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ROSATOM's VVER reactors are among the world's most widely used reactors. VVER plants have proved their high reliability over more than 1300 reactor-years of operation. Since the commissioning of the first VVER power unit in 1960s the technology has been providing safe and affordable electricity throughout the world: from Armenian mountains to the countryside of the Czech Republic, above the Arctic Circle and at the southern tip of India.

## in the world



The VVER reactor itself was developed by ROSATOM subsidiary OKB Gidropress, while the nuclear power stations employing the VVER have been developed by the power plant design organizations within ROSATOM: Moscow Atomenergoproekt, Saint-Petersburg Atomenergoproekt (a branch of VNIPIET), and Nizhniy Novgorod Atomenergoproekt.

The VVER is a pressurized water reactor (PWR), the commonest type of nuclear reactor worldwide,

employing light water as coolant and moderator. However there are some significant differences between the VVER and other PWR types, both in terms of design and materials used. Distinguishing features of the VVER include the following:

- Use of horizontal steam generators;
- Use of hexagonal fuel assemblies;
- Avoidance of bottom penetrations in the VVER vessel:
- Use of high-capacity pressurizers.

### A brief history of the VVER

### **FIRST VVER UNITS**

A total of 67 VVER reactors have been constructed since the 1960s. The first VVER unit was commissioned in 1964, at Novovoronezh nuclear power plant, in the Voronezh region, Russia. The first unit was called the V-210, the second the V-365 (the numbers were initially corresponding to electrical output). From that time the Novovoronezh nuclear power plant has been a testing ground for new VVER units. Today, ROSATOM continues the commitment to such an approach – export only the technology which has been thoroughly tested at home.



Novovoronezh 1, the world's first VVER unit, gave the green light for further technology development

### **VVER-440**

Successful commissioning and operation of these early units provided the basis for subsequent development of more powerful reactors. Implemented first at the same site, the VVER-440 was the first of the VVERs to be

Two VVER-440 reactors in Armenia continued to operate through the 0.7g Spitak earthquake in 1988





Loviisa NPP in Finland, with two VVER-440 reactors, has one of the best lifetime performance records in the world

constructed on a serial basis. VVER-440 units have been safely operating in many European Union countries: Slovakia (Bohunice 1-4, Mohovce 1-2), Hungary (Paks 1-4), Bulgaria (Kozloduy 1-4), Czech Republic (Dukovany 1-4), and Finland (Loviisa 1-2). Design of the Finnish Loviisa was completed in 1971-72 taking into account the General Design Criteria for Nuclear Power Plants, issued by the US AEC in 1971. After that, all VVER plants were designed to meet these safety principles.

These US criteria became the standard for all "Generation II" PWRs, and thus the safety features of operating VVER-440 units and other PWR types of similar vintage are quite similar.

The reliability of the design is appreciated by VVER-440 operating countries where regulators have approved the lifetime extension of the operating plants for decades ahead. Large safety margins in the VVER-440 design also provide the basis for safe and smooth uprate of these units.



Two VVER-1000/V-320 reactors at the Temelin NPP in the Czech Republic provide 20% of the country's electricity

### **VVER-1000 - V-320**

The VVER-1000 was a milestone not only in terms of generating capacity, but also because of the many safety innovations it incorporated. The VVER-1000 is the most common VVER design worldwide, 31 units are in operation, and have amassed about 500 reactor-years of operation.

The operating VVER-1000 plants are generally categorized in three broad groups as follows:

• pilot plant, Novovoroneh 5, commissioned in 1980;

• "small series" of four plants, commissioned in 1983-86; • "standard series" of 23 units, commissioned in 1985-2011. Design of these standard series VVER-1000 plants, called the V-320, was completed in the early 1980s, and implemented at 8 sites in Russia and Ukraine, as well as in Bulgaria (Kozloduy 5-6) and Czech Republic (Temelin 1-2). The safety record of VVER-1000 plants is good and there have been no incidents with significant safety impact.



China's Tianwan 1 and 2 are of the AES-91 type. Two more units are currently under construction

### VVER-1000 - AES-91 & AES-92

Drawing on the significant body of experience gained with the well established VVER-1000/V-320, the AES-91 (or VVER-1000/V-428) was developed by Saint-Petersburg Atomenergoproekt and the AES-92 (or VVER-1000/V-412 and 466) by Moscow Atomenergoproekt. Along with upgraded technology and improved economics, these designs deployed the concept of BDBA (beyond design basis accident) management based on a balanced

combination of passive and active safety systems. Only small modifications were made in the basic power production systems – reactor, primary cooling circuit, and turbine cycle. The main changes were in the safety systems and plant lay-out. The AES-91 power plant with V-428 reactor, as built at Tianwan, was an evolution from the AES-91 design initially proposed for Finland (but not constructed), which reflected Finnish regulatory

requirements, operating experience with the Loviisa VVER-440 units and international best practice. The Tianwan AES-91 units, commissioned in 2007, were the first reactors in the world to have "core catchers" installed.

A plant of the AES-92 design, certified to conform

with the technical requirements of European operating organizations (European Utility Requirements (EUR)), was planned for the Belene site in Bulgaria, but its construction was suspended. Two AES-92 units are in start-up testing at Kudankulam, India, where unit 1 achieved first criticality in 2013.



Two units at Novovoronezh II (of the V-392M type) are among the six AES-2006 projects currently underway in Russia

### **VVER-1200 - AES-2006**

The AES-2006 design is the latest evolution in the long line of VVER plants. It meets all the international safety requirements for Gen III+ nuclear power plants. The first AES-2006 units are now under construction in Russia: two units in Sosnovyi Bor (Leningrad II), two units in Novovoronezh

(Novovoronezh II) and two units in the Kaliningrad Region (Baltic project). In addition, construction contracts have been signed and site preparation is ongoing for four units in Turkey and two units in Belarus. It is also proposed for Temelin 3-4 (Czech Republic) and Hanhikivi 1 (Finland).

### **VVER GENERATIONS**

GEN I VVER	GEN II VVER-440	GEN II/GEN III VVER-1000	GEN III+ VVER-1200
V-210	V-179	V-187	V-392M
RUSSIA:	RUSSIA:	RUSSIA: Novovoronezh 5	RUSSIA: Novovoronezh II 1-2
Novovoronezh 1	Novovoronezh 3-4		(under construction)
(decommissioned)		V-302	
	V-230	UKRAINE: South Ukraine 1	V-491
V-365	RUSSIA: Kola 1-2		RUSSIA: Baltic 1-2
RUSSIA:		V-338	(under construction)
Novovoronezh 2	Decommissioned:	UKRAINE: South Ukraine 2	Leningrad II 1-2
(decommissioned)	EAST GERMANY:	RUSSIA: Kalinin 1-2	(under construction)
	Greifswald 1-4		BELARUS: Belarus 1 (under
	BULGARIA:	V-320	construction)
	Kozloduy 1-4	RUSSIA: Balakovo 1-4, Kalinin 3-4,	
	SLOVAKIA:	Rostov 1-2, Rostov 3-4 (under construction)	
	Bohunice I 1-2	UKRAINE: Rovno 3-4, Zaporozhe 1-6,	
		Khmelnitski 1-2, South Ukraine 3	
	V-213	BULGARIA: Kozloduy 5-6	
	RUSSIA: Kola 3-4	CZECH REP.: Temelin 1-2	
	UKRAINE: Rovno 1-2		
	HUNGARY: Paks 1-4		
	CZECH REP.:	V-428	
	Dukovany 1-4	CHINA: Tianwan 1-2,	
	FINLAND: Loviisa 1-2	Tianwan 3-4 (under construction)	
	SLOVAKIA:		
	Bohunice II 1-2	V-412	
	Mochovce 1-2	INDIA: Kudankulam 1,	
	Mochovce 3-4	Kudankulam 2 (under construction)	
	(under construction)		
		V-466	
	V-270	IRAN: Bushehr 1	
	ARMENIA:		
	Armenia-1		
	(decommissioned)		
	Armenia-2		
1960	1970	1980 1990 2000	2010

