Status report 107 - VVER-1200 (V-392M) (VVER-1200 (V-392M))

Overview

Full name	VVER-1200 (V-392M)
Acronym	VVER-1200 (V-392M)
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-392M 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, 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 a radioactive release;
- determination of the measures to control the beyond-design basis accidents.

The VVER-1200 (V-392M) design was developed by the Organization of General Designer, “Atomenergoproekt”, Organization of General Designer Organization 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-392M 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, which ensures a possibility of a long-term post-accident residual heat removal in case of a primary LOCA in a combination with the station blackout;
- 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-392M design and the Novovoronezh NPP-2 design are provided in the Technical data.

Description of the nuclear systems

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 into 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

Parameter	Value
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 is 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.

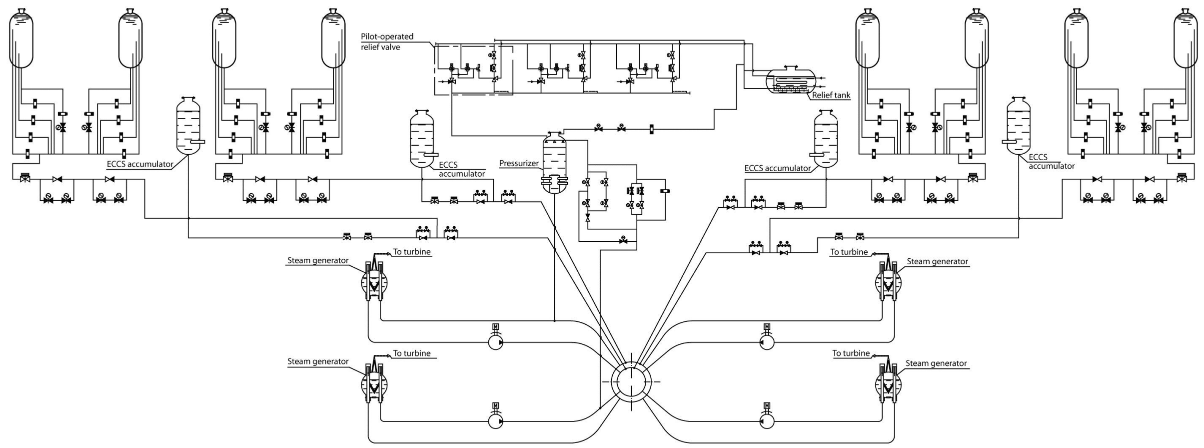

Figure 1 – Schematic flow diagram of the primary circuit

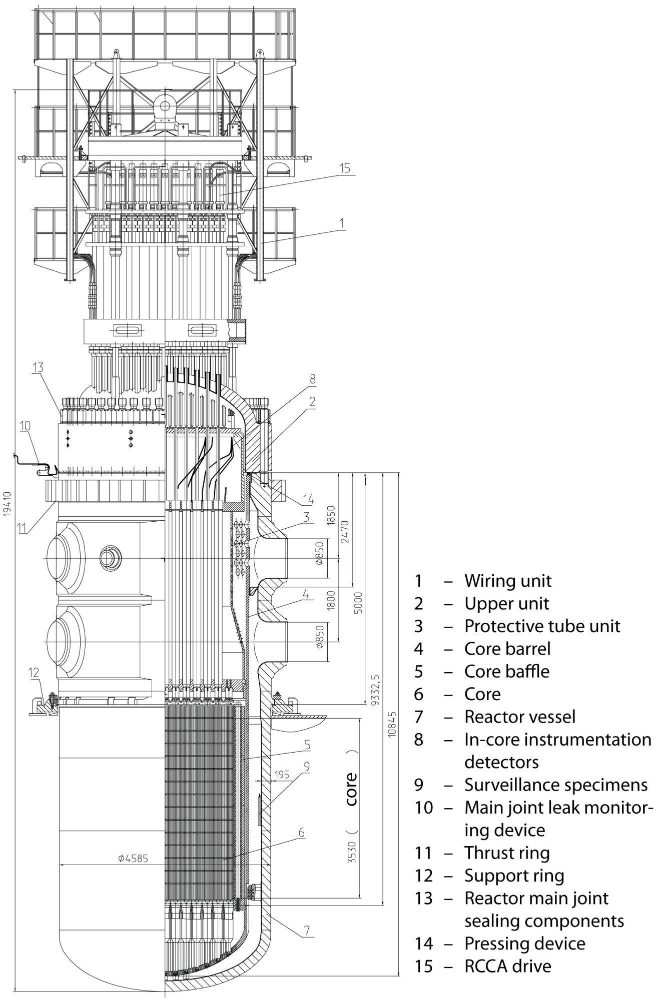

Figure 2 – Reactor

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 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 corridor 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 compressed air supply system for valve pneumatic drives is designed to feed the compresses air at pressure from 4.5 to 5.0 MPa to the pneumatic drives in order to keep the valves in the open position.The system of nitrogen supply to the blowdowns in the reactor building equipment is designed to supply nitrogen to the gas blowdown system.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 process blowdown radioactive gas purification.The filtering and venting systems provide the rated environmental conditions for the operation of the personnel under different conditions of NPP operation and cleanness of the air in the rooms, prevent the fouling of the atmosphere with radioactive and toxic substances, maintain the optimum conditions of the process equipment operation.