# Closed fuel cycle technologies based on fast reactors as the corner stone for sustainable development of nuclear power

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**Abstract.** This article analyzes problems and approaches to modern nuclear power development using closed nuclear fuel cycle and fast reactors. It describes specified technical requirements for nuclear power systems in large-scale nuclear power industry. Targets and scientific problems solved by Rosatom's "PRORYV" Project which is a part of the Federal State Program "Nuclear Power Technologies of New Generation in the Period of 2010-2015 and up to 2020" are examined.

**Key words:** nuclear reactor, closed nuclear fuel cycle, nitride fuel, radioactive waste.

## 1 Introduction

The concept of nuclear power development in Russia that was clearly stated in the year 2000 document [1] and further developed in papers [2-3] suggests the development and introduction of fast reactors with inherent safety and closed nuclear fuel cycle as a priority target. After the period of conceptual R&D at the turn of 2012 the PRORYV Project [4, 5] started first practical steps to the implementation of the concept and creation of the new technological base for large-scale use of nuclear power technologies.

Analysis of the results achieved within the frameworks of this project during 4 years since FR13 conference [5] is presented in this paper.

## 2 Goals and tasks of the project

PRORYV Project suggests the development of fast reactors with inherent safety and on-site closed nuclear fuel cycle that should meet the following requirements:

- elimination of accidents requiring population evacuation and resettling;
- maximum possible use of energy potential of uranium resources;
- gradual approach to radiation equivalent (compared to natural raw materials) RAW disposal;
- technological support to nonproliferation;
- competitiveness of nuclear power compared to other means of energy generation.

Experimental testing and demonstration of new technological solutions are planned to be carried out at the pilot demonstrator energy complex (PDEC) in Tomsk. It consists of pilot demonstrational power unit with BREST-OD-300 lead-cooled reactor operating with mixed nitride fuel, fuel (re)fabrication and fuel recycling facilities.

The project of power unit with BN-1200 reactor should prove that new reactors can be competitive to the best NPPs on thermal neutrons. The project uses as high as possibly all knowledge gained during the development and operation of BN-600 and BN-800 reactors. At the same time a commercial project of BN-1200 should be highly innovative and apply new technical solutions that will ensure fulfillment of abovementioned requirements of inherent safety, excluding those that are naturally tied with the use of sodium coolant.

## 3 Pilot demonstrator energy complex project (PDEC)

PDEC should be the first in the world to demonstrate technical-economical parameters of the whole set of facilities of the closed nuclear fuel cycle (CNFC) at one site.

The on-site CNFC allows perfecting “short fuel cycle” technology with minimum spent fuel (SF) cooling before reprocessing. Table I shows the main PDEC parameters, and Fig. 1 – a general scheme of phased commissioning.

It should be noted that all the developed technologies (excluding logistics) and components can be used also for centralized arrangement of a fuel cycle provided the optimum technical and economical effectiveness is attained with acceptable logistics. Module organization of production facilities seems to be the most logical one.

TABLE I: Main PDEC parameters.

| Rated output power, gross | 300 MW |
|---|---|
| Fuel type | Mixed U-Pu nitride (MNIT) |
| Design lifetime, year | 30 |
| Design lifetime of fuel cycle equipment, year | 30 |
| Fuel fabrication/refabrication capacity, t/year | 14.75 |
| Spent fuel reprocessing capacity, t/year | 5 |

### 3.1 BREST-OD-300 reactor design and power unit on its basis

In 2016 the basic design of the innovative BREST-OD-300 lead-cooled reactor facility with mixed nitride fuel was finalized. Its main characteristics are shown in Table II and the overall outlook in Fig. 2. Main parameters of the power unit (its design is also completed) are shown in Table III and its view is presented in Fig. 3.

TABLE II: BREST-OD-300 main parameters.

| | |
|---|---|
| Thermal power, MW | 700 |
| Number of FA in the core | 169 |
| Fuel | MNIT |
| Fuel load, t | 20.6 |
| Breeding ratio (BR) | 1.05 |
| Number of loops | 4 |
| Primary coolant | Lead |
| Maximum coolant pressure in the primary circuit, MPa | 1.17 |
| Coolant temperature at the core inlet/outlet, °C | 420 / 535 |
| Average temperature of SG working medium, °C | 340 / 505 |
| SG outlet pressure, MPa | 17 |
| Steam output, t/h | 1500 |

