Status report 102 - VVER-600 (V-498) (VVER-600 (V-498))

Overview

Full name	VVER-600 (V-498)
Acronym	VVER-600 (V-498)
Reactor type	Pressurized Water Reactor (PWR)
Coolant	Light Water
Moderator	Light water
Neutron spectrum	Thermal Neutrons
Thermal capacity	1600.00 MWth
Gross Electrical capacity	600.00 MWe
Design status	Conceptual Design
Designers	Gidropress
Last update	22-07-2011

Description

Introduction

The results of the reactor plant (RP) VVER-600 design studies for a unit of medium power 600 MW(e) are presented in the document.

These studies were purposed to demonstrate the possibility of RP VVER-600 design development and also receiving of initial data required for evaluation of the nuclear power plant (NPP) VVER-600 design competitiveness in Russian and in the world market.

Studies provided to the present day in cooperation with the St. Petersburg Research and Design Institute ATOMENERGOPROJECT (SPbAEP) and RRC "Kurchatov Institute" show that design bases of Unit with VVER-600 allow developing of safe and economically competitive NPP of medium power. It is promoted with:

- High degree of design availability;
- Reference character of Unit design and engineering solutions;
- High characteristics of fuel utilization;
- Minimum quantity of RP main equipment units;
- Main equipment compact lay-out;
- Main equipment transportability using vehicle of any type;
- Safety and economic characteristics at the level of requirements for NPP specified in EUR requirements;
- Corium localization within the reactor vessel under beyond design basis accident;
- Maximum use of R&D results provided for VVER RP verification;
- Availability of proved technologies for designing, manufacturing, construction, mounting, operation,

maintenance and repair of VVER type NPPs.

2.1 RP and power unit goals

The following goals are realized in the design of VVER-600 Unit:

- High degree of RP and Unit engineering advancement (maximum high efficiency, load factor, availability factor);
- Maximum achievable RP technical and economic characteristics regarding fuel burn-up, RP metal consumption, reactor building construction and mounting work scope and terms, operating costs;
- Justification of operating characteristics meeting the requirements for NPP specified in EUR requirements;
- Safety analysis is conducted according to the requirements of Russian Federation (RF) regulations, IAEA and EUR requirements;
- VVER-600 RP design is developed on the basis of equipment for VVER-1200 (LAES-2), including equipment design direct similarity; the reference character of RP main engineering and design solutions is provided;
- Lay-out and conditions of VVER-600 Unit are based on NPP-2006 (LAES-2) design solutions considering prospective development and improvement;
- Design service life is 60 years;
- The design provides corium catching within the reactor vessel during severe beyond design basis accidents and, respectively, non-availability of corium ex-core catching facility is justified;
- The equipment is manufactured commercially using proved technology;
- Results of R&D provided for VVER RP verification are maximally used;
- The design is developed according to RF regulatory document requirements, considering IAEA and EUR requirements.

The design of the RP VVER-600 incorporates OKB "GIDROPRESS" long-term experience of RP design activities, and also accounts for the operating experience accumulated for NPPs of high, medium and low power. RP VVER-600 design has the following VVER traditional features providing NPP efficiency, reliability and safety proved by NPP Units long-term operating experience (~1400 reactor-years):

- Core structural materials – zirconium alloys;
- Oxide fuel of low enrichment (<5% U^{235});
- Top arrangement of control rods;
- Hexahedral jacket-free FAs with a rigid skeleton;
- RP loop-type lay-out with equipment arrangement in separated boxes;
- Horizontal steam generators for dry saturated steam with tube bundle corridor arrangement;
- Steam pressurizer of large volume;
- RCP set with shaft tightening and remote electric motor comprising a flywheel for coastdown time increase under loss of power;
- RP cooldown using primary and secondary natural circulation under loss of power;
- Full testability of equipment and pipelines base metal and welded joints using the advanced test methods; proved methods providing relatively low dose commitments and short time of equipment and pipelines test procedures, maintenance and repair;
- Possibility of RP main equipment transportation by rail, road, river and sea transport.

2.2 Design studies

The decision was made to develop a two-loop VVER—600 RP on the basis of VVER-1200 RP with maximum use of the VVER-1200 equipment (reactor, steam generator, pressurizer etc.). Advantages of two-loop lay-out in comparison with a 4- or 6-loop layout are:

- Reduction in RP metal content;
- Time reduction of equipment mounting;

- Decrease in containment internal diameter;
- Reduction of time and dose commitments for equipment in-service testing, maintenance and repair (according to the performed evaluations the RP medium and overhaul repair period will be decreased by 20-25% as compared with VVER-1200).

Two-loop RP with pressurized water reactors have not been developed by domestic atomic power engineering before. Efficient mixing at the core inlet of two loop flows with varied temperature, boric acid concentration, etc. is required to provide two-loop reactor cooling. Maximum possible similarity of RP equipment to VVER-1200 equipment is provided: primary and secondary parameters are taken as for VVER-1200; the number of FAs in the core is 163;

RP VVER-600 main performance parameters are given in Table 1.

Table 1. VVER-600 RP main parameters.

Parameter	**VVER-600**
Core thermal power, MW	1600
Number of circulation loops	2
Number of FAs in the core	163
Operating pressure at the core outlet, MPa	16.2
Primary system design pressure, MPa	17.64
Core inlet/outlet temperature, °C	299/325
Steam operating pressure at SG steam header outlet, MPa	7.0
Secondary design pressure, MPa	8.1
Containment minimum internal diameter ensuring primary equipment and spent fuel pool arrangement, m	36.6

2.3 Reactor

Reactor engineering solutions are typical for VVER-1000 and VVER-1200 recent designs, considering reduced number of nozzles Dnom 850 for connection of main coolant pipeline legs (see Fig.1).

Table 2. Reactor design main parameters

Parameter	**VVER-600 reactor**
Number of FAs in the core	163

Parameter	VVER-600 reactor
External diameter of the reactor vessel in the beltline region, mm	4645
Reactor height overall dimension, mm	19315
Number of control and protection system (CPS) drives	79

