Status report 93 - VVER–1000 (V-466B) (VVER–1000 (V-466B))

Overview

Full name	VVER–1000 (V-466B)
Acronym	VVER–1000 (V-466B)
Reactor type	Pressurized Water Reactor (PWR)
Coolant	Light Water
Moderator	Light water
Neutron spectrum	Thermal Neutrons
Thermal capacity	3000.00 MWth
Gross Electrical capacity	1060.00 MWe
Design status	Under Construction
Designers	Gidropress
Last update	21-07-2011

Description

Introduction

This design description presents information on the "Belene" Nuclear Power Plant (NPP) design in Bulgaria with reactor plant V-466B, which is based on the AES-92 evolutionary NPP design, with reactor plant V-392 having 1000 MW electric power. This design has been elaborated within the framework of the State Program "Environmentally safe power engineering" and it has been described in several publications, including IAEA-TECDOC-1391, which was published in 2004. The basic objective of the NPP design of a new generation is elaboration of a unified competitive NPP design satisfying the present-day safety requirements and Operators' requirements.

The AES-92 design has been developed according to the requirements of Russian safety regulations in force, and with the objective of meeting the IAEA Safety Standards.

To a great extent this design incorporates the accumulated experience in design, manufacture and operation of NPPs with VVER-440 and VVER-1000 reactors.

It should be emphasized that Russia has a 50-year experience in designing and building NPPs with VVER reactors in Russia and abroad (Bulgaria, Finland, Hungary, Germany, Czechiă, Slovakia, etc.), and this is the base for implementation of objectives and tasks of NPP new designs. Operation of NPPs with VVER-1000 reactors amounts to over 500 reactor-years.

Now a license of the State Safety Regulatory Authority of Russia is available for construction of the 2-nd phase of Novovoronezh NPP under AES-92 design, and construction of two Units of "Kudankulam" NPP in India is under way implementing basic design solutions of this design.

In addition to the licensing procedure by the Russian regulatory authorities, this design has been analyzed by European Utility Requirements (EUR) Club experts for compliance with the European Utilities Requirements for new NPPs with light-water reactors (LWRs). This design has got the certificate of EUR Club by the results of the review.

At the same time, corresponding to AES-92 design, "Belene" NPP design is focused on more complete fulfillment of the requirements for enhancing its economic efficiency. These requirements, as compared to the V-392 design, include, first of all, the requirements for service life extension of the main equipment, requirements for improvement of fuel utilization and other operational characteristics.

The design uses experience of creation and operation of the Units with reactors VVER- 440 and VVER-1000 to the utmost as well as the results of the present-day R&D.

As a result, the "Belene" NPP design with RP V-466B, elaborated on the basis of AES-92, has the improved technical-and-economic indices and is refered to by the design organization as a Generation 3+ design. The basic technical performances and safety indices are given in the Appendix.

Description of the nuclear systems

2.1 Basic characteristics of the reactor coolant system

The reactor coolant system is intended for production of thermal energy released as a result of the controlled nuclear fission reaction of the fuel, heat removal from the reactor core, and steam generation in steam generators using this heat.

The reactor coolant system involves: water-cooled water-moderated power reactor and four circulation loops consisting of steam generators, reactor coolant pumps and pipelines Dnom 850 as well as the pressuring system connected to one of circulation loops.

The reactor coolant system is a barrier to the release of radioactive substances from the core into the secondary circuit system and the containment volume.

2.1.1 Main coolant pipeline (MCP)

The main coolant pipeline connects the reactor, steam generators and main coolant pump sets between themselves forming a circulation system, and it is intended for coolant circulation through the reactor and steam generators.

MCP consists of four circulation loops, each loop has two sections of tubes. A section between a reactor outlet nozzle and steam generator inlet collector is a ^{2}hot^{2} leg. A section between steam generator outlet collector and RCP set inlet (suction) nozzle and between RCP set outlet (discharge) nozzle and reactor inlet nozzle is a ^{2}cold^{2} leg. The size of internal diameter (850 mm) is chosen providing MCP acceptable pressure loss under design coolant flowrate 21500 m^3/h (in each loop).

The line of 426x40 mm connects the hot leg of Loop 4 and the pressurizer.

"Cold" leg No.3 is connected to the pressurizer by the line 219x20 mm (injection line).

MCP is made of 10ГН2МФА alloy structural steel. The internal tube surface is clad with corrosion-resistant stainless steel 04Х20Н10Г2Б not susceptible to intergranular corrosion in coolant.

Parameters	Value

Inside diameter of hot leg/cold leg, mm/mm	850
Wall thickness, mm	70
Coolant flowrate in the loop, m^3/h	21500
Hot leg length, m	10
Cold leg length, m	26

Table 2 - 1 MCP parameters

Schematic diagram of the primary equipment is given in Figures 2-1 and 2-2.

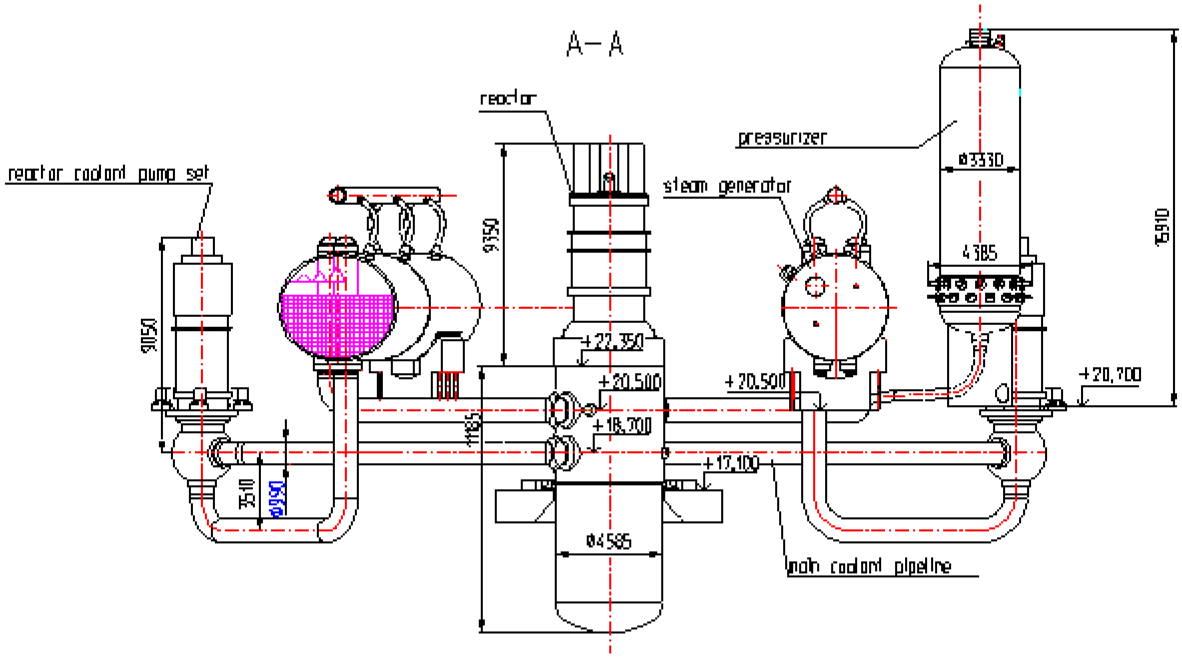

Figure 2-1. Lay-out of the reactor coolant system

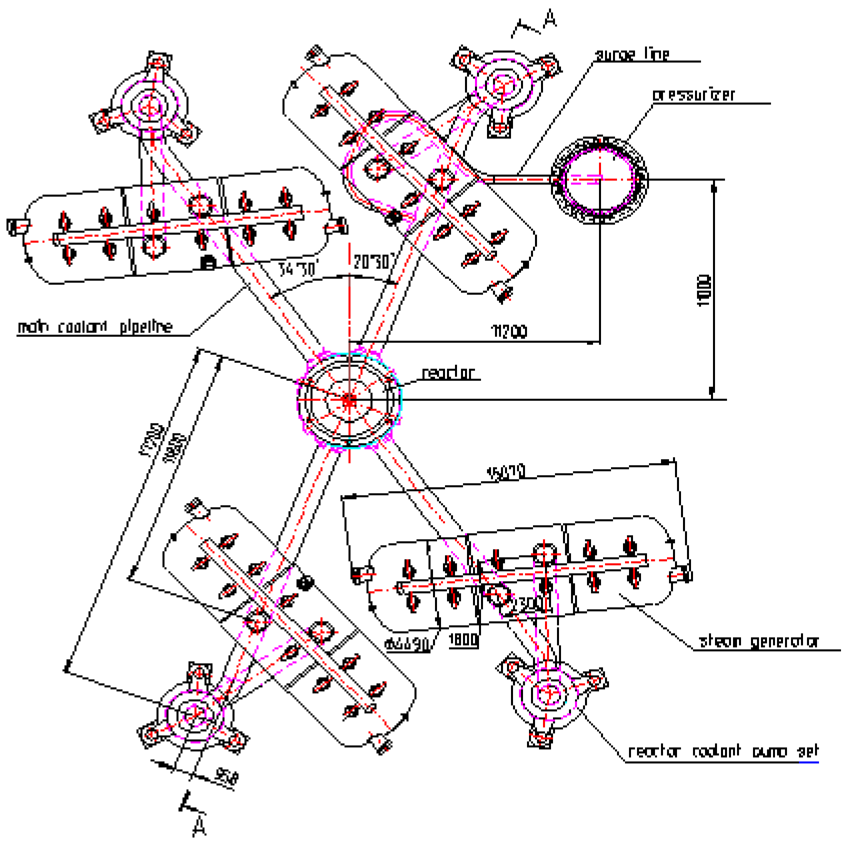

Figure 2-2. Lay-out of the reactor coolant system

2.2 Reactor core and fuel design

The core includes 163 fuel assemblies, identical in design, but different in fuel enrichment.

The TVSA fuel assembly (FA) is considered as a base version of fuel assembly (FA) design and as an alternative version is TVS-2. Both versions of FA are interchangeable and are of reference character.

The core design is developed for the generalized version of FA design (both base and the alternative) providing its operability in using several FA types.

