Status report 84 - VVER-300 (V-478) (VVER-300 (V-478))

Overview

Full name	VVER-300 (V-478)
Acronym	VVER-300 (V-478)
Reactor type	Pressurized Water Reactor (PWR)
Coolant	Light Water
Moderator	Light water
Neutron spectrum	Thermal Neutrons
Thermal capacity	850.00 MWth
Gross Electrical capacity	300.00 MWe
Design status	Detailed Design
Designers	Gidropress
Last update	07-06-2011

Description

Introduction

The design of the two-loop Reactor Plant (RP) with VVER-300 is based on engineering solutions for the equipment of previous designs of RP with VVER. The design of the V-407 is taken as reference.

The design of the VVER – 300 (V-478) is based on the concepts:

- design is developed for the regions with smaller power grids;
- structure, materials, heat-engineering parameters of loop main equipment – Steam Generator, Reactor Coolant Pumps and Main Coolant Pipelines (SG, RCP set, MCP) is optimally unified with similar equipment used in the design of VVER-640 (V-407);
- core is designed on the basis of Fuel Assemblies (FA) similar to the VVER-1000 FA with long-term experience of nuclear power plant (NPP) operation;
- corium catching within reactor vessel due to vessel external cooling under severe accidents;
- isolation of primary-to-secondary leak without radioactive releases into atmosphere.

Description of the nuclear systems

2.1. Reactor coolant system main parameters

Reactor coolant system provides core heat removal via coolant closed circuit and heat transfer to the second side. Reactor coolant system comprises a reactor, a pressurizer with pipelines and valves and two circulation loops, each comprising a steam generator, a reactor coolant pump set and a main coolant pipelines connecting loop equipment with the reactor. Pressurization system is a constituent part of the reactor coolant system and provides its pressurization, pressure maintenance under stationary conditions, pressure control under heat-up and cooldown, limitations of pressure deviation under transients and accident conditions.

Reactor coolant system main design and thermal-hydraulic parameters under RP normal operation conditions are presented in Table 1. Reactor coolant system schematic flow diagram is presented in Figure 1.

Parameter	Value
Reactor nominal thermal power, MW	850
Coolant volume in the reactor coolant circuit (without PRZ system), m^3	133
PRZ volume, m^3	31
Coolant pressure at the core outlet in design conditions, absolute, MPa	16.2
Steam pressure at SG steam header outlet, MPa	7.0
Coolant temperature at the reactor inlet, °C	295
Coolant temperature at the reactor outlet, °C	325
Coolant flowrate through the reactor, m^3/h	22730
Primary calculated parameters: - overpressure, MPa; - temperature, °C	 17.64 350

Table 1

2.2.Reactor core

Reactor core comprises 85 fuel assemblies (FA). FA are proposed designed for heat generation and transfer from fuel rod surface to the primary coolant during design service life without exceeding of fuel rod damage permissible design limits. Fuel stack nominal height is 3550 mm. Each FA contains 312 fuel rods. FA frame is assembled from 18 guiding channels with 13 welded spacing grids, instrumentation channel, lower grids. Fuel rod cladding are tubes of zirconium alloy. Sintered pellets with a maximum U^{235} enrichment of 3.3% are within the claddings.

According to the chart diagram, the core comprises 34 control and protection system control rod (CPS CR) proposed for fast termination of core nuclear reaction, power maintenance at assigned level and transfer from one level to another, core axial power field equalization, prevention and suppression of xenon oscillations. Pitch electromagnet drives with position indicators are used as drive mechanisms of CPS CR assemblies. CPS CR drives are installed on the reactor top head. Average burn-up of unloaded fuel amounts to 38 MW·day/kg U. The number of fresh FAs loaded annually into the core for base fuel cycle is 24 assemblies.

2.3. Nuclear fuel handling systems

The complex of nuclear fuel storage and handling systems is a group of systems, facilities, components proposed for nuclear fuel storage, loading, unloading, transfer and monitoring.

The complex includes the following systems for fulfilment of all nuclear fuel procedures within power station:

- system of fresh (non-irradiated) nuclear fuel storage and handling;
- core refueling system;
- system of spent nuclear fuel ex-pile storage;
- system of nuclear fuel transportation (transfer) within NPP site, starting form the receiving of special transport with fresh fuel and ending by the shipment of special transport with spent nuclear fuel;
- organization of nuclear fuel account and monitoring in NPP power unit.

2.4. Description of reactor coolant system components

2.4.1. Reactor

The reactor is a vertical high-pressure vessel (with reactor top head) comprising internals (protective tube unit, core barrel, core baffle), core, control rods and in-core instrumentation detectors. Vessel is tightened with the reactor top head comprised in the top head unit using main joint studs. Drive mechanism housings are installed on top head nozzles (CPS nozzles). Electromagnet units proposed for CPS control rod core axial displacement are fastened outside the housings.

The reactor is housed inside the concrete cavity provided with biological and thermal shields and the cooling system. The reactor vessel is supported with a supporting shoulder on the supporting ring, and fastened in the supporting truss. The reactor is prevented from lateral displacements by the thrust ring installed on the vessel flange.

Reactor fastening in the concrete cavity at two levels provides safe prevention from displacements at seismic impacts and pipeline breaks. Cooling of concrete cavity, electric equipment, in-core instrumentation nozzles and drives is provided with air.

Reactor design is presented in Figure 2.

Reactor comprises the following main units:

- vessel with main sealing components;
- upper unit with CPS drives;
- internals (core barrel, core baffle, protective tube unit);
- core comprising 85 FAs.