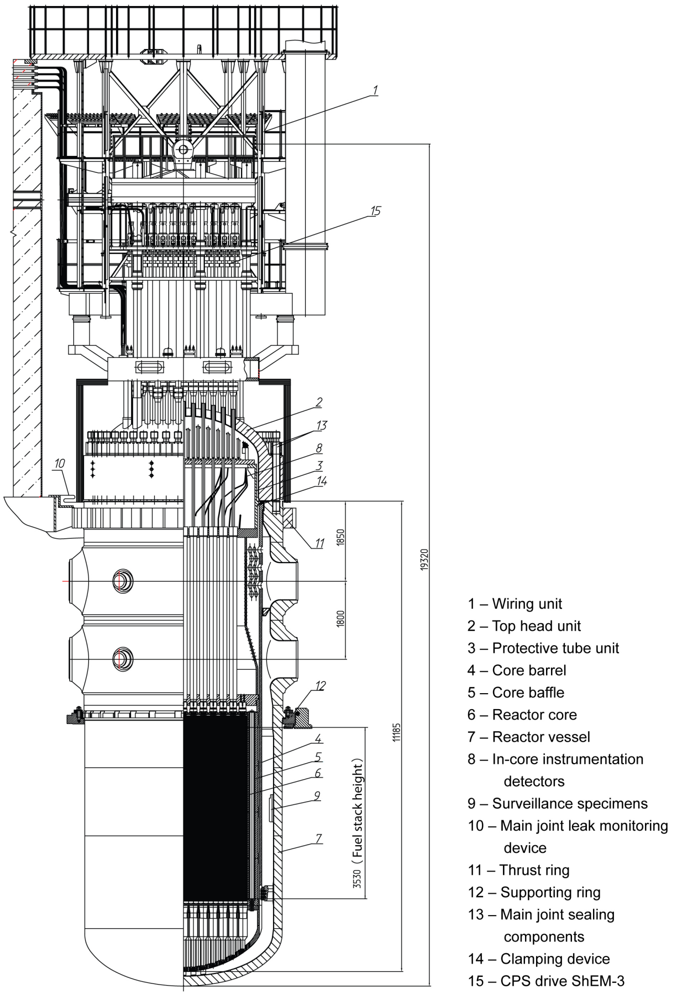

Figure 1. VVER-600 reactor

2.4 Fuel assembly

The design of fuel assemblies (FAs) for VVER-600 is based on design solutions of FAs for VVER-1200 and VVER-1500. FA design enables to ensure stability of FA shape and dimensions; a possibility of FA assembling-disassembling at the Unit is ensured for replacing the leaky fuel rods. The difference of the fuel assembly from an alternative fuel design (TVS-2) is the shortened top and bottom nozzle.

Fuel rods with uranium-gadolinium fuel (gadolinium fuel rods), installed in FAs, are used for flattening the core radial power field, compensating the reactivity margin in the beginning of fuel burnup cycle, and ensuring negative coolant temperature coefficients during reactor operation within. Gadolinium oxide (Gd_2O_3) is used as integrated burnable absorber.

Table 3. Main parameters of FA for VVER-600

Parameter	VVER-600 FA
Number of FAs in the core	163
FA-prototype	TVS-2M
Number of fuel rods and Gd-fuel rods in FAs, pcs.	312
Number of guiding channels for RCCA, pcs.	18
Height of fuel column in cold state, mm	3530
FA spacing in the core, mm (hexagonal grid)	236
FA axial overall dimension, mm	4330

2.5 Steam generator (SG)

Horizontal steam generators designed in OKB "Gidropress” are characterized by large secondary water inventory (to mitigate accident scenario); primary vertical collectors and tube bundle corridor arrangement enable to reduce the intensity of heat exchanging tube corrosion damages as compared to vertical SGs. The design of SG provides for access to heat exchanging tubes for their inspection and plugging of individual tubes.

Steam generator PGV-1000MPK taken from VVER-1200 design is used for VVER-600 RP.

2.6 Reactor coolant pump set (RCP)

Reactor coolant pump set GTSNA-1391 taken from VVER-1200 design is used for RP.

2.7 Main coolant pipeline (MCP)

Tube units and sharply bent stamped elbows Dnom 850, approved in commercial production and used in MCP of all RPs of VVER-1000 and VVER-1200, are used in a set of the main coolant pipeline. Routing of MCP legs in VVER-600 RP differs from that in VVER-1000 due to some other mutual arrangement of the reactor, steam generator and RCP set.

2.8 Pressurizer (PRZ)

The pressurizer is also taken from VVER-1200 design. The pressurizer of large volume enables:

- Compensation of primary coolant temperature expansions under power variation within the whole control range is an ensured passive principle (without inventory make-up or decrease);
- PRZ coolant inventory passively provides for minimum permissible level of the primary coolant under transients with reactor scram.

2.9 Safety system equipment

Hydroaccumulators of the emergency core cooling system, tanks of the core passive flooding, PRZ pilot-operated relief valves, SG pilot-operated relief valves, SG passive heat removal system, and design of sprinkling system for the reactor plant, are taken from VVER-1200 design.

2.10 RP equipment arrangement in the reactor building.

2.10.1 Evaluation of containment dimensions

The RP equipment arrangement in the reactor building considers the requirements for minimization of the containment dimensions. Unit lay-out design solutions are presented below:

- Reactor building containment comprises RP equipment and pipelines operating under primary pressure and tanks of passive safety systems with water inventory for the emergency core cooling;
- Spent fuel pool is also arranged within the reactor building containment;
- Spent fuel is reloaded from the reactor into the pool under protective water layer;
- The containment is isolated from the equipment and pipelines operating under primary pressure to protect the containment structures against damages in case of primary LOCA accidents;
- The channels of the reactor heat removal system are physically separated and, hence, absence of dependent failures is ensured;
- Radiation protection of the equipment and personnel from ionizing radiation of the core and radioactive coolant is provided both during operation at power, and during shutdown;
- Cavities and wells are arranged inside the containment for inspection of the reactor components, as well as and transport-engineering equipment for handling these components;
- Testability and maintainability of the equipment and pipelines is ensured within the scope of 100%;
- The possibility of in-service replacement of the main equipment is provided (except for the reactor vessel).

VVER-600 RP equipment arrangement in the reactor building is shown in Figures 2, 3, 4.

Layout solutions for RP equipment provide the containment internal diameter of the order of 36-37 m; the area of rooms within the containment is thus decreased by 30 % in comparison with AES-2006. Height of the containment from the under-reactor sump to the bonnet top is 52 m. This dimension is considerably decreased in comparison with the respective dimension of AES-2006 containment (by »12 m) due to the elimination of an under-reactor corium catcher and owing to decrease in the radius of the bonnet hemisphere in VVER-600 design.

Decrease in the containment dimensions provides for positive economic effect explained by cut down of construction scope.

Still, the containment dimensions are governed not only by specific features of the reactor plant equipment design and arrangement, but also by capabilities of the containment to withstand pressure under accidents. Further optimization of the containment dimensions will be provided at the stage of RP and Unit detailed design.