FA design (both base TVSA and alternative TVS-2) consists of the following components:

- top nozzle;
- bundles of fuel rods (fuel rods and Gd fuel rods);
- bottom nozzle.

FA top nozzle provides necessary force of FA compression in the core.

Bundles of fuel rods (fuel rods and Gd fuel rods) consist of a skeleton that houses 312 fuel rods (Gd fuel rods).

The skeleton consists of guiding channels and spacing grids and provides the following throughout the FA service life:

strength and ‘always” geometry (small bowings) of FA and fuel rods (Gd fuel rods) and spacing of fuel rods (Gd fuel rods).

FA bottom nozzle provides conjugation of FA lower part with the support of the reactor core barrel and presents a guiding device for coolant supply into bundles of fuel rods and Gd fuel rods.

General views of the base and the alternative FA are given in Figures 2-3 and 2-4.

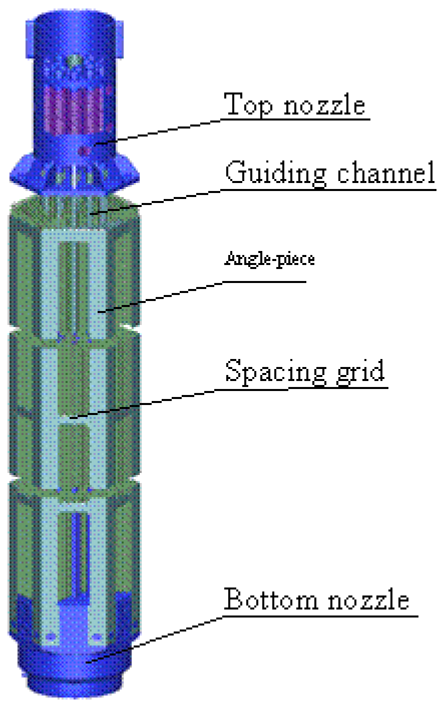

Figure 2-3. TVSA general view

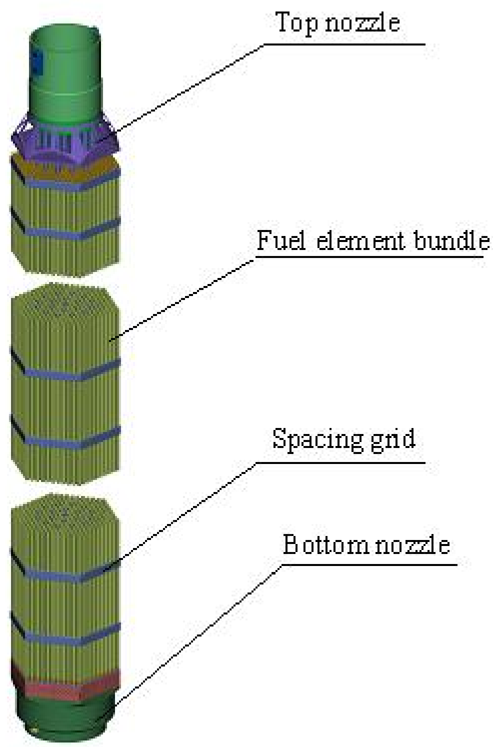

Figure 2-4. TVS-2 general view

The rod control cluster assembly (RCCA) consists of 18 absorbing elements (AEs), the grip head, springs of an individual suspender.

The RCCA AE is a tube with outer diameter $8{,}2\times10^{-3}$ m and wall thickness $0{,}5\times10^{-3}$ m filled with absorbing material and sealed with end pieces by means of welding. Boron carbide B_4C and dysprosium titanate (Dy_2O_3 TiO_2) are used as absorbing material. Dysprosium titanate in the AE lower part enables to extend RCCA service life under maintenance of sufficient worth of emergency protection.

The basic characteristics of the equilibrium fuel cycle are presented in Table 2-2.

Table 2-2 Main characteristics of the equilibrium fuel cycle

Characteristic	Value
Number of loaded fresh FA, pcs. - total	 42
Average enrichment of make-up fuel in U^{235}, % wt.	4.45
Duration of reactor operation between refuellings, eff. days	325
Burn-up of unloaded fuel, MWD/kg U:	

Characteristic	Value
- average	52.8

2.3 Fuel handling systems

The complex of systems for refueling and nuclear fuel storage represents a set of systems, devices, components intended for storage, loading, unloading, transportation and the inspection of nuclear fuel.

The complex of systems for refueling and fuel storage provides:

- receiving, storage and incoming inspection of a fresh nuclear fuel before loading into the reactor;
- nuclear fuel reloading in the reactor core;
- storage of spent nuclear fuel in spent fuel pool of NPP reactor building;
- a long-term storage of spent nuclear fuel in the storehouse at NPP site;
- on-site transportation of nuclear fuel on NPP territory starting from receiving a special transport with fresh fuel and finishing with a long-term storage of spent nuclear fuel in SNFS (spent nuclear fuel storehouse).

2.4 Description of primary components

2.4.1 Reactor and internals

A reactor is a vertical pressurized vessel that houses core barrel with the baffle, protective tube unit, fuel assemblies, RCCA, in-core instrumentation detectors. The general view of the reactor is given in Figure 2-5.

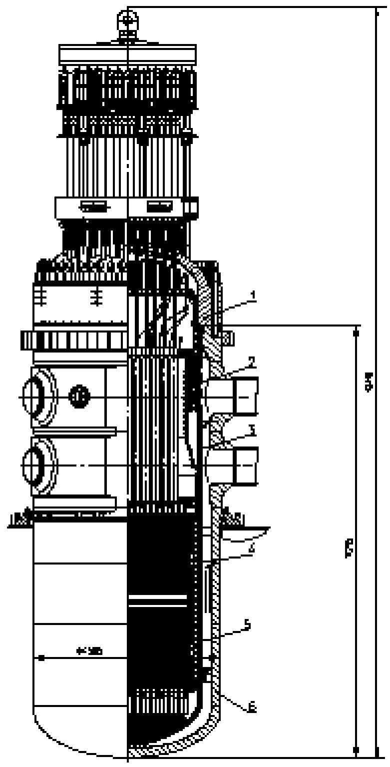

Figure 2-5. Reactor

1 – upper unit, 2 – protective tube unit, 3 – core barrel, 4 – core baffle, 5 – core, 6 –nuclear reactor vessel

The VVER-1000 reactor vessel design is based on the following principles:

- proven manufacturing process and structural materials;
- complete in-shop manufacture of the vessel, tests included;
- possibility of vessel transportation by rail and by sea;
- possibility of periodic in-service inspection of the vessel.

The reactor vessel consists of several forged shells welded to each other, elliptic bottom head and flange, sealed with solid ring sealing gaskets and tightened with 54 M170 studs.

Two nozzle shells of the vessel have four nozzles Dnom 850 each that are connected to the main coolant pipeline of reactor coolant system.

The ring welded to the vessel internal surface clad with austenitic steel, serves to separate the inlet (coolant) and outlet chambers, core barrel-mated keys that keep it from radial displacements, and arms to install hermetic casks for the vessel steel surveillance specimens.

The reactor vessel is made of heat-resistant alloy steel, grade 15Х2НМФА. Reactor vessel steel and welding materials were chosen on the basis of the analysis of mechanical properties, lack of susceptibility to brittle fracture, durability and irradiation stability.

The core barrel is a welded cylindrical shell with a supporting bottom and a flange to be supported on the vessel shoulder.

The perforated elliptical bottom of the core barrel, together with 163 perforated support tubes and spacing grid, makes up a structure to support and space the FAs.

Core baffle placed in the core barrel at the core level is at the distance of a structural gap from FA periphery row and serves as a displacer and a protective screen.

The core baffle is made of several massive rings that are mechanically attached to each other and to the core barrel bottom.

A lot of longitudinal channels for coolant passing ensure effective cooling of the core baffle metal.

The PTU is installed on the top and pressed to the core barrel flange by the force of the elastic element installed between PTU shoulder and UU top head and compressed as reactor main joint is sealed.

The perforated shell of the PTU with plates and protective tubes make a rigid support structure to space the FAs and keep them from lifting. PTU protective tubes house CPS CRs and ICID (In-core instrumentation detectors).

The internals are made of corrosion-resistant steels of austenitic grade.

The upper unit structure includes the elliptic top head with a flange and nozzles, CPS CR drives, metalwork and cross-piece.

Pitch electromagnet drive of SHEM-3 type is used as CPS CR drive that provides motion of control rods with the velocity of 2 cm/s.

For V-446B reactor the number of drives installed on the upper unit can be from 85 to 121 pcs. depending on the fuel loading.

The general view of the reactor vessel, core barrel, protective tube unit, upper unit is given in Figure 2-6.

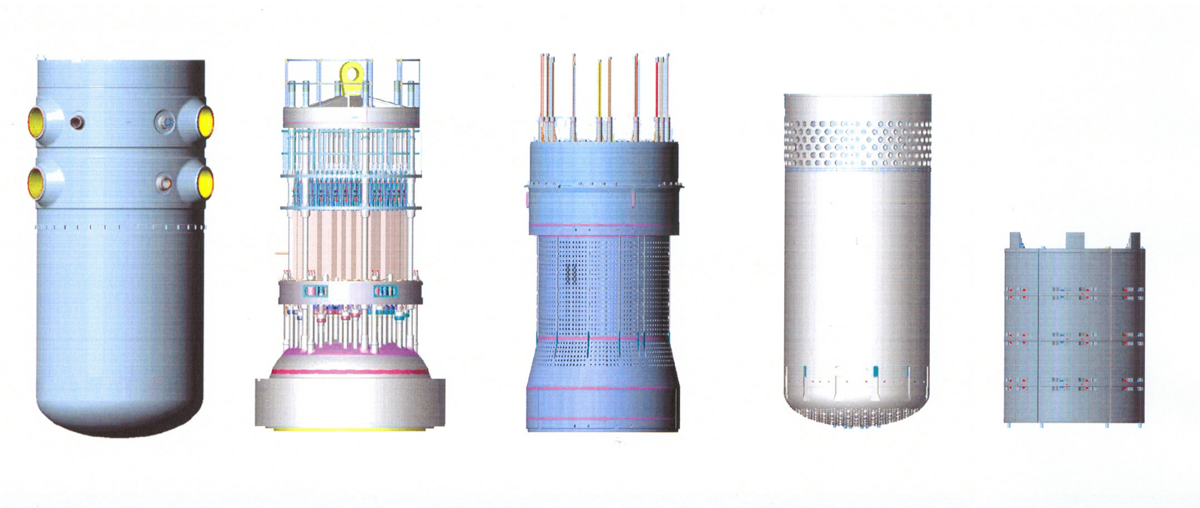

Figure 2-6. Reactor vessel, upper unit, protective tube unit, core barrel and core baffle

2.4.2 Reactor coolant pump set

GCNA-1391 is a vertical centrifugal one-stage pump set that contains a hydraulic casing, internals, electric motor, top and bottom spacers, supports and auxiliary systems.

