IRC MBIR
ROSATOM

ADVANCED SCIENTIFIC RESEARCH PROGRAMME FOR THE 2028-2040 PERIOD

ADDRESS OF THE CEO

Dear colleagues!

It is impossible to build new nuclear power industry without scientists. Scientists need for this purpose large modern research reactors allowing to improve technologies and create new materials for nuclear plants of the future that will make nuclear generation safe.

Multipurpose fast research reactor (**MBIR**) having modern research infrastructure will allow to create safe Generation IV nuclear power plants. Its unique features will make it possible to develop and improve technologies for two-component nuclear energy systems and closure of NFC.

The modern research infrastructure of the **MBIR** will allow:

• to test new structural and absorbing materials to justify the development of new generation reactors,

• to study promising fuels, fuel elements and develop technologies for closure of nuclear fuel cycle (NFC),

• to produce radioactive isotopes and raw materials for radiopharmaceuticals to solve industrial and medical issues, as well as to accumulate modified materials,

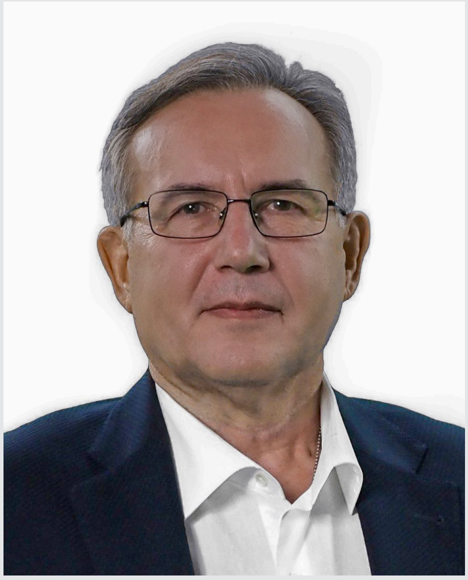
• conduct research using neutron beams in the field of medicine, fundamental and applied physics.

To perform a wide range of studies, the MBIR reactor project provides for autonomous/independent serpentine passages for modeling the operating conditions of active zones with various heat carriers.

On the basis of **MBIR**, Russia is creating the most modern research platform for the whole world. By the time the reactor is put into commercial operation, the International Research Center (**IRC**), the international fast reactors competence center, will start working on its basis, with scientists from all over the world taking part in its activities. Such cooperation will make it possible to participate in international scientific programs, in the implementation of educational programs for the development of competencies, and, as a result, to gain access to a common database of experimental data. The involvement of many participants representing various scientific and technical schools will create a synergistic effect for all project participants.

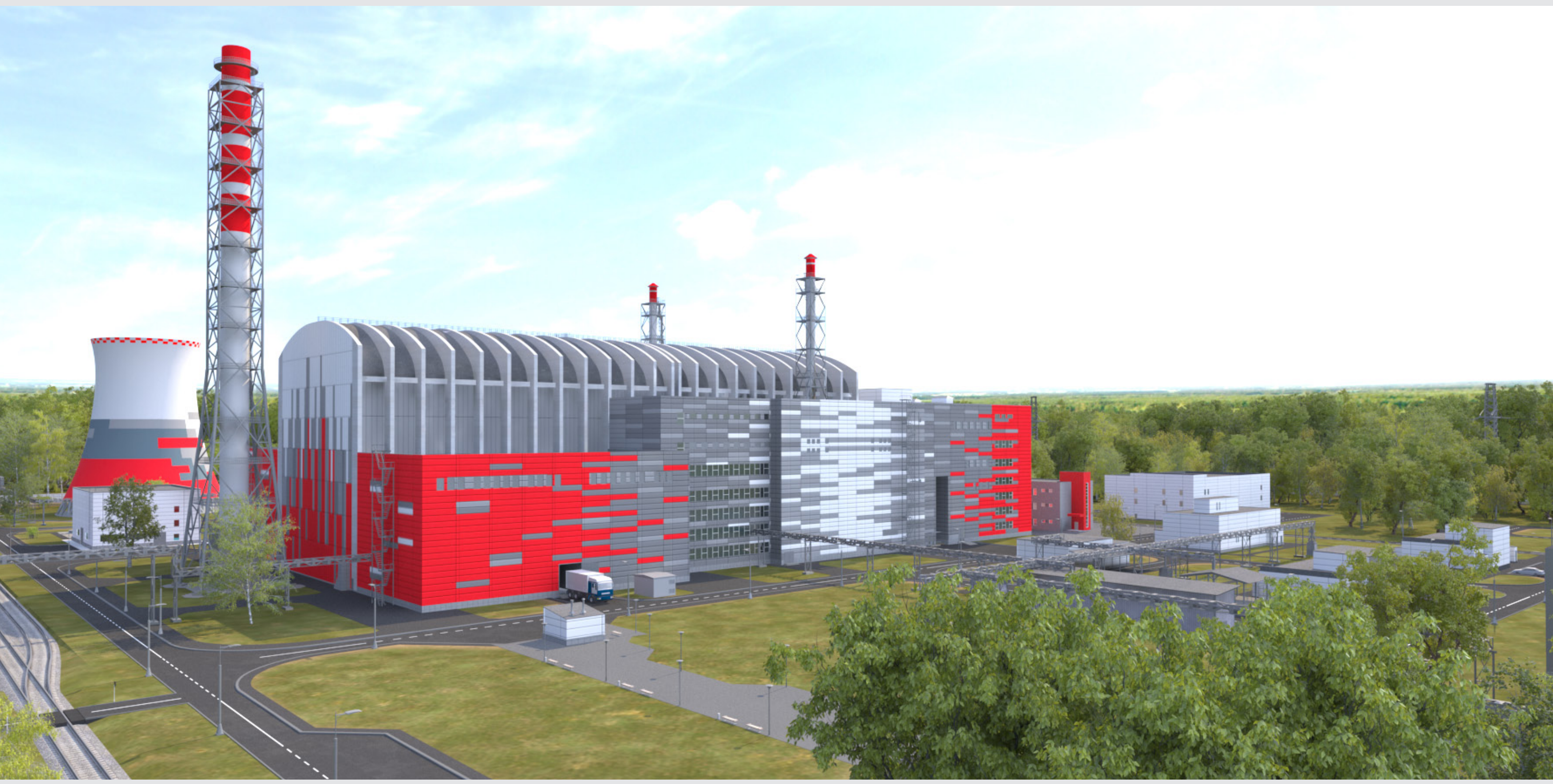
IRC MBIR is a unique platform for strengthening the engineering and scientific personnel potential of the participating country and mastering modern research methods. All the work of the future Center will be correlated very closely with the UN Sustainable Development Goals, the promotion of which is an integral condition of Rosatom's work. The power industry of the future is impossible without CO2-free nuclear technologies that will help solve the issue of climate change.

MBIR will become one of the most popular fast research reactors in the world and the most high-precision one of the research facilities existing and under construction. It will provide the nuclear industry in the current century with a modern and technologically advanced research infrastructure.



Vasily Konstantinov
CEO of IRC MBIR LLC

TABLE OF CONTENTS



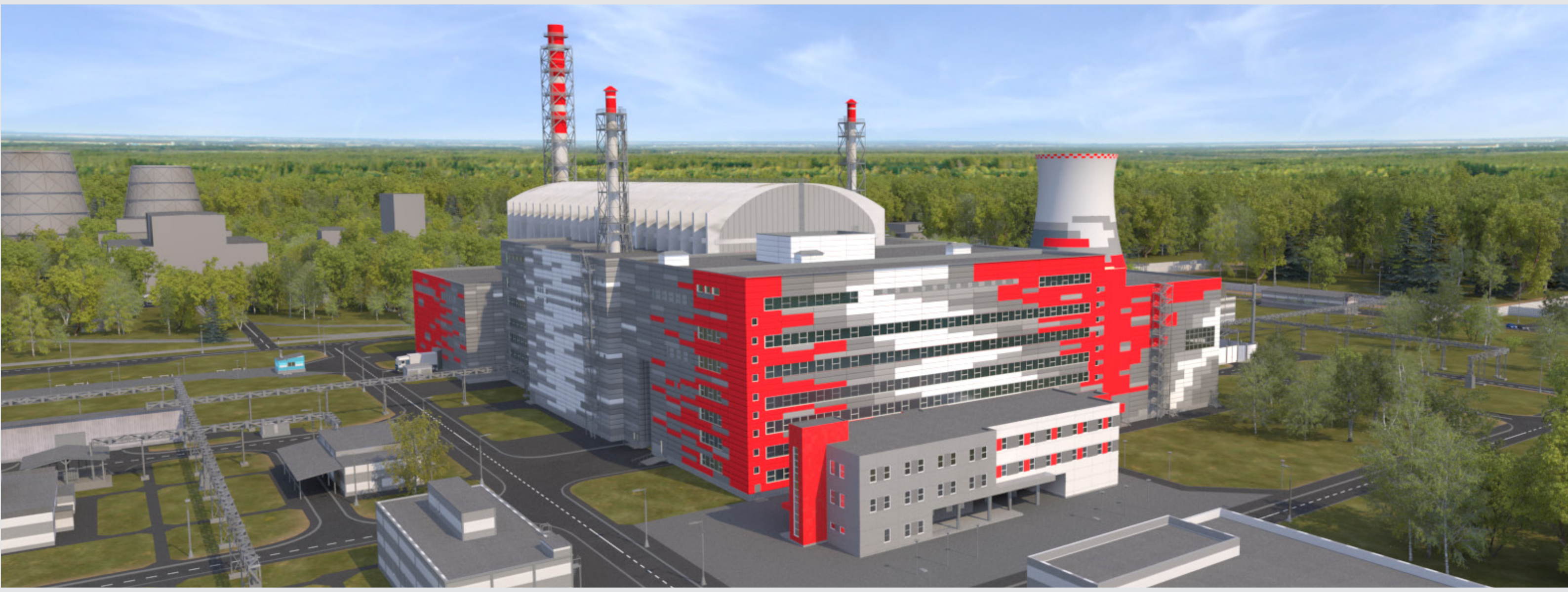
Definitions and abbreviations

AE	- absorber element
AHE	- air heat exchanger
BN	- sodium-cooled fast-neutron reactor
BREST OD-300	- lead-cooled fast-neutron reactor with on-site fuel cycle
CCEJP	- comprehensive computational and experimental justification programme
CLC	- central loop channel
CPS	- control and protection system
CR	- control rods
dpa	- displacement per atom
EFA	- experimental fuel assembly
EMP	- electromagnetic pump
EPS	- emergency protection system
FA	- fuel assembly
FE	- fuel element
FNS	- fusion neutron source
FTP	- Federal Target Programme
GFP	- gaseous fission products
GIF	- Generation IV International Forum
HTRE	- high-temperature radiation embrittlement
ID	- irradiation device
ILF	- independent loop facility
ISLF	- independent sodium-cooled loop facility
LBC	- lead-bismuth coolant
LBCILF	- independent loop facility with lead-bismuth coolant
LC	- loop channel
LF	- loop facility

Definitions and abbreviations

LTRE	- low-temperature radiation embrittlement
MA	- minor actinides
MBIR	- multipurpose research fast reactor
MNU-Pu	- mixed nitride uranium-plutonium fuel
MOX	- mixed oxide (fuel)
MP	- materials science package
MTA	- material test assembly
NP	- nuclear power
NPF	- nuclear power facility
ODS	- oxide dispersion strengthened (steels) / ODS
PBHTMSR	- pebble bed high-temperature molten salt reactors
PIE	- post-irradiation examination
R	- research
R&D	- research and development
REA	- JSC “Rosenergoatom”
r.c.	- (reactor) core
RF	- reactor facility
RI	- refueling interval
RVI	- reactor pressure vessel internals
RW	- radwaste
SM	- structural material
STC	- science and technology council
SVBR	- fast reactors with lead-bismuth coolant
TA	- test assembly
TFA	- test fuel assembly
VVER-SCP	- pressurized water reactor with supercritical coolant parameters

EXPLANATORY MEMORANDUM



In accordance with Russia's strategy of nuclear power development until 2050 and with a view to the year 2100, nuclear power in Russia will develop as a two-component system based on thermal and fast reactors working in a unified closed nuclear fuel cycle.