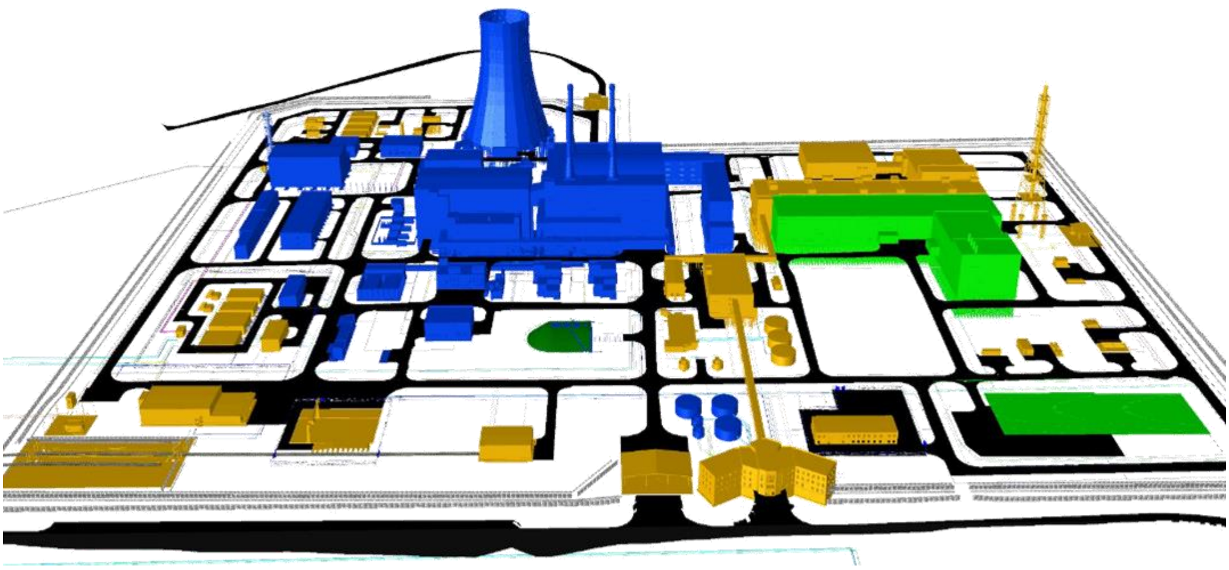


*Fig. 1. PDEC general scheme of phased commissioning: yellow – fabrication/refabrication facility – first stage, blue – BREST-OD-300 reactor unit – second stage, green – SF recycling and RAW handling facility.*

The use of non-boiling and not interacting with water and air lead coolant allowed to implement double-circuit reactor layout that significantly differs from more traditional three-circuit design of Na cooled reactors and therefore improves technical-economical parameters of liquid metal-cooled reactor. Special reports at the Conference are dedicated to the state of development of this power unit and related equipment.

Power unit design has passed the State Expert Assessment and is filed to Rostechnadzor to be licensed for construction.

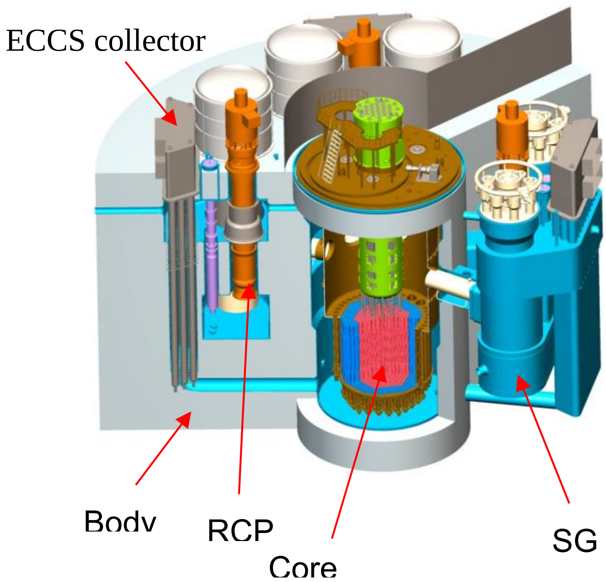



*Fig. 2. Integral design scheme of BREST-OD-300 reactor unit.*

TABLE III: Main parameters of BREST-OD-300 power unit.

| Parameter | Value |
|---|---|
| Reactor thermal power, MW | 700 |
| Turbogenerator electrical power, MW | 300 |
| Efficiency (gross), % | 43 |
| Capacity factor | 0.8 |
| Reactor service life, year | 30 |
| Design-basis seismic, point | 7 |

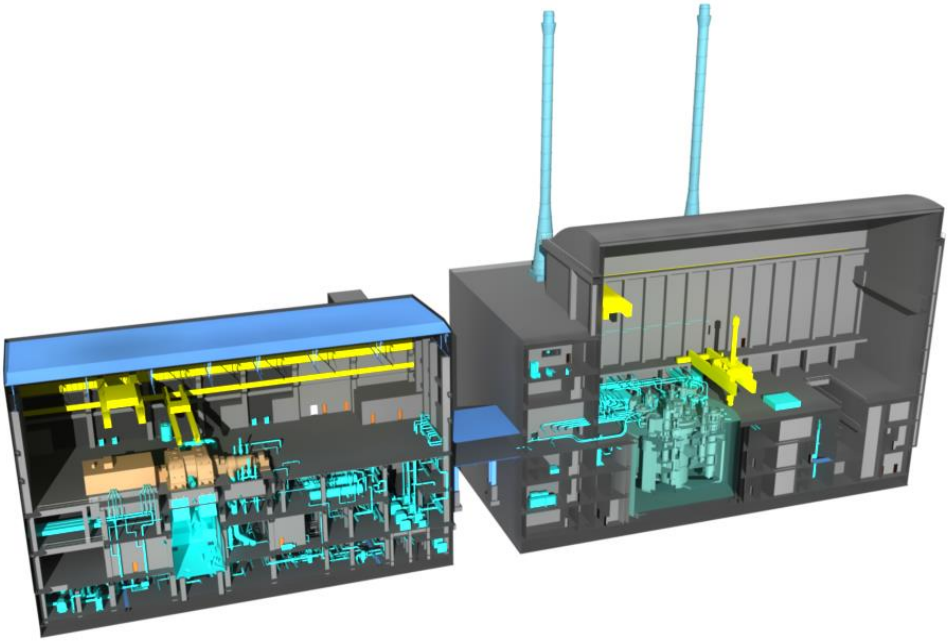


*Fig. 3. Power unit view (left – central hall, right – turbine hall).*

## 3.2 On-site fuel cycle design

On-site fuel cycle consists of two main facilities – Fuel Fabrication Facility (FFF) and Fuel Recycling Facility (FRF), that includes RAW handling system. At the first facility a pilot demonstrational production of mixed nitride fuel with power Pu and depleted U is created for the first time in the world with the use of carbothermal preparation technology.