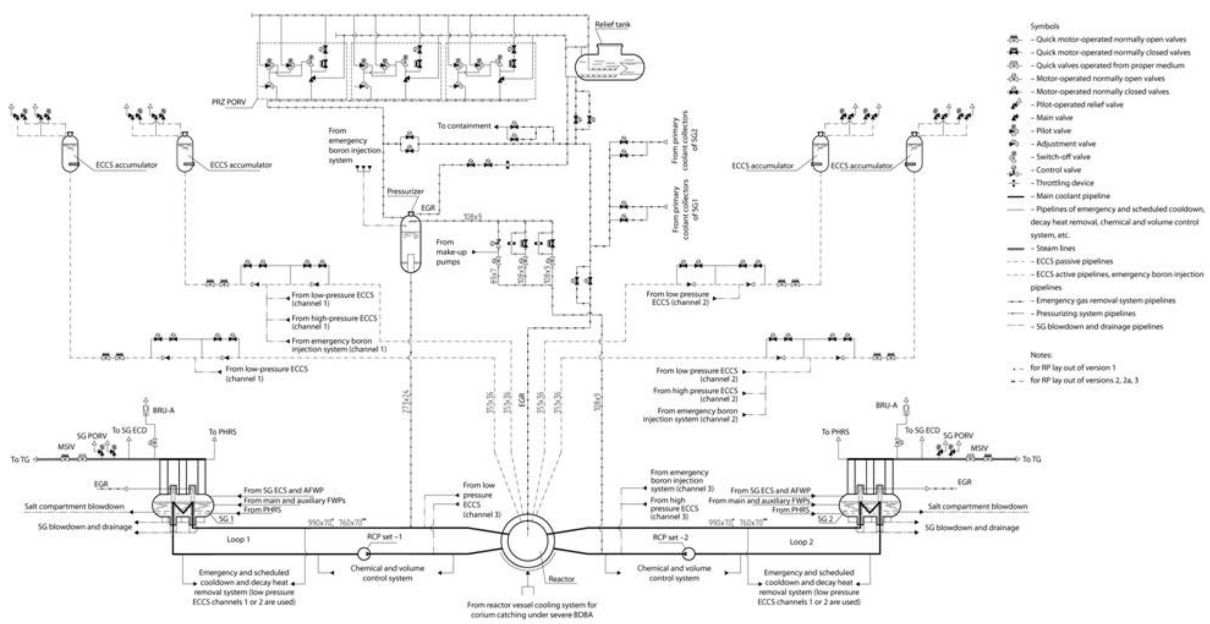


Figure 1 – Reactor coolant system schematic flow diagram

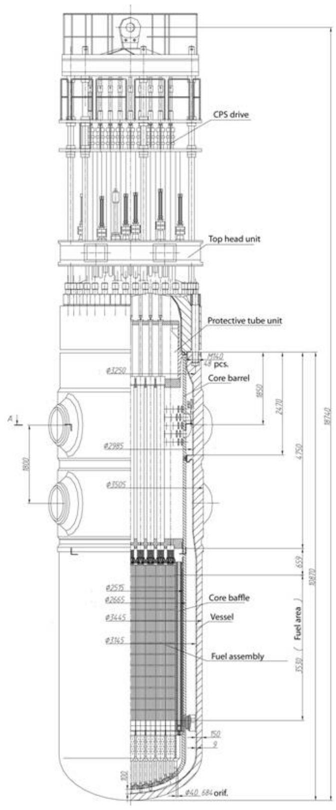


Figure 2 – Reactor

2.4.2. Steam generator

The steam generator (model PGV-640) is proposed for generation of dry saturated steam removing heat from the reactor coolant system. Design of steam generator PGV-640 (Fig. 3) is based on analysis, experimental examination and operational experience of steam generators PGV-1000 (PGV-1000M).

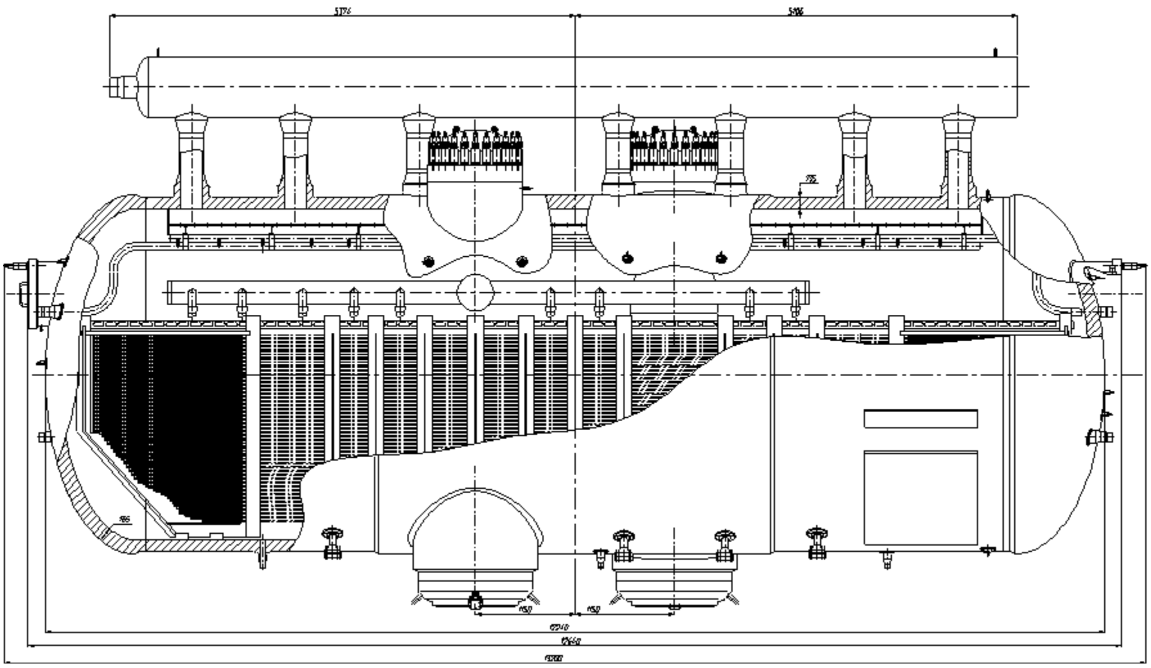


Figure 3 - Steam generator PGV-640

In-line arrangement of heat-exchanging tubes in the tube bundle designed in steam generator PGV-640 and meeting of SG and secondary side water chemistry standards are the conditions of SG tube bundle reliable operation within 60 years.

