Status report 104 - VVER-640 (V-407) (VVER-640 (V-407))

Overview

Full name	VVER-640 (V-407)
Acronym	VVER-640 (V-407)
Reactor type	Pressurized Water Reactor (PWR)
Coolant	Light Water
Moderator	Light water
Neutron spectrum	Thermal Neutrons
Thermal capacity	1800.00 MWth
Gross Electrical capacity	645.00 MWe
Design status	Basic Design
Designers	Gidropress
Last update	01-08-2011

Description

Introduction

The reactor plant (RP) of medium power V-407 with water-cooled water-moderated power reactor VVER-640 is a constituent part of new generation NPP with the goals of improved technical-and-economic indices. The NPP Unit with VVER-640 is the first Russian Unit of evolutionary class with completely passive safety.

The NPP design is based on the principle of evolutionary development of the solutions confirmed by the positive operation experience of VVER plants, and on the principle of safety assurance: limitation of radiation influence on the personnel, population and environment to the established limits under normal operation and anticipated operational occurrences, as well as under a wide spectrum of accidents with a principally new approach to decay heat removal from the reactor core using only passive safety systems.

Safety assurance in RP V-407 design is provided by:

- implementation of the defense-in-depth principle based on application of the system of barriers to the release of ionizing radiation and radioactive substances into the environment, and also on the use of the system of technical-and-organizational measures on protection and maintaining the efficiency of each barrier;
- use of the main RP equipment (reactor, main coolant pipeline, steam generator (SG), pressurizer, emergency core cooling system (ECCS) hydro-accumulators, control and protection system (CPS) drives, fuel handling machine) developed on the basis of technical solutions for standard equipment manufactured and operated at NPP with VVER-440 and VVER-1000 (V-320);
- use of the fuel assemblies as nuclear fuel in the core of VVER-640 that are similar to VVER-1000 fuel assemblies with long-term operating experience at NPP, and also decrease in power density of the core as

compared with VVER-440 and VVER-1000;
- application of materials for main equipment of RP V-407 approved by long-term operating experience at VVER-440 and VVER-1000;
- use of double containment;
- application of diagnostic systems for the equipment of systems important to safety, for periodic inspections during shutdown, and for on-line monitoring during operation.

Advances in the reactor plant V-407 design, as compared with the designs of reactor plants VVER-440 and VVER-1000, consists, mainly, in the following:

- application of the advanced, more reliable reactor core that excludes positive reactivity effects and provides subcriticality by means of mechanical controls without boric acid at coolant temperatures above 100°C at any time during the cycle, including beginning-of-cycle (BOC);
- application of the safety systems based on the passive principles, using such physical properties as heat capacity, heat conduction, heat transfer, gravitation, energy of the compressed gas and other similar principles, instead of support on continuously rotating machine equipment (pumps, fans, coolers, diesel engines) for emergency heat-and-mass transfer;
- mitigation of consequences of design basis accidents (DBA). The design limit for such accidents as to fuel rod cladding temperatures does not exceed 993 K;
- solution of the problem of primary-to-secondary leaks at the expense of placing the SG safety valves inside the containment with simultaneous increase in the secondary-side design pressure;
- prevention of the core melting through the reactor vessel under severe accidents owing to application of the in-vessel corium retention concept;
- multi-purpose use of the engineered safety features and the systems provided in the design for mitigation and management of both design basis, and beyond design basis accidents, including severe accidents;
- mitigation of consequences of a wide spectrum of beyond design basis accidents (BDBA) using the standards applied to estimate DBA consequences;
- possibility for connecting the normal operation systems, important to safety, to reliable power supply from diesel-generators that are not loaded with bulky active safety systems in the given design, that, in its turn, enables to enhance stability of RP and NPP, as a whole, under incidents caused by loss of offsite power;
- increase in efficiency of fuel utilization by 15-20 % as compared with VVER-440 and VVER-1000;
- possible use of NPP with VVER-640 as nuclear cogeneration plant with heat output to 1000 GCal/h and for the purposes of water demineralization;
- possibility of 100 % core loading with MOX-fuel;
- increase in thermal efficiency owing to pressure increase at SG outlet to 7.06 MPa;
- decrease in the scope of monitoring for the purpose of readiness due to less number of tests and inspections;
- possibility for fuel storage until RP decommissioning;
- substantial improvement of fire safety of RP and NPP, as a whole, owing to the elimination from the design of combustible isolation, lubricating and cooling media, and improvement of fire protection and fire extinguishing means.

The R&D program for operability justification of VVER-640 reactor plant, in view of its evolutionary development and lowered core power density, provided for complete use of the R&D results obtained for design justification of VVER-1000 reactor plant. In experimental study of V-407 RP design under this program, the emphasis is shifted towards the new safety systems, such as the SG passive heat removal system (PHRS), containment PHRS, the system of the primary circuit de-pressurization and long-term cooling down through the emergency pool, that are based on the passive principles of action and having no analogues in the previous designs, and towards justification of the in-vessel corium retention concept under severe accidents. Experimental verification of the mentioned systems was successfully performed on the bench equipment in the following organizations: SPbAEP ("Scientific Research Design and Engineering Survey Institute", St.Petersburg), NPO CKTI (Scientific-Industrial Association, Central Research, Boiler and Turbine Design Institute"), SSC RF IPPE (State Scientific Centre Russian Federation, The Institute of Physics and Power Engineering), OKB "GIDROPRESS", OKBM (Experimental and Design Organization of Mechanical Engineering). Justification of joint operation of the passive safety systems will be completed at the test facility KMS-500 in Sosnovy Bor, simulating VVER-640 RP in scale 1:28.

One objective during the design process for the reactor plant and process systems of the station was to reach the estimated probability of the core severe damage not exceeding 1.0E-6 reactor/year, and a probability of emergency accident release, prescribed by the standards, not exceeding 1.0E-7 reactor /year.

The design was developed by St.Petersburg AEP ("Scientific Research Design and Engineering Survey Institute") and OKB "GIDROPRESS" with Russian Research Center (RRC) "Kurchatov Institute" as a scientific leader.

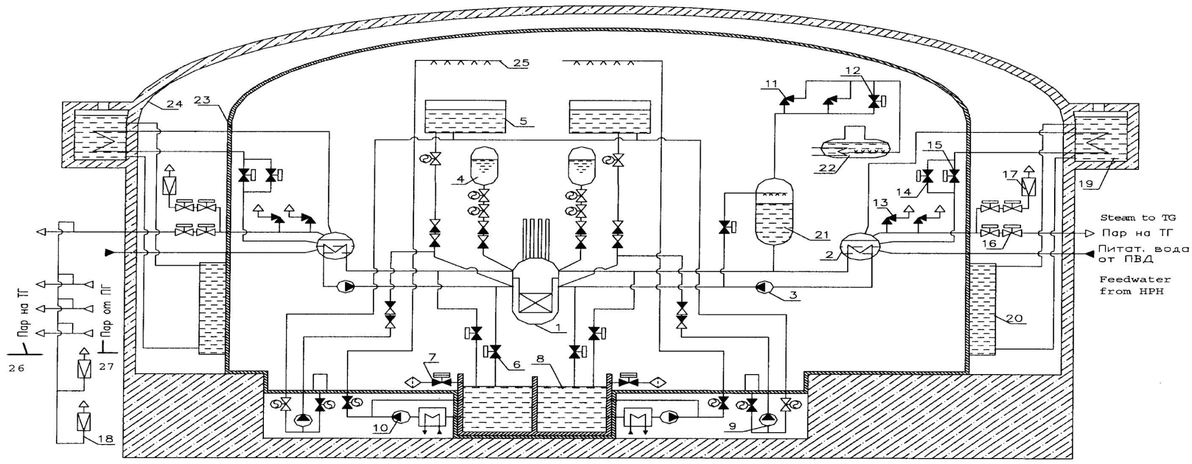
The design took into account the quality assurance requirements of IAEA and ISO 9000. Main technical solutions for VVER reactors are confirmed by operating experience of these plants during approximately 1000 reactor-years, including more than 300 reactor-years of VVER-1000 operation.

Description of the nuclear systems

2.1 Reactor coolant system and its main characteristics

The simplified diagram of the reactor plant and its main systems is presented in Figure 2.1.

The reactor plant includes the reactor coolant system (the primary circuit) and the associated systems required for its normal operation, emergency cooling, emergency protection and maintaining the safe state. The primary circuit consists of the reactor, 4 loops, each containing the main coolant pipeline, steam generator of horizontal type and reactor coolant pump set (RCP set), and the pressurizing system as a part of the system of pressure creation and maintenance, and the primary circuit overpressure protection system.
The main technical characteristics of the Unit are presented in the Appendix.



1 -реактор -reactor
2 -парогенератор - steam - generator
3 -ГЦНА -RCP set
4 -гидроемкость САОЗ -ECCS hydraulic accumulator
5 -бак САОЗ - ECCS tank
6 -арматурный блок разгерметизации - depressurization valve unit
7 -клапан мембранный -memdrane valve
8 - бассейн выдержки топлива -spent fuel pond
9- насос аварийной подпитки -emergency make-up pump
10- насос системы охлаждения топлива -fuel cooling system pump
11- ИПУ КД - PRZ PORV
12-устройство разгрузки -relief device
13 - ИПУ ПГ - SG PORV
14 - "малый" клапан СПОТ - PHRS «small» - valve
15 - "большой" клапан СПОТ PHRS «large» valve
16 - БЗОК - MSIV
17- "малое" БРУ-А - «small» BRU-A
18- "большое"БРУ-А - «large» BRU-A
19- бак СПОТ - PHRS tank
19 - теплообменник СПОТ ГО - TC PHRS heat - exchanger
21 - компенсатор давления - PRZ
22 - барботер - tank relief
23 - герметичная оболочка - sealed metal envelope
24 - защитная оболочка - protective concrete envelope
25 - спринклерное устройство - sprinkler device
26 - Пар на TG - Steam to TG
27 - Пар от ПГ - Steam from SG

Figure 2.1 Schematic diagram of V-407 reactor plant

2.2 Reactor core and fuel design

Requirements for the core, as a whole, and for the core components, as stated in corresponding sections of the Russian regulatory documents/1.2/, are the design bases of the reactor core. The reactor core design reflects operation experience of the existing reactors. The reactor core is designed to have stabilizing coefficients of reactivity in all operational modes to assure the main safety objectives. Reactivity control is accomplished by means of changing boron concentration in the primary coolant and by changing the position of CPS control rods. Slow reactivity changes due to burnup and changes in xenon amount are usually compensated by change in boron concentration, whereas quick reactivity changes for driving the power level towards the current loading are usually made by insertion/withdrawal of CPS control rods.

