Status report 85 - VVER-1500 (V-448) (VVER-1500 (V-448))

Overview

Full name	VVER-1500 (V-448)
Acronym	VVER-1500 (V-448)
Reactor type	Pressurized Water Reactor (PWR)
Coolant	Light Water
Moderator	Light water
Neutron spectrum	Thermal Neutrons
Thermal capacity	4250.00 MWth
Gross Electrical capacity	1560.00 MWe
Design status	Detailed Design
Designers	Gidropress
Last update	21-07-2011

Description

Introduction

The design of the large power NPP with water-cooled water-moderated power reactor VVER-1500 (RP VVER-1500, model V-448) is based on the following main objectives:

- safety improvement in comparison with the best NPPs, considering international trends and criteria stated in the requirements of European users (European User's Requirements) and in the requirements for prospective foreign power units of similar power level, accepted for new NPP designs, simultaneously complying with up-to-date requirements for ecology and international standards;
- reduction in NPP design cost for the purpose of ensuring the competitiveness with electric fossil-fuelled power stations; and
- development of the design oriented to the internal and foreign markets and meeting the up-to-date technical-and-commercial requirements.

In design of an NPP with VVER-1500 the essential optimization was made for the combination of the systems and equipment proved in the domestic and foreign practice at operating VVER and PWR reactors with the use of new solutions and approaches both in the field of safety and in the field of designing.

The design bases of new generation NPP with VVER-1500 reactor are based on the extensive experience in designing, manufacturing, constructing and operation of NPPs with VVER-440 and VVER-1000 reactors. With this, qualitative improvements of VVER is made owing to use of the proved reliable systems, layouts and equipment, consideration and elimination of shortfalls identified in operation of NPPs with VVERs and PWRs of the previous generation.

Safety assurance in the RP V-448 design is provided by:

- implementation of the defence-in-depth principle based on application of the system of barriers to the release of ionising radiation and radioactive substances into the environment, and also by use of the system of technical-and-organizational measures on protection and keeping of the efficiency of each barrier;
- use of the RP main equipment (reactor, main coolant pipeline, steam generator, pressurizer, ECCS hydroaccumulators, control and protection system (CPS) drives, fuel handling machine) that are similar to the equipment commercially manufactured and operated at NPPs with RP V-213 (VVER-440) and V-320 (VVER-1000);
- use of the nuclear fuel assemblies in the core of V-448 that are similar to VVER-1000 fuel assemblies with long-term operating experience at NPPs, and also a decrease in specific loads of the core as compared with VVER-440 and VVER-1000;
- application of materials for main equipment of RP V-448 proved by long-term operating experience at VVER-440 and VVER-1000.

Innovations of the reactor plant V-407 design, as compared with the designs of reactor plants VVER-440 and VVER-1000 (in particular, with the design of RP V-392) consists, mainly, of the following:

- application of the advanced, more reliable emergency protection that excludes positive reactivity effects and provides reactor subcriticality by means of mechanical control members without boric acid at coolant temperature above 100°C at any moment of the cycle operation, including beginning of cycle (BOC)_;
- application of the safety systems based on the active and passive operation principles, using such physical properties as heat capacity, heat conduction, heat transfer, gravitation, energy of the compressed gas etc.;
- solution of the problem of primary-to-secondary leaks at the expense of implementation of the corresponding algorithm of overcoming this accident [1] and increase in design pressure of the secondary circuit;
- multipurpose use of the engineered safety features and systems provided in the design for mitigation and management of both the design basis accidents, and beyond design basis accidents, including severe accidents;
- mitigation of consequences of the wide spectrum of beyond design basis accidents using the criteria applied to estimate the consequences of design basis accidents;
- ensuring reliable power supply (from diesel-generators) to the normal operation systems, important to safety, that, in turn, enable to enhance stability of RP and NPP, as a whole, under incidents caused by loss of off-site power;
- increase in efficiency of fuel utilization by 15~20 % as compared with VVER-440 and VVER-1000;
- retaining the reactor core corium during severe accidents within the special trap located outside the reactor pressure vessel;
- increase in thermal efficiency up to 35.3% owing to pressure increase at the steam generator (SG) outlet up to 7.34 MPa;
- substantial improvement of fire safety of the RP, and of the NPP as a whole, owing to rejection of combustible isolation, lubricating and cooling media, improvement of fire protection and fire extinguishing means.

For emergency core cooling under the design basis accidents and beyond design basis accidents the passive systems were applied, such as: SG Passive Heat Removal System (PHRS), emergency core cooling system, and for the accidents coincident with reactor scram, incorporation of a quick boron injection system ensuring the reactor shutdown due to insertion of an absorber into reactor using reactor coolant pump (RCP) set pressure differential.

Efficiency of the reactor emergency protection with 118 rod cluster control assembly (RCCA) provides the core subcriticality with reactor scram considering sticking of one most effective CPS CR at decrease in coolant temperature to 100 $^{\circ}$C. Such an increase in emergency protection efficiency essentially improves safety in the course of accident conditions, accompanied by uncontrolled cooling down of the primary circuit.

The core is made in such a way that for any accident the reactor has negative temperature feedbacks, that together with the action of mechanical reactivity control system will provide the core subcriticality without additional supply of boric acid.

Fast neutron fluence with energy of more than 1 MeV to the reactor vessel for designed service life of the reactor vessel of 60 years does not exceed the value of 0.92×10^{19} n/cm^2, that is less than fluence to VVER-1000 reactor vessel for 40 years.

In the safety design of the RP V-448 the "leak before break" concept is used in designing the primary circuit pipelines. Application of this concept has enabled the elimination of supports-restraints and guards from the design for addressing the case of guillotine break of the main coolant pipe (MCP) and other large-diameter pipes.

Improvement of RP V-448 safety is also reached by:

- improvement of quality of equipment and systems at all stages of RP life cycle;
- elaboration and implementation of Quality Assurance programs at each stage of NPP life cycle, and strict adherence and control over execution of these programs;
- application of diagnostic monitoring systems of production processes and material conditions in the reactor, steam generators and other equipment and pipelines of the reactor plant, including the systems important to safety;
- equipping the NPP Units with VVER-1500 reactors with up-to-date I&C system possessing an enhanced reliability, having the self-diagnostic and expert system that gives recommendations to the operator.

Characteristic	Value	
1 Reactor power, thermal, MW	4250±170	
2 Coolant flow rate through reactor, nominal, m^3/h	113740	+1940 -3060
3 Core outlet coolant pressure, MPa	15.7±0.3	
4 Average coolant temperature at the reactor inlet, °C	298.1	+1.2
5 Average coolant temperature at the reactor outlet, °C	329±2.9	
6 Pressure of generated steam at SG outlet , MPa	7.34±0.1	
7 Steam capacity of one SG, t/h	2152^{+170}	
8 Temperature of feed water (under nominal conditions), °C	230±5	
9 Number of circulation loops (RCP set), pcs.	4	
10 Total number of FAs in the core, pcs.	241	
11 Number of assemblies with CPS rods, pcs.	118	
12 Design service life, years:		

Characteristic	Value
- of reactor pressure vessel;	60
- of equipment	50
13 Average burnup fraction in the equilibrium loading of five-year fuel cycle, MWD/kg(U)	57.2
14 Average specific power density of the core, kW/L	87
15 Specific consumption of natural uranium in the equilibrium loading of five-year fuel cycle, gram U/(MWD)	180
16 Average enrichment in make-up fuel in the equilibrium loading of five-year fuel cycle, %	4.5
17 Design parameters of the primary circuit:	
- gauge pressure, MPa;	17.64
- temperature, $^{\circ}C$	350
18 Design parameters of the secondary circuit:	
- gauge pressure, MPa;	9.5
- temperature, $^{\circ}C$	325

RP V-448 main characteristics

[1] An automatic algorithm of coping with the primary-to-secondary LOCA that actuates in response to appropriate initiating signals is envisaged in the V-448 design. The algorithm is a definite sequence of signals in response to which appropriate safety systems are actuated.

2.1. Process scheme

The process scheme of RP V-448, is a traditional double-circuit scheme with four loops of energy conversion and transfer from reactor to the turbine generator for RP with VVER-1000 includes:

- the primary circuit;
- the secondary circuit;
- auxiliary systems for normal operation connected to the primary and secondary circuits;
- safety systems.

2.2. The primary circuit and its main characteristics

The reactor plant VVER-1500 is a part of large capacity Unit. The reactor plant (RP) consists of the pressurized primary circuit and the associated systems and equipment for its maintenance, located within the containment, and also the complex of systems for the monitoring, control, regulation, protections, interlocks, alarm and diagnostics, forming the process instrumentation and control system (I&C).

The primary system includes a circulation circuit at operating pressure, together with the pressurizing system. The primary circuit together with associated systems provides heat removal from the reactor core by coolant under normal operation conditions, anticipated operational occurrences and design basis accidents, and transfers heat to the secondary circuit through heat exchanging surface in the steam generator. The circulation circuit is a traditional four-loop configuration including the reactor, four horizontal steam generators, four reactor coolant pump (RCP) sets and the related pipelines. The process chart of the reactor plant is given in Figure 1.

Boric acid solution serves in the reactor as coolant and moderator.

The primary equipment and pipelines and associated systems are placed within the containment, which consists of an internal protective sealed envelope (made of prestressed reinforced concrete) of 49 m in diameter, which is designed for potential internal impacts, and which, in its turn, is protected by an external protective envelope (made of unstressed concrete), ensuring protection from external impacts and decrease in parameters of dynamic impacts on the equipment located within the internal protective envelope. Layout of RP equipment within the containment is shown in Figures 2, 3.

Steam generators, RCP set and a pressurizer are located in the isolated boxes to provide radiation protection and prevent damages to the equipment caused by missiles.

All RP equipment is installed and fastened in civil structures at different elevations of the reactor building with regard for probably more uniform distribution of loads onto the civil structures.

2.3. RP V-448 and NPP safety systems involve:

- passive part of the emergency core cooling system;
- system of passive heat removal from SG;
- primary circuit overpressure protection system;
- system of emergency gas removal from the primary circuit [2];
- quick boron injection system;
- emergency boron injection system;
- emergency and planned cooling down system;
- steam generator emergency cooldown and blowdown system.