### **VVER OF THE FUTURE - VVER-TOI**

The goal of the VVER-TOI (typical, optimized, with enhanced information) project is to create a standardized VVER power plant, optimised in terms of technology and economics. It is being developed

by Moscow Atomenergoproekt and is based on the AES-2006/V-392M design. It represents a further evolution of the VVER-1200 design and is designated V-510.





Design described in this booklet is St Petersburg Atomenergoproekt AES-2006 version, which is under construction at Leningrad phase II and at Kaliningrad (Baltic project), is under construction at Ostrovets in Belarus, is the only remaining contender for the Hanhikivi project in Finland, and is in the bidding process for Temelin 3 and 4 in the Czech Republic. Basic data for Moscow Atomenergoproekt AES-2006 version is provided in tables.

# The VVER-1200 (AES-2006), a Gen 3+ design



### Antecedents of the VVER-1200 (AES-2006)

The first set of Russian General Regulations on Ensuring Safety of Nuclear Power Plants was issued in 1988 and revised in 1997 (OPB 88/97). The safety requirements were based on the discussions that had been going on in the IAEA's INSAG group since 1986 and on the INSAG reports that led to development of Generation III nuclear power plants. INSAG reports issued between 1988 and 1996, with significant Russian participation, essentially represented the state-of-the art in global nuclear safety and were ahead of development of the IAEA Safety Standards and of most of the various national safety requirements. Therefore already in the 1990's the Russian General Regulations provided a sound basis for designing new Generation III plants. Design of Gen III VVER-1000 plants started around 1990 in co-operation with the Finnish utility Fortum, which wanted to build an advanced VVER-1000 plant on its Loviisa site where two smaller VVER units have been in operation since 1977. However, the Finnish Parliament halted plans for any new nuclear plant in 1993. The same plant concept was further developed for China, and the IAEA safety evaluation of this design, in 1995, was very positive. A slightly different variant, providing more passive safety features, was developed for India.

The AES-91 design with VVER-1000/V-428 reactor was accepted for construction in China in 1997. As

well as conforming to Chinese requirements (for example, with respect to earthquakes), the design also incorporated recommendations from more than twenty expert reviews conducted by the IAEA between 1995 and 2005. Relative to the standard V-320 it included improved physical separation of redundant safety systems, double containment, 4 x 100% redundancy in main safety systems, core catcher, passive hydrogen recombiners, and advanced water-cooled and water-lubricated reactor coolant pumps that remain virtually leak-tight even during loss of power.

To participate in the 1999 competitive bidding for a new nuclear plant in Finland, development of a modernised version of the AES-91 design was initiated, called AES – 91/99, with the V-466 reactor. The AES-91/99 design represented another evolutionary step for the VVER-1000, offering a reactor service life of 60 years.

In parallel with the AES-91, the AES-92 design was developed. It was the basis for the Belene plant in Bulgaria, which would have employed the V-466 reactor. But the Belene project was suspended by a Bulgarian government decision. The proposed design achieved certification by the European Utility Requirements (EUR) organization in 2007. EUR certified that the AES-92 design had successfully passed all the steps of compliance analysis with the European Utility Requirements.



Commissioning of Kudankulam unit 1, the first AES-92 plant, started in 2013



Leningrad phase II under construction

The first realization of the AES-92 design is at Kudankulam 1 and 2 in India, which is employing the V-412 version. Like the AES-91 the AES-92 makes extensive use of passive safety features: double containment shell for the reactor building; eight additional hydraulic accumulators for passive core flooding; and 12 heat exchangers for passive decay heat removal for an unlimited time without operator intervention.

The safety concept of the AES-91 and AES-92 designs is based on a preferential use of active safety systems to manage design basis accidents and the optimal combination of active and passive systems to manage more serious accidents. Particular attention was given to withstanding external hazards, using passive means for heat removal from the containment, avoiding common cause failures, and realistic assessment of the probabilities of operator errors.

Development of a yet further modernised design, the AES-2006, began in the middle 2000s. A main goal was to reduce costs without substantially changing the basic configuration of the nuclear steam supply system, while at the same time increasing safety. Thermal power was increased to 3200 MW and additional passive safety systems introduced for management of beyond design basis accidents. As well as extensive VVER-1000 operating experience, the AES-2006 design also takes

onboard lessons from Tianwan, and expert evaluations performed in relation to the Finnish bidding process of 2003.

There are two families of VVER-1200/AES-2006 plant designs. One, the V-392M version, was developed by Moscow Atomenergoproekt on the basis of the AES-92 design. It is this version that is being built at Novovoronezh phase II in Russia. It is also the design being offered for the Akkuyu project in Turkey. The other family of VVER-1200/AES-2006 designs, the V-491 version, was developed by St Petersburg Atomenergoproekt on the basis of the AES – 91 design developed for China, as successfully deployed at Tianwan. The St Petersburg Atomenergoproekt version is under construction at Leningrad phase II and at Kaliningrad (Baltic project), has been chosen for Ostrovets in Belarus (where construction is imminent), is the only remaining contender for the Hanhikivi project in Finland, and is in the bidding process for Temelin 3 and 4 in the Czech Republic.

For the latter, a consortium consisting of ROSATOM subsidiaries Atomstroyexport, Gidropress and Skoda JS is proposing a design called MIR.1200 (MIR standing for Modernized International Reactor), which is based on the AES-2006 concept. MIR.1200 has been shown to meet the Czech bid specifications, which in general are based on EUR requirements, but are more demanding in some areas.

### VVER-1200 (AES-2006) design basics (SPbAEP version)

The main principles underpinning the VVER-1200 (AES-2006) design are:

- · maximum use of proven technologies;
- minimum cost and construction times;
- balanced combination of active and passive safety systems in general to manage beyond design basis accidents:
- reduction in the influence of human factors on overall safety.



