuit with medium is carried out by KBC10-30 distillate system through collecting drains of the intermediate circuit channels.

A distillate product is automatically delivered to the KAA system channels upon

activation of blockage in case of a level decrease in a volume control tank down to 0,3 m, provided that increase of a level in the volume control tank up to 1,6 m will result in closing the valves at a fill line.

Heat is removed from the intermediate circuit heat exchangers by cooling water from respective channels of the cooling water pipeline system (PEB).

In a blackout condition, working pumps of the intermediate circuit are started automatically according to the stepped start program and these pumps are energized by emergency diesel generators.

Isolating valves are closed by alarm signals (ΔT_s < 8 °C and P_{cnt} > 30kPa) that are attributed to a loss-of-coolant accident.

In emergency conditions associated with the loss of coolant in the primary circuit, the system ensures removal of residual heat deploying the following chain: a system of emergency and planned shutdown cooling of the primary circuit and cooling of the spent fuel cooling pond (1 JNA) plus steam generator emergency shutdown cooling system (1 JNB10-40) → system 1 KAA → essential consumer cooling water system (1 PEB) → spray cooling pond.

In emergency conditions associated with leakage from the primary circuit to the second circuit, the system ensures the shutdown cooling of the steam generator at the rate of 60°C/hour deploying the following chain: steam generator emergency shutdown cooling system (JNB10-40) → KAA system → essential consumer cooling water system (PEB) → spray cooling pond.

Upon decrease of temperature of the primary circuit down to 150 °C due to shutdown cooling of the steam generator emergency shutdown cooling system JNB10-40 initiated by the heat exchangers, further heat from the primary circuit is removed through the heat exchangers of the system of emergency and planned shutdown cooling of the primary circuit and cooling of the spent fuel cooling pond (JNA).

Intermediate circuit ensures cooling of the boron solution supplied by JND system pumps to the reactor through JNA heat exchangers in case of transition of JND pump suction from tanks to JNA system.

Fan cooling tower ensures removal of at least 20 MW of heat irrespective of the environmental conditions. A portable pump unit (PPU) is used to arrange circulation of water in the intermediate circuit with a flow rate of approximately 500 cubic meters per hour.

KAA system by its designation is a safety support system that performs normal operation functions in normal operating conditions of the power unit.

Elements of KAA systems located outside the air-tight enclosure up to the isolating valves, other valves, such as intermediate circuit heat exchangers, intermediate circuit pumps, volume control tanks, pipes and valves of safety system consumers, are considered as critical to safety elements in respect of their influence to safety, and as to the nature of their functions they are considered as elements of the safety support system performing normal operating functions in normal operating conditions of the power unit. These elements belong to the

2nd safety class. Classification marking – 2NO.

Valves and piping isolating groups as to influence the safety are considered as elements critical to safety and as to the nature of their functions – as elements of the isolating safety system. These elements of the system belong to the 2nd safety class. The classification marking - 2L.

The volume of KAA system pipelines is 145 cubic meters. KAA system is filled and replenished by KBC10-30 distillate system with a flow rate up to 50 cubic meters per hour.

Technical specifications of KAA system equipment are presented in Table 2.3.5.2.

Table 2.3.5.2 – Technical specifications of KAA system equipment

Description	Quantity	Basic technical specifications of the equipment
KAA10,20 intermediate circuit heat exchanger	2	Type: T50-MFD Design: plate-type Heat exchanging surface, square meter: 1 562. **Cooled (heating) surface:** Medium: Water of the intermediate circuit (distillate product). Flow rate: 2 500 cubic meters per hour. Working pressure: 0,3 MPa. **Cooling (heated) surface:** Medium: service water. Flow rate: 3 000 cubic meters per hour. Working pressure: 0,38 MPa.
KAA11,12 KAA21,22 intermediate circuit pump	4	Type: A 500-QVD-750-100-ZC TY N 530 964 Pumped medium: water of the intermediate circuit. Working point consumption: 2 500 cubic meters per hour. Working point pressure: 70 m.
KAA10,20 volume control tank	2	Tank volume: 4.2 cubic meters

The design of KAA system uses plate-type heat exchangers, which are widely utilized in designs of safety systems of foreign nuclear plants, as well as in the modernization projects of operating power units with VVER reactors in Russia.

The plate-type heat exchanger Alfa Laval T50-MFD consists of a pack of corrugated metal plates that form channels for two fluids involved in the heat exchange process. The plate pack is placed between the supporting and clamping plates and fixed by tie bolts. Each plate is fitted with a gasket, which hermetically seals the channel and directs the fluids into alternate channels. Figures 2.3.5.3 and 2.3.5.4 show the design of the plate-type heat exchanger Alfa Laval T50-MFD.

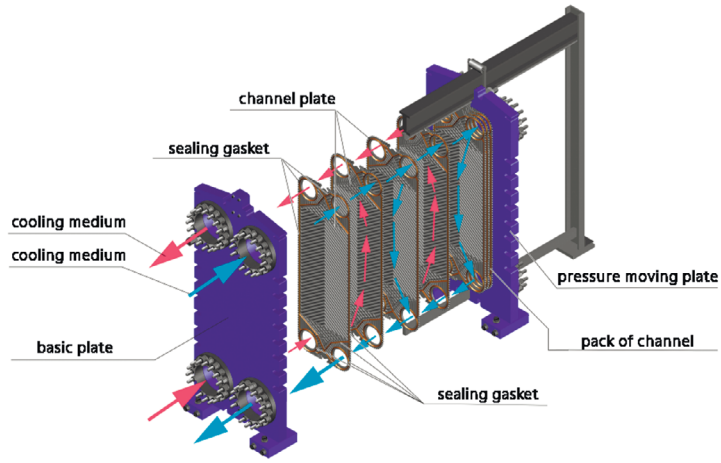


Figure 2.3.5.3 – Design of Alfa Laval T50-MFD plate-type heat exchanger

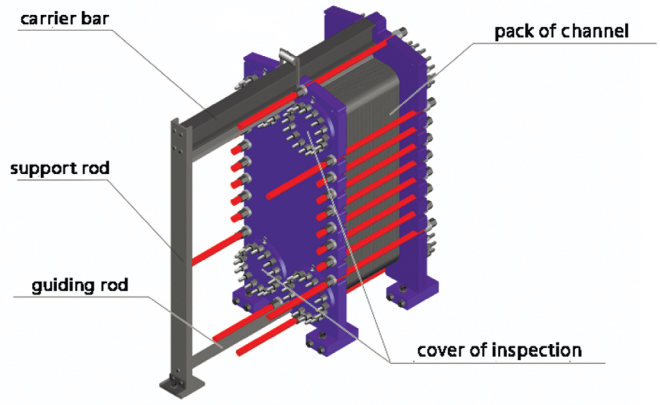


Figure 2.3.5.4 – Design of Alfa Laval T50-MFD plate-type heat exchanger

Pumps of the intermediate circuit are the elements of the safety support system and are designed for circulation of cooling water to remove heat from the system consumers located in the UJA building to a service water supply system supplying water to safety systems in all operating conditions of the power unit, including emergency conditions. Pumps operate in normal operating condition of the power unit and in emergency conditions, including blackout, and these pumps are installed in the UKA building at -5,4 m level.

Single-stage centrifugal horizontal pump. Shaft seals – end-type, self-contained. An electric motor has water-based cooling. Heat is removed through an external shell and tube heat exchanger. Cooling is performed by a pumped medium. The pump design allows for the repair under NPP conditions. Figure 2.3.5.5 shows the installation diagram of a intermediate circuit pump.

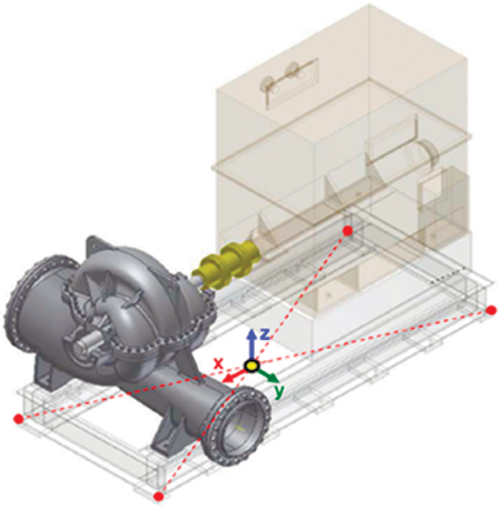


Figure 2.3.5.5 - Installation diagram of a intermediate circuit pump

A system layout and positioning of components comply with the following requirements:

— reducing process communications to a minimal level;

— equipment of each channel of KAA intermediate circuit should be placed in separate safety building compartments, which are isolated from each other by physical fireproof barriers;

— pipes of each channel located within a containment must be routed in different parts of the containment, to avoid their simultaneous damage;

— equipment located outside the containment must be accessible and available for maintenance and repairs during operation of the reactor at power;

— system layout should be performed so that failures in the normal operation systems do not result in disturbances in the system operation, and a failure of

one system channel should not result in failure of other channels through the ventilation system, engineering structures, transportation routes, cooling water channels and cable communication lines;

— provision of safe operation conditions for the staff.

Operating limits of 1 KAA system are:

— maximum flow of the intermediate circuit pump 3000 m^3/h;
— minimum flow of the intermediate circuit pump 1600 m^3/h;
— maximum level in the tank 1,7 m;
— minimum level in the tank 0,15 m;
— temperature downstream the intermediate circuit heat exchanger 33 °C.

2.3.6 ESSENTIAL PE COOLING WATER SYSTEM

Purpose and functions of PE cooling water system

Essential PE cooling water system performs the following functions:

— in the normal operating conditions, the system provides for the removal of excess heat from the intermediate circuits of the reactor compartment located in 10UJA and 10UKC buildings, and from the stand-by diesel power plants and from the consumers of the essential refrigeration complex located in the emergency power supply buildings 11 (12) UBN;

— in the emergency conditions, the system provides for removal of heat from the reactor core through the intermediate circuit heat exchangers, stand-by diesel power plants and coolers of the refrigeration complex;

— in the blackout condition, PEC pumps are stopped and automatically re-started according to the stepped start program. The electric power to the pumps during such blackout conditions is supplied from the stand-by diesel generators of the emergency power supply system.

System (PE) functions under all normal operating conditions, including start, stop, power operation, refueling, staying in “standby” and DGS testing modes, and in emergency conditions in the event of the following disruptions of the power unit operation:

— leakage in the primary circuit;

— de-energized sections of the reliable electric power supply;

— loss of normal heat removal from the reactor core through the second loop, in case of leak of water from the feed water pipelines connected to the steam generator.

Essential cooling water system (1 PE) consists of two closed-circuit and independent channels. Each channel supplies cooling water to the equipment

of the safety systems and normal operation systems with a design flow rate. This a closed-loop recirculation system, which is isolated from the external sources of water supply, groundwater and other cooling systems. Isolation of the system is conditioned by the need to comply with the environmental requirements, because there is a risk of contamination of the cooling water with radiation in case of loss of tightness of the equipment in PO heat exchanging system.

Essential cooling water system (PE) in 10UJA, 10UKC, 11 (12) UBN buildings is designed for:

— removal of heat from the intermediate circuits PGB50 (normal operation intermediate circuit system of the reactor compartment), PGB60-70 (normal operation intermediate circuit system of 10UKC building), KAA (essential intermediate circuit system of 10UJA building) through the intermediate circuit heat exchangers to the essential cooling water system and further to the spray cooling ponds. Water in the spray cooling ponds is sprayed through centrifugal spray nozzles and then returned to the system in the form of chilled water for a subsequent removal of heat from the equipment.

— partial (periodic) exchange of water in the ponds is carried out to maintain the water chemistry necessary for keeping the cleanliness of the heat exchanging plate-type surfaces;

—discharge of water from the spray cooling ponds to the storage container 00URX in case of technological need, as well as during the planned preventive maintenance.

Description of the cooling water pipeline system PE

Essential cooling water system (PE) (Figure 2.3.6.1.) is designed to perform its functions in any initiating event, accident with overlapping of one event that is independent from an initiating event, failure of any active or passive elements or human factor. It is fed from normal power supply sections with redundancy provided by the emergency power supply systems.

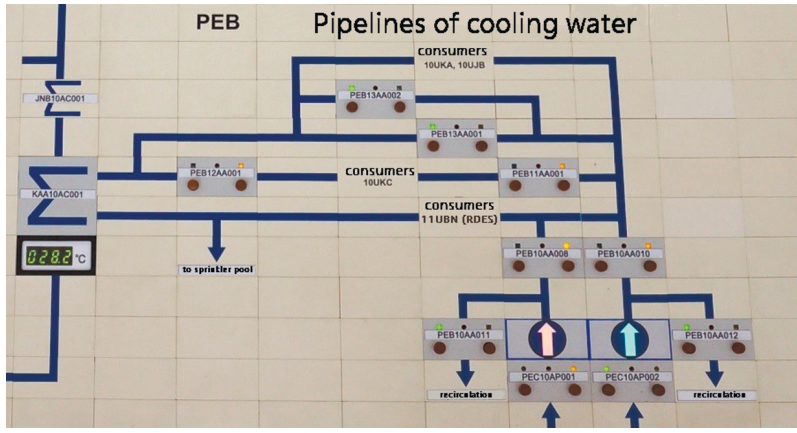


Figure 2.3.6.1 – Essential cooling water pipeline diagram (PE)

Electrical power supply of the system mechanisms (in case of de-energizing of the normal power supply sections) is delivered from the emergency power system of the second consumer group. According to the stepped start program, first essential cooling water pumps 11PEC10AP001, 12PEC20AP001 will be started automatically. They are fed from the emergency diesel generators, a delay time for the start of PE system pumps does not exceed 30 seconds after such pumps are de-energized.

In accordance with classification of PE system in respect of designation and influence on safety and according to the requirements of NP-001-15, circulation pumps 11PEC10AP001,11PEC10AP002; (12PEC20AP001, 12PEC20AP002) in the essential cooling water system, pipelines and valves, as well as water purification plants 11REA10AT001 (12REA20AT001) are marked as 2HO, they are safety important elements, and as to nature of their functions – they are elements of the safety support system combining normal operation functions when the power unit operates under the normal operating conditions. PE system is additionally protected from exposure of extremely high and low ambient temperatures, because the equipment is placed in premises furnished with a ventilation system.

The nominal flow rate of the cooling water in one channel is equal to 4 055,6 cubic meters per hour. The parameters of cooling water supplied to the consumers are listed in Table 2.3.6.1.

Table 2.3.6.1 – Parameters of cooling water supplied to consumers

Supply of water to consumers	**Parameters**		
	Water flow in one channel, minimum (m^3/h)	Inlet temperature (°C)	Working pressure (excessive), minimum (MPa)
For consumers of 10UJA and 10UKC buildings	3250,0	9 ÷ 31	0,6
For consumers of 11UBN and 12UBN building, including:	805,6		
a) to diesel generator coolers	600		
b) to the essential refrigeration supply complex	205,6		

The system maintains temperature of water to be supplied to the reactor compartment within a range from 9 to 31 ^{0}C in all operating modes, where a design-basis water level in the spray cooling pond is:

maximum - minus 1,0 m from the upper generating line of the pond wall;

minimum - minus 2,8 m from the upper generating line of the pond wall.

To avoid subcooling of water in the winter season and to reduce droplet carryover from the ponds at high wind speeds, there are sterile spill pipelines with distributing heads (to eliminate spraying of water on nozzles) from the drain pipes to each section of the pond. In the winter mode of system operation, supply of water to spray nozzles is adjusted to reduce the total flow rate.

The supply of water to the premises of 10UJA, 10UKC buildings and 11 (12) UBN buildings is organized via pressure and drain piping laid in the passes (tunnels) 11 (12) URZ.

Before the start all major pipelines upstream of 10UJA and 10UKC buildings are almost completely filled with water by gravity from the spray cooling ponds, subject to a maximum water level in the ponds. When the level in the spray cooling ponds is at the minimum, then pipelines are filled with water by half the cross-section only. Air volume of about 215 cubic meters remains in the upper pipeline section. There are vent valves and air inlet-outlet valves located on the heat exchangers for removal of such air in the process of pipeline filling.

An air inlet-outlet system for PEB and heat exchangers includes:

— manual tap-valves for air inlet-outlet when the pipelines and heat exchangers are discharged and filled with water after repair;

— automatic motor operated air inlet-outlet valves, the parameters of which and their location are defined on the basis of calculation of transient processes of the circulation pumps and pipelines PEB (Figure 2.3.6.2).

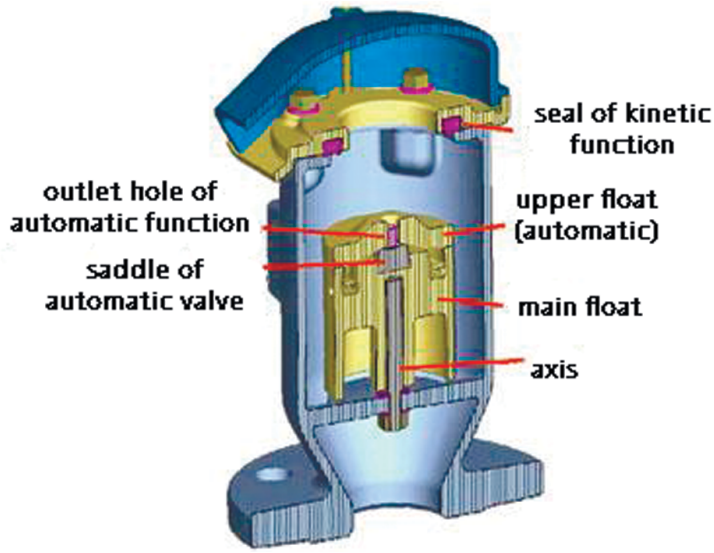


Figure 2.3.6.2 - Automatic motor operated valve

The approved parameters of the air inlet-outlet valves and their location on the pipeline route make it possible to avoid creation of dangerous negative pressure after the stop of the pump and soften the maximum pressure increase in the system after the start of the circulation pump.

After the stop of the circulation pump electric motor, the impeller speed is drastically reduced, pressure in the pump falls to almost zero and the pump flow rate is reduced respectively. Reduction of the flow rate results in a subsequent wave of pressure reduction along the whole pipeline route. Vacuum is generated at the upper points of the pipeline route, air valves open and air intake occurs. The water flow in the PEB pipeline is divided into two parts, i.e. there will be two water columns. One column continues to move toward the spray cooling ponds with a continuously decreasing stream speed as the water flow rate changes at the nozzles. The second column (towards the pump) is abruptly decelerated with the decreasing water flow through the pump and starts to flow in a reverse direction and goes down through the pump to the tail bay of the spray cooling ponds.

A check valve on the pump delivery pipe will abruptly close on the backflow, which results in a short surge of pressure downstream of the valve from the side of the pressure pipeline.

After the start of the pump the system pressure goes up, air released earlier by the valves will be compressed and this compression provides the damping at the time of occlusion of the separated columns of liquid with gradual increase of pressure in the system.

The required pressure and control of an overall flow is provided by partial opening of DN600 butterfly valves in the switch distribution chamber of the spray cooling ponds. DN800 check valve (on a circulation pump head) prevents the emptying of the pressure system and limits the frequency and time of the reverse rotation of the pump when the electric motor is stopped.

Filling and feeding of spray cooling ponds 11 (12) URR:

— by partially demineralized water (PDW, main feed) from GCF system, from water purification plants located in the demineralization plant building (00UGD). Basic feeding is implemented by the supply of the partly demineralized water to each section of the spray cooling ponds. There are wells on the southern side of the ponds, and basic feeding valves are installed in such wells;

— by domestic wastewater from a controlled access area (00UGW), from the system of the waste treatment facilities GRB21, amounting to 5,7 m^3/hour or 0,05 million m^3/year.;

— by redundant water of the water cooling system from the auxiliary reactor building (10UKC), amounting to 0,456 m^3/hour or 0,004 million m^3/year. Water is pumped into PEB drain pipe through the control tanks with a prior radiation monitoring.

— by artesian water from GKD potable water pipe of Kamenno-Verhovskii intake (stand-by feeding from the essential pump station buildings 11 (12) URF).

PE cooling water in operating conditions has the following chemical characteristics (Table 2.3.6.2)

Table 2.3.6.2 - Chemical characteristics of PE cooling water

Total salt content, mg/l, max	800
Hydrogen ion concentration indicator, pH	6,5-8,5
Total hardness, mEq/l	up to 7,5
Rigidity carbonate, mEq/l	up to 2,5
Chlorides, mg/l	up to 150
Sulfates, mg/l	up to 500
Nitrates, mg/l	up to 15
Phosphates, mg/l	up to 4
Oxidation characteristic, mg O_2/l	up to 20
Suspended substances, mg/l	up to 50

Reactant treatment by Akvarezalt 1010 is continuously used to maintain water chemistry of the recycling water supply system of normal operating conditions of the reactor. Water treatment is subject to results of pond inspections aimed at identification of local areas of slime fouling. In the warm season there are weekly dosage feeding of "Akvarezalt B1" and "Akvarezalt B2" depending on the current water chemistry. Reagents are fed into the system gradually over a period of 1 hour using a metering pump, with an exposure of 24 hours.

Description of PE cooling water system equipment design

Essential cooling water system (PE) consists of two closed-circuit and independent channels. Each cooling system channels includes:

— pond bowl 11 (12) URR (a spray cooling pond consists of two separate sections) with piping connections, spray nozzles and heads of winter discharge;

— switching chamber 13 (14) URR (adjacent to the pump station) with isolation valves installed in the gravity pipelines leading to sections of the spray cooling ponds, with drain pumps for drainage of leaks from the switching chamber and for internal drainage of spray cooling ponds;

— pumping station 11 (12) URF with water-cleaning mesh, two circulation pumps, pipelines with installed valves and expansion bellows;

— closed process tunnel 11 (12) URZ with the feeding pipeline and automatic wash filter installed at the PO inlet and the discharge pipeline from PO to the spray cooling pond with the valves.

Spray cooling ponds 11 (12) URR have a form of a subsurface rectangular reinforced concrete bowls, with width and length dimensions - 74 x 60 meters, and wall height - 4000 mm (figure 2.3.6.3).

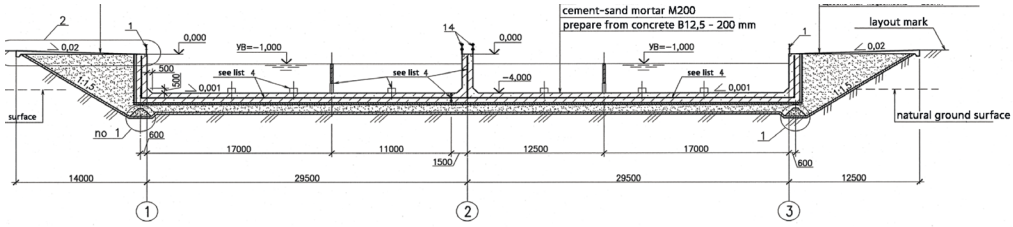


Figure 2.3.6.3 - Spray cooling pond design

Cooling of water in the pond is carried out by way of controlled water injection into the pond bowl through nozzles Цо-85 with rotary supply of water. Heating or heat insulation of aeration devices is not provided by the initial design.

Dimensions and design of the pond of one channel are determined with consideration of the following factors:

— cooling down of water in the operating and emergency conditions of the power unit with water irrigation density at the nozzles equal to 1,10 $m^3/h \cdot m^2$;

— maintenance of the cooling water temperature in the summer at the level of not more than + 31 ^0C (provided that an average ambient temperature of the hottest five-day period per year is equal to + 27 ^0C) and in the winter not less than + 9 ^0C;

— emergency shutdown cooling of the power unit in the hot season of the year without supply of water for feeding of the system during 48 hours;

— maintaining a nominal water volume in the ponds in case of a tornado passing along the NPP site.

Switching chamber 13(14) URR 11 is adjacent to the building (12) URF from the pond's side. Dimensions of the switching chamber are adopted to the existing control and isolation valves, arrangement of feed and discharge cooling water pipelines, location of pumps of the switching chamber leak drain system and pumps of the pond internal drainage system.

Pump station 11 (12) URF is intended for installation of electrical equipment, industrial cooling air conditioners, pumps, valves, pipelines, drainage system and suction and pressure duct of the pumps.

Equipment installed in the pump station 11 (12)URF:

— two circulation pumping units 11PEC10AP001, 11PEC10AP002 (12PEC20AP001, 12PEC20AP002) with winding air cooling electric motors.

— water treatment plant 11PEA10T001 (12PEA20T001) with frontal water supply, which are used to remove impurities in the cooling water coming from the pond;

— two vertical semi-submersible drainage pumps 10PEC51AP001, 10PEC51AP002 (10PEC52AP001, 10PEC52AP002) which are used for automatic pumping drainage pits and for pumping of water from sections (CK-1,2 or CK-3,4) of the spray cooling pond to the storage reservoir 00URX with simultaneous emptying of the PEC pump intake chamber;

— two industrial air conditioners 11SAQ01AN001, 11SAQ01AN002 (12SAQ01AN001, 12SAQ01AN002) which are used to establish and maintain the required temperature in the electrical rooms.

Technical specifications of the circulation pumps 11PEC10AP001, 11PEC10AP002 (12PEC20AP001, 12PEC20AP002) are described in Table 2.3.6.3.

Table 2.3.6.3 - Technical specifications of the circulation pumps

Pump		A 600-HVBW-755-97-OU	
Position	**Parameters**	**Measuring units**	**Value**
1	Nominal output	m^3/hour	4200
2	Nominal discharge	m	65
3	Maximum discharge	m	94

Coolant water is supplied from the ponds to the reactor island PO by two circulation pumps installed in the buildings 11(12) URF. Each channel (№1 and №2) has a working pump and a stand-by pump.

Technical description of pumps A 600-HVBW-755-97-OU produced by Sigma (Czech Republic).

Water is supplied to the impeller through the suction sector elbow and is discharged through the spiral housing via the pump discharge pipe located horizontally at an angle of 90° to the pump axis. The water is supplied to the circulation pumps from the water intake chamber via elbow-bent suction pipes (elbows). Water is discharged from the pumps via horizontal pressure pipes with a diameter of 800 mm with installed bellows expansion joints (Figure 2.3.6.4a-b).

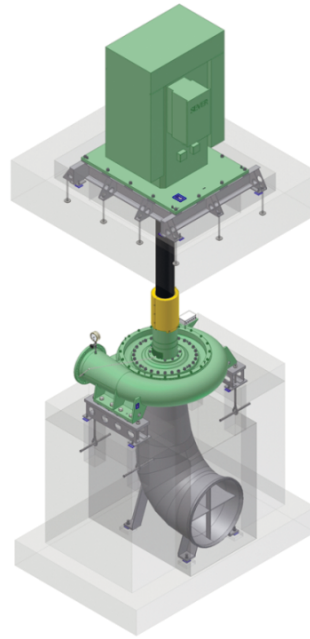


Figure 2.3.6.4a - A 600-HVBW-755-97-OU pump layout



Figure 2.3.6.4b - A 600-HVBW-755-97-OU pump general layout

Additional check valves and pressure valves with a retractable spindle are installed on pressure pipelines (downstream from pumps):

— with 800 mm diameter for water supply from the pump to the system;

— with 250 mm diameter (idle discharge valves) for unloading of pumps after connection to the system.

The circulation pump electric motors are connected to the electrical installations from the auxiliary busbars of the power unit, which are connected to the electrical installations of the reliable power supply section from diesel generators. In case of loss of auxiliary power supply i.e. when an electric section is de-energized, the first pumps are started to feed the de-energized section according to the stepped start program after connecting to a diesel generator's electric section.

Technical description of water treatment plants

In the intake chambers of the circulation pumps of each channel, upstream of the suction elbows of PEC pumps, there are water treatment plants 11REA10AT001 (12REA20AT001) installed to remove impurities from water supplied to the suction part of PEC pumps from separate sections of the spray cooling ponds. The water treatment plant (Figure 2.3.6.5) is a closed circuit consisting of separate metal frames (sections) sealed between each other by rubber bands and mounted on two closed ring-type roller-sleeve chains. Meshwork is made from 1,2 mm thick stainless steel wire with 5 × 5 mm mesh cells. Links of the chain transfer the forward motion to the gear wheels seated on one drive shaft.

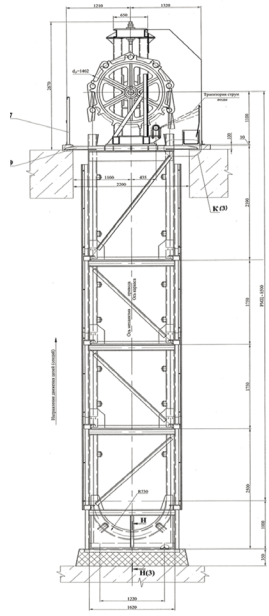


Figure 2.3.6.5 –Water treatment plant

Meshwork of the water treatment plant is flushed by water cooled in PEB pond, which is fed into the flushing perforated collecting channel inside the plant housing. The plant is flushed every 8 hours, as well as when a level difference of more than 100 mm occurs in the treatment circuit of the plant. When the water level difference increases to 300 mm, a warning light appears on the control panel in the building 11 (12) URF, and a relevant signal is sent to the main control room and the stand-by shutdown panel as part of the general "Fault in URF" signal.

Bulkhead (single-section) gates are installed in the guides of the stoplog windows (from the suction side of the pump). They are used when there is a need to carry out repair of the circulation pump without emptying the intake chamber. The dimensions of the bulkhead gates are 1500 × 2100 mm.

Drainage pumps

Drainage pumps 10PEC61 (Figure 2.3.6.6) are used in the internal drainage systems of the spray cooling ponds. Pumps 10PEC71 are designed for use in the leak drainage systems of the switching chambers 13 (14) URR. All pumps for pumping drainage water are installed in 13 (14) URR. Each building has two pumps, one for each system. Pumps are centrifugal, monoblock, horizontal, with packing seals of the shafts, single-stage with axial fluid supply and spiral outlets.

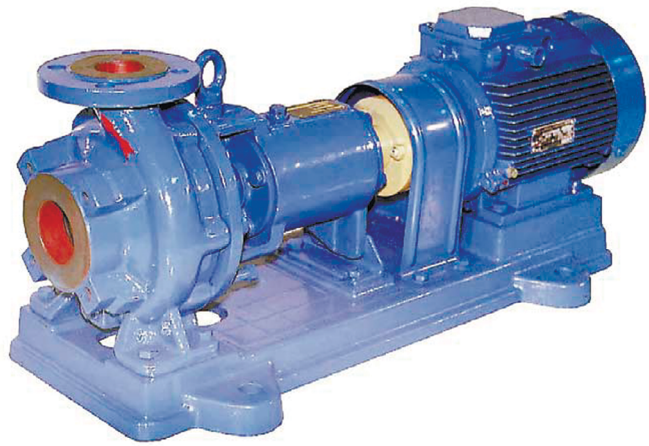


Figure 2.3.6.6 – Overall view of a drain pump

Drainage pumps 10PEC51 (figure 2.3.6.7) are intended for:

— automatic pumping of water leaks from two drain collection pits N°. 1 and N°. 2 URF. Pits are interconnected by a single pipeline D300 laid in the concrete basement of the pumping station;

— pumping of water (during planned preventive maintenance in the manual control mode) by pumps from the suction intake chamber for the purposes of repairing PEB pipelines in a spray cooling pond and an underwater part

of a water treatment machine. Water is pumped out of the intake chambers of the circulation pumps by two working pumps to a reserve tank 00URX through an integrated underground collector. Capacity of the reserve tank allows pumping out water only from one of the two sections of the spray cooling pond.

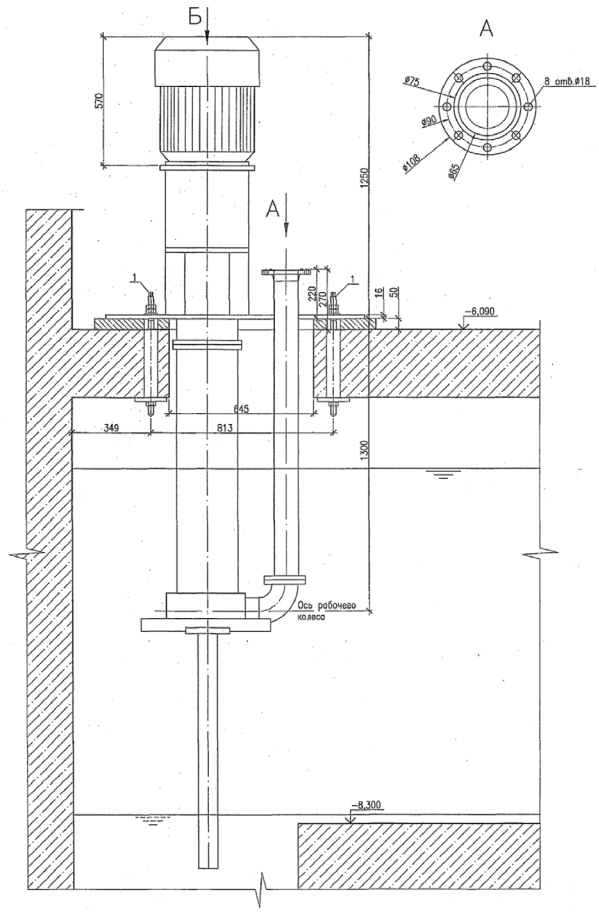


Figure 2.3.6.7 - Drain pump 10PEC51

Additional purification filters 11PEB01AT001, 12PEB01AT001, filter-type ALF60-R, intended for additional purification of chilled water PEB downstream of the water treatment nets from small debris with a diameter of up to 2,5 mm. The need for additional water purification on with the use of filters is justified by a small gap between the cooling surfaces of the plate-type heat exchangers installed in the reactor compartment. Filters are installed in channels 11 (12) URZ, in the pressure main line of the PEB system, at the inlet of the cooling water system to 10UJA building.

Filters are equipped with an automatic flushing system to remove mechanical impurities (filtrate) without interrupting the main flow.

There are eight subsurface wells around the perimeter of spray cooling ponds 11URR and 12URR. Six collecting wells are connected to a drainage network of water leaks from the bowl of the spray cooling pond. The other two wells have electric drive isolation valves supplying partially demineralized water to fill up corresponding sections of ponds from GCF system (from chemically demineralized water plants).

PEB system pipelines

The following PEB pipelines are installed within 11 (12) URF (Figure 2.3.6.8):

— gravity feed water supply pipeline from spray cooling ponds through the switching chamber to the water intake chamber of the pumping station with a diameter of 1220 x 11 mm;

— pressure feed water supply pipeline to supply water to the system consumers with a diameter of 1020 x 10 mm;

— hot water outlet pipeline from PE system consumers to spray cooling ponds through the switching chamber with a diameter of 1020 x 10 mm and other auxiliary pipelines.

The following PEB pipelines are installed outside of 11 (12) URF:

— pressure line (from the pumping station to the UBN forked (Y-tube) tube), with diameter of Dy1000 and length of 203 m, laid in 11 (12) URZ tunnel. A forked tube (Y-tube) has two branches:

a) Dy500 branch with a length of 60 m for the supply of water to consumers of UBN building. The main consumers in the UBN building include refrigeration units and XJG heat exchangers installed on the diesel-generating set.

b) Dy800 branch with a length of 44 m for the supply of water to the 10UJA building, extension of the pressure line (between the 10UJA building and the 10UKC annex), a pipeline section with a length of 39m and a diameter of Dy600;

— PEB drain pipe (from the fork tube on UBN to the distribution chamber of the URR spray cooling pond) has a length of 246 m and a diameter of Dy1000. In the URR switching chamber, the drain water pipeline is split into 4 branches with a diameter of Dy600, of which two branches supply water to the spray nozzles, and the other two branches supply water to the idle (winter) discharge through the connecting valves. Two pipes 63 m long extend from each pipe Dy600 (water supply to the nozzles into the spray cooling pond section) through a Y-tube, the diameter of these two pipes diminishes from Dy500 to Dy300 as water is dispensed at the nozzles. The distance between the nozzles in the pipeline is 8,7 m. Цо-85 centrifugal spray nozzles in one section of the spray cooling pond are arranged in two linear rows with 8 nozzles in each row, i.e. 16 nozzles per section. The number of installed nozzles in both sections of the pond is equal to 32 nozzles.

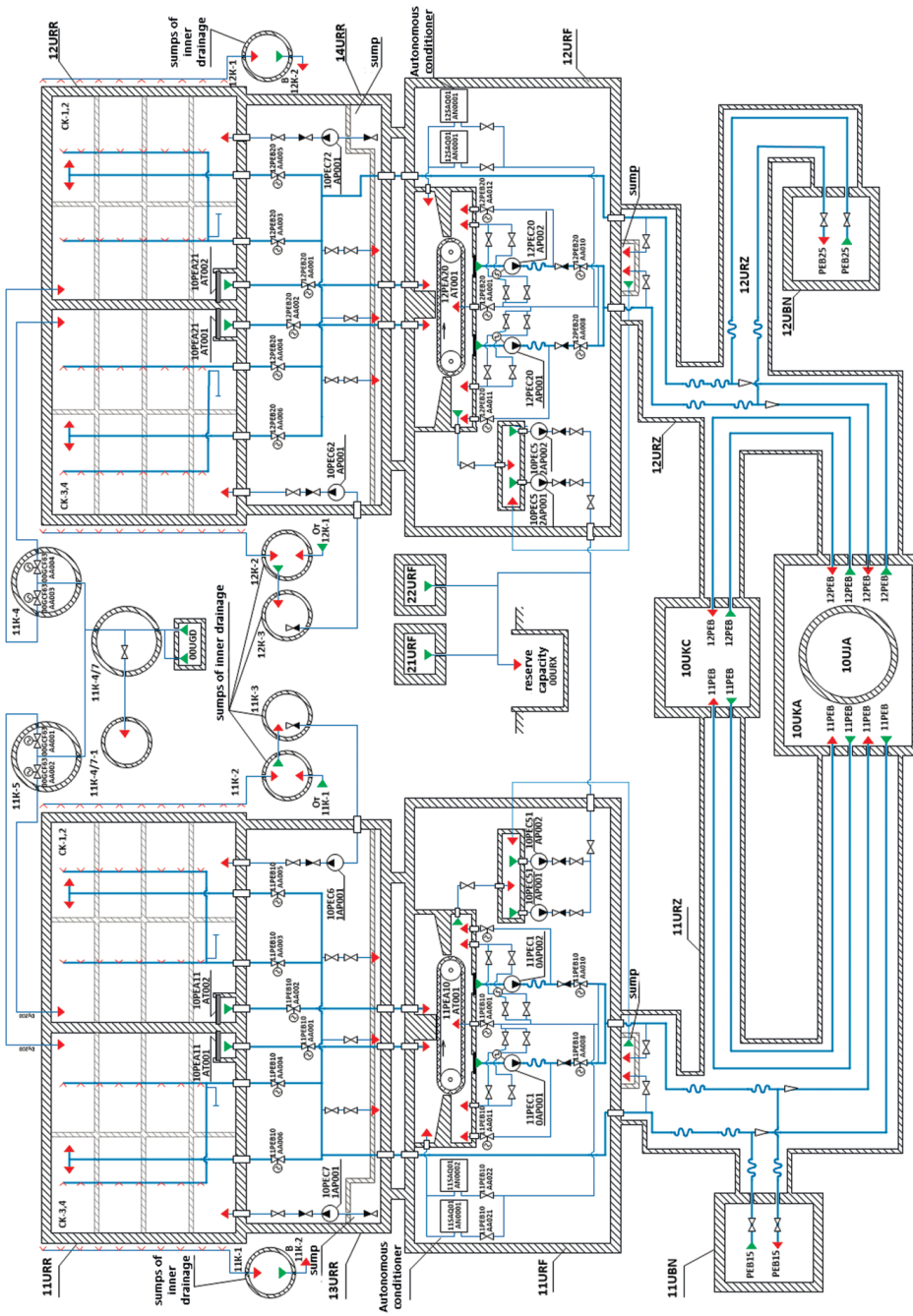


Figure 2.3.6.8- PEB system pipelines

Normal operation of the system

In normal operation conditions both channels of the PE system ensure heat removal through the following heat exchangers:

— Plate-type heat exchangers 11KAA10AC001, 12KAA20AC001, which are installed in the essential intermediate cooling circuit;

— Plate-type heat exchangers 10PGB50AC001, 10PGB50AC002, 10PGB60AC001, 10PGB60AC002, which are installed in the normal operation intermediate cooling circuits.

— Plate-type heat exchangers 11XJG10AC001, 11XJG10AC002; 12XJG20AC001, 12XJG20AC002, which are installed in the intermediate cooling circuits of diesel generator sets.

In normal operation conditions each channel of the PE system has an ability of automatic start of the essential backup cooling water circulation pump.

In the modes of planned and repair shutdown cooling of a power unit, it is allowed to decommission one cooling channel for the purposes of repair works or technical inspection of the equipment.

If the reactor is operating at power, it is allowed to decommission one circulating pump (in one of the channels) for the purposes of repair works for repair for a period not exceeding 10 days. At the same time, the remaining serviceable active elements of the system must be subjected to an extraordinary additional inspection.

The cooling ability of the spray cooling ponds helps to maintain water temperature at the inlet to consumers in the reactor containment and the standby diesel electrical power plant within the range from + 10 °C to + 25 °C. To exclude the formation of ice on the rims of the bowls and on the reinforced concrete structures of the spray cooling ponds, the water distribution should be fully switched over to the "winter discharge" mode when the water temperature at the outlet from the reactor containment falls below + 10 °C.

The reverse switching of the water distribution to the "summer discharge" mode should be fully implemented when the water temperature at the outlet from the reactor containment exceeds + 25 °C. At intermediate temperatures at the inlet of cooling water to the reactor containment within the range from + 10 to + 25 °C, distribution of water is controlled by the height of the fountains (torches) from the spray nozzles.

Design-basis accident mode

In design-basis accident conditions, circulation pumps in both channels of the PE system remain in operation. Supply of cooling water to heat exchangers removing heat from normal operation consumers in the 10UKC building (10PGB50AC001, 10PGB60AC001) is stopped by automatic closure of the isolation valves installed on the pressure and drain headers of the water supply to the 10UKC building, the whole flow of the cooling liquid is used to cool down the essential intermediate circuit heat exchangers 11KAA10AC001, 12KAA20AC001.

If the emergency power supply system is energized from the auxiliary transformer, an automatic standby power supply circuit breaker is provided between mutually redundant mechanisms.

If power supply is provided from the diesel generator, only the first mechanism is activated, whereas the use of an automatic standby power supply circuit breaker is prohibited.

2.3.7 AIR-CONDITIONING AND VENTILATION COOLING SYSTEMS

Refrigerating machines 11,12QKB

Refrigerating machines 11,12QKB are intended to cool down water circulating in refrigeration pipelines of essential consumers to the required parameters before delivery to ventilation systems heat exchangers to cool air in the rooms with the process or electrotechnical equipment of safety systems (Figure 2.3.7.1).

System 11,12QKB provides cold water to refrigeration supply system of essential consumers 11,12QKJ of reactor building 10UJA and to refrigeration supply system of essential consumers 11,12QKD of 11UBN and 12UBN emergency power supply buildings.

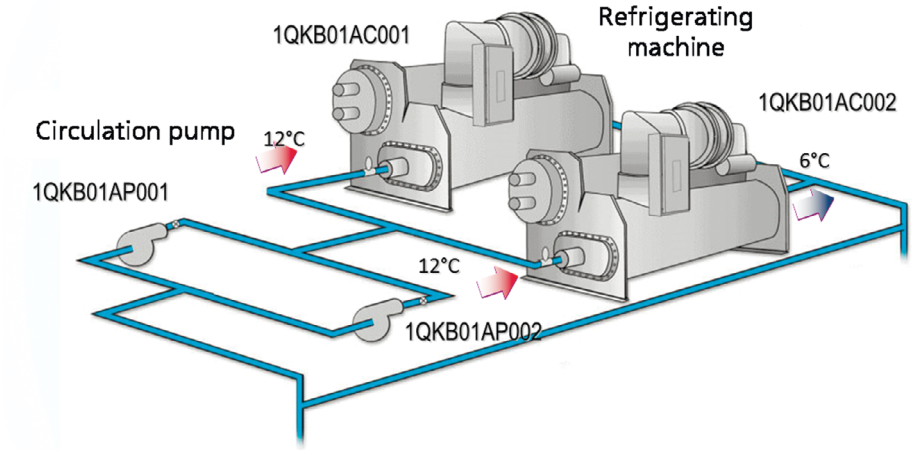


Figure 2.3.7.1 - Refrigerating machines 11,12QKB

11,12QKB system components are located in buildings 10UJA, 11,12UBN and 11,12URZ.

The following equipment is located in buildings 11UBN and12UBN:

— refrigeration supply pumps 11,12QKB01AP001,002;

— refrigerating machines 11,12QKB01AC001,002;

— surge tanks 11,12QKB07BB001;

— pipelines and valves;

— pipelines and valves between buildings 11UBN – 12UBN and building 10UJA are located in tunnels for essential consumers pipelines 11URZ and 12URZ.

The heat exchangers of ventilation system of emergency power supply buildings 11UBN and 12UBN (11,12SAC01, 11,12SAD01, 11,12SAE41, 11,12SAE45, 11,12SAC44) are consumers of 11,12QKB system.

The heat exchangers of ventilation system of reactor building (11,12KLB02, 11,12KLC01, 11,12KLC03, 11,12SAE42, 11,12SAC50, 11,12SAC40, 11,12SAC43, 11,12SAC45, 11,12SAC46, 11,12SAE44, 11,12SAE56, 11,12SAA41) are located in building 10UJA.