The electric motor is of vertical asynchronous double-speed type. To prevent the reverse rotation of the rotor, RCP set is equipped with anti-reversing mechanism presenting a cam-and-ratchet mechanism. RCP set is equipped with a flywheel to provide inertia coastdown. General view of RCP set is provided in Figure 2-7.

Parameter	**Table 2-3. RCP set main parameters**	Value

GCNA-1391 is a further evolution of GCN-195M design and has the following improvements:

- main thrust bearing is water-cooled and water-lubricated;
- application of double-speed electric motor decreases grid loading at pump start-up;
- the applied shaft sealing ensures unlimited outage under cold and hot standby proving intermediate circuit sealing and cooling water is supplied and outage at NPP blackout (loss of cooling) under nominal parameters of coolant for not less than 24 hours providing leaks through the seal not above 50 L/h.

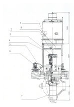

Figure 2 – 7. Reactor coolant pump set

1 – electric motor, 2 – laminated coupling torsion bar, 3 – internals, 4 – top spacer,5 – bottom spacer, 6 – support, 7 – pump casing, 8 - flywheel

2.4.3 Pressurizer

The primary system pressurizer is a vertical vessel with electric heaters to increase primary pressure.

The pressurizer vessel is made of carbon steel with corrosion-resistant austenitic cladding of internal surfaces.

Two independent nozzles with spray devices are mounted in the upper part of the pressurizer, they ensure water injection into steam space from the following sources:

- from RCP set discharge line under normal operating conditions and under anticipated operational occurrences;
- from discharge line of high-pressure emergency injection pumps under design basis and beyond design basis accidents.

The pressurizer is connected to the hot leg of the main coolant pipeline through the lower nozzle with surge line of Dnom 350.

There is also a nozzle in the top section of the pressurizer to attach the primary circuit overpressure protection system that consists of three pilot-operated relief valves.

General view of the pressurizer is given in Figure 2-8.

Parameter	Value
Pressure in PRZ, MPa	15.7
Capacity (full volume), m^3	79
Water inventory under nominal conditions, m^3	55
Power of electric heaters (total), kW	2520
Quantity of electric heaters, pcs.	28

Table 2-4 Pressurizer main parameters

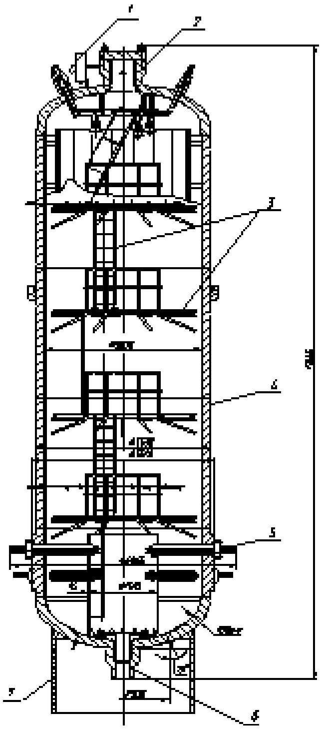

Figure 2 – 8. Pressurizer

1 – surge bottle, 2 – neck, 3 – internals, 4 – vessel, 5 – tubular electric heater unit, 6 – nozzle, 7 – support

2.4.4 Steam generator

The steam generator PGV-1000MK is a single-vessel recuperative heat-exchanger of horizontal type with submerged heat exchanging surface. General view of steam generator is presented in Figure 2-9.

The steam generator vessel is designed to house the internals. The steam generator vessel consists of forged shells, stamped elliptic bottoms and forged nozzles connected by welding. Vessel design provides easy access to examine the internals on the secondary side.

The heat exchanging surface comprises 10978 U-tubes 16x1.5 mm in diameter in U-coils, positioned horizontally in corridor arrangement. The coils are connected to primary-side collectors. Their edges are hydraulically expanded over the entire collector wall thickness and welded on the inside surface of the collectors by argon-arc welding. Austenitic steel is the tube material.

Primary-side collectors are designed for coolant distribution in heat exchanging tubes, its collection and removal.

The internal surface of the collector is a two-layer corrosion-resistant cladding.

The steam-distribution plate is installed in the top section of the vessel.

The perforated plate arranged under the SG water level serves for equalization of steam load.

Inside SG, near one of the bottoms, the conditions are created for accumulation of water with increased concentration of salt and other impurities (the so-called “salt cell”) owing to the appropriate arrangement of supply of feedwater and blowdown of SG.

A large water inventory in the vessel is the merit of PGV-1000MK steam generator that provides favourable dynamic characteristics of the entire reactor plant in accident processes caused by loss of feedwater.

Parameter	**Value**
Steam capacity, t/h	1470
Steam pressure at SG outlet, MPa	6.27
Primary coolant temperature at the SG inlet, ^{o}C	321
Primary coolant temperature at the SG outlet, ^{o}C	291
Feedwater temperature, ^{o}C	220
Feedwater temperature when high-pressure heaters are switched off, ^{o}C	164
Steam moisture at SG outlet, %, not above	0.20

Table 2 - 5. Parameters of PGV-1000MK steam generator under nominal conditions

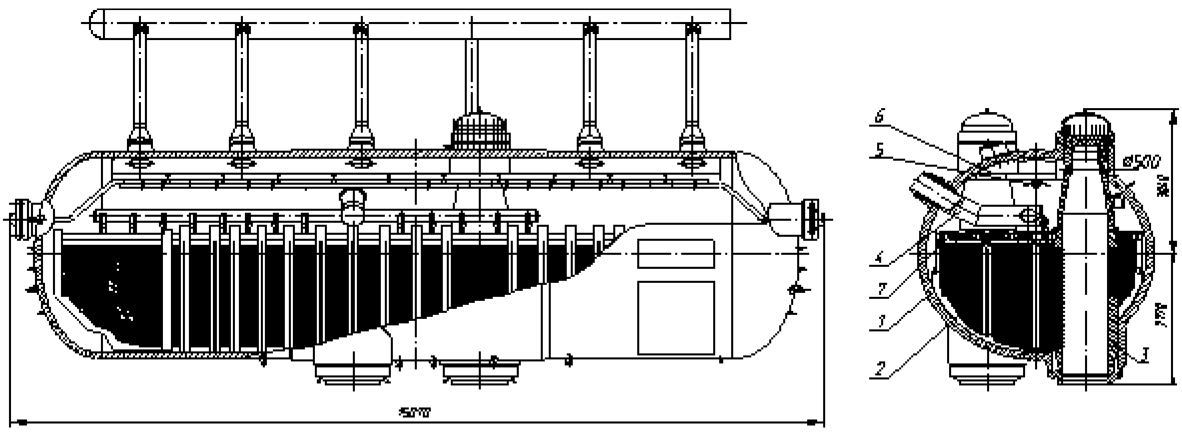

Figure 2-9. General view of PGV-1000MK steam generator

1 – vessel, 2 – heat exchanging surface, 3 – primary-side collectors, 4 – main feedwater distribution devices, 5 – emergency feedwater distribution devices, 6 – steam-distribution perforated plate, 7 – submerged perforated plate (SPP)

2.5 Auxiliary systems

2.5.1 Chemical and volume control system

The system is intended for supply of blowdown water to the system low-pressure filters and its return to the circuit through blowdown-make-up deaerator, replenishment of non-identified leaks, return of identified leaks into the primary circuit, including leaks from RCP seals.

At the same time the chemical and volume control system is used for changing boric acid concentration in the coolant to provide boron control of reactor reactivity during power operation and subcriticality under reactor shutdown, maintaining necessary quality of the coolant by introducing chemical reagents into the primary circuit (hydrazine-hydrate, ammonia, caustic potassium).

2.5.2 Distilled water system

The system is intended for supply of the required amount of distillate to the equipment and the primary circuit systems.

2.5.3 High temperature purification system of primary coolant

The system is intended for purification of high temperature primary coolant without drop of pressure on high-temperature filters from radioactive products of corrosion by removing products of corrosion due to sorption on filtering bed of filters. Capacity of the system – 400 m^3/h.

2.5.4 Low temperature purification system of primary coolant

The system is intended for purification blowdown water of the primary circuit and identified leaks, removal of the rests of boric acid at the end of operating period.

Capacity of the system - to 60 t/h.

2.5.5 Boron concentrate system

The system is intended for supply of boron concentrate with concentration 40 g/dm^3 into the primary circuit systems through chemical and volume control system KBA, and for changing boric acid content in coolant and under compensation of losses of boric acid with non-identified leaks, inclusive.

2.5.6 System for supply of reagents to the primary coolant

The system is intended for receiving and supply of chemical reagents into the chemical and volume control system with the aim of maintaining coolant controlled parameters.

The information on other systems is not given because of the limited scope of the document.

2.6 Operating conditions

The main operating condition is a condition with base load of 100 % of nominal power. The RP equipment is designed considering load-follow mode specified in Table 2-6.

The design confirms the possibility of load shedding to the level of station auxiliaries and to turbine idle running without reactor trip.

Conditions	**Remarks**
Steady-state conditions (with regard for frequency variation within the range of 49.0-50.5 Hz) including power variation $\pm$1 % N_{nom} at the rate of 1 % of N_{nom}/s	Change of power and return to the initial level. Operation with two or three RCP sets is allowed
Variation of Unit power by not more than $\pm$5 % N_{nom} (condition of maintenance of grid frequency) at the rate of 1 % N_{nom}/s	Change of power and return to the initial level
Variation of Unit power at the rate not more than 5 % N_{nom}/min under deviation from the current value not more than $\pm$10 % N_{nom}	Change of power and return to the initial level
Variation of Unit power at the rate not more than 5 % N_{nom}/min within the range from 20 % N_{nom} to 100% N_{nom} (without regard for limitations in terms of fuel)	Change of power and return to the initial level

Table 2-6

2.7 Standard fuel cycle

The standard fuel cycle is single, open. Time between refuellings amounts to 12 months. Length of the standard fuel cycle is 4 years.

