Status report 108 - VVER-1200 (V-491) (VVER-1200 (V-491))

Overview

Full name	VVER-1200 (V-491)
Acronym	VVER-1200 (V-491)
Reactor type	Pressurized Water Reactor (PWR)
Coolant	Light Water
Moderator	Light water
Neutron spectrum	Thermal Neutrons
Thermal capacity	3200.00 MWth
Gross Electrical capacity	1170.00 MWe
Design status	Under Construction
Designers	Gidropress
Last update	01-08-2011

Description

Introduction

The design of AES-2006 of Generation 3+ with V-491 reactor plant is an evolutionary development of the designs with the VVER-1000 water cooled and water moderated reactor proved by a long-time operation. The AES-2006 design is based on the principle of safety assurance for the personnel, population and environment. The principle meets the requirements for the standards of radioactive substance releases into the environment and their content at normal operation, at anticipated operational occurrences including the design basis conditions (i.e. design events of Category 1-4) as well as at the beyond-design basis events during the entire service life of the nuclear power plant. One of the requirements during the reactor plant and process system design elaboration was not to reach the estimated value of a severe core damage 1.0E-6 reactor/year and for the probability of emergency radioactivity release not to exceed 1.0E-7 reactor/year. Level I and II PSA predict that these given values are not exceeded.

NPP safety is achieved by a comprehensive realization of the principle of defense-in-depth based on the application of a system of barriers in the way to the possible propagation of the ionizing radiation and radioactive substances into the environment as well as on the application of a system of the engineered safety features and organizational measures to ensure the integrity of these barriers. Below are provided the activities for a comprehensive realization of the principle of defense-in-depth:

- installation of a sequential system of physical barriers in the way of radioactive substance propagation: fuel matrix, fuel rod cladding, primary coolant boundary, and containment;
- consideration of the postulated initiating events that can lead to loss of efficiency of these barriers;
- determination of design measures and operative personnel activities for each postulated event that are required

to keep the integrity of the above barriers and for elimination of the consequences of the barrier destruction;
- minimization of the probability of the accidents that lead to radioactive release;
- determination of the measures to control the beyond-design basis accidents.

The VVER-1200 (V-491) design was developed by the Organization of General Designer “Atomenergoproekt”, (St.Petersburg), Organization of General Designer of reactor plant, OKB Gidropress (Podolsk), with the scientific supervision of the RRC “Kurchatov Institute”(Moscow), in line with the Russian Regulatory Documents and considering the requirements of the IAEA and the European Utilities Requirements (EUR). The design was performed using the ISO 9001:2000 Quality Assurance International Standard. The main engineering solutions for reactor plants with VVER reactors have been corroborated by the experience of operating these installations for about 1400 reactor-years (decommissioned reactors included) considering about 500 reactor-years of operating the RPs with VVER-1000.

The safety concept of VVER-1200/V-491 considers the up-to-date world tendencies in the field of NPP safety enhancement in order to meet the requirements for the NPP safety that are continuously made more stringent. The requirements for improving the economic efficiency of the NPP were also considered.

Below are described the specific features of the NPP design:

- Service life of the main irreplaceable equipment of the RP is 60 years;
- Application of horizontal type steam generators with a large water inventory and improved conditions of the primary-side natural circulation in comparison with the vertical-type steam generators;
- Application of ECCS based on the principles of active and passive operation;
- Application of double envelope concrete containment;
- Application of the diagnostics systems for safety-related system equipment for periodic tests in shutdown power Unit as well as for the operative monitoring of the operating reactor;
- Application of enhanced reliability I&C with the self-diagnostics functions;
- Reactor vessel manufactured of forged shells free from longitudinal welds with a diameter larger than the Generation III vessel, which ensures a minimum number of welds and hence reduces the time taken by inspection;
- Depending on the requirements of the Customer the RP equipment can be transported by rail (except for the reactor vessel), by water and by road;
- Reactor vessel is free from incuts and holes below the reactor main nozzles and respectively, below the upper mark of the core the way it has been made in all the VVER designs;
- Application of passive components, isolation, restraints and discharge devices;
- Usage of inertia coastdown of special rotating masses of RCP set to provide the required decrease in the flowrate through the core at loss of power the way it has been made in all VVER-1000 designs.

Work is underway on construction of a VVER-1200 (V-491) at the Leningrad-2 site and the Baltic sites. Work is underway on the construction of a VVER-1200 (V-392M) at the Novovoronezh-2 site.

Significant differences in these two plant designs include;

- Incorporation of a system of passive heat removal from the containment, and system of passive heat removal via steam generators [V-491 design];
- Incorporation of a passive core flooding system [V-392M];
- Incorporation of active systems for high pressure emergency injection system and for low pressure emergency injection [V-491];
- Differences in systems to cope with beyond design basis accidents;
- Differences in predicted core damage frequencies;
- Differences in the I&C system, the feedwater system, and in the layout of the main control room; and
- Differences in the NPP layout

The reason for these differences is, in whole, the fact of development of the above mentioned designs by two independent design organizations. The General Designer of LNPP-2 (V-491) is JSC “SPb Atomenergoproekt” (Saint-Petersburg) and the General Designer of NVNPP-2 (V-392M) is JSC “Atomenergoproekt” (Moscow). With that, use of a large number of passive safety systems is provided for the V-392M design in comparison with the V-491 design based mainly on application of active safety systems. In spite of the differences in these two designs, each of them meets the current safety requirements, meets the requirements of Russian rules and standards, considers the recommendations of IAEA and EUR guidelines as well as has a unit construction permit issued by Rostehnadzor.

The main technical performance data of the RP V-491 design and the Leningrad NPP-2 design are provided in the Appendix.

2.1 Reactor coolant system performance

The Reactor coolant system removes the heat from the reactor core by coolant circulation in a closed circuit and provides heat transfer to the secondary side. The reactor coolant system comprises a reactor, a pressurizer and four circulation loops, each one comprising a steam generator, reactor coolant pump set and main coolant pipelines that provide the loop equipment-to-reactor connection. A steam generator links the primary and the secondary sides. The steam generator headers and heat-exchange tubes are a barrier between the primary coolant and the working medium of the secondary side and prevent the radioactive substance penetration out of the primary to the secondary side.

The pressurizing system is a constituent part of the primary side and performs the functions of primary-side pressurization, pressure maintenance under the steady-state conditions, pressure control at heatup and cooldown, pressure deviation limitation under transients and accidents.

The main design and thermal-hydraulic performance of the primary side with reactor plant nominal operation is provided in Table 1. The primary-side flow diagram is provided in Figure 1.

Table 1

Parameter	Value
Reactor nominal thermal power, MW	3200
Coolant inventory in reactor coolant system (PRZ system not considered), m^3	290
Coolant inventory in PRZ at nominal power operation, m^3	55
Primary pressure at the core outlet, absolute, MPa	16.2
Coolant temperature at reactor inlet, °C	298.2
Coolant temperature at reactor outlet, °C	328.9
Coolant flowrate through reactor, m^3/h	86000
Primary-side design parameters: - gauge pressure, MPa; - temperature, °C	 17.64 350
Pressure of the primary-side hydraulic tests, MPa: - for tightness; - for strength	 17.64 24.5

2.2 Reactor core and fuel design

The reactor cores contain 163 fuel assemblies (FA). The FAs are intended for heat generation and its transfer from the fuel rod surface to coolant during the design service life without exceeding the permissible design limits of fuel rod damage. The FAs are 4570 mm high (nominal value). When the reactor is in the hot state the height of the power-generating part of the fuel rod is 3750 mm. Each FA contains 312 fuel rods. The FA skeleton is assembled of 18 guide channels, 13 spacer grids welded to them, an instrumentation channel and a support grid. The fuel rod cladding is a zirconium alloy tube. Sintered UO_2 pellets with a 5% (4.95±0.05) maximum enrichment are stacked inside the cladding. The average linear heat rate of a fuel rod is 167.8 W/cm.

According to the cartogram up to 121 rod cluster control assemblies (RCCAs) are placed inside the core. They are intended for quick chain reaction suppression, maintaining power at assigned level and its level-to-level transition, axial power field leveling, xenon oscillation suppression. Pitch electromagnet drives with pitch position indicators are used for RCCA drive mechanisms. The drives are installed on the reactor top head. The maximum effective time of FA operation between refuelings for a 12-month fuel cycle is 8400 effective hours. The average burnup of unloaded fuel is up to 60 MWD/kg U. Annually 42 fresh FAs are loaded into the core for the basic fuel cycle.

2.3 Nuclear fuel storage and handling systems

The nuclear fuel storage and handling system complex is a set of systems, equipment and components designed for nuclear fuel storage, loading, unloading, transfer and monitoring.

The complex comprises a number of systems and equipment to implement all the fuel handling procedures on the site:

- Fresh (non-irradiated) nuclear fuel storage and handling system;
- Core refueling system;
- In-containment spent fuel storage system;
- On-site nuclear fuel transfer system that comprises all the handling procedures, beginning with the fresh fuel delivery vehicles acceptance up to spent fuel vehicles dispatch;
- Arrangement of on-site nuclear fuel management record system at the Unit.

