DESIGN FEATURES OF BREST REACTORS. EXPERIMENTAL WORK TO ADVANCE THE CONCEPT OF BREST REACTORS. RESULTS AND PLANS.

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SUMMARY

Principle designs of 300 MW(th) and 1200 MW(th) lead-cooled fast reactors are presented. Reactors of various output are shown to be built using the same principles.

In conjunction with increased output and to implement inherent safety concept in BREST-1200 reactor design a number of new solutions, which may be used in BREST-300 concept too, has been taken including:

- pool-type reactor design not requiring metal vessel, hence, not limiting reactor power;
- new handling system allowing to reduce central hall and building dimensions as a whole;
- emergency cooling system using Field pipes, immersed directly in lead, which may be used to cool down reactor under normal conditions;
- by-pass line incorporated in coolant loop allowing to refuse the actively actuating valve initiated in pumps shut down.

I. DESIGN FEATURES OF BREST REACTORS.

We guess, large-scale power will demand reactors of various sizes, but centralized power generation using large-scale nuclear power plants, will be the thoroughfare of the development. This has resulted in necessity to consider large-scale reactor concepts satisfying the criteria of new technology, the BREST-1200 reactor is the example of which. The same principles as ones used in BREST-300 reactor was applied in developing BREST-1200 reactor, while BREST-300 may be considered both as mid-size series power reactor, and test and experimental reactor, designated to gain operating experience, finally develop and check new engineering solutions, specifying safety and economy of lead-cooled fast reactors. The reactor power (1200 MW(e)) was chosen due to the fact that there are turbines of supercritical parameters, such as K-1200-240LMZ, being manufactured in Russia.

BREST-1200 reactor facility (as well as BREST-300) represents dual-cycle cooling, steam generating power unit, comprising reactor itself with steam generators, pumps, refuelling equipment, control and protection system, concrete well with thermal insulation, steam turbine plant, heat removal system being used in reactor cooldown, reactor heatup system, reactor overpressure protection system, primary coolant cleanup system, gas cleanup system and other subsidiary systems.

High-density (14.3 g/cm^3) and highly conductive (20 W/m·K) mononitride mixed fuel (UN-PuN) is discussed as the fuel of high compatibility with the lead and fuel cladding material, chromium ferrite-martensite steel will be used as fuel cladding.

Lead, ensuring good thermal fuel-coolant interaction, is filled between the fuel and cladding in a gap provided by fuel element design to reduce fuel temperature resulting in relatively low fission product release from the fuel inwardly the cladding.

For the purpose of providing large cross section for coolant flow passage, increasing the power removed by lead natural convection, reducing coolant heatup value, and mainly avoiding loss of cooling the affected fuel assemblies caused by local flow blockage of the fuel assemblies, all the in-core fuel assemblies are can-free. Such fuel assembly design allows coolant to flow across the core preventing affected fuel assembly from burnout. Fuel assemblies installed in reactor reflector have leak-tight cans. The first line of these fuel assemblies is used as control member channels, and fuel assemblies in the lines from 2 to 4 may content I and Tc for transmutation, as well as Sr and Cs as stable heat generator.

Three-zone flattening of lead incremental heating and fuel cladding temperatures by shaping fuel assembly power density and lead flow rate by the use of fuel elements with different diameters but similar plutonium content in the fuel to be loaded is applied in the core instead of usual flattening of radial power density distribution by fuel enrichment.

Uranium screens, used conventionally in fast reactors, have been replaced for efficient lead reflector, albedo characteristics of which is better than that of uranium dioxide, to reduce neutron leakage, better neutron flux flattening, provide operating conditions without reactivity change during core cycle, and achieve complete in-core fuel breeding.

The use of chemically inert, high-boiling lead as the primary coolant, enabled to refuse three-circuit heat removal system, and adopt simpler dual-cycle cooling system with steam superheating by steam and feed water preheating up to 340 °C by live steam at supercritical parameters.

Heat removal from reactor core is carried out by lead coolant forced convection using pumps. Lead coolant is pumped to 2 m height about suction chamber lead level and fed to free level of the ring discharge chamber. Then the lead comes down to core support grid, through fuel assemblies from the bottom upwards, to be heated up to 540 °C temperature, and is fed to common dump chamber of «hot» coolant, further the coolant comes up and through distributing header pipes flows to steam generator inlet cavities and shell space. While coming down the shell space, the lead coolant gives its heat to secondary coolant, which flows through steam generator tubes. Cooled to ~420 °C lead coolant flows up through the ring gap and pours out to pump suction chamber, from which it again is pumped to the discharge chamber (Figs. 1, 2).