Bird's eye view of VVER-1200 (AES-2006) typical power unit layout

- 1 Reactor building
- **2** Turbine building
- 3 Vent stack
- 4 Treatment plant

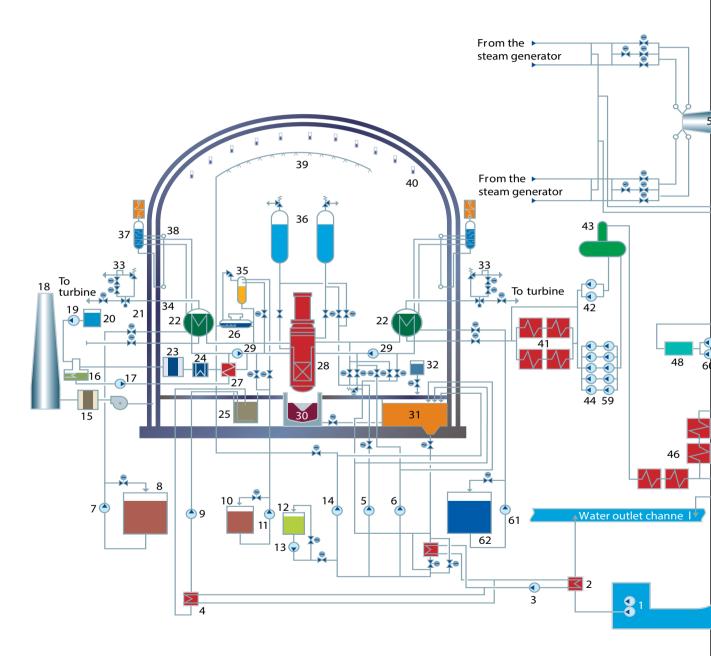
- 5 Unit diesel generator building
- 6 Nuclear service building
- 7 Auxiliary building
- 8 Emergency diesel generator station
- **9** Safety building (4 trains)
- 10 Transportation lock trestle sling
- 11 Steam cell
- 12 Water treatment building
- 13 Power supply building
- 14 Unit transformers
- 15 Control building
- 16 Fuel storage

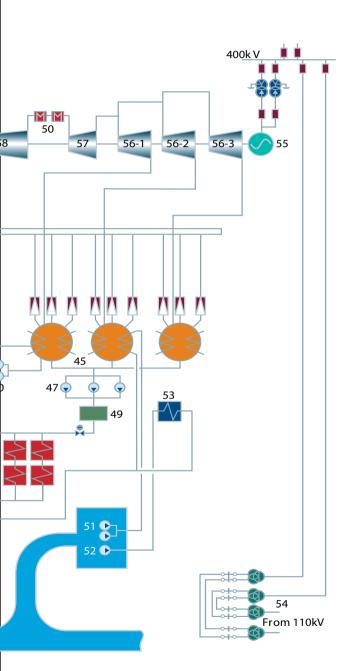
### **VVER-1200 (AES-2006) - THE BASIC DATA,** FOR THE ST PETERSBURG AEP AND MOSCOW AEP VERSIONS

	ST PETERSBURG AEP	MOSCOW AEP
Service life (years)	60	60
Unit output, electric, low-sea-temp. site (MWe gross)	1198	1198
Reactor thermal output (MWt)	3212	3212
Heat supply capacity (MWt)	300	300
Availability (%)	>90	>90
House load (including power for re-circulating cooling water supply) (%)	7	6.83
Power plant efficiency (turbine in condensing mode) (%)	37.0 gross, 34.5 net	37.0 gross, 34.5 net
Unplanned automatic scram per year	<1	< 1
Planned outage duration (annual) over seven years of operation (days, max)	4 x 16, 2 x 24, 1 x 30	4 x 16, 2 x 24, 1 x 30
Duration of outage required every eight years to include turbine disassembly (days, max)	40	40
Number of operating personnel (person/MW)	0.42	0.37
Design basis maximum fuel burn-up (average per fuel assembly) (MWd/kgU)	60	60
Fuel campaign duration (ie fuel life in the core) (years)	4	4
Refuelling frequency (months)	12(18)	12(18)
Primary coolant temp. at core inlet (°C)	298.2	298.2
Primary coolant temp. at core outlet (°C)	328.9	328.9
Primary coolant flow rate through reactor vessel (m³/hour)	86 000	86 000
Primary coolant pressure at reactor vessel outlet (MPa)	16.2	16.2
Steam pressure at the steam generator outlet (MPa)	7	7
Steam production rate per SG (t/hour)	1602	1602
Feed water temperature at SG inlet (°C)	225	225
Steam moisture content at SG outlet (%)	< 0.2	< 0.2
Total probability of core damage due to internal initiating events (per reactor year)	< 7.37 x 10 <sup>-7</sup>	< 4.16 x 10 <sup>-7</sup>
Total probability of accidental sequences involving large releases caused by containment bypass or initial lack of leak tightness	< 3.71 x 10 <sup>-9</sup>	< 1.77 x 10 <sup>-8</sup>

	ST PETERSBURG AEP	MOSCOW AEP		
DOUBLE CONTAINMENT DIMENSION				
EXTERNAL, PROTECTIVE, CONTAINMENT (REINFORCED CONCRETE):				
Internal diameter (m)	50	50.8		
Height of dome (m)	71.4	65.4		
Thickness (cylindrical section) (m)	2.2	≤1.5		
Thickness (dome part) (m)	0.8	1.5		
INTERNAL, HERMETIC, CONTAINMENT (ALSO REINFORCED CONCRETE):				
Internal diameter (m)	44	44		
Height of the dome (m)	67.1	61.7		
Thickness (cylindrical section) (m)	1.2	1.2		
Thickness (dome part) (m)	1.1	1.2		