2.4. The main auxiliary systems associated with the primary circuit include the following systems for normal operation:

- chemical and volume control system;
- gas blowoff system (air removal when filling the primary circuit);[3]
- nuclear equipment leak collection system;
- sampling system.

2.5. The primary circuit components

2.5.1. Reactor

The reactor part of RP V-448 is intended for generation of thermal energy at NPP due to nuclear fuel fission reaction in the reactor core.
The principal design of the reactor is shown in Figures 4 and 5.

Design of reactor V-1500 (V-448). The reactor consists of the following main components:

- vessel with the main joint components;
- upper unit with 118 CPS drives;
- reactor internals (core barrel, core baffle, protective tube unit);
- reactor core consisting of 241 fuel assemblies (FAs).

Seventy two (72) in-core instrumentation detectors (ICIDs) are installed into the central tubes of FA without control and protection system control rod (CPS CR) to ensure monitoring, control and protection of the reactor core, besides, forty two (42) thermoelectric temperature transducers located within the reactor protective tube unit (PTU) are installed into the reactor to control temperature at the outlet of FA with CPS CR.

The surveillance programme is laid down in the design.
The forced coolant supply (driven by the RCP set) into the reactor is provided via four inlet nozzles of the vessel, then it flows down through the annulus between the reactor vessel and core barrel, and upward into the fuel assemblies through the perforated elliptical bottom of the core barrel and holes in the supports .

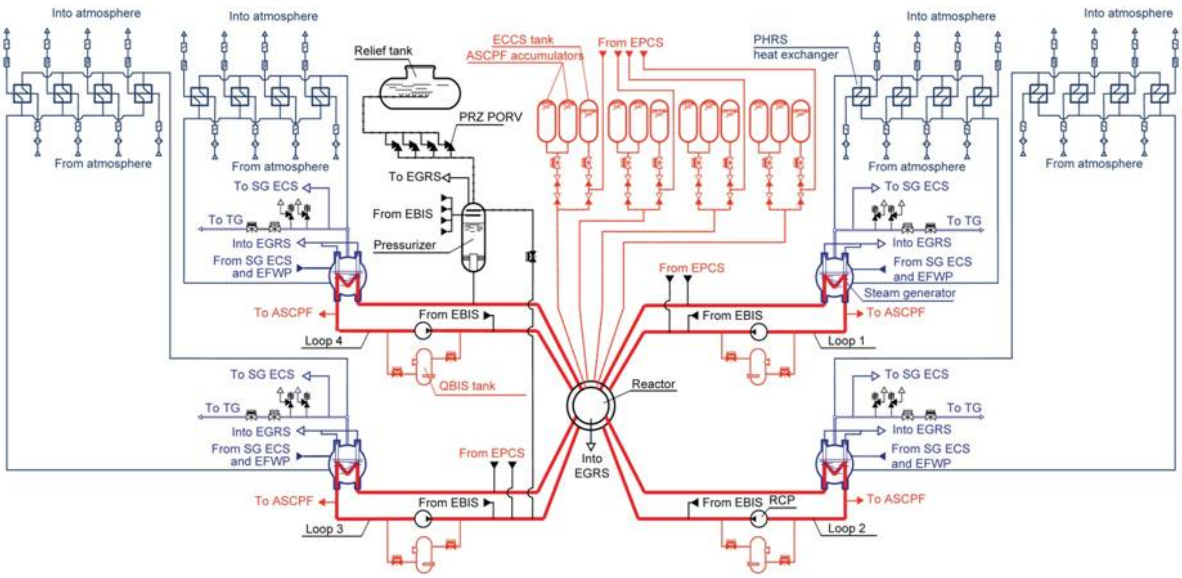

Figure 1. Process chart of RP V-448 primary circuit

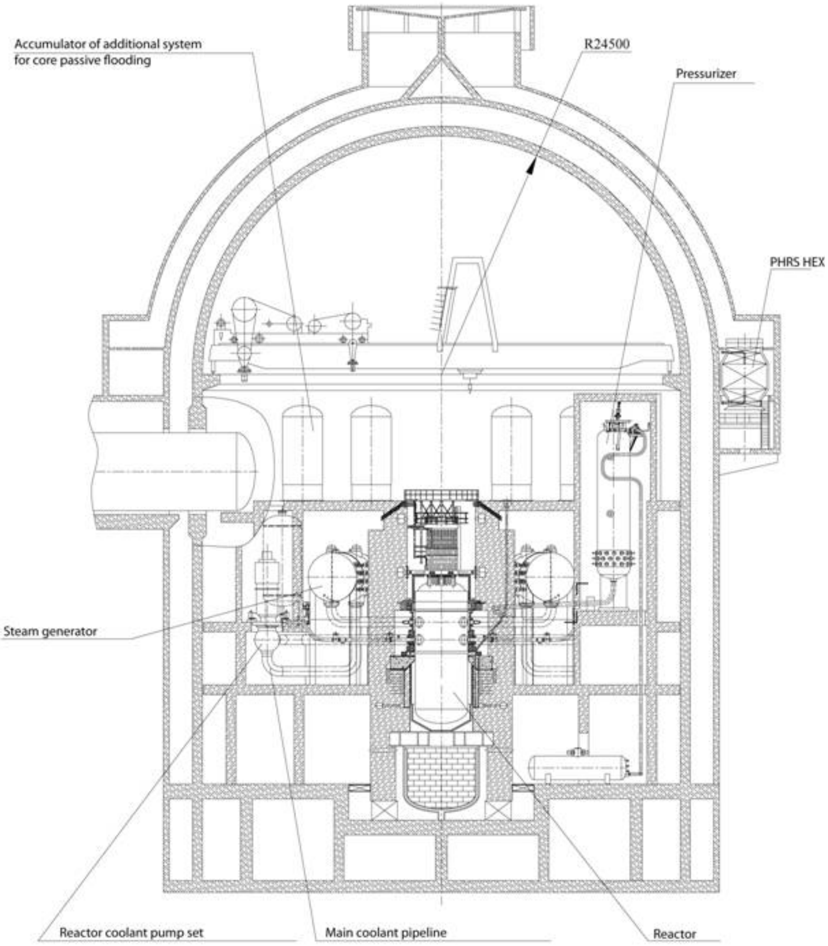

Figure 2. Layout of RP V-448 equipment (longitudinal section)

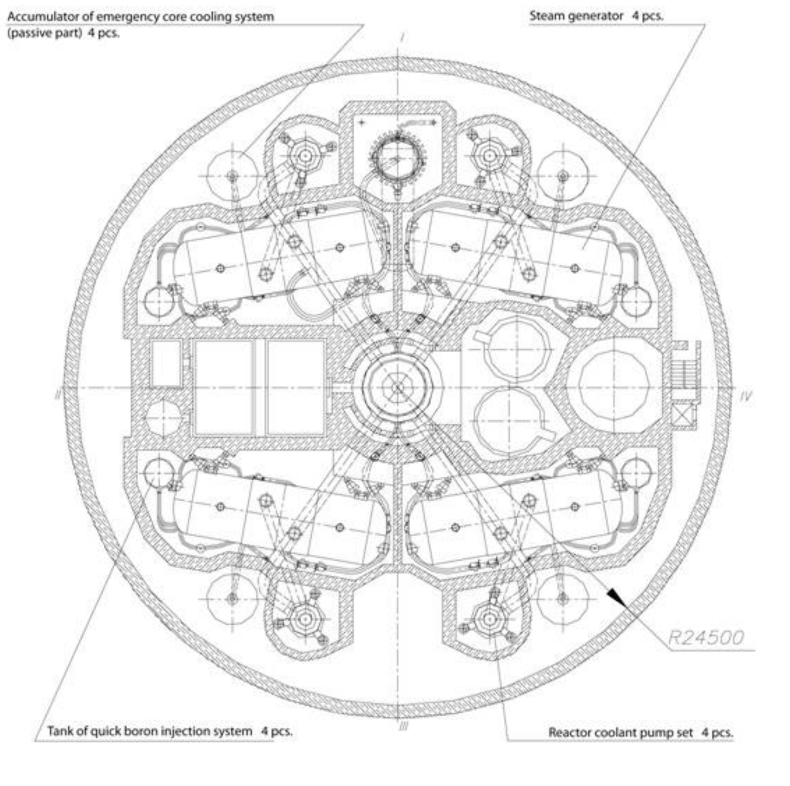

Figure 3. Layout of RP V-448 equipment (cross-section)

While flowing through the FAs, the coolant is heated up due to heat from nuclear fuel fission reaction and flows out of FA top nozzle into the inner cavity of PTU tube and through perforations of PTU tube the coolant flows into PTU tube space and then through perforations in PTU and core barrel shells flows out of the reactor through four outlet nozzles of the vessel.

The reactor is housed in the cylindrical concrete cavity, having a biological shielding in the core region. The supporting truss and the thrust truss are used to fasten the reactor.

2.5.1.1. Reactor pressure vessel

The reactor pressure vessel consists of the following main components: a flange, upper and lower shells of the nozzles area, cylindrical shells and elliptical bottom, welded between themselves by circumferential welds. The vessel has two rows of nozzles with inner diameter of 850 mm - four nozzles in each row, and Emergency Core Cooling System (ECCS) four nozzles Dnom 300 - two nozzles in each row. The instrumentation nozzle with inner diameter of 300 mm is provided on the vessel.

The vessel is installed and fastened on the supporting ring placed on the supporting truss of the concrete cavity. The vessel is protected against seismic loads with the help of thrust ring placed on the flange.

Service life of the vessel is 60 years.

2.5.1.2. Upper unit

The upper unit comprises: elliptical top head with nozzles and housings for arrangement of CPS drives (118 pcs.), cross piece, metalwork, nozzles for outlet of ICIDs (28 pcs.), air vent nozzle and level gauge nozzle.

2.5.1.3. Reactor internals

The reactor internals include: core barrel, protective tube unit and core baffle. The core barrel is intended for arrangement of the reactor core and formation of coolant channel. The protective tube unit is intended for protection of CPS drive elements and RCCA against impact of coolant at the core outlet, provides fixation of FA by coordinates in the core. The core baffle is intended for protection of the vessel against impact of neutron flux to the reactor vessel walls.

2.5.1.4. CPS drive

ShEM-3M CPS drive is applied as an actuator of RCCA.