2.8 Alternative fuel

Possibility of using MOX-fuel could be considered as an alternative fuel.

2.9 Spent fuel and plans for its transfer

The spent fuel is unloaded from the reactor and put for storage in the spent fuel pool located in the reactor compartment within the containment near the reactor core barrel.

Spent fuel pool lay-out excludes necessity of transfer of transport containers above the stored spent nuclear fuel during transfer of fuel and other cargoes from the reactor compartment.

For transfer of transport containers with fuel the polar crane is used installed in the containment building.

During reactor refuelling the transfer of fuel stored in spent fuel pool into the spent nuclear fuel storehouse (SNFS) is provided.

SNFS is intended for dry storage of spent nuclear fuel (SNF) at NPP site in the dual-purpose containers intended for transportation and storage.

SNFS capacity is designed for a long-term storage of the spent nuclear fuel accumulated during 10 years of operation of two Units with a possibility for the building capacity to be extended in future for fuel storage accumulated during the total operating time of Units.

3.1. Safety concept, main principles of designing and methods of licensing

The design was developed on the basis of safety requirements, codes and standards in nuclear power engineering of Russian Federation and Bulgaria, considering IAEA codes and standards and also with regard for EUR requirements for NPP with LWR that made up ToR basis for design. The design applies the concept of defense-in-depth.

3.1.1 Provision of design simplicity and reliability

Simplicity and reliability of the equipment are provided due to elimination of excess components and quality assurance in designing, manufacturing and operation.

3.1.2 Active and passive systems, inherent safety features

"Belene" NPP design provides for the active and passive safety systems and control systems for management of beyond design basis accidents. Passive systems enhance VVER inherent safety features. Their description is given in item 3.4.

3.2. Description of defense-in-depth

Defense-in-depth concept, applied in the design, is based on application of the system of physical barriers on the way of propagation of ionizing radiation and radioactive substances to the environment and the system of engineering and organizational measures oriented to protection of barriers and maintaining their effectiveness, and also to protection of the personnel, the population and the environment, does not involve some specific features as compared to its realization in design V-392 and AES-2006.

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

The system of engineering and organizational measures forms five levels of defense-in-depth and includes the following levels.

Level 1. Conditions of NPP siting and prevention of anticipated operational occurrences;

Level 2. Prevention of design basis accidents by normal operation systems;

Level 3. Prevention of beyond design basis accidents by safety systems;

Level 4. Beyond design basis accident management;

Level 5. Emergency planning.

3.3. Safety indices

The probability of core damage for one-year fuel cycle, as given in regulatory documents amounts to:

operating conditions at power	$3.1 \cdot 10^{-7}$ /reactor/year
standby conditions	$3.0 \cdot 10^{-7}$ /reactor/year
Probability of large releases	$1.77 \cdot 10^{-8}$ /reactor/year

3.4. Safety systems

3.4.1 Destination of safety systems

The strategy of copying with design basis accidents is based on using both active and passive safety systems. The strategy of copying with beyond design basis accidents is based on using mainly passive safety systems and systems of beyond design basis accident management.

Safety systems are designed to be stable to failures, including dependent failures and common-cause failures, and capable to fulfill the function under loss of power supply.

3.4.2. Structure of safety systems and systems of beyond design basis accident management

According to the concept of defense-in-depth, the NPP design provides safety systems intended for fulfillment of the following main safety functions of:

- reactor scram and maintaining it in subcritical state;
- emergency heat removal from the reactor, and also from the spent fuel in cooling pool;
- confinement of radioactive substances within the established limits.

The main safety functions are provided by operation of active and passive safety systems.

The structure and components of active and passive parts of safety systems and systems of beyond design basis accident management are indicated in Table 3-1. Schematic process diagram of connections of the main safety systems is given in Figure 3-1.

Table 3-1 - Structure and components of active and passive parts of safety systems

Safety systems (SS) and systems of BDBA management	Effectiveness			
	Subsystem 1		Subsystem 2	
	Channel 1	Channel 2	Channel 3	Channel 4
1 Active part of safety systems				

Safety systems (SS) and systems of BDBA management	Effectiveness			
	Subsystem 1		Subsystem 2	
	Channel 1	Channel 2	Channel 3	Channel 4
1.1 Protective safety systems				
1.1.1 High-pressure safety injection system	100 %	100 %	100 %	100 %
1.1.2 Safety boron injection system	50 %	50 %	50 %	50 %
1.1.3 Emergency and planned cooling down of primary circuit and fuel pool cooling system	100 %	100 %	100 %	100 %
1.1.4 SG emergency cooldown and blowdown system:	100 %	100 %	100 %	100 %
1.2 Localizing systems				
1.2.1 Sprinkler system	100%	100%	100%	100%
1.2.2 Containment isolation system	100%			
1.3 Supporting safety systems				
1.3.1 Intermediate circuit and service water supply:				
1.3.2 Ventilation	100 %	100%	100 %	100 %
1.3.3 Reliable power supply to valves				
- power supply to valves of SS active part channels;	100 %	100 %	100 %	100 %
- power supply to localizing valves and valves of SS passive part;	200 %		200 %	
1.3.4 Reliable power supply to pumps	100 %	100 %	100 %	100 %
1.4 Control safety systems				

Safety systems (SS) and systems of BDBA management	Effectiveness			
	Subsystem 1		Subsystem 2	
	Channel 1	Channel 2	Channel 3	Channel 4
1.4.1 Reactor trip system	100 %		100 %	
1.4.2 Sensors of control safety system	2×100 %	2×100 %	2×100 %	2×100 %
1.4.3 Control and regulation of safety systems	100 %	100 %	100 %	100 %
2 Passive part of safety systems				
2.1 Protective				
2.1.1 Quick boron injection system	4´25 %			
2.1.2 1-st stage hydroaccumulators	4´33 %			
2.1.3 2-nd stage hydroaccumulators	4´25 %			
2.1.4 Passive heat removal system	4´33 %			

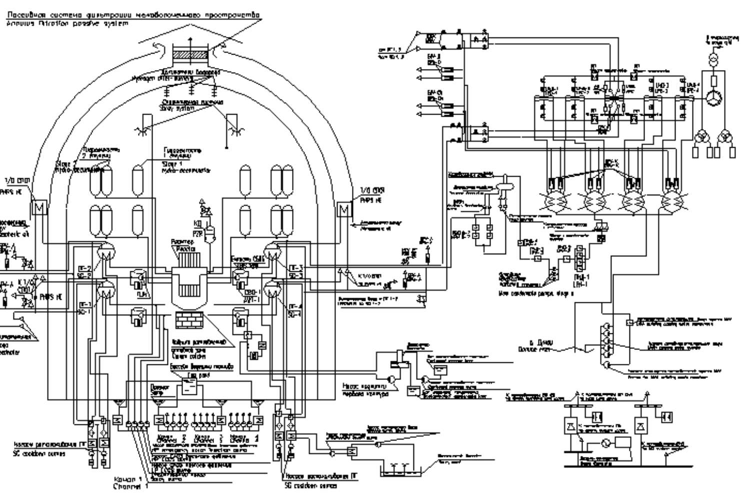

Figure 3 – 1 Schematic process diagram of connections of main safety systems of “Belene” NPP, power Unit

3.4.3 Protective safety systems and systems of beyond design basis accident management

3.4.3.1 Reactor control and protection system

The reactor control and protection system is intended for emergency and preventive protection of the reactor, automatic and manual reactor power control, provision of monitoring of parameters and CR position, documenting of events and interchange of signals with I&C interfaced subsystems. Control and protection system is a special system of the reactor plant and provides maintaining reactor power without violating operational limits under normal operating conditions, reactor power level limitation under anticipated operational occurrences and reactor scram in accident situations and during accidents.

3.4.3.2 Emergency and planned cooling down of primary circuit and fuel pool cooling system

The structure of emergency and planned cooling down of primary circuit and fuel pool cooling system includes four channels, each including: a sump, a heat exchanger of emergency and planned cooling down of primary circuit and fuel pool cooling, the pump of emergency and planned cooling down of primary circuit, storage tank of solution of reagents, pipelines, valves, throttle washers and restrictive insertions in the place of connection to primary pipelines.

The heat exchanger is common for emergency and planned cooling down of primary circuit and fuel pool cooling systems, sprinkler system, high-pressure ECCS.

3.4.3.3 Emergency core cooling system, passive part

The emergency cooling system, passive part, is intended for fast injection of boric acid solution into the reactor for core cooling and its flooding under loss-of-coolant accidents when primary pressure drops below 5.9 MPa and it consists of 4 hydroaccumulators (HA-1) and pipelines with valves.

3.4.3.4 High-pressure safety injection system and safety boron injection system

The high-pressure safety injection system consists of four independent channels. The structure of each channel of the system includes a high-pressure safety injection pump, pipelines, valves. Suction lines of high-pressure safety injection pumps are connected to the pipings of the respective channels of emergency and planned cooling down of primary circuit and fuel pool cooling system downstream of heat exchanger. The heat exchanger is common for emergency and planned cooling down of primary circuit and fuel pool cooling systems, sprinkler system, high-pressure safety injection, safety boron injection system. Water in the refueling pool above the normal level of fuel storage serves as a storage tank of borated water with concentration of 16 g/kg for the above-mentioned systems.

Discharge lines of each channel of the system are connected to "cold" legs of the respective loops of the primary circuit. Isolation valves and check valves are mounted in discharge line of the pump.

The safety boron injection system is designed for fulfillment of the following functions:

- providing core subcriticality under the conditions of failure of reactor trip system together with quick boron injection system (QBIS);
- injection of boron solution into pressurizer under the conditions of primary-to-secondary leaks to reduce pressure.

The safety boron injection system is made up of four-channels. The structure of each channel of the system includes safety boron injection pump, pipings and valves.