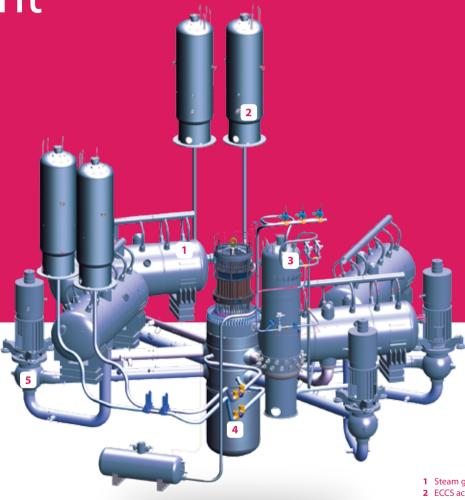
### Schematic diagram, simplified





- 1 essential cooling water (or "service water") pump
- 2 intermediate cooling circuit heat exchangers for priority consumers
- 3 intermediate circuit pump
- 4 spent fuel pool heat exchanger
- 5 emergency injection system, low pressure pump
- 6 emergency injection system, high pressure pump
- 7 emergency feed water pump
- 8 storage tanks for high concentration boric acid
- 9 spent fuel cooling pump
- 10 storage tanks for boric acid solution
- 11 emergency boration system pump
- 12 storage tank for chemical reagents
- 13 supply pump for chemical reagents
- 14 containment spray system pump
- 15 filter
- **16** volume and chemical control system deaerator
- 17 volume and chemical control system pump
- 18 ventilation stack
- 19 controlled-leak pump
- 20 controlled-leak tank
- 21 external containment
- 22 steam generator
- 23 water treatment plant
- 24 after-cooler
- 25 spent fuel pool
- 26 bubbler tank
- 27 regenerative heat exchanger for the volume and chemical control system
- 28 reactor
- 29 reactor coolant pump
- 30 core catcher
- 31 emergency core cooling system sump and refuelling water storage tank
- 32 alkali (NaOH) emergency reserve tank
- 33 MSIV, safety and relief valve assembly
- 34 containment
- **35** pressuriser
- **36** ECCS hydroaccumulators
- 37 passive heat removal system tank
- 38 condenser for the containment passive heat removal system
- 39 spray system
- 40 passive hydrogen recombiner
- 41 high-pressure heaters
- 42 electric-powered auxiliary feed water pump
- 43 deaerator
- 44 electric-powered feed water pump
- 45 condenser
- 46 low-pressure heaters
- **47** condensate pumps, first stage
- 48 unit demineralised water plant
- **49** main condensate treatment
- **50** superheater
- **51** circulation cooling water pumps
- **52** cooling water pump for turbine hall consumers
- 53 turbine hall consumers
- 54 stand-by step-down transformer
- **55** generator
- **56** turbine low-pressure cylinders
- 57 turbine intermediate-pressure cyl.
- 58 turbine high-pressure cylinder
- **59** boost pump
- 60 condensate pumps for unit demineralisation plant
- **61** emergency feed water pump
- **62** demineralised water storage tank

Main components of the VVER-1200 (AES-2006) plant



- 1 Steam generator
  - **ECCS** accumulator
- 3 Pressurizer
- 4 Reactor pressure vessel
- 5 Reactor coolant pump

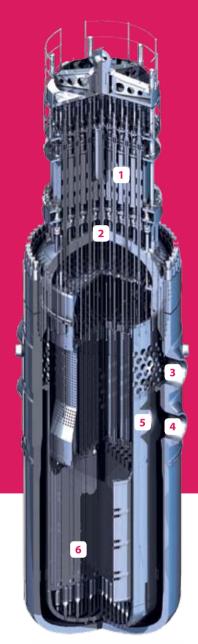
### **REACTOR PRESSURE VESSEL**

The design service life of the reactor pressure vessel is 60 years, with maximum neutron flux of 4.22 x 10<sup>19</sup> neutron/cm² (> 0.5 MeV) at the level of the surveillance specimens for the whole service life, and at the level of the top of the core of 1.28 x 10<sup>19</sup> neutrons/cm². Measures to extend vessel service life have included: limitation of nickel content in welds; limitation of impurities in base metal and welds; decrease in ductile to brittle transition temperature of the nozzle area shell material to minus 35 °C; and reduction of neutron flux at vessel walls by increasing vessel diameter.

The reactor vessel internals include core barrel, core baffle, protective tube unit and in-core instrumentation system parts in the upper plenum of the vessel, the core itself, control rods and ICID (in-core instrumentation detector) sensors. The reactor vessel head is integrated structurally with the upper assembly. The CPS (control and protection system) drive housings are installed on the reactor vessel head. The core barrel, core baffle and protective tube unit are kept from lifting under normal operating conditions by their weight plus hold-down assemblies employing elastic components made of thermo-expanded graphite. This performs better than the materials used in the V-320 reactor, with a service life of at least four years without replacement.

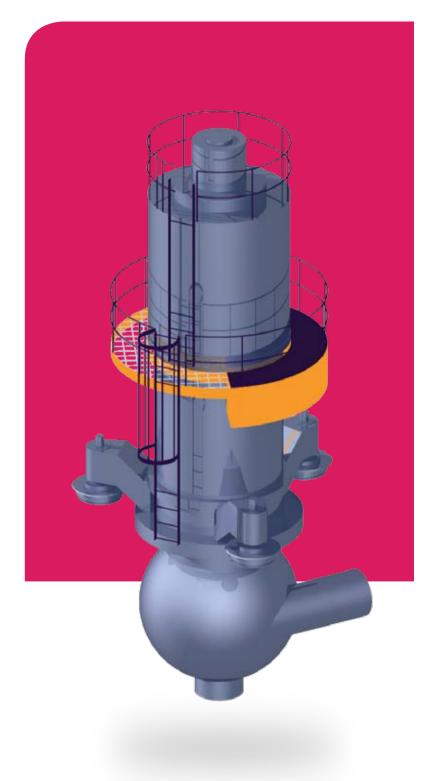
### **MAIN COOLANT PIPING**

The main coolant (primary) pipework connecting the reactor, steam generators and reactor coolant pumps has a nominal diameter, Dnom, of 850 mm, with a service life of 60 years. The nominal external diameter is 990 mm and the nominal wall thickness is 70 mm. The total length is 146 m. The main coolant piping is designed to meet all necessary conditions of the "leak-before-break" concept: material properties, stress analysis, in-service inspections and leak monitoring.



- 1 Control and protection system drives
- 2 Vessel head
- 3 Outlet nozzle
- 4 Inlet nozzle
- 4 iniet nozzie5 Core baffle
- 6 Fuel assemblies

VVER-1200 reactor vessel



GCNA-1391 type reactor coolant pump

### **REACTOR COOLANT PUMP**

The reactor (main) coolant pump is of the GCNA-1391 type. The RCP is equipped with a flywheel providing smooth main coolant circulation rundown during accident scenarios with loss of power. This permits adequate reactor cooling until the reactor is shutdown and decay heat has dropped to level where it can be safely removed by natural circulation. The design incorporates experience gained from the GCN-195M type pumps employed in the V-320 reactor and from the first GCNA-1391 pumps, which have operated reliably at Tianwan since 2007. Basic data for the GCNA-1391 pump is as follows: capacity, 22 000 m<sup>3</sup>/h; head, 0.588 MPa; nominal suction pressure, 16.02 MPa; rpm, 1000; power consumption, hot, < 5 MW; power consumption, cold, < 6.8 MW; supply current frequency, 50 Hz; weight (without motor), 75.5 t; service life, 60 years. The GCNA-1391 is a vertical pump set, consisting of a centrifugal single-stage pump with a mechanically sealed shaft and spherical weldedforged casing and asynchronous two-speed electric motor with a flywheel. The cooling of pump motor and lubrication of all bearings is provided by water. Oil-free cooling and lubrication eliminates the risk of oil fire inside the reactor containment building. The seal design ensures that rated leakage through the pump is very small when the pump is stopped and no active cooling or water injection to the seal is provided. This eliminates the pump as a possible cause of a loss of reactor coolant accident in the event of long term loss of all electrical power. When developing the pump design special attention was paid to repairability and reduced maintenance requirements. The pump design allows replacement of main components without unsealing the main joint, which considerably facilitates maintenance and repair.



### **STEAM GENERATOR**

Horizontal steam generators are traditionally used in VVER reactors. The horizontal design means the steam generators do not face such problems as primary water stress-corrosion cracking, fouling and denting. Several VVER-440 plants have had steam generators in operation for more than 35 years with no corrosion of heat exchange tubes requir-

ing tube plugging.

The steam generators employed in the VVER-1200 are of the PGV-1000MKP type. As well as being horizontal the steam generator also uses a "corridor" layout for the heat-exchange tubes in the tube bundle. Proven design with incremental improvements such as effective sludge removal from the

steam generator bottom, the use of secondary side ethanolamine water chemistry and elimination of copper-bearing components on the secondary side, enable an expected service life of 60 years to be achieved.

