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BREST REACTOR AND PLANT-SITE NUCLEAR FUEL CYCLE

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The fundamental design of BREST-OD-300 and BREST-1200 reactors is presented. It is shown that it is necessary to develop nuclear energy complexes based on the BREST nuclear system. The stages, the status of scientific research and development work over the entire nuclear power complex, and the results of an expert analysis of the BREST-OD-300 design are presented.

Nuclear Power Complex. To meet the requirements imposed on large-scale nuclear power, the questions of a closed nuclear fuel cycle and fuel regeneration become fundamental, since the duration of the extra-reactor part of the cycle must be as short as possible.

The residence time of fuel in the BN-800 fast sodium reactor is 1.4 years. In water-chemical fuel recovery at a centralized plant, the duration of the part of the fuel cycle outside the reactor is ~7 yr, i.e., there will be five times more fuel outside than inside the reactor. In addition, it will be possible to put six times fewer fast reactors into operation using the plutonium from the irradiated fuel of thermal reactors. The number of fast reactors operating on excess plutonium from fast reactors will also decrease by a factor determined by the ratio $T_{in}/(T_{in} + T_{out})$, where T_{in} is the fuel run in the reactor and T_{out} is the duration of the extra-reactor part of the fuel cycle, as if the excess breeding (BR − 1) was less by the same factor. Then, plutonium breeding in fast reactors becomes pointless.

It follows that a closed nuclear fuel cycle for fast reactors must be on-site and fuel recovery must permit working with high-level fuel. This is also supported by the shipment of fresh and irradiated uranium–plutonium fuel to a centralized plant. Shipping fuel from one BREST-1200 reactor to a single plant (based on experience VVÉR-440) will require 10 railroad cars per year, and correspondingly 100 reactors will require 1000 cars (approximately 6 cars per day: three in and three out). In so doing, 8000 containers with high-level nuclear fuel (50–100 kg plutonium per container) will be used, which will require accident measures, radiation safety measures, and adherence to a nonproliferation regime, all of which are difficult to ensure at the present time.

When long-lived actinides are burned in a reactor, only fission products will be present in the repository for radioactive wastes. After a holding time of 150–200, years the fission products can be buried at the uranium mining site without destroying the natural level of radioactivity and without long-distance shipments of high-level materials.

Thus, a nuclear power complex must consist of a nuclear power plant, a nuclear fuel cycle on the plant site, and a waste repository on the plant site.

Design of the BREST-1200 Reactor System. The BREST-1200 reactor system is a two-loop steam-generating power-generating unit, which contains a reactor with steam generators, pumps, equipment for reloading fuel assemblies, a safety and control rod system, a concrete shaft with a heat shield, a steam-turbine unit, systems for removing heat during

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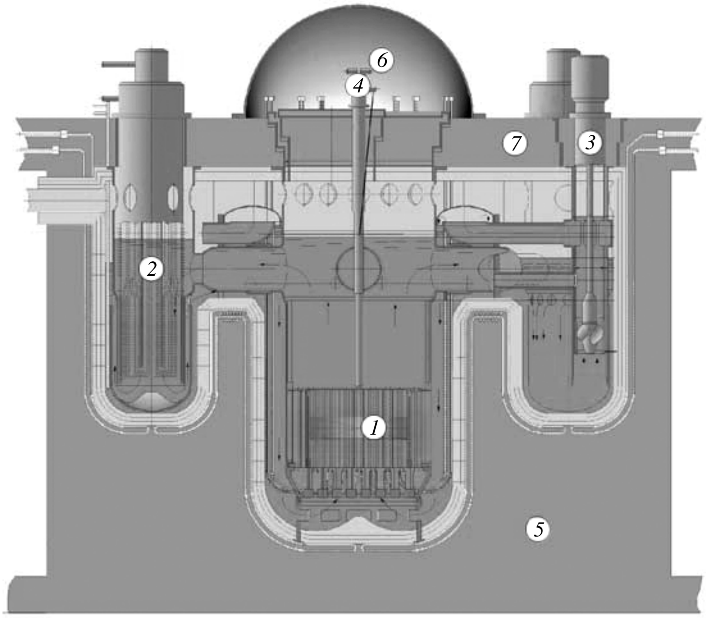