To justify technologies for closing the fuel cycle at the experimental and engineering levels and to validate the use of structural, fuel and absorbing materials for fast reactors in nuclear power industry, it will be necessary to carry out a programme of experimental studies, including in-core studies. That will be one of the tasks for the research fast-neutron reactor MBIR.

This document is the Programme of advanced scientific research proposed to be performed at the research reactor facility MBIR in the period from 2028 to 2040. It includes description of the RF MBIR's experimental capabilities for the foreign partners participating in the International Research Centre based on the MBIR reactor facility.

1. NATIONAL FAST REACTOR DEVELOPMENT PROGRAMME FOR THE 2020 - 2050 PERIOD.

Russia's strategy of NP development involves integration of fast reactors into the two-component nuclear power system.

1. The near-term stage of 2020 - 2028 reflects the current state of NP when the pilot demonstration BREST OD-300 reactor is being built and the first commercial power unit with BN-1200 reactor is being designed. Reactor tests of innovative materials and core component mock-ups for the Generation 4 nuclear energy systems are carried out at BOR-60.

2. The midterm stage of 2028 - 2040 suggests the completion of construction and the start of operation of the first BN-1200 reactor, as well as the development of the commercial BREST-1200 reactor project (if the lead-cooled fast BREST OD-300 reactor demonstrates its safety and competitiveness). Reactor tests of innovative materials and core component mock-ups for the Generation 4 nuclear energy systems are carried out at MBIR.

3. The long-term stage of 2040 - 2050 suggests building a small series of BN-1200 power units and the power unit with BREST-1200 reactor. Reactor tests of innovative materials and core component mock-ups for advanced nuclear energy systems and for operating maintenance of production prototypes of innovative reactor facilities are carried out at MBIR.

It is expected that the BN-1200 reactor will have been built by 2030, with MOX fuel and a layer of depleted uranium chosen as reference fuel. After the dense nitride fuel technology has been tried out, the core can be converted to new MNU-Pu fuel (presumably after 2035-40).

In the medium term, justification of MNU-Pu fuel both for lead-cooled and sodium-cooled fast reactors is to be completed. To justify the fuel, “A Comprehensive Computational and Experimental Justification Programme (CCEJP) for the Dense Fuel of Fast Reactors” has been developed.

Thus, materials science programmes for these reactors have already been developed.

Besides, in the medium and long term, such innovative projects as SVBR-100 (fuel: uranium oxide, uranium nitride, uranium-233-thorium), BREST-1200 (MNU-Pu fuel), advanced BN-GT (with gas-turbine cycle), research molten salt and thermonuclear reactors can be implemented.

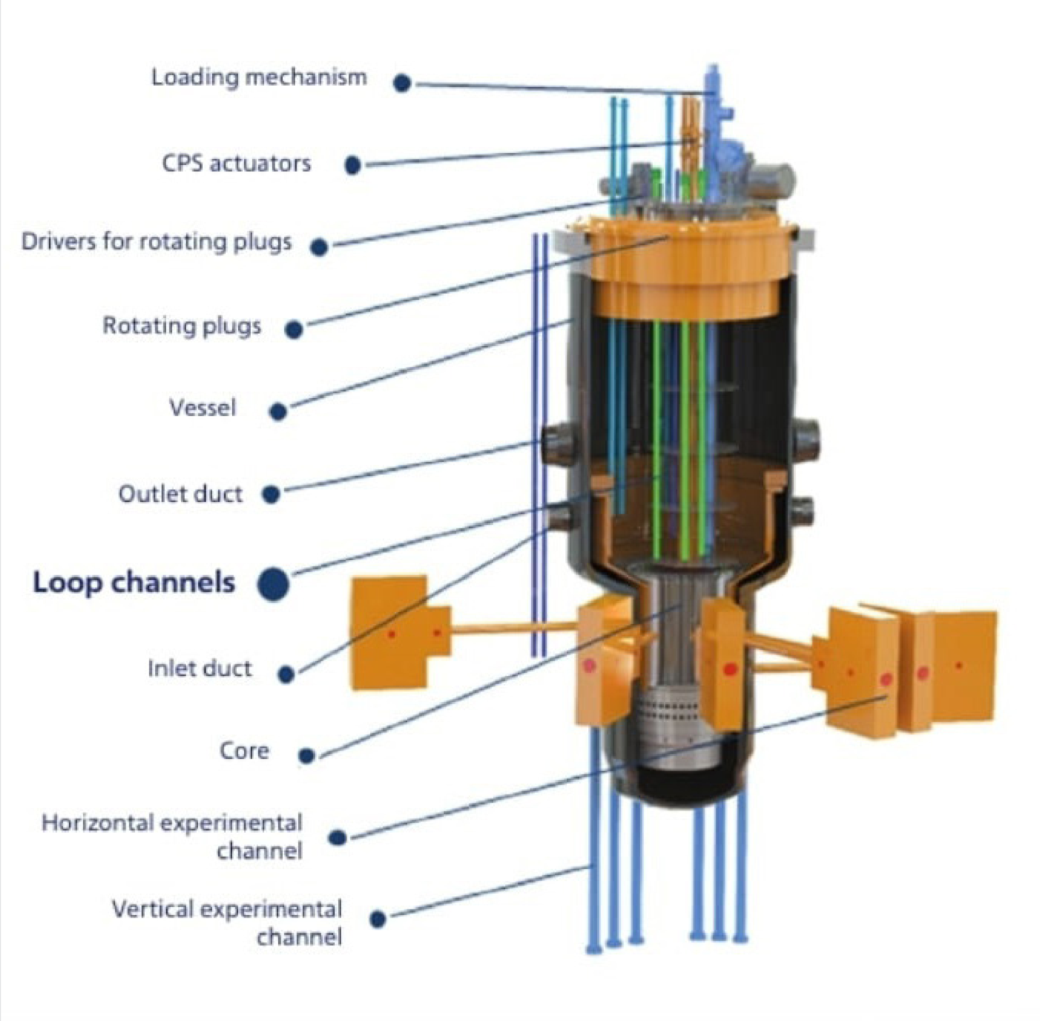
2. RATIONALE BEHIND THE PROGRAMME OF SCIENTIFIC RESEARCH IN SUPPORT OF ADVANCED REACTOR FACILITY PROJECTS.

For sufficient support of the national advanced fast reactor projects and for justification of technologies for closing the nuclear fuel cycle, it is necessary to provide the reactor test facility resource for a long period. In view of this need, the MBIR research reactor is considered as the principal research reactor facility till at least 2080.

It is planned to complete the construction of the multipurpose fast research reactor MBIR by 2027. The reactor is supposed to achieve its first criticality in the same year, with scheduled research work expected to start in 2028.

In accordance with the Concept and the Technical Assignment, one of the main purposes for the multipurpose research fast reactor MBIR is to perform a series of reactor tests of innovative materials for the Generation 4 nuclear energy systems including fast reactors operated in the closed fuel cycle together with thermal reactors, taking in small and medium power thermal reactors. The RF MBIR’s design sets up a wide range of reactor experimental capabilities for conducting experimental studies within the scope of the national nuclear industry development programmes and for fulfilling works commissioned by foreign customers under the auspices of the International Research Centre on the basis of MBIR.

This Programme defines prospective areas of experimental work and research studies at the MBIR research reactor for the midterm period of 2028 - 2040. The Programme has been drawn up with regard to the current state of RIAR's test facilities and strategic goals of the industry, namely:



– operation of the IRC MBIR as the backbone experimental reactor centre for creation of Rosatom's new technological platform, which will make it possible to conduct a wide range of reactor tests in order to develop and implement next-generation nuclear energy systems;

– support for the development of spent nuclear fuel reprocessing technologies for closing the nuclear fuel cycle and technologies for disposal of radioactive and nuclear materials at the new multifunctional radiochemical complex under construction;

– support for the development of methods for regenerated fuel fabrication, fuel element fabrication and for the development of control methods, including control of fuel assembly fabrication;

– support for scientific and technological activities in the field of reactor materials science by means of research reactors, post-irradiation examination of reactor materials and core components of innovative nuclear reactors;

– support for the development of new technologies for production of radionuclides to be used as various sources of ionizing radiation for industrial and medical purposes;

– support for the introduction of results received from the research and development of nuclear nuclear fuel, structural and absorbing materials into innovative products and technologies of fuel elements and fuel assemblies, absorbing elements, CPS rods, radionuclide-based sources and medications.

Construction of safe and economically viable G4 nuclear power units will demand a large amount of in-core examination of new materials and reactor component mock-ups in special experimental devices and loops equipped with modern control and management tools. Structural materials are supposed to guarantee safe operation of core components at damaging doses of up to 170 dpa, at the very least, and even higher values in the future. Reactor safety has to be justified experimentally in transient, power cycling and emergency reactor operation modes. During the implementation of the "Proryv" ("Breakthrough") project, tests and studies of new structural materials, various options of FE designs and currently developed nitride uranium-plutonium fuel will be needed for justification of their safe operation at high burnup levels (up to 12% h.a. and more) in BREST OD-300 and BN-1200 reactors. These innovative projects are planned to be brought to the stage of demonstration and pilot facilities by 2030. It means that most reactor research in support of these projects will be conducted within the period from 2028 to 2035. Beyond this period, a variety of experimental programmes may be launched to improve and maintain the operation of prototypes of innovative reactor facilities.