11,12QKB system consists of two physically separated channels which are designed as independent in terms of process and electric connections. Each channel includes the following elements:

— two refrigerating machines 11(12) QKB01 (one in operation, another in standby under nominal conditions);

— two pumps 11(12) QKB01 ЦНМ 100-65-200 (one in operation, another- in standby);

— valves and pipelines.

The cold source is provided by the air-cooled refrigerating machines CVHE 0270 (Figure 2.3.7.2) (refrigerating capacity Q = 749,96кW) which are equipped with electric hardware, protective automation, control panel and which consist of:

— compressor with in-built electric drive motor with lubrication system;

— evaporator;

— condenser;

— freon pipelines with valves and pipe fitting;

— refrigerating machine instrumentation and control system.

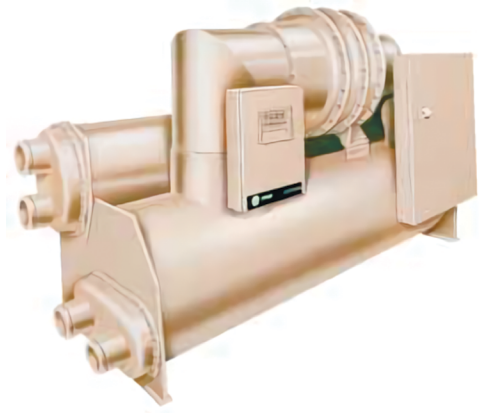


Figure 2.3.7.2 – Water-cooled refrigerating machines CVHE 0270

R123 serves as a refrigerant agent. Demineralized water of plus 6 °C/12 °C supplied to consumers of a safety train is used as the coolant fluid.

For the purpose of temperature changes compensation as well as for maintaining pressure at pumps suction, the surge tank 11,12QKB07 is connected to the return pipeline to maintain the required pressure due to nitrogen cushion in the tank.

The capability of essential consumers refrigeration system 11,12QKB and its function fulfilment depends on the following systems:

— 11,12PEB Cooling water of 11,12PEB system coming from spray ponds is used for heat removal from refrigerating machines condensers. Under the nominal mode no less than 128,5 m^3/h is required (in accordance with technical characteristics of the refrigerating machine). When the Complex operation is provided from Diesel-Generator (two refrigerating machines at reduced capacity) no less than 205,6 m^3/h is to be delivered;

— 10GCF. Demineralized water from 10GCF system is used to fill-in the process components of 10GCF11,12QKB system;

— 10LDN30,40. System to maintain the intermediate circuits water chemistry with refrigerant tri-sodium phosphate corrective treatment.

Technical characteristics of refrigerating machines system 11,12QKB are given in Table 2.3.7.1.

Table 2.3.7.1 - Technical characteristics of refrigerating machines system 11,12QKB

Name	Technical characteristics
11(12)QKB01AC001(002)	Refrigeration capacity,750 кW
11(12)QKB01AP001(002)	Supply – 107,6, m^3/ч, head – 0,44 MP
11(12)QKB07BB001	Cubic capacity – 2,7 m^3, refrigerant volume – 0,8 m^3, nitrogen volume – 1,9 m^3, operating pressure – 0,4 MP

Refrigerating machines 11,12QKB01 are related to the support safety system, safety class 2.

Surge tanks 11,12QKB07 are related to safety class 2 (2NO designation), group «B».

Refrigeration pipelines of essential consumers ventilation systems 11,12QKB, 11,12QKD, 11,12QKJ are related to safety class 2 (2NO designation), group «B», and are manufactured from carbon steel.

Monitoring of process parameters as well as preventive and emergency alarms and the whole Complex control are provided from the Main Control Room (Emergency/Back-up Control Room).

System and equipment operation mode

The refrigeration supply system for essential consumers 11,12QKB operates under all normal operation modes including a Power Unit start-up, shutdown and under blackout.

When the Complex is in operation, the heated water of plus 12 °C comes from consumers of 10UJA, 11,12UBN buildings to the heated water header where water temperature and velocity are balanced. Then water is pumped by 11,12QKB01 pumps to evaporators of refrigerating machines 11,12QKB01 correspondently, where it is cooled down from plus 12 °C to plus 6 °C. After cooling down in evaporators water of plus 6 °C enters the common cold water header from which it is supplied to consumers. Constant values of cold water at refrigerating facilities outlet – regardless of heated water inlet temperature and flow rate – are provided by control system of cooling machines refrigerating capacity.

The pressure maintaining system consists of two surge tanks 11QKB07BB001, BB002 filled up with nitrogen and make-up valves which provide the automatic pressure keeping at pumps suction within the range from 0,3 MPa to 0,4 MPa (with automatic water make-up or discharge of excessive water).

Cooling water circuit 11,12PEB is intended for heat removal from condensers of cooling machines of essential consumers refrigeration Complex.

For filling-in and make-up of intermediate cooling circuit for nonessential loads of 11,12QKB ventilation system, the demineralized water is used with addition of tri-sodium phosphate $Na_3PO_4 \cdot 12H_2O$ as a corrective agent (to reduce corrosion wear).

Reactor building ventilation systems (KLA, KLB)

Ventilation of internal containment rooms is designed based on the concept of two zones segregation:

— Unattended zone: the staff access is allowed only during outage. It is the assigned area in the internal containment which includes the SGs and RCPs boxes, reactor vessel cavity, valves chamber and other rooms to house primary circuit equipment and pipelines;

— Controlled access area where the staff access and short stay is allowed during the operation at power level. It includes the Central Hall, RCPs electric motors compartments and other rooms. Video frame of the Reactor building ventilation system is shown at Figure 2.3.7.3.

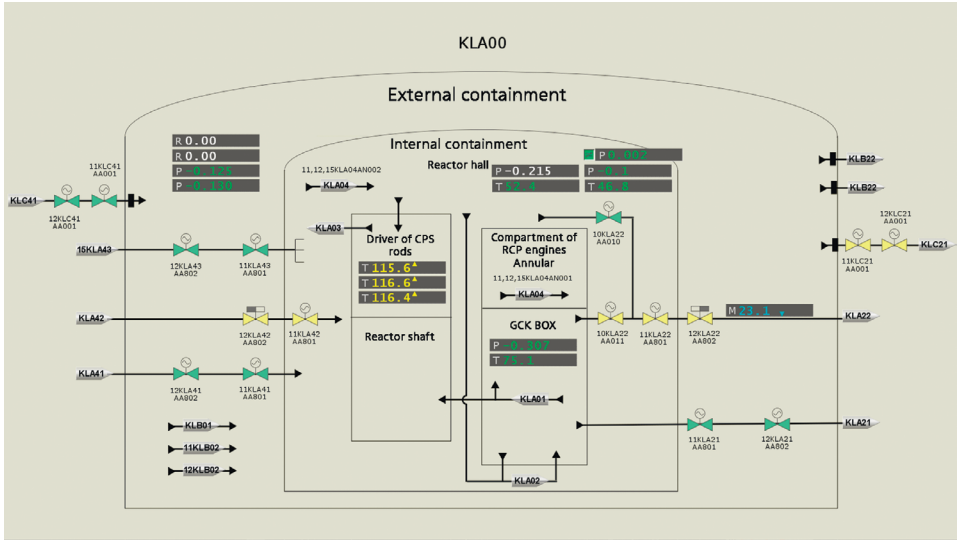


Figure 2.3.7.3– Video frame of Reactor building ventilation system

Duration of the staff stay within the controlled access area is identified by the Plant Health Physics Service depending on radiation situation in the internal containment.

Ventilation system of the above rooms is intended to perform the following functions:

— To remove excessive heat and moisture and to maintain optimal ambient conditions for equipment normal operation;

— To create normal climatic conditions for the personnel during outages and refueling operations;

— To provide air purification from aerosols and iodine;

— To provide under pressure no less than 200 Pa in unattended area and 150 Pa in the Controlled access area to prevent the active air propagation outside the sealed circuit through possible leaks under the Plant normal operation;

— To provide dilution of radioactive noble gases in rooms air.

For the purpose of heat and moisture removal from the pipelines and process equipment and in order to maintain the design room temperature, the air cooling recirculation pumps are provided (systems: 11KLA01, 12KLA01, 15KLA01, 11KLA03, 12KLA03, 15KLA03, 11KLA04, 12KLA04, 15KLA04) which are related to safety systems (identified as 11,12 as per KSS code) and to normal operation systems (identified as 10,15 as per KSS code).

Namely the ventilation systems related to safety support systems will be considered in more detail. Description of normal operation system is provided for general idea of ventilation in the containment and auxiliary structures.

Recirculation filtering system (10KLA02) is provided for air purification from radioactive contamination. This system ensures the permissible concentration of radioactive substances in the air under operating values of primary coolant total activity limit established for the Power Unit normal operation. Air is drawn from under the dome zenith as well as from the rooms of Main Circulation Circuit boxes, then it is purified at filter facilities and is delivered to the Main Circulation Circuit boxes. The system operation ensures purification of internal containment air from radioactive aerosols and iodine under normal operation conditions.

The emergency & maintenance intake-and-exhaust ventilation systems (10KLA41, 10KLA21) are provided for the internal containment aeration to enable maintenance activities as well as for post-accident air cleaning. 10KLA21 system is connected to KLA01 system pressure air line. Air is drawn from internal containment and after purification at filtration units is directed to the ventilation stack (10UKH) on the reactor auxiliary building (10UJA). 10KLA41 system supplies the outside air into the internal containment. Operation of 10KLA21, 10KLA41 systems ensure the hygiene-and-sanitary conditions for maintenance activities. The recirculation mode of 10KLA21 system operation also provides the post-accident air purification.

The intake-and-exhaust ventilation systems (10KLA42, 10KLA22) are provided for internal containment aeration and underpressure conditions under normal operation. Under normal operation, 10KLA22 system draws air from the primary circuit loop room, and during outages the air is taken from under the dome zenith. After purification on filtration facilities the air is directed to the ventilation stack (10UKH) on the reactor auxiliary building (10UKA). 10KLA42 system supplies the outside air to the internal containment. Operation of 10KLA22, 10KLA42 systems under normal operation ensures maintaining the underpressure of no less 200 Pa in the Main Circulation Circuit compartments and 150 Pa in the controlled access rooms of the internal containment during power operation (except blackout), purification of internal containment air from radioactive aerosols and iodine and prevents atmospheric contamination. 50 Pa underpressure in Main Circulation Circuit compartments in relation to controlled access rooms is maintained by means of pressure relief valves mounted on the compartments wall.

Ventilation systems are intended to ensure normal functioning of the containment equipment which is in operation or in standby.

For removal of excess heat generated by the process equipment:

— 11KLA01, 12KLA01, 15KLA01 — cooling system for unattended zone rooms (MCP boxes, reactor vault, valve chamber, etc.). The air from the MCP boxes is directed to the common header; after cooling on recirculation facilities it is delivered to the valve chamber rooms and for blowdown of supporting truss of dry containment and shielding plate, thrust truss and reactor branch piping. Operation of 11KLA01, 12KLA01, 15KLA01systems ensures maintaining the air temperature up to plus 60 °C under normal operation and up to 75 °C under violation of heat removal;

— 11KLA03, 12KLA03, 15KLA03 – CPS drives cooling system. The Reactor Hall air

coming through the CPS drives enters the common header. After cooling-down on recirculation facilities it is discharged into the facilities rooms. Operation of 11KLA03, 12KLA03, 15KLA03 systems ensures maintaining the air temperature at the system outlet up to plus 30 °C under normal operation and up to plus 45 °C under blackout;

— 11KLA04, 12KLA04, 15KLA04 – cooling system for controlled access rooms in the internal containment including ventilation centers, rooms for RCP motors maintenance, Reactor Hall. KLA04 system consists of two groups to support walkways and Reactor Hall. In the first group the air after cooling-down on recirculation facilities is delivered to the common cooling ring duct directed to walkways zone. In the second group the air is drawn through recirculation facilities and is delivered to Reactor Hall rooms. Operation of systems 11KLA04, 12KLA04, 15KLA04 ensures maintaining of air temperature in the controlled access zone up to plus 33 °C (in summer time) under normal operation and up to plus 60 °C under blackout;

— 11KLB02, 12KLB02 –air cooling system for Safety Systems penetrations rooms. The air is drawn through recirculation facilities and is delivered to penetrations sector rooms. Operation of systems 11KLB02, 12KLB02 ensures maintaining the air temperature in cable rooms up to plus 40 °C and not higher than plus 45 °C under emergency.

The video frame of KLA03 system of CRDM cooling is shown at Figure 2.3.7.4.

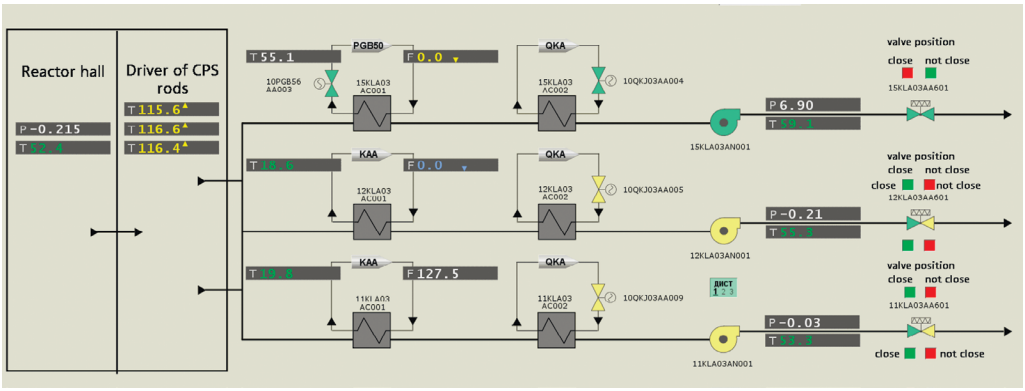


Figure 2.3.7.4- Video frame of KLA03 system of CRDM cooling

The ventilation systems are located in buildings 10UJA, 10UJB, 10UKA, 10UBB.

The following systems are located inside the internal containment 10UJA:

— 11,12,15KLA01; 11,12,15KLA03; 11,12,15KLA04

The following systems are located in the external containment 10UJB:

— 11,12KLB02

Isolating sealed valves are mounted in the rooms 10UJA, 10UJB, 10UBB, 10UKA.

In the points where air lines go through the fire partitions, the fire dampers are installed.

Technical characteristics of 10UJA ventilation system equipment are given in Table 2.3.7.2.

Table 2.3.7.2- Technical characteristics of ventilation system equipment

Name	Q-ty, pcs.	Main technical features
Recirculation cooling facility KLA01 consisting of :		
— air handling unit (made of corrosion resisting steel)	4	L = 90000 m^3/h H= 3000 Pa; N = 150 кW
— air cooling unit;	4	Q_{cold} = 860 кW G_{water} = 185 m^3/h
— air cooling unit;	4	Q_{cold} = 130 кW G_{water} = 18,6 m^3/h
— separator	4	—
Recirculation cooling facility KLA03 consisting of :		
— air handling unit (made of corrosion resisting steel);	3	L = 20000 m^3/h H= 5500 Pa; N = 75 кW
— air cooling unit;	3	Q_{cold} = 480 кW G_{water} = 103 m^3/h
— air cooling unit;	3	Q_{cold}= 480 кW G_{water}= 68,6 m^3/h
— separator	3	—
Recirculation cooling facility KLA04 (I Group) consisting of :		
— air handling unit (made of corrosion resisting steel);	3	L = 60000 m^3/h H = 3000 Pa; N= 110 кW

Name	Q-ty, pcs.	Main technical features
— air cooling unit;	3	Q_{cold}= 200 кW G_{water} = 42,9 m^3/h
— air cooling unit;	3	Q_{cold} = 282 кW G_{water} = 40,3 m^3/h
— separator	3	—
Recirculation cooling facility KLA04 (II Group) consisting of :		
— air handling unit (made of corrosion resisting steel);	3	L = 20000 m^3/h H = 800 Па; N= 11 кW
— air cooling unit;	3	Q_{cold}= 25 кW G_{water}= 5,4 m^3/h
— air cooling unit;	3	Q_{cold}= 90 кВт G_{water}= 12,9 m^3/h
— separator	3	—
Recirculation cooling facilities 11,12KLB02 consisting of :		
— air handling unit	4	L = 2000 m^3/h H = 800 Pa; N = 1,5 кW
— air cooling unit;	4	Q_{cold} = 7,0 кW G_{water} = 3,36 l/s
— air cooling unit;	4	Q_{cold} = 6,9 кW G_{water} = 2,74 l/s
— separator	4	—

As per the design, the system elements KLA01, KLA03, KLA04 have classified designation «3N», air cooling units and hydro-valves of systems KLA01, KLA03, KLA04 are related to Group «C».

11,12KLB02 system elements have designation «2NO».

Systems 11,12KLA01; 11,12KLA03; 11,12KLA04; 11,12KLB02, 11,12KLB22 are provided with emergency power supply of Group II (from safety system DGs).

Systems 15KLA01; 15KLA03; 15KLA04 are energized from normal operation reliable power supply from common-plant Diesel Generator. The isolating valves of 10KLA21, 10KLA22, 10KLA41, 10KLA42, 15KLA43 systems are provided with emergency power supply of Group II.

For ventilation systems UJA, UJB the operating parameters monitoring is provided from AWP (automated working place) of Digital Control Upper Level System (SBBU) of Ventilation Control&Monitoring System.

Ventilation systems UJA, UJB consist of recirculation, exhaust and draw-in air systems.

The diagram and general view of radial type fan is given on Figure 2.3.7.5.

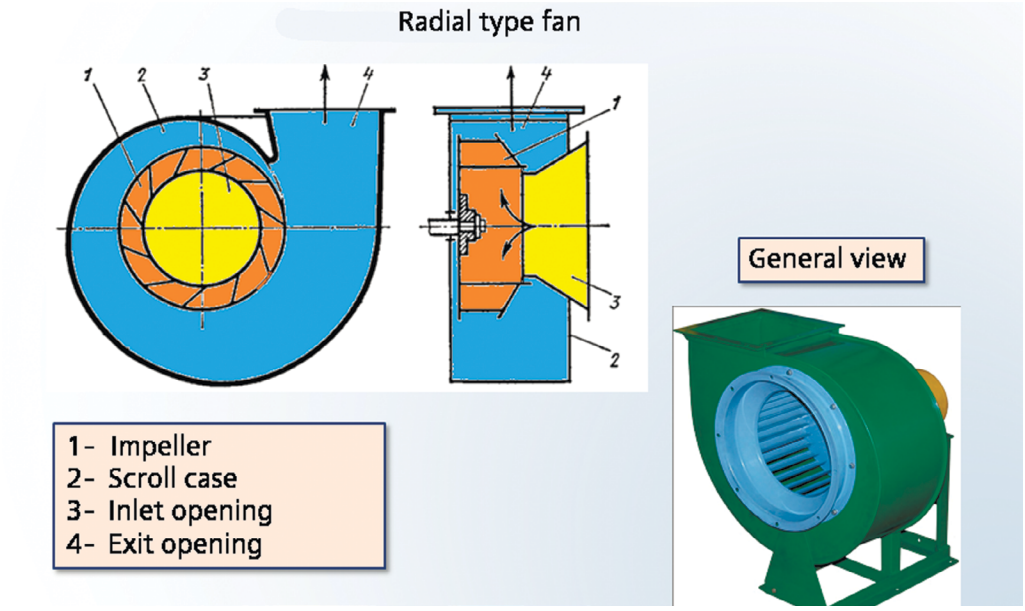


Figure 2.3.7.5 – Diagram and general view of the Radial-type Fan

In terms of design, the recirculation facilities KLA01, KLA03, KLA04, KLB02 consist of:

— Ventilation unit;

— Stop-check valve or check valve;

— Two sequentially installed groups of air-cooling units. The first group (along with the air flow) receives water from intermediate circuit of Reactor compartment consumers KAA or from intermediate circuit normal operation consumers PGB with coming water temperature up to 33 °C. Water from non-essential consumers refrigeration supply system 10QKJ with coming water temperature

up plus 6 °C and return water up to plus 12 °C is delivered to the second group.

In the cold period the cooling water of 11PEB, 12PEB essential consumers is supplied to KLB02 recirculation facilities to the first (in the direction of air flow) group with coming water temperature plus 17 °C and return water plus 21 °C. During the warm period water is supplied to the second group from essential consumers refrigeration system of appropriate safety system train 11QKJ, 12QKJ with coming water temperature up plus 6 °C and return water up to plus 12 °C.

The view of combined filtration facility is given on Figure 2.3.7.6

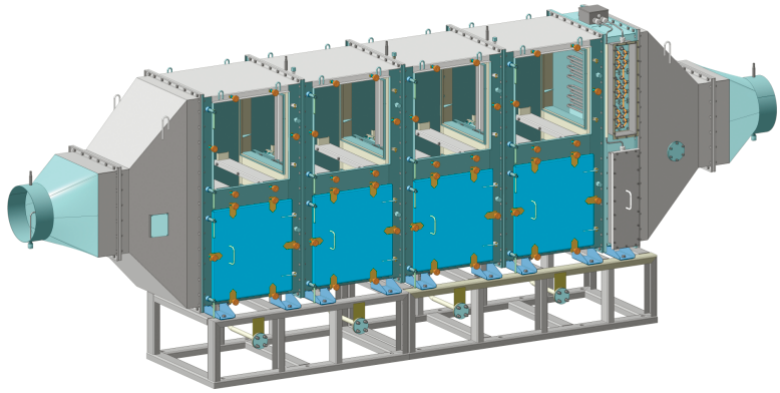


Figure 2.3.7.6 - Combined filtration facility (UFKPE)

The combined filtration facility consists of the following sections:

— Drop/mist extractor;

— Air heater;

— Aerosol purification section (of High-efficiency aerosol filter);

— Two sections of FAI iodine purification with granulated carbon sorbent performing functions of adsorbent-type filter;

— Section of aerosol purification at the facility outlet which is intended for entrapping the secondary aerosols which can't be excluded in case of carbon sorbent wear-out.

The air lines from corrosion resistant steel are used for KLA01, KLA02, KLA03, KLA04 systems. The air lines for KLB01, KLB02 system are manufactured from carbon steel with corrosion-resistant coating.

Normal operation system parameters

— KLA01 — two facilities are in operation and two in standby. Inflowing air temperature in summer time is plus 20 °C, in winter- plus 30 °C and up to 46 °C in case of heat removal violation;

— KLA02 – one facility is in operation; another is in standby. The contaminated air is drawn from under the dome of internal containment in a volume of 1000 m^3/h and is mixed with 2000 m^3/h of air taken from the SG box. After purification on aerosol and iodine filters the air in volume of 3000 m^3/h is discharged to the SG box;

— KLA03 — two facilities are in operation; one is in standby. Inflowing air temperature is plus 30 °C, and plus 45 °C under blackout;

— KLA04 — consists of two groups of equipment. In each group two facilities are in operation; one is in standby. The air temperature in the controlled access zone in maintained plus 33 °C in summer time, plus 40 °C in winter time and plus 60 °C under blackout. In case of temperature rise in the controlled access zone, the Operator can additionally switch on the standby facility, and switch it off in case of temperature decrease;

— KLB01 — consists of two groups of equipment. In each group one facility is in operation, one is in standby. The air temperature in cable rooms is maintained up to plus 40 °C;

— KLB02 — in each system channel one facility is in operation, one- in standby. The air temperature in cable rooms is maintained up to plus 40 °C, under emergency – up to plus 45 °C;

Equipment and system operation mode

The mode of UJA, UJB ventilation systems operation is defined in terms of the NPP Power Unit main operating modes.

Under normal NPP power operation all ventilation facilities operate in accordance with their designated function except those which are in standby and emergency&maintenance ventilation systems.

The planned shutdown for maintenance: (rated air parameters for personnel in the internal containment are achieved by means of inflow and exhaust emergency&maintenance systems (10KLA41, 10KLA21), which perform aeration of internal containment after its unsealing and by 10KLA22 exhaust system.

Emergency modes depending on the primary coolant leak size:

— «small leak» – pressure increase up to 0,17 MPa (abs.) and steam-gas mixture temperature rise up to plus 90 °C in sealed compartments designed for overpressure (in response to signal «internal pressure containment» is above 0,0003 MPa (exc.) the system of internal containment air purification 10KLA02 and exhaust and inflow system maintaining underpressure in the internal containment (10KLA42, 10KLA22) are turned off; isolating valves of ventilation system on internal containment sealing line close down; exhaust system to produce underpressure in external containment KLB22 starts working. In response to signal «internal pressure containment» is above 0,03 MPa (exc.), all recirculation systems located in the internal containment are turned off);

— «large leak» – rupture of main circulation pipeline with unimpeded two-sided coolant leakage and pressure surge up to 0,40 MPA (exc.) inside the internal

containment (all ventilation systems intended for internal containment rooms do not operate under this mode). After the accident elimination the exhaust emergency & maintenance system 10KLA41 starts to operate as per the set algorithm).

Loss of external power supply — (ventilation systems 10KLA42, 10KLA22, 10KLA02, 10KLB02 which perform functions of normal operation systems, not safety important, are not in operation. Ventilation systems 11,12KLA01, 11,12KLA03, 11,12KLA04, 11,12KLB02, 11,12KLB22, which perform functions of safety systems and safety important systems, continue operation in accordance with their designated purpose).

Depending on weather conditions, systems 10KLA41, 10KLA42, 15KLA43 follow winter or summer algorithm set up in Ventilation Control&Monitoring System (SKUV) («SUMMER KEY» or «WINTER KEY»).

Fast-acting isolating valves 11KLC41AA001, 12KLC41AA001, 11KLC21AA001, 12KLC21AA001mounted on the air branch line into the external containment, close down automatically following closure of isolating valves of KLA22 system. During outage this closure interlock is deactivated.

Ventilation systems of Reactor compartment auxiliary building (KLC)

UKA ventilation systems are intended to perform the following functions:

— Excessive heat and moisture removal and maintaining optimal ambient conditions for equipment normal operation;

— Provision of normal climatic conditions for the staff during outage and refueling during shutdown;

— Air purification from aerosols and iodine;

— Provision of 50 Pa underpressure in rooms with possible radioactive leakages;

— Maintaining underpressure under normal operation and ventilation for maintenance works in the external containment.

Systems operating designation and function

Systems 11,12KLC01, 11,12KLC03 are intended for removal of excessive heat generated by the process equipment. The air is drawn through recirculation facilities and is directed to primary circuit emergency cooldown compartments (for 11,12KLC01 systems) and secondary circuit rooms (for 11,12KLC03 systems). The systems operations provide maintaining the process rooms air temperature up to plus 40 °C and not above plus 45 °C under emergency.

11KLC01, 12KLC01 — recirculation systems of air-cooling of primary circuit emergency cooldown compartments.

11KLC03, 12KLC03 — recirculation systems of air-cooling secondary circuit emergency cooldown compartments.

10KLC21, 10KLC22, 10KLC23, 10KLC24, 10KLC25, 10KLC26, 10KLC27, 10KLC28, 10KLC41, 10KLC42, 15KLC43 systems are intended to provide the required

meteorological conditions in the rooms of the Reactor compartment auxiliary building.

10KLC21, 10KLC22 – exhaust systems of safety systems trains rooms with aerosol and iodine purification draw air from periodically attended rooms and after purification on aerosol and iodine filters discharge it into the ventilation stack 10UKH. Air comes inside the rooms from corridors through pressure relief valves. In case of pressure increase in emergency cooldown compartments ≥ 0,03 кPa, the isolating valves-disk shutters 11,12KLC21,22 mounted on exhaust ducts, are actuated.

10KLC25, 10KLC26 – exhaust systems of primary circuit I&C rooms of 1 and 2 trains remove air from primary circuit I&C sensors rooms where pressure can go up to 0,13 MPa in case of gauge line rupture. Air is discharged into the ventilation stack without purification. In case of pressure increase in I&C sensors rooms ≥ 0,03 кPa, the isolating valves-disk shutters 11,12KLC21,22 mounted on exhaust ducts, are actuated.

The following systems facilities are located in the reactor compartment enclosing building 10UKA:

— 11,12KLC01

— 11,12KLC03

Systems elements: 11,12KLC01, 11,12KLC03 – have classified designation 2NO.

Systems 11,12KLC01; 11,12KLC03 are energized from emergency power supply system, group II.

Isolating valves of 10KLC21, 10KLC22, 10KLC25, 10KLC26 systems are energized from emergency power supply system, group II.

Ventilation systems UKA consist of recirculation, exhaust and inflow systems.

Recirculation cooling facilities 11KLC01, 12KLC01, 11KLC03, 12KLC03 consist of:

— Radial type fans;

— Two sequentially installed groups of air-cooling units with separators and trays. During the cold period the cooling water of 11PEB, 12PEB essential consumers is supplied to the first (in the direction of air flow) group with coming water temperature plus 17 °C and return water plus 21°C. During the warm period water is supplied to the second group from essential consumers refrigeration system of appropriate safety system train 11QKJ, 12QKJ with coming water temperature up plus 6 °C and return water up to plus 12 °C.

Equipment and system operation mode

The mode of UKA ventilation system operation is specified based on the Power Unit main operating modes.

The UKA ventilation system is considered to be in normal operation if the Power Unit is under normal power operation.

Recirculation systems 11,12KLC01, 11,12KLC03 periodically operate when the temperature in attended rooms rises above plus 40 °C. If the temperature in

attended rooms goes lower than plus 25 °C, the facility turns off. When «WINTER KEY» mode is activated in Ventilation Control&Monitoring System (SKUV) during cold season, air cooling is provided by primary air coolers powered from essential consumers refrigeration system (1PEB). When «SUMMER KEY» is activated during warm season, the secondary air coolers powered from essential consumers refrigeration system 11,12QKJ, are linked up following the interlocks.

In case of loss of external power, the recirculation cooling systems of process safety system compartments, 11,12KLC01, 11,12KLC03 go on working. Other ventilation systems UKA of normal operation are out of operation.

Under all operating conditions of UKA ventilation systems, the ventilation systems are in «automatic» mode.

When ventilation intended for individual rooms is turned ON, the exhaust ventilation starts first and the inflow mode comes next.

Under pressure increase in the primary circuit emergency cooldown compartments ≥ 0,03 кPa, 11,12KLC01 are turned off following the interlocks with AVR (automatic load transfer) inhibit switch.

All systems are interlocked with fire extinguishing sensors. In response to smoke detectors signal, exhaust and intake facilities are turned off automatically with AVR switch on inhibit, and forced smoke suppression facility in turned on. Since the smoke detectors actuation is resolved, the Ventilation Shop operating staff put the ventilation systems in operation.

The fire dampers are open in normal (protection) position. The blades surface is perpendicular to the open flow area. The dampers turn close automatically by the signal of smoke detectors.

Safety support Ventilation systems UBP, UBB, UJE

Ventilation systems of Safety support system UBP, UBB, UJE are divided into the following main groups in regard to compartments for which they are intended:

— ventilation systems of safety process system rooms (steam lines rooms) (11,12SAA41-21);

— ventilation systems of electrotechnical safety means (11,12SAC43-23; 11,12SAC45-25; 11,12SAC46-26; 11,12SAC47-27; 11,12SAE44-24; 15SAC43-23; 10SAC43-23; 15SAE45-25; 10SAC70);

— ventilation and air conditioning systems of MCR and ECR (11,12SAC40-20; 11,12SAC41; 11SAC42; 11,12SAC50-30; 11,12SAC51; 12SAC52; 10SAC48; 11,12SAE56-36; 11,12SAE42-22);

Systems operating designation and function

Air intake 11SAA41, 12SAA41 and exhaust ventilation systems 11SAA21, 12SAA21 of process safety system rooms (steam chambers compartments) are designed to perform the following functions:

— Excessive heat and moisture removal;

— Provision of the required conditions for equipment normal operation;

— Outside air dust elimination;

— Provision of the required conditions for maintenance activities.

The air intake systems 11SAC43, 12SAC43 and exhaust systems 11SAC23, 12SAC23 of safety system batteries rooms are intended for:

— To keep sulfuric acid vapor value within the permissible concentration (not more than 2 mg/m^3);

— To keep hydrogen content within the limits of explosion-proof concentration (no more 0,7 % by volume);

— To maintain the rooms air temperature within the range from plus 20 °C up to plus 25 °C under normal operation and from plus 10°C up to plus 35 °C under blackout.

The air intake systems 11SAC45, 12SAC45 and exhaust systems 11SAC25, 12SAC25 of safety systems electrotechnical rooms (uninterrupted power supply unit and electrical switchgear) and are intended for:

— Excessive heat removal;

— Maintaining rooms inside air temperature not higher than plus 35 °C;

— Outside air dust elimination;

— Provision of the required conditions for maintenance activities.

The air intake systems 11SAC46, 12SAC46 and exhaust systems 11SAC26, 12SAC26 of USB system (Safety Control System) rooms are intended for heat removal and maintaining rooms inside air temperature in the range of plus 20 °C up to plus 30 °C under normal operation and in the range of plus 10 °C up to plus 40 °C under extreme conditions.

The air intake systems 11SAC47, 12SAC47 and exhaust systems 11SAC27, 12SAC27 of USB rooms under blackout conditions without outside air contamination are intended for maintaining temperature in USB rooms and serve as BDBA management technical tool.

The air intake systems 11,12SAE44(42,56) and exhaust ventilation systems 11,12SAE24(22,36) of safety system cable rooms (MCR) are intended for:

— Excessive heat and removal;

— Maintaining cable rooms inside air temperature not higher than plus 40 °C;

— Outside air dust elimination;

— Provision of the required conditions for maintenance activities.

Intake-recirculation system 11,12SAC40(50) and exhaust system 11,12SAC20(30) of air conditioning in MCR (ECR) are intended for:

— Maintaining the MCR (ECR) temperature from plus 21 °C up to plus 25 °C and moisture from 40 % to 60 %;

— Maintaining the personnel rooms temperature from plus 21 °C up to plus 25 °C and moisture from 40 % to 60 %;

— Providing pressurization in MCR (ECR) not lower than 20 Pa;

— Outside air supply in the amount not less than 60 m^3/h per person under normal operation and 20 m^3/h per person under filter-ventilation mode;

— Outside air dust elimination;

— Providing MCR (ECR) habitability under emergency situation (NPP accident with radioactive aerosols and gases release to the environment, fire, appearance of toxic substances in atmospheric air of air intake area).

MRC (ECR) life support systems of MCR (ECR) 11,12SAC41(51), 11SAC42(12SAC52) are intended for:

— Provision of personnel operation conditions as per the sanitary norms;

— Provision of personnel protection against external penetrating radiation;

— Provision of protection of the personnel involved in radiation accident management at the Power Unit and under external events related to appearance of toxic, chemical and radioactive substances in the outside air;

— Provision of personnel protection in case of fire outside MCR (ECR) and on site.

Equipment of UBP, UBB, UJE ventilation systems (ventilation facilities, air-supply plants, stop and recirculation valves) is located at elevations plus 4,800 m, plus 16,500 m, plus 21,300 m and plus 26,100 m, within the area of their operation.

Technical characteristics of ventilation systems UBP, UBB, UJE equipment are given in Table 2.3.7.3.

Table 2.3.7.3 - Technical characteristics of ventilation systems UBP, UBB, UJE equipment

Name	Q-ty, pcs.	Main technical features
Intake air facility SAA41,21 including:		
— Coarse mesh filter	4	P_{min} =60 Pa; P_{max} =130 Pa
— Hot water heater	4	Q_{warm} = 255 кW; $G_в$ = 0,79 l/s
— Cold water cooler	4	Q_{cold}= 124 кW, $G_в$ = 5,41 l/s
— Air supply fan	4	L =16200 m^3/h, H = 1770 Pa,
— Exhaust air plant 11SAA21	4	L = 16200 m^3/h, H = 833 Pa
Intake air facility SAC43,23 including:		
— Coarse mesh filter	4	P_{min} =60 Pa; P_{max} =130 Pa
— Electric heater	4	Q_{warm} = 104 кW

Name	Q-ty, pcs.	Main technical features
— Hot water heater	4	Q_{warm} = 120 кW; $G_в$ = 0,37 l/s
— Cold water cooler	4	Q_{cold} = 53,7 кW, $G_в$ = 2,14 l/s
— Air supply fan	4	L =6550 m^3/h, H = 2047 Pa
— Exhaust air plant SAC23	4	L = 6000 m^3/h; H = 869 Pa
Intake air facility SAC45,25 including:		
— Coarse mesh filter	4	P_{min} =60 Pa; P_{max} = 130 Pa
— Hot water heater	4	Q_{warm} = 347 кW; $G_в$ = 1,08 l/s
— Cold water cooler	4	Q_{cold} = 136 кW, $G_в$ = 5,42 l/s
— Air supply fan	4	L =24000 m^3/h, H = 1866 Pa,
— Exhaust air plant 11,12SAC25	4	L = 24000 m^3/h; H = 940 Pa
Intake air facility 1SAC46,47,26,27 including:		
— Coarse and fine filter	4	Filter class G3,F9 P_{min} = 51 Pa; P_{max} = 185 Pa
— Hot water heater	4	Q_{warm} = 173 кW; $G_в$ = 0,54 l/s
— Cold water cooler	4	Q_{cold} = 55,5 кW, $G_в$ = 2,21 l/s
— Air supply fan SAC46	4	L =12000 m^3/h, H = 2000 Pa
— Exhaust air plant SAC26	4	L = 11600 m^3/h; H = 1003 Pa
— Air ventilation unit SAC47	4	L = 1000 m^3/h; H = 650 Pa
Intake air facility SAC40,20 including:		
— Coarse mesh filter	4	P_{min} =50 Pa; P_{max} = 120 Pa
— Fine mesh filter	4	P_{min} =83 Pa; P_{max} = 183 Pa
— Hot water heater	4	Q_{warm} = 128 кW; $G_в$ = 0,4 l/s

Name	Q-ty, pcs.	Main technical features
— Cold water cooler	4	Q_{cold} = 31 кW, $G_в$ = 1,22 l/s
— Cold water cooler	4	Q_{cold} = 34,7 кW, $G_в$ = 2,1 l/s
— Air supply fan	4	L = 8650 m^3/h; H = 1908 Pa
— Exhaust air plant SAC20	4	L = 9007 m^3/h; H = 1225 Pa
Intake air facility SAC50,30 including:		
— Coarse mesh filter	4	P_{min} = 34 Pa; P_{max} =104 Pa
— Fine mesh filter	4	P_{min} =57 Pa; P_{max} =157 Pa
— Hot water heater	4	Q_{warm} = 96 кW; $G_в$ = 0,3 l/s
— Cold water cooler	4	Q_{cold} = 29 кW, $G_в$ = 1,15 l/s
— Cold water cooler	4	Q_{cold} = 25,7 кW, $G_в$ = 1,53 l/s
— Air supply fan	4	L = 6300 m^3/h; H = 1627 Pa
— Exhaust air plant SAC30	4	L = 6000 m^3/h; H = 1030 Pa
MCR (ECR) life supporting facility SAC41 (Reserve control room SAC51) with air purification from aerosols and iodine , including:		
— Filtration -plant HS-F1	2 (2)	L = 300 m^3/h, P_{min} = 1000 Pa, P_{max} = 1700 Pa
— Centrifugal fan	2 (2)	L = 306 m^3/h; H = 2610 Pa
Intake air facility SAE56,36 including:		
— Coarse mesh filter	4	P_{min} = 35 Pa; P_{max} = 105 Pa
— Hot water heater	4	Q_{warm} = 28,9 кW; $G_в$ = 2,0 l/s
— Cold water cooler	4	Q_{cold} = 8,3 кW, $G_в$ = 0,36 l/s
— Air supply fan SAE56	4	L = 2100 m^3/h, H = 968 Pa

Name	Q-ty, pcs.	Main technical features
— Exhaust air plant SAE36	4	L = 2100 m^3/h; H = 739 Pa
Intake air facility SAE42,22 including:		
— Coarse mesh filter	4	P_{min} = 26 Pa; P_{max} = 96 Pa
— Hot water heater	4	Q_{warm} = 12,1 кW; $G_в$ = 0,04 l/s
— Cold water cooler	4	Q_{cold} = 4,1 кW, $G_в$ = 0,33 l/s
— Air supply fan SAE42	4	L = 800 m^3/h, H = 1220 Pa
— Ventilation unit SAE22	4	L = 800 m^3/h; H = 801 Pa

Design classification of UBP, UBB, UJE ventilation systems

Elements of systems 11,12SAC47, 11,12SAC27 have classified designation – **4**.

Elements of systems 11,12SAA21, 11,12SAA41, 11,12SAC23, 11,12SAC43, 11,12SAE24, 11,12SAE44, 11,12SAC25, 11,12SAC45, 11,12SAC26, 11,12SAC46, 11,12SAC20, 11,12SAC40, 11,12SAC30, 11,12SAC50, 11,12SAE36, 11,12SAE56, 11,12SAE22, 11,12SAE42 are designated as **2NO**.

Elements of systems 11,12SAC41, 11SAC42, 11,12SAC51, 11SAC52 — are designated as **2O**.

Isolating valves 11,12SAA21,41;11,12SAC20,40,41,30,50,51 – are designated as **2NL**.

Ventilation systems UBP, UBB, UJE are loads of different power sources.

Systems 11,12SAA21, 11,12SAA41, 11,12SAC23, 11,12SAC43, 11,12SAE24, 11,12SAE44, 11,12SAC25, 11,12SAC45, 11,12SAC26, 11,12SAC46, 11,12SAC20, 11,12SAC40, 11,12SAC41, 11,12SAC30, 11,12SAC50, 11,12SAC51, 11,12SAE36, 11,12SAE56, 11,12SAE22, 11,12SAE42, are energized from emergency power supply system, II Group of appropriate safety train.

Elements of systems 11SAC27, 12SAC27, 11SAC47, 12SAC47 and sealed fact-acting stop valves 11SAA21, 12SAA21, 11SAA41, 12SAA41 at the boundary of steam line compartments are energized from emergency power supply system, Group I of appropriate safety train.

System and equipment operation mode

The outdoor air from the intake facilities is supplied into the attended rooms in the following way:

— Inflowing air is drawn from the outside, passes through the blast protection

damper (UPVK) to the air intake facilities suction;

— It is purified on the air filters;

— It is delivered to the working area of attended rooms through the air lines system .

During the warm season (if needed) it is cooldown in the surface-type air cooler. During the cold season it is heated up in the surface heater or electric radiator (air intake-recirculation systems maintain the inflowing air temperature by regulating of valves opening rate on the outdoor air line and on recirculation line. When temperature goes down, the valve on the outdoor air line is partially closed and the amount of recirculation increases. When the temperature goes up, the valves are regulated in the reverse order.

The rooms air is drawn by exhaust facilities and discharged into atmosphere through UPVK (blast protection damper).

The smoke protection system provides the outdoor air supply through motor-driven heat-insulated valves, UPVK and fans head into the stair wells. Valves opening is interlocked with actuation of fans.

Operation of the CLSS (personnel life-supporting systems) ensures the purified air (on aerosol and iodine filters) supply into the MCR (ECR) air lines and air conditioning system.

The air intake systems 11,12SAA41, 11,12SAC43, 11,12SAC45, 11-12SAC46, 11,12SAC40, 11,12SAC50, 11,12SAE56, 11,12SAE42 are industrial air conditioners.

The industrial air conditioner is a multiple module system which consist of different units and mechanisms:

— Receiving module (with air heat-insulating valve);

— Receiving -mixing module (with air heat-insulating and control valve);

— Module of course and fine filters;

— Air heater with connection of hot water from heat supply system SBJ with temperature plus 150 °C/70 °C;

— Electric air heater;

— Air cooler with connection of cold water from refrigeration system QKJ with temperature plus 6 °C/12 °C;

— Fan module.

Facilities of systems 11,12SAC47, 11,12SAC27 consist of ducted fan-assisted units.

The personnel life-supporting facilities 11,12SAC41; 11,12SAC51 consist of:

— Filtration plant HS-F1 and centrifugal fan;

— Check valves for isolation of back-up facility from shared network of air lines;

— Control valves at discharge head to distribute air flow between recirculation lines and dump into the ventilation stack;

— Zink-coated air lines, duct flange joint.

System and equipment operation mode

The mode of UBP, UBB, UJE ventilation system operation is specified based on the Power Unit main operating modes.

The UBP, UBB, UJE ventilation systems are considered to be in normal operation if the Power Unit is under normal power operation. This is the operation mode of all working facilities except smoke exhaust ventilation systems and personnel life supporting systems 11,12SAC41, 11,12SAC51, 11SAC42, 12SAC52.

During preventive maintenance the ventilation systems of building UBP, UBB, UJE continue working as under the normal operation. If the scope of maintenance activities is big, the backup exhaust facilities can be used.

Under emergency modes («small leak» and «large leak») ventilation systems UBP, UBB, UJE continue working.

In case of loss of external power supply, ventilation systems 11,12SAA41,21; 11,12SAC43,23;11,12SAE42,22;11,12SAE44,24;11,12SAE56,36;11,12SAC45,25; 11,12SAC46,26; 11,12SAC40,20, 11,12SAC50,30 continue working. Other ventilation systems UBP, UBB, UJE are not in operation.

In case of blackout, systems 11SAC47-11SAC27, 12SAC47-12SAC27, 10SAC48 are activated.

Air conditioning systems of MCR (ECR) 11SAC40-11SAC20, 12SAC40-12SAC20, 11SAC41, 12SAC41, 11SAC42 (11SAC50-11SAC30, 12SAC50-12SAC30, 11SAC51, 12SAC51, 12SAC52) are functioning under normal operation and under emergency (filter system of ventilation and full isolation mode).

The filter-ventilation mode is activated automatically following readings of radiation monitoring sensors when photon radiation dose rate within the air intake area exceeds 3 x 10^{-7} Gy/h. The sensors are installed at the air intake locations. Sealed valves 11,12SAC40,20 close down, valves of filter-ventilation system 11SAC41– drop open and 11SAC41 fan turns on. In case 11SAC41 fan fails or turns off, or in case of 11SAC41 system failure, the Shift Supervisor of Ventilation Shop remotely switches on 12SAC40 and 12SAC41 systems.