2.5.1.5. Reactor core and fuel part of the design

The reactor core is intended for generation of heat and its transfer from FA fuel rod surface to the coolant during the design service life without exceeding the permissible limits of damage to fuel rods.

The reactor core structure consists of 241 FAs (Figure 6), into which 118 RCCAs are located.

V-1500 FA is an evolutionary development of the jacket-free advanced fuel assembly (UTVS) designs applied at the commercial VVER-1000 reactor. The fuel assembly has a demountable structure and consists of the following parts: framework made from 25 guiding channels with the spacing grids, a top nozzle with the spring unit, a fuel rod bundle, and a bottom nozzle.

The FA structure provides a possibility for free temperature and radiation expansion of its elements. The FA radial expansion as a function of temperature and radiation does not exceed the gap value between fuel assemblies in the core. Availability of spring stroke range in FA top nozzle provides compensation of process tolerances and difference in temperature expansions between FA elements and reactor internals, as well as damping of CPS CRs drop (RCCA and CPS drive extension shaft engaged with it). FA compression through the movable cylindrical shell of top nozzle and spring unit prevents fuel assembly from lifting under the action of pressure differential.

The fuel rod consists of the cladding filled with pellets from the sintered uranium dioxide UO_2, and sealed by plugs. The inner volume of fuel rod is filled with helium under pressure.

RCCA is intended to control nuclear reaction in the reactor core, maintain the reactor power at prescribed level and change it over from one level to another, smooth power field over the core height, prevent and suppress xenon oscillations. RCCA consists of a head and suspension. RCCA head is a sleeve with console ribs, wherein the holes for AR suspension are made.

AE consists of the cladding filled with the absorbing material and sealed with end pieces by welding. Boron carbide B_4C and dysprosium titanate $Dy_2O_3 \times TiO_2$ are used as the absorbing material.

The main characteristics of the versions of equilibrium fuel cycles considered in the reactor core design are presented in Table 1 (neutronics indicated in the Table are given without regard for engineering safety margins, measurement errors and control of reactor thermal power).

Table 1 provides the neutronics of the fuel cycles that consideres all the reactivity effects. One of them is the power reactivity effect, used to extend the fuel cycle through the erduction of reactor plant thermal power and coolant temperature at the core inlet. This is shown in Table 1, where the characteristics of the fuel loadings with different length of operation are given.

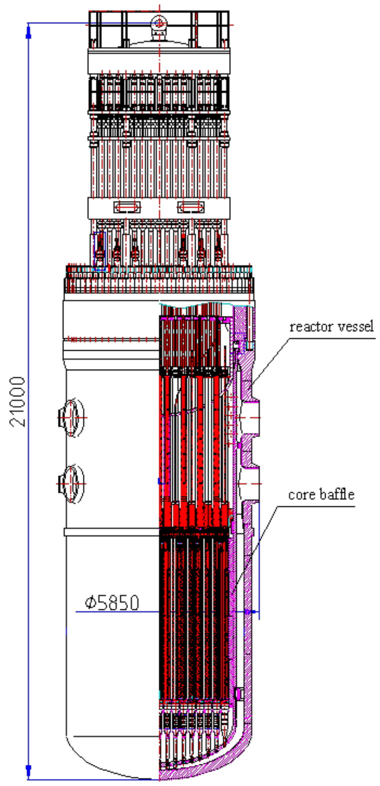

Figure 4. Reactor Longitudinal section.

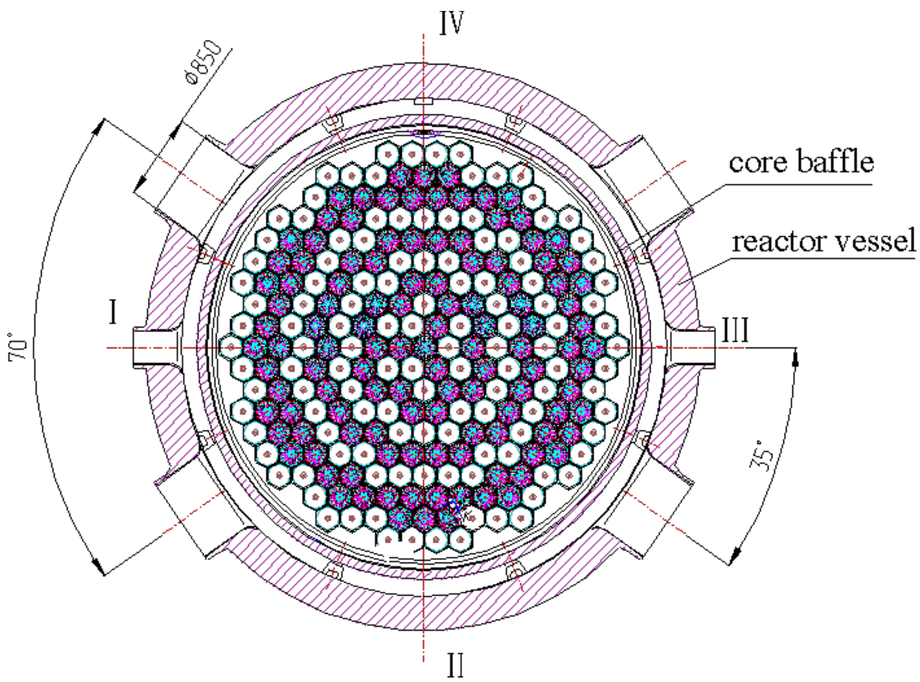

Figure 5. Reactor Cross section.

Parameter	Value			
	"Base" fuel cycle		"Prospective" fuel cycle	
	Ignoring operation at reactivity power effect	Considering operation at reactivity power effect	Ignoring operation at reactivity power effect	Considering operation at reactivity power effect
Number of fresh FAs loaded during refueling, pcs.	48		42	
Number of withdrawn FAs, pcs.:				
- after five fuel cycles	47		11	
- after six fuel cycles	1		31	
Average enrichment of makeup fuel in U-235, % mass.	4.502		4.850	
Duration of RP operation at nominal power, eff. day	327	295.1	318	281.8
Duration of RP operation between refuelings, eff. days	327	342.3	318	333.3
Burnup of withdrawn fuel, MWD/kgU:				
- average over all FAs	57.2	59.9	63.6	66.6
- maximum over FA	64.9	67.8	68.7	72.0
- maximum over fuel rod	70.1	73.1	73.6	77.1
- maximum over fuel rod pellet	75.2	78.1	79.7	83.0
- maximum over Gd fuel rod	61.0	64.0	63.3	66.9

- maximum over Gd fuel rod pellet	65.4	68.4	68.3	71.7
Maximum value of FA relative power	1.35	1.37	1.37	1.40
Maximum value of fuel rod relative power	1.51	1.55	1.55	1.59
Maximum value of Gd fuel rod relative power	1.23	1.24	1.27	1.29
Maximum value of linear heat rate of fuel rod, W/cm	275	290	280	299
Maximum value of linear heat rate of Gd fuel rod, W/cm	225	230	227	231

Table 1

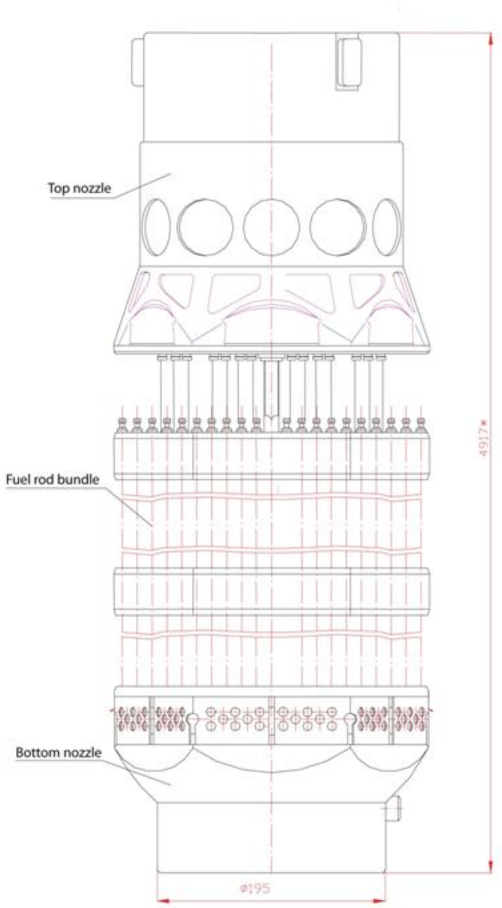

Figure 6. Fuel assembly

2.5.2. Reactor coolant pump set (RCP set)

RCP set is intended for creation of coolant circulation in the primary circuit. RCP set has an additional function of ensuring coolant circulation during coast down under different accidents with loss of power.

Spherical casing of pump is used in RCP set design (Figure 7); the pump and motor bearings operate with water lubricant excluding use of oil in the pump set.

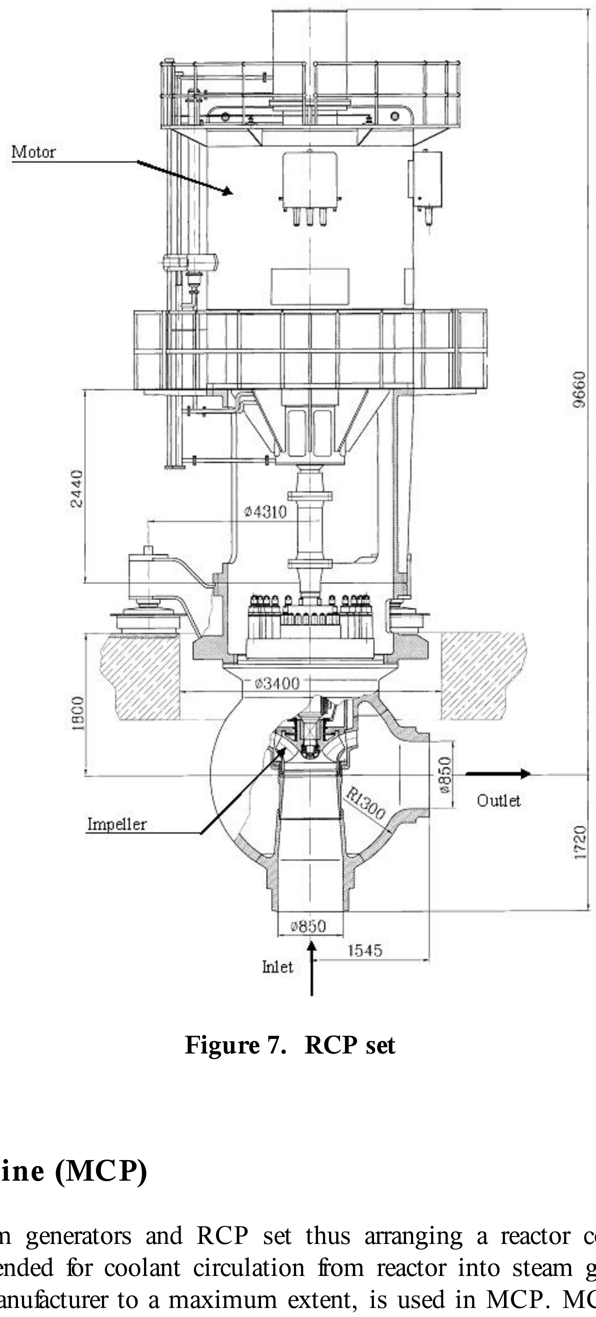

Figure 7. RCP set

2.5.3. Main coolant pipeline (MCP)

MCP connects a reactor, steam generators and RCP set thus arranging a reactor coolant system involving four circulation loops. MCP is intended for coolant circulation from reactor into steam generators and back. Pipelines Dnom 850, fabricated at the Manufacturer to a maximum extent, is used in MCP. MCP is designed with regard for implementation of " leak before break" concept in the reactor plant design.

