BREST-OD-300 REACTOR FACILITY: DEVELOPMENT STAGES AND JUSTIFICATION

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Abstract. BREST-OD-300, an innovative inherent safety fast reactor, is being developed as a pilot and demonstration prototype for the basic commercial reactor facilities of future nuclear power with a closed nuclear fuel cycle.

Coolant in the reactor facility is lead, the layout of the primary circuit is integral, the reactor vessel material is multilayer metal concrete. The reactor core design uses mixed uranium-plutonium nitride as the fuel, and the fuel elements are contained in shroudless fuel assemblies (FA). Small reactivity margin, excluding prompt-neutron runaway is provided in the core.

Decisions are based on a computational and experimental justification. To confirm the fuel serviceability, radiation tests of fuel elements are conducted in fast reactors. Full-scale fuel-free mockups of FA are tested. Tests have been conducted of the vessel elements. Experiments have confirmed the absence of a dependent break of steam generator tubes. Neutronic codes have been verified, including with the use of BFS critical assemblies. Loop facilities have been built on which studies are conducted to determine the radionuclide release from the coolant.

It has been shown based on the calculation results that the probability of the core damage (without core melting) for nuclear power plants with the BREST-OD-300 reactor facility does not exceed $8.65 \cdot 10^{-9}$ 1/year.

Key Words: calculation experiments, fast reactor, inherent safety.

Manuscript

An innovative fast reactor BREST-OD-300 with inherent safety is being developed as a pilot and demonstration prototype for the basic commercial reactor facilities of future nuclear power with a closed nuclear fuel cycle [1].

The tasks of such systems are:

– elimination of nuclear accidents requiring evacuation and, more than that, resettlement of the public;

– closure of the nuclear fuel cycle (NFC) for the full use of the energy potential of uranium raw material;

– consecutive approach to radiation-equivalent (relative to natural raw materials) RW disposal;

– technological strengthening of nonproliferation regime (consecutive abandonment of uranium enrichment for nuclear power, of weapon-grade plutonium production in the blanket and of its extraction during SNF processing, reduction in the nuclear material transportation volume);

– ensuring competitiveness against other energy generation types.

The lead coolant properties make it possible to implement in fast reactors the following:

• in combination with application of (U-Pu)N fuel, complete breeding of fissile materials in the reactor core, which provides for a constant small reactivity margin preventing the disastrous effects of an uncontrolled power increase when implementing the reactivity margin because of equipment failures and personnel errors [1-2];
• to avoid the void effect of reactivity due to a high boiling point and high density of lead;
• to prevent coolant losses from the circuit in an event of the vessel damage due to high melting/solidification points of the coolant and the use of an integral layout of the reactor;
• to provide for high heat capacity of the coolant circuit which decreases a possibility of fuel damage;
• to allow for utilization of the high density of lead and its albedo properties for flattening the FA power distribution and the fuel pin temperatures respectively, as well as in the safety systems;
• to facilitate larger time lags of the transient processes in the circuit, which makes it possible to lower the requirements to the safety systems' rate of response [2].

One of the BREST-OD-300 development objectives is practical justification of the main design approaches applied to the reactor facility with the lead coolant based on the closed nuclear fuel cycle (CNFC), and of the foundations of the inherent safety ensuring concept, on which these approaches are based [2]. For this reason, special attention is paid to justification of serviceability of the reactor core and its components.

A special attention in the reactor development is paid to justification of the reactor core and its components capacity. Mixed uranium-plutonium nitride is used to ensure complete breeding of fissionable materials in the reactor core and a constant small reactivity margin preventing prompt-neutron runaway during the reactor operation. A low-swelling ferrite-martensite steel is used as the fuel cladding.

To confirm the fuel serviceability, radiation tests of fuel elements are conducted in the BN-600 power reactor and in the BOR-60 research reactor. At the present time, 8 FAs with nitride fuel elements are being irradiated in the BN-600 reactor, and the fuel elements of a previously withdrawn two FAs are subjected to post-irradiation studies. Seven FAs with nitride fuel elements are being irradiated in the BOR-60 research reactor.

