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Open Joint-Stock Company “N.A. Dollezhal Research and Development Institute of Power
Engineering”

• Lead-Cooled Fast Reactor BREST – Project Status and Prospects

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International Workshop on Innovative Nuclear Reactors Cooled by Heavy Liquid metals:
Status and Perspectives

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Requirements to Be Met by Large-Scale Nuclear Power of the 21st Century

- Future large-scale nuclear power (NP) based on fast reactors (FR) in a closed nuclear fuel cycle (CFC), can arrest the growth of fossil fuel consumption, provide the bulk of electricity production increase, and
- “... resolve the problems of energy supply for sustainable human development, non-proliferation of nuclear weapons and environmental improvement of the planet...” – as stated by Russian President V.V. Putin at the UN Millennium Summit in 2000.



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Requirements to Be Met by NP of the 21st century

- Nuclear power will not be socially acceptable unless it gains high safety and security, interpreted broadly as:
 - - freedom from the constraints of fuel resources;
 - - impossibility of severe accidents with uncontrollable power growth, loss of cooling, fires and explosions, accompanied by radioactive and toxic releases at a level requiring public evacuation;
 - - technological support of the non-proliferation regime;
 - - environmentally safe closing of the fuel cycle and final waste disposal without upsetting the natural radiation equilibrium;
 - - ability to compete economically with alternative energy sources.



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Reactor BREST Concept

- All the above requirements are met by the innovative nuclear technology under development at NIKIET with contributions made by a number of institutes and research centres.
- This technology relies on the concept of BREST – a *naturally safe* FR with nitride fuel, lead coolant and a special onsite closed fuel cycle.
- The BREST concept emerged more than a quarter of a century ago with a promise to reconcile the conflicting requirements of safety and economic efficiency of NP.
- The appearance of this concept is associated with the name of Viktor Orlov.



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Elimination of fuel constraints for NP

- Owing to neutron excess, any FR can convert U-238 (> 99% in natural uranium) into fissionable Pu-239 with the breeding ratio of $BR \geq 1$.
- U-238 involvement in the FR fuel cycle allows increasing by a factor of more than 100 the efficiency of natural uranium use. With such efficiency, it becomes economically feasible to exploit lean but widespread uranium ore deposits.
- The above facts add immeasurably to the NP resources, even without considering the reserves of Th-232 which is more plentiful than uranium.



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Natural safety

- Accidents that are viewed as public hazards due to their toxic and radioactive releases should be ruled out primarily by the *natural safety* features of nuclear technologies in use.
- A reactor facility of a new generation will be naturally safe if it relies in its design on the physics, thermal and hydraulic characteristics of its various components and processes to eliminate the accidents leading eventually to public evacuation.



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Absolute safety is nonexistent

- Natural safety is not totally free from incidents caused by failure of systems and components or by human errors, but development of such events should be limited by defence in depth as well as by safety systems so that they present no threat to human life.
- Such accidents, which can lead to loss of the whole power unit in the worst case, are economic in nature, with their consequences limited to the cost of the nuclear unit.



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Natural safety of lead coolant and nitride fuel

- Use of high-boiling ($T_{\text{boil}} > 2000 \text{ K}$), radiation-resistant, low-activated lead coolant, which is also inert to water and air, offers freedom from high circuit pressure and rules out LOCAs, fires, steam and hydrogen explosions.
- Use of high-density ($\gamma = 14.3 \text{ g/cm}^3$), heat-conducting ($\lambda \approx 20 \text{ W/(m}\cdot\text{deg)}$) nitride fuel compatible with lead coolant and steel fuel claddings, affords relatively low average fuel temperature ($T \leq 1000^\circ\text{C}$), low heat storage in fuel, and small release of fission gas under the cladding.
- The combination of lead coolant with nitride fuel makes it possible to achieve full reproduction of fissile nuclides in the core ($\text{CBR} > 1$) and to keep the breeding properties within $1 \beta_{\text{eff}}$, which allows operation with a small reactivity margin and rules out accidents with uncontrollable power growth.



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Natural safety of lead coolant and nitride fuel

- Owing to the low neutron moderation by heavy lead, it is possible to expand the fuel lattice without affecting the reactor neutronics, to increase the coolant flow section in fuel assemblies, and to raise the level of power removed by natural lead circulation. The lead circuit itself giving off its heat continuously to naturally circulating air and thereby to the atmosphere, will never be overheated by residual heat.
- Thus, the neutron budget features in a chain reaction and the natural properties of the key BREST components – lead, fuel and structural materials, together with the engineering solutions to support them, can naturally exclude the two worst classes of severe accidents: with uncontrollable power growth and with loss of heat removal. It is such a natural solution of the safety problems posed by extremely severe accidents that lies at the heart of the reactor's ***natural safety***.