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 ± 2% N_{nom} and by not more than ±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 ±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 %	15000	A cycle implies a change of power and return to initial level

N_{nom}/min within the range from 50 % N_{nom} to 100% N_{nom}		
Change of Unit power within the range from 50 % N_{nom} to 100% N_{nom} under accident situations in the power grid: power increase at a rate of 5 % N_{nom}/min and further power decrease at a rate of 20 % N_{nom}/min	100	
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 VVER-1000 NPP 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-392M design is under way in Russia at the Novovoronezh NPP-2 according to the permit granted by the Rostechnadzor Regulatory Authority of Russia.

Description of safety concept

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-392M design:

- Emergency steam generator cooldown systems;
- Emergency gas removal system;
- Emergency boron injection system;
- System of the primary circuit emergency and planned cooldown and spent fuel pond cooling;
- Main steamline isolation system.

The following passive safety systems are incorporated into the V-392M design:

- Emergency core cooling system, passive part;
- Emergency core passive cooling systems;
- Systems of passive heat removal;

- 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 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 (CDF) probability for a one-year fuel cycle, as predicted by the design organizations, is provided in Table 4.

Table 4

Conditions	CDF probability, reactor/year
Operating conditions	$3.1x10^{-7}$
Standby conditions	$3.0x10^{-7}$
Total	$6.1x10^{-7}$

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 and the beyond design basis management system.

The following active and passive safety systems are implemented in V-392M design:

- ***system of the primary circuit emergency and planned cooldown and spent fuel pond cooling*** is designed, in particular, for residual heat removal from the fuel in reactor to the system of the reactor compartment consumers (intermediate circuit) under all the design basis conditions of the Unit operation, maintenance of the required coolant inventory in the reactor in case of a large-break LOCA, emergency primary make-up in the conditions of a small-break LOCA (D nom 25-80) and medium supply to the spay system;
- ***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 of 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;
- ***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 part of the system of emergency and planned cooldown and spent fuel pond cooling actuate in design-basis loss-of-coolant accidents;
- ***passive core flooding system*** is designed to provide the boric acid solution supply with a concentration of 16 g/dm^3 into the reactor core at primary pressure decrease to 1.5 MPa and less in order to replenish the water inventory in the core to a safe level, thus providing its reliable cooling together with the passive heat removal system in loss of coolant accidents in reactor coolant system including those coinciding with loss of AC power sources for a period of 24 hours and more. The boric acid solution supply in the tanks of passive core flooding system is used to fill up the reactor core barrel and reactor internals inspection well for the period of refueling;
- ***emergency 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 create the required 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 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 the shut-off electric motor-operated gate valves, considering the advance actuation of BRU-A and reactor trip system;
- ***passive heat removal system*** is designed for long-time residual heat removal from the core at a beyond design basis accident - loss of all the sources of AC power supply on condition of the primary and secondary side integrity retention. Besides, under certain scenarios of beyond design basis accidents with a primary or secondary leak combined with a simultaneous loss of all the sources of AC power supply, the PHRS contributes to providing the required coolant inventory in the primary circuit by the steam condensation in the SG tubing, the steam being generated in the reactor due to heat removal from the SG on the secondary side and to the PHRS and the condensed primary coolant is returned into the reactor.
- ***steam generator emergency cooldown system*** performs the safety functions to remove residual heat from the core and cool the reactor plant down via the secondary side;
- ***main steamline isolation system*** is designed for quick and reliable steam generator isolation from a leaky section. The system is designed for work under all the accidents that require a SG isolation:
    - At pipeline breaks downstream of the SGs as far as the turbine stop valves in the pipeline sections that either can be isolated and 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*** is 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 in AES-2006 design for the Novovoronezh NPP-2.

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 and 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 decease 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
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
Design temperature of cooling water, °C	20
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

Under normal operating conditions the system is designed to perform the functions of maintaining the level in steam generators and providing them with feedwater.

Under the nominal and partial power operation the system provides the steam generators with continuous water supply at a rate of 300 – 6480 t/h through the main control feedwater valves and at a rate of not less than 350 t/h when the supply is realized through the bypass control valves if the main steam pressure upstream of the steam generators is equal to 6.9 MPa. Under the conditions of residual heat removal one bypass line with a control valve provides water pass at a rate of not less than 250 t/h at the main steam pressure upstream of the steam generators equal to 7.9 MPa.

The system provides the control of water supply to the steam generator under all the design-basis conditions (including start-up and shutdown modes).

The system provides a quick termination of feedwater supply in case of steam generator overfilling and in case of the breaks of feedwater pipelines, SG steam line in the section that cannot be separated and in case of a primary-to-secondary leak.

The system of the main feedwater pipelines comprises four main feedwater pumps (4x25%), one backup feedwater pump (1x25%), valves and pipelines.

The quality analysis of the system shows that it meets the imposed safety requirements of the regulatory documents and performs its functions under all the conditions that require its operation. There are no deviations from the technical documentation and standards.

6.2.2 Auxiliary feedwater system

The auxiliary feedwater system is designed:

- to provide steam generators with feedwater under the conditions of normal operation (commissioning hydraulic tests, main steam line heat up) shutdown to the state of hot standby, cooldown);
- to provide steam generators with feedwater under anticipated operational occurrences in case of a shutdown and subsequent cooldown, under the conditions of loss of normal heat removal through the secondary side (through the turbine condenser) as well as in case of loss power to the station auxiliaries;
- to provide steam generators with feedwater under failures that are accompanied with limitations in water supply from the system of main feedwater at the Unit power operation;
- to provide the pre-startup deaeration of feedwater (with the secondary-side deaerator);
- to provide the post-cooling of steam generators in water-to-water conditions at the Unit repair shutdown.

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

6.3 Auxiliary systems

The systems below belong to the turbogenerator auxiliary systems:

- Turbogenerator oil equipment system;
- Hydraulic hoisting and shaft rotation system;
- Turbine sealing system;
- Turbine drain system;
- Oil feeding system for turbine adjustment;
- Main condenser vacuumization system;
- High pressure regeneration system;
- Low pressure regeneration system;
- Steam separation and intermediate heating system;
- Main steam line system;
- steam dump valve to the turbine condenser (BRU-K) system;
- Demineralized water system;
- Turbine condensate demineralization and iron removal system;
- Secondary-side sampling system and Unit demineralization plant;
- Secondary-side automated water chemistry control system.