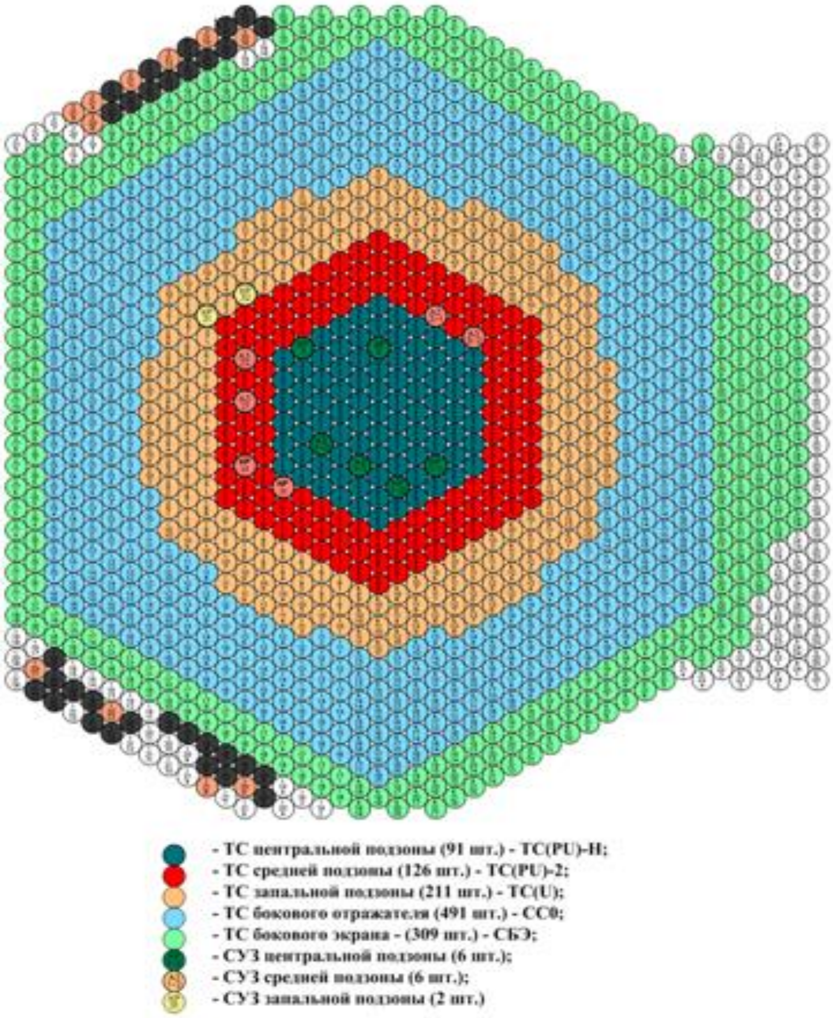
In the design of the reactor core items, novelty was coupled with reference solutions. The FA has a shroudless hexagonal design. Such solution eliminates a possibility of fuel melting when the FA flow area is blocked; because as per calculations if it is even occurred at the inlet of a 7 FAs group being blocked, the safe operation limits in terms of the fuel cladding temperature are not exceeded. Another positive point is a 30% reduction of the metal content of the shroudless FA as compared to the shrouded option. As regards to a manufacturing technology, the adopted design makes it possible to use the experience gained when fabricating the FAs for the VVER reactors.

To justify the FA design serviceability, the full-scale mockups (Fig. 1) had been manufactured, which were subjected to mechanical, hydraulic and vibration tests in the air and water environments. The mechanical tests included transverse bending, torsion, axial tension and compression. The vibration tests were conducted using running and stagnant water. Also, the vibration strength tests were performed in the air. The hydraulic tests of FA mock-up were conducted using the lead coolant.

FIG. 1. Full-scale FA mock-up and FA mock-up with a retort for testing.