The steam generator is a single-unit heat-exchanging device of horizontal type with submerged heat exchanging surface comprising the following main units:

- vessel with multipurpose nozzles;
- heat-exchanging tube bundle with fixing and spacing components;
- primary steam headers;
- devices of feedwater supply and distribution under accident conditions;
- distribution perforated sheet;
- submerged perforated sheet;
- chemicals supply devices.

2.4.3. Reactor Coolant Pump

The reactor coolant pump set (RCP set) proposed provides plant primary coolant circulation. RCP set has a sub-function to provide coolant circulation during coastdown under various accidents with loss of power that allows smooth transition to natural circulation.

GTSNA-1455 is a vertical pump unit comprising the pump with shaft mechanical seal, drive-operated electric motor with a flywheel and auxiliary systems. Pump bearing units lubrication and cooling is provided with water and electric motor – with flameproof oil of Fyrquel type.

GTSNA-1455 has an axial flow part, suction is provided sideways and pressure from below. Pump case is solid-forged with delivery and suction nozzles Dnom 620.

Parameter	**GTSNA-1455**

	(VVER-640)
Current	Three-phase alternate
Nominal voltage feed current, V	6000
Feed current frequency, Hz	50
Rotation speed (synchronous) of RCP set drive motor, rpm	1500
Motor nominal power, kW	2800
Lubrication type	Fyrquel, oil bath
Nominal supply (four RCP sets in operation), m^3/h	13420
Pressure (at nominal supply), MPa	0.349±0.025
Coolant temperature at the pump inlet, °C	300
Suction pressure, MPa	15.3

2.4.4. The main coolant pipeline (MCP) Connects the reactor, steam generators and reactor coolant pumps comprised into the reactor coolant system proposed for coolant circulation from the reactor to steam generators and in reverse direction. Long-term experience of MCP design and operation in NPP has been used for MCP development.

MCP comprises two circulation loops. Each loop has three tube sections. The section between reactor discharge nozzle and steam generator inlet nozzle is a "hot" leg. The section between steam generator outlet nozzle and RCP set inlet (suction) nozzle and the section between RCP set outlet (pressure) nozzle and reactor inlet nozzle is a "cold" leg. Internal diameter (620 mm) is selected from the condition of primary coolant admissible velocity and MCP flow resistance for coolant flow design range. MCP material is stainless steel.

"Hot" leg of loop 1 is connected with the pressurizer by pipeline 273x24 mm (connecting pipeline).

2.4.5. PRESSURIZER (PRZ)

The Pressurizer is a cylindrical vessel with elliptic pressurized bottoms of vertical position, installed on the cylindrical support. PRZ main dimensions and technical data are presented in Table 2.

Parameter	Value
Nominal pressure of stationary conditions, MPa	16.2
Nominal temperature of stationary conditions, ^oC	348
Internal diameter, mm	1984
Accumulator (total volume), m^3	31
Water volume at operation under the rated power, $м^3$	21.5
Working medium	Steam, water, under heat-up/cooldown conditions – nitrogen
PRZ TEH total power, kW	1500

Table 2

2.5 Auxiliary systems

Complex of nuclear fuel storage and handling systems is described in section 2.3.

Compressed air system for pilot–operated valves is proposed for quick-acting isolation valve air actuators supply with compressed air under pressure from 4.5 to 5.0 MPa.

System of nitrogen supply to blowoffs from reactor building equipment is proposed for nitrogen supply to gas blow-off system.The gas blow off system is proposed for collection of air and gases from the primary equipment (in filling the primary circuit) and their transfer to the hydrogen burning system. On completion of hydrogen burning the blowoffs are transferred to the decontamination system for reducing the activity of inert gases, gaseous iodine and aerosols to the permissible levels.

Filtering and vent systems provide normalized weather conditions for personnel operation under plant various operating conditions, room atmosphere cleanness, prevent atmospheric air pollution with radioactive substances and toxicants, maintain optimum operating conditions of process equipment operation.

Other auxiliary systems are also proposed to provide NPP normal operation.

2.6. Operating conditions

List of normal operation design conditions is presented in Table 3.

Conditions		Remarks
1	Stationary conditions (considering net frequency variation in the range 48,0-51,5 Hz), including power variations ±1 % Nnom at the rate of 1% Nnom/s	The cycle means power variation and return to initial level. One RCP set operation is permissible
2	Unit power variation is not below ±2% Nnom and not above ±5% Nnom (network frequency maintenance mode)	Initial power is lowered to 95-98 %Nnom depending on control range. The cycle means power variation and return to initial level as per the pattern 95-100-90-95% Nnom or 95-90-100-95% Nnom
3	Unit power variation is at the rate not below 1% Nnom/min and not above ±5% Nnom/min with deviation from current value not above ±10% Nnom	Initial power is lowered to 90-95% Nnom depending on control range. The cycle means power variation and return to initial level as per the pattern 90-100-80-90% Nnom or 90-80-100-90% Nnom
4	Unit power variation according to scheduled (operations schedule) and to not scheduled load variation at the rate not above 5% Nnom/min in the range from 50 to 100% Nnom	The cycle means power variation and return to initial level as per the pattern 100-50-100% Nnom.