Reactor power level is monitored using the ex-core instrumentation detectors, and the core volume power distribution is determined by indications of the in-core instrumentation detectors (ICID) wherein the self-powered neutron detectors are used.

The reactor core is arranged of 163 fuel assemblies (FA). Each FA contains 311 fuel rods, 18 guiding tubes for the absorbing elements, a central tube and a channel for placing the ICID primary transducers. A fuel rod is a zirconium-alloy tube with the end pieces filled with the pellets of uranium dioxide of various initial enrichment. The average fuel enrichment for the first loading is 1.72 wt. % which gives maximum cost reduction of the initial core loading. The length of the first fuel cycle is 309.3 effective days (7423 h). The CPS CR clusters in the number of 121 can be inserted into the core either for the purpose of control, or for using in the reactor scram system for the reactor trip. The pitch electromagnetic drives with position indicators are used as the driving mechanisms for CPS CR clusters. The CPS CR drives are installed on the reactor top head.

The number of fresh FAs, to be loaded annually in each equilibrium cycle, is 30 pieces. The length of equilibrium fuel cycle is 296.7 effective days.

Reactor core design can be changed in order to meet the requirements for the reactor cores of new generation.

2.3 Nuclear fuel handling systems

The fuel handling and transfer systems are intended to load the fresh FAs and to replace the spent FAs and absorbing rods. Nuclear fuel reloading and spent fuel cooling in the spent fuel pool are performed in the reactor compartment, whereas the cooled spent nuclear fuel, before its off-site shipping to the regeneration factory, is temporarily stored in the spent fuel storehouse. Transfer of fresh nuclear fuel into the reactor compartment, in-pile refuelling, transfer of the spent fuel from the reactor compartment are performed during preventive maintenance (PM). The fuel handling procedures are performed under water layer to provide radiation protection. To perform fuel handling procedures the reactor shall be shut down and depressurized, the reactor top head shall be removed, the protective tube unit shall be withdrawn from the reactor vessel, and the reactor concrete cavity, as well as the spent fuel pool, shall be filled with water.

The fuel handling and transfer systems include the following compartments with the necessary equipment: reactor concrete cavity, spent fuel pool, transfer compartment. The spent fuel pool (SFP) is arranged near the core barrel and connected to it by the transition channel intended for transfer of one FA at a time. The SFP is equipped with the structures for storage of spent FAs which consist of separate sections for storage of FAs and of sealed containers for storage of failed FAs. The transition compartment is equipped with a universal cell for placing the in-site transport packing set (ITPS) for fresh FAs and the transport set for spent and cooled FAs. The transition compartment is connected with SFP by means of the transition channel through which the fresh FAs are loaded. The fresh FAs are withdrawn from ITPS by the fuel handling machine and installed into the core according to loading cartogram.

After withdrawal from the core the spent FA is placed into a bottle of the defective assembly detection system (DADS) and then transferred to the storage facilities or into the sealed container for the spent and cooled fuel in SFP, depending on the results of inspection for presence of defects. Procedures on replacement of CPS CR clusters are the same as the fuel handling procedures: the CPS CR clusters withdrawn from the core are placed into the empty cells for FAs or into the storage facilities in SFP. The fuel handling machine performs fuel handling procedures only with one FA at a time, with one CPS CR or one FA with CPS CR bundle placed in it. Heat removal from the reloaded fuel assemblies is performed by the SFP cooling system.

2.4 Description of the reactor coolant system components

2.4.1 Reactor vessel

The schematic drawing of the reactor assembly is shown in Figure 2.2. The reactor vessel design is similar to that of the standard VVER-1000 reactor.

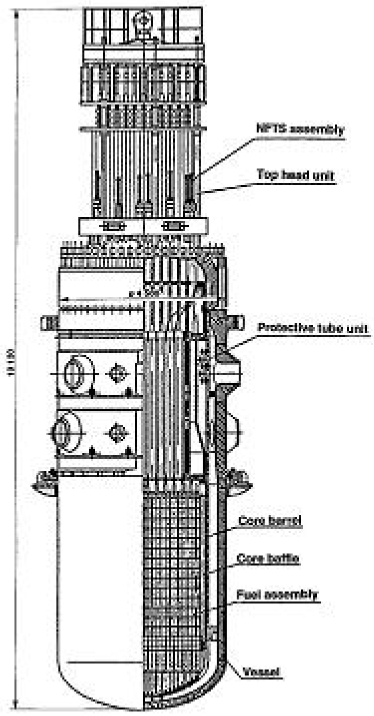


Figure 2.2 Reactor VVER-640

The reactor vessel has a flange, upper and lower nozzles zones, supporting shell, cylindrical shell and elliptical bottom welded together with circumferential welds. On the reactor vessel there are two rows of nozzles with internal diameter 620 mm (four in each row); at a level of each row there are two nozzles available for joining the ECCS hydro-accumulators, besides, in the upper row there is a nozzle for instrumentation penetrations. Internal surfaces of the reactor vessel and nozzles have corrosion-resistant coating. The separating ring is welded to the reactor vessel on the inside for division of the inlet and outlet coolant flow. On the reactor vessel supporting shell a supporting ring is installed for attaching the reactor vessel to the supporting structure. The reactor top head comprises a truncated ellipsoid and a flange joined with circumferential weld. On the reactor top head ellipsoid the penetrations are provided for in-core instrumentation detectors, as well as nozzles for gas removal lines, and the holes for CPS CR housings.

2.4.2 Reactor internals

Inside the reactor vessel there are structures intended for the core and CPS CRs fastening and proper installation. The reactor internals include the core barrel that houses the core baffle and the protective tube unit. The core barrel provides coolant flow separation and arrangement in the reactor through the core FAs installed into the core baffle and spaced in plan by the protective tube unit. The protective tube unit, in its turn, ensures the protection of the CPS CRs against coolant flow effect, as well as arrangement of ICID outlets.

2.4.3 Steam generators

The steam generator (SG) is a horizontal one-vessel heat exchanging apparatus with a bundle of horizontal heat exchanging tubes, submerged into water volume, and the internals. The internals provide distribution of the main and emergency feedwater, washout of deposits from the vessel bottom surface as well as equalization of steam loading on evaporation mirror. This SG is a modernized version of the standard steam generator of PGV-1000 type. When developing the steam generator for V-407 RP the positive operating experience of the steam generators at the operating plants with VVER-1000 and VVER-440 was considered. In particular, the bottom and perforated parts of the primary-side collector are made of stainless steel 0X18H10T-BД proved to be good during operation of SG collectors at VVER-440 plants. To provide for a possibility of examination of SG internals the manholes 500 mm in diameter are arranged on the elliptical bottom.

2.4.4 Pressurizer

For the V-407 reactor plant the PRZ will be the same as the PRZ used for standard reactors VVER-1000. The pressurizer, together with the primary pressure control system, is intended for maintaining the primary pressure within the assigned limits under all design basis conditions of the reactor plant. The PRZ proper is a vertical vessel installed on the cylindrical supporting structure. At the upper end there are nozzles for connection of injection lines and the lines for steam dumping through PORV. On the PRZ bottom there is a nozzle for PRZ connection to the hot leg. The nozzles for level gauges, and the submerged jacket for the primary transducers for coolant temperature measurement inside the vessel, are arranged on the vessel cylindrical part. Inside the PRZ there are injection device and tubular electric heaters. The injection device is intended for water injection into steam volume and for steam condensation. The injection device represents the exhaust manifold attached on the top of PRZ vessel. The PRZ internal surface is coated with resistant-resistant material. All PRZ internals are made of stainless steel of austenitic grade.

2.4.5 Reactor coolant pumps

The RCP sets are of new design. When developing RCP set, the R&D results were considered with long-term inspection of service life of the main units. The RCP set is a modernized pump whose prototype is successfully operated at standard reactor plants VVER-1000/V-320. It is a vertical one-stage centrifugal pump with independent lubrication system in a spherical casing. Proceeding from the specific features of heat removal from the core of V-407 reactor plant the pump flow path is made axial that provided in the design of the pumps used at NPPs with VVER-type reactors. The model flow part of such pump is fabricated and tested at the test facility. Besides, in the given pump a necessity of water supply from special systems for lubrication of radial-thrust bearings is excluded due to change in their positions. The RCP set subsystems prevent the radioactive coolant leak outside the reactor coolant system boundary. The asynchronous, three-phase electric motor of vertical type is applied for the RCP set. Nonflammable lubrication is used in the electric motor. The shaft seals are designed such as to prevent coolant leak in case of loss of power during 24 h and in case of loss of sealing water and other cooling media.

2.4.6 Pipelines of reactor coolant system

The main coolant pipelines (MCP) are a part of four circulation loops being identical in pairs. The MCP of each loop consists of straight pipes and sharply bent elbows in places of MCP connection to steam generators and pumps. The MCPs (hot and cold legs) have internal diameter of 620 mm.

2.5 Reactor auxiliary systems

The reactor auxiliary systems, connected to the primary circuit, include the chemical and volume control system and the fuel cooling system. The chemical and volume control system, alongside with the classical solution on providing the coolant inlet-outlet for maintaining the required boric acid concentration in the primary circuit and PRZ water level as well as coolant purification in special water purification (SWT) system, is intended for the primary coolant degassing in the relief tank-degasser by using the losses of PRZ blowoff energy that makes a positive effect on the Unit profitability. The fuel cooling system fulfills the function of the residual heat removal from the primary circuit under the conditions of refuelling and of heat from the refuelling pool under normal operation and in case of the primary circuit depressurization in case of accident.

2.6 Operating conditions

The requirements for a wide spectrum of conditions are provided in the design, including the station modes both under normal operation with technical-and-economic indices assigned in the design, and for anticipated operational occurrences with reliability and safety indices justified in the design. The main operating mode for the Unit equipment operation is accepted to be the base mode with annual refuelling and PM. The Unit equipment is developed with account for electric power production in the daily load-follow mode. Evaluations of the core and Unit systems operation showed sufficiency of technical solutions accepted in the design for implementation of the load-follow mode enabling to use the Unit for a wide spectrum of power grid operating modes.

Reliability of the equipment is assured by the corresponding calculations and meeting the design requirements for manufacturing, mounting and operation of the equipment, systems and NPP, as a whole.