Fig. 1. BREST-1200: *1*) core; *2*) steam generator; *3*) pump; *4*) reloading machine; *5*) reactor shaft; *6*) sealing cap; *7*) top plate of cover.

cooldown and heating the reactor, a system for protecting the reactor system from excess pressure, a system for purifying the first-loop coolant, a gas purification system, and other auxiliary systems (Fig. 1).

The fuel being considered is high-density (14.3 g/cm^3) and high thermal conductivity (20 W/(m·K)) mononitride mixed fuel (UN–PuN–Np, Am, Cm, and others), which is compatible with lead and the cladding material of the fuel elements. The cladding material consists of ferrite–martensite chrome steel [1].

To decrease the fuel temperature and the migration of fission products from the fuel to beneath the cladding, the gap between the fuel and cladding is filled with lead, which provides good thermal contact between the fuel and coolant. To increase the coolant flow section, increase the power removed by natural circulation of lead, and eliminate cooling losses in the fuel assemblies when the flow is covered, all fuel assemblies are made without jackets.

Instead of the standard smoothing of the radial distribution of energy release by enriching fuel, three-zone variation of heating of the lead is used and the temperature of the fuel element cladding is changed by shaping the energy release and the flow rate of lead in a fuel assembly by using fuel elements with different diameter but with the same plutonium content. This gives good equalization and stability of the lead temperature at the core exit and the temperature of the fuel element cladding.

The use of chemically inert melted lead, which has a high boiling point, makes it possible to do without a three-loop scheme for removing heat and switch to a two-loop scheme with steam superheating of steam and additional heating of the feed water to 613 K by live steam.

Heat is removed from the reactor core by forced circulation of lead by pumps. Circulation through the core and the steam generators is effectuated not by the head provided by pumps but rather by the difference between the “cold” and “hot” coolant levels created by the pumps. In the process, the nonuniformity of the lead flow rate through the steam generators when one or several pumps are switched off is eliminated and inertial flow is produced with rapid shutdown of the pumps as a result of equalization of the coolant levels in the head and intake chambers (~20 sec).

A central-circuit arrangement of the first loop is used to decrease the consequences of an accident where the pipes of the steam generators rupture. In this arrangement the steam generators and the main circulation pump are placed outside the main reactor vessel. This arrangement, together with the scheme chosen for lead circulation and release of steam from the reactor vessel into bubbler, prevents a dangerous quantity from flowing into the core and pressure on the reactor vessel.

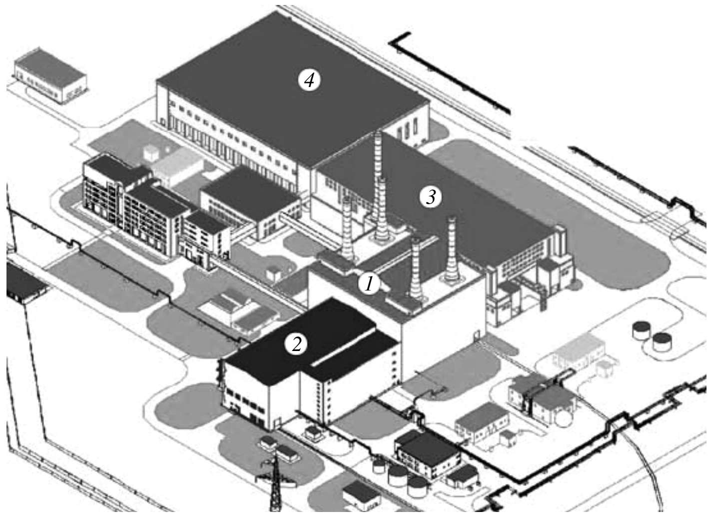

Fig. 2. Nuclear power complex with a BREST-OD-300 reactor: *1*) reactor; *2*) turbines; *3*) complex for storing and reprocessing radioactive wastes; *4*) complex for the on-site nuclear fuel cycle.