Lead circulation through reactor core and steam generator is carried out by the difference between the levels of «cold» and «hot» coolant developed by the pumps, rather than the head developed by the pumps. Nonuniformity of lead flow through the steam generators with one or few pumps shut down is excluded, in so doing flow inertia in fast pump shutdown is provided by equalizing coolant levels in discharge and suction chambers (for about 20 sec).

As for proposed primary coolant flow system, while flowing the coolant reaches free level twice, that results in surfacing and escaping the majority of lead vapour bubbles, being generated in accidents caused by steam generator tube depressurization.

Integral-loop primary circuit layout is applied to reduce consequences of potential accident caused by steam generator tube ruptures, in which the steam generators and main circulating pumps are placed outside reactor main vessel. Such layout together with the chosen lead flow circulation system and steam relief units from the reactor vessel to the pressure-suppression pools will prevent critical quantity of steam from intake by reactor core and reactor vessel from overpressure. In comparison with integral layout conventional for fast reactors, BREST reactor design enables to reduce reactor dimensions and lead loop volume.

In-vessel spent fuel storage, spaced from the core and protected from being irradiated, enables to accelerate and simplify spent fuel unloading from the reactor by precooling the spent fuel to the level of decay heat release, allowing its handling without being cooled by force.

The lack of high pressure in the lead circuit and relatively high lead freezing point contribute to crack self-healing, that prevents loss of coolant, fuel element melting, radioactive lead melt blowdown to reactor room.

In conjunction with increased power and to implement completely the concept of inherent safety in BREST-1200 reactor design, a number of new solutions has been taken as compared to BREST-300 reactor.

Increased reactor dimensions and weight is a problem for vessel manufacturing, shipment and installation, seismic stability. Pool-type layout of the reactor and steam generators is adopted in BREST-1200 design, where those are installed inwardly concrete well with thermal insulation without metal vessel. Natural convection of air, circulating through pipes (120 mm pipe size), the downcomer and upcomer legs of which are placed inwardly the load bearing concrete, is used to maintain the concrete temperature within the admissible limits. Reinforced concrete body is lined inwardly with 8-10 mm thick anchored steel. Air pipes are secured to the lining from outside, and thermal insulation is anchored from inside, being made in the form of stainless steel clad thermal insulating units of 3×200=600 mm total thickness. «Keramvol» plate is chosen as the thermal insulation. The reduced design conductivity of the thermal insulating units was taken 0.2 W/m·degree.

Regarding BREST-300 reactor, the lack of screens and relatively small size of reactor core allowed one to place control rods outside the core, around its side surface, implement the potential to control the reactor by influence on neutron leakage through varying lead column level in the reflector, consider the potential to refuse control and protection system having conventional mechanical drives.

Large dimensions of BREST-1200 core in radial direction has reduced the importance of side neutron leakage significantly. In comparison with BREST-300 reactor the efficiency of lead columns in the lateral reflector has reduced by a factor of three, that is why making BREST-1200 reactor subcritical in outages ($\Delta K = 2$ %) is carried out by absorber rods, placed inwardly the tubes of the fuel assembly structures and held above the core in upper position by lead flow. As forced convection ceases, heavy rods come down the core. As circulation starts, rod groups (each having its own lead inlet) subsequently reset to upper position. The main functions in reactivity control (ΔK~0.3 %) are carried out by the members placed in the lateral blanket, mainly by lead columns, the level of which in the channels is regulated by gas pressure.

To simplify installation and dismantling of the gas driven control members due to increased number of gas driven regulators in BREST-1200 reactor (from 28 in BREST-300 up to 64 in BREST-1200) it has resulted in necessity to route the feeding mains from the top, that is a feature of BREST-300, supply the drive gas from core bottom. A flexible pipe was used to supply the gas, with the pipe inserted through the guide channel on the side of the coolant downcomer path in the guide pipe of the control member, installed in the center of reflector unit.