2.7 Standard fuel cycle

The fuel cycle of VVER-640 reactor is arranged in the following way:

- the fuel cycle of VVER-640 reactor core, as given in the design, has the refuelling factor of 5.43 that corresponds to annual loading of 30 FAs, of them 13 FAs are irradiated for six years, and 17 FAs - for five years. Provision is made for seven transient loadings enabling to reach the equilibrium fuel cycle by the ninth year of reactor operation. Starting with the ninth fuel cycle the neutronics remain unchanged. The FAs with average fuel enrichment of 1.72 % shall be loaded into the core during the first cycle. The length of the first fuel cycle is 309.3 effective days.
- the refuelling patterns provide a small neutron leakage by placing the FAs of the fifth and sixth years of irradiation over the core periphery. The equilibrium refuelling mode is started with the seventh loading, i.e. the seventh and the subsequent refuellings are identical. During each equilibrium refuelling 30 fresh FAs are to be loaded.
- length of equilibrium cycle is 296.7 days, average burnup fraction of the unloaded fuel is 41.4 MWD/kgU. Maximum linear heat rate in fuel rods does not exceed 265 W/cm;
- storage and handling of VVER-640 FAs at NPP are similar to storage and handling of VVER-1000 FAs. Spent FAs are cooled in SFP for three years at least. The spent FAs, to be shipped from NPP, shall be transferred by the fuel handling machine from SFP racks into transport container intended for storage of the mentioned fuel at NPP and its further transportation to the fuel regeneration factory.

2.8 Alternative fuel

Low power intensity in the core of VVER-640 enables to arrange 100 % of its loading with MOX-fuel.

Specific features of VVER-640 fuel cycle with 100 % MOX-fuel core:

- equilibrium fuel cycles are developed on the basis of VVER-1000 FA design which is applied to VVER-640 also;
- MOX-fuel component in the pellets is 100 %. Here the principle of direct replacement of uranium-oxide fuel with MOX-fuel only is implemented with FA design unchanged;
- burnable absorber in the form of fuel elements containing gadolinium with uranium is available in all MOX loadings for suppression of excessive main reactivity and for power field flattening;
- when refuelling in the equilibrium fuel cycle, 13 fuel assemblies with MOX-fuel are loaded for operation during six-year fuel cycle, and 17 fuel assemblies with MOX-fuel are loaded for operation during the five-year fuel cycle. Thus, refuelling factor for the core with MOX-fuel is 5.43, the same as for the cores with uranium-oxide fuel. In total, 30 assemblies are to be changed in the course of refuelling;
- average burnup of MOX assemblies is 40 MWD/kg.

Analysis of possible usage of MOX-fuel in the reactor plant showed the necessity of upgrading a number of equipment and systems because fresh MOX-fuel is much more active than fresh uranium-oxide fuel.

3.1 Safety concept, basic design principles

The design solutions on safety for NPP with VVER-640 reactors are aimed at development of NPP with the enhanced safety level so that to minimize the general risk related to station operation to reasonably achievable level. Development of the safety concept is based on the detailed analysis of operating experience and PSA results for the operating NPPs with VVER-type reactors and on the requirements of Russian standards as well as on the recommendations of IAEA safety documents. The fundamental principle of the safety concept for VVER-640 plants is the defense-in-depth principle based on application of the system of barriers to the release of ionizing radiation and radioactive substances to the environment and of the system of engineering-and-organizational measures on protection of the barriers and keeping their efficiency as well as on direct population protection. The main deterministic principles included into the safety concept are:

- single failure principle;
- diversity;

- physical separation;
- protection against common-cause failure;
- protection against operator's errors;
- application of passive safety system;
- protection against influence of internal and external hazards;
- assurance of improved reliability level for fulfilling the safety functions;
- decrease in frequency of catastrophic failure to negligible level.

The main principle of creating a new Unit is the principle of maintaining the evolutionary solutions confirmed by positive operating experience of VVER plants, providing safety enhancement VVER-640 Unit, that is assured by upgrading the components of the reactor plant and NPP systems and equipment concerning the following basic trends:

- decrease in core power density;
- reliability improvement of the primary and secondary boundaries;
- reliability improvement in decay heat removal;
- reliability improvement in power supply;
- improvement of RP protection under accidents of anticipated transient without scram (ATWS) type;
- improvement of RP and NPP protection against external hazards;
- decrease in radioactive releases.

According to the Russian safety standards the target values of provability indices are limited by the following values: the rate of core damage shall not exceed 1.0E-5 reactor/year and frequency of high emergency release shall be less than 1.0E-7 reactor/ year.

Proceeding from the mentioned safety concept the following technical solutions have been accepted in V-407 plant design:

- advanced emergency protection system with the number of CPS CRs two times more than in V-320 reactor plant that is capable to maintain the reactor under subcritical state to the temperature of 100°C ;
- overcoming of all design basis accidents using passive systems only; with this, the active safety system, when actuated, is used to minimize and mitigate the accident scenario;
- active and passive systems for heat removal through the secondary circuit shall remove heat as long as possible even in case of loss of both normal, and emergency power supply; the special system of emergency feedwater supply into steam generators for heat removal from the core through the secondary circuit is not provided;
- some channels of the active safety systems can be used for fulfilling the functions of normal operation, with this, most components of these systems are in the state similar to that when the required emergency functions shall be fulfilled. Use of such modes of the mentioned systems enables to raise the indices of their readiness and to provide additional protection against common-cause failures;
- applying in the design of the principle of overcoming the accidents with the use of passive systems only, when no operator's actions are required for their actuation, as well as of active safety systems with a high level of automation used only for minimization and mitigation of accident scenario, will allow to provide the deeper protection against personnel errors;
- reducing the frequency of LB LOCAs and large primary-to-secondary leaks to negligible level owing to application of "leak before break" concept;
- corium retention under beyond design basis accident is provided in the reactor vessel without ex-core corium confining device on passive principles;
- in V-407 design the double containment is provided with the passive system of heat removal from the containment, the system of hydrogen removal, the system of ventilation, the system of decontamination of the volume between envelopes.

3.2 Description of defense-in-depth

According to the defense-in-depth principle the NPP is designed, constructed and operated in such a way that radioactive materials turn to be surrounded by a number of physical barriers. The system of physical barriers at VVER-640 plants includes:

- fuel matrix;

- fuel element claddings:
- boundary of the circuit coolant cooling the core;
- system of sealed enclosures.

The system of barriers, limiting the release of radioactive products and ionizing radiation, is arranged depending on location of the main source of radiation hazard, namely, FUEL ROD.

“A” - Fuel matrix

The barrier presents a pellet of the sizes (diameter and height) fixed within the limits of technological tolerances and the central design orifice. Mass fraction of chemical elements in the composition of a pellet as regards to uranium is limited by the requirements of the manufacturing documentation. Low-enriched uranium dioxide, applied now at VVER plants, is used as fuel for the fuel rods. Usage of uranium dioxide enables to limit the release of gaseous radioactive fission products from the fuel matrix.

“B” – Fuel rod claddings.

The barrier presents a tube with the end pieces made of zirconium alloy and filled with uranium dioxide pellets. Volume fraction of helium in cladding is provided by the manufacturing process. In the course of manufacturing the tubes are tested for their dimensions, chemical composition, mechanical properties, corrosion resistance. During NPP operation in base-loading mode the residence of fuel rods in the reactor is 5-6 years.

Internal gas pressure in fuel rods, determined by release of gaseous fission products from the fuel matrix for maximum burnup, does not exceed the primary coolant nominal pressure.

“C” – Boundary of coolant circulation circuit for core cooling.

The core-cooling coolant circuit (the primary circuit) is a circuit wherein the coolant circulates through the core. Besides, the pressurizing system is referred to the primary circuit.

The core-cooling coolant circuit boundary incorporates:

- vessel, top head, CPS drive housings and the reactor sealing units;
- SG collectors, collector sealing units, steam generator tubing;
- pressurizer vessel and sealing units;
- casings and sealing units of the reactor coolant pumps;
- bodies and sealing units of pipeline valves of various systems;
- pipelines of normal operation systems and safety systems.

Stainless steel is chosen to be the base material for the barrier, except the reactor vessel and top head, by the operating experience of the previous generation NPPs.

The primary circuit is a barrier during NPP power operation, startup and shutdown. When refuelling the primary circuit is unsealed and does not present a barrier on the way of spreading of ionizing radiation and radioactive substances.

“D” - System of sealed enclosures.

The sealed enclosure is a barrier wherein the RP main equipment and the spent fuel pool are arranged.

The sealed enclosure includes:

- internal protective steel envelope;
- isolation devices;
- locks and manholes;
- sealed penetrations.

The internal protective steel envelope presents a smooth cylindrical envelope with a hemispherical dome of low-alloyed steel with bottom embedded into mass reinforced concrete of internal pedestal. On the outside of the envelope cylindrical part a system of the closed channels-coolers filled with water is provided for fulfilling the functions of passive heat removal from the envelope.

To create the closed sealed volume, the steel envelope is connected over bottom to sealed facing of the floor.

The bottom part of the closed sealed volume, to be filled with water during ECCS operation in case of loss of integrity of "C" barrier, forms the emergency pool.

In the envelope cylindrical part a number of the following technical inlets to the closed sealed volume is provided:

- openings for locks (for cargo - 1 and for the personnel -2);
- openings for process penetrations;
- for the personnel for electrotechnical penetrations.

All the mentioned inlets are equipped with special sealed penetrations through the steel envelope and arranged above the emergency pool elevation.

Power-operated or air-operated quick-acting isolation valves, pressurized valves, as well as manual isolation valves (normally closed and locked up), are used as the isolation devices.

The locks of containment area in number of 2 pieces are intended to provide possible access of the attendants into the reactor compartment rooms of the containment during refuelling. In this case one of the locks is working (main), and the other is the reserve (emergency).

The lock vestibule is intended for delivery (off-site shipping) of the equipment and the materials required for preventive maintenance during reactor shutdown into (from) the central hall of the reactor compartment.

The manhole of refulling well is used for delivery (off-site shipping) of fuel to (from) the spent fuel pool and at the same time it is a component of NPP localizing safety system.

The system of engineering and organizational measures is provided for the efficient protection of VVER-640 barriers, the system consists of five levels of protection. Each "level" of NPP protection provides certain efficiency of barriers protection against the effects typical for the given level. The corresponding engineering and/or organizational measures on prevention and/or mitigation of consequences of the effects or sources of the effects are provided for each "level" at the expense of limitation of normal operation up to termination of NPP operation, and also on NPP return from the lower "level" of protection to the higher "level" of protection.

The first level (Prevention of anticipated operational occurrences).

Characteristics of the engineered features and the regulated actions of the personnel assure the integrity of barriers and NPP functioning within the operational limits stated in the design up to reaching the specified limits.

The second level (Prevention of design basis accidents by normal operation systems).

Characteristics of the engineered features, including those specially provided for the reactor shutdown, as well as the regulated actions of the personnel on observing the safe operation conditions, assure the efficiency of barriers owing to prevention of the effects within the limits of safe operation up to reaching the specified limits.

The third level (Prevention of beyond design basis accidents by safety systems).