Conditions		Remarks
5	Power lowering at the rate of 20 % Nnom/min in accident situations of power supply system. Initial state – operation at power in the range from 20 to 100% Nnom	Power increase after the mode completion is provided with scheduled velocity to 5% Nnom/min
6	Unit power variation by ±10% Nnom at the rate of 5% Nnom/s: - power variation by -10% Nnom; - power variation by +10% Nnom	The cycle means power variation and return to initial level
7	Unit power variation by ±20% Nnom at the rate of 10% Nnom/min: - power variation by -20% Nnom; - power variation by +20% Nnom	The cycle means power variation and return to initial level
8	Spurious scram	-
9	Spurious operation of accelerated preventive protection (drop of CPS CR group)	Reaching of N=60% Nnom
10	Drop of CPS control rods (single)	N=80% Nnom (after flaw elimination N=100% Nnom)
11	RCP set startup according to the schedule	-
12	RCP set scheduled trip after power level lowering to required limits	-
13	HPH bypass connection	-
14	End of life operation with overshoot of reactivity	Coolant temperature at the reactor inlet is nominal

Table 3

Considering possible operation of NPP with VVER-300 in the regions with power grids of limited capacity, meeting of the requirements for equipment and fuel operation under network load follow mode and frequency automatic maintenance is required according to the following conditions:

- power variation range is ±2.5% Nnom, number of the cycles of power variation at the rate of 2.5% Nnom/min is not limited, core inherent control features are used, CPS CRs are not used, boron control system is not used;
- power variation range is 5-10% Nnom, number of the cycles of power variation at the rate £ 2.5% Nnom/min is not limited, CPS CRs are used, boron control system is not used;
- power variation range is 100-80% Nnom (daily control), permissible number of the cycles of power variation at the rate 1% Nnom/min - 2000, CPS CRs are used, boron control system is not used;
- power variation range is 100-50% Nnom (daily control), permissible number of the cycles of power variation at the rate 1% Nnom/min - 500, CPS CRs are used, boron control system is used.

VVER-300 RP fuel and equipment load following capabilities are not lower than VVER-440 RP capabilities (with power variation increased velocity and number of permissible cycles under daily and weekly control).

2.7 Standard fuel cycle

The standard fuel cycle is not closed. The length of the cycle is 12 months. The time of fuel residence in the core (fuel life) is 8 years.

2.8. Spent fuel and plans of fuel removal

Spent fuel is unloaded from the reactor and installed for storage into spent fuel pool (SFP) located in the reactor building under the containment near reactor cavity.

Besides, test of defective fuel elements and a separate area for leaking fuel assemblies storage in tight bottles are provided in the spent fuel pool.

SFP lay-out prevents displacement of transport containers and other consignments above stored spent nuclear fuel during fuel transfer from the reactor building.

Polar crane installed in containment building is used for fuel transport containers transfer.

Transfer of the fuel stored in SFP into spent nuclear fuel storage (SFS) is provided during reactor refuelling.

SFS is proposed for spent nuclear fuel (SNF) dry storage at NPP site in dual-purpose containers for transportation and storage.

SFS volume is aimed for long-term storage of spent nuclear fuel accumulated within 10 years of two power units operation with the possibility of building volume increase in future for storage of nuclear fuel accumulated within the total period of power units operation.

For the description of SNF handling systems refer to Section 8.

2.9 Examples of power systems with NPP of similar type

An experience base for VVER reactor plant operation of about 1400 reactor-years (including decommissioned VVERs) has been gained up to now.

3.1. Safety concept, designing philosophy and licensing methods

3.1.1. Design Guidelines

The design is developed according to the requirements of Russian Federation current regulations and standards for atomic energy and considering IAEA safety regulations and other recommendations as well as requirements of European operators for LWR NPP (Requirements EUR Volume 2, Chapter 2.1).

Generally recognized integrated safety purposes, principles and criteria are realized in NPP design according to requirements /1/. Defense-in-depth concept is used in the design.

3.1.2. Design simplicity and reliability

Simplicity and reliability of safety systems is provided by use of reliable equipment and operation of passive safety systems alongside with active systems. The passive systems do not require operation of other systems for their work.

3.1.3. Active and passive systems, inherent safety features

The following active, passive safety systems and beyond-design control systems of power unit are used in V-478 design:

- steam generator emergency cooldown systems;
- emergency gas removal system;
- emergency boron injection system;
- reactor primary coolant and spent fuel pool emergency and scheduled cooldown system;
- main steam lines isolation system.
- emergency core cooling system, the passive part;
- core cooling system with high-pressure pumps;
- core cooling system with low-pressure pumps
- primary overpressure protection system;
- secondary overpressure protection system
- passive heat removal system;
- double containment
- the system of corium catching within the reactor vessel for severe beyond-design accidents.

Besides, the design stipulates primary and secondary overpressure protection systems comprising the components (PORV) capable of operating by automatic actions, by reaching specified setpoints (active and passive operating actions).

3.1.4. Defence-in-depth description

NPP safety shall be provided due to successive realization of the defence-in-depth concept using the system of physical barriers from propagation of ionizing radiation and radioactive materials into the ambient air and the systems of engineering and organizational measures for barriers protection and maintenance of their efficiency, and also for protection of staff, population and environment.

The system of NPP power unit physical barriers includes fuel matrix, fuel rod cladding, reactor coolant pressure boundary, reactor plant pressurized containment and biological shielding.

The system of engineering and organizational measures shall provide five levels of the defence-in-depth and to include the following levels.

Level 1 (Conditions of NPP site and prevention of anticipated operational occurrences):

- available NPP site assessment and selection;
- provision of sanitary buffer and NPP adjacent radiation-control area on which protective measures are planned;
- development of the design using conservative assumptions approach and RP developed inherent safety;
- maintenance of required quality of NPP systems (components) and provided work;
- operation of NPP according to the requirements of regulatory documents, process specifications and operating manuals;
- maintenance of safety important systems (components) operability conditions by detection of flaws, provision of preventive measures, replacement of the equipment with finished service life and organization of efficient system of work and inspection results documentation;
- selection and provision of NPP staff required skill level for the work under normal operation conditions and anticipated operational occurrences, including pre-accident and emergency conditions and also safety standards development.

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

- detection of deviations from normal operation and their elimination;
- control of operation with deviations.

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

- prevention of initiating events evolution to design-basis accidents and design basis accidents - into beyond-design-basis accidents, using safety systems;
- mitigation of accident consequences which could not be prevented by isolation of released radioactive materials.