As for BREST-1200 reactor, heat is transferred to the emergency cooling system by air circulating due to the natural draft in Field pipes, immersed directly in liquid lead in steam generator wells. Atmospheric air enters internal Field pipe being a downcomer portion, and comes upward through the gap between the internal and external pipes, which is the upcomer portion. Heated air enters exhaust stack through the collecting headers and is released into the atmosphere. Emergency cooling system incorporates 264 Field pipes with internal and external pipe diameter 140 and 210 mm respectively. Interior side of the external pipe has 16 fins of 200 mm width each extending along the heated portion. With reactor operated at normal conditions the emergency cooling system is in hot standby and the system capacity in this mode of operation is as low as possible from viewpoint of maintaining the temperatures of outlet circulation circuit at the level allowing one to start the system at full capacity immediately. The emergency cooling system may initiated by either opening a shut valve by active or passive signal, or passive unit actuation by air temperature growth at Field pipe outlet. At 420 °C temperature the capacity of emergency cooling and reactor well cooling system (this system may be used as a train of emergency cooling) is ~11 MW with 70 m exhaust stack height. Field pipes of the emergency cooling system may be also used in standard reactor cooling with forced air convection. In this case fans are used instead of blowers. Power output taken by this cooling system is ~1 % full power.

As radial size of reactor core increases, negative component of positive reactivity effect at end reflectors decreases. Neutron absorbers are proposed to be placed in fuel assemblies at 200 ÷ 300 mm distance from the ends of fuel column to increase this component, that enables to reduce core sensitivity to lead density variation in essence to the value characteristic for BREST-300 reactor core.

New fuel assemblies and core components handling pattern is proposed for BREST-1200 (Fig. 3) as compared with BREST-300 reactor. The handling operations at these reactors are carried out in two steps: moving components in/out of reactor, in-reactor handling. A conveyor, having two ramp conveying pipes and movable tungsten-loaded container located on the side of the reactor to install and fix from surfacing core components (spent fuel assemblies, fresh fuel assemblies, lead reflector unit, control member) in it, is used in BREST-1200 to move in and out of reactor the spent fuel assemblies, fresh fuel assemblies and other core components. Floor-mounted handling machine is used in BREST-300 reactor to carry out this operations. By moving the bridge and trolley the handling machine installs fuel assembly enclosing booth coaxial with discharging channel-lock and joins the booth with lock chamber by lowering the booth, the chamber being installed on the rotable plug. In-reactor handling includes operations to load fresh fuel assemblies, reflector units and control members from the in-reactor storage to the core, as well as unloading the spent fuel assemblies and other components from the core to the in-reactor storage; regarding the BREST-1200 reactor this operations cover also insertion of the spent fuel assemblies into the conveyor container and withdrawal of the fresh fuel assemblies out of the conveyor container. In-reactor handling is carried out using rotable plugs and in-reactor handling machine, installed onto the small rotable plug. The rotable plugs are designed for guiding the in-reactor handling machine to the co-ordinate of the specific cell of the core and in-reactor storage; regarding BREST-1200 reactor the plugs are used also for guiding the in-reactor handling machine to conveyor container co-ordinate. Handling pattern used in BREST-1200 reactor allows one to reduce significantly the central hall dimensions in comparison with those of BREST-300 reactor, where upper handling pattern with handling through ratable plug is used.

Check cargo valves interconnecting pump suction chamber with reactor discharge chamber are used in BREST-300 reactor to provide coolant natural convection. All check valves open in pump operation at rated capacity and the lead is pumped to the reactor discharge chamber through these valves and through upper overflow pump pipes too. The check valves become closed by hydrodynamic forces arising in moving the coolant from the reactor discharge chamber to the discharge chambers of shutdown pumps with one or two pumps shut down. Thus shutdown of one or two pumps will not result in lead coolant back flow through the pumps.

As clear from BREST-type reactor incident design study including severe accidents, the reactors is stable to the incidents and large-scale fission product releases are impossible.

Comparative economic estimates allows one to hope to the expenses not higher than those for light water reactors.

Conceptual design development has confirmed the potential to build BREST-type reactors of various power with inherent safety for large-scale nuclear power in future.

Table 1 outlines the specifications of 300-1200 MW lead-cooled fast reactors within the framework of conceptual design.