Characteristics of the special engineered features and the regulated actions of the personnel assure the efficiency of barriers, or of the intact part of barriers, owing to prevention and mitigation of the effects in the frame of non-exceeding the maximum design limits established for all the accidents considered in the design. Effectiveness of the system of barriers is assured; however, service life of some barriers may be reduced up to impossibility of their further operation that governs a possibility of restoration of NPP normal operation.

The fourth level (Management of beyond design basis accidents).

Characteristics of the additional engineered features and the regulated actions of the personnel assure the effectiveness of the system of sealed enclosures, or the intact part of other barriers, by preventing and mitigating the effects so that they do not result in destruction or loss of properties of the sealed enclosure system as to limitation of the effects of radioactive products and ionizing radiation on the personnel, population and environment. Possible restoration of NPP normal operation is determined on the basis of separate analysis of consequences of such effects.

The plans of measures at NPP site and surroundings on protection and elimination of consequences of such effects make further contribution to protection of the population, personnel and environment.

The fifth level

Off-site emergency measures to mitigate consequences of radioactive products release into the environment.

For each NPP protection “level” a set of radiation criteria are provided for achieving the main safety objective, as well as the corresponding process criteria of efficiency for each barrier. The criteria are established on the basis of regulatory requirements and experience in development and operation of VVER plants.

3.3 Active and passive systems, inherent safety features

To provide the required degree of NPP safety, alongside with the reactor trip system – emergency protection, a complex of the special engineered safety features is provided in the design for fulfilling the safety functions, and the design itself and its systems make use of such “inherent safety” properties as self-control, heat capacity, heat conduction, heat transfer, gravitation, compressed gas energy and other natural processes.

Structure of the passive safety systems

System	Number of channels and capacity
Pressurized ECCS hydro-accumulator systems	4 x 50%
Hydrostatic tank system	4 x 50%
Primary circuit depressurization system	2 x 100%
System of passive heat removal from SG	4 x 50%

Structure of the active safety systems

System	Number of channels and capacity
Emergency makeup system	2 x 100%
Emergency gas removal system	1 x 100%
Primary circuit overpressure protection system	2 x 100%
Secondary circuit overpressure protection system	2 x 100%
Main steam isolation system (MSIV)	2 x 100%

The ECCS hydro-accumulator system is intended for cooling water supply into the core and maintaining the primary coolant inventory during LOCA when primary pressure decreases to P1c< 4.0 MPa.

The ECCS hydrostatic tank system is a four-channel system of the tanks under atmospheric pressure that is intended for the reactor core flooding for decay heat removal. The system actuates when the difference between the primary pressure and containment pressure becomes 0.29 MPa.

The primary circuit depressurization system is a circulation circuit connecting the MCP to the fuel pool and intended for the reactor pressure decrease for the purpose of connecting the ECCS tanks to the reactor and of subsequent transfer of the core to the stage of long-term cooling due to natural circulation through the spent fuel pool. The system channel includes depressurization valve units (DVU), the membrane valve for connection of the refuelling pool with the emergency pool, the repair fittings, screen filter and pipelines.

The SG PHRS system is intended for emergency heat removal from the steam generators in case of loss of all active systems of heat removal from the core, as well as for quick cooling down of the reactor plant under LOCAs. Besides, the PHRS, together with the containment heat removal system (CHRS), is intended for long-term removal of residual decay heat under accidents with complete loss of all power sources including diesel generators. The SG PHRS removes heat over the closed circuit: steam generator - steam path pipeline - submerged heat exchanger - condensate pipeline - steam generator. Operability of components of the system of passive heat removal from steam generators and containment is verified at full-scale models.

The emergency gas removal system is intended for removal of non-condensable gases released from the coolant in upper points of the primary circuit under accidents related to primary pressure decrease in order to prevent loss of natural circulation of the primary coolant. The system is made with redundancy of the valves being the active elements of the system.

The emergency makeup system starts to supply boric acid solution into the primary circuit when its pressure decreases below 3.0 MPa. This system has reliable power supply from the emergency diesel generators. The main safety function of the emergency makeup system is engineered safety feature for ATWS-type management. The system is also used for BDBA management. The second high-pressure stage of this system is used for ATWS-type management and actuated only in response to ATWS signals and other signals related only to BDBAs.

In VVER-640 plant no provision is made for a special secondary circuit emergency makeup system intended for feedwater supply into the steam generators under anticipated operational occurrences and design basis accidents to remove heat from the core through the secondary circuit. However, there is a number of systems fulfilling these functions and duplicating each other.

For these purposes the following systems can be used:

- *the auxiliary feedwater system* supplying water into the steam generators at the first stage of the accident with loss of power to the station auxiliaries. Water is supplied from the deaerator and then from tanks of the demineralized water system. The system is fed from the main diesel-generators. System redundancy is 2x100 %.

- *the system of planned cooling down and residual heat removal* is intended for RP planned cooling down under the steam conditions and then under the water-water conditions over the closed circuit: cooldown pump – steam generator – steam line – cooldown heat exchanger – cooldown pump. The system consists of two independent channels with redundancy 2x100 %. Each channel is provided with two high-pressure pumps and two low-pressure pumps. The high-pressure cooldown pump has the flow rate of 100 m^3/h at SG pressure 9.0 MPa. The low-pressure cooldown pump has the flow rate of 400 m^3/h at pressure 0.7 MPa. The system is reliably fed from the main diesel-generators. The rate of cooling down is regulated by the amount of cooling water supplied into the heat exchanger of this system. Besides, this system is used for residual heat removal in case of complete or partial failure of the feedwater channel or system. With this, any of the high-pressure system pumps is connected to the sections with the failed auxiliary feedwater pump. The water supplied by these pumps is taken from the PHRS tanks (emergency heat removal tanks).

It is worth adding to the above mentioned that the secondary circuit is provided with a complex of dumping devices used to fulfill the functions of pressure decrease and to minimize operation of the secondary circuit overpressure protection system. These devices include:

- 4 steam dump valves to the turbine condenser (BRU-Ks) – normal operation system;
- 2 "large" steam dump valve to the atmosphere (BRU-As) installed on the main steam header;
- 4 "small" BRU-As installed on steam lines of each SG;
- 2 PORVs per a steam generator of the secondary circuit overpressure protection system.

In fact, each of the dumping systems consistently fulfills the functions of secondary circuit overpressure unloading.

3.4 Safety systems for overcoming the consequences of design basis accidents.

1. Under blackout accidents (with loss of power supply to the station auxiliaries including the main and emergency diesel-generators), the residual heat removal and RP cooling down are provided by the SG passive heat removal system (SG PHRS) and the containment heat removal system (CHRS) during at least 24 hours taking into account a single failure principle.

2. Under LOCA, with superimposing of complete loss of NPP power, the boric acid solution is supplied into the reactor from the ECCS hydro-accumulator system, and further on, with decrease in pressure, from hydrostatic pressure tanks. Pressure of actuation of ECCS hydro-accumulator is 4.0 MPa. With further decrease in the primary pressure to 0.3MPa the core is spilled with water from the ECCS tanks. The leaking coolant is collected on the containment area floor in the emergency heat removal pool.

For the guaranteed spillage of ECCS tanks and assurance of natural circulation at the primary pressure decrease to 0.6 MPa the passive opening is made of the primary depressurization system valves connecting the reactor with the refuelling pool. As the core is filled with water from the ECCS hydro-accumulators and tanks the water level is created in the emergency pool above elevation of the reactor outlet nozzles, thereby providing natural circulation over the circuit: reactor core - depressurization valve above the core - refuelling pool - depressurization valve under the core - reactor core.

Heat removal from the refuelling pool is accomplished either by the fuel pool cooling system (FAK) with power supply to the station auxiliaries available, or due to heat transfer to the emergency pool. The coolant of emergency and fuel pools is partially vaporized with condensation on the walls and within the metal sealed envelope owing to operation of CHRS, and then it returns into the emergency pool. Makeup of the refuelling pool is accomplished from the emergency pool through the valve connecting the refuelling and emergency pools.

Heat is removed from the containment through the wall to the CHRS water due to natural circulation between the emergency heat removal tank and the containment cooling channels. Water inventory in the tank is designed for heat removal during at least 24 hours taking into account a single failure principle.

After passing of 24 hours the decay heat is removed by normal operation systems important to safety and having the reliable power supply from the main diesel-generators. Loss of electric power for more than 24 hours is considered as the accident when the personnel shall take an active part in its elimination performing additional actions such as, for example, connection of additional sections of the PHRS tanks (that provides heat removal for about 72 hours), or shall arrange water supply to compensate the evaporated part (for example, from the fire prevention water supply system).

3.5 Safety systems for overcoming the beyond design basis accidents.

In the design of VVER-640 plant there is a number of systems fulfilling the functions of engineered safety features for management of beyond design basis accidents. With this, this role can be played both by the safety systems, and by normal operation systems important to safety. These systems incorporate the passive heat removal system, ECCS tank system, active emergency makeup system, emergency gas removal system and the system of planned cooling down through the secondary circuit. All the mentioned active systems can fulfill these functions with the power supply available from diesel-generators, and the emergency gas removal system can fulfill the functions with power supply available from storage batteries. Besides, the means for management of beyond design basis accidents include also the relief valve of the primary circuit overpressure protection system. Practically all these systems (except for ECCS tanks) are used for forced decrease in the primary pressure in order to connect this or that system, with subsequent fulfilling of functions on heat removal to the ultimate sink and heat removal from the reactor vessel on the outside.

Besides, the emergency makeup system, in the part of high-head stage of the pump set, fulfills the function of ATWS-type accident management supplying boric acid solution with concentration at least 8 g/kg into the primary circuit within the pressure range from 0 to 20 MPa.

To cope with the accident caused by loss of power supply, both normal and emergency (common-cause failure of the main diesel-generators), and resulted in failure of two secondary circuit systems: planned cooling down and residual heat removal system and the auxiliary feedwater system, the SG PHRS is actuated.

Unlike VVER-440 and VVER-1000 plants the design of VVER-640 plant makes provision for the system enabling to manage severe beyond design basis accidents related to core melting. This system includes the mentioned systems of ECCS tanks, the primary circuit depressurization system, CHRS and emergency pool volume whose joint actions (without any sources of power supply, including emergency sources) enable to create the reactor vessel cooling circuit on the outside and to provide in-vessel corium retention (even when the whole core is melted).

3.6 Safety assurance during earthquakes

Special requirements for NPP seismic stability are defined in "Standards", /3/, whose requirements shall be considered in the NPP design by all means.

According to the "Standards", to be met unconditionally, the presented NPP is seismically stable because its safety is assured according to /2/ under seismic impacts up to SSE, inclusive, and power and heat production is assured under earthquake up to OBE inclusive.