2.4 Primary-side component description

2.4.1 The reactor

The reactoris a vertical pressure vessel (a vessel and a top head) that houses the internals (protective tube unit, core barrel, and core baffle), the core, control rods and in-core instrumentation sensors. The main joint of the vessel-to-top head that is structurally integrated into the top unit is sealed with the main joint studs. The drive housings (RCCA nozzles) are installed on the top head nozzles. Electromagnet units designed for RCCA axial motion in the core are fastened outside the housings.

The reactor is positioned in the concrete cavity with a biological and thermal shielding and a cooling system. The reactor vessel is supported and fastened by the support ring fixed in the support truss. The thrust ring, installed on the vessel flange keeps the reactor vessel from transverse displacements.Reactor fastening inside the concrete cavity at two levels safely keeps it from displacements at seismic impacts and pipeline breaks. The concrete cavity, electric equipment, in-core instrumentation nozzles and the drives are cooled by the air. The reactor design is given in Figure 2.

The reactor comprises the following components:

- Reactor vessel;
- Support ring;
- Thrust ring;
- Main joint sealing components;
- Internals (core barrel, core baffle, PTU);

- Upper unit;
- SHEM-3 CPS drive;
- Wiring unit;
- Reactor core;
- In-core diagnostics system;
- Main joint leak monitoring system;
- Surveillance specimens;
- Clamp ring.

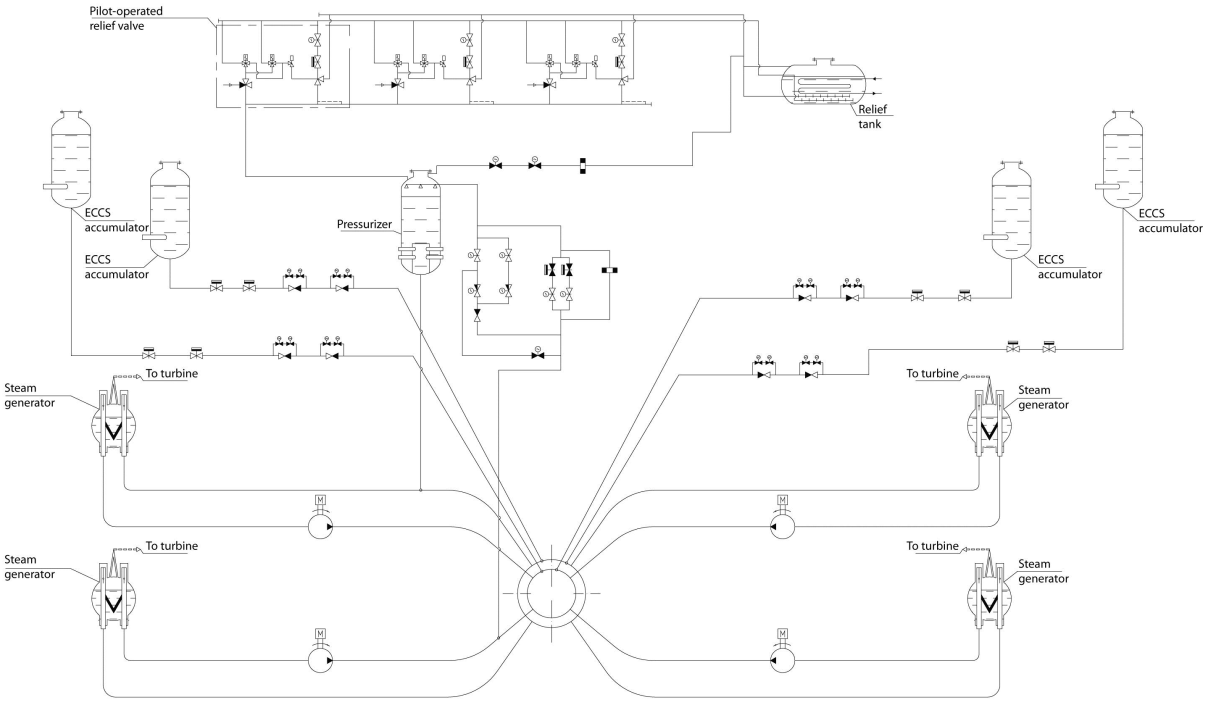

Figure 1 – Schematic flow diagram of the primary circuit

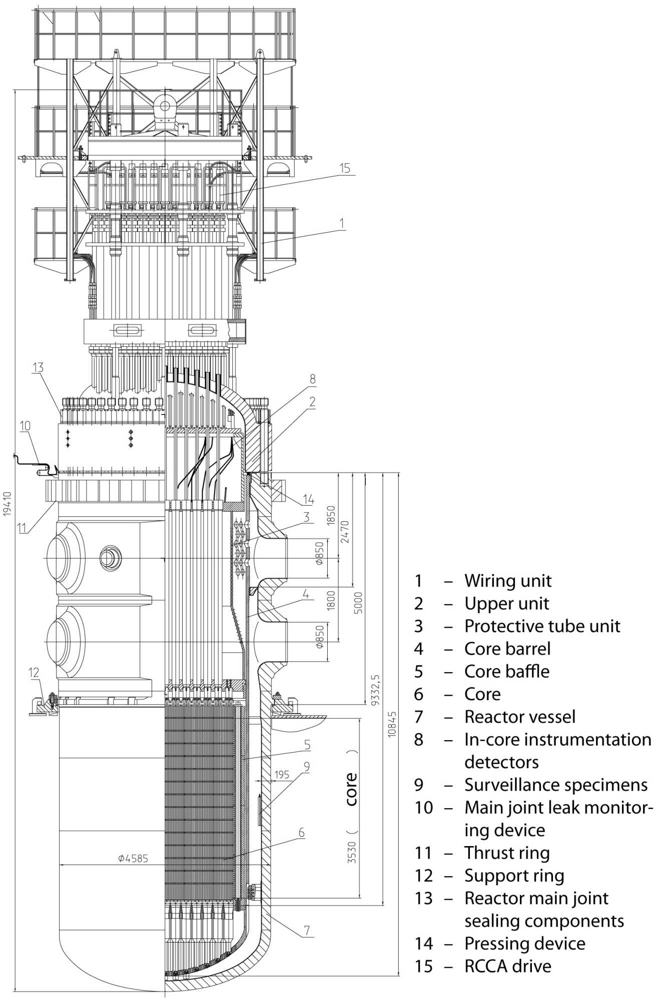

Figure 2 – Reactor

2.4.2 Steam generator

Steam generator PGV-1000MKP with supports comprises the following components: steam generator, steam header, supports, shock absorbers, one- and two-chamber surge tanks, embedded components for supports and shock absorbers.

The steam generator itself is a single-vessel heat exchange apparatus of horizontal type with submerged heat-transfer surface and comprises the following components:

- a vessel with different-purpose nozzles;
- a heat-exchange bundle with fastener and spacer components;
- primary coolant collectors;
- feedwater supply and distribution systems;
- emergency feedwater supply and distribution systems;
- distribution perforated plate;
- submerged perforated plate;
- chemicals feeder.

The application of the horizontal type steam generators makes it possible to reduce the height of the building, thus improving its seismic resistance.

SG vessel is a 13820 mm long welded cylindrical vessel with a 4200 mm internal diameter. Elliptic bottoms are welded to its both edges. The SG vessel houses the primary collectors, heat-transfer surface and the internals. The vessel is made of 10GN2MFA steel.

The primary coolant collector is a thick-walled cylinder of variable diameter and thickness. The total height (without the cover and displacer) is 5100 mm, the maximum diameter is 1176 mm at wall thickness 171 mm. The collector is made of electroslag remelting 10GN2MFA steel. The internal surface has protective corrosion-resistant cladding.

There are 10978 holes of 16.25 mm nominal diameter in the central cylindrical part of the collector. The holes are positioned in a staggered order: the vertical pitch of the holes is 44 mm and the horizontal pitch of the holes is 30.8 mm (on the external surface).

The joint in the top part of the collector connects the Dnom 500 collector flange with threaded seats and a flat cover with a displacer. 20 joint studs of M60 type are made of steel 38KhN3MFA and the nuts are made of steel 25Kh1MF.

There are two pipe sleeves on the reducing ring for continuous and periodic blowdown out of the collector pockets and for flushing device installation and one D nom 30 pipe sleeve to attach flushing devices. The pipe sleeves for continuous and periodic blowdown out of the collector pockets and for flushing device installation are provided with detachable flange joints that are sealed with expanded graphite gaskets and fixed with six bolts. The D nom 30 pipe sleeve to attach the flushing device is also provided with a detachable flange joint that is sealed with expanded graphite gaskets and fixed with six bolts, too.

The heat-transfer surface consists of 10978 tubes 16x1.5 mm (08Kh18N10T U steel). The heat-exchange tubes are U-coils, assembled into a heat-exchange tube bundle. The tubes in the bundle are placed in a staggered order: the vertical pitch is 22 mm and the horizontal pitch is 24 mm. The coils are slanting towards the collectors (20 mm on the coil length) to provide a possibility for complete drain of the tubes.

The tubes are fixed in the collectors by welding the edges over with the internal surface of the collector (the penetration depth is not less than 1.4 mm). The tubes are hydraulically expanded with additional mechanical expansion on the outside surface until the gap (slot) is completely closed.