The single facility can work with both raw materials and BREST-OD-300 SF recycling products. It also suggests including minor actinides into fuel for further transmutation. FFF is under construction (Fig. 4) as the first PDEC construction stage with commissioning in 2020.

It’s worth noting that an alternative technology of direct hydration is at R&D stage within the frameworks of PRORYV Project as well.

*Fig. 4. General view of PDEC construction site.*

Gradual implementation of this technology is considered: the first stage – hydrometallurgical treatment with further introduction of the combined option that includes pyro-chemical processing at the initial stage and further purification by hydro-metallurgical techniques. The option of switching to single pyro-chemical technology is also considered in case R&D results will prove the required purification goals can be met.

### 3.3 Major tasks for R&D at PDEC

Construction of PDEC will allow demonstrating the capacity of technologies based on FR with inherent safety in CNFC and conducting a series of R&D cycle that will open the way to their implementation in an industrial scale, including:

- Demonstration of BREST-OD-300 operation with the minimum reactivity margin that eliminates possibility of reactivity-induced accidents and optimization of BREST-OD-300 reaching of the equilibrium mode;
- Verification and demonstration of full fuel breeding in the reactor core (CBR~1) and utilization of full power potential of natural U without the use of blankets;
- Mastering of the lead coolant technology in the real conditions of large NPP;
- Testing of innovative technical solutions for reactor unit, FFF, FRF and acquiring of component endurance characteristics;
- Validation and demonstration of CNFC with MNIT;
- Proof of effectiveness of MA transmutation in FR and determination of RAW characteristics in order to ensure U raw material radiation-equivalent disposal.

## 4 R&D main results and inherent safety implementation in the Project

The results of accomplished R&D show a qualitatively new level of safety of being developed technologies (in the first turn, the reactor ones) that was specified at the initial stage by the conceptual requirements of inherent safety. The key requirement is deterministic elimination of the possibility of severe accidents that lead to population evacuation and furthermore resettling.

## 4.1 Integral design of the primary circuit and elimination of accidents with loss of heat removal from the reactor core

Unlike water- or gas-cooled reactors the fast reactors with liquid metal coolant are capable of deterministically eliminating the possibility of accidents with loss of coolant and/or heat removal from the core. The key part in this is played by non-boiling coolant, integral reactor design and passive systems for decay heat removal (or ECCS) directly from the primary circuit and passive coolant flow feedback (SPFF).

In the integral BREST-OD-300 design the primary coolant inventory is encased inside a multilayer metal-concrete vessel (see Fig. 2). The design-basis probability of its depressurization is estimated at $10^{-9}$ year$^{-1}$.

Coolant circulation path in BREST-OD-300 is ensured by difference in free levels. Such a scheme eliminates the need to use isolation valves. This prevents the possibility of coolant flow stop with RCPs in operation and assures continuous circulation under power loss until cooling will be ensured by natural circulation through ECCS. Reactor power loss with coolant flow decrease and malfunction of active reactivity controls (ULOF) is ensured by SPFF (system of passive flow feedback) that is triggered by coolant flow rate.

All this eliminates melting of fuel cladding (the temperature reaches ~890°C just for a short time) and fuel and preserves the integrity of circulation circuit (Fig. 5). The design-basis probability of such an accident is equal to ~$3{\cdot}10^{-9}$ year$^{-1}$.

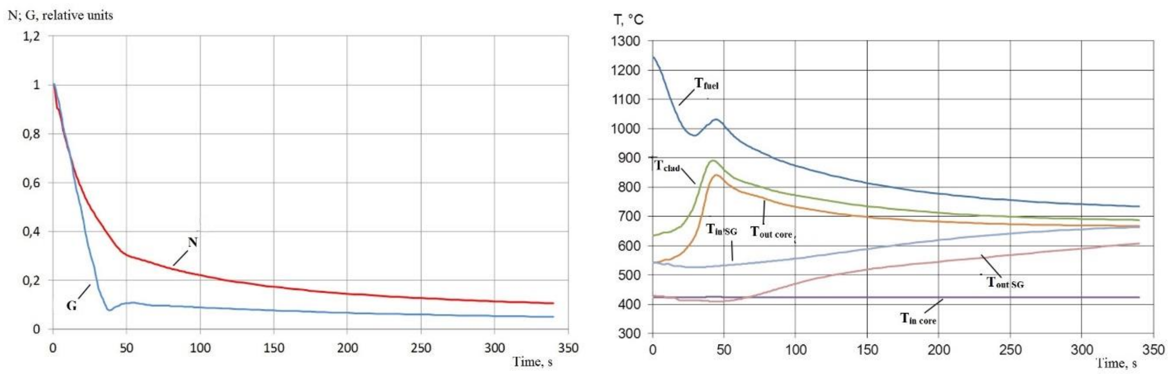



*Fig. 5. Major reactor characteristics change under ULOF.*

## 4.2 Low reactivity margin and elimination of reactivity-induced accidents

FRs are capable to operate without significant reactivity change that is their obvious advantage. Besides the absence of such effects as iodine pit the reactivity stability during operation (fuel burn up) of an equilibrium core with CBR~1 allows eliminating the root of potential danger of uncontrolled prompt neutron runaway – the respective reactivity margin.

Design research in this idea showed that it's practically possible for implementation. In BREST-OD-300 the possibility to contain the reactivity margin of ~$0.65\beta_{eff}$ (~0.23%$\Delta k/k$) during power operation is shown when using startup load made of Pu from PWR SF after long-term cooling and reprocessing. This is basically 100 times less than thermal reactor reactivity margin and 10 times less than sodium FR of first generations (BN-350, BN-600, BN-800) margin.