The full isolation mode is introduced under emergency situations for a time required for the off-site Radiometric and Chemical surveillance services to identify concentration of the hazardous substances in the outside air taken-in by MCR air conditioners.

In addition, the full isolation mode shall be introduced when the outside air is contaminated with toxic agents, carbon oxide (in case of fire) and other hazardous substances which are not entrapped by filters-absorbers. Under the full isolation mode, the outside air sealed valves SAC40,20,41 are closing down, SAC40,20,41and MCR staff put in operation the CLSS air cylinder station which structure is described later.

All systems are interlocked with fire extinguishing detectors.

In response to smoke detectors signal, exhaust and intake facilities are automatically turned off with AVR inhibit switch on, and forced smoke suppression facility in

turned on. Since the smoke detectors actuation is resolved, the Ventilation Shop operating staff put the ventilation systems in operation.

The fire dampers are open in normal (protection) position. The blades surface is perpendicular to the open flow area. The dampers turn close automatically by the signal of smoke detectors.

Sealed valves SAA41,21; SAC40,20 mounted on the air lines, are in open position, electric schemes are set and interlocks are entered. When temperature in attended rooms goes above plus 60 °C (in case of steam line rupture), the disk dampers SAA41,21 turn close by interlock.

The ventilation systems are intended not only for provision of equipment operation conditions but for provision of Control boards habitability and survivability to create conditions for the staff stay.

Habitability — is a combination of factors which characterize conditions of personnel stay in a room and which enable personnel to conduct their normal professional activities.

Survivability — property of a system or element including control boards, to fulfil their designated functions despite of sustained damage.

CLSS Air Bottle Station

The CLSS (controlled life support system) compressed air bottle station is intended for storage of the air reserve and provision of controlled air flow rate into 11(12) SAC40 ventilation system to create the air head in MCR (ECR) under full isolation mode.

The Air Cylinder Station provides:

— Permissible sanitary-based staff working environment;

— MCR (ECR) habitability under emergency (accidental release of radioactive aerosols and gases into the environment, fire at NPP, appearance of toxic substances in the atmospheric air within the air intake zone).

The Air Bottle Station consist of two parts: pneumatics and electric equipment.

Pneumatic part in intended for storage of the air reserve (in waiting mode) and its supply to MCR (ECR) ventilation system with design air flow rate (in air delivery mode)). Pneumatic part consists of:

— Multi-purpose cylinders for compressed air, gases and liquids;

— Connection piping with valves;

— Flow metering unit.

Electric equipment is used for control of the Air Cylinder Station conditions and its operation. Electric equipment consists of:

— Local control board;

— Junction box.

The Air Cylinder Station is located in several rooms related to the appropriate Control Board:

11SAC42 – for MCR, 12SAC52 – for ECR.

Technical characteristics of habitability system equipment are given in Table 2.3.7.4.

Table 2.3.7.4- Technical characteristics of habitability system equipment

Name	Value
Operating media	Compressed air
Nominal volume of air reserve, nm^3	693
Air flow rate into SAC40 system, nm^3/h	80 - 130
Time to support the air flow, not less than, h	4
Nominal operating pressure, MPa, excessive	24,5
Number of compressed air cylinders, pcs.	15

Life support systems 11SAC42 и 12SAC52 refer to Safety Support Systems and are classified as 2O as per NP-001-15.

11SAC42 и 12SAC52 systems mechanisms receive power from emergency power supply of relevant safety train, Group II. The uninterruptible power supply (UPS) and energy accumulator consisting of two series connected lead-acid storage batteries 12 V and 7,2 A·h capacity are provided inside the Local Control Panel (LCP). Availability of UPS and energy accumulator provides the guaranteed operation of life-support system under loss of external power supply approximately for 5 hours and approximately for 20 hours in waiting mode.

In addition, terminals for connection of DC external power supply, 60 V (e.g. a car battery) are provided in LCP.

In case of problems with internal power supply (loss of external power, failure of batteries, etc.), the LCP generates a signal «Loss of power» to Digital Control System. The signal is also generated in case of additional external power source connection.

In case of full absence of LCP power supply, the automatic or manual control is impossible.

Monitoring of working parameters is provided from the AWP (automated working place) of Digital Control System Administrator.

Example of video frame for working parameters monitoring of Compressed Air Bottle Station of life-support system is given on Figure 2.3.7.7.

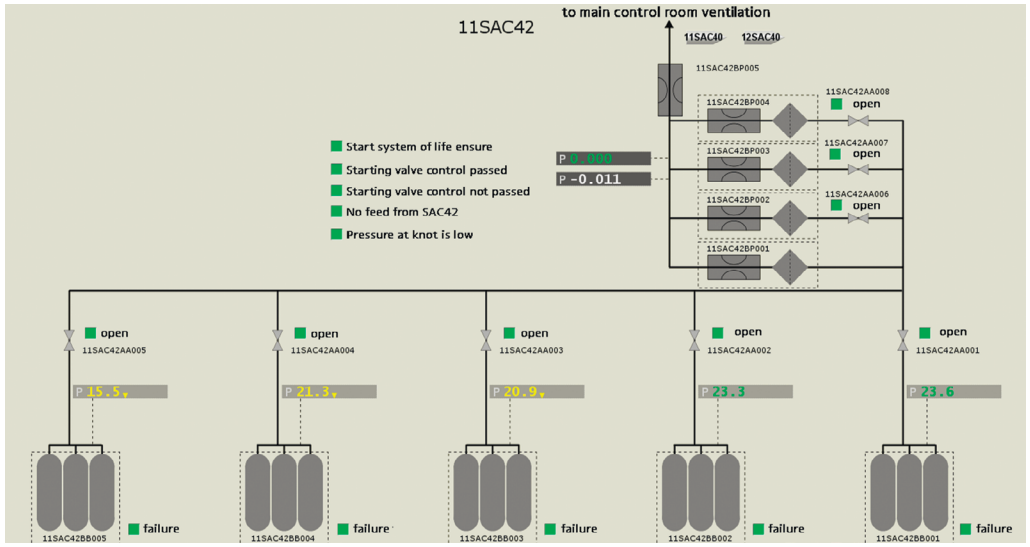


Figure 2.3.7.7 - Video frame for working parameters monitoring

The following parameters are under monitoring on the above video segment:

— Pressure in each unit of cylinders;

— Pressure upstream the bleed throttle;

— Start-up indicators of CLSS (controlled life-support system), opening of starting valves, low pressure in valve assembly, correct functioning of starting circuits of start valves.

CLSS has a computer-aided control (remote start-up, shutdown, operation based on pre-set algorithms) from the Digital Control System.

CLSS, if needed, can be started-up and controlled manually in accordance with cycle timing (provided by LCP).

Decision of a System actuation and its start-up is made by Unit Shift Supervisor from MCR mosaic panel 11CWF03 or from ECR safety mosaic panel 12CWC03. The manual system start-up is performed in field by the Ventilation Shop Operator of refrigeration equipment on the instruction of Ventilation Shop Shift Supervisor; the step-by-step instruction is available on the LCP front panel.

CLSS consist of:

— 5 gas bottle (cylinders) units;

— 5 valves assemble;

— Flow metering unit;

— Local control panel;

— Set of support equipment.

The gas bottle unit is intended for mounting of three high pressure composite compressed air cylinders of 185 liters capacity and 24,5 MPa working pressure.

External view of gas bottles unit is given on Figure 2.3.7.8.

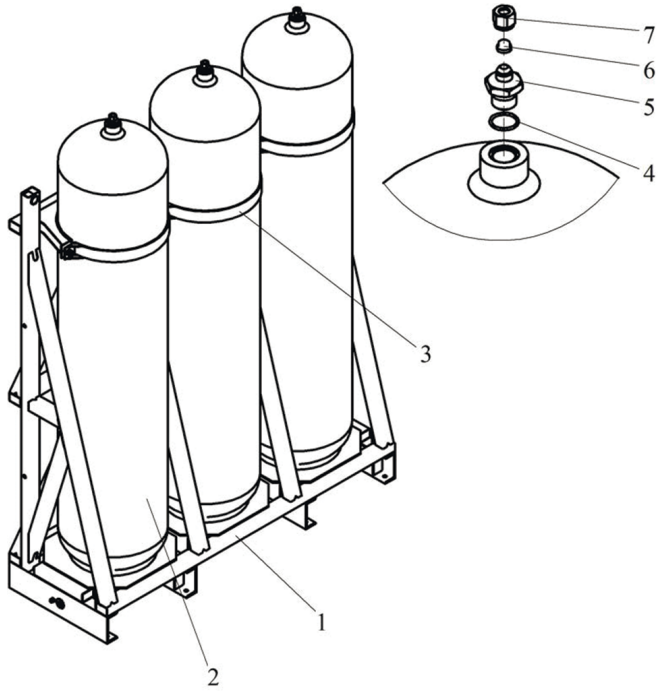


Figure 2.3.7.8 – The exterior of gas bottled unit

CLSS has two modes of operation:

— Waiting mode;

— Air Supply mode.

Most of the time the CLSS system is the waiting mode ready for start-up. The waiting mode is characterized by availability of LCP external power supply (must be energized) and excess pressure gauges.

Under the waiting mode the Digital Control System continuously monitors the excess pressure of cylinders compressed air by means of excess pressure gauges, and performs periodic testing of starting circuits of start valves operability by means of «PC TEST» commands to LCP.

Air supply mode is the mode when the air from CLSS (full isolation mode) is supplied to MCR (ECR) ventilation system.

Under the air supply mode, the Digital Control System generates commands for start valves opening following the readings of excess pressure gauges in accordance with the pre-set algorithms.

2.4 SAFETY CONTROL SYSTEMS

2.4.1 REACTOR CONTROL AND PROTECTION SYSTEM –ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (CPS-ESFAS)

2.4.1.1 General principles

NPP should have safety systems designed to perform the following basic safety functions:

— emergency shutdown of the reactor and maintaining it in a subcritical state ;

— emergency heat removal from the reactor;

— keeping the radioactive substances within established boundaries (NP-001-2015)/1/.

The NPP safety design is based on the strict requirement of adhering to the protection principle in depth. The basic idea is that there must be several layers or layers of protection to prevent unsafe conditions and that preference is always given to preventing unsafe conditions than mitigating their effects. Since the operation of the NPP and maintaining its safety requires a large number of functions, which increases even more due to the application of the protection principle in depth, it is important to know the significance of each safety function (IEC 61226/2 /).

The hierarchy of the formation of requirements for the functions of the safety systems (and, accordingly, the functions of the CPS-ESFAS) of the NPP-2006 project was carried out in accordance with the requirements of IEC 61226-05 and is shown in Figure 2.4.1.1.

The composition of the first (upper) level of the hierarchy of the formation of requirements for functions is determined by the in-depth protection concept of a nuclear power plant, which involves the implementation of engineering tools (protective, localizing and supporting safety systems) and safety control systems to achieve the following basic safety goals aimed at preventing accidents or their limitations providing safety systems) and safety and control systems to achieve the following basic safety objectives aimed to prevent or limit the accidents:

— emergency shutdown of the reactor and maintaining it in a subcritical state;

— emergency heat removal from the reactor;

— keeping the radioactive substances within established boundaries.

The second level of the hierarchy of the requirements for the functions of the safety system involves the development of the functions of the instrumentation

and control system necessary for engineering safety tools to perform the corresponding safety functions. The set of functions of the instrumentation and control system must be necessary and sufficient to ensure the fulfillment of the required safety functions, and at the same time, it must take into account the availability of appropriate engineering means of protective, localizing and supporting safety systems.

The plant instrumentation and control systems perform functions that have different levels of importance for safety. The safety importance of each instrumentation and control function depends on its role to achieve and maintain safety, the potential consequences of a function to fail at the time it is necessary to perform it, and the likelihood of these consequences. Therefore, the initial safety analysis of a particular NPP design should be completed before the classification of control and management functions (IEC -61226).

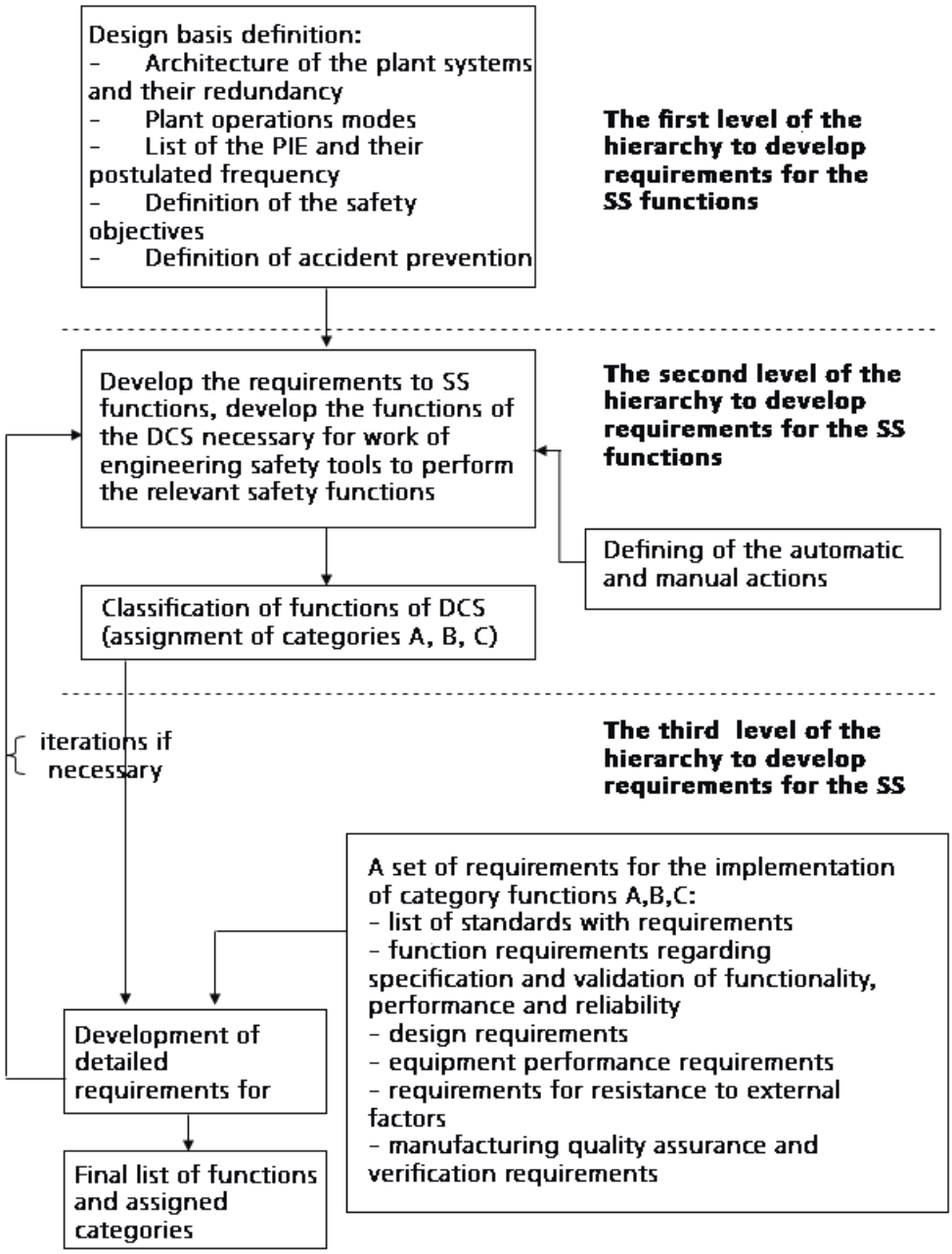


Figure 2.4.1.1 - The hierarchy of the requirements to the safety systems functions

An important role in the development of the list of functions of the instrumentation and control system is played by their classification depending on the degree of influence of the control function on plant safety. When developing the second level of the hierarchy of functions, it is important to indicate the appropriate category for each function of the instrumentation and control system, since the category will determine during further development a set of requirements in terms of reliability, design, requirements for qualifications, verification and validation, which this function of instrumentation and control system must satisfy.

From the point of view of the impact on safety, the functions of the control system are divided into three categories — A, B and C.

The presented three categories of functions A, B, and C are implemented on physically different equipment. This separation provides different levels of protection and is the basic principle of defense in depth.

The classification of individual functions of the instrumentation and control system takes into account their effect on the safety of the plant, as well as the consequences of a postulated violation of the function of the control system.

Category A is used to denote those control functions that realize emergency shutdown of the reactor and its maintenance in a subcritical state and play the main role in achieving or maintaining the safety of nuclear power plants with the aim to prevent evolution of the maximum design basis accident to unacceptable consequences.

These functions are required to stabilize the transient, i.e. bringing the station into a controlled state, in which the reactor is in a subcritical state, heat removal is provided, and radioactive emissions are limited.

Category A also involves functions whose failure can directly lead to emergency conditions that can cause unacceptable consequences if emergency conditions are not mitigated by other functions of category A.

Category A features are subject to enhanced reliability requirements.

Category B includes functions that play an additional role in relation to the functions of category A in achieving or maintaining the safety of nuclear power plants, in particular the functions necessary for operation after reaching a controlled state in order to prevent evolution of the design-basis events up to unacceptable consequences or to mitigate the consequences of the design-basis events .

The performance of functions of category B may eliminate the need to perform the functions of category A. (for example, warning or preventive protection).

Category B functions can improve or complement the performance of Category A functions in the mitigation process of the substation in such a way that damage to the plant or equipment or the release of activity can be eliminated or minimized.

Category B also involves functions which failure can cause design-basis events or increase its severity. Since category B functions are designed to complement or improve the operation of category functions, the safety requirements for a category B function may not be as stringent as for a category A function.

The third level of the hierarchy of the formation of requirements for safety functions involves a detailed development of the functions of the instrumentation and control system, the structure of the algorithms with a complete description of the measuring channels and signalization.

At an early stage of the design process, it is impossible to determine in detail all the functions, since the characteristics of the power plant are not yet fully defined. Consequently, the process of identifying and assigning categories to functions can be iterative and continue throughout the design phase (IEC 62269).

In accordance with the above approaches, in the process of safety analysis and development of the detail design of NPP-2006, the final classification list was formed in the form of "Terms of Reference for the design of safety control systems, based on TELEPERM XS.

Task description of the level 1-3 to implement safety functions».

The objectives of the safety functions, the list of functions and their interface with the classification of the instrumentation and safety by design categories formed in the above terms of reference for the AES-2006 project are shown in the Figure 2.4.1.2.

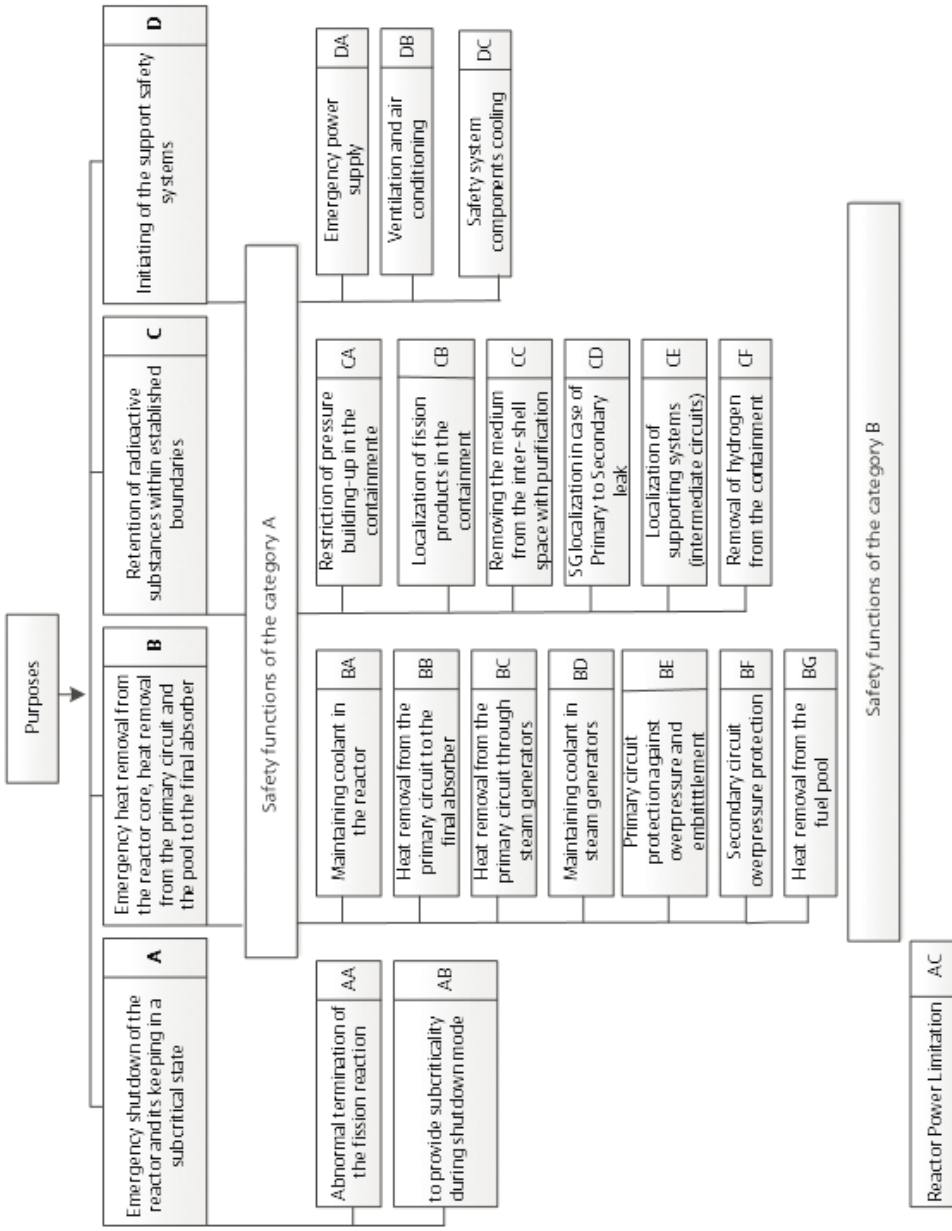

Purposes
A
Emergency shutdown of the reactor and its keeping in a subcritical state
B
Emergency heat removal from the reactor core, heat removal from the primary circuit and the pool to the final absorber
C
Retention of radioactive substances within established boundaries
D
Initiating of the support safety systems
Safety functions of the category A
AA
Abnormal termination of the fission reaction
AB
to provide subcriticality during shutdown mode
BA
Maintaining coolant in the reactor
BB
Heat removal from the primary circuit to the final absorber
BC
Heat removal from the primary circuit through steam generators
BD
Maintaining coolant in steam generators
BE
Primary circuit protection against overpressure and embrittlement
BF
Secondary circuit overpressure protection
BG
Heat removal from the fuel pool
CA
Restriction of pressure building-up in the containmente
CB
Localization of fission products in the containment
CC
Removing the medium from the inter-shell space with purification
CD
SG localization in case of Primary to Secondary leak
CE
Localization of supporting systems (intermediate circuits)
CF
Removal of hydrogen from the containment
DA
Emergency power supply
DB
Ventilation and air conditioning
DC
Safety system components cooling
Safety functions of the category B
AC
Reactor Power Limitation

figure 2.4.1.2

2.4.1.2 Control and protection system

Purpose of the control and protection system

The control and protection system is part of the automatic control system of the power unit and is designed to control the parameters necessary for the protection and control of the reactivity and power of the reactor installation, as well as controlling the power and reactivity of the reactor, including planned and emergency shutdown, maintaining it in a subcritical state.

The CPS fulfills its functions in collaboration with the main controllers of the reactor installation and boron regulation system.

In the modes of the fuel loading / reloading, due to the fact that the drives of the control rods are disconnected from the control rods, the impact on the reactivity of the CPS is excluded. In these modes, the CPS provides only control of the neutron-physics parameters by refueling monitoring system.

Safety assurance principles

When designing the system, the following safety assurance principles were taken into account:

— principle of single failure;
— principle of redundancy;
— principle of diversity;
— principle of independence;
— principle of safe failure;
— principle of actuation

Single failure

The principle of a single failure is the principle according to which the system must perform the specified functions for any initial event requiring its operation and for the failure of one of the elements of this element, which is taken into account in the design of the power plant, independent from the initial event in case of failure of one component of this system (NP-001-2015/1/).

The principle of a single failure is ensured through the implementation of the principles of redundancy, diversity and independence in such a way that any single failure in the safety control system does not violate the performance of the safety control system.

Redundancy

The principle of redundancy is the principle of increasing reliability by applying several identical or unequal elements (channels, systems) in such a way that each

of them can perform the required function regardless of the state, including failure, of other components (channels, systems) designed to perform this function (NP-001-2015 /1/).

To implement the principle of redundancy, the initiating part of the emergency protection system consists of two sets, each set consists of three redundant channels in the initiating part. Each redundant channel of each set has its own primary transmitters and power. In this case, the analog signal from the primary converter is propagated in the device for galvanic separation of signals (UGRS) and then separately goes to the same channels of the subsystem RTS-ESFAS and warning protection.

Diversity

The principle to improve reliability by applying two or more systems or elements to perform a single safety function having different designs or operating principles, with the goal of reducing the probability of common-cause failure (NP-001-2015/1/).

The principle of diversity was implemented in the AES-2006 project when designing the entire complex of safety systems as a whole. The maximum efficiency is achieved by applying functional diversity. This provides for the implementation of each given safety function, the widespread use of mutually redundant elements (channels or systems), based on various (passive or active) principles of action and / or on the use of active systems with different active elements in the technological part of the safety system (e.g. active system of emergency and planned cooling of the primary circuit and cooling the spent fuel pool JNA and number of the passive systems: system JNG50-80, and hydraulic accumulators Stage IIJNG10-40 and passive residual heat removal system JNB50-80, operating in the cooldown mode).

Diversity as a way to reduce the probability of common-cause failure in instrumentation and control system of SCSPP of the AES-2006.

Diversity is specifically created as protection against common cause failure (CCF). It can be achieved by the presence of systems that are physically different from one another, or by using functional diversity, if similar systems achieve their goals in various ways (GOST R IEC 60880-2011 «Nuclear power plants. Monitoring and control systems significant to safety. Software for computer systems performing category functions A»/3/).

Review of the requirements stipulated by NUREG/CR-7007 («Diversity Strategies for Nuclear Power Plant Instrumentation and Control Systems»), taking into account the design experience of Control Safety SystemCSS NPP, it is possible to identify six basic directions for implementing the principle of diversity :

— functional diversity;

— signal diversity;

— design diversity:

- architectural diversity;
- algorithmic diversity;

— human diversity;

— equipment diversity;

— software diversity.

To implement the principle of diversity, as one of the ways to prevent CCF (including in software), there is a diverse protection system (DSS) independent of the AZ-USBT system,), implemented on non-programmable hardware different from the SCSP. This system enables the power unit to be transferred to a controlled and safe state for various design initiating events. The decision on the application of diversity (diversification) was made taking into account the provisions [4].

«The design features to prevent CCF are related to the architecture of the instrumentation and control systems, which includes at least two instrumentation and control systems that perform category functions A. Proof that any individual monitoring and control system does not have errors is impossible, and therefore the existence of hidden defects and associated triggering mechanisms cannot be ruled out in principle.»

Independence

The principle of independence is the principle of increasing reliability by applying functional and (or) physical separation of channels (components), in which failure of one channel (component) does not lead to failure of another channel (component) NP-001-15/1/.

To exclude dependent failures, as well as to exclude the influence of any types of work performed on the equipment of one of the channels (repair, maintenance), on other channels, equipment of individual channels of the system outside the containment located in different, physically separated rooms, and inside the sealed enclosure, pipelines, cable routes of the system channels are spatially spaced. Channels are completely independent of each other, independent technological parts, control systems, supporting systems, locations of equipment, pipelines, cables, control elements.

The initiating part of the reactor scram system is located in rooms separated from the premises of the executive part of the reactor scram system.

The equipment of the initiating part of the emergency protection system, which ensures the implementation of the emergency protection function and the equipment, which ensures the implementation of the preventive protection functions, are implemented in different cabinet (since they belong to different categories A and B, respectively). Thus, the failure of equipment providing the implementation of the functions of preventive protection does not affect the function of emergency protection.

Safe failure

The safe failure principle is the principle according to which, when a system or component fails, the nuclear power plant (Power Unit) enters a safe state without the need to initiate any action through the safety controlling system /1/.

In the initiating part of the emergency protection-ESFAS subsystem in the emergency protection part, the principle of safe failure is laid down, consisting in the following failures in the signal generation path in the emergency protection channel:

— loss of power supply;

— loss (invalidation) of signals from NFMS channels ;

— loss (invalidation) of signals from process control channels;

— malfunction in TXS equipment (output modules, processing processor);

bring to generation of a signal equivalent to the emergency protection signal only by this channel.

Actuation

To minimize the influence of the human factor, an automatic inclusion of the system into operation is realized, which does not require operator actions. The system is automatically actuated at:

— power supply loss;

— Primary leaks.

In addition, manual actuation of the emergency protection by the operator is possible.

The composition of the control and protection system

The control and protection system is comprised of:

— two three-train sets of the initiating part of the emergency protection subsystem and the controlling safety system ;

— two three-train sets of the initiating part of the subsystem of the preventive reactor protection (with function of reactor power limiting device) ;

— two sets of the executive part of the subsystems of emergency and preventive protection ;

— two four-channel NFMS sets, including refueling monitoring system, physical start-up equipment and reactivity control equipment ;

— diverse protection system, comprised of two sets;

— system of group and individual control for control rods drives;

— information and diagnostic network (SHC-IDN);

— automatic power controller.

The general structure of one train of the initiating control safety system of the CPS is presented on the figure 2.4.1.3.

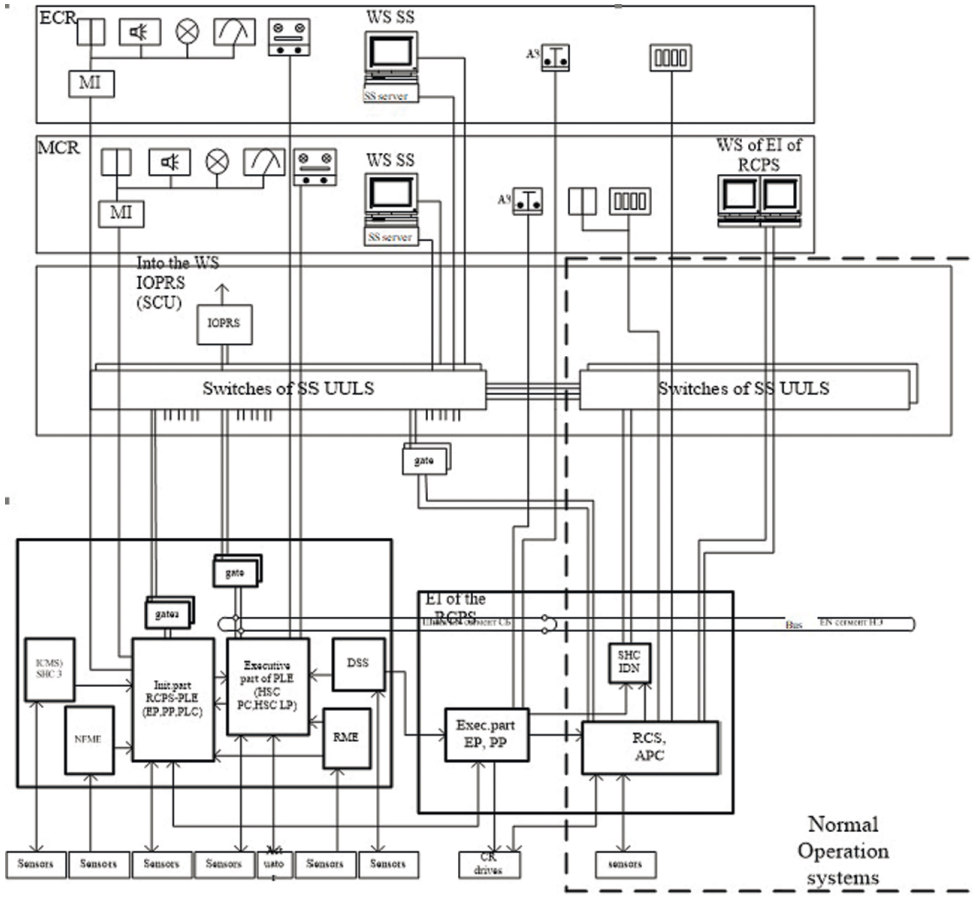


Figure 2.4.1.3 - The general structure of one train of the initiating control safety system of the CPS

Description of the safety controlling system

Each subsystem of the emergency protection-ESFAS and preventive protection comprises:

— initiating part;

— executive part.

Initiating parts of the subsystem of the emergency protection-ESFAS and preventive protection are designed to :

— monitor neutron-physics and technological parameters, calculating complex parameters by which the reactor is protected and launching process safety equipment, and comparing them with the corresponding setpoints;

— process discrete signals for reaching the controlled parameters of the corresponding setpoints in accordance with the algorithms and generation of protection signals to the executive part of the emergency protection and preventive protection, priority logic equipment ;

— generate signals to other control subsystems and information into UULS.

The initiating parts of the subsystems of emergency and preventive protection, as well as the initiating parts to actuate process safety-related systems, are implemented in the form of two independent three-train sets located in two rooms of the safety controlling system.

Both sets of emergency protection are implemented on the basis of hardware and software TELEPERM XS.

To minimize the probability of CCF due to errors in the application software by providing different signal paths with the same initiating event, the following types of diversity are used:

— functional (parametric) diversity within each set, namely, the presence of at least two triggering criteria for emergency protection and priority logic equipment for each initial event;

— a variety of algorithmic implementations of the same safety functions (different signal paths) in different sets;

— usage as a condition for the implementation of the emergency protection manual action of the operator entering directly into the executive part of the emergency protection ;

— usage of mechanical contactors produced by different manufacturers in the cabinets of power breakers to minimize the probable failure of the executive part of the emergency protection due to CCF.

To implement the principle of diversity in the function of residual heat removal (USBT function), various technical means are used to initiate protection commands. FPGA (hard-wired programming of the assigned logic functions) is used to generate the signals by Priority Logic Equipment to actuate Passive Heat Removal System. This instrumentation is implemented in a separate cabinet and completely isolated from the equipment for collection and processing of signals of emergency protection-ESFAS, implemented on the basis of software and hardware TXS.

There are four cabinets TXS MO in each set of the initiating part of the Priority Logic Equipment (ESFAS). Two main and two in hot reserve. The main and backup cabinets receive the same information for voting of the same discrete signals in accordance with the logic two out of the three. Voting results are compared according to the logic “OR” and are received in priority control devices for each criterion to actuate safety systems. Thus, a single failure in one TXS MO cabinet will not lead to a failure to generate a protection command to actuate Priority

Logic Equipment subsystem .

In each pair (primary-backup), a separate part of the total number of signals for starting safety systems is voted. These signals are distributed over each pair of one set so that different signals corresponding to the same initial event fall into different pairs. Thus, the failure of one pair will lead to the loss of only part of the signals to automatically start the safety systems in this set. The received commands to actuate safety systems from the first set go to the first safety train of the executive part of the Priority Logic Equipment, and from the second to the second.

Each of the sets of the initiating part of the EP generates emergency protection signals in the corresponding set of the executive part of the EP. Together with the set of the executive part, the set of the initiating part is able to provide the emergency protection function independently of the other set.

Preventive protection instrumentation is also implemented on the basis of TELEPERM XS software and hardware. Two sets of preventive protection subsystems are identical and implement the same initiation criteria.

The equipment of the initiating part of the EP, which ensures the implementation of the emergency protection function and the equipment, which ensures the implementation of the preventive protection functions, are implemented in different cabinets, since the functions of the EP and preventive protection belong to different categories (Category A and B respectively). Failure of equipment providing the implementation of the preventive protection functions does not affect the operation of equipment performing emergency protection functions.

The preventive protection signals generated in each set also go to the corresponding set of the executive part of the EP-preventive protection separately for each type of protection for further majority processing and the formation of commands to control the control rod drives and rod control system, where, depending on the type of protection, the corresponding algorithm and priority of control of the groups of the control rods are provided.

The option to manually actuate preventive protection and Accelerated preventive protection is provided by a command from the corresponding buttons on the operator panel in eth MCR. The commands from the buttons are sent directly to the rod control system, which generates an appropriate command to control rod drives.

There is a possibility to generate manually emergency protection commands by the operator from the console in the MCR or ECR. To do this, two EP buttons are provided in the MCR or stand-by shutdown panel. The EP signals from the buttons go directly to the control cabinets of the corresponding sets of the executive part of the EP. Circuit solutions exclude of false EP signals to be generated in case of single failures in the signals generation chains.

Setting the setpoints in the subsystems EP-ESFAS and preventive protection is provided promptly (for neutron power of the reactor) and impromptly (for other parameters). In the first case, the operator on the control panel in the MCR,

channel-by-channel, using the setpoint unit which is a part of the NFMS, sets the EP setpoint value required by the procedure. For other parameters within the limits of each complete set of initiating parts of subsystems of the EP, Engineered safety feature actuation system and preventive alarm values can be made only channel-by-channel. Authorization for this operation is provided by the commands from keys " variation of a mode of operation of appropriating complete set TXS EP-ESFAS» (« variations of a mode of operation of appropriating complete set TXS of the preventive signalization » for subsystem of the preventive signalization), placed on safety panels of the dedicated channels of the control safety system. The voltage of 24V at the same time enters the cabinets of the subsystems EP-ESFAS or preventive signalization. In the cabinets there are special keys “PARAMETRIZATION”, providing 24V voltage switching to the corresponding service inputs of information collection and processing processors. The possibility of transferring to the “PARAMETRIZATION” mode only one channel is provided programmatically in the service workstation. The operating personnel, by calling up the corresponding function diagrams, can change the settings. The structure of the technological safety means launch subsystem as part of EP -ESFAS solves the following problems:

— remote control of technological safety systems and control of parameters of emergency monitoring and normal operation;

— technological protections and interlocks, the initiation of a control ban from the automatic power controller in case of accident;

— priority management;

— to send information and receive commands from the safety panels and from the automatic power controller from the MCR (SSP) in terms of systems combining the functions of normal operation and safety functions.

The initiating part of the control safety system is structurally combined with emergency protection and implemented on the TXS platform, and all the principles and approaches described above for this platform form and implementation of the EP are also valid for the control safety system functions. The main difference between the functions of EP and safety control system is the level of decision-making (voting) and execution of the commands. In the case of ESFAS, actuators are controlled by priority control functions through priority control modules. The purpose of this control method is the need to ensure at the level of actuators (shutoff valves, control valves, etc.) the operation of normal operation system and safety systems at the corresponding levels of the in-depth protection.

The general block diagram of the initiating and executive parts of the subsystems EP, preventive signalization, ESFAS is shown in the figure 2.4.1.4.

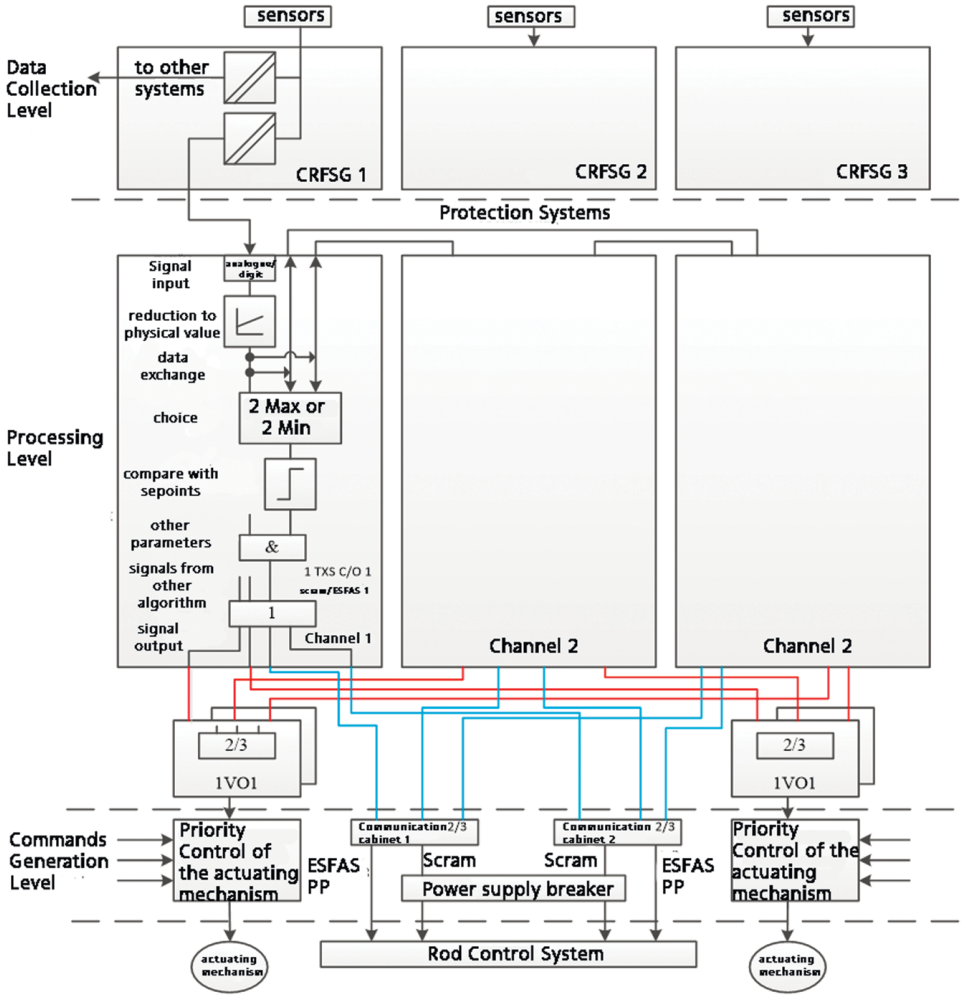


Figure 2.4.1.4 - The general block diagram of the initiating and executive parts of the subsystems EP, preventive signalization, ESFAS

The executive part of the emergency protection

To implement the emergency protection function, two sets of the executive part of the EP are provided. Each set of the executive part is associated with one set of the initiating part and is located in a separate room.

The block diagram of the executive part of the EP is shown in the figure 2.4.1.5.

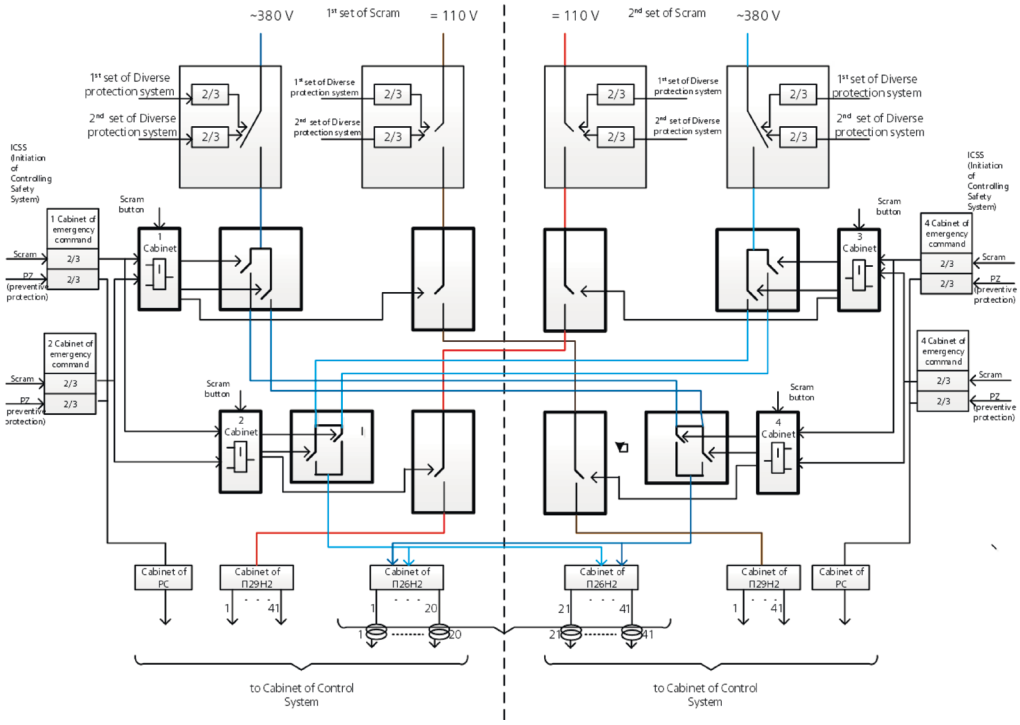


Figure 2.4.1.5 - Flow diagram of the executive part of the EP

Each set of the executive part of the EP consists of:

— two cabinets of the emergency commands performing majority processing according to the “two out of three” logic of the generalized EP signals coming from the corresponding set of the initiating part of the EP, and the formation of executive commands for actuation;

— two power interrupters for drives for the control rods, each of which is made in the form of ShP6 cabinets with disconnecting contactors, and control cabinets that control the contactors of power supply cabinets ShP6. One of them turns off the AC voltage, the second DC voltage;

— two power interrupters of drives of regulatory bodies controlled from a diverse protection system. One of them turns off the AC voltage, the second DC voltage. Each of them, unlike current breakers controlled by EP commands, has two opening electromagnets. One electromagnet is controlled from the first set of the diverse protection system, the second electromagnet is controlled from the second set of diverse protection system.

