CONSORTIUM

DICON – ACCIONA ING.

ENVIRONMENTAL IMPACT ASSESSMENT REPORT

for Investment Proposal:

BUILDING A NEW NUCLEAR UNIT OF THE LATEST GENERATION AT THE KOZLODUY NPP SITE

CHAPTER 2: ALTERNATIVE LOCATIONS AS ASSESSED BY THE CLIENT (WITH SKETCH AND COORDINATES OF THE CHARACTERISTIC POINTS IN THE APPROVED COORDINATE SYSTEM FOR THE COUNTRY) AND/OR ALTERNATIVE TECHNOLOGIES AND REASONS FOR THE CHOICE MADE FOR STUDY, WITH REGARD TO THE ENVIRONMENTAL IMPACT, INCLUDING ZERO ALTERNATIVE

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2.1 ALTERNATIVES IN TERMS OF LOCATION

In accordance with the Terms of Reference, the assessment covers four alternative locations (Error! Reference source not found.**)**.

Site 1 is situated northeast of NPP Kozloduy Units 1 and 2 between the Outdoor Switchgears and locality Valyata, near the existing warm and cold channels (northward). The surface area of the site is approx. 55 ha. The land is flat, slightly graded from southwest to northeast. Usage of this site will require embankments to elevate the ground level. Certain draining channels cross the site and will need to be reconstructed. Part of the site is used for growing agricultural crops.

Site 2 is situated east of NPP Kozloduy Units 1 and 2 in the direction of v. Hurlets. The surface area of the site is approx. 55 ha. The land is hilly, with significant inclination from south to north, more expressed in the southeast part of the site. Usage of this site will require earth excavation works. The site is crossed by open-air draining channels and overhead power lines, which will have to be reconstructed. The site is used for growing agricultural crops.

Site 3 is situated north-northwest of NPP Kozloduy Units 5 and 6 near the bypass road of the plant. The surface area of the site is approx. 53 ha. The land is hilly, slightly graded from south to north, more expressed in the southeast part of the site. Usage of this site will require embankments to elevate the ground level. The site is crossed by open-air draining channels and overhead power lines, which will have to be reconstructed. The site is used for growing agricultural crops.

Site 4 is situated west of NPP Kozloduy Units 3 and 4 and the NPP Spent Fuel Storage Facility, north of the cold and hot channels. The available space is approx. 21 ha within the lands acquired by NPP Kozloduy. The terrain encompasses existing service facilities – Equipment Storage Facility, Vehicle Maintenance Workshop and Assembly facility. Usage of this site will require reconstruction and relocation of major underground utilities and service buildings of NPP Kozloduy.

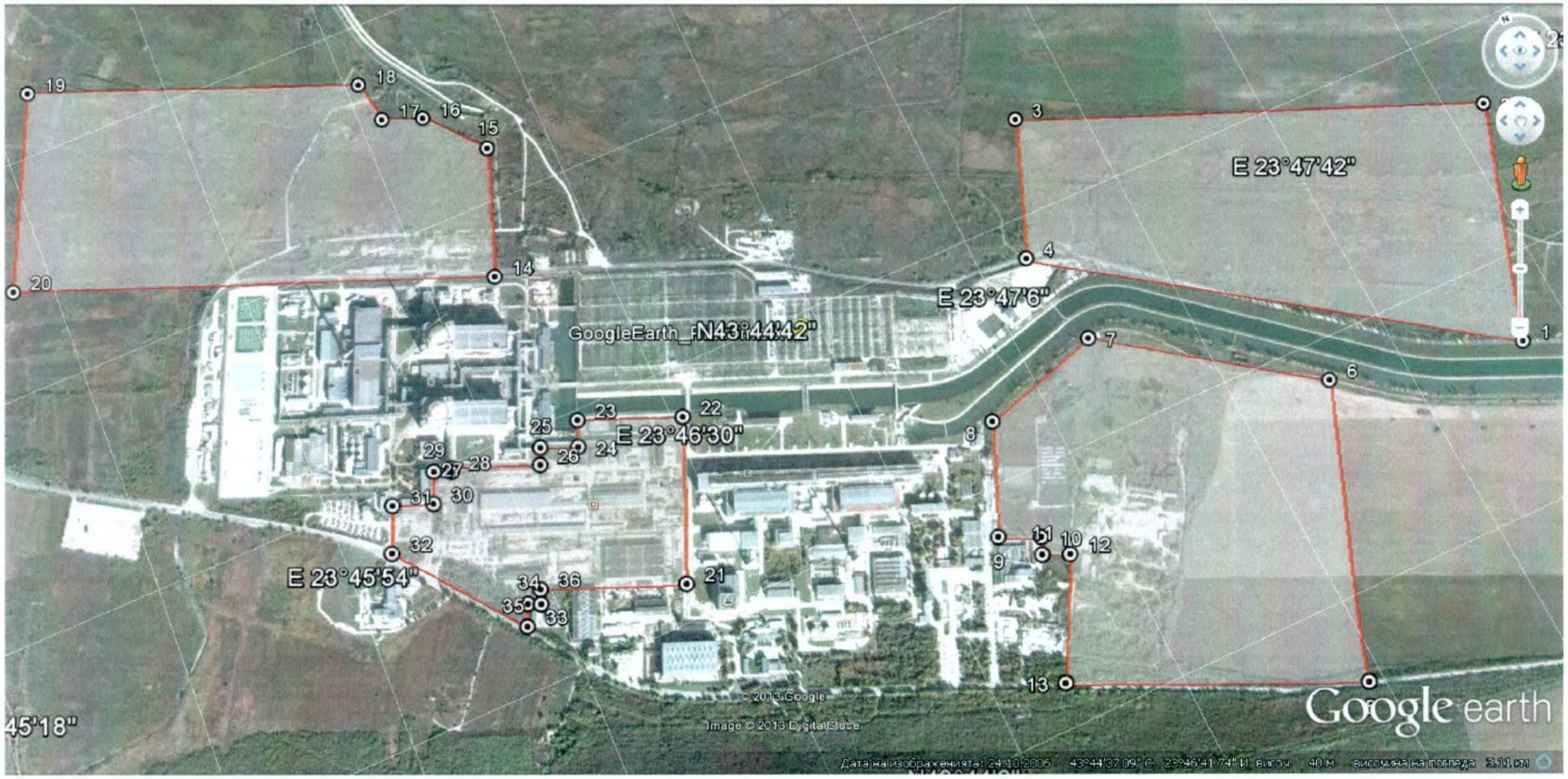


FIGURE 2.1-1: AERIAL VIEW OF THE FOUR ALTERNATIVE SITES PROPOSED FOR CONSTRUCTION OF THE NNU

2.2 ALTERNATIVES FOR ASSOCIATED INFRASTRUCTURE DURING THE CONSTRUCTION AND OPERATIONS PHASE

The core construction works at the proposed site should be preceded by application of methods for improvement of the ground base to ensure that its load bearing capacity will be adequate to the loads and that any settling of earth will be within the permissible limits.

The proposed site will have to be pre-developed with temporary facilities such as storage for bulk materials, prefabricated steel, concrete, ferroconcrete, metal and other structural elements, fuels and lubricants, on-site offices, on-site accommodation for workforce permanently stationed at the work site, on-site sanitation and medical facilities, freshwater pipelines and sewers for discharging of gray and black water, as well as rainwater drains and a dewatering system. Each of the alternative locations is large enough for setting up a construction site. In the case of Site 4 the design as such should take into account the space proposed by the Customer for temporary facilities as they will have to be deployed outside the area in which the new unit will be built.

The vertical planning of the proposed site will be aligned to the working level of the existing site of the Plant, which is +35.00 m by the Baltic System. This is predetermined by the fact that the utilities will have to be connected to the existing Cold Channel (CC) and Hot Channel (HC). Thus for example, if Site 1 or 3 is selected, the preparatory works should include reconstruction or relocation of the draining channels that cross these sites, or, if Site 4 is selected, demolition and relocation of existing service buildings to another area. Selection of Site 3 will also require relocation of the fan the 400 kV Overhead Power Line (OPL).

All sites can be supplied with drinking water from the existing pipeline system of the Plant.

Access of road vehicles can be provided to all sites by building branches from the existing road infrastructure.

Liquid radioactive waste that will be generated during the operation of the power unit from the primary circuit as a result of equipment leakages, by the equipment decontamination and ion-exchange filters generation and flushing facilities, the laundry for special purpose clothing and the Sanitary loops, the radio-chemical laboratories, etc., will be treated within the territory of the respective site in accordance with the requirements of the *Regulation for safe management of radioactive waste*.

According to the EUR requirements, the solid radioactive waste generated during operation, including conditioned liquid RAW, must mot exceed 50 m^3 per 1000 MW of installed capacity on annual basis.

The solid RAW that will be generated will belong mainly to Category 1 and 2a.

The RAW management activities will be carried out on the basis of established administrative structures, having defined statuses, between the NNU operator and

Radioactive Waste State-Owned Company with clearly defined functions, tasks and allocations of rights, obligations and responsibilities.

Certain specific technical measures in respect of Outdoor Switchgear 400 kV will be necessary only if the choice is for Site 3 northwest of NPP Kozloduy Units 5 and 6 near the bypass road of the existing Plant. In terms of engineering utilisation and connectivity to the national power grid this option would require many activities and complicated reconstructions of the 400 kV OPL.

2.2.1 Possible layouts of the electrical systems

Under all alternatives, the generated electricity must be fed to the national power system of Bulgaria in accordance with *Regulation no. 4 of 21.05.2001 on the scope and contents of investment projects* and *Regulation no. 6 of 9.06. 2005 on the connection of electric power producers and users to the transmission grid and to the distribution grids* such as to satisfy the *Rules for management of the power system* issued on the grounds of Art. 21(1)(7) of the *Energy Act* by the Chairman of the State Commission of Energy and Water Regulation, being an Annex to item 1 of Decision no. P-5 of 18.05.2007, published in State Gazette (SG) no. 68 of 21.08.2007.

- *Connection to the power system of the Republic of Bulgaria* – the connection to the national power system will be one dedicated Overhead Power Line (OPL) 400 kV to the 400 kV Outdoor Switchgear of NPP Kozloduy, which at present is connected to the national power system by eight 400 kV OPLs (including two intersystem lines) and one automatic transformer 400/220 kV. Backup houseload power will be provided by one 220 kV OPL, also from the 220kV Outdoor Switchgear of NPP Kozloduy. In the event of external and internal failures of the power system the so constructed backup power supply will minimise the disturbance of the normal operation of the reactor. The heat removal system and the other loads that are important to the operation of the power plant will be provided with power from two independent sources (own generator and the electric power grid).
- *Working power supply* – Service transformers for houseload will be the source of working power to each generation unit. The service transformers will be connected by a branch between the generator breaker and the house step-up transformer.
- *Backup power supply* – groups of two backup transformers will be designed as sources of backup houseload power. These transformers will receive power from the 220 kV Outdoor Switchgear of NPP Kozloduy. The backup sources will be used in rated and emergency operating modes and also in emergencies caused by partial or complete loss of working power.
- *Emergency power* – secure power supply systems will be designed for providing power to the systems that are important to nuclear safety. The emergency systems will be activated automatically by connection to emergency power sources and/or rechargeable batteries.

The options for connecting the alternative sites to the Outdoor Switchgear are analysed in **Table 2.2-1**.

TABLE 2.2-1: ANALYSIS OF THE SITES IN RESPECT TO THE OUTDOOR SWITCHGEARS

	Site 1	Site 2	Site 3	Site 4
Position of the lines between the site and the Outdoor Switchgear	Connection with the Outdoor Switchgear is difficult because it is behind the site. This will make it necessary to create a course for the incoming OPLs in the Outdoor Switchgear, opposite to the access fields, and even modifications in the Outdoor Switchgear such as installation of new fields for better access to the 400 kV bus.	Much easier and shorter electrical access to the Outdoor Switchgear compared to Sites 1 and 3, and less conflicts with infrastructure. The electrical connection to the Outdoor Switchgear is easy because the distance is short and the site is on the front side, so it will not be necessary to modify the fields of the Outdoor Switchgear.	This site has the most complicated access by overhead power lines to the Outdoor Switchgear, because it will intersect the OPLs of Units 5 and 6. The electrical connection to the Outdoor Switchgear is difficult, because the site is on the back side and this will make it necessary to create a course for the incoming OPLs in the Outdoor Switchgear, opposite to the access fields, and even modifications in the Outdoor Switchgear such as installation of new fields for better access to the 400 kV bus.	Much easier and shorter electrical access to the Outdoor Switchgear compared to Sites 1 and 3. The electrical connection to the Outdoor Switchgear is easy because the distance is short and the site is on the front side, so it will not be necessary to modify the fields of the Outdoor Switchgear.
OPLs at the site	This site has the least intersections; no overhead lines at the site.	2x 110 kV OPLs and 1x 20 kV OPL.	This site is crossed by five OPLs 400 kV and one OPL 220 kV.	There are no overhead connecting lines at this site.

2.2.2 SUPPLY OF SERVICE WATER

The following alternatives are available for supply of service water:[1]

2.2.2.1 SITE 1

A Service water supply facilities

The on-site facilities for supply of fresh service water from the Danube and for discharging the hot processed (used) water are the following ones:

- → Connections to the channels which supply cold water and discharge the hot water;
- → Fore chamber;
- → Circulation Pump Station (CPS);
- → Power control building;
- → Pressurized pipelines;
- → Filter house;
- → Low-pressure channels.

The NNU can be connected to CC-1 and HC-1 at this site by means of a branch from the cold channel and a new fore chamber for circulation water. The distance to CC-1 is about 60 m. The connection to the hot channel will be difficult, but still possible. The distance to HC-1 is about 120 meters. Another arrangement includes a connecting channel and a bypass channel for cold water, and low-pressure and open-air channel for hot water. The bypass channel will be needed in case the connection will be built while the channels are in operation, which would make it necessary to lower their water level there during pitch-in.

B Auxiliary facilities

Generally, the auxiliary facilities will be used to back up and supplement the conventional stock required for rated and design basis operating modes, and for protection of the environment from various wastes generated by the processes, and for supplying the required amounts of drinking water.

These facilities will include:

- → Water pretreatment plant – Chemical water treatment process;

[1] Information provided by the Client with letter no. 416 of 13.05.201; Protocol of delivery and acceptance no. 31 of 13.05.2013.

- → Treatment plants for grey and black water from “controlled” and “clean” areas, and for industrial waste water, as well as various local treatment installations;
- → Diesel fuel storage and diesel fuel supply;
- → Other general-purpose and auxiliary facilities;
- → Connection to the existing drinking water supply system will be possible at a suitable location;
- → New, separated sewerage system discharging the wastewater from the site in the receiving water body, namely Danube river, after treatment and upon obtaining of Discharge Permit for the IP issued in accordance with the Waters Act.

2.2.2.2 SITE 2

A Service water supply facilities.

The on-site facilities for supply of fresh service water from the Danube and for discharging the hot processed (used) water are the following ones:

- → Connections to the channels which supply cold water and discharge the hot water;
- → Fore chamber;
- → CPS;
- → Power control building;
- → Pressurized pipelines;
- → Filter house;
- → Low-pressure channels.

The NNU connection to the service water facilities (CC-1 and HC-1) at this site will not be substantially different from the connection at Site 1, the difference being that a cold water sag pipe (Düker) from CC-1 will be built or CPS-1 will be used, and a bypass channel for hot water from HC-1 will be built. It will be possible to build the connection during operation of dual channel by lowering the water levels in the channels during pitch-in. The distance to CC-1 is 75 m. The other option for connection to CC-1, namely via CPS-1, will increase the length of the feeding pipelines. The detailed technical solution will be developed during the next design phase, if this site is selected for the NNU.

B Auxiliary facilities

Generally, the auxiliary facilities will be used to back up and supplement the conventional stock required for rated and design basis operating modes, and for protection of the environment from various wastes generated by the processes, and for supplying the required amounts of drinking water.

These facilities will include:

- → Water pretreatment plant – Chemical water treatment process;
- → Treatment plants for grey and black water from "controlled" and "clean" areas, and for industrial waste water, as well as various local treatment installations;
- → Diesel fuel storage and diesel fuel supply;
- → Other general-purpose and auxiliary facilities;
- → Connection to the existing drinking water supply system at a suitable location within the main site;
- → New, separated sewerage system discharging the wastewater from the site in the receiving water body, namely Danube river, after treatment and upon obtaining of Discharge Permit for the IP issued in accordance with the Waters Act.

2.2.2.3 SITE 3

A Service water supply facilities.

The on-site facilities for supply of fresh service water from the Danube and for discharging the hot processed (used) water are the following ones:

- → Connections to the channels which supply cold water and discharge the hot water;
- → Fore chamber;
- → CPS;
- → Power control building;
- → Pressurized pipelines;
- → Filter house;
- → Low-pressure channels.

At this site, the connection to CC-1 and HC-1 can be made without disturbing the operation of the other units if there would be an additionally built cold channel CC-2. After the decommissioning of Units 1 to 4, approx. 100 m^3/s of fresh service water from the Danube have become available. This inventory makes it unnecessary to build additional facilities for service water supply, i.e. CC-2, therefore other technical solutions will be needed in respect of the connection to the existing CC-1. For example, a connection with CC-1 would require construction of a new CPS at its end, which would not interfere with the operation of the existing Units. The distance from the site to CC-1 is ca. 235 m.

This site is situated near HC-2, which has been built in order to convey a flow of $Q=110m^3/s$ from Units 5 and 6, therefore the present IP does not consider usage of that channel.

