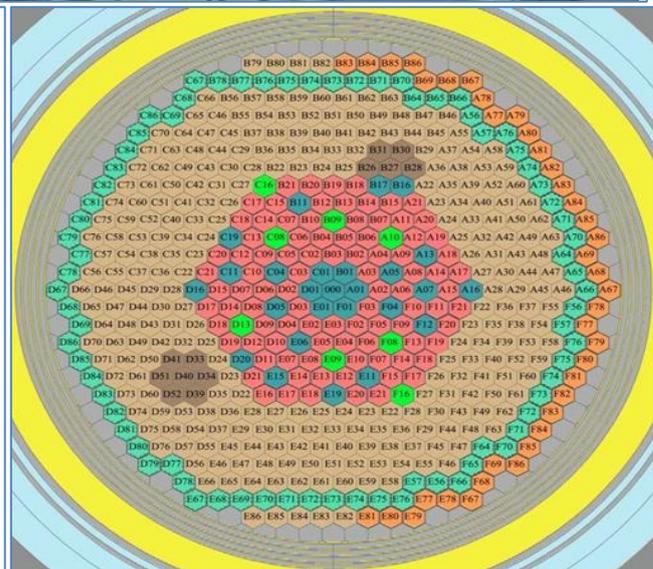




POCATOM STATE ATOMIC ENERGY CORPORATION «ROSATOM»



**Multilateral Research Program
International Research Center MBIR
Proposal on R&D activities (Fuels and Materials) for the Ten Year
Period 2025-2035**

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1. INTRODUCTION.

FAST REACTOR DEVELOPMENT PROGRAMME

Development of nuclear technologies at the turn of the 21st century has brought us nearer to the closed nuclear fuel cycle with probable positive economic impact on the basis of multiply recycling of nuclear fuel. In this regard the global interest in fast reactors has been growing because they can provide efficiency of natural uranium usage in comparison to thermal reactors and improvements in nuclear spent fuel and waste management. And the long term development of nuclear power is associated primarily with incorporation of fast reactor technology and closed fuel cycle in the existing nuclear infrastructure.

International cooperation on fast reactor technology is carried out in the frame of the Generation IV International Forum (GIF) and the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO).

Members of the Generation IV International Forum identified and selected 6 (six) advanced nuclear energy systems for further deployment (four of them are fast reactors): Sodium-cooled Fast Reactor (SFR), Lead-cooled Fast Reactor (LFR), Gas-cooled Fast Reactor (GFR), Molten Salt Reactor (MSR), Supercritical Water-cooled Reactor (SCWR) and Very High Temperature Reactor (VHTR).

The most mature reactor technology (from the abovementioned) is the sodium cooled fast reactor as it has the significant experience of operating experimental, prototype, demonstration and commercial units. Nowadays the Russian Federation, France, the United States, China, India, Japan and the Republic of Korea are working on this technology.

Lead-cooled (and lead-bismuth) fast reactor is under development in the Russian Federation, China and Europe (European consortium ALFRED and Belgian project MYRRHA).

The most active project on gas-cooled fast reactor is considered to be joint European project ALLEGRO implemented by consortium V4G4. The consortium includes scientific institutes of the Czech Republic, Hungary, the Slovak Republic and Poland with CEA's (France) association.

Molten salt reactor is being carried out in the Russian Federation, France and the United States.

Supercritical water-cooled reactor is run by the Russian Federation, the United States and Japan.

Very high temperature reactor is studied in China, France, Japan, the Republic of Korea, and the United States.

The International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) under the IAEA's auspices also supports sustainable development of nuclear energy by means of fast reactor technology among others.

Implementation of fast reactors is included in the long-term strategy and program development of the following countries: China, the Russian Federation, India, the Republic of Korea (along with pyroprocessing), the United States, France, and Japan. In particular, the Republic of Korea and Japan plan to use fast reactors for improvement the nuclear waste management by reducing the radiotoxicity level of nuclear waste. Meanwhile, China, India and the Russian Federation believe in the necessity to develop two-component nuclear system with closed nuclear fuel cycle.