In the reactor core composed of the shroudless FAs, important in terms of the fuel element temperature determination is the knowledge of local flow rates in hydraulic cells. To determine the intercell and intercassette mixing coefficients, respective experiments in liquid metal and air were carried out. A mockup 37-rod fuel bundle was used in the liquid metal experiments to refine the heat transfer coefficients. Thus, a large bulk of data was obtained, which allows for verification of the calculation codes intended for thermal-hydraulic calculations of the reactor core. To justify the corrosion resistance of the FA elements in the lead coolant, tests using small-scale fuel-free mockups of the FAs at different temperatures were conducted.

Absence of the data on the physical experiments at the critical assemblies with nitride fuel led to the necessity of carrying out an experiment using the large physical test bench BFS (Fig. 2) at Institute of Physics and Power Engineering (Obninsk). In the simulation lead, plutonium, and uranium nitride were used. Based on the results of the new experiments and the data obtained from the previous experiments, the calculation codes were verified and validated for neutronic calculations.



1 – FAs of central subzone (91) – FA(PU)-H;
2 – FAs of middle subzone (126) – FA(PU)-2;
3 – FAs of driver subzone (211) – FA(U);
4 – FAs of side reflector(491) – CC0;
5 – FAs of side screen (309) – SSA;
6 – CPS members of central subzone (6);
7 – CPS members of middle subzone (6);
8 – CPS members of driver subzone (2)

FIG. 2. Map of BFS with BREST-type fuel composition.

Results of the calculations carried out using the verified software tools show the possibility to achieve a small reactivity margin during the reactor start-up loading and operation. A practically stable power density field is ensured during the fuel life.

An integral layout is used in the reactor facility to avoid coolant losses [1, 3]. The reactor vessel where are located the main components of the primary circuit and lead coolant is made of multilayer metal concrete (Fig. 3).

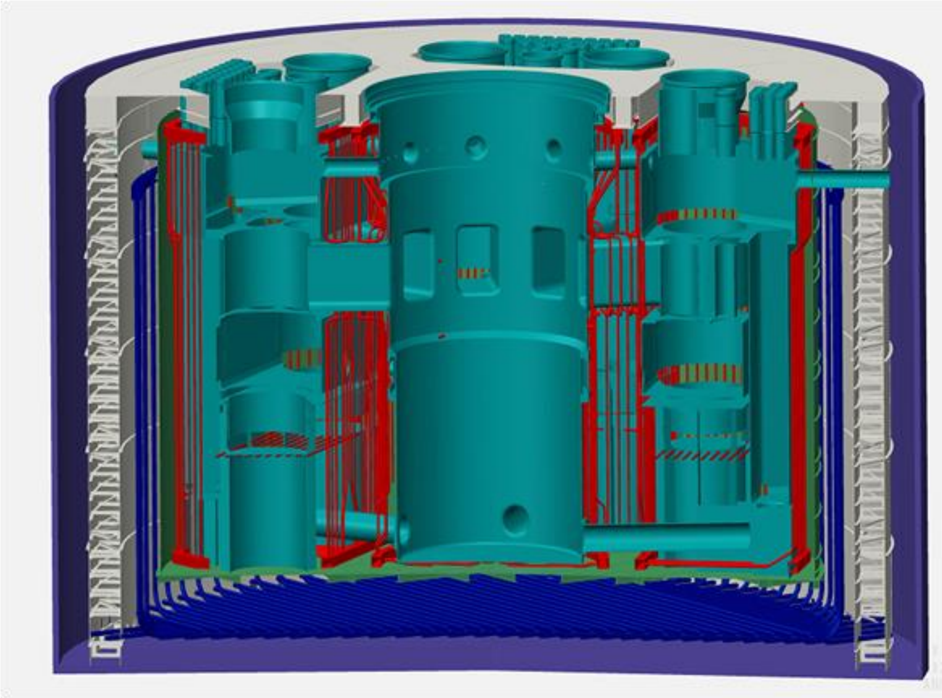


FIG. 3. Vessel of BREST reactor facility.