It is proposed to build a new channel from Site 3 to the open-air stretch of HC-1, and in this case it is recommended to avoid any connection in the underground stretch of the channel. The best solution will be developed during the next phases of the investment project.

B Auxiliary facilities

Generally, the auxiliary facilities will be used to back up and supplement the conventional stock required for rated and design basis operating modes, and for protection of the environment from various wastes generated by the processes, and for supplying the required amounts of drinking water.

These facilities will include:

→ Water pretreatment plant – Chemical water treatment process

→ Treatment plants for grey and black water from “controlled” and “clean” areas, and for industrial waste water, as well as various local treatment installations;

→ Diesel fuel storage and diesel fuel supply;

→ Other general-purpose and auxiliary facilities;

→ Connection to the existing drinking water supply system at a suitable location within the main site;

→ New, separated sewerage system discharging the wastewater from the site in the receiving water body, namely Danube river, after treatment and upon obtaining of Discharge Permit for the IP issued in accordance with the Waters Act.

This site will also require additional construction and installation works for reconstruction and/or relocation of existing draining channels forming part of the draining and irrigation system in the Lowland of Kozloduy, which is important for the NNP site. There will also be a need for major earth-moving works in order to reach Ground Zero of the existing NNP site; for new technical solution for service water supply as well as for relocation of the fan of the 400 kV OPL.

2.2.2.4 SITE 4

A Service water supply facilities and auxiliary facilities

The construction of the main and auxiliary facilities will only be possible after demolition of all existing buildings, facilities and related utilities at the site. New lots of land will be needed if these would have to be rebuilt.

Service water can be supplied to the Unit by means of sag pipe from the Fore chamber of Units 5 and 6. The sag pipe will intersect the main mixed wastewater pipe and the low pressure channels of Unit 5 and 6. The discharge pipes of the recirculation water system will be connected in the low pressure channels of Units 5 and 6 or Unit 4:

- → Connection to the existing drinking water supply system will be possible at a suitable location.
- → New, separated sewerage system discharging the wastewater from the site in the receiving water body, after treatment and in accordance with the Discharge Permit for the IP issued in accordance with the Waters Act.

According to the IP, service water for cooling of the NNU can be captured from the existing intake of Units 3 and 4 (decommissioned), and in this case the course of the pipeline will pass above the underground stretch of the hot channel (HC-1) in the northern corner of the site. The distance to CC-1 is about 75m.

Although the underground stretch of HC-1 is in the northern corner of the site, the connection should preferably be in the open-air stretch in order to avoid any connection in the underground stretch.

The best solution will be developed during the next phases of the investment project.

2.3 ALTERNATIVE OPTIONS FOR BUILDING THE NNU

In the field of nuclear energy, Generation III and respectively III+ units represent the very best of state-of-art technology. These are latest nuclear power plant designs, which demonstrate better technological, economic and safety performance compared to the old generations.

The nuclear energy development phases are shown on the next **Figure 2.3-1.**

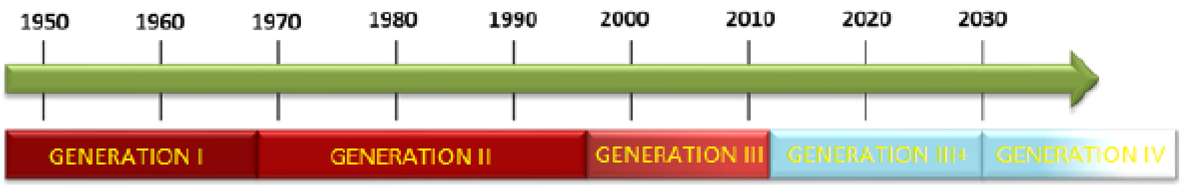


FIGURE 2.3-1: DEVELOPMENT OF THE NUCLEAR ENERGY SECTOR FROM THE PERSPECTIVE OF REACTOR GENERATIONS

Generation III power plants presently use the best available technologies based on the proven Generation II types. The main differences from Generation II are:

- → Standardized design, which reduces the time required for licensing the individual plants, the investment requirements and the construction period;
- → Simplified, but nevertheless solid design providing for easier maintenance and for increasing of operational margins;
- → Higher availability (90% or more), greater net efficiency (up to 37%) and longer service life (at least 60 years);
- → Lower risk of accidents involving significant core damage (much below 10^{-5}/year);
- → Greater resilience to external impacts;
- → Possibility for higher fuel burn-up rate (greater fuel utilisation, up to 70 GWd/tU) and for reduction of the amount of generated waste;
- → Extension of fuel residence time in the core by using burnable absorbers (up to 24 months).

Generation III+ was developed immediately after Generation III. It involves reactors with improved operational economics. Examples of PWR Generation III+ reactors are the EPR units from the Finnish company *Olkiluoto* and the French manufacturer *Flamanville*, the new Russian reactor AES-2006, the Japanese EU-APWR or the reactor units AP-1000 of *Westinghouse*. The reactor (and respectively the power plant) considered by the present investment proposal also belongs to this generation.

According to the Customer's Terms of Reference, there are two possible options for implementation of the IP and accordingly for building a new nuclear capacity with a reactor of the latest generation (Generation III or III+), which is compliant with the contemporary requirements for safe operation:

- → **A-1**: (Hybrid) Maximum usage of the nuclear island equipment ordered for NPP Belene and turbine island from another supplier.
- → **A-2**: An entirely new design.

Both options envisage the usage of Pressurized Water Reactor (PWR) of the latest generation (Generation III or III+) with installed capacity of approx. 1200 MW. The next sections describe a range of reactor models and scenarios for the two alternatives.

2.3.1 DESCRIPTION OF A-1 (HYBRID)

NPP Belene (**Figure 2.3-2**) has been designed with a Water-Cooled Water-Moderated Energy Reactor (WWER) of the WWER-1000/V466B type with four circulation loops, based on a standard design for a WWER AES-92 power plant, which in 2006 passed successfully all analysis stages for compliance with the EUR requirements, supported by the major European energy companies for next generation NPPs with light pressurized water.

The main differences between this design and the previous NPP designs with previous-generation WWERs are the following ones:

- Four independent trains of the safety systems;
- Residual heat removal and maintaining the reactor in safe condition by combination of active and passive systems, which do not require operator intervention;
- System for containment and cooling the melt from core components;
- System for control and reduction of hydrogen contents in the containment;
- Double containment designed for wide spectrum of internal and external events: the primary (inner) containment with leak-proof envelope is designed in prestressed ferroconcrete with steel liner, and the external containment is made of ferroconcrete.

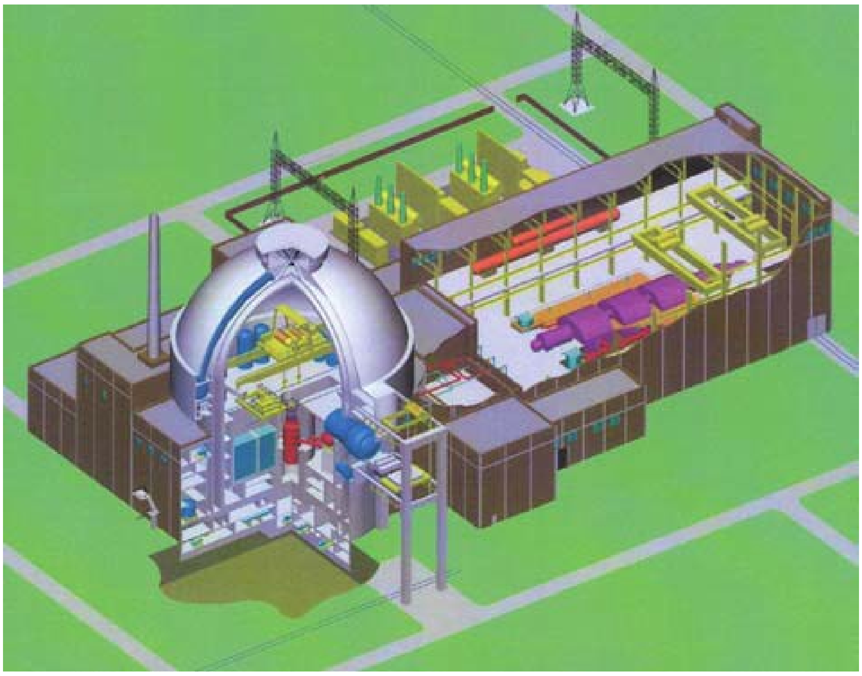


FIGURE 2.3-2: LAYOUT OF AES-92 (V-466B – BELENE)

Essential improvement of the leak-tightness has been made, providing maximal barrier for the emission of radioactive products into the surrounding environment. A double-containment structure is designed, wherein the primary containment is made of prestressed ferroconcrete with leak-tight metal lining, and the external containment is made of non-prestressed ferroconcrete. The external containment has been designed to withstand external forces such as crash of large passenger or military aircraft, external explosion waves, tornadoes, snow, extreme temperatures, and earthquakes. The inner containment has been designed to withstand seismic impacts and all security systems comply with the seismic safety requirements. NPP Belene has been designed to withstand Safe Shutdown Earthquake (SSE) with a^{max} = 0.24 g and probability of occurrence 1 in 100000 years.

The coolant circulation system consists of reactor and four circulation loops, see **Figure 2.3-3.** Each circuit has one steam generator and one main circulation pump. The pressurizer is situated in one of the loops.

In the primary circuit, the cold water (291°C) enters the bottom of the core at mass flow rate 86000 m^3/h and is heated up to 321°C, but does not boil. The so heated water is fed directly to the four hot legs and then to the U-pipes of each steam generator, where it is again cooled down to 291°C. The cooled water is pumped in each cold leg to the inlet of the reactor vessel in order to run down through the core.

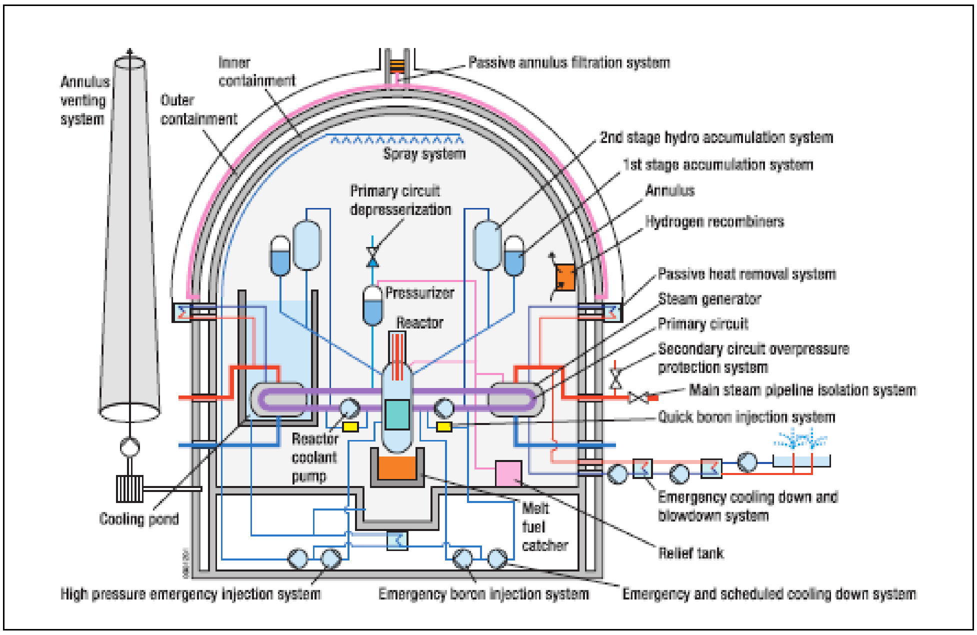


FIGURE 2.3-3: BLOCK DIAGRAMME OF AES-92 (V-466B – BELENE)

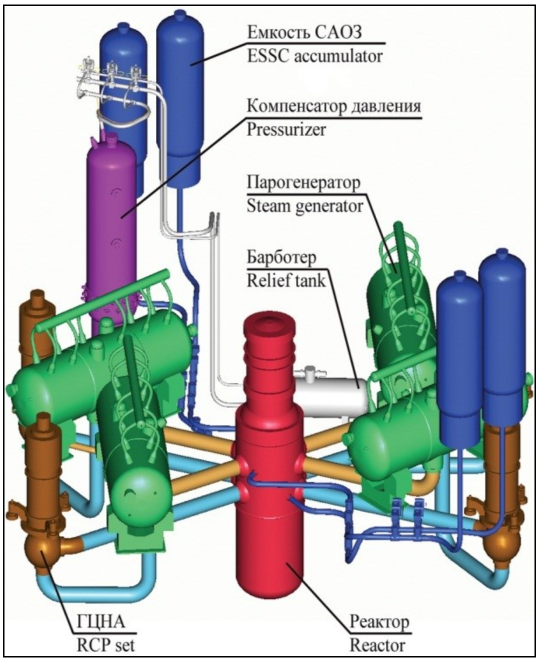
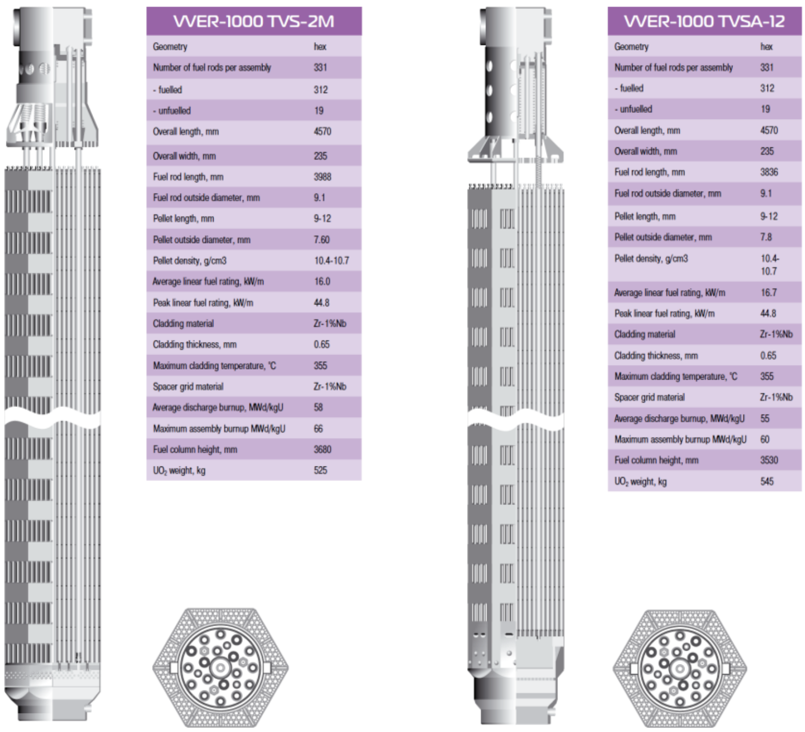


FIGURE 2.3-4: BLOCK DIAGNRAMME OF THE REACTOR SYSTEM

The feed water enters the secondary circuit of the stream generator at 220°C and cools down the water in the U-shaped pipes in the primary circuit from 321°C to 291°C – **Figure 2.3-4**. This cooling leads to generation of steam as a result of boiling in the secondary circuit. The steam is driven through separators and desiccators in order to reach minimum moisture content (<0.2%) to protect the impellers of the turbine rotor from damage. All steam generators deliver steam at mass flow rate of 5880 t/h to the power conversion system.

2.3.1.1 REACTOR VESSEL, CORE AND FUEL

The reactor core has 163 hexagonal Fuel Rod Assemblies (FRA). The WWER-1000/B466B reactor can use fuel type TVSA or alternatively TVS-2. The specific characteristics of each fuel type can be seen in **Figure 2.3-5.**



VVER-1000 TVS-2M	
Geometry	hex
Number of fuel rods per assembly	331
- fuelled	312
- unfuelled	19
Overall length, mm	4570
Overall width, mm	235
Fuel rod length, mm	3988
Fuel rod outside diameter, mm	9.1
Pellet length, mm	9-12
Pellet outside diameter, mm	7.60
Pellet density, g/cm3	10.4-10.7
Average linear fuel rating, kW/m	16.0
Peak linear fuel rating, kW/m	44.8
Cladding material	Zr-1%Nb
Cladding thickness, mm	0.65
Maximum cladding temperature, °C	355
Spacer grid material	Zr-1%Nb
Average discharge burnup, MWd/kgU	58
Maximum assembly burnup MWd/kgU	66
Fuel column height, mm	3680
UO_2 weight, kg	525

VVER-1000 TVSA-12	
Geometry	hex
Number of fuel rods per assembly	331
- fuelled	312
- unfuelled	19
Overall length, mm	4570
Overall width, mm	235
Fuel rod length, mm	3836
Fuel rod outside diameter, mm	9.1
Pellet length, mm	9-12
Pellet outside diameter, mm	7.8
Pellet density, g/cm3	10.4-10.7
Average linear fuel rating, kW/m	16.7
Peak linear fuel rating, kW/m	44.8
Cladding material	Zr-1%Nb
Cladding thickness, mm	0.65
Maximum cladding temperature, °C	355
Spacer grid material	Zr-1%Nb
Average discharge burnup, MWd/kgU	55
Maximum assembly burnup MWd/kgU	60
Fuel column height, mm	3530
UO_2 weight, kg	545

FIGURE 2.3-5: TVSA AND TVS-2M FUEL ROD ASSEMBLIES FROM TVEL

Each fuel rod assembly consists of head, tail (bottom nozzle) and stacks of fuel rods. One assembly contains 331 elements (rods): 18 control rods, one metering rod and 312 fuel rods (FRs).