The steam generator produces saturated steam that flows through the holes of the perforated sheet submerged below the evaporation surface. Steam is dried in the vapour space by gravitation and flows to the perforated distribution sheet in the upper part of the steam generator and then, via ten nozzles, enters the steam header, which is located above the steam generator. Steam production rate equalisation takes place in the perforated sheets along the length of the steam generator. The steam, free of moisture, flows out of the steam header into the steam lines. The feed water flows, via Dnom 400 mm pipework, into the feedwater distribution header of the steam generator. During emergency cooldown the feed water supply is provided by the emergency feed water system. Water circulation in the steam generator secondary side is natural. The heat transfer surface of the steam generator consists of 10 978 stainless steel tubes of 16 mm diameter and 1.5 mm wall thickness. The tubes and their support structures are very robust when compared with those typically used in vertical PWR steam generators.

The heat exchange tubes are arranged in a U-shaped bundle. The "corridor" layout of the tubes has vertical spacings of 22 mm and 24 mm horizontally. The tube bundle slopes downwards (20 mm over the full length) to provide full drainage of tubes. The tubes are mounted by welding of the ends to the inside surfaces of the main coolant inlet and outlet (collectors).

The steam header, 630 mm diameter, 25 mm wall thickness, is located above the steam generator. The main steam line (630 mm diameter, wall thickness 25 mm) and PHRS (passive heat removal system) pipe (219 mm diameter, 13.5 mm wall thickness) is connected to this steam header.

### **PRESSURIZER**

In VVER reactor plants, high volume pressurizers have always been used, thus assuring a high level of reactor safety owing to the large coolant inventory in the primary circuit. In the VVER-1200 design, a modernized system for even better pressure control under transients is used. The new system has an additional line of controlled water injection to the steam space.

### REACTOR CORE AND FUEL ASSEMBLIES

The reactor core comprises 163 fuel assemblies. Reactor output is controlled using the 121 control rods of the control and protection system, by burnable neutron absorber in the fuel rods, and by change of boric acid concentration in the primary circuit water. In the VVER-1200 design hexagonal fuel assemblies are used with these characteristics:

- 13 spacer grids (including the anti-vibration one);
- located with a gap of 340 mm;
- height of fuel column 3.73 m;
- weight of fuel UO<sub>3</sub> up to 534 kg;
- enrichment up to 4.95 %;
- quickly removable fastening of the top nozzle with the use of collets enabling quick disassembly and assembly for the replacement of leaky fuel rods;
- collet fastening of fuel rods:
- anti-debris filter (option).



- 1 Top nozzle
- 2 Spacer grid
- **3** Fuel rod
- 4 Guide tube5 Bottom nozzle

VVER-1200 fuel assembly

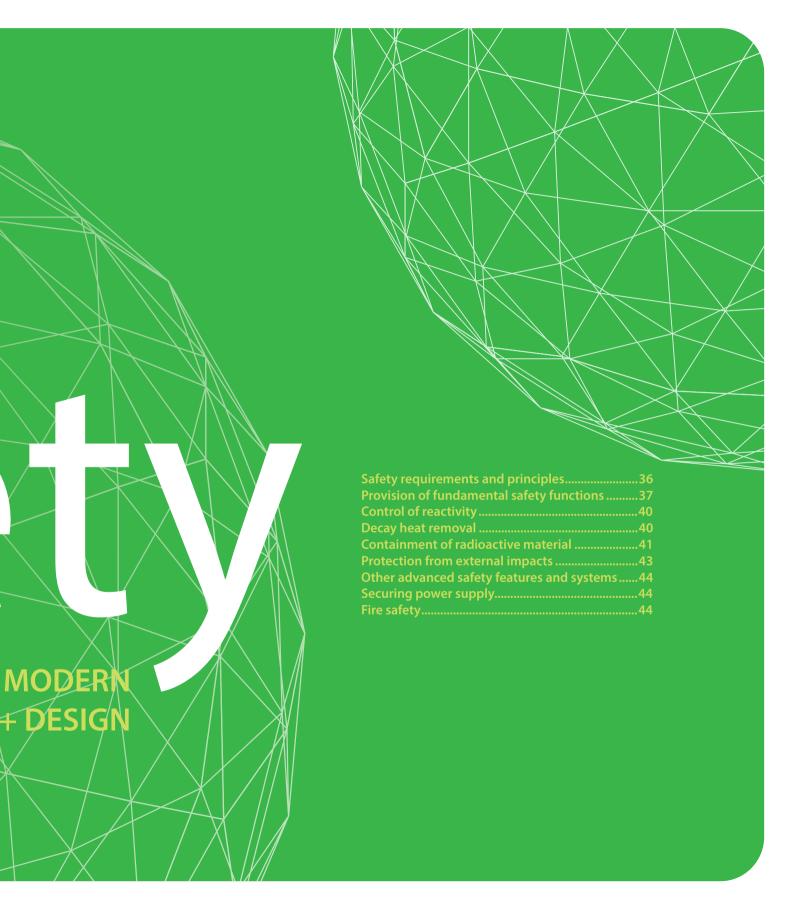


LMZ turbine at Tianwan NPP

### **TURBINE**

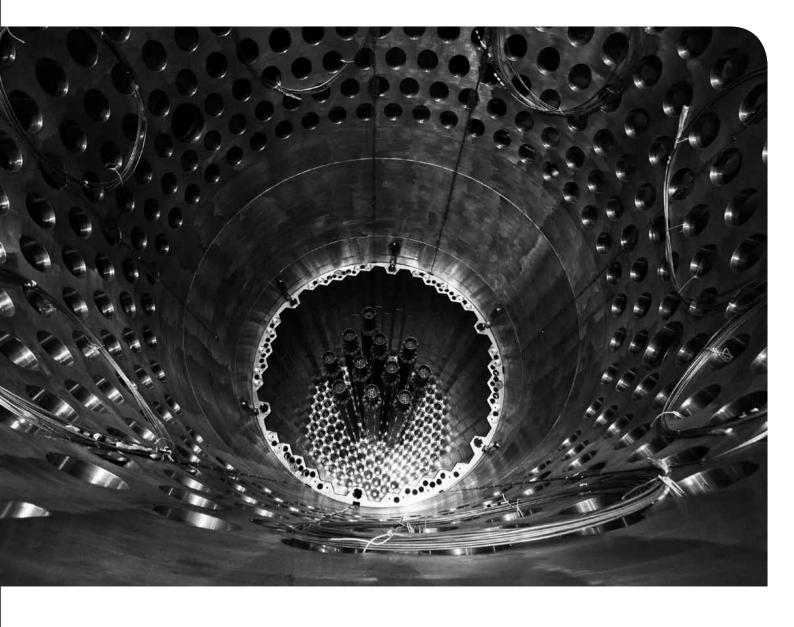
The VVER-1200 is offered with two steam turbine variants: Russian (LMZ) design full speed turbine manufactured by Power Machines; and Alstom Arabelle half speed design, which is to be used for the Baltic project, for example, and is proposed for MIR.1200 and Hanhikivi.