Tabl. 1

Technical Characteristics Reactors BREST-300 and BREST-1200

Characteristic	BREST−300	BREST−1200
Thermal power, MW	700	2800
Net electric power, MW	300	1200
Number of FA in the core	185	332
Core diameter, mm	2300	4755
Core height, mm	1100	1100
Fuel element spacing, mm	13.6	13.6
Fuel element diameter, mm	9.1;9.6; 10.4	9.1; 9.6; 10.4
Core fuel	UN+PuN	UN+PuN
Core fuel load, t	16	63.9
Load of Pu/(^{239}Pu+^{241}Pu), t	2.1/1.5	8.56/6.06

Characteristic	BREST–300	BREST–1200
Fuel lifetime, yrs	5	5÷6
Refueling interval, yrs	1	1
Inlet/Outlet lead temp., °C	420/540	420/540
In.(water)/Out.(steam) temp., °C	340/520	340/520
Maximum cladding temp., °C	650	650
Maximum lead velocity, m/s	1.8	1.7
Lead flow rate, t/s	40	158.4
Efficiency, %	43	43
Core breeding ratio (CBR)	~1	~1
Effects of reactivity, % $\Delta K/K$:		
power	0.16	0.15
total (max) reactivity margin	0.35	0.31
b_{ef}	0.36	0.34
Lifetime, yr	60	60

II. Experimental work to advance the concept of BREST reactors

The experimental works conducted during 1990-1994 at Institute of Physics and Power Engineering (Obninsk, Russia), Central Research Institute of Structural Materials (St.-Petersburg, Russia), All-Russian Research Institute for Technical Physics (Russia), Institute of Steel and Alloys (Moscow, Russia), Polytechnical Institute (N. Novgorod, Russia), IGR reactor (Semipalatinsk, Russia), are discussed.

1. NEUTRONIC EXPERIMENTS TO JUSTIFY NEUTRONIC ANALYSIS

Different BREST-300 reactor core characteristics have been studied in experiments at a series of three critical assemblies (BFS-61, BFS-61-1, BFS-61-2) differed from each other with side reflector configuration, Institute of Physics and Power Engineering (Obninsk, Russia). The critical assembly core height was 86.7 cm, and the core radius changed with the variation of the side reflector configuration and ranged from 44.6 cm to 49.8 cm. Such characteristics as critical parameters, mean cross section ratio, reactivity coefficients for different materials, Doppler reactivity effect in sample heating, control member prototype worth, reaction rate distribution, void effect in Pb, reactivity effect in hydrogen insertion, reactivity effect in fuel melting simulation, distribution of fission neutron weights, delayed neutron effective fraction were measured. Different codes with different constant's systems were used by experts of Institute of Physics and Power Engineering (Obninsk, Russia), and Kurchatov Institute (Moscow, Russia) to compare the experimental and analytical values.

For verification purpose of constant support for BREST-300 reactor analyses the following critical experiments were analyzed by Monte-Carlo method at Kurchatov Institute:

- three configurations at BFS-2 rig (Institute of Physics and Power Engineering);
- critical experiments at PF assembly (Institute of Physics and Power Engineering)with Pb blocks and highly enriched uranium;
- six compositions with cylindrical core and highly enriched uranium and Pb disks in the core and reflector at critical assembly ROMB (All-Russian Research Institute for Technical Physics);
- six compositions with spherical core of semi-nuclear grade plutonium and reflector of highly enriched uranium at assembly ROMB (All-Russian Research Institute for Technical Physics).

The spectrum of neutron leakage from the Pb sphere was measured using Cf-source in the range from 0.1 to 10 MW, and inelastic scattering cross sections for Pb were measured by time-of-flight method at Institute of Physics and Power Engineering to define more accurately the differential nuclear data.

CONCLUSION

Nuclear data for Pb have been defined more accurately.

2. HYDROTHERMAL EXPERIMENTS TO JUSTIFY CORE HYDROTHERMAL ANALYSIS

The following studies have been performed at sodium-potassium (22 % Na + 78 % K) eutectic rig, Institute of Physics and Power Engineering [2, 3, 4]:

- heat transfer coefficient in square fuel element grids with relative pitch S/d=1.28-1.46 has been defined;
- the effect of fuel element square spacer grids on heat transfer has been defined;
- heat transfer at initial portion of fuel element square grids has been defined;
- the effect of power density varying with height on heat transfer coefficient has been defined.

The experiments for defining the effect of coolant technology and corrosion films on heat transfer coefficient have been conducted at Pb rig, Institute of Physics and Power Engineering.

CONCLUSIONS

Necessary heat transfer coefficients and empirical dependencies for the core and loop hydrothermal analysis have been obtained as a result of experiments.

