Joint Stock Company (JSC) “N.A.Dollezhal Research and Development Institute of Power Engineering”

РОСАТОМ

ГОСУДАРСТВЕННАЯ КОРПОРАЦИЯ ПО АТОМНОЙ ЭНЕРГИИ «РОСАТОМ»

Lead-Cooled Fast-Neutron Reactor (BREST)

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Basis for the development and creation of the BREST-OD-300 reactor

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- The development and construction of the BREST-OD-300 reactor is included in the framework of tasks in the:
- – “Development Strategy of Nuclear Power in Russia in the First Half of the 21st Century” approved by the Russian Government in 2000;
- – the Federal Target Program “Nuclear Power Technologies of a New Generation for the Period of 2010-2015 and Up to the Year 2020” approved by the Russian Government in 2010;
- – the “Proryv” Project (2011) that integrates projects on the strategic solution of target tasks on the creation of natural-safety nuclear power technologies based on fast-neutron reactors and a closed nuclear fuel cycle (CNFC).

Natural safety

- The principle of natural safety means excluding the potentiality of severe accidents (TOP, LOF, fires and explosions) with a radioactivity and toxicity release leading to evacuation of the population.
- This principle is achieved, for the most part, due to feedbacks, natural dependencies, the neutron balance in fast reactors, and the inherent physical and chemical behavior and properties of fuel, coolant and other reactor components rather than thanks to building up expensive engineered barriers and complicated safety systems.
- Of essential importance for realization of the natural safety features are fast reactors design properties.

Goals and objectives of the reactor BREST-OD-300 creation

- ***Goal*** – practical confirmation of realization of the "Natural Safety" concept of the lead-cooled fast reactor, operating in NPP mode with closed NFC.

Objectives:

- **Elimination of severe accidents** (provoked by reactivity runaways, loss of cooling, by superposition of faults), which may require temporary or permanent evacuation of local inhabitants.
- **Complete fuel breeding** (equilibrium mode) for self-sustaining.
- **Elaboration of operational experience** of commercial NPPs of this type on all stages of their life cycle.
- **Minimization of operation costs** for of experimental demo-type unit under condition of maintenance fulfill the objective.

BREST-OD-300: Design concept

- Lack of excess reactivity enough for realization of severe reactivity accident
- Integral-type arrangement of the first circuit to avoid output of coolant outside the reactor vessel, to eliminate lost of coolant
- Using of low-activated coolant with high enough boiling temperature, without rough interaction with water and air in the case of depressurizing of the circuit
- Realization of full breeding of fuel within the core solely, transmutation of minor actinides
- Simplifying of the safety systems due to physical features of used materials and design solutions

Rationale behind the choice of lead coolant and nitride fuel (1)

- An analysis shows that the BREST-OD-300 of thermal power 700 MW, a pilot demonstration fast reactor is the most attractive choice in the range of innovative naturally safe reactor technologies under consideration worldwide.
- The use in integral design of the BREST reactor of a high-boiling (~2000 K), radiation-resistant, low-activated lead coolant, which is inert when contacting water and air, does not require high pressure in the primary circuit, and excludes the potentiality of accidents with a loss of coolant and heat removal, fires and explosions in a contact with the environment (water and air).
- The use of a dense (γ=14.3 g/cm^3), heat-conducting ($\lambda \approx 20$ W/(m·deg.)) nitride fuel, which is compatible with the lead coolant and the fuel cladding steel, permits operation with relatively low working temperatures of the fuel (T<1000°C), small thermal energy store, small release of fission gases from the fuel and their low pressure under the fuel cladding, which contributes to keeping it intact.
- In a combination, the properties of the heavy lead coolant and the highly heat-conducting, dense nitride fuel provide conditions for complete plutonium breeding to be achieved in the core (CBR≥1) even in case of a moderate reactor power, along with minor power effect, results in a small total reactivity margin and enables power operation with a small operating reactivity margin ($\Delta\rho<\beta_{eff}$) that rules out prompt-neutron reactor power excursions.

Values of reactivity effects in the BREST-OD-300 reactor with (U-Pu)N fuel

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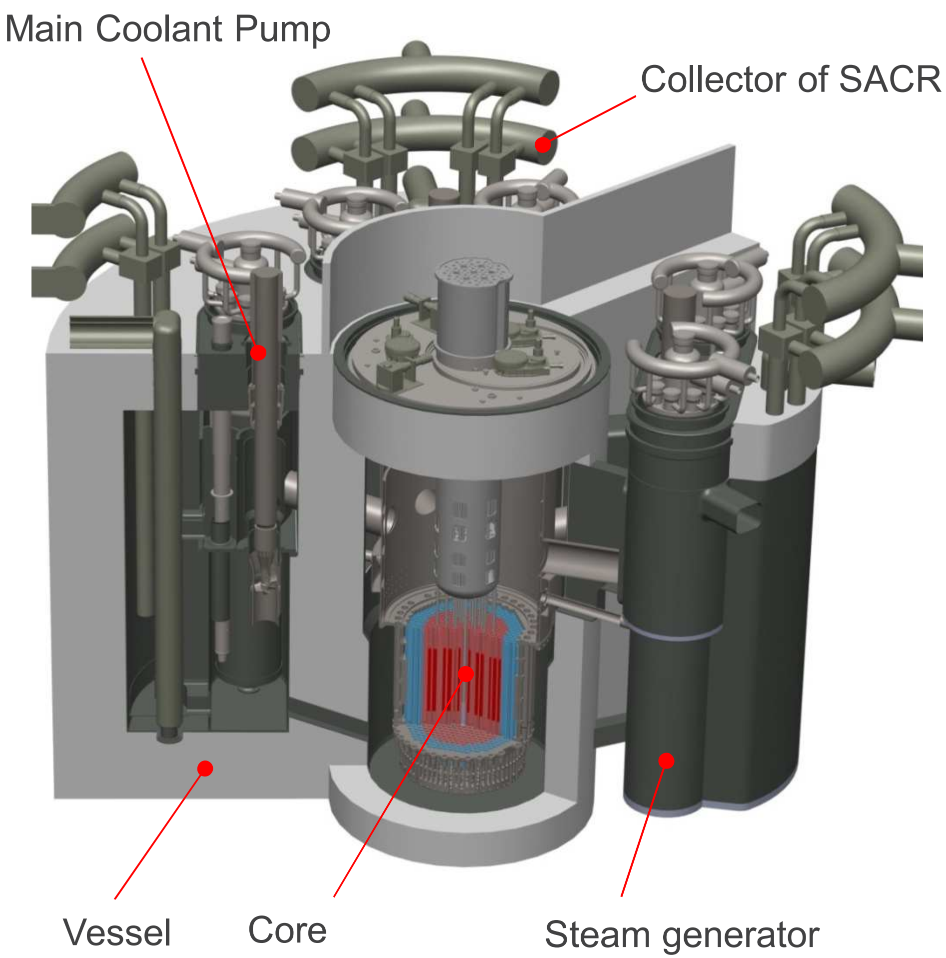
Reactivity effect	Value, $\Delta(1/K_{eff})$
Power effect when changing over from $N=0$ to $N=N_{nom}$	$-4.4\cdot10^{-3}$ / $(1.2\cdot\beta_{eff})$
Power effect when changing over from $N=0.3\cdot N_{nom}$ to $N=N_{nom}$	$-2.2\cdot10^{-3}$ / $(0.6\cdot\beta_{eff})$
Neptunium effect	$-0.7\cdot10^{-3}$ / $(0.2\cdot\beta_{eff})$
Full effect when ascending to $N=N_{nom}$	$-5.1\cdot10^{-3}$ / $(1.4\cdot\beta_{eff})$
Operating reactivity margin at N_{nom}	$1.5\cdot10^{-3}$ / $(0.4\cdot\beta_{eff})$
Total reactivity margin at T=380°C	$6.6\cdot10^{-3}$ / $(1.8\cdot\beta_{eff})$
Effective delayed neutron fraction, β_{eff}	$3.68\cdot10^{-3}$