After the successful commissioning of BN-800, it is generally recognized within the Russian nuclear community that nuclear power in Russia will further develop as a two-component system based on thermal and fast reactors operated in unified closed nuclear fuel cycle.

The Russian side priority areas for research are aimed at solving the

following strategic objectives:

1. Development of technologies for closing the fuel cycle at the experimental and engineering levels, followed by its practical implementation at the demonstration and pilot levels.
2. Assurance of competitiveness of fast reactors against steam-gas and alternative energy as well as NPPs with thermal reactors.

Fast reactors will be incorporated into Russian two-component nuclear power incrementally.

In the short term, BN-600 and BN-800 reactors are certain to be operated. Programs of increasing the core life of BN (up to 2024) and converting BN-800 to MOX fuel have been approved by Rosenergoatom for these reactors. It is also supposed that the first BN-1200 type reactor will be constructed. Its base fuel will be MOX fuel with a layer of depleted uranium, with dense nitride fuel considered as an option after its technology has been tried out (after 2035 or later). It is expected that the demonstrator lead-cooled fast reactor will be available in 30-40s years of the 21st century. At the moment the pilot-demonstration BREST OD-300 reactor is planned.

In the medium term, justification of dense MNU-Pu fuel for lead cooled reactors is to be completed. To justify the fuel, a comprehensive program of computational and experimental substantiation (CPCES) and a program of increasing fuel burnup have been developed.

The Russian Federation already has materials science programs for these reactors and customers, financing and implementation terms are known.

In the medium or long term, such projects as SVBR-100, BREST-1200 (MNU-Pu), advanced BN-GT (with gas-turbine cycle), molten salt and thermonuclear reactors can also be carried out. Engineering and materials programs designed to implement these projects needed further developments

and may be conducted at the beginning on multilateral basis for risks mitigation and response to general technical issues.

Development, implementation and safe operation of fast reactors requires comprehensive work on computer codes verification, structural materials research, advanced fuel material and fuel types validation, as well as experiments under emergency conditions along with materials and fuel tests in transient conditions in different coolants. Tasks on improvement materials and fuel for thermal reactors are also important.

Intense research is being conducted into advanced reactor technologies in the international scientific community for the Generation 4 Project within the Generation IV International Forum (GIF). Much of the research is concentrated on the development and justification of new materials and fuel types; such research is already being undertaken in the BOR-60 research reactor, for example:

- Irradiation of structural materials which were irradiated in FFTF;
- Irradiation of U-Zr fuel rods;
- Works on minor actinide incineration.