In-vessel storage of spent fuel, removed from the core and protected from neutrons, permits accelerating and simplifying the removal of irradiated fuel from the reactor by first holding the fuel until the heat is released (~0.5 yr), which admits reloading and transport operations without forced cooling.

The low pressure in the lead loop and the relatively high freezing temperature of lead promote self-healing of cracks, and together with sinking the reactor in a manner where the lead is at ground level, eliminates accidents with loss-of-coolant, melting of fuel elements, and leakage of radioactive lead into the hall of the reactor facility.

The large size and mass of the reactor create difficulties with fabricating, transporting, assembling, and seismic stability of the structure. In BREST-1200, a pool arrangement of the reactor, pumps, and steam generators directly in the concrete shaft with heat shielding by the metal vessel is used. The temperature of the concrete is maintained within admissible limits by natural circulation of air along Du120 pipes. The inner of the reinforced concrete mass is lined with 8–10 mm thick steel facing.

A floor machine reloads fuel in the BREST-1200 reactor. In-reactor reloading includes the operations for loading fresh fuel assemblies, reflection blocks, and safety and control rods from in-reactor storage into the core and removing irradiated fuel assemblies and other components from the core which are placed into in-reactor storage. In-reactor reloading is conducted with rotating plugs and an in-reactor machine, which is placed on a small rotating plug.

Calculations of emergency situations, including serious accidents, confirmed that the reactor is resistant to such situations and that radioactive emissions requiring evacuation of the population are eliminated [2].

To substantiate the design of the nuclear power complex with a BREST-1200 reactor, it is necessary to develop an experimental demonstration complex, whose main problems are comprehensive physical, thermohydraulic, and technological studies under reactor conditions, service life tests, demonstration of reactor stability with respect to serious initial emergency events, including without actuation of the safety and control system, adoption of an on-site nuclear fuel cycle, and handling of radioactive wastes. In this connection, a design was developed for a nuclear power complex with a BRET-OD-300 reactor system (Figs. 2 and 3) with an on-site nuclear fuel cycle for the site of the Beloyarskaya nuclear power plant, including the following:

- technical designs of the reactor facility, a steam generator, pump, cover, reactor shaft, reloading machine, systems for heating, receiving, preparing, and filling with coolant, pressure compensation, purification of radioactive gas, coolant treatment with gas mixtures, air cooling of the shaft, normal and emergency cooldown, and localization of a leak in a steam generator;

• designs of a nuclear power plant and an on-site nuclear fuel cycle at the plant (general plan, technological solutions, many vessel, machine hall and second loop, construction solutions, construction organization, preliminary report substantiating safety, assessment of environmental effects, design-research work, technical designs of equipment used in the nuclear fuel cycle at the plant site –dismantling of fuel assemblies, fuel recovery, fabrication and fuel elements and fuel assemblies, equipment for reprocessing radioactive wastes).

Main technical characteristics of the reactors:

	BREST-1200	BREST-OD-300
Power, MW:		
thermal	2800	700
electrical	1200	300
Core:		
diameter, mm	4750	2100
height, mm	1100	
Fuel elements diameter, mm	9.4–9.8–10.5	
Fuel element spacing in a square lattice, mm	13	
Fuel	UN + PuN	
Fuel load (U + Pu)N, tons	60	17.6
Plutonium, Np, Am, Cs, and other contents, % h.a.	13.5	
Interval between refuelings, eff. days	300	
Average burnup of extracted fuel assemblies, % h.a.	6.2	
Maximum burnup in extracted fuel assemblies, % h.a.	10.2	
Run time, eff. days	1500	
BRA	~1.05	
Capacity utilization factor	0.82	
Net efficiency of power generating unit, %	~43	