Thus, the executive part of the EP-PP provides:

— implementation of emergency protection operation upon receipt of at least two of the three generalized EP signals from any set of the initiating part of the EP-PP, any set of the diverse protection system or manual controls in the MCR or shutdown control panel by removing the power supply of alternating and direct current from drives of the CPS;

— implementation of the operation of preventive protection upon receipt of at least two out of the three PP signals (accelerated preventive protection, PZ-1 and PZ-2) from any set of the initiating part of the EP-PP or from manual controls in the MCR.

In addition, the EP generates a signal to turn off the thyristors of the power control devices in the rod control system. Since the speed of thyristors is higher than the speed of opening the drive power circuit by mechanical power contactors, the opening of the power contactors of the drive power supply occurs without a power load.

In case of a power interruption or when the voltage drops to 0,8 U nom on two AC inputs, the entire load switches to a 110 V DC network from the battery. The battery is rated for 3 s. At the same time, an emergency protection initiating signal is generated in the equipment of the initiating part of the EP by the loss of AC power supply via two inputs simultaneously..

Executive part of the Priority Logic Equipment

SB-1.2 executive mechanisms are controlled by SB-1.2 automatic and remote commands, operator's commands from the MCR (ECR) and normal operation system in all operating modes of the power unit provided for by the project through the software and hardware complex. The interaction scheme of the initiating part of the Priority Logic Equipment with actuators is shown in the figures 2.4.1.3-2.4.1.4.

Functions of the executive part of the Priority Logic Equipment during normal operation:

— continuous monitoring of the working ability of the technical means of the system and periodic, with the participation of personnel, checks of its functioning;

— control over the mechanisms of the safety systems and the presentation of information to the operational personnel;

— implements local technological protections and interlocks for normal operation, systems combining the functions of normal operation and safety, and issues control actions on mechanisms;

— equipment control and management;

— automatic control;

— remote control, if needed;

— receiving a tasks from the operator of the operating mode of redundant units and automatic regulators.

In the event of emergencies by the signals of the parameters of the initial events, the executive part of the Priority Logic Equipment operates for its intended purpose and implements the following functions:

— on the initiating signals received from the CPS-Priority Logic Equipment, initiates a control action on the actuators;

— DG starts-up and its step loading;

— implements the necessary priorities for automation and operator actions;

— generates signals regarding the state of mechanisms into the UULS;

— control the performance of the specified functions;

— prohibits control from the UULS and action from local protection;

— if necessary, remote control from the safety panels.

The executive part of the Priority Logic Equipment is implemented in the form of two complexes located in the premises of the Priority Logic Equipment channels.

The executive part of the Priority Logic Equipment involves:

— Hardware and software complex of the priority control for the safety equipment, consisting of cabinets with priority control modules;

— Hardware and software complex of the local protections, where local protection is implemented using TPTS-EM;

— primary measuring transducers;

— means of individual control located in the MCR and ECR;

— devices for transmitting commands and information.

Software and hardware complex of the priority control-1,2 is intended:

- to interact with systems combining safety functions with functions of normal operation, in accordance with accepted agreements;

— to transfer control actions of the initiating part of CPS-USBT to the executive mechanisms of the safety systems ensuring the priority of commands from the initiating part of CPS-USBT by remote controls from the MCR (ECR) and over the normal operation commands;

— to transmit remote control commands from the panels of the safety systems in the MCR (ECR) to the actuators of the safety systems and the monitoring of the status of the executive mechanisms of the safety systems on the panels in the MCR and ECR;

— to transfer control actions of local protections and interlocks generated in the software-hardware complex of the local protections to actuators and providing the possibility of remote control with actuators EP UULS and state control to EP UULS.

The main element of the software and hardware complex of the priority control is a priority control module. It is intended to be applied in the safety systems for individual control of one actuator (shutoff valves, electric motors, solenoid valves, control valves) on automatic and remote control commands generated by the

safety system and normal operation system.

The priority module receives control commands from the safety systems by wired communications:

— automatic commands from the initiating part of the safety systems;

— diverse protection system;

— remote commands from safety panels in the MCR, ECR .

The priority module receives control commands from normal operation systems via bus connections (via the EN bus, through the CM-E module and the cubicle I/O bus) :

— protective and automatic commands from the software-hardware complex of normal operation system ;

— remote commands from UULS.

The priority of input commands (TXS equipment, safety control panels of the MCR and SSP, software-hardware complex of normal operation system, UULS) to generate the output commands to the actuator through the switching equipment in top-down prioritization is shown in the figure 2.4.1.6.

In addition to the main function associated with providing priorities for controlling the actuator, the priority control module also performs some auxiliary functions related to testing hardware communications with TXS equipment, monitoring feedback signals from the actuator and switching equipment, generating state signals and faults of the actuator and issuing them to the control room, safety control panels in the MCR and ECR, to the TXS equipment and to the software-hardware complex of normal operation system.

Command source	Operation logic	Command type	Command target	Top-down prioritization	Comments
functional modules tester	→	remote	close/off open/on stop	1	not used by NVNPP in operation (used during precommissioning)
Priority Logic Equipment	→	automatic	close/off	2	The commands from the Priority Logic Equipment, DPS, SS panels MCR/ECR to the priority control module are transferred by the wire connections
			open/on	3	
Diverse protection system	→	automatic	close/off	4	
			open/on	5	
SS panel MCR/ECR Prohibit for normal operation and remote control MCR/ECR from Priority Logic Equipment	& →	remote	close/off open/on stop	6	
<u>Commands from the normal operation systems</u> Protections P1, P2, AY UULS SS panel MCR/ ECR Prohibit for normal operation by Priority Logic Equipment	& →	Protection P1	close/off	7	The commands from normal operation systems to the priority control module are transferred via the EN communication bus
		Protection P2	close/off	8	
		Protection P2	open/on	9	
		automatic	close/off open/on	10	
		remote	close/off open/on stop		
		remote	close/off open/on		

Figure 2.4.1.6 - Priorities of control commands received by the priority control module

In addition to the software and hardware complex of the priority control, each safety system channel has a software and hardware complex of local protection .

Software and hardware complex of local protection -1,2 is intended:

— to generate control actions of technological protections and interlocks of normal operation and sending into the Software and hardware complex of the priority control-1,2 to preserve the safety systems equipment in case of violations in its operation in the absence of initial accident events;

— to remote control of actuators according to commands transmitted via the EN communication bus with EP UULS;

— to implement automatic regulation functions related to the SSNE of the WB and organization of their control from the SB panels, including signaling of the status;

— monitoring and control functions of technological systems as part of SS-1,2 of the Automatic Process Control System of the Power Unit.

Software and hardware complex of the priority control and software and hardware complex of local protection provide information via the EN communication bus proactively to the UULS for archiving with the assignment of a time stamp of the value of the technological parameters, the fact of the transition of the value of the technological parameter through the setting, signs of inaccuracy of analog signals, the fact of the operation of protections and interlocks, remote operator commands, diagnostic information.

Software and hardware complex of the priority control and software and hardware complex of local protection exchanges information with the normal operation system via a communication bus EN.

Local protection and blocking of safety system mechanisms are implemented in separate cabinets (software and hardware complex of local protection). In case of technological accidents, protection data, as well as commands from the UULS, are automatically blocked by commands from the initiating part of the Priority Logic Equipment.

Software and hardware systems for priority management and local protection are implemented at TPTS –EM.

2.4.2 DIVERSE PROTECTION SYSTEM

The design of APCS NPP-2006 (ASU TP-2006) includes using programmable hardware as in safety control systems (SCS) and in normal operating systems (NOS). The information in the main control room (MCR) is presented with the unified software and hardware of the digital control system (DCS). These means have many advantages as compared to the analog technique, including self-diagnostic testing, reliability, accuracy increase, accessibility in operation, which facilitate the work of MCR operation personnel. Centralization of control and management is carried out from workstation monitors in the MCR and the emergency control room (ECR). Notably, control redundancy from the ECR is carried out in full for SCS and in a minimum sufficient volume for NOS.

However, along with the advantages of using programmable hardware, the severity of the consequences increases in case of an automatic process control system failure for a common reason as a result of an error in the software (SW).

2.4.2.1 Justification of the need to implement the Diverse Protection Systems (DPS)

APCS NPP-2006 (ASU TP-2006), engineered safety feature actuation system (ESFA system), emergency and preventive protection are implemented with TELEPERM XS (TXS) programmable hardware.

When considering the impact of failures on the functioning of the system, the consequences of software failures should be also analyzed.

By a program error (fault) is meant any fact of program non-completion of any functions assigned to it.

Software defects are systematic, and not random, therefore the principle of a single failure cannot be applied to the system software as it is applied to hardware [3-4].

It should be recognized that absolutely reliable programs do not exist, since the absolute degree of reliability cannot be theoretically proved

Currently, there is no recognized method of assessing the probability or frequency of failures appearing due to software defects.

Therefore, when calculating the reliability indicators for two sets of programmable ESFA system tools, it is postulated and, in accordance with the recommendations of the Scientific and Engineering Centre for Nuclear & Radiation Safety, a probability of failure is counted as 10^{-5}; as a result the probability < $5*10^{-7}$ (according to the technical requirement for safety control systems – ESFA system) was not achieved.

The possibility of a common cause failure (CCF) due to the software (SW) should be already considered in the design. If the postulated conditions for the occurrence

of a CCF **can be predicted**, then to prevent the CCF due to the use of SW, it may be necessary to change the design and **use special protection tools, including software diversity** [4].

If the required reliability exceeds what is actually achievable with the use of software, then additional independent systems are used, each of them is capable of performing the assigned security function.

Russian and international regulatory documents clearly confirm that reliable protection against a common case failure (CCF) due to the possible SW errors can only be implemented based on the principle of diversity.

To decrease the possibility of CCF it is recommended to apply the combination of programmable and non-programmable devices (field programmable gate array — FPGA).

Diversity as means to decrease the possibility of CCF

Diversity is specially designed to protect against CCF. It can be achieved by the presence of systems that are physically different from one another, or by using functional diversity, if similar systems achieve their goals in different ways /3/.

The analysis of the requirements of NUREG/CR-7007 «Diversity Strategies for Nuclear Power Plant Instrumentation and Control Systems», taking into consideration the experience of designing NPP SCS, allows to identify six basic areas of implementation of the principle of diversity:

— functional diversity;
— parametric diversity;
— design diversity:
  • architectural diversity,
  • algorithmic diversity,
— diversity, determined by the human factor;
— hardware diversity;
— software diversity.

Despite the fact that the SW development of the ESFA system is carried out according to the requirements of the IEC standard, 60880 convincing evidence of the impossibility of failure or the assessment of reliability indicators of software there were not presented by the system developer.

According to GOST R ISO/IEC 9126-93/5/, the evaluation of software products, software reliability is a set of attributes related to the ability of software to maintain its level of quality of functioning under specified conditions for a specified period of time.

The U.S. Nuclear Regulatory Commission issued an explanation in which CCF software is defined as the result of multiple failures that were not detected during inspections, and which should be attributed to an beyond-design accident (ODA).

To implement the principle of diversity, as one of the methods for CCF failover,

it is necessary to have a diversion protection system (DPS) independent from the Reactor safety ESFA system, implemented with other technical means. This system should ensure that the power unit is in a controlled and safe state for various design initiating events.

The decision on the application of diversity (diversification) was made taking into account the regulations ("The measures of the project for the prevention of failure for a common reason are related to the architecture of control and management systems, which includes at least two control and management systems that perform category A. To prove that any individual control and management system has no errors, is impossible, and therefore the existence of hidden defects and associated triggering mechanisms cannot be excluded in principle").

In this respect CCF software is postulated as an beyond-design accident and it is decided to create a failover by implementing a diverse protection system (DPS) in the design.

DPS in the event of a failure of the initiating part of the Reactor safety ESFA system or the executive part of the reactor safety system for a common reason should ensure the fulfillment of the following basic safety functions upon the occurrence of design initiating events of category 2, 3, 4 modes:

— emergency shutdown of the reactor and maintaining it in a subcritical state;

— emergency reactor heat removal;

— retention of radioactive substances within established boundaries.

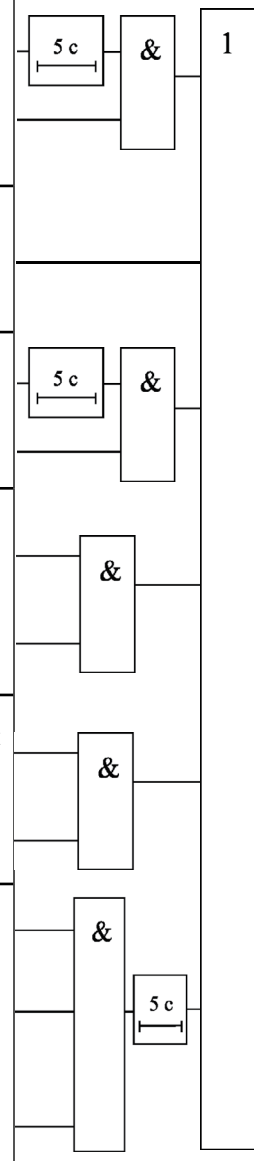
Here we consider the implementation of compensatory measures to address the comments of the expert report by performing some functions with an independent system of protection.

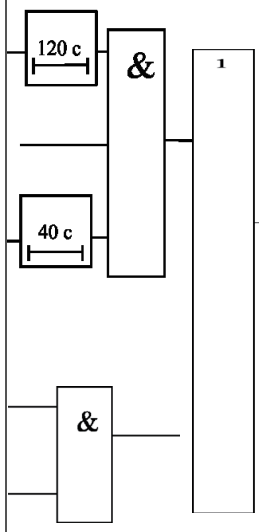
Determination of the volume of compensatory measures

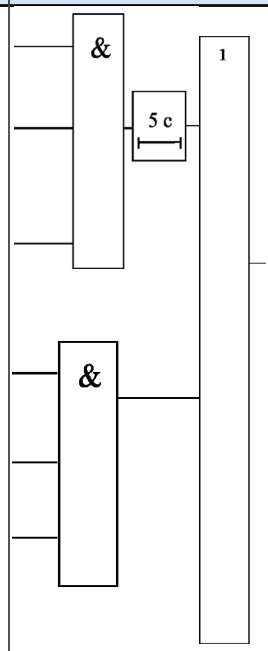
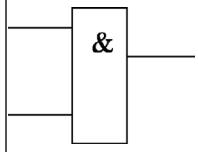
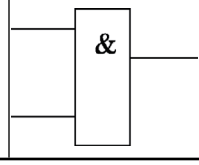
To determine the scope of compensating measures, the organizations of the Chief Designer of reactor installation and the scientific supervisor conducted an analysis of the initiating events with the application of CCF, leading to the BDA with the most serious consequences, considering fulfilling the safety functions and not meeting the acceptance criteria for off-design accidents without fuel melting. The necessary actuators are determined, the operation of which must be initiated by additional signals. Algorithms for diverse protection have been developed to perform all safety functions that force on the reactor shutdown and implement the functions of the ESFA system. Verification calculations of the consequences of the development of the initial event with CCF have been performed taking into account the work of additional DPS algorithms and personnel actions.

Protection algorithms, implemented with DPS, are presented in Table 2.4.2.1.

Table 2.4.2.1 – Algorithms of organizing of diverse protection against de-energizing of the CPS actuating device

Name of parameter	Actuation setpoint	Logic	Protective action
1 The pressure over the reactor core, more than The temperature in one of the hot legs of the loop, more than	18,01 MPa 260 °C		The de-energizing of the actuators of the CPS absorber rods
2 Neutron flux power, more than	110 % N_{nom}		
3 The pressure over the reactor core, less than Neutron flux power, more than	13,4 MPa 5 % N_{nom}		
4 The level in the pressurizer, less than The temperature in one of the hot legs of the loop, more than	3,6 m 150 °C		
5 The water level in the steam-generator, less than The temperature in one of the hot legs of the loop, more than	(normal height – 0,7) m 260 °C		
6 The pressure in the steam-generator, less than The temperature in one of the hot legs of the loop, more than The saturation temperature difference of the 1st and 2nd circuits of the steam generator	5,04 MPa 200 °C 70 °C		

Name of parameter	Actuation setpoint	Logic	Protective action
7 The pressure in the steam-generator, less than The temperature in one of the hot legs of the loop, more than Flowrate at the condensate return line of the steam generator automatic control system, less than 8 The water level in the steam-generator, less than The temperature in one of the hot legs of the loop, more than	8,0 MPa 150 °C 20 m^3/h (normal height –0,975)m 150 °C		The deenergizing of the electromagnets, which keep closed the air seals of the corresponding PHRS (Passive Heat Removal System) channels
9 Boiling margin (the difference between the saturation temperature and the temperature of any of the four hot legs of the loops), less than	3 °C		The deenergizing of the electromagnets, which keep closed the air locks, and opening the regulating devices of all PHRS channels to the full cross section. Disabling all the MCPs

Name of parameter	Actuation setpoint	Logic	Protective action
10 The pressure in the steam-generator, less than The temperature in one of the hot legs of the loop, more than The saturation temperature difference of the 1st and 2nd circuits of the steam generator, more than 11 The water in the steam-generator, more than The temperature in one of the hot legs of the loop, more than The level in the pressurizer, less than	5,04 MPa 200 °C 70 °C (normal Height + 0,5) m 150 °C 3,6 m		Closing the MSIV of the corresponding Steam Generator. Closing of the motor-operated valves: in the steam lines of the corresponding steam generators; before fast-acting atmospheric reducing station; in the line of the feed water and of the blowdown of the corresponding steam generator. Disabling MCP of the corresponding loop
12 The water level in the steam-generator, less than The temperature in one of the hot legs of the loop, more than	(H_{normal} – 0,975) m 150 °C		Closing MSIV of the corresponding steam generator Closing of the motor-operated valves in the steam lines of the corresponding steam generators
13 The water level in the steam-generator, less than The temperature in one of the hot legs of the loop, more than	(H_{normal} – 0,7)m 150 °C		Disabling MCP of the corresponding loop

Name of parameter	Actuation setpoint	Logic	Protective action
14 The level in the tank of the emergency core cooling system, less than	1,15 m		Closing of the slam-shut valve in the pipe lines of the passive part of the reactor core cooling system (Stage I accumulators)
15 Pressure(gauge) under sealed enclosure, more	0,4 kPa		Closing isolating valves in the vent pipes
16 Pressure(gauge) under sealed enclosure, more	40 kPa		Closing isolating valves in the industrial process pipes

In total, 16 algorithms have been developed and justified, the implementation of which meets the acceptance criteria for design basis accidents without fuel melting in case of CCF of the design ESFA control and protection system.

To eliminate the risks associated with the reliability of the use of other programmable tools when fulfilling the principle of diversity, it was decided to implement DPS through non-programmable hardware.

Calculation of the probability of a failure of a requirement during the joint operation of the CPS systems — ESFA and DPS — taking into account the software failure in two sets and the superposition of the failure of sensors or signal propagation devices in one SCS channel was $4,74 * 10^{-7}$. This satisfies the requirements of the technical specifications for the CPS ESFA system ($< 5 * 10^{-7}$).

Description of diverse protection system

One identical set of diverse protection system is installed in every safety control system. Every set includes 3 cabinets of a set of automation tools (SAT) — DPS, manufactured by OOO “Moscow Plant Fizpribor”. Algorithm 7 in Table 2.4.2.2 is implemented through non-programmable tools (FPGA) – TXS.

SAT DPS is designed for:

— input and processing of analog and discrete signals from primary converters (sensors) and related systems (neutron flux monitoring equipment, MCR panels, devices for galvanic isolation of signals (DGIS), software and hardware systems of local protection (SHS LP) and TXS for issuing FCR signals and implementation of protection functions through non-programmable tools;

— presenting information on the controlled parameters and condition of parts of the complex in the MCR and in the DCS (digital control system) of the power unit using non-programmable and programmable means.

Each cabinet has one protection channel powered by two sources: direct and alternating current.

The formation of the “2 out of 3” protection logic for reactor shut down is carried out by two power interruption circuit breaker of the CPS regulating element (CPS RE) — for the direct current and two — for the alternating current.

The formation of the logic “2 of 3” of the protection of the initiating part of the ESFA system is carried out in the SAT cabinets and wired to the existing executive part of the ESFA, software and hardware complex of priority control (hereinafter SHC PC) to the priority control modules of the actuators (hereinafter PCM). One PCM controls one mechanism. To eliminate failures of PCM for a common reason, the principle of diversity is implemented inside the PCM.

Prioritization of the execution of the protection commands of the ESFA system implemented in the PCM in descending order:

— automatic control from TXS (standard operation in the absence of a failure);

— from DPS (in case of TXS failure (failure to issue commands));

— from keys of MCR, ECR;

— from control systems for normal operation.

When DPS is triggered, as well as when the TXS is triggered, there is imposed a ban on 30 minutes to complete the operator's commands.

To execute protection commands to stop the reactor, commands from TXS and DPS are of equal priority.

As analogical parameter sources for TXS and DPS the existent sensors are used. To exclude the influence of the operation of TXS and DPS, galvanic isolated signal propagation was performed through the non-programmable galvanic isolation modules available in the CPS ESFA system in the DGIS cabinets.

This decision has been made to provide a strict sequence of operation: first in TXS protection with the absence of failure, later in DPS protection because of the fact

that trip settings of TXS and of DPS differ slightly. In case of using different sensors the possibility of DPS triggering earlier than TXS triggering is not excluded, which can provoke incorrect operator's actions.

For example, a trip set point of level in a steam generator in a CPS ESFA system is H_{normal} — 650 mm and in DPS is H_{normal} — 700 mm. The difference is less than three measurements errors, which with different sensors can lead to a disruption in the sequence of system operation.

Thus the main reason for using common sensors is the need to ensure the sequence of triggering at first CPS ESFA system and then DPS system when both systems function correctly.

At the same time, in terms of the performance of the emergency protection function, the DPS and CPS-ESFA systems are of equal priority. As for the management of safety systems, priority is given to the commands of CPS-ESFA. According to the algorithms, the DPS generates commands to de-energize the CPS drives for the reactor shutdown, to localize the containment volume and put into operation the passive heat removal system (PHRS).

The implementation of protection functions with non-programmable tools eliminates the unpredictable reliability of software and the impact of information security aspects on the protection function. Reliability calculations are possible for non-programmable hardware, including taking into account FCR, and are impossible for software.

SAT DPS is based on the following innovative approaches:

— protection algorithms implementation through non-programmable tools (logic gates, counters, registers and so on) without using micro controllers and FPGA of any degree of integration;

— implementation of digital processing of analog signals by non-programmable means (scale transformation, screening, 50 Hz, damping, linearization, cold junction temperature compensation, the necessary correction of parameter values) through table-driven processing in non-volatile memory microcircuit;

— periodical automated verification (testing) of the major part of the equipment, including algorithm testing, intra-system communication lines, communication lines with the actuators through non-programmable means;

— reservation (triplication) of not only sensors, but also hardware for the implementation of protection algorithms with possibility of testing equipment without decomissioning and without loss of protection function at the power unit under operation;

— the presence of an internal redundant local area network and a redundant communication network with the information and control system of the power unit for issuing information on discrete and analog signals, system actuation, state of hardware, while the impossibility of the influence of programmable network tools on the means of non-programmable logic is provided by hardware.

Built-in verification tools provide diagnostics from sensor outputs to communication lines with actuator control modules included in ESFA.

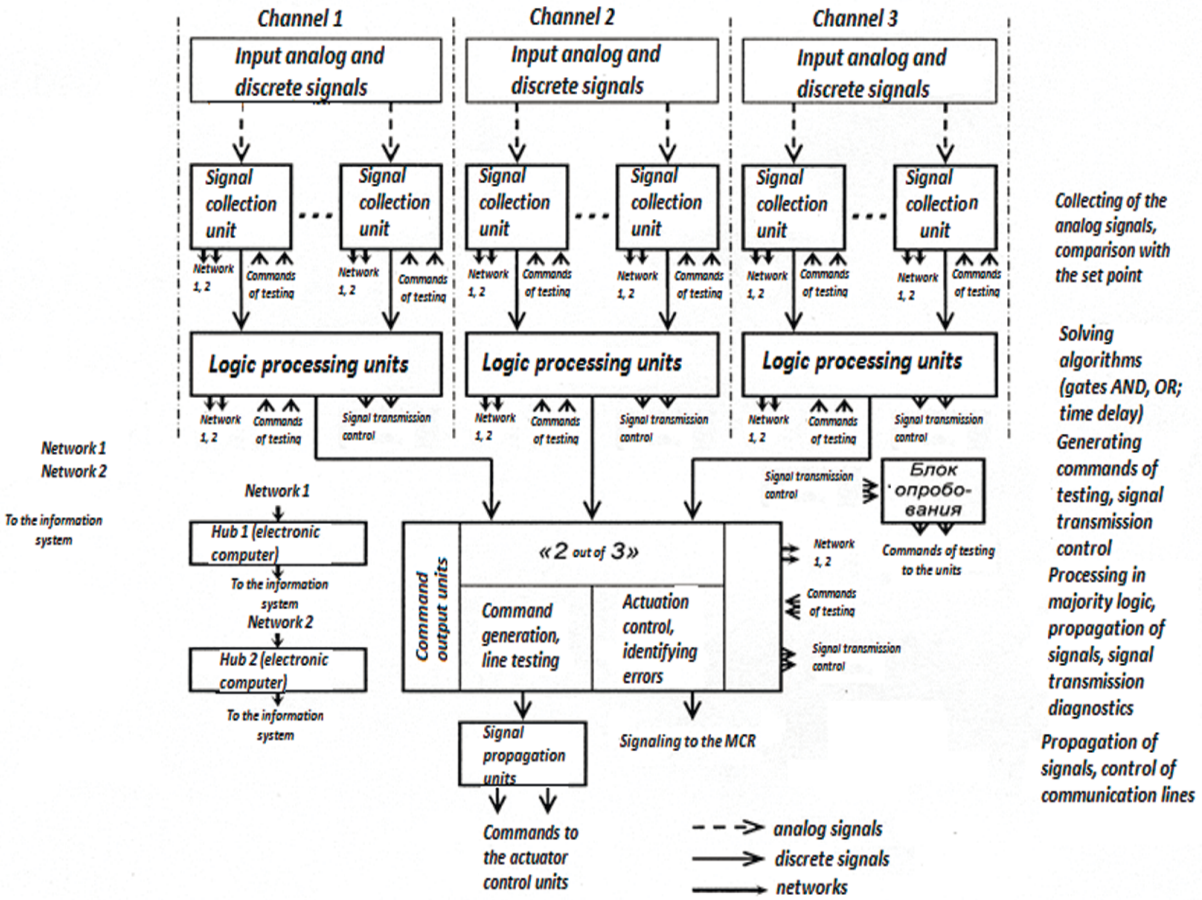


Figure 2.4.2.1- The structure of the Diverse protection system, set of automation tools (DPS SAT)

Building the systems

Structure of system (Figure 2.4.2.1) includes reservation (triplication) of not only sensors, but also hardware for the implementation of protection algorithms with possibility of testing equipment without decomissioning and without loss of protection function at the power unit under operation.

The signal collection blocks carry out the input of signals from sensors (primary transducers) and related systems.

Blocks of collecting analog signals on non-programmable tools provide:

— input of thermocouple signals, resistance thermometers, unified current signals,

— scale transformation, glitch filtering of industrial frequency (50 Hz),

— damping (with a time constant comprised between 50 millisecond - 10 sec individually for each input of the block),

— linearization of signals according to standard (or non-standard as agreed by the customer) nominal static characteristics of thermocouples and resistance thermometers (type of characteristic and signal range are set individually for each input of the block),

— compensation of the temperature of the cold junction of thermocouples, level correction;

— comparison with the set point for increasing or decreasing with the regulated return area (return area is set individually for every set point and can be of any kind within the range of input signals);

— if necessary, sensor power supply with an individual galvanic isolation of power sources. The mismatching of the analog signals is identified and the signal range is controlled through the functional unit microcontrollers and hubs (industrial computers).

Collecting discrete signals from the neutron flux monitoring equipment involves the input of “dry contact” type signals with sensor power supply and majority processing according to the “2 out of 4” logic.

Logical signal processing includes:

— majority processing according to the logic “2 of 3” (“2 of 4”),

— logic gates (AND, OR, NOT),

— time delay.

Each of analogic signals is input from three sensors of the same name into three cabinets of DPS. In the DPS cabinets (in the blocks of collecting and processing of the unified current signals) every signal is processed: glitch filtering at 50 Hz frequency, damping, correction, cold junction temperature compensation, comparison to the set point including the return area.

The results of comparison with the set point from every cabinet are spread to the other two cabinets and are processed in majority logic as “2 out of 3” (in the blocks of majority logic “2 out of 3”, logic processing and in the control blocks

of power keys) for generation of commands to de-energizing of CPS regulating element. The generated commands are sent to the power keys – dry contacts (the blocks of power keys). The process described is implemented in all the three cabinets in the same way. For the power keys the majority processing “2 of 3” is implemented.

DPS algorithm operation

The example of the DPS algorithm structure of control of the actuators in the context of steam generator protection is given in the Figure 2.4.2.2

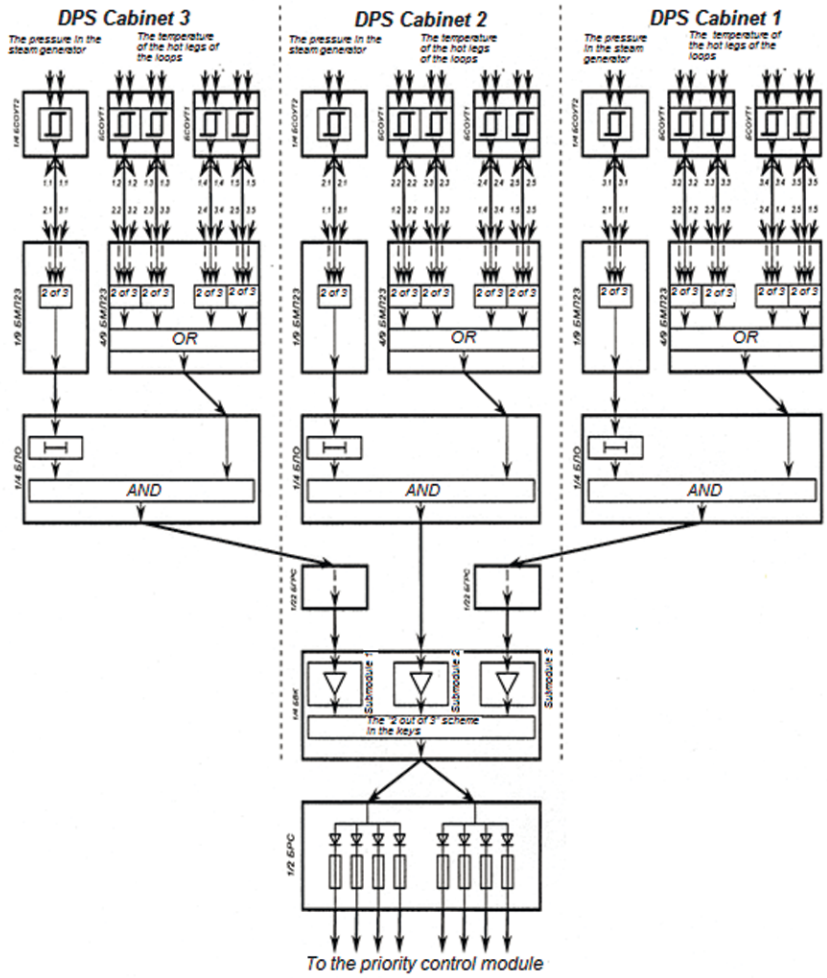


Figure 2.4.2.2 - DPS algorithm structure of control of the actuators

Each of analogic signals is input from three sensors of the same name into three cabinets of DPS. In the DPS cabinets every signal is processed: glitch filtering at 50 Hz frequency, damping, level correction, cold junction temperature compensation, comparison to the set point including the return area.

The results of comparison with the set point from every cabinet are spread to the other two cabinets and are majoritarly processed as “2 out of 3” and logic processing for generation of commands. The process described is implemented in all the three cabinets in the same way.

The generated commands are sent to the command issue units. (CIU). The command at the exit point of the CIU is formed according to the logic “2 out of 3” at the 6 key scheme (optoelectronic relay). The command is propagated to several priority control modules (PCM) by the diode propagator of the signal propagation unit (SPU). One SPU is connected with one or more PCM.

Collecting analog and discrete signals from the hardware of the complex and their delivery to digital control system DCS in two interredundant local networks is carried out by microcontrollers built into functional units.

The local networks operate under the control of two interredundant hubs - industrial computers. Hubs are installed directly in the cabinets of the complex. Two interredundant hubs are provided. Hubs implement the following software functions:

- collection and processing of signals from the DPS equipment through two interredundant local networks;

— checking the mismatch of two analog signals of the same name, checking the signal ranges in order to detect failures of the nodes of the analog-to-digital conversion of blocks and / or sensors;

— diagnostics of hardware and software of the complex;

— technological process and hardware information status transfer to the power unit information system.

The interaction of the standard and diverse control systems.

The interaction scheme of the standard and diverse control system is presented in Figure 2.4.2.3.

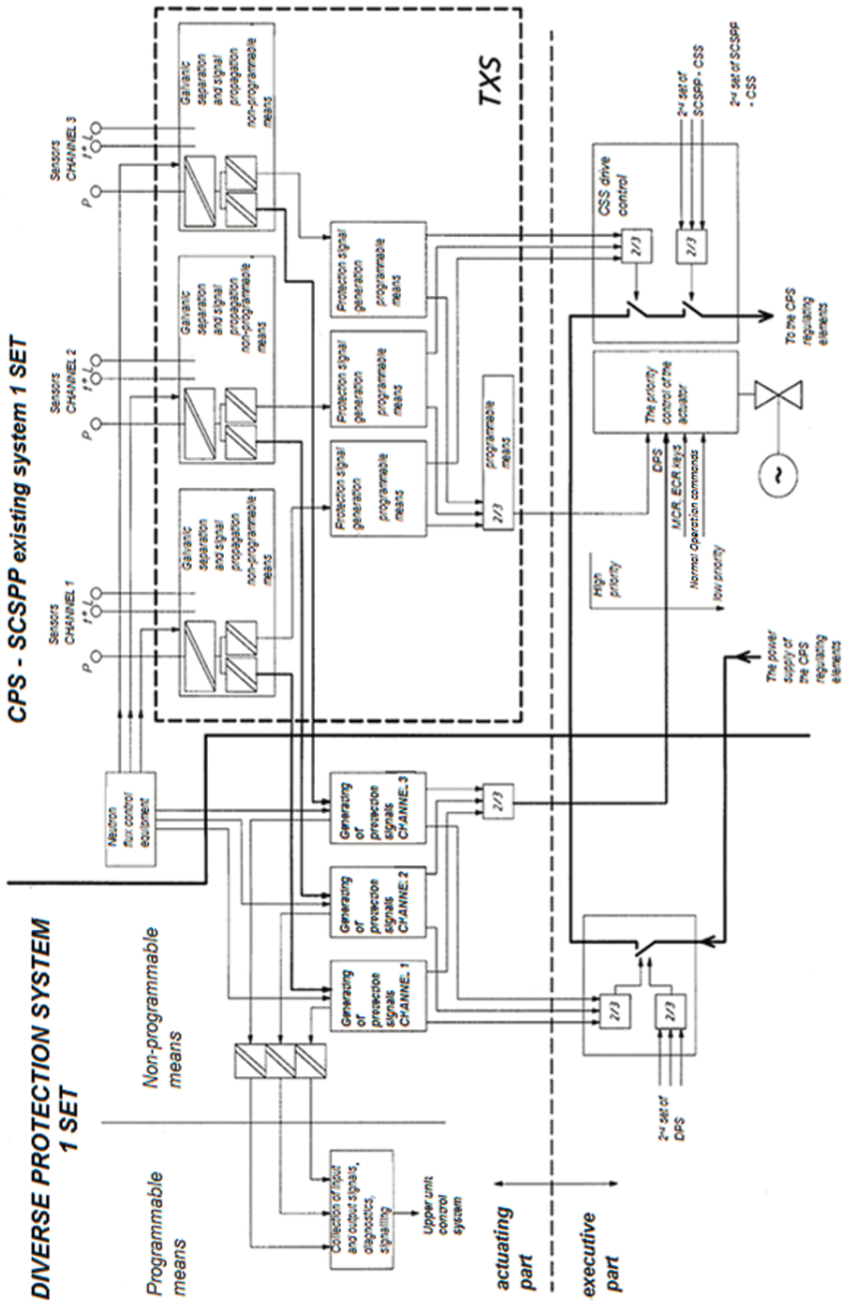


Figure 2.4.2.3 - The interaction scheme of the standard and diverse control system

As an actuator for removing voltage from the CPS regulating element the toggles with two separate solenoids are used. Each solenoid has a command from one set of DPS. The toggles controlled by the DPS are connected in series with the standard toggles. Thus, the command action on the reactor shutdown from the DPS and from the standard CPs are of the equal priority. The command formation to the disconnecting solenoid in the logic "2 out of 3" is carried out by the wireline of 6 commands of power key units (2 out of each channel) on the solenoid terminals.

The actuators are controlled by the present priority control modules. And the commands from the standard SCSPP are of higher priority. Since the set points of the standard SCSPP triggering are lower than the set points of the DPS triggering. When there is no CCF the control is carried out from the standard SCSPP (safety control system by process parameters), and if the SCSPP is a failure, the control is carried out from the DPS (diverse protection system).

The conclusions

1. The implementation of the principle of diversity through creating a DPS without using software is an effective means of overcoming the CCF (common cause failure).

2. The use of non-programmable tools in ESFA together with programmable ones allows to achieve the required provable reliability indicators and to implement all regulatory requirements for protection systems that perform the functions of safety protections.

2.4.3 MAIN CONTROL ROOM AND EMERGENCY CONTROL ROOM

To organize centralized instrumentation and control of the power unit the main and emergency control rooms are designed.

The main control room is the main place for controlling the power unit and is designed for instrumentation and control of the process equipment of the normal operating system and protection systems of all design basis conditions of the power unit operation, including accidents.

The emergency control room (ECR) is used when there is no possibility to control out of the MCR (for example, because of the fire in the MCR).

Out of the ECR there can be performed a limited set of functions:

— control of the protection systems;

— retention of the reactor in the subcritical state;

— organizing the reactor heat removal;

— monitoring the reactor state and the state of the protection systems;

— fire protection control.

The arrangement of the MCR and ECR relative to one another (Table 2.4.3.1), and special design choices make the common cause failure of the MCA and ECR impossible.

Table 2.4.3.1 – The arrangement of the MCR and the ECR

Name of the control room	Arrangement/Location		
	Building	Level	Premises
MCR	UJA	+ 3,60	UBB13R011
ECR	UJA	+7,75	UBB17R002

The design excludes the possibility of simultaneous control from the MCR and ECR. There is an "MCR/ECR switch" installed in the ECR, the circuit of which is constantly controlled – and the personnel is immediately informed if any troubles occur.

In the MCR and ECR there are video cameras of monitoring systems installed which register operator actions.

The MCR and ECR are fully equipped as for normal operation conditions and for accident situations. For this purposes, the air conditioning systems in the premises of the MCR/ECR and the life support systems of the MCR/ECR are provided.

Just next to the MCR (Figure 2.4.3.1) there is a technical support center (TSC) situated, which is designed for involving additional personnel in emergency and accident situations. If necessary, through the TSC communication with external organizations is carried out to solve diagnostic problems of a high degree of complexity. The design includes a break room UBB13R014 with all the facilities for the MCR operators, where they can consume food.

2.4.3.1. Main control room

The project conception of the NVNPP-II MCR includes as a computer-based method of control and a traditional method, based at individual means.

In normal operation conditions and in the violation of normal operating procedure the operators monitor and control the power unit equipment, using one interface – the displays of the workstations.

In accidental situation, safety systems are managed and controlled from the safety panels with the help of individual hardware, devices and indicator boards.

For the possibility of a quick overview of the equipment state and parameters in transient modes, a CPS panel and a large screen display (LSD) are provided, with the help of which all the operators, using common information, can quickly assess the situation.

For the unit shutdown (in case of an Upper Unit control System failure, more than two hours, for common case) the standby zone is provided, including panels of Reactor, Turbine and Electrical parts. From these panels the operators monitor and control the technological systems, of most importance for the safe operation of the NPP in the two-hours limited time period with the following unit shutdown, which is realized with the help of individual hardware, devices and indicator boards.

Control from the traditional means and control from the displays are of equal priority.

The MCR Layout, presented in the Figure 2.4.3.1, includes three areas:

— operational control area;

— Unit shift supervisor area;

— non-operational control area.

Operational control area is designed for instrumentation and control of the main equipment of the 1st and 2nd circuits, and of the electrical part of the power unit.

In the operational power unit control area the duty personnel work:

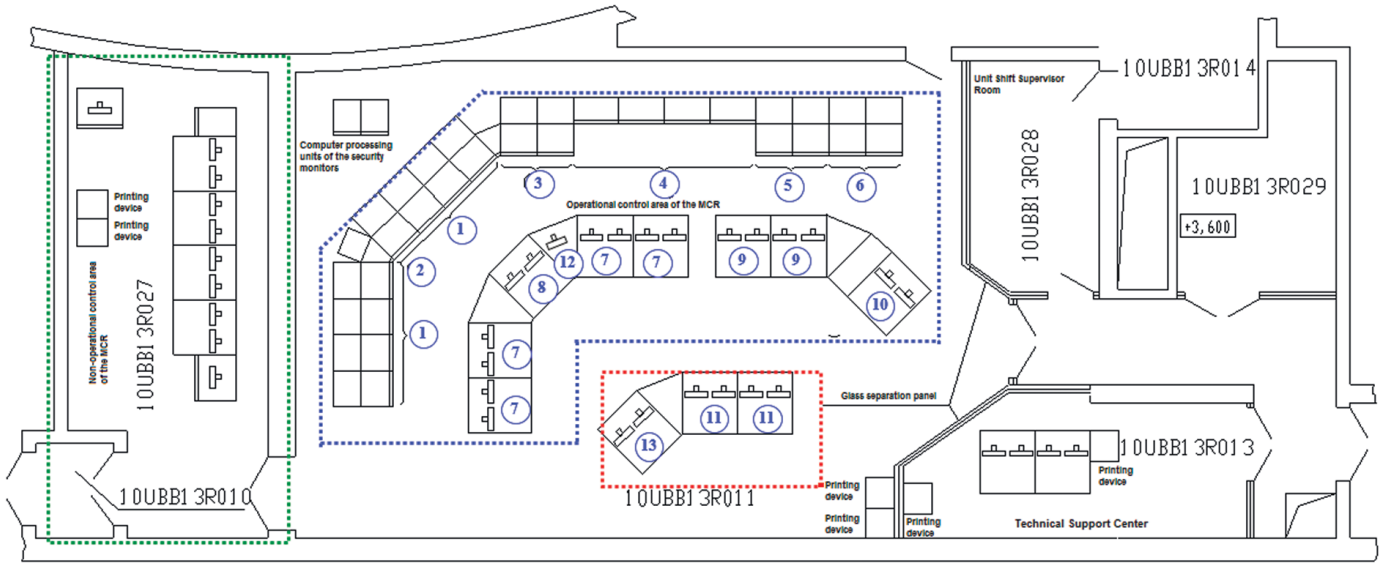
— the operators of the Reactor part (Reactor Control Leading Engineer (RCLE), Reactor Shop Shift Supervisor);

— the operators of the Turbine part (Turbine Control Leading Engineer (TCLE), Turbine Shop Shift Supervisor).

The design does not include the presence in the MCR of the regular operator - electrical part specialist for controlling the power unit electrical part equipment. The general supervision of the electrical part state is carried out by the Turbine part operator and the Unit Shift supervisor. All the switches of the electrical part are carried out by the electrical engineer, called out of the Electrical Shop shift staff, and if necessary in the accidental situations they are carried out by the Unit Shift Supervisor.

The work places of the RCLE and TCLE for controlling the systems of normal reactor and turbine operation are organized in similar way: each workplace includes dual-display work stations (WS). Each work station has one set of actuating members (track ball and a keyboard).

At the RCLE's workplace (besides 4 dual-display automated work places of the Reactor part Operator (item 7 in Figure 2.4.3.1)) there is one dual-display work station, designed for implementing the function of monitor control of Electrical equipment complex CPS and for presenting information about EEC CPS and In-Core Monitoring system (item 8 in Figure 2.4.3.1).



Operational control area

Unit Shift Supervisor Area

Non-operational control area

1. Safety system panels - 1,2
2. CPS panel
3. Panels of the Normal Operation Actuators
4. Large screen display
5. Turbine Shop panels
6.Electro technical equipment panel
7. Reactor shop operator Automated work place (AWP) (4 dual-display workstations)
8.CPS of Electrical equipment complex and In-Core Monitoring system AWP (1 dual-display workstation)
9.Turbine Shop Operator AWP (2 dual-display workstation)
10. Electrical part AWP and generator control panel
11. Unit Shift Supervisor AWP (2 dual-display workstation)
12. Reactor simulator
13. Closed Circuit TV Workstation and Fire safety instrumentation and control system

Figure 2.4.3.1 - The MCR layout

The mentioned above work station (EEC CPS AWP) can work in 3 modes:

— the first -presenting information from the EEC CPS,

— the second - presenting information from ICMS,

— the third mode – when one display presents EEC CPS and the other –ICMS.

In case of one display failure, the other one shows the combined video frame EEC CPS and ICMS with the function degradation.

The control panel add-on unit 10CWA09 is situated in front of the displays of the EEC CPS AWP and is used for arranging (Figure 2.4.3.2):

— keys of group and individual control of CPS regulating elements,

— the key “Switch on/off – automatic power controller of the reactor”;

— buttons «Emergency Protection», «Accelerated Preventive Protection», «Fire Protection-1», «Emergency Protection actuating»;

— Setpoint adjustment units of the neutron flux control equipment .