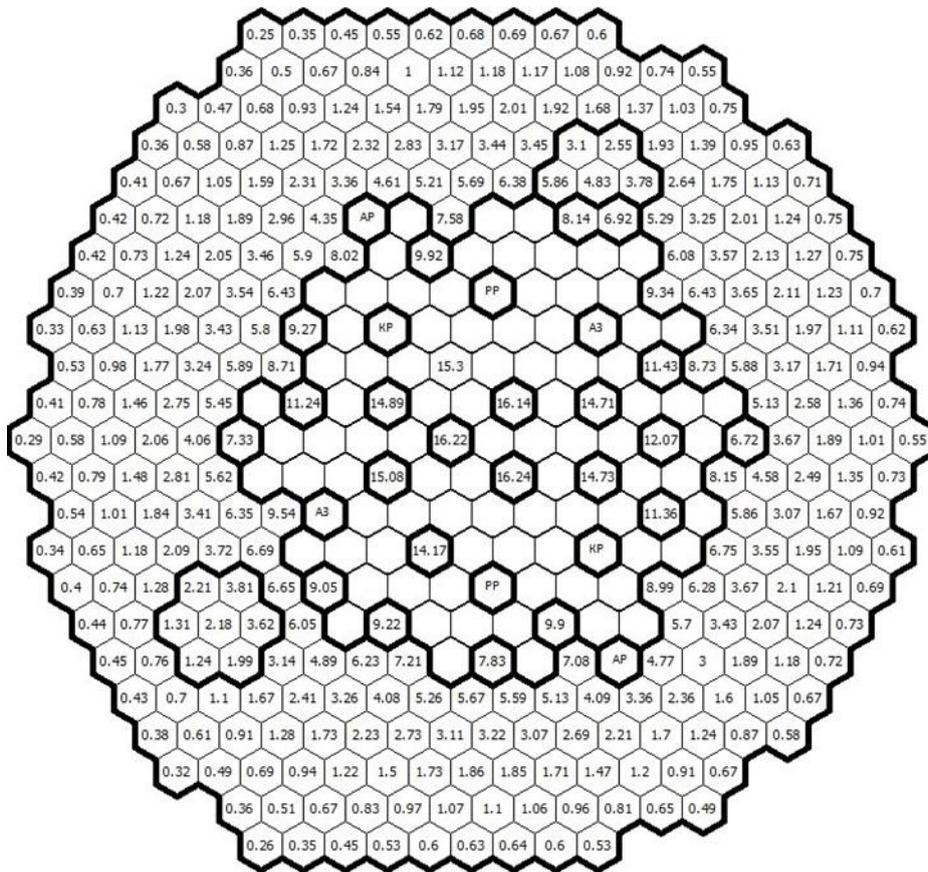
To justify the task solutions, an extensive experimental program will have to be implemented. The program will be carried out in the fast multipurpose MBIR research reactor which is being constructed in our country to replace BOR-60.

The present report describes the main areas of experimental studies planned in the MBIR reactor for justification of the international research programs of our foreign partners to be implemented at the International Research Center MBIR as well as for justification of the two-component nuclear power in the short and medium term (till 2035) .

2. MBIR DESIGN AND EXPERIMENTAL CAPABILITIES

MBIR is unique multipurpose sodium-cooled fast reactor of 150 MW (t) capacity. Using of thin fuel elements (diameter is 6.0*0.3 mm) will allow MBIR to reach high heat intensity and high neutron flux up to $5,3 \cdot 10^{15}$ n / $\text{cm}^2 \cdot \text{s}$. Reactor core (fuel column height is 55 cm) is comprised of 93 fuel assemblies with width across flats 72.2*1.5 mm. Every fuel assembly contains 91 fuel elements.

Figure 1. Core and lateral blanket



Damaging dose rate (dpa per 100 effective days) in the irradiation cells and blanket assemblies.

The core has 8 control rods of different purpose. The core provides 17 cells for experimental and material test assemblies. Besides the project is supposed to have 3 loop channels (central and 2 in the side screen) each of which takes 7 reactor cells.

Figure 2. Loop channels technical parameters

Parameter / Loop name	LCh-Na	LCh-Pb	LCh-Pb-Bi	LCh-Gas (He)	LCh-Salt
Coolant	Sodium	Lead	Lead-bismuth alloy	Gas (high purity helium)	Metal fluorides melt
Neutron fluence in LCh, $\text{cm}^{-2}\cdot\text{s}^{-1}$	$\geq 3 \cdot 10^{15}$	$2 \cdot 10^{15}$	$(2\div 3) \cdot 10^{15}$	$(0.4\div 1) \cdot 10^{15}$	Up to $3.5 \cdot 10^{15}$ to
Power, MW	Up to 1.0	≥ 0.3	Up to 0.8	Up to 0.15	Up to 0.15
External diameter, mm	≥ 190	≥ 190	≥ 190	≥ 130	≥ 150
Fuel length	Core height	Core height	Core height	Side reflector height	Core height
$T_{\text{in}}/T_{\text{out}}$ of working fluid, $^{\circ}\text{C}$	320/550	Up to 350/ up to 750	Up to 350/ up to 500	≥ 950	750/ 800

At the beginning the reactor could work without loop channels and the core will differ from the project one. In particular the number of experimental and material test assemblies may be increased up to 21. Besides for research purposes it will be possible to use 45 steel assemblies of the 1st blanket row.