2.5.4. Steam generator

Steam generator (Figure 8) is a constituent part of reactor plant V-448 and intended for generation of dry saturated steam owing to heat transferred by the primary coolant from the nuclear reactor core into steam generator. Dry saturated steam, generated by steam generator, is used in turbine generator plant for production of electric energy.

Horizontal steam generator (Figure 8) is used in VVER-1500 design. Design of steam generator is based on the domestic experience in development, manufacturing and operation of horizontal steam generators with steam distribution perforated sheet and corridor arrangement of heat-exchange surface. Service life of steam generator is 50 years.

2.5.5 Pressurizing system

Pressurizing system is a component of RP V-448 primary circuit and intended to:

- create pressure in the primary circuit and maintain it at nominal power;
- control pressure during RP startup and shutdown;
- protect the primary circuit equipment and pipelines against overpressure under design basis accidents and beyond design basis accidents;
- control the beyond design basis accidents (power governor).

Implementation of "feed and bleed" procedure, intended for control of beyond design basis accident and mitigation of its consequences is provided by main valve additional control line, permitting to decrease pressure in the primary circuit to 1.0 MPa at the remote control from MCR and ECR panels.

Pressurizing system involves: pressurizer (Figure 9), relief tank (Figure 10), surge lines with valves as well as safety valves. The pressurizer is a vertical cylindrical vessel with two elliptical heads and electrical heaters, located in the bottom part of the vessel, installed and fastened immovably on the lower support and fixed in the upper movable support for compensation of temperature expansions. The relief tank is intended for receiving steam-gas mixture from the primary circuit under operation conditions and consists of a horizontal vessel with two elliptical heads and vessel internals.

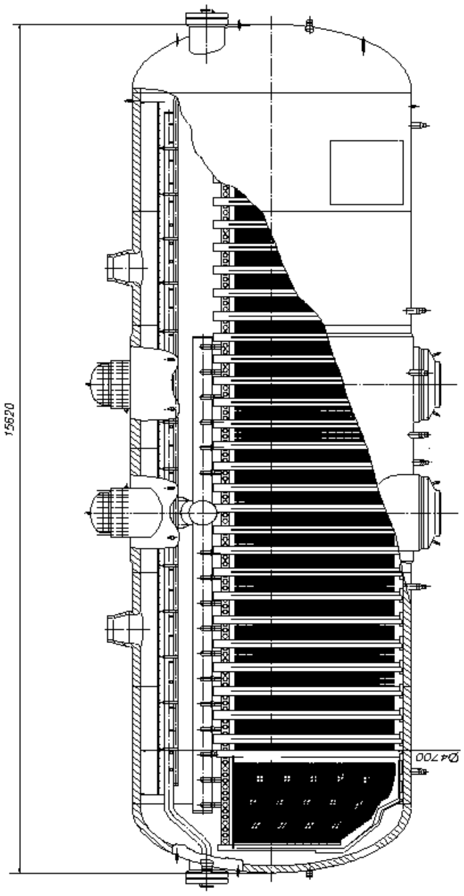

Figure 8. Steam generator

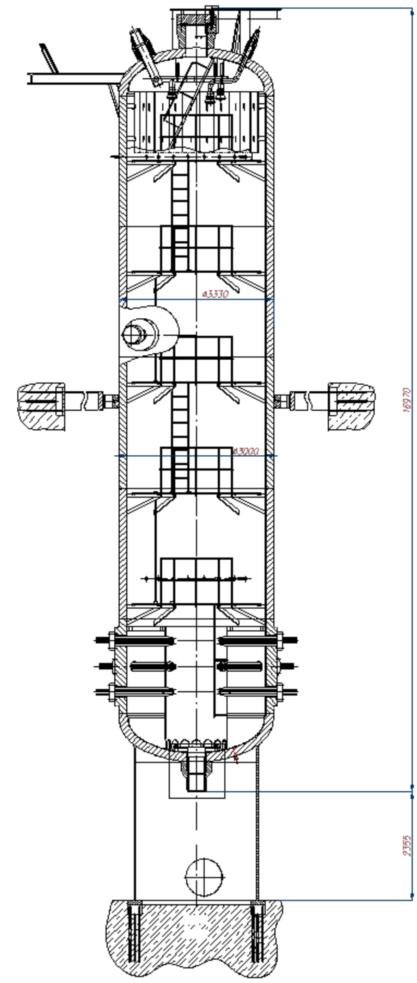

Figure 9. Pressurizer

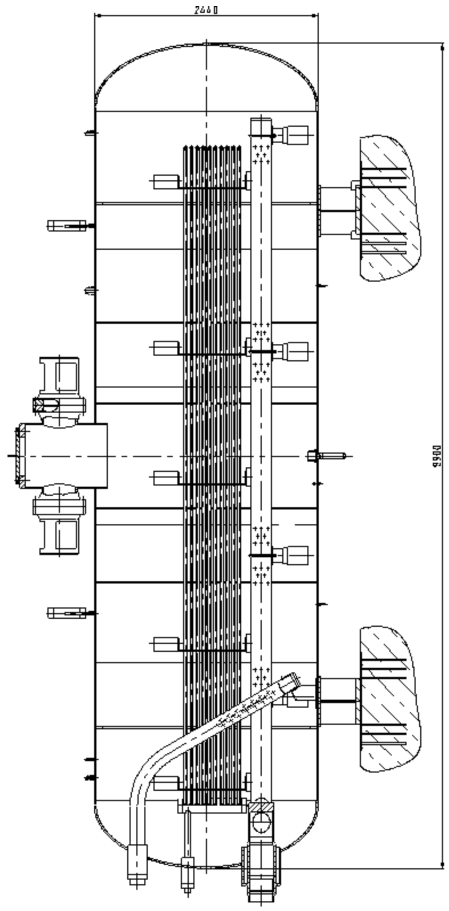

Figure 10. Relief tank

2.6. Fuel handling system

The fuel handling systems involve a fuel handling system for the fresh and spent fuel and refueling system of the reactor.

The fuel handling systems involve:

- spent fuel pool;
- equipment for reloading, storage and transfer of fuel, RCCA within the limits of reactor-spent fuel pool, and for FA inspection;
- defective assembly detection systems and equipment for storage of defective assemblies.

The main safety functions of fuel handling systems are:

- ensuring the nuclear safety (effective neutron multiplication coefficient should not exceed 0.95);
- preventing the transfer of cargoes above the stored fuel, if they are not the parts of lifting and refueling facilities;
- seismic stability of equipment under safe shutdown earthquake (SSE) and other natural phenomena;
- preventing the FA fall and uncontrolled transfer of mechanisms under SSE and loss of power.
- FA reloading under water.

The spent fuel pool is arranged within the sealed protective envelope along one axis with the reactor and the inspection wells for reactor internals, so that the pool and wells are located on different sides of the reactor.

The fuel assemblies are transferred by fuel handling machine under water and the depth of the water is sufficient to reduce the radiation level from the reloaded fuel assembly to the permissible level. The spent fuel pool can be isolated from the reactor core barrel by special lock for simultaneous implementation of procedures with the dry reactor and for withdrawal of assemblies from the spent fuel pool.

Capacity of the spent fuel pool is designed considering the requirement that simultaneous arrangement of the reactor core and storage of spent fuel in it are ensured within 10 years.

Two types of leak check of claddings (LCC) of fuel rods are provided for detection of defective assemblies – when the reactor is in operation and when the reactor is shut down. LCC in the reactor during operation permits identification, of the leaky fuel rods and degree of their leakiness by sampling the primary coolant.

The check of FAs is provided during transfer process in the mast of fuel handling machine on the shutdown reactor, as well as the check by bottle method with the help of analysis of water samples taken from the circuit of defective assembly detection system (DADS) where the bottle with the checked FA is located.

The leaky FAs can be repaired at the inspection and repair bench.