Rationale behind the choice of lead coolant and nitride fuel (2)

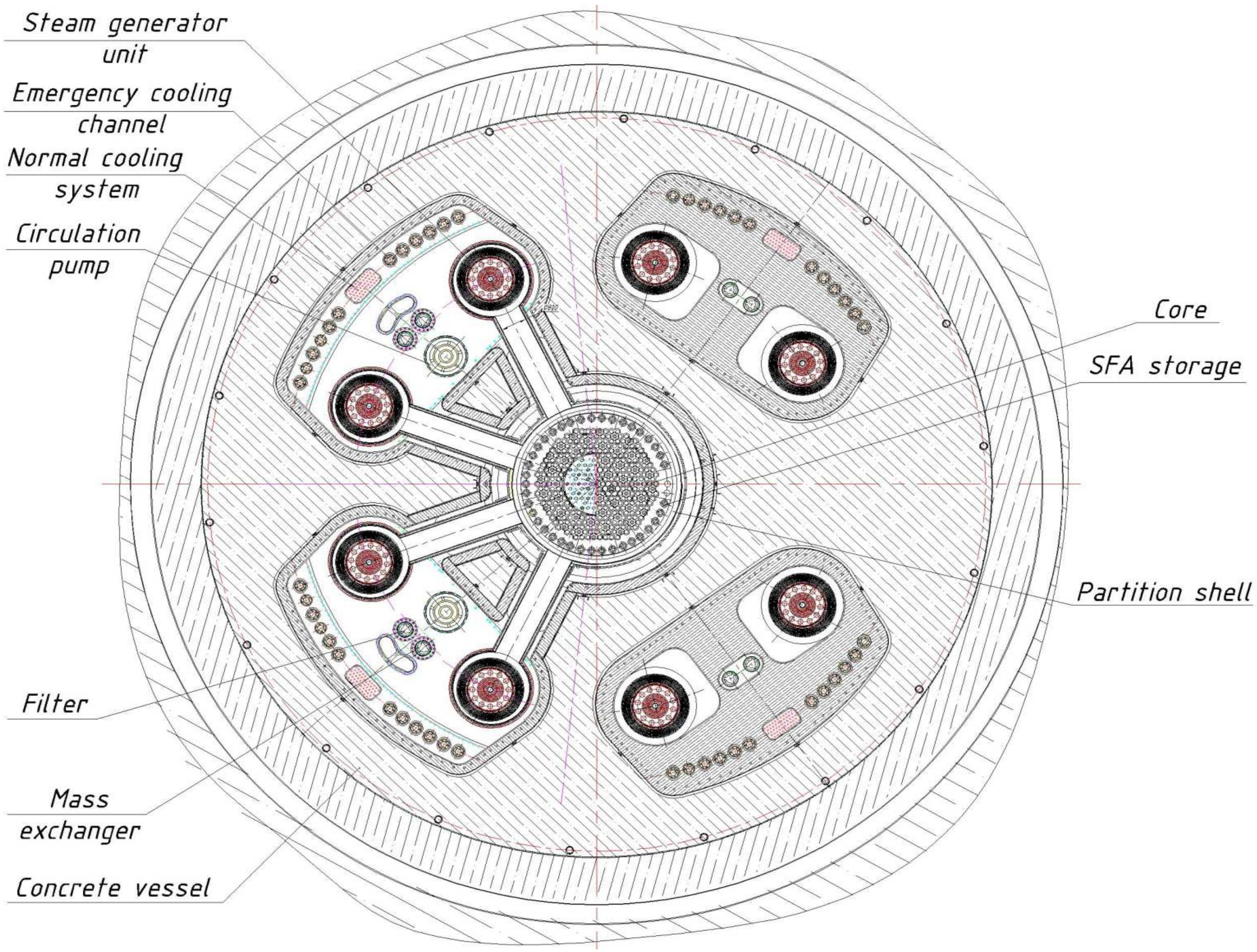
- A small neutron moderation by the lead makes it possible to expand the fuel element lattice, widen the coolant flow path and increase the amount of heat removed by natural lead circulation without worsening physical characteristics.
- High heat capacity of the lead circuit that accumulates the released heat ensures that emergency and transient processes occur smoothly without a major growth in the circuit temperature.
- The integral lead circuit with passive and time-unlimited direct residual heat removal from the circuit via natural air circulation with heat discharge in the atmosphere excludes accidents with fuel and coolant overheating during the reactor cooldown.
- Therefore, only thanks to the natural peculiarities of the chain reaction in fast reactors, the properties and qualities of lead and fuel (the BREST major components), as well as the designs that help implement these, two classes of the most severe accidents (caused by an uncontrolled power growth and a loss of heat removal) are excluded in natural manner. And it is in such approach to addressing the problem of the BREST safety in potentially severe accidents that the ***natural safety*** of this reactor consists in.