The fuel rods are sustained by a frame consisting of guiding grooves and spacing grilles. The tail (bottom nozzle) supports the rod in the core plane and the head ensures structural stability. The tail minimizes the penetration of foreign objects in the FRs. The fuel tablets have central coaxial orifices.

The Control Rod Assemblies (CRAs) have 18 absorber rods and pipes filled with absorber material (dysprosium titanate and boron carbide) and their tops are sealed. The CRAs are also equipped with components for shifting the absorber rod (springs and suspenders).

The usage of dysprosium titanate has a number of benefits, namely:

- → Negligible expansion/bulging;
- → Lack of external gas emissions under neutron exposure;
- → High efficiency rate;
- → High melting point (~1870°C);
- → Lack of interaction with the FR shell at temperatures higher than 1000°C;
- → Simple manufacturing;
- → Convenient management as radioactive waste;
- → Can be used in the control rods in powder or tablet form.

2.3.1.2 COOLANT CIRCUALTION SYSTEM

The coolant circulation system is the main cooling system of the plant, which removes the heat generated in the reactor core. Similar to other PWR designs, this system consists of cooling loops, each loop having one heat exchanger and one main circulation pump – **Figure 2.3-6**.

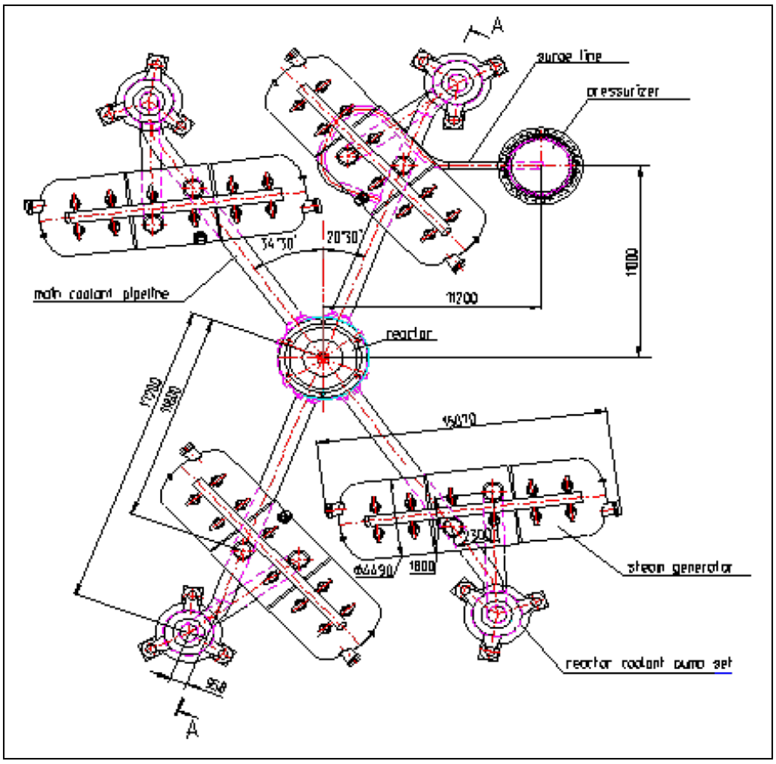


FIGURE 2.3-6: LAYOUT OF THE COOLANT CIRCUALTION SYSTEM

2.3.1.3 REACTOR VESSEL

The reactor vessel consists of internal devices, reactor head (lid), inlet and outlet couplings and the vessel itself.

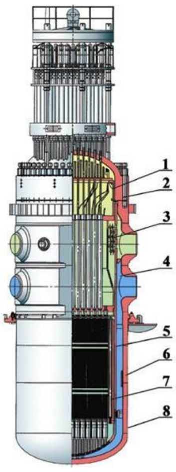
The vessel is made of forged rings, elliptic bottom and flange, welded together. The head is sealed and tightened to the vessel by means of stud bolts (dual end bolts).

2.3.1.4 INTERNAL DEVICES

An internal barrel of austentic steel is installed inside the vessel. It has a perforated bottom with 168 support pipes (one for each fuel rod assembly) and a spacing grille which serves as a frame of the fuel rod assemblies.

The internal barrel has a core reflector around the fuel rods, designed to protect the vessel from radiation. The reflector consists of several rings joined together by mechanical means. Longitudinal grooves are provided for proper cooling. There is also a ring of austentic steel, welded to the inner surface of the vessel in order to separate the flows of cold and preheated water.

The reactor vessel (**Figure 2.3-7**) is connected to the cold and hot leg by means of couplings DU 850. In the case of WWER reactors the hot leg outlet and the cold leg outlet are at different levels.



(1) Internal devices
(2) Head
(3) Protective pipes block
(4) Internal barrel
(5) Core reflector
(6) Witness samples
(7) Core
(8) Reactor vessel

FIGURE 2.3-7: REACTOR VESSEL

The Main circulation pumps (MCP) are vertical, centrifugal, single-stage pumps as in most pressurized water reactors. The pump model is GCNA-1391 – **Figure 2.3-8**.

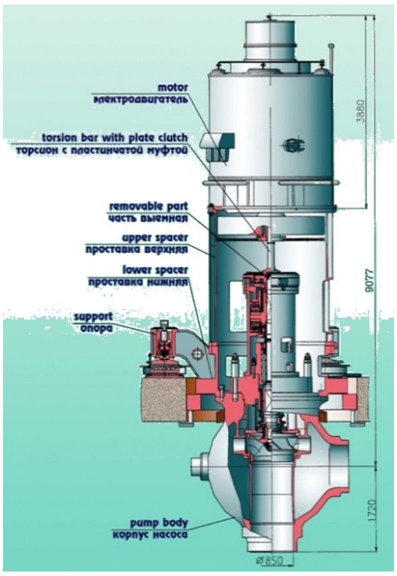


FIGURE 2.3-8: MAIN CIRCULATION PUMP

The pressurizer controls the pressure in the primary circuit. This is achieved by means of electric heaters, on one side, and an injection system on the other side. The pressurizer heaters heat the water to boiling point if it is necessary to increase the pressure. The pressurizer injection system injects cold water, which condenses the steam in the pressurizer and thus reduces the pressure if necessary.

The pressurizer is also equipped with three pivotal safety (relief) valves, which are opened if there is overpressure in the primary circuit.

2.3.1.5 STEAM GENERATOR

The steam generator model is PGV-1000MK, horizontal type – **Figure 2.3-9.**

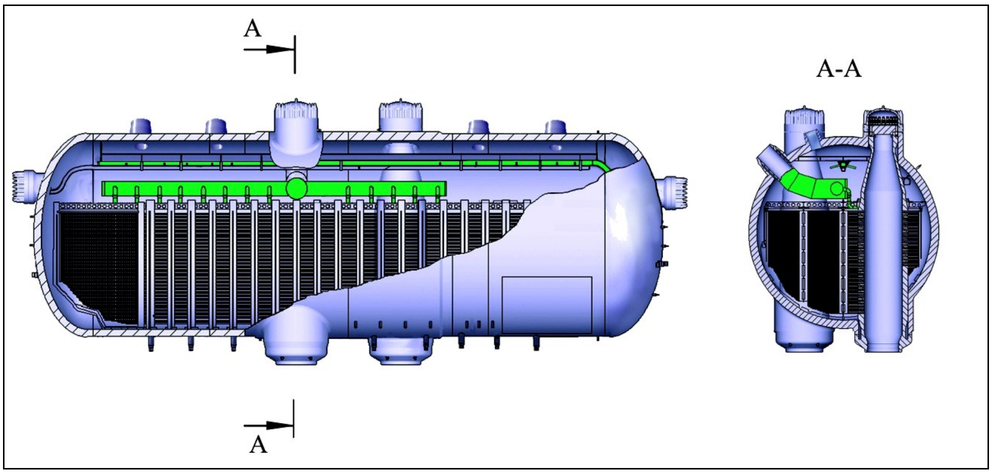


FIGURE 2.3-9: STEAM GENERATOR

The vessel accommodates the pipes forming part of the primary circuit (10978 U-shaped pipes of austentic steel) and the pipes forming part of the secondary circuit, in which the steam is generated as a result of the heat emitted by the pipes. The steam generator is made of forged rings, couplings and welded elliptical stacks of pipes.

The pipe stacks are arranged in circular form and connected to the cold and hot leg of the primary circuit though manifolds. The U-shaped pipes are rolled hydraulically into openings in the walls of the manifold and then welded. The walls of the manifold are resistant to corrosion and are coated in two layers.

The main advantage of the horizontal model is the large inventory of water present in the secondary circuit during transient processes such as loss of feed water and de-energizing of the plant

Apart from the large water inventory in the secondary circuit, this type of reactors are equipped with 8 additional two-stage hydro-accumulators with water inventory of 120 m^3 each, and an extended spent fuel pond, which provides much greater additional time reserve for fuel cooling in case of accident. A distinguished feature of the WWER design is the location of the spent fuel pond (SFP) inside the containment.

2.3.1.6 SAFETY SYSTEMS

The safety systems include: protective, localizing, support and control systems. The protective systems perform functions related with emergency cooling and residual heat removal from the reactor core. They have active and passive parts. Each safety system has four independent trains, the efficiency of the trains is selected on the basis of the "single failure" principle and the trains of the safety systems are physically separated.

The safety systems are designed to meet the objectives of the fundamental safety functions:

- Reactivity management;
- Core and spent fuel cooling;
- Containment of the radioactive products within the established limits.

The AES-92 design provides multiple protection levels for mitigation of accidents (deep defense), which result in very low probability of core failures. The deep defense concept of the AES-92 design is associated with the following six aspects:

- **Operational stability** – This is achieved by selection of materials, quality assurance during the design and construction phase, well trained operators and improved plant management system and design ensuring significant operational margins of the plant before the safety limits are even approximated.
- **Physical barriers of the plant** – The spread of radioactive products is prevented by means of designed physical barriers: fuel cladding, boundaries of the reactor coolant circuit and leak-tight design of the nuclear system.
- **Passive safety systems** – The passive systems and equipment of the AES-92 reactor are sufficient to establish and maintain automatic cooling of the core, integrity of the containment, allowance for least restrictive single failure and loss of AC power sources in and beyond the site.
- **Systems important to safety** – The next deep defense level are the systems important to safety, which reduce the probability of occurrence of events leading to core failure. These highly reliable systems are activated automatically upon occurrence of higher-probability events in order to provide first tier of protection and reduce the probability of unnecessary activation and operation of the safety systems.
- **Localization of core failure** – The AES-92 design enables the operators cool down the melt catcher in the event of reactor vessel failure. The double containment minimizes the probability of releases beyond the boundaries of the site.

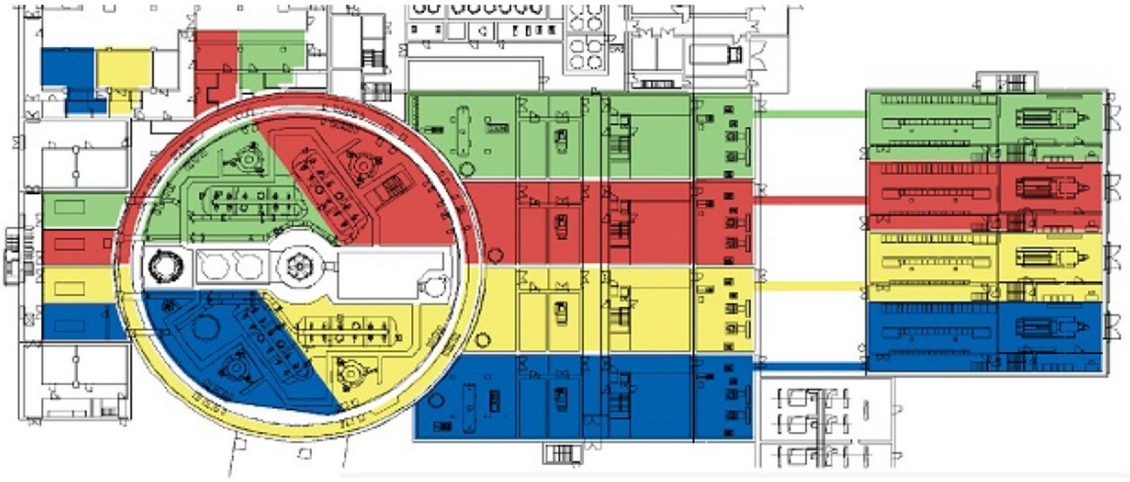


FIGURE 2.3-10: QUADRUPLE REDUNDANCY *(IN DIFFERENT COLOURS)* OF THE SAFETY SYSTEMS

The safety systems are designed with redundancy taking into account the requirements in respect of general failure and the probability of loss of power.

An important concept of the safety systems design is that each safety system has four redundant trains and each train is sufficient to ensure the safety of the reactor.

2.3.1.6.1 Active safety systems

- → Spray system – this system is designed to reduce the pressure and temperature in the containment in accident conditions by condensation of the steam coming from the depressurized primary or secondary circuit. The system has four redundant trains, each one with pump and related piping and fittings.
- → Containment isolation systems – this system is designed to minimise the spread of radioactive materials beyond the containment in accident conditions.
- → System for emergency injection of boron solution in the core under high pressure – the system has four redundant trains, each one with pump and related piping and fittings. The pressurized lines of the pumps are connected to the cold legs and the suction lines are connected to the four trains of the System for cooling of the primary circuit and the fuel pond in accident and rated conditions, after the heat exchanger. The water for the emergency boron injection system is taken from the refuel pond and the concentration of boron in it is 16 g/kg.
- → System for cooling of the primary circuit and the fuel pond in accident and rated conditions – the system has four redundant trains, each one with pump and related piping and fittings. The main task of this system, together with the System for emergency injection of boron solution in the core under high pressure, is to bring the reactor to subcritical state if the reactor emergency protection systems has failed. The other purpose of this system is to inject boric water in the pressurizer in the event of leak from the primary circuit. The pressurized lines are connected to those of the System for emergency injection of boron solution in the core under high

pressure, before the connections to the cold legs of the Coolant circulation system in the primary circuit. There is also a connection with the head of the pressurizer.

- SG emergency cooldown and blowdown system – this system serves as an alternative system for residual heat removal and as a reactor cooling system in accidents involving failure of normal cooling in the secondary circuit such as leaks from steam pipes, loss of feed water and loss of power. It also provides cooling in accidents involving loss of coolant in the primary circuit. The system has four trains, one for each steam generator. Each train has two heat exchangers (HE) – one economizer and one for emergency cooling, a pump and related piping and fittings. The steam from the steam generator condenses in the heat exchanger and the condensation water is returned to the same generator.
- Primary circuit overpressure protection system – this system serves to prevent situations where a hypothetical transient process involving overpressure may cause failure of the primary circuit due to mechanical stress. The system protects the primary circuit from overpressure in rated, warm-up and cool-down conditions. It consists of three pivoted safety valves in the upper part of the pressurizer. These valves open when the pressure in the primary circuit exceeds a preset value.
- Secondary circuit overpressure protection system – this system serves to protect the secondary circuit from overpressure. It consists of three pivoted safety valves in the main steam pipelines of each generator.
- Emergency gas exhaust system – the task of this system is to exhaust the steam mixture formed at various places in the primary circuit: in the head of the reactor vessel, in the steam generator and in the pressurizer. It can be activated by an operator either in rated conditions or in beyond design basis accidents.
- Main steam lines isolation system – this system serves to isolate the steam generators in the event of leak from SG line or rupture of feed water line.

2.3.1.6.2 Passive safety systems

- Emergency core cooling system, passive part – the passive system is the Quick boron injection system, which has the task to inject boric water in the reactor core and flood the core to disrupt the fission process in the event of accident or accidental loss of coolant. The system has four hydro-accumulators and related piping and fittings.
- The independent Passive emergency core cooling system is the passive part of the Emergency heat removal system. The task of this system is to exhaust the steam from the steam generator, condense it by means of air-water heat exchangers and take the condensation water back to the generator. The system cools the reactor core in the event of power or coolant loss in the primary or secondary circuit. In the event of accident involving loss of coolant, it operates together with the passive core flooding system. Pumps are not needed because the heat exchangers operate on the principle of natural circulation. The air stream comes directly from the atmosphere by natural draft.

- Passive core flooding system – this system can be regarded as second stage of the hydro-accumulators. The first stage is the passive emergency core cooling system. Its task is to cool and maintain the core in subcritical state in the event of full de-energizing coincident with coolant loss for more than 24 hours (if the Passive Heat Removal System succeeds to start) or 8 hours (if the Passive Heat Removal System fails to start). The system consists of four independent hydro-accumulators of capacity 120 m^3 containing boric water at atmospheric pressure. The total inventory of this system is 8x120 = 960 m^3, which is considered sufficient for the functioning of the system.
- Independent Passive Heat Removal System (PHRS)
- Quick boron injection system – this system is intended for beyond design basis accidents when the emergency protection system fails to start. It provides highly concentrated boric solution which disrupts the fission reactions and brings the reactor in subcritical state. It has four independent trains with a tank for boric acid solution and related piping and fittings for each train.
- Containment system – the system consists of two protective shells or containments. The primary containment is a prestressed ferroconcrete structure with carbon steel lining of the inner surface. Its task is to contain the spread of radionuclides in accident conditions. The secondary containment is executed in concrete and serves to protect the reactor from external threats – either natural or caused by man.
- Hydrogen control system and system for reducing the hydrogen level in the containment (hydrogen recombination system) – these systems reduce the risk of formation of flammable and explosive hydrogen mixtures in the containment. The selected strategy for managing this risk is by passive auto-catalytic recombiners. The recombiners are designed to operate in accident conditions and beyond design basis accidents.
- Passive annulus filtration system – the task of this system is to trap any leaks before they are released in the atmosphere as a result of major accident. The passiveness of this system means that it operates even if AC power supply is lost.
- Melt containment and cooling system – the task of this system is to contain and cool the molten material consisting mainly of molten core, internal reactor vessel devices and reactor vessel material in situations where the vessel is damaged and molten material escapes from the vessel. It is designed to cooldown the melt with water and maintain subcriticality.