Design of a Nuclear Power Plant with a BREST-OD-300 Reactor System. The BREST-OD-300 power-generating unit was designed as an experimental–demonstration unit. After the tests have been completed, it will be put into commercial operation delivering power into the power system. The thermal scheme is a two-loop scheme, the first loop containing lead with the required purity and the second loop containing water–steam with supercritical parameters. The second loop is nonradioactive and consists of steam generators, the main steam pipes, a feed-water system, and a single turbine unit. The water chemistry in the second loop, adopted on HPP blocks with supercritical pressure, is a neutral-oxygen regime with no de-aerator. The K-300-240-3 serially produced turbine facility developed by the Leningrad Metal Works contains a steam turbine with systems for rotating the shaft, lubrication, regulation, and hydraulic lift of the rotors, and others. In contrast to existing designs of nuclear power plants, the second loop is not required to remove heat in an emergency situation.

The buildings of the main loop were designed taking account of the norms for designing earthquake-resistant nuclear power plants. For design purposes, an earthquake of magnitude 6 was assumed and the maximum earthquake was taken to be magnitude 7 on the MSK-64 scale. The spatial system of the reactor buildings is designed in monolithic reinforced concrete to ensure reliable operation of the structures. To decrease inertial seismic forces, the building was designed to be symmetric with dimensions in the plane 65 × 74 m, separated by crossed vertical bearing diaphragms.

On-Site Nuclear Fuel Cycle [3]. The technical design of the BREST-OD-300 on-site nuclear fuel cycle was developed on the basis of natural safety principles:

• deterministic elimination of serious radiation and nuclear accidents during reprocessing and fabrication of nuclear fuel by creating nuclear-safe equipment; the critical mass of a sphere of fuel with equilibrium composition and a concrete reflector is ~ 1100 kg, up to three irradiated fuel assemblies with a total mass of nuclear material about 373 kg are in reprocessing;

• high level of radioactivity of the fuel 50–500 Ci/kg, facilitating theft protection;

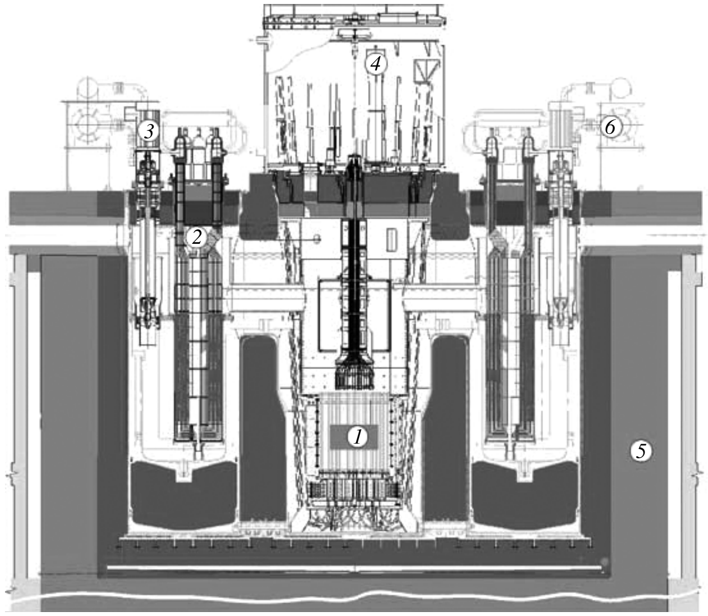

Fig. 3. BREST-OD-300 reactor: *1*) core; *2*) steam generator; *3*) pump; *4*) reloading machine; *5*) shaft; *6*) cooldown system.