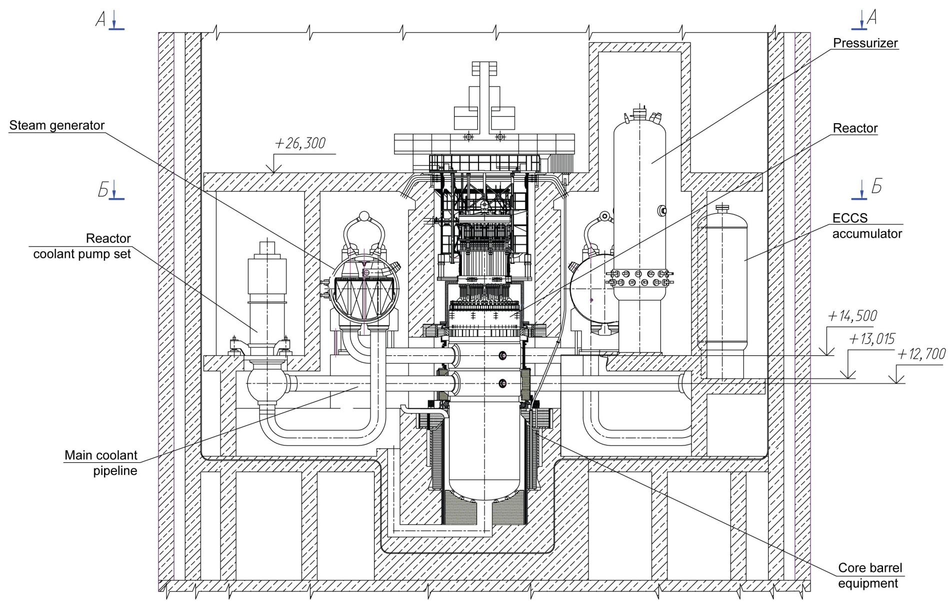

Figure 2 – Reactor plant main equipment arrangement

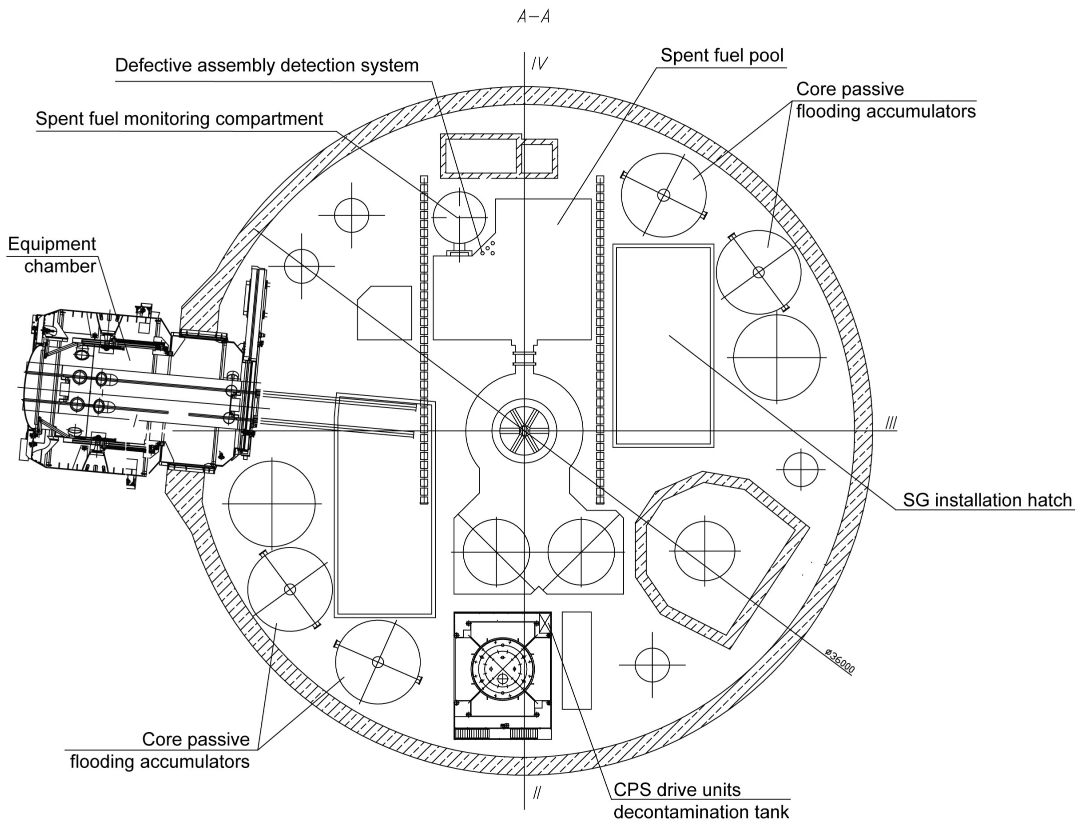

Figure 3 – Reactor plant main equipment arrangement

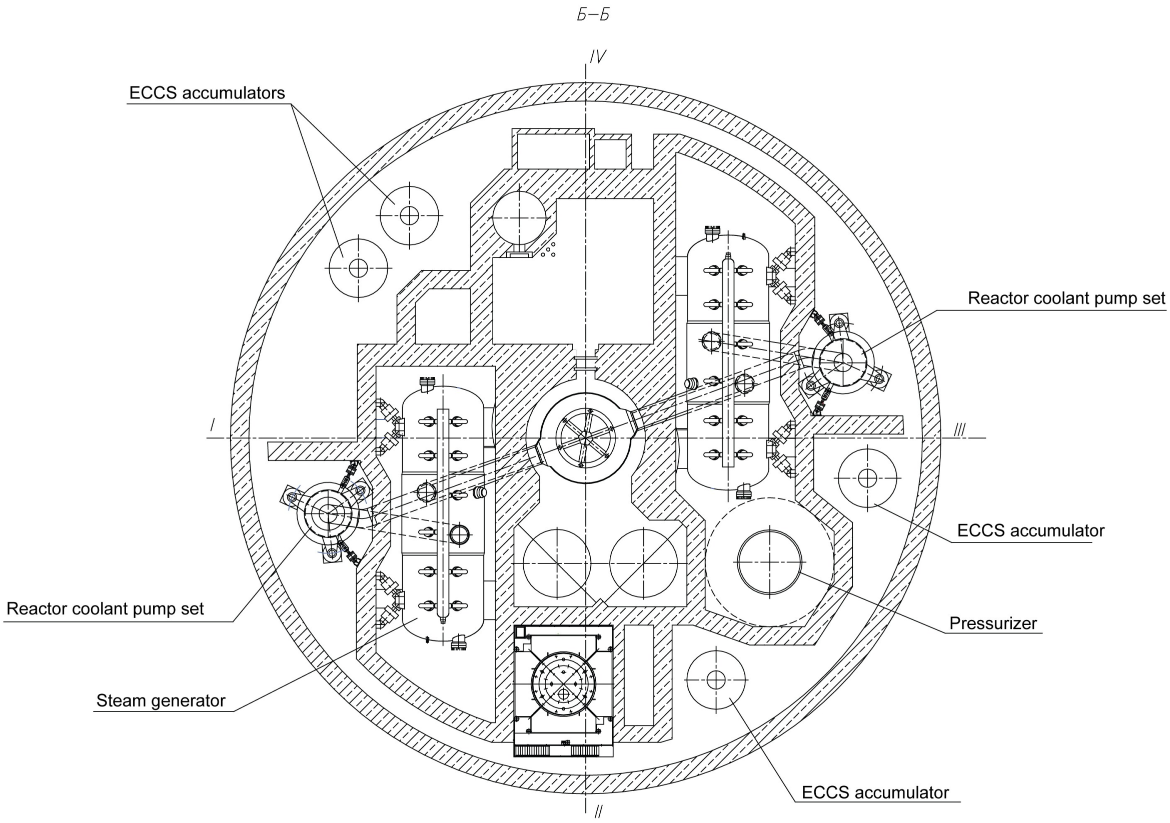

Figure 4 - Reactor plant main equipment arrangement

Description of safety concept

The following goals are established for optimization of VVER-600 RP protective safety systems:

- Keeping the acceptable safety level;
- Reference character of engineering solutions;
- Reduction in the number of equipment, pipelines, valves, etc. of safety systems.