BREST-OD-300: key components and technical characteristics



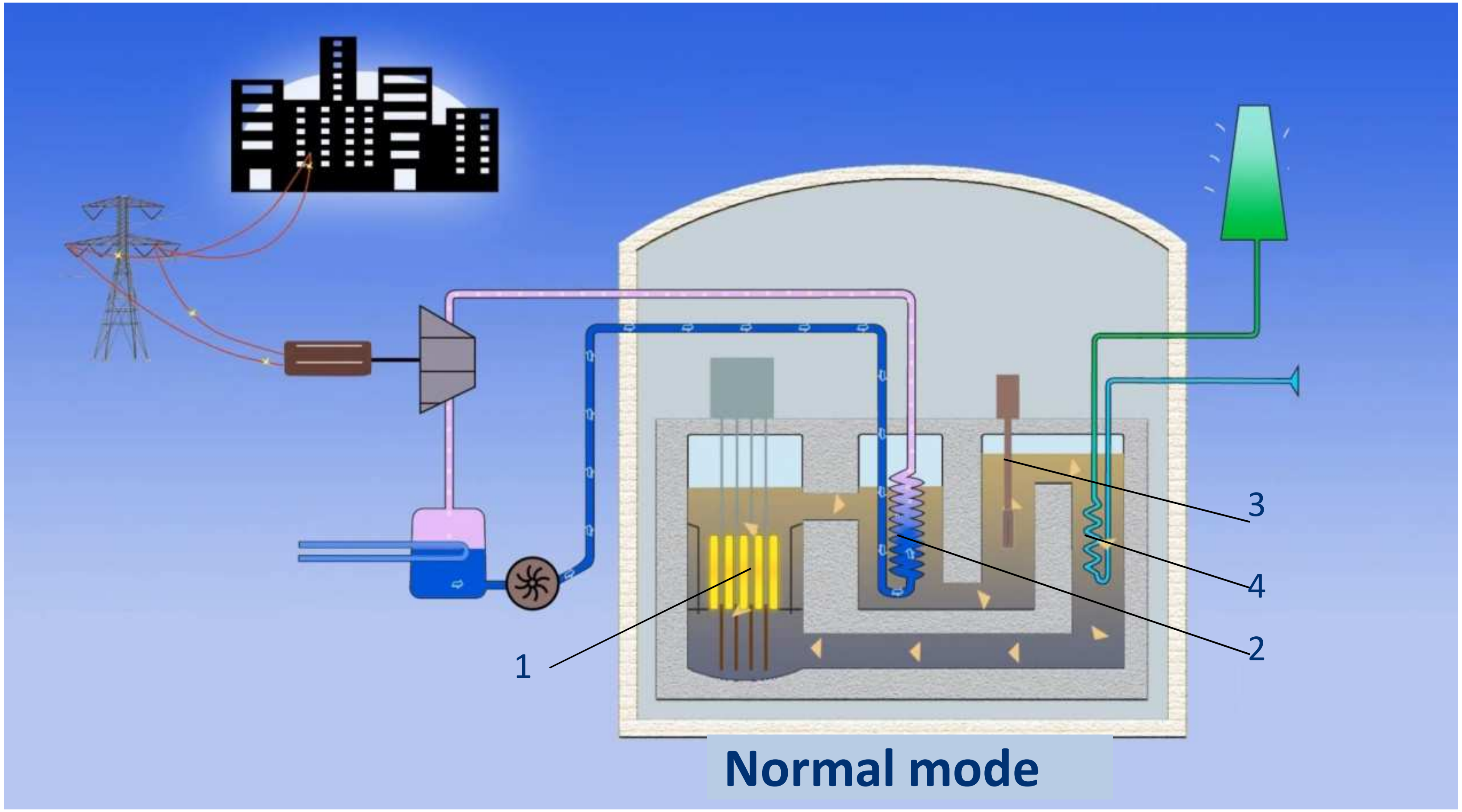
Thermal power, MW	700
Electric power, MW	300
Steam production rate, no less than, t/hour	1480
Coolant of the first contour	Lead
Gas pressure above the lead level: - exceed, MPa - maximal, MPa	 0,003-0,008 0,02
Average temperature of the lead coolant on the active zone entry/ exit, °C	420/540
Average temperature of the lead coolant on the steam generator entry/ exit, °C	340/505
Loop number	4
FA number in the active zone	169
Core height, mm	1100
Fuel load, t	20,6
Fuel campaign, years	5
Burn-up of unloaded fuel (maximum/ average), % HM.	9,0/5,5

BREST-OD-300



Steam generator unit
Emergency cooling channel
Normal cooling system
Circulation pump
Filter
Mass exchanger
Concrete vessel
Core
SFA storage
Partition shell