3. STRUCTURAL STEEL CORROSION TESTS FOR SERVICE LIFE AT LEAD-COOLED NONISOTHERMAL RIGS

Structural steels for reactor cores were tested for service life during 8500 hours at two rigs with test section temperatures 620 and 650 °C, and lead flow velocity 1.6 m/sec, Institute of Physics and Power Engineering. The effect of temperature, coatings, alloying elements, steel structure, coolant technology and other parameters on steel corrosion resistance in Pb was studied at the rigs. On completion the in-Pb tests, oxide films and mechanical properties of the steels have been studied.

Structural steels for the vessel and steam generators were tested for service life during 12500 hours at the rig with test section temperature 550 °C, and lead flow velocity 1.7 m/sec, Central Research Institute of Structural Materials. The effect of temperaturs, coatings, alloying elements, steel structure, coolant technology stress and other parameters in Pb was studied at the rig. Welded joints and steels in zones of stagnation were tested. On completion the benchmark in-Pb tests steel corrosion resistance and mechanical properties were studied.

Fuel element prototypes simulating the fuel by molybdenum and uranium nitride with Pb sublayer and different additions to it were tested for service life in furnaces during 5000 hours, at temperature drop through the fuel element from 540 to 700 °C, and constant temperatures 650 and 700 °C, Institute of Physics and Power Engineering. The effect of temperature, coatings, additions to the Pb sublayer, alloying elements, steel structure and other parameters on steel corrosion resistance was studied at fuel element prototypes. On completion the in-furnace tests of the fuel element prototypes, the effects of Pb-sublayer on steel corrosion resistance were studied.

CONCLUSION

As evident from the results of the conducted tests and study, there is a possibility to use Pb-Bi coolant technology for the Pb. To achieve steel corrosion resistance as high as possible the parameters of oxygen treatment have been chosen preliminary. Structural steels for the core, vessel and steam generator have been chosen. Steel corrosion rate for service life of the core, steam generator and reactor has been predicted preliminary.

4. LEAD INFLAMMATION

When discussing lead-cooled reactors at seminars and scientific and technical councils, any time an issue on its combustion arisen. As referred to [5], which was written on the base of GOST 12.1.044-88 «Fire and explosion hazard of substances and materials. List of parameters and methods of their definition» lead is classified as combustible substance. With dust size 74 μm self ignition point of aerogel is 270 °C, aerosuspension is 580 °C. Regarding iron parameters, it also it the combustible substance with dust size 74 μm, the self ignition temperature of aerogel is 170 °C, aerosuspension is 320 °C. Therefore studies and experiments were carried out on liquid lead ignition.

Lead ignition processes with slow air leak in and prompt air breaking through, as well as oxygen supply to the gas space over the lead were simulated at liquid lead temperature 1200 °C were simulated in Institute of Steel and Alloys. Ignition, flash and combustion both with and without additional ignition source in the form of powerful spark discharge were not observed.

In lead metallurgical fabrication and refining using open kettles or reverberatory furnace at temperature 1200 °C with oxygen-enriched air blow through, at temperature of the exhaust gases and vapours ~1100 °C no ignition and combustion of lead and vapour were observed.

Thermodynamic analysis and modeling of liquid lead ignition possibility at 900-1200 °C temperature have shown that there no liquid lead ignition, even if all the initial agents are taken at high temperature equal to 1200 °C, since lead and lead oxide evaporation reaction requires more heat, than oxidation reaction and the cycle of lead oxidation reactions may provide — evaporation processes will be endothermic and the main thermodynamic ignition condition will not be fulfilled.

CONCLUSION

As resulted from the experiments and thermodynamic analysis liquid lead is not combustible.

5. THE EXPERIMENTS ON THE RUPTURE OF STEAM GENERATOR TUBES

Experiments on bubbling water and steam at pressure up to 24.0 MPa and temperature 140-350 °C through liquid lead at temperature 550-600 °C and 50-3000 mm thickness were conducted in Polytechnical Institute, N. Novgorod. The possibility to abolish «vapour hammer» in tube rupture was studied.

Experiments on rupture of the tube containing water of supercritical parameters were conducted at All-Russian Research Institute for Technical Physics (Russia), and the effect of the rupture on the adjacent tubes was studied.

CONCLUSION

Simple engineering solutions allowing to avoid «vapour hammer» have been found. It has been shown that the rupture of one steam generator tube will not cause the rupture of adjacent tubes. A possibility to use double-cycle cooling system in the reactor has been demonstrated.