The tubes are spaced with bent and flat plates that provide a regular tube positioning in the heat exchange bundle. The sub-slime corrosion in the gap between the tube and the spacers is prevented due to the venting slots in the plates that are designed to avoid the gap stagnation.

2.4.3 Reactor coolant pump

The reactor coolant pump is designed to create the primary coolant circulation in the reactor plant. The RCP set has an additional function of providing coolant circulation at the coastdown under any loss-of power accidents, which allows a smooth passing to the natural circulation mode.

The RCP set is a vertical centrifugal one-stage pump set GTsNA-1391 that consists of a hydraulic casing, pump internals, electric motor, upper and lower insertions, supports and auxiliary systems.

The reactor coolant pump set has the following parameters:

- nominal supply, m^3/h 21500
- pressure head, MPa 0.588
- coolant temperature, °C 298.2
- design temperature, °C 350
- design pressure, MPa 17.64
- rotational rate, rev/min 1000/750
- current frequency, Hz 50
- sealing water leakages, m^3/h 1.2

An alternative implementation of the RP design with a one-speed RCP set.

2.4.4 Main coolant pipeline

The main coolant pipeline connects the reactor, steam generators and reactor coolant pump sets thus making up the reactor coolant system and is designed to implement the reactor-to-steam generator coolant circulation and backwards. The "Leak before break" concept is used in the MCP design. The long-term experience in the MCP design development and operation at the NPPs was used in the course of design elaboration. At this, a guillotine break of D nom 850 pipeline with double-ended coolant flow through the break was considered as a design-basis accident in the frameworks of safety analyses.

The main coolant pipeline comprises four circulation loops. Each loop is divided into three tube sections. The reactor outlet nozzle-to-SG inlet collector pipeline section is the hot leg. The cold leg comprises two sections: the SG outlet collector-to-RCP set suction nozzles pipeline and RCP set discharge nozzle-to-reactor inlet nozzle pipeline section. The 850 mm internal diameter was chosen proceeding from the condition of providing the acceptable coolant velocity and pressure loss in the MCP within the design range of coolant flowrate.

The hot leg of loop 4 via pipeline 426´40 mm is attached to the pressurizer (surge line).

The hot leg of Loop 3 is connected to the pressurizer via a 219´20 mm pipeline (injection line).

The detachable modular heat insulation is installed on the MCP external surface to reduce the heat losses during reactor operation.

2.4.5 Pressurizer

The pressurizer is a vertically positioned pressurized cylindrical vessel with elliptic bottoms installed on a cylindrical support. The pressurizer performance and main dimensions are given in Table 2.

Table 2

Parameter	**Value**
Nominal pressure of the steady-state conditions, MPa	16.1
Nominal temperature of the steady-state conditions, °C	347.9
Design pressure (gauge pressure), MPa	17.64
Design wall temperature, °C	350
Internal diameter, mm	3000
External diameter, mm	3330
Capacity (full volume), m^3	79
Water level at nominal power operation, m^3	55
Water level under steady-state conditions (level gauge readings), m	8.17±0.15
Working medium	Steam and water. Nitrogen under heatup/cooldown conditions

Parameter	Value
Quantity of TEH groups, pcs.	4
Total power of PRZ TEHs, kW	2520

2.5 Auxiliary systems

The nuclear fuel storage and handling complex is described in Section 2.3. The process air supply system is designed to feed the low pressure air to the NPP users located in different buildings. The high pressure nitrogen system is designed to supply nitrogen at 5.9±0.05 MPa to the users of reactor building at the NPP normal operation. The gas blowdown and purification systems are designed to remove and purify the gas medium from the unit equipment. The hydrogen ignition system for the hydrogen of radioactive process blowdowns is designed to prevent explosive concentration in the system of radioactive gas purification. The system of blowdowns for reactor building equipment is designed to discharge and evacuate the gas media out of reactor building equipment and to supply nitrogen for hydrogen dilution. The other auxiliary systems are also designed to provide the normal NPP operation.

2.6 Operating conditions

The preferable operating mode is the condition with the basic load of 100 % nominal power. The RP equipment is designed with account for the conditions of load maneuvering specified in Table 3. A possibility of load shedding to the station auxiliaries and turbine idle run without reactor trip is also envisaged in the design. The permissible rate of reactor power increase is 0.017…3 % Nnom/min depending on the initial values.

Table 3

Conditions	Quantity of cycles	Note
Steady-state operation (considering the change of frequency in the grid within the interval 49.0- 51.0 Hz), including the power oscillations ±1 % N_{nom} at the rate of 1 % N_{nom}/s	Not limited	A cycle implies a change of power and return to initial level. Operation of two or three RCP sets is permitted
Change of Unit power by not less than $\pm$ 2% N_{nom} and by not more than $\pm$5 % N_{nom} (grid frequency maintenance mode) at a rate of 1 % N_{nom}/s	$7x10^6$	A cycle implies a change of power and return to initial level
Change of Unit power at a rate of not less than 1% N_{nom}/min and not more than 5 % N_{nom}/min at a deviation from the current value not above $\pm$10 % N_{nom}	$5x10^6$	A cycle implies a change of power and return to initial level
Change of Unit power as per scheduled (dispatcher schedule) and unscheduled load change at a rate not more than 5 % N_{nom}/min within the range from 50 % N_{nom} to 100% N_{nom}	15000	A cycle implies a change of power and return to initial level
Change of Unit power within the range from 50 % N_{nom} to 100% N_{nom} under accident situations in the power grid:	100	

power increase at a rate of 5 % N_{nom}/min and further power decrease at a rate of 20 % N_{nom}/min		
Change of Unit power by ±10 % N_{nom} at a rate of 5% N_{nom}/s:		A cycle implies a change of power and return to initial level
- change of power by -10 % N_{nom};	1000	
- change of power by +10 % N_{nom}	1000	
Change of Unit power by ±20 % N_{nom} at a rate of 10% N_{nom}/min:		A cycle implies a change of power and return to initial level
- change of power by -20 % N_{nom};	65	
- change of power by +20 % N_{nom}	65	

2.7 Standard fuel cycle

A standard fuel cycle is once-through and open. The length between the refuelings is 12 months. The standard fuel cycle length is 4 years. 18-20 month long fuel operation is possible along with the fuel operation as per the standard fuel cycle.

2.8 Alternative fuel

A possibility to apply MOX fuel as alternative fuel can be considered.

2.9 Spent fuel (SF) and SF handling schedule

Spent fuel is taken out of the reactor to be installed for storage in the spent fuel pond. It is located in the reactor hall inside the containment close to reactor cavity. Besides, leak check of defective fuel rods is envisaged in the spent fuel pond as well as a dedicated storage area for leaky fuel assemblies in leak-tight bottles. The spent fuel pond layout excludes the necessity to move the transport casks above the spent fuel when taking away fuel and other cargoes from the reactor hall. The polar crane of the containment building is used to transfer the transport casks loaded with fuel. During reactor refueling the fuel cooled in the spent fuel pond is taken to the spent fuel storage. The spent fuel storage is designed for dry storage on the NPP site in double purpose casks designed both for transport and storage. The spent fuel storage tank is designed for long-time storage of the spent fuel accumulated for 10 years of operation of two units with a possibility for the capacity of the building to be expanded in the future to keep the fuel accumulated during the Unit service life. The spent fuel handling systems are described in section 8.

2.10 Example power systems with NPPs of similar type

So far the operating experience of VVER-reactor equipped NPPs has been about 1400 reactor-years (including the commissioned ones), of which about 500 reactor years account for the operation of NPPs equipped with VVER-1000. The NPP to VVER-1000 designs were implemented at the following sites:

- Kozloduy NPP in Bulgaria;
- Tianwan NPP in China;

- Bushehr NPP in Iran;
- Kudankulam NPP in India.
- Rovno NPP, Zaporozhe NPP, Khmelnitskaya NPP and South-Ukraine NPP in the Ukraine;
- Nononoronezh NPP, Balakovo NPP, Kalinin NPP and Rostov NPP in Russia.

The nuclear power plant construction to AES-2006 V-491 design is under way at the Leningrad NPP-2 as well as the Baltic NPP in Russia according to the permit granted by the Rostechnadzor Regulatory Authority of Russia.

3.1 Safety concept, main design principles and methods of licensing

The design was developed on the basis of the requirements of the up-to-date safety rules and standards in the nuclear power engineering of Russia considering the Safety Guides and other recommendations issued by the International Atomic Energy Agency and the requirements of the European Utilities for the NPPs with LWR (EUR Requirements, Volume 2, Chapter 2.1).

In the course of NPP design elaboration in line with the requirements /1/ the generally recognized targets, principles and safety assurance criteria are met that are mutually dependent and form a single complex.

The determination of the safety system configuration in the present design is based on the application of the following principles:

- single failure principle;
- redundancy principle;
- diversity principle;
- principle of physical separation;
- protection against the operator's errors;
- RP inherent safety principle.

3.2 Design safety and reliability assurance

The simple design and reliability of the safety systems are enhanced due to the application of the active systems, passive systems that do not require involving other systems for their operation and the application of reliable equipment.