Uncontrolled power ramp at insertion of the full design reactivity margin is blocked at $1.4N_{nom}$ level. At that the temperature of fuel cladding doesn't exceed ~815°C (Fig. 6), fuel rod meltdown is ruled out. Fission product release for the first 24 hours will not exceed $6.1 \cdot 10^{10}$Bq (below the control release level during normal operation). The probability of such a scenario is $\sim 3 \cdot 10^{-9}$ year$^{-1}$.

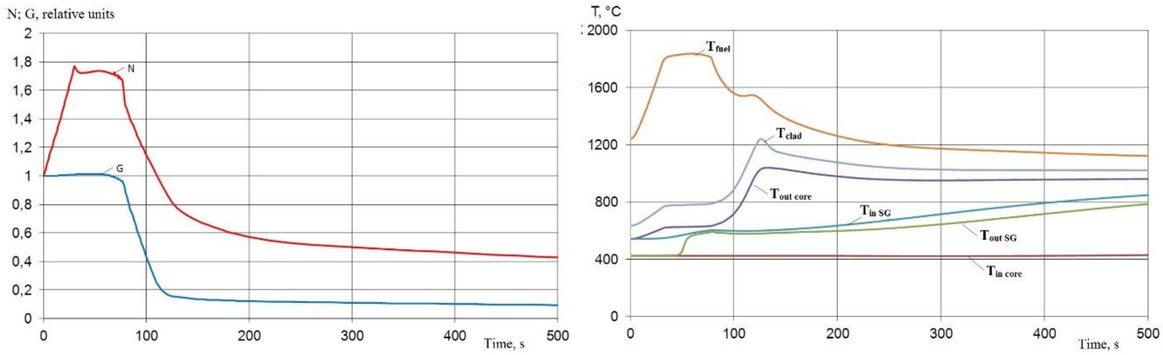



*Fig. 6. Major reactor characteristics change under UTOP.*

### 4.3 Approaching radiation equivalent RAW management

Radiation equivalent principle with its main propositions developed in RF at the turn of the 20th century [6-9] is accepted as the ground requirement for PRORYV Project. It is shown that at certain conditions a situation is possible in nuclear power when RAW sent for final disposal in geological formations would have potential biological toxicity (PBT) equal (i.e. equivalent) or less than the consumed natural U which means that the concept of radiation equivalence (RE) will be fulfilled. Radiation equivalence can be achieved at the moment of disposal or over a certain, historically short, easily forecasted period of time (200-500 years).

Radiation equivalence can be reached if transmutation fuel cycle is implemented in nuclear power with the following major constituents:

- Reprocessing of the whole volume of irradiated fuel from thermal reactors with preset fractioning for transfer of Pu, MA and long-lived fission products to a fast reactor fuel cycle.
- Fast reactors operating in CNFC where most of actinides burn up and long-lived fission products transmutation occur while generating electricity.
- Deep purification of long-lived radioactive waste from Pu, Am and certain other long-lived nuclides to be disposed (loss of actinides in RAW don't exceed 0.1-0.01%).
- Temporary storage of high-active waste before final disposal for a period of 200 years in order to decrease their biological danger by 100 times.

It's preferable to implement a new technology for U ore extraction that wouldn't pollute the environment with concurrent extraction of Ra and Th with U for further transmutation in FR fuel.

In order to decrease RAW long-lived radioactivity it's most important to remove actinides (from U to Cm) from the material that is to be disposed. It decreases PBT of remaining fission products by 1000-10000 times. So the goal of actinides transmutation is to transfer them into fission products, but not to change one actinide into another. The contribution of main fuel nuclides in PBT of BREST irradiated fuel is shown in Fig. 7, the main one being from Pu and Am.

$^{90}$Sr and $^{137}$Cs with daughter nuclides are worth noting out of fission products with half-life more than 25 years. Due to small cross sections of neutron interaction these nuclides cannot be effectively subjected to transmutation and the only way of their management is controlled storage, possibly – useful utilization in isotope devices, or disposal.

The recommended share of actinides lost in RAW of 0.1% will remain acceptable to implement the radiation equivalence until the end of this century.

To guarantee radiation equivalence of nuclear power for distant times it is necessary:

- to implement simultaneous co-extraction of Ra and Th along with U from the ore in the next few decades;
- to keep decreasing in the 22$^{nd}$ century the share of actinides lost in RAW;
- decrease $^{14}$C accumulation in fuel (switch to nitrogen enriched with $^{15}$N isotope);
- solve the problem of transmutation of long-lived fission products.

The possibility of achieving radiation equivalence in Russian nuclear power by the end of 21$^{st}$ century was shown for scenarios of nuclear power development with existing and planned thermal reactors and with developing system of fast reactors. An example of growing power scenario is shown in Fig. 8. RAW radiation balance accumulated by the end of the 21$^{st}$ century in case of implemented fuel cycle with transmutation and consumed U raw material (U and its decay products, $^{226}$Ra and $^{230}$Th) is shown in Fig. 9. Radiation equivalence will be reached after 200 years of RAW cooling.