Discharge lines of each channel of the system are connected to discharge lines of the respective channels of high-pressure safety injection system just upstream of their joining the RCS pipings of the respective "cold" loops of the primary circuit, the line for coolant injection into steam volume of pressurizer is also provided.

3.4.3.5 Quick boron injection system (QBIS)

The quick boron injection system is a special system for beyond design basis accident management without reactor scram and intended for changing the reactor core into subcritical state by injection of concentrated boric acid solution into the primary circuit under the conditions of reactor trip system failure.

The quick boron injection system consists of four channels independent from one another, each including a tank with boric acid solution, connecting pipings that join the tank with the main coolant pipeline, pipings Dnom 25 and valves.

Channel of quick boron injection system is mounted in RCP set bypass and supplies concentrated boric acid solution into the primary circuit due to pressure differential on the suction side.

3.4.3.6 Emergency gas removal system (EGRS)

The system is intended for removal of a steam-gas mixture from the primary equipment and for pressure decrease under beyond design basis accidents caused by formation of a steam-gas volume above the reactor core, and if necessary, it can be used by the operator under normal operating conditions and design basis accidents.

The system consists of pipings and valves mounted on them for removing steam-gas mixture into the relief tank from reactor top head, from primary collectors of steam generators, from pressurizer.

3.4.3.7 Core passive flooding system

The system is intended for passive supply of boron solution with concentration of 16 g/kg into the reactor core with the purpose of decay heat removal and maintenance of the core in subcritical state in case of the primary leak with blackout, including diesel-generators, during the maximum possible period of time (not less than 24 h considering PHRS operation). The system consists of eight hydroaccumulators (HA-2), combined in four groups. The structure of each group of the system involves two 2nd stage hydroaccumulators (volume 120 m^3, each), pipings and valves. The hydroaccumulators subjected to atmospheric pressure under normal operating conditions, contain boric acid solution with concentration of 16 … 20 g/kg. The total water inventory in tanks is accepted to be 960 m^3 that provides the required amount supplied into the core.

The 2-nd stage hydroaccumulators are connected via discharge line to the pipings of the 1-st stage hydroaccumulators connected to the reactor in the part non-isolated from the reactor, boron solution is supplied from hydroaccumulators into pressure and collection chambers of the reactor.

3.4.3.8 SG emergency cooldown and blowdown system

SG emergency cooldown and blowdown system is intended for:

reactor core decay heat removal and cooldown of the reactor plant in accident situations caused by loss of power or loss of possibility of normal secondary-side heat removal, including leaks of steam lines and feedwater lines of SG;

reactor core decay heat (or portion) removal and cooldown of the reactor plant in accident situations caused by loss of integrity of the primary circuit, including RCS piping break (through the intact loops), primary-to-secondary leaks, inclusive.

Each channel of system includes the pipeline for connection with a steam line of steam generator, a regenerative heat exchanger, an emergency cooldown heat exchanger (process condenser), a pump of emergency cooldown and the pipeline for condensate return into the steam generator, the quick-acting valves providing connection of each channel to steam lines of steam generators in case of the accident initiating event.

3.4.3.9 Primary circuit overpressure protection system

The system consists of three pilot-operated relief valves independent of each other that are connected to pressurizer (PRZ PORV). PRZ PORVs also provide implementation of "feed and bleed" procedure and, together with EGRS, make possible the primary pressure decrease to 1.0 MPa under beyond design basis accidents caused by melt-through of the reactor vessel.

Primary circuit overpressure protection system also fulfils overpressure protection at low temperatures (under the conditions of heating-up and cooling down).

3.4.3.10 Secondary circuit overpressure protection system

The system consists of two pilot-operated relief valves independent of each other and mounted on the main steam lines of each steam generator.

3.4.3.11 Isolation system of main steam lines

The system is designed for quick and reliable isolation of steam generators from the leak under the break of SG steam lines as well as under the break of feedwater lines within the section from SG to the check valve.

3.4.3.12 Passive heat removal system (PHRS)

The system is designed for decay heat removal from the reactor core during the accidents with loss of all a.c. sources of electric power both with the primary circuit sealed, and under occurrence of leaks in the primary or secondary circuits. In case of leaks in the primary circuit the system operates together with ECCS 2nd stage hydroaccumulators after primary pressure drops to the value corresponding to operation of hydroaccumulator.

The system consists of four independent circuits of natural circulation, one per each circulation loop.

Each circuit includes two heat-exchanging modules, piping of steam-condensate system with the valves, air duct system for air supply and removal, air gate valves and regulating devices.

The piping of steam path from a steam line of each SG goes as far as the collector that distributes steam over individual pipings per two heat exchanging modules. Condensate from each heat exchanger comes back into the steam generator via the pipings.

Steam in heat exchangers of the system is cooled by atmospheric air. The cooling air is taken from atmosphere outside the containment building. Air passes through protective meshes due to natural draught and gets into the annular corridor located around the containment building. Then air comes to heat exchanging modules via individual air ducts. A cooling air takes heat from the steam in heat exchangers and goes to draught sections of air ducts that end with a common collector – deflector.

Air gate valves are provided upstream and downstream of each heat exchanger in the direction of the air flow. Between the heat exchanger and the upper air gate valve the regulating device is provided that is intended for regulation of air flowrate in the mode of operation of the system and air gate valves, mounted at the heat exchanging module inlet and outlet, are intended for operation of heat exchanger in case of their opening. In the mode of operation the air gate valves open completely, in standby mode the air gate valves are in closed position to reduce heat losses.

During Unit operation the PHRS is permanently connected to steam generators.

3.4.5. Localizing safety systems

Localizing safety systems (LSS) are the systems intended to prevent or restrict propagation of radioactive substances, released during accidents, and of radiation beyond the limits established in the design as well as prevention of their release into the environment.

3.4.5.1 System of protective enclosures

The system of protective envelopes consists of the primary (internal) and the secondary (external) envelopes. The primary (internal) envelope is made of prestressed concrete and intended for confinement of radioactive substances within the limits established in the design with the aim of limiting their propagation into the environment under design basis accidents. External envelope is intended for protection of systems and components of the reactor building against specific natural and man-induced impacts. Both envelopes provide biological shielding against ionizing radiations.

3.4.5.2 Sprinkler system

Sprinkler system is intended for fulfillment of the function of decreasing parameters (pressure and temperature) in the containment.

3.4.5.3 System for corium retention and cooling

The system is intended for retaining liquid and solid fragments of the damaged core, reactor vessel parts, internals under severe accident with core melting and reactor vessel melting-through.

3.4.5.4 Emergency hydrogen control and emergency hydrogen removal systems in containment

Emergency hydrogen control and emergency hydrogen removal systems in containment fulfill the functions of prevention against formation of fire-explosive mixtures in the area of localization of accidents.

3.4.5.5 Annulus passive filtering system

Annulus passive filtering system is intended for the organized purification of leaks through the NPP internal protective envelope in annulus before their discharge into atmosphere under beyond design basis accidents caused by loss of all a.c. sources of electric power and also under severe accidents with core melting.

3.4.5.6 Containment isolation system

The system is intended for isolation of pipings containing various process media passing through the boundary of the containment to prevent fission products release as a result of loss of coolant accident in the primary circuit.

3.4.6. Supporting safety systems

Supporting safety systems are the systems (components) intended for supply of power, working medium to safety systems and for creation of the conditions for their functioning.

3.5. Safety assurance during earthquakes

All NPP buildings, systems and components have their own seismic category because in case of earthquake all objects are subject to seismic impacts.

Categorization of NPP systems and components is done depending on the degree of their importance in terms of safety assurance under seismic impacts and operability after the earthquake. Three seismic categories are established considering safety class.

The primary equipment and safety systems fall into seismic category I and shall be able to fulfill the functions related to safety assurance during and after earthquake with the intensity up to SSE, inclusive.

Components of seismic category II shall be operable after earthquake with intensity up to OBE, inclusive. NPP systems and their components whose failure could bring about a break in generation of electric energy and heat can be referred to seismic category II.

For the site of "Belene" NPP the following parameters of seismic impact are taken:

For seismic impact of SSE level with the frequency of recurrence every 10000 years, not more:

- maximum horizontal acceleration on a free ground surface amounts to 0.24 g;
- maximum and spectral vertical acceleration of a free ground surface is determined by multiplication of horizontal acceleration by coefficient 0.645.

For seismic impact of OBE level with the frequency of recurrence every 1000 years, not more:

- maximum horizontal acceleration on a free ground surface amounts to 0.15 g.

The automatic reactor trip is provided in the design for the case of earthquake with the intensity of OBE and more.

3.6 Probabilistic risk assessment

The results of probabilistic safety assessment, level I, confirm adequacy of choice of safety systems and their performances for the design of "Belene" NPP and confirm high reliability of safety functions.

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.

5.1. Physical security system (PSS)

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

5.2. Tasks of physical security system

PSS realizes the following tasks:

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

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

5.3. Structure of physical security system

PSS is realized on the basis of a complex automated system of physical security including:

- the complex of engineered-and-alarm control and display system of physical security of buildings and structures;
- the complex of engineered-and-alarm control and display system of physical security of NPP perimeter;
- the complex of engineered-and-alarm control and display system of physical security of company security.

6.1. Turbine

The turbine K-1000-60/3000 is a steam turbine of condensation type, one-shaft, five-cylinder (2 LPC + HPC + 2 LPC) with intermediate separation and steam superheating, of nominal power 1000 MW, rotational speed 50 1/s

(3000 rpm) is used for direct driving of a.c. generator TBB-1000-2У3 mounted on the same foundation with the turbine

Turbine length is ~ 51.45 m.

Total length of turbine with generator is ~ 68.8 m.

Steam under pressure 6.27 MPa is coming to the turbine from steam generators via four steam lines Dnom 600 with stop and control valve on each of them. Downstream of the valves steam is coming to HPC via four steam lines Dnom 600.

Steam is supplied from the HPC via four Dnom 1600 steam lines to four separator-superheaters (SSH). Separation and steam superheating are of single stage. Downstream of the SSH the steam is supplied to each of the four LPCs. The butterfly gates mounted in the path fulfill the functions of stop and control valves.

Steam is dumped from each LPC to its own condenser. The condensers are of surface type, basement-positioned, two-flow and pressure sectioned. Steam receiving devices built in the condenser take the SG steam dumped to condenser via BRU-K under start-up and transient conditions.