Fuel cycle is supposed to be 100 effective full power days and installed capacity utilization factor is planned to be 0.65 (237 effective full power days per year).

Parameters on maximum neutron flux and damaging dose rate on structural materials are given in the table below.

Maximum neutron flux, $\text{n}/\text{cm}^2\text{s} \times 10^{15}$	5.30
Maximum neutron flux with $E > 0.1 \text{ MeV}$, $\text{n}/\text{cm}^2\text{s} \times 10^{15}$	3.6
Maximum fluence per 100 EFPD, $10^{23} \text{ n}/\text{cm}^2$	0.46

Maximum fluence per 100 EFPD flux with $E > 0.1$ MeV, 10^{23} n/cm ²	0.31
Maximum fuel cladding damaging dose rate per 100 EFPD, dpa	15.2

Depending on specific location maximum neutron flux in the core may vary from maximum (5.3×10^{15} n/cm²s) to 2×10^{15} n/cm²s and 1.8×10^{15} n/cm²s in the 1st blanket row.

3. MAIN AREAS OF EXPERIMENTAL WORK AND MATERIALS RESEARCH

3.1. Research into advanced fuel materials performance

Objective: To justify reliability and safety of different fuel types including (U, Pu) N-based fuel for Generation 4 reactors.

Main areas:

- Studies into (U, Pu) N fuel swelling and gas release, depending on its composition, porosity, temperature (900-1200⁰C), burnup (5-15%).
- In-core studies into (U, Pu) N fuel creep, depending on its composition, porosity, temperature, burnup.
- Studies into heat transfer of (U, Pu) N fuel irradiated to different burnup levels, depending on the fuel composition, porosity, temperature, burnup.
- Reactor testing of regenerated fuel with different purification efficiency.
- Studies into different (U, Pu) N-based fuel elements with high burnups and of different technologies used for fuel fabrication.
- In-core studies into composite ceramic (U, Pu) C-based fuel characteristics (gas release, thermal conductivity and diffusivity, swelling, creep).
- Studies into composite ceramic (U, Pu) C-based fuel elements with SiC ceramic coating.
- Studies into MOX fuel having coarse-grain texture (35÷40 µm) with nano-additives (20÷40 nm UO₂) and controlled porosity.
- In-core studies into characteristics (gas release, thermal conductivity and diffusivity, swelling, creep) of dense metal fuel based on alloys (U-Zr, U-Mo and the like)
- Tests on experimental fuel elements of different designs, with dense metal fuel based on alloys (U-Zr, U-Mo and the like).
- Tests on fuel elements with different kinds of dense fuel containing minor actinides.

Expected results

- To get experimental data on the properties of nuclear fuel irradiated in different conditions for its certification and for computer code development. To make recommendations with regard to improvement of fuel fabrication technologies, fuel element and FA designs.
- To enlarge databases of the properties of nuclear fuel irradiated in different conditions. To make appropriate recommendations.
- To create safe and efficient technologies for nuclear fuel reprocessing.
- To improve performance characteristics when regenerated fuel is produced.
- To get experimental data on the properties of nuclear fuel irradiated in different conditions for its assessment/certification and for computer code development.
- To create new fuel materials for high-temperature reactors.
- To increase fuel burn-up.
- To create new fuel compositions and FE designs for fast reactors for improvement of economic characteristics.

3.2. Testing of advanced fuel element materials in transient, power cycling and emergency conditions

Objective: To choose and justify operational modes, to get data for improving FE design and fabrication technology with the aim of achieving high burnups and conforming to the daily (NPP) power control mode of plant operation.

Main areas:

- Studies into behavior of (U, Pu) N containing fuel elements in transient conditions, including degraded heat removal (reduction and loss of the coolant flow, growth of the coolant temperature, etc.) and power ramp due to accidental insertion of positive reactivity.
- Studies into behavior of FEs containing dense metal fuel based on (U-Zr, U-Mo and the like) alloys in transient conditions, including degraded heat removal

(reduction and loss of flow, coolant temperature rise, etc.) and power ramp due to accidental insertion of positive reactivity.