Electrical and I&C systems

7.1 Power supply systems

The accepted electric wiring circuitry is:

- For the 500 kV switchgear – "breaker-and-a-half" scheme;
- For the 220 kV switchgear – two working busbar systems.

The turbogenerator is connected the 500 kV switchgear according to the generator – transformer block scheme. Power generator switches are installed between the generator and each transformer block in the generator circuit. Turbogenerators with complete water cooling of 1200 MW with rotational speed of 3000 rpm are envisaged in the design. The generators are made in a six-phase embodiment and have two three-phase windings mutually shifted by 30 electrical degrees. Each generator wiring is connected to the transformer via busducts 24 kV, 20 kA and a switch. The transformer is connected to the 500 kV switchgear via a flexible tie. When the block is switched off from the 500 kV switchgear with an instantaneous power Unit load shedding, the control system of each power Unit provides maintaining it at the power level for station auxiliaries.

7.2 Safety-related electric systems

The emergency power supply system is a system to supply power to the Unit safety system consumers under all the conditions of operation including those of loss of working and backup sources supplied by the power grid. The system contains the independent power supply sources, distribution and commutation devices.

7.3 Layout of the main control room (MCR)

The MCR is conventionally divided into 2 control circuits: operative and inoperative. A technical support center is envisaged in an adjacent room. The inoperative control circuit (MCR-I) with individual workstations takes off part of the work on auxiliary system control from the leading operators (shift engineer for reactor control and shift engineer for turbine control). It allows paying more attention to providing the power Unit safety and control of the main process of power generation. The operative part of the MCR is designed to arrange the control of the main equipment of the primary and secondary circuit and electrical facilities of the power Unit. It houses three automated workstations of shift engineer for reactor control, shift engineer for turbine control and the working area for the Unit chief of the shift. The Unit chief of the shift coordinates the work of all the operative personnel of the power Unit. The working area of the Unit chief of the shift is considered as a backup workstation for situations when the chief exercises control under the conditions of support rendered to the shift engineer for reactor control and shift engineer for turbine control at start-up, shutdown and under the transients. The working area of the Unit chief of the shift is equipped with two workstations, means of industrial television for the control of the situation in the main rooms of the power Unit and with special communication, including the communication with the external organizations as well as the means of control for the jointly used screen to call the mode videoshot.

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 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-392M 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 and internals inspection well pipeline system;
- system for spent fuel pond water supply for purification;
- 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 water cooling system is designed for:

- Residual heat removal from the spent fuel located in the spent fuel pond under all the operating conditions of the Unit;
- Pressure and temperature decrease in the area of accident localization in the course of design basis accidents.

The leak monitoring system for spent fuel pond lining is designed:

- to monitor the integrity of the pond lining, internals inspection well, refueling pond and cask section lining;
- water seal tightness check in the spent fuel pond.

In the course of NPP refueling procedures, spent fuel cooling and storage the contamination on the equipment and system structure is borne with borated water that fills up the reactor cavity and spent fuel pond. There is no induced activity in the equipment and system materials due to the deep subcriticality at spent fuel storage.

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.

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 houses the reactor plant, process and electric equipment of the safety and normal operation systems, control systems, venting and central heating systems. The dimensions of the reactor building are 78.00 x 72.00 m in the axes in plan. It is a double-envelope cylindrical concrete containment that houses the accident localization area and the surrounding structures positioned on the common foundation slab.

The containment is an element of the accident localization system and is made up of two envelopes, internal and external to protect it from external impacts. There is an annular space between the envelopes where the safety system cable channels are located. The space serves to collect the radioactive media leaking through the internal envelope in case of accidents.

The construction of the containment is described below. The internal envelope is a pre-stressed concrete structure, 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.

Reactor plant, spent fuel pond, the systems that operate under the primary-side parameters, the venting systems and equipment for fuel handling procedures and repair are located inside the accident localization area in the containment.