Main trends in optimization to achieve the mentioned goals are:

- Consideration of total contribution of passive and active systems into fulfilling the required safety function in choosing the safety system (SS) characteristics and design bases;
- Consideration of contribution of all available accident management hardware into deterministic and probabilistic safety analyses with meeting the requirements for the analyses stated in the valid regulatory documents;
- Consideration of mutual redundancy factor for fulfilling the required safety function of SS participating in overcoming the initiating event based on the diversity requirements;
- Optimization of lay-out solutions preventing simultaneous failures, dependent on initiating event, of safety systems required for fulfilling the safety function, on the basis of the analysis of specific governing initiating events;
- Optimization of SS equipment repair schedule purposed to prevent simultaneous failure of the whole SS channel (including redundancy of the diesel-generator to be brought into repair).

Development of the design considering the mentioned trends results in more particular statements enabling to determine properties and requirements for specific systems, and namely:

- constructing the complex of safety systems using active and passive systems;

- mutual redundancy of active and passive systems if required;
- determination of safety system design bases proceeding from specific initiating events and failure analysis considering mutual redundancy of active and passive systems if required;
- two-channel structure of active safety systems;
- two-channel structure of passive safety systems connected to the secondary circuit;
- four-channel structure of passive safety systems connected to the primary circuit;
- prevention of simultaneous, dependent on initiating event, failures of channels of passive and active systems;
- prevention of simultaneous, dependent on initiating event, failures of low and high pressure safety injection systems;
- redundancy of two-channel system of emergency power supply in bringing of common-station diesel-generator into repair;
- retaining of corium within the reactor vessel with internal and external cooling.

The passive safety systems and beyond design basis accident (BDBA) management systems include:

- ECCS passive part;
- core passive flooding system;
- SG passive heat removal system (PHRS) (water);
- passive sprinkling system;
- SG pilot-operated relief valve (PORV);
- PRZ passive heat removal system (PORV);
- corium retaining system.

The active safety systems and BDBA management systems include:

- emergency protection;
- high pressure safety injection system;
- low pressure safety injection system;
- emergency boron injection system;
- emergency gas removal system;
- BRU – A – steam dump valve to the atmosphere (BRU-A) (in each steam line upstream of the main steam isolation valve (MSIV)).
- emergency feedwater system;
- sprinkling system;
- boric acid solution storage system;
- decay heat removal system.

Supporting safety systems include:

- system of intermediate circuit for essential consumers cooling;
- system of cooling water for essential consumers;
- system of emergency power supply;
- common-station emergency power supply system.

Characteristics of passive safety systems and beyond design basis accident management system at the Unit

Choosing the parameters of passive safety systems for V-600 RP is made proceeding from the condition of overcoming the design basis accidents with primary coolant leaks without consideration of active systems operation and provision of long-term (not less than 24 h) heat removal from the reactor core by passive systems.

The following passive systems are used for such conditions in V-600 RP design:

- ECCS passive part consists of the accumulators with water inventory to be supplied into the reactor under nitrogen blanket pressure;
- core passive flooding with water inventory of the accumulators to be supplied into the reactor under leveling pressure of water column;
- SG PHRS provides energy removal from SG to the water stored in special tanks arranged outside the containment. Coolant circulation in tanks between SG and heat exchangers is provided under leveling pressure of water column.

The concept of corium retaining within the reactor vessel

Corium retaining within the reactor vessel is assured in VVER-600 design as the means of management of beyond design basis accidents with core melting. This enables to omit the under-reactor corium catcher from the design (being the obligatory NPP component in the latest designs of VVER-1000 and VVER-1200) thus reducing the capital costs of Unit construction.

Duration of decay heat removal with the use of passive safety systems can be from 24 to 72 hours depending on water inventory in the tanks of core passive flooding system. The number of tanks in this system is governed by RP layout solutions.

Complete core melting (100%) is postulated in simulating the degradation mechanism.

Reactor vessel cooling is assured owing to outside cooling of the vessel bottom part with water from the emergency pool. Emergency pool water is supplied from the spent fuel pool and inspection wells. Design solutions for reactor vessel civil structures assure increase in the containment emergency pool water level to the elevation required for reliable cooling of the reactor vessel. This is achieved owing to arrangement of the reactor bottom part in the specially designed containment sump (see Fig.5). The steam resulting form the cooling of the outer surface of the vessel is discharged via the supporting truss special channels into the containment where it is condensed. In development of the Unit design, measures are implemented to prevent cold water getting onto the reactor vessel during operation at power.