The turbine is also equipped with a regenerative plant to heat-up the condensate and feedwater, it can make uncontrolled steam extractions to the heaters of regeneration system and to the station auxiliaries.

Main parameters of the turbine plant

Control range of operation, %	20 - 100
Steam temperature upstream of HPC valves, ^{o}C	274.3
Steam pressure upstream of HPC valves, MPa	5.88
Steam flow to turbine, t/h	5880
Height of operating blades of the last stages, mm	1200
Blade material	titanic alloy

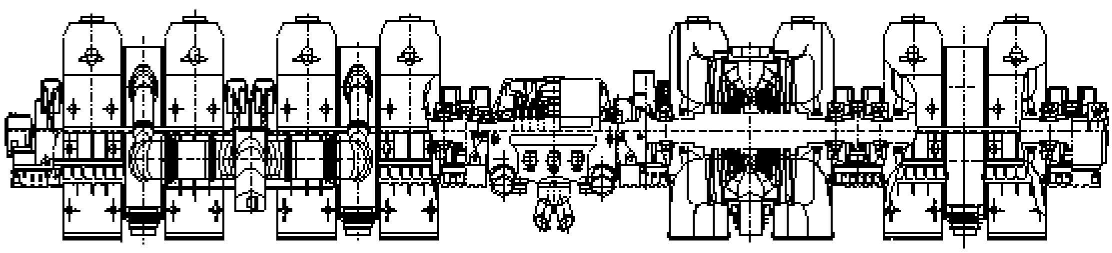

Figure 6-1 Steam turbine K-1000-60/3000

6.2 Generator

The generator of type TBB-1000-2У3 is implicit-pole selsyn three-phase electric machine. It consists of the stationary part (stator) including a core and a winding, connected to the grid, and a rotating part (rotor) with excitation winding powered by rectified current. Rotor winding and stator core are cooled with hydrogen; stator winding is cooled with water.

Main parameters of generator:

full power, MV•A	1111
active power, MW	1000
voltage, kV	24

6.3. Main feedwater system

Main feedwater system comprises:

- main turbine-driven feedwater pump;
- deaerator plant;
- pipelines and valves (isolation, control, safety).).

Feedwater into the steam generators is supplied from deaerator plant whereto condensate is supplied from the turbine.

Two feedwater pumps are connected to the deaerator plant.

Feedwater is supplied from the pressure collector of feedwater pumps through two lines Dnom500 to the turbine high-pressure heaters (HPH). Downstream of HPH the feedwater is supplied to Dnom 500 collector and then to steam generators via four lines Dnom 400.

Main parameters of feedwater pump:

Capacity, m^3/h	3000
Pressure head, MPa	9.1
Pumped liquid temperature, ^oC	172

Deaerator is designed to remove corrosive gases and to heat-up the turbine condensate under nominal, start-up and transient conditions of operation.

Deaeratorparameters:

Capacity, t/h	6000
Operating pressure, MPa, abs.	0.824
Operating temperature, $^{\circ}C$	172
Geometrical volume of tank, m^3	400
Tank useful capacity, m^3	250

6.4. Auxiliary feedwater system

Auxiliary feedwater system is intended for feedwater supply to steam generators under the conditions of start-up (pre-starting hydrotests, heat-up of main steam lines), shutdown and being in the state of "hot" standby.

Auxiliary feedwater system consists of two electrical feedwater pumps, each of 150 m^3/h capacity, valves and pipelines.

7.1. Power supply systems

NPP power output to the grid is realized through switchgear 400 kV. The following is connected to the switchgear:

two turbine generators, 1000 MW each. Connection of generators will be made to 400 kV switchgear for Unit 1 by means of two main step-up transformers 630 MV·A, connected in parallel;

air lines 400 kV;

two transformers 400/110 kV.

The generator and transformers are connected between themselves with shielded wireways with the generator switch provided between themselves, capable to disconnect a current of short circuit. Two 63 MV·A working transformers are connected to the tap between the generator switch and the main transformer.

Two 250 MV·A autotransformers are provided for connection of switchgear-400 kV and switchgear-110 kV.

Lines of 110 kV and two groups of emergency transformers are connected to 110 kV switchgear to meet the demand of the region in the electric power.

Functioning of power supply system to the station auxiliaries is required under all operating conditions of NPP Unit, including loss of power (loss of power from working and emergency transformers) to NPP station auxiliaries.

In case of loss of connection with the grid at voltage 400 and 110 kV and generator switch-off, electric power to critical loads of station auxiliaries is provided from diesel generators, connected to 6 kV switchgears, and from storage batteries.

The system of power supply to the station auxiliaries includes:

power supply system of normal operation for common-station consumers that is provided in the auxiliary building and consists of two 6 kV sections and low-voltage mains for all common-station consumers inside the auxiliary building and other common-station buildings;

power supply system of normal operation that consists of four 6 kV sections of low-voltage mains for all consumers and uninterruptible a.c. power supply system. Auxiliary diesel-generator of 6.3 MW is provided for power supply of 6 kV sections of normal operation under loss-of power.

emergency power supply system (EPSS).

7.2 Safety-related electrical systems

Emergency power supply system is intended to provide the Unit safety system consumers with power supply under all operating conditions, including loss of power to the station auxiliaries from working and standby power supply sources (under loss of power).

According to the design solutions, the process safety systems are divided into four channels that are standby to each other in terms of equipment composition, the similar division into four independent channels is done in the emergency power supply system (EPSS). All four EPSS channels are identical. According to equipment composition and power, each EPSS channel is capable to fulfill its function.

Power supply to EPSS loads is provided from four 10 kV sections of emergency power supply that are fed from respective sections of normal operation under normal condition, and in case of loss of power it is done from respective sections of diesel-generators.

Independent diesel electric power station is provided per each section. Power of one diesel-generator and load ascension are chosen such that to provide 100 % power of the consumers necessary for operation of safety systems under Unit shutdown with no voltage of station service working and emergency transformers.

For power supply to a.c. consumers with 380 V voltage the safety system channels are provided with uninterruptible power supply sets (UPSS) that consist of a rectifier and an inverter.

The following 220 V storage batteries (SB) are provided for the Unit:

- for power supply of NO d.c. consumers – 2 sets;
- for power supply of d.c. loads of EPSS channels– 12 sets (three sets per SS channel).

NO batteries are designed for 120 minutes of operation. Besides, in each SS channel the storage battery, designed for 24 hours of operation, is provided to supply power to reactor monitoring devices.

Independent 110V storage batteries are provided as a standby source for retaining CPS drives in the assigned position in case of short-term voltage drop in station service mains.

7.3. Lay-out of the main control room

Main control room is the center of control and process monitoring for Unit operator using I&C equipment. It is arranged in the control building.

The main control room is divided into several working areas. It enables a clear separation between the process control panels and the panels of other functions needed for NPP operation.

Operators and shift supervisor perform monitoring of all the events with the help of the displays of top-level system of the Unit (TLS-U).

The system consists of a set of processors making processing of the signals dealing with the man-machine interface, as well as a short-term data recording. The server saves data for the TLS-U OM and a number of operator stations are designed to display all the monitored functions in the VDUs. Each monitor is capable of performing all the I&C functions (control, emergency signal presentation, information display, archiving and digital registration). The terminal bus provides communication between the processors and the operator terminals.

In case of permissible loss of TLS-U system, the information necessary and important to safety is displayed on control boards or panels. On the basis of the information presented the operator can maintain the plant under steady-state operating conditions during a certain period of time or shutdown the reactor and change it over into the safe state. Control desks and control panels are mainly used to perform this task. The system provides the operator with all the information to monitor the systems important to safety and to make the manual process control. In case it is impossible to realize monitoring with the TLS-U system the mosaic panels provide basic information to exercise control with traditional hardware in the main control room with indicators, recorders, signaling system elements and control keys in the mosaic panel in the scope sufficient to implement the safety functions, as a minimum. MCR is also furnished with back-up panel for control of safety systems in accident situations.

7.4 Emergency protection system and other safety systems

Control and protection system (CPS) is intended for monitoring the reactor parameters, control of reactor power, including planned shutdown and scram, maintaining the reactor in subcritical state.

Control and protection system consists of the following components:

- subsystem of emergency and preventive protection of reactor (EP-PP) including initiating and actuating parts;
- rod control and indication system (RCIS) of CPS CR drives;
- automatic power controller (APC).

EP-PP subsystem of reactor includes an initiating part as a part of NFME, industrial aseismic protection system (IAPS), the equipment of acquisition and processing of the information on the basis of Teleperm XS hard-and-software, and actuating part, representing power supply interruption system of CPS CR drives.

The initiating part is realized as four sets placed in the rooms of control safety system (CSS) channels.

For protection against common-cause failures in case of software failure in initiating part for fulfillment of EP functions, the principle of functional diversity is realized that involves availability of two subchannels (of diversity A and B) in each channel, and also the variety of the equipment involved in possible use of primary transducers, based on various physical principles of measurement or usage of sensors of different companies.

RCIS is intended for group and individual control of CR drive, automatically or in response to operator's commands from MCR, and also for monitoring and displaying the information in MCR and ECR about CRs position. With this, diagnostics of the system and the controlled equipment is provided.

The RP incorporates monitoring, control and diagnostics system (MCDS). It fulfils the following tasks:

- monitoring of neutronics and thermal-hydraulic characteristics of the core, of the primary and secondary circuits at power operation of the Unit under base and load-follow modes, and presentation of information on the core state to the operator;
- generation and transfer of emergency and preventive protection signals on local core parameters to CPS (fuel rod linear heat rate, DNBR);
- generation and transfer of signals for automatic control of the core power field to CPS;
- in-service diagnostics of the RP main process equipment regarding monitoring of mechanical integrity and reliability of fastening of the equipment components, RCS integrity, assessment of residual service life.

The following subsystems are provided in MCDS structure to cope with its tasks:

- in-core instrumentation subsystem (ICIS);
- primary and secondary coolant leak detection subsystem (CLDS);
- vibration diagnostics subsystem (VDS);
- RCS loose parts monitoring subsystem (LPMS);
- system of integrated analysis (SIA).

I&C block-diagram, CPS and MCDS detailed, is given in Figure 7-2.