• elimination of uranium enrichment and plutonium separation, no inter-object transport of fresh and irradiated nuclear fuel;
• simplified handling of radioactive wastes as a result of fractionation of the wastes, transmutation of the actinides and long-lived fission products in the nuclear reactor;
• uranium-plutonium fuel with the equilibrium composition with added depleted or natural uranium.

The yearly production program of the cycle provides for regeneration and fabrication of 29 BREST-OD-300 fuel assemblies and 259 BN-800 fuel assemblies. In the process, three size types of fuel elements and fuel assemblies for the BREST reactor and two types of fuel elements for a fast sodium reactor are disassembled, regenerated, and fabricated.

The technical design of the equipment used in on-site nuclear fuel cycle includes apparatus for dissolving fuel assemblies, setups for regenerating fuel (electrolyzer), obtaining mononitrides, a cassette press, continuous-operation furnaces for removing binding agents and sintering, equipment for assembling, sealing, and monitoring few elements, equipment for fabricating fuel assemblies, and designs of control systems.

The BREST-OD-300 on-site nuclear fuel cycle is divided into two phases:
• the first phase includes sections for fabricating powder from the initial nuclear materials, pellets, few elements, and fuel assemblies and is intended for preparing the first fuel loads for BREST-OD-300 (145 fuel assemblies, 17.6 tons) and BN-800 (776 fuel assemblies, 31.7 tons) in the course of a year;
• the second phase includes sections for disassembling and regenerating irradiated fuel, and it will be put into operation three years after the first phase.

The components of the nuclear power complexes for BREST-1200 and BREST-OD-300 are in different stages of development: BREST-1200 is at the conceptual level, with respect to the nuclear power plant, on-site nuclear fuel cycle, and reprocessing wastes. Technical and economic studies have been done for the site of the Novovoronezh nuclear power plant for subsequent economic comparison. The investigations were performed assuming new norms will be developed for natural safety reactors.

Status of Research and Development Work. Full-scale research and development work was performed on all components of the nuclear energy complex before 2002. However, because of cutbacks in financing, at the present time only reactor tests of fuel elements with mononitride uranium–plutonium fuel are being performed on the BOR-600 reactor, tests are also being performed to study the influence of lead on the long-time mechanical properties and characteristics of the cyclic strength of the structural materials 10Kh15N9S3B and 10Kh9NSMFB (TsNII KM Prometei) and on developing the working documentation for an automatic assembly line for manufacturing fuel elements (Sverdlovsk Research Institute of Chemical Machines). All other research and development work was stopped at the following stages. For coolant technology [4], all research was completed, a second edition of the rules for handling lead coolant was prepared, and a representative stand for a final substantiation of the coolant technology was half completed. The research and development work on the development of specific units of the reactor need to be continued.

Corrosion tests were completed on the SM-2 stand at the Physics and Power Engineering Institute (~$16.6 \cdot 10^3$ h), and positive results were obtained on substantiation of the corrosion resistance of steel. The depth of the oxidized surface layer of ÉP-823 steel was 2–10 µm with an admissible value of 100 µm on the inner and outer sides over 40000 hours. In the future, the corrosion tests will have to be continued under load (heat flux, pressure) and tests on individual units of the reactor will have to be continued also.

The Research and Design Institute of Electrical Technology together with the Physics and Power Engineering Institute, the All-Russia Research Institute for Standardization in Machine Engineering, the Research Institute of Nuclear Reactors, and other enterprises developed, fabricated, and tested in a BOR-600 reactor an autonomous channel with lead coolant (ACLC) in which fuel elements with nitride uranium–plutonium fuel can be tested under conditions approximating BREST-OD-300 conditions (inner sublayer of a fuel element, circulating lead flowing on the outside with the oxygen regime maintained). The tests showed that different types of metallic coolants – sodium and lead – can be combined. No corrosion damage to the pump wheel, fuel assembly jacket, spacing lattices and fuel elements was found even though the oxygen level in the coolant fluctuated, and the fuel elements remained airtight. The construction of the channel showed that in the future it will be possible to perform complex tests on it [5].