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Additional safety features

- Special passive devices providing coolant flow feedback to reactivity (and power). These are lead-filled channels surrounding the core, with their levels keeping track of inlet lead pressure to change neutron leakage.
- Emergency protection rods brought into action by a passive direct initiator in response to impermissible coolant temperature rise at the core outlet caused by inadvertent increase of reactor power or by coolant flow reduction.
- Heat storage capacity of the lead circuit which is high enough to overcome transients within safe operation limits.



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Grounds for BREST development and construction

- *Strategy for Development of Russian Nuclear Power in the First Part of the 21st Century*. Approved by RF Government in 2000.
- Federal Target Programme *Nuclear Energy Technologies of a New Generation for the Period of 2010-2015 and to 2020*. Approved by RF Government in 2010.
- Programme objective – “...to develop and build a pilot demonstration power unit with a lead-cooled fast reactor...”.



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BREST-OD-300 Project

- A pilot demonstration 700 MWt / 300 MWe fast lead-cooled reactor with U-Pu nitride fuel and a two-circuit heat removal system with subcritical water-steam as secondary fluid.
- BREST-OD-300 is considered as a prototype of a future commercial innovative reactor to be provided for naturally safe large-scale nuclear power, which will be able to deal with all the above problems.



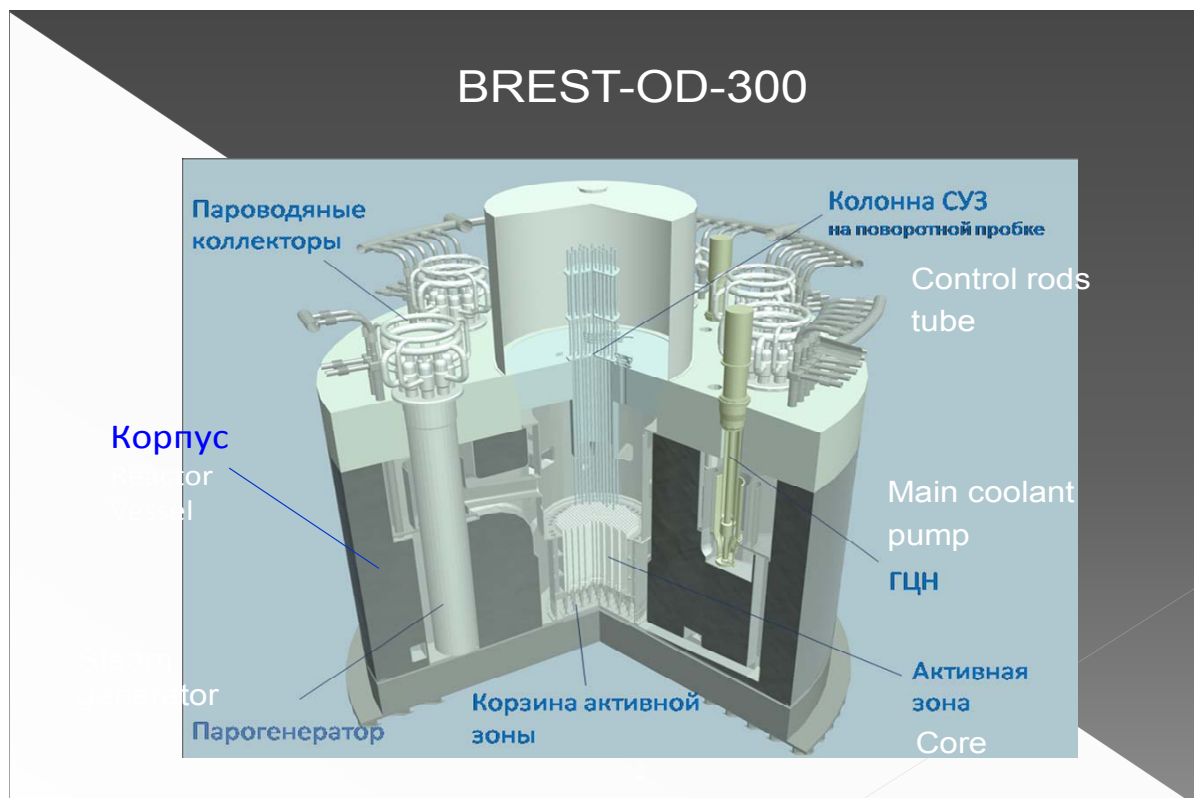
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BREST design