The reactor, steam generator, RCP set, MCP and pressurizing system including PRZ, valves and pipelines are referred to the equipment of seismic category 1 according to the "Standards" and can be applied at NPPs constructed on the sites with seismicity (SSE) to magnitude 8 as per MSK-64 scale, with this:

- under earthquake with the intensity up to magnitude 7 (0.14 g) inclusive (design), normal operation of the equipment without shutdown is ensured;
- under earthquake with the intensity of magnitude from 7 to 8 inclusive, the automatic trip and cooling down are ensured.

The RCP set motor is referred to seismic category II.

The CPS-CSS equipment, participating in generation of emergency commands, is made seismically stable, referred to seismic category I and provides fulfillment of its functions under seismic impacts with the intensity up to magnitude 8.

Other equipment is referred to seismic category II and provides fulfillment of its functions under seismic impacts with the intensity up to magnitude 7.

Reactor, PRZ, ECCS tanks and accumulators are fastened steadily on their supports so that not to prevent thermal vertical displacements.

The reactor vessel is installed by its supporting shoulder and fixed on the supporting ring fastened on the supporting truss in the reactor concrete cavity. For additional reactor fastening against lateral displacement under seismic impacts a supporting ring is installed on the reactor vessel flange on the outside.

For fastening the steam generator the supporting structures, snubbers and embedded components are used. The supporting structure includes a supporting bed, two-tier roller support and a supporting base. The supporting structure provides for a possibility of SG displacement at thermal expansions of pipelines in longitudinal and lateral directions. To take up the seismic loads acting on SG, a system of snubbers is provided, a part of them are arranged in the region of supporting structures, and another part is arranged above the SG vessel axis between the RCP set box wall and the SG vessel.

Pump set GCNA-1455 with the help of load-bearing spacer, connecting the pump casing to the motor, rests upon three spherical bearings providing displacement of the pump set during thermal expansions of RCS pipelines. The RCP set is additionally fastened against seismic impacts by means of viscoelastic snubbers in the region of the supporting structures and motor.

The CKTI viscoelastic snubbers with high damping properties are supposed to use as the main means for assurance of equipment and pipelines seismic stability, the snubbers can be controlled and optimized. Providing protection against seismic, impact and other emergency dynamic loads, the CKTI snubbers enable to reduce vibration of the equipment and pipelines several times under NOC.

The refuelling system equipment is designed so that to withstand the normal operation loads and remains operable after seismic impacts because this equipment is referred to seismic category 1.

Under seismic impacts the cranes of FFS, reactor compartment, SFS, as well as the fuel handling machine, are disconnected from the station seismic sensor reliably keeping the transported cargo. Skidding movements of the bridge, crane trolley and fuel handling machine are excluded.

All the equipment of the SRW handling system, as the components of the system important to safety, is considered by the seismic categories according to the “Standards”.

As to the auxiliary structures, referring to seismic categories 2 and 3, the seismic load of W=0.05g or magnitude 6 by MSK-64 scale is accepted in the present design proceeding from the fact that they are greatly subject to changes depending on specific-plant conditions.

With a view of expanding the geography of possible NPP sites for areas with SSE intensity up to magnitude 10 by MSK-64 scale (roughly), the work has been done within the frame of this NPP to determine the expediency and possibility of making seismic isolation of the structure of category 1 (using the example of the reactor building). As a result of this work, a principal possibility for development of the multicomponent low-frequency system of seismic isolation based on application of the serial domestic-made supporting shock-absorbers with stable dynamic and operational characteristics between two foundation plates is demonstrated by the analysis of loading conditions of the reactor building by seismic waves of intensive earthquake (magnitude 9-10). The proposed seismic isolation devices (supporting shock-absorbers) do not require continuous control and adjustment.

Proliferation resistance

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

Safety and security (physical protection)

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

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

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

The engineered physical security features of the complex include:

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

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

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

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

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

Description of turbine-generator systems

6.1 Description of the turbine generator

The turbine set of LMZ design is delivered in a set with the turbine generator of "Electrosila" design.

Turbine set K-640-6.9/50 is a single-shaft set with one high and medium pressure cylinder (HMPC) and two double-flow low-pressure cylinder (LPCs). The structural system: HMPC + 2LPCs + generator. Low-pressure cylinders are unified with LPC of turbine K-1000-60/3000 being in operation now.

The steam condensation turbine set of K-640-6.9/50 type without controlled steam bleeding, with intermediate separation and single-stage steam superheating is intended for direct driving the a.c generator of T3B type, and also for heat output in the amount to 250 GCal/h for demands of district heating of the industrial and social consumers at temperature range of 150/70 °C.

The schematic thermal circuit arrangement includes four low-pressure heaters (LPH), deaerator of increased pressure (1.3 MPa abs.), operating at sliding pressure under the transients, and two high-pressure heaters (HPH).

The intermediate steam separation and superheating system is implemented in the turbine set. Unlike the system with single-stage steam separation and intermediate superheating downstream of HPC in the separator-reheater previously used in the turbine K-1000-60/3000, here the two-stage (downstream of HPC and downstream of MPC) steam separation and single-stage steam superheating downstream of MPC using main steam (S+S+RH; separation+ separation+ reheating).

Generator T3B-630-2У3 is completely cooled with water and does not require usage of hydrogen. Power supply to the station-service consumers is made with d.c. voltage of 6 kV, 0.4 kV and 220 V.

Main performances of the turbine set are given in the Appendix.

6.2 Feedwater system

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

- deaerator;
- three feedwater pumps;
- two auxiliary feedwater pumps;
- feedwater heaters;
- feedwater lines with isolation and control valves.

Feedwater pump capacity is 1470 m^3/h, pressure head is 900 m of water column, auxiliary feedwater pump capacity is 250 m^3//h, pressure head is 830 m of water column. Feedwater temperature at the SG inlet during power operation is 230 - 180 ^oC and 164 ^oC under HPH switching off.

Electrical and I&C systems

7.1 Main electric system

The VVER-640 plant design makes use of the synchronous turbine generator of T3B-630-2 type with rated power 645 MW, voltage 20 kV, rotational speed 3000 rpm which is completely cooled with water, the internal space is filled with air at pressure close to atmospheric pressure. It is possible to use fire-resistant fluid OMTI in the bearing lubrication system.

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

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

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

7.2 Station-service system

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

- normal operation system;
- system of reliable power supply to systems of normal operation important to safety including two main diesel-electric sets of 4 MW each and storage batteries;
- emergency power supply system including two emergency diesel plants of 1 MW each and storage batteries. Storage battery capacities are chosen such that under complete loss of power of the Unit the safe state is ensured during 24 hours.

7.3 Emergency protection system and other safety systems

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

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

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

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

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

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

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

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

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

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

7.4 Layout of the main control room

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

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

The NPP control room concept provides:

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

The MCR provides the following areas of control:

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

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

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

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

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

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

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

Spent fuel and waste management

Handling of spent nuclear fuel at NPP provides:

- unloading of spent FAs from the reactor into SFP;
- check of leak-tightness of fuel rod claddings in spent FAs;
- holding of spent FAs in SFP;
- loading of spent FAs into the transport container cask;
- temporary storage of spent fuel in the transport container in SFS rooms;
- biological shielding of maintenance personnel at all stages of refueling, fuel transfer and storage.

When refueling, the claddings of fuel rods of the FAs unloaded from the reactor is checked for integrity, if required. The fuel handling machines place them one-by-one into two DADS bottles being components of sampling part of this system. Depending on the results of checking the defective FAs, withdrawn from DADS bottle, are placed into the leak-tight bottle arranged in the SFP racks, and FAs without defects are placed either into the SFP rack cells, or again into the reactor.

Storage (cooling) of spent nuclear fuel in the reactor compartment is provided in the refuelling pool during three years at least, with this, the capacity of the refuelling pool racks enables to provide storage (cooling) of spent nuclear fuel during five years with additional space for the case of emergency reactor unloading. The spent fuel, cooled in SFP for at least three years, is loaded into the transport container for transfer to SFS. Temporary storage of the cooled spent nuclear fuel, before its off-site shipping to a regeneration factory, is provided in the SFS rooms that presents a intermediate storage of container type. At Customer’s request, the spent fuel can be stored till NPP decommissioning. The SFS capacity enables keeping six transport containers at a time, each containing 12 FAs.

8.1 Waste handling

Waste handling provides for reprocessing of liquid and solid radwaste and their recycling.

In VVER-640 design a number of technical solutions is made for minimization of liquid radwaste (LRW) and reduction of their salt content:

- separate collection of LRW depending on their activities with re-using of low-active media;
- maximum application of reagent-free technologies;
- application of low-waste decontamination methods;
- giving up the regeneration of low-salt water clean-up filters;
- use of purified circuit water only for primary circuit makeup.

Providing observance of the operational specifications by the maintenance personnel of NPP the calculated amount of sump water is 8000 m3/year for one Unit, and not more than 10 % of this amount has salt content of 1.5 g/L and more.

Technical solutions on the LRW collection and reprocessing systems are made such that owing to arrangement of water closed circuits with re-using of purified flows in NPP cycle, the LWR releases outside NPP boundaries are eliminated. An important aspect is practically complete elimination of oil from the reactor compartment systems.

During NPP operation the solid radwaste (SRW) are formed:

- during Unit operation at power - from the process systems for reprocessing and decontamination of liquid and gaseous radioactive media (solidified salt concentrates, dried ion-exchange resins and sorbents, spent filters of special ventilation, etc.);
- during repair work (process equipment, instrumentation, tools, overalls, etc.);
- when eliminating accident consequences.

According to the concept of new generation NPP with VVER-640, accepted and approved in the established order, the radwaste of the 1st and 2nd groups are to be stored on NPP territory during five years before their off-site shipping to regional storage. Radwaste of the 3rd group are to be held in the on-site storage throughout the NPP service life and are to be shipped from the territory during NPP decommissioning.

The design makes provision for the system ensuring collection, grading, conditioning, transfer and temporary on-site storage of SRW.

The system of SRW collection, transportation, conditioning and storage is used as required during NPP operation at power in the rooms that are accessible for the personnel during maintenance or current repair of the equipment, and also during refuelling shutdown and preventive maintenance.

The system is not used under accidents and in the post-accident period it is used for collection, classification and conditioning of the SRW formed in the course of accident.

The layout solutions and operating characteristics of the SRW collection, transportation, conditioning and storage system are based on ensuring the normal radiation situation both in the SRW conditioning and storage room, and at the NPP site, proceeding from the following initial principles and provisions:

- provision for the required conditions of normal process scenarios as stipulated in the design;
- possible maintenance and repair of "dirty" equipment with low dose commitments;
- minimization of service lines;
- storage of solid radwaste, packed beforehand in closed bottles (metal barrels, capsules, concrete containers), in the reinforced-concrete storage compartments;
- SRW transportation within the NPP site using specially equipped trucks and electrocars;
- SRW preparation for reprocessing using special-purpose equipment in the specially equipped room.