2.7. Operation conditions

Preferred operation condition is the condition of base load operation 100 % of the nominal power. RP equipment is designed with regard for load following modes. The list of RP design operation conditions in a set with NPP:

2.7.1 List of normal operation conditions

Conditions	Number of RP loading cycles for 50 years	Finite state
1 Filling the equipment with working medium, sealing the equipment of:		Cold
- reactor;	170	
- the rest of equipment	100	
2 Separate hydraulic test of the primary and secondary circuits:		
- for tightness;	170	
- for strength	80	
3 Heat up from "cold" state to "hot" state: - after shutdown for refueling; - after shutdown by process causes	60 200	Hot
4 Planned cooling down to "cold" state: - at the rate of 30 ^{o}C/h; - at the rate of 60 ^{o}C/h	130 130	Cold

Conditions	Number of RP loading cycles for 50 years	Finite state
5 Startup (turbine startup, load ascension at the rate of 60 % of current power per minute within 0-10 % N_{nom}, at the rate of 3-4 % N_{nom}/min within 10-70 % N_{nom}, at the rate of 1-1.5 N_{nom}/min within 70-100 % N_{nom})	3800	100 % N_{nom}
5.1 Startup from “hot” state to N_{nom}	3700	
5.2 Startup from the lower limit of control range to N_{nom} *	100	
6 Load decrease		
6.1 Load decrease to “hot” state at the rates specified in condition 5	1900	Hot
6.2 Load decrease to lower limit of control range at the rate of 20 % N_{nom} /min (under abnormal situations in the power grid) *	100	20 % N_{nom}
7 Operation under planned (dispatcher’s schedule) and unplanned load following modes		Steady state at power level within load range
7.1 Operation within the range of load variation from 100 to 50 % N_{nom} at the rate of variation in Unit power (electrical) 3 % N_{nom}/min	10000	
7.2 Operation within the range of 100-20 % N_{nom} (Unit operation in the power system at night and on days off) at the rates of load change for 24 hours of operation by the following program: 14-2-6-2. Digits mean the following cycle of operation: 14 h at the load of 100 % N_{nom}, two hours of load decrease from 100 % N_{nom} to partial load within the range of 100-20 % N_{nom}, six hours of operation at partial load and further load variation from the partial load to 100 % N_{nom} at the rates specified in condition 5	5200	
8 Stepwise load variation within the limits of control range * by 20 % of current power	250	Steady state at power level within control range*
9 Steady-state conditions including operation with partial number of loops	Unlimited	
9.1 Steady-state conditions at nominal power level over the range of frequency 49.0-50.5 Hz		

Conditions	Number of RP loading cycles for 50 years	Finite state
Conditions	Number of RP loading cycles for 50 years	Finite state
20 Trial operation of SG cooldown and blowdown system	60	10 % N_{nom}
21 Trial operation of emergency gas removal system	60	Cold
22 Trial operation of emergency pressure decrease system	60	Hot
23 Trial operation of emergency boron injection system		Cold
24 Drainage and unsealing of: - reactor; - the rest of equipment	170 100	Cold
25 Supply of nitrogen to the equipment and removal of gas blow off	340	
26 Fulfillment of handling procedures with fuel and reactor internals	60	
27 Supply of makeup water, with the temperature differing from the loop temperature by 100 °C, into the primary circuit	180	
28 Test of envelope tightness by special program: - for tightness;	50	
- for strength	20	
29 The secondary control of the Unit power variation within the limits of control range[2)] not less than ±10 % N_{nom} (electrical) at the rate of not more than 5 % N_{nom} /min	5×10^6	Steady state at power level within control range*
* The lower limit of control range - 20 % N_{nom} (electric power). The upper limit of control range - 100 % N_{nom} (electric power)		

2.7.2 List of anticipated operational occurrences

Conditions	Number of RP	Finite state

	loading cycles for 50 years	
1 Trip of one or several RCP sets from the operating ones (partial loss of the forced coolant flow rate)	100 per each set	68 N_{nom} 48 N_{nom} 40 N_{nom}
2 Trip of all RCP sets	30	Hot
3 Trip of the turbine by stop valves	130	Hot
4 Loss of power to the Unit auxiliaries	10	Hot
5 Loss of normal feedwater flow rate. Trip of all feedwater pumps.	40	Hot
6 Loss of normal feedwater flow in one SG	30	Hot
7 Injection of cold feedwater into SG nozzle of the emergency cooldown and blowdown system from the emergency feedwater storage system (at temperature of 5 °C)	20 (five cycles per SG)	100 % N_{nom}
8 Malfunction in feedwater system leading to decrease in feedwater temperature	40 (10 cycles per SG)	100 % N_{nom}
9 Increase in feedwater flow in SG due to malfunction of feedwater system	60 (15 cycles per SG)	100 % N_{nom}
10 Improper operation or failure of steam pressure controller resulting in increase of steam rate to the turbine	60	Hot
11 Improper operation or failure of steam pressure controller resulting in decrease of steam rate to the turbine	60	Hot
12 Inadvertent closure of isolation valve on one of steam lines	40 (10 cycles per SG)	Hot
13 Spurious injection of coolant into PRZ under RCP set pressure head	30	Hot
14 Malfunction of emergency pressure decrease system leading to water injection into PRZ at decreased temperature (to 20 °C)	20	Hot
15 Malfunction of emergency boron injection system that leads to water injection into MCP with decreased temperature	10	Hot
16 Malfunction of makeup and blowdown system leading to increase	10	Hot

Conditions	Number of RP loading cycles for 50 years	Finite state
in the primary coolant inventory as a result of water injection at decreased temperature		
17 Malfunction of chemical and volume control system that decreases boron concentration in the primary circuit	30	Hot
18 Spurious actuation of QBIS at power operation	30	Hot
19 Inadvertent actuation of ECCS accumulator during RP cooldown	8 (two per each accumulator)	Cold
20 Spurious startup of passive heat removal system	30	Hot
21 Spurious actuation of SG emergency cooldown and blowdown system	30	Hot
22 Spurious startup of "Primary-to-secondary leak" accident management algorithm	5	Cold
23 Inadvertent opening of steam bypass valve to turbine condenser (BRU-K)	16 (two per valve)	100 % N_{nom}
24 Inadvertent opening of steam dump valve to the atmosphere (BRU-A)	8 (two per each valve)	Hot
25 Inadvertent opening of SG PORV	16 (two per each valve)	Cold
26 Malfunction of PRZ level control system	10	Hot
27 Spurious startup of operation algorithm of emergency gas removal system (from PRZ, SG, reactor)	5	100 % N_{nom}
28 Loss of heat removal from the sealed envelope	60	-
29 Operator's error (failure of interlockings) in case of loop startup	10	Hot
30 Spontaneous withdrawal of RCCA group with operating rate from subcritical state or critical state at HZP	20	Hot
31 Spontaneous withdrawal of RCCA group with operating rate at power operation	20	Hot

32 Inadvertent drop of RCCA. Drop of one RCCA	150	Hot
33 Inadvertent drop of RCCA. Drop of RCCA group	50	38 - hot 12 - cold
34 Operator's error in xenon oscillation suppression	10	Hot
35 Incorrect loading of FA into the core and operation in improper position	1	Cold
36 Conditions of emergency frequency deviation in the grid: - from 50.5 to 51 Hz –till 10 s; - from 49 to 48 Hz –till 300 s; - from 48 to 47 Hz –till 60 s; - from 47 to 46 Hz –till 10 s.	-	100 % N_{nom} Or intermediate
37 Small compensable leaks of the primary circuit	50	Cold
38 Inadvertent opening of PRZ safety valve followed by their failure to seat	8 (two per each valve)	Cold

2.7.3 List of emergency situations (design basis accidents)

Conditions	Number of RP loading cycles for 50 years	Finite state
1 Small break loss of coolant accidents as a result of pipeline breaks with diameter up to 100 mm inclusive in the reactor coolant system	25	Cold
2 Large break loss of coolant accidents as a result of pipeline breaks in the reactor coolant system with diameter above 100 mm and up to MCP diameter	1	Cold
3 Ejection of AR in case of CPS drive housing rupture	2	Cold
4 Instantaneous seizure of one RCP set out of the operating ones	one per RCP set	Cold
5 Break of one RCP set shaft out of the operating ones	one per RCP set	Cold
6 Break of SG feedwater pipeline	4	Cold
7 Steam line rupture inside and outside the protective envelope	4	

Conditions	Number of RP loading cycles for 50 years	Finite state
8 Rupture of one SG heat exchanging tube	20 (five per SG)	Cold
9 Primary-to-secondary leak in SG as a result of collector cover lift-off	one per SG	Cold
10 FA drop during refueling	1	Cold
11 Drop of the spent fuel container	1	Cold
12 Accident with drop of the fresh fuel cask	1	Cold

[2] The system is designed to take the steam-gas mixture away out of the primary equipment (reactor top head, pressurizer and steam generator steam headers in case of beyond design basis accidents when coolant level appears inside the reactor vessel. It operates together with the PRZ PORV to reduce the primary pressure in order to mitigate the consequences of the DBA and BDBA.

[3] The gas blowoff system provides the purification and treatment of the controlled releases of the air and gases out of the primary equipment and directs them to the decontamination system.

[4] Power level after the appropriate conditions are over.

3.1. Safety requirements and design philosophy

The design is developed on the basis of requirements of up-to-date safety regulations and standards in nuclear power engineering of Russian Federation.

Safety criteria and design limits are established in accordance with the regulatory documents on safety valid in nuclear power engineering.

Determination of safety systems configuration in the present design is based on the following principles :

- single failure principle (acceptance of failure of one of system elements or operator's error);
- redundancy principle (the number of redundant elements not less than two in respect to any initiating events);
- diversity principle (functional, structural and diversity of operational states of systems);
- separation principle (arrangement of equipment of SS separate channels in separate rooms);
- protection against erroneous actions of operating staff (use of passive SS and the high level of automation of active SS);
- RP inherent safety principle (capability of RP to prevent development of initiationing events and accidents,

mitigate their consequences without environmental effects for a long time).

Safety Guides and other recommendations issued by International Atomic Energy Agency as well as requirements of European Utilities for NPP with PWR (Safety regulations EUR Volume 2, Chapter 2.1) were taken into account during design work.

3.2. Ensuring convenience and reliability of design

Convenience and reliability of safety system designs are enhanced due to use of passive systems not requiring participation of other systems for their actuation for ensuring safety alongside with active systems.

3.3. Safety systems and safety characteristics (active, passive and inherent safety)

The active systems are the systems the functioning of which depends on normal operation of other system.

The following active safety systems are provided in the design:

- quick boron injection system;
- steam generator emergency cooldown system;
- emergency gas removal system;
- emergency boron injection system;
- emergency and planned cooling down of primary circuit and fuel pool cooling system;
- main steam line isolation system.

The passive systems are the systems the functioning of which is related to the events causing their operation and don't depend on operation of other active systems.

The following passive safety systems are provided in the design:

- CPS of the reactor (emergency protection);
- emergency core cooldown system, passive part;
- systems of the core passive flooding;
- passive heat removal system;
- double protective envelope as well as a trap for the meltdown core.