3.3 Active and passive systems, inherent safety characteristics

The following active systems are incorporated into the V-491 design:

- High pressure emergency spray system;
- Low pressure emergency spray system;
- Emergency gas removal system;
- Emergency boron injection system;
- Emergency feedwater system;
- Residual heat removal system;
- Main steamline isolation system.

The following passive safety systems are incorporated into the V-491 design:

- Emergency core cooling system, passive part;
- System of passive heat removal from containment;
- System of passive heat removal via steam generators;
- Double-envelope containment and core catcher.

Besides, the primary- and secondary side overpressure protection systems that are envisaged in the design contain

Pressure Operated Relief Valves (PORVs) capable of operating both in response to the appropriate functions of the automation and in response to the set parameter values achieved (in active and passive modes of operation).

3.4 Defense-in-depth

NPP safety shall be provided by the realization of the defense-in-depth concept based on the application of a system of physical barriers to the release of the ionizing radiation and radioactive products into the environment and the systems of engineering and organizational measures to protect the barriers and maintain their efficiency, as well as to protect the personnel, the population and environment.

The system of physical barriers of a NPP Unit comprises a fuel matrix, a fuel rod cladding, reactor coolant circuit boundary, reactor plant enclosure and biological shielding.

The systems of engineering and organizational measures shall form five levels of defense-in-depth, namely:

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

- assessment and choice of the place applicable for NPP siting;
- establishment of control area and off-site surveillance area to realize the planned protective measures;
- design elaboration on the basis of a conservative approach with well-developed inherent safety of RP;
- provision of the required quality of systems (components) of the NPP and performed work;
- NPP operation in accordance with the requirements of regulatory documents, process specifications and operating instructions;
- keeping operable the systems (components) important for safety by timely detecting the defects, taking preventive measures, replacement of equipment with expired service life and arrangement of efficient system of recording for work and monitoring results;
- NPP personnel recruitment and provision of the required level of their qualification for activities under normal operating conditions, anticipated operational occurrences including the pre-accident situations and accidents, instillation of safety culture.

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

- detection of deviations from normal operating conditions and their elimination;
- management at deviated operation.

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

- prevention of an initial event development into design basis accidents, a DBA development into the beyond design basis accidents through using the safety systems;
- mitigation of the consequences of the accident in case of a failure to prevent them by localization of the releasing radioactive substances.

Level 4 (Management of beyond design basis accidents):

- prevention of beyond design basis accident development and mitigation of their consequences;
- protection of leak-proof enclosure from destruction at beyond design basis accidents and keeping up its operability;
- bringing the NPP back to a controlled state of chain fission reaction termination, provision of continuous nuclear fuel cooling and retention of the radioactive substances in the established boundaries.

Level 5 (Emergency actions planning):

- preparation of on-site and off-site emergency action plans and their realization, if necessary.

3.5 Safety indices

The core damage frequency probability (CDF) for a one-year fuel cycle is provided in Table 4.

Table 4

Conditions	CDF probability, reactor/year
Operating conditions	2.24×10^{-7}
Standby conditions	3.7×10^{-7}
Total	5.94×10^{-7}

3.6 Safety systems to cope with the design basis- and beyond design basis accidents

The strategy of coping with the design basis accidents is based on using both the active and passive safety systems. The strategy of coping with the beyond design basis accidents is based on using preferably the passive safety systems.

The following active and passive safety systems are implemented in V-491 design:

- ***Low pressure emergency injection system*** is designed for boric acid solution supply to the reactor coolant system in case of loss-of-coolant accidents including the break RCS with a maximum D nom 850 when the pressure in the system goes below the working parameters of the given low pressure emergency injection system;
- ***High pressure emergency injection system*** is designed for boric acid solution supply to the reactor coolant system in case of loss-of-coolant accidents that exceed the compensatory capability of the normal make-up system at the pressure in the reactor coolant system below the working pressure of the high pressure emergency injection system (below 7.9 MPa);
- ***Residual heat removal system*** is designed for the decay heat removal and reactor plant cooldown during a normal NPP trip, under the conditions of anticipated operational occurrences and under design basis accidents on condition of retaining the primary-side integrity together with the low-pressure emergency injection system;
- ***Emergency core cooling system, passive part*** is designed for boric acid solution supply with a concentration not less than 16 g/kg at primary pressure below 5.9 MPa in the amount sufficient for reactor core cooling before the low-pressure emergency injection pumps actuate in design-basis loss-of-coolant accidents;
- ***Quick boron injection system*** is designed for boric acid injection into the pressurizer in case of a primary-to-secondary leak to reduce the primary pressure and to create the necessary concentration of boric acid in the primary coolant under a BDBA without scram;
- ***Emergency gas removal system*** is designed to remove the steam-gas mixture out of the RP primary side (reactor, PRZ and SG collectors) and to reduce the primary pressure in order to mitigate the consequences at design basis- and beyond design basis accidents;
- ***Primary overpressure protection system*** is designed to protect the RP equipment and pipelines from the gauge pressure on the primary side under the design basis conditions of Category 2 – 4 and beyond-design basis accidents due to the operation of the PRZ pilot-operated relief valves installed on the line for steam discharge out of the PRZ steam space into the relief tank;
- ***Secondary overpressure protection system*** is designed to protect the RP equipment and pipelines from the gauge pressure on the secondary side under the design basis conditions of Category 2 – 4 and beyond-design basis accidents due to the operation of the SG pilot-operated relief valves installed on the steamline sections between the steam generators as far as shut-off electric motor-operated gate valves, considering the advance actuation of BRU-A and reactor trip system;
- ***Passive heat removal system via steam generators*** is designed for long-time residual heat removal from the core to the ultimate heat sink via the secondary side at beyond design basis accidents. The system of passive heat removal through the steam generators backs up the appropriate active system of heat removal to the ultimate sink in case it is impossible for it to perform its design functions.

- ***Emergency feedwater system*** is designed to supply the steam generators with feedwater under the conditions of anticipated operational occurrences and design basis accidents when feedwater supply by the standard system and auxiliary systems is impossible;
- ***System of passive heat removal from the containment*** refers to the engineered safety features for coping with BDBA and is designed for long-time heat removal from containment at beyond design basis accidents;
- ***Main steamline isolation system*** is designed for quick and reliable steam generator isolation from a leaky section:
  - At pipeline breaks downstream of the SGs as far as the turbine stop valves in the pipeline sections that either can be isolated or cannot be isolated from the SG;
  - At feedwater pipeline breaks downstream of the SGs as far as the check valves;
  - At primary-to-secondary leak;
- ***Double-envelope containment* and *core catcher*** are designed to retain the radioactive substances and ionizing radiation within the limits envisaged in the design.

3.7 Safety assurance under seismic impacts

The seismic input parameters are assumed in accordance with the values provided below:

- SSE with a frequency once every 10000 years magnitude 8 to MSK-64 scale;
- Operating basis earthquake with a frequency once every 100 years magnitude 7 to MSK-64 scale.

According to the requirements of the Code for designing earthquake-resistant nuclear power plants the NPP provides safety at seismic impacts up to SSE inclusive and generation of electric and thermal power up to the level of OBE inclusive /2/.

According to the requirements of the Code for designing earthquake-resistant nuclear power plants /2/ the safety systems retain:

- The capability to perform the functions related to NPP safety assurance during and after the earthquake of the intensity up to SSE inclusive;
- Operability at earthquakes of the intensity up to OBE inclusive and after it is over.

A system of seismic monitoring and signaling is envisaged in the AES-2006 design that provides command generation for automatic reactor trip in case of a seismic input on the ground that corresponds to OBE.

3.8 Probabilistic risk assessment

The results of the probabilistic safety analysis of Level I confirm meeting the main engineering principles of the up-to-date concept of defense-in-depth including the principles of functional and design diversity, protection from common-cause failures, proofness against the operator's error, physical separation and assurance of a high reliability of performing the safety functions.

3.9 Emergency measures

Emergency measures include the development of Emergency Action Plans to protect the personnel and the population.

The Emergency Action Plans to protect the personnel and the population are developed on the basis of design basis characteristics and parameters of a NPP Unit, the criteria to take the decision on the population protection in case of an accident at a NPP considering the economic, natural and other characteristics, specific features of the locality and the degrees of actual hazard of the emergency.

The Emergency Action Plans to protect the personnel and the population are the main regulatory documents to carry out the protective, organizational, engineering, preventive health-protection and other measures to be taken in case of an accident in order to protect the personnel and the population, localize the accident and cope with it.

When the Emergency Action Plans to protect the personnel and the population are committed to action in case of an accident, the following steps are taken:

- Information of the personnel and the population;
- Commitment of control authorities to action;
- Radiological and general reconnaissance;
- Radiological protection;
- Medical protection;
- Physical protection;
- Public order protection;
- Evacuation activities.

Proliferation resistance

The management and storage of the available nuclear materials at the NPP, as well as control of their storage and transfer, along with the assurance of reliable security excludes an outflow of nuclear materials outside NPP boundaries.

Safety and security (physical protection)

5.1 Designation of physical security system (PSS)

The PSS is a part of a set of engineering and organizational measures to provide nuclear and radiological safety at AES-2006 operation.