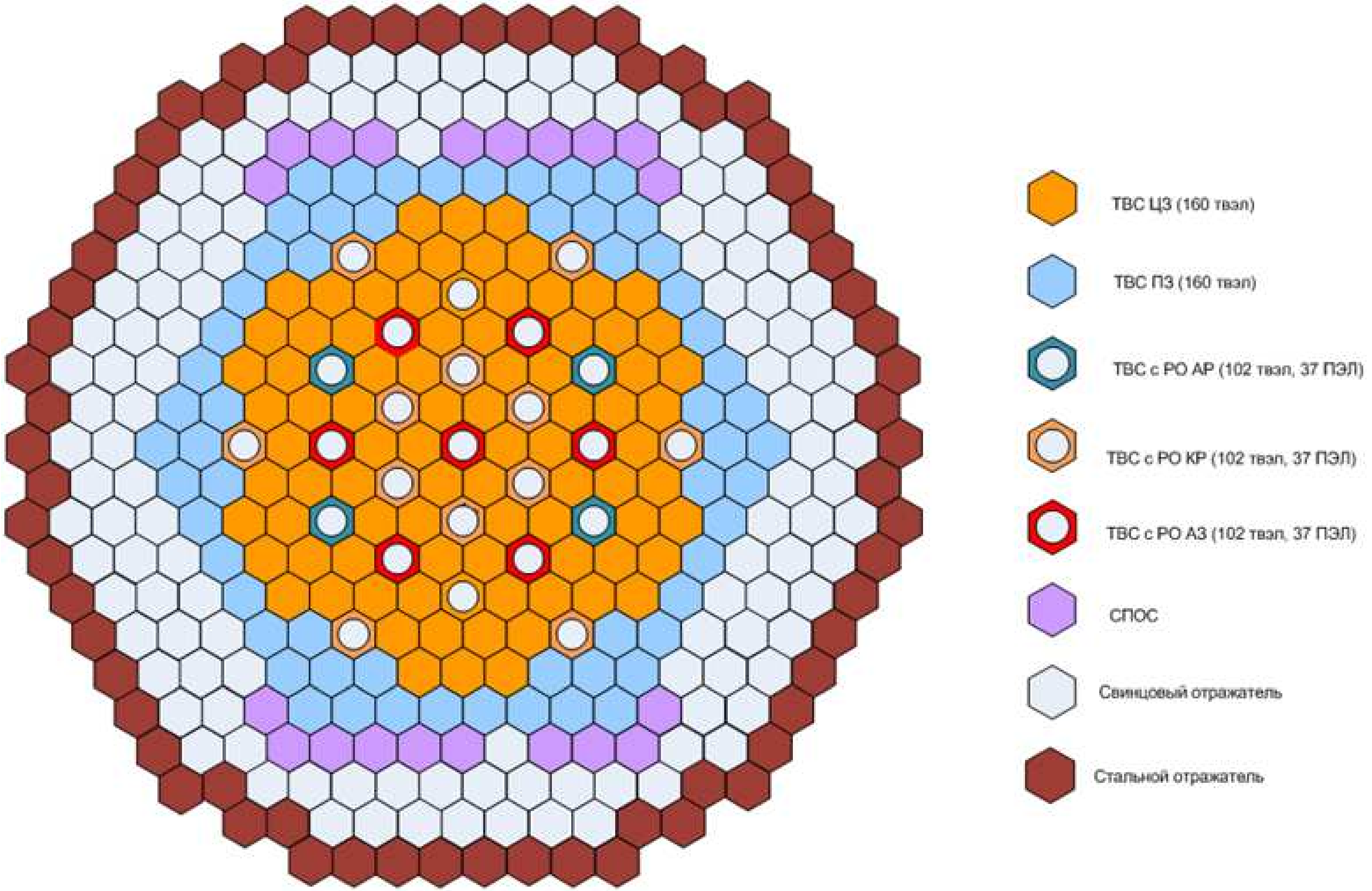
BREST-OD-300



BREST-OD-300 coolant circulation circuit

1 – core; 2 – steam generator; 3 – reactor coolant pump (RCP); 4 – ECS channel

BREST-OD-300: Core map



Assumed fuel element temperature limits

Based on experimental data and expert estimates, the following values have been assumed to be the temperature limits for the fuel element damage, given the overheating factors:

Max. cladding temperature: T=650°C – operating limit;

T=800°C – safe operation limit.

Max. fuel temperature: T=2800°C – safe operation limit.

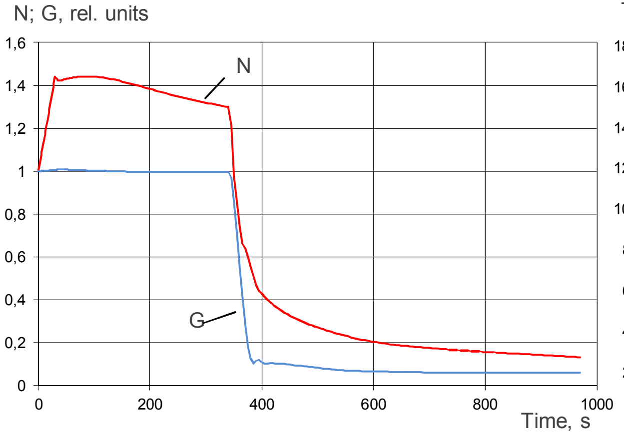
Two classes of the most severe accidents were considered:

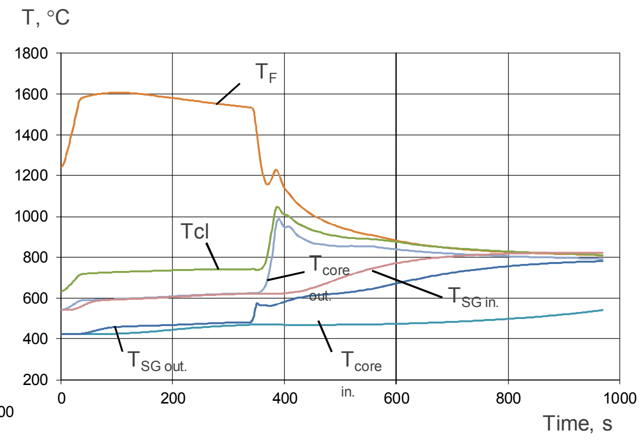
- an introduction of full reactivity margin (UTOP);
- a loss of forced heat removal (ULOF)

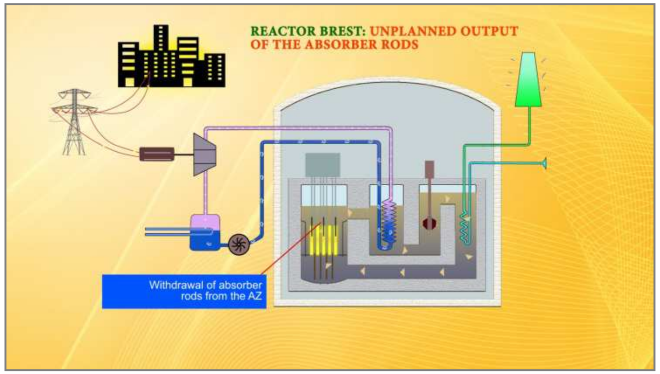
Power removed by the passive emergency core air cooling system – 5 MW

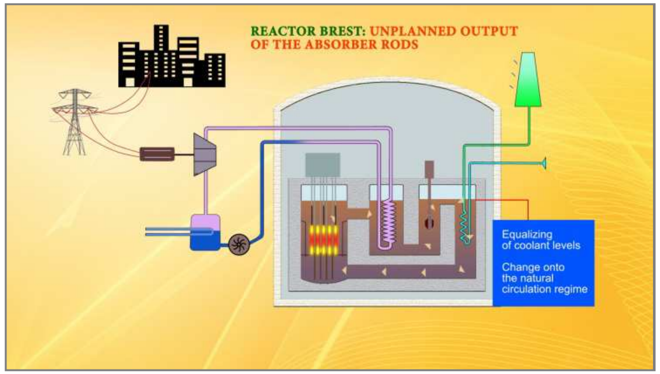
(with two of four heat removal loops in operation)

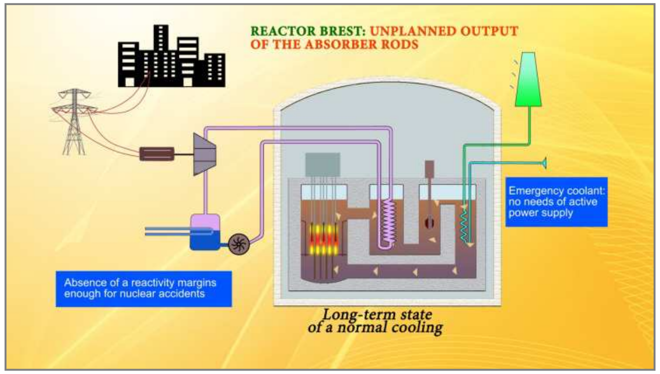
BREST-OD-300:
Analysis of the design-basis excess reactivity realization at rated power











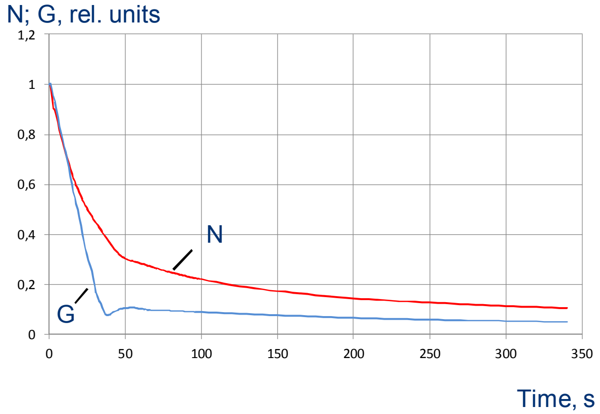
This mode cannot be realized when systems and components operate as designed.