- Studies into behavior of FEs containing composite ceramic (U, Pu) C-based fuel with SiC ceramic cladding in transient conditions, including degraded heat removal (reduction and loss of flow, coolant temperature rise, etc.) and power ramp due to accidental insertion of positive reactivity.

Expected results

- To get experimental data on the properties of nuclear fuel irradiated in different conditions for its certification and for computer code verification.
- To make recommendations for improving fuel fabrication technologies, fuel element and FA designs.

3.3. Testing of advanced structural materials

Objective:

- To justify reliability of FE claddings and internals based on new ferritic-martensitic and austenitic steels for doses up to 170 dpa. To increase the core life to 5 years, to increase the capacity factor. To ensure 50÷60 years of lifetime for irremovable reactor components.
- To study radiation resistance of special heat-resisting materials to be used in reactors for production of hydrogen fuel and other advanced technological processes. To increase thermal efficiency (during the transition to gas coolant – working medium as well).
- To study radiation resistance of advanced low-absorbing, corrosion-resisting materials including those based on silicon carbide and advanced ceramics for justification of the optimal fuel cladding material in SWCR reactors.
- To justify radiation and corrosion resistance of fuel claddings in sodium cooled fast neutron reactors.
- To increase heat resistance of the claddings up to 900⁰C.

- To provide experimental justification of the internals durability for BREST-OD-300 type reactors with heavy metal coolant.
- To conduct experimental studies in support of creating a new class of radiation-resisting materials characterized by high heat resistance and dimensional stability when the damaging dose does not exceed 170 dpa.
- To conduct high-dose experimental studies into the properties of nanocomposite coatings in support of higher corrosion resistance and endurance of cladding materials for CPS rods.
- To conduct high-dose experimental studies on vessel and in-vessel steels for justifying a long lifetime (more than 80 years) of NPPs with thermal reactors.

Main areas:

- High-dose irradiations (170 dpa) of advanced cladding materials (ferritic-martensitic and austenitic steels) in dismantlable devices at temperatures of (350÷700) °C to study mechanical properties, swelling, irradiation creep.
- In-core examination of creep rupture strength/durability and creep of advanced ferritic-martensitic and austenitic steels at temperatures of 350÷700 °C.
- In-core examination of strain capacity and irradiation embrittlement of advanced ferritic-martensitic and austenitic steels at temperatures of 350÷700 °C and different loading rates.
- High-dose irradiations up to 170 dpa of special heat-resisting materials operational at temperatures of (750÷950) °C and (1000÷1100) °C to study their mechanical properties, swelling, irradiation creep, creep rupture strength/durability; studies on carbon-graphite material degradation.
- In-core examination of new low-absorbing, corrosion-resisting materials including those based on silicon carbide and advanced ceramics and operational at pressures of 25÷30 MPa and temperatures of 570÷580 °C for determining their radiation resistance.
- Studies on radiation resistance in order to select and justify the optimal structural materials for the lithium loop, first wall and blanket of the thermonuclear

reactor.

- Studies on materials characterized by radiation-, heat- and corrosion resistance (relative to lithium coolant). Endurance tests on the materials of the first wall, blanket and experimental modules of the thermonuclear reactor.
- Tests on the claddings based on vanadium alloys, including alloys with different coatings.
- Tests on EP823 steel samples to high damaging dose rates to study mechanical properties.
- Tests on structural and functional purpose nanodispersed materials, oxide dispersion strengthened (ODS) ferritic-martensitic steels.
- Tests on nanocomposite coatings based on TiCrNi/Ni-Cr-Fe-Si-B and TiAlN/Ni-Cr-Fe-Si-B.
- Tests on vessel and in-vessel steels for advanced thermal reactors.