PSS prevents the unauthorized actions in relation to nuclear and radioactive materials, physical barriers in the way of the propagation of the ionizing radiation and radioactive substances, and also in relation to the process systems, their equipment and operational personnel that exercises control over 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 video monitors;
- call of response group by the alarm-calling signals from the guard posts and from the secure rooms, buildings, and structures;
- detain (slow-down) of intruder's advance;
- suppression of unauthorized actions;
- monitoring, recording and assessment of the operators' and first-line group actions;
- automated monitoring of the people's access to secure areas, buildings and rooms;
- automated reporting of the staff location;
- remote round-the-clock TV monitoring of the situation in the secure areas, buildings and rooms;
- documenting of the event by video recording;
- on-line broadcasting of voice data through wire- and radio communication operative channels;
- detention of people involved in preparation or implementation of unauthorized actions.

The PSS functions under normal operating conditions. Under accident conditions and when the emergency actions are performed, the PSS shall not prevent the evacuation of the personnel and make obstacle to the access to the secure areas for the specialized units who take part in emergency actions (suppression of fires, decontamination of buildings, constructions and territory).

5.3 Structure of physical security system

The PSS for the AES-2006 is realized on the basis of a complex automated system of physical security that includes:

- a complex of engineered-and-alarm control and display system for physical security features of buildings and constructions designed in the framework of AES-2006 Project;

- a complex of engineered-and-technical physical security features for the NPP boundary perimeter;
- a complex of engineered-and-technical physical security features of the branch security.

6.1 Turbine generator

The main solutions for the thermal diagram and additional equipment have been well proven by the operation of K-1000-60/3000 commercial turbine unit at the Rovno NPP, Khmelnitsky NPP, South-Ukraine NPP and Kalinin NPP and will be used at LMZ K-1200-6.8/50 turbine unit.

K-1200-6.8/50 turbine is a single-shaft five-cylinder set and comprises a double-flow high pressure cylinder (HPC) and four double-flow low pressure cylinders (LPC). without the generator the turbine is
~ 52.3 m long, and with the generator it is ~74.5 m long.

The schematic thermal diagram of the turbine plant comprises four stages of low pressure heaters, a deaerator, two stages of high pressure heaters. The accepted thermal diagram of the turbine plant is designed for a regenerative heatup of the main condensate and feedwater under normal operating conditions.

The schematic diagram of the turbine is 2LPC+HPC+2LPC.

The given structural diagram has a number of advantages of turbine manifold and allows:

- A symmetrical steamline manifolding of all the turbine cylinders which assures a uniform loading of supports, decrease in the forces on the turbine nozzles, simplifies the compensation of the steamline thermal expansion;
- A symmetrical manifolding of separator-superheaters, which provides a reliable operation of these sets.

The accepted position of HPC contributes to a decrease in the values of relative axial displacements of the turbine rotors. A double-wall construction of HPC allows reducing the pressure differentials on the vessel walls as well as static and thermal stresses in the structural components, especially under transients and considerably decreases the time for the turbine plant start-up from different thermal states. The HPC rotor is solid-forged.

The design conditions of operation of the turbine itself are the conditions of normal operation including start-ups, beginning with the start-up of the jacking gear, power operation within the entire range of the loads and NPP shutdown up to jacking gear trip following turbine and reactor plant cooldown by shedding the secondary coolant into the condenser via steam dump valve to the turbine condenser (BRU-K). The turbine itself also functions under the anticipated operational occurrences if these AOOs do not bring about a loss of the system operability. Under accidents the turbine functioning is not required and it does not operate.

The main parameters of the turbine plant operation under nominal conditions with reactor plant thermal power 3212 MW are presented in Table 5.

Table 5

Parameter	Parameter value
Electric power under the warranty conditions (no steam bleeding above regeneration and no chemically demineralized water added to the cycle), MW	1170
Nominal flow of main steam, t/h	6408
Nominal pressure of main steam, MPa (abs.)	6.8

Parameter	Parameter value
Nominal temperature of main steam, °C	283.8
Nominal humidity of main steam, %	0.5
Post-intermediate superheating steam parameters: • Pressure, MPa (abs) • Temperature, °C	0.54 260
Calculated temperature of cooling water, °C	18
Nominal absolute steam pressure in condenser, kPa	4.9
Nominal mass flow of cooling water in condenser, t/h	170000
Nominal absolute steam pressure in deaerator, MPa	0.81
Feedwater temperature, °C	227
Cogeneration load, MW	300

6.2 Feedwater system

6.2.1 Main feedwater system

The main feedwater system is designed to perform the functions enumerated below:

- Turbine condensate deaeration in order to maintain the design standards of the content of oxygen and carbon dioxide in feedwater;
- Feedwater supply to steam generators under normal operating conditions and at anticipated operational occurrences;
- Feedwater regenerative heatup;
- Maintaining level in steam generators;
- Reactor plant cooldown at the steam stage;
- Feedwater line isolation in case of a feedwater line break or steam generator overfilling;
- Termination of feedwater supply to the damaged steam generator from all the possible sources under the conditions of uncontrolled steam removal from the steam generator and at primary-to-secondary leaks.

The engineering and organizational solutions taken to provide a safe operation of the system of main feedwater have been proven by the previous experience in design elaboration, tests, research and by the operating experience of similar systems at the operating VVER-1000 NPPs Units both in Russia and abroad.

6.2.2 Auxiliary feedwater system

The auxiliary feedwater system is designed to provide steam generators with feedwater under Unit transients (start-up, shutdown, cooldown) and anticipated operational occurrences.

The engineering and organizational solutions taken to provide a safe operation of the system of auxiliary feedwater have been proven by the previous experience in design elaboration, tests, research and by the operating experience of similar systems at the operating VVER-1000 NPPs Units both in Russia and abroad.

The auxiliary feedwater system comprises two pumps, valves and pipelines.

6.3 Auxiliary systems

The systems below belong to the turbogenerator auxiliary systems:

- Turbogenerator lubrication system;
- Steam condenser system;
- Steam separation and superheating system;
- High pressure heater system;
- Low pressure heater system;
- Main steam line system;
- Secondary-side water chemistry control system;
- Turbine condenser purification system;
- Secondary-side process media sampling system.

Electrical and I&C systems

7.1 Power supply system

Installation of a 1200 MW turbogenerator is envisaged at the power Unit, its rotational speed being 3000 rpm in a six-phase embodiment with two three-phase windings of 24 kV voltage, mutually shifted one versus the other by 30 electric degrees calculated at 50% of the generator nominal power.

The connection of the turbogenerators of the power Units of AES-2006 is realized according to the diagram: generator - transformer block via the Unit step-up transformers.

The Unit step-up transformer ensures performing the functions under all the normal operating conditions and also in the combination with a seismic impact up to the OBE inclusive. The design of the Unit step-up transformer corresponds to the standards of electric and fire safety.

The main schematic diagram of connections assumed in the design makes it possible to exclude a simultaneous trip of more than one generator – transformer block and more than one set of the backup transformers both for the "complete" scheme and for the emergency and repair schematic modes when the repair of one of the connecting switches coincides with a failure of the other switch.

7.2 Safety-related electric systems

The emergency power supply system is a system to supply power to the Unit safety system consumers, support safety systems, I&C systems to control and monitor the operation of the given systems including the reactor plant system instrumentation.

7.3 Layout of the main control room (MCR)

The control of the I&C of the main process, NPP safety monitoring under all the conditions of normal operation, at design- and beyond design basis accidents, including severe accidents, is exercised in the MCR.

The Main Control Room comprises a set of:

- Mimic panel and panel devices to locate the I&C hardware envisaged in the design (control keys, operational

terminals) that are required for information representation and process control;
- Means of industrial TV, telephone, radio and loud-speaking communication of the operative personnel-to-operative personnel, NPP administration, fire-engine house etc.;
- Rooms to house the above hardware, documentation, and operative personnel who are provided with the conditions required for the operation of the personnel and automation hardware.

The MCR provides performing the following I&C functions:

- Monitoring and control of the NPP operation in accordance with the assigned function (normal operating conditions and accidents);
- Failure detection in all the systems important for safety as well as taking the measures to maintain a safe state and availability of the plant;
- Detection of events that have lead to design extension in accordance with the assigned function and taking measures aimed at their coping and consequence mitigation;
- Measures to be taken to protect the personnel.

The complex of the main control room comprises:

- Room for the operative I&C areas as well as the area of shift supervisor;
- Room of the engineering support center;
- I&C service support room;
- Room for the Unit shift supervisor;
- Personnel refreshment area;
- Computer rooms.

7.4 Reactor trip system and other safety systems

Reactor trip and preventive protection systems are designed for reactor parameter monitoring, reactor power control including the scheduled shutdown and scram, quick transfer to a subcritical state, maintaining the reactor in the state and generation of the signals to other I&C subsystems.