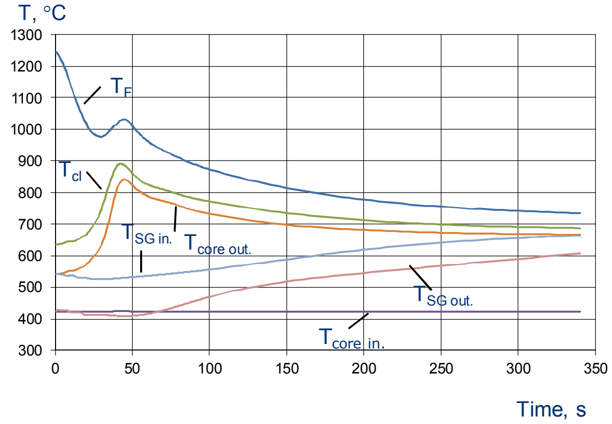
When programmed controlling safety systems fail, the reactor power reaches 1.45 N_{nom}. The cladding temperature of the highest heated fuel elements reaches ~ 1025 °C, and the SG inlet lead coolant temperature is 815 °C. Highly burnt up fuel elements may be partially damaged.

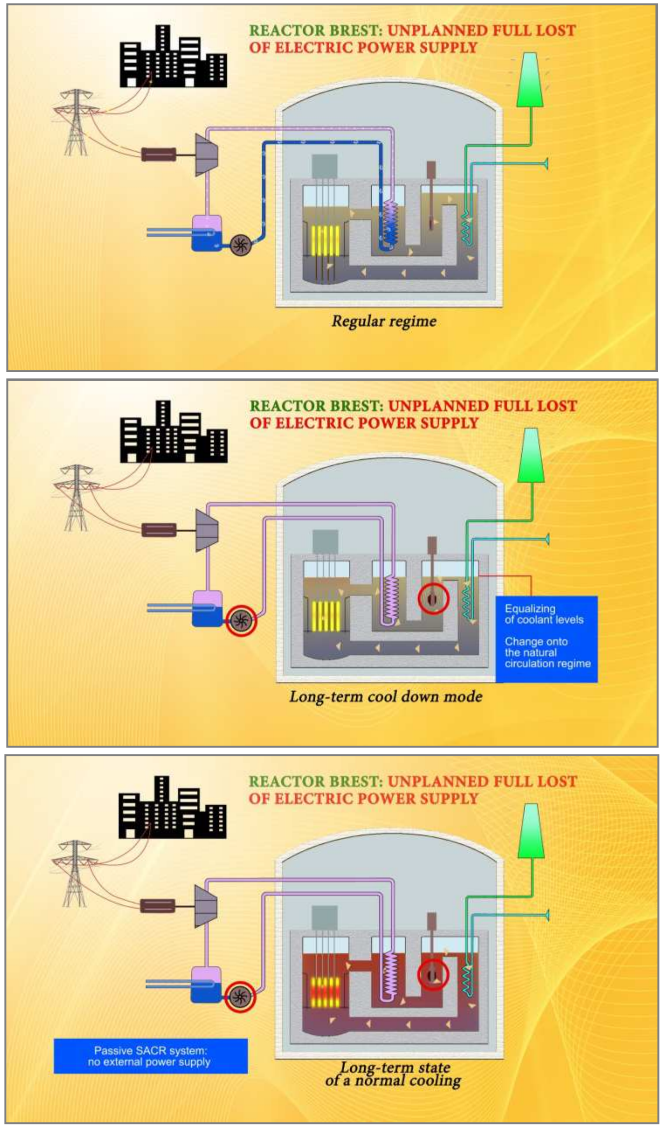
There is no fuel cladding or fuel melting. The circulation circuit remains integral.

BREST-OD-300:
Analysis of a unit blackout scenario

The worst case of a loss of normal heat removal from the reactor core is considered, namely blackout with four MCPs tripped and the feedwater supply lost during operation at the initial rated power. Failures of two reactor shutdown systems are postulated. Residual heat is removed by two out of four ECCS loops (a failure of the two other ECCS loops is postulated).







The maximum cladding temperature of the highest heated fuel element for ~ 45 s exceeds 800 °C and reaches ~ 890 °C.

<u>There is no fuel cladding or fuel melting</u>. <u>The circulation circuit remains integral</u>.

Inference

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- Thus, It has been shown that no even most severe accidents involving full reactivity margin introduction (UTOP) and full loss of power for all reactor facility systems (ULOF) lead to fuel melting and coolant boiling.
- Even in case of a loss of integrity of all fuel element claddings in the core, while keeping the lead circuit leak-tight and operating of the accident localization system as designed the accident may be classified as a medium incident of the level 2 or 3.
- So it is required no population sheltering, preventive iodine administration, evacuation or resettlement.

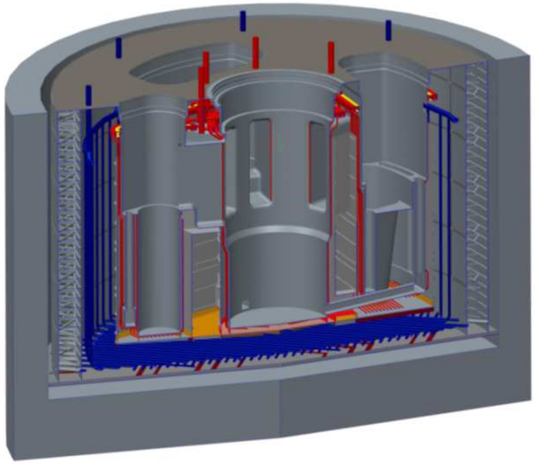
BREST-OD-300 reactor development

- The detailed design of the BREST-OD-300 reactor is being developed as part of the State Target Program and the PRORYV project, including calculations and experiments conducted to justify the design and technological approaches.

- The respective R&D activities are carried out in a broad cooperation of NIKIET with such nuclear organizations as: IPPE, ATOMPROEKT, VNIINM, ZIOMAR, TsKBM, TsNIITMASh, KBSM, TsNII KM Prometey, VNIITF and others.