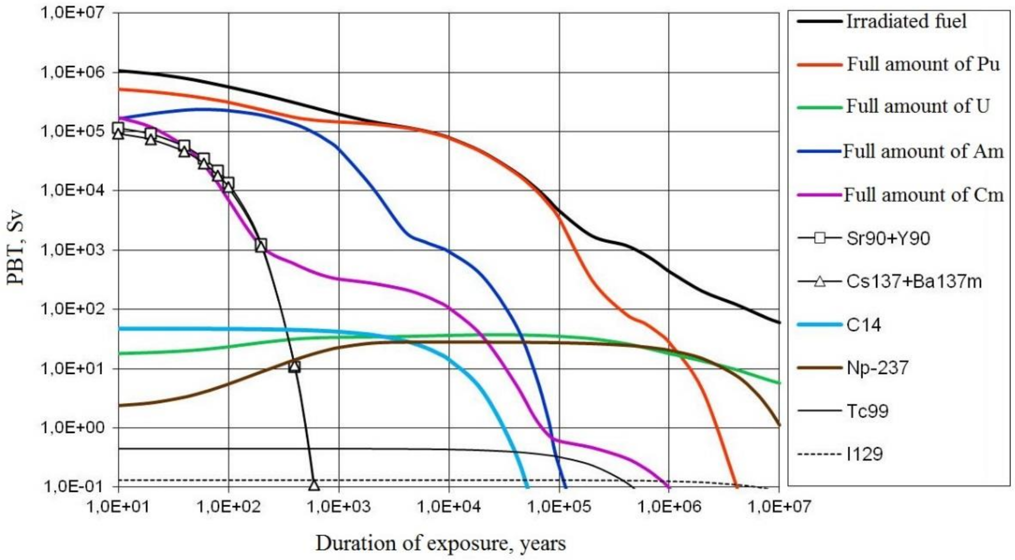



*Fig.7. Contribution of certain elements and nuclides in PBT of BREST spent fuel normalized per 1 kg of irradiated actinides (1.06 kg of nitride fuel).*

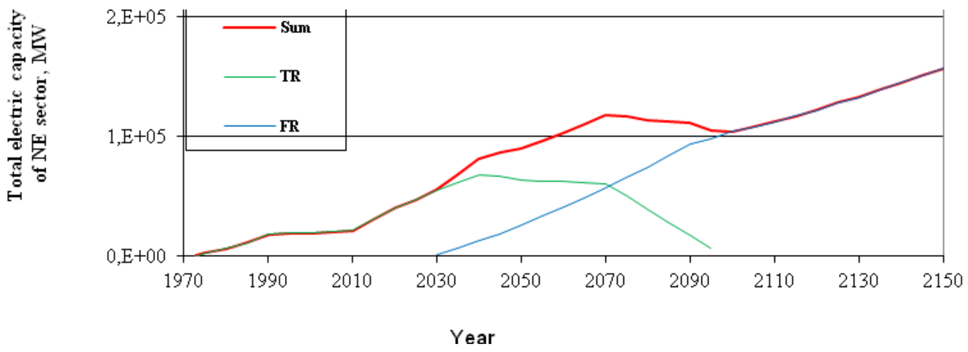



*Fig.8. Model scenario of nuclear power development in RF.*

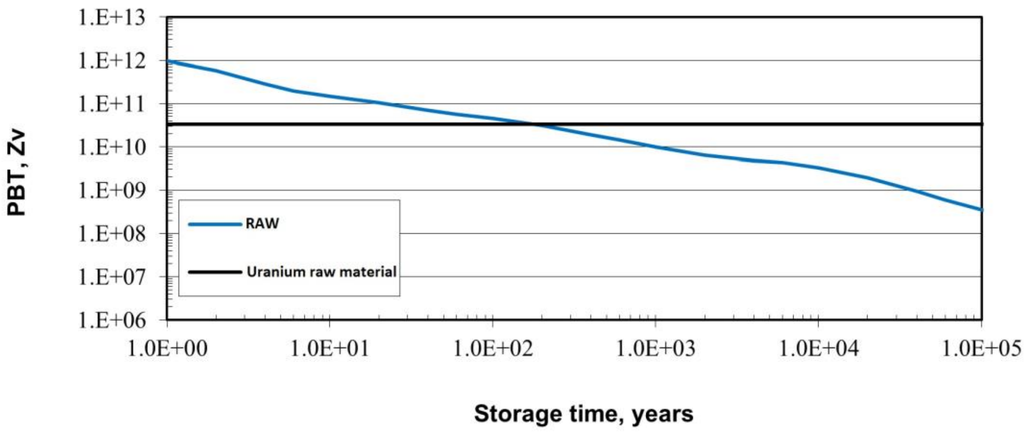



*Fig.9. Full potential biological danger of consumed raw U and RAW.*

# 5 Dense nitride fuel

## 5.1 Advantages of nitride fuel

At initial stages when physical grounds and first designs of fast reactors were developed the attention was paid to reaching the highest fuel breeding ratio. This was guided by small fuel source for FR as naturally absent Pu and the need to produce it with large rate. Metal fuel fits this strategy's implementation in the best way.

At present the list of priorities defining the place of FRs in nuclear power has significantly changed. The first place is now occupied by safety problems including ecology, competitiveness, accumulation of SF and RAW, nonproliferation, optimal utilization of natural resources. Results of comparative analysis of two main types of dense fuel – nitride and metal in order to determine the best option for this strategy showed the following:

- Nitride fuel has high density and heat conductivity (1.4 and 10 times higher than oxide).
- Despite metal fuel has higher theoretical density the nitride fuel is comparable to alloyed metal fuel with Zr and higher porosity that are required to decrease swelling and increase creep resistance, which decrease its density.
- Phase changes of metal fuel and especially its interaction with steel cladding (with generation of easily-melted eutectic) define low destruction margins in accidents with increasing temperature or otherwise decrease of coolant temperature is required.