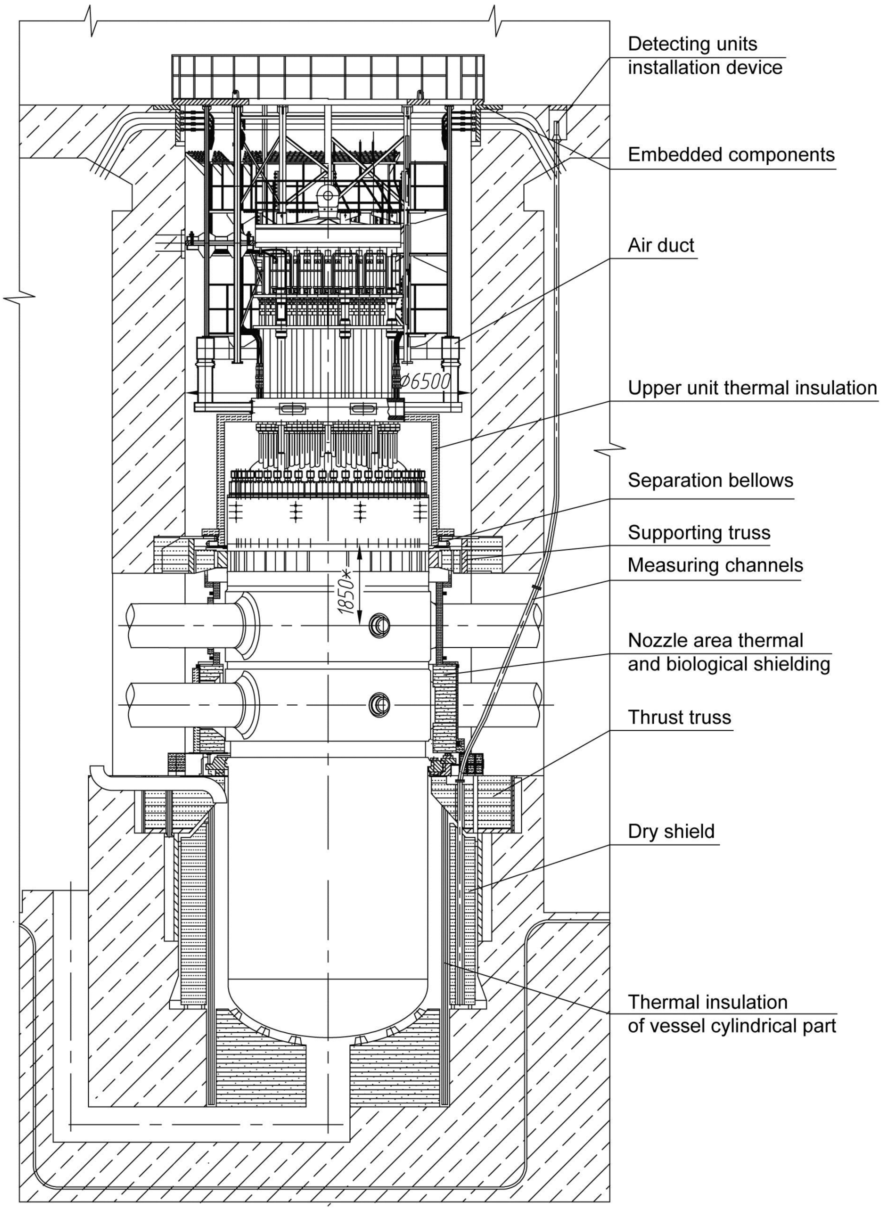

Figure 5. Reactor concrete cavity equipment

Proliferation resistance

Accounting of NPP available nuclear materials, control of their storage and transfer provided alongside with reliable security prevents nuclear materials release outside NPP boundaries.

Safety and security (physical protection)

5.1 Purposes of physical security system (PSS)

PSS is a part of integrated engineering and organizational measures for nuclear and radiation safety during NPP operation.

PSS prevents the unauthorized actions in relation to nuclear and radioactive materials, physical barriers being on the way of propagation of ionizing radiation and radioactive substances, and also in relation to process systems, their equipment and the operational personnel that fulfills control of the process.

5.2 Tasks of physical security system

PSS realizes the following tasks:

- prevention of unauthorized actions;
- detection of unauthorized invasion of an intruder into the secure areas, buildings, rooms and structures;
- objective confirmation of the information obtained from the discovering facilities using videomonitors;
- call of response group by the alarm-calling signals from the guard posts and from the secure rooms, buildings, structures;
- detain (slow-down) of intruder motion;
- suppression of unauthorized actions;
- monitoring, registration and assessment of actions of operators and first-line groups;
- automated monitoring of persons' access to secure areas, building and rooms;
- automated reporting of the staff location;
- a round-the-clock remote TV monitoring of the situation in NPP territory, in the secure areas, buildings and rooms;
- video-documenting of events;
- on-line broadcasting of information through wire communication operative channels;
- detention of the persons involved in preparation or fulfillment of unauthorized actions.

PSS functions under normal operating conditions. Under accident conditions and during realization of emergency actions PSS shall not prevent evacuation of the personnel and access of the specialized units taking part in emergency actions (suppression of fires, decontamination of buildings, constructions and territory) to the secure areas.

5.3 Physical security system structure

PSS is realized on the basis of integrated automated system of physical security - including:

- integrated engineered safety features of physical security within design boundaries for buildings and constructions;
- integrated engineered safety features of physical security over NPP perimeter:
- integrated engineered safety features of local security.

Description of turbine-generator systems

The turbine unit of LMZ design is supplied in the set with turbine generator of "Electrosila" design.

Turbine unit K-640-6.9/50 is a single-shaft set with one high and mean pressure cylinder (HMPC) and two double-line low-pressure cylinders (LPC). The structural diagram: HMPC + 2LPC + generator. Low-pressure cylinders are unified with LPC of operating turbine K-1000-60/3000.

Steam condensation turbine unit K-640-6.9/50 without controlled steam bleeding with intermediate separation and single-stage steam superheating is purposed for direct driving of a.c power generator T3V and also for heat release to 250 GCal/h for the needs of district heating for industrial and social consumers with time-temperature chart 150/70◦C.

Schematic thermal system comprises four low pressure heaters (LPH), deaerator of increased pressure (1.3 MPa abs.) working under transient sliding pressure and two high pressure heaters (HPH).

The turbine unit is designed with the system of steam intermediate separation and superheating. Two-stage steam separation (downstream HPC and MPC) and single-stage intermediate steam superheating downstream of MPC with live steam (S+S+SH) are used in the design in contrast to the system with a single-stage separation and intermediate steam superheating downstream of high-pressure cylinder (HPC) in separator-superheaters (SSH) used previously in turbine K-1000-60/3000.

Generator T3V-630-2UZ has total water cooling and does not require usage of hydrogen. Auxiliaries power supply is provided with voltage 6 kV, 0.4 kV and 220 V of d.c. power.

Turbine unit main performances are presented in the Appendix.

6.2 Feedwater system

The feedwater system consists of the main and auxiliary feedwater systems including:

- deaerator;
- 3 feedwater pumps;
- 2 auxiliary feedwater pumps;
- feedwater heaters;
- feedwater lines with isolation and control valves.

Feedwater pump capacity is 1470 m^3/h, pressure head is 900 m of water columm, auxiliary feedwater pump capacity is 250 m^3/h, pressure head is 830 m of water columm. Feedwater temperature at steam generator inlet during operation at power is 230-180 ^oC, and 164 ^oC with disconnected high-pressure heater (HPH).

Electrical and I&C systems

7.1 Main electric system

Synchronous turbine generator of T3B-630-2 type with rated power 645 MW, voltage 20 kV, rotational speed 3000 rpm which is completely cooled with water, the internal space is filled with air at pressure close to atmospheric pressure are applied in the design. It is possible to use fire-resistant liquid OMTI in the bearing lubrication system.

Turbine generator T3B-630-2 is explosion-proof and fire-safe, and meets the requirements for the equipment applied in production of explosion-and-fire hazard group “B”.

For operation in a set with turbine generator T3B-630-2 the main step-up three-phase double-wound transformer of 800 MVA is provided.

For station-service redundancy two standby station-service transformers of 63 MVA are installed at NPP.

7.2 Station-service system

The following station-service power supply systems are provided at the Unit:

- normal operation system;
- system of reliable power supply to systems of normal operation important to safety including two main diesel-electric sets of 4 MW each and storage batteries;
- emergency power supply system including two emergency diesel plants of 1 MW each and storage batteries.

Storage battery capacities are chosen such that under complete loss of power of the Unit the safe state is ensured during 24 hours.

7.3 Emergency protection system and other safety systems

The integrated control and protection system and control of safety systems (CPS-CSS) is an electrical system intended for continuous monitoring of all critical operating parameters, and when the limiting values are reached it calls for actions aimed at safety assurance. The reactor plant CPS-CSS is a part of the Unit I&C. The CPS-CSS includes the reactor control and protection system (CPS) and the engineered safety features actuation system (ESFAS).

It is based on the systems directly related to safety assurance: two emergency protection-preventive protection (EP-PP) sets (as a part of CPS) and two sets of ESFAS equipment. The sets are arranged in different rooms for their physical separation. Functionally, the sets are completely separated from one another owing to absence of direct (galvanic) coupling between them and of coupling to other I&C subsystems.

The CPS and ESFAS have an independent set of primary converters per two channels of safety systems.

The control and protection system is intended for reactor emergency and preventive protection, automatic and manual control of reactor power, monitoring of parameters and CR positions, documenting of events and interchanging of signals with the integrated I&C subsystems.