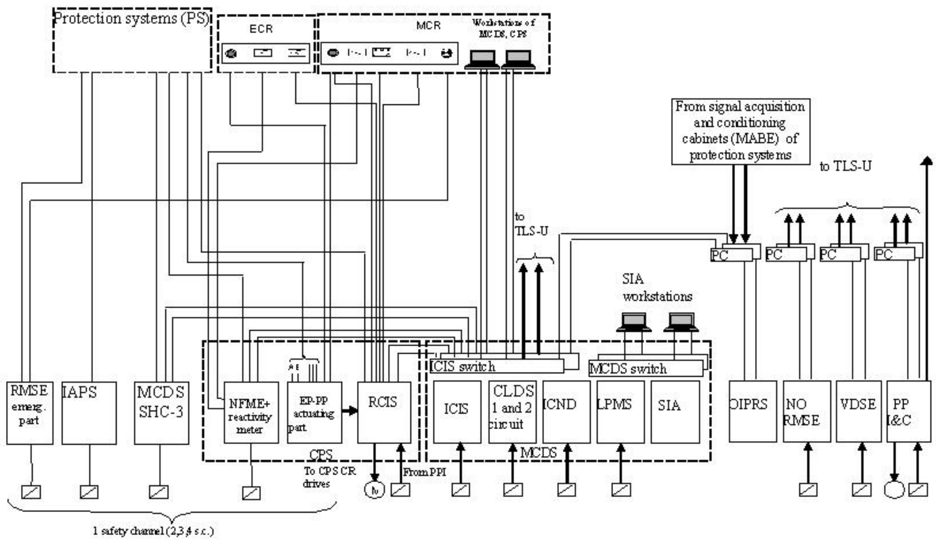

Figure 7 – 2 I&C block-diagram, CPS and MCDS detailed

8.1 Spent Fuel Pool

The spent fuel is unloaded from the reactor and placed for storage in spent fuel pool (SFP) located in the reactor compartment within the containment near the reactor core barrel.
Besides, inspection of failed fuel elements is provided in the spent fuel pool and a separate area for storage of leaky fuel assemblies in tight bottles is arranged.

Lay-out of spent fuel pool excludes necessity of transfer of transport containers during fuel transfer above the stored spent nuclear fuel and other cargoes from the reactor building. For transfer of transport containers with a fuel the polar crane is used installed in the containment building.

During reactor refuelling the fuel stored in spent fuel pool is transferred into the spent nuclear fuel storehouse (SNFS).
SNFS is intended for dry storage of spent nuclear fuel (SNF) at NPP site in the dual-purpose containers intended for transportation and storage.

SNFS capacity is designed for a long-term storage of the spent nuclear fuel accumulated for 10 years of operation of two Units with a possibility for the building capacity to be extended in future for fuel storage accumulated for all operating time of the Units.

8.2 System for liquid radwaste handling

The system for liquid radwaste handling includes systems for temporary storage, reprocessing and solidification of liquid radwaste (LRW).

The system for LRW handling is designed so that to provide receiving, temporary storage and reprocessing of LRW in full scope.

The liquid radioactive media, subjected to temporary storage, present ion-exchanging resins (filtering materials), salt concentrate (vat residue), sludge.

To receive LRW in the full volume and to provide their intermediate storage for decay of short-lived radionuclides before further reprocessing, the tanks of LRW intermediate storage are provided.

Media from tanks of intermediate storage are transferred to the system of processing and solidification.

Liquid radioactive media are solidified by inclusion of the spent filtering materials, of the solutions, formed during reprocessing of sump water, and of sludge into cement mass in grouting plant.

8.3 Gaseous waste

The system for gaseous radwaste handling includes systems of special ventilation and collection and purification systems of gaseous radwaste.

In order to reduce the releases of radioactive substances into the environment and their uncontrolled propagation over the station, the design provides for the purification systems as a part of ventilation systems for radioactive substances released into the air of process rooms during normal operation and accidents at the Unit.

The system of collection and purification of gaseous radwaste consists of two subsystems:

- system of combustion of hydrogen from radioactive process blow-offs;
- system of purification of radioactive process blow-offs.

8.4 Reprocessing system of solid radwaste

The system of reprocessing of solid radwaste is intended for reprocessing solid radwaste and also for temporary storage of solid radwaste and solidified liquid radioactive media.

Solid radwaste (SRW) are formed both during normal NPP operation and during carrying out repair work and also during the accidents.

Solid radwaste of middle and low activity are sorted in the places of formation into groups depending on contamination and composition and they are ground, if necessary. Then SRW are packed into barrels or collecting containers and transported into reprocessing and storage building (RSB) of solid radwaste.

Solid middle-and-low level radwaste are subject to grinding, pressing and burning, burning at plasma facility to reduce their volume transferred to the storehouses

RP high-activity solid radwaste (detection units, ICID, RCCA and waste from cutting of surveillance specimens) are transported into RSB in special containers.

The volume of radwaste after reprocessing, and of those not subject to reprocessing, with the exception of the waste formed as a result of repair and maintenance, does not exceed 50 m^3 per Unit annually. There is a problem of reducing radwaste and dose commitment when handling.

9.1. Buildings and structures

The design provides the NPP complex incorporating a set of buildings and structures being a permanent unchangeable part that forms actually the Unit (depending on the number of Units) and a set of site-specific common-station structures.

The site of "Belene" NPP arranges two Units whose unchangeable part involves a reactor building, a turbine building, an auxiliary reactor building and four buildings of emergency electric supply and control safety systems, being doubled.

The central place at the site incorporates the doubled turbine buildings and the reactor building arranged in series along the longitudinal axis oriented from the north to the south.

The main buildings of free and controlled access areas are connected with sanitary-household buildings with the help of foot galleries and, further, the gallery of free access area is going as far as the complex of administrative buildings with the canteen.

The solid waste reprocessing and storage building of the controlled access area, the shops of the controlled access area, sanitary-household building of the controlled access area, engineering-laboratory building, building of chemical water treatment, building of auxiliary systems are located to the west of the main buildings.

The buildings of emergency electric supply and control safety systems, pump stations of essential consumers, fresh fuel storehouse, spent fuel storehouse, spraying pools, emergency control room building are located to the south of the main buildings.

Water intake constructions, common-station diesel-generator building, heat distribution point, building of 400 kV switchgear, purification plant, storehouse and other auxiliary constructions, main control room building are located in the north of the site.

Reactor building consists of a double containment that houses an accident confinement area (ACA). Annexes adjoin the external envelope on both sides. All constructions rest on the common base part with size 42´72 m. General view of the main building is given in Figure 9-1.

Containment is a component of accident localization system and consists of two reinforced concrete cylindrical envelopes with a hemispherical dome: internal envelope of prestressed reinforced concrete, designed for emergency pressure and temperature, and external envelope protecting from external impacts.

Reactor occupies a central place in accident confinement area. The lower part of the reactor concrete cavity arranges a trap for retaining fuel melt. There is a spent fuel pool and inspection well of the internals, two RCS compartments with steam generators, RCP sets, main coolant pipelines, pressurizer, relief tank and tanks of quick boron injection are located on different sides of the reactor concrete cavity.

9.2. Containment

Taking into account safety requirements, the design provides for a double containment, external envelope is of non-stressed reinforced concrete and internal envelope is of prestressed reinforced concrete.

External envelope of cast-in-situ reinforced concrete presents the cylinder with internal diameter 50,8 m, overlapped with a dome as a hemisphere. In the places not closed with adjoining civil structures, the external envelope has thickness of 1500 mm. External envelope is subject to loads from external impacts: tornados, external shock wave, crash of aircraft and ventilation tube, extreme wind and temperature impacts, seismic impact of SSE level.

The space between external and internal envelopes is 2.2 m considering requirements for maintenance of the prestressing system of internal envelope, equipment available in the annulus and for possibility of examination of envelope surfaces.

The internal envelope is made of prestressed reinforced concrete cylinder overlapped with a dome as a hemisphere, lined with carbon steel on the internal side.

The basic geometrical dimensions of the envelope are governed by lay-out of the equipment inside the sealed volume. The basic dimensions are given below:

internal diameter of the cylinder and dome - 44.0 m;

height of the cylindrical part - 38.5 m

total envelope height - 61.7 m

thickness of walls and dome, proceeding from design requirements, as well as biological shielding requirements - 1.2 m

For prestressing the orthogonal scheme of arrangement of prestressed bundles is used. Horizontal prestressed bundles are arranged in the cylindrical part of the envelope and in the lower part of the dome. For vertical prestressing of the cylinder and dome the U- bundles are used that cover the envelope in two mutually perpendicular directions.

The containment provides three tight locks:

- transport lock for fuel and the large-sized equipment with entrance to maintenance elevation of the reactor hall +31.700 m;
- basic operational lock for personnel and small cargoes to pass. It is arranged on the opposite side of the containment in respect to transport lock at the axis elevation level +28.250 m;
- emergency lock is arranged under transport lock with axis elevation +21.280 m.