During the final stage of MBIR's construction in 2025 - 2027, i.e.at the commissioning stage, comprehensive research will be required to study reactor neutronics and thermohydraulics parameters. Test procedures will be transferred from BOR-60 to the MBIR reactor. New experimental devices and improved methods will be developed for testing materials and component mock-ups of advanced reactors in instrumented and non-insrtumented EFAs, ampoule devices and independent loop channels. The continuity condition, applying to experimental studies started at BOR-60, shall be fulfilled. The experimental studies to be continued at MBIR are scheduled to begin in 2028. When experimental devices and test procedures of the MBIR reactor are developed, it is necessary to provide for the possibility to conduct research in support of all the advanced G4 reactors. Main types of G4 reactors and their key parameters based on recommendations drawn up by the Generation IV International Expert Committee are shown in ***Table 2.1.***

MBIR REACTOR DESIGN FEATURES CIRCUIT

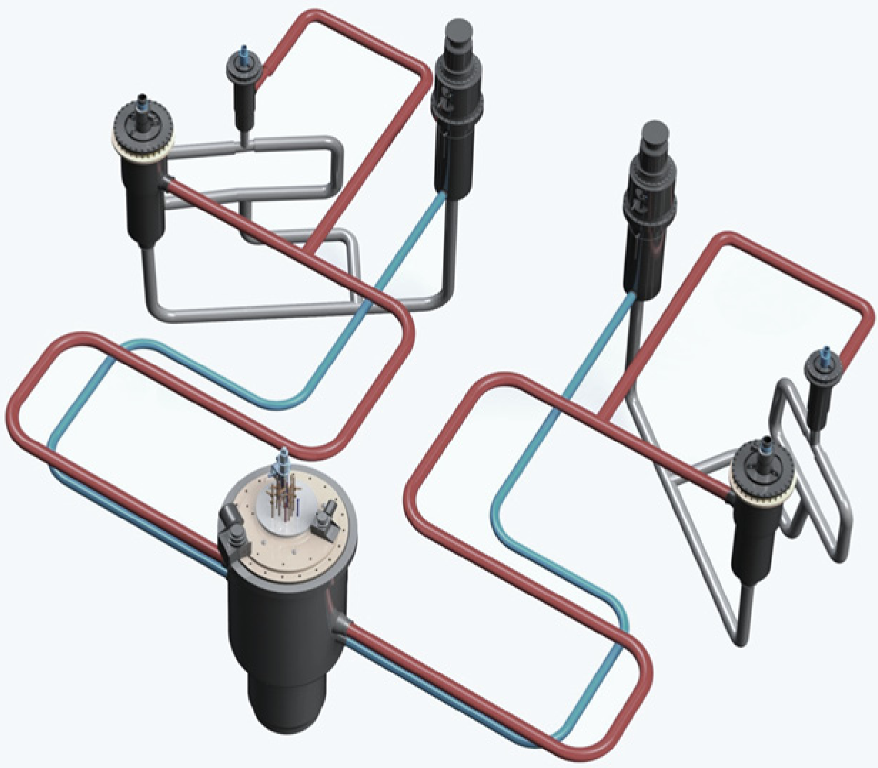


***Table 2.1** – Proposed materials for G4 reactors*[1]

Reactor	*Coolant*	*Maximum temperature, °C*	*Materials of core components*		
			Fuel	*Cladding*	*Absorber*
Gas-cooled fast	helium	850	(U, Pu)C/SiC composite ceramic, fuel particles with ceramic coating	SiC [2], ceramic	• high-temperature ceramics (metal carbides or borides) based on boron with a content of ^{10}B isotope up to 95 % - $^{10}B_4C$, $W^{10}B_2$, $Hf^{10}B$ [2] • $Dy_2O_3 \cdot HfO_2$, $Dy_2O_3 \cdot HfO_2$ + B_4C[2]
LFR (fast)	lead or lead-bismuth	620	(U-Pu) (U, Pu)N (U, Pu) N+ actinides UO_2, UN	– ferritic-martensitic steel (9...12 % Cr) – ODS steels	
SFR (fast)	sodium	600	– U-Pu-Zr – U-Pu-Zr + actinides – $(U, Pu)O_2$ – $(U, Pu)O_2$ + actinides	– improved austenitic steels [2] – ferritic-martensitic steel (9...12 % Cr) – ODS – coated vanadium alloys [2]	
SCWR fast (thermal)	supercritical water	550 (P=25 MPa)	– $(U, Pu)O_2$ – dispersion – (UO_2)	– Fe Ni Cr Ti – ODS – Inconel 690, 625, 718 – ceramics	
MCR epithermal	molten salt	700	salt	–	–
VHTR thermal	helium	1000	TRISO UOC in a graphite matrix with ZrC coating	– ZrC-coated graphite – $^{11}B^{15}N$ [2], – $^{11}B^{15}N$ + $^{11}B_4C$ [2]	pyrocarbon-impregnated boron carbide[2]

[1] A Technology Roadmap for Generation IV Nuclear Energy System//Issued by the US DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, GIF- 002-00, December 2002,

[2] – proposed by the Russian side.

After the MBIR reactor (and its in-pile experimental devices, i.e. instrumented and non-instrumented channels) is commissioned and its experimental operation is carried out, it is planned, in accordance with mutually consistent specific research programmes of the Customers, to begin the construction of independent high-power loop facilities in the period from 2028 to 2030. Such LFs with different types of coolant will be used for testing FA mock-ups in the simulated stationary, transient and emergency operation modes.

3. EXPERIMENTAL EQUIPMENT OF THE MBIR REACTOR

3.1. The main areas of research at MBIR

The following are the main areas where the MBIR reactor will be used:

– mass reactor tests and studies on advanced structural materials at different temperatures (in the range of 350 to 1800 °C) in different coolants for the next-generation nuclear power facilities (NPFs) including thermonuclear reactors;

– mass reactor tests of fuel elements prototypes based on advanced fuels of the next-generation NPFs;

– reactor experiments aimed at solving the problems of the closed fuel cycle including transmutation of minor actinides and radwaste reduction;

– studies on FE performance and justification of their operability for various NPFs in transient, power cycling and emergency operation modes at loop facilities with different coolant types;

– radionuclide production;

– creation of neutron beams for fundamental, applied and medical purposes;

– production of heat and electrical energy.

3.2. Requirements for the in-pile experimental devices

The chief purpose of MBIR is high-dose irradiation of structural and fuel materials in the reactor core and radial reflector. *Table 3.1* shows basic characteristics of the four types of reactor irradiation cells and their number.

LOOP CHANNELS

Table 3.1 *– Quantitative characteristics of MBIR's irradiation cells*

Reactor cell type	Number
1 - 128-mm turnkey cells for the channels of in-pile loop facilities and channels for ampoule instrumented tests	3
2 - 72-mm turnkey cells for ampoule channels and in-pile loop facilities of instrumented test channels.	3
3 - 72-mm turnkey cells for non-instrumented tests of structural materials and for production of isotopes in the core	~ 27
4 - 72-mm turnkey cells for irradiation of materials and production of radionuclides in the radial reflector	Not limited

In-pile loop facilities can be installed into cells of Types 1 and 2. They need integral layout of equipment inside the channel and serve to test FE and absorbing rod mock-ups and SM samples in stationary, transient and emergency operation modes in different working fluids (sodium, lead, lead-bismuth, helium, etc.). Preset thermodynamic parameters of the coolant are maintained due to natural or forced circulation.

Cells of Types 1 and 2 for instrumented tests, which have their own leads of service lines (instrumented and/or power lines, gas lines of sample loading systems, etc.) out of the irradiation zone, allow testing of structural and fuel materials in a given fluid, with irradiation temperature measured and controlled and mechanical loading of samples provided. These cells are fixed in the core.

Non-instrumented tests of structural materials can be done in cells of Type 3 in dismountable material test assemblies and ampoule devices with different coolant types. The option of on-line data readouts of irradiation parameters is not envisaged, but it is possible to perform intermediate materials science research and replace samples. Cells of this type are not fixed in the core, their location is determined according to the required neutron flux values in regular FA cells. The cells can also be used for dismountable "isotopic" assemblies with initial materials for production of isotopes.

Cells for design fuel assemblies can be used for installing dismountable experimental FAs with different fuel types and sodium coolant to study the fuel performance when irradiated.

Dismountable MTAs and ampoule devices with different coolants as well as isotopic assemblies will be installed into cells of Type 4 located in the radial reflector. When needed, special assemblies with moderating materials for degrading the spectrum can be installed in the radial reflector so that the required irradiation conditions are satisfied.

3.3 Requirements of the out-of-pile experimental devices

To provide irradiation experiments, the reactor facility is designed with the following necessary equipment and instrumentation:

- pick-and-place mechanisms and machines for loading and reloading experimental devices and loop facilities;

- stands for washing and storing various kinds of experimental devices and loop facilities with controlled temperature conditions;

- research stands in the spent fuel pool for detecting fuel cladding failure and for carrying out tests simulating conditions of deteriorating heat dissipation.

- research safety hot cells with equipment and facilities for assembly and disassembly of irradiation devices, as well as for initial investigations (geometrical measurements of irradiated samples, filming and photography, gamma scanning, measurements of corrosion thickness).

When using reactor irradiation for applied purposes outside the reactor vessel, it is planned to have:

- horizontal and tangential channels to get neutron beams out of the reactor for use in nuclear medicine, neutron radiography and tomography;

- vertical experimental channels for irradiation tests of products whose irradiation resistance is moderate and for neutron transmutation of materials (neutron-transmutation doping of silicon);

4. DESIGN CAPABILITIES FOR TESTING MATERIALS IN THE MBIR REACTOR.

The reactor design provides for 20 irradiation cells including 1 central loop channel (occupying 7 cells), 2 peripheral loop channels, 14 non-instrumented channels, 3 cells to place assemblies for production of radionuclides. Altogether, 24 cells (17 + 7 CLC cells) with various irradiation assemblies are allowed within the core boundaries. The effective volume of a single assembly like that is ~ 2280cm^3 (the inner size of a turnkey cell is 6.92 cm; the FE height is 55 cm).