Besides, overpressure protection system of the primary circuit and overpressure protection system of the secondary circuit are provided in the design, and these systems contain the valves (PORVs) which can be activated both by corresponding actions of automatics, and directly by reaching the prescribed values of parameters (active and passive actions).

3.4. Description of defense-in-depth

NPP safety shall be ensured owing to sequential realization of defense-in-depth concept, based on application of the system of physical barriers to the release of ionizing radiation and radioactive substances into the environment, and of the system of technical-and-organizational measures for protection and maintenance of the efficiency of these barriers as well as for protection of personnel, population and the environment.

The system of NPP Unit physical barriers involves the fuel matrix, fuel rod cladding, reactor coolant pressure boundary, sealed enclosure of the reactor plant and biological shielding.

The system of technical-and-organizational measures comprise five levels of defense-in-depth and involve the following levels.

Level 1 (Conditions of NPP siting and prevention of anticipated operational occurrences):

- NPP siting and assessment;
- establishing the buffer area and surveillance area around NPP with planning of protective measures;
- design development using conservative approach with improved inherent safety of RP;
- ensuring the required quality of NPP systems (components) and activities to be performed;
- NPP operation in compliance with the requirements of regulatory documents, process regulations and operation manuals;
- maintaining serviceable condition of systems (components), important to safety, by timely detection of flaws, adoption of preventive measures, replacement of equipment with expired service life and organization of effective system for documenting the results of work and inspection;
- NPP staff recruitment and provision of the required qualification level for the actions under normal operation and anticipated operational occurrences, including pre-accident situations and accidents, formation of safety culture.

Level 2 (Prevention of design basis accidents by the normal operation systems):

- identification of deviations from normal operation and their elimination;
- control during deviated operation.

Level 3 (Prevention of beyond design basis accidents by safety systems):

- prevention of development of initiating events into design basis accidents and of design basis accidents into beyond-design basis accidents by using safety systems;
- mitigation of accident consequences, failed to be prevented, by localizing the released radioactive substances.

Level 4 (Management of beyond design basis accidents):

- prevention of developing the beyond design basis accidents and mitigation of their consequences;
- protection of sealed enclosure against damage under beyond-design basis accidents and maintenance of their serviceability;
- return of NPP to the controlled state at which the chain fission reaction stops, provision of continuous cooling of nuclear fuel and retention of radioactive substances within assigned limits.

Level 5 (Emergency planning):

- preparation and realization, if necessary, of emergency plans at NPP site and outside its boundaries.

Safety systems for the reactor core protection and cooling down, that perform the RP main safety functions, involve:

- emergency boron injection system;
- emergency and planned cooling down of primary circuit and fuel pool cooling system;
- core passive flooding system (ECCS I stage hydroaccumulators);
- additional system for core passive flooding (ECCS II stage hydroaccumulators);
- passive heat removal system;
- steam generator emergency cooldown and blowdown system;
- emergency gas removal system;

3.5. Safety indices

The probabilistic safety analysis performed during the design has predicted that the probability of severe damage to the core is not higher than 10^{-5} per reactor-year, and a probability of the extreme release of radionuclides is not higher than 10^{-7} per reactor-year.

3.6. Safety systems for overcoming the design basis accidents and

beyond design basis accidents

The strategy of elimination of design basis accidents is based both on the use of the active, and passive safety systems. The strategy of elimination of beyond design basis accidents is based on the use of passive safety systems preferably.

The following active and passive safety systems are provided in the design:

The reactor CPS provides sufficient negative reactivity to the core to achieve a subcritical state and to maintain this state. In combination with the reactor core cooling systems and the reactor inherent safety properties, the efficiency of emergency protection is sufficient for achieving subcritical state under accidents up to temperature of 100 °C with pre-accident current boric acid concentration in the primary coolant as well as maintaining this state for a long time.

Quick boron injection system is intended for changeover of the reactor into subcritical state under the conditions without reactor scram (beyond design basis accidents, anticipated transient without scram [ATWS] type).

Emergency boron injection system is intended for emergency supply of high-concentrated boric acid solution into the reactor core under accidents with keeping high pressure in the primary circuit, as well as for pressure decrease by injecting cold water into the pressurizer steam space.

Emergency and planned cooling down of primary circuit and fuel pool cooling system is the active part of ECCS intended for fulfillment of the following safety functions under design basis accidents:

- supply of water into the reactor core under the conditions related to loss of integrity of the primary circuit;
- reactor core cooling down and subsequent long removal of decay heat during the accidents related to loss of integrity of the secondary circuit or failure of the normal heat removal system through the secondary circuit;
- supply of water into the spent fuel pool to ensure the fuel cooling under accident conditions.

The core passive flooding system involves the system of ECCS accumulators of the first stage and the system of ECCS accumulators of the second stage.

ECCS system of the first stage is a traditional system included in all designs of NPP with VVER-1000 that should provide the recovery of loss of coolant and reactor core cooling during initiating events with loss of the primary coolant.

The ECCS second stage accumulator system is an additional system intended for passive supply of boric acid solution into the reactor core at the primary pressure of 1.5 MPa and less for the purpose of flooding the core during loss of coolant accidents under the conditions of loss of all a.c. power for a long time.

The passive heat removal system is intended for long term removal of the reactor decay heat under the conditions of loss of all a.c. power both under the condition of maintaining the primary circuit integrity and in case of occurrence of leaks in the primary and secondary circuits. In case of the primary circuit leak, the system will operate together with the second stage hydro accumulators.

SG emergency cooldown and blowdown system is an active system for heat removal to the ultimate heat sink and it is intended for:

- removal of the reactor core decay heat and cooling down of the reactor plant during accident situations related to loss of power or loss of ability for normal heat removal through the secondary side including the leaks of SG steam lines and feedwater pipelines;
- removal of the reactor core decay heat and cooling down of the reactor plant during accident situations related to loss of integrity of the primary circuit, including the break of RCS pipelines (through intact loops).

Emergency gas removal system is intended to remove the permanent gases released from coolant in the top points of the primary circuit during accidents related to decrease in the primary circuit parameters, to prevent explosive concentrations and hydrogen explosions as well as to prevent the loss of natural coolant circulation in the primary circuit.

3.7. Earthquakes

The seismic impact of the following intensity is accepted for development of the base Design of NPP with VVER-1500:

- - SSE – of magnitude 7 as per scale MSK-64;
- - OBE – of magnitude 6 as per scale MSK-64.
- Under the impact of SSE the horizontal acceleration on the ground surface is 0.12g, the vertical one is 0.08g. Under the impact of OBE – 0.06g and 0.04g respectively.

3.8. Probability assessment of risk

The results of probabilistic safety analysis of the first level confirm, that fulfillment of all basic engineering principles of up-to-date defence-in-depth concept, including the principles of the functional and structural diversity, protection against common-mode failures, protection against erroneous actions of the personnel, physical division and assurance of higher reliability of fulfilling the safety functions is ensured in the design.

Proliferation resistance

Alongside with the assurance of reliable security the stock-keeping of / accounting for / the available nuclear materials at NPP, as well as control of their storage and transfer, excludes an outflow of nuclear materials outside NPP boundaries.

Safety and security (physical protection)

The complex of engineered-and-technical physical security features includes the engineered physical security features and alarm control and display system.

In the designed complex the installation of all kinds of engineered physical security features is provided according to the regulatory requirements. The engineered physical security features of the complex fulfill the following functions:

- creation of physical barriers that prevent an unauthorized man (intruder) crossing the security area boundaries including the access points;
- creation of the conditions for immediate response and arrival of security service to any place of the protected room;

The engineered physical security features of the complex include:

- security fence of the internal areas;
- window grates at the ground and first floors of the special building with the walls being boundaries of the internal areas.

Internal access control points (points of access to the exclusion area of the reactor compartment) are equipped with the turnstiles and lattice gate.

Alarm control and display system of the complex provide for solution of the following tasks of the object security:

- automatic detection of unauthorized access of an intruder into the security area, buildings, rooms and structures,
- objective confirmation of the information obtained from the detection equipment by the video surveillance equipment;
- emergency alarm calling signals from guard posts and from the secure rooms, buildings, structures;
- monitoring, registration and assessment of actions of operators and first-line groups;
- automated monitoring of people access to NPP territory, to the protected areas, building and rooms;

- around-the-clock remote TV monitoring of the situation at NPP territory, in the secure areas, buildings and rooms;
- automated reporting of the staff location and moving;
- detection of attempts of unauthorized carrying of nuclear materials and arms (metal objects);;
- video documenting of events;
- on-line broadcasting of information through communication lines;
- emergency warning;
- safety lighting of the security area perimeters, internal areas and access control points;
- non-interruptible power supply to the alarm control and display system.

Description of turbine-generator systems

A steam condensation turbine, type K-1500-7.1/50, working with saturated steam, is installed in the Unit with VVER-1500 reactor. The turbine has four cylinders without controlled steam extractions with separation and two-stage steam intermediate overheat (bled and main steam). The turbine is a principal drive of the generator.

Structural scheme of turbine 2 low-pressure cylinder (LPC) + high-pressure cylinder (HPC) + 2LPC.

The main technical characteristics of the turbine are given in Table:

Characteristic	Value
1 Nominal electric power of turbine, MW: (with no steam extractions to auxiliaries and heating unit)	1550
2 Thermal power of the reactor plant, MW (th); 3 Nominal steam parameters before turbine: pressure, MPa; temperature, ^{o}C; dryness fraction	4265 7.1 286.4 0.995
3 Temperature of superheated steam (after moisture separator/reheater) of water, °C	260
4 Temperature of feedwater, °C	225 ±5
5 Design temperature of cooling water, °C	20
6 Flow rate of cooling water into condensers, m^3/h	170 000
7 Steam pressure in the condenser at design temperature of cooling water, kPa	0.06
8 Heating load, GCall/h	250

7.1. Power supply system

Auxiliary electric power system of NPP is intended for power supply to the consumers that ensure:

- NPP operation under normal operation conditions;
- Unit operation under accident conditions including the loss of the emergency and standby power sources.

Auxiliary power consumers according to the requirements made for reliability of electric power are divided into the following groups:

- The first group – a.c. and d.c. power consumers not allowing, by the safety conditions or safe-keeping conditions of the main equipment, power outage more than by 20 ms under all conditions including the mode of complete collapse of a.c. voltage.
- The second group – a.c. power consumers that allow power outage for the period determined by the safety or safe-keeping conditions of the equipment. For safety systems the consumers of the second group require obligatory presence of power supply after reactor scram.
- The third group – consumers that permit power outage for the time of automatic changeover to standby power source and not requiring an obligatory existence of power supply after reactor scram.