The reactor occupies the central place in the accident localization area. The spent fuel pond and the internals inspection well are located on the both sides of the reactor cavity as well as the RCS compartments that house steam generators, RCP sets and main coolant pipelines, the pressurizer and the relief tank.

The rooms for water treatment systems, vent chambers of the contaminated pipelines, venting plants are placed on the foundation slab of the containment and the allocated place to house the core catcher structures is under the reactor cavity.

The hydro accumulators of the passive part of the ECCS and passive core flooding systems are located at the elevation mark for servicing (elevation +26.300 m). The contaminated equipment washing facilities are located close to the spent fuel pond on the side of the transpiration lock.

The passive heat removal system is located on the external side of the dome and the upper cylindrical part of the external envelope.

The passive heat removal system exchangers are located over the containment perimeter. The cooling medium (atmospheric air) is supplied to the heat exchangers to via the air duct that provides uniform cooling as the direction of the wind changes. The exhaust tubes of the passive heat removal system are located on the dome of the external envelope. The filter of medium discharge out of the annular space (deflector) is situated on the apex of the envelope.

The turbine complex houses the systems and equipment of the secondary side related to power generation. The complex contains the turbine building, building for the normal operation power supply systems and the Unit demineralization plant.

The turbine building is designed to house the systems and equipment of the secondary side related to power generation. The turbine building comprises the turbine hall and deaerator hall. The turbine building houses the turboplant, feedwater pumps, deareator and auxiliary equipment.

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 equipment and storage for chemicals for the Unit demineralization plant are situated in a single-span building that is adjacent to the edge of the turbine building. The dimensions of the building are 45.0 x 21.0 m. It is a three-storey building without a basement floor.

The auxiliary reactor building is designed as a separate building that houses the rooms for normal operation process systems, water purification systems, process media tanks, exhaust venting center and electro-technical equipment that refer to the controlled access areas, and also the input ventilation center that refer to a free access zone.

The building is situated at a distance of 9.0 m from the reactor hall and is connected with it by pedestrian, transport and communication galleries and tunnels. The dimensions of the building are 60x60 m in the plan. There are six overground and one underground floors in the auxiliary reactor building.