The main technical specifications of the V-466B (AES–92) design for NPP Belene are as follows:

GENERAL SPECIFICATIONS

Output, gross [MWe]	1068
Output, net [MWe]	1000
Heat output [MW]	3000
Efficiency [%]	33.1

GENERAL SPECIFICATIONS

Operating mode	Base load and load tracking
Design service life [years]	60
Availability [%]	85 or > 90 (target)
Construction period [months]	60
Core damage frequency, 1/year	$<6.1 \times 10^{-7}$
Early major releases frequency 1/year	$<1.77 \times 10^{-8}$
Residual heat removal systems	4 trains – active and passive
Feed systems	4 trains – active and passive
Melt catcher	Yes
MDE [g]	0.24
Primary circuit	
Number of main circulation loops	4
Primary circulation flow [m^3/s]	23.9
Operating (rated) pressure [MPa]	15.7
Secondary circuit	
Rated steam flow [kg/s]	1633
Steam temperature/pressure [°C/MPa]	278.5 / 6.27
Reactor core	
Core height [m]	3.53
Core equivalent diameter [m]	3.16
Fuel rod assemblies	163
Fuel assembly	Hexagonal
Maximum fuel enrichment [%]	4.28
Number of absorber stacks	121
Average discharge burnup [MWd/kg]	54.6
Fuel	UO_2
Duration of burnup campaign [months]	12÷18
Fuel amount [t UO_2]	79.8
Reactor vessel	
Inner diameter of the barrel [mm]	4195
Barrel wall thickness [mm]	195
Gross height [mm]	11185
Main circulation pumps	
Number of pumps	4
Rated flow [m^3/h]	21500
Pressurizer	
Gross capacity [m^3]	79
Design pressure [MPa]	17.3
Steam generators	
Number of steam generators	4
Type	Horizontal, with U-shaped pipes
Maximum outer diameter [mm]	4490
Gross length [mm]	13820
Inner containment	
Make	Prestressed concrete with steel lining

GENERAL SPECIFICATIONS	
Capacity [m³]	63000
Outer containment	
Make	Ferroconcrete

2.3.1.7 TURBINE PLANT

The Hybrid reactor scenario includes two options for steam to energy conversion, taking into account the rated operational parameters of the WWER-1000 AES-92 Nuclear island and the conditions of the final absorber at the NPP Kozloduy site. The suppliers are ***Toshiba*** and ***Alstom***.

These suppliers offer the following turbine generator alternatives:

- ✓ Turbine generator ARABELLE 1000 from ***Alstom***
- ✓ Turbine generator TC6F from ***Toshiba***

The features of the two turbine generators are described briefly below together with the steam to energy conversion system.

*2.3.1.7.1 Description of **Alstom** steam to energy conversion system*

Alstom offers the model ARABELLE 1000 – a compact 3-cylinder steam turbine consisting of one combined high/medium pressure module and two dual flow low pressure modules. The turbine is directly coupled to the generator (model TA1200-78, 24 kV, with brushless exciter). Both of them spin at lower speeds (i.e. 1500 rpm for Bulgaria) and the total length of the shaft is only 55 meters. Assuming that the condenser will be cooled with water coming directly from the Danube (open cycle cooling system), at average annual temperature 12.5°C the gross electric output of the generator would be at least 1100 MWe at condenser pressure 5 kPa(a).

The main components are:

- ✓ Turbine (1 combined HP/MP cylinder and 2 dual flow LP cylinders)
- ✓ 2 steam separators-superheaters between the HP and MP modules
- ✓ Main turbine condenser
- ✓ 4-stage low pressure preheaters
- ✓ 1 feed water tank
- ✓ 2-stage high pressure preheaters

The steam from each of the four steam generators (SG) enters a *high pressure* (HP) cylinder via four valve boxes, each one containing one stop valve and one check valve. After expanding in the *high pressure* cylinder, the spent steam passes through two external steam separators-superheaters (SSS), where the moisture is removed and the steam is reheated. After the SSS the steam passes through the stop valve and check valve of the SSS and then enters the *low pressure* (LP) cylinder where it expands further.

Two connecting pipes intersecting underneath guide the steam from the *low pressure* exhaust pipe to the *low pressure* (LP) inlet pipe, where the steam is distributed to each of the two cylinders in two even flows and expands up to the condenser of the main turbine.

Various steam extractors in the HP, MP and LP cylinders deliver steam to the feed water heaters.

The condensation and feed water systems deliver to the steam generators (SG) heated feed water in a closed cycle using regenerative heating of feed water. The condensation system collects the condensed steam from the condenser and pumps it to the deaerator through 4-stage LP preheaters. The feed water system sucks in water from the deaerator and pumps it to the SG through 2-stage HP preheaters.

2.3.1.7.2 Description of Toshiba steam to energy conversion system

For the Hybrid reactor, Toshiba propose TC6F turbine with 52-inch (1.32 m) impellers in the last stage. The turbine generator consists of one (1) dual flow HP cylinder and three dual flow LP cylinders. Its calculated electrical output is approximately 1100 MW for 3.89 kPa(a) at reference Danube water temperature 12.5°C measured at inlet. Both the turbine and the generator spin at rated speed 1500 rpm. Thus, the operation of the system includes:

- Turbine (1 HP cylinder + 3 LP cylinders)
- 2 steam separators-superheaters between the HP and LP modules
- Condenser of the main turbine. Three sections, open loop
- 3 condensation pumps of 50% capacity
- LP preheaters no. 1 and 2
- LP superheaters no. 3 and 4
- 1 deaerator
- 3 main feed pumps of 50% capacity
- 2-stage high pressure feed water preheaters, connected in parallel

The steam generated in the steam generators is fed to a *high pressure* (HP) cylinder from a Main steam manifold (MSM). After expansion in the HP turbine, the steam passes through two steam separators-superheaters (SSS) and then enters three *low pressure* (LP) cylinders. Part of the steam is extracted through steam extractors in the HP and LP cylinders and fed to a 7-stage feed water preheating system.

The spent steam from the LP turbines condenses and deaerates in the condenser of the main turbine. The condense pumps suck-in condense from the hot leg of the condenser and feed it through 4-stage LP feed water preheaters to the fifth stage – the open deaerating preheater. Then the condense proceeds to the suction side of the feed water booster pump of the steam generator and then to suction side of the main feed water pump. The feed water pumps of the steam generator deliver the water through a 2-stage HP preheater to the steam generators.

2.3.1.8 GENERATOR AND MAIN ELECTRICAL EQUIPMENT

2.3.1.8.1 Generator

2.3.1.8.1.1 Alstom generator

The turbine ARABELLE 1000 is coupled directly to a 3-phase synchronous generator TA1200-78, 24 kV with brushless exciter. The winding of the stator of the generator is cooled directly by deionized water, which circulates in grooves of stainless steel, and the excitation coil of the rotor is directly cooled with hydrogen,

Main characteristics:

- Rated capacity: 1100 MW
- Power factor
  - Input power factor: min. 0.95
  - Output power factor: 0.90
- Short circuit ratio: >0.5 for power factor 0.85 and 0.90.

2.3.1.8.1.2 Toshiba generator

The generator supplied by Toshiba is a 3-phase cylindrical rotor with spinning field and synchronous generator. It is designed to operate at base load with power rating factor or at full load within the rated capacity of the generator.

The stator winding is cooled directly with water. The rotor coil is cooled directly with hydrogen.

The main characteristics are:

- ✓ Rated capacity: 1110 MW
- ✓ Efficiency: 98.9%
- ✓ Excitation system: static
- ✓ Power factor:
  - Input power factor: min. 0.95
  - Output power factor: 0.90
- ✓ Short circuit ratio: >0.5.

2.3.1.8.2 House AC power supply systems

The main power supply system delivers AC power from the generator to the electric system and the auxiliary system of the power plant. The voltage rating values at the power supply buses are 1.5 kV, 24 kV, 6.3 kV, 400 V, 230 V.

In power generation mode, the turbine generators are connected to the electric system by two house power transformers connected in parallel. Each transformer has power rating

630 MVA[2] and voltage rating 24/400 kV. The voltage rating of the generators is 24 kV. They are connected to two auxiliary transformers by screened trunks. The auxiliary transformers (also known as *houseload* working transformers) have capacity 63 MVA, three coils, power rating 63/31.5/31.5 MVA and voltage rating 24/6.3/6.3 kV. There are four 6.3 kV buses for distribution of the rated power. Each busbar has two outlets: one powered by the secondary coil of the auxiliary transformer and the other one from the secondary coil of the *houseload* backup transformer. Each unit has two backup houseload transformers of capacity 63 MVA, power rating 63/31.5/31.5 MVA and voltage rating 110/6.3/6.3 kV. One transformer (of capacity 63 MVA, power rating 3/31.5/31.5 MVA and voltage rating 110/6.3/6.3 kV), connected to one busbar 110 kV and two 6.3 kV buses delivers power to the general-purpose loads of the plant. These loads are backed up by backup *houseload* transformers.

The power supply system for rated operation consists of 4x 6.3 kV buses powered from the secondary coils of the backup *houseload* transformers. These 6.3 kV busbars supply 0.4 kV busbars for low voltage loads.

2.3.1.9 DIESEL GENERATOR FOR RATED OPERATION

One auxiliary diesel generator of power rating 6.3 MW and voltage rating 6.3 kV is installed and serves as power source in the event of power loss during rated operation. The diesel generator powers two emergency low voltage busbars, which receive power from a diesel generator and one secondary coil of the backup *houseload* transformer. If the power from the backup *houseload* transformer is lost, these busbars are automatically switched over to diesel generator power.

2.3.1.9.1 Emergency power supply

The emergency power supply system consists of four trains. The power supply of each train consists of:

- ✓ Power supply to loads of category 2 – these are AC loads, which do not require high reliability of power supply and can tolerate a blackout of max. 10 minutes. These loads tolerate blackout periods determined by safety conditions such as diesel generator ramp up time and startup sequence period.
- ✓ Power supply to loads of category 1 – these are DC and AC loads, which require high reliability of power supply and do not tolerate blackouts for more than a fraction of a second under any conditions.

Each train consists of the following equipment:

- ✓ One diesel generator of power rating 6.3 MW and voltage rating 6.3 kV;

[2] The cosine of the phase displacement angle between the current and voltage in sinusoidal systems determines the active (MWA) and reactive (MVAr) component of the rated capacity (MW).

- ✓ One output from busbar for rated operation, which receives power from auxiliary transformers or backup *houseload* transformers in rated operating conditions;
- ✓ Two Integrated Switchgears (ISG) 6.3 kV;
- ✓ ISG 0.4 kV and secondary equipment;
- ✓ One power transformer of voltage rating 6.3/0.4 kV;
- ✓ Uninterrupted Power Supply equipment;
- ✓ Rechargeable battery 220 V.

The category 2 loads are powered from the 6.3 kV bus of the train. These loads can receive power from auxiliary transformers or backup *houseload* transformers in rated operating conditions. If the rated power supply from these power transformers is lost, the trains will receive power from the relevant diesel generator with one generator assigned to each busbar of the train. Each diesel generator is equipped with main independent services (fuel, oil, cooling water, startup air, etc.) for autonomous operation up to 21 days.

2.3.1.9.2 Rechargeable batteries

Emergency power sources are fed by four systems, one for each train. Each system is equipped with two rechargeable batteries, each one coupled to one rectifier and one inverter for uninterrupted power supply.

One rechargeable battery is assigned to the safety systems (battery life is 2 hours) and the other one provides power in a fully de-energized situation (battery life is 1 hour).

2.3.2 DESCRIPTION OF A-2

The second option considered for the construction of a new nuclear capacity is an entirely new PWR design of Generation III or III+ reactors, with 1200 MW output. Generation III and III+ are advanced reactors, which have been developed based on the experience with the operation of Generation II reactors.

The design will comply with the basic Safety Fundamentals Safety and Requirements of the International Atomic Energy Agency (IAEA).

The reactor models under review should comply with the safety criteria determined by the Bulgarian legislation, the IAEA documents and the European Utility Requirements (EUR) for LWR Nuclear Power Plants (requirements of the European operators for NPPs with light water reactors). The WWER/PWR types of reactors were chosen based on the following considerations:

- ✓ The reactors of this type (WWER) have been successfully operated in Bulgaria since 1974.
- ✓ Utilising the highly qualified personnel at Kozloduy NPP site with their knowledge of many years in the respective technology.

- ✓ The proposed technology is the most widely used one worldwide for electricity generation from nuclear source and approximately 80% of the reactors are precisely of that type.

Currently, over 430 nuclear power reactors with total installed capacity of about 370 GW_e are operating worldwide. Several dozens of nuclear power plant units are at different construction phases – **Figure 2.3-10.**

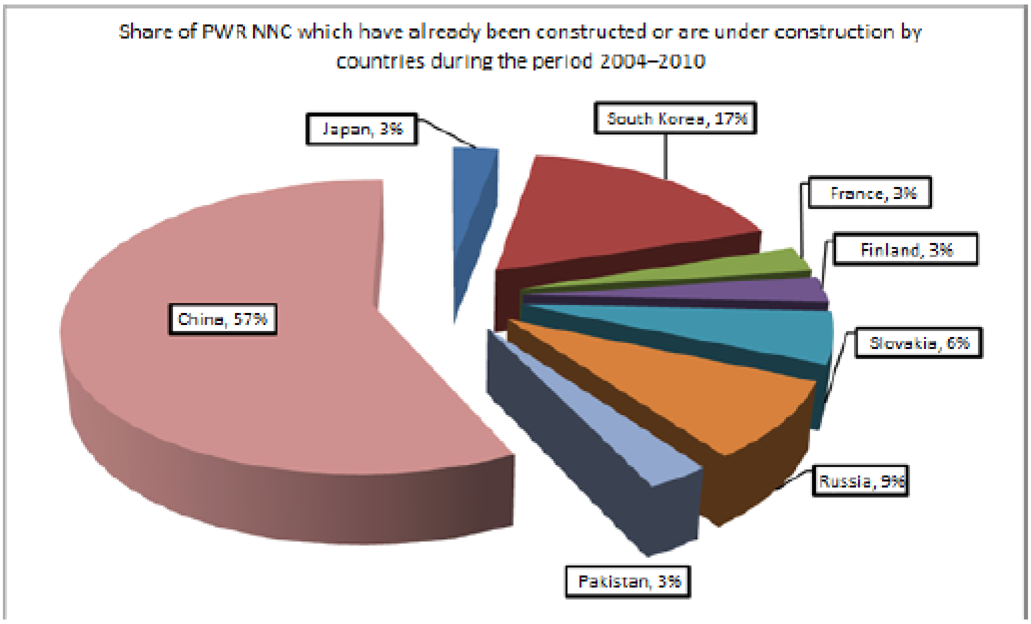


FIGURE 2.3-11: SHARE OF PWR NNU WHICH HAVE BEEN ALREADY CONSTRUCTED OR ARE UNDER CONSTRUCTION BY COUNTRIES DURING THE PERIOD 2004-2010

In parallel with the EIA project, a project for "Techno-economic analysis to justify the construction of a new nuclear capacity at the site of NPP Kozloduy" (TEA) is carried out. With regards to alternative **A-2** (entirely new design), the TEA Terms of Reference document sets out two requirements:

1. The installed single-unit capacity should be around 1200 MW. This requirement is defined by the fact that a number of regulatory documents recommend that the single-unit installed capacity does not exceed 10% of the total installed capacity in the country. Currently, the total installed capacity in Bulgaria is around 12200 MW. The emergency shutdown of a single unit of capacity greater than 1200 MW would threaten the integrity of the electrical energy system of the country.

2. Having regard to the fact that practically no new NNP have been built in the recent years, the construction of a Generation III and III+ reactor is considered an advantage at this time.

According to the interim results of the "Techno-economic analysis to justify the construction of a new nuclear capacity at the site of NPP Kozloduy", a summary of currently available on the market models of PWR/WWER reactors of Generation III or III is presented on **Table 2.3-1.**

TABLE 2.3-1: ***GENERATION III OR III+ OF CAPACITY GREATER THAN 1000 MW***

Model	EPR	EU-APWR	APR-1400	AES-2006	ATMEA1	AP-1000
Manufacturer	Areva France	Mitsubishi Japan	Kepko Japan	Atomstroy-export Russia	Areva + Mitsubishi	Westing-house USA
El. capacity *(Gross)*	1770 MW	1700 MW	1455 MW	1170 MW	1200 MW	1200 MW
El. capacity *(Net)*	1650 MW	1620 MW	1400 MW	1082 MW	1150 MW	1117÷1154 MW
Certificate	EUR; URD-in progress	EUR-in progress; URD-in progress	URD-in progress	Designed to EUR requirements	Designed to EUR requirements	URD; EUR
License	Франция; iDAC; NRC-in progress	NRC-in progress	KINS; NRC-предстои	Rostehnadzor	In progress	NRC; iDAC
Construction	France Finland China	None	Korea UAE	Russia	None	China USA

Legend: *EUR-European Utility Requirements; URD-Utility Requirements Document (US); iDAC – Interim Design Acceptance Confirmation (UK); NRC – Nuclear Regulatory Commission (US); KINS – Korean Institute for Nuclear Safety*

In view of the objectives set out in the Terms of Reference document for the TEA, there are several nuclear reactors available on the market that meet the above requirements.