The following auxiliary electric power systems are provided at power unit:

- normal operation system;
- reliable power supply system of normal operation;
- emergency electric power system.

The following levels of voltage and systems are accepted in the auxiliary power systems:

- alternating-current grid of voltage 10 kV for power supply to motors of 500 kW and higher and to transformers of 10/0.4 and 10/0.69 kV;
- alternating-current grid of voltage 0.66 kV, for power supply to motors from 500 kW to 200 kW;
- alternating-current grid of voltage 380/220 V, for power supply to motors below 200 kW and lighting;
- alternating-current grid of voltage 220 V and 380/220 V, for power supply to I&C facilities and equipment which operates off the inverters of uninterrupted power supply units;
- direct-current grid of voltage 220 V;
- direct-current grid of voltage 110 V for holding CPS drives in the prescribed position during short-term drop of voltage in the a.c. auxiliary power systems of normal operation.

7.2. NPP Unit Instrumentation and Control system

The following systems and subsystems are used for monitoring and control of the Unit equipment under normal operation conditions:

- top level system of Unit (TLS-U);
- control and protection system (RCIS and other systems and subsystems);
- MCDS;
- engineered safety feature-actuation system (ESF-AS) (in the part of systems uniting the functions of Normal Operation and safety)
- radiation monitoring system (RMS);
- I&C NO.
- automatic video surveillance and diagnostic system;
- fire protection equipment;

- MCR control boards and panels;
- LCR;

For monitoring the parameters characterizing safe operation there is used information output to personnel from the control safety system.

Spent fuel and waste management

Spent nuclear fuel storage system in the reactor building is intended for cooling the spent nuclear fuel withdrawn from the reactor.

The functions of SNF storage system in the reactor building are:

- arrangement of SNF, discharged from the reactor during refueling as well as arrangement of emergency discharge of the core fuel;
- holding (storage) of spent fuel for not less than three years before removal from the reactor compartment;
- decay heat removal from SNF;
- assurance of biological shielding of the staff from the fuel stored in the spent fuel pool.

Capacity of the spent fuel pool of the reactor compartment is accepted from the condition of simultaneous arrangement in it of SNF in the amount of not less than five scheduled reactor reloadings, arrangement of leak-tight bottles for storage of the leaky and defective FAs of 25 pcs. At least as well as arrangement of FA skeletons for installation of excess number of RCCA. Besides, the capacity of spent fuel pool should permit to arrange complete discharge of the core at any moment of NPP operation. During annual refueling the replacement of 49 FAs is performed and, in all, for nine years of operation it will amount 441 FAs, and the core consists of 241 FAs, i.e. the required capacity of the spent fuel pool racks should amount not less than 441+241=682 FAs.

The maximum (constructively obtained) capacity of the racks for compacted fuel storage amounts to 734 cells.

Registration and control of FA arrangement, number and transfer is carried out by the serial numbers marked on FA. Each FA has a marking with designation of FA No. and its enrichment. Identification of FA in the reactor and in the spent fuel pool is provided with the help of television system of fuel handling machine.

Safe operation of in-containment SNF storage system is provided by maintaining water temperature in SFP of not more than 60 °C as well as by using the following systems, components, equipment and structures:

- spent fuel pool;
- spent fuel pool cooling system, including filling, draining and feeding;
- fuel pool water purification system;
- process control system;
- radiation monitoring;
- ventillation;
- leak collection and return check;
- compacted fuel storage racks;
- leak-tight bottles;
- stop logs of spent fuel pool;

Plant layout

9.1. General arrangement

Scheme of general arrangement is presented in Figure 11.
Scheme of general arrangement is developed for two Units with regard for further extension.

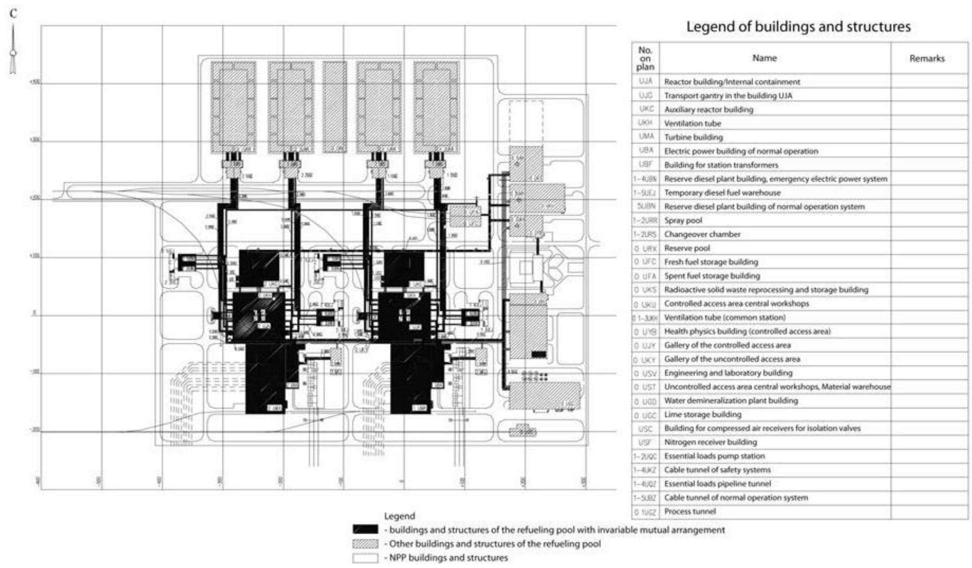

Figure 11. Scheme of general arrangement

9.2 Reactor building

Reactor building consists of double protective cylindrical reinforced concrete envelope inside thereof there is an accident confinement area – ACA and civil structures, located on the common foundation part of the building.

Protective envelope is an element of accident localization system and consists of two envelopes: internal sealed and external one protecting it from external impacts.

Between two envelopes there is formed an annulus space through which the cable channels of safety systems penetrate. The annulus space serves for collection of radioactive leaks through the internal protective envelope under emergency conditions.

Inside the sealed envelope in the accident localization area there is placed the reactor plant, spent fuel pool, the systems working with the primary circuit parameters, ventilation systems and the equipment for fuel handling procedures and repair.

The central place in the accident localization area is occupied with the reactor. On different sides of the reactor core barrel there are the spent fuel pool and inspection well for reactor internals, RCS boxes, which contain steam generators, RCP and main coolant pipelines, pressurizer, relief tank and the tanks for quick boron injection.

On the supporting slab of sealed envelope (elevation 0.000) the rooms of special water purification system, the valve chambers of "contaminated" pipelines, ventilation facilities are located as well as the trap for meltdown fuel in the lower part of reactor cavity.

ECCS hydroaccumulators of I and II stages are located at maintenance elevation (elevation +29.700 m). The washer of the "contaminated" equipment is located near the spent fuel pool as viewed from the transport lock.

The passive heat removal system (PHRS) is located on the dome-shaped part of the external envelope. PHRS heat exchangers are fastened to the upper cylindrical part of envelope from the outside.

PHRS heat exchangers are located around the perimeter of envelope (at elevation +43.260 m). Supply of cooling medium (atmospheric air) to PHRS heat exchangers is carried out through air duct at elevation +36.500 m, ensuring the uniform cooling in case of direction change of wind. PHRS draft tubes are placed on the dome of external protective envelope. At the high point of the dome of external envelope the filter of medium discharge from annulus space (deflector) is placed.

According to operation condition the volume inside the sealed envelope is referred to controlled access area and it is divided into two categories of rooms: unattended (personnel access during power operation is prohibited) and periodically attended rooms. The reactor vessel cavity, rooms of RCS boxes, rooms wherein auxiliary equipment of the primary circuit (SWP-1 and SWP-2 filters, heat exchangers of blowdown) and their valve chambers are referred to the unattended rooms. The remaining volume of sealed envelope is referred to periodically attended rooms.

These categories of rooms have separate ventilation. RCS box walls have unloading apertures for pressure release during accident situations.

The floors of periodically attended rooms are connected by ladders, the elevator from the maintenance elevation is provided.

Access of the personnel into the sealed envelope is carried out onto intermediate annular maintenance platform (elevation +10.800 m) through the basic locks from the controlled access area of civil structure (elevation of the lock axis +11.700 m).

For emergency exit from the sealed envelope there is an emergency lock, also placed at elevation +10.800 m (elevation of axis +11.700 m).

All handling procedures with fuel and equipment are carried out through the transport lock with elevation of axis +32.800 m and a transport gantry outside the envelope.

The basic load-lifting and fuel handling procedures within the sealed envelope are performed by the bridge electrical polar crane.

The sealing circuit passes over the internal surface of sealed envelope and over the upper plane of the supporting slab of envelope (elevation 0.000 m).

Unsealed part of building contains:

- foundation part;
- civil structure above elevation 0.000 m of the building.

The foundation part of the reactor building is a space between the supporting slab of the sealed envelope and the foundation slab of the building. Elevation of the top part of the foundation slab is minus 10.050 m. Elevation of the top part of the sumps of emergency cooling down pumps is minus of 12.750 m.

In the foundation part of the envelope there are the planned cooling down of primary circuit and fuel pool cooling system, the emergency cooling down pumps of primary circuit, the rooms of I&C and RM systems, the rooms and corridors of safety channels, common annular corridor, serving also for vertical stressing of the internal envelope.

The active process safety systems located in the foundation part of the building are broken-down into two independent safety subsystems with two independent channels in each subsystem with total-lot redundancy. The rooms for different channels are separated from each other by walls.

In the civil structure above elevation 0.000 from the side opposite to the turbine building there are special gas purification system and exhaust ventilation center of controlled access area. This part of civil structure together with foundation part is the area of controlled access.

MCR and ECR rooms are located from the opposite sides of protective envelope within civil structure.

From the side of the turbine building within civil structure there are a cable floor and CPS board, main steam lines and feedwater pipelines, the blowing and exhaust ventilation center. These rooms are referred to the uncontrolled access area.

Areas are completely isolated.

Access of the personnel into the unattended and periodically attended rooms of the controlled access area is carried out from the controlled access area of auxiliary reactor building at elevations +10.800 m and minus 5.550 m.