- This pool-type reactor has an integral configuration of the lead circuit components located in the 1 central and 4 peripheral steel-lined sections of the concrete vault.
- The central section accommodates the reactor core with the reflector, CPS rods, SFA storage, and a structure separating hot and cold lead flows.
- The 4 hydraulically connected peripheral sections (making 4 loops) contain SG-MCP units, emergency and normal heat exchangers, filters, and other auxiliary components.



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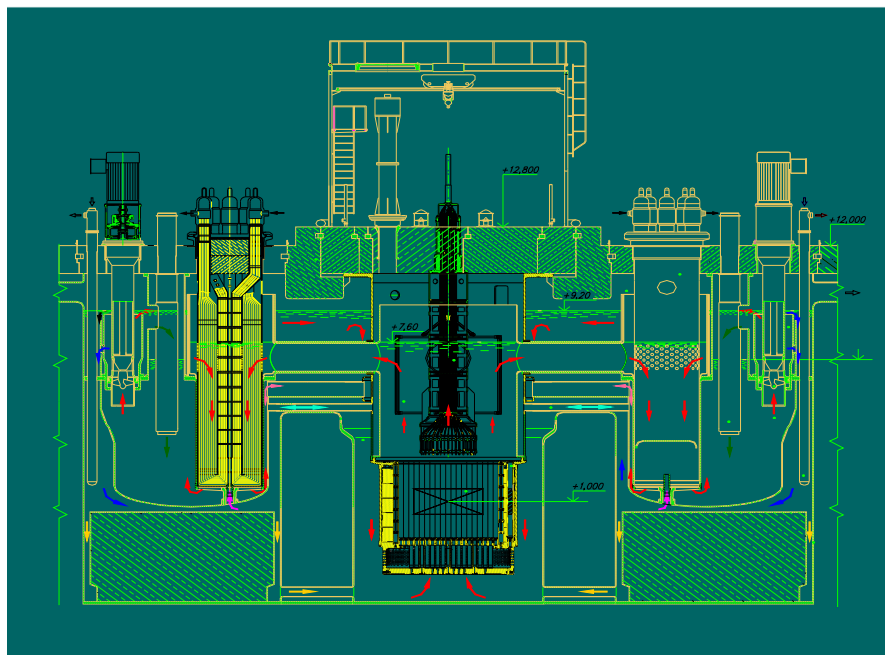
Lead circulation circuit

- The lead circulation system is designed to minimise the risk of steam or gas bubbles entering the core and resulting in a positive effect of reactivity and power growth.
- Lead circulation in the core and steam generators does not need pressure of the pumps but is set up by the difference they provide in the levels of “hot” and “cold” coolant.
- In its travel through the circuit, lead comes into contact with the free gas level three times, making it possible to separate the bubbles entrapped by the flow.
- This circulation arrangement reduces the non-uniformity of coolant flow if one or several pumps trip and provides for inertial flow in the core if all the pumps stop quickly.