The design loading map of MBIR is presented in ***Figure 4.1.***

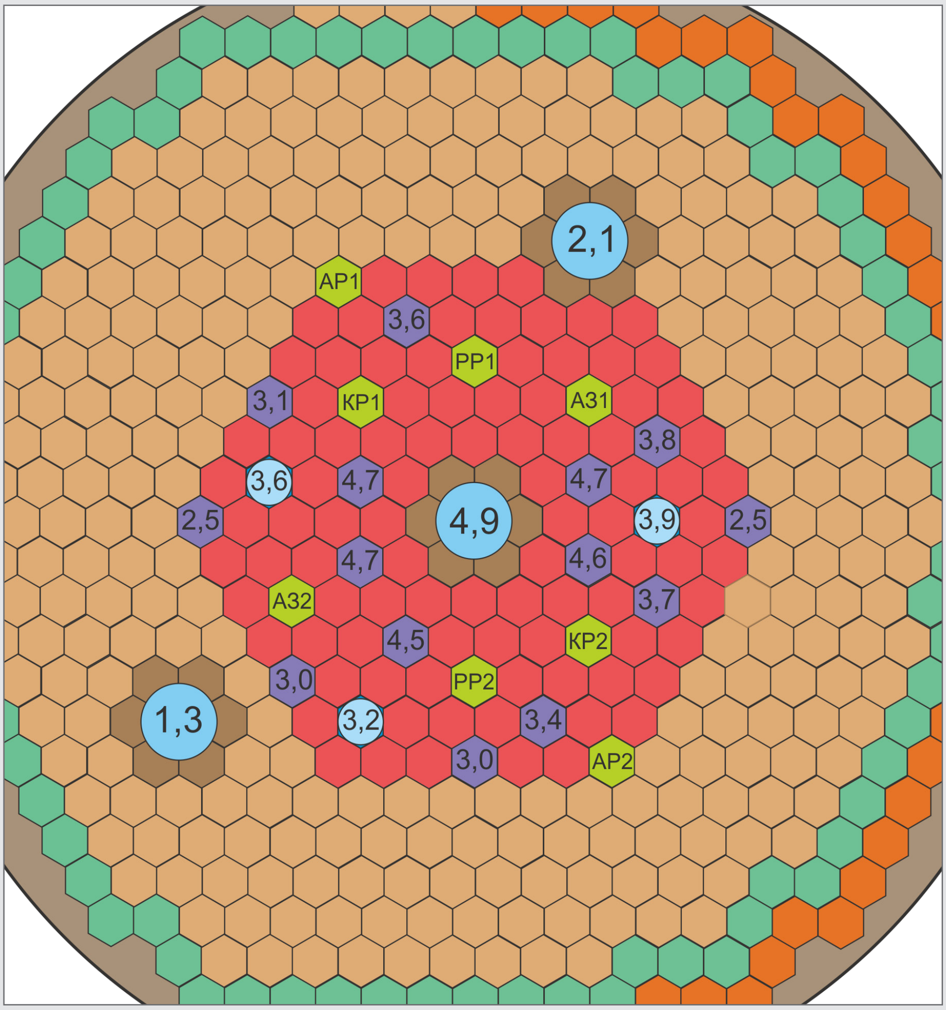


Figure 4.1 – Design loading map of the MBIR reactor

The core is considered to be in the steady-state mode of regular reloadings. FAs with different burnup levels are located in it at the same time. Maximum heat production rate is achieved in fresh FAs. In the course of fuel burnup, heat production rate in these FAs is decreasing. The distribution of the maximum linear heat production rate averaged over the fuel assembly is shown in Figure 4.2. It is different for different FAs, depending on their fuel burnup.

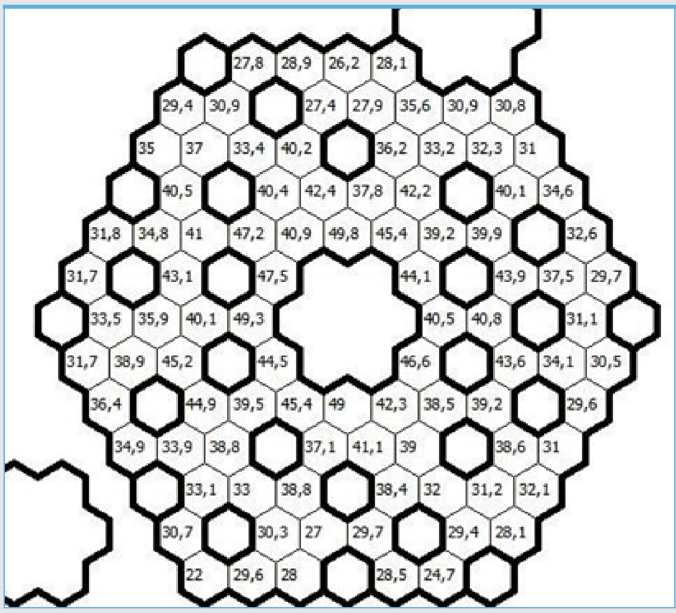


Figure 4.2 – Maximum linear heat production rate in the design option

Neutron flux level in irradiation cells (averaged over the refuelling interval) is shown in Figure 4.3a. Figure 4.3b shows the dpa rate. At present, cladding steel ЧС-68 х.д. brought to a technological level remains operational at up to 85 dpa, advanced steel ЭК-164 ИД х.д. remains operational at up to 110 dpa. Such a damaging dose can be reached by irradiation in MBIR during 7 RIs of 100 eff. days, or ~ 3 years.

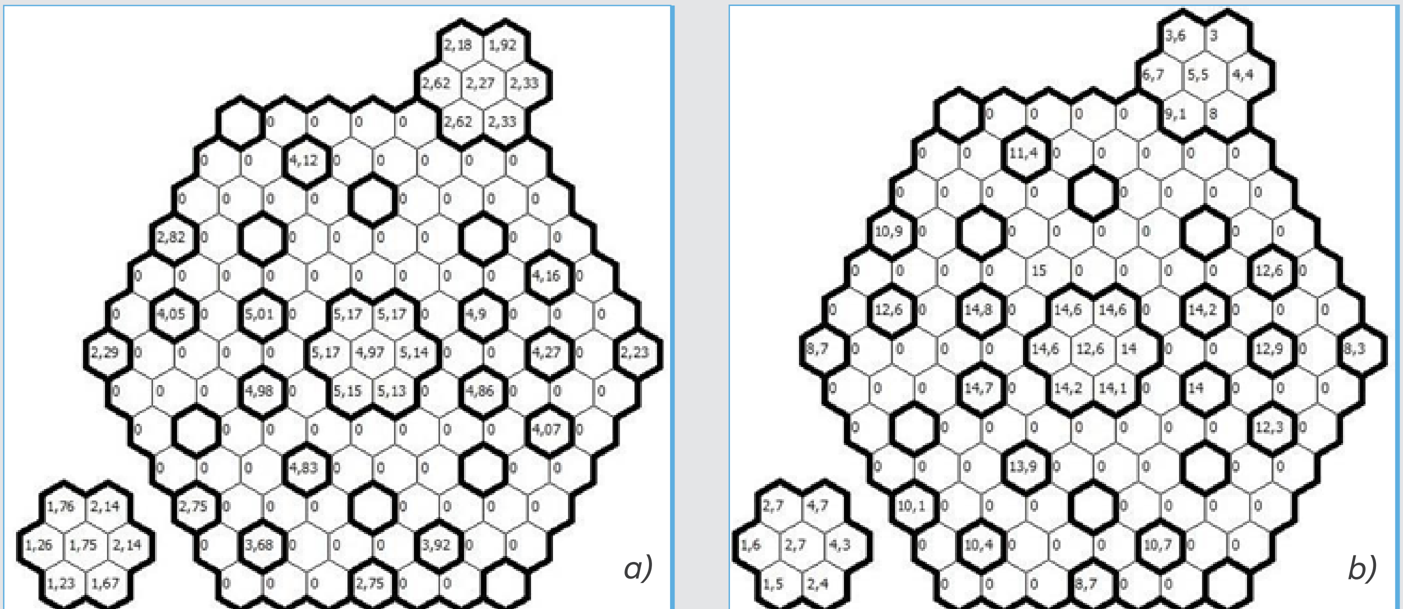


Figure 4.3 – Neutron flux in irradiation cells, 10^{15}n/cm² s (a) and dpa rate (displacements per atom during 100 eff. days) in irradiation cells (b) for austenitic cladding steels (dpa rate for ferritic-martensitic cladding steels of ЭП-450 type is by 3% less)

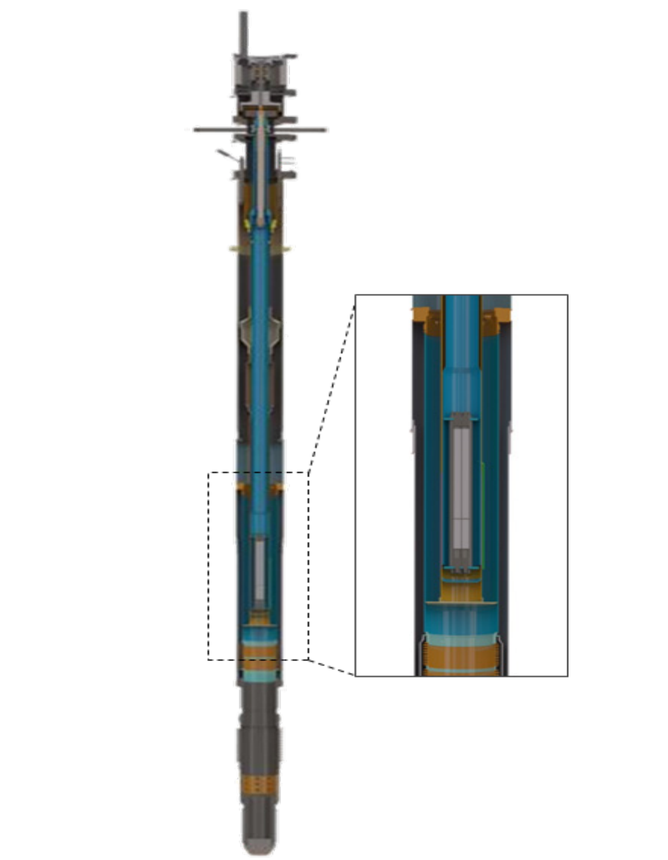
Loops can be used for carrying out experimental programmes including tests of FEs with advanced fuel types and various simulations in transient, power cycling and emergency modes on condition that these experiments are consistent with experimental programmes in other channels.

5. ESTIMATIONS OF IRRADIATION POTENTIAL FOR TESTING MATERIALS IN MBIR AT THE INITIAL STAGE OF OPERATION (2028 - 2033).

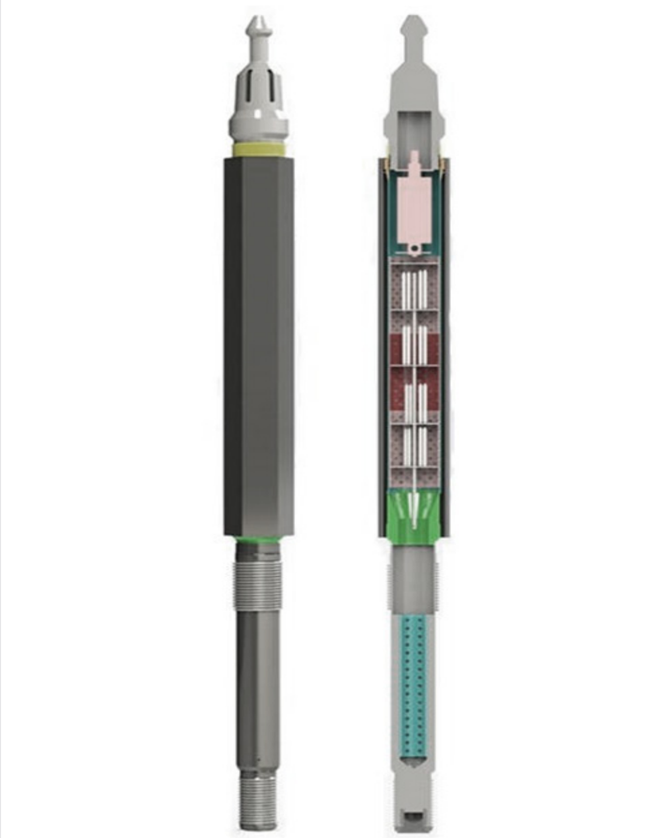
Peculiarity of the initial stage of MBIR's operation is the absence of loop channels, which are complex engineering structures. There is a probability that they will not be ready for operation at the initial stage. However, the cells where these channels are located have to be filled. In this respect, CLC occupying 7 cells in the core centre is of special importance. It is not advisable to use all these cells for irradiation assemblies because of their strong impact on each other. To eliminate this impact, it is proposed to place 3 additional irradiation assemblies and 4 regular fuel assemblies into CLC cells.