Access of the personnel into the rooms of uncontrolled access area is provided from pedestrian gallery of uncontrolled access area at elevation +7.800 m.

The "contaminated" ladders with emergency exits to the street are also used for evacuation of the personnel.

9.3 Auxiliary reactor building

Auxiliary reactor building is designed as the separate building which comprises the rooms for process systems of normal operation, a special water purification, tanks with the process media, exhaust ventilation center and electrical rooms, which are referred to the controlled access area, as well as the blowing ventilation center and electrical rooms, that are referred to the uncontrolled access area.

The building is located at a distance of 9.0 m from the reactor building and connected to it by the pedestrian, transport and communication galleries and channels.

9.4 Turbine Building

The turbine building is intended for arrangement of the secondary side systems and equipment related to generation of power. The turbine building involves the turbine plant, feed pumps, deaerator, auxiliary equipment. The turbine building is located on the reactor axis. Between the turbine building and reactor building there is a fire-safety passage.

The turbine building of Unit K-1500 MW consists of turbine hall and deaerator stack. The equipment of oil supply system to the turbine plant and hydraulic hoisting of rotors is located within the room, which is separated from the basic volume of turbine hall by fire wall, type 1.

All the rooms of the turbine building are referred to the "uncontrolled" access area.

The turbine building equipment is attended by two bridge cranes of load carrying capacity 125/20 t and 15 t.

9.5 The electric power building of normal operation

The electric power building of normal operation is intended for arrangement of the electrical systems and I&C systems of normal operation as well as it involves the support systems for ventilation, water supply, sewage and fire-fighting.

9.6. The building of reserve diesel generator station (RDGS)

The building of RDGS of emergency power supply system with temporary warehouse of diesel fuel.RDGS is a part of the emergency power supply system.Two RDGS buildings are provided for one Unit.The equipment of each safety system channel is placed in the respective isolated cell. Two cells are for each building.RDGS are located from the opposite sides of reactor building that makes provision for impossibility of common-cause failure as a result of aircraft crash.External doors and gates are designed for external air shock wave.

9.7 Fresh fuel storage building

The building is referred to the controlled access area. The fresh fuel storehouse is intended for fuel reception and storage. In the storehouse the fresh fuel is checked and prepared for loading into the reactor. The storehouse is located above zero elevation in order to exclude possibility of flooding with water. The building structure is designed as the rigid construction from monolithic reinforced concrete united with the spent fuel storehouse.

9.8 Spent fuel storage building

The building is intended for the storage of spent fuel before its removal outside NPP for reprocessing.

9.9 Spray pools with changeover chambers

Two pools are provided for each Unit.

Each spray pool was intended for cooling the water of two channels and is included in an underground structure, divided into two parts (by number of channels) with the open and closed sections. Such division makes provision for keeping water volume in the pool under the impact of blast wave or tornado.

Changeover chamber is intended for arrangement of the unit of the spray pool inlet and outlet pipelines with isolation valves, operation of these valves ensures the different operation conditions of the spray pool sections.

The pool water tightness is provided by using concrete of respective class as well as by the device of internal and external waterproofing.

9.10 Essential loads pump stations

Two essential loads pump stations are designed for each Unit by one per each independent subsystem of cooling water.

Each of two pump stations is placed not far from the spray pools with changeover chambers. The pump stations for their protection against possible aircraft crash are spatially arranged in the plan.

The pump station supplies two physically distributed channels of cooling water that have independent process parts, and respectively consist of two separate sections.

9.11. Solid radioactive waste reprocessing and storage building

The building is intended for reprocessing and storage of solid radioactive waste (SRW).

The building comprises two volumes located on the common foundation slab:

- SRW storage unit;
- SRW reprocessing unit.

SRW reprocessing unit is intended to reprocess the solid radioactive waste coming for storage and further to reduce them in volume.

Plant performance

10.1. RP operation

Operation of power units is carried out in accordance with the requirements of regulatory documents, process specifications and operation manuals. RP operation may occur both during the base-load operation and maneuvering mode.

10.2. Reliability

Reliability indices:

- frequency of emergency shutdowns not more than 1 for the year of operation;
- availability factor, not less than 0.9.

The given data are confirmed by the availability and reliability analyses of the equipment carried out within the limits of detailed project report.

10.3. Load factor indices

Capacity factor amounts to 0.93.

Annual load factor is – 0.9.

Development status of technologies relevant to the NPP

Composition and structure of the basic components, equipment and systems of NPP with VVER-1500 have, basically, solutions being laid down in the previous designs and improved in accordance with the up-to-date requirements, which allow to enhance the operational characteristics of NPP with ensuring the required safety level.

The basic engineering solutions for NPP are confirmed by operating experience of RP with VVER for more than 1400 reactor-years, including more than 500 reactor-years of operation of RP with VVER-1000.

Use of the evolutionary approach to solution of target problems of designing has allowed to develop design AES-2006 for a short time that meets the very up-to-date Russian standards and regulations.

Design of NPP with VVER-1500 has a necessary level of competitiveness and has the justified prospects for serial construction both in Russia and abroad.

Deployment status and planned schedule

Duration of the construction of power unit with VVER-1500 is established on the basis of the construction schedule that involves the work of the base period, starting from building the foundation slab of reactor building and finishing by the commissioning stage of the Unit.

The composition of work and the duration of preparatory period are determined with reference the basic design to the to the specific construction site.

Duration of the base period of the Unit construction is determined by the period of construction of the reactor building and amounts, according to the schedule:

- 50.5 months - before physical startup,
- 51.9 months - before power startup,
- 60 months - before preliminary acceptance of the Unit for service.

Duration of the building and erecting works is established with regard for man-hour thereof calculation is made by respective transactions GESN-2001 [1] and using the transactions guidelines.

[1] GESN is the acronym for the Russian State Standard of estimated prices for construction budget in different sphers.

References

1. Правила ядерной безопасности реакторных установок атомных станций (ПБЯ РУ АС-89). ПНАЭ Г-1-024-90, Москва, 1999.(Safety regulations for reactor plants of nuclear power plants (PBYa RU AES-89), PNAE G-1-024-90, Moscow, 1999.)
2. Установка реакторная В-448 Техническое задание на разработку технического проекта реакторной установки ВВЭР-1500. 448-ТЗ-001, ОКБ «ГИДРОПРЕСС», 2006. (Reactor plant V-448 Technical assignment for elaboration of technical design of reactor plant VVER-1500. 448-TZ-001, OKB "GIDROPRESS", 2006.)
3. Общие положения обеспечения безопасности атомных станций. ОПБ-88/97. НП-001-97(ПНАЭ Г-01-011-97), Москва, 1997. (General provisions for safety assurance of nuclear power plants (OPB-88/97). NP-001-97 (PNAE G-01-011-97), Moscow, 1997.)

Technical data

General plant data

Reactor thermal output	4250 MWth
Power plant output, gross	1560 MWe
Power plant efficiency, net	35.7 %
Mode of operation	Baseload and Load follow
Plant design life	50 Years
Plant availability target >	93 %
Seismic design, SSE	0.2
Primary coolant material	Light Water
Secondary coolant material	Light Water
Moderator material	Light water
Thermodynamic cycle	Rankine
Type of cycle	Indirect

Safety goals

Core damage frequency <	10E-6 /Reactor-Year
Large early release frequency <	10E-7 /Reactor-Year

Nuclear steam supply system

Steam flow rate at nominal conditions	2390 Kg/s
Steam pressure	7.34 MPa(a)
Steam temperature	289 °C
Feedwater temperature	230 °C

Reactor coolant system

Primary coolant flow rate	31594 Kg/s
Reactor operating pressure	15.7 MPa(a)
Core coolant inlet temperature	298 °C
Core coolant outlet temperature	330 °C
Mean temperature rise across core	32 °C

Reactor core

Active core height	4.2 m
Equivalent core diameter	3.85 m
Average linear heat rate	12700 KW/m
Fuel material	UO_2 and UO_2 + Gd_2O_3
Cladding material	Zirconium alloy
Outer diameter of fuel rods	9.1 mm
Number of fuel assemblies	241
Enrichment of reload fuel at equilibrium core	4.92 Weight %
Fuel residence time	342 Months
Average discharge burnup of fuel	57.2 MWd/Kg
Burnable absorber (strategy/material)	B_4C + $Dy_2O_3TiO_2$
Control rod absorber material	Dysprosium titanate, Boron carbide
Soluble neutron absorber	Boron

Reactor pressure vessel

Inner diameter of cylindrical shell	4960 mm
Wall thickness of cylindrical shell	230 mm
Design pressure	17.65 MPa(a)
Design temperature	350 °C
Base material	Steel 15H2NMFA
Total height, inside	12500 mm

Transport weight	430 t

Steam generator or Heat Exchanger

Type	Horizontal
Number	4
Total tube outside surface area	9490 m^2
Number of heat exchanger tubes	15261
Tube outside diameter	16 mm
Tube material	08H18N10T
Transport weight	560 t

Reactor coolant pump (Primary circulation System)

Pump Type	GCNA-1667
Number of pumps	4
Pump speed	1500 rpm
Flow at rated conditions	7.5 m^3/s

Pressurizer

Total volume	103 m^3
Steam volume (Working medium volume): full power	76 m^3
Steam volume (Working medium volume): Zero power	38.7 m^3
Heating power of heater rods	6300 kW

Primary containment

Type	Dry, double
Overall form (spherical/cylindrical)	Cylindrical
Dimensions - diameter	49 m
Dimensions - height	67.2 m

Residual heat removal systems

Active/passive systems	Passive

Safety injection systems

Active/passive systems	Active and Passive

Turbine

Type of turbines	K-1500-7.1/50
Number of turbine sections per unit (e.g. HP/MP/LP)	2LPC+HPC+2LPC
HP turbine inlet pressure	7.1 MPa(a)
HP turbine inlet temperature	286 °C

Generator

Type	TVV-1500-4UZ
Rated power	1500 MVA
Active power	1650 MW
Voltage	27 kV
Frequency	50 Hz

Condenser

Condenser pressure	6.08 kPa

Feedwater pumps

Number	4
Pump speed	5500 rpm
Head at rated conditions	750 m
Flow at rated conditions	3300 m^3/s