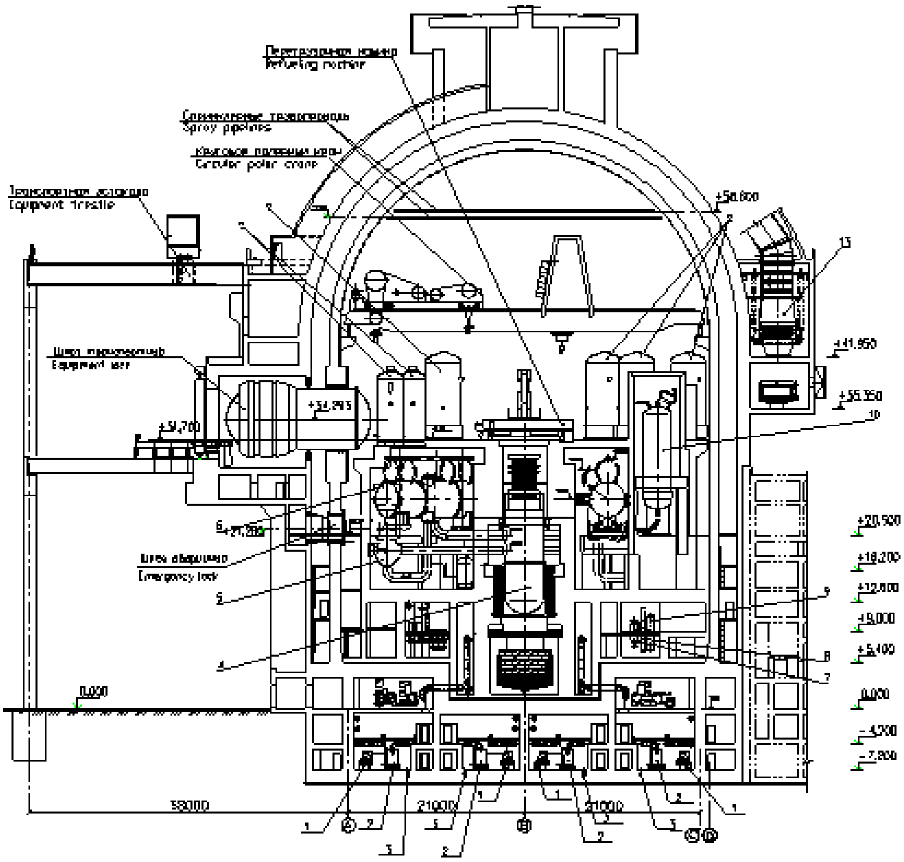

Figure 9-1 Section of the main building of the Unit

1 –high-pressure safety injection pump, 2 –pump of emergency and planned cooling down of primary circuit, 3 – submerged pump of water pumping from the sump tanks, 4 –reactor, 5 – reactor coolant pump, 6 – steam generator,

7 – aftercooler of blowdown water of the primary circuit, 8 – heat exchanger of identified leaks, 9 – regenerative heat exchanger of blowdown of the primary circuit, 10 – pressurizer, 11 – 1st stage ECCS hydroaccumulator, 12 – 2nd stage ECCS hydroaccumulator, 13 – PHRS heat exchanger.

10.1. RP operation

Units are operated according to the requirements of regulatory documents, process specifications and operating manuals. RP operation is possible both in base and load-follow modes.

10.2. Reliability objectives

Reliability indices:

- frequency of scrams not more that once a year;
- availability factor, not less than 0.9.

The given values are confirmed within the framework of analyses of reliability and readiness of the equipment performed in basic design.

10.3. Load factor indices (objectives)

Capacity factor is 0.92.

Annual load factor is 0.9.

The given indices are achieved because of improvement in the design of some RP equipment, optimization of repair cycles of some equipment and RP (reduction to uniform 8-year cycle), optimization of the schedule for each shutdown, implementation of progressive system of preparation and arrangement of repair and maintenance (R&M), use of the refueling machine operating at higher velocities of motion, automatic multiposition power nut-drivers for simultaneous elongation of studs of the equipment flange joints (of the reactor main joint, steam generator collectors, all RP tanks, etc.), loading of fresh fuel (FA, RCCA) in the spent fuel pool during RP primary circuit cooldown, combining refueling work in the reactor with steam generator R&M work owing to usage of the isolation devices placed in steam generator collectors etc..

10.4. Provision of reduction in capital, construction and fuel costs

The following basic components were taken into account when making an assessment and optimization of electric power generation cost (production costs):

- capital costs;
- operation and maintenance costs;
- fuel costs (BOC and EOC);
- decommissioning costs.

When developing the design of "Belene" NPP a special attention was drawn to the problem of capital costs reduction. Solution of this problem is made in the following basic directions:

- reduction in labour input for designing;
- reduction in R&D scope for design verification;
- cutting down the construction time;
- arrangement of construction with maximum use of construction and industrial capacities, raw-materials and labour resources of the region;
- space-saving arrangement of buildings and structures in the territory of the project and of process systems and equipment inside rooms that provides for space-saving of the areas;
- reduction in specific consumption of materials.

Design solutions are optimized so that to reduce fuel costs and thereby to improve performances of the project. Studies are conducted in the following trends:

- improvements of fuel cycle performances. It concerns the problems of fuel enrichment, increase in fuel burn-up fraction, decrease in neutron escape, and also optimum fuel arrangement in fuel assembly;
- optimization of fuel cycle and cycle length (both in base load and in load-follow modes);
- increase in reactor plant efficiency;
- optimization of performances of passive safety systems and the systems of beyond design basis accident management.

Design solutions are improved so that there would be possibility to reduce operational costs and, thereby, to enhance competitiveness of the project. Solution of this problem is made both technically, and economically in the following basic directions:

- reduction in depreciation deductions owing to extension of equipment service life, reliability improvement, rejection of in-service repair or bringing the number of repairs to minimum, etc.;
- reduction in operational costs in respect to RW and SNF handling;
- reduction in the resources consumed and, first of all, of energy-and-water use;
- reduction in chemical and other industrial waste products;
- reduction in station service power consumption;
- heat saving;
- reduction in the number of the personnel and reduction in annual dose commitments;
- reaching the maximum possible NPP service life that will enable to reduce the current allocations to the fund of project decommissioning.

10.5. Construction schedule

Planned duration of construction of the standard Unit (from the "first concrete" till provisional take-over) does not exceed 60 months with a 12 month interval for commissioning the next Unit on the specific site.

11.1. Information on R&D status

For justification of RP and "Belene" NPP designs, R&D is carried out referring to licensing stages of the Unit.

In carrying out expert's evaluation of compliance of designs RP V-392 and AES-92 with the EUR requirements, incomplete R&D are revealed. The necessity of their completion is governed by the ToR requirements for designing and construction of "Belene" NPP.

They are, first of all, R&D for justification of performances of passive safety systems and control systems of beyond design basis accidents not having reference prototypes at operating NPPs, efforts for justification of design margins that could be used by personnel during NPP operation and other kinds of work.

11.2. Companies/institutes involved in R&D and project elaboration

The following companies take part in elaboration of conceptual and basic design and Interim Safety Analysis Report (ISAR):

- JSC " Atomenergoprojekt" - Organization of General Designer of "Belene" NPP;
- OKB "Gidropress" - Organization of RP General Designer;
- Organization of Scientific Leader of the design – RRC " Kurchatov Institute ";
- JSC TVEL;
- Framatome ANP;
- JSC "Silovye Mashiny".

11.3. Provisional assessment of project implementation period

According to the accepted concept of elaboration of the Project schedule, construction period of the Unit from the "first concrete" till provisional take-over is 60 months, stage of commissioning of the Unit is 12 months. The first Unit of "Belene" NPP is planned to put into trial operation in 2015.

11.4. Information on the basic R&D, stages of licensing and their duration

The design of "Belene" NPP was developed in several stages:

- in 2007 at the first stage, NPP conceptual design was developed and approved by National Electric Company EAD;
- in 2009 at the second stage, NPP basic design and Interim Safety Analysis Report (ISAR) were developed.
- incomplete R&D are referenced to the next stages of licensing.

At the beginning of 2010, the NPP basic design and the ISAR were approved by the Bulgarian Customer.

Technical data

General plant data

Reactor thermal output	3000 MWth
Power plant output, gross	1060 MWe
Power plant output, net	1011 MWe
Power plant efficiency, net	33.7 %
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
Non-electric applications	District heat

Safety goals

Core damage frequency <	6.1E-7 /Reactor-Year
Large early release frequency <	1.77E-8 /Reactor-Year
Occupational radiation exposure <	0.5 Person-Sv/RY
Operator Action Time	6 Hours

Nuclear steam supply system

Steam flow rate at nominal conditions	5880 Kg/s
Steam pressure	6.27 MPa(a)
Steam temperature	278.5 °C
Feedwater flow rate at nominal conditions	6000 Kg/s
Feedwater temperature	220 °C

Reactor coolant system

Primary coolant flow rate	23888 Kg/s
Reactor operating pressure	15.7 MPa(a)
Core coolant inlet temperature	291 °C
Core coolant outlet temperature	321 °C
Mean temperature rise across core	30 °C

Reactor core

Active core height	3530 m
Equivalent core diameter	3.16 m
Average linear heat rate	15.73 KW/m
Average fuel power density	108 KW/KgU

Fuel material	UO_2 and $UO_2 + Gd_2O_3$
Cladding material	Alloy E-110
Outer diameter of fuel rods	9.1 mm
Enrichment of reload fuel at equilibrium core	4.45 Weight %
Fuel cycle length	18 Months
Average discharge burnup of fuel	52.8 MWd/Kg
Burnable absorber (strategy/material)	Gd_2O_3
Control rod absorber material	Dysprosium titanate, Boron carbide
Soluble neutron absorber	H_3BO_3

Reactor pressure vessel

Inner diameter of cylindrical shell	4195 mm
Wall thickness of cylindrical shell	195 mm
Design pressure	17.64 MPa(a)
Design temperature	350 °C
Base material	Steel 15H2NMFA
Total height, inside	11235 mm
Transport weight	322 t

Steam generator or Heat Exchanger

Type	Horizontal
Number	4
Number of heat exchanger tubes	10978
Tube outside diameter	16 mm
Tube material	Steel 08X18H10T У
Transport weight	330 t

Reactor coolant pump (Primary circulation System)

Pump Type	GCNA - 1391
Number of pumps	4
Pump speed	1000 rpm

Pressurizer

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

Primary containment

Type	Sealed envelope of prestressed reinforced concrete with lining
Overall form (spherical/cylindrical)	Cylindrical part and hemispherical dome
Dimensions - height	38.5 m
Design pressure	0.4 MPa
Design temperature	150 °C
Design leakage rate	0.3 Volume % /day

Residual heat removal systems

Active/passive systems	Active and passive systems

Safety injection systems

Active/passive systems	Active and Passive

Turbine

Type of turbines	K-1000-60/3000
Number of turbine sections per unit (e.g. HP/MP/LP)	2LPC+HPC+2 LPC
Turbine speed	3000 rpm
HP turbine inlet pressure	5.88 MPa(a)
HP turbine inlet temperature	274.3 °C

Generator

Type	ТВВ-1000-2У3
Rated power	1000 MVA
Active power	1110 MW
Voltage	24 kV

Frequency	50 Hz

Condenser

Type	Recuperative
Condenser pressure	3.4 kPa

Feedwater pumps

Type	TBN Driven
Number	2
Head at rated conditions	800 m
Flow at rated conditions	3000 m^3/s