In the ToR for EIA of investment proposal "Building a new nuclear capacity of the latest generation at the NPP Kozloduy site" the following two models of reactors are considered as examples:

- AES-2006;
- AP-1000.

The Atomstroyexport AES-2006 model is an evolutionary design of the AES-91/92, which was developed for NPP Belene as well. The AES-02 design has passed EUR conformity assessments. Currently, the AES-2006 is constructed in Leningrad, Novovoronezh, Kaliningrad..

The Westinghouse AP-1000 has passed EUR conformity assessments and has obtained a license from NRC. Currently, it is under construction in China (4 units foreseen to be put into operation by 2015) and USA (14 units have received combined license for construction and operation from NRC).

These different engineering solutions are options of the Investment proposal which will be subject to environmental impact assessment. The environmental and safety requirements for all reactor types are identical and their impact will be assessed at their maximal potential values.

For the purposes of the EIAR, the so-named **conservative approach** has been chosen, meaning that the values which result in the least favourable environmental effects will be considered throughout the assessment.

2.3.2.1 REACTOR AP-1000

Westinghouse AP-1000 (**Figure 2.3-12)** is a Generation III+ PWR with thermal neutrons moderator and water cooling. The actual electric output of AP-1000 depends on the specific conditions at the site, the choice of turbine, the electrical systems and other site-specific parameters. The heat output is 3415 MW with net electric output in the range of 1117-1154 MW. The actual net electric output is correlated to the site conditions. The availability of AP-1000 is around 93%.

AP-1000 uses open fuel cycle with refuelling interval of 18 months and three fuel cycles. The design service life is 60 years.

The design is licensed in the US and China, and is currently being licensed in Europe by the UK Nuclear supervision authority. The first four units are already under construction in Sanmen and Haiyang, China.

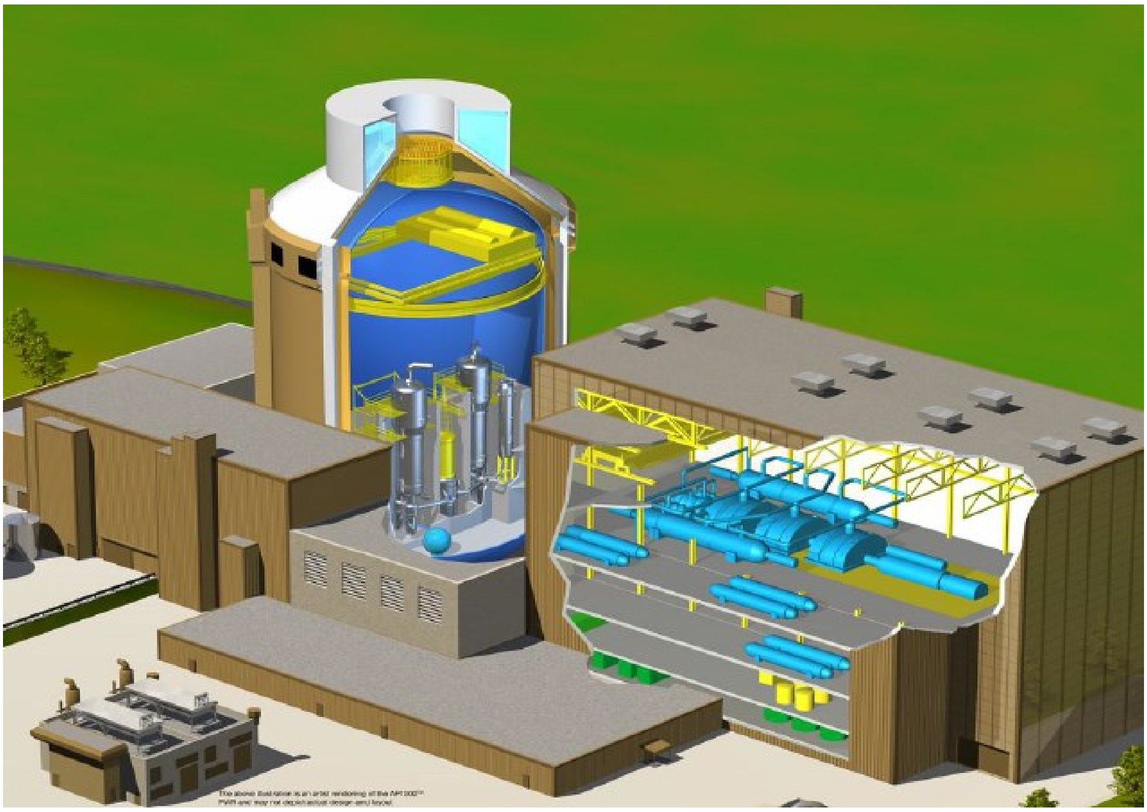


FIGURE 2.3-12: LAYOUT OF AP-1000

Compared to a standard power plant with similar production capacity, AP1000 has 35 % less pumps, 80 % less high safety class pipings, and 50 % less valves of safety class ASME. There are no high safety class pumps. This makes the AP1000 power plant much more compact compared to the older designs. Since the equipment and pipings have been reduced, the greater part of the safeguarding equipment is assembled within the containment. Consequently, AP1000 has approximately 55 % less pipe connections to the containment compared to the power plants of the current generation. The amount of structures conforming to seismic class 1 is about 45 % less compared to older designs for commensurate capacities. AP1000 features a relatively larger pressurizer, which allows it to adapt easier to different modes.

AP-1000 features two circulation loops, each having one hot leg (outer diameter 790 mm) and two cold legs (outer diameter 560 mm), steam generator with vertical U-shaped pipe evaporator with integrated moisture separator and two circulation pumps. AP-1000 uses two vertical steam generators, model Delta-125, and four circulation pumps mounted two by two directly on each steam generator to eliminate the pipe link between the steam generator and the main circulation pump. The circulation pumps include flywheel to maintain primary circuit circulation in de-energized conditions – **Figure 2.3-13**.

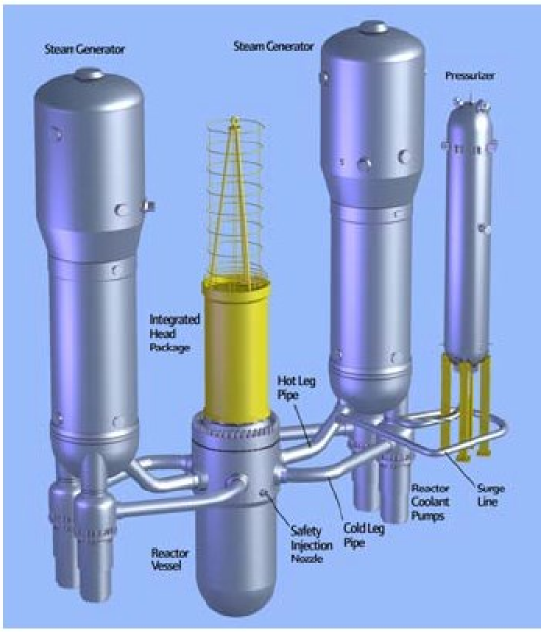


FIGURE 2.3-13: AP-1000 PRIMARY CIRCUIT

The reactor vessel (**Figure 2.3-14**) is barrel-shaped with welded hemispheric bottom and removable hemispherical head.

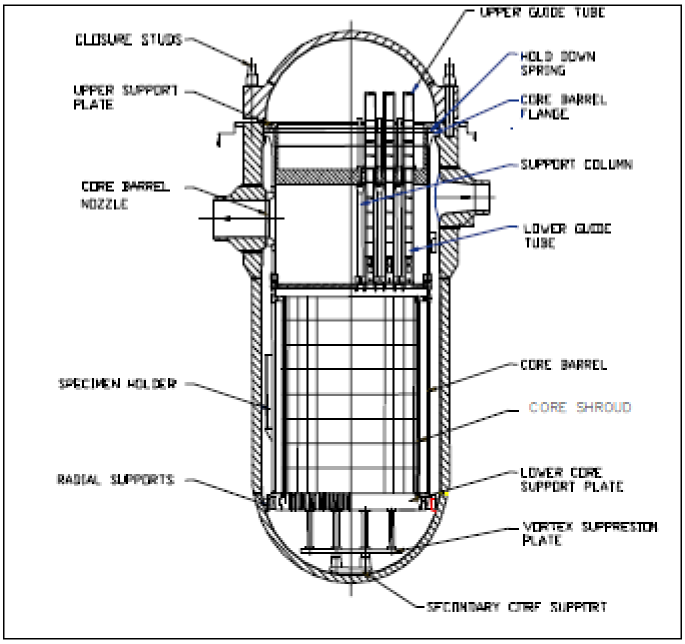


FIGURE 2.3-14: AP-1000 REACTOR VESSEL

The cold and hot manifolds are attached to the reactor vessel. The head has openings for the instrumentation which controls the drives in the core. The reactor vessel (including the head) is around 12.192 m long and its inner diameter in the core area is 4.0386 m. AP-

1000 is designed to withstand pressure 17.1 MPa and temperature 343°C in the course of 60 years. For better safety performance, the design does not provide openings under the core level of the reactor. This eliminates the probability of leaks from the reactor vessel that may lead to exposure of the core.

The cooling and moderating agent is light water at rated operating pressure 15.51 MPa. The reactor core consists of 157 fuel assemblies 426.7 cm in length. The fuel assembly contains 264 fuel rods, namely tubular elements ZIRLO, which accommodate the fuel – enriched Uranium in the form of cylindrical granules Uranium Dioxide. The core arrangement features three radial zones with different fuel enrichment levels varying from 2.35% to 4.45%.

2.3.2.1.1 Safety concept

The AP-1000 design features various protection levels (deep defense), which leads to extremely low probabilities of core damage. The deep defense concept is defined by the following six aspects:

2.3.2.1.1.1 Rated operation

At rated operation, the most basic deep defense level ensures that the plant can operate in a stable and reliable manner. This is achieved by selection of appropriate materials, quality assurance during the design and construction phases, training of operators and other measures to provide significant margins during the operation of the unit before the safety limits are even approximated.

2.3.2.1.1.2 The physical barriers

One of the most recognizable aspects of the deep defense concept is the safeguarding of public safety by the physical boundaries of the unit. Radiation leakage is prevented by the fuel rod shells, the reactor boundary and the containment.

2.3.2.1.1.3 Passive safety systems

The passive systems of AP-1000 suffice for automatic activation and maintenance of cooling and preservation of the integrity of the core for 72 hours following maximal design basis accident, limited single failure, lack of operator action and unavailability of local and external AC sources.

2.3.2.1.1.4 Diversity of safety systems

An additional level of protection is provided by diversity of the principles which define the collaboration of the various safety systems. Such diversity exists, by way of example, in respect to the residual heat removal function. In the event of multiple failures of the passive residual heat removal systems, the removal of residual heat during transient processes is ensured by other passive functions such as automatic depressurization and coolant injection and drainage.

2.3.2.1.2 Systems important to safety

The next deep defense level are the systems important to safety, which reduce the probability of occurrence of events that may lead to core failure. These highly reliable systems are activated automatically by many of the probable events in order to provide first tier of protection and reduce the probability of unnecessary activation and operation of the safety systems.

2.3.2.1.3 Core damage

The design of AP-1000 enables the operators use the refuel water in the pond situated in the containment and discharge it in the reactor if the core is not flooded and is melting down. This prevents damage of the reactor vessel and subsequent spill of molten core in the containment. The retention of the core in the reactor vessel ensures high security in terms of avoidance of containment failures and radioactive releases in the environment as a result of major accidents involving container vessel damage.

The deep defense of AP-1000 increases the safety performance such that serious release of fission products with an intact containment can be expected more than 100 hours after the core damage occurs assuming that no recovery action is to be taken. This provides time for undertaking accident management measures to mitigate their consequences and prevent containment failure. The recurrence of serious radioactive releases in the environment is $1.95x10^{-8}$ per reactor/year – a value much lower compared to the existing plants.

2.3.2.1.4 Safety systems and components

AP-1000 uses passive safety systems to increase the safety of the plant and ensure compliance with regulatory safety criteria. The employment of passive safety systems provides an advantage vs. the conventional designs in terms of significant and measurable improvements such as simplified designs, reliability and protection of investments. Passive safety systems do not require operator intervention. They use only natural forces such as gravity, natural circulation and pressure of compressed gases in order to secure the operation of the system. These systems do not require any pumps, fans, diesel generators, cooling plant or other active equipment.

The passive safety systems of AP-1000 include:

→ **Passive core cooling system** AP-1000 (**Figure 2.3-15**), which includes:

- System for emergency introduction of water in the core – high pressure: two tanks connected via valves to the cold and hot legs provide additional coolant if necessary. The pressure in the tanks is equal to the pressure in the primary circuit;
- System for emergency introduction of water in the core – medium pressure: two tanks with boric water automatically deliver coolant to the core in the event of pressure drop. The tanks are pressurized by compressed nitrogen.

- System for emergency introduction of water in the core – low pressure: if the primary circuit pressure drops to atmospheric level, the refuel water can be used as core coolant. The water is discharged by the force of gravity.
- Emergency core cooling system – a heat exchanger in the refuel tank can be used in the event of steam generators failure. The heat exchanger is connected to the hot and cold leg and the temperature differential between the two provides natural circulation. The tank can sustain residual heat removal in the course of two hours until the boiling point is reached.
- Passive primary circuit pressure relief system – this system consists of three valves connected to the pressurizer and one valve connected to the hot leg. Their function is to relieve the pressure in the primary circuit and enable the supply of coolant from the emergency tanks.

→ **Passive containment cooling system** – an air cooling zone is configured in the upper part of the containment. In the event of accident involving elevation of the inner temperature, the heat exchanging surface is sprayed, by the force of gravity, with water from a tank on the roof of the building, the evaporation of which improves the heat exchange.

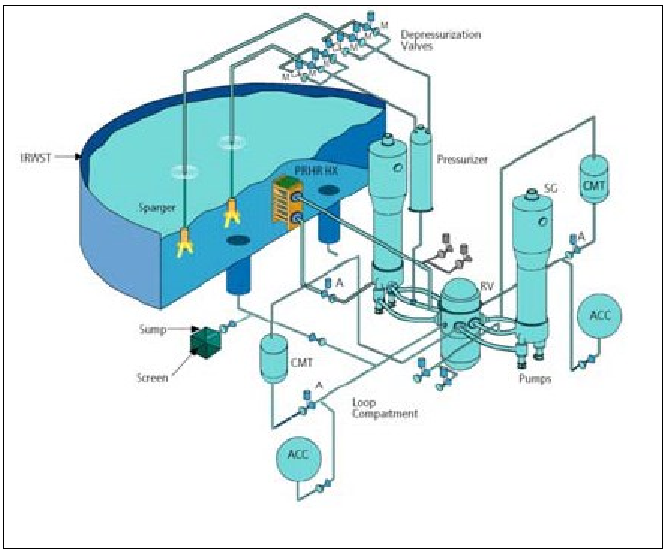


FIGURE 2.3-15: AP-1000 PASSIVE CORE COOLING SYSTEM

→ **Emergency inhabitance system** for the Unit Control Room (UCR);

→ **Isolation function** – includes isolation of all communications between the hermetically sealed bodies and all other compartments, as well as systems for control of hydrogen concentration, recombination and, if necessary, incineration.

AP-1000 is designed to retain pieces from the molten core within the reactor vessel, so that no melt can penetrate the bottom of the reactor vessel. In case of major accident, cooling water from the large reserve water tank used during refueling may be used to flood the reactor cavity and to cool the reactor vessel on the outside. The device is shown in **Figure 2.3-16**. The special-design insulation of the reactor vessel forms a ring which allows the cooling water to come in direct contact with the vessel. Vents have been provided allowing the steam to escape from the ring.

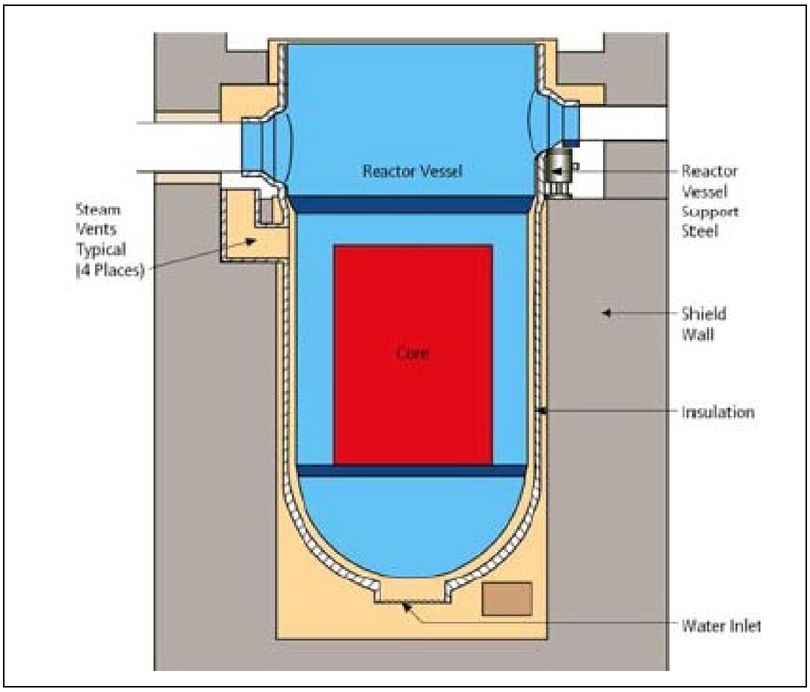


FIGURE 2.3-16: AP-1000 INTERNAL VESSEL COOLING DESIGN

The vented steam condenses on the containment walls to be directed afterwards back to the cavity (**Figure 2.3-17**).