- Relative disadvantage of nitride fuel is its neutron absorption in $^{14}N(n,p)^{14}C$ reaction that results in somewhat worsening of neutron balance and generation of $^{14}C$ with long half-life.

The results of the research lead to a conclusion that nitride fuel is the best option. It allows to reach basically new qualities of reactor core with CBR~1, decrease a reactivity margin to minimum values and at the same time keep other effects and reactivity coefficients in admissible limits.

One point in favor of metal fuel could be higher achievable fuel breeding parameters if necessary. Considered scenarios of nuclear power development in Russia and the world don't see such necessity. The fundamental point of PRORYV Project is the priority of safety and therefore the choice of CBR~1 which doesn't prohibit the use of blanket if higher BR is necessary (proliferation resistance is a concern in this case). All this determines the choice of nitride fuel as having better overall safety characteristics.

The choice of fuel for lead-cooled fast reactors is significantly influenced by lead interaction with metal fuel with generation of U and Pu plumbates. Disadvantages of metal fuel are high swelling with burn up that requires deep alloying and high porosity as well as phase changes at relatively low temperatures – near the lead working temperatures. Under these circumstances the choice of the same fuel for BN-1200 looks logical too.

It was also taken into consideration that Russia has more experience with nitride fuel operation and its technology is more developed than for metal fuel. In particular one can remember 18 years of operation in BR-10, BORA-BORA experiment in BOR-60, where 12.1% h.a. burn up was achieved.

For large power reactors the nitride fuel allows to implement the advantages of CBR~1 reactor cores and suitable fuel cycle: low reactivity margin for fuel burn up, self-sufficient fuel source, lack of need to separate U and Pu, and the required feedback parameters that define reactor safety (reactivity coefficients and effects).

### 5.2 Reactor testing of nitride fuel in BN-600 and research reactors

A comprehensive program of numerical and experimental research (CPNER) [11] and validation of working capacity of mixed U-Pu nitride (MNIT) fuel is performed for BN-1200 and BREST-OD-300 reactors within the frameworks of PRORYV Project. The goal of CPNER is validation of working capacity of fuel rods, ensuring stability of parameters reproduction and required quality of MNIT fuel, fuel assemblies and experimental fuel assemblies production technology. At the current stage CPNER has a goal to verify the initial stage of operation: for BREST-OD-300 max. fuel burn up of ~6%, max. damaging dose up to ~85 dpa, for BN-1200 max. fuel burn up of ~7.5%, max. damaging dose up to ~95 dpa.

The program includes improvement of production technology, composition and structure of MNIT fuel, pre-reactor research of MNIT properties, reactor tests of fuel rods in research reactors (MIR, BOR-60) and commercial reactor BN-600, post-irradiation research of all experimental assemblies. Reactor research is accompanied by pre-test analysis using fuel codes.

In order to study fuel behavior produced by carbothermal synthesis JSC "VNIINM" has fabricated experimental fuel rods that are being tested in BOR-60 reactor. Fuel pellet parameters are as follows: density from 12.0 g/cm$^3$ to 13.0 g/cm$^3$, Pu content from 12% to 20%, $O_2$ content <0.15%, C content <0.15%, pellet diameter from 5.9 to 10.2 mm depending on fuel rod type. Total number of fuel rods manufactured – 65.

Lab technology of mixed nitride fuel with carbothermal synthesis of initial powders developed in JSC "VNIINM" is implemented on a large scale at JSC "SHK" in Seversk, where the capability to manufacture full scale experimental fuel assemblies for testing in BN-600 is available. This technology was used to manufacture fuel pellets for fuel rods of all experimental fuel assemblies (EFA) that are being irradiated in BN-600 core. The total of 500 fuel rods was manufactured.

Fifteen large-scale FAs were loaded into BN-600. Irradiation of 7 EFAs has been finished by the fall of 2016 with max. burn up of 7.4% h.a. All fuel rods are intact. Nine EFAs of detachable type with 7 fuel rods in each were placed into BOR-60 reactor for irradiation.

An irradiation device consisting of 7 fuel rods, 6 of which are equipped with sensors for measurement of gaseous fission products pressure under the cladding, fuel rod extension and fuel temperature, is loaded into the MIR research reactor. According to the program, post-irradiation study will be completed in 2017.

Post-irradiation examination was conducted on BOR-60 fuel rods at maximum burn up of 1.3% h.a. and 3.2% h.a., irradiated within EFA-1 experimental fuel assembly, and CEFA-1 combined experimental fuel assembly of BN-600 at 5.5% h.a. burn up. The obtained results enable the verification of fuel codes and justification of possible extension of assumed service life of experimental fuel assemblies in BN-600 reactor.

## 6 Design of BN-1200 commercial sodium-cooled reactor

The BN-1200 design developed within the framework of the PRORYV project makes full use of Russian experience in development and operation of BN-350, BN-600 and BN-800 reactors and was aimed at ensuring its competitiveness with the best designs of thermal neutron reactors. The list of verified and tested design approaches includes:

- integral primary circuit layout (Fig. 10);
- three-circuit reactor unit scheme;
- a variety of equipment technical solutions.

However innovative solutions enabling new safety qualities and CAPEX reduction are of the most interest:

- decreased power density in the reactor core (from 450 down to 230 kW/m$^3$) with the CBR~1 and using nitride fuel;
- 2-3 times longer fuel campaign and 2 times longer refueling interval;
- complete integration of all sodium systems of the primary circuit in the reactor vessel;
- emergency system for decay heat removal from the primary circuit which is built in the reactor vessel;
- system of passive reactivity feedback based on the thermal principle;
- pressure vessel steam generator;
- refueling system without accumulation drums, flushing chamber and spent FA drums;
- fewer auxiliary systems.