Expected results

- To get experimental data on the properties of advanced ferritic-martensitic and austenitic steels as well as special heat-resisting materials for their certification and computer code development.
- To get experimental data on the new low-absorbing, corrosion-resisting materials including those based on silicon carbide and advanced ceramics for their certification and computer code development.
- To get experimental data on materials characterized by radiation-, heat- and corrosion resistance (relative to lithium coolant) for their certification and computer code development.
- To get experimental data on EP-823 steels for their certification and computer code development.
- To increase fuel burnup to 18÷20 % in fast reactors with no coolant performance degradation.
- To enhance lifetime characteristics, reliability and operational safety of movable CPS rods in various nuclear reactors.

- To enlarge databases of material properties.

3.4. Testing of absorbing, moderating and composite materials for innovative nuclear systems

Objective:

- To justify radiation resistance of the absorbing kernels and CPS rods with a long lifetime for advanced fast reactors.
- To justify operability of the CPS rods to be recycled after prolonged operation by retrieving/recovering absorbing kernel inserts and producing strong gamma sources.
- To justify radiation resistance of the moderating block and operability of the CPS rods with improved physical efficiency for fast reactors.
- To justify radiation resistance of the absorbing kernels at the damaging dose rate of up to 170 dpa.
- To study radiation resistance of new composite absorbing materials having specified working parameters and irradiated to the damaging dose rate of up to 170 dpa.
- To study radiation resistance of new composite materials having specified working parameters.
- To study radiation resistance of porous beryllium of different manufacturing technologies, to justify the product performance.

Main areas:

- Tests on CPS rod mock-ups containing hafnium hydride (HfH_x) of different composition, stoichiometry and manufacturing technologies in static and emergency conditions.
- Tests on dual-purpose dismountable CPS rod mock-ups based on the trap-type Eu₂O₃+Co absorbing composition to high damaging dose rates in static and emergency conditions.
- Tests on CPS rod mock-ups and irradiation devices with ¹¹B₄C moderating block

to high damaging dose rates.

- Tests on CPS rod mock-ups containing rare-earth element hafnates ($\text{Ln}_2\text{O}_3+\text{HfO}_2$, where Ln-Dy, Eu,Gd,Er) of different composition and manufacturing technologies to high damaging dose rates in static and emergency conditions of operating various nuclear reactors.
- Tests on SiC- SiC-, B4C-type composite materials with graphite nanotubes, BN with pyrocarbon at high temperatures and damaging dose rates.
- Tests on nanostructured boron steel dispersion strengthened samples (to 2 % B) of different size and geometry to high damaging dose rates.
- Tests on porous beryllium of different manufacturing technologies.

Expected results

- To get experimental data on the properties of hafnium hydride (HfH_x) of different composition, stoichiometry and manufacturing technologies in different conditions for their certification and computer code development. To increase CPS lifetime from 2÷3 to 8÷10 years.
- To increase CPS lifetime in fast reactors, including BN-600, BN-800, BN-1200, from 2÷3 to 8÷10 years and to produce gamma sources characterized by specific activity of over 100 Ci/g and improved performance characteristics, to recycle high-level waste represented by spent CPS rods from nuclear reactors.
- To increase CPS rod effectiveness, to reduce the amount of absorbing materials in the product, to have the possibility of radioisotope production.
- To get experimental data on the properties of rare-earth element hafnates ($\text{Ln}_2\text{O}_3+\text{HfO}_2$, where Ln-Dy, Eu, Gd, Er) in different conditions for their certification and computer code development. To increase CPS lifetime of thermal reactors from 10 to 25÷30 years.
- To get experimental data on the properties of materials like SiC-SiC, B4C with graphite nanotubes, BN with pyrocarbon in different conditions for their certification and computer code development. To increase lifetime of reactor core components.

- To enhance operating parameters of CPS rods and other reactor core components, including neutron shield of reactor vessels.
- To get experimental data on the properties of porous beryllium of different manufacturing technologies in different conditions for their certification and computer code development. To enhance performance characteristics of the blanket and wall in thermonuclear reactors and moderating blocks in research reactors.