The safety philosophy embodied in the VVER-1200 (AES-2006) is unique among reactors on the market in deploying a full range of both active and passive systems to provide fundamental safety functions. Its safety systems can thus handle complicated situations that go beyond the traditional design basis accidents.

# Safety concept of VVER Gen 3+ design



### Safety requirements and principles

The VVER-1200 (AES-2006) plant was designed to meet the Russian general safety requirements issued in 1997, which were consistent with the IAEA's International Nuclear Safety Group (INSAG) recommendations. The INSAG group recommendations led to the development of what were called "Generation (Gen) III" nuclear power plants, and the current IAEA safety standard on nuclear power plant design safety, issued in 2012, builds on the same principles. The Russian general safety requirements are also consistent with the safety objectives specified by WENRA (Western European Nuclear Regulators Association) in 2010 for new nuclear power plants.

The VVER-1200 (AES-2006) design takes account of Design Extension Conditions (DEC), in accordance with the current IAEA safety standard. Thus all new VVER-1200 plants under construction already have design features that take fully into account the main "Fukushima lessons learned", including:

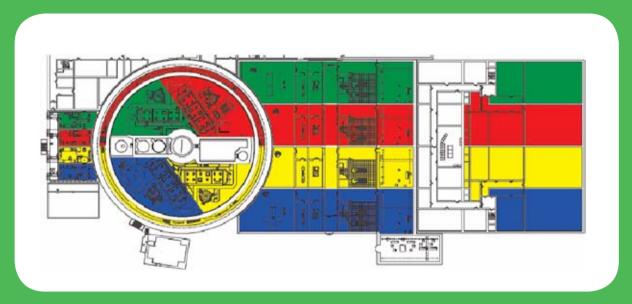
- long term cooling of reactor core without electrical power;
- long term decay heat removal that does not rely on primary ultimate heat sink (sea, river, cooling tower;
- protection of reactor containment integrity with dedicated systems after a core meltdown accident. The safety systems are designed to have the capability for stable operation under adverse conditions due to natural phenomena such as earthquakes, floods, storm winds, hurricanes, snowfalls, tornadoes, low and high extremes of temperature, as well as such man induced events as aircraft crash

(or impact from aircraft parts), air shock wave, fire, and flooding caused by water pipe breaks.
The main principles include:

- The inherent-safety principle, that is, the ability of the reactor to ensure safety based on natural feedback processes and characteristics.
- Defence in depth principle, that is, use of successive barriers preventing the release of ionising radiation and radioactive substances to the environment as well as a system of technical and organizational measures for protection of these barriers.

The main concept for providing fundamental safety functions are:

- Passivity: Passive means are used to deal with "design extension conditions" and "beyond design basis accidents" (passive SG cooling system, passive containment cooling system) and provide back up for active safety systems.
- Multiple train redundancy: The plant utilises four trains for safety systems and for their control systems.
- Diversity: The backup systems for the systems providing basic safety functions use different equipment from the backed up safety system and if possible also a different operating principle.
- Physical separation: All four trains of safety systems and their control systems are physically separated, which addresses common mode failures due to fire, aircraft accident and terrorist act. Unit control rooms (main control room and emergency control room) are also physically located in separate rooms/buildings.



Multiple train redundancy principle implemented in the VVER-1200 (AES-2006)

# Provision of fundamental safety functions

RELIABLE PROVISION OF THE THREE FUNDAMENTAL SAFETY FUNCTIONS HAS BEEN THE LEADING PRINCIPLE IN THE DESIGN OF VVER-1200 (AES-2006) PLANTS:

- 1. Control of reactivity
  - preventing uncontrolled reactor power increase
  - ensuring fast safe shutdown of the reactor when needed
- 2. Removal of decay heat to the ultimate heat sink
- cooling of the shutdown reactor
- cooling of used nuclear fuel
- 3. Containment of radioactive materials
- preventing significant radioactive releases to the environment

### **VVER-1200 (AES-2006) SAFETY SYSTEMS, FOR THE ST PETERSBURG AEP** AND MOSCOW AEP VERSIONS (NUMBER OF TRAINS AND CAPACITY)

	ST PETERSBURG AEP	MOSCOW AEP
PROTECTION, ISOLATION, SAFETY AND SAFETY CONTROL SYST	EMS	
High pressure safety injection system	4 x 100 %	2 x 100 %
Low pressure safety injection system	4 x 100 %	2 x 100 %
Emergency boration system	4 x 50 %	2 x 100 % (each train includes 2 pumps, each pump capacity is 50%)
Emergency feedwater system and heat removal via atmospheric steam dump valves	4 x 100 %	4 steam dump valves, no EFS
SG emergency cooldown and blowdown system	-	2 x 100 % (each train includes 2 pumps, each pump capacity is 100%)
Containment emergency spray systems	4 x 50 %	2 x 100 %
Residual heat removal system and reactor cooling	4 x 100 %	2 x 100 %
Intermediate cooling system (component cooling system)	4 x 100 %	2 x 100 % (each train includes 2 pumps, each pump capacity is 100%)
Essential cooling water system (service water system)	4 x 100 %	2 x 100 % (each train includes 2 pumps, each pump capacity is 100%)
HVAC system for safety systems rooms	4 x 100 %	2 x 100%
Containment isolation valve ystem	2 x 100 %	2 x 100 %
Borated water storage system	2 x 100 %	Fuel pond
Emergency gas removal system	2 x 100 %	2 x 100 %
Primary circuit overpressure protection	3 x 50 %	3 x 50 %
Secondary circuit overpressure protection	2 x 100 %	2 x 100 %
Main steam line isolation system (fast isolation valve + valve with electric actuator)	2 x 100 %	2 x 100 %
Emergency diesel generator power system	4 x 100 %	2 x 100%
Safety system activation	4 sensors/ parameter, 4 logic trains, each with 2/4 logic	2 x 100% 3 sensors in train, each with 2/3 logic

	ST PETERSBURG AEP	MOSCOW AEP
Emergency reactor shut down system	4 sensors/ parameter, 4 logic sets 2/4 for 1st level selection and 2 logic sets 2/4 for 2nd level selection	2 x 100% 3 sensors in train, each with 2/3 logic
HVAC system to maintain under-pressure in the containment annular space	2 x 100 % (active)	2 x 100 % (passive)
PASSIVE SAFETY SYSTEMS FOR DESIGN BASIS ACCIDENTS		
Emergency reactor core cooling hydroaccumulator system, 1st stage	4 x 33 %	4 x 33 %
Emergency reactor core cooling hydroaccumulator system, 2nd stage	-	4 x 33 %
Containment hydrogen removal system (1st subsystem)	1 x 100 %	1 x 100 %
Double containment	✓	✓
AUXILIARY MEANS FOR BEYOND DESIGN BASIS ACCIDENT MA	NAGEMENT	
Passive heat removal system through steam generators (SG PHRS)	4 x 33 % (water-cooled)	4 x 33 % (air-cooled)
Passive heat removal system from containment (containment PHRS)	4 x 33 %	-
Core catcher	✓	✓
Containment hydrogen removal system (2nd subsystem)	✓	-
Volatile iodine chemical retention system	✓	-
Reactor core inspection shaft water emergency use system	✓	✓

System for primary circuit overpressure protection and emergency gas

System for the supply of water to provide a long term external cooling source for the core catcher

System for the supply of water to emergency heat removal tanks located

Emergency electrical supply system using a mobile diesel generator and

Emergency I&C equipment and control panel in the main control room

System for the supply of water to the spent fuel cooling pool

on the external wall of the containment

removal

batteries

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### **CONTROL OF REACTIVITY**

All VVER-1200 reactors have a unique safety feature, when compared with other PWR types or older VVERs: if the control rods are inserted into the core the reactor will stay in shutdown state even at low temperature over the long term. This means that reactor can be cooled down to safe shutdown condition without any concern that it might become critical again and without a mandatory requirement to inject boron into the reactor coolant system.