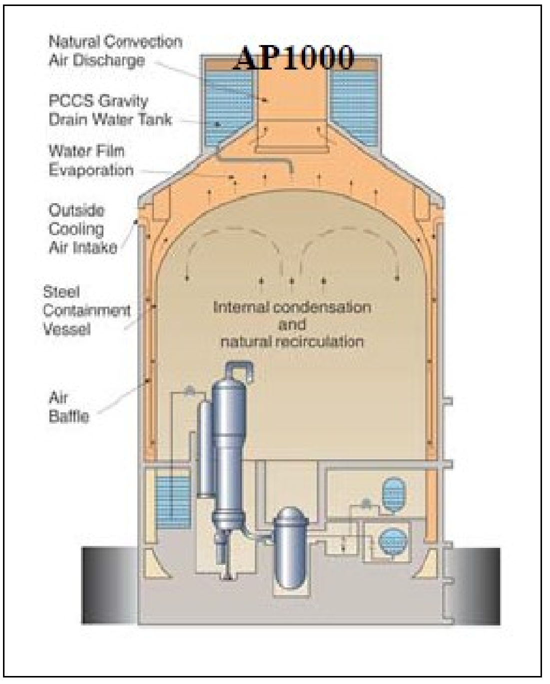


FIGURE 2.3-17: AP-1000 PASSIVE CONTAINMENT COOLING SYSTEM

The main technical specifications of the **AP-1000** design are as follows:

GENERAL SPECIFICATIONS

Output, gross [MWe]	1200
Output, net [MWe]	1117÷1154
Heat output [MW]	3400
Efficiency [%]	33÷34
Operating mode	Base load and load tracking
Design service life [years]	60
Availability [%]	> 93
Construction period [months]	54
Core damage frequency, 1/year	5.11×10^{-7}
Early major releases frequency 1/year	5.94×10^{-8}
Residual heat removal systems	Passive
Feed systems	Passive
Melt catcher	Yes, in the vessel
MDE [g]	0.3
Primary circuit	
Number of main circulation loops	2 hot / 4 cold
Primary circulation flow [m³/s]	19.87
Operating (rated) pressure [MPa]	15.5
Secondary circuit	
Rated steam flow [kg/s]	1886
Steam temperature/pressure [°C/MPa]	272.78 / 5.76
Reactor core	

GENERAL SPECIFICATIONS

Core height [m]	4.267
Core equivalent diameter [m]	3.04
Fuel rod assemblies	157
Fuel assembly	Square
Maximum fuel enrichment [%]	4.8
Number of absorber stacks	69
Average discharge burnup [MWd/kg]	60
Fuel	UO_2 or MOX
Duration of burnup campaign [months]	18÷24
Fuel amount [t UO_2]	95.97
Reactor vessel	
Inner diameter of the barrel [mm]	4038.6
Barrel wall thickness [mm]	203
Gross height [mm]	13944
Main circulation pumps	
Number of pumps	4
Rated flow [m^3/h]	17886
Pressurizer	
Gross capacity [m^3]	59.5
Design pressure [MPa]	17.1
Steam generator	
Number of steam generators	2
Type	Vertical U-shaped pipes
Maximum outer diameter [mm]	6096
Gross length [mm]	22460
Inner containment	
Make	Steel
Capacity [m^3]	58333
Outer containment	
Make	Ferroconcrete

2.3.2.1.5 Turbine plant

The steam to energy conversion system is designed to extract the heat from the reactor cooling system by means of two vertical steam generators and convert it to electricity in the turbine generator. The steam turbine condenser extracts the unused heat of the steam by the water circulation system. The turbine regenerator heats up the condensation water in the condenser and in the feed system and takes it back to the steam generators.

2.3.2.1.5.1 Overview

The steam from each steam generator enters a high pressure cylinder via 4 stop valves and 4 check valves. After expansion in the high pressure cylinder, the spent steam enters an external steam separator-preheater. There the moisture content of the spent steam from the high pressure cylinder is reduced from ~10% to 13% to relative humidity 0.5% or less.

The turbine uses two-stage heating wherein the first stage uses steam from the high pressure cylinder (HPC) and the second stage uses main steam from the main tract to heat up the steam to superheated condition. The superheated steam enters three low pressure cylinders (LPC) via separate stop valves in each of the 6 steam lines, which deliver steam to the inlets of the cylinders. Seven steam extractors extract from the turbine steam used by the system for regenerative heating of feed water for the steam generators.

2.3.2.1.5.2 Description of the turbine generator

The turbine of the AP-1000 reactor unit is a power conversion system designed to transform the thermal energy of the steam in mechanical energy by means of a turbine, which in turn spins the generator to produce electrical energy. The turbine plant is a TC6F module with turbines, generator, external steam separator-superheater, control and service systems.

The turbine spins at 1500 RPM for electric current with frequency 50 Hz. It consists of one dual-flow HPC, three dual-flow LPCs and two external two-stage steam separator-superheaters. The other service systems include one common system for lubrication of the turbine generator bearings, digital electro-hydraulic control system, turbine bearing packing system, turbine overspeed protection system, stator water cooling system, hydrogen cooling system and oil-based generator packing system, generator CO_2 supply system, generator excitation system and voltage governor.

The turbine generator is designed to operate at base load, but can also be used in load tracking mode. The mechanical design of the flows in the turbine uses an optimized circuit, which significantly reduces the stresses during operation. The steam flow from each steam generator to the HPC is controlled by two stop valves and two check valves. The turbine generator and all service and control systems are situated in the generator hall.

The turbine generator is supported by a spring system. The springs of the turbine generator isolate the other structures from the vibrations occurring during operation, which allows integration of the structure under the turbine platform. The condenser is rested on spring support and is connected by rigid link to the outlet pipes of the low pressure cylinder.

The foundations design comprises a concrete platform rested on springs and secured by a structural steel frame, which forms the integral part of the turbine hall structure. This type of structure reduces the need for additional reinforcement and the number of required columns of the building. Furthermore, the spring-suspended structure ensures dynamic isolation of the turbine generator fundaments from the other structures. This type of structure is more independent from the specificities of the particular site insofar as the ground is decoupled/isolated from the dynamic forces of the turbine. The structures under the springs are designed independently from the vibration considerations. The foundations of the turbine generator and the securing of the equipment are designed to the same seismic performance as the turbine hall.

2.3.2.1.6 Steam generator condensation and feed water systems

The systems for condensation and feeding the SGs with preheated water use a closed cycle with regenerative heating. The condensation system collects the condensed steam from the condensers and pumps is back to the deaerator. The SG feed water system uses high pressure feed pumps to deliver the water from the deaerator to the SG system. It consists of pipelines and safety valves, which deliver feed water to the steam generators. The condensation and feed water systems are situated in the turbine building, while the SG systems are in the service building and in the containment.

The SG feed water system consists of three electric pumps operating in parallel, with intake pressure secured by booster pumps. After the pumps, the feed water is delivered to the steam generator by high pressure preheaters (HPP). The latest designs feature three main feed pumps of 50% capacity each, wherein two are operational and one is redundant.

The coolant inventory in the intake of the condenser and in the deaerator allows/defines the margin of the design. This margin, combined with three condensation pumps of 50% each allows greater flexibility and leeway for operators to control transient process in the unit and the related operation of the feed water and condensation systems.

The system for delivery of chemical compounds in the tract of the turbine island dispenses reagents for reduction of hydrogen concentrations and for adjustment of the pH-value of the coolant in the secondary circuit.

2.3.2.1.7 Generator and main electrical equipment

The main AC power system supplies the non-safety systems and the Class 1E battery charging systems (these are electrical systems which are responsible for power supply of safety systems), wherein Class 1E systems are separated from the other systems. The main system also supplies the control transformers in rated and emergency modes.

The rated voltages at the power buses are 11 kV, 400 V, 230 V, 208 V and 120 V.

The unit is designed to cope with 100% loss of load, wherein the turbine generator will continue to deliver houseload power in sustainable manner.

There is no power supply to the generator during startup, shutdown and maintenance of the unit. Main AC power is provided from the high voltage system via Outdoor Switchgears and a house step-up transformer, which provides power to two service transformers of the unit. Each service transformer provides 50% of the unit's power requirements.

400V switchgears supply power to certain electrical drives (EDs) and ED controllers. Each reactor cooling pump is powered from two Class 1E breakers connected in series. These are the only Class 1E breakers in the main AC power system designed to deliver safety-related power to these pumps. The breakers belong to seismic Category 1 and can withstand design basis earthquake without loss of their safety function.

The controls of Class 1E breakers are powered by the relevant Class 1E group. The main power system ensures separation between the Class 1E groups and Class 1E groups and other non-Class 1E power cables.

The main AC power system of the unit also supports the following basic non-safety functions:

- ✓ Optional separation of non-Class 1E AC power from on-site sources for loads not important to safety;
- ✓ Tripping of breakers in ECS-ES-1 and ECS-ES-2 switchgears upon receipt of signal from the on-site backup power system;
- ✓ Turning on the breakers in each redundant DG upon receipt of signal from the on-site backup power system;
- ✓ Each redundant DG is dimensioned to deliver long-term power to safety related loads which support the control functions in post-accident periods, the control room lighting and ventilation systems and one recirculation pump of the passive containment cooling system.

2.3.2.1.8 Backup house power

Two backup diesel generators of power rating 4000 kVA each provide power to certain non-nuclear safety loads such as the residual heat removal system in situations involving partial loss of coolant, and to loads responsible for the industrial safety of the unit. This backup power is not required for nuclear safety purposes and does not perform safety functions.

2.3.2.1.8.1 Auxiliary diesel generators

When other sources are not available, power to Class 1E systems for post-accident monitoring, lighting and ventilation systems in the control room and the instrumentation room, for filling the main water tanks and the spent fuel pond is provided by two auxiliary DGs (each one rated 35 kVA) situated in a separate building. These generators are not required during the first 72 hours after complete loss of all external power sources.

2.3.2.1.8.2 DC power systems

Uninterrupted DC power supply to non-safety non-Class 1E loads.

The system provides power to loads important for the operation of the unit and to those responsible for industrial safety.

The system consists of two independent DC and UPS trains, each one with two subsystems situated in separate rooms in the auxiliary building. Each subsystem consists of battery chargers, fixed batteries, switchgears and related control and protection devices.

DC lines 1, 2, 3 and 4 of the subsystems provide 125V power to the inverters responsible for uninterruptible power supply to non-Class 1E loads. Alternative DC power is provided by the relevant control transformers. DC line 5 powers large EDs, thus avoiding impacts on the other power lines.

The main DC loads are:

- ✓ Pumps of the turbine generator bearings emergency lubrication and packing system;
- ✓ Local control and display instrumentation;
- ✓ Various remote control devices and instruments.

The main UPS loads are:

- ✓ The radioactive monitoring system (non-safety part);
- ✓ Equipment forming part of communication systems;
- ✓ The firefighting system;
- ✓ Control data displays and processing systems components;
- ✓ Components of unit control systems.

2.3.2.1.8.3 DC and UPS systems for Class 1E loads

This system provides DC power to the loads important to safety as well as uninterrupted DC and AC power in rated and accident conditions.

The components of the Class 1E loads power supply systems are situated in structures belonging to seismic resistance Category 1.

The system consists of four independent Class 1E DC trains of voltage rating 250V. The individual trains supply Class 1E loads for 24h or 72h depending on their safety functions so that operator intervention for ensuring their power supply is not required during the first 24h. The battery chargers can be also be powered from the backup DGs as each one has the capacity to charge a fully discharged battery for 24 hours.

2.3.2.2 REACTOR AES-2006

AES-2006 is a water-cooled water-moderated 1200 MW power reactor. This is the latest design of the Russian company *Atomstroyexport*, owned by the Russian State company *Rosatom*. This design is based on the design and operational experience with WWER-1000 reactors and upgrades the design of AES-92. The AES-2006 design is licensed in Russia.

At present, different versions of AES-2006 projects are under construction in Leningrad – Mod. V-491 and in Novovoronezh – Mod. V-392M.

Although the main specifications of V-392M and V-491 are similar, there are several important differences between the two versions such as:

- ✓ Incorporation of passive containment heat removal system and passive steam generators heat removal system in Mod. V-491;

- ✓ Incorporation of passive core flooding system in Mod. V-392M;
- ✓ Incorporation of active emergency coolant injection systems, high and low pressure, in Mod. V-491;
- ✓ Differences in the systems for management of beyond design basis accidents;
- ✓ Differences in the estimated core damage frequencies;
- ✓ Differences in the control and management systems, the feed water system, the design of the control rooms, etc.

The main reason for these differences is that V-392M is designed by *Atomenergoproekt* (Moscow), while V-491 is designed by *Atomenergoproekt* (St. Peterburg).

Despite these differences, both models satisfy the advanced safety requirements and the requirements of the Russian rules and standards. The two models are designed in conformity with IAEA and EUR recommendations in order to obtain building permits from *Rostehnadzor*.

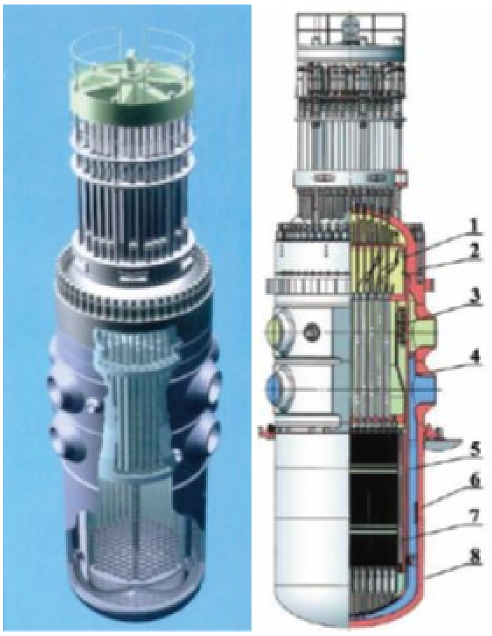
The safety functions of AES-2006 have been improved compared to the AES-92 power plants. With the AES-2006 power plant, both the active and passive systems are used to perform safety functions. Moreover, AES-2006 is equipped with major accident control systems. The rated service life of the power plant is 60 years. With AES-2006, the structural protection against large aircraft crash is concentrated in the external containment and the fresh fuel storage facility.

A power plant with AES-2006 reactor type comprises the following main equipment and systems:

- → Pressurized water reactor with thermal power output of 3200 MW with primary circuit coolant pressure 16.2 MPa, where the water with boric acid acts as a coolant and moderator in the reactor. The concentration of boric acid varies during operation. Slightly enriched uranium dioxide is used as a fuel;
- → Four horizontal steam generators PGV-1000MKP type with pipe arrays, placed at large distances in corridor arrangement. Each steam generator produces (1602+112) t/hour of dry saturated steam with pressure 7.0 MPa;
- → Four sets of main circulation pumps type GTsNA-1391;
- → Main circulation pipelines of rated diameter Dn 850;
- → Pressure compensation system;
- → Equipment of the concrete containment of the reactor;
- → Safety systems.

The reactor vessel is made of forged rings, elliptical bottom and peripherally welded flanges. The head is tightened to the barrel by studs and leak-tightness of the joint is ensured by solid ring gasket. The vessel material is steel alloy 15H2NMFA, which is used due to its mechanical properties, durability and resistance to neutron radiation.

A "basket" made of austentic steel is placed in the vessel. 163 casing pipes for the fuel assemblies are situated at the elliptical bottom of the "basket" – **Figure 2.3-18**. The tops of the assemblies are pressed by the so named "protective pipes block" (PPB) of the reactor, which has sockets for the head of each assembly. The PPB accommodates the controls of 121 drives passing through it.



(1) Internal devices
(2) Head
(3) Protective pipes block
(4) Internal barrel
(5) Core reflector
(6) Witness samples
(7) Core
(8) Reactor vessel

FIGURE 2.3-18: AES-2006 REACTOR VESSEL

2.3.2.2.1 Reactor vessel

The inlets and outlets of the four circulation (coolant) pipelines, each one of diameter 850 mm, are incised in the vessel. Each circulation loop comprises a hot section, steam generator, cold section and circulation (coolant) pump. The steam generators are horizontal, Mod. PGV-1000MPK. The advantage of horizontal generators is the large inventory of water. The heated water (328.9°C) from the reactors is driven through 10978 U-shaped pipes of austentic steel and heats up the water in the steam generator (secondary circuit) to evaporation temperature to produce steam at the rate of approx. 1780 kg/s. A circulation pumps takes the cooled-down coolant (298.2°C) back to the reactor for reheating. AES-2006 uses Mod. GTsNA-1391 – vertical centrifugal pumps with flywheel. The flywheel ensures circulation of the coolant in the event of power failure. The rated capacity of each circulation pump is 21 500 m^3/h.

The AES-2006 provides for the usage of two types of fuel – from the producer TVEL or alternatively from Gidropress.