Level 4 (Beyond design-basis accidents control):

- prevention of beyond design-basis accidents development and mitigation of their consequences;
- protection of pressurized containment from damage under beyond design-basis accidents and maintenance of its serviceability;
- recovering of NPP controlled conditions under which the chain fission reaction is stopped, nuclear fuel continuous cooling and radioactive materials isolation within specified boundaries are provided.

Level 5 Emergency schedules:

- preparation and realization if required of emergency schedules for NPP site and adjacent areas.

3.1.5. Safety objectives

Probability of core severe failure per reactor/year: less than 10^{-6}

Probability of limiting emergency release per reactor/year, less than 10^{-7}

3.1.6. Safety systems for design and beyond design-basis accidents

The strategy of design basis accidents elimination is based both on active and passive safety systems. The strategy of beyond design-basis accidents is mainly based on passive safety systems and beyond design-basis accident control systems.

The following active and passive safety systems are realized in the design:

- steam generator emergency cooldown system for protective functions of core decay heat removal and RP cooldown via the secondary circuit;
- emergency gas removal system for RP primary steam-gas mixture removal (from the reactor, PRZ and SG collectors) and primary pressure decrease alongside with PRZ PSD to mitigate the consequences of design and beyond design-basis accidents;
- emergency boron injection system proposed to provide boric acid solution injection into the pressurizer under primary-to-secondary leak conditions for primary pressure decrease and required primary coolant boric acid solution concentration under beyond design basis accident (BDBA) without scram;
- reactor primary coolant and spent fuel pool emergency and scheduled cooldown system (E&S cooldown) is proposed in particular for decay heat removal from reactor fuel to the system of reactor consumers (intermediate circuit) in all Unit design operating conditions, maintenance of reactor coolant system required inventory under "LB LOCA", primary circuit emergency make-up under "SB LOCA" (Dnom 25-80) and medium supply to sprinkler system;
- main steam lines disconnection system is proposed for SG fast and reliable isolation from leak. The system of main steam lines disconnection is proposed for operation under all accident conditions requiring SG isolation:

1. under the break of steam lines from SG to turbine stop valves in SG isolated and non-isolated parts;
2. under the break of feed lines in the section from SG to the back pressure valve;
3. under primary-to-secondary leaks;

- emergency core cooling system, the passive part is proposed for supply into the reactor of boric acid solution with concentration not below 16 g/kg at primary pressure below 5.9 MPa, the supply shall be sufficient for core heat removal before connection of low pressure part of the reactor primary coolant and spent fuel pool emergency and scheduled cooldown system under design LOCA;

- ECCS with high-pressure pumps is proposed for core heat removal under primary LOCA when compensation capacity of normal make-up is not sufficient or unavailable.
- ECCS with low-pressure pumps is proposed for;

1. core decay heat removal under primary LOCA resulting in primary pressure decrease below 2.5 MPa;
2. reactor plant decay heat removal and cooldown alongside with E&S cooldown system during Unit normal shut-down (as per scheduled or repair scheme) under the conditions of anticipated operational occurrences and design basis accidents;
3. SFP decay heat removal alongside with E&S cooldown system (including reactor core complete unloading).

- primary overpressure protection system is proposed for protection of RP equipment and pipelines from excessive primary pressure increase under anticipated operational occurrences, design basis accidents and primary pressure decrease under beyond design-basis accidents due to operation of PRZ pilot valves installed in the pipeline of steam discharge from PRZ steam space into the relief tank;
- secondary overpressure protection system is proposed for protection of RP equipment and pipelines from excessive secondary pressure increase under anticipated operational occurrences, design basis accidents and primary pressure decrease under beyond design-basis accidents due to operation of SG pilot valves installed in steam pipeline areas downstream steam generators and upstream isolation motor-operated valves considering steam dump valve to the atmosphere (BRU-A) and reactor emergency protection advanced actuation;
- passive heat removal system which main function is reactor decay heat continuous removal under beyond design-basis accident - loss of all a.c. power supply sources with primary and secondary pressurization kept.
- Besides PHRS provides required primary coolant inventory due to reactor steam condensation in SG tubing by SG secondary side heat removal to PHRS and primary condensed coolant returning to the reactor under certain scenarios of beyond design-basis accidents with primary or secondary coolant leak with simultaneous loss of all a.c. power supply sources;
- double containment for radioactive materials and ionization confinement within design boundaries;
- system of corium catching within the reactor vessel under severe beyond design-basis accidents.

3.1.7. Earthquake safety

Earthquake load parameters are assumed according to the values:

- SSE with frequency 1 per 10000 years of magnitude 7 by MSK-64 scale;
- operating-basis earthquake with frequency 1 per 100 years of magnitude 6 by MSK-64 scale;

NPP ensures safety at seismic impacts to safe shutdown earthquake (SSE) inclusively and generation of electrical and thermal energy to operating basis earthquake (OBE) level inclusively /2/ according to the requirements of Design Standards for Earthquake-Proof Nuclear Power Plants.

Safety systems keep according to the requirements of Design Standards for Earthquake-Proof Nuclear Power Plants /2/:

- functions related to NPP safety provided during and after the earthquake with intensity to SSE inclusively;
- serviceability under the earthquake of intensity up to OBE inclusively and subsequently.

3.1.8. Emergency measures

Emergency measures include elaboration of scheduled measures for personnel and population protection under NPP accidents.

The scheduled measures for personnel and population protection are developed on the basis of power unit design characteristics and parameters, criteria for making decision of the measures for population protection under NPP accident conditions considering economic, natural and other characteristics, site specific features and the degree of emergency situation actual hazard.

The Scheduled Measures for Personnel and Population Protection are the main regulatory document for protective, organizational, engineering-technical, medical-preventive and other emergency measures for NPP personnel and population protection, accident localization and elimination.

The following is provided for realization of the Scheduled Measures for Personnel and Population Protection :

- personnel and population informing;
- control means activation;
- radiation and general survey;
- radiation protection;
- medical protection;
- physical protection;
- maintenance of public order;
- evacuation provisions.