It took a wide range of calculation and experimental activities to justify serviceability of such vessel type, which is novel for the nuclear power industry [3-7]. The experimental justification is based on investigations and testing of the small- and full-scale components. Using the developed full-scale mockup of the vessel bottom a capability to ensure the required temperature of the building structures has been demonstrated, and joint thermal movements of the components have been determined. Using the developed full-scale mockup of the central part of the vessel (Fig. 4), heating-up modes have been optimized, and the gas emission parameters have been determined.

FIG. 4. Full-scale mockup of reactor vessel's central part.

To carry out the strength calculations, it was necessary to obtain the properties of the used materials, which required performance of several experimental works. Properties of the concrete have been determined in the conditions of operating temperatures and irradiation as well. With respect to the metal, corrosion resistance experiments in the lead coolant environment have been conducted. To justify safety, the concrete behavior was studied in the conditions of its direct contact with lead. The depth of lead penetration was experimentally determined to be no more than 0.5 mm without chemical interaction.

The structural analysis of the vessel was performed using the newly developed techniques [5]. The analysis took into account of the actual geometrical and physical mechanical properties of the vessel components and complex three-dimensional contact interaction between them, the non-linear concrete properties and the formation of cracks in it (Fig. 5).

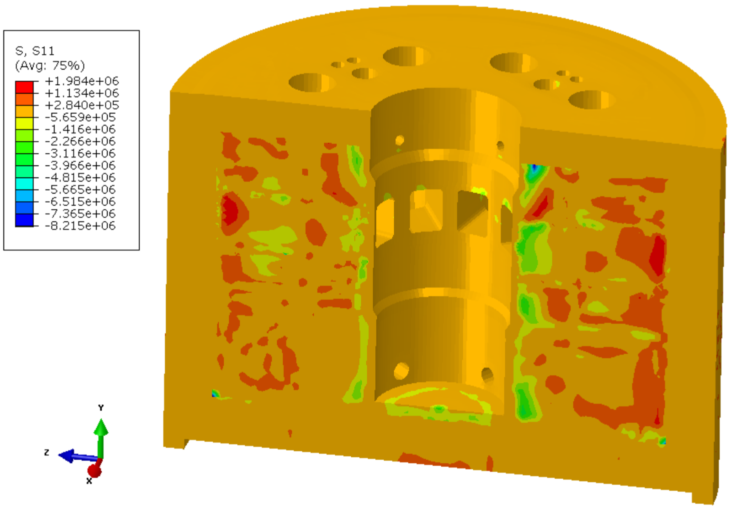


FIG. 5. Distribution of first primary stresses σ_1 in concrete filler of a fast reactor vessel by the end of heating-up.

The analytical justification showed that the adopted reactor vessel design ensures the probability of leak occurrence with partial coolant loss of no more than $9.7 \cdot 10^{-10}$ 1/year.

The integral layout with a steam generator (SG) located in the reactor unit vessel imposes a high responsibility on the developers, designers and experimentalists involved in the justification of serviceability and safety of the SG (Fig. 6). Therefore, a thorough justification of the steam generator components and the processes taking place in the steam generator has been planned and is being carried out.

In the course of the SG experimental justification several mockups had been developed, which were used to verify (check) the parameters, which was laid down in the detailed design.

To determine the thermohydraulic characteristics including the impact of centrifugal force on the thermohydraulic stability, a SG 18-tube model was developed (Fig. 7).

From the results of the 18-tube model tests, the heat transfer coefficients and hydraulic characteristics in the steam-water and lead circuits were obtained, as well as the temperature distribution in the lead circuit. Thermohydraulic stability was demonstrated in the investigated ranges [8]. Based on these data the SG operating parameter ranges were determined in terms of fluid flow rate.

FIG. 6. Steam generator of BREST reactor facility.

FIG. 7. The 18-tube model of SG.

Because of a high specific weight of lead, the probability analysis of dependent failure of the steam generator tubes if one of them breaks. The dependent failure and the subsequent ingress of steam into the coolant may in turn affect the circulation in the circuit and consequently impair thermal condition of the fuel elements. Based on the series of experiments (Fig. 8), it was demonstrated that it is impossible for a single SG tube rupture to develop into a multiple tube rupture (dependent rupture unavailability) [9].