Nevertheless, the VVER-1200 also has reliable

boron injection systems. These can add liquid with high boron concentration to the reactor coolant in the event that the control rods do not drop into the core for any reason.

The boron injection system has four identical parallel pumps. Operation of two pumps is enough for fast shutdown of the reactor so that the fuel is not damaged in any anticipated scenario where fast control rod insertion fails. If there is no urgent reason requiring fast shutdown of the reactor, operation of one pump is enough.

### **DECAY HEAT REMOVAL**

In the VVER-1200, decay heat can be removed in three different ways:

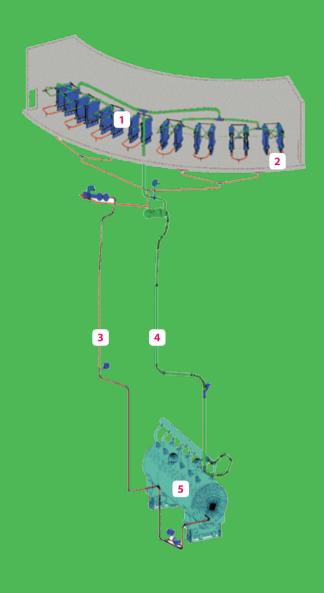
- 1) by active systems to the main ultimate heat sink or to a separate dedicated "spray pond",
- 2) by active systems to atmosphere (feed and bleed from steam generators), or
- 3) by passive systems to atmosphere. Passive decay heat removal is an important advanced feature for ensuring safety of the VVER-1200.

The steam generator passive heat removal system (SG PHRS) is intended for implementation of the following functions:

- residual heat removal and reactor shutdown cooling when there is complete loss of power;
- residual heat removal and reactor shutdown cooling when there is complete loss of feedwater supply;
- prevention of release of radioactive coolant to atmosphere via steam bypass to atmosphere or SG safety valves during an accident involving a coolant leak from primary to secondary circuit;

• minimisation of radioactive coolant release during an accident with simultaneous coolant leak from primary to secondary circuit and steam pipeline break between containment wall and pipeline isolation valve.

Validation of the performance and efficacy of the SG PHRS required a good deal of analysis, calculation, estimation and experiment. Analytical simulation of beyond design basis accident modes was performed using the Russian certified thermo-hydraulic best estimate code KORSAR. Experimental validation of the performance of the SG PHRS has been done using a large-scale test rig at NPO CKTI JSC (St Petersburg), which includes a large-scale model of the SG PHRS actually employed in the VVER-1200 (AES-2006) plant. The rig allows experiments to be carried out using full scale secondary circuit coolant conditions, with the steam condensate circuit having an overall height of 23.6 m. The geometry and configuration of the emergency cooldown heat exchangers (ECHE) are also close to full-scale.



- 1 ECHE (emergency cooldown heat exchangers)
- 2 PHRS tank
- **3** Downcomer
- 4 Riser
- **5** Steam generator

SG PHRS of VVER-1200 (AES-2006) - SPb AEP version

### CONTAINMENT OF RADIOACTIVE MATERIAL

A target already established after the Chernobyl accident was that dedicated systems have to be developed for the new generation of VVER plants for protecting the reactor containment after possible core meltdown accidents. Following the Fukushima Daiichi accident, this target was widely judged to be well founded and the requirements are now included in the IAEA Safety Standard SSR2.1, "Safety of Nuclear Power Plants: Design," issued in 2012. Protection of the reactor containment even in the event of a core meltdown accident has been one of the original design principles used for AES-2006 plants, and supporting experimental research in this area has been done for more than 20 years. The strategy for protection of the VVER-1200 containment after a postulated reactor core meltdown is that all phenomena that could occur in connection with a core meltdown that might threaten containment integrity are taken into account and dedicated means are provided to ensure containment integrity. These phenomena are as follows:

- reactor core meltdown with high primary circuit presssure;
- containment overpressure due to steam generated inside the containment;
- accumulation of hydrogen inside the containment and consequent hydrogen explosion;
- steam explosion;
- penetration of the molten reactor core through the containment bottom;
- re-criticality of the molten core.

Ensuring VVER-1200 containment integrity in the event of these circumstances is based on systems that are completely independent and separated from the systems that are intended to prevent severe reactor core damage. Containment overpressure is prevented by the containment passive heat removal system (CPHRS). Accumulation of hydrogen is prevented by passive hydrogen recombiners, with some contribution from the core catcher, which is also the main system for elimination of steam explosion, containment bottom penetration and recriticality of the molten core.

The containment PHRS is intended to

- reduce and maintain pressure within design limits inside the containment during beyond design basis accidents, including those accompanied by severe core damage;
- transfer to the ultimate heat sink the heat released into the containment during beyond design basis accidents, including those accompanied by severe core damage;
- serve as a redundant system for the containment spray in order to enhance safety.

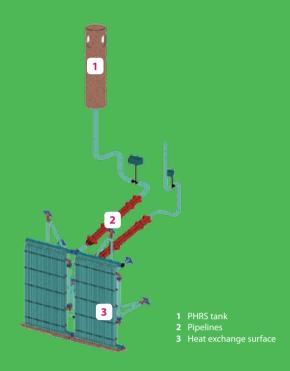
To confirm the performance of the containment PHRS and validate its design features a considerable body of testing and analytical work has been amassed, including use of large-scale test rigs modified or built specially for the purpose.

The functions of the core catcher:

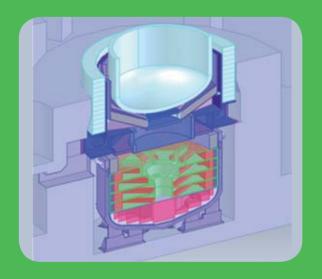
- retaining the reactor vessel bottom in case of detachment or deformation;
- protection of reactor shaft structure elements from corium;
- retention of liquid and solid corium components, core fragments and structural materials;
- heat transfer to cooling water;
- maintenance of corium in sub-critical state;
- minimisation of releases of radioactive substances/ hydrogen to the containment.

The core catcher is a cone shaped metal structure weighing about 800 t. It is double walled, with the gap between the two walls filled with FAOG (ferric and aluminium oxide granules). The core catcher is filled with sacrificial material, which is a ceramic mixture also including ferric oxide and aluminium oxide. Tianwan 1 and 2 were the first nuclear plants in the world to be equipped with core catchers.