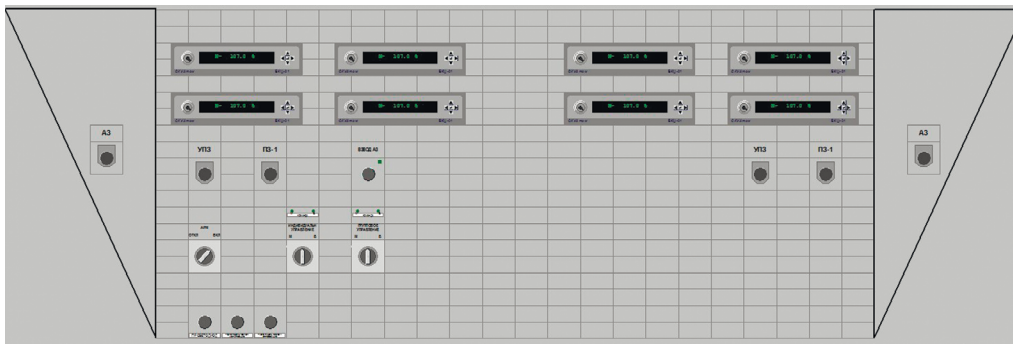


Figure 2.4.3.2 - The control panel add-on unit 10CWA09

Reactor simulator display 10CWA12 is arranged at the corner insert of the Reactor Control Leading Engineer work place (item 12 in the Figure 2.4.3.1).

The Unit Shift Supervisor work place is situated in the Unit Shift Supervisor area. The USS work place consists of two dual-display work stations, included in the upper level unit control; the workstations of the Closed Circuit TV and Fire safety instrumentation and control system; and also the communication equipment.

The closed circuit TV equipment included in the USS workplace serve for

— Observation over the technical process and over the power unit equipment state for visual control and early detection of accidental situations with videographing of their development in all cases covered by the project;

— For carrying out search and rescue efforts at the power unit;

— For personnel information support.

The communication equipment included in the USS work place is the main center for organizing communication with all the outside (controlling and administrative authorities, electric power system) and internal parties. The maximum number of direct call buttons are organized here.

At the other workplaces in the MCR there are only communication phones within the power unit site with a limited number of direct call buttons (15-20 buttons) to connect with the subordinate operational personnel.

Non-operational control area is designated for monitoring and controlling of the systems of special water treatment, of ventilation, of In-core monitoring, of access monitoring and control. This designating of the non-operational control area and organizing there separate work places allows to take away a part of workload of controlling auxiliary systems from the regular operators of the Reactor and Turbine Shops. This allows them to pay more attention to the power unit safety and security arrangements and to maintenance of the power generation technological process.

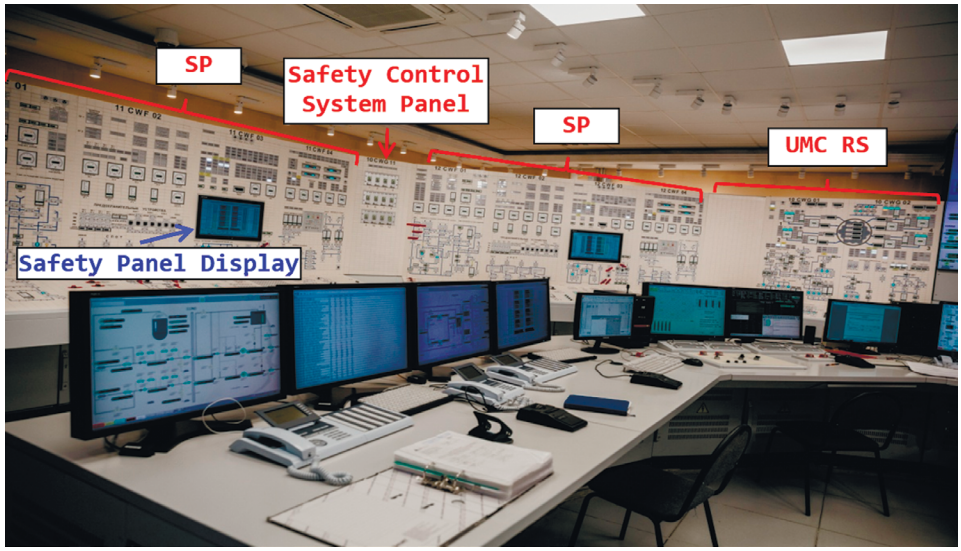
At the NvNPP-II the Special water treatment control Engineer's workplace is situated in the non-operational control area and it is based at the dual-display work stations.

Safety panels (SP)

The panels of the 1st and 2nd safety channels are identic and consist of the sections. Each safety channel has 4 sections. The Novovoronezh NPP-II (NvNPP-II) design does not include separate sections for severe accident management.

SP is rated as the equipment safety class 2У according to НП-001-15.

The section has the form of (Figure 2.4.3.3) a vertical construction of 2,2 meters height with an inclined add-on unit of the control panel. At the vertical and at the inclined surfaces of the SP section there are tile instrumentation and control panels situated.



SP – Safety Panels

UMC RS – the united mimic chart of the Reactor shop

Figure 2.4.3.3 – The safety panels and the united mimic chart of the Reactor shop

The SP section is designed for:

— Individual control of the protective, localizing and support safety systems;

— Individual control of the technological parameters;

— Process signaling, including notification on the choice of the control point (MCR/ECR), on SCSPP hardware failure, triggering of emergency protection, fire protection, accelerated preventive protection, operation modes of the actuating and performing parts of SCSPP emergency protection, of NFCE, of industrial anti-seismic protection system and of the in-core monitoring system.

The SP section provides for:

— Transmission and receipt of the commands and discrete signals on the process equipment state through the wired circuits;

— Presenting information on the process equipment state to the NPP personnel with the help of the digital and discrete-analog measuring devices, single indicators, alarm and warning message transparency, passive mnemonic symbols and sound warning;

— Presenting information on the basic safety parameters to the NPP personnel to the liquid crystal display.

The foreside of the safety system panels includes the specially arranged monitoring and controlling means connected with the certain process systems, which are grouped according to the functional features.

The limited “emergency” set of the control and measurement instruments is arranged in the vertical part of the SP, here are the individual devices with the extended control range, and also indicating instruments of neutron-flux monitoring.

The actuation duplicate keys of the essential functions of the Instrumentation and control system (ICS) are arranged in the inclined add-on control panel of the SP. With the help of these keys the operator can start the essential functions of the ICS in case of fault or failures occurrence in the triggering circuit of these functions in the auto mode.

Between the SP sections of the first channel of the safety system and the SP sections of the second channel of the safety system there is a corner insert panel of control and safety system (CSS). It includes: (Figure 2.4.3.4):

— The display with the signals on failures of the Group and Individual Control System (in the upper part);

— The integrated displays, presenting the arrangement of the 12 groups of the CSS regulating elements (in the middle part of the corner insert panel).



Figure 2.4.3.4 - CPS panel

2.4.3.2 Emergency control room

The emergency control room has the same design and the same safety area structure (see Figure 2.4.3.5) as the safety panel of the MCR. The corner insert panel of CPS is situated in the ECR between the safety panels of the two safety system channels in the same way it is situated in the MCR. There are two dual-display work stations – Reactor shop work station (RSWS) and Turbine shop work station (TSWS) situated at the workplace of the operators.

Besides, there is a fire safety instrumentation and control system (FSI CS) work station is situated in the ECR (Fig. 2.4.3.5).

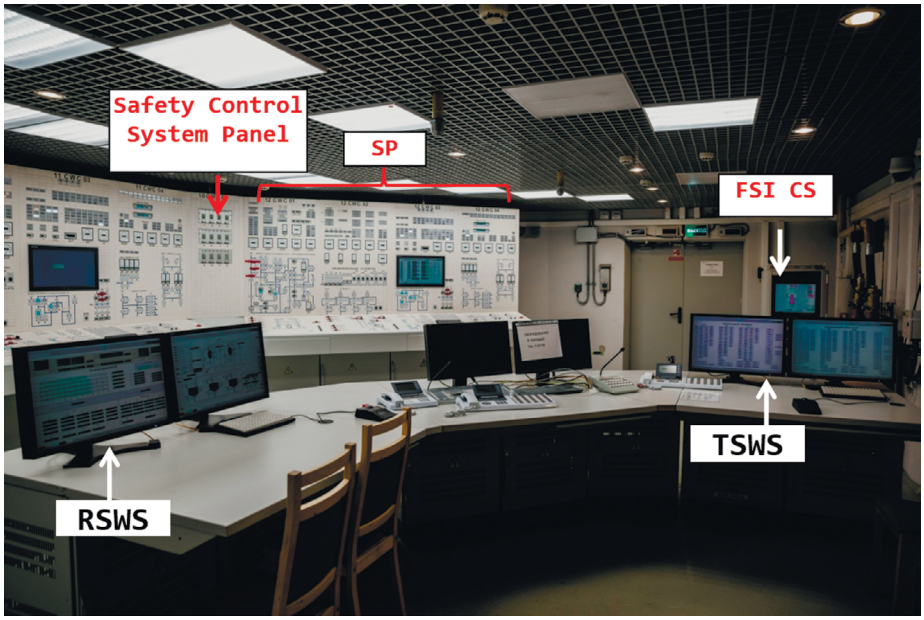


Figure 2.4.3.5 - ECR design

LIST OF REFERENCES (CHAPTER 2)

1. NP-001-15 General Safety Regulations for Nuclear Power Plants

2. GOST P IEC 61226-2011 Nuclear Power Plants. Instrumentation and control systems important for safety. Classification of instrumentation and control functions. ORDER from September 26, 2011 N-St of the Federal Agency for Technical Regulation.

3. GOST P IEC 60880-2011 “Nuclear power plants. Instrumentation and control systems important for safety. Software aspects for computer-based systems performing category A functions” approved and came into force by the Order of the Federal Agency for technical Regulations and Metrology from November 30, 2010 N 739-St GOST R-MAK 62340-2011. Nuclear power plants. Instrumentation and control systems important to safety. Requirements for coping with common cause failure. Pages 6-14.

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6. “VVER-1200: Evolution of the Classics. Operation Physics, Systems and Elements, Nuclear Fuel, Safety” by Andrushechko S.A., Favorov B.Yu, Vasiliev B.Yu, Kosourov K.B, Semchenkov Yu.M, Kuchumov A.Yu., Ukraintsev V.F. published in Moscow, “Logos” 2019, 672c, ISBN 978-5-98704-852-8.

3 TECHNICAL AND ENGINEERING MEANS FOR MANAGEMENT OF AN BEYOND-DESIGN ACCIDENT (BDA)

3.1 THE REQUIREMENTS FOR ADDITIONAL TECHNICAL AND ENGINEERING MEANS FOR BDA MANAGEMENT

According to the requirements of paragraph 2.5 NP-001-15 the NPP design should provide for systems (elements), essential for providing safety, including systems (elements), provided by the NPP design for managing accidents within the first 72 hours after the accident initiating event (or within another defined in the NPP design time interval, which should not be less than 72 hours). The Novovoronezh NPP-II design was created according to NP-001-98, which didn't include the equivalent requirement.

Due to the events in March, 2011 at the Fukushima NPP there were created working parties for working out a list of possible accident initiation and development scenarios and steps of response, eliminating the accident aftermath for the population and the environment. The analysis was carried out and the list of scenarios of possible accident development at Russian NPPs was prepared.

In the beginning of 2012 there was organized the similar work process for the power units under construction.

The following actions were taken:

— additional assessment of the design technical solution sufficiency, of the safety system efficiency, of the reliability of barriers of defense in depth to ensure the NPP safety in the event of extreme external impacts, which are provided by the project;

— assessment of efficiency and sufficiency of the hardware, available at the NPP, and of the preventive organizational measures in the event of extreme external influence threat;

— assessment of efficiency and sufficiency of the hardware, available at the NPP, and of the preventive organizational measures of controlling severe beyond-design accidents and mitigation of their consequences;

— assessment of the NPP safety in the event of extreme external impacts, exceeding boundary limits, provided by the NPP design.

Safety analysis was carried out for the following initial events:

— major NPP blackout (total loss of all the alternating-current supplies, including emergency generators);

— loss of all the water ultimate heat sink;

— the combination of all the above mentioned events.

The listed events were considered as in case of primary circuit leak and without primary circuit leak.

The calculation is performed to estimate time limit until the fuel is damaged in the initiating event of heat transfer failure. It demonstrates that under the worst case scenario (the reactor is depresserised, the fuel is not defueled, the level in the reactor is decreased) the time till fuel uncovering is 64,3 hours.

The results of the assessments and analyses and the hardware suggested were listed in the corresponding report on Novovoronezh NPP-2 safety analysis in the event of extreme external impacts. It demonstrated that for the accident with the total loss of all the alternating-current supplies with the reactor operation at power to eliminate fuel damage in the reactor for a long time continuing until the resumption of power unit supply, the operating personnel needs no longer than 24 hours since accident initiation (maximum time, during which guaranteed power supply from the accumulators of instrumentation and control systems of the power unit is provided) to take the afterheat removal passive system off into cooldown mode.

For other accidents, putting into operation no later than 24 hours after the beginning of the accident, an alternative intermediate circuit, which, by using of a mobile diesel engine and a mobile cooling tower with a fan, provides water supply to the SFP and to the reactor with a flow rate of 50 m^3/h and 270 m^3/h from one SFP cooling pump, as well as to the cold legs of the main circulation piping with a flow rate of 2 x 14,5 m^3/h from two emergency boron injection pumps from the containment sump with a temperature not exceeding 70 °C, will ensure reliable cooling of the reactor installation, the fuel tank and the containment for an unlimited time .

For the situations connected with prolonged smoke in the NvNPP-II site (with extensive burning of the forests adjacent to the NPP site), the general designer has proposed the option of providing habitability of the control room / emergency control room in emergency situations using an additional balloon life support station.

Based on the report of Concern Rosenergoatom, there was worked out “Decision to supplement the design of the Novovoronezh NPP-2 with technical means for managing the beyond-design accidents”, in accordance with which the project was supplemented with the following means for managing the BDA:

— an alternative air-cooled diesel power station;

— an alternative intermediate circuit with an air cooling tower;

— a mobile pumping unit for supplying water from external sources;

— cable lines for the operational connection of consumers used to manage the BDA.

The performance of additional hardware was specified on the basis of the calculated analysis of the modes with the loss of all the alternating-current supplies and with the simultaneous primary circuit leak.

3.2 INFORMATION MANAGEMENT, MONITORING AND ACQUISITION IN CASE OF BEYOND-DESIGN ACCIDENT

With the purpose of information support to operating personnel while accident management and conduction of post-accident arrangements at power units №1 and №2 of Novovoronezh NPP-2, it was implemented the Emergency I&C.

The Emergency I&C is consisted of technical means that ensure measurement and displaying of the information on power unit conditions while accident and within post-accident period (herein just beyond-design accident is considered as an accident).

Information support is carried out through provision to operating personnel of information on values of the key parameters of power unit in order to ensure the operating personnel is able to undertake the effective measures on accident management and the support personnel arriving for help rendering is informed in a proper way.

Based on its designation, the Emergency I&C provides the NPP personnel with information required to manage the beyond-design accident (levels 3, 4 of defense-in-depth) including the following:

— prevent the beyond-design accidents progression (accident transition to the severe stage) and mitigate their consequences (in case of transition to the severe stage);

— protect containment from damaging and maintain its operability (maintain the forth barrier);

— prevent fuel melting in SFP, and also reduce consequences of fuel melting and radioactive substances escape;

— NPP returning to the controlled state under which:

1) it is stopped the self-sustained chain reaction;

2) it is ensured the continuous fuel cooling (heat removal from reactor core);

3) radioactive substances and media are restrained within the established limits.

Set of power unit parameters controlled with Emergency I&C should provide the NPP operating personnel with the possibility to execute the following functions:

— monitor integrity of the physical barriers on the path of ionizing radiation and radioactive substances propagation into environment;

— control performance of the basic safety functions. The following critical safety functions are considered:

1) «Provision of reactor core subcriticality» — fast shut-down and maintaining of the reactor core in subcritical condition);

2) «Nuclear fuel cooling» (in reactor) — provision of the reliable heat removal from reactor core while accident, and also after parameters stabilization in post-accident condition;

3) «Nuclear fuel cooling» (in SFP) – provision of the reliable heat removal from fuel in SFP;

4) «Provision of heat removal from the primary circuit» – provision of heat removal from reactor core by the secondary circuit systems;

5) «Provision of the primary circuit integrity» – protection of reactor vessel from the "cold" embrittlement;

6) «Provision of leak-tight enclosure integrity» – provision of leak-tight enclosure integrity to minimize the radiological consequences;

7) «Provision of the primary circuit coolant inventory» – provision of the reliable heat removal from reactor core.

To develop the technical requirements to Emergency I&C in case of beyond-design accident in terms of resistance to the external acting factors, there were used the results of the available for now calculations of the beyond-design accidents conditions that are the most representative from the view of maximum values of reactor facility and values of the external acting factors parameters (for example, «Large leakage while breakage of the primary circuit piping with the total loss of all the AC power supplies»). Herewith, the worst scenarios of beyond-design accidents progression are conventionally assumed (in order to develop the most rigorous requirements to technical means of Emergency I&C) that lead to the possible damage of defense-in-depth physical barriers.

The defining parameters of the external acting factors based on which the requirements to technical means of Emergency I&C should developed, are the following parameters of environment in containment at the places of the Emergency I&C technical aids arrangement:

— pressure;

— temperature;

— humidity;

— radiation parameters (absorbed dose rate, ionizing radiation flux).

Besides the values of the external acting factors parameters within the total volume of containment, there were considered the possible values of the external acting factors parameters within the separate areas of containment (in case of Emergency I&C technical means arrangement near to the primary circuit equipment or in the other local areas where the values of the external acting factors parameters could significantly differ from the common ones within containment. For example, the space within the upper unit where the sockets of CPS, in-core monitoring system, the channels of ionizing chamber in dry shield and the others are located. To monitor the in-core parameters there are considered the parameters of environment in reactor vessel.

So, at power units № 1 and № 2 of Novovoronezh NPP-2 it is ensured the required monitoring with Emergency I&C means in condition of beyond-design accident. 48 parameters (Table 3.2.1):

Table 3.2.1 List of parameters ensuring the required monitoring with Emergency I&C means in condition of beyond-design accident

Parameter description	Measurement range	Variation range	Required measurement accuracy in accident conditions
1 Neutron features: — reactor power, % Nnom; — reactor period, sec *	 10^{-4} — 1 5 — 999	 10^{-8} — 1 5 — 999	 ± 10% N_{act} ± 20 % of the actual level
2 Temperature of coolant in «cold» legs of loops, °C*	60 — 300	0 — 400	± 5
3 Temperature of coolant in reactor vessel at core outlet, °C*	100 — 1400	0 — 1400	± 15
4 Temperature of coolant in «hot» legs of loops, °C	60 — 350	0 — 400	± 5
5 Margin to saturation temperature of the primary circuit coolant, °C	70 — 0	100 — 0	± 5
6 Pressure at reactor outlet, MPa*	0 — 23,8	0 — 25	± 0,3
7 Level of water in reactor under accident conditions, mm (discrete)*	—	—	± 50 in three discrete points along the reactor height
8 Level of water in standby modes (below reactor main joint), m	0 — 3,8	0 — 4,0	± 80
9 Coolant level in reactor pit (indicator) (discrete)	—	—	± 50
10 Level of boron solution in sump, m			
11 Temperature of reactor vessel metal, °C	60 — 1400	0 — 1400	± 30
12 Temperature of reactor pit concrete, °C	0 — 70	0 — 150	± 2
13 Concentration of boron acid at reactor inlet, g/dm^3	0 — 16	0 — 25	± 1
14 Position of the pilot operated relief valve of pressurizer (discrete)	closed/ open	—	—
15 Position of valves along the additional control line of pilot operated relief valve of pressurizer (discrete)	closed/ open	—	—

Parameter description	Measurement range	Variation range	Required measurement accuracy in accident conditions
16 Level in pressurizer, m	0 — 12,7	0 — 12,7	± 300
17 Level of boron solution in ECCS accumulator, m	0 — 8,5	0 — 10,0	± 250
18 Pressure in ECCS accumulator, MPa	0 — 6,9	0 — 10	± 0,1
19 Position of check valves and fast-actuating valves of ECCS accumulator (discrete)	closed/ open	—	—
20 Levels of steam generator SG1-4, m	0 — 4,0	0 — 4,0	± 120
21 Pressure at outlet of steam generator SG1-4, MPa	0 — 10	0 — 10	± 3
22 Position of pilot operated relief valve of SG (discrete)	closed/ open	—	—
23 Level in tanks of emergency boron injection system, m			
24 Position of shut-off valves at suction and discharge of emergency boron injection system pumps (discrete)	closed/ open	—	—
25 Flow from emergency boron injection system pumps, m^3/h			
26 Position of check valves, double check valves and shut-off valves of the system of reactor core passive flooding system (discrete)	closed/ open	—	—
27 Level in tanks of reactor core HA-2 system, m	0,3 – 9,7	0,3 – 9,7	± 0,5
28 Position of the air gate valves and regulating devices of PHRS (discrete)	closed/ open	—	—
29 Position of fast-acting pressure reducing valve to atmosphere BRU-A (discrete)	closed/open, level of opening /closing	—	—

Parameter description	Measurement range	Variation range	Required measurement accuracy in accident conditions
30 Position of gate valve before fast-acting pressure reducing valve to atmosphere (discrete)	closed/ open	—	—
31 Position of main steam isolation valve and motor-operated isolation valve behind main steam isolation valve (discrete)	closed/ open	—	—
32 Level in SFP, m	0 — 6,9	0 — 12,5	± 250
33 Temperature of boron solution in SFP, °C	20 — 150	0 — 150	± 5
34 Position of the shut-off valve at pumps suction and discharge of SFP cooling system	closed/ open	—	—
35 Flow from pumps of SFP cooling system, m^3/h			
36 Dose rate in containment, Gy/h	0 — 7*104	0 — 7*104	not above ± 10 %
37 Pressure in containment, MPa	0 — 0,5	0 — 10	± 1,5
38 Temperature in containment, °C	0 — 250	0 — 260	± 3
39 Hydrogen content in containment, vol. %	0 — 20	0 — 25	± 1,25
40 Oxygen content in containment, vol. %	0 — 25	0 — 25	± 12
41 Position of valves of automatic gas removal system (discrete)	closed/ open	—	—
42 Position of localisation valves (discrete)	closed/ open	—	—
43 Dose rate at NPP site			
44 Water level in inspection cavity of protective tubes, m			
45 Water level in in-vessel internals inspection cavity, m			
46 Temperature in core catcher, C			

Parameter description	Measurement range	Variation range	Required measurement accuracy in accident conditions
47 Position of valves on water supply piping to core catcher (discrete)	closed/ open		
48 Water level in core catcher (indicator), m (discrete)			

The technical means of Emergency I&C include the following:

— impulse piping with valves and penetrations;

— sensors and sensors benches;

— normalizing transducers;

— cable stubs, cable ways and cable and cable penetrations;

— information transformation devices (if required);

— information displaying, signalizing and recording;

The technical means of Emergency I&C are important in terms of impact on safety and related to the safety class 3 according to NP-001-15 and Seismic Category I.

Electrical power supply of the Emergency I&C technical means is organized from the autonomous systems of power supply in the way that interruption of power supply to Emergency I&C technical aids would be possible just for a fleeting second (automated load transfer). As power supplies there are used the reliable power supply systems of the first group with provision of power supply to Emergency I&C in case of total loss of all the AC power supplies within the total period of accident and post-accident period considering application of the special technical means of beyond-design accidents management.

For all the parameters of Emergency I&C it is ensured the continuous online displaying, signalizing and recording of information, and also information transmission into «Black box» and NPP Research Center.

Recording of the information from Emergency I&C is carried out with the recording devices with ensuring of the possibility to unload at any time the recorded information to operating personnel. Recording shall be ensured (at least) 30 min before the initial event and within 72 hrs. after the initial event.

3.3 ADDITIONAL TECHNICAL MEANS OF BEYOND-DESIGN ACCIDENTS MANAGEMENT

3.3.1 COMMON ORGANIZATION SCHEME FOR THE TECHNICAL MEANS OF BEYOND-DESIGN ACCIDENTS MANAGEMENT

Figure 3.3.1.1 shows the organization scheme for the technical means of beyond-design accidents management.

The goal of the suggested arrangements is to remove heat from reactor and spent fuel pool, to exclude water boiling inside them, to maintain sub-critical condition of fuel in reactor and in spent fuel pool, and to maintain the pressure in containment within the allowed limits for the long time period.

In case of beyond-design accidents not related to the primary circuit leakages, the pump of the SFP cooling system (FAK10 20) executes intake and return of the SFP water with the flow of 300 m^3/h according to the standard procedure. Herewith, it is ensured removal of heat from SFP to fan cooling tower through the alternative industrial circuit that cools down the heat exchanger of FAK10 20 system. Heat removal from reactor facility is carried out through operation of the passive heat removal system. The similar heat removal scheme is used in case of the complete fuel unloading from spent fuel pool.

In case of accidents with the primary circuit leakage, the pump of FAK10-20 system supplies water to spent fuel pool with the flow of 50 m^3/h from the emergency sump along the existing piping. It ensures heat removal from fuel located in SFP and prevents water boiling in it. After the alternative industrial circuit is put into operation and it is controlled the initial supply of the cooled water into SFP, there is a possibility to open valves on emergency water storage drain line from the SFP into emergency sump for the guaranteed availability of the required water level in the sump.

In parallel, the pump of FAK10-20 system and the pumps of JND system supply water to reactor from the emergency sump with the total flow of at least 300 m^3/h that excludes the process of coolant boiling in reactor.

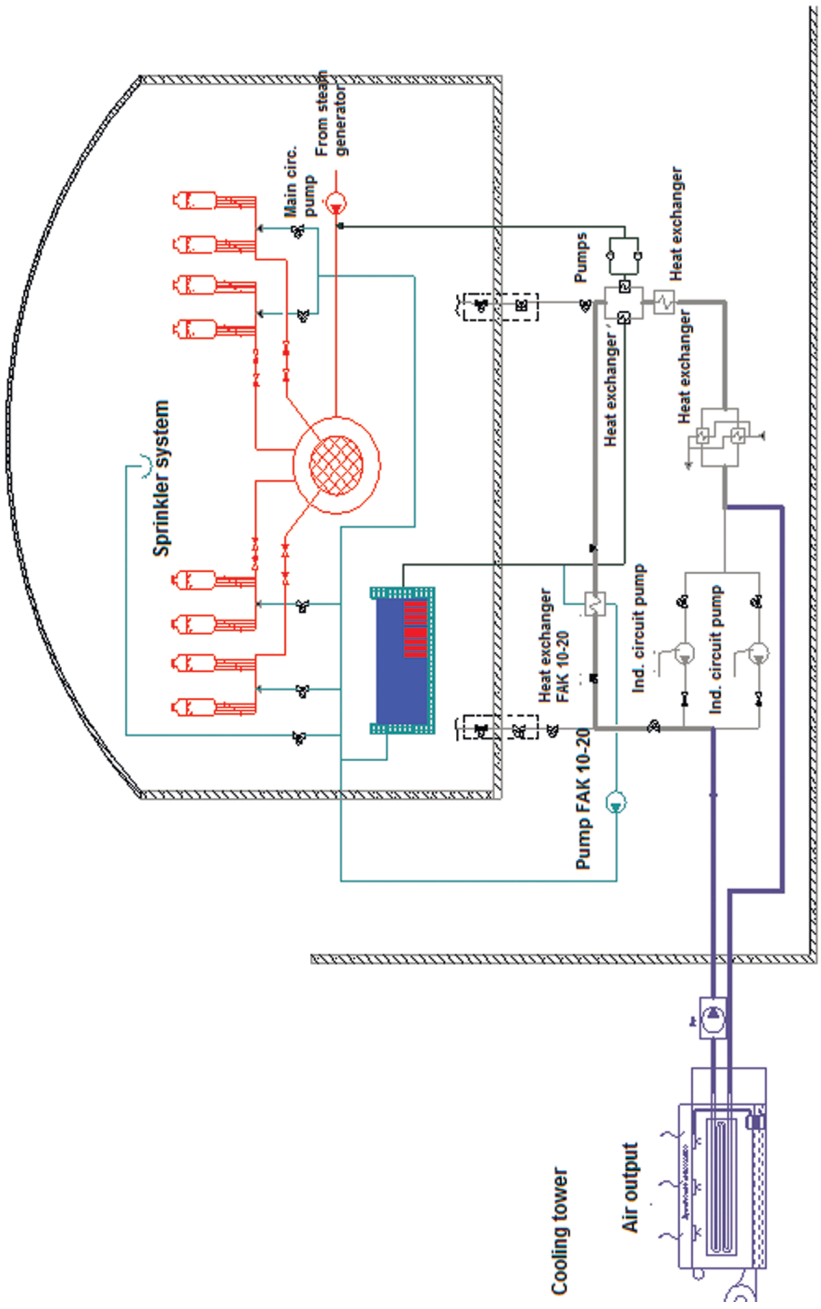


Figure 3.3.1.1 - Organization scheme for the technical aids of beyond-design accidents management

In all the mentioned modes it is excluded supply of the steam into containment. Consequently connection of the special means for pressure reduction in containment is not required. Though, the design suggests the possibility to supply water to sprinkler headers and thereat heat is removed in the following way: emergency sump – heat exchanger of FAK10-20 system – pump of FAK10-20 system - sprinkler nozzles – containment space – emergency sump. Such scheme of heat removal could be performed for example if it has not been possible to open valves on piping of water supply from the pumps of FAK10-20 system into reactor.

The performed design-basis justification showed that for the considered accidents activation not later than 24 hrs. since the accident of the alternative industrial circuit that ensures with the movable diesel generator and fan equipped movable cooling tower water supply to SFP and reactor under the flow of 50 m^3/h and 270 m^3/h accordingly from one cooling pump of SFP, and also into cold legs of main circulating piping with the flow of 2 x 14,5 m^3/h from two emergency boron injection pumps from containment sump at the temperature not above 70 °C, ensures the reliable cooling of reactor facility, SFP and containment during unlimited time period.

Herewith, for the accident with the total loss of AC power supplies and with the coolant leakage in parallel within the considered time range (100 hrs. of accident process), it is obtained the stable condition for all the parameters of reactor facility and containment that exhibits slow but stable cooling and pressure reduction in reactor facility and containment.

Considering the mentioned above, the solution suggested to be implemented in design including application of the alternative industrial circuit provides the reliable cooling of reactor facility and containment in case of this accident within the unlimited time period.

3.3.2 DESCRIPTION OF THE TECHNICAL MEANS FOR BEYOND-DESIGN ACCIDENTS MANAGEMENT

3.3.2.1 Intermediate circuit of KAA

Based on the results of works on proposals described before, the following components are implemented in the intermediate circuit of essential consumers, UJA building of the 1st safety train of every unit's safety system:

— fan cooling tower;

— movable pump unit (to organize circulation of industrial circuit water through cooling tower);

— movable pump unit for fan cooling tower make-up.

Cooling tower and movable pump unit are installed on the concrete site that consists of concrete plate with asphalt concrete coat.

Fan cooling tower, movable pump unit, movable pump unit of fan cooling tower are suitable for open-air storage and operation at the temperature from minus 37 up to plus 38,9 °C.

Thermal power removed with air cooling tower is equal to 21,2 MW at the ambient air temperature of + 24 °C (residual-heat output of reactor core 24 hrs after actuation of reactor core is equal to 20,4 MW).

Diagram for connection of fan cooling tower, movable pump unit and movable pump unit of fan cooling tower to KAA industrial circuit is shown in the Figure 3.3.2.1.1.

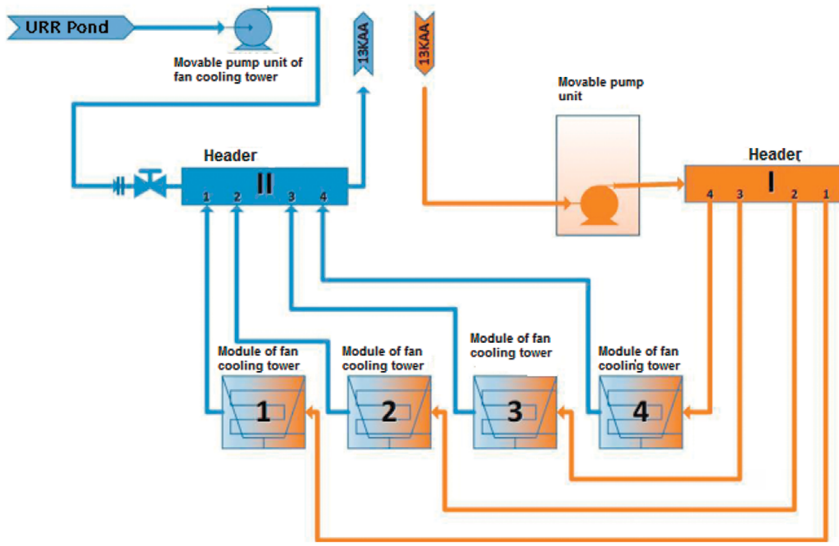


Figure 3.3.2.1.1 – Connection diagram for beyond-design accidents management technical aids

The procedure of putting into operation of fan cooling tower cooling circuit includes the following steps:

— connect power supply from diesel generator facility;

— flexible hoses laying, their fastening, connection to terminals from reactor building (to the KAA circuit system);

— move the movable pump unit of fan cooling tower from the storage place to operation place and its connection to the flexible hoses system;

— fill the fan cooling tower circuit;

— switch the movable pump unit of fan cooling tower to fan cooling tower in

order to fill in the ponds of cooling tower cells and for their further make-up (water is used for spraying of heat exchanging surface and evaporated while cooling tower operation);

— provide transportation of fuel for movable pump unit and motorized pump

The mentioned operations are carried out within 24 hrs since accident. The scope of works that should be performed almost doesn't exceed the scope of the works on connection of the additional technical aids performed by personnel at the power units under operation with VVER-reactors in the Russian Federation. Based on the results of personnel drills conducted at the power units under operation, the time required to connect diesel generator facility and movable pump unit is equal to about 2 hrs.

3.3.2.2 Fan cooling tower

Fan cooling tower consists of evaporative cooler with centrifugal fans, REF-C type manufactured by DECSA (Italy) (Figure 3.3.2.3.1).



Figure 3.3.2.3.1 – Fan cooling tower

Fan cooling tower includes four cells connected between each other with parallel flexible hoses in the single facility. Piping of fan cooling tower is performed with flexible hoses along the cooled water; cooled water supply/takeout to all the cooling tower cells is arranged through one hose/connection (distributing device) with the further uniform delivery/removal of cooled water from the cells.

Every cell of fan cooling tower consists of totally enclosed evaporative cooler,

pump that transfers the liquid for spraying of heat exchanging surface, fans and pumps electrical motor, tray for the liquid spraying the heat exchanging surface (pond) behind the fans, and the grids for air intake that prevents foreign materials entrance in cooling tower.

With the fan cooling tower it is supplied the set of flexible hoses Du 200 with the total length of 260 m and connectors that provide make-and-break joint of cooling tower to piping of KAA system and to elbows of movable pump unit.

The cooling towers are manufactured as climate version equipment. So for the storage period of the assembled cooling towers (stand-by condition) the covers for cells, equipment to protect from climate factors impact are not required.

3.3.2.3 Movable pump unit for fan cooling tower make-up.

The movable pump unit includes: platform on which there are installed pump drive motor, pump, container, motor control panel, pump I&C board, module of the pumped media flow measurement, fuel tanks (main tank and storage tank).

Movable pump unit of fan cooling tower is manufactured as a container-type facility with internal reheating and equipped with flow meter having the independent power supply.

Movable pump unit of fan cooling tower is placed on transporting platform installed on trolley for its further moving. The movable pump unit of fan cooling tower includes: transporting platform on which there are installed pump drive motor, pump, fuel tank. On the discharge and suction elbows there are installed manometer and mano-vacuum meter for the pumped media pressure control.

Driving of the movable pump unit of fan cooling tower is performed from combustion engine. The diesel facility of the movable pump unit of fan cooling tower is hand triggering.

3.3.2.4 Alternative diesel generator (Alternative DG).

At Novovoronezh NPP-2 as an alternative power supply for the systems consumers in case of design-beyond accidents related to loss of power supply from the external and internal sources (including the diesel generators of Emergency Power Supply System) and for all the final heat absorbers for the time above 24 hrs., it is suggested to use the alternative container-type DG with air cooling.

The diesel power plant manufactured by LLC «VTEZ» type ED2000-T400-3RN (rated power of 2 MW, voltage of 0,4 kV) is used as an alternative diesel power plant.

The alternative diesel power plant is trailer mounted and could be operated for the long period being installed in this way. The trailer is installed on outriggers being located on the special concrete basement.

To maintain the thermal conditions ensuring operation availability within not above 15 seconds and capability to the fast load acceptance, the diesel power plant is equipped with the system of cooling liquid reheating with electrical heater (tubular heating element) and with the heating system for the air inside the block-container with power supply from the independent tension source of 220 (380) V, frequency of 50 Hz while staying in «Stand-by» condition. In de-energized unit condition, reheating is carried out with independent diesel reheating facility with the own diesel fuel storage tank for the period of 48 hrs.

It is suggested to use the alternative diesel generator to supply power to the following consumers:

— technological mechanisms of the 1st train of safety system:

1) emergency boron injection pump of the 1st channel of JND system;

2) cooling pump of the FAK spent fuel pool;

— motor-operated valves;

— fan cooling tower;

— ventilation facilities of KLB22 system.

— DC boards of the 1st and 2nd channels of emergency power supply system (through rectifiers).

To provide power supply to consumers it is considered to use the switchgear of 0,4 kV tension as a part of the alternative diesel generator.

Connection of the alternative diesel generator under beyond-design accident conditions is carried out with the flexible cables:

— to the cells of 0,4 kV of emergency power supply system - 11BMA, 11BMC and 12BMB, 12BMD provided lack of voltage at the 0,4 kV cells;

— to terminals of fan cooling tower.

The supplied flexible cables make a part of alternative diesel generator assembly. They are not laid in the fixed condition; they are connected just in case of necessity to connect the alternative diesel generator.

Voltage supply to the 0,4 kV cells is carried out through actuation of disconnector-switch located in cabinet № 8 of every cell. Actuation of the disconnector-switch is executed manually. While connection of the alternative diesel generator from the 0,4 kV cells, all the consumers should be disconnected except the mechanisms engaged in beyond-design accident elimination, and rectifiers. In normal NPP operation conditions and in case of design accident, the disconnector-switch is positioned in switched-off state.

Alternative diesel power plant start-up is carried out by personnel manually with modular launcher supplied as a part of whole assembly.

3.4 MOLTEN CORE CATCHER

The system of the molten core catching and cooling outside the reactor-JKM, the main component of which is molten core catcher, is designated for restraining and cooling of the liquid and solid fragments of the damaged reactor core, reactor vessel parts, in-vessel internals in case of severe accident with reactor core melting.

Overall dimensions and layout of the molten core catcher units in reactor pit are shown in the figure 3.4.1.

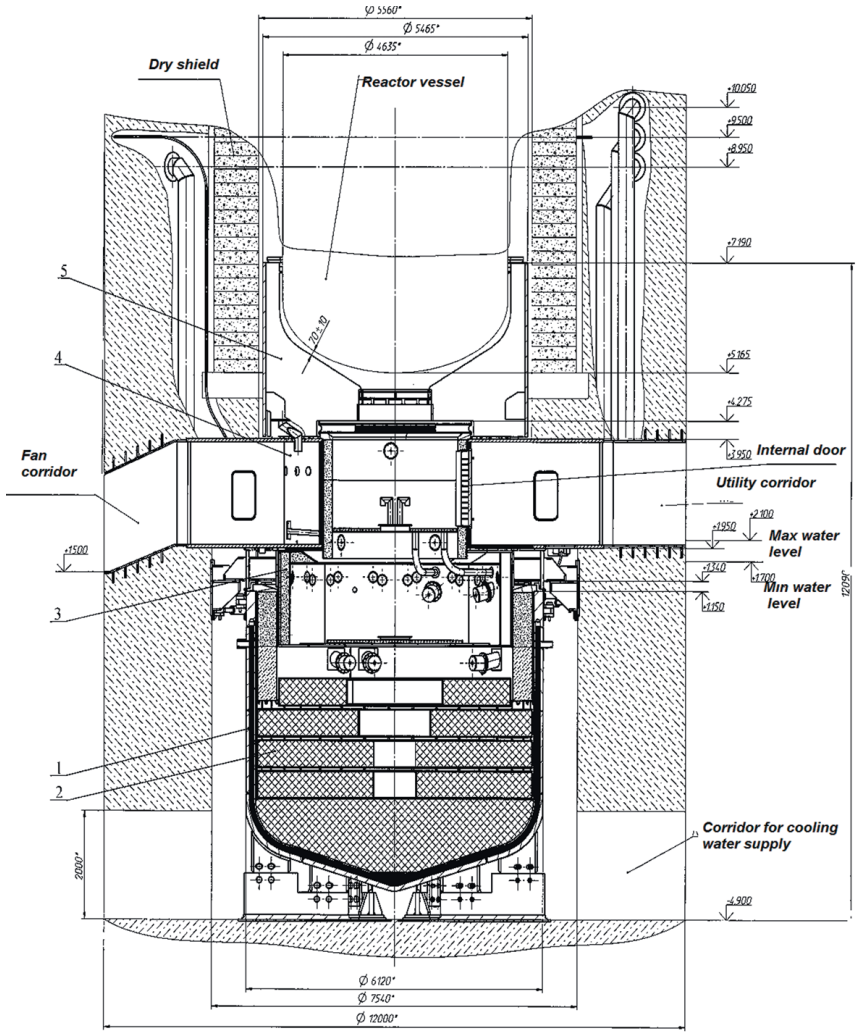


Figure 3.4.1– Layout of the molten core catcher in reactor pit

The molten core catcher performs the following basic functions:

• receiving and positioning of the molten core, solid fragments of core and reactor construction materials (corium);

• stable heat removal from corium to cooling water;

• exclusion of radiation activity release beyond the catching area limits;

• exclusion of the self-induced chain reaction in corium;

• minimization of radioactive substances and hydrogen escape into containment space.

Molten core catcher consists of the following functional units positioned (Figure 3.4.1 – Top-downward vision) in the direction of corium movement from the reactor vessel to the pit concrete basement:

— Lower plate;

— Cantilever truss;

— Filler (sacrificial material);

— Vessel

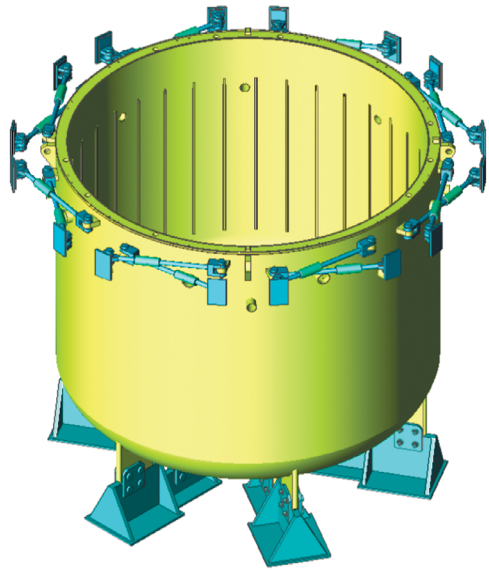


Figure 3.4.2 – Vessel of the molten core catcher with supports and installed double rods

The lower plate guides the corium motion after reactor vessel damage or melting. Thermal insulation ensures protection of cantilever truss and service platform from damaging and protects the vessel with supports form damaging.

Cantilever truss provides protection of the molten core catcher structure and communication aids from damage on the part of corium. It is a support for the lower plate and includes jacket-tubes that ensure connection of I&C sensors aimed to organize monitoring of temperature of media in the molten core catcher, cooling water level in concrete pit, water availability in concrete pit in case of normal operation.

Filler of molten core catcher (Figure 3.4.3) consists basically of iron oxide and aluminum oxide. It permits to prevent floating of the coarse-meshed filler up to its complete solving with corium and protects the vessel with supports from thermal and mechanical damage. It ensures the volume distribution of the molten core within the vessel with supports and is designated to dilute corium in order to reduce volume energy output, to enlarge heat exchanging surface and to ensure floating of fuel containing fractions of corium over the steel layer.

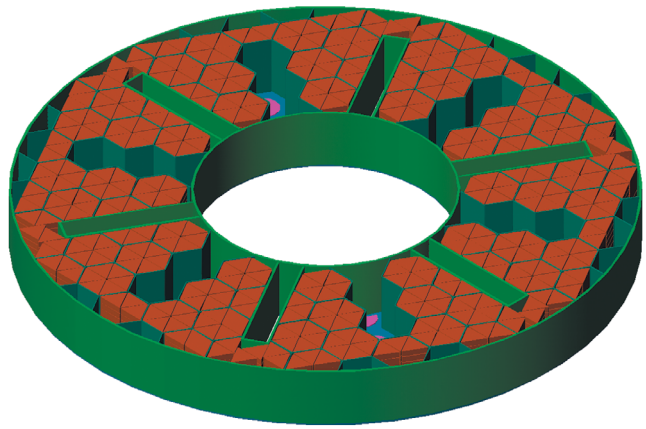


Figure 3.4.3 – Diagram of cartridges block with filler

The vessel with supports ensures restraining and cooling of the melted core in under-reactor building of the concrete pit in case of reactor vessel melting or damaging. The vessel with supports includes jacket-tubes that ensure connection of I&C sensors aimed to monitor filling of the molten core catcher vessel with corium.