CONCLUSIONS

The R&D works have shown that the concept of lead-cooled fast reactor BREST is vital.

III. DEVELOPMENT OF LEAD-COOLED FAST PILOT REACTOR OF 300 MW POWER WITH ON-SITE FUEL CYCLE AT RDIPE SVERDLOVSK BRANCH SITE.

Technical documentation for NPP with lead-cooled fast pilot reactor of 300 MW power with on-site fuel cycle at RDIPE Sverdlovsk branch site is planned to develop in 1998-1999. The reactor is aimed at

1. New coolant management.
2. Neutronic and hydrothermal study.
3. Tests for service life.
4. Demonstration of stability to accidents with and without scram:

- insertion of total reactivity margin;
- primary and secondary pumps off;
- steam generator tubes broken;
- freezing - unfreezing;
- imposition of simultaneous accidents;
- limiting accidents;

– «devised» accidents.

The following experimental works are planned to carry out to justify the technical documentation:

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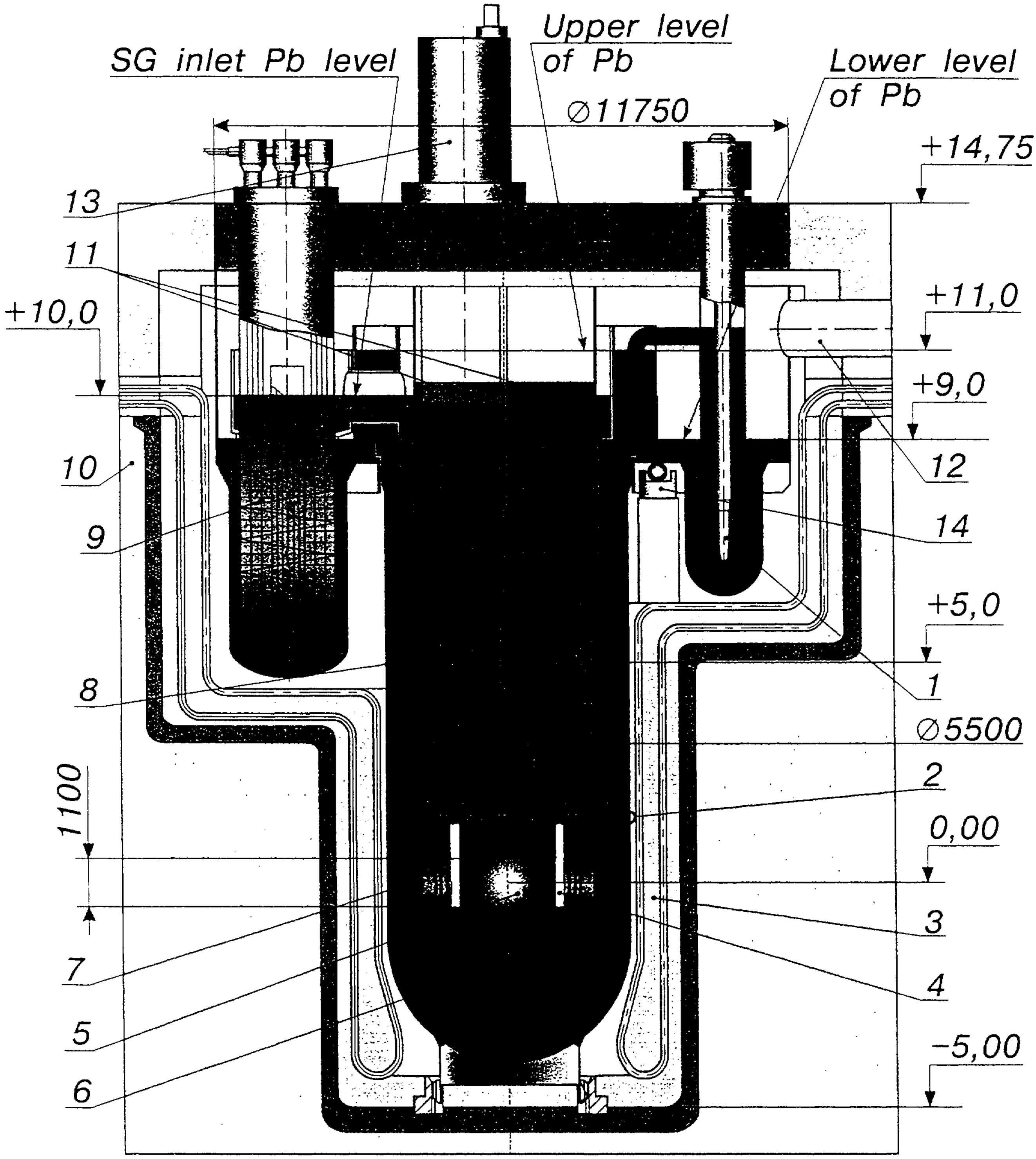