MBIR's irradiation volume of the initial stage increases by 4 cells due to the partially used CLC volume, with the maximum damaging dose rate of ~ 16 dpa per RI (100 eff. days). The damaging dose rate in the remaining 17 irradiation assemblies (provided for by design) ranges from 15,3 to 6,7 dpa per RI. The volume of each assembly like that is 2280 cm³. The total number of irradiation cells at the initial stage is 21. It does not affect the neutron flux and the dpa rate values. The reactor's full irradiation potential can be estimated at around 1200 dpa×l/year[1].

IRRADIATION TEST RIG (TR)



MATERIALS TEST RIG (MTR)



[1] ***dpa×l/year*** *– conventional non-system unit that characterizes experimental capabilities of the core to accumulate the annual damaging dose rate (1 displacement per atom) in austenitic steel of ЭК-164 type in an irradiation cell whose volume is 1 liter. Multiplied by the volume of all the cells, this value shows integral irradiation capabilities of the reactor.*

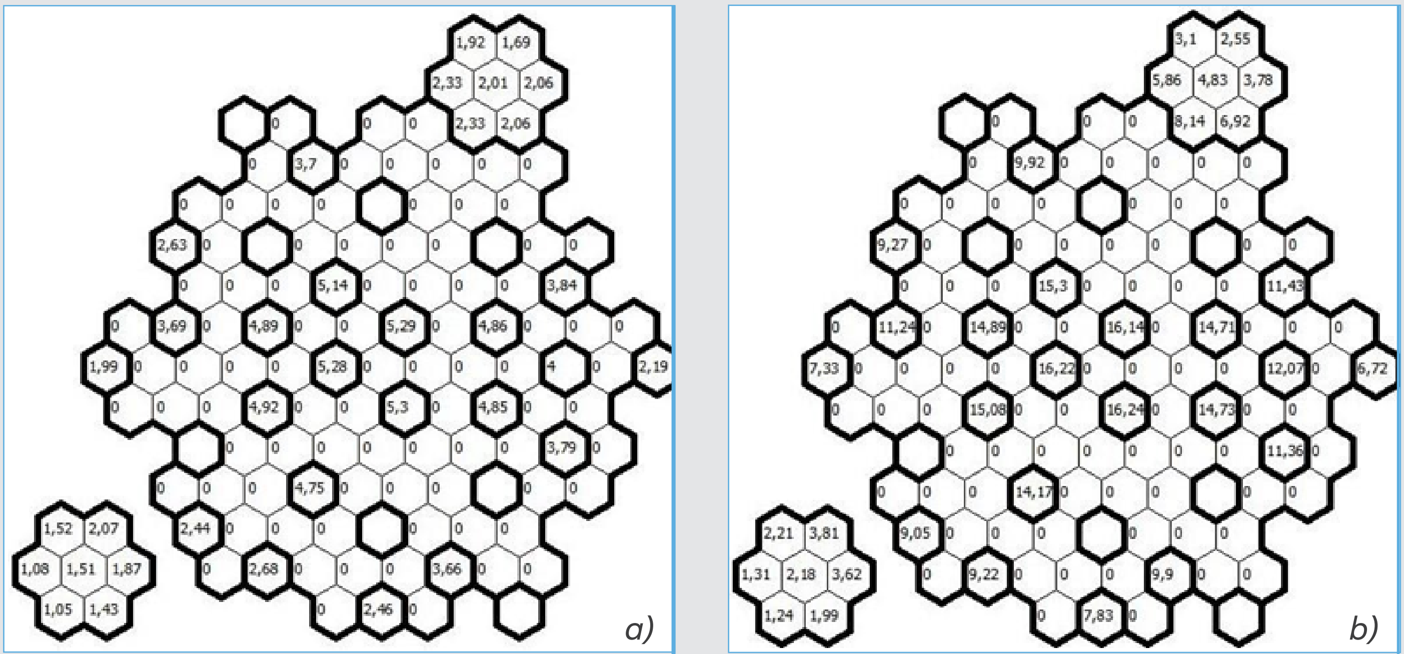


Figure 5.3 – Neutron flux in the irradiation cells of the modified core at the initial stage, 10^{15} n/cm² (a) and damaging dose (displacements per atom during 100 eff. days) in the irradiation cells (b) of the modified core at the initial stage.

6. PROGRAMME OF WORK AT THE INITIAL STAGE OF OPERATION FOR THE 2028 - 2033 PERIOD FROM IN THE INTERESTS OF RUSSIAN AND FOREIGN CUSTOMERS, INCLUDING STUDIES TO BE TRANSFERRED FROM BOR-60 TO MBIR

After MBIR's power start-up and coming into steady-state nominal power operation, it is planned to conduct experiments aimed at trying out various engineering procedures and to carry out comprehensive studies on neutronics and thermohydraulics parameters of experimental devices in order to continue experimental studies initiated in the BOR-60 reactor.

Independent loop facilities are going to be put into operation 1-2 years after the start of MBIR's operation.

6.1. Materials science package (MP) of the MBIR reactor.

Typical design of a materials science package for irradiation of materials and products in the core and the radial reflector of the MBIR reactor is presented in Figure 6.1. There are 4 modifications of the package, depending on the irradiation environment. It was developed based on the experience gained with similar products which were used in BOR-60. That makes it possible to "transfer" tests of materials and products to the MBIR reactor after the decommissioning of BOR-60 without major engineering and technological constraints.

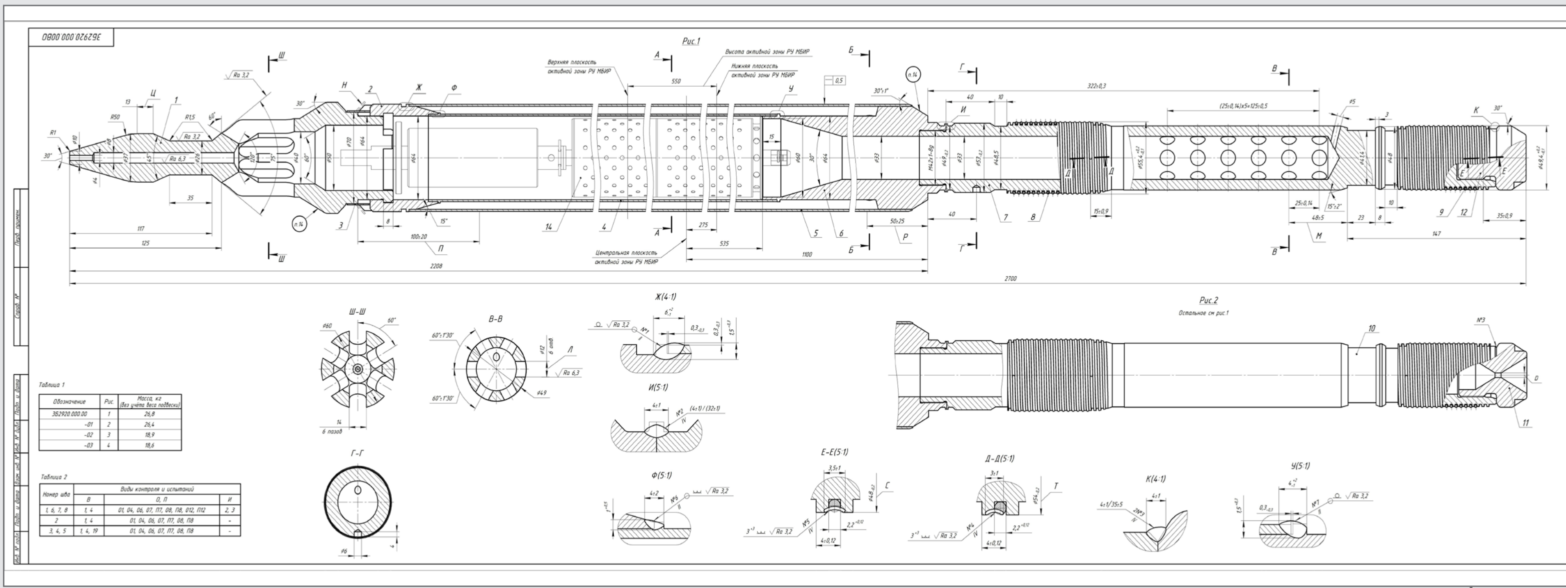


Figure 6.1 - Typical design of MBIR's materials science package

6.2. The core of the MBIR reactor.

The cross section of the MBIR reactor core with designated cells is shown in Figure 6.2. MP locations are designated by the blue colour. As there are no loop facilities at the initial stage, their places will be taken by FAs and materials science packages. 21 cells of the MBIR reactor core are going to be used at the initial stage.