To prevent breakdown of the walls of the concrete pit and reactor fastening module in result of the thermal irradiation from the molten material and also to speed up its cooling-down on corium surface that is located in the vessel with supports, it is additionally supplied the cooling water from wells of the internals inspection cavity along piping positioned in cantilever truss. Water is supplied manually on corium in 30 minutes after reactor vessel destruction. Reactor vessel destruction and corium delivery to the molten core catcher is detected based upon temperature rise or sensors failures.

In case of severe beyond-design accidents (primary circuit leak with the total loss of power supply, diesel generators included), for heat exchanger cooling it is used water from containment sump that is delivered there in result of leakage from the primary circuit. In addition, in the sump through the primary circuit leak besides the content of the main circulating circuit piping it is supplied water of accumulator tanks of emergency core cooling system. It is also used the spent fuel pool. As far as in this case the volume of the boron solution emergency storage kept in spent fuel pool has not been used by the active systems of the emergency core cooling system, the water is drained along the pipelines specified for such purposes. Thereat, operator in the Unit Control Room (Emergency Control Room) opens the motor-driven valves which are fed with emergency power supply system (accumulator batteries). The total volume of water in sump and on containment floor is equal to above 1500 m^3.

After corium entrance in molten core catcher, temperature of the molten core catcher vessel walls is gradually getting increased. Water around the vessel of molten core catcher in reactor pit is getting heated and boils. While rising up, the saturated steam goes to containment space through steam discharge channels that are positioned in cantilever truss and in reactor pit walls. The conditions of longstanding cooling of the melted material are starting. The steam generated in containment is partially condensed; the condensate is supplied once more in sump connected to reactor pit to cool the molten core catcher.

In 2-3 hrs. after corium enters the molten core catcher, there are actuated the passive valves of water supply through the upper side of corium into vessel of molten core catcher. These valves ensure the cooling water supply from reactor pit where the vessel of molten core catcher is located.

The design suggests that within 24 hours there would be performed the organizational measures to restore operation availability at least for one channel of emergency and planned cooling system of the primary circuit and of spent fuel pool cooling (JNA). After power supply is reactivated, water circulation is carried out with pumps of JNA system. Heat removal from corium is performed in heat exchangers of JNA system. Under these conditions corium can be kept in the catcher in safe state for the long period.

Additionally, to ensure the safe restraining of the melt material in catcher after use of storage water in internals inspection cavity, it is suggested to supply water to corium from the external sources (fire engines).

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4 SAFETY SYSTEMS IN THE DESIGNS OF NOVOVORONEZH NPP-2 WITH REACTOR FACILITY V-392 AND OF LENINGRAD NPP-2 WITH REACTOR FACILITY V-491

4.1 GENERAL PROVISIONS

The designs of Novovoronezh NPP-2 with reactor facility V-392 and of Leningrad NPP-2 with reactor facility V-491 are developed considering the experience on designing and operation of the commercial project U-87 (reactor facility V-320) and of the Units 1, 2 of Tianwan NPP (reactor facility V-428).

Safety of Novovoronezh NPP-2 and Leningrad NPP-2 is based on the defense-in-depth concept. Nevertheless, there are some differences in implementation of this concept in the different designs.

Novovoronezh NPP-2 (reactor facility V-392M)

The safety functions in case of design-basis and beyond-design accidents are supported with the comprehensive operation of the active and passive safety systems. The active safety systems have two trains structure with backing-up of the active elements inside the trains. Failure of the active part of safety system doesn't mean automatically the safety function non-performance.

Let's consider for example supporting of the safety function «Coolant inventory maintenance in reactor» in case of the maximum design-basis accident.

In the design as the maximum design-basis accident it is considered the guillotine rupture of the main circulating piping Du 850 at reactor inlet (the first and the third loop) with superimposement of the auxiliary power supply loss and postulated failures of:

— one emergency diesel generator

— one primary accumulator tank and one channel of the secondary accumulator tank connected to the coupling duct of the failed tank of the emergency core cooling system.

Therewith, it is considered the depended failure (operation with leakage) of one ejector-pump connected to the emergency loop.

In this case, operation of three primary accumulator tanks and of one low pressure channel of the emergency core cooling system support performance of the safety function «Coolant inventory maintenance in reactor» (Figure 4.1).

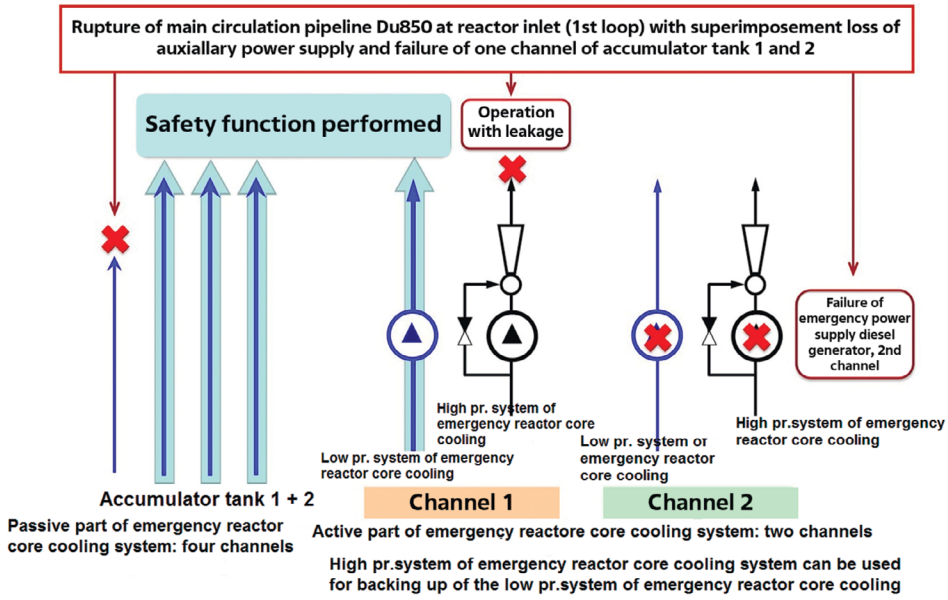


Figure 4.1 – Operation diagram of the active and passive parts of emergency reactor core cooling system in case of the maximum design-basis accident

If at the time of the initial event occurrence the pump of the 1st channel low pressure system of emergency reactor core cooling is under maintenance, the total failure of the active part of emergency reactor core cooling system happens.

In this case the safety function «Coolant inventory maintenance in reactor» is supported with operation of three channels of accumulator tanks 1 and 2 (figure 4.2).

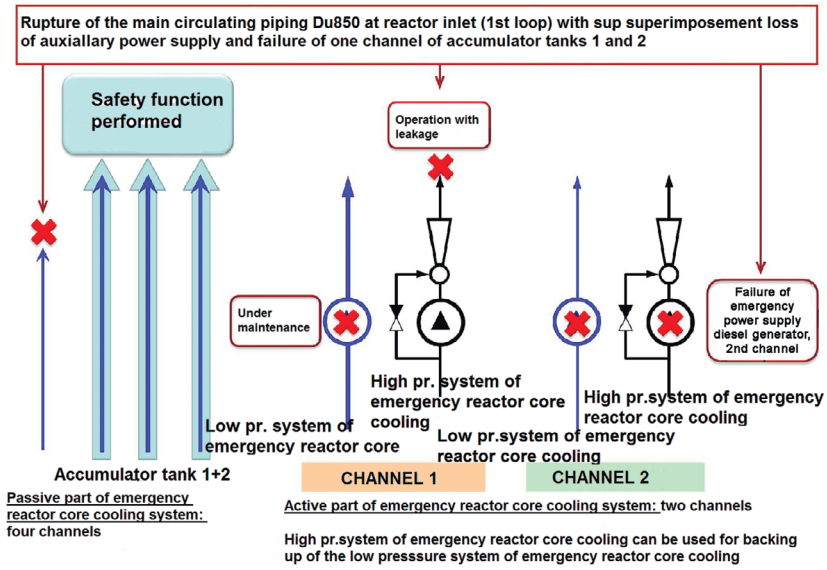


Figure 4.2 – Operation diagram of the active and passive parts of emergency reactor core cooling system in case of the maximum design-basis accident and removal of the pump of low pressure system of emergency reactor core cooling for maintenance

Water amount (weight) in the 2nd stage accumulator tanks permits to keep the coolant inventory in reactor at least for the period of 24 hrs provided the heat removal passive system of steam generator is under operation.

During the period 24 hrs after, the design suggests the additional technical means: alternative diesel generator, alternative industrial circuit, additional pipeline for water supply to connecting ducts of «2nd stage accumulator tank– reactor» from the 1st channel of cooling system of the spent fuel pond FAK10.

These technical means permit to stop water boiling in reactor core and in fuel pool, and also to carry out heat removal within unlimited period (at least 21 MW) from accident confinement area through alternative industrial circuit to ambient area which is a final heat absorber.

Leningrad NPP-2 design (Reactor facility V-491)

Safety concept of the design is based on prior use of the active safety systems aimed to manage design-basis accidents and to prevent accidents transition to severe stage.

Combined use of the active and passive systems is suggested to manage severe accidents.

Active safety systems have four-channels structure which from one hand permits to remove one safety train for maintenance if required for the long period, and

from the other hand increases scope of procedural inspections.

Independence of the passive safety systems in case of beyond-design accidents is equal to 24 hrs and defined by water inventory in emergency heat removal tanks.

During the period 24 hrs after, it is required to make up the tanks of emergency heat removal and fuel pool with movable equipment and stand-by water inventory located at the site.

4.2 SAFETY FUNCTIONS

Safety systems and technical means aimed to manage beyond-design accidents support performance of three fundamental safety functions (goals):

— Shutdown fission reaction and maintain reactor in subcritical condition (reactivity control);

— Remove the residual heat from nuclear fuel positioned in reactor and spent fuel pond to the final absorber;

— Confine activity within the physical barriers.

The Table 4.1 shows a structure of functions of protective and localizing safety systems management that is implemented in safety control system procedures of Novovoronezh NPP-2 (Reactor facility V-392M) and Leningrad NPP-2 (Reactor facility V-491) designs.

Table 4.1 - Structure of functions of protective and localizing safety systems management implemented in safety control system procedures of Novovoronezh NPP-2 and Leningrad NPP-2

Novovoronezh NPP-2 (reactor facility V-392M)			Leningrad NPP-2 (reactor facility V-491)		
A. Reactivity control	B. Emergency heat removal from reactor core, heat removal from the primary circuit and spent fuel pool to the final absorber	C. Restraining of radioactive substances within the established limits	A. Reactivity control	B. Reactor core cooling	C. Prevention of radioactivity release
AA: Fission reaction shut down	BA: Coolant inventory maintaining in reactor	CA: Pressure limitation in containment	AA:Fission reaction shut down	BA: Water supply to reactor core	CA:Pressure limitation in containment
AB: Subcriticality provision in the stopped condition	BB: Heat removal from the primary circuit to the final absorber	CB: Confinement of fission products in containment	AB: Reactor power limitation	BB: Heat removal from the primary circuit	CB: Confinement of fission products in containment
AC: Reactor power limitation	BC: Heat removal from the primary circuit through steam generator	CC:Flow removal from inter-containment space with cleaning	AC: Subcriticality provision in the stopped condition	BC: Heat removal from the primary circuit through steam generator	CC: Confinement in annular region
	BD: Coolant inventory maintaining in steam generator	CD: Steam generator confinement in case of leakage of the primary circuit into the secondary one		BD: Coolant inventory maintaining in steam generator	CD: Confinement in steam generator in case of leakage of the primary circuit into the secondary one

Novovoronezh NPP-2 (reactor facility V-392M)			Leningrad NPP-2 (reactor facility V-491)		
A. Reactivity control	B. Emergency heat removal from reactor core, heat removal from the primary circuit and spent fuel pool to the final absorber	C. Restraining of radioactive substances within the established limits	A. Reactivity control	B. Reactor core cooling	C. Prevention of radioactivity release
	BE: Protection of the primary circuit from pressure exceeding and brittle rupture	CE: Confinement of the supporting systems		BE: Protection from pressurization of the primary circuit	CE: Confinement in auxiliary systems
	BF: Protection of the secondary circuit from pressure exceeding	CF: Hydrogen removal from containment		BF: Protection from pressurization of the secondary circuit	
	BG: Heat removal from spent fuel pond				

4.3 SAFETY FUNCTION «CORE SUBCRITICALITY»

The reactor core subcriticality at Novovoronezh-2 (B-392M reactor facility design) and Leningrad-2 (B-491) is provided by the following systems:

— Reactor scram (set of CPS drives);

— Make-up and boron control system (KBA);

— emergency boron injection system: JND (Novovoronezh-2)/JDH (Leningrad -2).

4.3.1 REACTOR EMERGENCY PROTECTION SYSTEM

Under normal operation the reactor power is controlled by changing the CRs working axial position in the core. The reactor power control is executed automatically following the signals of APC (Automatic Power Controller) or manually by an MCR Operator using the group control key.

In case normal operation violation, the reactor power is reduced automatically up to the permitted power level by actuation of APP (accelerated preventive protection) and/or PP-1 (preventive protection).

Under emergency, the SHEM type (Pitch electromagnets) electromagnets are de-energized by the signal of emergency protection actuators and 121 CPS CRs fall into the core by gravity force. The control rods are inserted within two to four seconds. The reactor emergency protection efficiency ensures the recriticality temperature lower than 100 °C – at any time of campaign.

Design and technical features of CPS actuators as well as AZ (fast acting emergency shutdown) actuating signals in B-392M and B-491 types are identical.

4.3.2 MAKE-UP AND BORON CONTROL SYSTEM - KBA

The make-up and boron control system- KBA is the safety important normal operation system.

The KBA active elements are energized from normal operation power supply systems and normal operation reliable power supply.

Since the reactor emergency protection is actuated, KBA provides the required boric acid content and concentration in coolant by means of primary circuit water exchange.

Novovoronezh NPP-2 (B-392M)

KBA system includes three centrifugal multi-step pumps with capacity of 80 m^3/h. At an Operator's choice one of three pumps connected in parallel is in operation, the second is in standby and the third one may be in maintenance. Since the reactor emergency protection is actuated, the boron concentrate system pumps provide boron concentrate (40 g/dm^3) to the suction of make-up pumps.

Leningrad NPP-2 (B-491)

The primary circuit make-up is provided by high capacity pumps (KBA20,30AP001) and low capacity pumps (KBA51,52,53AP001).

Each KBA20,30AP001 pump capacity is up to 80 m^3/h, note that one from the pumps connected in parallel is in operation (as chosen by the Operator), the second one is in standby.

Each KBA51,52,53AP001 pump capacity is up 9 m^3/h. One from the pumps connected in parallel (as chosen by the Operator) is in operation, the second is in standby and the third one may be on maintenance.

Each KBA51,52,53AP001pump has a frequency converter to control the flow rate from 1 to 9 m^3/h.

Depending on the reactor operating mode, the automatic controllers by means of the frequency converters maintain either the level in pressurizer, or the design flow rate for primary circuit make-up.

The control valves are mounted at KBA20,30AP001 pump discharge outside the containment which, depending on the reactor operating mode, automatically maintain either the design flow rate for primary circuit make-up or the level in pressurizer.

Intake headers of KBA make-up pumps are connected to deaerator during power operation, reactor start-up and shutdown and during "small" boron control with supply of boron solution or "pure" condensate to deaerator.

In water exchange modes (increase and decrease of boron acid concentration in the coolant) the suction line of KBA pumps is connected to JNK10,40BB001 tanks (16 g/dm^3) or JNK10,40BB002 (40 g/dm^3) or to pure condensate tanks KBC11,12BB001. The deaerator is isolated from the suction line of KBA pumps by stop valves.

In response to signals «Scram ("AZ")» or «Pressurizer level is minus 0,8 m» the suction line of KBA pumps is connected to JNK10,40BB001 (16 g/dm^3) tanks and the deaerator is isolated.

4.3.3 EMERGENCY BORON INJECTION SYSTEM JND (NOVOVORONEZH-2) AND JDH (LENINGRAD-2)

The functions performed by JND and JDH are identical:

— primary circuit pressure reduction to the estimated pressure of secondary circuit under the accident conditions of primary-to-secondary leaks (injection of boron solution into pressurizer steam section);

— increase of boron concentration in primary circuit and compensation of primary circuit coolant decrease when the Power Unit is brought to the safe state under the accidents with sealed primary circuit (injection of boron solution of high concentration into the reactor and pressurizer steam section);

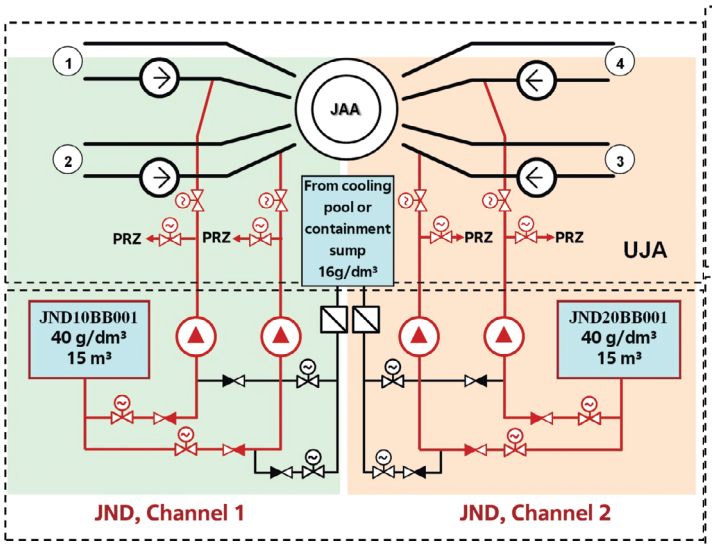
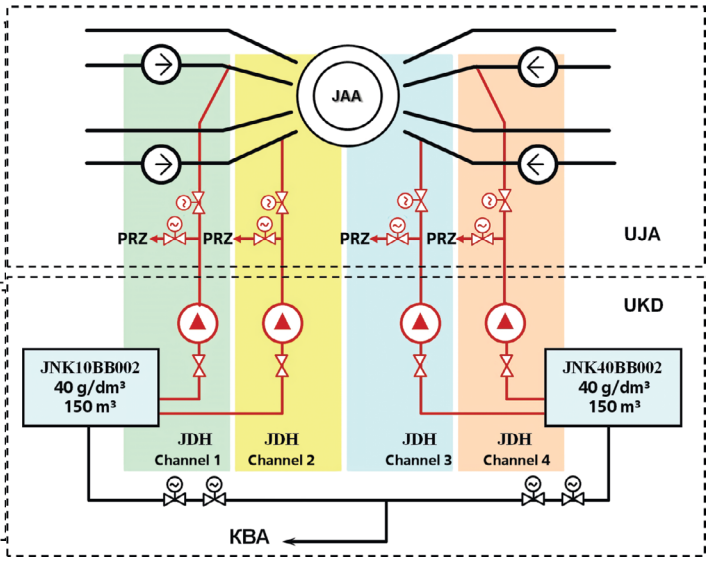
Under BDBA modes (DEC-A category) the emergency boron injection system performs the following functions:

— to bring the reactor facility in subcritical state in the modes with violation of normal operating conditions and emergency modes with ATWS failure of reactor emergency shutdown systems (injection of boron solution of high concentration into reactor);

— to maintain reactor facility in subcritical state and to compensate coolant decrease in cooldown mode with overlap of failure of normal operation systems.

Simplified process diagrams of JND and JDH systems are given in Table 4.2.

Table 4.2 – Comparison of emergency boron injection systems JND (Novovoronezh NPP-2) and JDH (Leningrad NPP-2)

Emergency boron injection systems JND (Novovoronezh NPP-2)	Emergency boron injection systems JDH (Leningrad NPP-2)
	
2 channels system with channel redundancy 2x100 %. Each channel consists of: — two legs with 50 % efficiency (pump capacity is 14,5 m^3/h); — one storage tank for boric acid solution (RBK) (40 g/dm^3) 15 m^3. In case of water reserve depletion in tank JND10(20) BB001 the pumps are switched over to supply the boric solution 16 g/dm^3 from the cooling pool or containment sump.	4 channels structure with channel redundancy 4 x 50 % (pump capacity is 14,5 m^3/h). 2 JDH channels have one common RBK (40 g/dm^3, 150 m^3). No possibility to switch-over the JDH pumps to other tanks. KBA system is also connected to the tanks. When JDH system is automatically actuated, KBA system is isolated from tanks (FSKU BA17 «Disconnection of Normal Operation System»).

Two boron concentrate storage tanks (40 g/dm^3) each of 150 m^3 are provided in Leningrad NPP-2 design, whereby KBA system is also fed from these tanks. As a result, there is the functional link of normal operation systems working at DiD levels 1 and 2 and safety systems working at DiD level 3.

Hanhikivi1 design (Finland), which is the further development of B-491 reactor type (Leningrad NPP-2) provides 4 storage tanks for high concentration boric acid (40 g/dm^3). Therefore, complete independence of JDH system channels from one another is assured: each channel is connected to its own borated water storage tank (RBK).

4.4 REACTOR COOLANT RESERVE MAINTENANCE - BA FUNCTION

The following systems are provided for the primary circuit make-up and maintaining the coolant inventory:

— KBA system;

— borated water storage system;

— emergency core cooling system система: active and passive ECCS.

4.4.1 MAKE-UP AND BORON CONTROL SYSTEM KBA

Under normal operation and under violation of normal operation modes the system is intended to maintain the primary circuit coolant inventory by return of identified leaks to the circuit, replenishment of non-identified leaks (compensated leaks) and compensating the coolant decrease as a result of the reactor cooldown.

4.4.1.1 Novovoronezh-2 (B-392M)

In case of primary circuit leaks, the KBA system maintains the level in pressurizer by changing rate of control valve opening at pumps discharge.

In case of non -compensated leak (Figure 4.3):

— KBA standby pump is put in operation;

— borated water 16 g/dm^3 is provided to KBA pumps suction part (by FAL system pumps);

— Scram (AZ) is actuated in response to pressure decrease above the core or decrease of pressurizer level;

— «$\Delta T_{s-1} < 8$ °C» trigger signal is generated to activate ECCS mechanisms and to localize the containment.

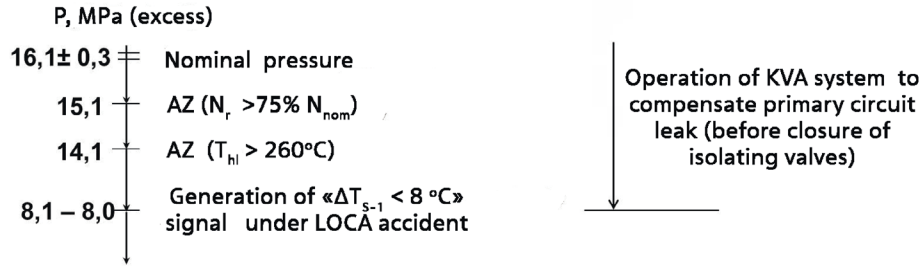


Figure 4.3 –Decrease of pressure above the core and generation of emergency signals in case of primary circuit leaks (LOCA)

4.4.1.2 Leningrad NPP-2 (B-491)

In case of primary circuit leaks, KBA system responds to the pressurizer level drop — high capacity pumps are turned on and the control valve at pumps discharge opens up, thus the pressurizer level is restored and maintained. Pumps suction is connected to containment sump tank with boron solution concentration 16 g/dm^3.

In case of non -compensated leak, «AZ» is actuated and the signal «ΔT_{S-1} < 8 °C» to trigger ECCS is generated (refer to Figure 4.3).

4.4.2 BORATED WATER STORAGE SYSTEM

4.4.2.1 Novovoronezh NPP-2 (B-392M)

This design does not provide the separate system for borated water storage. Under the modes with primary circuit leak, the ECCS high pressure and low-pressure pumps initially use the SFPl excess water inventory (not less than 750 m^3). Since this amount is pumped out, the ECCS pumps following the signal of SCSPP (Safety Control system by process parameters) are switched over to operation from the containment sump tank which by that time accumulates the required water reserve.

Such a solution has its advantages and disadvantages.

Advantages are as follows: borated water reserve temperature is maintained within the range 40 – 60 °C due to SFA heat removal and SF cooling system operation (FAK10-20).

Disadvantage : safety system pumps are to be switched over to containment sump under the low level in the cooling pool, which degrades the safety system reliability.

4.4.2.2 Leningrad-2 (B-491)

Two sump tanks of boron solution storage system JNK are located within the containment and provide storage of the required inventory of boron solution 16 g/dm^3. Each tank volume is 1000 m^3. The chosen structure (2 x 100 %) assumes that the system has no failures dependent on an initiating event.

The decision to locate the low concentration boric acid storage tanks in Reactor building at the containment low elevation but not in the Safety Building as provided in Tainwan NPP design, allows:

— to use them as sump tanks for operation of safety system pumps under DBAs;

— to exclude (under DBA) switching-over the safety system pumps to operate from the containment sump following the signal of low level in the tank located in the Safety Building, which improves safety systems operation reliability under accidents.

Water intake lines to supply borated water to the different Plant consumers are connected to each sump tank:

1. Four water intake lines (equal to the number of safety trains) are part of JNG-1 system and provide borated water supply to safety systems (JNG-1, JND, JMN). Water lines of first and second trains (JNG10 and JNG20) are connected to one JNK tank, water lines of third and fourth trains (JNG30 and JNG40) - to the second tank JNK.

2. Two water intake lines JNK supply borated water to the systems of normal operation and to the systems operating under BDBA (DEC modes):

— KBA system;

— KBB system → to ensure primary circuit filling-in during maintenance under normal operation mode and make-up of primary circuit system under DEC mode associated with coolant leakage;

— pure condensate delivery system (KBC-1) → to ensure delivery of make-up water into the reactor coolant system during cooldown under the shutdown conditions and for filling-in the hydro accumulators;

— Fuel pool cooling system (FAK) → to fill and feed the pool;

— Purification system of the fuel pool coolant and water of borated water storage tanks (FAL);

— Blow-off and drainage system of auxiliary building (KTC) → for tanks emptying.

3. Two lines of Core Catcher → to supply water to the Core Catcher compartment under the severe accident.

The isolating valves are installed in sequential order on each JNK pipeline which pass through the boundary of the containment.

Recirculation lines and safety system pumps tests lines (JNG, JND, JMN) as well as drains and overflow lines of cooling pool, internals inspection cavity, reactor vault and ECCS hydro-accumulators are connected with the sump tanks.

Installation of active filters in sump tanks structures is provided to ensure reliable operation of JND, JNG, JMN pumps under accidents with primary coolant leaks.

4.4.3 ECCS ACTIVE PART

4.4.3.1 Novovoronazh-2 (B-392M)

In B-392M design the system of emergency and planned primary circuit cooldown and fuel pool cooling (JNA) performs the function of emergency core cooling active system.

The system structure includes 2 channels (Figure 4.4), each including a high-pressure pump with ejector and one low pressure pump.

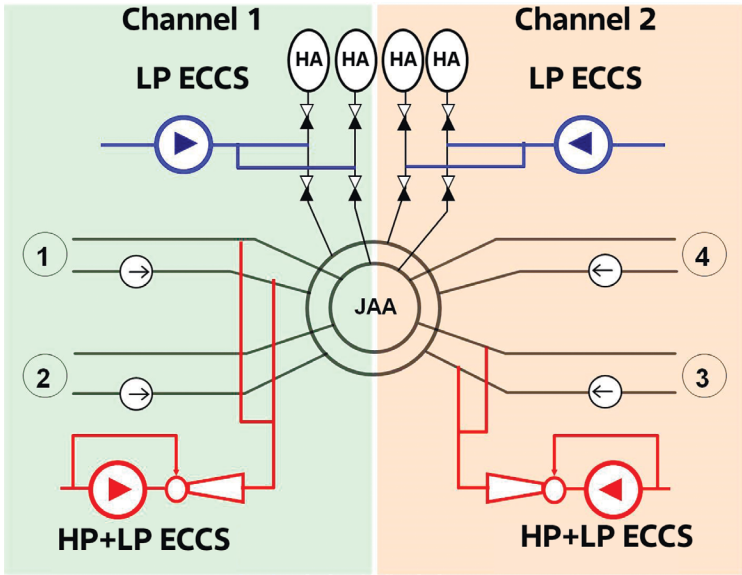


Figure 4.4 –JNA system structure in B-392M design

When required, the high-pressure ECCS pump-ejector may also perform functions of low-pressure ECCS pump. At the same time, high-pressure ECCS pumps have no such a reserve.

Under conditions of complete failure of high — pressure ECCS pumps (overlap of single and dependent failure), the reactor design features allow cooldown through the SG, which ensures timely decrease of pressure above the core up to 5,89 MPa and reactor make-up by operation of ECCS hydro accumulators and by ECCS low-pressure pumps when the pressure is less than 2,45 MPa – (Figure 4.5).

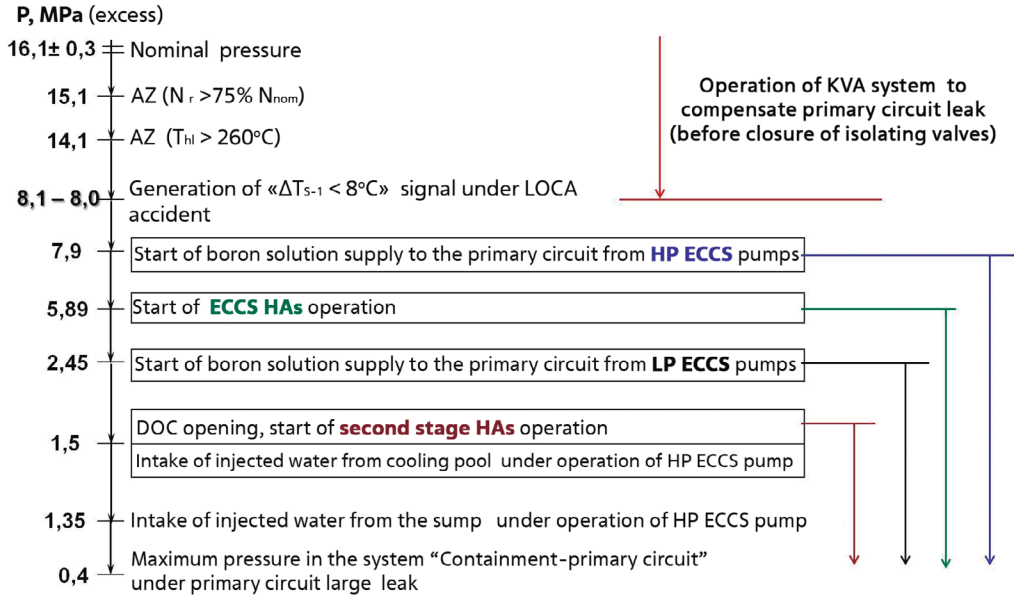


Figure 4.5 – Maintaining reactor coolant inventory under «LOCA» accidents (B-392M)

4.4.3.2 Kursk NPP-2 (VVER-TOI)

Combined high pressure ECCS pump-ejector is not used in Kursk-2 design. They use conventional multi-step centrifugal pump with B-491 head-capacity curve (refer to Table 4.4).

Active part of ECCS (JNA system) has two-channel active system with 2 x 100 % each. Low-pressure ECCS pump - as compared to Novovoronezh-2 - has no reserve inside the channel.

The design concept postulates a capability to back-up the execution of the same safety function by different sets of active and passive safety systems/elements.

In case of large LOCA, the reactor heat removal and core make-up can be done both by JNA low pressure system pumps and through functioning of the set of active

and passive systems: HP pump of JNA system jointly with Hydro Accumulators (HAs) of first and second stages. In case of "small" leaks, the core make-up is provided by HP pump of JNA system and reactor heat removal is done through the secondary circuit heat removal systems. Under the same mode the safety function can be performed without the high pressure pump owing to operation of HAs of the first stage at the initial step and by reactor quick cooldown by heat removal systems through the secondary circuit to decrease the parameters down to the setpoint when JNA low pressure pumps or HAs of second stage are connected to the primary circuit.

4.4.3.3 Leningrad -2 (B-491)

Emergency injection system of high pressure (JND) and low pressure (JNG-1) constitute the ECCS active part in B-491 design.

Low pressure safety injection system (JNG-1) is intended to inject boron solution from the containment sump tank to the reactor under LOCA accidents (including rupture of main circulation circuit, Dy 850) when pressure above the core goes lower than working parameters of JNG-1 system (2,45 MPa).

The system consists of four channels with 4 x 100 % redundancy.

The structure of each channel of the system includes (Figure 4.6):

— One centrifugal pump (10 кV);

— Two heat exchangers installed in parallel at the pump suction;

— valves;

— pipelines.

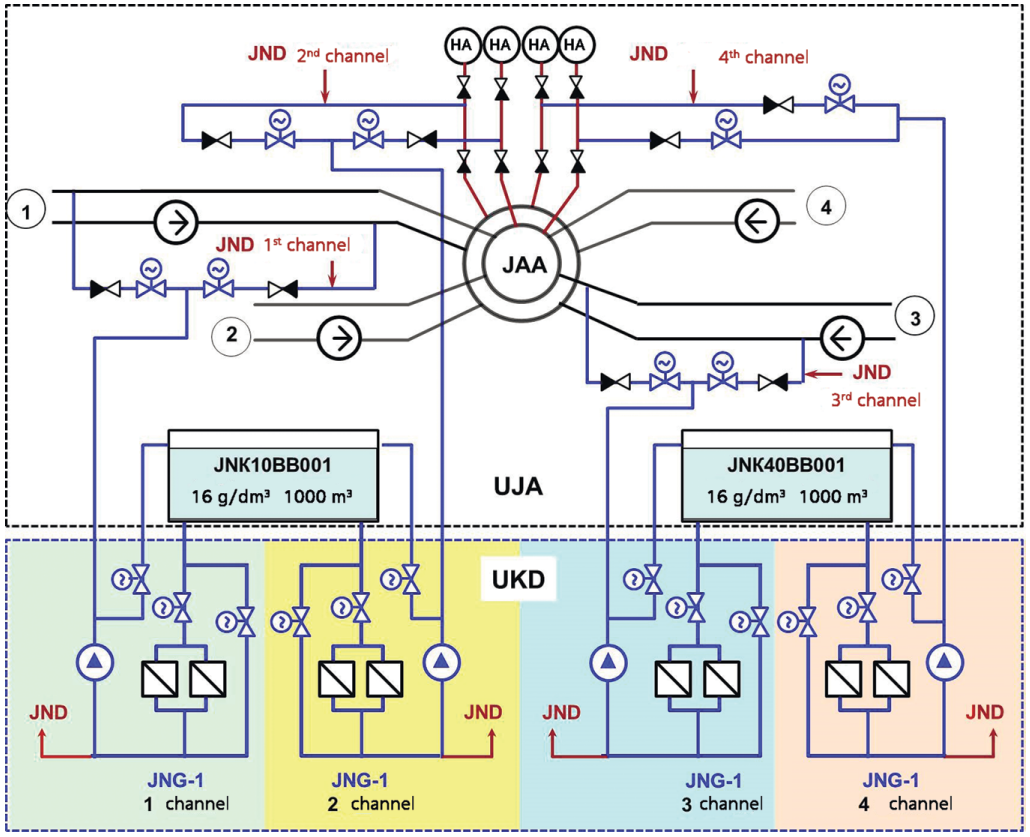


Figure 4.6 – Simplified process diagram of low pressure emergency injection system JNG-1 (UJA – reactor building; UKD – safety building)

Suction lines of each channel are connected to the containment sump tanks of borated water reserve of low concentration 16 g/dm^3 (JNK10,40BB001).

From the sump tank water enters the distribution header either through heat exchangers or through bypass line bypassing heat exchangers. The distribution header provides connection of suction lines of JNG-1, JND and JMN pumps.

Discharge lines of JNG10 and JNG30 channels are connected to the circuit loop legs JEC10, 30 in such a way that half of JNG-1 outputs are directed to “cold” leg and another half to the “hot” leg.

JNG20 channel connection is done between two check valves installed in the line coming from JNG50 and JNG60 ECCS HAs to the reactor, and channel JNG40 connection is done between two check valves mounted in the line coming from JNG70 and JNG8 ECCS HAs.

Under LOCA accidents the JNG-1 pumps are activated in response to the following emergency process signals:

— primary coolant boiling margin is less than 8 °C;

— containment overpressure is more than 30 кPa and there is no signal on secondary circuit leak inside the containment.

When JNG pumps are activated by emergency signals and when primary circuit pressure is higher 2,5 MPa, the pumps operate on the recirculation line. When pressure in primary circuit goes lower 2,45 MPa and the required output is achieved in the discharge line (refer to Table 4.3), the valves on recirculation line close down and the pumps start to deliver the boron solution from the sump tank to the reactor. Water delivery from each system channel is done simultaneously: both to “cold” leg and “hot” leg of the primary circuit system.

Table 4.3 JNG-1 pumps flow rate versus primary circuit pressure

Primary circuit back pressure, MPa	2,5	2,45	2,25	1,96	1,47	0,93	0,44	0
JNG-1pumps flow rate under supply from the sump tank, m^3/h	0	121	311	434	591	724	823	900

Initially, when water temperature in the sump tank is lower than 45 °C, JNG-1 heat exchangers are isolated from ECCS pumps but the valves on bypass line are open. When the sump tank water temperature rises higher than 45 °C (due to entry of the hot primary coolant), the pumps suction is automatically switched over from the bypass line to JNG heat exchangers: by opening the valves at heat exchangers inlet and closing them at bypass line.

Under minimum intermediate circuit water temperature 6 °C (winter season), temperature of boron solution cooled down in heat exchangers JNG10(20,30,40) AC001,002 and delivered to the primary circuit is not lower than 20 °C.

The JND high-pressure emergency injection system is designed to supply a solution of boric acid to the reactor in case of accidents with a loss of coolant exceeding the compensation capacity of the normal make-up system, at a pressure above the core below 7,9 MPa (Table 4.4).

Table 4.4 - Flow rate from the pumps of the JND system depending on the pressure in the 1st circuit

Primary circuit pressure, MPa	7,88	7,4	4,9	2,45	1,96	0,44	0
JND pumps flow rate under supply from the sump tank, m^3/h	0	24	156	215	225	253	260

In BDBA modes (category DEC-A), the system maintains the reserve of the primary coolant in case of large leaks, with the imposition of a common cause failure of the low pressure injection system: at a pressure in the 1st circuit of 0,44 MPa, the performance of three JND channels is equal to the performance of one JNG-1 channel (see tables 4.3 and 4.4).

The coolant supply is maintained by feeding the primary circuit with a solution of boric acid with a concentration of 16 g/kg and a temperature of at least 20 °C.

The system has a four-channel structure with 4 x 100 % channel redundancy.

Each channel includes (Figure 4.6):

– centrifugal, multistage pump (10 kV);

– fittings;

– pipelines.

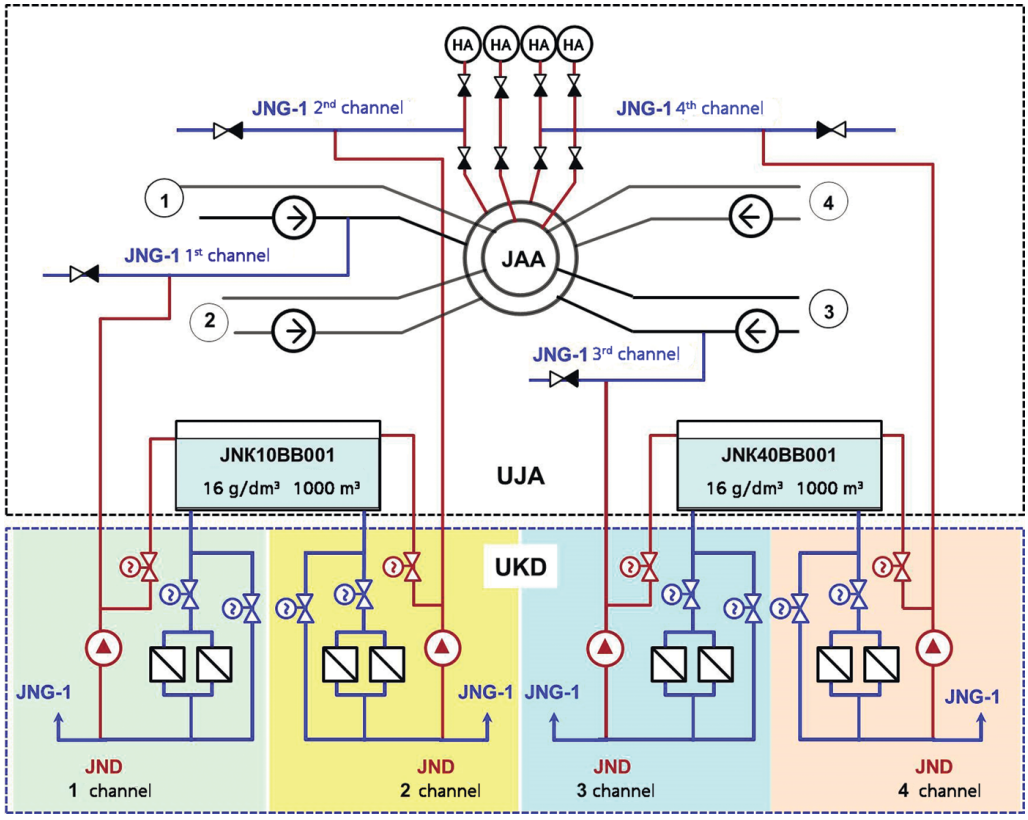


Figure 4.7 - Simplified process diagram of high pressure safety injection system JND (UJA – reactor building ; UKD – safety building)

Suction lines of each channel are connected through the pipelines and heat exchangers of low pressure safety injection system to the containment sump tanks of borated water reserve of low concentration 16 g/dm^3 (2 channels to one tank).

Discharge lines of the first and third channels are connected to “cold” legs of primary circuit loops through low pressure safety injection system pipelines. Discharge lines of the second and fourth channels are connected to the reactor discharge chamber through discharge lines of low-pressure safety injection system and ECCS HAs pipelines.

Therefore, all channels of high pressure ECCS in Leningrad-2 design supply borated solution only below the core.

4.4.4 ECCS PASSIVE PART

4.4.4.1 Novovoronezh-2 (B-392M)

The ECCS passive part includes the system of hydro accumulators of 1-st stage and core passive flooding system (HAs of 2-nd stage). Water inventory in 1-st and 2-nd stage HAs is sufficient (considering single failure principle) for repeated core flooding and decay heat removal during the period not less than 24 hours under the primary coolant leak of 850 mm equivalent diameter and failure of ECCS active part.

Throttling devices are provided on the ECCS HAs- Reactor connecting duct (Figure 4.8) to restrict the flow rate of water supplied from Hydro accumulators for core cooling during the period until the start of 2-nd stage HAs (116-th second).

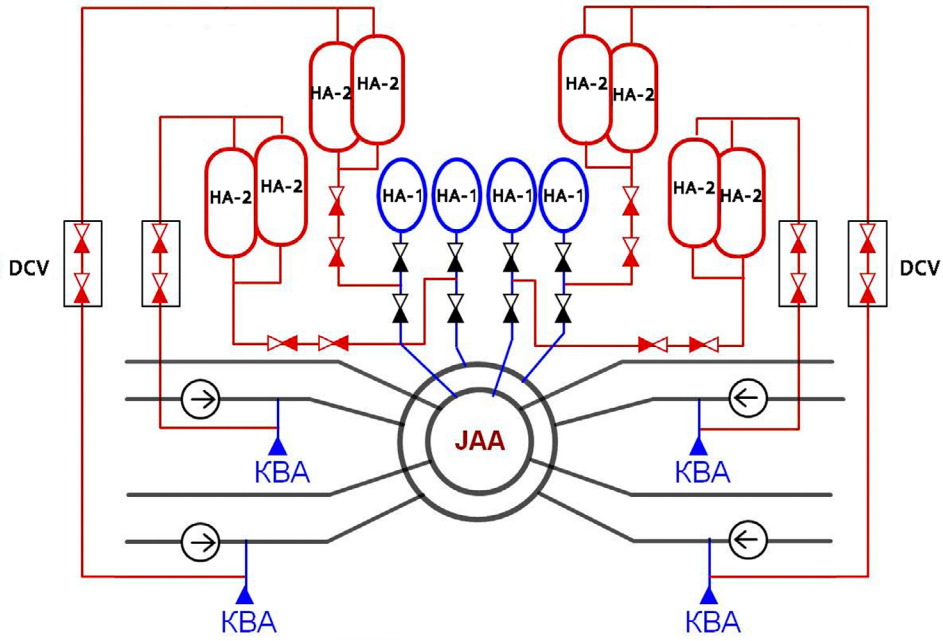


Figure 4.8 –B-392M. Passive ECCS simplified diagram

4.4.4.2 VVER-TOI design (Kursk-2)

Kursk NPP design was supplemented by hydro accumulators of the third stage (HA-3) in addition to HA-1 and HA-2 (Novovoronezh-2). HA-3 operation is required only under BDBA with complete failure of ECCS active part. The system is activated after depletion of boric solution reserve in HA-2 and provides reactor core passive cooling within the interval from 24 to 72 hours from initiating event onset.

The HA-3 system consists of 4 physically separated groups. Each group has three tanks of 60 m^3/h capacity each. Thus, the total amount of HAs-3 is 12 and the total boron solution inventory (16 g/dm^3) is equal to 720 m^3. Such HA-3 water amount is enough for residual heat removal during the period not less than 48 hours (considering SG PHRS is in operation). Each HA-3 group ensures the flow rate to the reactor not less than 1,6 kg/s and not less than 1,1 kg/s at the final stage.