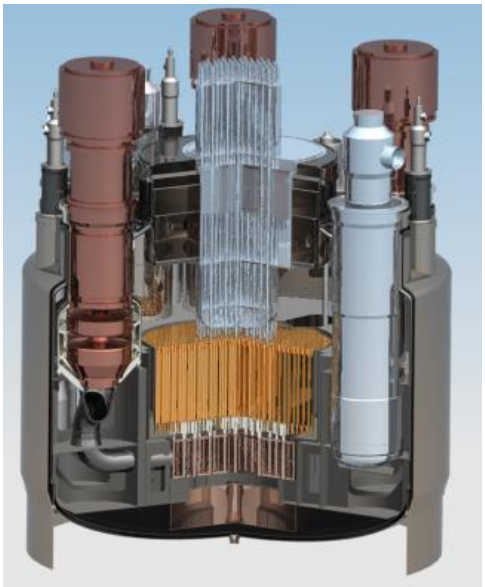


*Fig. 10. BN-1200 reactor layout.*

Results obtained in the course of design implementation confirm the progress achieved in CAPEX reduction and the technical and economic performance parameters of fast reactors approaching the similar indices of thermal reactors.

At the same time, the PRORYV project is aimed at ensuring the competitiveness not only and, most likely, not predominantly versus thermal reactors, but rather versus alternative power sources including Combined Cycle Power Plants (CCPP) and renewable sources of energy.

The requirements that follow from such a formulation of the objective are detailed in the following section as applied to the BR-1200 fast reactor of 1200 MWe power.

## 7 New generation computation software for validation of design approaches and safety of NPPs with BREST-OD-300 and BN-1200 in CNFC

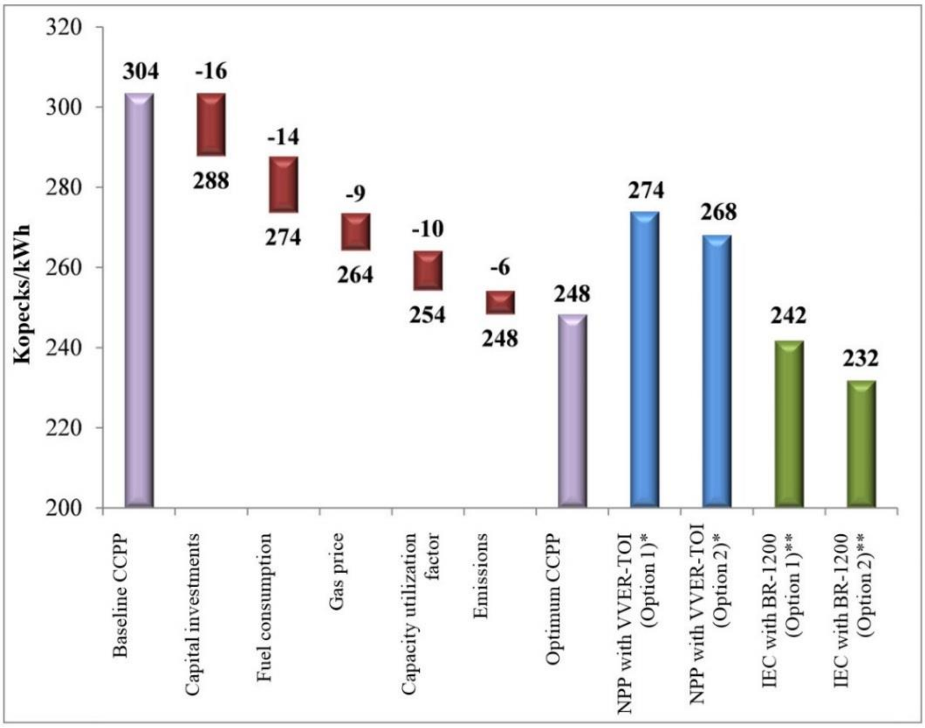
A topic devoted to developing and using new generation computation software for validation and justification of design solutions is opened within PRORYV Project besides the large-scale experimental program [12, 13]. The list of software codes of new generation includes a full range of software necessary to validate design approaches and safety of NPPs with BREST-OD-300 and BN-1200 reactors, including codes to simulate reactor operation, fission products transport inside the premises of NPP and in the environment, technological processes of CNFC and RAW handling, validated as to both experiments on certain phenomena and the results of BN-600, BN-800, BOR-60 operation in the experimental conditions.

By the end of 2016 the following 18 codes have been developed, verified and validated: neutron physics (MCU-FR, ODETTA, NDP-ACE), thermal hydraulics (HYDRA-IBRAE/LM, LOGOS, CONV-3D), fuel rod behavior (BERKUT), heat and mass transfer and fission products transport inside premises of NPP (KUPOL-BR), fission products transfer in the environment (Sibilla, ROM, ROUZ), RAW disposal safety validation (GeRa), integral codes for NPP safety validation (EUCLID/V1, EUCLID/V2, SOCRAT-BN/V1, SOCRAT-BN/V2), probabilistic safety analysis (CRISS 5.3), balance of materials and nuclide flows in CNFC (VISART), and 3 out of them are certified by Rostechnadzor, 10 others have just entered the certification procedure.

The results obtained with the help of this new generation software have confirmed the high safety level of BREST-OD-300 and BN-1200 designs and allow confirming the main proposals for radiation equivalent RAW disposal.