Table 6.1. shows parameters of experimental cells and their intended use at the initial stage. The layout of the core is shown in Figure 6.2. Characteristics of irradiation devices for the core are given in the explanatory memorandum on the materials science package for testing materials in MBIR's experimental cells

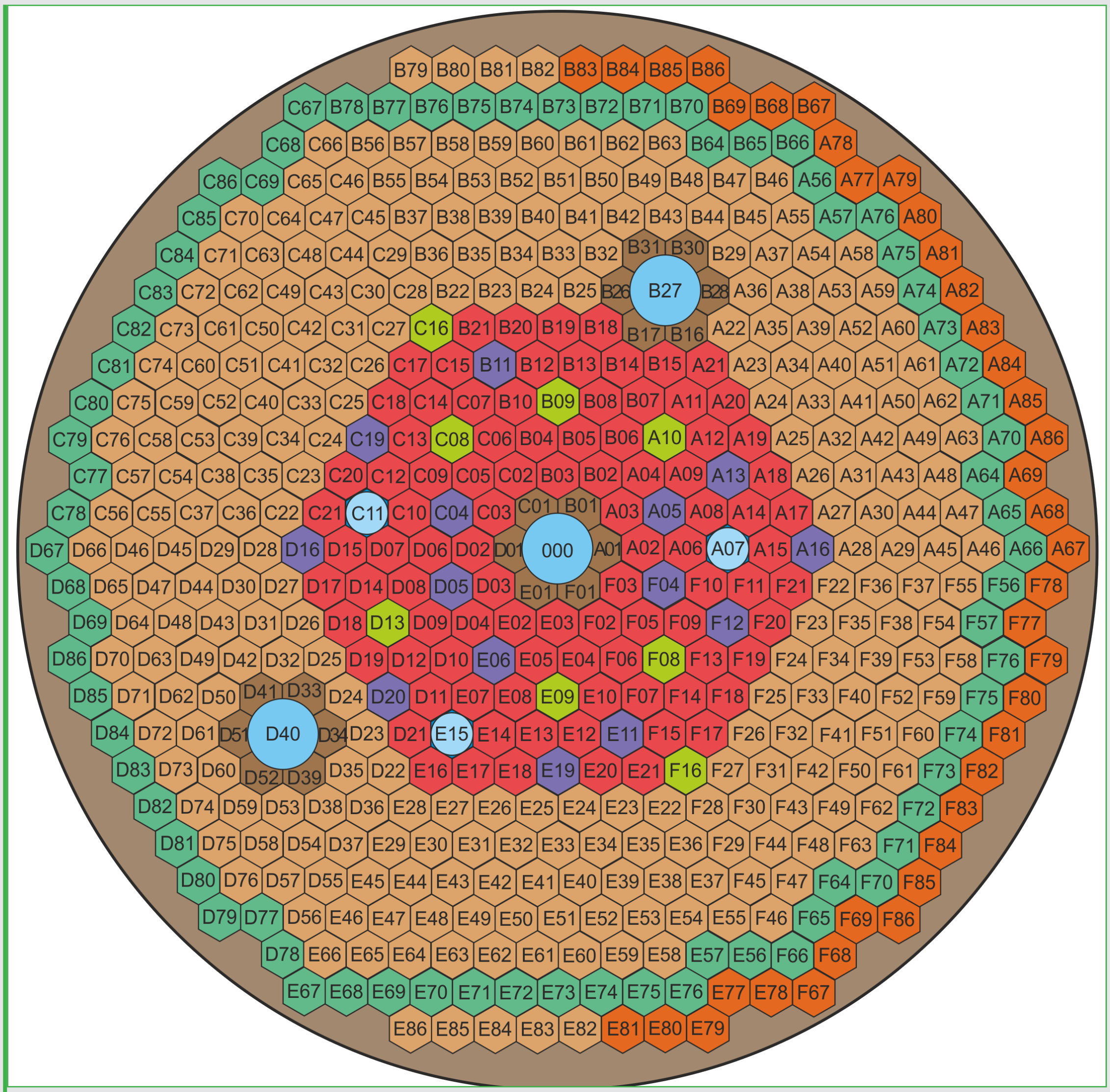


Figure 6.2 - Layout of the MBIR core

- Package of in-pile storage assemblies
- Lateral blanket assembly
- SFA
- MTA
- Fuel assemblies
- Experimental devices
- CPS drives

Table 6.1 - Parameters of experimental cells and their intended use at the initial stage of MBIR operation

№	Experimental cell parameter	Neutron flux density, 10^{15} cm^{-2} s^{-1}	Damaging dose, dpa/year	Intended use
1	000, 128-mm turnkey central cell	4,88	32,5	Tryout of high-dose SM irradiation procedures and independent in-pile loop test procedures
2	B27, 128-mm turnkey peripheral cell	2,07	13,8	Tryout of independent in-pile loop test procedures
3	D40, 128-mm turnkey peripheral cell	1,27	8,5	Tryout of independent in-pile loop test procedures; production of isotopes
4	A07, 72-mm turnkey core cell	3,95	26,3	EFA with (UPu)N-based fuel for BN-1200 reactors
5	C11, 72-mm turnkey core cell	3,62	24,1	EFA with (UPu)N-based fuel for BN-1200 reactors
6	E11, 72-mm turnkey core cell	3,44	22,9	EFA with fuel based on U-Zr alloys
7	E19, 72-mm turnkey core cell	2,97	19,8	EFA with fuel based on U-Zr alloys

7. MAIN AREAS OF EXPERIMENTAL WORK AT MBIR

R&D	Implementation terms, year	Objective	Expected result
7.1 Research into characteristics of advanced fuel materials			
7.1.1 Studies of fuel swelling and outgassing (UPu) N, depending on the composition, porosity, temperature, burnup.	2028 - 2040	To justify reliability and safety of (UPu) N-based fuel for BREST-300, BN-1200 reactors.	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, FE and FA designs.
7.1.2 In-core studies into (UPu)N fuel creep, depending on the composition, porosity, temperature, burnup.	2028 - 2040	To justify reliability and safety of (UPu) N-based fuel for BREST-300, BN-1200 reactors.	To enlarge databases of the properties of nuclear fuel irradiated in different conditions. To make appropriate recommendations.
7.1.3 Studies into heat transfer of (UPu) N fuel irradiated to different burnup levels, depending on the composition, porosity, temperature, burnup.	2028 - 2040	To justify FE operability for BREST-300, BN-1200 reactors.	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.
7.1.4 Reactor tests of regenerated fuel of different degrees of purification from fission products (from 108 to 102 times).	2028 - 2040	To support studies dealing with the development of fuel regeneration technologies.	To create safe and efficient nuclear fuel reprocessing technologies. To improve regenerated fuel performance characteristics during its fabrication process.

R&D	Implementation terms, year	Objective	Expected result
7.1.5 Studies into fuel elements at high burnup levels, with different (UPu)N fuel types and fabrication technologies.	2033 - 2040	To justify MNU-Pu fuel burnup of 15÷20 % h.a. in fast-neutron reactors BREST-300, BN-1200. In future, to develop metal fuelled reactors.	To improve NPP performance characteristics
7.1.6 In-core studies into composite ceramic (U,Pu)C -based fuel characteristics (outgassing, heat transfer, swelling, creep)	2028 – 2040	Reactor tests of new types of fuel for advanced nuclear reactors.	To get experimental data on the properties of nuclear fuel irradiated in different conditions for its certification and for computer code development. To create new fuel materials for high-temperature reactors.
7.1.7 Studies into fuel elements with composite ceramic (U, Pu)C -based fuel and SiC ceramic coating.	2028 – 2040	Reactor tests of new fuel and cladding materials and FE designs for high-temperature reactors. To justify FE operability for helium-cooled reactors.	To get experimental data on the properties of new materials and FE designs in different irradiation conditions for their certification and computer code development for high-temperature reactors. To create new FE designs for high-temperature reactors.

R&D	Implementation terms, year	Objective	Expected result
7.1.8 Tests of nuclear MOX fuel with nano additives (20÷40 nm UO2) and a coarse-grain structure (35÷40 µm) with controlled porosity.	2028 - 2040	To justify FE operability.	To increase fuel burnup.
7.1.9 In-core studies into characteristics (outgassing, heat transfer, swelling, creep) of dense metal fuel based on alloys (U-Zr, U-Mo).	2028 – 2040	Reactor tests of dense metal fuel based on alloys (U-Zr, U-Mo) and designs of the fuel elements based on these alloys for sodium-cooled fast-neutron BN-1200 reactors.	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, FE and FA designs. To create new fuel compositions and FE designs for sodium-cooled fast reactors
7.1.10 Tests of experimental fuel elements of different designs, with dense metal fuel based on alloys (U-Zr, U-Mo).	2028 – 2040	Reactor tests of different designs of the fuel elements based on dense metal fuel (U-Zr, U-Mo) for sodium-cooled fast-neutron BN-1200 reactors.	To make recommendations with regard to improvement of fuel fabrication technologies, FE and FA designs. To create new fuel compositions and FE designs for sodium-cooled fast reactors.

R&D	Implementation terms, year	Objective	Expected result
7.1.11 Tests of fuel elements with different types of dense fuel containing minor actinides.	2028-2033	To justify the operability of products that ensure burning of long-lived and high-level radionuclides.	To create a database of the new nuclear fuel properties. To recycle long-lived and high-level radionuclides.
7.2 Tests of fuel elements with advanced fuel types in transient, power cycling and emergency modes			
7.2.1 Studies into behaviour of (UPu) O2 containing fuel elements in transient modes.	2030-2040	To choose and justify reactor operating modes, to get data for improving FE design and fabrication technology with the aim of achieving high burnups and conforming to the daily power control mode of plant operation.	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, FE and FA designs

R&D	Implementation terms, year	Objective	Expected result
7.2.2 Studies into behaviour of (UPu) N containing fuel elements in the simulated operating conditions of degraded heat removal (reduction and loss of the coolant flow, growth of the coolant temperature).	2033-2040	To justify safe reactor operation, to get data for improving FE design and fabrication technology in the operating conditions of degraded heat removal (reduction and loss of the coolant flow, growth of the coolant temperature, etc.).	To get experimental data on the properties of nuclear fuel in the operating conditions of degraded heat removal (reduction and loss of the coolant flow, growth of the coolant temperature) for its certification and for computer code development. To make recommendations with regard to improvement of fuel fabrication technologies, FE and FA designs.
7.2.3 Studies into behaviour of (UPu) N containing fuel elements in the simulated operating conditions of power ramp due to accidental insertion of positive reactivity.	2033-2040	To justify safe reactor operation in the operating conditions of power ramp due to accidental insertion of positive reactivity for certification of the fuel and development of computer codes.	To get experimental data on the properties of nuclear fuel irradiated in the operating conditions of power ramp due to accidental insertion of positive reactivity for its certification and for computer code development. To make recommendations with regard to improvement of fuel fabrication technologies, FE and FA designs.

R&D	Implementation terms, year	Objective	Expected result
7.2.4 Studies into behaviour of FEs containing dense metal fuel based on (U-Zr, U-Mo) alloys in transient conditions.	2033-2040	To choose and justify reactor operation modes, to get data for improving FE design and fabrication technology with the aim of achieving high burnups and conforming to the daily power control mode of plant operation.	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, FE and FA designs.
7.2.5 Studies into behaviour of FEs containing dense metal fuel based on (U-Zr, U-Mo) alloys in the simulated operating conditions of degraded heat removal (reduction and loss of the coolant flow, growth of the coolant temperature, etc.).	2033-2040	To justify safe reactor operation, to get data for improving FE design and fabrication technology in the operating conditions of degraded heat removal (reduction and loss of the coolant flow, growth of the coolant temperature).	To get experimental data on the properties of nuclear fuel in the operating conditions of degraded heat removal (reduction and loss of the coolant flow, growth of the coolant temperature) for its certification and for computer code development. To make recommendations with regard to improvement of fuel fabrication technologies, FE and FA designs.
7.2.6 Studies into behaviour of FEs containing dense metal fuel based on (U-Zr, U-Mo, etc.) alloys in the simulated operating conditions of power ramp due to accidental insertion of positive reactivity.	2033-2040	To justify safe reactor operation in the operating conditions of power ramp due to accidental insertion of positive reactivity for certification of the fuel and development of computer codes.	To get experimental data on the properties of nuclear fuel irradiated in the operating conditions of power ramp due to accidental insertion of positive reactivity for its certification and for computer code development. To make recommendations with regard to improvement of fuel fabrication technologies, FE and FA designs.