The development, design and implementation of the core catcher concept has been underpinned by a significant programme of testing and analysis, with participation by the following organisations: NITI (Sosnovy Bor), Kurchatov Institute (Moscow), Obninsk Institute for Physics and Power Engineering, Gidropress, St Petersburg State Institute of Technology, and Institute for Silicate Chemistry of the Russian Academy of Sciences (St Petersburg).



Containment PHRS of VVER-1200 (AES-2006) – SPb AEP version



Core Catcher

## Protection from external impacts



**SEISMIC LOADS.** Protection against seismic hazards is provided taking into account the site specific seismic conditions. For instance, in the Temelin 3-4 VVER-1200 offer, the buildings are designed to stay intact if the horizontal maximum ground acceleration caused by an earthquake does not exceed 0.25g. (Note: The Temelin site is in a non-seismic area and Temelin 1-2 were designed for 0.1g horizontal maximum ground acceleration.)



**AIRCRAFT CRASH.** The VVER-1200 (AES-2006) plants currently under construction are designed to withstand a small airplane crash, with a design basis aircraft weight of 5.7 t. Protection against the impact of larger aircraft, depending on customer requirements, can be achieved by increasing the thickness of the outer containment wall and of some other buildings. A large (400 t) passenger plane crash was included in the design basis for the Temelin 3-4 offer, and the ability to withstand such a crash demonstrated by detailed analysis supported by model testing.



**HURRICANES AND TORNADOS.** Safety related components are designed to cope with a wind load corresponding to a wind velocity of 30 m/s at a height of 10 m. Design loads are those corresponding to a whirlwind of class 3.60 according to the Fujita scale.



**SNOW AND ICE LOADS.** Design peak snow load is 4.1 kPa.



**EXTERNAL EXPLOSIONS.** VVER-1200 safety-related components are designed taking into account a shock wave arising from an external explosion. The pressure at the shock wave front is taken to be 30 kPa, with a compression stage time of 1 second.

## Other advanced safety features and systems

### SECURING POWER SUPPLY

In spite of the extensive use of passive safety systems, the VVER-1200 also has very robust electrical power supply systems, both from the external grid and from on-site independent power sources. Thus the VVER-1200's active safety systems are as reliable as most "conventional" PWRs in operation today, which do not have the added benefit of its passive systems.

Offsite power connections are site specific but there will always be at least two separate high voltage transmission lines from different directions, bringing power to the plant via three separate transformers.

Onsite power sources comprise four identical diesel generators, plus one or two additional units of different design.

### **FIRE SAFETY**

For suppressing oil and electrical fires the VVER-1200 (AES-2006) is equipped with passive fire protection system (PFPS). PFPS of fire zones is used for the following:

- •To exclude the simultaneous effect of fire on the equipment and the elements of the main and backup options for safe emergency shutdown and cooldown of the reactor, and thereby ensure the implementation of project functions by these systems during and after the fire;
- If necessary, to ensure localization and control of radioactive releases into the environment in case of fire;
- •To protect personnel/population from excess of the established radiation doses.



### **KEYS**

- 1 Reactor building
- 2 Annulus
- 3 Reactor vessel
- 4 Steam generator
- 5 Reactor (main) coolant pump
- 6 Main coolant piping
- 7 Bubbler tank
- 8 Pressuriser
- 9 Passive heat removal system tanks
- **10** Core catcher
- 11 Emergency core cooling system accumulators
- 12 Passive heat removal system heat-exchangers
- 13 Sprinkler system
- **14** Polar crane
- 15 Turbine building
- **16** Deaerator
- 17 Moisture separator reheater
- 18 HP cylinder
- 19 LP cylinder
- **20** Generator
- 21 Gantry crane

### **MAIN DATA:**

Rated thermal power of the reactor (MWt)	3200
Installed capacity per unit (MWe, gross)	1190-1270
Thermal efficiency (%)	~37
Primary circuit loops, with one	
steam generator per loop	4
Coolant flow rate through the reactor (m <sup>3</sup> /h)	86 000
Coolant temperature, reactor inlet (° C)	298.2
Coolant temperature, reactor outlet (° C)	328.9
Reactor coolant pressure (MPa)	16.2
Steam pressure, steam generator outlet (MPa)	7.0
Number of fuel assemblies	163
Number of control rods	121

### **SAFETY SYSTEMS**

Internal containment	Prestressed concrete
Number of safety trains	4
Reactor trip	Passive
Emergency injection	Active
Emergency core cooling	Active
Emergency containment cooling	Active

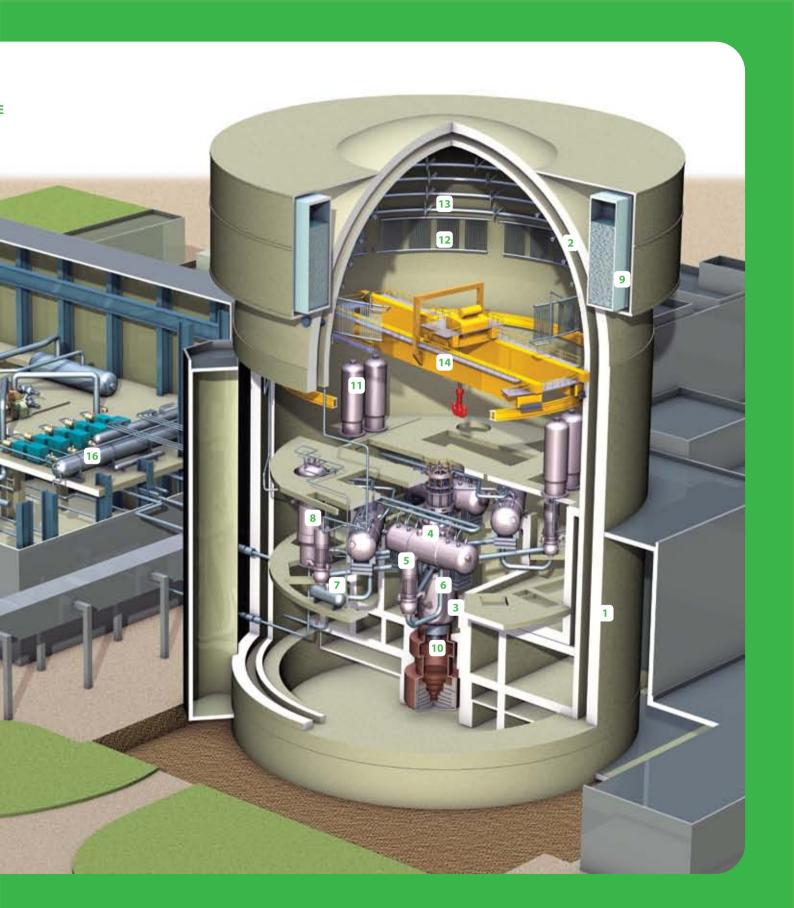
### BEYOND DESIGN BASIS ACCIDENT MANAGEMENT

Core catcher	Yes
Passive heat removal from steam generator	Yes
Passive heat removal from the containment	Yes
Hydrogen removal system	Yes

### **AES-2006 (VVER-1200)**

### ST PETERSBURG AEP DESIGN, WITH ARABELLE™ STEAM TURBINI





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