All handling operations related to SRW collection and transportation are performed with the help of standard and returnable lifting devices.

The accepted methods of radwaste storage provide for a possibility of SRW withdrawal from the storage compartments and their off-site shipping to regional disposal area.

Considering SRW reprocessing from one NPP unit annually, their pure amount is:

- radwaste of the 1st group: **46 m^3**
- radwaste of the 2nd group: **14 m^3**
- radwaste of the 3rd group: **0.5 m^3**

Besides, there are solidified radwaste (from liquid radwaste) to be packaged in the concrete containers with biological shielding to ensure fulfillment of radiation safety criteria.

For NPP with two Units the total number of containers will be 88 containers annually (42 and 46 containers for low-level and medium-level radwaste, respectively).

Collection and classification of SRW of the 1st and 2nd groups are provided on places of their formation with their loading into the respective reusable containers, collecting containers or single-use tanks (paper or plastic bags).

Collection of SRW of the 3th group ("IC" and "NTMC" indicators and lines, etc.) is provided during Unit shutdown for PM using a special-purpose equipment.

Transfer of SRW into the conditioning room to be used at the final stage for arrangement of the waste packages and placing for temporary storage is provided with the help of the standard lifting devices and special transport means. Final SRW reprocessing is provided in the regional storage.

During NPP operation, the SRW collection, transportation, conditioning and storage system is used in the following sequence:

- during PM the SRW, formed at the place of work performance, are collected into the single-use containers and/or packages;
- SRW, collected into the single-use containers and/or packages, are transferred to the SRW storage room using the specially equipped vehicle (motor vehicles, electrocars);
- after SRW are delivered into the SRW storage room the following procedures shall be performed depending

on the group of activity:

   1. radwaste of the 3rd group of activity are encapsulated and placed for storage into the respective cell of the storage compartment;
   2. SRW of the 1st and 2nd groups of activity are transferred for final classification, then pressing or grinding are performed and packing into metal barrels to be placed into the stands and put for storage in the respective storage compartments;
   3. solidified LRW of the 1st and 2nd groups in protective reinforced-concrete containers are at once put for storage into the respective compartments;

- when off-site shipping of SRW to the regional storage, the radwaste shall be withdrawn from the storage compartments, placed into transport containers of the regional storage, loaded into special vehicles of the regional storage and removed from NPP site.

Plant layout

NPP Unit with VVER-640 reactor consists of the reactor, turbine and auxiliary buildings, the building of control and electrical systems, station auxiliary buildings.

The buffer are for the given design is within the site fence. According to the requirements of regulatory documents the radii of the radiation control area and the area of planning the emergency measures are defined to be 30 km and 1.5 km, respectively.

The reactor compartment presents itself a cylindrical building with a hemispherical dome. It has double protective envelope – internal sealed envelope made of metal and external protective envelope made of reinforced concrete. The lower elevation of reactor building is minus 4.500, elevation of the sealed envelope dome is 61.6 m, elevation of the protective envelope dome is 64.15 m.

The sealed area of reactor compartment is located from elevation 10.500 (including reactor cavity from elevation 0.000) up to elevation 61.6 of the sealed envelope dome. From elevation +16.000 to elevation +10.500 the sealed envelope presents a boundary of emergency heat removal and is fastened to reinforced-concrete civil structures on the outside. Below elevation +10.500 the sealed envelope in the form of cylindrical vessel with spherical bottom of 12.0 m diameter passes through the reinforced concrete mass surrounding the reactor vessel.

The whole volume inside the sealed envelope presents itself the interconnected air space separated by walls and floors, partially of lattice structure, i.e. assembled from beams (above steam generators at elevation +23.500 and +28.000), into the separate boxes wherein all RP equipment is arranged.

The reactor is installed in the concrete cavity, located from elevation 0.000 to elevation 28.000, and fastened to it at the level of SG box floor at elevation 10.500.

All RP main equipment, except for a polar crane, is installed and fastened in the civil structures at elevations from +10.500 to +28.000.

At elevation +10.500 within EHRP, alongside with MCP and ECCS pipelines, there are the primary depressurization system pipelines and valves connecting MCP with SFP and the SFP with EHRP.

At elevation +16.000 the steam generators and RCP sets are installed, with this, RCP sets are separated from SG by walls. In SG box, above the protected corridor connecting staircases with the RCP set box, and under the floor at elevation +23.500, the relief tank-degasser of the pressurizing system is fastened on beams.

Elevation +23.500 is occupied mainly with the recirculation ventilation center, ECCS valves and PHRS. Safety valves of the secondary circuit and the PRZ are located here as well. Neutron flux measuring channels and cable connections from the reactor are led through the concrete cavity wall to a separate room adjacent to the reactor concrete cavity.

At elevation +28.000 there are ECCS hydro-accumulators and tanks, protection container for transfer of the reactor internals, fuel handling machine, upper unit support, opening for CPS drives stand.

The low-pressure ECCS tanks are under pressure of the sealed envelope medium and owing to its axial arrangement provides for hydrostatic pressure head of at least 15 m sufficient for their reliable actuation after high-pressure tanks under loss of primary circuit integrity.

The ECCS accumulators and tanks are connected on the above by the crossovers on which the long equipment (extension shafts, wrenches, grips, etc.) can be hanged.

At elevation +28.000 the core barrel and PTU inspection wells are located. One of the wells has the bottom within the floor at elevation +23.500 and is intended for PTU installation during short-term refuelling. The core barrel (but only together with the protective container) can be also installed into this well. The second well is combined with the refuelling well into which both the core barrel and the PTU can be installed.

In the reactor building two smoke-proof staircases are provided. The main staircase is equipped with the lift and provide for an access from elevation +16.000 to elevation +37.000. The auxiliary staircase provides an access from elevation +16.000 to elevation +17.500. Transport lock for the equipment is located at elevation +28.000.

Near the pressurizer box and the main staircase (with the lift) there is a lean-to wherein the inspection room, the box of injection valves, the box of pilot-operated relief valves and offloading devices are arranged by floors.

In under-reactor space of concrete cavity, above elevation 0.000, the biological shielding and channels for the reactor vessel cooling medium inlet and outlet are arranged. The concrete cavity plays a role of a sump of the sealed envelope, therefore there are no problems of loss and ingress of water into lower rooms.

The polar crane of load carrying capacity 4000 kN rests upon the cylindrical concrete wall distant from the metal shell by 1.5 m and forming the corridor near the shell that enables to inspect the containment state.

Delivery of fresh fuel and removal of spent fuel are provided through the bottom of the receiving well arranged near the refuelling pool at elevation +10,500 and equipped with leak-tight manhole. The manhole design enables hermetic attachment to transport container through the bellows device without removing the container from the trolley at elevation 0.000.

Fuel transfer outside RP boundaries is provided at elevation 0.000 through the transport corridor.

The reactor plant layout is shown in Figure 9.1.

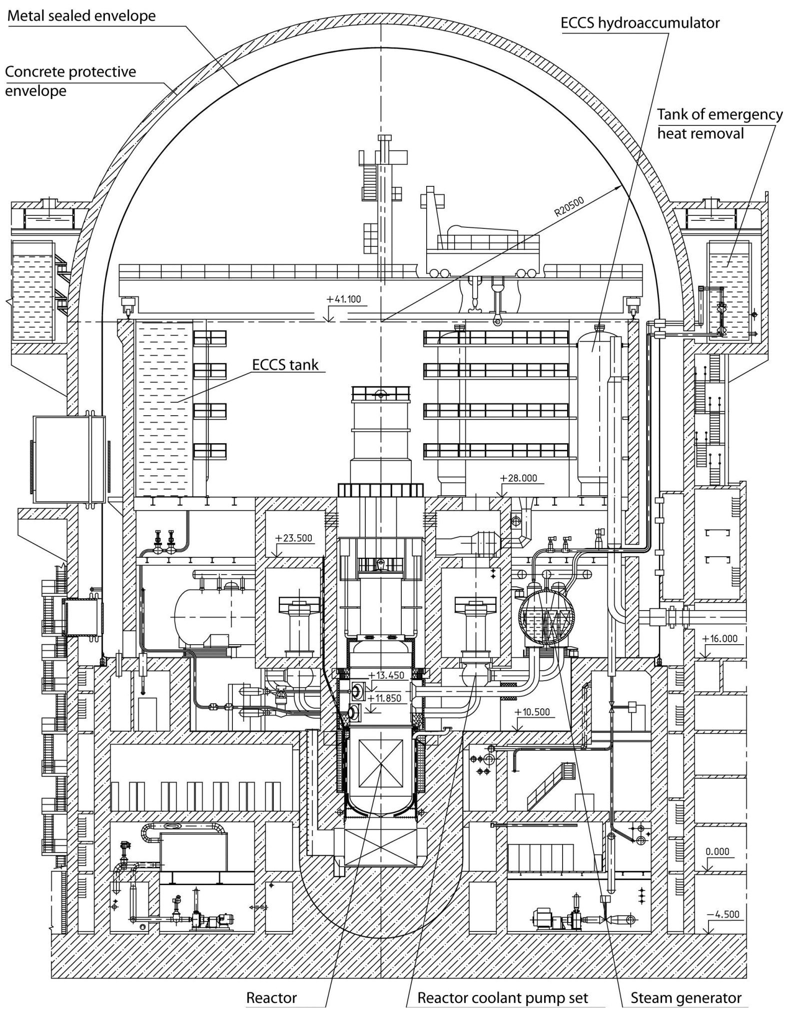


Figure 9.1 – Longitudinal section of the reactor plant building

The reactor building containment is a structure of two concentrically located envelopes, and their simultaneous damage is practically excluded. The internal envelope is intended for receiving loads occurring under accidents in the reactor plant and for confining the fission products under accidents at the reactor plant itself. The internal sealed envelope is a thin-walled steel envelope with a spherical dome of 41.0 m diameter and embedded over bottom at elevation +16.000 into the reinforced-concrete of internal foundation. The envelope thickness is 34-36 mm.

The external envelope is intended for protection of the internal envelope against external impacts. In the gap between the internal and external envelopes a low rarefaction is maintained and provision is also made for the environment monitoring and cleaning. The main parameters of internal sealed envelope are given herein below.

Inside diameter, m	41
Total volume, m^3	60 000
Design pressure, MPa	0.5
Design temperature, ^oC	150
Design leak	not more than 0.1 % of sealed volume for 24 hour

The external envelope of reinforced-concrete is designed for external impacts from:

- aircraft crash (light airplane-5t, heavy airplane-20t);
- explosion wave;
- seismic impacts up to magnitude 8 as per MSK-64 scale (SSE).