R&D	Implementation terms, year	Objective	Expected result
7.2.7 Studies into behaviour of FEs containing composite ceramic (U, Pu)C-based fuel with SiC ceramic coating in transient conditions.	2033-2040	To choose and justify reactor operation modes, to get data for improving FE design and fabrication technology with the aim of achieving high burnups and conforming to the daily power control mode of plant operation.	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, FE and FA designs.
7.2.8 Studies into behaviour of FEs containing composite ceramic (U, Pu)C-based fuel with SiC ceramic coating in the simulated operating conditions of degraded heat removal (reduction and loss of the coolant flow, growth of the coolant temperature).	2033-2040	To justify safe reactor operation, to get data for improving FE design and fabrication technology in the operating conditions of degraded heat removal (reduction and loss of the coolant flow, growth of the coolant temperature, etc.).	To get experimental data on the properties of nuclear fuel in the operating conditions of degraded heat removal (reduction and loss of the coolant flow, growth of the coolant temperature) for its certification and for computer code development. To make recommendations with regard to improvement of fuel fabrication technologies, FE and FA designs.

R&D	Implementation terms, year	Objective	Expected result
7.2.9 Studies into behaviour of FEs containing composite ceramic (U, Pu)C-based fuel with SiC ceramic coating in the simulated operating conditions of power ramp due to accidental insertion of positive reactivity.	2033-2040	To justify safe reactor operation in the operating conditions of power ramp due to accidental insertion of positive reactivity for certification of the fuel and development of computer codes.	To get experimental data on the properties of nuclear fuel irradiated in the operating conditions of power ramp due to accidental insertion of positive reactivity for its certification and for computer code development. To make recommendations with regard to improvement of fuel fabrication technologies, FE and FA designs.
7.3 Tests of advanced structural materials			
7.3.1 High-dose irradiations (over 110 dpa) of advanced cladding materials (ferritic-martensitic and austenitic steels) at temperatures of 350÷700 C in dismountable devices, to study their mechanical properties, swelling, irradiation creep.	2028-2040	To justify the operability of fuel claddings and internals based on new ferritic-martensitic and austenitic steels for doses over 110 dpa. To increase the fuel lifetime to 5÷10 years, to increase the load factor. To ensure 50÷60 years of service life for irremovable reactor components.	To get experimental data on the properties of new ferritic-martensitic and austenitic steels for their certification and computer code development. To enlarge databases of the properties of materials.

R&D	Implementation terms, year	Objective	Expected result
7.3.2 In-core studies into creep rupture strength and creep of advanced ferritic-martensitic and austenitic steels at temperatures of 350÷700 C.	2028-2040	To justify the operability of fuel claddings and internals based on new ferritic-martensitic and austenitic steels for doses up to 170 dpa. To increase the fuel lifetime to 5÷10 years, to increase the load factor. To ensure 50÷60 years of service life for irremovable reactor components.	To get experimental data on the properties of new ferritic-martensitic and austenitic steels for their certification and computer code development. To enlarge databases of the properties of materials.
7.3.3 In-core studies into strain capacity and irradiation embrittlement of advanced ferritic-martensitic and austenitic steels at temperatures of 350÷700 C and different loading rates.	2028-2040	To justify the operability of fuel claddings and internals based on new ferritic-martensitic and austenitic steels for doses up to 170 dpa. To increase the fuel lifetime to 5÷10 years, to increase the load factor. To ensure 50÷60 years of service life for irremovable reactor components.	To get experimental data on the properties of new ferritic-martensitic and austenitic steels for their certification and computer code development. To enlarge databases of the properties of materials.

R&D	Implementation terms, year	Objective	Expected result
7.3.4 High-dose irradiations up to 170 dpa of special heat-resistant materials operational at temperatures of (750÷950) °C and (1000÷1100) °C to study their mechanical properties, swelling, irradiation creep, creep rupture strength; studies into carbon-graphite material degradation.	2033-2040	To study radiation resistance of special heat-resistant materials to be used in reactors for production of hydrogen fuel and other advanced technological processes. To increase thermal efficiency, also during the transition to gas coolant (coolant).	To get experimental data on the properties of special heat-resistant materials for their certification and computer code development. To enlarge databases of the properties of materials.
7.3.5 High-dose irradiations and studies into new low-absorbing, corrosion-resistant 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.	2033-2040	To study radiation resistance of new low-absorbing, corrosion-resistant materials including those based on silicon carbide and advanced ceramics, to justify the choice of the optimal fuel cladding material for VVER-SCP reactors.	To get experimental data on the properties of new low-absorbing, corrosion-resistant materials including those based on silicon carbide and advanced ceramics for their certification and computer code development. To enlarge databases of the properties of materials.

R&D	Implementation terms, year	Objective	Expected result
7.3.6 Studies into materials characterized by radiation-, heat- and corrosion resistance (relative to lithium coolant). Endurance tests of materials for the first wall, blanket and experimental modules of the thermonuclear reactor.	2033-2040	To study radiation resistance, to select and justify the optimal structural materials for the lithium circuit, first wall and blanket of the thermonuclear reactor.	To get experimental data on the properties of materials characterized by radiation-, heat- and corrosion resistance (relative to lithium coolant) for their certification and computer code development. To enlarge databases of the properties of materials.
7.3.7 Tests of claddings based on vanadium alloys with different coatings.	2033-2040	To justify radiation and corrosion resistance of fuel claddings for sodium-cooled fast-neutron reactors. To increase heat resistance of the claddings.	To enlarge databases of the properties of materials. To improve FE performance characteristics of BN reactors.
7.3.8 Tests of ЭП-823 steel samples to high damaging dose rates to study mechanical properties.	2033-2040	To justify experimentally the internals durability for BREST-OD-300 reactors with heavy metal coolant.	To get experimental data on the properties of the ЭП-823 type steels for their certification and computer code development. To enlarge databases of the properties of materials.

R&D	Implementation terms, year	Objective	Expected result
7.3.9 Tests of structural and functional purpose nanodispersed materials, oxide dispersion strengthened ferritic-martensitic steels.	2033-2040	Experimental studies to justify creation of a new class of radiation-resistant materials characterized by high heat resistance at 700 °C and dimensional stability at damaging doses of up to 170 dpa.	To get experimental data on the properties of the ЭП-823 type steels for their certification and computer code development. To increase fuel burnup to 18÷20 %, keeping the coolant performance characteristics of fast reactors.
7.3.10 Tests of nanocomposite coatings based on TiCrNi/Ni-Cr-Fe-Si-B и TiAlN/Ni-Cr-Fe-Si-B.	2033-2040	High-dose experimental studies into characteristics of nanocomposite coatings for justification of higher corrosion resistance and endurance of cladding materials for CPS rods.	To enlarge databases of the properties of materials. To enhance lifetime characteristics, reliability and operational safety of movable CPS rods in various nuclear reactors.
7.3.11 Tests of vessel and in-vessel steels for advanced thermal reactors.	2033-2040	High-dose experimental studies into advanced vessel and in-vessel steels for justification of a long-term design life (over 80 years) of NPP with thermal reactors.	To enlarge databases of the properties of materials. To improve NPP performance characteristics.

R&D	Implementation terms, year	Objective	Expected result
7.4 Tests of absorbing, moderating and composite materials for innovative nuclear reactors.			
7.4.1 Tests of CPS rod mock-ups containing hafnium hydride (HfHx) of various composition, stoichiometry and fabrication technology in steady-state and emergency conditions.	2033-2040	To justify radiation resistance of the absorbing kernels and CPS rods with an increased lifespan for advanced fast reactors.	To get experimental data on the properties of hafnium hydride (HfHx) of various composition, stoichiometry and fabrication technology in different conditions for their certification and computer code development. To increase CPS lifetime from 2÷3 to 8÷10 years.
7.4.2 Tests of dual-purpose dismountable CPS rod mock-ups based on the trap-type Eu_2O_3+Co absorbing composition to high damaging dose rates in steady-state and emergency conditions.	2033-2040	To justify operability of the CPS rods to be recycled after long-term operation by removal of absorbing kernel inserts and producing strong gamma sources.	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.
7.4.3 Tests of CPS rod mock-ups and irradiation devices with $^{11}B_4C$ moderating block to high damaging dose rates.	2033-2040	To justify radiation resistance of the moderating block and operability of the CPS rods with improved physical efficiency for fast reactors.	To improve physical efficiency of the CPS rods, to reduce the amount of absorbing materials in the product, to have the possibility of radioisotope production.

R&D	Implementation terms, year	Objective	Expected result
7.4.4 Tests of CPS rod mock-ups containing rare-earth element hafnates ($Ln2O3+HfO2$, where Ln-Dy, Eu, Gd, Er) of various composition and fabrication technology to high damaging dose rates in steady-state and emergency conditions of various nuclear reactors.	2033-2040	To justify radiation resistance of the absorbing kernels at the damaging dose rate of up to 170 dpa.	To get experimental data on the properties of rare-earth element hafnates ($Ln_2O_3+HfO_2$, 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.
7.4.5 Tests of composite materials like SiC- SiC, B_4C with graphite nanotubes, BN with pyrocarbon at high temperatures and damaging doses.	2033-2040	To study radiation resistance of new composite absorbing materials which have specified working parameters and are irradiated to the damaging dose rate of up to 170 dpa.	To get experimental data on the properties of materials like SiC-SiC, B_4C with graphite nanotubes, BN with pyrocarbon in different conditions for their certification and computer code development. To increase lifetime of reactor core components.
7.4.6 Tests of nanostructured dispersion strengthened boron steel samples (to 2 % B) of various size and geometry to high damaging dose rates.	2033-2040	To study radiation resistance of new composite materials with specified working parameters.	To enhance performance characteristics of CPS rods and other core components, including neutron shield of reactor vessels

R&D	Implementation terms, year	Objective	Expected result
7.4.7 Tests of porous beryllium of different fabrication technologies.	2033-2040	To study radiation resistance of porous beryllium of different fabrication technologies, to justify the product operability.	To get experimental data on the properties of porous beryllium of different fabrication 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.
7.5 Research into new and modified liquid metal coolants and molten salt compositions			
7.5.1 Experimental studies into lead coolant technologies in a loop facility with simulated operating conditions of advanced reactor cores.	2031-2040	To develop instrumentation and methods that will allow for presence of impurities, activation of circuit equipment in BREST reactors.	To improve operational modes. To enhance reliability and operational safety of the BREST-300 reactor.
7.5.2 Experimental studies into lead-bismuth coolant technologies in a loop facility with simulated operating conditions of advanced reactor cores.	2031-2040	To develop instrumentation and methods that will allow for presence of impurities, activation of circuit equipment in SVBR reactors.	To improve operational modes. To enhance reliability and operational safety of the SVBR-100 reactor.