Proliferation resistance

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

Safety and security (physical protection)

5.1. Purposes of physical security system (PSS)

PSS is a part of integrated engineering and organizational measures for nuclear and radiation safety during NPP operation.

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

5.2. Tasks of physical security system

PSS realizes the following tasks:

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

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

5.3. Physical security system structure

PSS is realized on the basis of integrated automated system of physical security - including:

- integrated engineered safety features of physical security within design boundaries for buildings and constructions;
- integrated engineered safety features of physical security over NPP perimeter:
- integrated engineered safety features of local security.

Description of turbine-generator systems

Feedwater system

Feedwater system is designed for:

- water supply to steam generators to provide required steam capacity under normal operating conditions and anticipated operational occurrences;
- steam generator level maintenance;
- prevention of feedwater supply in case of steam generator overflow;
- prevention of feedwater supply into damaged steam generator in the cases of SG steamline not-isolated part or feedwater pipeline breaks and primary-to-secondary leaks;
- power unit cooldown in case of auxiliary feedwater pumps failure;
- steam generators feedwater initial supply.

The system comprises: main feedwater pumps, stand-by feedwater water pump, feedwater deaerator, valves (isolation, control), pipelines.

Feedwater supply is provided by feed water pumps from the deaerator plant.

Three feed water pumps are connected to the deaerator plant.

Each pump is designed with the recirculation line into deaerator providing pump tests and operation at low flowrates.

Temperature of the feedwater supplied into steam generators is 225°C at 100 % turbine power. The temperature of feedwater supplied into steam generators is not below 160°C under load decrease and high-pressure heater (HPH) disconnection.

Spent fuel and waste management

Complex of spent nuclear fuel (SNF) handling systems is stipulated in the design of NPP with RP V-478. The following systems are included into the complex of SNF handling:

- SNF ex-pile storage;
- fuel rod leak-check in shutdown reactor;
- spent fuel pool and reactor core barrel pipelines;
- spent fuel pool water supply for purification;
- spent fuel pool water cooling;
- spent fuel pool liner leak-check;

SNF ex-pile storage system is proposed for unloaded spent fuel cooling for SFA activity and decay heat lowering to acceptable values for transportation.

SNF ex-pile storage system functions are:

- receiving of SNF unloaded from the reactor during transportation and accidents;

- spent fuel cooling (storage) before transportation from the reactor building;
- SNF decay heat removal;
- personnel biological shielding for protection from SFP nuclear fuel.

Spent nuclear fuel storage system provides nuclear fuel storage and cooling in power unit reactor building within 10 years considering scheduled refuellings and reactor core full unloading at any moment of NPP operation.

SNF storage system is a spent fuel pool (SFP) supplied with all required equipment and systems.

fuel rod leak-check in shutdown reactor

The functions of fuel rod clad leak-check (CLC) system in shutdown reactor are:

- FA leak-check in the extension shaft of fuel handling machine (FHM) using CLC system;
- FA leak-check using defective assembly detection systems (DADS);
- leaking FA check and repair using check and repair bench (CRB).

Necessity of CLC in shutdown reactor is determined from operating reactor CLC data.

CLC is performed using the analysis of gas or water samples.

Spent fuel pool water cooling system functions are:

- SFP fuel decay heat removal under all operating conditions of power unit;
- pressure and temperatures lowering in the accident localization area (ALA) under design accident conditions.

Spent fuel pool liner leak-check system functions are:

- liner leak-check of SFP, fuel pond, reactor internals inspection wells, reactor cavity and container compartment;
- leak-check of SFP loop seals

Radioactive contamination is transferred to equipment and system structures in borated water of reactor cavities and spent fuel pool during SNF refuelling, cooling and storage procedures under NPP power unit operating conditions. Induced activity of equipment and system materials is not available because of high subcriticality during SNF storage.

The following work is provided in spent nuclear fuel handling equipment under power unit decommissioning after assigned service life.

At the preparatory stage:

- NPP unit spent nuclear fuel removal;
- coolant discharge and drainage of the reactor and spent fuel pool;
- coolant regeneration in the special water treatment facility;
- equipment decontamination as per standard procedure.

At the stage of power unit preparation for controlled preservation:

- equipment preservation in standard place.

At the stage of controlled preservation:

- equipment cooling within a long-term period.

Equipment decontamination and long-term cooling result in 3-order decrease of the activity level which is mainly determined by availability of long-living radionuclides, such as Cs^{137}.

Liquid radioactive media (LRM) as well as solid and gaseous radwastes are released during NPP operation.

Liquid radioactive media treatment. NPP design stipulates the systems for liquid radioactive media collection, regeneration and temporary storage in all operating conditions of nuclear units, exactly:

- special sewage system;
- sump water regeneration system;
- liquid radioactive media interim storage system.

The following main requirements are considered for LRM regeneration systems design:

- decrease of LRM total volume supplied for regeneration;
- decrease of LRW volume proposed for temporary storage, supplied for solidification facility and subsequently for disposal in RAW storages;
- decrease of unbalanced water volume in the area of controlled access.

NPP design stipulates main technological solutions proposed at:

- LRM separated collection and regeneration depending on chemical composition and activity;
- localization of main radionuclides in minimum volumes;
- low-waste technologies of LRM and LRW regeneration;
- low-waste decontamination technologies;
- decrease of filter regenerations number;
- enhanced requirements for supplied equipment quality and operation to decrease uncontrolled leaks;
- use of decontaminated water for primary coolant make-up and process system auxiliaries in controlled access area.

LRM regeneration system designs consider designing, operation and improvement experience of RF and foreign NPPs as well as prospective technologies.

Liquid radioactive media temporary storage system is proposed for LRM full volume temporary storage not below three months for short-lived radionuclides decay before subsequent regeneration.

Liquid radioactive media supplied into the temporary storage system are:

- concentrated salts of evaporation plants;
- waste ion-exchange resins (filtrating materials);
- sludges.