2.3.2.2.2 Deep defense

The deep defense concept is a major contributor to safety. The objective is to prevent the spread of radioactive materials to the population and the environment. The deep defense concept features three major safety functions: capacity control, fuel cooling and containment of radioactive material within the controlled zone. To compensate potential human errors and equipment failures, the deep defense concept focuses on several protection levels, including consecutive barriers preventing the release of radioactive material to the environment. The concept includes protection of barriers by preventing damage to equipment/components and to the barriers proper. It includes additional deferred measures for personnel protection in case one barrier is not efficient enough. The metering instrumentation and the control systems supporting the deep defense help maintain the integrity of the physical barriers (fuel array, fuel rod shells, the boundary of the reactor coolant and of the containment). The safety-related control and management systems are also designed on the basis of the deep defense principle. The multistage defense uses independent subsystems and is designed to perform all essential functions in all operating modes, taking into account the existence of general failures.

Multistage defense is ensured by means of:

- Multiple redundancy;
- Use of reliable safety systems equipment;
- Operation by combined use of automatic devices and operator action;
- Use of systems and equipment designed to mitigate the consequences of accidents.

The design ensures endurance to internal and external impacts that may lead to general failure by means of:

- Physical separation and installation – the subsystems are installed in different compartments and use separated cable routes;
- Independence, i.e. electrical and physical isolation between safety trains and systems, and between the safety-related systems and rated operation systems;
- High fire resistance performance of the materials used;
- Certification of the safety-related systems and equipment in accordance with the Russian standards and with the IEC series of standards;
- Seismic certification of the safety-related systems and equipment in accordance with the Russian standards and with the IEC series of standards;
- Electromagnetic compatibility of the equipment in accordance with the Russian standards and with the IEC series of standards.

AES-2006 is designed in consideration of the single failure principle, meaning that each safety system would be able to perform its function upon the failure of one active or passive component. The safety systems design also takes into account other principles such as: diversity, redundancy, independency, separation and reliability.

The AES-2006 design includes active and passive safety systems. The objective of the safety systems is to achieve the safety objectives by applying the deep defense principle.

- subcriticality maintenance:
  - ✓ Negative reactive power factor
  - ✓ Negative reactive steam factor
- Core and spent fuel cooling:
  - ✓ Quadruple redundancy of each safety system;
- Containing the radioactive materials below the regulatory limits:
  - ✓ Dual containment

The safety systems of AES-2006 (B 491) consist of four completely independent trains. The power, the quick activation, and other line characteristics have been chosen based on the condition for providing nuclear and radiation safety under all initiating events envisaged in the design. Due to the fact that the safety system trains are located in separate bodies, high degree of physical separation is achieved. Each safety system train is separated from the others by fire-proof physical barriers located along their entire length, including the connections from one compartment to another (**Figure 2.3-19***)*. Direct connections between the different safety trains are not permitted. The safety systems are provided with physical protection against access by unauthorized personnel.

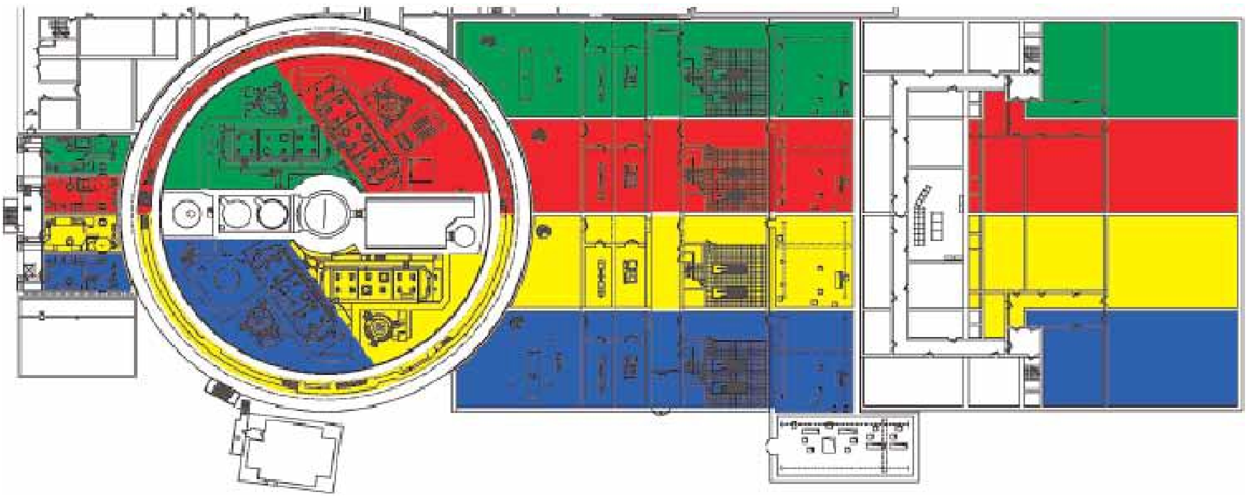


FIGURE 2.3-19: AES-2006 (B 491) PHYSICAL SEPARATUION OF SAFETY TRAINS
(shown in different colours)

The technical solutions used in the AES-2006 design with WWEP-1200 preclude the occurrence of major beyond design basis accidents in case of occurrence of several single failures and subsequent failures of the safety system components. Such accidents may occur only in case of failure of several safety system trains, which is a rare event.

The models V-392M and V-491 feature different sets of active and passive safety systems.

2.3.2.2.2.1 V-392M Safety systems

Active systems

- Emergency core cooling system via the steam generators – cools down the reactor core upon loss of power and loss of coolant in the primary or secondary circuit. In the event of leakage from the primary circuit, the system operates together with the main passive core flooding system;
- Emergency gas evacuation system – serves to evacuate the concentrations of steam mixture in the upper parts of the primary circuit – reactor, steam generator and pressurizer. Can be activated by an operator during rated operation or in accident conditions;
- Emergency high-concentration boric solution injection system – the main purpose of this system is to bring the reactor in subcritical condition upon failure of the reactor protection system. The other task of this system is to inject boric water in the pressurizer in the event of minor to medium leaks;
- Emergency and scheduled primary circuit and spent fuel pond cooldown system – the task of this system is to remove the residual heat and provide emergency boric water injection upon partial loss of coolant. It also delivers water to the spraying (sprinkling) system;
- Primary circuit overpressure protection system – this system is designed to prevent overpressure conditions during rated operation and warmup/cooldown processes. The system comprises three pressure relief valves mounted on the pressurizer, which open if the primary circuit pressure violates a preset level;
- Main steam lines isolation system – it serves to isolate the steam generator(s) in the event of steam line or feed line leak before the return valve.

Passive systems

- Emergency core cooling system, passive part – injects boric water in the reactor core in order to cooldown and flood the core and helps disrupt the fission process in accident situations such as loss of coolant. The trigger pressure is 5.9 MPa in the primary circuit;
- Emergency core flooding system – second operating stage of the hydro-accumulators: this is a passive first stage emergency cooling system. Its purpose is to cool the core and sustain subcriticality upon full loss of power coincident with partial loss of coolant for a minimum period of 24 hours (if the passive heat removal system succeeds to start) or 8

hours (if the passive heat removal system fails to start). In rated operating conditions it serves to fill the primary circuit with boric water during refueling;

- Passive heat removal systems;
- Dual containment and melt catcher.

2.3.2.2.2.2 V-491 safety systems

Active systems

- Emergency high pressure boron injection system – delivers boron at the concentration equal to that in the primary circuit upon leaks and ruptures which exceed the capacity of rated operation systems. The trigger pressure of the system is 7.9 MPa;
- Emergency low pressure boron injection system – this system delivers coolant with appropriate concentration of boric acid in the reactor system upon partial loss of coolant up to a rupture equivalent to the diameter of the main circulation line, i.e. 850 mm;
- Emergency gas evacuation system;
- Emergency high-concentration boric solution injection system – the main purpose of this system is to bring the reactor in subcritical condition upon failure of the reactor protection system. The other task of this system is to inject boric water in the pressurizer in the event of minor to medium leaks;
- Emergency feed water system – this is an additional feed water system to the main and auxiliary steam generators makeup water systems. It is designed to deliver makeup water to the steam generator if all the other feed water systems are not available;
- Residual heat removal system – this system is designed to cool the reactors during shutdown. In accident conditions the system is used for long-term cooling;
- Main steam lines isolation system – it serves to isolate the steam generator(s) in the event of steam line or feed line leak before the return valve.

Passive systems

- Emergency core cooling system, passive part – injects boric water in the reactor core in order to cooldown and flood the core and helps disrupt the fission process in accident situations such as loss of coolant. The trigger pressure is 5.9 MPa in the primary circuit;
- Passive core cooling system via the steam generators – cools down the reactor core upon loss of power and loss of coolant in the primary or secondary circuit. In the event of leakage from the primary circuit, the system operates together with the main passive core flooding system;
- Passive containment heat removal system;
- Dual containment and melt catcher.

Dual containments and melt catchers are provided in both models. The localization systems are intended to prevent or restrict the spread of radioactive substances (as a result of accidents) within the NPP and their release into the surrounding environment. Basically, the primary (inner) containment is a cylinder made of prestressed ferroconcrete, with hemispheric head and ferroconcrete base slab. The inner side is lined in welded carbon steel sheets to ensure leak-tightness. The external containment is a cylinder made of ferro-concrete, with hemispheric head. The inlets of all pipings are firmly fixed to the walls of the inner body and are welded to the steel lining. All inlet pipes are equipped with localization valves (**Figure 2.3-20**).

FIGURE 2.3-20: AES-2006 CONTAINMENTS

The main technical specifications of AES-2006 are:

GENERAL SPECIFICATIONS

Output, gross [MWe]	1170
Output, net [MWe]	1082
Heat output [MW]	3200
Efficiency [%]	34
Operating mode	Base load and load tracking
Design service life [years]	60

GENERAL SPECIFICATIONS

Availability [%]	> 90
Construction period [months]	54
Core damage frequency, 1/year	< 1 x 10^{-6}
Early major releases frequency 1/year	< 1 x 10^{-7}
Residual heat removal systems	4 trains – active and passive
Feed systems	4 trains – active and passive
Melt catcher	Yes
MDE [g]	0.25
Primary circuit	
Number of main circulation loops	4
Primary circulation flow [m^3/s]	23.9
Operating (rated) pressure [MPa]	16.2
Secondary circuit	
Rated steam flow [kg/s]	1780
Steam temperature/pressure [°C/MPa]	286 / 7
Reactor core	
Core height [m]	3.73
Core equivalent diameter [m]	3.16
Fuel rod assemblies	163
Fuel assembly	Hexagonal
Maximum fuel enrichment [%]	5
Number of absorber stacks	121
Average discharge burnup [MWd/kg]	60
Fuel	UO_2
Duration of burnup campaign [months]	12÷24
Fuel amount [t UO_2]	87
Reactor vessel	
Inner diameter of the barrel [mm]	4250
Barrel wall thickness [mm]	200
Gross height [mm]	11185
Main circulation pumps	
Number of pumps	4
Rated flow [m^3/h]	21500
Pressurizer	
Gross capacity [m^3]	79
Design pressure [MPa]	17.6
Steam generators	
Number of steam generators	4
Type	Horizontal, with U-shaped pipes
Maximum outer diameter [mm]	5100
Gross length [mm]	13820
Inner containment	
Make	Prestressed concrete with steel lining
Capacity [m^3]	74169
Outer containment	

GENERAL SPECIFICATIONS

Make	Ferroconcrete

2.3.2.2.3 Turbine plant

The steam turbine is Mod. K-1200-6.8/50, condensation type, with one shaft and five cylinders. Initially the steam expands in a HPC and then in four LPCs. The steam undergoes intermediate superheating and separation before in enters the LPCs.

The steam produced in the steam generators is conveyed to the turbine by four steam lines provided with check valves and stop valves. The steam pressure after the SGs is 6.8 MPa. After expansion in the HPC, the steam proceeds to 4 steam manifolds and thence to 4 individual single-stage steam separators-superheaters, where the moisture is separated from the steam and the steam is reheated before it enters the LPC. After expansion in the LPC the steam enters the condenser. The condenser is designed to receive steam during transient processes and unit startup as well. There is also a system for regenerative heating of SG feed water using steam exhausted from steam extractors in the turbine. The turbine spins at 3000 RPM. The turbine is coupled to an asynchronous AC generator type T ZV-1200-2UZ. The arrangement of the turbine generator is shown on **Figure 2.3-21.**

TABLE 2.3-2: MAIN CHARACTERISTICS OF THE TURBINE

Feature	Value
Type	K-1200-6.8/50
Number and type of individual sections	2LPC-HPC-2LPC
Length	52.3 m
Speed	3000 RPM
Temperature at HPC inlet	283.3°C
Pressure at HPC inlet	6.8 MPa
Steam flow	6408 t/h

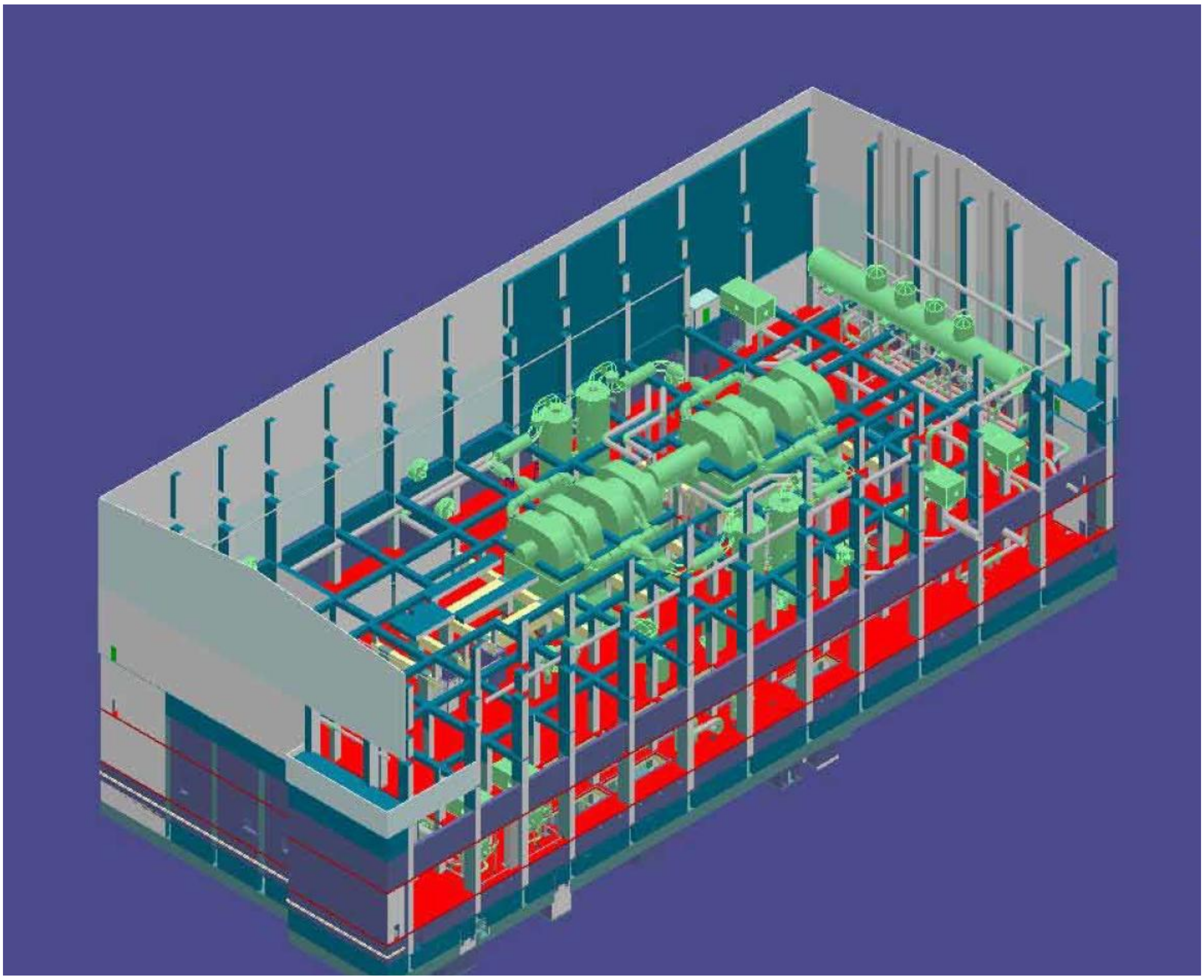


FIGURE 2.3-21: AES-2006 TURBINE GENERATOR ARRANGEMENT

2.3.2.2.4 Steam generators feed system

The feed system is provided with 4 electric pumps (25% capacity each) and one redundant pump (25% capacity). All pumps are connected by separate pipelines and valves. The system feeds the SGs during rated operation, startup and shutdown of the unit. The system is designed to trip automatically if a SG is overfilled or pipeline ruptured.

TABLE 2.3-3: MAIN CHARACTERISTICS OF SG FEED PUMPS

Feature	Value
Type	Electric
Number of operating pumps	4/25%
Number of redundant pumps	1/25%
Thrust at rated parameters	850 m
Speed	3000 RPM
Flow at rated parameters	0.569 m^3/s

The condenser is dual-flow type, pressure rating 4.9 kPa.

2.3.2.2.5 Auxiliary SG feed system

The auxiliary system is used to feed the SGs when the reactor system operates at low capacity: startup, shutdown and hot reserve. The system comprises two pumps and related pipelines and fittings. The main reason for using these pumps at low capacity is that their flow rates are commensurate with the low steam output from the SGs.