An assembly line was developed at the All-Russia Research Institute of Standardization in Machine Engineering, and 15 fuel elements with uranium–plutonium fuel were fabricated [6]. Two experimental fuel assemblies were fabricated at the Rresearch Institute of Nuclear Reactors and tested in the BOR-60 reactor [7]. In June 2007, the first fuel element with 3% burnup was extracted to perform tests and regenerate fuel by the electrochemical method. After the tests in BOR-60, tests must be performed on uranium–plutonium fuel in BN-600.

A full-scale stand was built at the Physics and Power Engineering Institute to study heat transfer and its stability in a steam generator (water with transcritical parameters–lead), thermohydraulic experiments were performed on electrically heated models of fuel assemblies to obtain closure relations and the hydraulic characteristics of lattices [8], and production of sensors for monitoring oxygen and lead was restored. Experimental studies were performed on BFS to develop a constants library and software and to verify and certify them for evaluating the analysis of accuracy in predicting the main neutron-physical parameters of BREST-OD-300 [9].

A full-scale automatic model of an industrial assembly line for fabricating fuel pellets was developed at the Sverdlovsk Research Institute of Chemical Machines and tested [10].

A stand on IVV-2M for performing experimental work to study the properties of structural materials and the activation of the coolant and mass transfer of radionuclides when performing simulations in the system lead–gas phase–structural materials was built at the Institute of Reactor Materials (Zarechnyi) but not mounted [11].

Full-scale stands for studying a mixing feed-water heater and a steam–steam heat exchanger were built at the Dzerzhinskii All-Russia Heat Engineering Institute, and the first series of experiments was conducted [12]. A stand for testing a reloading machine was developed and partially built by the Central Machine Design Bureau.

Experiments on the cyclic stability of fuel elements with freezing and melting of a lead sublayer and accidental temperature jumps were performed at the Research Institute of Nuclear Reactors to substantiate the serviceability fuel elements in an experimental fuel assembly for tests in the BOR-60 reactor.

Investigations of the effect of the oxygen concentration in lead on heat transfer [13] and experiments on the behavior of lead accompanying ruptures of steam generator tubes [14] were performed at NGTU (Nizhnii Novgorod).

A technology for manufacturing 10Kh15N9S3B and 10Kh9NSMFB steels and the technical requirements for the required assortment were developed at factories.

Analysis by Experts. In 2001, experts analyzed the design of a nuclear energy complex with a BREST-OD-300 reactor. One hundred seven specialists and independent experts participated in this analysis. It did not reveal any fundamental problems which would impede implementation of the design. The final conclusions of the experts are as follows:

1) for developers of the BREST-OD-300 design:

- complete the substantiation of the strength and earthquake resistance of the reactor facility, analyze an accident with rupture of steam generator tubes, finish the work on the design of the core taking account of the salient features of the first load;
- prepare a substantiation of the representativeness of the planned comprehensive experimental analysis of the coolant and hydrodynamics in the first loop;
- prepare suggestions concerning the schedule of scientific research and development work with a description of the main stands and the best sequence for developing the design;

2) examine the materials of the BREST-OD-300 design, the results of their analysis by experts and the suggestions made in paragraph 1 of the recommendation part of the present protocol on the Scientific and Technical Council (STC) No. 1 of the agency with a preliminary analysis of the corresponding questions on STC No. 4.

A previous expert analysis performed by the energy division of the Russian Academy of Sciences arrived at similar conclusions.

Conclusions. The research and development work performed to substantiate the nuclear energy complex of a BREST reactor facility confirmed the possibility of demonstrating and finishing the work on this reactor concerning the physics and technology of fast reactors with natural safety, but this work must be completed.

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