LRM regeneration technologies provide the finished product which satisfies valid regulatory document requirements for radwaste storage and disposal.

Preferably Russian enterprises shall be designers of the technologies, prospecting inclusive, and manufacturers of respective equipment.

Solid radioactive mediua treatment. Solid radwastes (SRW) are:

- overalls, footwear, individual protection means not subject to decontamination;
- metal wastes;
- construction and thermal insulation materials;
- filting components of gas purification and vent system filters;
- solidified liquid radioactive media.

Quantity of the radwaste, including solidified supplied for storage annually – not above 50 m^3/year (per 1000 MW) from one power unit. The quantity specified does not include the contaminated equipment and tools after repair and maintenance as well as the volume of containers НЗК-150-1.5П (reinforced concrete container 150-1.5P).

Treatment of all types of radwaste is provided at all NPP life stages to ensure safety during their handling and storage.

LRW collection and classification by activity level and regeneration methods is provided in radwaste locations by loading in respective containers or casks of single use. Test classification is provided in the building of waste reprocessing with storage.

Solid radwaste supplied for storage shall be subject to the following regeneration types for decrease of their volume:

- decomposition;
- combustion;
- compression.

Power unit design solutions shall provide for normal operation:

- radwaste temporary storage in the form of conditioned SRW in containers НЗК-150-1,5П (reinforced concrete container 150-1.5P). Container capacity is assumed for 5 years of NPP operation with the possibility of extension to 10 years. The containers shall be subsequently transported to regional depositories.
- storage of high-activity SRW in specially equipped volumes for high-activity wastes during NPP full service life. High-activity SRW are supplied into depositoris packed into special capsules using the facilities comprised in the set of NPP fuel handling equipment.

The systems are equipped with process and radiation monitoring facilities required for operation monitoring, evaluation of integrity and environmental release monitoring.

Gaseous radwaste treatment system. The system of gaseous radwaste treatment comprises special ventilation and gaseous radioactive blow-off purification systems.

Process blow-off purification system is proposed for decrease of radioactive noble gas releases, gaseous compounds of iod and aerosols in gaseous blow-offs from reactor building process equipment to permissible limits.

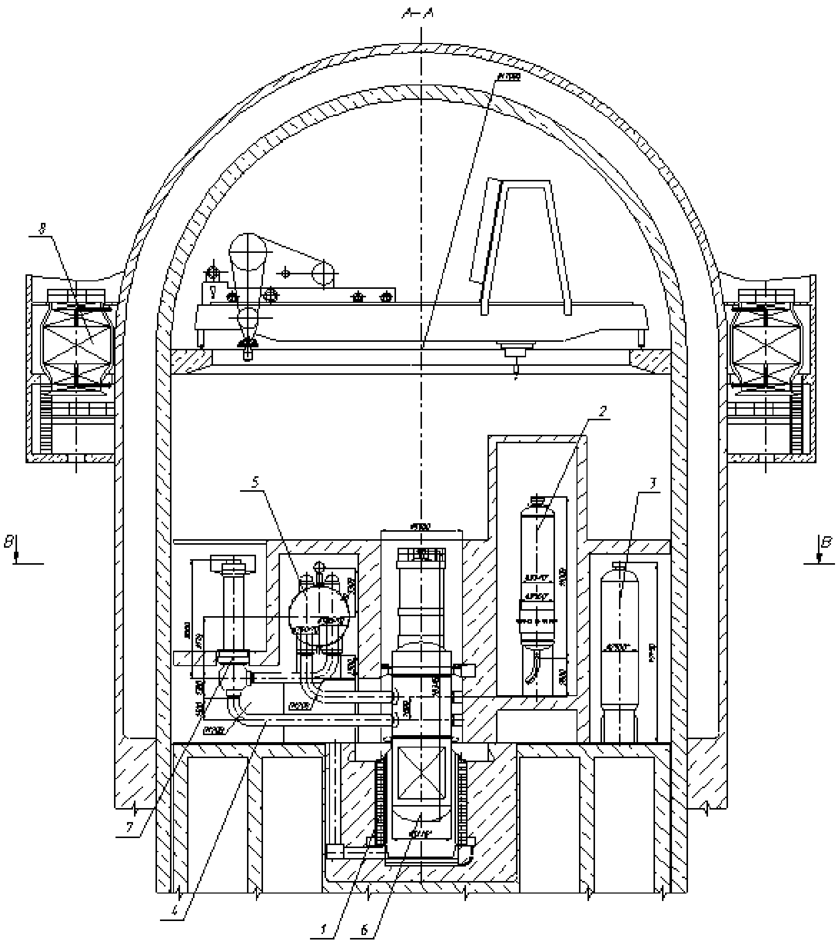
The system provides:

- continuous purification of gas blow-off from the system of hydrogen combustion;
- continuous purification of gas blow-off from the system of tank facilities and reagents.

Plant layout

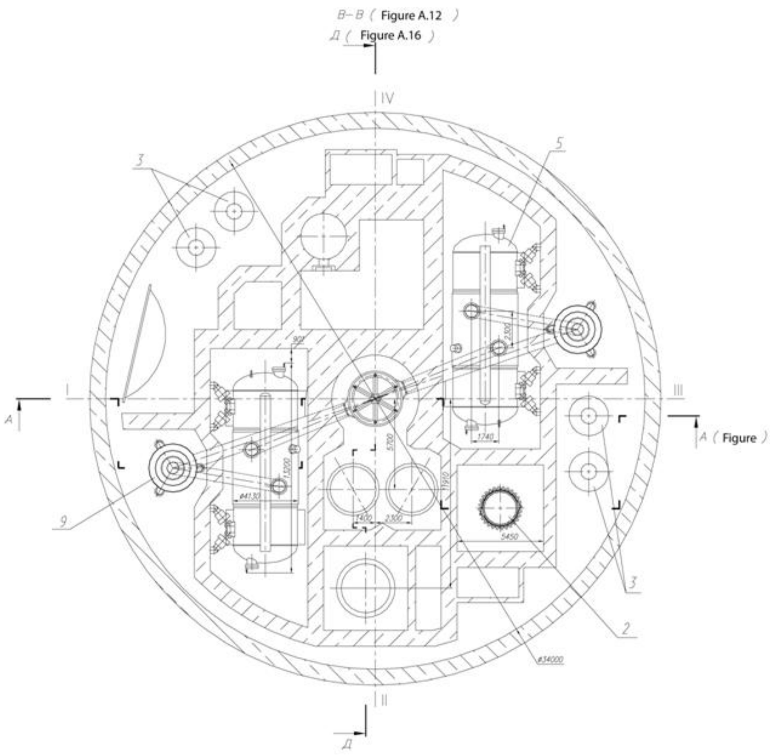
Lay-out of the commercial reactor plant V-320 is used as reference for development of main equipment lay-out in reactor plant V-478 (Fig. 4.5). The only difference is that the number of circulation loops (and, respectively, the number of SG and RCP sets) is decreased from four to two.