Computational and experimental substantiation of reactor vessel

The vessel



The vessel bottom mockup

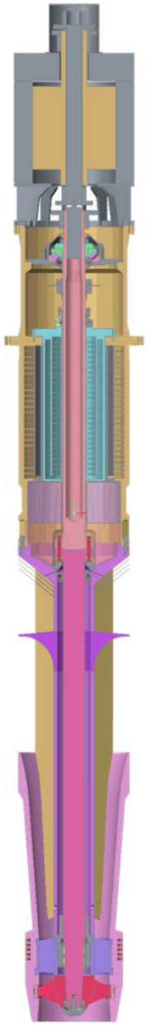
The concrete species

Results: Carried out tests on short-term mechanical properties of the concrete, developed methods of basic strength and thermal computations, mounted a mockup of the vessel bottom, determined recommendations on the drying modes

Coming results: Mechanical (including after irradiation) and thermophysical properties of the selected concrete compounds, determination of thermal conductivity coefficients in the concrete filler, experimental determination of temperature profile for verification of computational methods, development of mounting, filling and drying technologies for the reactor vessel

Computational and experimental substantiation of MCP

MCP



The middle-scale facility

The pump mockup

Results: Created the middle scale facility and a mockup of MCP (lead), created flowing part mockup for water tests, obtained water hydraulic characteristic, obtained lead hydraulic characteristic, determined the ways of refinement. Tests of the actual lower radial bearing are under preparation

Coming results: Determination of cavitation, energy, erosion characteristics of the improved flowing part mockup, correction of computational methods; substantiation of material and operational modes of the lower bearing; development, manufacture of full-scale mockup of MCP and putting on resource tests.

Computational and experimental substantiation of SG

Steam generator

Studied elements of the steam generator

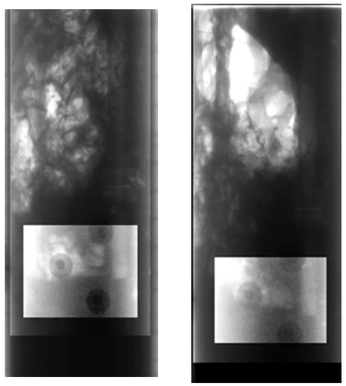




Results: Confirmed characteristics of lowering part of SG, shown the impossibility of transforming single SG tube depressurization into the multiple one, analyzed the dynamics of the destruction accidents, created facility for study of fretting-corrosion in SG, shown the durability of tubes made of a new steel in the conditions of lead and water-vapor environment.

Coming results: Set on production of SG parts, experimental specification of heat emission coefficients, preparation of a middle-scale mockup for study of vibration characteristics.

Justification of the impossibility of a single SG tube break growing into a multiple break



X-ray photographs of jet-efflux experiments

The purpose is to justify experimentally the BREST-OD-300 safety in the event of the steam generator's heat-exchange tubes losing integrity (steam escape into the lead).

The work has been performed to prove that a single BREST-OD-300 SG tube break cannot grow into a multiple break and to obtain experimental data for the software verification.

It has been experimentally demonstrated that a single SG tube break cannot grow into a multiple break.
The internal pressure for the SG tube failure at 700 ^{0}C is 185 MPa (or 53 MPa for tubes with a residual thickness of 1 mm).
The value of the SG tube breaking pressure, even where there is a weak point (a 2mm flat), exceeds 40 MPa at the coolant temperature of 950 °C.

For the SG tubes from EP302-M steel, the allowable temperature and breaking pressure values exceed the levels (including with regard for damageability) expected in the worst cases of severe accident accompanied by a power growth.

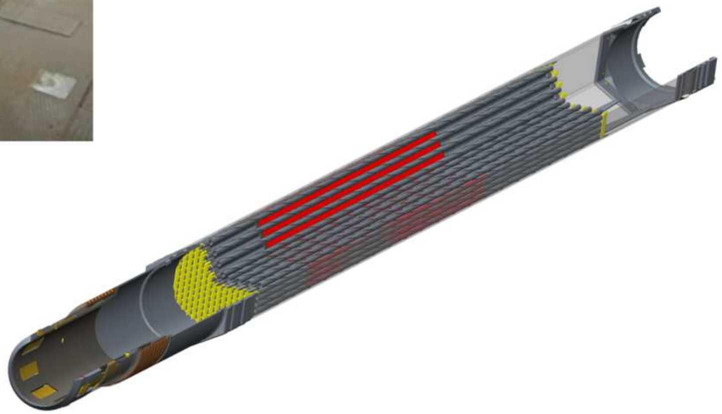
Core components

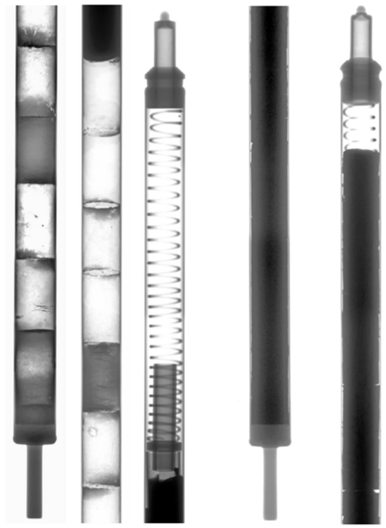
Results:

Developed the mockup drafts of the core components.

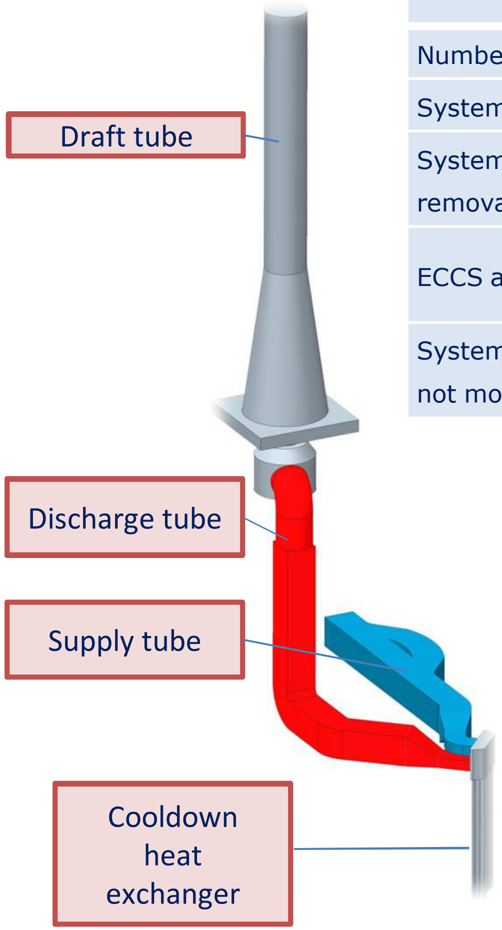
The mechanical properties of the fuel cladding have been obtained to determine the safe operating limits, and the initial 300-hour series of the erosion tests of plain fuel elements in grids has been conducted (no damage detected).

Mockups of absorber elements have been irradiated, post-irradiation tests conducted, and representative parameters of the absorbers have been determined (up to a burn-up of 12% in B-10).