As to protection functions the CPS provides the reactor emergency protection by de-energizing all CR drives and dropping of CR under gravity to the extreme lower position in response to respective initiating signals. EP action does not cease at removal of initial cause signal.

As to control functions the CPS provides the reactor startup, change in the reactor power during startup, shutdown, changing from one mode over to another, reactor power automatic control, preventive protection ensuring automatic power decrease by sequential downward motion of CR groups, starting from the working group, when the monitored parameters reach the corresponding setpoints, or reactor off-loading and power limitation over PP channels depending on the equipment state. Action of preventive protection ceases at removal of initial cause signal.

The ESFAS makes an automatic control of active components of engineered safety features in occurrence of DBA initiating events and also monitoring of passive components of safety systems.

The ESFAS functions are subdivided into control functions, monitoring functions and information functions.

As to control functions the ESFAS provides generation of control commands by the necessary actuators of safety systems and systems important to safety, as well as remote control of the actuators from the I&C located in MCR and ECR.

As to monitoring function the ESFAS provides monitoring of values of current process parameters, their preprocessing and comparison of their current values with the assigned values of setpoints; calculation of values of composite parameters and their comparison with the setpoints, monitoring the EP discrete signal, monitoring the state of actuators.

7.4 Layout of the main control room (MCR)

The CPS-CSS is controlled by operators in MCR and emergency control room (ECR) from workstations of the operator-process engineer of reactor compartment.

At VVER-640 plant the unified system of on-line control and monitoring system is provided with the use of video terminals, including data support integrated system OM 650. Process servicing in the OM 650 system is provided using the mouse through the so-called “windows of service”.

The NPP control room concept provides:

- use of video terminals for the systems important to safety (state of Unit safety, information on safety systems state and operation) and for the normal operation systems (startup, power operation, shutdown);
- use of remote viewing displays (general survey information);
- use of traditional mosaic control panels as the reserve for the systems important to safety.

The MCR provides the following areas of control:

- use of video terminals for the systems important to safety (state of Unit safety, information on safety systems state and operation) and for the normal operation systems (startup, power operation, shutdown);
- use of remote viewing displays (general survey information);
- use of traditional mosaic control panels as the reserve for the systems important to safety.

The MCR provides the following areas of control:

area of shift supervisor	responsibility for supervision of the shift work;
operative area	control of Unit technological process;
inoperative area	control of the power supply and waste processing, system maintenance;
safety area	Unit shutdown, subcriticality, reactor cooling and long-term reactor transfer into safe state in case of OM 650 failure;
information area	Centralized and continuous representation of information for coordination of joint actions of maintenance personnel.

The following supplementary areas are arranged near the main control room:

service area	I&C diagnostic problems, actualization of I&C software and documentation, CSS in-service inspection;
admission area	tasks on assurance of physical security of nuclear installation.

The tasks of process control are divided between two operators – the reactor plant operator and the turbine plant operator.

At each workstation an access is provided to the information on control of normal operation systems and control of safety systems. Each workstation for the process control area is equipped with OM 650 color monitors.

The safety area is realized with the use of mosaic technique. The control commands from the safety areas of higher priority among all commands of OM650 on-line control system, with this, a priority of CSS automatic commands is kept.

In the shift supervisor area there are two color monitors to provide access to all information represented in the area of process control.

Spent fuel and waste management

Complex of spent nuclear fuel (SNF) handling systems is provided in the design of NPP with RP V-498. The following systems are included into the complex of SNF handling:

- SNF ex-pile storage;
- fuel rod leak-check in shutdown reactor;
- spent fuel pool and reactor core barrel pipelines;
- spent fuel pool water supply for purification;
- spent fuel pool water cooling;

- spent fuel pool liner leak-check;

SNF ex-pile *storage system* is purposed for unloaded spent fuel cooling for SFA activity and decay heat lowering to acceptable values for transportation.

SNF ex-pile *storage system* functions are:

- receiving of SNF unloaded from the reactor during transportation and accidents;
- spent fuel cooling (storage) before transportation from the reactor building;
- SNF decay heat removal;
- personnel biological shielding for protection from SFP nuclear fuel.

Spent nuclear fuel storage system provides nuclear fuel storage and cooling in power unit reactor building within 10 years considering scheduled refuellings and reactor core full unloading at any moment of NPP operation.

SNF storage system is a spent fuel pool (SFP) supplied with all required equipment and systems.

fuel rod leak-check in shutdown reactor

The functions of *fuel rod clad leak-check (CLC) system* in shutdown reactor are:

- FA leak-check in the extension shaft of fuel handling machine (FHM) using CLC system;
- FA leak-check using defective assembly detection systems (DADS);
- leaking FA check and repair using check and repair bench (CRB).

Necessity of CLC in shutdown reactor is determined from operating reactor CLC data.

Leak-check is provided by gas or water samples analysis.

Spent fuel pool water cooling system is purposed for:

- SFP fuel assembly decay heat removal under all design operating conditions, design and beyond design basis accidents;
- providing of radiation protective layer above reactor core fuel assemblies in the core barrel, fuel pool and refueling well;
- core barrel loading during refueling work;
- core barrel, inspection well unloading;
- fuel pool and refueling well unloading during the work on fuel pool and refueling well liner repair;
- spent fuel pool cooling pipelines are used for water supply from sprinkler pumps for reactor internals inspection well filling during refuelling;
- spent fuel pool cooling pipelines are used for water supply from sprinkler pumps for reactor internals inspection well filling during post-accident measures after beyond design-basis accident related to core melt-down and corium release outside the reactor vessel.

Spent fuel pool liner leak-check system is purposed for leak-check of SFP liner, reactor internals inspection well, refueling well and determination of water availability in the gap between liner and wall in each linear volume*:*

The following work is provided in spent nuclear fuel handling equipment under power unit decommissioning after specified service life.

At the preparatory stage:

- removal of spent nuclear fuel NPP Unit;
- coolant discharge and drainage of the reactor and spent fuel pool;
- coolant regeneration in the special water treatment facility;
- equipment decontamination as per standard procedure.

At the stage of power unit preparation for controlled preservation:

- equipment preservation in standard place.

At the stage of controlled preservation:

- equipment cooling within a long-term period.

Equipment decontamination and long-term cooling result in 3-order decrease of the activity level which is mainly determined by availability of long-living radionuclides, such as Cs^{137}.

At dismantling stage:

- radiation monitoring which at this stage comes to evaluation of radioactive situation using standard radiation indicators, taking of "smear" test samples from contaminated surfaces of equipment and pipelines for isotopic composition and a radioactivity level analysis;
- dismantling of equipment and engineering structures;
- radiation monitoring of dismantled equipment and construction wastes;
- metal residual radioactivity level analysis;
- metal "hard" decontamination and remelting.

Liquid radioactive media (LRM) as well as solid and gaseous radwaste are released during NPP operation.

Liquid radioactive media treatment. NPP design stipulates the systems for liquid radioactive media collection, regeneration and temporary storage in all operating conditions of nuclear units, exactly:

- special sewage system;
- sump water regeneration system;
- liquid radioactive media interim storage system.