The reactor plant is housed within the double leak-tight reinforced-concrete containment of cylindrical shape with top restriction of hemispherical vault. The containment internal wall is also a support for a polar crane.



1 - Reactor cavity equipment, 2 - Pressurizer, 3 - emergency core cooling system (ECCS) accumulator, 4 - MCP, 5 - SG, 6 - Reactor, 7 - RCP set, 8 - passive heat removal system (PHRS) equipment.

Figure 4 - VVER-300 RP lay-out



1- RCP set, 2 - Pressurizer, 3 - ECCS accumulators, 4 - MCP,5 - SG with supports, 6 - Reactor,

Figure 5 - VVER-300 RP lay-out

Plant performance

Power unit operation is provided according to the requirements of regulatory documents, process specifications and operating manuals. RP operation is possible both in base and load-follow conditions.

The reactor plant has two circulation loops. The coolant temperature at the reactor outlet is 325°C. The design primary pressure is 17.64 MPa. Each loop has one RCP set with electric motor of required inertia parameters and one horizontal steam generator with submerged heat exchange surface.

The pressure of the generated steam at nominal load at SG steam header outlet is 7.0±0.1 MPa (design pressure is 8.1 MPa). The reactor plant total steam capacity is 900 t/h.

Development status of technologies relevant to the NPP

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

Basic technical solutions for NPP are proved by operating experience of VVER reactor plants operation which is more than 1400 reactor-years.

Application of an evolutionary approach to solution of target design tasks allows developing NPP VVER-300 design, meeting the requirements of the up-to-date Russian standards and regulations within the shortest time.

Deployment status and planned schedule

The time of NPP construction from the initial stage to commissioning for commercial operation is expected by the designer to be 4.5 years.

12.1 Research and development

The following researches and developments shall be provided for verification of design and engineering solutions:

integrated work for RP hydraulic and hydrodynamic characteristics determination:

1. VVER-300 flow part testing using aeromodels;
2. examination of hydrodynamic instability and vibration mode using reactor simulator in two-loop test bench;
3. test of RP VVER-300 operation at respective power with one RCP set (one loop);

- integrated work for verification of fuel load-follows conditions and I&C requirements;
- verification of TVS-2M application in VVER-300 RP;
- analyses of VVER-300 RP hydrogen safety for design-basis accidents;
- severe accident safety analyses.

References

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

Technical data

General plant data

Reactor thermal output	850 MWth
Power plant output, net	300 MWe
Power plant efficiency, net	35.3 %
Plant design life	60 Years
Plant availability target >	90 %
Primary coolant material	Light Water
Secondary coolant material	Light Water
Moderator material	Light water
Thermodynamic cycle	Rankine
Type of cycle	Indirect

Safety goals

Core damage frequency <	10E-6 /Reactor-Year
Large early release frequency <	10E-7 /Reactor-Year

Nuclear steam supply system

Steam flow rate at nominal conditions	472 Kg/s
Steam pressure	7.0 MPa(a)
Steam temperature	295 °C
Feedwater temperature	164 °C

Reactor coolant system

Primary coolant flow rate	6314 Kg/s
Reactor operating pressure	16.2 MPa(a)
Core coolant inlet temperature	295 °C
Core coolant outlet temperature	325 °C
Mean temperature rise across core	30 °C

Reactor core

Active core height	3550 m
Average linear heat rate	0.0903 KW/m
Fuel material	UO2
Cladding material	Alloy E-110
Outer diameter of fuel rods	9.10 mm
Number of fuel assemblies	85
Enrichment of reload fuel at equilibrium core	4.79 Weight %
Fuel cycle length	24 Months
Average discharge burnup of fuel	65 MWd/Kg
Burnable absorber (strategy/material)	Gd2O3
Control rod absorber material	Dysprosium titanate, Boron carbide
Soluble neutron absorber	H3BO3

Reactor pressure vessel

Inner diameter of cylindrical shell	3145 mm
Wall thickness of cylindrical shell	150 mm

Design pressure	17.64 MPa(a)
Design temperature	350 °C
Base material	Steel 15H2NMFA

Steam generator or Heat Exchanger

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

Reactor coolant pump (Primary circulation System)

Pump Type	GTSNA-1455
Number of pumps	2
Pump speed	1500 rpm
Flow at rated conditions	3.731 m^3/s

Pressurizer

Total volume	31 m^3
Steam volume (Working medium volume): full power	11 m^3
Steam volume (Working medium volume): Zero power	21 m^3
Heating power of heater rods	2520 kW

Primary containment

Type	Pre-stressed concrete
Overall form (spherical/cylindrical)	Cylindrical
Dimensions - diameter	34 m
Dimensions - height	52 m
Design pressure	0.5 MPa
Design temperature	150 °C
Design leakage rate	0.2 Volume % /day

Residual heat removal systems

Active/passive systems	Active and passive systems

Safety injection systems

Active/passive systems	Active and Passive

Feedwater pumps

Type	Motor Driven
Number	3