1 — pump
2 — reactor vessel
3 — termal protection
4 — CPS
5 — core
6 — support pillars
7 — dividing barrel with a support grid
8 — in-pile FA storage
9 — steam generator
10 — concrete vault
11 — rotating plugs
12 — emergency to throw of steam
13 — transfer machines
14 — vessel support

Fig. 1 General view of BREST–300

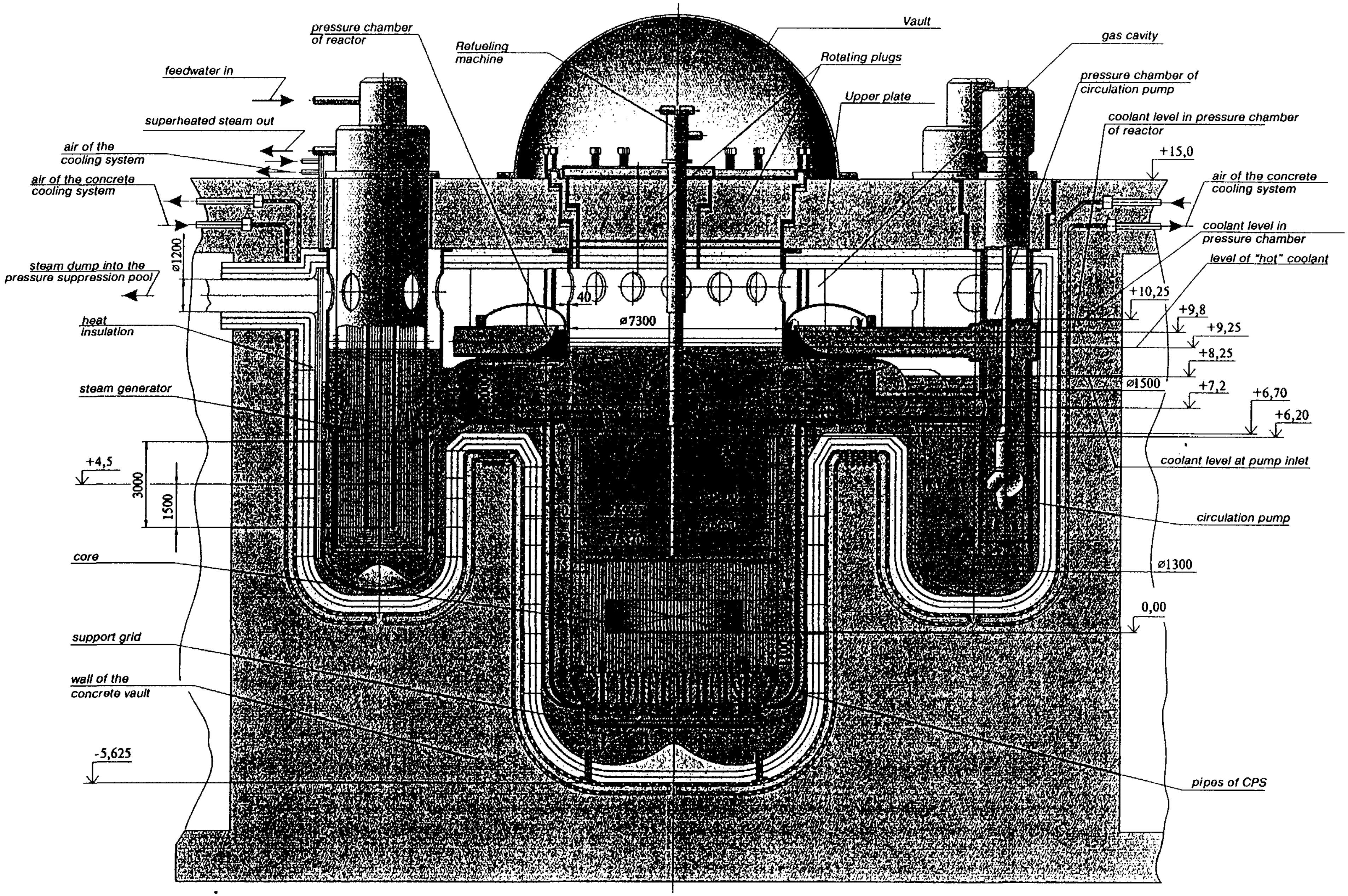

Fig.2 Reactor BREST–1200

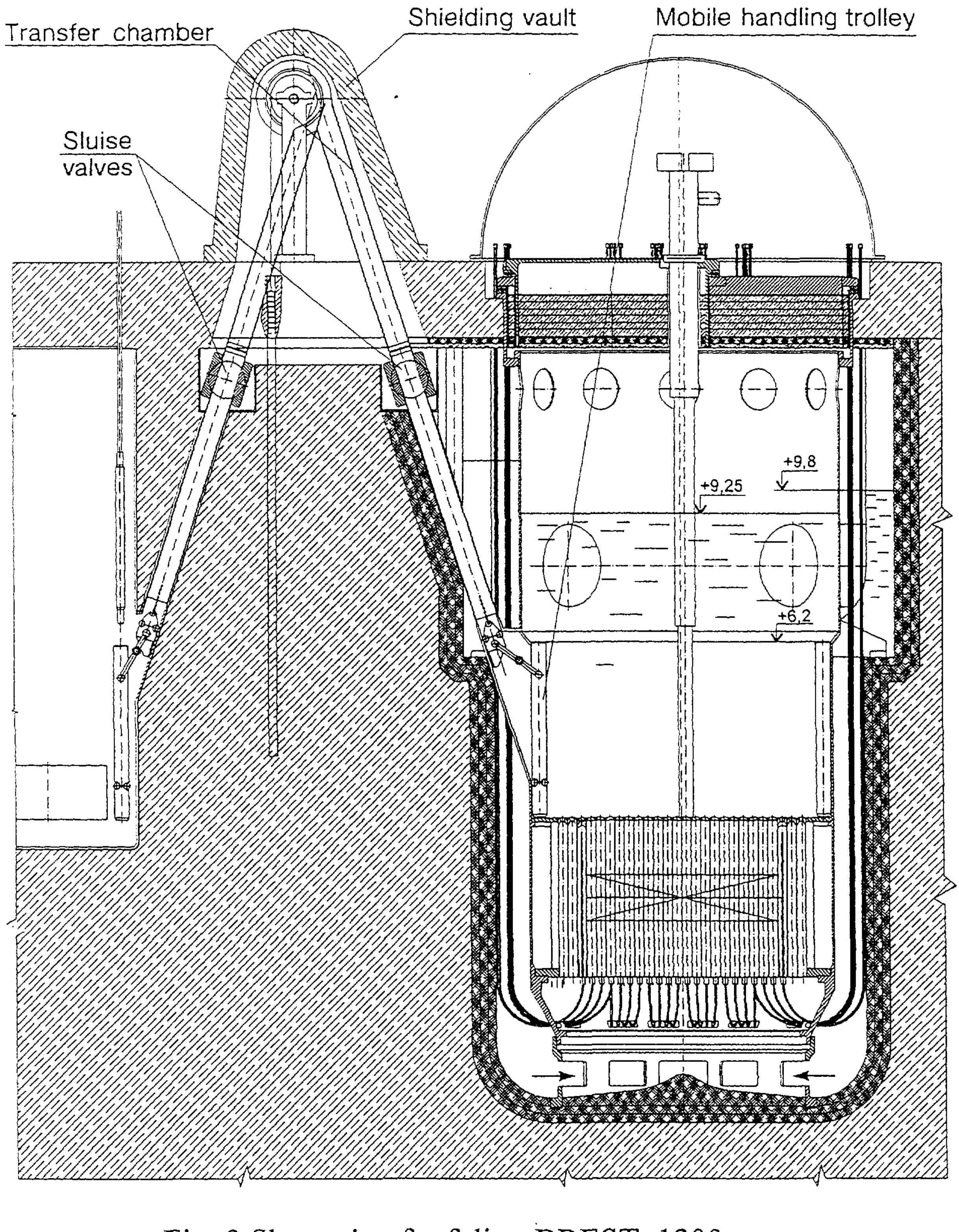

Fig. 3 Shematic of refuling BREST–1200 reactor