The following main requirements are considered for LRM regeneration systems design:

- decrease of LRM total volume supplied for regeneration;
- decrease of LRW volume purposed for temporary storage, supplied for solidification facility and subsequently for disposal in RAW storages;
- decrease of unbalanced water volume in the area of controlled access.

NPP design stipulates main technological solutions purposed at:

- LRM separated collection and regeneration depending on chemical composition and activity;
- localization of main radionuclides in minimum volumes;
- low-waste technologies of LRM and LRW regeneration;
- low-waste decontamination technologies;
- decrease of filter regenerations number;
- enhanced requirements for supplied equipment quality and operation to decrease uncontrolled leaks;
- use of decontaminated water for primary coolant make-up and process system auxiliaries in controlled access area.

LRM regeneration system designs consider designing, operation and improvement experience of RF and foreign NPPs as well as prospective technologies.

Liquid radioactive media temporary storage system is purposed for LRM full volume temporary storage not below three months for short-lived radionuclides decay before subsequent regeneration.

Liquid radioactive media supplied into the temporary storage system are:

- concentrated salts of evaporation plants;
- waste ion-exchange resins (filtering materials);
- sludge.

LRM regeneration technologies provide the finished product which satisfies valid regulatory document requirements for radwaste storage and disposal.

Preferably Russian enterprises shall be designers of the technologies, prospecting inclusive, and manufacturers of respective equipment.

Solid radioactive media treatment. Solid radwaste (SRW) are:

- overalls, footwear, individual protection means not subject to decontamination;
- metal wastes;
- construction and thermal insulation materials;
- filtering components of gas purification and vent system filters;
- solidified liquid radioactive media.

Quantity of the radwaste, including solidified supplied for storage annually – not above 50 m^3/year (per 1000 MW) from one power unit. The quantity specified does not include the contaminated equipment and tools after repair and maintenance as well as the volume of containers НЗК-150-1.5П (NZK-150-1.5P).

Treatment of all types of radwaste is provided at all NPP life stages to ensure safety during their handling and storage.

LRW collection and classification by activity level and regeneration methods is provided in radwaste locations by loading in respective containers or casks of single use. Test classification is provided in the building of waste reprocessing with storage.

Solid radwaste supplied for storage shall be subject to the following regeneration types for decrease of their volume:

- decomposition;
- combustion;
- compression.

Power unit design solutions shall provide for normal operation:

- radwaste temporary storage in the form of conditioned SRW in containers НЗК-150-1,5П. Container capacity is assumed for 5 years of NPP operation with the possibility of extension to 10 years. The containers shall be subsequently transported to regional depositories.
- storage of high-activity SRW in specially equipped volumes for high-activity wastes during NPP full service life. High-activity SRW is supplied into depositories packed into special capsules using the facilities comprised in the set of NPP fuel handling equipment.

The systems are equipped with process and radiation monitoring facilities required for operation monitoring, evaluation of integrity and environmental release monitoring.

Gaseous radwaste treatment system. The system of gaseous radwaste treatment comprises special ventilation and gaseous radioactive blow-off purification systems.

Process blow-off purification system is purposed for decrease of radioactive noble gas releases, gaseous compounds of iodine and aerosols in gaseous blow-offs from reactor building process equipment to permissible limits.

The system provides:

- continuous purification of gas blow-off from the system of hydrogen combustion;
- continuous purification of gas blow-off from the system of tank facilities and reagents.

Plant layout

The Unit consists of the following structures surrounding the reactor building: safety structures building, steam chamber, turbine compartment, auxiliary vessel, nuclear service building, building of electrotechnical systems and control building. Buildings lay-out satisfies the requirement for physical separation of the systems with redundant safety functions. Thus safety building, steam chamber and emergency diesel-generators building are divided into two buildings each (by the number of channels of active safety systems) located on different sides of the reactor building.

Main and emergency control rooms are also physically separated in relation to the reactor building. Buildings legend VVER-600 Unit is given in Figure 6.

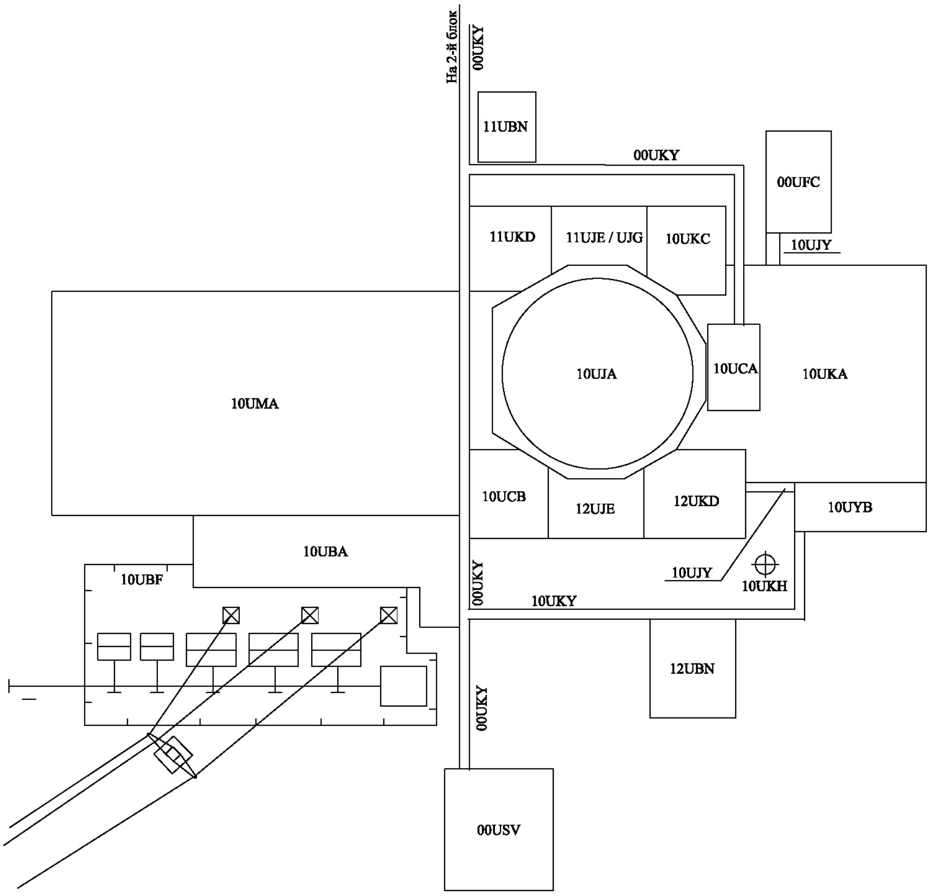

Figure 6 Buildings legend of power unit with VVER-600

The following buildings are conventionally presented in the legend: UBA – power supply of normal operation, UBF – structures for Unit transformers, UBN – stand-by diesel station of emergency power supply system, UCA – main control room, UCB – emergency control room, UJA – reactor building, UJE – steam chamber UJG – transport platform, UKA – auxiliary vessel, UKC –nuclear service building, UKD – safety systems, UKH – vent duct, UKY –free access pathway, UMA – turbine building, UPC –fresh fuel storage building, USV – engineering-utilities, UYB –buffer area of restricted access.