Reactor trip system and preventive protection systems comprise the following kinds of safety systems:

- Reactor trip system actuated by de-energizing all the reactor rod cluster control assembly (RCCA) drives;
- Accelerated preventive protection system that actuates when a definite set of equipment is in operation at definite RP parameters through a drop of a RCCA group (or a combination of RCCAs out of one group) either automatically or with a key by the operator in the MCR;
- Preventive protection system I that actuates when the controlled parameters reach appropriate setpoints. At this, automatic reactor power decrease is achieved either by RCCA groups downward motion one-by-one beginning with the working group or with a key by the operator in the MCR by RCCA groups downward motion one-by-one beginning with the working group;
- Preventive protection system II that actuates when the controlled parameters reach appropriate setpoints and also at uncontrolled drop of one single RCCA. PP-II is realized through a prohibition for RCCAs to move upwards (meanwhile, the RCCAs are allowed to move downwards);
- Reactor unloading and power limitation depending on the state of RCP set, feedwater pumps and the RCP set power frequency.

Spent fuel and waste management

A complex of spent fuel handling systems was realized in the RP AES-2006 V-491 design. The complex comprises the systems that are enumerated below:

- in-containment spent fuel storage system;
- system of fuel rod leak check performed when the reactor is shut down;
- spent fuel pond cooling system;
- leak monitoring system for spent fuel pond lining.

The in-containment spent fuel storage system is designed to cool the spent fuel taken out of the reactor in order to reduce the former's activity and residual heat to the values that are permissible at transportation.

The functions of the in-containment spent fuel storage system are described below:

- placement of the spent fuel taken out of the reactor during refueling as well as the placement of the fuel urgently taken out of the core;
- spent fuel cooling (storage) before it is taken out of the reactor compartment;
- residual heat removal from the spent fuel;
- provision of the biological shielding for the personnel from the fuel kept in the spent fuel pond.

The in-containment spent fuel storage system is designed to keep and cool the spent fuel inside the reactor building for 10 years considering the scheduled fuel reloadings and the whole core unloading at any moment of NPP operation.

The in-containment spent fuel storage system is a spent fuel pond furnished with appropriate equipment and systems.

The system of fuel rod leak check for the reactor in a shut down state performs the following functions:

- FA check in the fuel handling machine mast with the system of leak check of cladding;
- FA check with a system of defective assembly detection (DADS);
- Leaky FA check and repair at the inspection and repair bench.

The necessity of a leak check of cladding when the reactor is shut down is determined on the basis of the data on the leak check of cladding when the reactor is in operation.

The leak check is performed by analyzing the gas or water samples.

The spent fuel pond cooling system is designed for:

- Residual heat removal from the spent fuel that are located in the spent fuel pond under all the operating conditions as well as under the design basis accidents and beyond design basis accidents;
- Creation of a shielding layer above the fuel assemblies in reactor core barrel, spent fuel pond and refueling pond;
- Reactor core barrel filling-up in the course of fuel handling procedures;
- Core barrel and internals inspection well emptying;
- Spent fuel pond and refueling pond emptying when the spent fuel pond lining and the refueling pond is repaired;
- The pipelines of the spent fuel pond cooling system are used to supply the water fed by the spray pumps to fill up the internals inspection well and refueling pond during refueling;
- The pipelines of the spent fuel pond cooling system are used to supply the water fed by the spray pumps to fill up the internals inspection well during the post-accident activities following the core melting and corium release outside the reactor vesse

The leak monitoring system for spent fuel pond lining is designed to monitor the integrity of the pond, internals inspection well and refueling pond lining and the determination of water availability in the lining-to-wall gap in each lining section.

When the service life of a NPP Unit has expired and it is decommissioned, the spent fuel handling equipment is subjected to the procedures that are provided below.

At the preparation stage:

- Removal of spent fuel from the NPP Unit;
- Coolant discharge and reactor and spent fuel pond drying;
- Coolant treatment in the dedicated water purification unit;
- Equipment decontamination as per standard process procedure.

At the stage of the Unit preparation for the observation period:

- Equipment conservation at the standard places.

At the observation stage:

- Equipment remains in their places for a long period of time.

Following the equipment decontamination and retention during the observation period, the level of structure activity and contamination is going to decrease by 3 orders and will be mainly defined by the presence of such long-living nuclides as Cs-137.

At the liquidation stage:

- Radiological monitoring that is confined at the given stage to the estimation of radiological assessment with standard radiation monitoring instrumentation to assess the total radioactivity, sampling and analysis of smears from the contaminated surfaces of equipment and pipelines for isotope composition and radioactivity level;
- Disassembly of equipment and structures;
- Radiological monitoring of dismantled equipment and building refuse;
- Estimation of the level of residual radioactivity in metal;
- Rigid decontamination and remelting of metal.

Liquid radioactive media as well as solid and gaseous wastes are generated in the course of a NPP operation.

Liquid radwaste management. The NPP design envisages the systems for collection, reprocessing and temporary storage of liquid radioactive media in all the station operating conditions, namely:

- dedicated sewage system;
- sump water recycling system;
- liquid radioactive media intermediate storage system.

The requirements below were taken into consideration in designing the liquid radioactive media recycling systems:

- reduction in the total volume of the liquid radioactive media to be sent for reprocessing;
- reduction in the volume of liquid radioactive waste for temporary storage to be sent to the solidification plant for subsequent burial in the radwaste storages;
- reduction in the volume of disbalanced water in the controlled access area.

The basic process solutions envisaged in the NPP design allow realizing the set targets:

- separate collection and recycling of the liquid radioactive media depending on their chemical composition and activity;
- basic radionuclide localization within minimum volumes;
- introduction of low-refuse processes in liquid radioactive media and liquid radioactive waste recycling;
- application of low-refuse decontamination methods;
- reduction in the number of filter regeneration;
- imposing more stringent requirements on the quality of the supplied equipment and its operation in order to reduce the non-identified leaks;
- usage of the purified water to make-up the primary circuit and for service water of the process systems in the controlled access area.

The experience in designing, operation and modernization of the systems of liquid radioactive media recycling at the NPPs in Russia and worldwide, as well as perspective technologies have been considered in the course of the recycling system design development.

The system of liquid radioactive media intermediate storage is envisaged to take up the full scope of the media and its storage for not less than three months for the short-living radionuclides to decay before the subsequent recycling.

The liquid radioactive media that are collected in the system of intermediate storage comprise:

- salt concentrate from the evaporation plants;
- spent ion-exchange resins (filtering materials);
- crud.

The processes of liquid radioactive media recycling allow obtaining the final product that meets the requirements of the valid regulatory and technical documentation for radwaste storage and burial.

Solid radioactive waste management. The items below are the solid radioactive wastes:

- Overalls, footwear, means of individual protection not to be subjected to decontamination;
- metal refuse;
- construction and thermal insulation materials;
- gas purification and venting system filter components;
- solidified liquid radioactive media.

The amount of the radioactive wastes, including solid radwastes that come for storage per year does not exceed 50 m^3/year (per 1000 MW) from one Unit. The value does not consider both the contaminated equipment and tools that had been applied in repair and maintenance activities and the volume of the NZK-150-1.5P containers.

At all the stages of the NPP service life management of all kinds of radwastes has been arranged to ensure safety at their transportation and storage.

Depending on the level of activity and recycling methods, the solid radwaste collection and sorting shall be performed in the places they have been generated by distributing in appropriate containers or disposables. Check sorting is carried out in the waste recycling building furnished with storage.

To reduce their volume before they go to the storage the solid radwastes are subjected to treatment:

- chopping;
- burning;
- pressing.

A set of design solutions for the power Unit normal operation is to provide:

- temporary storage of conditioned radioactive wastes in the NZK-150-1,5P containers. The storage capacity is intended for a 5-year period of the NPP operation with a possibility to extend the capacity for being used up to 10 years. Further on, the containers are supposed top be taken away to the regional storage to be buried;
- storage of high-level radioactive wastes in special cells during the entire service life of the NPP. The high-level radioactive wastes come to the storage packed into special capsules on the equipment that is part of the reactor plant fuel handling equipment.

The systems are equipped with the process and radiation monitoring devices needed to control the system functioning, estimation of the system integrity and environmental release monitoring.

Gaseous radwaste management system. The system of gaseous radioactive waste management comprises the special venting systems and gaseous radioactive blowdown decontamination system.

The system of the process blowdown decontamination is designed to reduce the releases of radioactive inert gases, gaseous compounds of iodine and aerosols from the gas blowdowns of the process equipment in reactor compartment to permissible levels.

The system provides:

- Continuous purification of gas offs from the system of hydrogen ignition;
- Continuous purification of gas offs from the system of tank facilities and reagents.

Plant layout

The reactor building is a construction combining the double-envelope containment and internal structures on a common foundation. The foundation is a 2.4 m thick reinforced concrete slab 51.6 m in diameter.

The double-envelope containment comprises and internal envelope made of pre-stressed reinforced concrete, which makes it a load-bearing structure that absorbs well the tensile stresses of the overpressure in case of an accident. The metal lining on internal surface only provides air-tightness in this case. The external envelope is made of ordinary concrete, that is not pre-stressed as the application of the latter is inexpedient in case of an external impact when additional compressive forces arise that lead to concrete destruction in the compressed area.

The internal envelope is a pre-stressed concrete structure in the form of a cylinder with a semi-spherical dome. The pre-stressed concrete structure of the reactor building is mounted into the structures at the elevation mark +0.00 m. The internal diameter of the cylinder is 44.0 m, it is 42.2 m high and the internal surface of the dome apex is at the elevation mark +68.60 m.