FIG. 8. Tube rupture experiment.

To justify the steam generator life, thermal cyclic strength tests of the unit for securing the tubes between the tube sheets have been carried out (Fig. 9). Degree of reliability of the "tube-tube sheet" joints has been determined for superheated steam removal and feedwater supply chambers in the SG modules, and the fulfillment of the thermal cyclic strength conditions has been confirmed for the heat exchange tubes and the points where they are welded to the tube sheet.

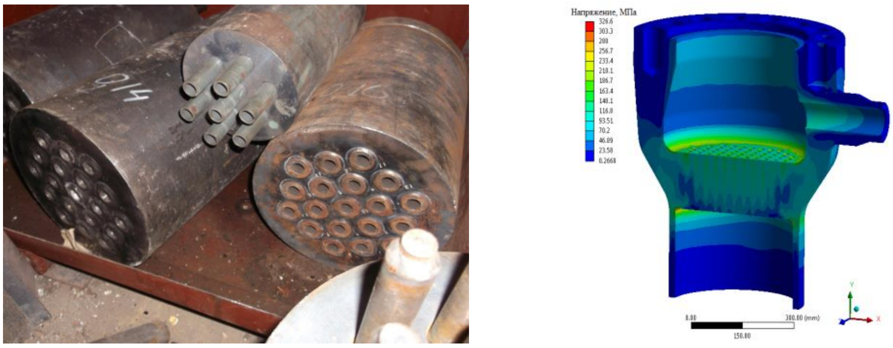


FIG. 9. Study in thermal cyclic strength of "tube-tube sheet" joint.

Tribological tests of the "tube-spacer grid" contact points in the lead coolant environment have been performed [10-11]. As a result, experimental data were obtained on the wear of the friction couples of the specimens in the characteristic range of forces and movements within the contact areas.

A complex three-dimensional analytic justification of the steam generator serviceability has been carried out, which included thermohydraulic calculations (Fig. 10), strength calculations for all operating conditions, vibration strength calculations, seismic effect, aircraft crash and air shock wave effect calculations, and other design analyses.

To verify the vibration calculations, a mockup of the SG with actual geometric parameters is being created.

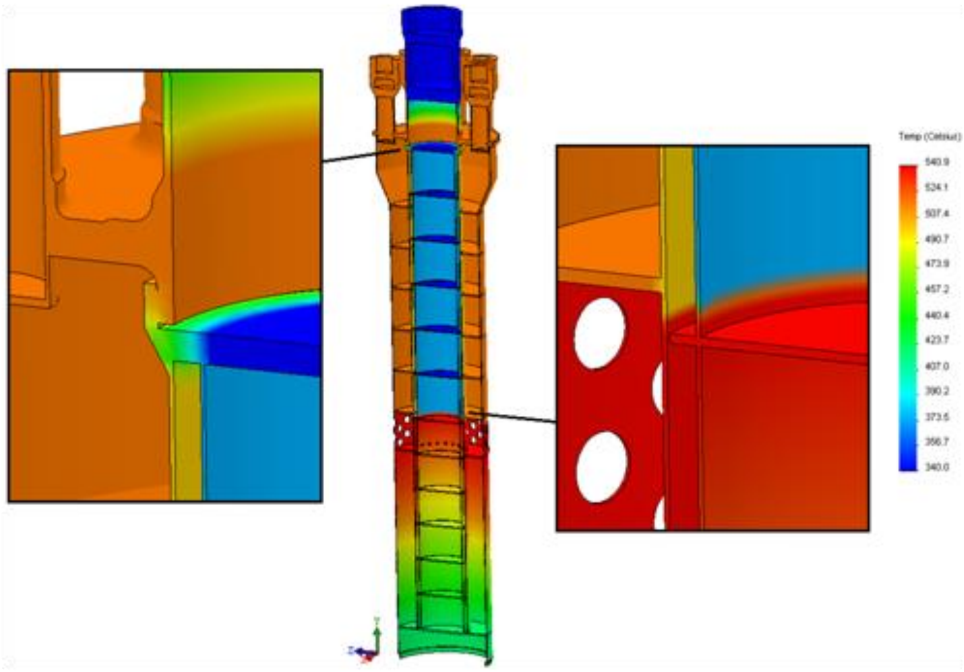


FIG. 10. Results of SG temperature field calculations.