BREST-OD-300:
Emergency core cooling system (ECCS)



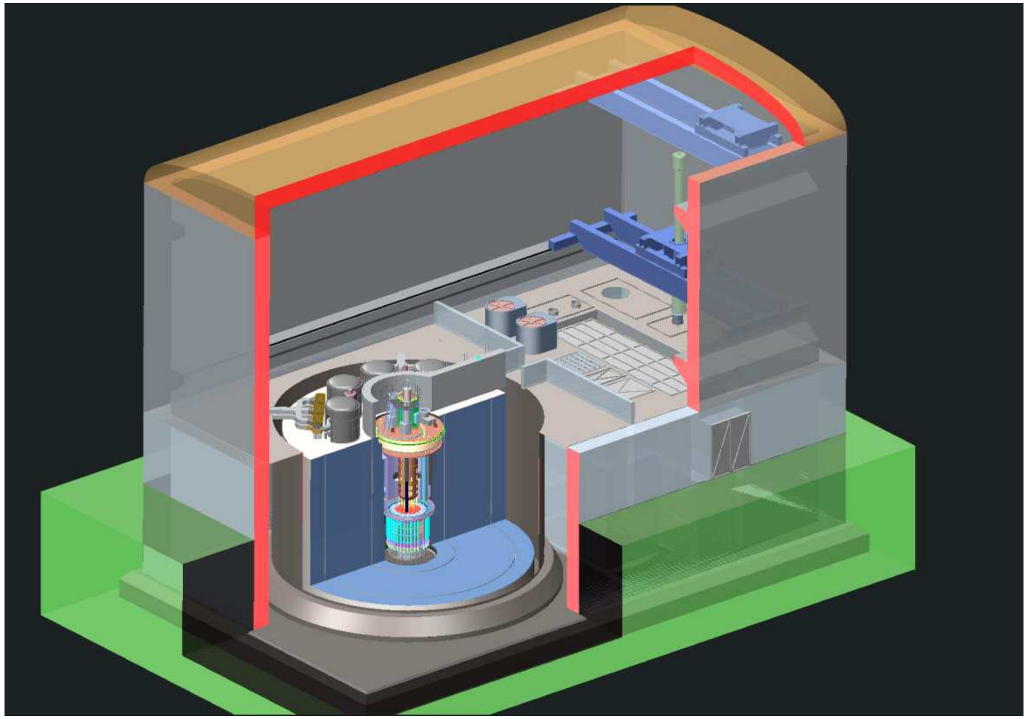
Description	Value
Number of branches (loops in the system)	4
System coolant (cooling)	Atmospheric air
System coolant circulation in emergency heat removal mode	Natural
ECCS air circuit inlet air temperature	min -55 ºC max 37.4 ºC
System's loop thermal power in waiting mode, kW, not more than	120

Heat tests of a short-length ECCS heat-exchange tube were performed at the test facility.

Data has been obtained for verifying the respective codes and optimizing the methods for the calculation of systems with Field-type air heat exchangers in nearly realistic operating conditions of a standard ECCS heat exchanger.

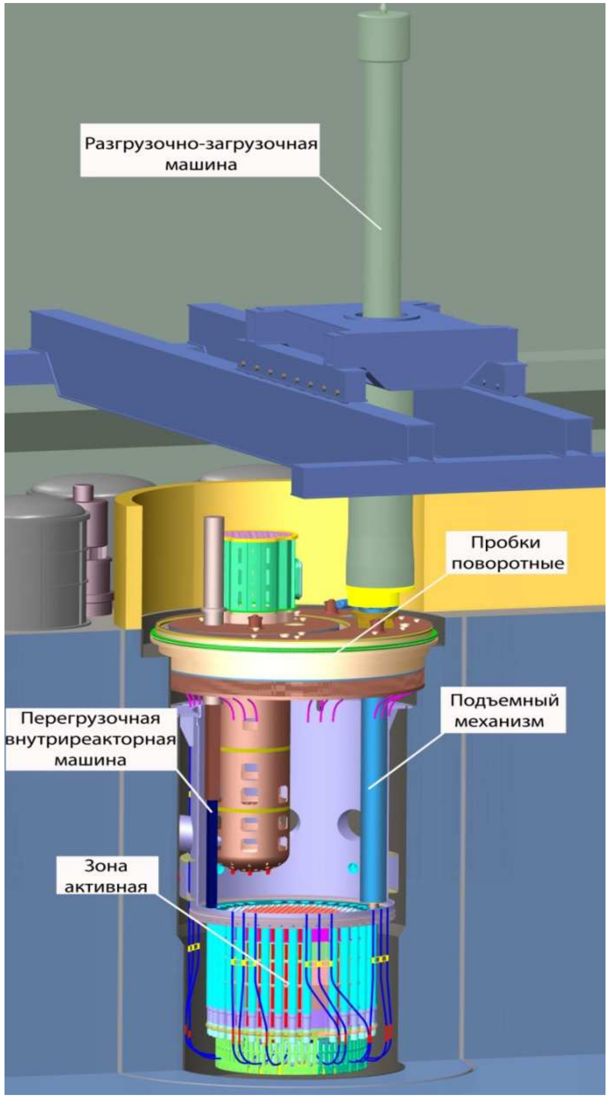
The experimentally found extracted power of 1 ECCS tube at a coolant temperature of 600 ºC is sufficient (~100 kW).

Computational and experimental substantiation of FA load and upload system



Results : Designed and manufactured mockups of the capture units, facilities for testing the captures, chains, screws, seals

Coming results: Durability and workability characteristics of the captures, chains, screws, seals in actual conditions



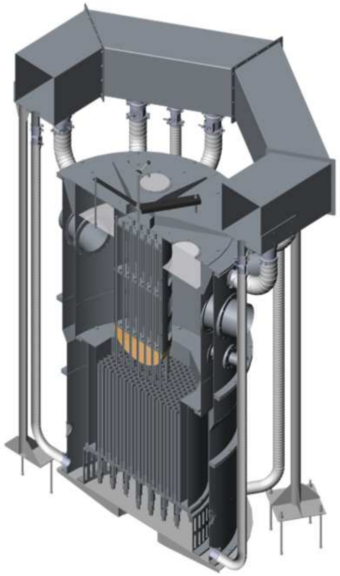
R&D works for substantiation lead coolant technology

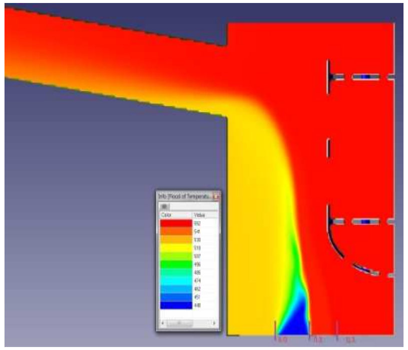
Results: Designed, manufactured and studied mockups of sensors for control of the coolant characteristics, mockups of equipment

Coming results: Set on production the reagents of the system of maintaining coolant quality, certified methods of the coolant quality control, developed quality standards, manufactured and tested experimental samples of the technology equipment