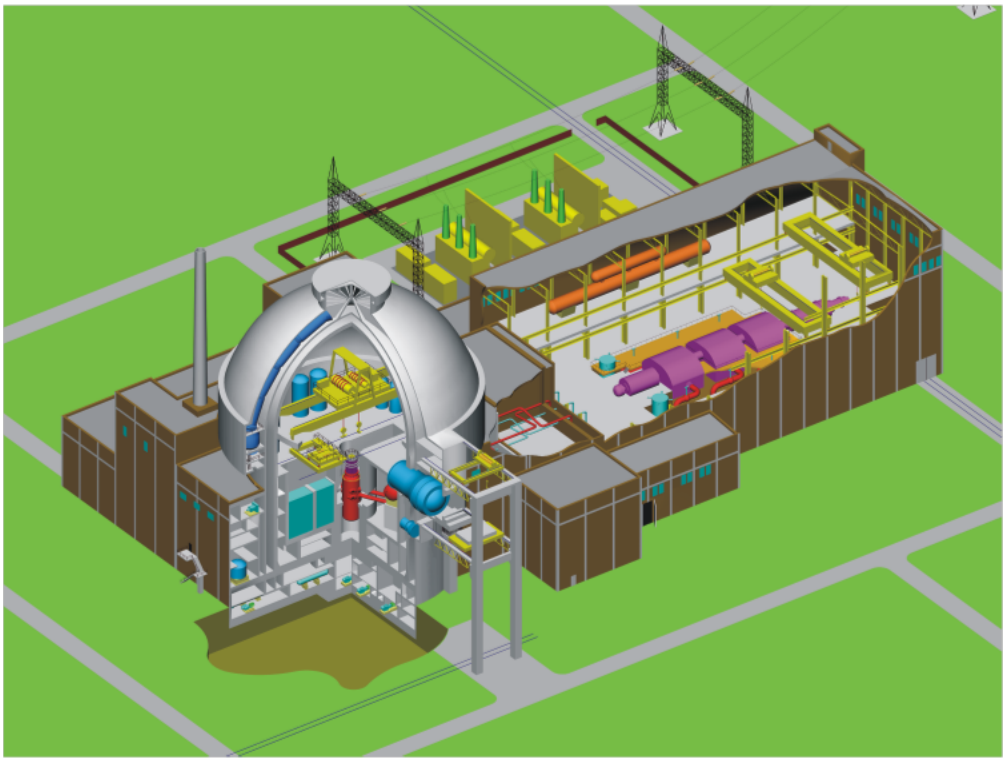

Figure 3 – NPP layout

The standby diesel power plants are situated on the opposite sides of the reactor building which guarantees the impossibility of a common cause failure resulting from the impact caused by an aircraft crash.

The standby diesel power plant refers to the free access area. In the underground area of the building process cells for the auxiliary equipment and venting equipment. The generator hall is served by an overhead crane with a 5 ton lifting capacity.

The interim fuel storage is situated close to the standby diesel power plant. It houses pump and underground metal tanks for fuel storage, clean and spent oil drain sumps, drain water sumps.

The tanks are placed into reinforced concrete chambers to protect the environment from possible leaks. The tanks are mounted before the upper floor of the chamber is concreted.

The fresh nuclear fuel storage is designed to receive and store the fuel. Fresh fuel is inspected and prepared for loading into the reactor. Fresh fuel is not a source of direct radiation, but all the possible precaution is taken that excludes and ingress of water or water-containing substances into the room for fresh fuel storage. Air humidity in the room for fresh fuel is limited. Radiation monitoring is mandatory in the fresh fuel storage.

The building for solid radwaste reprocessing and storage mainly refers to the controlled access area and is designed to reprocess and store solid radioactive wastes. The building is divided into a section for solid radwaste reprocessing and a section for solid radwaste storage.

The tunnels that connect the building of the standby diesel power plant and the reactor building are intended for cable routing for the systems of emergency power supply. The cable routes of each Unit are divided into two safety channels that pass in different tunnels. The cable tunnels go along the process tunnels for fire extinguishing and venting systems. The tunnel sections are made of monolithic reinforces concrete of increases water tightness and their dimensions are 1.7x2.7 m.

Two *spray ponds* are envisaged in the design of each NPP Unit. Each spray pond serves to cool water of two channels and is a subterranean construction divided into two parts (according to the number of the channels) with open and closed sections. Such a division ensures the retention of water volume in the pond in case of a shock wave or a tornado. The structural constructions of the pond are made of monolithic reinforced concrete. The pond water tightness is ensured by the application of appropriate grade concrete and external and internal dampproofing. The changeover chamber is designed to house the supply and discharge pipelines with the shut-off valves of the spray pond that provide the operation of different sections of the spray pond in different modes.

Each of the two *pump stations* is situated close to the spray ponds with changeover chambers. The pump stations are separated in the plan for the sake of protection against possible aircraft crash.

A pump station supports the operation of two physically separated channels of cooling water that have independent process parts and respectively consist of two different sections. The constructions of the pump stations are made of monolithic reinforced concrete.

10.1 NPP operation

The power 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 is a four-loop plant. The coolant temperature at the reactor outlet is 328.9±5.0°C, the design pressure of the primary side is 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) is predicted by the design and construction organizations to 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 power 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.

12.1 Design Status

So far the elaboration of RP V-392M design has been completed. The Rostechnadzor of the Russian Federation has issued a construction permit for the power Units with V-392M reactor plants. The work is under way on the NPP construction at the Novovoronezh NPP-2 site.

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 solutions 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 (AEP);
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
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	UO2 and UO2 + Gd2O3
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-UO2
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
Number	5
Pump speed	3000 rpm
Head at rated conditions	850 m
Flow at rated conditions	0.569 m^3/s