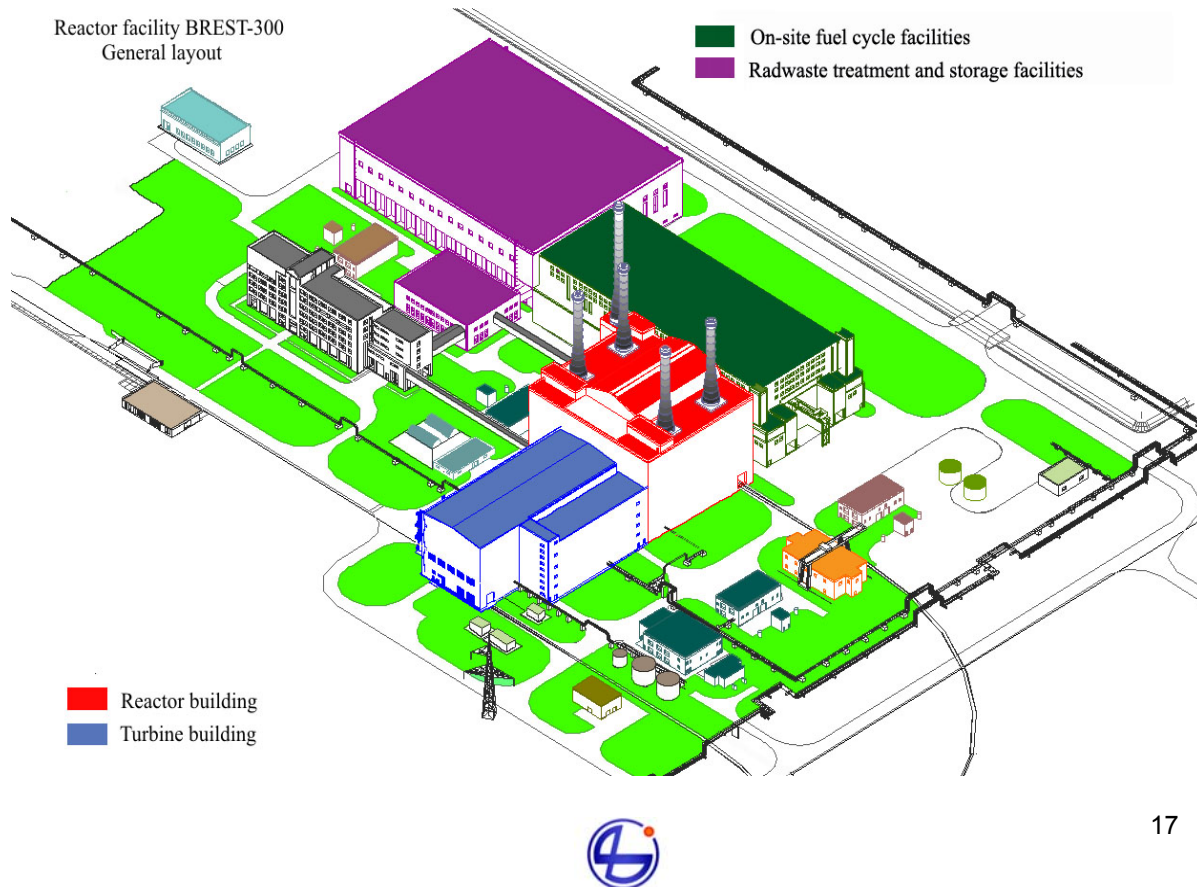
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Lead coolant circulation diagram



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Power production complex in plan



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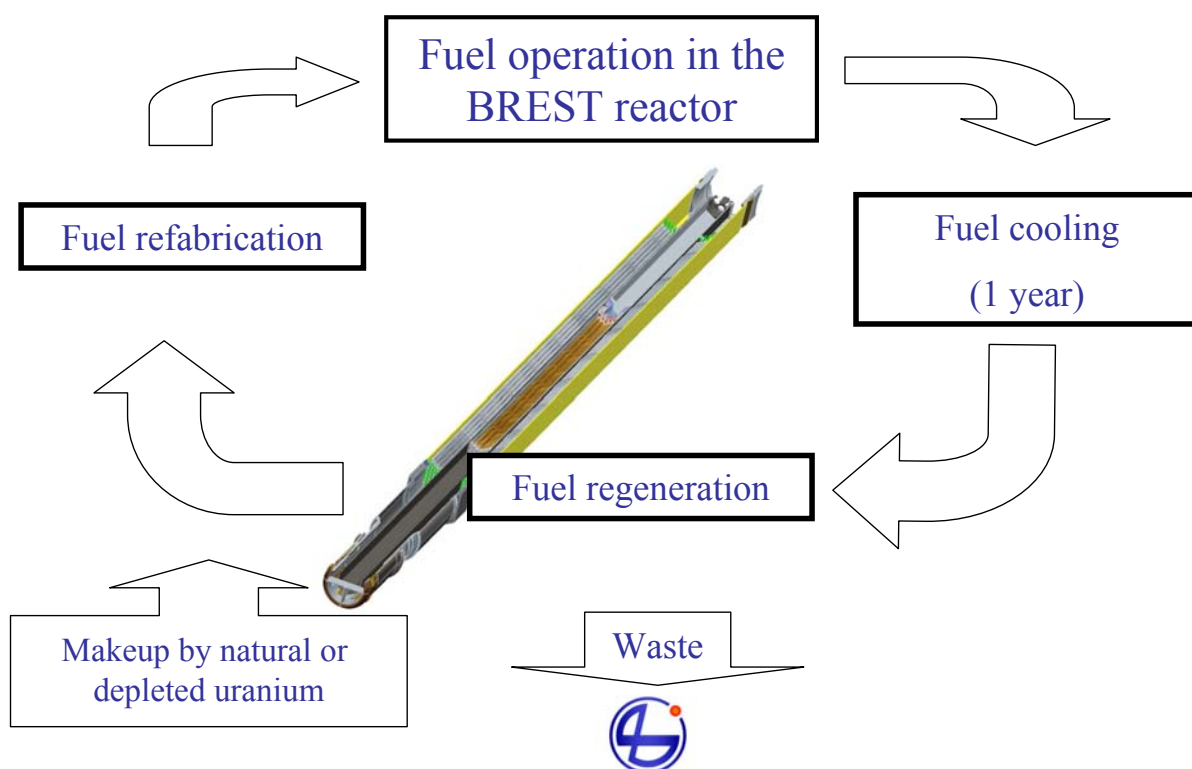
Closed fuel cycle of BREST-OD-300