The boron solution from HA-3 is delivered to the reactor through HA-2 system. The HA-3 equipment, valves and pipelines are located outside the containment in annex building UJA (in building UKA). The isolating valves are installed on HA-2 and HA-3 connecting lines.

The HA-3 tanks are emptied in turn in the following way: When isolating valves on HA-2 and HA-3 connecting lines become open, the excess pressure is generated in the first HA-3 tank (downstream). The first HA-3 tank is discharged into the discharge line of HA-2 system. After depletion of the first tank, the pressure rises in the second one and it is discharged into the rector through HA-2. The same procedure is repeated for the third tank.

The motor operated valves on HA-2 and HA-3 connecting lines are powered from Group 1 emergency power supply – from the battery designed for 72 hours of operation. As per the design, the battery can be recharged from alternative DG.

4.4.4.3 Leningrad-2 (B-491)

The passive core cooling system JNG-2 (ECCS HA) implemented at the Leningrad-2 is similar to Novovoronezh-2 system of 1-st stage HAs design in regard to structure, technical features and equipment design (Figure 4.9).

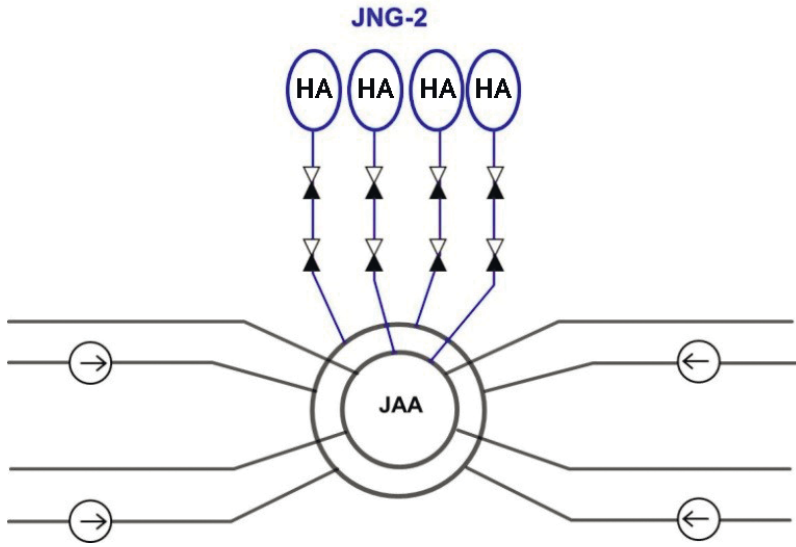


Figure 4.9 – B-491 Simplified diagram of passive ECCS

ECCS HAs provide delivery of boron solution into the reactor with concentration not lower than 16 g/dm^3 and temperature not lower than 20 °C under the primary pressure less than 5,9 MPa, in the amount sufficient for the rector core cooling before activation of low pressure safety injection pumps under MDBA (maximum design basis accident).

The design conservative assumption considers actuation of boron solution delivery by ECCS pumps to start on the 40-th second, therefore no orifice gages are installed on ECCS HA-Reactor connection lines, and ECCS HA discharge time under MDBA is approximately 60-100 seconds other than 180-190 seconds as in Novovoronezh-2 design.

Under BDBA with small coolant leak and CCF (common cause failure) of High-Pressure injection system (JND), ECCS HA provide make-up of the primary circuit up to the point when active low-pressure safety injection system (JNG-1) can be activated.

B-491 design does not include 2-n stage HAs. Therefore, ECCS passive part can't be used as a reserve for ECCS active systems under the large primary coolant leaks as in Novovoronezh-2 and Kursk-2 design.

For "Hanhikivi" NPP it was proposed to use the coolant storage system (KBB) as a diverse tool under LOCA with overlap of common cause failure of ECCS active part (JND and JNG-1).

For this purpose, Leningrad-2 design was modified as follows (Figure 4.10):

— additional lines for connection of KBB system pumps to primary circuit cold legs (JEC) were developed;

— power supply for KBB pumps was provided from independent emergency power supply system.

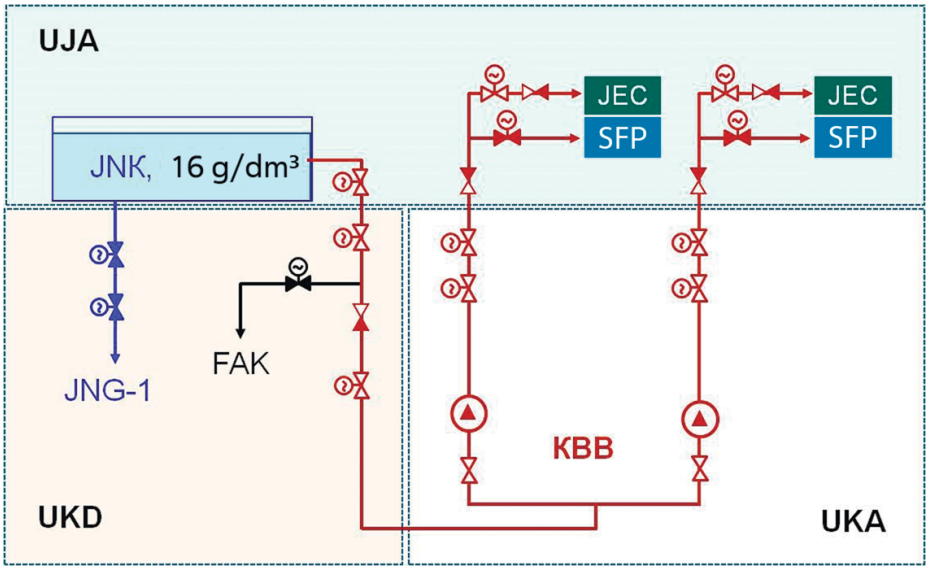


Figure 4.10 – Diagram of JNK tank sump switch-over from JNG-1 to KBB pumps under the primary circuit leaks and failure of ECCS active part (UJA – reactor building; UKD – safety building, UKA –auxiliary building)

As a result: under full failure of emergency power supply (NPP blackout with failure of DG of emergency power supply) the KBB pumps maintain their operability as they receive power from their own DGs.

Under CCF of ECCS active part, the diverse system of protections generates trigger signals to automatically start KBB pumps and to switch-over the JNK sump tank from JNG-1 to KBB pumps (Figure 4.10).

The KBB pump provides supply of the boron solution (16 g/dm^3) into the reactor with the flow rate from 20 m^3/h (under primary pressure ≈ 1,5 MPa) to 80 m^3/h (under primary pressure less than 0,5 MPa). For execution of BA function «Maintaining the primary coolant inventory» operation of one pump is sufficient, the second one is in standby.

Figure 4.11 presents the diagram of safety system operation to maintain the reactor coolant inventory under LOCA accidents (B-491 type).

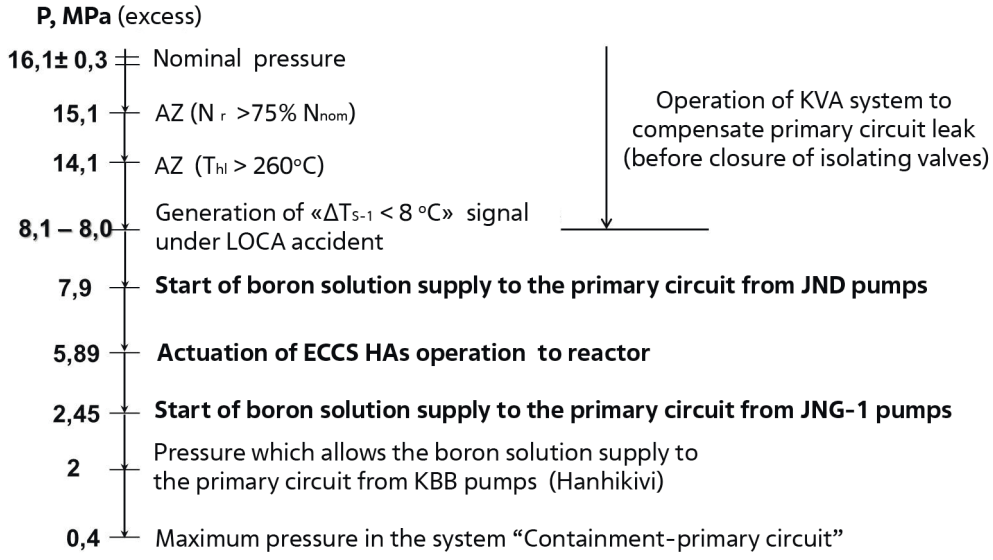


Figure 4.11 – Maintenance of reactor coolant inventory under LOCA accident (B-491)

4.5 PRIMARY CIRCUIT COOLANT HEAT REMOVAL– BB FUNCTION

AES-2006 design provides two ways for heat removal from primary circuit coolant
— through heat exchangers of ECCS active part (BB safety function);
— through steam generators (BC safety function).

4.5.1 NOVOVORONEZH-2 (B-392M)

Borated water storage tanks are united with the SF pool where the temperature is maintained within the range from 40 °C to 60 °C. Such a decision provides the boron solution supply to the primary circuit with a guaranteed temperature not lower than 20 °C with permanently connected heat exchangers of active ECCS.

BB control function provides for operation of cooling water support systems for heat removal to the ultimate heat sink in response to signal of active ECCS actuation:

— BB11 «Activation of intermediate circuit of essential consumers (KAA)»;

— BB12 «Activation of service water system of reactor compartment essential consumers».

4.5.2 LENINGRAD-2 (B-491). RESIDUAL HEAT REMOVAL SYSTEM JNA

Under DBA conditions with loss of primary circuit coolant, the heat exchangers JNG10(20, 30, 40)AC001 and JNG10(20, 30, 40)AC002, installed on the suction header of safety system pumps JND and JNG-1, provide cooling of the primary circuit coolant delivered from the containment sump tanks to the reactor (Figures 4.8 and 4.9).

Control function BB11 ensures connection of ECCS heat exchangers if temperature in the containment sump tanks is higher than 45 °C and disconnection if the temperature is less than 41°C. Such an algorithm excludes the possibility to supply boron solution to the primary circuit with the temperature less than 20 °C.

Under the DBA modes with the sealed primary circuit, the function of primary coolant heat removal from 130 °C to 60 °C is fulfilled by joint operation of low-pressure safety injection system JNG-1 and residual heat removal system JNA.

JNA residual heat removal system has 4 x 50 % channels. Two system channels ensure 84 MW heat removal under the intermediate circuit water temperature equal to 33 °C .

JNA system is connected to the primary circuit in the following way (Figure 4.12): JNA10, JNA20, JNA30, JNA40 suction lines are connected to the header which is connected by two pipelines to primary circuit loops № 2 and № 4, and the connection is done both to cold and hot legs of Main Circulation Circuit (MCC).

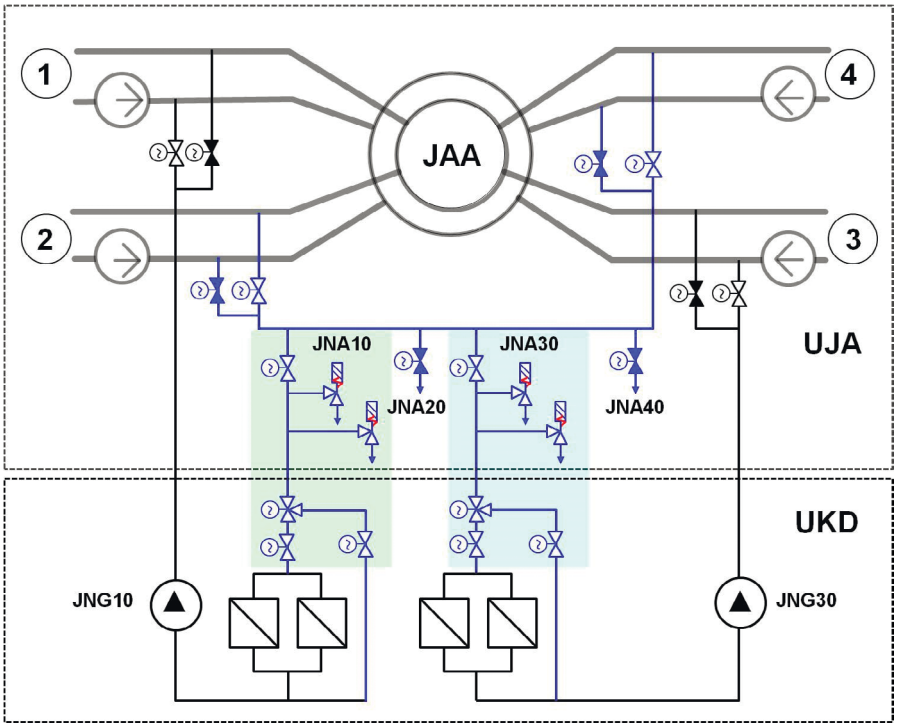


Figure 4.12 – Simplified diagram of two (of four) channels of residual heat removal system JNA (UJA – reactor building; UKD – safety building)

Pipeline D_u 300 of each JNA10,20, 30, 40 channels is connected to the pump suction line of the corresponding JNG-1 system channel. The connection is done either through the heat exchanger of emergency and planned cool-down or through its bypass. Such a decision allows to control the rate of primary circuit cool-down without changing the coolant flow rate through JNG pump.

Three-way control valve of JNA system provides:

— Reactor cooldown at the rate of 30 °C/h,

— Non-exceedance of difference between the temperature of the primary circuit input media and primary circuit coolant temperature 50 °C,

— Maintaining constant temperature of refueling pool water during core refueling.

JNA relief valves are intended for reactor overpressure protection under low temperatures (lower than 130 °C) and for protection of equipment and pipelines of residual heat removal system itself as well as the sprinkler system from overpressure.

The basis to identify the relief valves efficiency is:

— Two JNA channels are in operation;

— Three of four relief valves are available (failure of one valve in relation to single failure criteria);

— Two KBA20, 30AP001 pumps are in operation under fully open control valves at pumps discharge and three KBA51, 52, 53AP001 pumps with maximum efficiency.

Relief valves consist of two valves connected in series in single enclosure. Under normal operation the first valve (relief valve) is closed and provides overpressure protection. The second valve (isolating valve) installed after the first one, is adjusted for lower pressure, hence it is open. In case of violation of the first valve operation, when it is stuck, the second valve closes and thus provides protection against the uncontrolled LOCA accident.

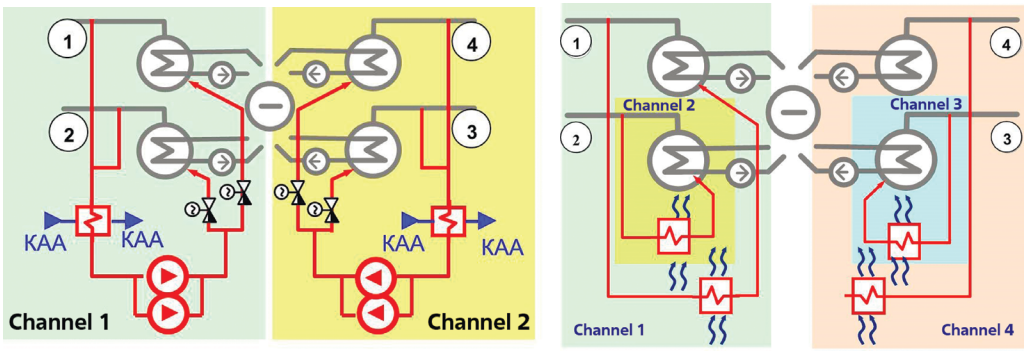
Each relief valve is provided with two independent control systems (open/close possibility) – by spring valve and electromagnetic valve.

4.6 SG HEAT REMOVAL SYSTEM (BC FUNCTION) AND MAINTENANCE OF SG COOLANT INVENTORY (BD FUNCTION)

AES-2006 design provides active and passive systems for the purpose of heat removal from the primary circuit through SGs.

4.6.1 NOVOVORONEZH-2 (B-392M)

In Novovoronezh-2 design the ACS SG (Automatic Control system) is the heat removal active system (SG - Emergency Cooling System) and PHRS SG – is the passive one. Both systems are of closed-circuit operation and do not require the SG constant make-up by demineralized water (Figure 4.13).



ACS SG. Two- channel structure with 2 x 100 % channel redundancy. Inside the channel — pumps redundancy. One channel for two SGs.	**PHRS SG.** Four-channel structure with 4 x 33,3 % channel redundancy. Each SG has its RHRS channel.

Figure 4.13 – Systems of residual heat removal through steam generators in B-392M design

Atmospheric air is the ultimate heat sink in these schemes:

— Indirectly via systems of intermediate circuit (KAA) and service water → for ACS SG;

— directly → for RHRS.

The particular design feature of ACS SG (Automatic Control System) for Novovoronezh NPP is one ACS channel operation for two SGs, which leads to interconnections between SG-1 and SDG-2; SG-3 and SG-4. These interconnections provide impact on the transients dynamics.

For example:

— SG pressure rise in case of spurious MSIV (BZOK Fast Acting Isolating Valve) actuation is abated since a portion of steam escapes to the nearby SG through SG-ECS pipelines;

— under different rate of control valves opening on the condensate return line, the coolant may be pumped over from one SG to another.

The particular design feature of Novovoronezh NPP PHRS-SG is that steam in heat exchangers is cooled by atmospheric air. Such a solution makes the heat removal process time — unlimited — upon availability of pressure in SG. From the other hand, the air cooling is less efficient against water cooling implemented at Leningrad-2. Moreover, the atmospheric air used for cooling makes the heat removal process dependent upon season conditions (outside air temperature), therefore it is not possible to adjust the system for cooldown for the determined rate (30 °C/h, 60 °C/h) compared to Leningrad-2.

In VVER-TOI (Kursk-2) the PHRS-SG capacity was enhanced as compared to Novovoronezh-2 due to increase of overall dimensions of RHRS heat exchangers (Table 4.5).

Table 4.5 – Comparison of Novovoronezh-2 and Kursk-2 PHRS-SG design

Parameter	Novovoronezh-2	Kursk-2
The number of heat exchangers in the system	8	8
Heat exchanger dimensions (length, width, height), mm.	8736 x 5010 x 2776	9822 x 5985 x 2689
One heat exchanger capacity, MW	8	9,6
Total RHRS capacity, MW	64	76,8

4.6.2 LENINGRAD-2 (B-491)

Leningrad-2 design provides 2 ways of primary circuit heat removal through SGs (Figure 4.14):

— by steam dump through steam relief systems (under overpressure in SG and primary coolant temperature more than 130 °C) and concurrent SG make-up by demineralized water (open circuit).

— with the use of PHRS-SG (closed circuit).

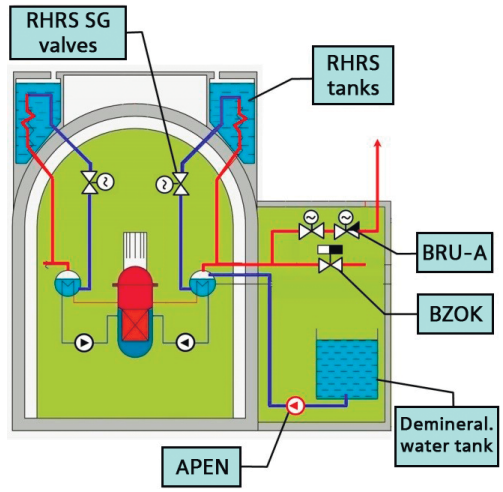


Figure 4.14 – Primary circuit heat removal system through SG in B-491 design

4.6.2.1 Live steam system within the nuclear island boundaries LBA-1

The live steam system within the LBA-1 nuclear island boundaries includes steam lines, main steam valve unit and isolating valves downstream BZOK (Figure 4.15).

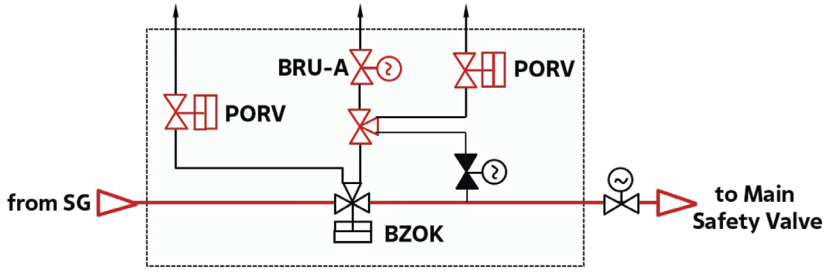


Figure 4.15 – Main steam valve unit

The main steam valve unit includes:

— BZOK — fast acting isolating valve;

— BRU-A — fast acting steam dump facility (to atmosphere);

— Isolating valve upstream BRU-A;

— SG two pilot — operated relief valves (PORV).

The LBA system performs normal operation and safety functions. The safety functions include:

— steam dump to atmosphere under the Power Unit emergency cooldown (BB function);

— SGs overpressure protection (BF function);

— isolation of steam line under accidents which require SGs localization (for example, primary-to-secondary leak – CD function).

The following is provided by the system for BB function fulfilment:

— Primary coolant heat removal at DiD level 2 by steam dump from the SG in dynamic modes through BRU-K or BRU-A with subsequent maintaining the design pressure;

— Primary coolant heat removal by SG cooldown through BRU-A with the rate 15 °C/h; 30 °C/h; 60 °C/h;

— Residual heat removal before actuation of passive heat removal system under BDBA.

BF function fulfilment provides SGs overpressure protection by BRU-A and/or SGs relief valves (refer to Chapter 4.8).

CD function provides localization of emergency SG by:

— Isolating valves upstream BRU-A;

— BZOK and/or isolating valves downstream BZOK.

4.6.2.2 SG emergency cooling down system

The SG emergency cool down system includes the emergency feedwater system and BRU-A. The system includes 4 x 100 % (Figure 4.16). For emergency cooldown one SG heat removal is sufficient.

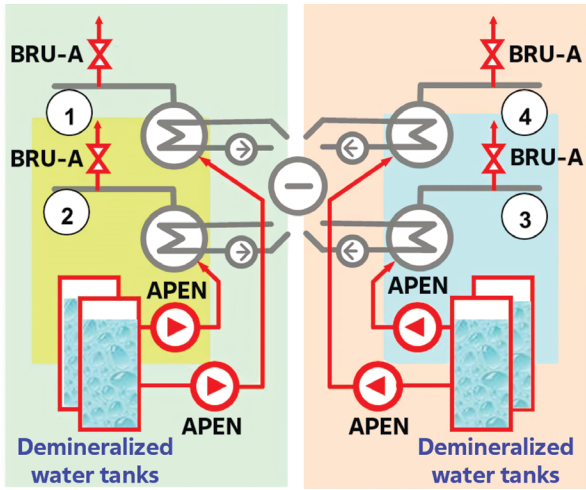


Figure 4.16 – Primary circuit heat removal through SG emergency cooldown system (APEN + BRU-A)

Reactor core heat removal is performed by discharge of steam generated in SG through BRU-A to atmosphere. The SG make-up is effected by pure condensate from the demineralized water tanks by means of Emergency Feedwater Electric Pumps (APEN). Air is the ultimate heat sink.

Emergency feedwater system (LAR/LAS)

Emergency feedwater system (LAR/LAS) is intended for feedwater supply to steam generators under the conditions of violation of normal operation and under DBA when feedwater supply from normal and auxiliary systems is not possible. The system operates under initiating events related to SGs water level decrease and which require emergency cooldown and maintaining the Power Unit in hot standby.

Under the mode of emergency make-up, the system provides water to SGs under the following backpressure values (Table 4.5):

Table 4.5 - APEN flow rate versus SG backpressure

SG overpressure, MPa	9,4	7,8	7,3
Emergency make-up flow rate, m^3/h	77	138	165

The system consists of four identical and fully independent channels LAR10, LAR20, LAR30, LAR40. The design provides water supply from each channel to one SG. SG water supply is arranged through dedicated emergency feed pipe connection.

The structure of each channel includes (Figure 4.17):

— Emergency feedwater pump LAS10(20,30,40) AP001;

— Valves;

— Pipelines.

The control valve is mounted on the discharge line to maintain the target level in the steam generator.

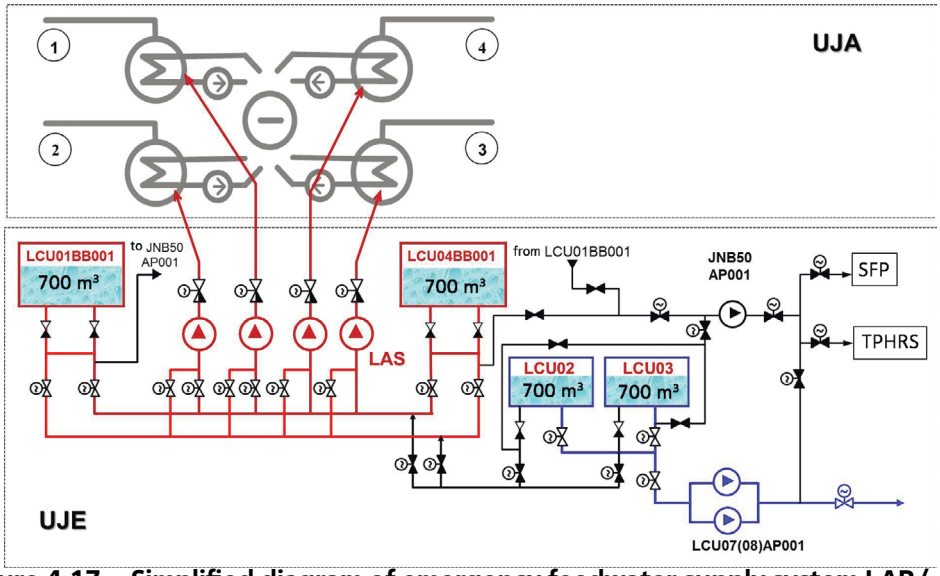


Figure 4.17 – Simplified diagram of emergency feedwater supply system LAR/LAS. Connection with LCU tanks and JNB50AP001 pump (UJA – reactor building; UJE – steam chamber)

The feedwater system tanks LCU01,02,03,04BB001 each of 700 m³ capacity are used for the demineralized water storage. And LCU01,04BB001 tanks are provided to supply demineralized water only for emergency feedwater supply systems. LCU02,03BB001 tanks are used for secondary circuit make-up under normal operation conditions, however, it is possible to connect them to suction headers of emergency feedwater pumps. The demineralized water target temperature is maintained in the tanks in the range of +20 °C to + 25 °C.

Under BDBA conditions LCU01,02,03,04BB001 tanks can be used for emergency make-up of PHRS tanks and cooling pool by JNB50AP001 pump.

Under normal operation conditions the LAR/LAS emergency feedwater system functioning is not required.

The system is actuated automatically by the SCSPP (Safety control system by process parameters) command in case of the following process signals concurrency:

— SG level is lower than менее (H_{nom} – 900) mm;

— Primary circuit temperature exceeds 150 °C.

Under the accidents with steam and feedwater lines rupture in the part which is not isolated from SGs, the inhibiting signal for operation of emergency feedwater system channel for failed SG is generated.

If the de-energizing signal is generated, the system pumps are turned on as per the step-by-step startup program of emergency power supply system DGs.

4.6.2.3 The passive heat removal system through the JNB steam generators

The passive heat removal system through the steam generators works due to the natural circulation and turns up to be the technical mean of the beyond-design accident overcoming.

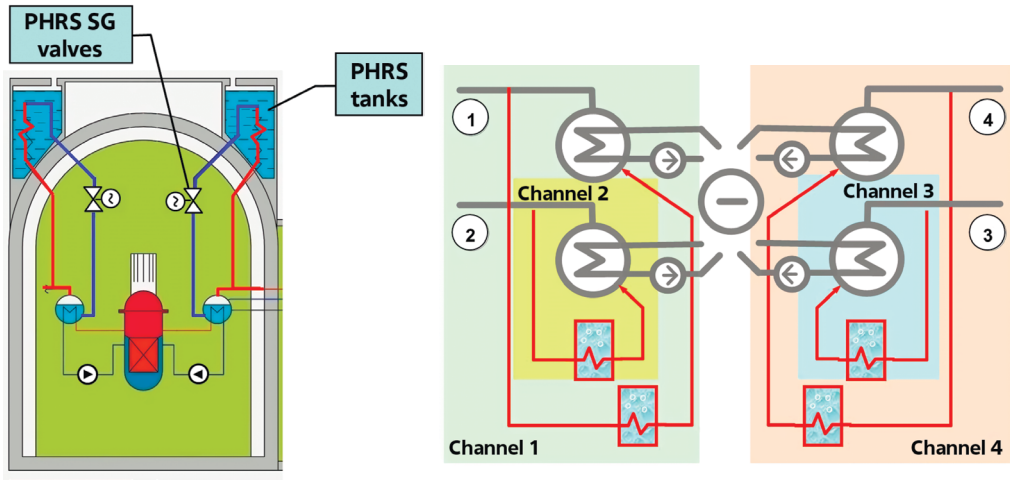
The system provides:

— the residual heat removal and reactor cooldown in a blackout condition of the NPP (with the diesel-generator unstart);

— the residual heat removal and reactor cooldown in the complete loss of feed water conditions (feed electric pump, additional feed electric pump and motor-driven emergency feed electric pump);

— the reserve to the safety active system (the atmospheric pressure relief valve BRU-A) in the event of its failure.

The simplified scheme of the system is shown in the figure 4.18.



The passive heat removal system through the steam generators (PHRS SG).

The four-channel structure with the reservation 4 x 33,3 %.

Each steam generator is provided by its own passive heat removal system channel.
Heat removal closed circuit.

Figure 4.18 - The simplified scheme of the passive heat removal system through the steam generators

The system consists of four totally independent from each other channels. The channels are dimensionally spread along the external protection cover and are separated from each other with the building constructions.

The system productivity is 4 x 33,3 %, it means that three operational circulation circuits JNB are enough for the system to fulfil its functions to the full extent at any accidental condition.

The simplified scheme of the system channel is shown in the figure 4.19.

Each channel of the PHRS SG includes:

— one emergency heat removal tank (EHRT);

— sixteen emergency cooling down heat exchanger sections;

— «major» starter valve (Dy100) и «small» starter valve (Dy50);

— isolating (cut-off) valves;

— steam and condensate pipelines.

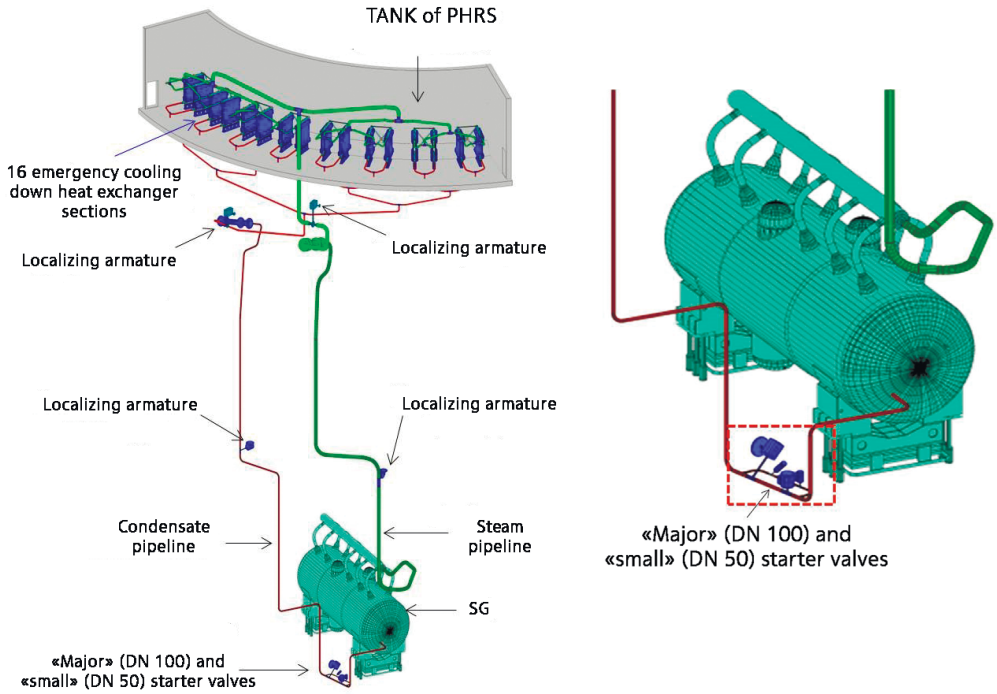


Figure 4.19 – The simplified scheme of PHRS SG channel

The emergency cooling down heat exchangers (ECDHE) are intended to heat transition from the steam-generator unit to the cooling water in the emergency heat removal tank. The heat removal from EHRT to the ultimate heat sink is accomplished through the water evaporation in the tanks during at least 24 hours from the accident startup.

EHRT is made as the reinforced-concrete covered stainless steel construction and placed in the separate facility of the circular reactor construction above +59,850 mark. The emergency heat removal tank position is higher than the position of the steam generators. This position provides the natural circulation in the PHRS SG contour.

EHRT's sixteen emergency cooling down heat exchanger sections of each channel are placed inside the emergency heat removal tanks of each channel.

Full water volume of the emergency heat removal tank in each JNB system channel is 538 m^3. The heat exchanging sections of emergency cooling heat exchange unit are placed under the water level (H = 5,54 m) in the lower part of EHRT. The total heat transfer surface of each out of four PHRS SG channels is ~ 226 m^2.

Each PHRS SG channel is connected to the steam and water generators with the compatible capacity. The PHRS SG steam pipe is connected to the steam collector SG from the side of its dumb but end through the special connecting leg. The condensate drain pipe from the PHRS precool exchangers is connected to the special connecting leg of the water region steam-generator. The end of the condensate circuit is lead under the water level at the depth of the lower third of the SG tube.

The system is thermal insulated. At the standby condition the lift line is heated until the secondary coolant circuit temperature level. In the standby condition the lift line is heated to the level of the secondary coolant circuit temperature as the steam path of the system is constantly connected to the steam pipe of the steam generator.

In the full power operation mode of the nuclear power unit the isolation technology of the steam supply pipeline and the condensate drain pipe is always in the open position. This provides PHRS SG from overpressure due to the safety valves of the steam generator. The cut-off devices are designed for the automatic proper shut-down of the PHRS SG emergency channel in case of loss of integrity of a circuit.

The «major» and the «small» starter valves are set up parallel to each other on the downflow pipe of each PHRS SG channel which are in front of the steam generators. At the standby condition the startup valves are closed.

The PHRS SG switching approach is based on the automatic starting opening of the valves: the «small» (electromagnetic) starter valve and the «major» (electrically driven) starter valve according to the system control algorithm.

At the accident with the station blockout the electromagnet power loss takes place and the «small» starter valve opens on the passive mode.

Under any other initial circumstances for the PHRS SG startup the activating signal flow from automatic process control system is required in accordance with the PHRS SG system control algorithm.

The «small» starter valve provides cooldown for the reactor unit equipment at a speed of «30 °C/hr» at the following conditions:

— all electric power-supply sources failure (blackout) at the current density of the primary coolant circuit;

— complete loss of feedwater, including the auxiliary feedwater system;

The «major» starter valve provides cooldown for the reactor unit equipment at a speed of «60 °C/h» at the conditions that require the accelerated cool down of the reactor facility:

— the leak from the nuclear plant primary system to the secondary system with the failure of all atmospheric pressure relief valves BRU-A at the non-emergency steam-generators;

— the leak from the nuclear plant primary system with the failure of all channels in emergency core cooling system of high pressure (ECCS HP) and BRU-A (BRU-K).

The system output time for the maximal power after full opening of the «small» starter valves is ≈ 80 s, for the «major» starter valves is ≈ 40 s.

The maximal power of one PHRS SG channel in the available by this project modes at the water temperature of 30 °C in the emergency heat removal tank is

— after opening of the «small» starter valve is ≈ 28 MW;

— after opening of the «major» starter valve is ≈ 52 MW.

We shall consider the system working using an example. In case of the failure of all the AC power sources upon condition that the integrity of the primary circuit is saved: the reactor protection system is activated. The pressure rise in the secondary system lead to the activation of the atmospheric pressure relief valve BRU-A (or it may lead to the single activation of Stem Generator Pilot Operated Relief Valve in case of the failure of the atmospheric pressure relief valve BRU-A). The PHRS SG automatic switching-in occurs upon condition that the diesel-generator doesn't start at the reactor unit blackout by means of the «small» starter valve opening. The atmospheric pressure relief valve BRU-A closes after the residual heat level is equal to the power of the PHRS SG heat output.

The EHRT cooling water heatup begins until boiling. As a result of the steam generation in the PHRS Tank capacity the gradual water level lowering begins in the tank. The EHRT stored water capacity in three channels of PHRS is enough for the reactor shutdown cooling down during 24 hours from the beginning of the accident.

The initial refill and the periodical flooding of the emergency heat removal tanks are accomplished from the LCU02,03BB001 tanks with the help of the emergency feedwater makeup pumps LCU07,08AP001 (Figure 4.20).

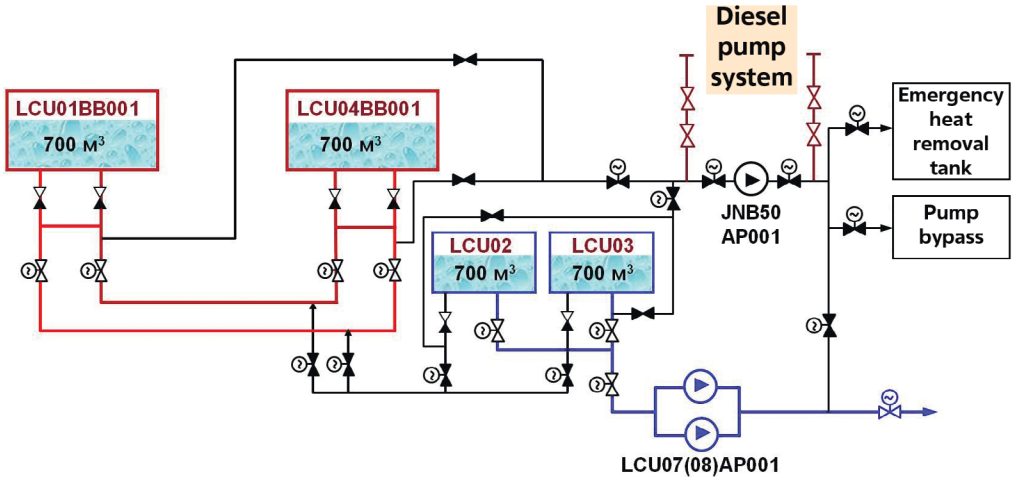


Figure 4.20 – The scheme of the initial refill and the EHRT safety injection

For the emergency heat removal tanks refilling after 24 hours after the beyond-design accident with the full failure of all AC sources (blackout), a low powered high-pressure pump JNB50AP001 is provided, which is placed in the steam cell and connected to the LCU system tanks. The electric power supply of this pump is provided by the beyond-design accident electric power supply channel which is oriented to the isolated work of 72-hours (without regard to the JNB50AP001 pump work). The channel has an opportunity to be connected to the mobile diesel-generator for the rechargeable batteries makeup and further system operation.

With the purpose of the additional mean which doubles the refilling function of the emergency heat removal tanks, during the beyond-design accident with the full failure of all AC sources, there is a high pressure diesel pumping unit (DPU), which is connected to the special pipelines in the pump bypass JNB50AP001.

In the «Hanhikivi» NPP project the capacity of the chemically demineralized water (CDW) tanks is increased from 700 m^3 до 1000 m^3. Moreover, the emergency feed water pump JNB50AP001 is duplicated – in parallel two pumps are set: a working one and an auxiliary one.

4.7 THE NUCLEAR PLANT PRIMARY SYSTEM PROTECTION FROM THE OVERPRESSURE AND THE BRITTLE DESTRUCTION FAILURE IN THE PROJECT V-491 (FUNCTION BE)

There are technical facilities which keep non-exceedance of pressure for the integrity protection of the nuclear plant primary system. The nuclear plant primary system protection from the overpressure is provided:

— by the venting process from the the nuclear plant primary system to the relief tank or to the containment envelope;

— by the steam condensation in the steam zone;

— by the urgent abstract of heat from the primary coolant through the secondary coolant circuit.

During maintaining natural circulation and during the operational occurrences related to the failure of the reactor plant heat removal system, the pressure limitation control in the nuclear plant primary system is provided by the recirculation spray into the reactor pressurizer using the reactor coolant pump set or the pumps of the KVA system if the reactor coolant pump set spray becomes ineffective.

Under conditions of the design-basis accident related to the primary coolant leak to the secondary system the pressure suppression in the nuclear plant primary system is provided by the reactor primary system cooldown at a speed of 60 °C/hr and the pressurizer spray into the steam zone by the pumps of the JDH emergency boron injection system.

For the venting process from the nuclear plant primary system in the V-491 project the following technical means are provided:

1. The pressurizer safety valves. They are operated: by the means of the engineered safety feature accusation system, by the spring; through the additional control line - the main control room operator or the auxiliary control room operator.

2. The safety valves of the JNA residual-heat removal system: at the tempeartures of the reactor coolant of the primary circuit below 130 °C. They are operated by engineered safety feature actuation system, by spring, by the main control room operator or the auxiliary control room operator.

3. The emergency gas removal system. It is controlled by the main control room operator or the auxiliary control room operator.

The pressure safety valves of the pressurizer protect the nuclear plant primary system equipment from the high pressure. The structural redundancy of the pressurizer pulse safety device: 3 x 50 %. The quantity of the installed pilot operated safety devices is determined according N + 1 (where N - is the quantity of pulse safety devices, which provide necessary discharge rate for the pressure decrease), it guarantees that the nuclear plant primary system pressure will not

exceed the operating pressure by more than 15 % even if the operated safety devices break with the minimum set point for the opening.

Each operated safety device represents a building-block design and has the additional control line for the forced opening operation for the pressurizer safety-valve by the main control room operator or the auxiliary control room operator.

The safety-valves in the JNA residual heat removal system

During the unauthorized rise of pressure in the nuclear plant primary system which is higher than the proposed pressure at low temperatures and in the circumstances of a break-down danger of the nuclear plant primary system because of the brittle fruction the protection function for the nuclear plant primary system (together with the pressurizer safety valves) is layed on the safety-valves of JNA system.

The emergancy gas removal system is analogous to the same system in the project of Novovoronejskaya NPP-2. The KTP reactor system provides the opportunity of steam gas mixture release to the relief tank or in the containment volume of a nuclear power plant: from the reactor, pressurizer, internal spray heaters of the steam generators. The system consists of the pipe lines and the equipement. All equipement framework is duplicated (2 x 100 %). The steel framework control is distant — by the main control room operator or the auxiliary control room operator.

The control procedure of the hydrogen situation in the containment envelope of a nuclear power plant during the reactor severe accidents requires the opening of the equipement framework system of the emergency gas removal and pulse safety device of the pressurizer by a signal when there is an exceed temperature over 400 °C above the active zone. The reactor destruction is impossible at high-pressure.

In «Hanhikivi» NPP project there is an emergency pressure decrease system JEG-30 (figure 4.21) in addition to the above mentioned ones.

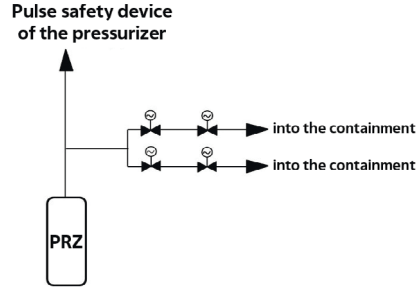


Figure 4.21 – «Hanhikivi» NPP. The emergency pressure decrease system JEG-30

The system is aimed for the pressure decrease in the reactor till 1 MPa in the reactor vessel distruction modes. The isolation valves opening is carried out by the main control room operator or the auxiliary control room operator.

4.8 SECONDARY CIRQUITSYSTEM PROTECTION FROM THE OVERPRESSURE IN THE PROJECT V-491 (FUNCTION BF)

The nuclear plant secondary system protection from the overpressure prevents the pressure rise below the allowable value in the steam-generators and in the main steam supply piping. The system elements are the parts of the main steam valve units which are set separately to each pipeline.

In the project V-491 each system channel consists of:

— the atmospheric pressure relief valve BRU-A with the check valve in front of it;

— the control steam generator pilot operated relief valve;

— the operating steam generator pilot operated relief valve.

The algorithms and protection set points from the over pressure of the nuclear plant secondary system are identical in the Novovoronezh NPP-2 and Leningradskaya NPP-2 (please refer to figures 2.1.12.5 and 4.22).

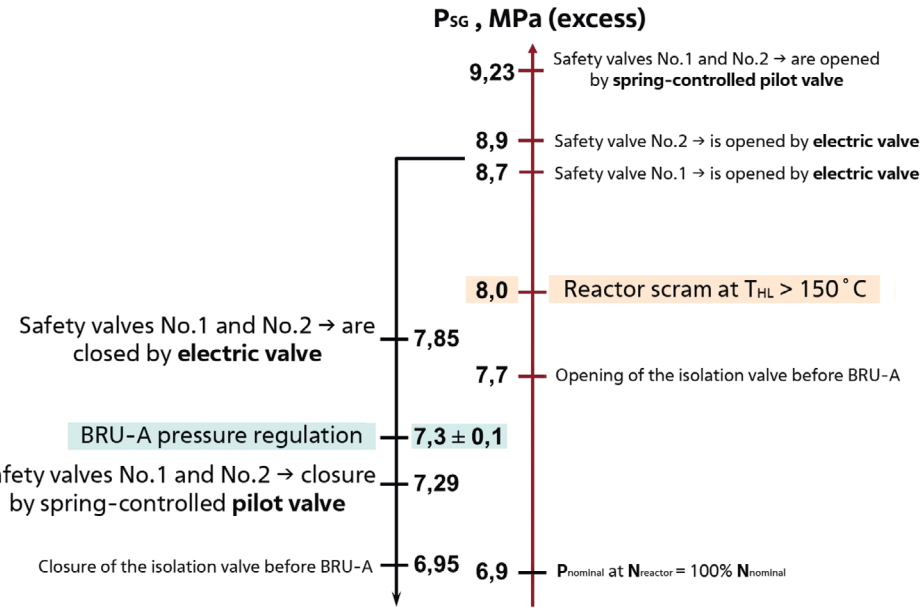


Figure 4.22 - Scheme of Set point of the secondary circuit from overpressure

4.8.1 THE ATMOSPHERIC PRESSURE RELIEF VALVE BRU-A AND THE CHECK VALVE BEFORE THE ATMOSPHERIC PRESSURE RELIEF VALVE BRU-A

The atmospheric pressure relief valve BRU-A is meant for the air discharge to the atmosphere for the purpose of fluctuating operational state prevention of safety-valves in the sharp load release modes and in the station blackouts conditions, also with the purpose of the steam discharge for the residual heat removal and the reactor system cooldown when there is a violation of the standard environment and in the case of the design basis accidents.