## 8 Competitiveness of NPP with fast neutron reactors in a closed nuclear fuel cycle

Calculation of LCOE for FR and CCPP under Russian conditions was based on a discount rate of 10%. Fig. 11 shows the results of LCOE calculations based on gradual improvement of CCPP performance versus LCOE of considered NPP units. LCOE values for NPPs are provided for different values of fuel cost.




*(* - for VVER-TOI – different price levels for NFC, ** - for BR-1200 – different burn up)*

*Fig. 11. LCOE of CCPP and NPP in Russian conditions (10% discount rate), kopecks/kWh.*

Table IV shows the results of the LCOE calculations for competing generation types developed in Russia with the optimal (best) technical and economic performance for different discount rates.

The following conclusions may be made based on the calculations performed for Russian power generation facilities:

1. NPPs with thermal reactors operating in an open NFC cannot guarantee further efficient competitive development of the nuclear power sector in Russia.
2. Achievement of the established technical and economic performance requirements for industrial energy complexes with BR-1200 will enable maintenance of the Russian nuclear power sector's competitiveness even against CCPPs with an optimum technical and economic performance as well as renewable energy sources.

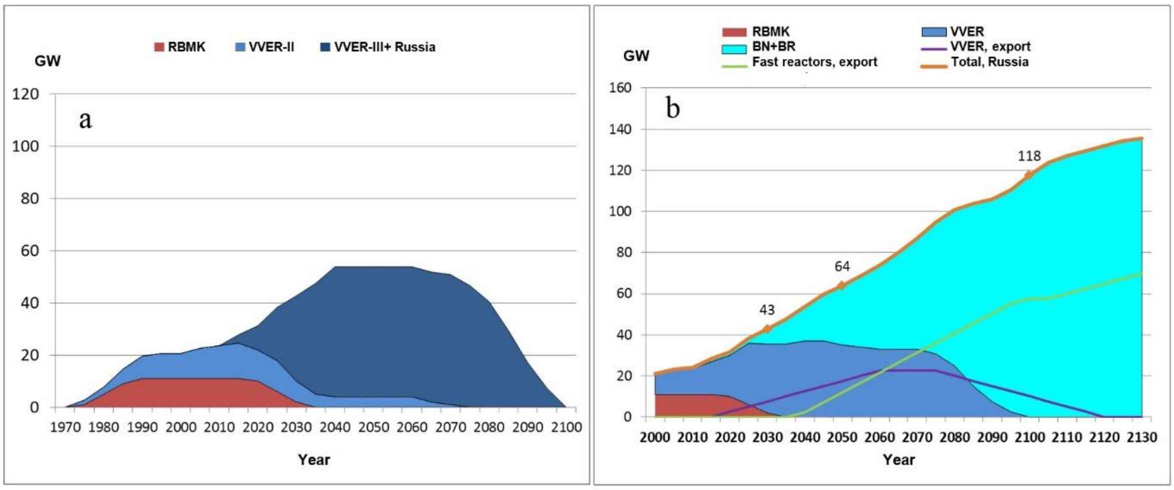
TABLE IV: Results of LCOE calculations for competing energy technologies with optimal technical and economic performance, kopecks/kWh.

| | Discount 10% | Discount 7% | Discount 3% |
|---|---|---|---|
| Solar Power Plant | 485,4 | 284,0 | 228,8 |
| Wind Power Plant | 322,5 | 189,9 | 152,9 |
| CCPP | 248,3 | 152,9 | 136,2 |
| VVER-TOI | 268,1 | 151,8 | 116,6 |
| BR-1200 | 231,8 | 129,2 | 96,7 |

## 9 Two-component nuclear power sector and its development prospects

The nuclear power sector has a future only if FR technologies and a CNFC are successfully mastered (Fig. 12).

Scenario (a) implies that the objective of developing a large-scale nuclear power sector will not be achieved if VVERs with an open NFC are used, so because of uranium resource restriction at the level of 700 kt the introduction of new generating facilities must cease between 2040 and 2045 with power level of the nuclear power sector at <55 GW. In this case, the nuclear power sector will deplete its resources and cease existing by the end of the century. If fast reactors with closed NFC are introduced in due time based on scenario (b) (first 3-5 units based on fast-neutron technology currently in use followed by inherently safe FR with lead coolant), and if installed capacity growth rate till the year 2040 is the same, there will be no resource restrictions for development of nuclear power, and by the end of the century the level of ~120 GW may be achieved with possible future growth.




a – single-component nuclear power based on VVER

b – two-component nuclear power based on VVER→FR

*Fig. 12. Possible dynamics of nuclear power development in Russia.*

Regardless of the overall Russian nuclear power growth forecast, development of a two-component nuclear power sector should be transient in this century and will end with a transfer to a new technological platform with domination of inherently safe FR and CNFC. Duration of this stage should be minimized, if possible, based on the following key aspects:

- preservation of acceptable nuclear power safety level in general with significant increase in NPP capacities;
- uranium resource saving;
- resolution of the problem with accumulated spent nuclear fuel from thermal reactors;
- reduction of system-wide electricity cost (ultimately also in the nuclear power sector).

## 10 Summary

The results achieved over a relatively small period of time (5 years) within the PRORYV project have confirmed technical and technological feasibility of its basic provisions and enable a transition to the practical implementation stage and transfer to a new technological platform of the nuclear power sector based on a closed NFC at the cusp of the 2030's.

Implementation of the developed design, engineering and process solutions and performance of the scheduled R&D program at the PDEC (first phase startup in 2020) will ensure a high probability of appearance of a new prototype of competitive industrial energy complex capable of operating within the framework of two-component nuclear power sector by 2030.

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