The reinforced-concrete envelope is a cylinder with inside diameter of 44.6 m and wall thickness of 800 mm with a spherical dome. This envelope is fastened on a base slab of the reactor building foundation at the elevation of minus 4.500 and separated from other civil structures with gaps. On the cylinder outer side, in its upper part in the annular pathway the sections of the emergency heat removal tank are arranged with the passive heat removal system heat exchangers.

The unsealed area of reactor building includes the rooms located within the foundation part from the elevation of minus 4.500 to the elevation of 10.500, including rooms for the isolation valves at the elevation of 10.500 around the sealed envelope outer perimeter.

Proceeding from the concept of application of independent safety system channels "2x2" at NPP, the equipment of each channel is located in separate rooms distant in the territory and arranged in mirror position about axis 0° - 180°. The equipment of high-pressure emergency injection system (JND), fuel pool cooling pipelines system (FAK), reactor plant cooling down system (LAH), component cooling systems of reactor building (KAA20), chemical and volume control system (KBA), main feedwater and main steam pipeline system (LAB, LBA), ventilation system of the respective rooms, instrumentation and radiation and chemical monitoring instruments, are also arranged in this area.

The PHRS tanks (emergency heat removal) are arranged in the annular room in cantilever outside the protective reinforced-concrete envelope at the elevation of 39.500.

The building of process systems and special water purification (SWP) with dimensions in plan 45.0 x 66.0 m (safety class II) is a process lean to the containment building and adjoins it over the long wall on the side of a scaffold bridge. This building contains auxiliary systems of the primary circuit including SWP systems. The building is made of monolithic reinforced concrete.

At the building lower elevations the special wastewater disposal systems and the borated and service water drain systems are arranged. Laboratories are situated at elevation 7.2 m. At the building upper elevations the rooms of instrumentation, control and protection systems, exhaust ventilation center and vent stack are arranged. The lock for personnel access to the containment area is located at elevation 31.8 m; entrance and exit of the building are provided through the sanitary-utility module.

The turbine building layout

The main turbine building is a single-span building with the dimensions in plan 36x72m and elevation of the floor truss bottom of 34.6m.

The operating elevation of the turbine hall is + 15.6m.

The condenser basement floor: - 4.2m. The main elevations of the equipment service platforms: 0.00, 6.00, 6.90, 11.40.

Longitudinal arrangement of the turbine is in the middle of 36-meter span. A total length of the turbine set from the front bearing end to the back bearing end of the exciter is 47.325m. The turbine is installed on the reinforced-concrete base and its axis elevation 16.52m. Steam superheaters are installed on their own bases with elevation of the base top –1.00 on the both sides of HMPC.

At the elevation of 6.90 m the vertical heating-system water heaters are installed. Deaerator-feedwater plant is located upstream of the turbine front in cell 12x36m. The deaerator is placed at elevation 21.00 m. Three FWPs and two AFWPs are installed at elevation 0.00 m. An individual bridge crane with load carrying capacity 20/5t is provided for their maintenance. The entry for motor transport is also provided at elevation 0.00m.

Plant performance

For VVER-640 the equipment is used maximum unified with that for VVER-1000 (including reactor vessel), therefore the manufacturers have confirmed all the components as stipulated in the specifications, except for a small list of the equipment for which the run-in of new modifications of standard components is necessary.

The design of VVER-640 plant is completely blended with the work on standardization of the VVER plant designs.

Economic efficiency of NPP operation is governed by the quantity of the produced energy depending on the reliability figures of power generation. The main reliability figure of power generation is the load factor which is a ratio of quantity of energy to be produced during a definite time period (for example, during one year) to quantity of energy which could be produced if the Unit would operate that period at nominal power. Loss of power generation can result from Unit shutdown for scheduled repairs and maintenance, including replacement of spent nuclear fuel, or may be caused by random failures resulting in Unit shutdown or power decrease. For VVER-640 plant designs the target value of load factor was established to be 0.90 within the time interval of one year. To reach this value the design provides for certain measures ensuring duration of Unit planned shutdowns for refueling and repairs within the time interval of 16-40 days annually at the specified interval within the service life. Measures for cost reduction of the refueling are:

- optimization of the schedule for each shutdown (as regards details of operations, their sequence and duration);
- using of fuel handling machine operating at higher velocity of motion to perform fuel handling procedures;
- fresh fuel (FAs, CPS ARs) loading into the spent fuel pool during RP primary circuit cooling down;
- LCC of FA fuel rods in the course of refueling with the use of FHM LCC;
- combining the activities on reactor refueling and on repair and maintenance of steam generators owing to usage of isolation devices installed in steam generator collectors.

Alongside with the mentioned measures the technical solutions are used in the design ensuring cost reduction for repair:

- installation of the reliable and tested equipment;
- increase in the intervals between scheduled inspections;
- increase in the equipment service life;
- consideration and implementation of measures on prevention of failures recorded during operation of similar equipment.

Revisions have been made recently by the results of examinations and activities concerning VVER-640 plant design for the third countries intended to optimize the technical solutions, to reduce costs of NPP construction and to reduce the unit costs of electric power produced. The technical solutions have been optimized regarding the following sections of the design:

1. Process and layout solutions as regards the main buildings and structures of the Unit.

The following provisions are made to reduce costs of construction work:

- combining of a part of Unit structures into common buildings (fresh fuel storage and laboratory-utility module are incorporated into the auxiliary building, the diesel generator buildings are attached to the building of electrical and control systems). The building of spent fuel temporary storage is excluded from the design. Optimization of the general plan enabled to reduce the construction area by 40-50 %;
- the underground cable and pipe routes are planned mainly; communication lines in tunnels are significantly reduced.

Considering a high reliability of the passive safety systems, as well as the criteria applied in western designs, the conservative margins incorporated in VVER-640 for BDBA management are removed:

- requirements for seismic protection of the cooling water systems, intermediate circuit and main diesel-generators are made less stringent because these systems are not required for safety assurance under design basis accidents and there is enough time to restore these systems;
- according to Russian standards the active system of boron injection into the reactor, fulfilling its functions under ATWS, can be brought from class 23 into class 3H (class of systems important to safety). Accordingly, the power of small emergency diesel-generators is reduced and usage of standard diesel-generators run in Russian industry became possible;
- in connection with optimization of the systems of handling boron-containing water the number and volume of the tanks in the system of sump and circuit water treatment and storage are reduced.

2. Solutions used in electrical and I&C systems.

Optimization made in the part of electrical systems and I&C enabled to reduce the scope of construction work of the building for electrical and control systems by 50 %. It is achieved due to:

- application of the up-to-date switchgears permitting to arrange up to 80 low-voltage connections in one cabinet of sizes 2200x1000x1000 and to give up the current-limiting devices and secondary assemblies of low-power consumers due to high resistance of commutation equipment to short-circuit currents. It enables three times reduction of the number of low-voltage cabinets;
- increase in the unit capacity of auxiliary transformers 10/0.69 and 10/0.4 kV enabling two time reduction in the number of transformers. It becomes possible at the expense of usage of commutation equipment with high resistance to short-circuit currents;
- reduction of in the number of I&C cabinets almost twice due to usage of the up-to-date digital and fiber optic technologies that resulted in simplification of the I&C structural scheme;
- decrease in the volumes of ventilation systems owing to the above mentioned and usage of the up-to-date equipment of improved reliability that resulted in reduction of the number of reserve and simplification of the structural scheme.

Usage of the copper cables with insulation free of halogens and not sustaining combustion instead of aluminum cables enables to reduce a fire hazard essentially and to reduce costs of cable mounting.

Change over to average voltage 10 kV instead of 6 kV and additional input of low voltage 0.69 kV, alongside with 0.4 kV, enabled to reduce short-circuit currents essentially and, as consequence, to reduce cost of electric equipment and to decrease the cable cross-sections that also results in price reduction of NPP construction.

Changes in the process part enabled also to decrease scopes and cost of electric equipment. Thus, for instance, transfer of the active ECCS systems to the lower safety class enabled to reduce power of the standby diesel-generators considerably and to arrange them in the protected area of the building of electric and control systems and not in separate buildings designed for all external impacts.

The construction schedule developed by the design and construction organizations provides for constructing the NPP forerunner Unit during 5 years, the standard Unit – during 4.5 years, and the preparatory period on site is 2 years. At present, the state of VVER-640 design corresponds practically to the construction starting stage (the working documents for the beginning of construction have been already given to the site) and additional studies of design solutions are under way concerning clarification of the proposed deliveries, and preliminary estimations of financial possibilities are provided. The construction schedule is presented in Figure10.1.

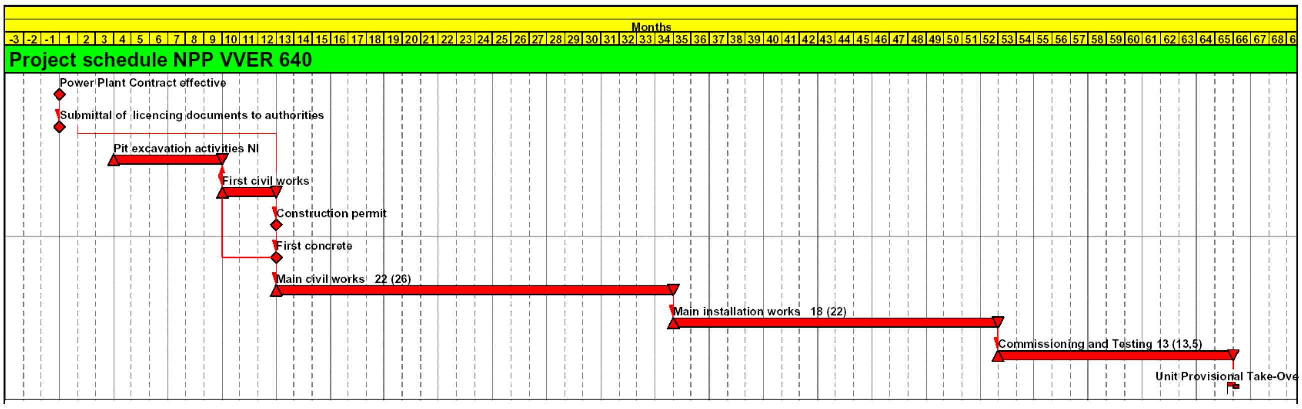


Figure 10.1 Construction schedule

(*) The licensing terms as accepted in EEC are given in the schedule

Development status of technologies relevant to the NPP

Composition and design of the main components, equipment and systems of NPP with VVER-640 are based on the solutions laid down in the preceding designs and improved according to the up-to-date requirements that enable to improve NPP performances and ensure the required safety level.