The protection function from the overpressure is provided by the co-working of the atmospheric pressure relief valve BRU-A and the check valve in front of it.

The check valve is the valve in front of the atmospheric pressure relief valve BRU-A. It represents the valve with the piston drive operated from its own working substance. The control is provided by two motor-operated valves mounted on the primary-valve body. The check valve opening time is less than 5 seconds.

When the unit is operating on power and there are no signals from the engineered safety feature actuation system the BRU-A check valve is 50 % opened. The check valve in front of the BRU-A is closed. The upper and the lower drive piston chambers are pressure — loaded by the system. The closed position of the valve is provided by the piston surface area variation in the upper and the lower chambers.

In case of the pressure rise in the steampipe up to 7,7 MPa the control valves open, the upper piston chamber of the check valve drive discharges to the atmosphere through the pulse valve. Then the major valve opens due to the system pressure.

In such case the BRU-A supports the pressure maintaining conditions at (7,3 ± 0,1) MPa.

At the pressure decrease till 6,95 MPa the control valves close, the upper drive piston chamber is loaded by the system, the check valve closes, the BRU-A casts back. With the objective of ensuring the function performing two electromagnetic impulse valves are provided on the major check valve.

The BRU-A saturated steam transfer capacity is 1000 tons per hour under a pressure of 7,7 MPa.

4.8.2 STEAM GENERATOR PILOT OPERATED RELIEF VALVE (SG PORV)

Two following impulse valves with different operating principles are provided on each PORV for the variety purpose and for increasing the reliability of function execution of SG PORV:

— the electromagnetic impulse valve which works due to the signal from the engineered safety feature actuation system (ESFAS);

— spring pulse valve with direct action.

In accidental conditions related to the overpressure in the secondary circuit when the SG PORV set point is actuated, the opening of the valves take place. When the pressure quantity of 8,7 MPa is reached the control steam generator safety valve opens after the signal from ESFAS. Both valves close when the pressure in the secondary circuit decreases till 7,85 MPa. When the pressure rises till the operate value the main valve piston chamber is loading through the impulse valve and it opens through the effort that occurs as the result of the shift piston surface area differences.

The PORV saturated steam transfer capacity is 1050 tons per hour under a pressure of 8,7 MPa.

For the valve closing the drive piston chamber discharges into the atmosphere through the impulse valve so the major valve closes under the action of the system pressure.

4.9 COOLING DOWN POOL HEAT REMOVAL– FUNCTION BG

4.9.1 NOVOVORONEZH NPP-2 (REACTOR FACILITY V-392M)

In the project the safe operation limits are provided at the cooling down pool water temperature (80 °C) and the water level in the cooling down pool (6 meters when in full production).

At the design basis accidents and non-design basis accidents water boiling in the cooling down pool may be admitted.

The following technological systems are provided for cooling the spent fuel assembly in the cooling down pool (Figure 4.23):

— FAK — is the twin duct cooling system of the cooling pool. The heat exchanger of FAK10,20AC001 system is used for the heat abstraction from the cooling pool.

— JNA – is the twin duct cooling system of the emergency and planned cooling down of the primary circuit and for cooling down of the cooling pool. For the heat removal from the cooling down pool JNA system heat exchanger is used.

For the cooling pool level control the safety injection lines (Dy50) are used from FAK10,20 system (in response to a signal from the engineered safety feature actuation system) or from JNA system (controlled by an operator from the main control room).

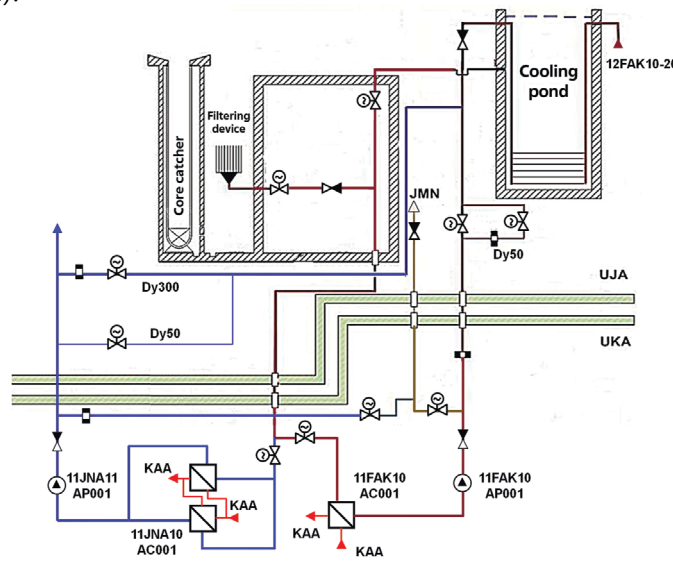


Figure 4.23 – The simplified technological scheme of the first channel cooling down pool system (11FAK10 and 11JNA)

During the heat removal from the cooling pool the ultimate heat sink is the air according to the following scheme:

— the heat-exchangers of FAK and JNA systems are cooled down by the component cooling loop water of the essential consumer KAA;

— the heat-exchanger of the component cooling loop KAA is cooled down by the plant service water system of essential consumer;

— in the spray cooling pool the plant service water is cooled down by the air.

The concept of normal operation functions adjustment is used for FAK system (cooling down of the cooling pool) together with the safety functions (used as the sprinkler system under conditions of the accidents related to the leak in the primary and the secondary circuits in the containment).

At the accidents with localization in the containment the removal of heat from the cooling down pool is carried out in the vapourizing mode. In this case the cooling pool make-up is carried out by FAK system (which simultenuosly takes the role of the sprinkler system). In case of the FAK system breakdown the operator has an opportunity to feed off the fuel tank by the emergency and planned low-pressure cooling down pumps. In both cases the make up takes place through the Dy50 line at rate 20 m^3/h.

At the accidents with total failure of AC sources (blackouts and unstart of all diesel generators) the staff is required to fulfil the following actions: within 24 hours from the beginning of the accident it is obligatory to bring into action the alternative component cooling loop and ensure water supply into the cooling pool from the sump of the containment with the temperature less than 70 °C at rate ≈ 50 m^3/hr from the first channel of FAK system.

4.9.2 LENINGRAD NPP-2 (REACTOR FACILITY V-491)

The following temperature criteria are provided by the project in the spent fuel storage pool:

— not more than 50 ºC at NPP power operation and during scheduled refuelling;

— not more than 60 ºC at scheduled refuelling and full discharge of all reactor fuel assemblies of the reactor core into the cooling pool;

— water boiling is acceptable in the modes with the loss of the authorized cooling equipment of the cooling pool.

Less than 8 meters level decrease in the cooling pool is not allowed.

The following technological systems are provided for the heat removal from the cooling pool (Figure 4.24):

— FAK10 (FAK40) cooling down system for the cooling pool. The FAK system heat exchanger is used for the heat removal from the cooling pool.

— The second and the third channels of JNA /JNG-1/JMN systems. The NG20(30) AC001 system heat exchanger is used for the heat removal from the cooling pool.

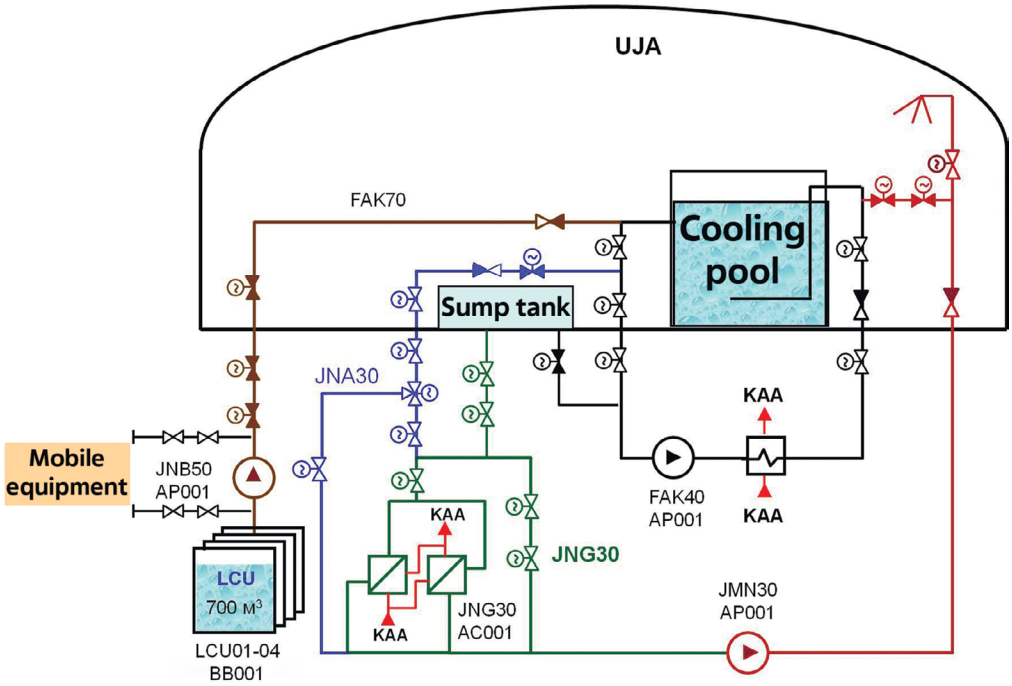


Figure 4.24 - The simplified scheme of the fuel pool cool: channel FAK40, channel JNA30 → JNG30 → JMN30. The safety injection of the cooling pool at line FAK70.

The FAK fuel pool cooling system is the system of the normal operating system and consists of two channels FAK10 and FAK40.

For the cooling function of the fuel pool in the power operation mode of the NPP one FAK channel operation is enough.

For the cooling function of the fuel pool in the planned reloading mode two FAK channels operation is enough.

Two FAK channels operation is enough during the full core unloading into the fuel pool. In case of FAK channel breakdown one FAK10(40) channel together with JMN/JNG30(20) channel operation is enough

Two FAK channels operation is enough during the emergency core unloading into the fuel pool. In case of FAK channel breakdown one FAK10(40) channel together with JMN/JNG30(20) channel operation is enough.

For the fuel pool cooling the channels connection of JNG20,30, JNA20,30, JMN20,30 systems is allowed only through the pipelines of the non-operated channel FAK10 or channel FAK40. The concurrent operation of three channels

FAK10, FAK40 and JNG20(30), JNA20(30), JMN20(30) is not allowed.

Upon the signal «P_{GO} > 30 kPa»:

— FAK system isolates from the containment and begins operating never-ending circle for cooling the cool pool;

— the isolating valve of JNA system loses and isolates the spray pump suction from the cool pool. The same way closes the isolating valve on the water supply line into the cool pool from the JMN pumps. The spray pumps begin working from the sump-tank onto the spray burner of the containment.

At the cool pool level less than 8,4 meters the system FAK isolates from the cooling pool by closing the isolating valve nearest to the cool pool.

At the blackouts the earlier operating FAK pumps begin operating again in accordance with the Diesel-Generators Step by Step Programme and the system operates the same way as in the normal operation mode.

During the beyond-design accidents FAK system is used in its direct function if the system KAA, the air discharge purification system and the power supply system operate.

The KAA component cooling loop system failure leads to the loss of the cooling equipment efficiency of the cooling down pool. In this case the residual heat removal of the reactor fuel assembly in the fuel pool is performed by water expulsion in the cooling down pool. For preventing SFA denudation in such circumstances it is obligatory to provide the make-up fuel supply of the cooling down pool.

In the project the make-up fuel supply of the fuel pool is provided from the sump tanks of the containment using the pumps of the following systems:

— **JMN** sprinkler system (the pumps are energized by the emergency power system - EPS);

— **FAK** cooling system of the cooling pool (the pumps are energized by the emergency power supply system EPS);

— **FAL** fuel-pool cleanup system (the pumps are energized by the uninterruptible power system of the normal operation).

At the accidents related to the failure of all sources of the AC current for the period more than 24 hours the safety injection is provided for the fuel pool at the following lines:

— four tanks LCU01-04BB001 the capacity of 700 m^3 each;

— high pressure pump JNB50AP001 (the pump is energized by the electrical power supply channel for the beyond-design accident control tools);

— FAK70 interface piping with foot valve, isolating and shut-off valves (at the normal operation conditions and during the design basis accidents the valve is closed on the line FAK70).

In case of JNB50AP001 pump failure the opportunity of connecting mobile high pressure diesel pumping system is provided.

4.10 FUNCTION CA: PRESSURE LIMITATION CONTROL INSIDE THE CONTAINMENT IN THE PROJECT LENINGRAD NPP-2 (REACTOR FACILITY V-491)

Project V-491 provides for the derating factors inside the containment in the circumstances of the design basis accident and beyond design basis accident the following technological systems:

— JMN sprinkler system;

— passive heat-removal system for the reactor containment JMP.

4.10.1 SPRINKLER SYSTEM JMN

The sprinkler system serves a function for the normal operation and safety function in case of the design basis accidents and it serves safety function for the severe accidents.

In the environments of the design basis accidents the sprinkler system provides:

— temperature fall and pressure supression in the containment under conditions of the accidents related to the leak of the primary coolant and the secondary coolant (boron solution spray into the containment for the steam gas mixture);

— discharge of the fission products from the containment atmosphere;

— chemical control of water in the sump tanks of the containment for the long-time holding of iodine and for the corrosion prevention of the construction materials under conditions of the accidents related to the leak of the primary coolant circuit.

Under the conditions of the beyond design basis accident related to the loss of a fuel pool coolant system (FAK), система JMN system keeps the reserve for the fuel pool coolant system for the time required for the operational cooling system by-pass redeveloping of the fuel pool of the FAK10(40).

Under the conditions of the beyond design basis accident related to the reactor core melting and the corium material output beyond the reactor vessel, the backfilling of the internals inspection cavity from the sump-tanks of the containment is carried out by the system pump JMN20 (after the power restoration).

JMN system has the structure: 4 x 50 %. In each channel the following items are provided:

— sprinkler pump;

— internal spray sprinkler header with spray nozzles;

— pipeline control valve;

— interface piping.

The suction pipelines of each system channel are connected through the low-pressure safety injection pipelines to their sump tanks of the containment with the low-consentrated 16 g/kg borated water storage Two system channels are connected to one tank of JNK system, two other system channels are connected to the second one (Figure 4.25).

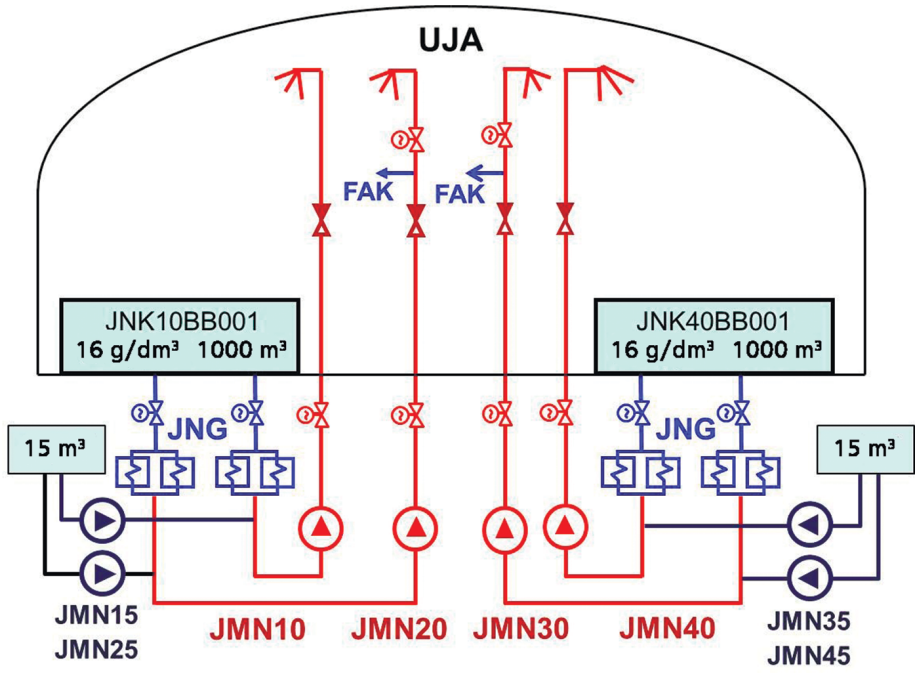


Figure 4.25 - The simplified technological scheme of the sprinkler system with the chemical tanks.

The pressure system pipelines are connected to the collecting channels with spray nozzles positioned under the containment (UJA).

The pressure lines of chemical injections pumps are connected to the suction lines of the sprinkler pumps. It provides the chemical supply from the alkaline storage tanks.

Under conditions of the design-basis accident related to the primary coolant leak two chemical tanks and four chemical injection pumps are provided in the system. JMN system must pipe the chemicals into the containment or iodine capture in steam air medium.

JMN20 and JMN30 channels are placed inside the containment and they are connected with the fuel pool through the FAK system pipelines: having both suction and pressure operations. (Figure 4.24).

JMN system provides the fulfilment of CA function «Pressure limitation control inside the containment» upon an activating signal of the safety control system by process parameters (SCSPP) «Overpressure growth under the containment is more than 30 kPa». Upon this signal: the sprinkler pumps switch on, the containment support fixture opens for the supply pipelines of the working substance into the containment atmosphere, then the support fixture of the pipeline which serves for the working substance intake from the sump tanks opens and the chemical injection pumps switch for the suction mode of the sprinkler pumps.

The diffuse working substance in the containment transfers back into the sump tanks. In case of the water temperature over 45 °C in the sump tanks the heat-exchangers of JNG-1 system connect: the support fixture opens on entering the heat-exchanger and the of the valves on the by-passes close.

For the safe function fulfilment JMN system must ensure the water supply into the air space of the containment dome at the rate of 600 tons per hour (one pump discharge rate is 300 tons per hour) and not above 80 °C.

4.10.2 THE JNB PASSIVE HEAT REMOVAL SYSTEM OF THE CONTAINMENT (PHRS C)

The containment dome passive heat removal system (JNB) refers to the technical tools which help to overpass the beyond-design accidents and it is designed for long—lasting heat removal (grid-isolated mode min 24 hours) from the containment dome upon the the beyond-design accidents conditions.

The system provides the pressure supression and the pressure maintenance within the project prescribed limits inside the containment and bend to the ultimate heat sink upon the beyond-design accidents conditions including the accidents with the core severe damage.

The system operation is based on the passive concepts and does not require an outside feedline for either the electric power starting operation or any control actions.

Passive Heat Removal System of the Containment (PHRS C) consists of 4 full independent from each other channels with the performance 4 x 33,3 %.

Each channel includes four heat-exchangers — condencers connected by the pipeleines with the energency heat removal tanks EHRT (Figure 4.26).

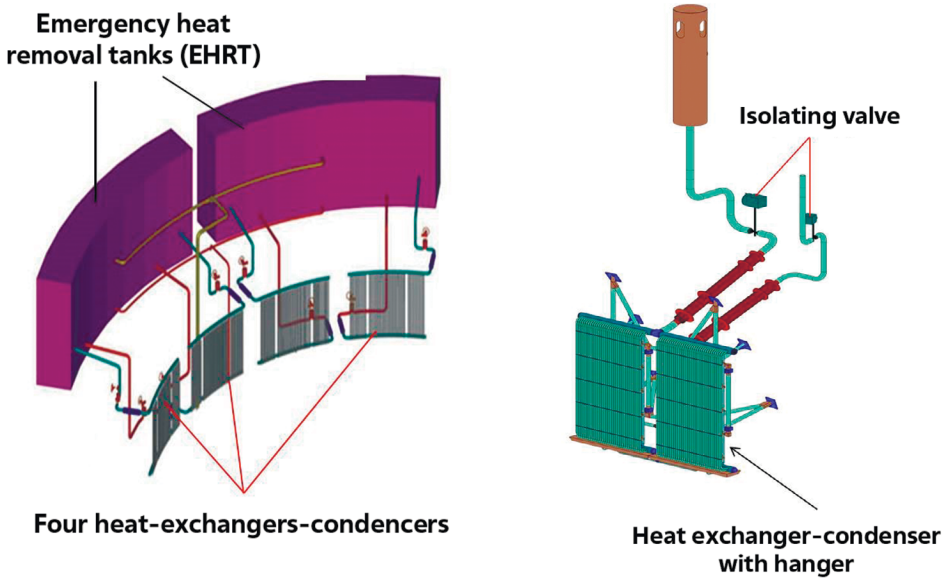


Figure 4.26 - The simplified scheme of the Passive Heat Removal System of the Containment (PHRS C) channel.

On the communication pipelines of the heat-exchangers-condencers and EHRT the isolating valve (opened in a normal position) is installed. It is placed for isolating the emergency heat-exchanger - condencer in case of its leak. The isolating valve is operated in the remote mode by the main control room operator or the auxiliary control room operator. For the on-time isolation of the emergency circulation loop from EHRT the project provides the leak detection system.

Upon conditions of the emergency closing of isolating valve for preventing the in-reactor circuit PHRS C overpressure the relief valves discharge into the emergency heat removal tank are installed on the heat-exchangers - condencers.

Technical characteristics of a separate heat-exchanger - condencer and the system in general are presented in the table 4.7.

Table 4.7 – Technical characteristics PHRS C

Characteristics	Heat-exchanger - condencer	One channel of PHRS C	Four channels of PHRS C
Quantity of the heat-exchanger - condencers, pcs	—	4	16
Heat-exchange, m^2	75	300	1200
Thermal power, MW	max 1,33	max 5,3	max 21,28

For comparison - thermal power removed by the air-cooling tower in the Novovoronezh NPP-2 project reachers **21,2 MW** at air temperature + 24 °C.

The heat-exchanger — condencers are placed on perimeter of the containment inner wall under mark 49,30 m. PHRS C tanks are placed higher — in the ring shaped auxiliary building envelope at mark + 59,85 m. (Figure 4.27).

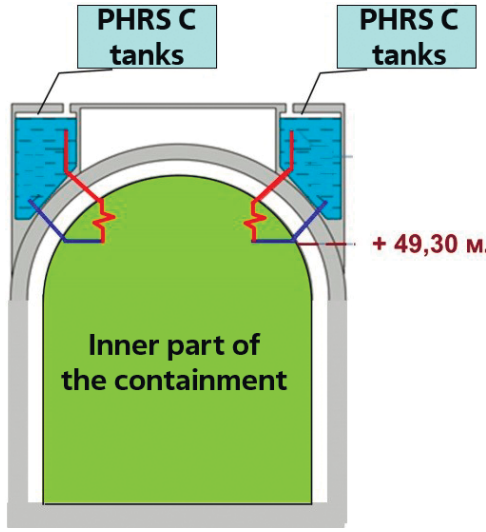


Figure 4.27 – Passive Heat Removal System of the Containment Dome (PHRS C) hight dimensioning

The heat from the containment is removed due to the steam condensation on the heat-exchangers — condencers. After that the heat is passed to the emergency heat removal tanks by means of the natural coolant circulation. The heat removal to the ultimate heat sink from the emergency heat removal tanks (EHRT) is fulfilled due to the water evaporation in the tanks during at least 24 hours after the accident occurs.

In the period beyond 24 hours different schemes of EHRT injection are provided including the schemes with the help of mobile equipment (Figure 4.20).

On standby mode the system is in the constant-ready state for fulfilling its functions.

The system starts-up in a passive way when the pressure increases under the containment dome without energy supply from outside and without any control actions.

The passive principle of the system start-up in case of either the pressure increase or the temperature increase under the containment dome results the JMP system operation under the conditions of design basis accidents and beyond-design basis accidents with leak under the containment dome.

4.11 FUNCTION CB. FISSION PRODUCT LOCALIZATION INSIDE THE CONTAINMENT DOME IN THE PROJECT OF LENINGRAD NPP-2 (REACTOR FACILITY V-491)

4.11.1 PRINCIPLE SCHEME OF THE ACCIDENT LOCALIZATION AREA (ALA)

Safety function «Fission product localization inside the containment dome» is provided by the following systems:

— leakproof enclosure (LE);

— sprinkler system (JMN);

— system pipelines: JNG, JND, JDH;

— sump tanks of the containment dome JNK10,40BB001;

— cooling down pool protection;

— leakage isolation system of the containment (KLC11/21/31/41).

The interspace limited by the leakproof enclosure which keeps the radioactive materials creates the accident localization area (ALA).

The leak-tight enclosure fulfills the following general functions:

— prevention or localization of the spreading radioactive materials outside ALA;

— the protection of the system and (or) its elements from enviromental impacts which are responsible for keeping the radioactive materials and in case of the break-down they perform let go the radioactive release which is higher than the leakage design values.

The leakproof enclosure consists of the following basic elements (Figure 4.28):

— ferroconcrete building structures of the inner and external parts of the containment dome;

— steel lining (with anchoring system), made on the inner part of the containment dome;

— leakproof standpipes with embedded items;

— locks with embedded items;

— parts of the distribution pipelines crossing the leakproof enclosures or connected to the leakproof enclosures within the bounds of the isolating devices;

— equipping and distribution pipelines that go beyond the bounds of the building structures of the leakproof enclosures and are the part of ALA;

— isolating valves.

ALA consists of two volumes. The first one is leakproof and consists of (Figure 4.28 – «V1»):

— inner part of the containment dome;

— isolating valve placed beyond UJA building,

— containment penetration parts placed beyond the inner part of the containment dome;

— pipelines parts from the leakproof standpipes till the isolating valves, off-board doors of the locks;

— embedded items of the containment penetration and locks placed in the external part of the containment dome.

The second volume (Figure 4.28 – «V2») consists of:

— inner and external parts of the containment dome;

— leakproof standpipes;

— inner and external lockdoors;

— embedded items of the containment penetration and hatchways placed in the inner and external parts of the containment dome.

Thus, the common boundary of ALA consists of the following items (Figure 4.28):

— external part of the containment;

— isolating valve placed beyond UJA building;

— containment penetration parts placed beyond the containment dome,

— pipelines parts from the containment penetrations till the isolating valve,

— external doors of the hatchways,

— embedded items of the containment penetration and hatchways placed in the inner and external parts of the containment dome.

In accordance with the article 4.3.14 of the Normative Rules NP-061-05 the cooling down pond coat is the separate accident localization area which must be accomplished as the element of the accident localization system.

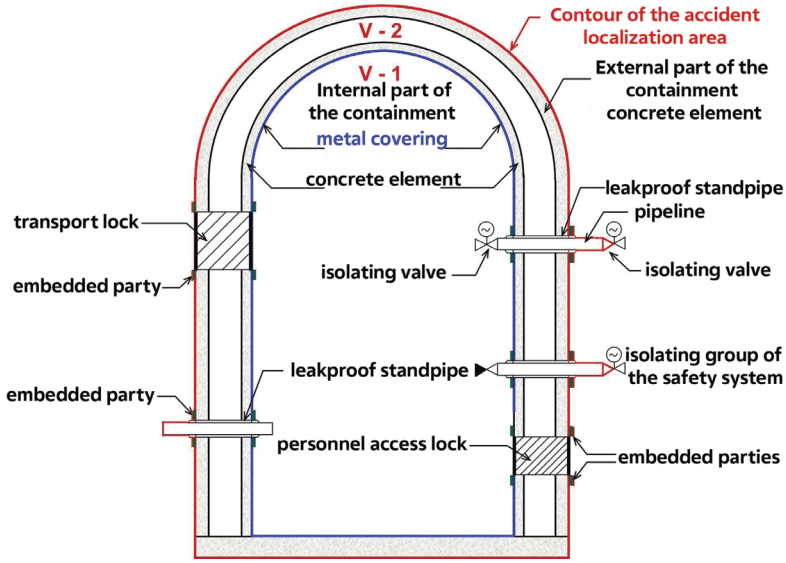


Figure 4.28 – Principle scheme of ALA

As long as during the accidents with the steam generator collector rupture or heat-transfer tube rupture the primary circuit and the secondary circuit combine, the accident localization boundary zone becomes the area from the by-passes till quick-acting pressure-reducing valve and from the feed water pipelines till isolating valves.

4.11.2 ISOLATING DEVICES

Isolating devices are made for isolating pipeline systems with the process media which go through the containment dome. They also serve for prevention fission products outflow into the environment during the design basis accident with loss of coolant.

In the project of Leningrad NPP-2 in comparison with Novovoronrzh NPP-2 pneumatic valves are not used with isolation function. In most cases on the pipelines with fission products two motor-operated valves are installed — one is installed inside the containment and the other is installed outside of the containment. The motor-operated valves are supplied with the power from the system of the first group steady electric-power supply (emergency battery). For this purpose the separation principle is used — two isolated valves placed on the same pipeline have the power supply from different power supply system trains.

The additional safety valve is set up for the protection of the pipelines with the medium liable to the temperature expansion (Figure 4.29).

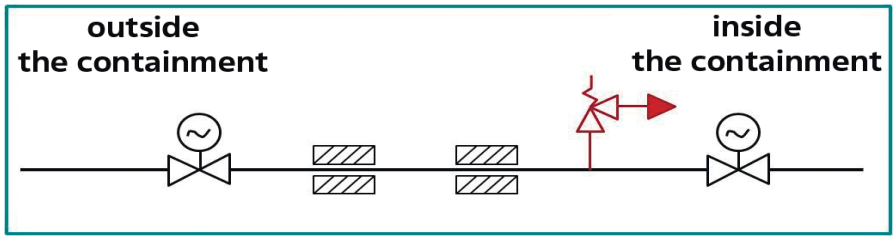


Figure 4.29 – Isolating scheme with motor-operated valves

According to the project of «Hanhikivi» NPP the reverse valves are provided instead of the safety valves for protection from the elements overpressure. They are set up on the bypass circuits of the isolating valves inside the containment (Figure 4.30).

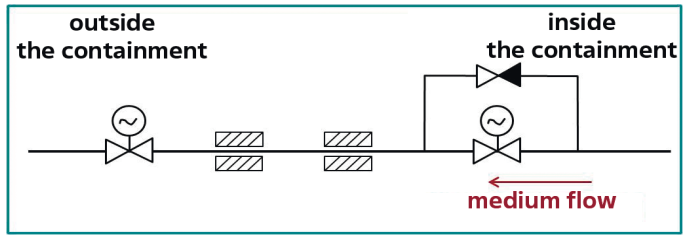


Figure 4.30 – Isolating group scheme with the motor operated valves

With rising temperature of the water medium in the space between the isolating valves the excessive pressure is released into the pipelines of the containment dome using the reverse valve.

Hand/automatic isolating valve is set up on the lines which are used only during the repair process (for example, on the line of the defective assembly detection system). When operated with full power the manually-driven valve is on the closed position («locked») and its opening is fulfilled with special permission when the reactor is out of work and cooled.

On the pipelines of the safety system, which bring up the medium into the containment when the accidents related to the leak of the coolant take place, one isolating valve with the electric motor drive is set up outside the containment (Figure 4.31).

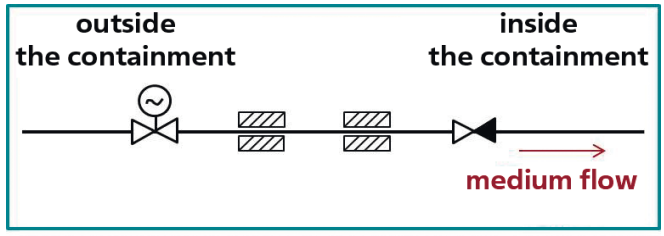


Figure 4.31 – Isolating group scheme of the safety system

The isolating valve of the safety system is opened in a normal mode and it doesn't close on a signal of the containment isolation. In case of the safety system channel break-down the operator closes the isolating valve on the required line.

During the operation of the safety system channels when the accidents of feeding into the containment and/or out of the containment take place the accident localization boundary passes the pipelines and the channels equipment.

The isolating valve inside the containment can not be installed on the by-passes and the feedwater pipelines.

The feeding pipeline from the emergency leakproof pulling chamber and including the isolating valve has the special construction — it is made as the part of the containment and is placed inside the metal envelope (capsule) designed according to the parameters of the containment counting the static pressure during the accident.

The transmitters which are connected by the impulse lines from (leakproof explosure) LE are installed in the special constructions in the itercontainment space. Such special constructions have coverings and traps in case of depressurization.

The self-start algorithm «SV-11 — Isolation of the containment» is made for two groups of the isolating valves: «Isolation of the leakproof exposure (LE)-1» and "Isolation of the leakproof exposure (LE)-2».

For the first group of the isolating valve the activating signal for the operation is «Overpressure in the containment above **5 kPa**». This group consists of the following elements: the isolating valve of the air discharge purification system, radioactive drains and equipment drain system of the reactor building.

For the second group of the isolating valve the activating signal for the operation is one of two signals: «Overpressure in the containment above **30 kPa» or «$\Delta T_{S-1} < 8$ °C».**

«Isolation of the leakproof exposure (LE)-1» is fulfilled with the help of the isolating groups which are divided onto two categories 2a and 2b.

For 2a category isolating valve the opportunity of its opening by the operator is provided in 30 minutes after the activating signal. This category includes the structural reinforcement of the component cooling loop, the feed and bleed systems of the primary circuit, the auxiliary systems of the reactor coolant pump set, the equipment drain system of the reactor building.

For category 2b of the isolating valve operation becomes possible only after signals «$P_{GO} < 20$ kPa» и «$\Delta T_{S-1} > 12$ °C».

4.12 FUNCTION CC. LOCALIZATION IN THE ANNULAR REGION IN THE PROJECT LENINGRAD NPP-2 (REACTOR FACILITY V-491)

KLC11/21/31/41 leakage isolating system which protects the containment dome is made for underpressure in the intercontainment space and in the safety structures in emergency operation.

System function criteria are:

— underpressure in the intercontainment space minimum 100 Pa, in the safety structures — minimum 50 Pa;

— air cleanup from the dust radiation and radio-iodine removed from the structures into the atmosphere (as for the dust radiation - minimum 99, 9 %; as for radio—iodine — minimum 99, 9 %).

KLC11/21/31/41 system has four-channel structure with 50 % equipping capacity of each channel. The rated performance is 3600 m^3/h.

Each channel installation is supplied by the appropriate emergency power supply channel and includes the following elements:

— reverse valves;

— filter and integrated cleaning facility for the dust radiation and radio-iodine;

— leakproof gag valve with the electric motor drive;

— fire safety valves with the electric motor drive;

— electric air heater;

— fan.

In case of overpressure in the containment dome above 30 kPa the leakproof isolating valves close on the border of the auxiliary building and the safety building. Two filter channels are switched on automatically. The regulation valves on the supply air piping of the intercontainment space close when the leakproof isolating valves close on the border of the auxiliary building and the safety building.

The air is removed from the intercontainment space and the safety building through the filter channels and then it is outthrown through the exhaust air duct and the ventilating high-level pipe.

The air heaters are switched on and operate upon very humid environment which is for the air clean-up (above 60 %).

In case of activation of a fire detector the operating air fans are switched off and the fire dampers are closed automatically. They are place in the intercontainment space on the border of the safety channels.

4.13 SEVERE ACCIDENT MANAGEMENT SYSTEM IN THE PROJECT OF LENINGRAD NPP-2 (REACTOR FACILITY V-491)

The severe accident management system operates with the purpose of the severe accident mitigation. The main objectives of the severe accidents operating are:

— preserving the integrity of the forth physical barrier of the containment;

— limitation of the radioactive materials and/or radioactive emission outflow to the environment.

In the project of Leningrad NPP-2 for the severe accident control the following technical means are provided:

— containment hydrogen removal system;

— core melt localization device;

— system of passive heat removal from containment;

— emergency water usage system from the internal inspection cavity.

The containment hydrogen removal system is intended for maintaining the hydrogen explosive safety in the accident localization zone during the beyond-design basis accidents and design accidents.

The technological scheme is the distribution of hydrogen recombiner combined into eight technological groups around the confinement area.

The system consists of 44 passive autocatalytic hydrogen recoils by AREVA:

— FR90/1-1500 (28 pcs);

— FR90/1-750T (16 pcs).

Total system capacity under a pressure of 0,15 MPa and bulk concentration of the hydrogen 4 % equals approximately 188 kilogram of hydrogen per hour.

For comparison - JMT system in the project of Novovoronezh NPP-2 consists of:

— autonomous PCRV models PVK-1000 in the quantity of 200 pcs;

— autonomous PCRV models PVK-500 in the quantity of 7 pcs.

In the table 4.7 hydrogen recombiner productivity is shown which is used in Novovoronezh NPP-2.

Table 4.7 – Hydrogen recombiner productivity in the project Novovoronezh NPP-2

Hydrogen recombiner productivity (at a temperature of 100 °C and pressure 0,1 MPa)	Types of passive catalytic hydrogen recombiner	
	RVK-500	RVK-1000
at bulk concentration of the hydrogen 3 %	0,16 kg/hr	0,39 kg/hr
at bulk concentration of the hydrogen 5 %	0,32 kg/hr	0,77 kg/hr
at bulk concentration of the hydrogen 8 %	0,67 kg/hr	1,58 kg/hr

The core melt localization device is made for the fuel-containing mass localization on the out-of-core stage of the severe accident. The core catcher design of Leningrad NPP-2 is analogous to the core catcher design of Novovoronezh NPP-2.

The corium subcriticality in the core catcher is provided by gadolinium addition into the sacrificial material. For cooling down the melt water is used. The water is taken from the internals inspection cavity or through the valves of passive water supply of in-vessel internals The external part of the core catcher housing is cooled down by water. The water comes into the concrete vault from the floor with the zero mark or from the sump tanks of the containment.

The system of the emergency water usage from the internals inspection cavity (JNB90) is made for:

— flooding the in-vessel internals unit with the mark of 0,00 (in case of the beyond-design basis accidents related to the reactor coolant damage) or from the samp tanks (in case of the beyond-design basis accidents related to the core melt);

— 42 % NaOH alkali solution feeding into the sump tanks of the containment for slowing down the rate formation of flying iodine forms inside the containment;

— for borated water feeding from the internal inspection cavity into the core melt localization device in case of the severe accidents related to the fuel-containing material output beyond the reactor vessel.

The automatic system operation by a signal from the safety control system is not provided. The motor-operated valve of JNB90 system is operated by the operator of the mosaic panel of beyond-design basis accidents which is placed in the main control room (MCR) or in the auxiliary control room (ACR).

The project provides the following action algorythm for the MCR operator:

— If reached above the limiting index (600) of the steam-gas mixture on the way out of the fuel assemblies the operator must open the valve JNB90AA101, AA103 on the pipelines, combining the core catcher structure with the sump tanks (Figure 4.32).

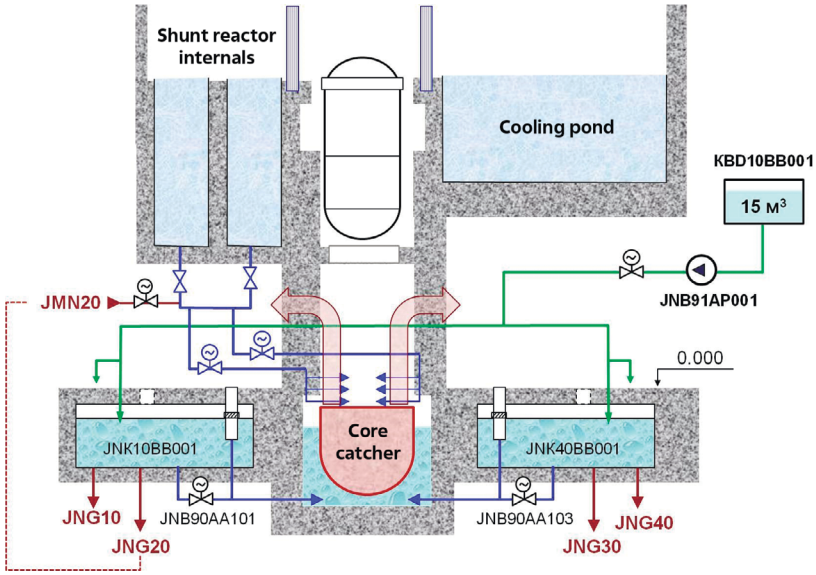


Figure 4.32 – The simplified scheme of JNB90 system

If reached 100 Gy/hr gamma dose in the atmosphere of the containment (such capacity of dose testify about 5 % of the fuel in the reactor core) the operator must supply alkali solution from KBD10BB001 tank into the sump tank of the containment using JNB91AP001 pump. KBD10BB001 tank volume is 15 m^3, the pump flowrate capacity is 30 m^3/hr.

If reached 400 °C temperature in undercore space the alarm is generated on the panel-board of the beyond-design accident. It informs the operator about the beginning of the core-melt output. In 3 hours after that the operator must open the support fixture on the water supply line from the internals inspection cavity into the core catcher frame line. The water is carried onto the surface of the core-melt with the outflow capacity of 3 kg/s. Water supply in the inspection cavity is enough for 24 hours of continuous feeding onto the core-melt.

Residual heat removal from the core catcher is fulfilled due to water boiling. The steam goes to the upper part of the containment dome from the core catcher. In the upper part of the conrainment dome the steam becomes condensed on the heat-exchange surface of PHRS CD. After that the condensed water is collected in the sump tank of the containment dome and after that the condensed water is fed back into the core catcher passively.

For the further over 24 hour heat-extraction from the core catcher and containment dome the project provides:

— filling the internals inspection cavity from the pump of JMN20 system (Figure 4.31);

— EHRT emergency injection from a high-pressure diesel pumping system.

CONCLUSION

The first 3rd generation light water reactors were designed in 1990-s. Although they have already proven to be reliable, they are being continuously modernized in both safety and performance. Some power units with modernised reactors are currently being constructed and in operation in various parts of the world. One such example is 1600 MW Evolutionary Pressurized Reactor (EPR) by AREVA, two of which have already been commissioned in China, while two more are under construction in Finland and France as of 01.05.2020. Also, Toshiba-Westinghouse's four AP-1000 units (based on modernized PWR design) were commissioned in China.

As for Russian Nuclear industry, Novovoronezh NPP-2 Unit 1 became the world's first operating nuclear power unit in the latest generation, which was an undoubted accomplishment. As of 01.05.2020, three power units with VVER-1200 have been commissioned: Novovoronezh NPP-2 Units 1 and 2, as well as Leningrad NPP-2 Unit 1. Leningrad NPP-2 Unit 2 is planned to be commissioned in 2021, two units in Belarus are on the final stage of commissioning, while similar units at Kursk NPP-2 in Russia, as well as at Akkuyu NPP in Turkey and Rooppur NPP in Bangladesh, are currently under construction.

Novovoronezh NPP-2 Units 1 and 2 were constructed per a state-of-the-art Russian 3+ generation design AES-2006 and are equipped with VVER-1200 reactor plants. The design is based on half-a-century safe and accident-free VVER-type reactors' operation experience. The current project complies with all requirements of Russian National legislation, as well as the European Utility Requirements for LWR Nuclear Power Plants (EUR) and IAEA Guidelines.

The state-of-the-art units of Novovoronezh NPP-2 show improved performance indicators, ensure exceptional operational safety and comply with the post-Fukushima requirements of the Western European Nuclear Regulators Association. The units feature an optimal combination of active and passive safety systems, which improves unit safety by preventing severe beyond-the-design accidents for at least 24 hours in case of losing all power supply both with leak-tight and leaky primary circuit.

Novovoronezh NPP-2 design allows for unit construction in diverse environmental and man-made conditions. The design covers various construction site soil options – from rocky to soft ground. For units constructed on sites with severe environmental exposure, the design provides structural reinforcement without changing the layout arrangement of buildings and facilities.

As for further improvement potential of units with VVER-1200 reactors, there are the following options:

— expansion of engineering storage for process medium discharge and heat transfer surfaces facilities for milder passing through the transients in case of the operation disorder including cases of non-redundant equipment outage;

— circuit design improvement with more effective and maintenance-friendly equipment servicing;

— optimization of technological protection and interlocks for increasing the dynamic stability of new generation power units;

— validation of control and management facilities for operating personnel, human-machine interface implemented at UULS, large screen displays, auxiliary and normal operation safety panels;

— development and application of the intelligent assistance system to decrease traffic load for nuclear plant operator;

— setting diurnal power maneuvering modes for power units (100%-50%-100%);

— enabling the possibility for power unit operation with an increased fuel cycle (up to 18 month);

— creation of a 'digital twin' nuclear power unit which comprises 3D information model with established connections and capability for navigation between system and component characteristics, passports, operating documentation, and also includes a simulation programme of the nuclear power unit capable of performing neutronic calculation of the reactor core, thermohydraulic calculation of primary and secondary coolant circuits and a model for the power unit's main process equipment automatics;

— development of predictive analytics systems as the basis for the condition-based equipment maintenance.

The continuously modernized designs of power units with VVER-1200 serve as the foundation for further development of both Russian and foreign nuclear industry.

The authors express their gratitude to those who would leave comments on this manual and suggestions for its improvement.