The internal structures of the building are housed inside the space limited by the internal envelope and the foundation slab and they are made of monolithic reinforced concrete. The building contains the reactor cavity, spent fuel pond and internals inspection well, three intermediate floors and vertical structures of the walls and columns. The construction makes up a single structure free of temperature seams.

The safeguards building is designed to house the equipment and pipelines of the emergency core cooling system, spray system, essential consumer cooling system, emergency boron supply system, system for the creation and maintenance of rarefaction in the containment annular space. The building houses the systems important for safety.

The *auxiliary building* houses the primary-side auxiliary systems, water- and gas purification systems, liquid radioactive waste collection and storage systems and equipment of venting systems of the controlled access area. The building is a many-storied rectangular construction made of monolithic reinforced concrete structures with the dimensions in the plan 44.90 x 51.20 m.

The auxiliary building is part of the nuclear island. Below are enumerated the buildings of the nuclear island that are adjacent to the building on the three sides:

- nuclear service building with the refreshment area of the controlled access zone;
- control building;
- reactor building;
- safety building;
- fresh fuel and solid radwastes storage or fuel handling equipment and solid radwastes storage.

The auxiliary building is separated from adjacent buildings with aseismic seams.

The control building is designed to house the electric engineering and instrumentation and communication systems that provide control of the power Unit both under normal operating conditions and in accidents. The building is divided into four independent areas that house the safety-related equipment.

The control building is placed between the reactor building, steam cell, nuclear service building with the refreshment area of the controlled access zone and auxiliary building. It is separated from the adjacent buildings with aseismic seams.

The building is 50.8 m long and 43.2 m wide as the maximum and there are eight floors in it.

The steam cell is designed to house the equipment and pipelines of steam generator overpressure protection system, feedwater system and demineralized water supply system. The equipment and pipelines of all the systems are divided into four independent safety channels.

The steam cell is a part of the nuclear island. The building is situated between the reactor building and turbine building. It is separated from the reactor building with an aseismic seam.

The building is 33.95 m long and 15.7 m wide as the maximum.

The *fresh fuel and solid radwastes storage building* contains the solid radwastes storage on the ground floor and the fresh fuel storage on the first floor.

The building is situated at the NPP Unit 1 site but it services two power Units as it contains the fresh nuclear fuel storage intended for fuel receiving and storage, its capacity being designed for two units.

The fresh nuclear fuel storage is intended for fuel receiving, storage and preparation for core loading and serves two power units.

The solid radwastes storage is designed to store the solid and solidified wastes and control rods. Solid wastes are sorted out, chopped and pressed in the storage. Liquid radwaste are solidified by concreting.

The nuclear service building with the refreshment area of the controlled access zone is designed:

- to control the access of the personnel to the controlled area;
- to perform the decontamination and repair of the controlled area equipment;
- to analyze the process media in the controlled area;
- to decontaminate the overalls of the Unit personnel;
- to repair and maintain the I&C hardware;
- to prepare and supply the chemical reagents for the chemical cleaning of the steam generators on the secondary side, for decontamination systems, for the primary and secondary water chemistry maintenance system.

The nuclear service building with the refreshment area of the controlled access zone is part of the nuclear island. The building is situated between the auxiliary building and control building. It is separated from the adjacent buildings with aseismic seams. There are four overground and one underground floors in the five-floor building.

The Unit diesel power plant with the reserve of diesel fuel is designed to house the diesel-generator that is an independent source of power supply to the station auxiliaries important for the NPP safety that ensure the preservation of the main equipment under the loss of power conditions. The building also houses the switchgear of the diesel plant, control panel and venting system rooms.

The Unit diesel power plant is standing at a distance from the other buildings and close to the chemical cleaning tanks for station auxiliaries. The building of the Unit diesel power plant comprises the diesel-generator room and a two-floor semidetached building.

There are two overground floors and a basement in the building.

The back-up diesel power plant with an intermediate warehouse for diesel oil is designed to house the equipment of the emergency power supply system. The building is divided into four sections that house the equipment of completely independent safety channels.

The building of the back-up diesel power plant is a separate building situated opposite to the safety building.

The building is 39.0 m wide and 68.4 m long as the maximum.

The *turbine building* is designed to house the systems and equipment of the secondary side related to power generation. The turbine building houses a turbine plant, feedwater pumps, deareator, auxiliary equipment. The turbine building is located along the axis of the reactor. A fire protection passage is envisaged between the turbine building and the reactor building.

Water treatment, normal power supply system building and heating system building are adjacent to the turbine building. All the buildings are separated from each other with aseismic seams.

The turbine building is a rectangular single-span construction with the dimensions in the plan in the axes 121×51 m.

The normal power supply system building is designed to house the electric engineering systems and I&C systems of normal operation. It also houses the support systems of venting, water supply, sewage and fire extinguishing.

The normal power supply system building is adjacent to the turbine building and to the heating system building and it is separated from them with aseismic seams.

Two cable tunnels are adjacent to the normal power supply system building.

The dimensions of the building in the plan are 81.60x14.10 m. There are three overground and two underground floors in the building.

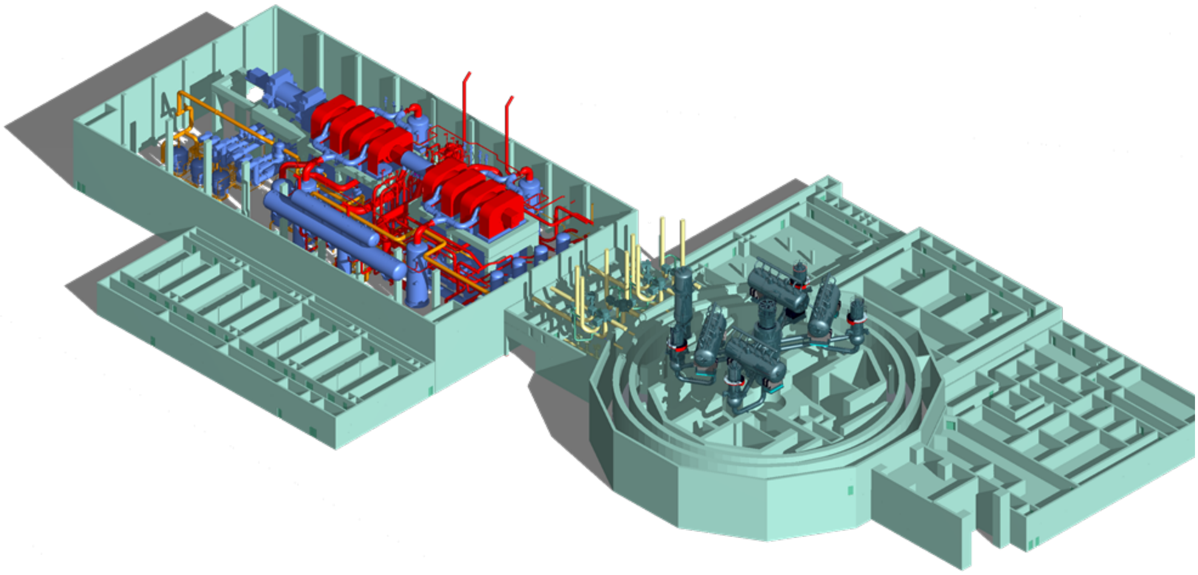

Figure 3 – NPP layout

10.1 NPP operation

The Unit operates in accordance with the requirements of the regulatory documents, process specifications and operating instructions. The RP operation is possible both in the basic mode and in the mode of load maneuvering.

The reactor plant has four loops. The coolant temperature at the reactor outlet is 328.9±5.0°C, the design pressure of the primary side is equal to 17.64 MPa. There is one motor-operated RCP set in each loop with the required inertia parameters and one horizontal steam generator with submerged heat exchange surface. The pressure of the generated steam at nominal load at the steam header outlet is 7.0±0.1 MPa (the design pressure is 8.1 MPa). The total steam capacity of the reactor plant is 6408 t/h.

10.2 Reliability

Reliability indices:

- frequency of emergency shutdowns does not exceed 1 event per a year of operation;
- equipment availability factor is 99%.

The values are confirmed in the framework of analyses of equipment reliability and availability.

10.3 Load factor indices

The capacity factor is 0.92. Annular load factor is 0.9. The given indices are achieved by improving the design of certain equipment of the RP, optimization of repair cycles for individual equipment and RP (bringing them to a single 8-year cycle), optimization of the schedule of each shutdown, introduction of a progressive system of the preparation and organization of maintenance and repair, usage of the fuel handling machine operating at elevated displacement rates, automated multi-position power nut drives for simultaneous stud extraction from the equipment flange joints (reactor main joint, steam generator collectors, all the RP tanks etc.), fresh fuel loading (FAs, RCCA absorber rods) into the spent fuel pond during the primary side cooldown, combination of core refueling and the activities on maintenance and repair of SGs thanks to the application of the isolation devices installed into the steam generator collectors etc.

10.4 Reduction in capital, construction and fuel costs

When assessing and optimizing the cost of power generation (production costs) the following items were considered:

- capital costs;
- running costs and maintenance costs;
- fuel cost (beginning and end of fuel cycle);
- costs of facility decommissioning.