3.5. Research into new and modified liquid metal coolants

Objective:

- To develop instrumentation and methods that will allow to define presence of impurities, activation of loop equipment in BREST-OD-300 and SVBR-100 type reactors with heavy coolant.
- To develop instrumentation and methods that will allow to define presence of impurities, activation of loop equipment in molten salt reactors.

Main areas:

- Experimental studies into lead and lead-bismuth coolant technologies in a loop facility with simulated operating conditions of advanced reactor cores.
- Experimental studies into the molten salt coolant technology in a loop facility with simulated operating conditions of advanced reactor cores.

Expected results

- To improve operational modes. To enhance reliability and operational safety of BREST and SVBR type reactors with heavy coolant.
- To get experimental data on the molten salt coolant technology, for example corrosion of structural materials, for controls and circuit equipment development.

3.6. Life tests of new equipment types for innovative nuclear systems.

Objective:

To get experimental data on the functionality of the innovative equipment, sensors and devices for reactors with lead, lead-bismuth and molten salt coolants.

Main areas:

- Tests of the mock-ups of innovative loop equipment, sensors and controls for reactors with lead, lead-bismuth and molten salt coolants.

Expected results

- To get experimental data for creating innovative equipment, sensors and loop monitoring devices for new type nuclear reactors.
- To enhance reliability and operational safety of BREST-OD-300, SVBR-100 type reactors and molten salt reactor.

3.7. Conducting of reactor physics, materials, thermal hydraulics and other research for computer code verification

Objective:

To get experimental data for development of integral codes to simulate reactor cores with various types of coolant, including sodium, lead, lead-bismuth and molten salt coolants in different operating conditions.

Main areas:

- Conducting of calibration and benchmark model experiments for multiscale simulation of the core component mock-up behavior in reactors with various types of coolant, including sodium, lead, lead-bismuth and molten salt coolants in different operating conditions.

Expected results

- To create and verify integral codes for simulating reactor cores with various types of coolant, including sodium, lead, lead-bismuth and molten salt coolants in different operating conditions.

3.8. Applied experimental work, using reactor radiation.

Objective:

- To develop and to justify experimentally neutron therapy technologies.
- To develop technology for producing low neutron capture cross-section radionuclides
- To develop and to justify experimentally improved technologies for producing ^{153}Gd , ^{89}Sr , ^{62}Ni , ^{60}Co radionuclides.
- To develop and to justify experimentally methods of neutron radiography and tomography of irradiated materials and products.

Main areas:

- Experimental studies in support of neutron therapy technologies.
- Tests on trap-type irradiation devices for producing various low neutron capture cross-section radionuclides.
- Creation of improved technologies and production of ^{60}Co , ^{153}Gd , ^{89}Sr , ^{62}Ni in the radial blanket of the MBIR reactor.
- Experimental studies in support of the facility for neutron radiography and tomography of irradiated materials and products.

Expected results

- To create technologies for neutron therapy and to bring neutron beams into practical use for medical purposes.
- To set up production of low neutron capture cross-section radionuclides.
- To produce ^{153}Gd , ^{89}Sr , ^{62}Ni , ^{60}Co radioisotopes and sources based on them.
- To study fuel, absorbing and structural materials, mock-ups of various reactor core components irradiated in the MBIR reactor.
- To study the content of toxic elements in the environment (Fe, Zn, Cr, Sb, W, Sc, La, Yb, Th, Na, Rb, Cs).

4. CONCLUSION

The multipurpose fast research reactor MBIR is needed for the implementation of development strategy of the two-component nuclear power based on fast and thermal reactors as well as for justification of technologies for closing nuclear fuel cycle.

Construction of the multipurpose fast research reactor MBIR and achieving first criticality in it are scheduled for 2024, research work on materials is scheduled to start in 2025.

The main function that the multipurpose fast research reactor MBIR is supposed to serve is performing a series of reactor tests of innovative materials and core component mock-ups for the Generation 4 nuclear energy systems including fast reactors with closed fuel cycle and low- and medium-power thermal reactors.

5. CONTACTS

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