The reactor coolant pump set (RCPS) (Fig. 11) is intended for generation of the lead coolant head and provision of its circulation in the circuit.

Fig. 11. Reactor coolant pump set of BREST reactor facility

To justify its serviceability, several mockups of the pump set have been developed, as well as the test sections to check their performance:

• a medium-scale test section operating with liquid lead and a RCPS mockup have been developed;

• energy characteristics of the lead coolant flow part have been obtained in the order of 80% of the required ones (test bench limitations);

• serviceability of a hydrostatic bearing unit has been demonstrated in the conditions of the medium-scale test bench (over 300 start-up-shutdown cycles);

• first test run of a natural-scale bearing was successfully performed in the lead environment with demonstration of its high capability to incur the radial loads;

• energy performance of the flow part in water has been optimized; the required flow, head and positive suction head have been obtained.

In the future, a test bench base will be set-up for the tests of the full-scale prototype RCPS, including the endurance tests.

Other main and ancillary components are being justified at the small- and medium-scale test benches; the properties of the structural materials in the operating temperature ranges and rated operating conditions, including irradiation, are being obtained. The main (largest) components developed for the BREST reactor facility have been justified through the experiments and calculations and are now being prepared for the prototype testing.

Another critically important direction of safety justification is the acquisition of data on the radionuclide transport in the reactor facility. To investigate the processes of radioactivity transport in the liquid-metal phase and the radionuclide exchange between the liquid-metal and gaseous phases, the following components were developed:

- an ex-reactor loop facility with lead coolant and gas circuit (Fig. 12);

- reactor loop facility with gas coolant;

- reactor loop facility with lead coolant and gas circuit.

Transport of coolant activation products (impurities in lead) ^{110m}Ag, ^{123m}Te, ^{124}Sb, ^{210}Po, ^{65}Zn and ^{210}Hg, as well as the fission products ^{131}I, ^{137}Cs and inert radioactive gases was investigated. The experimental results made it possible to perform the justified calculation of the reactor facility's irradiation characteristics.

FIG. 12. Ex-reactor loop facility with lead coolant and gas circuit.

The lead-cooled reactor facility is innovative one due to the engineering approaches applied to obtain the required technical and economical ratios in terms of the design, are different from those specified by the Russian regulations in the field of atomic energy use. The conducted experiments and completed calculations make it possible to proceed with licensing of the reactor facility development using these principles and to establish a foundation for the generation of the regulatory framework for the development of the commercial plants. Development and introduction of the new regulatory framework proceed in a staged manner as the experience is gained at each life-cycle stage defined by the licensing.

At the current stage the developed detailed design of the BREST-OD-300 reactor facility justified using the small- and medium-scale test benches and test sections, as well as on the basis of the analytical justification using the verified software tools, met the key parameters and requirements specified in the technical assignment and undergoes the licensing procedure as a part of the power unit project. At the following stages there are planned R&D completion, construction and operation of a power unit as a part of the pilot and demonstration energy complex.

It has been shown by the calculations that the probability of the reactor core damage (without core melting) does not exceed $8.65 \cdot 10^{-9}$ 1/year, and a maximum external radiation dose in case of heat removal loss by the reactor systems is generated at the distance of 1 km and does not exceed $5 \cdot 10^{-2}$ μSv during first 10 days, and in case of SG tube ruptures imposed on the isolation valve failures does not exceed 1 mSv. These circumstances do not require to implement the measures for population protection and ensure the acceptable level of safety when reactor facilities of such type are used for the nuclear power industry development.

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