In the course of AES-2006 Project elaboration special attention was paid to the issue of capital costs reduction, because the capital costs are the main component of power generation cost and hence, their reduction leads to an increase in the object competitiveness. A solution to the problem was searched in the following main directions:

- reduction in man-hours for designing;
- decrease in R&D scope to justify the design;
- decrease in the length of construction;
- construction management using to the utmost the local construction and industrial facilities, raw materials and human resources available;
- compact layout of buildings and constructions on the site, process systems and equipment inside the buildings to save up the space;
- decrease in specific materials consumption;
- usage of the sites of the operating NPPs to construct new facilities.

The design solutions have been improved in a way to considerably decrease the running costs and thus increase the competitiveness of the object. The problem was solved using the economically and technically substantiated methods in the following directions:

- reduction in depreciation allocations due to an increase in the equipment service life, increase in its reliability, exclusion of the necessity or reduction to a minimum of the amount of in-service repair etc.;
- reduction in the running costs of radwaste and spent fuel management;
- reduction in consumed resources, first and foremost, power- and water consumption;
- reduction in chemical and other industrial wastes;
- reduction in the power to the station auxiliaries;
- saving up the heat;
- reduction in the quantity of the personnel and annual dose commitments;
- reaching the maximum possible NPP service life to reduce the current allocations to the foundation for the facility decommissioning.

10.5 Construction schedule

Length of construction for a series power Unit (beginning with the initial concrete placement until connection to the grid) does not exceed 54 months with a 12 months interval for commissioning of each next power Unit at the given site.

10.6 Refueling cost reduction

The design solutions are optimized in a way to reduce the fuel costs as part of the expenditures and thus improve the characteristics of the facility. The studies are performed in the area of:

- improving fuel cycle characteristics, including the issues of fuel enrichment, burnup increase as well as its optimum placement in fuel assemblies;
- decrease in neutron leakage;
- optimization of fuel loading and cycle length (both in the basic load mode and in the mode of load maneuvering);
- increase in the Unit efficiency, etc.

Development status of technologies relevant to the NPP

The composition and the structure of the main components, equipment and systems of the NPP to the AES-2006 Project are based on the solutions used in the previous designs and improved in accordance with the demands of today. It allows increasing the operational characteristics of the NPPs providing a required safety level.

The principal engineering solutions for a NPP have been proven by the operating experience of RPs with VVERs for more than 1400 reactor-years including more than 500 reactor-years of VVER-1000 reactor plant operation.

The usage of an evolutionary approach in solving the key problems in the course of design development made it possible to prepare the AES-2006 design within a short period of time, the design meeting the newest Russian standards and rules.

AES-2006 design has been developed at the required level of competitiveness and its prospects for commercial application both in Russia and abroad are well-grounded.

Deployment status and planned schedule

12.1 Design Status

So far the elaboration of RP V-491 design has been completed. The Rostechnadzor of the Russian Federation has issued a construction permit for the power Units with V-491 reactor plants. The work is under way on the NPP construction at the Leningrad NPP-2 and the Baltic NPP sites.

12.2 Information on the situation around R&D.

R&D activities are carried out to substantiate the design engineering solutions, optimize the system and equipment characteristics and other important operational parameters within the framework of AES-2006 design elaboration in the following directions:

- A complex of activities to eliminate the design solution conservatism in aspects dealing with the requirements for the external systems on the part of the RP (considering the increase in RP power and parameters);
- A complex of activities to increase the fuel burnup;
- A complex of activities to substantiate the increase in power, efficiency and load factor;
- A complex of activities to increase the reliability of equipment and extend its service life including the complex of work in materials study;
- A complex of activities to provide the conditions of load maneuvering;
- A complex of activities to substantiate the permissibility of deviations from the standards and technical documentation;
- A complex of activities in radiation and nuclear safety;
- A complex of activities to improve the I&C systems;
- A complex of activities to improve the primary- and secondary water chemistry and the process of their maintenance including the activities to substantiate the hydrogen safety.

12.3 The companies/institutes involved in R&D and design development:

A number of the leading organizations and institutes of the Russian Federation took part in the design development and R&D execution, among them:

1. Experimental and Design Organization OKB “Gidropress” (OKB GP);
2. Atomenegroproekt of Saint-Petersburg (SPbAEP);
3. Russian Research Centre Kurchatov Institute (RRC “ Kurchatov Institute ”);

4. All-Russia Research Institute of Inorganic Materials named after Academician A.A.Bochvar (VNIINM);
5. All-Russia Research Institute for Nuclear Power Plant Operation (VNIIAES);
6. TVEL;
7. Central Design Organization for Machine-Building (TsKBM);
8. Research Institute for Atomic Reactors (RIAR);
9. Institute for Safe Nuclear Power Industry Development (IBRAE);
10. Institute for Physics and Power Engineering named after Academician A.I.Leipunksy (IPPE);
11. Special Research Institute for Instruments (SNIIP);
12. Research Technological Institute named after A.P.Alexandrov (NITI);
13. Research-and-Production Association "Luch" (NII NPO "Luch");
14. Electrogorsk Research and Engineering Centre (EREC);
15. Central Research and Development Institute for Process Machine-Building (TsNIITMASH);
16. Central Research and Development Institute for Structural Materials "Prometey";
17. Engineering Centre for Strength and Materials Study of Components for Nuclear Power Engineering (TsP MAE);
18. Moscow Complex Metal Plant (MZP).

References

1. General provisions for safety assurance of nuclear power plants (OPB-88/97). NP-001-97 (PNAE G-01-011-97), Moscow, 1997
2. Code for designing earthquake-resistant nuclear power plants. NP-031-01, Moscow, 2001.
3. Safety regulations for reactor plants of nuclear power plants (PBYa RU AES-89), PNAE G-1-024-90, Moscow, 1999

Technical data

General plant data

Reactor thermal output	3200 MWth
Power plant output, gross	1170 MWe
Power plant output, net	1082 MWe
Power plant efficiency, net	33.9 %
Mode of operation	Baseload and Load follow
Plant design life	60 Years
Plant availability target >	90 %
Primary coolant material	Light Water
Secondary coolant material	Light Water
Moderator material	Light water
Thermodynamic cycle	Rankine
Type of cycle	Indirect

Non-electric applications	District heat

Safety goals

Core damage frequency <	1E-6 /Reactor-Year
Large early release frequency <	1E-7 /Reactor-Year
Occupational radiation exposure <	0.39 Person-Sv/RY
Operator Action Time	6 Hours

Nuclear steam supply system

Steam flow rate at nominal conditions	1780 Kg/s
Steam pressure	6.8 MPa(a)
Steam temperature	283.8 °C
Feedwater flow rate at nominal conditions	1780 Kg/s
Feedwater temperature	227 °C

Reactor coolant system

Primary coolant flow rate	23888 Kg/s
Reactor operating pressure	16.2 MPa(a)
Core coolant inlet temperature	298.2 °C
Core coolant outlet temperature	328.9 °C
Mean temperature rise across core	30.7 °C

Reactor core

Active core height	3.75 m
Equivalent core diameter	3.16 m
Average linear heat rate	16.78 KW/m
Average fuel power density	36.8 KW/KgU
Average core power density	108.5 MW/m^3
Fuel material	UO_2 and $UO_2 + Gd_2O_3$
Fuel element type	Fuel rod
Cladding material	Alloy E-110
Outer diameter of fuel rods	9.10 mm
Rod array of a fuel assembly	hexahedral

Number of fuel assemblies	163
Enrichment of reload fuel at equilibrium core	4.79 Weight %
Fuel cycle length	12 Months
Average discharge burnup of fuel	60 MWd/Kg
Burnable absorber (strategy/material)	Gd2O3
Control rod absorber material	B4C + Dy2O3TiO2
Soluble neutron absorber	H3BO3

Reactor pressure vessel

Inner diameter of cylindrical shell	4232 mm
Wall thickness of cylindrical shell	197.5 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	4
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	4
Pump speed	1000 rpm
Head at rated conditions	62.2 m
Flow at rated conditions	5.97 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
Active devices used	Heaters

Primary containment

Type	Hermetic containment, pre-stressed reinforced concrete
Overall form (spherical/cylindrical)	Cylindrical part and hemispherical dome
Dimensions - diameter	44.0 m
Dimensions - height	42.2 m
Design pressure	0.5 MPa
Design temperature	150.0 °C
Design leakage rate	0.2 Volume % /day

Residual heat removal systems

Active/passive systems	Active and passive systems

Safety injection systems

Active/passive systems	Active and Passive

Turbine

Type of turbines	K-1200-6.8/50
Number of turbine sections per unit (e.g. HP/MP/LP)	1/0/4
Turbine speed	3000 rpm
HP turbine inlet pressure	6.8 MPa(a)
HP turbine inlet temperature	283.8 °C

Generator

Type	TZV-1200-2UZ
Rated power	1333 MVA

Active power	1200 MW
Voltage	24 kV
Frequency	50 Hz

Condenser

Type	Single-pass, double flow
Condenser pressure	4.9 kPa

Feedwater pumps

Type	Motor Driven - PEA 2050-85
Number	5
Pump speed	3000 rpm
Head at rated conditions	900 m
Flow at rated conditions	0.502 m^3/s