R&D	Implementation terms, year	Objective	Expected result
7.5.3 Experimental studies into molten salt composition technologies for fusion neutron source (FNS) blankets and minor actinides (MA) burner reactors in a loop facility with simulated operating conditions of FNS blankets and advanced reactor cores.	2033-2040	To develop instrumentation and methods that will allow for presence of impurities, activation of circuit equipment of reactors with molten salt fuel compositions for FNS blankets and MA burner reactors.	To get experimental data on the molten salt coolant technology for development of circuit equipment and control devices.
7.5.4 Experimental studies into helium coolant technologies in a loop facility with simulated operating conditions of advanced reactor cores.	2035-2040	To develop instrumentation and methods that will allow for presence of impurities, activation of circuit equipment of high temperature gas-cooled reactors.	To get experimental data on the molten salt coolant technology for development of circuit equipment and control devices.
7.6 Life tests of new types of equipment for innovative nuclear reactors			
7.6.1 Tests of mock-ups of innovative circuit equipment, sensors and controls for reactors with lead coolant.	2031-2038	To get experimental data on performance characteristics of innovative equipment, sensors and devices for reactors with lead. To get experimental data on performance characteristics of innovative equipment.	To get experimental data for creating innovative equipment, sensors and control devices. To enhance reliability and operational safety of BREST-300 reactors.

R&D	Implementation terms, year	Objective	Expected result
7.6.2 Tests of mock-ups of innovative circuit equipment, sensors and controls for reactors with lead-bismuth coolant.	2031-2040	To get experimental data on performance characteristics of innovative equipment, sensors and devices for reactors with lead-bismuth coolant.	To get experimental data for creating innovative equipment, sensors and monitoring devices. To enhance reliability and operational safety of SVBR reactors.
7.6.3 Tests of mock-ups of innovative circuit equipment, sensors and controls for reactors with molten salt coolant.	2031-2040	To get experimental data on performance characteristics of innovative equipment, sensors and devices for reactors with molten salt coolant.	To get experimental data for creating innovative equipment, sensors and control devices for reactors with molten salt coolant.
7.7 Conducting of reactor physics, materials, thermal hydraulics and other research for computer code verification.			
7.7.1 Conducting of calibration and benchmark model experiments for multiscale simulation of the core component mock-up behavior in reactors with lead coolant in different operating conditions.	2033-2040	To get experimental data for development of integral codes to simulate reactor cores with lead coolant in different operating conditions.	To create and verify integral codes for simulating reactor cores with lead coolant in different operating conditions.

R&D	Implementation terms, year	Objective	Expected result
7.7.2 Conducting of calibration and benchmark model experiments for multiscale simulation of the core component mock-up behavior in reactors with lead-bismuth coolant in different operating conditions.	2033-2040	To get experimental data for development of integral codes to simulate reactor cores with lead-bismuth coolant in different operating conditions.	To create and verify integral codes for simulating reactor cores with lead-bismuth coolant in different operating conditions.
7.7.3 Conducting of calibration and benchmark model experiments for multiscale simulation of the core component mock-up behavior in reactors with sodium coolant in different operating conditions.	2033-2040	To get experimental data for development of integral codes to simulate reactor cores with sodium coolant in different operating conditions.	To create and verify integral codes for simulating reactor cores with sodium coolant in different operating conditions.
7.7.4 4 Conducting of calibration and benchmark model experiments for multiscale simulation of the core component mock-up behavior in reactors with molten salt coolant in different operating conditions.	2033-2040	To get experimental data for development of integral codes to simulate reactor cores with molten salt coolant in different operating conditions	To create and verify integral codes for simulating reactor cores with molten salt coolant in different operating conditions.

R&D	Implementation terms, year	Objective	Expected result
7.8 Applied experimental work with the use of reactor radiation			
7.8.1 Experimental studies for justification of neutron therapy technologies.	2033-2040	To develop and to justify experimentally neutron therapy technologies.	To create technologies for neutron therapy and to bring neutron beams into practical use for medical purposes.
7.8.2 Tests of trap-type irradiation devices for production of various radionuclides with low neutron capture cross-sections.	2033-2040	To develop technologies for producing radionuclides with low neutron capture cross-sections.	To set up production of low neutron capture cross-section radionuclides.
7.8.3 Creation of improved technologies and production of ^{60}Co, ^{153}Gd, ^{89}Sr, ^{63}Ni in the radial reflector of the MBIR reactor.	2033-2040	To develop and to justify experimentally technologies for producing ^{153}Gd, ^{89}Sr, ^{62}Ni, ^{60}Co radionuclides.	To prepare licence agreements on the technologies. To produce ^{153}Gd, ^{89}Sr, ^{62}Ni, ^{60}Co radioisotopes and sources based on them.

R&D	Implementation terms, year	Objective	Expected result
7.8.4 Experimental studies in support of the facility for neutron radiography and tomography of irradiated materials and products.	2033-2040	To develop and to justify experimentally methods of neutron radiography and tomography of irradiated materials and products.	To study fuel, absorbing and structural materials and mock-ups of various reactor core components irradiated in the MBIR reactor.
7.8.5 Neutron transmutation doping of silicon	2033-2040	To develop technologies for neutron transmutation doping of silicon.	To create technologies for neutron transmutation doping of silicon and to set up production of doped silicon of up to 30 tonnes/ year.
7.8.6 Mass neutron activation analysis.	2033-2040	To perform mass multi-element analysis of biological, environmental and geological samples.	To study the content of toxic elements in the environment (Fe, Zn, Cr, Sb, W, Sc, La, Yb, Th, Na, Rb, Cs). To study new medicines and absorbers in biotechnology, the quality and safety of food products

8. SUBSTANTIATING THE NEED FOR EQUIPPING THE MBIR REACTOR WITH INDEPENDENT LOOP FACILITIES.

To ensure reliable and safe operation of fast reactors with any coolant, reactor tests of various fuel types, including those with minor actinides, should be carried out. To actually confirm the fast reactor lifetime characteristics not only in normal operating conditions, but also in emergency modes, tests with coolant quality deviations from the specified one and tests of fuel elements with deep power manoeuvres should be carried out. In addition, NPP performance characteristics are largely determined by the coolant temperature level at the reactor outlet and fuel burnup, which can be adequately substantiated only with the integrated effect of all damaging factors in conditions of FE reactor testing.

The remaining service life of BOR-60 will not allow long-term lifetime tests of fast reactor fuel elements with advanced fuel types and new structural materials of the fuel cladding (including composite ceramic materials), which ensure a higher level of coolant temperature, or tests for verification of emergency codes, or tests for the thermonuclear reactor. Moreover, tests in independent “channel-loops” that were carried out in the BOR-60 reactor and planned for the MBIR reactor do not allow to obtain important results due to the limited experimental capabilities and unstable heat removal from the “channel-loop” coolant to sodium through the gas gap between tubes in comparison with full-scale independent loop facilities with various coolants.

The experimental research programme for the MBIR reactor, which is also designed to solve the problems of creating a nuclear power system that meets the principles of sustainable development, is aimed at assisting in the development of a nuclear power system capable of using uranium 238 and thorium 232 efficiently. This implies the creation of a multicomponent nuclear power system resting not only on thermal and fast solid-fuel reactors but also on liquid-fuel reactors helping to effectively close the nuclear fuel cycle for minor actinides. Besides, there will be fusion neutron sources with liquid fuel blankets for production of uranium 233 from thorium 232 and gas-cooled reactors for production of hydrogen and for the needs of high-temperature petrochemistry.

Experiments for justification of the creation of appropriate solid-fuel and liquid-fuel compositions and corresponding structural materials, as well as those performed for a justified choice of operating modes of such systems, are best carried out in special loops for experiments with liquid-fuel (molten-salt) compositions or in independent gas loop facilities. At the initial stages of research, it is possible to irradiate capsules with corresponding fuel compositions and structural materials. The research programme should also provide experiments for justifying the development of metal and thorium fuels.

Experimental capabilities of the reactor, including loop facilities, should be sufficient for conducting experimental research under development programmes not only for the national nuclear industry, but also for performing work for foreign customers, including joint international projects within the framework of the International Research Centre on the basis of MBIR.

In this regard, after the MBIR reactor with its in-pile experimental devices (instrumented and non-instrumented channels) is commissioned and its experimental operation is carried out, it is planned, in accordance with mutually consistent specific research programmes of the Customers, to start work on the construction of independent high-power loop facilities with various coolant types for testing FA mock-ups in simulated steady-state, transient and emergency operation modes in the period from 2028 to 2030.

Creation of LFs at the MBIR reactor, as well as the experimental capabilities of the future multifunctional radiochemical complex (MRC), will make it possible to do extensive tests of fuel elements and structural components of G4 reactors with different coolants, to be followed by the studies conducted in the hot cells of the JSC “SSC RIAR”.

CONCLUSION

The MBIR reactor is required for implementation of the strategy of the two-component nuclear power in Russia for the period up to 2050 based on fast and thermal reactors as well as for justification of the technologies for nuclear fuel cycle closure.

It is planned to complete the MBIR reactor construction and to gain first criticality by 2027 and to commence scheduled activities on materials research in 2028.

The main purpose of the research reactor facility MBIR is to conduct extensive reactor tests on innovative materials and component mock-ups for 4G nuclear systems, including fast neutron reactors with a closed nuclear fuel cycle, as well as small and medium power thermal reactors.

Experimental capabilities of the reactor, including loop facilities, should be sufficient for conducting experimental research under development programmes not only for the national nuclear industry, but also for performing work for foreign customers, including joint international projects within the framework of the MBIR–based collective use international centre.

The initial stage of the MBIR reactor operation assumes the absence of loop channels, one of which (CLC) occupies 7 cells in the core centre. It is impractical to use all these cells for materials science packages because of their strong impact on each other. To eliminate this impact, it is proposed to modify the MBIR core by placing 3 materials science packages in the CLC cells and one more material science package for power distribution flattening. As a result, the MBIR reactor irradiation volume at the initial stage of operation increases to 20 cells, the number of fuel assemblies is reduced from 93 (design) to 85 (at the initial stage).

The damaging dose rate in materials science packages located in the core is from 16 to 6.7 dpa per refuelling interval (100 eff. days). The internal volume of one MP is 2300 cm3.

A further increase in MBIR’s irradiation capabilities is possible if radial reflector cells are used.