2.3.2.2.6 Generator and main electric equipment

The main characteristics of the generator are:

Feature	Value
Type	T ZV-1200-2UZ
Length	22.2 m
Power rating	1333MVa
Active power	1200 MW
Voltage	24 kV
Frequency	50 Hz

2.3.2.2.7 Main AC system

The house power system provides rated voltages 24 kV, 10 kV, 400 V, 230 V.

When the unit is operated at rated conditions, the turbine generator is connected to the grid via the main transformer. It consists of four separate transformers rated 533 MVA. The generator powers two additional transformers rated 80 MVA, which supply the loads via four 10 kV sections. Each section can also be supplied by the backup transformer of the unit.

2.3.2.2.8 Diesel generator for rated operation

One auxiliary diesel generator of voltage rating 10 kV is installed in order to supply certain loads from three sections. Upon loss of power from the auxiliary or redundant transformers, these sections are automatically switched over to power supply from this DG.

2.3.2.2.9 Emergency power supply

The power plant has the following power supply systems:

- Rated power supply to Group 3 loads;
- Secure power supply for loads important to safety (Group 2 and Group 1) during rated operation;
- Emergency power supply to Group 2 and Group 1 loads.

Once the emergency reactor protection is activated, the system shuts down the unnecessary Group 3 loads. In accident conditions, Group 2 loads receive emergency power after a certain period of time, while Group 1 loads do not tolerate blackouts and the system provides them with uninterrupted power in all operating modes.

The system for emergency power supply of loads included in the safety systems has 4 trains, each train consisting of:

- One diesel generator of voltage rating 10 kV;
- One 10 kV section, normally supplied from auxiliary or redundant transformers in rated operating conditions;
- Uninterrupted power supply equipment;
- Rechargeable batteries.

In rated conditions, Group 2 loads are supplied by the auxiliary or redundant transformers. Upon loss of external power, these loads are supplied from relevant DGs – one in each of the four trains. The diesel generators are autonomous facilities equipped with all services required for their autonomous operation.

Group 1 loads that do not tolerate blackouts are connected to the uninterrupted power supply system. The system has four independent trains, each one consisting of rechargeable battery, rectifier and inverter, which supplies the AC loads.

2.3.3 SNF

According to the provisions of the project, each of the considered nuclear unit alternatives includes a spent fuel pond. The fuel resides in such ponds for 3 to 5 years after which it may be transfered outside the facility. The spent fuel pond (SFP) provides space for placement of fuel assemblies during unit outage works and for underwater storage of activated components. The safety requirements for the SFP include maintenance of 5% subcriticality in normal operating mode and in case of design basis accidents. This requirement is met by:

- → Limitation of the spacing of SNF assemblies storage seats;
- → Controlling the location of the SNF assemblies and restricting the possible displacements during transportation at the site, SNF handling and storage under normal operating conditions and under external impacts;
- → controlling the parameters of the systems (elements) which affect nuclear safety during SNF management.

Details of the NNU alternatives concerning SNF storage are provided in **Table 2.3-4.** As seen from the table, the available capacity is sufficient for SNF storage in the course of at least 10 years. This period of time is considered sufficient for deciding the next steps to be taken in respect of SNF management.

TABLE 2.3-4: *SPENT FUEL POND*

MODEL	Location of SNF pond	Number of fuel assemblies in the reactor core	Number of seats in the SNF pond	Approx. duration of fuel campaign and proportion of core refueled	Approximate capacity
AP-1000	**Dedicated building outside the containment**	157	889 (157 will be left redundant for emergency evacuation of the core)	18 months 1/3 of the core	17 years
AES-92 Hybdrid	**Inside the containment**	163	580 (163 redundant for emergency evacuation of the core)	12 months 1/3 of the core	10 years
AES-2006	**Inside the containment**	163	(*) (163 redundant for emergency evacuation of the core)	18 months 1/3 of the core	10 years

(*) No data available at present

The spent nuclear fuel and radioactive waste management strategy of the Republic of Bulgaria envisages an *open fuel cycle*/*once-through fuel cycle*. Essentially, this solution is not a cycle. After the fuel has been used, it is deposited in storage facilities, without any further processing other than packaging to provide better insulation of the radioactive substances from the biosphere. This is the preferred method for six countries: the United States, Canada, Sweden, Finland, Spain, and South Africa. Some countries, in particular Sweden and Canada, have designed such storage facilities to enable future use of the nuclear material in case of need, while other countries plan permanent disposal in a geologic depot. In the Republic of Bulgaria, spent nuclear fuel is considered a usable resource, which may be processed to benefit the country. The Strategy envisages storing the SNF in intermediate storage facilities, using dry storage as preferred technology.

The next **Table 2.3-5** shows the estimated number of casks required for dry storage of spent nuclear fuel during the service life of the new capacity (60 years), depending on cask models. As assessment of the applicability of these casks for each technology in the specific Bulgarian circumstances will be undertaken at a later stage.

TABLE 2.3-5: ESTIMATED NUMBER OF SNF DRY STORAGE CASKS REQUIRED DURINGTHE OPERATION OF THE NNU

Cask model	Fuel assemblies per cask	Most frequently used refueling strategy (fuel cycle months, approx. proportion of core refueled)	Number of fuel assemblies in the reactor core	Number of dry storage casks
GNS CASTOR®1000/19 for WWER	19	12 months 1/3 of the core	163	216
GNS CONSTOR®1000/19 for WWER	19	12 months 1/3 of the core	163	216
VSC-VVER-24 for WWER	24	12 months 1/3 of the core	163	171
Holtec International HI-STORM 100 MPC-24 for AP1000	24	18 months 1/3 of the core	157	97
NAC's MAGNASTOR for AP1000	37	18 months 1/3 of the core	157	63
Holtec International HI-STORM 100 MPC-31 for the Hybrid reactor	31	12 months 1/3 of the core	163	133

Availability of a dry spent fuel storage facility for the proposed models is important, especially until a national decision for the future use of SNF is taken and/or until a storage facility for highly radioactive waste is constructed.

The considered NNU alternatives feature different SNF dry storage solutions:

With **AES–92** the dry storage facility is a separate building with capacity covering 20-year service life of one unit (72 casks). In the existing WWER-1000/V466B designs, the facility is common for the two units, assuming that it may be expanded further on to provide sufficient capacity for the entire service life. In the by the reactor pools, the fuel is placed in CASTOR1000 casks which are then transferred to a dry storage facility.

AP-1000 provides conceptual designs for dry storage facilities. With AP-1000, the cask handling equipment can be adapted to various cask types. The SNF dry storage system will vary depending on plant preferences, regulatory requirements and technological developments. Nevertheless, the AP-1000 version selected for the United Kingdom is based on the HI-STORM 100U underground dry storage system developed by *Holtec International*.

HI-STORM 100U is a vertical, ventilated dry storage system. Holtec and Westinghouse have confirmed that the Holtec equipment can be deployed such that the spent fuel can be loaded and transferred from the spent fuel pond to an underground facility for interim storage. The system consists of three major components:

1. Underground vertical ventilated module (UVVM)

The UVVM is designed for storage of Multi-Purpose Canisters (MPC) in vertical configuration, situated entirely under the ground surface, in a subsurface cylindrical void. The main purpose of the UVVM structure is to ensure biological protection and cooling.

2. HI-TRAC transfer cask, which holds the MPC during handling operations

HI-TRAC is the name of a transfer cask or "shuttle cask" of the HI-STORM 100U system. HI-TRAC is a small-diameter cylinder with removable top and bottom hatch. HI-TRAC is used for transfer and extraction of MPCs.

3. MPC containing SNF assemblies

The MPC and HI-TRAC components of the HI-STORM 100U are 100% identical to those of Holtec's aboveground system, which is being used for several years already. MPC is a single package, equally suitable for on-site storage, transportation and final disposal in the future storage facilities. The MPC is made entirely of stainless steel alloys except the fixed neutron absorber Metamic™, which is installed inside the canister for criticality control.

Conclusion

The experience of NPP Kozloduy in the area of SNF management (detailed description provided in **Chapter 1** of the EIAR), including the constructed infrastructural facilities, can be used for managing the SNF from the NNU. Their specific application must be supported with relevant safety assessments.

2.3.4 GRID FOR EVALUATION OF THE EXPECTED IMPACTS IN TERMS OF RELEASES (EMISSIONS) FROM THE ALERNATIVE TYPES OF REACTORS ON ENVIRONMENT COMPONENTS AND FACTORS

The expected impacts from the operation of the NNU in terms of releases (emissions) from the alternative types of reactors, as considered in this section, on the various components and factors of the environment are summarised in the next grid.

GRID FOR EVALUATION OF THE EXPECTED IMPACTS IN TERMS OF RELEASES (EMISSIONS) FROM THE ALERNATIVE TYPES OF REACTORS ON ENVIRONMENT COMPONENTS AND FACTORS

Environment component/factor		AES-92	AP-1000	AES-2006
Ambient air	Non-radiation aspect		☐	
	Radiation aspect	☒	☒	*Similar to AES-92*
Conventional water		☒ *Based on data from: *Similar NPP project Kudankulam, India; *NPP Kozloduy – Units 5 and 5*	☒	*Similar to AES-92*
Earth and soil			☐	
Earth interior			☐	
Conventional waste			☐	
Radioactive waste		☒	☒	☒

Environment component/factor	AES-92	AP-1000	AES-2006
Dangerous substances		☐	
Biological diversity		☐	
Landscape		☐	
Health/sanitary aspects and radiation risk to population	☒	☒	*Similar to AES-92*
Cultural monuments		☐	
Harmful physical factors		☐	

☒ – *perform detailed assessment and modeling*

☐ – *use assessments and forecasts for other components/factors*

Similar to AES-92 – the similarity in respect to each factor or component is explained in EIAR Section 4

On the basis of the analysis in the grid, detailed assessment and modeling of the three proposed reactors will be undertaken in respect of:

- ✓ Ambient air in radioactive aspect;
- ✓ Groundwater in radioactive aspect;
- ✓ Radioactive releases;
- ✓ Health/sanitary aspects of the environment and risk to the population

The assessments and conclusions made in respect of the above components/factors in Section 4 of this EAIR, will also be used to estimate the impact on:

- ✓ Earth and soil;
- ✓ Earth's interior;
- ✓ Landscape;
- ✓ Biodiversity;
- ✓ Non-radiation waste;
- ✓ Dangerous substances;
- ✓ Harmful physical factors,
- ✓ Immovable cultural heritage.

2.4 THE ZERO ALTERNATIVE

Taking into account the Governmental decision to renounce the Belene NNP Project and build a NNU at Kozloduy NPP site instead, using the nuclear equipment produced for the Belene NNP Project, as well as the Decision of the Council of Ministers adopted by Protocol No. 14 of April 11, 2012 for consent in principle to undertake actions required for the construction of a new nuclear capacity at Kozloduy NNP, the zero alternative has become a practically implausible option.

In this context, the following two options are theoretically available:

1. Try to find another site for construction of the required nuclear capacity elsewhere in the country;

2. Completely put an end to all surveys and activities for building a new nuclear capacity anywhere in the country.

The first option may be considered on purely theoretical basis. NPP Kozloduy is the only operating licensed site, where the greater part of the associated infrastructure required for the implementation of a new nuclear unit has already been built.

Actually, the "zero" alternative, or a decision to not undertake any actions for the implementation of this proposed investment project at the NPP Kozloduy site, is tantamount to relinquishing the construction of any new nuclear capacity in the country in the foreseeable future. Such decision contradicts the objectives laid down in the country's National Energy Strategy for launching new nuclear capacities and increasing the share of electric energy generated by nuclear power plants by 2020.

Of the two options outlined above, the second option remains a plausible one, but only if considered detached from the country's energy needs. From the viewpoint of the electric energy sector, abandoning the possibility to build a new nuclear unit means to build a new non-nuclear capacity of electric output 1000–2000 MW. Taking into account the country's energy resources, the required new energy capacity will most probably have to be provided by thermal power stations, which will be located elsewhere. This will require surveying a new site and new planning, technical works, preparation of the site and construction to a tight schedule, in view of building a thermal power plant with output of 2000 MW.

Environmental consequences of the Zero alternative

The building of new capacity replacing NPP Kozloduy, in case of abandoning the option for construction of nuclear capacity, could theoretically be achieved following different choices, the most probable of which is a new thermal power plant, taking into account the energy resources and the fuel-energy balance of the country.

The building of a new thermal power plant of 1000–2000 MW output will cause problems in the energy sector at least in two aspects:

- A new thermal power plant with the mentioned capacity means a new large combustion plant (LCP), which should comply with the requirements of Directive 2001/80/EC on the limitation of emissions of certain pollutants into the air from large combustion plants. A more practical and cost-effective choice for the country is to build a thermal power plant (TPP) fired by lignite coal, but the research revealed that, with such fuel, the existing TPPs exceed the allowed emission limit. Therefore, it is not practical and feasible to build a new lignite-fired TPP with such capacity, since such a project would not comply with the Directive's requirements. A TPP of such capacity based on imported good-quality coal or imported gas would hardly be an economically preferable solution;
- The building of a new LCP – a lignite-fired TPP with capacity of 1000–2000 MW means more GHG emissions and reduction of Bulgaria's chances to honour its commitments under the UN Framework Convention on Climate Change and the Guide of the Intergovernmental Panel on Climate Change (IPCC), and other international treaties and programs for protection of the environment.

For the purposes of analysing the environmental consequences of these two aspects, it would be expedient to present some global and national data.

On the one hand, it should be emphasized that energy demand and installed power capacity worldwide are expected to increase during this century. This will result in abrupt increase of the combustion of conventional fuels, which will aggravate further the global negative effects related to greenhouse gases and climate change.

Against this background, the minimization of environmental impacts and, in particular, the savings of greenhouse gases with each new power capacity becomes essential, especially on continental/regional and national scale. The latter becomes even more evident taking into consideration the commitments under the UN Framework Convention on Climate Change, the Kioto Protocol, etc.

In this context, the consequences of electric power generation by TPP and NNP will be benchmarked.

The controlled nuclear fission reaction produces vast amounts of electrical power out of small amounts of uranium fuel, with relatively small amount of generated radioactive waste. Compared to the waste from coal-fired plants, the volume of NPP waste is more than 30 000 times less than the volume of TPP waste.

The effect of large-scale use of nuclear energy on emission reduction becomes particularly evident, if we compare the generated national product per unit of emitted carbon dioxide. The data about 6 countries worldwide, which account for over 60 % of world economy and in which nearly half of the world's population lives, demonstrate that the amounts of CO_2 emissions are inversely proportional to the relative share of nuclear power in the relevant national energy sector. Moreover, the experience of the greatest emitter of carbon dioxide in the world, the USA, shows that more than 75 % of all "emission savings" in the production of electrical energy are due to nuclear power, and less than 1 % – to solar and wind energy.

These examples of the role of nuclear energy in the "power engineering – environment" system, as well as the numerous related publications of a number of international organizations, such as the IAEA, Euratom, the World Nuclear Association and more are indicative of environmental impact ratios and the conclusions derived therefrom are largely applicable to each country.

With respect to nowadays conditions in our country, the commitments of the Bulgarian Government in the field of power engineering and environment relating to the closure of Chapter *Environment* and the negotiations on the preparation for Bulgaria's accession to the European Union, and more specifically, the transposition into the national law of Directive 2001/80/EC on the limitation of emissions of certain pollutants into the air from large combustion plants should be emphasized. On the other hand, the building of a new 2000 MW capacity is related with certain time limits under the *Plan for cost-efficient development the electrical energy sector of the Republic of Bulgaria during the period 2004–2020*.

To compare the environmental impacts of a new lignite-fired TPP of commensurate capacity, if such were to be built in our country, to the one proposed by the Investment Project for building of NPP Kozloduy, the actual impact parameters of the largest TPP operating in the country, Maritsa-East 2 TPP, will be used, notwithstanding that its installed capacity of 1587 MW is approx. 24.4 % greater than the envisaged 1 200 MW for the NPP Kozloduy site.

The conducted national-scale survey regarding the possibility to apply Directive 2001/80/EC demonstrates that Maritsa-East 2 TPP is the number one energy source of SO_2 emissions and number four source of NO_x emissions, generating accordingly 44 % and 11 % of these emissions in the country. Obviously, from this point of view, it is not acceptable to build in the country yet another emitter of such capacity, especially in view of Bulgaria's commitments to reduce SO_2 emissions.

Only one example with Maritsa-East 2 TPP would suffice to assess the "GHG savings" effect from the operation of Kozloduy NNP compared to a lignite-fires TPP of equal capacity:

Following the rehabilitation of Units 1-6, the installed electrical capacity of Maritsa-East 2 TPP was raised to 1600 MW[3]. During the period 1996–2000, the TPP has had 6.995 million tons of average coal consumption from the Maritsa–East coal-field. With average thermal capacity of the coal of 6.12 kJ/g and C=18.4 %, the calculated average annual CO_2 emissions amount to about 4.724 millon tons. These emissions account for 38 % of the greenhouse gas emission reduction required by the Kioto Protocol for Bulgaria – 12.4 million tons of CO_2-equivalent. Obviously, from this point of view as well, it is not acceptable to build in the

[3] http://www.tpp2.com/page/equipment-and-installations.html

country yet another emitter of such capacity, which would increase the annual CO_2 emissions from Bulgaria by approximately 4.7 million tons.

Consequently, it could be summarized that, from the viewpoint of GHG, SO_2, NOx, dust, and other emissions, the "zero" alternative of replacing the new nuclear capacity at the NPP Kozloduy site, which does not emit into the atmosphere such harmful substances, by a TPP of equivalent and even lesser capacity, is not advisable.