- In-pile fuel irradiation (for 5-6 years) with 1/5 – 1/6 of irradiated assemblies replaced by regenerated fuel;
- In-pile cooling of irradiated fuel assemblies (IFA) for ~1 year;
- Irradiated fuel discharge from the reactor and transfer to the on-site reprocessing building;
- IFA cutting with fuel separated from steel components;
- (Non-aqueous) pyrochemical reprocessing (regeneration) with removal of fission products from fuel isotopes (U-Pu-MA);
- Fuel composition adjustment with FP replacement by depleted uranium;
- Fabrication of nitride pellets;
- Fabrication of fuel elements and assemblies;
- Interim fuel storage;
- Fuel delivery to the reactor and its loading.
- External (on-site) fuel cycle as short as ~1 year.
- All the processes involved in fuel regeneration and fabrication are designed to be remotely controlled and the associated equipment to be tended in hot cells.

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Fuel cycle flow chart



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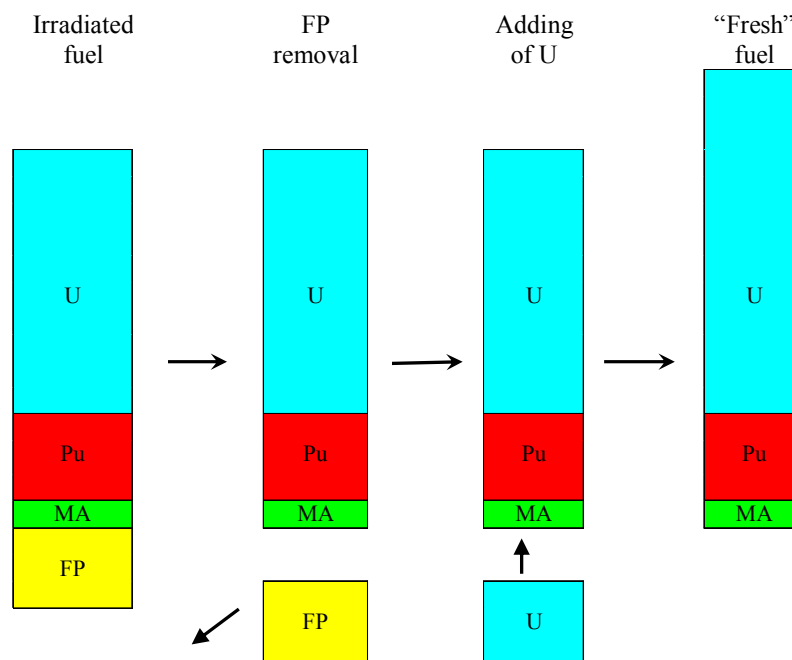
Provision of radiation equivalence

- Reprocessing of all irradiated fuel from thermal reactors with fractionation designed to ensure transfer of Pu, MA and long-lived FP to the fuel cycle of fast reactors.
- Recycling of all fissile materials (U, Pu, MA) with reproduction of Pu, burning of U and MA, and transmutation of FP (Tc, I).
- Thorough treatment of the residual RW to remove Pu, Am and some other long-lived nuclides (loss of actinides to RW no greater than 0.1-0.01%);
- Long-term monitored cooling of HLW in a special facility (~200 years) to reduce its biological hazard (by a factor of ~100) with subsequent RW inclusion in stable mineral-like compositions, and its final disposal without upsetting the natural radiation balance of the Earth (calculated with allowance for migration).