Basic technical solutions for NPP are confirmed by operation experience of VVER reactors during more than 1400 reactor-years including more than 500 reactor-years of VVER-1000 reactor operation.

Application of evolutionary approach to solution of target design tasks enabled to prepare AES-2006 design in a shortest time with meeting the requirements of the up-to-date Russian standards and regulations.

VVER-640 reactor design is considered by its developers to possess the required level of competitiveness and the justified prospects for standard construction both in Russia, and abroad.

Deployment status and planned schedule

On the basis of the Basic design for the conditions of the sites in Sosnovy Bor, Kola NPP-2 and Far-East NPP the following documents have been elaborated: the feasibility study of construction, preliminary safety analysis reports (PSAR), environmental impact evaluation, probability safety assessment (PSA) and detailed project report of Kola NPP-2 . The mentioned documents were reviewed and examined by the state bodies and expert organizations at the regional and federal levels (of Gosatomnadzor with issue of the construction permit for NPP forerunner and Unit 1 at Kola NPP-2 site, of Ministry of Health, Ministry of Environmental Protection and Natural Resources, Sanitary Inspectorate, Fire Inspectorate, Ministry of Extraordinary Situations, etc.) as well as by experts from GRS (Germany) and DoE (USA).

The scope of R&D activities made for the design of VVER-640 plant is sufficient for the beginning and performing the Unit forerunner in accordance with the licenses and permits obtained and the examinations conducted at the federal and regional levels.

As to the reactor plant, the main part of the detailed project report documentation has been prepared that enabled to obtain the construction permits. The work on the reactor plant detailed project can be completed together with NPP construction activities and does not exert essential influence on putting into production of the equipment with long-run manufacturing cycle.

The detailed project reports were prepared on the turbine plant and turbine generator, steel protective envelope, equipment of the sump and boron-containing water treatment systems together with the cementation plant, and the technical proposals were developed on the pump, heat exchanging and ventilation equipment and valves.

The organizational structure on VVER-640 plant design implementation includes the following leading enterprises:

- Organization of NPP General Designer – SPbAEP;
- Organization of RP General Designer – OKB "Gidropress";
- Organization of Scientific Leader of the design – RRC " Kurchatov Institute ";
- General Manufacturer of "Nuclear Island" equipment– JSC "Izhorskie zavody ";
- General Manufacturer of the turbine and low-pressure regeneration equipment – JSC LMZ;
- General Manufacturer of the generator – JSC "Electrosila";
- Manufacturer of pump equipment – CKBM;
- General Manufacturer of cable products – VNIIKP;
- General Manufacturer of the NPP I&C– NITI named after A.P.Alexandrov;
- General Contractor for construction – to be defined as a result of site-specific Tender.

The experimental facilities of the organizations involved in R&D:

- models of containment V=1 m^3, 8 m^3, 100 m^3, 3200 m^3 – SSC RF IPPE;
- models of SG PHRS and PHRS of containment – NPO CKTI, SSC RF IPPE, OKBM;
- test facility IST (NC) – NPO CKTI, NITI;
- test facility KMS - NITI;
- test facility PSB - EREC;
- test facilities for verification of in-vessel corium retention:
    1. study of DNB – NITI, SSC RF IPPE, OKB GP
    2. processes in the melt bath and interaction with the vessel – NITI, RRC KI, SSC RF IPPE.

Operability of passive components of the primary circuit depressurization system has been verified at OKB GP mockups and prototypes. Some passive components of the emergency core cooling system (hydro-accumulators, check valves) used in V-407 reactor plants are approved by operating experience of the existing VVERs.

Before putting RCP set into production the manufacturing and testing of a prototype are planned at JSC "Kirovsky zavod”.

For I&C systems the developments are used unified with VVER-1000 and made by the program of creation of I&C hardware at VNIIA in co-operation with Siemens.

Manufacturing of the main process equipment is mastered at the Russian industrial enterprises.

R&D for making equipment-specific engineering solution have been completed. Verification of joint operation of the passive safety systems needs to be completed at the test facility KMS-500, VVER-640 prototype, in Sosnovy Bor.

To finalize R&D of the design the following activities are to be performed:

- preparing and conducting the experiments at the test facility KMS;
- experimental after-studies for verification of in-vessel corium retention;
- verification of applicability of engineering solutions for the core and passive systems of AES-2006;
- R&D for verification of process solutions;
- modification of designs of passive safety systems, metal protective envelope and other unique equipment.

The supposed time to complete activities on this design is about two years. The activities are performed in parallel with NPP construction.

References

1. Safety regulations for the nuclear power stations (PBYa RU AES-89), PNAE G-1-024-90, Moscow, Atomizdat, 1990.
2. General provisions for safety assurance at nuclear power stations (OPB-88/97). NP-001-97 (PNAE G-01-001-97), Moscow, Energoatomizdat, 1997.
3. Code for designing earthquake-resistant nuclear power stations.NP-031-01, Moscow, 2001.

List of abbreviations

AFWP - emergency feedwater pump;

AR - absorbing rod;

ATWS - anticipated transient without scram;

BDBA - beyond design basis accident;

BOC - beginning of cycle;

BRU-A - steam dump valve to the atmosphere;

BRU-K - steam dump valve to the turbine condenser;

CHRS - containment heat removal system;

LCC - leak-check of claddings;

CPS - control and protection system;

CR - control rod;

CSS - control safety system;

DADS - defective assembly detection system;

DBA	- design basis accident;
DVU	- depressurization valve unit;
ECCS	- emergency core cooling system;
ECR	- emergency control room;
EHRP	- emergency heat removal pool;
EP	- emergency protection;
ESFAS	- engineered safety feature actuation system;
FA	- fuel assembly;
FFS	- fresh fuel storage;
FHM	- fuel handling machine
FWP	- feedwater pump;
HMPC	- high and medium pressure cylinder;
HPC	- high-pressure cylinder
HPH	- high-pressure heater;
HRS	- heat removal system;
I&C	- instrumentation and control system;
IAEA	- International Atomic Energy Agency;
IC	- ionization chamber;
ICID	- in-core instrumentation detectors;
KMS	- large-scale test facility;
LB	- large break;
LBB	- leak before break;
LOCA	- loss of coolant accident;
LPC	- low-pressure cylinder;
LPH	- low-pressure heater;
LRW	- liquid radwaste;
MCP	- main coolant pipeline;
MCR	- main control room;
MPC	- medium-pressure cylinder
MSIV	- main steam isolation valve;
NC	- natural circulation
Nnom	- nominal power;

NPP	- nuclear power station;
NTMC	- neutron and temperature measuring channel;
PHRS	- passive heat removal system;
PM	- preventive maintenance;
PORV	- pilot-operated relief valve;
PP	- preventive protection;
PRZ	- pressurizer;
PSA	- probabilistic safety assessment;
PTU	- protective tube unit
R&D	- research and development;
RH	- reheating
RCP	- reactor coolant pump;
RP	- reactor plant;
S	- separation
SB	- small break;
SFP	- spent fuel pool;
SFS	- spent fuel storage;
SG	- steam generator;
SRW	- solid radwaste;
SWP	- special water purification;
TG	- turbine generator
VVER	- water-cooled water-moderated reactor;

Technical data

General plant data

Reactor thermal output	1800 MWth
Power plant output, gross	645 MWe
Power plant output, net	603.7 MWe

Power plant efficiency, net	33.3 %
Mode of operation	Baseload and Load follow
Plant design life	60 Years
Plant availability target >	90 %
Seismic design, SSE	0.2
Primary coolant material	Light Water
Secondary coolant material	Light Water
Moderator material	Light water
Thermodynamic cycle	Rankine
Type of cycle	Indirect
Non-electric applications	District heat, Industrial, Cogeneration

Safety goals

Core damage frequency <	1E-6 /Reactor-Year
Large early release frequency <	1E-7 /Reactor-Year
Operator Action Time	6 Hours

Nuclear steam supply system

Steam flow rate at nominal conditions	1000.0 Kg/s
Steam pressure	7.06 MPa(a)
Steam temperature	286.4 °C
Feedwater flow rate at nominal conditions	1000 Kg/s
Feedwater temperature	230.0 °C

Reactor coolant system

Primary coolant flow rate	14945.0 Kg/s
Reactor operating pressure	15.7 MPa(a)
Core coolant inlet temperature	294.3 °C
Core coolant outlet temperature	322.7 °C
Mean temperature rise across core	28.4 °C

Reactor core

Active core height	3.53 m

Equivalent core diameter	3.16 m
Average linear heat rate	10.0 KW/m
Average fuel power density	26.2 KW/KgU
Average core power density	64.5 MW/m^3
Fuel material	UO2 and UO2 + Gd2O3
Fuel element type	Fuel rod
Cladding material	Alloy E-110
Outer diameter of fuel rods	9.1 mm
Rod array of a fuel assembly	Triangular
Number of fuel assemblies	163
Enrichment of reload fuel at equilibrium core	3.18 Weight %
Fuel cycle length	12 Months
Average discharge burnup of fuel	47.38 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	4132.0 mm
Wall thickness of cylindrical shell	192.5 mm
Design pressure	17.6 MPa(a)
Design temperature	350 °C
Base material	Steel 15H2NMFA
Total height, inside	10655.0 mm
Transport weight	288 t

Steam generator or Heat Exchanger

Type	PGV-640 horizontal
Number	4
Total tube outside surface area	4222.6 m^2
Number of heat exchanger tubes	8310
Tube outside diameter	16.1 mm
Tube material	08H18N10T
Transport weight	210 t

Reactor coolant pump (Primary circulation System)

Pump Type	GTSNA-1455
Number of pumps	4
Pump speed	1500 rpm
Head at rated conditions	0.349 m
Flow at rated conditions	3.72 m^3/s

Pressurizer

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

Primary containment

Overall form (spherical/cylindrical)	Cylindrical
Dimensions - diameter	41.0 m
Dimensions - height	51.0 m
Design pressure	0.5 MPa
Design temperature	150.0 °C
Design leakage rate	0.1 Volume % /day

Residual heat removal systems

Active/passive systems	Active

Safety injection systems

Active/passive systems	Active

Turbine

Type of turbines	K-640-6.9/50
Number of turbine sections per unit (e.g. HP/MP/LP)	1/0/2

Turbine speed	3000 rpm
HP turbine inlet pressure	6.9 MPa(a)
HP turbine inlet temperature	284.5 °C

Generator

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

Condenser

Condenser pressure	4.8 kPa

Feedwater pumps

Type	PEA 1470-78
Number	3
Head at rated conditions	900 m
Flow at rated conditions	1475 m^3/s