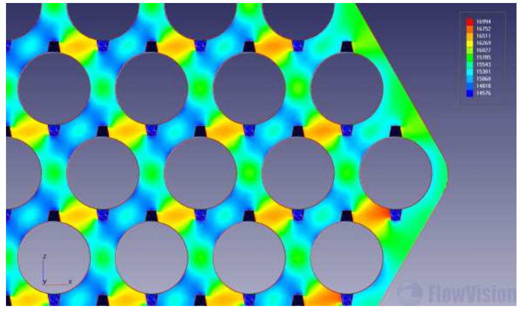
Experimental and design work for substantiation thermo hydraulic characteristics

Aerodynamic model and CFD calculation of the vessel cavities





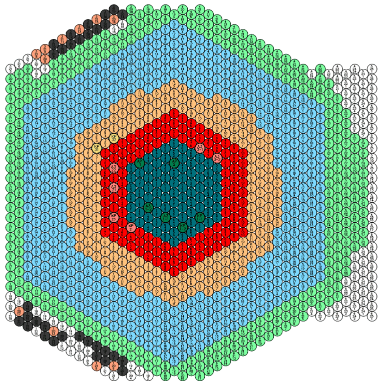
CFD calculation of FA



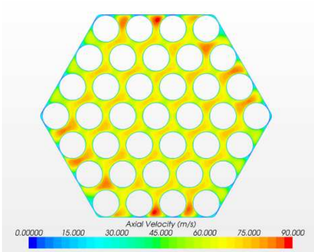
Results: Designed and manufactured aerodynamic and liquid metal facilities with reactor elements for examination the design decisions and verification cellular and CFD codes, accomplished experimental substantiation of mass transfer for self-spacing fuel rods and FAs with smooth fuel rods.

The coming results: Verification of computational cellular and CFD codes, obtaining verification data for circuit problems.

Computational Codes:
Justification of neutronic and thermal-hydraulic characteristics



BFS critical assembly with nitride



FA test mockup and CFD calculation

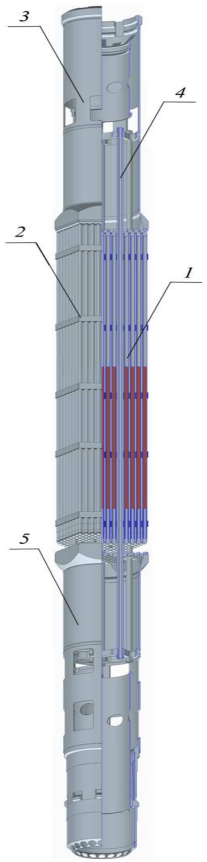
- MCU-BR and FACT-BR neutronic codes have been developed and verified as part of earlier experiments at the BFS test facility (assemblies 61, 64, 77, 85, and 95) and at the BN-350, BN-600 and JOYO reactors.

- A critical assembly with nitride (500 kg) has been created for the first time at the BFS-1 test facility, a benchmark model of the BREST-OD-300 reactor core with a central insert having the composition and the spectral characteristics close to those of the simulated reactor.
  The deviation of the calculated values of Keff from the experimental values is 0.19 % of Δk/k, and the root-mean-square deviation in the comparison of the radial and axial fission reaction rate distributions for ^{239}Pu, ^{238}U and ^{235}U is less than 4.0 %.

- The constants of the release of ^{210}Po, ^{124}Sb, ^{110m}Ag, ^{123m}Te, ^{131}I and ^{115}Cd from lead with entering the gas phase at 500 °C have been obtained at the facility for justifying the radiation characteristics. The data obtained will help make the radiation calculations using the TARUSA code less conservative by two orders.

- Data has been obtained at aerodynamic and liquid-metal test facilities for verifying subchannel and CFD codes (PUChOK-ZhMT, FLOWVISION). The errors of the drag and heat-transfer coefficient determination using the PUChOK-ZhMT code were 10 % and 16 % respectively. It has been shown that the maximum fuel cladding temperature does not exceed the design limit with the given error taken into account.

BREST-OD-300:
Shroudless FA design



Shroudless FA
1 – fuel element; 2 – spacer grid; 3 – head;
4 – support tube; 5 – tail

The reactor core FA has a shroudless hexagonal design with plain-rod fuel elements.

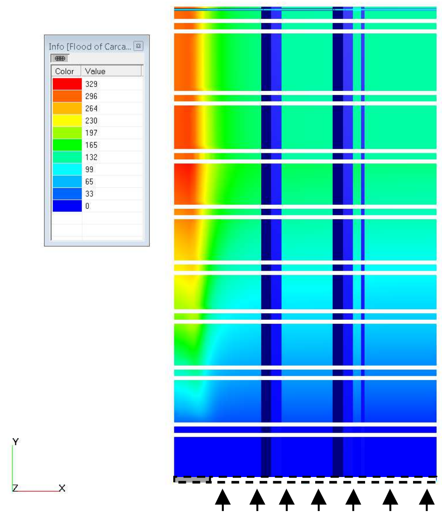
A higher safety level of the reactor core with shroudless FAs, as compared to the shrouded-FA design.

For a shroudless FA, heat is removed, when the inlet coolant flow is blocked, at the expense of the coolant inflow from the 'neighboring' FAs.

Calculations:

Postulated blocking of the coolant flow at the inlet of 7 shroudless FAs in the central core

Cladding temperature temperature does not exceed 800 °C.



Cladding temperature distribution inside the reactor core with the flow blocked in a part of the core

BREST-OD-300:
Automated control and protection system (ACPS)

ACPS simulator

To be checked for serviceability, the control and protection system (CPS) operation algorithms require to be implemented at the BREST-OD-300 ACPS simulator.

Automated control and protection system (ACPS) simulator has been built and integrated tests have been conducted on its basis to support the development of the CPS detailed design.

Tasks solved:

- Protection and automatic regulation algorithms developed and tested
- CPS regulators checked for stability of operation in conditions of different transients
- Video frames of the data display system for the main and backup control rooms developed
- Operator interface, controls and data displays refined

CONCLUSION

1. **Project BREST-OD-300 creates a base** for development of commercial reactor for NP on the basis of new nuclear technologies.

2. **Analysis of transient processes in BREST-OD-300** shows a possibility of exclusion of heavy accidents, demanding evacuation and displacement of inhabitants while using first physical properties of coolant, fuel, other reactor components, and also technical design, directed at it realization.

3. **Substantiation of adopted decisions** for design in 2014 is built on experimental substantiation on small- and middle-scale mockups and facilities, and also on computational substantiation with verified codes. In the further the substantiation will be held on large-scale mockups.

4. **The results of experimental and design works** points at possibility of realization in power complexes with BREST in closed fuel cycle basic demands on safety, volume of consumption of fuel raw materials, efficiency, solving problem of spent fuel.

РОСАТОМ

Thank you for your attention