Reactor building is a double containment comprising all primary equipment and pipelines, spent fuel pool, hydro accumulators of ECCS passive part and core emergency flooding system, and also passive heat removal equipment. PHRS heat exchangers with water inventory are outside the containment.

Internal containment is made of prestressed steel concrete with carbon steel facing and is designed for the whole spectrum of accidents with increase in pressure and temperature within the containment. External containment is designed for all external impacts both of natural, and man-induced character, including aircraft crash.

Plant performance

10.1 NPP operation

Unit operation is provided according to the requirements of regulatory documents, process specifications and operating manuals. RP operation is possible both in base and load-follow conditions.

10.2 Reliability predictions by the designer

Reliability factors:

- scram frequency is not more than once a year.

10.3 Capacity factor predictions by the designer

Load factor is 0.92.

Annual capacity factor is 0.9.

Specified factors are reached also due to RP individual equipment design improvements, optimization of individual equipment and RP repair cycles (designing of integrated 8-year cycle), optimization of each shut-down schedule, implementation of advanced maintenance and repair system, use of fuel handling machine operating at increased motion speeds, automatic multi-position power-nut drivers for concurrent elongation of equipment flange joint studs (of reactor main joint, steam generator headers, all RP tanks, etc.), loading of fresh fuel (FA, CPS absorbing rods) into spent fuel pool during reactor coolant system cooldown, reactor refueling concurrent with maintenance and repair of steam generators due to usage of isolation devices installed in steam generator headers, etc.

Development status of technologies relevant to the NPP

Composition and design of VVER-600 main components, equipment and systems are based on the solutions laid down in the preceding designs and improved according to the up-to-date requirements that enable to improve NPP performances and ensure the required safety level.

Basic technical solutions for NPP are proved by operating experience of VVER reactor plants operation during more than 1400 reactor-years including more than 500 reactor-years of VVER-1000 RP operation.

Application of evolutionary approach to solution of target design tasks enables to develop NPP VVER-600 design, with the objective of meeting the requirements of the up-to-date Russian, IAEA and EUR standards and regulations within the shortest time.

VVER-600 design is of the required level of competitiveness and has justified prospects for commercial construction both in Russia and abroad.

Deployment status and planned schedule

Unit construction is expected by the design organization to be within 48 months (the term can be specified for a specific site) considering decrease in equipment units and simplified lay-out solutions (in comparison with VVER-1200).

List of R&D required for implementation of VVER-600 Unit design

Results of design, research and development work provided for VVER-1000 and VVER-1200 are used for VVER-600 design since VVER-600 Unit is developed on the basis of maximum possible use of equipment, technologies, lay-out and mode solutions of VVER-1200. The scope of new researches and developments required for VVER-600 includes:

- Complex of analyses justifying the corium retention within the reactor vessel under beyond design basis accidents with the core melting;
- Studies of mixing the flows with different temperature and boric acid concentration in the reactor coolant path;
- Studies of the core flow nonuniformities during operation at nominal power and in case of RCP set trip;
- Calculational analyses of reactor flow path hydraulic characteristics using finite element models and CFD-codes;
- Justification of permissible power level during operation with one RCP set;
- Justification of the core passive flooding system tank flow characteristics;
- Study of the expediency of using the nitrogen accumulating blanket in the core passive flooding system tanks;
- Study of the possibility of remaining within the limits of safe operation of fuel rods for all design basis accidents;

- Integrated work on justification of RP operation under load-follow conditions;
- Study and justification of FAs operation in the core up to 10 years.

Technical data

General plant data

Reactor thermal output	1600 MWth
Power plant output, gross	600 MWe
Power plant efficiency, net	35 %
Mode of operation	Baseload and Load follow
Plant design life	60 Years
Plant availability target >	90 %
Seismic design, SSE	0.2
Primary coolant material	Light Water
Moderator material	Light water
Thermodynamic cycle	Rankine

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	444.4 Kg/s
Steam pressure	7.1 MPa(a)
Steam temperature	299 °C
Feedwater temperature	230 °C

Reactor coolant system

Primary coolant flow rate	13861 Kg/s
Reactor operating pressure	16.2 MPa(a)
Core coolant inlet temperature	299 °C

Core coolant outlet temperature	325 °C
Mean temperature rise across core	26 °C

Reactor core

Active core height	3.50 m
Fuel column height	3.53 m
Equivalent core diameter	3.16 m
Average fuel power density	89.1 KW/KgU
Fuel material	UO_2 and UO_2 + Gd_2O_3
Fuel element type	Fuel rod
Cladding material	Zirconium alloy
Outer diameter of fuel rods	9.1 mm
Lattice geometry	Triangular
Number of fuel assemblies	163
Fuel cycle length	18 Months
Average discharge burnup of fuel	56.6 MWd/Kg
Burnable absorber (strategy/material)	B_4C + $Dy_2O_3TiO_2$
Control rod absorber material	B_4C + $Dy_2O_3TiO_2$
Soluble neutron absorber	Boron

Reactor pressure vessel

Inner diameter of cylindrical shell	4232 mm
Wall thickness of cylindrical shell	197 mm
Design pressure	17.64 MPa(a)
Design temperature	350 °C
Base material	Steel 15H2NMFA
Total height, inside	3855 mm
Transport weight	330 t

Steam generator or Heat Exchanger

Type	PGV-1000MPK
Number	2
Total tube outside surface area	6105 m^2
Number of heat exchanger tubes	10978

Tube outside diameter	16 mm
Tube material	08H18N10T
Transport weight	330 t

Reactor coolant pump (Primary circulation System)

Pump Type	GCNA - 1391
Number of pumps	2
Pump speed	1000 rpm
Head at rated conditions	61 m
Flow at rated conditions	6.0 m^3/s

Pressurizer

Total volume	79 m^3
Steam volume (Working medium volume): full power	24 m^3
Steam volume (Working medium volume): Zero power	45 m^3
Heating power of heater rods	2520 kW

Primary containment

Type	Dry, double
Overall form (spherical/cylindrical)	Cylindrical
Dimensions - diameter	36 m
Dimensions - height	56 m
Design pressure	0.5 MPa
Design temperature	150 °C
Design leakage rate	0.1 Volume % /day

Residual heat removal systems

Active/passive systems	Passive

Safety injection systems

Active/passive systems	Active and Passive

Turbine

Type of turbines	K-640-6.9/50
Number of turbine sections per unit (e.g. HP/MP/LP)	1/0/2
Turbine speed	3000 rpm
HP turbine inlet pressure	6.9 MPa(a)
HP turbine inlet temperature	284.5 °C

Generator

Type	T3V-630-2U3
Active power	645 MW
Voltage	20 kV
Frequency	50 Hz

Condenser

Condenser pressure	4.8 kPa

Feedwater pumps

Type	PEA 1470-78
Number	3
Head at rated conditions	900 m
Flow at rated conditions	0.41 m^3/s