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Basic diagram of BREST fuel regeneration



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Technological support of non-proliferation

- BREST demonstrates its performance under equilibrium fuel conditions with $CBR \sim 1.05$, small reactivity margin ($\Delta\rho \sim \beta_{eff}$), self-sufficiency for Pu, and steady power distribution;
- U blankets are excluded (replaced by Pb) to rule out production of weapons-grade Pu.
- Pu separation is ruled out by using special fuel regeneration and refabrication processes which amount basically to coarse removal of FP, addition of depleted U to the cleaned (U-Pu-MA) mixture, nitration, and production of new fuel.
- Fuel theft is barred by presence of MA and some FP in regenerated fuel.
- On-site fuel reprocessing obviates the need for its off-site transportation.
- The U enrichment process is phased out.
- *In the closed fuel cycle under consideration, U-238 burns, while Pu-239 in fuel acts as a catalyst.*



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Competitive ability

- A plant with a BREST-type reactor is expected to be economically competitive owing to the simpler design of the facility and its safety systems, as well as to efficient utilisation of nuclear fuel and generated heat.
- Low lead pressure in the circuit allows using an integral configuration of the circuit components in a concrete pool, which was tentatively shown to reduce the construction costs.
- On-site fuel cycle arrangement is also likely to be economically beneficial owing to the shorter out-of-pile cooling and transportation time, which will eventually lead to a reduction in the recycled fuel quantity – one of the greater contributors to the fuel cycle costs.
- BREST-OD-300 being a prototype of the prospective commercial BREST-1200 plant, both facilities are quite similar in their design and performance as shown by the following table.



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Characteristics of BREST–OD-300 and BREST–1200

Characteristic	BREST–OD-300	BREST–1200
Thermal power, MW	700	2800
Electric power, MW	300	1200
Fuel assembly design	Shrouded	Shroud-free
Number of fuel assemblies	199	569
Core diameter, mm	2650	4710
Core height, mm	1100	1100
Fuel rod diameter, mm	9.7; 10.5	9.1; 9.7
Fuel rod pitch, mm	13.0	13.0
Core fuel	(U+Pu+MA)N	(U+Pu+MA)N
Fuel inventory, (U+Pu+MA)N, t h.m.	24.1	58.7
Mass of (Pu)/ (Pu ²³⁹ +Pu ²⁴¹), t	3.1/2.2	7.8/5.,6
Fuel lifetime, eff.days	1800	1800
Cycle-averaged CBR	~1.05	~1.05
Average/max. fuel burnup, MW·d./kg	53/84	71/124
Inlet/outlet lead temperature, °C	420/540	420/540
Maximum fuel cladding temperature, °C	650	650
Water-Vapor temperature at SG inlet/outlet, °C	340/505	340/520
Pressure at SG outlet, MPa	18	24.5
Design life, years	30	60



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Lines of BREST - OD -300 development

The process of BREST engineering design includes:

- Development effort to validate the design of BREST core components;
- Development effort to validate the design of BREST facility components;
- R&D to assess the lead coolant technology;
- Development effort to prove the performance of structural materials;
- R&D to demonstrate the neutronics and safety of BREST;
- R&D to demonstrate the thermal hydraulics of BREST;
- R&D to validate the design solutions for the BREST power unit



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Project of BREST-OD-300 development stages

- 2009 – technical proposal;
- 2010 – outline design (with correction of the TA for the reactor facility);
- 2011-2013 – engineering design with validation of engineering solutions:
 - 2011 – inputs to engineering design, design management system (computer archive, 3-D design, etc.);
 - 2012 – test sections and rigs for trying out full-scale specimens, provision of makeups);
- 2014 – review and improvement of engineering design;
- 2017 – service life characteristics of reactor facility components;
- 2011-2017 – R&D for validation of the plant and fuel cycle design;
- 2011-2019 – development of fuel management technology;
- 2011-2019 – provision of a fuel (FA) fabrication line; development of the fuel regeneration process

•Final objective

- 2015 – detailed design for NPP construction; construction license;
- 2016-2020 – NPP construction and commissioning.

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Conclusion

- The BREST-OD-300 development efforts carried out to pave the way for commercial reactors of this type and thereby to lay a foundation for large-scale nuclear power, show that such reactors can provide:
- – radical improvement of safety with elimination of the most hazardous radiation accidents by combining the inherent properties of a fast reactor and its components with features of ***natural*** safety;
- – unlimited fuel resources and independence from U mining.
- – reduction of the proliferation risk;
- – conclusive solution of the radwaste problem;
- – ability to compete with other types of power generation.



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THANK YOU FOR YOUR
ATTENTION!



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